GERMAN (GRS) APPROCH TO ACCIDENT ANALYSIS (PART III)

Status of Uncertainty Evaluations of Thermal-Hydraulic Code Results in Germany

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CONTENT OF THE PRESENTATION

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1. INTRODUCTION

There is an increasing interest in computational reactor safety analysis to **replace the conservative evaluation model calculations by best estimate calculations supplemented by a quantitative uncertainty analysis.**

- Sources of uncertainties - code models, initial and boundary conditions, plant state, fuel parameters, scaling, and numerical solution algorithm.

- Measurements which are the basis of computer code model show a scatter around a mean value. For example, data for two-phase pressure drop show a scatter range of about ± 20 – 30%.

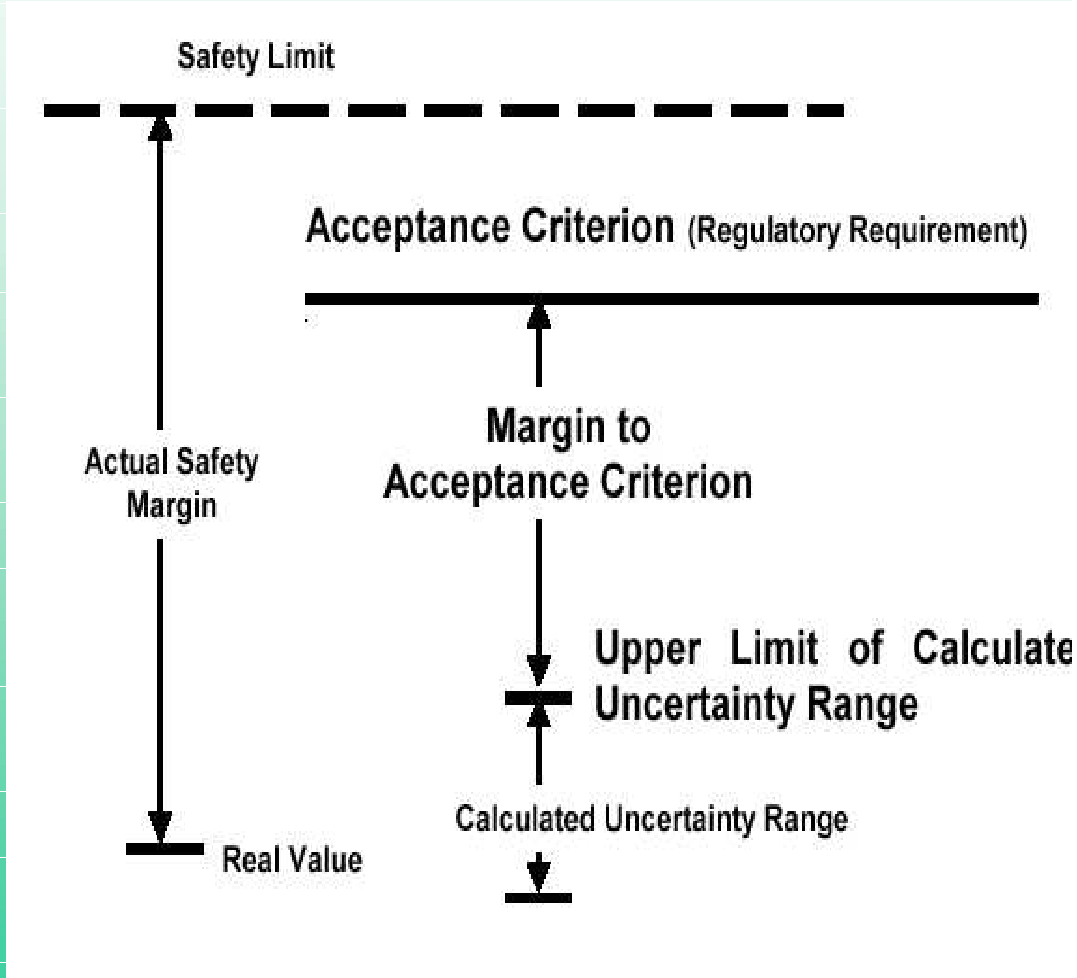
- A **range** of values should be taken into account for the respective model parameter **instead of one discrete value only**. **The state of knowledge** about all uncertain parameters is described by **ranges** and **subjective probability distributions**.

