Uncertainty analyses of Reactivity Initiated Accidents and ATWS events

Ph.D. thesis booklet

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Background

One of the most important duties of nuclear energy is to establish and to prove the safety of NPPs (**N**uclear **P**ower **P**lants). However, at NPP Paks, during the operation of the Soviet designed NPPs, used to be some changes in these reactors (e.g. utilization of new fuel types, power upgrade projects, etc.). Hence, from time to time, it is necessary to renew the safety analyses of these reactors.

In the KFKI Atomic Energy Research Institute there have been many research projects for DBA (**D**esign **B**asis **A**ccident) analyses of NPPs (especially for the VVER-440 type reactors of Paks) for several decades. These analyses are basically deterministic, and they are used for the events of normal operations, anticipated operational occurrences (the frequency of initial event > 10^{-2} per year) and postulated accidents (10^{-2} per year > the frequency of initial event > 10^{-5} per year) of NPPs.

The final goal of DBA analyses is to investigate the fulfillment of the predescribed acceptance criteria (e.g. max. fuel temperature), such a way the integrity of the fuel pins and their cooling conditions are investigated. Using the defense-in-depth philosophy (decreasing the frequency of initial event the criteria are less strict), the acceptance criteria are determined for the different initial events taking into account previously measured data.

Those accidents (or beyond Design Basis Accidents) - which have frequencies less than 10^{-5} per year - are not discussed in the DBA analyses. In these accidents the active zone could be injured to a great extent, and the level of radioactive release can exceed some limits.

Investigating the fulfillment of the acceptance criteria, the calculations are usually performed in two steps. Firstly, assembly-wise system calculations (global/nodal calculations) are performed and than (using the results of nodal calculations) the hottest sub-channels of the active zone are investigated (hot channel calculations).

As a background of my work, I have to clear that the traditional safety analyses follow the cumulative conservative method, which means that the extreme conservative values are supposed simultaneously for all the relevant input parameters.

In the conservative methodology the enveloping values of the respective input parameters are given such a way - partly to lead to the most adverse consequences - partly to cover the uncertainties of methodological origin, simultaneously. Nowadays, mainly the above mentioned conservative methods are accepted in the licensing processes of NPPs. However it is worthwhile to mention that recently even in these conservative methods BE (Best Estimate) calculation models are used and only the input parameters are treated in a conservative manner. This method is suitably conservative concerning the fulfillment of the acceptance criteria.

However, the development of algorithms applied in the simulations of thermo-hydraulic transients is leading to the propagation of the so called BE calculations (BE inputs and BE calculation model). But the supplement of best estimate calculations with safety margins is necessary, it is generally agreed.

Taking into account the above situation a demand had arisen to elaborate the uncertainty analysis methodology of RIA (**R**eactivity **I**nitiated **A**ccidents) and ATWS (**A**nticipated **T**ransient **W**ithout **S**cram) events and to apply the elaborated method for some representative transients.

The uncertainty analyses are one of the continually developing methods of safety analyses. These analyses are used in the licensing processes of NPPs only in a few countries. More research institutes develop own method, and these methods are very similar in their goals (e.g. mainly investigation of thermo hydraulic transients), but they can differ in the used mathematical methods to a great extent.

Performing the safety analyses, mainly four requirements must be achieved, which requirements determine the necessary researches in this topic:

- it is necessary to determine such **events (transients)** for which the safety analyses are used,
- it is necessary to select the neutronic and/or thermo hydraulic codes to be used, and these codes must be **well validated**,
- furthermore **fast and reliable numerical methods** have to be used in the selected codes,
- it is necessary to elaborate the execution methodology of the safety analyses (e.g. conservative methodology or uncertainty analysis), it has to meet the Hungarian and foreign (up-to-date) requirements.

At the beginning of my Ph.D. work, the above mentioned steps had been reviewed and the development of the last three steps seemed to be necessary and worthwhile for us.

The selected nodal neutronic code was the KIKO3D three-dimensional reactor dynamics code, which had been developed in the KFKI Atomic Energy Institute. It has been continually used for the safety analyses of RIA, from the beginning of 90's, the code has been continually developed and validated, the results of these validations were summarized at the beginning of my Ph.D. work in [16].

Goals

Before elaborating the methodology of uncertainty analyses, - in order to effectively perform the analyses - it was necessary to solve some preliminary problems.

