Uncertainty and sensitivity analysis of results of the post test calculation of a LSTF experiment with the code ATHLET

One of the main goals of GRS’s reactor safety research is the development of codes to calculate thermal-hydraulic processes. For the validation of such codes, experiments performed at test facilities are simulated. The comparison of the test results with the results of the calculation allows deductions on the code accuracy. Within the scope of the validation of the thermal-hydraulic code ATHLET, a post test calculation of the experiment SB-PV-09 carried out at the LSTF test facility has been performed. This calculation was complemented by an uncertainty and sensitivity analysis of the obtained results. The main goal was to investigate the influence of a combined variation of the input data upon the simulation of the main phenomena observed experimentally. In this analysis multiple sensitivity coefficients have been derived, allowing a comparison between the influence due to modelling uncertainties and those due to experimentally induced uncertainties.

The Experiment SB-PV-09

ROSA-V/LSTF test facility. The Japanese ROSA-V/LSTF test facility is the volumetrically scaled (1/48) model of a pressurised water reactor with four loops and a thermal power of 3,423 MW. The facility is designed for a full system pressure of 16 MPa. The four reactor loops of the reference plant are combined into 2 double loops. The horizontal legs of the loops are scaled by means of the Froude number in order that two phase flow regime transitions can be reproduced in a reactor-typical manner. The heights are scaled on a 1/1 scale to allow a realistic simulation of natural circulation. The power of the electrically heated core of the LSTF is 10 MW. This allows simulation of the reactor heating with the scaled-down decay heat of the real plant.

Test SB-PV-09 was conducted in November 2005 within the OECD/NEA project »Rig of Safety Assessment« (ROSA).
3.2 Uncertainty and sensitivity analysis of results of the post test calculation of a LSTF experiment with the code ATHLET

**Goals of test SB-PV-09.** The main goals of this test were the analysis of the thermal-hydraulic phenomena in the reactor coolant system in case of a postulated break of a control rod drive mechanism penetration nozzle, the evaluation of the impacts of symptom-oriented accident management measures on the coolability of the reactor core as well as the provision of experimental data for the validation of advanced numerical codes.

For this purpose, a small break at the pressure vessel upper head (Davis-Besse scenario) was simulated, under the assumption of a total failure of the high pressure injection system. A secondary side depressurization was foreseen as an accident management (AM) action. The leak size selected for the test setup corresponds to a 1.9 % break in the cold leg of the reference reactor.

**The test.** The test was initiated by opening the break valve. The break flow rates depended strongly on the water level in the upper head. The coolant in the upper plenum flowed via the control rod guide tubes into the upper head until the water level in the upper plenum sank below the penetration holes in the lower part of the control rod guide tubes. The relatively large break area led to a quick pressure drop in the reactor coolant system. Primary pressure dropped below secondary pressure at approx. 800 s, simultaneously with the beginning of core uncovering.

With the temperatures increasing in the upper core region, the limit (623 K) for the activation of the planned accident management measure was reached at approx. 1,090 s after opening of the break. The secondary side depressurization was initiated by manually opening the relief valves. However, this measure was not effective since the primary pressure was lower than the secondary pressure in this phase. The cladding tube temperatures kept increasing until the limit value for the response of the LSTF core protection system was reached, which caused an automatic reduction of the core power by approx. 75 % at t = 1,200 s.
3.2 Uncertainty and sensitivity analysis of results of the post test calculation of a LSTF experiment with the code ATHLET

After approx. 1,300 s, the actuation pressure of the accumulator was reached. The subsequent steam condensation in the cold legs led to the clearance of the pump suction lines which initiated the refilling of the reactor core. Approx. 1,400 s after break, the major part of the core was reflooded. In the further course of the transient, primary pressure dropped again until it reached the activation pressure of the low-pressure injection system at 2,900 s. The test was finished with the closing of the break valve at t = 3,265 s.

Reference calculation

ATHLET input dataset. The basis for the modelling of the LSTF facility was the ATHLET input dataset which was used for the post test analysis of test SB-CL-18 in the frame of the international standard problem ISP 26 (see Fig. 23–24 »NODALISATIONS«). The main modifications which were carried out in the test facility for the current phase of the ROSA test programme were taken into account.

In addition to the actual initial and boundary conditions of test SB-PV-09, the original dataset was changed in the following areas:

- Simulation of the break unit in the upper part of the upper head,
- Application of the swell level model in the upper head,
- Modelling of the new pressurizer and the surge line,
- Revision of the input regarding the accumulator connection lines,
- Simulation of the nitrogen injection after the emtping of the accumulator.

Overall, ATHLET could simulate satisfactorily the main phenomena observed experimentally. The influence of the water level in the upper head on the break mass flow was reproduced correctly by the code. Both the calculated time point of initiating the accident management measure and the beginning of the accumulator injection are in good agreement with the experimental values.

