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Statistical and sensitivity analysis of failing rods in EPR LB-LOCA

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Abstract. A statistical fuel failure analysis procedure has been developed [1], and now applied in full scale to a large break loss-of-coolant accident (LBLOCA) scenario in EPR-type nuclear power plant. The goal is to statistically determine the number of failing fuel rods. It is also important to bring out the underlying causes for rod failures, and therefore a sensitivity analysis procedure for the preceding complex calculation chain is outlined and adopted.

In order to produce a statistically reliable estimation on the number of failing rods, a large number of simulations are required. In the analysis, single rods are simulated with the coupled fuel performance - thermal hydraulics code FRAPTRAN-GENFLO. The statistically varied factors are divided into global and local by their range of influence. Transient thermal hydraulic and power history boundary conditions and model parameters are global and affect all the rods. Fuel rod manufacturing parameters and other rod position related factors are local and are related to a certain rod. The number of global scenarios is 59 as dictated by the Wilks' formula, and 1000 randomly sampled rods are simulated in each scenario. The thermal hydraulic and transient power boundary conditions for the fuel performance code are calculated with a statistical version of the system code APROS. The steady-state irradiation histories are simulated with FRAPCON, and the steady-state power histories used as boundary conditions for FRAPCON are obtained from SIMULATE calculations. Thus, a multistage calculation chain is required to consolidate the procedure. As an outcome, in the worst global scenario, 1.4% of the simulated rods failed. It can be concluded that the Finnish safety regulations, i.e. max. 10% of the rods allowed to fail, are met.

In order to find out which input parameters have significance to the outcome of the analysis, a sensitivity analysis is done. At the moment, only the worst global scenario is considered but later on it is possible to include also the sensitivities of the global factors. Due to complexity of the existing data, first the relevant input parameters for the sensitivity analysis have to be specified. Data visualization with a cobweb graph is used for the screening. Then, selected sensitivity measures are calculated between the chosen input and output parameters. The sensitivity indices calculated are the Borgonovo's delta measure, the first order Sobol' sensitivity index, and squared Pearson correlation coefficients. The first mentioned is a novelty in this context. As an outcome, the most relevant parameters with respect to the cladding integrity were determined to be the decay heat power during the transient, the thermal hydraulic conditions in the rod's location in the reactor, and the steady-state irradiation history of the rod as represented in this analysis by the rod burnup.

Keywords: loss-of-coolant accident, statistical fuel failure analysis, sensitivity analysis, FRAPTRAN-GENFLO, EPR

INTRODUCTION

The Regulatory Guides on nuclear safety [2] set by the Finnish nuclear safety authority STUK introduce a number of design criteria that the fuel has to fulfil in accident conditions. Among others, the following criteria are applicable in LOCA conditions: less than 10% of the rods in the reactor are allowed to be damaged, and the peak cladding temperature should not exceed 1 200 ºC. The estimation of the fraction of failing rods is traditionally based on conservative analyses but this approach has several downsides. It is hard to judge a priori whether the assumptions are conservative because the phenomena in the reactor are highly nonlinear. Therefore, a statistical calculation system has been developed and described in [1], and applied now in full reactor scale.

The tolerance interval theory [3] gives means to determine the number of simulations that are needed for the statistical analysis. The upper and lower limits in the result parameter distribution are chosen in a way that a given probability content γ would be inside those limits with a confidence level β. In the case of one-sided tolerance interval, the lower bound is chosen to be minus infinity and the upper bound is the highest value (in first order formulation) in the random sample picked from the distribution. Based on this, one can derive a well-known relation known as the Wilks' formula [4]:

$$\beta = 1 - \gamma^N. \tag{1}$$

When this formula is applied to deterministic safety evaluations, the acceptable level is a 95% probability with a 95% confidence that the number of failed fuel rods would not overstep the allowed limit [2]. When the corresponding values are inserted into Equation (1), the number of cases comes out as 59. The number of calculations is independent of the number of initial parameters included in the analysis. Furthermore, the figure is valid for one output quantity [4], and in this analysis we are interested in just one, the total number of failing rods.