My first goal was the further validation of KIKO3D three-dimensional neutron physics code, because earlier the validation of the code happened only by few comparisons between measurements done on original-size facilities and calculations. In the frame of the VALCO EU-5 project, I was going to simulate the transient states measured in an original-size VVER-1000 mock-up in order to investigate the efficiencies of the nodal and IQS (**I**mproved **Q**uasi **S**tatic) methods applied in KIKO3D code.

As the uncertainty analyses need fast and reliable codes, my second goal was to make KIKO3D faster applying a more effective numerical method for the solution of the large sparse-matrix inhomogeneous equation. This equation is coming from the two group diffusion approximation of the neutron transport equation, and my goal was the verification of the applied numerical methods, as well.

Before elaborating the methodology of uncertainty analyses, it was necessary to clarify some important and topical questions related to the safety and uncertainty analyses of the most important RIA and ATWS events of VVER-440 reactors. These questions were related to the sensitivity investigations and methodical questions of hot channel calculations.

My main goal was the elaboration of a new method based on consistent mathematical methods (uncertainty analysis) for the safety analysis of RIA and ATWS events, taking into account the literature. Up to now, this method has not been used for the RIA and ATWS events, although the 'frame parameter' concept - used in the Hungarian practice – could be a good basis. In the course of realization of the new method, my goal was to avoid the use of unnecessary cumulative conservative input parameters (e.g. initial and boundary conditions) in the uncertainty analyses. I wanted to solve this problem by using statistical methods. Firstly, I wanted to apply the elaborated method for nodal-wise uncertainty analyses assuming conservative hot channel methodology (mixed method).

After that, my further goal was to extend the above method for the hot channel calculations too, because the evaluation of the acceptance criteria was done at the level of hot channel. I was going to determine some parameters concerning the acceptance criteria and to estimate the number of the failed fuel rods, using the elaborated methodology of uncertainty analyses.

Of course, my goal was to test in practice the elaborated methods. I have planned an uncertainty analysis of RIA (a rod ejection transient) using conservative hot channel calculations and a hot channel-wise uncertainty analysis of an ATWS even. I have planned, the comparison of the results coming from the conservative methodology and from the uncertainty analysis, as well. The above mentioned 'sample' transients have been selected because in these cases some fuel pins could fail according to the conservative calculations, and than the use the use of uncertainty analyses can be reasonable.

New scientific results

1. I was the first to elaborate the uncertainty analysis methodology of Reactivity Initiated Accidents (RIA) using the KIKO3D and TRABCO codes. I have pointed out the advantages of uncertainty analyses against cumulative conservative method:

- Taking into account the small frequency of initial events the cumulative conservative safety analyses are unduly conservative in many cases.
- It is an advantage that the level of conservatism is quantified by statistical parameters (probability content and confidence level) in case of uncertainty analyses.
- We can perform the sensitivity investigations needed by the Nuclear Regulatory Guide. As a result of sensitivity investigations it can be seen which input parameters are important for the given transient, which parameters' probability density functions are needed to specify **[4, 5, 8, 12, 13]**.

2. I was the first to employ the elaborated method for the uncertainty analysis of an Reactivity Initiated Accident (namely for a rod ejection transient of a VVER-440 reactor) using the KIKO3D and TRABCO codes, and using the UAMRA[1] program package elaborated by me. Thus, it can be concluded:

- I have determined the upper and lower limits (with a probability content of 95% and a confidence level of 95%.) of some characteristic output parameters of the nodal calculations and of the hot channel calculations. In the case of hot channel calculations several parameters were treated in a conservative manner.
- I have compared the results with the results of conservative calculations, and by the comparisons I have proved that the cumulative conservative methodology leads to unduly conservative results many times.
- I have pointed out that the uncertainty analysis is applicable even if some conservative assumptions are used, although in that case the results must be interpreted by taking into account the applied conservatism **[8, 12, 13]**.