The term "**subjective**" is used here to distinguish uncertainty due to imprecise knowledge from uncertainty due to stochastic or random variability. Stochastic variability may be a time dependent ensemble of records for a given state that can be used to estimate the variability of measurements so that uncertainty of measurement readings can be derived. Stochastic variability may also be possible system failures.

- Stochastic variability due to possible component failures of the reactor plant is not considered in an uncertainty analysis. The single failure criterion is taken into account in a deterministic way.

- The **aim** of the uncertainty analysis is at first **to identify and quantify all potentially important uncertain parameters**

- Their propagation through computer code calculations provides subjective probability distributions (and ranges) for the code results. The evaluation of the margin to acceptance criteria, (= technical limit value) e.g. **the maximum fuel rod clad temperature, should be based on the upper limit of this distribution for the calculated temperatures**



Sources of information

- OECD/CSNI activities for the validation of best-estimate thermalhydraulic codes and the development of uncertainty and sensitivity analysis (SA).

- 'Sensitivity Analysis',
  by A. Saltelli, K. Chan, E.M. Scott, Wiley (2000)

  A book that presents basic ideas and a very good review of methods and work performed, not only in the nuclear field. It makes references to special issues of journals.

  - JSCS (1997)Journal of Statistical Computation Simulation 57 (1-4)

  - RESS (1997)Reliability Engineering System Safety 57, 1

  -SAMO (1998)Sensitivity Analysis of Model Output
  EUR Report 17758 EN, Luxembourg

  - CPC (1999)Computer Physics Communications, 117 (1-2)

- The NRC programme of code consolidation includes the development of an **Automatic Code Assessment Package (ACAP).**

Classification of sensitivity analysis methods

- Screening methods
  Dealing with models that are computational expensive to evaluate and have a large number of input parameters screening experiments can be used to identify the parameter subset that controls most of the output variability.
  They do not quantify the effects.

- Local methods
  Usually carried out by computing partial derivatives of the output functions.

- Global methods
  The input factors are described by probability distribution functions that cover the range of values. The sensitivity of output values is estimated by varying all input factors.
  These methods are quantitative methods (GRS method is such a method)

Example of a Sensitivity Analysis

The simulation by the computer code represents an input/output relation.

$y(t) = S(x,t)$

$y(t)$ output vector

$x(t)$ input vector or input factors consisting of

$x_e(t)$ external boundary conditions

$x_i(t)$ internal parameter of the physical model

Each factor is defined by a range of values or states and their probability distribution.

The design of the experiment is determined by a statistical sampling method which determines a set of values for input factors

$(x_{1,i}\ ,\ x_{2,I}\ ,\ldots,\ x_{K,i}),\quad i = 1,N.$

The simulation run is performed for each set of values, i.e. N runs of the code are performed. The result consists of N output vectors

$(y_{1,i}\ ,\ y_{2i}\ ,\ \ldots,\ y_{r,i})\quad i = 1,N$

These output vectors represent a sample of the system response. This sample is evaluated by statistical methods to determine sensitivity of output factors to input factors.