The applied simulation codes are a modified version of the FRAPCON code for the steady-state simulations [5] and FRAPTRAN-GENFLO for the transient calculations [6, 7]. The thermal hydraulics boundary conditions needed for the transient analysis with FRAPTRAN-GENFLO are obtained from the system code APROS [8]. In addition to transient boundary conditions, the steady-state power histories needed in FRAPCON simulations are generated with neutronics codes.

The simulated accident scenario may be summarized as follows. As the initiating event, a double ended break in a cold leg opens. Due to pressure decrease, reactor and turbine trips follow. Simultaneously with the turbine trip, the offsite power is lost and the main recirculating coolant pumps start to coast down. The core is uncovered. The content of the accumulators is injected to the primary loop when the accumulator pressure is reached. After a delay, the diesel generators are started and the medium, and later on the low head safety injections start operation and the core is quenched.

The cause of a fuel rod failure in a LB-LOCA is the sum of many interconnected phenomena. Thermal hydraulic conditions in the rod's location during a LOCA play an important role, as well as the decay heat power during the transient. In local level, the rod's irradiation history prior to the accident and the resulting state of the rod has an effect. Also rod design parameters such as fuel enrichment and gadolinia content and tolerances in the fuel manufacturing parameters may have a contribution. To determine the relevant input factors, and to quantify their relative importance, a sensitivity analysis is done by using the data from the statistical analysis.

APPLICATION OF THE CALCULATION SYSTEM

The initial parameters of the statistical analysis can be divided into two groups by their range. Global parameters have an effect on all the rods in the reactor, whereas local parameters bring variation only to individual rods. For example, the model parameters of a fuel performance code are global, whereas fuel manufacturing parameters are local. The above mentioned 59 simulation cases are constructed in a way that selected model parameters in the codes, and the fuel performance codes' boundary conditions are different between the scenarios. The values for fuel manufacturing parameters are random sampled from their distributions; those have the same values in each global scenario. The uncertainties related to the neutronics calculations used to produce the base irradiation pin power histories are left outside the study; that may be considered in future studies.

In practice, when using the Wilks' formula, one can state that when all the rods in the reactor are simulated 59 times with variations between each of the 59 scenarios, and if the number of failed rods in the worst case is below the allowed limit, then the safety requirements are rightly met with the probability of 95% and with the confidence level of 95%. As that kind of number of transient fuel performance code simulations, i.e. 59 times the 63 865 rods in EPR, is out of reach with the computer resources of today, some other method or approach is needed alongside in order to reduce the amount of simulations.

In our approach, a limited number of random sampled rods are simulated for each global scenario. The number of simulations is chosen to be 1000, in order to keep the computation time still reasonable (less than two weeks with 25-32 simultaneous simulations in cluster), and yet the number to be sufficient. In order to be able to pinpoint the worst global scenario unambiguously, the same rods are simulated in each global scenario. The number of failed rods in the worst global case can then be directly scaled to find out the number of failed rods in the whole reactor. This approach is on the conservative side because with a smaller number of cases, the deviation of the number of failures grows. Therefore, the highest failure number is likely to be higher than it would have been if all the cases had been calculated. In a thorough analysis, the rest of the rods among the global scenarios may be taken into account by fitting a neural network for each global scenario by using the data from the 1000 simulated rods per scenario, as explained in [9]. The trained network may then be applied to reduce the deviation resulting from the extrapolation.

In the LOCA scenario considered in this paper, the accident occurs after the 4th cycle. At the end of cycle 4, there are five different assembly types in the core. These types vary in U-235 enrichment, the number of Gd rods in the assembly, and the Gd_2O_3 content in the Gd rods. About half of the assemblies have been in the core only for the cycle 4. The rest have been irradiated for two cycles: one bundle during the cycles 1 and 4, and all the others during the cycles 3 and 4. The cycle lengths for cycles 1, 3 and 4 are 18, 24 and 24 months, respectively.

