UNCERTAINTY AND SENSITIVITY ANALYSIS OF THE OECD/NEA KALININ-3 BENCHMARK

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ABSTRACT

In the present work are discussed the results of performing uncertainty and sensitivity studies of the VVER-1000 reactor with respect to the uncertainty sources that are coming both from thermal-hydraulic input modeling as well as from the nuclear data. It is based on the analysis of the OECD/NEA coolant transient Benchmark K-3 on measured data at Kalinin-3 Nuclear Power Plant (NPP). The K-3 transient is a switch off of one main coolant pump (MCP) at nominal reactor power. For this purpose the GRS uncertainty and sensitivity software package XSUSA is applied to propagate uncertainties in nuclear data libraries to the full core coupled transient calculations. Nuclear data uncertainties are complemented by a set of most important thermal-hydraulic parameters.

A statistically representative set of coupled ATHLET-PARCS code calculations is analyzed and local output quantities are compared with the measurements available in the benchmark specification. Finally the sensitivity study shows most important contributors to the fuel assembly power in case of the studied asymmetric K-3 transient.

KEYWORDS
ATHLET, XSUSA, uncertainty, sensitivity, coupled code analysis, K-3 Benchmark

1. INTRODUCTION

In the present work the results of the performed uncertainty study of the VVER-1000 reactor with respect to the uncertainty sources that are coming both from thermal-hydraulics as well as from the nuclear data are shown and discussed.

This work is based on the analysis of the OECD/NEA coolant transient Benchmark (K-3). The transient benchmark analyses a switch off of one main coolant pump (MCP) at 98.6% nominal reactor power. The core load is the first one with fuel assemblies (FA) type TVSA, and the transient takes place at 130.6 effective days reactor operation [1]. The amount of available measured data allows a benchmarking of the calculations on both global and local reactor parameters. The scenario of the K-3 Benchmark transient is following: after switching off of the MCP #1 the reactor power is automatically decreased in 71 s to 67.8 % of the nominal value according to the measured data histories. That is done by insertion at first of control rod group (CRG) #10 and later on - CRG #9. At time t= 29 s a reverse flow in the affected loop #1 takes place. That leads to a rapid redistribution of the coolant flow through the reactor pressure vessel resulting in a spatially dependent coolant temperature change. This leads to an asymmetric power distribution that is being recorded by the local in-core temperature and neutron flux (SPND) measuring devices. Thereby, the simulation of the transient requires evaluation of the core response from a multi-dimensional perspective (coupled 3D neutronics/core thermal-hydraulics). After 300 s the reactor parameters stabilize.

In the frame of the OECD Benchmark for Uncertainty Analysis in Best –Estimate Modelling (UAM) for Design, Operation and Safety Analysis of LWRs [2] it is foreseen to continue activities in the K-3 benchmark and to carry-out a new Exercise 4 of uncertainty analysis as a part of Phase-III of the above mentioned UAM benchmark. GRS as a K-3 benchmark team leader took the responsibility to generate varied set of few-group homogenized cross section libraries. For this purpose GRS has used its code package XSUSA ("Cross Section Uncertainty and Sensitivity Analysis") which is successfully applied for a wide variety of calculations for fissionable systems.

The uncertainties of only the thermal-hydraulic model parameters and their influence on the transient results are already investigated in a number of publications [3-4]. However the performed analyses in those publications are in principle incomplete. This is due to the fact that the precision of nuclear data, which are used in the form of few-group homogeneous cross section libraries, is limited by the uncertainties of the underlying measurements and the theoretical parameters. Therefore it is crucial to extend the uncertainty model by including sampling of nuclear data variations. This work has already started in [5-6] where a new set of two group cross section libraries is generated with the help of SCALE package [7] and some stationary values before the beginning of the actual transient are calculated.

The following paper is devoted to the analysis of uncertainties influencing the fuel assembly power and its response to the variation of reactor model input parameters. After describing in Section 2 the simulation model used to study the K-3 benchmark transient, Section 3 presents a discussion of the obtained results.