3. Using the all available measurements measured in original-size VVER-1000 critical system I have validated the KIKO3D code at hot zero power in the frame of the VALCO EU-5 project. Thus, it can be concluded:

- However I was not able to prove the exact criticality of the measured stationary states (I have over-predicted the values of k_{eff} - effective multiplication factor - with more than 1 %), but I have pointed out that it is not possible to reach higher accuracy. I have explained this statement with the uncertainties (errors) coming from the measurements and with the uncertainties (errors) coming from the specialty of modeling (using nodal method): e.g. the

[1] UAMRA: **U**ncertainty **A**nalysis **M**ethodology of **R**IA and **A**TWS events

calculated reactivity-difference (due to the estimated uncertainty of boric acid measurement) is approximately ± 600 pcm.
- When control rods were fully out, I have simulated the stationer power distribution with acceptable accuracy. Using the results of my calculations I showed that if there is significant heterogeneity inside the assemblies (assemblies containing absorbers) than the use of the nodal method can lead to high error (even more than 50 %) in the simulation of central power of these assemblies.
- For the transient states I have demonstrated that the common use of the nodal and the IQS method –even in case of using relatively large assemblies – is applicable for the simulations of kinetic transients of VVER-1000 reactor at hot zero power **[1, 6, 7].**

4. I have built a novel solver (Bi-CGSTAB algorithm) to the KIKO3D code (which was fairly fast originally, as well) for solving the large sparse-matrix equation system coming from the two group diffusion approximation of the neutron transport equation. I have tested the improved code against special matrixes and I was validated it by the simulation of a measured transient at a VVER-1000 critical system. I have proved that the error of the solution has been estimated more accurately in case of using the new algorithm (Bi-CGSTAB). I have demonstrated that - using the Bi-CGSTAB algorithm and my new estimation for the solution's error – the calculated results converged safely to the real results. Additionally, I have reached that the improved code became 7%–12% faster than the old one (used GMRES algorithm) [2, 10, 14].

5. Using COBRA hot channel calculations and the heat transfer model of the KIKO3D code, I was the first to solve - for the most important RIA and ATWS events of VVER-440 reactors – the following important (in the point of view of safety and uncertainty analyses) and up to now unsolved problems:
- I have proved that the assumption of flat power distribution or power distribution corresponding to zero burnup in the pellet leads to non-conservative results concerning DNB (**D**eparture from **N**ucleate **B**oiling).
- It was shown for an ATWS transient that the sub-channel enthalpy limit alone is not an appropriate frame parameter for limiting the coolant thermal hydraulic state in the safety analyses, while if the maximal pin power is selected for limitation, the flat radial pin-wise power distribution inside the assembly is a conservative assumption.
- I have demonstrated by comparisons of multi-channel and closed single-channel calculations assuming of radially flat pin-wise power distribution that the single-channel calculations are BE or slightly conservative if the flow area is decreased by two percents in the single channel calculation.
- Concerning the results, it must be mentioned that the investigations had been performed for VVER-440 assemblies (first of all for the assemblies have recently used in Paks NPP). For different assembly geometry (e.g. water gap between the assemblies) and/or for different reactor types other conclusions cannot be excluded **[3, 11].**

6. I was the first to elaborate the uncertainty analysis methodology of hot channel calculations using the one sided tolerance limit method combined with the response surface method and using the TRABCO hot channel code. I was the first to solve the realistic determination of the number of failed fuel rods using the elaborated uncertainty analysis methodology. Using the UAMRA program package (developed by me), the method was applied for an ATWS transient of a VVER-440 reactor. Because of the small frequency of the initial event of this transient, in the course of licensing process some relevant parameters are treated in a BE manner in the hot channel calculations. I

compared the results with the results coming from different approaches. Thus, it can be concluded:

- I have pointed out that the estimated number of failed fuel rods depended to a great extent on the applied conservatism, on the applied probability characteristics: the estimated numbers of the failed fuel rods were between 250 and 8276.
- I have pointed out as well, that simple use of the best estimate power distribution (without taking into account the uncertainties) is non-conservative concerning the failed fuel rods.
- I have demonstrated that the 'full' hot channel uncertainty analysis was leading to the minimum number of (estimated) failed fuel rods in 0.95/0.95 sense.
- However, the more detailed analysis led to locally higher temperature values in case of uncertainty analysis than in case of conservative (used some BE parameters, as well) analysis, but the acceptance criteria were fulfilled.
- I have demonstrated the effectiveness of the uncertainty analyses in such cases, when fuel failure can happen or the acceptance criteria are approached.
- Concerning the results, it must be mentioned that they are relating mainly for the concrete transient. In case of other transients/reactor types, the concrete values of estimated number of failed fuel rods can differ to a great extent from the above ones, but the tendencies will be probably similar **[4, 5, 9, 15].**

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