Codes in the Calculation System

The calculation system is presented in Figure 1. The primary calculation tool is the coupled fuel performance–thermal hydraulics code FRAPTRAN-GENFLO. FRAPTRAN (version 1.4) is a single-rod transient fuel performance code developed by PNNL for U.S.NRC [5]. FRAPTRAN has been coupled with general thermal hydraulics code GENFLO [10], developed at VTT, to improve the thermal hydraulics modeling in LOCA. For each time-step and axial segment, GENFLO calculates the coolant temperature and the clad-to-coolant heat transfer coefficients. GENFLO contains a five-equation thermal hydraulics model (two energy and mass equations, one momentum equation) with drift-flux phase separation. The code is able to simulate the long-lasting reversed core flow which is possible in EPR. The default LOCA cladding failure criterion in FRAPTRAN is applied in rod failure predictions. The ballooning model of FRAPTRAN assumes that local non-axisymmetric cladding ballooning begins when the effective plastic strain in any axial segment of the cladding exceeds the instability strain given by the material properties package MATPRO. The rod is considered to fail when the indicator for the rod failure is triggered.

The steady-state initializations of the transient calculations are performed with the U.S.NRC/PNNL FRAPCON code (version 3.4), with statistical features introduced at VTT [11]. For the FRAPCON simulations, the steady-state power histories of all the rods in the reactor are needed. SIMULATE 3 [12] was used to simulate the core and to obtain the power histories. In order to obtain the time-dependent transient boundary conditions for the fuel performance code calculations, the progress of the accident was simulated 59 times with the system code APROS, developed jointly by VTT and Fortum. The relevant boundary conditions obtained from APROS are the enthalpy and the mass flows of both liquid and vapor at the channel inlet and outlet, the coolant pressure and the rod power evolution. Also axial power profiles are provided by APROS. The applied dynamic one-dimensional two-phase flow model of APROS simulates the behavior of a system containing liquid and gas phases, covering all heat transfer modes. The system is governed by six partial differential equations, from which pressures, void fractions and phase velocities and enthalpies are solved. The phases are coupled to each other with empirical friction and heat transfer terms.

Varied Parameters

As a first step of a statistical analysis, the varied parameters have to be chosen and their distributions defined. The parameters chosen to be varied in the VTT's system are collected into Table 1. The parameters that are important regarding the whole core are varied in APROS, and those that are important in fuel behavior analysis are varied in fuel performance codes. Generally, the parameters affecting the thermal hydraulics are varied in APROS. However, the parameters now varied in GENFLO cannot be currently varied in APROS and therefore those are taken into consideration in GENFLO. Meanwhile, fuel-related parameters are kept at their best-estimate values in APROS and varied only in the fuel performance codes. Selected model parameters are varied in FRAPCON. The lack of the current analysis is that no model parameters are varied in FRAPTRAN; this deficiency is mended in future analyses.

TABLE 1. Varied parameters and their distributions.

Global parameters	*Distribution*[1]	*Min.*	*Max.*	*Global scenario #47*[2]
APROS				
Containment pressure [MPa]	N (0.250)	0.200	0.300	0.263
Pump1 inertia [kgm^2]	N (5210.0)	5157.9	55262.1	5247.2
Pump2,3,4 inertia [kgm^2]	N (5210.0)	5105.8	5314.2	5164.9
Decay heat of normal [%]	N (100.0)	92.0	108.0	95.1
Accu2,3,4 pressure [MPa]	N (4.9800)	4.7800	5.1800	4.9484
Accu2,3,4 level [m]	N (4.4600)	4.3600	4.5600	4.5124
Emergency water temperature [°C]	N (50.0)	10.0	50.0	21.4
Emergency water flow [%]	N (100.00)	95.00	105.00	97.47
CCFL parameter	N (1.0000)	0.6900	1.0350	0.9232
Discharge coefficient, RPV side	N (0.8750)	0.7500	1.0000	0.8202
Discharge coefficient, pump side	N (0.8750)	0.7500	1.0000	0.9383
Upper plenum temperature [°C]	N (336.0)	331.0	341.0	332.9
FRAPCON				
Swelling parameter	N (1.0; 0.000144)	not defined		0.995770
Creep rate parameter	N (1.0; 0.25)	0.6	1.1	0.881100
Fission gas parameter	N (0.0; 0.25)	-1	1	0.363080
Thermal conductivity parameter	N (1.0; 0.01)	not defined		1.015300
Cladding corrosion parameter	N (1.0; 0.0004)	0.6	1.4	1.016000
GENFLO				
Basic drift flux velocity	Tri (1.2)	1.13	1.2	1.154200
Drift flux separation constant	Tri (1.1)	1.1	1.2	1.122200
Interphasial heat transfer tuning factor	Tri (0.3)	0.1	0.33	0.183950
Film boiling heat transfer tuning factor	Tri (0.2)	0.08	0.22	0.187390
Transition boiling heat transfer tuning factor	Tri (0.2)	0.18	0.22	0.192250
Local parameters (FRAPCON)	N			
Cladding inner diameter, Fuel pellet diameter, Cladding wall thickness, Cold plenum length, Fuel pellet density, Bottom plenum volume, Internal fill pressure				