2. ATHLET-PARCS SIMULATION MODEL

In this study the coupled code system ATHLET-PARCS [8-9] is applied to perform K-3 transient simulations. The coupled code system ATHLET-PARCS requires two separate inputs for thermal-hydraulics and neutronics. For the thermal-hydraulic (TH) part the ATHLET model, the so called ATHLET "Macromodel" which has the smallest possible set of elements (fluid objects) containing six parallel thermal-hydraulic channels (PCH) in the downcomer is applied. Both primary and secondary sides of the Kalinin-3 NPP are modelled as well as the balance-of-plant (BOP) system model is introduced. The latter system is modeled by the ATHLET GCSM module. The primary side model has seven PCH in the active core, seven PCH in the lower and upper plenums. Neighbouring PCHs are connected with cross connections to model the transversal coolant mixing. Thermal properties of the fuel are taken from the UAM benchmark specification [2]. Further details of ATHLET input deck can be found in [8-13].

The neutronic model depicts a full three dimensional core layout of 163 hexagonal fuel assemblies. The core is modeled using the fuel assembly types from the K-3 benchmark specification. The loading scheme is shown in Figure 1. The neutronic model consists of a set of hexagonal cells with a pitch of 23.6 cm (cold system), each corresponding to one fuel assembly (FA). Additionally the model contains reflector which radially surrounds the active part of the reactor and is considered as "reflector assemblies". The radial reflector nodes have the same pitch of 23.6 cm like the fuel assemblies. There are at total 211 assemblies; among them are 48 reflector assemblies. Axially the reactor core is divided into 10 layers with a height (starting from the bottom) of 35.5 cm, adding up to a total active core height of 355 cm. Both upper and lower axial reflectors have a thickness of 23.6 cm.

As a nodal core model PARCS requires homogenized few-group neutron energy cross sections. In this paper the two energy groups which are obtained by the collapsing of 44-group library with NEWT (part of SCALE-6.1 [7]) code are used. The dependence of homogenized cross sections on the nodal burnup is calculated using linear interpolation from the three dimensional depletion calculation of the whole core depletion with the diffusion code BIPR-8 [14]. To be able to perform transient simulation it is necessary to specify the branch conditions to generate the data required for these transient calculations. Branch conditions are performed from the reference uncontrolled state of the fuel assembly nodes with fuel temperature $Tf = 850K$, coolant temperature $Tc = 577.15K$, coolant density $\rho_c = 0.71885\ g/cm^3$ and boron concentration $c_B = 629.6\ ppm$. For the K-3 transient branch conditions are specified by a following set of fuel temperatures, moderator temperatures and moderator densities:

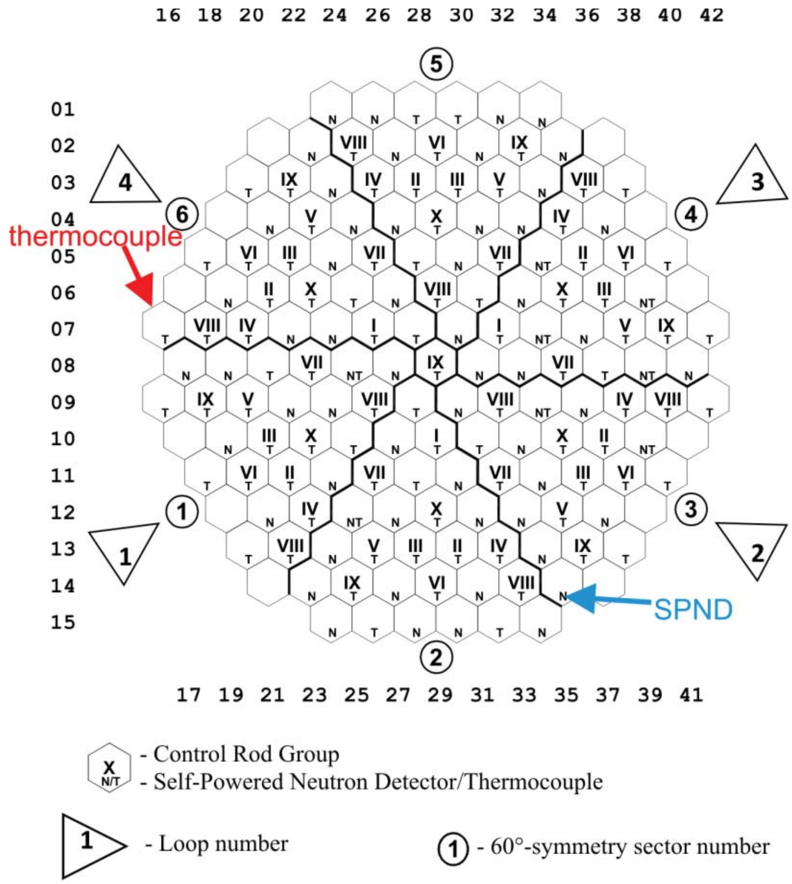

Figure 1. Core configuration and core instrumentation

1) $T_f \in \{540, 900, 1300\}$ in K
2) $T_c \in \{540, 560, 580, 600\}$ in K
3) $\rho_c \in \{660, 700, 740, 790\}$ in kg/m^3

For each above mentioned parameter set points two branches with uncontrolled and controlled states of the fuel assembly nodes are additionally included.

To consider the nuclear data uncertainties a set of $N = 200$ complete two-group cross section libraries is generated. This is achieved by applying the XSUSA algorithms within the spectral code NEWT from SCALE-6.1 [7]. By calculating each parametric point of this library (fuel assembly type, fuel temperature, moderator density, moderator temperature and controlled/uncontrolled state) N times with the same set of random numbers, the correlations are preserved naturally and are implicitly contained in the output data.

In case of uncertain thermal-hydraulic parameters of the most important modelled fluid objects of the NPP are identified and a set of $N = 200$ variations of 23 uncertain parameters using SUSA package [17] is generated. The complete list of 23 thermal-hydraulic parameters is being a priory defined in [6]. Finally using Python scripting language [18] a separate set of both ATHLET and PARCS input files is created. The complete input deck includes ATHLET and PARCS input files together with a two energy cross section library.

3. RESULTS

For the estimation of the output uncertainties there are $N = 200$ calculations in total. According to the Wilks formula already a set of $N = 90$ variations is enough to achieve with a confidence level is 99 % the probability level of 95% that the maximum code output will not be exceeded. However for the purposes of the sensitivity analysis a larger sample set is required. The generation of 2-energy group cross sections is very time consuming, therefore it is chosen to generate only N = 200 set. One calculation on a Windows HPC 2008R2 machine with Intel Xeon 2.5GHz processor takes roughly 8 hours of CPU time. After that all desired transient calculations can be performed for each set configuration of $N = 200$ libraries and evaluated statistically. As an example of the influence of uncertainties onto the local quantities the axial power distribution in two assemblies “06-27” and “02-31” are considered (see Figure 1). Both assemblies have installed self-powered neutron detectors (SPND) which allows a direct comparison of results with the measurements. The assembly “06-27” is placed in the central part of the core, whereas assembly “02-31” is located closer to the periphery of the core. Figure 2 and Figure 3 show axial power profiles in relative units for both assemblies. The points give measurements and solid lines present results of all $N = 200$ simulations. The width of the line gives a standard deviation axial power. The profiles are given at the beginning of the transient (t=0s) and the moment of the flow reversal (t=45s). One can see a good correspondence with measured results. At the same time the uncertainty of axial power is rather small and reaches 4% relative to the mean value at highest. The uncertainty band does not cover measurements points that can be an indication of the possible incompleteness of the model. The increase of the power in the lower part of the core at the moment of flow reversal is caused by the decrease of the coolant temperature due to the switch off of the MCP.