Simulation results. The multi-dimensional flow processes in the upper part of the reactor pressure vessel (RPV) which were observed in the test could be satisfactorily simulated by modelling parallel channels in the core and in the upper plenum (including the control rod guide tubes). An exception is the late draining of the upper plenum after uncovering the openings in the lower area of the control rod guide tubes which was calculated by the code. This led to a late beginning of core uncover in comparison with the test.

The deviation in core uncover may be caused by the modelling of the core bypasses. Parameter
studies conducted along with these calculations show that the value assumed for the form pressure loss of the spray nozzles between the downcomer and the upper head can have a great impact on the calculated moment at the beginning of core uncover. In addition, the calculated time point of the beginning of core uncover could only be achieved by way of assuming an additional bypass path between the upper part of the downcomer and the upper plenum. This indicates that either the bypass flow through the spray nozzles was considerably higher than the specified value or that there was an unforeseen leakage between downcomer and upper plenum.

Generally, ATHLET could reproduce the experimentally determined coolant distribution in the coolant loops well and in the RPV satisfactorily, which confirms the applicability of the models for determination of the interphase friction for different geometries and flow conditions.

3.2 Uncertainty and sensitivity analysis of results of the post test calculation of a LSTF experiment with the code ATHLET

Method to determine the uncertainty of numerical code results. The calculation of test SB-PV-09 was supplemented by an uncertainty and sensitivity analysis to investigate the influence of a combined variation of the uncertain input parameters on the calculated results. For this analysis, the method introduced by GRS to determine the uncertainty of numerical code results was used. This method is based on the simultaneous variation of uncertain input parameters of the calculation model together with a statistical evaluation of the code results. All potentially important input uncertainties can be considered in the analysis. The number of calculations to be performed does not depend on the number of parameters but on the desired probability level and the confidence level of the statistical tolerance limits which are used in the uncertainty statement of the results.

In addition, this method also allows the determination of sensitivity coefficients to quantify the influence of the individual uncertain parameters on the scatter range of the calculation results. From this, a ranking order of the input uncertainties according to their respective relative contribution to the uncertainty of the result ensues. Further developments within the scope of the GRS method also allow for the determination of multiple sensitivity indices which quantify the influence of the uncertainty of an entire parameter group on the uncertainty of the calculation result. With the aid of these multiple sensitivity indices, it is possible to estimate which portion of the uncertainty of results stem from the uncertainties in the experimental setup and which portion stems from modelling uncertainties.

Identification of the uncertain parameters and the probabilistic quantification of their uncertainty. The identification of the uncertain parameters and the probabilistic quantification of their uncertainty are essential for the uncertainty and sensitivity analyses. For the present study, a total of 50 potentially important uncertain parameters were identified and quantified. Among them, there are 40 parameters which describe the uncertainties of the physical modelling and the numerical simulation as well as another 10 parameters which refer to the uncertainties of the test facility and the experiment conducted.
The model uncertainties include:

- 4 parameters for the determination of the critical discharge flow,
- 20 parameters to describe the uncertainties in the momentum equations,
- 9 parameters for the heat transfer from fluid to structures,
- 4 parameters for the two-phase heat and mass exchange through evaporation and condensation,
- 1 parameter for the axial nodalization in the bundle area,
- 2 parameters to describe the form pressure losses in the facility.

The ten test-specific uncertain parameters consider the uncertainties regarding the core bypasses, the bundle power, and leakage through the venting pipe at the RPV upper head as well as the accuracy of the temperature measurements which are used for initiating the accident management measure and triggering the core protection system.

Generation of ATHLET datasets. On the basis of the identified and quantified uncertain parameters and by means of the code system SUSA, 208 ATHLET datasets with the combined variation of the input parameters were generated. The values were determined by simple random selection from the determined distributions (Monte Carlo simulation). Comparing to the usually applied minimum number of $N_{\text{min}} = 93$ calculations required for the 95%/95% tolerance limits, the limits calculated on the basis of 208 calculations are markedly less conservative. Furthermore, the accuracy of the derived sensitivity coefficients increases that way.

Uncertainty and sensitivity analysis results. Overall, eight scalar individual values and 15 time-dependent output quantities were chosen for the uncertainty and sensitivity analysis. Significant findings gained from this analysis will be exemplarily represented here by the water level in the active core region.

The double-sided (95%/95%) tolerance limits of the calculated water level in the core region, which were determined on the basis of the variation of all 50 parameters, are shown in Fig. 25 »TOLERANCE LIMITS«. During the core uncovering phase (between 800 and 1,300 s), the corresponding measured values are close to the lower tolerance limit of the calculation results. An additional analysis based only on the variation of the 10 experimental uncertain parameters shows that these parameters alone have less influence on the uncertainty range of the calculation results (see Fig. 26 »TOLERANCE LIMITS«).