[1]*N=Normal (μ,σ^2) or (μ); Tri=Triangular (mode)* [2]*Values in global scenario #47 that revealed to be the worst case*

The parameters and their ranges used in the APROS analysis are based mainly on the BEMUSE data [13, 14] but also other publically available data has been used [15, 16]. The varied model parameters and their ranges in FRAPCON and GENFLO are based on previous analyses done at VTT. Normal distribution is used for the fuel manufacturing parameters, but the exact values are proprietary information. Sampling of parameter values is conducted with the SUSA software developed by the German research organization GRS [17].

Boundary conditions from system code APROS

In APROS, the reactor core is divided into 20 axial nodes for the solution of thermal hydraulics and neutronics. The core is divided into 17 thermal hydraulic channels. The locations of the thermal hydraulic channels in relation to the fuel bundles are presented in Figure 1. As there are no shrouds around the fuel assemblies in EPR, there are cross flows in the core. It is important to take the cross flows into account in the simulations as those balance the flows between bundles with various power levels. However, a single FRAPTRAN-GENFLO simulation is done for a closed subchannel, and the cross flows cannot explicitly be taken into account. However, as the core is divided in APROS into thermal hydraulic channels consisting of several bundles instead of only one bundle per channel, the cross flows are being taken into account in the boundary conditions produced for FRAPTRAN-GENFLO.

The linear power histories and thermal hydraulic boundary conditions are available for 17 channels except the inlet enthalpy for liquid and vapor which are given for eight sectors. The eight sectors take into account the asymmetric temperature distribution in the core resulting from the break in one of the cold legs. The axial power profiles are available for all the 241 bundles. The same static beginning-of-transient axial power profiles are used in all the 59 global scenarios. The linear power histories available for the channels are further refined for each bundle by using a normalized multiplier defined from an APROS output file. The multiplying coefficients are defined using the average power in an APROS channel:

$$coeff_{bundle} = P_{bundle}\frac{total\ number\ of\ bundles\ in\ APROS\ channel}{P_{channel}}. \tag{2}$$

Channel power is obtained by adding up the bundle powers in that particular coolant channel. An approximation has to be made regarding the transient power evolutions in fuel rods: as the transient pin power is not known but only the bundle power, the latter is used in FRAPTRAN-GENFLO simulations as pin power. Similarly, bundle axial power profile is used as pin axial profile.

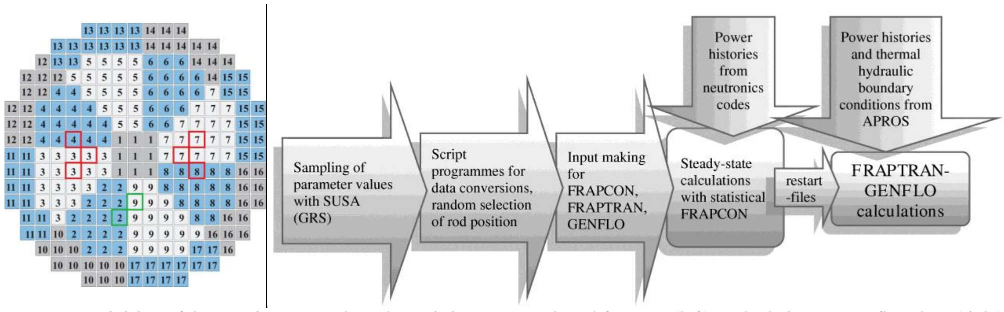


FIGURE 1. Division of the core into 17 coolant channels in APROS, altered from [9] (left). Calculation system flowchart (right).