XSUSA gives the possibility to identify the most important input parameters, performing a sensitivity analysis. It is done in terms of group sensitivities, where typically the nuclear multi-group data for a certain reaction of a certain nuclide are treated as one group [19]. Since the coupled analysis is performed, thermohydraulic uncertain parameters also participate in the sensitivity estimation in the form of the group as whole as well as unique contributions from each of the thermohydraulic parameter. The group sensitivity analysis is performed by determining the “squared multiple correlation coefficients (R^2)” as uncertainty importance indicators (sensitivity indicators) for parameter groups. The squared multiple correlation coefficient R^2, which quantifies the uncertainty importance of a group of input variables $(X_1, \dots, X_k)$ with respect to an output variable Y, is defined as the maximum (squared) simple correlation coefficient between the output variable Y and any linear combination of input variables from the group. It can be computed by the formula:

$$R^2 = (\rho(Y, X_1), \dots, \rho(Y, X_k)) \cdot C_X^{-1} \cdot \begin{pmatrix} \rho(Y, X_1) \\ \vdots \\ \rho(Y, X_k) \end{pmatrix}$$

where $\rho(Y, X_i)$ is the correlation coefficient between the output variable Y and the input variable X_i; C_X^{-1} is the inverse of the $(k \times k)$ -correlation matrix of the group of input variables $(X_1, \dots, X_k)$, i.e. inverse of the matrix of correlation coefficients $\rho_{ij} = \rho(X_i, X_j)$ between all the input variables X_i and X_j from this group [20].