Milestones of OECD/CSNI activity

- Pioneering work promoted by USNRC led to the proposal of CSAU-method (**Code Scaling, Applicability and Uncertainty**), published 1990. This initiated CSNI work:
- 1989 CSNI-Status Report on LOCACSNI-Report 161
- 1994 Review study on uncertainty methods for thermal-hydraulic computer codes EUR 15364 EN, Luxembourg, 1994
- 1994CSNI-Workshop on Uncertainty Analysis MethodsCSNI, R (94) 20, 2 Vol. It was recommendet to perform an Uncertainty Method Study (UMS)
- **1998 Report on the Uncertainty Methods Study, T. Wickett (Ed.) CSNI, R (97) 35, Vol 1-2, June 1998**
- 1998 OECD Seminar on Best-Estimate Methods on Thermalhydraulics, Ankara, 29 June – 1 July, 1998
- 2000 Int. Meeting on Best-Estimate Methods in Nuclear Installation Safety Analysis, BE-2000, Washington, Nov. 2000

Participant	Code Version Used	Method Name and Type
AEA Technology, UK	RELAP5/MOD3.2	AEAT Method. Phenomena uncertainties selected, quantified by ranges and combined.
University of Pisa	RELAP5/MOD2 cycle 36.04, IBM version CATHARE 2 version 1.3U rev 5	Uncertainty Method based on Accuracy Extrapolation (UMAE). Accuracy in calculating similar integral tests is extrapolated to plant.
Gesellschaft für Anlagen- und Reaktorsicherheit (GRS), Germany	ATHLET Mod 1.1 Cycle A	GRS Method. Phenomena uncertainties quantified by ranges and subjective probability distributions (SPDs) and combined. SUSA
Institut de Protection et de Sûreté Nucléaire (IPSN), France	CATHARE 2 version 1.3U rev 5	IPSN Method. Phenomena uncertainties quantified by ranges and SPDs and combined. SUNSET
Empresa Nacional del Uranio, SA (ENUSA), Spain	RELAP5/MOD 3.2	ENUSA Method. Phenomena uncertainties quantified by ranges and SPDs and combined. Statistical package MAYDAY

Summary of Methods

	Method	**Utility Class**
1	D'Auria FFT (DFFT)	Data Comparison
2	Mean Error (ME)	Data Comparison
3	Variance of Error (VE)	Data Comparison
4	Mean Square Error (MSE)	Data Comparison
5	Mean Error Magnitude (MEM)	Data Comparison
6	Index of Agreement (JA)	Data Comparison
7	Systematic Mean Square Error (SMSE)	Data Comparison
8	Unsystematic Mean Square Error	Data Comparison
9	Mean Fractional Error (MFE)	Data Comparison
10	Cross-Correlation Coefficient (ρ_{xy})	Data Comparison
11	Standard Linear Regression ($L_{2\text{-standard}}$)	Data Comparison
12	Origin Constrained Linear Regression	Data Comparison
13	Perfect Agreement Norm ($L_{e\text{-perfect}}$)	Data Comparison
14	Continuous Wavelet Transform (CWT)	Data Comparison
15	Percent Validated (PV)	Data Comparison
16	Resampling	Data Conditioning
17	Trend Removal	Data Conditioning
18	Time-Windowing	Data Conditioning

ACAP Methods

Two methods for the quantification of uncertainties are available in Germany at present:

- The designer's method by Framatome ANP (former Siemens/ KWU). It follows essentially th **Code Scaling, Applicability, and Uncertainty Evaluation Methodology (CSAU**), but differs in the application of some steps (Depisch, 1998). The CSAU methodology (Boyack, 1990, Young, 1998) has been proposed by the US Nuclear Regulatory Commission (NRC).

- **GRS method**. It has been developed for application of future confirmatory analyses conducte as part of the safety assessment by expert organisations (Glaeser, 2000).

2. Framatome ANP (former Siemens/ KWU) Method

- The **designer's methodology** was developed by U.S. Siemens Power Corporation (SPC) in 1993 and has been submitted to the US NRC for approval. It follows essentially the CSAU approach, but differs in the application of some steps. It was recently (1996/97) applied by Framatome ANP to a 4 loop PWR 1300 of KWU type. The approach consists of 3 steps.