RESULTS OF THE STATISTICAL ANALYSIS

The numbers of rods that failed in each global scenario are shown in Figure 2. The highest number of failures, 14 rods, occurred in global scenario 47. Thus in the worst case, 1.4% of the rods failed. In each scenario, the same 1000 rods were simulated, and consequently in many cases, the same rods failed in various global scenarios. Some of the simulations were not successfully terminated; in scenario #3, total of 62 simulations ended with an error, 58 of which were due to GENFLO. With respect to errors, this was the worst case. If all the erroneous simulations were conservatively assumed to stand for a rod failure, the total portion would be 6.6%, and the safety criterion is still easily met.

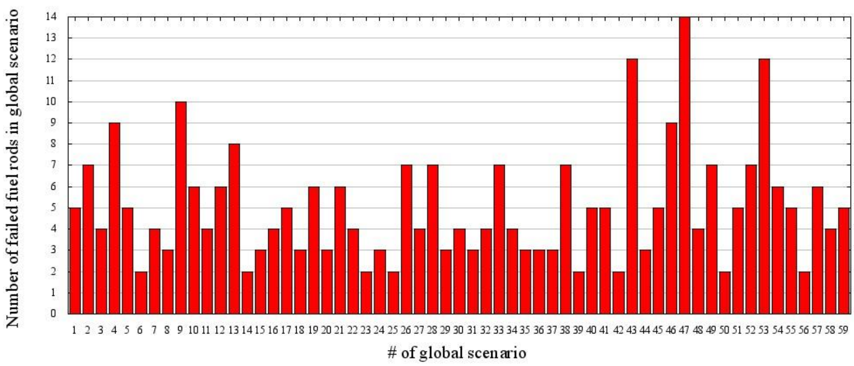


FIGURE 2. Numbers of failed fuel rods in each global scenario; 1000 rods were simulated per scenario.

The global parameter values applied in scenario 47 are presented in Table 1. The evolutions of selected output parameters in the global scenario 47 are plotted in Figure 3. The curves corresponding to the failed rods are indicated with markers. In each scenario, the highest peak cladding temperatures occurred always soon after the beginning of the accident, and were at the same level in all global variations, little below or above 900 ºC. The requirement concerning the highest peak cladding temperature, 1 200 ºC, is thereby met.

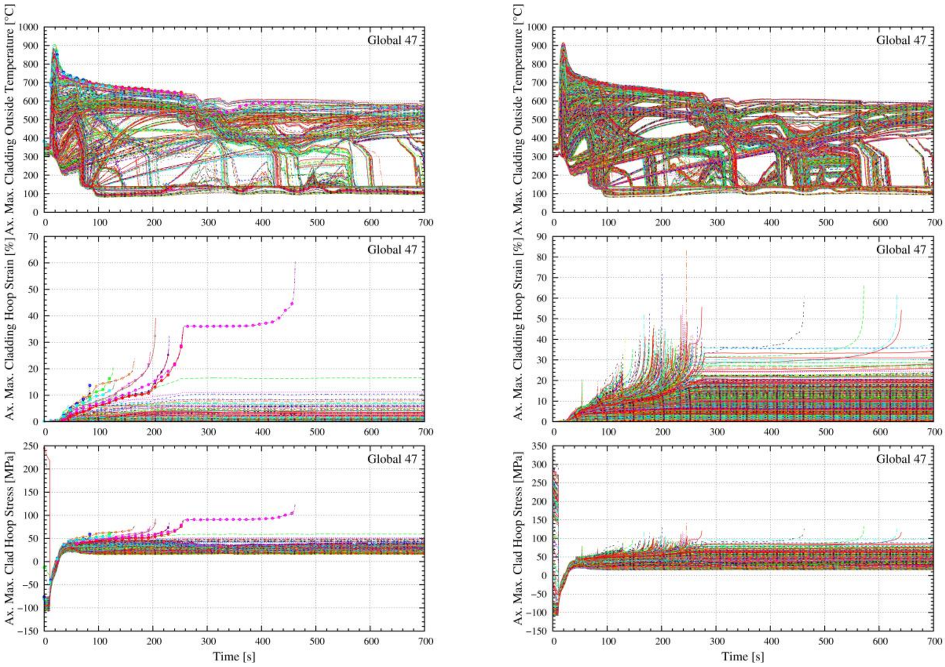


FIGURE 3. Evolutions of key output parameters in FRAPTRAN-GENFLO simulations in the worst global scenario with respect to the number of failing fuel rods. First, 1000 rods were simulated (left hand side), and then all the rods that had been in reactor during cycles 3 and 4 (right hand side).