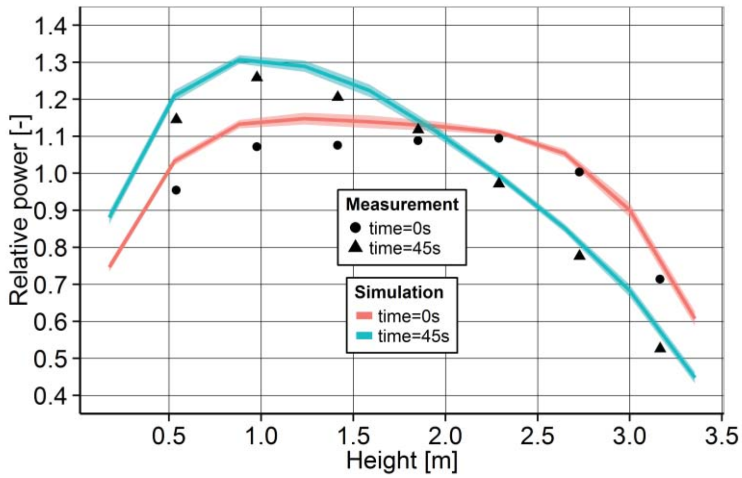

Figure 2. Axial power profile in the assembly 06-27

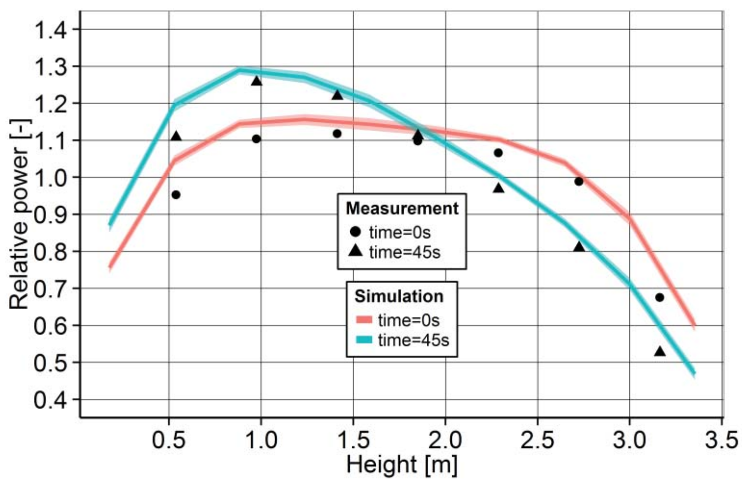

Figure 3. Axial power profile in the assembly 02-31

Sensitivity coefficients are calculated for the same fuel assemblies "06-27" and "02-31". The calculation of R^2 is performed at each time step for two axial positions: at the bottom and at the top of the active zone (heights of 0.176m and 3.353m correspondingly). Figure 4 — Figure 7 show sensitivity coefficients of the axial fuel assembly power distributions as time series during the whole transient. The solid lines show values of R^2as sensitivity measures, whereas ribbons give their 95 % confidence interval. Only values that are lying above the 95% significance level are plotted, which allows to present only statistically relevant results. Abbreviations used in the labeling of input variables resemble identifiers used in SCALE package. They consist of two numbers combined by hyphen. In case of nuclear data uncertain parameters the first number represents the nuclide and the second number defines the reaction type. In the presented study only five different nuclides appear to be important, namely "92235" – ^{235}U, "92238" – ^{238}U, "94238" – ^{238}Pu, "94238" – ^{238}Pu, "94239" – ^{239}Pu and "94241" – ^{241}Pu. The reaction sensitivity types are shown in Table 1.

Table 1. Reaction sensitivity types for nuclear data uncertainties

Identificator	Reaction
0002	Elastic scattering
0004	Inelastic scattering
0018	Fission
0102	n,γ
0452	$\bar{\nu}$

To be able to treat thermohydraulic uncertain parameters on the same footage, in this paper the identifiers are defined in the same manner, where the first number for each thermohydraulic parameter is always "00000" and the second number is the identifier of the thermohydraulic input variable. Additionally there is a special "reaction type" identifier "9999" which represents all thermohydraulic parameters as a whole group. As can be seen from Figure 4 — Figure 7 only two thermohydraulic parameters with identifiers "0002" (MCP volumetric flow rate) and "0016" (Initial reactor power) have major contributions among all other thermohydraulic parameters. It is also clear that all thermohydraulic parameters as an entire group significantly contribute at different phases of the transient.

The results show that for both assemblies after the time t=100 s the axial power profile is mostly dominated by thermohydraulic quantities. It is clearly seen that among all thermohydraulic parameters the MCP volumetric flow rate is most influential for power profile in the studied transient. The time t=100 s is the time where the plant shows an unstable behavior with respect to variation of input variables. Due to the modelling of GCSM signals the plant undergoes some cascading events which are mostly driven by thermohydraulic parameters. The similar behavior was already observed in similar studies performed in VALCO project [21]. On the other hand two assemblies show different sensitivity pictures in terms of axial power profiles in the initial phase of the transient. The bottom part of the central assembly "06-27" is clearly dominated by thermohydraulic uncertainties, where parameters "00000-0002" and "00000-0016"are most influential. But the power at the bottom of assembly "02-31" shows a strong response to the scattering of ^{238}U. The rather small sample size of $N = 200$ runs does not allow identifying clearly other contributions. But one can certainly say that the nuclear properties show its importance especially at the first part of the transient. To increase the significance of the results one needs a larger set of sample data.