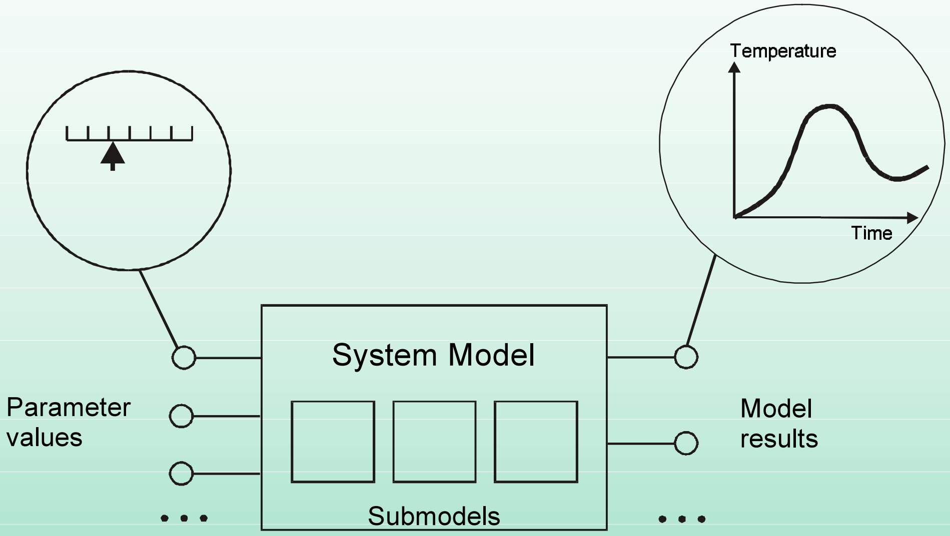
- In the first step the **P**henomena **I**dentification and **R**anking **T**able (PIRT) is established for each selected event sequence, e.g. for LBLOCA of a NPP.

- Within the second step the specific assessment of the applied code related to the selected event sequence is demonstrated. A cross reference matrix contains the experiments (separate effects tests and integral tests) used for the assessment of the phenomena of the PIRT. The validation results are documented also as part of the second step. The plant nodalisation for the analyses of the selected event sequence is established in consistence with the analyses of the experiments.

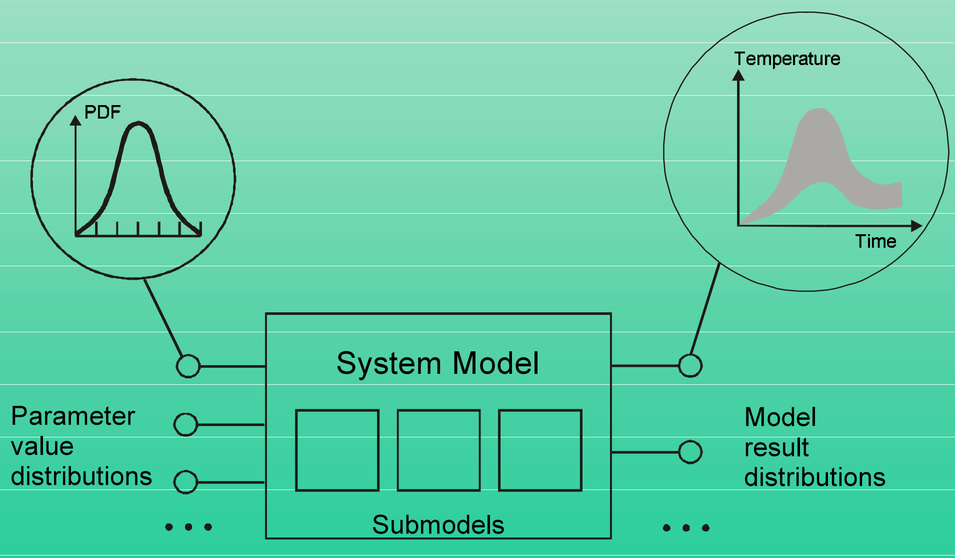
- A sensitivity analysis is performed within the third step.