Compared to the previously published results [9], the numbers of failed rods in almost all the scenarios are few units higher. The reason is that different cladding yield strength correlation in FRAPTRAN is applied between the analyses. The default model is used in this paper, and the older model from [18] was applied in the previous calculations [9]. The model from [18] was previously used as it had proved to give the best results in modeling Halden IFA-650 LOCA series with FRAPTRAN-GENFLO [19]. In addition, a bug in FRAPTRAN was corrected between the two analyses: in previous simulations, false indications of rod failures were given by the low-temperature PCMI cladding failure model during the high temperature phase in LOCA.

Among the analyzed global scenarios, all the failed rods had been in reactor during the 3rd and 4th cycle. In other words, no failures were simulated in the rods that had been in reactor only during the 4th cycle, or during the 1st and 4th cycle. Therefore, by applying the boundary conditions and the global parameter values of the worst global scenario (in terms of the number of failed rods), all the rods in the loading pattern that had been in reactor during 3rd and 4th cycle were simulated with FRAPTRAN-GENFLO. In this smart sample, 1.96% of rods were simulated to end up in fuel rod failure, and 0.44% of simulations crashed. Thus even in a whole reactor scale, taking account the most limiting rods, the safety criterion is met. The evolutions of the output parameters are presented in Figure 3.

SENSITIVITY ANALYSIS

A sensitivity analysis was done using the data of all those rods in the reactor that had been in core during 3rd and 4th cycle. In this analysis only one global scenario, i.e. the worst case is considered. The effect and importance of various local parameters, i.e. the location related parameters and the sampled fuel manufacturing parameters, to the outcome of chosen output parameters is studied. In the future, also global parameters may be addressed in the analysis.

The number of sensitivity analysis methods that are suitable to this problem is limited because the dataset is given, which excludes methods that take advantage of efficient sampling techniques [20]. In addition, the influential input parameters are highly correlated. This results from the facts that, firstly, the phenomena in a nuclear reactor are highly interlinked. Secondly, the data used for the sensitivity analysis originates from a complex calculation system consisting of several codes. If parameters are selected from successive phases of the calculation chain, their effect may be taken into account more than once when performing the sensitivity analysis. The analysis is further complicated by the differences in the level of detail in modeling in various codes.

Because the data used in the sensitivity analysis is complex, even defining the input parameters is not trivial. In the first phase, all the potentially important sampled or calculated input parameters of the LOCA analysis are distinguished. To help sorting out the relevant parameters, a cobweb graph [21] is used to visualize the input variables that lead to high hoop strains in the LOCA calculations. Finally, sensitivities are quantified by calculating various sensitivity indices.

Data pre-processing

Part of the data is scalar while some is in the form of time series. The latter include transient power histories, thermal hydraulic boundary conditions, and steady-state power histories. In order to perform the sensitivity analysis, the time series data needs to be simplified. Some of the thermal hydraulic boundary conditions of the 17 coolant channels vary significantly during the transient, and therefore condensing the boundary conditions into scalars was not attempted. Instead, the 17 thermal hydraulic boundary conditions are treated by the sequence number of the particular channel in the APROS model. When comparing the steady-state power histories of the rods during both of the cycles, it is seen that within a cycle the power evolution is quite steady. The exceptions are the gadolinia rods in which the power increases during the first cycle. Due to the simplicity of the histories, average values for power are calculated for each rod for both cycles.

The decay heat during the transient is directly proportional to the initial power prior to the transient, and therefore the transient power histories may be represented by the initial steady-state power values obtained from APROS. However, in the subsequent sensitivity analysis, the power coefficients for the bundles are considered instead of the actual transient power. This choice is made because that way the channel power may be grouped together with the thermal hydraulics boundary conditions, which are also given per channel. The normalized bundle power coefficients are calculated through Equation (2).

The rod failure criterion in FRAPTRAN for ballooning in LOCA is based on empirical stress and strain limits. In the sensitivity analysis, maximum values relative to time and axial position of the cladding plastic hoop strain, hoop stress and outer surface temperature are used instead. This makes the analysis more general and independent of the particular failure criterion adopted in FRAPTRAN. The continuous output is also easier to analyze statistically than the binary failed/non-failed output of the rod failure model.