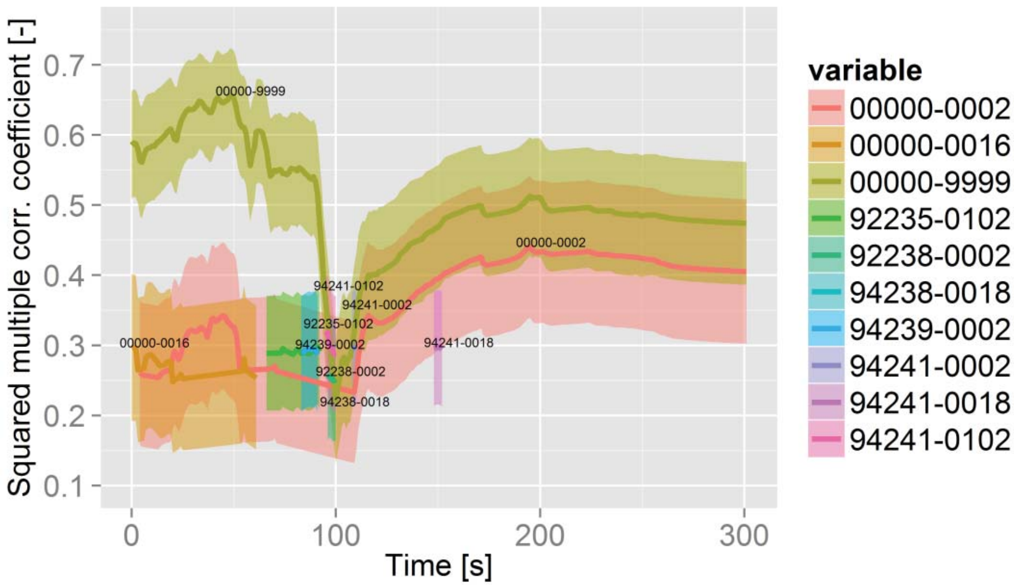

Figure 4. Sensitivity of 06-27 assembly power in first node (height 0.176m)

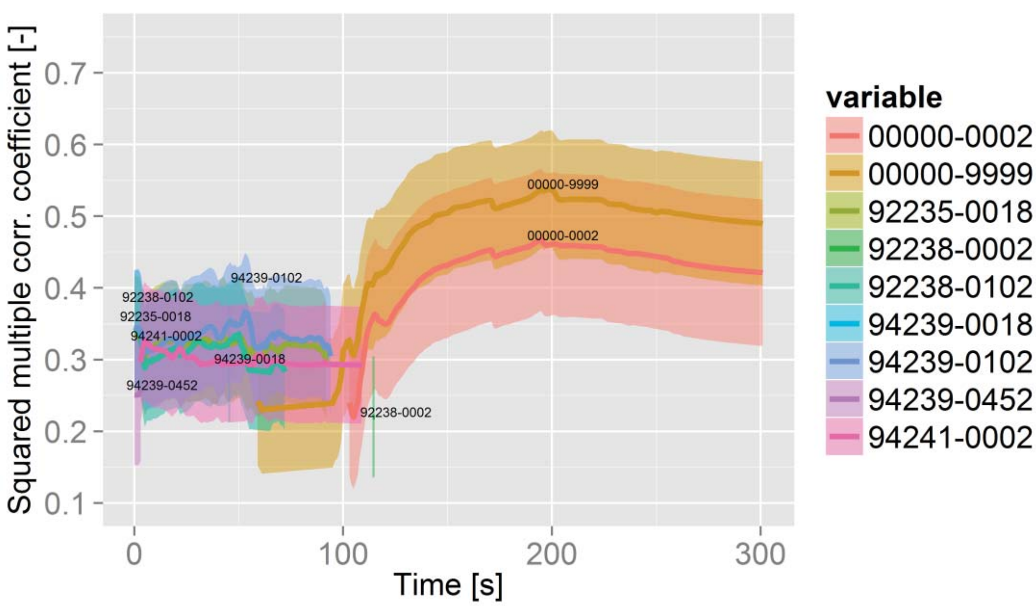

Figure 5. Sensitivity of 06-27 assembly power in last node (height 3.353m)