3. Description of the GRS Method

- GRS (Hofer 1993) has developed methods for the determination of uncertainties. The state of knowledge about all uncertain parameters is described by ranges and subjective probability distributions.



- In order to get information about the uncertainty of computer code results, a number of code runs have to be performed. For each of these calculation runs, all identified uncertain parameters are varied simultaneously.



- Uncertain parameters are uncertain input values, models, initial and boundary conditions, numerical values like convergence criteria and maximum time step size, etc. **Model uncertainties are expressed by adding on or multiplying correlations by a corrective term, or by a set of alternative model formulations**.

- Code validation results are a fundamental basis to quantify parameter uncertainties.

- The selection of parameter values according to their specified subjective probability distributions, their combination and the evaluation of the calculation results requires a method.

- Following a proposal by GRS the central part of the method is a set of statistical techniques. The advantage of using these techniques is that the **number of code calculations needed is independent of the number of uncertain parameters.**

- In each code calculation, **all uncertain parameters are varied simultaneously**. In order to quantify the effect of these variations on the result, statistical tools are used. Because **the number of calculations is independent of the number of uncertain parameters**, no a priori ranking of input parameters is necessary to reduce their number in order to cut computation cost.

- The number of code calculations depends on the requested probability content and confidence level of the statistical tolerance limits used in the uncertainty statements of the results. The required minimum number n of these calculation runs is given by the **Wilks' formula (**Wilks, 1941/ 1942).

	One-sided statistical tolerance limits				Two-sided statistical tolerance limits		
β/α	0.90	0.95	0.99		0.90	0.95	0.99
0.90	22	45	230		38	77	388
0.95	29	59	299		46	93	473
0.99	44	90	459		64	130	662

Minimum number of calculations n for one-sided and two-sided statistical tolerance limits

- Another important feature of the method is that one can evaluate sensitivity measures of the importance of parameter uncertainties for the uncertainties of the results. These measures give a **ranking** of input parameters. This information provides guidance as to **where to improve the state of knowledge in order to reduce the output uncertainties most effectively, or where to improve the modelling of the computer code**

- The different steps of the uncertainty analysis are supported by the software system **SUSA** (Software System for Uncertainty and Sensitivity Analyses) developed by GRS (Krzykacz 1994). They provide a choice of statistical tools to be applied during the uncertainty and sensitivity analysis

How to apply the GRS method?

Standard application:

- Identify relevant parameters including state of knowledge (parameter range, subjective probability density function).
- Determine by Simple Random Sampling (SRS) sets of parameter values. The minimum number of sets is defined by Wilk's formulae.
- Run the code for each set of parameters
- Collect results and make evaluations by SUSA.

Which variations are reasonable?

- The Wilk's formuale define the minimum number of runs which are needed to allow the correct determination of (a, b)-tolerance limits.
- A few runs (about 20) could be used to determine relevant sensitivities, though the sensitivity measures may show great scattering.
- The range of values is more important for SA than the distribution.

What are typical correlation coefficients and sensitivity measures?

The SUSA tutorial describes that following correlation coefficients are calculated:

- Pearson’s Rho
- Quadrant Measure
- Kendall’s Tau
- Spearman’s Rho
- Correlation of Bivariate Standard Normal

In addition, SUSA offers 4 groups of correlation related sensitivity measures.

- Pearson’s Product Moment
- Blomquist’s Medial
- Kendall’s Rank
- Spearman’s Rank

Within each group are calculated

- The ordinary and partial correlation
- Standard partial regression coefficient
- Coefficient of multiple determination (R2)

and in addition, sensitivity measures from

- Correlation ratios
- 2x2 contingency tables
- step wise rank regression

4. Application of GRS method to experiment LSTF-SB-CL-18

- **Test description:**

The test simulates a **small break loss of coolant accident (5%)** performed on the **Japanese LSTF facility** which is a 1/48 volumetrically scaled model of a Westinghouse-type 3423 MWth four loops PWR (Kumamaru 1989). The main components of LSTF have the same elevations as the reference PWR to simulate the natural circulation phenomena and large loop pipes to simulate the two-phase flow regimes and phenomena of significance in an actual plant. The **four primary loops of the reference PWR are represented by two loops** of equal volume (inlet diameter is 0.207 m).Both the initial steady state conditions and the test procedures were designed to minimize the effects of LSTF scaling compromises on the transient during the test.