Methods for visual and quantitative sensitivity analysis

A cobweb graph [21] is used for visualizing and screening the input variables that lead to extreme values of a selected output parameter. In the graph, the values of the input parameters have been scaled to the range 0...1, and the values associated with single simulations are connected with lines. The simulations that exceed a certain criterion are plotted with a different color than the rest of the simulations. In the LOCA analysis, the cladding ballooning is a key result and may be represented by the cladding plastic hoop strain. Therefore, the simulations in which the plastic hoop strain is equal or exceeds a chosen limit, fixed in this analysis to be 20%, are highlighted in the cobweb graph.

The sensitivity indices calculated for the data are the Borgonovo's delta measure [22, 23] and the first order Sobol' sensitivity index [24]. These sensitivity indices can be evaluated from pre-existing data. The first order Sobol' sensitivity index is chosen for its simplicity, and the Borgonovo's delta is used as it is a more comprehensive, moment-independent measure. Also, squared Pearson correlation coefficients, i.e. the R^2 values [25], are calculated for comparison whenever possible. Traditionally in sensitivity analysis of LOCA, various correlation coefficients are applied as sensitivity measures [26, 27]. However, the present work involves data presented as ordinal numbers (the coolant channel number) that cannot be analyzed with correlation coefficients. The Borgonovo's delta is defined for the input variable X_i as [22]:

$$\delta_i = 0.5\, E_{X_i}\left[\int |f_Y(y) - f_{Y|X_i}(y)|dy\right]. \tag{3}$$

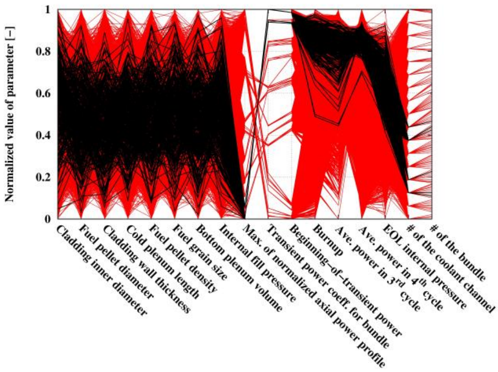
The term in the square brackets is the shift between the probability density function of an output and the probability density function of a conditional output with the value of X_i fixed. E_{X_i} stands for expectation value. The numerical estimation of δ_i is done using a Gaussian kernel estimator with Kolmogorov-Smirnov filtering to reduce spurious correlations [23]. The sensitivity index S_i for an input parameter X_i and output Y is formally defined as the ratio between the variance of the conditional expectation value and unconditional variance as:

$$S_i = var\left[E[Y|X_i]\right] / var[Y]. \tag{4}$$

Usually the calculation of S_i is done with a specific sampling design [20], which is very efficient but in the case of pre-existing data cannot be used. Instead, the numerical estimation of S_i follows the discrete cosine transformation method [28, 29]. With both sensitivity measures S_i and δ_i, the range of the indices is 0-1; zero means that the output is independent of the input parameter in question. The indices are calculated with two MATLAB scripts [30].

RESULTS OF THE SENSITIVITY ANALYSIS

The cobweb graph is shown in Figure 4. Parameter values of all the rods that have been irradiated during cycles 3 and 4 are plotted with red color and the simulations with equal to or higher than 20% plastic hoop strain are plotted with black. Enrichment and gadolinia content are not shown in figure due to confidentiality reasons.



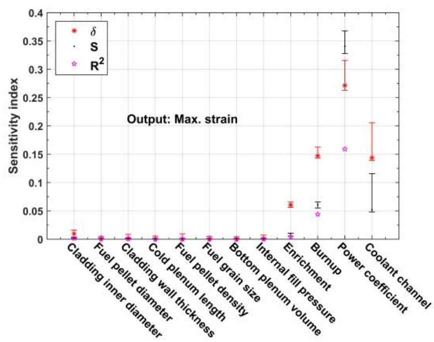


FIGURE 4. Cobweb graph used for screening the most important input parameters (left). Calculated sensitivity indices (right).