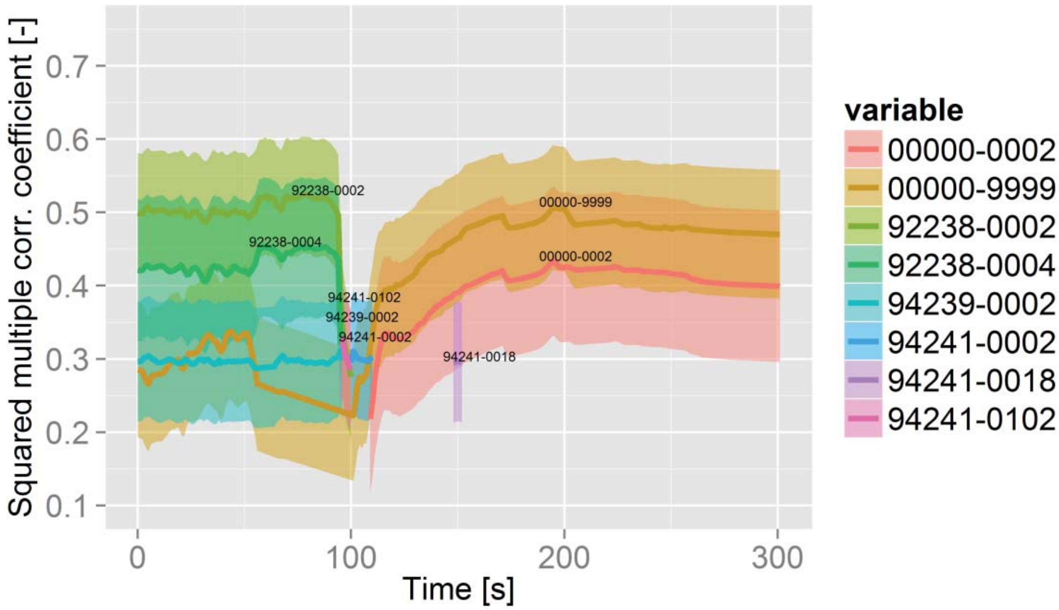

Figure 6. Sensitivity of 02-31 assembly power in first node (height 0.176m)

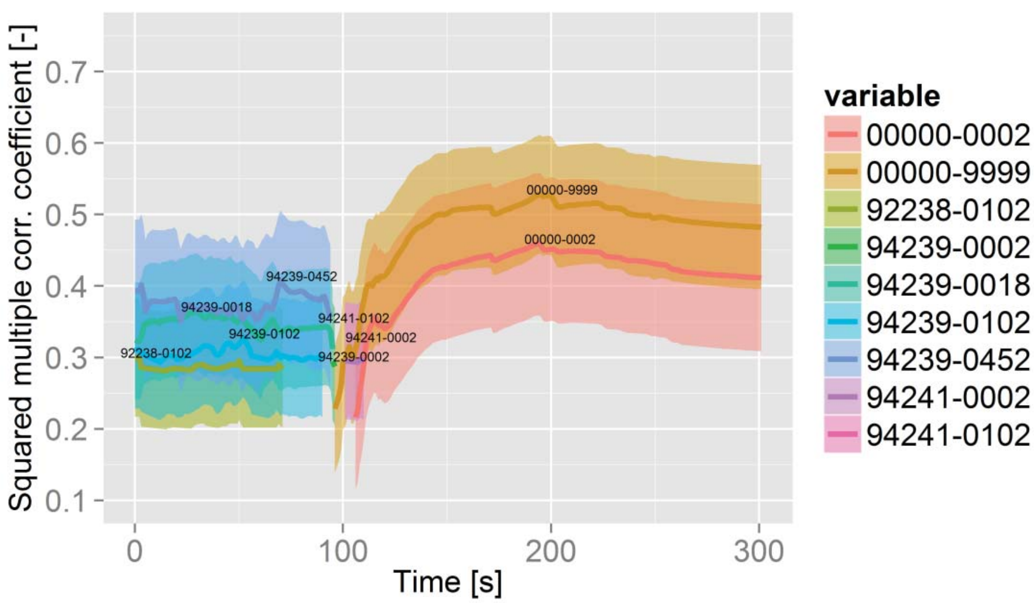

Figure 7. Sensitivity of 02-31 assembly power in last node (height 3.353m)

4. CONCLUSIONS

The paper investigates effects of nuclear data covariance and thermal-hydraulic uncertainties on local reactor parameters. The GRS uncertainty and sensitivity software package XSUSA is applied to propagate uncertainties in nuclear data libraries to the full core coupled calculations of the OECD/NEA Kalinin 3 transient benchmark. The representative set of thermal-hydraulic uncertain parameters is based on the knowledge of the most important fluid objects and explicitly specified after the choice of the ATHLET simulation model. A set of N = 200 coupled ATHLET PARCS code calculations is analyzed. Evaluated local output axial fuel assembly power profiles show large statistically relevant values of sensitivity with respect to the nuclear data uncertainties. Further studies must be concentrated on the analysis of contribution from different isotopes and reactions. This can be achieved by drastically increasing the number of simulations (about 1000).

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