Main operational conditions are:

- break opening at time zero,
- loss of offsite power at scram,
- high pressure safety injection not actuated,
- main feedwater termination at reactor scram,
- auxiliary feedwater not actuated,
- accumulator injection at 4.51 MPa,
- lower pressure injection at 1.29 MPa.

Main physical phenomena observed during this test:

Two uncoveries of the heater rod bundle representing the core. The first one due to water level depression (120-155 s) before the loop seal cleared (140 s), and the second one (420-540 s) due to loss of water inventory at the break which was finished by the accumulator injection (455 s). The whole transient lasted 900 s.

No	Parameter	Ranges		Reference	Distribution	Explanation
		min	max			
Critical break flow						
1	DSCON	0.5	3	1.3	Polygonal	Correction factor contraction length
2	FD	0.02	0.22	0.02	Polygonal	Weisbach-Darcy-wall friction coefficient
3	FF	0.7	1	0.775	Polygonal	Contraction coefficient for steam flow
4	PP	0.98	0.999	0.98	Polygonal	Transition of void fraction for contraction coefficient
Evaporation						
5	ZBO	10^8	10^{10}	$5x10^9$	Polygonal	Number of bubbles per unit volume (m^{-3})
6	ZT	10^8	10^{10}	$5x10^9$	Polygonal	Number of droplets per volume (m^{-3})
7	OMTCON	0.5	2	1	Uniform	Direct condensation
8	TURB	1	50	20	Log-normal	Turbulence factor for evaporation in critical break flow model
Drift models						
9	ODVRO	0.5	1.5	1	Polygonal	Correction factor for vertical pipe (Flooding-Based-Drift-Flux Model)
10	ODBUN	0.3	1.5	1	Normal	Correction factor for vertical bundle
11	ODVKU	0.7	1.3	1	Normal	Correction factor for vertical annulus
12	ODHPI	0.75	2.25	1	Polygonal	Correction factor for horizontal pipe
13	ODHBR	0.5	2	1	Uniform	Correction factor for horizontal core channel connections
14	ODENT	1	3	1	Uniform	Correction factor for water entrainment in vertical bundle
Two-phase pressure drop						
15	ITMPO	1 or 4				Correlation selection (parameter 16 and 17)
16	OFI2H	1			Log-normal Log-normal	Martinelli-Nelson correl. with constant friction factor, horizontal (ITMPO=1) Chisholm correlation with calculated friction using wall roughness, horizontal (ITMPO = 4)
17	OFI2	1			Log-normal Log-normal	Martinelli-Nelson correl. with constant friction factor, vertical (ITMPO=1) Chisholm-correlation with calculated friction using wall roughness, vertical (ITMPO = 4)