The following conclusions are made based on the cobweb graph:

- Tolerances in fuel manufacturing parameters do not significantly affect the transient strain.
- Average steady-state power during the 4th cycle is more correlated to high strains than the 3rd cycle power.
- High burnup is related to high strain, but the range of burnups in the high strain rods is fairly large.
- The range of end-of-life internal pressures resulting in high strains is large, but high strains are not reached in rods with the end-of-life internal pressure in the lower third.
- The distinction between the beginning-of-transient power and the transient power coefficients for the bundles is seen: highest strains are achieved with the highest power coefficients but not with the highest beginning-of-transient powers. Strong correlation with the high hoop strain is seen in both.
- High strains are associated to distinct coolant channels and bundles.

When comparing the cobweb graph with the locations of the bundles and coolant channels in the core (Figure 1), one can see that the rods with the highest permanent hoop strain (> 20%) are located in bundles that are situated in the middle circle of coolant channels in APROS model, namely in channels 3, 4, 7 and 8. Again, these rods are located in six bundles, marked with red squares in Figure 1 (according to FRAPTRAN, a few rods failed also in two more bundles with less than 20% strain, marked with green squares in Figure 1). This is explained by the fact that the powers in the outer circle channels are a lot lower compared to the middle circle and the central channel. In the central channel, the highest powers do not reach the level of those in the middle circle channels, and therefore the hoop strains are not high in that channel.

Next, the parameters to be included to the quantitative sensitivity analysis are chosen. The cobweb graph showed that the rod manufacturing tolerances are quite insignificant but the indices for those are calculated for comparison. Some of the input parameters are screened out due to strong correlation with each other, while some dependent parameters cannot be excluded, e.g. burnup and transient power. These are screened out: gadolinia content, as it is associated with the U-235 enrichment; end-of-life internal pressure, strongly correlated with burnup; average steady-state power in both cycles, correlated with burnup and decay heat; beginning-of-transient power, partly included in the channel number through the average channel power.

The sensitivity indices are presented in Figure 4 for chosen inputs and the output maximum strain. It is seen that the most relevant parameters, in order of importance, are the decay heat power during the transient represented by the bundle power coefficient, the steady-state irradiation history of the rod represented by the rod burnup, and the thermal hydraulic boundary conditions plus the coolant channel power in the rod's location as represented by the coolant channel number. Also fuel enrichment has a non-zero effect. Meanwhile, the tolerances in fuel manufacturing parameters have practically negligible effect, even though the delta indices are not exactly zero. As the inputs are not independent, caution should be used with further interpretation of the indices. Similar study is done for the two other outputs, maximum stress and temperature, and the conclusions are the same [31].

The effect of coolant channel thermal hydraulics is hard to sort out from the effect of coolant channel power, and the effect may only be seen indirectly. Based on high power, at least some of the rods located in channel 5 should reach high strains but that is not observed. Furthermore, the coolant channel power in that channel is higher than in those where the high strains are reached. According to the cobweb graph and the calculated sensitivity indices, the explanation for the lower strains in that channel remains to be searched from among the rod burnup and the coolant channel thermal hydraulics. Similar burnups are observed in channel 5 as in the channels where the strains are high, which excludes the burnup as the sole explaining factor. Therefore, it may be concluded that the thermal hydraulic effects dominate over the effect of the coolant channel power.

CONCLUSION

Using a calculation system for statistical fuel failure analysis, the number of failed fuel rods in LB-LOCA in an EPR was estimated. The number of failing fuel rods in 59 simulated global scenarios ranged from 2 to 14 rods per 1000 simulations. Thus, in the worst case, 1.4% of the simulated rods failed. It can be concluded that according to the statistical analysis performed, the requirement that less than 10% of the rods may fail in a LB-LOCA is met. Also, the highest peak cladding temperatures in all global variations were around 900 ºC, and the requirement concerning the highest peak cladding temperature, 1 200 ºC, is thereby met.

A sensitivity analysis was done in order to discover the most influential input parameters in the worst global scenario. Both visual and numerical methods were applied. A cobweb graph was used both to visually interpret the data and to perform final screening of the input variables. For example, the rod manufacturing parameters were identified as almost non-influential already using the cobweb graph. The burnup, the power coefficient during the transient and the coolant channel number were chosen for further numerical analysis. For comparison, the manufacturing parameters were also chosen to be included in the last stage. The results of the numerical analysis gave support to the conclusions made concerning the importance of the three key parameters.

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