The time history of the water level in the bundle can be divided into three different phases:

1. an initial phase of approx. 800 s with steam generation in the core and the formation of a swell level which is high enough to ensure the cooling of the heating rods;
2. the phase of core uncovering and heating of the fuel rods, and
3. the phase of core reflooding with the beginning of the accumulator injection at approx. 1,300 s.

In the initial phase, parameter 17 (calculation of the interphase friction in bundle geometries) and parameter 40 (number of the axial nodes in the heated core area) contribute the most to the uncertainty of the calculated water level (see Fig. 27 »SENSITIVITY MEASURES«). They are significant for the calculation of the void profile in the core region.
However, during the phase essential for the entire test procedure, i.e. the core uncovery and heating phase, parameter 41 (leakage between downcomer and upper plenum) is the most influential parameter: The bigger its value, the lower the water level in the core.

In the re-filling phase, parameter 3 (contraction factor for the steam discharge flow out of the leak) and parameter 41 (now with a positive sign) have the greatest impact on the calculation results. Increasing values of these parameters tend to cause a lower primary pressure and thus a fast re-filling of the core.

The multiple sensitivity measures show (see Fig. 28 »SENSITIVITY MEASURES«), that the model uncertainties (group 1) are the main contributor to the uncertainties of the calculated water level in the core. Nevertheless, the experimental uncertainties (group 2), particularly the leakage between downcomer and upper plenum, are significant in the important phase of core uncovery.

**Summary**

The results of the uncertainty and sensitivity analysis endorse the code capability to reproduce adequately the main experimental outcomes. The corresponding measured values lay mostly within the tolerance limits of the calculated results.

The main contributions to the global uncertainty of calculated results are due to the modelling un-
3.2 Uncertainty and sensitivity analysis of results of the post test calculation of a LSTF experiment with the code ATHLET

certainties, especially to the modelling of the critical discharge flow rates. On the other hand, it has been shown that the experimental uncertainties, particularly the unknown bypass flow between the downcomer and the upper plenum, are crucial for the determination of the coolant distribution within the reactor vessel and consequently for the correct simulation of the important phase of core depletion and heat-up of the fuel rods.

The application of methods of uncertainty and sensitivity analysis to validation calculations is a valuable complement to the assessment and validation process of a computer code.
**NODALISATIONS**

Fig. 23 (from 23-24)
ATHLET model of the LSTF facility

23 ATHLET
Nodalisation of the LSTF pressure vessel

---

**Fig. 23** Master file

LSTF Vessel Configuration for ATHLET
NODALISATIONS
Fig. 24 (from 23-24)
ATHLET model of the LSTF facility

24 ATHLET
Nodalisation of the primary and secondary system

Fig. 24 Master file

LSTF Loop Configuration for ATHLET
ANALYSIS RESULTS

Fig. 25
Double-sided tolerance limits of the calculated water level in the core with consideration of all 50 uncertain parameters

25 TOLERANCE LIMITS
All 50 uncertain parameters

Fig. 25
Master file

UaSA: ROSA/LSTF Versuch SB-PV-09
two-sided tolerance limits for consequence no. 6
sample size = 208, $\beta = 0.95$, $\gamma = 0.95$
Fig. 26
Double-sided tolerance limits of the calculated water level in the core with consideration of only the 10 experimental uncertain parameters.

**ANALYSIS RESULTS**

**TOLERANCE LIMITS**
Only experimental uncertain parameters

---

**Fig. 26**
Master file

UaSA: ROSA/LSTF Versuch SB-PV-09 (Exp. Uncert.)
two-sided tolerance limits for consequence no. 6
sample size = 208, β = 0.95, γ = 0.95

- two-sided lower limit
- two-sided upper limit
- reference
- experiment

Wasserstand im Kern [m]
Zeit [s]
ANALYSIS RESULTS

Fig. 27
Sensitivity measures of all uncertain parameters for the calculated water level in the core.

27 SENSITIVITY MEASURES
All 50 uncertain parameters

Fig. 27  Master file

UaSA: ROSA/LSTF Versuch SB-PV-09
Wasserstand im Kern [m]
consequence no. 6, sample size= 208
Fig. 28
Multiple sensitivity measures of Group (1) of the modelled parameters and of Group (2) of the experimental parameters for the calculated water level in the core.

28 SENSITIVITY MEASURES
Groups of modelled and of experimental parameters

UaSA: ROSA/LSTF Versuch SB-PV-09
Wasserstand im Kern [m]
consequence no. 6 , sample size = 208