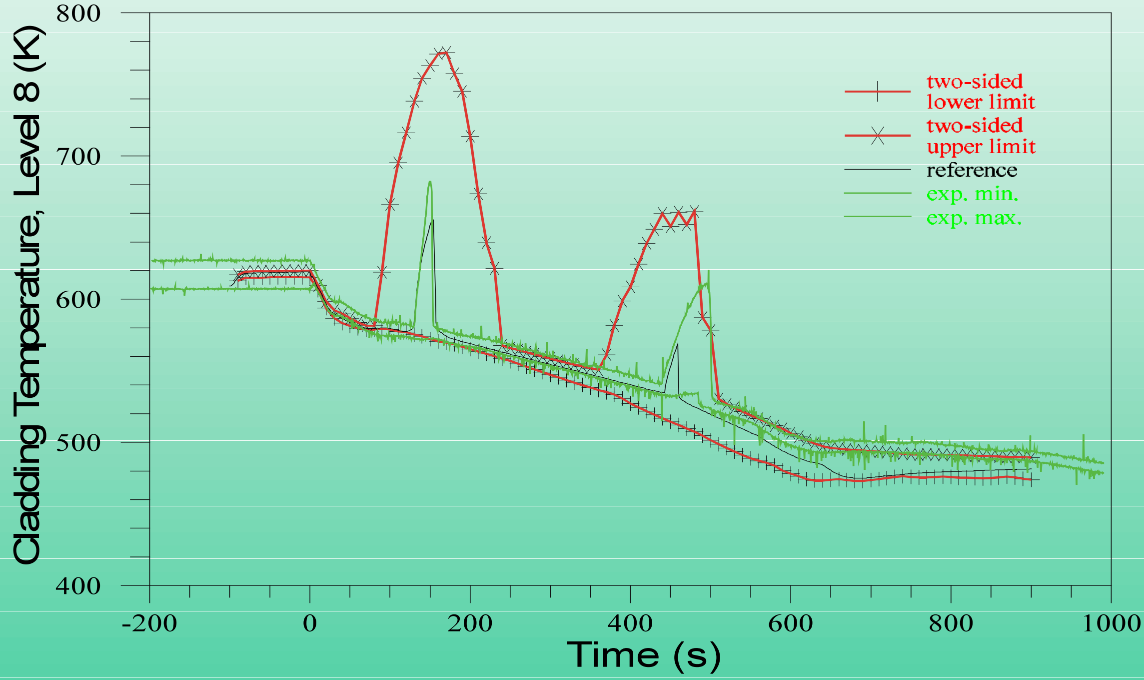
List of uncertain input parameters for LSTF calculations

Pressure drop, wall friction						
18	ALAMO	0.01	0.03	0.02	Triangular	Pipe wall friction (option ITMPO = 1
19	ALAMO	0.01	0.03	0.02	Triangular	Rod bundle wall friction (option ITMPO = 1)
20	ROUO	10^{-5}	10^{-4}		Polygonal	Pipe wall roughness (option ITMPO = 4)
21	ROUO	1.5 x10^{-6}	2x10^{-5}		Polygonal	Rod bundle wall roughness (option ITMPO = 4)
Main coolant pump						
22	YHS	Table	Table	Table	Uniform	Two-phase multiplier for head and torque
Bypass flow paths						
23	CSA	0.01	0.6	0.47	Uniform	Correction factor for bypass flow cross section between upper downcomer and upper plenum
24	CSA	0.2	1	0.62	Uniform	Correction factor for bypass flow cross section between upper downcomer and upper head
25	ZFFJ0/ZFBJ0	0.4	2.5	1	Uniform	Correction factor for bypass form loss between rod bundle and upper head
26	ZFFJ0/ZFBJ0	0.33	3	1	Uniform	Correction factor for bypass form loss between upper plenum and upper head
Pressure drop, momentum term						
27	JDPA				0.25	Momentum term hot leg/ upper plenum from HL only (25%)
	JDPA				0.25	Momentum term hot leg/ upper plenum not computed (25%)
	JDPA				0.5	Momentum term hot leg/ upper plenum in both directions (50%)
28	JDPA				0.25	Momentum term cold leg/ downcomer from CL only (25%)
	JDPA				0.25	Momentum term cold leg/ downcomer not computed (25%)
	JDPA				0.5	Momentum term cold leg/ downcomer in both directions (50 %)
29	JDPA				0.5	Momentum term at heater rod bundle inlet not computed (50 %)
	JDPA				0.5	Momentum term at heater rod bundle inlet in both directions (50 %)
Pressure drop, form losses						
30	ZFFJ0/ZFBJ0	0.667	1.5	1	Uniform	Correction factor for form loss at branch
31	ZFFJ0/ZFBJ0	0.5	2	1	Uniform	Correction factor for form loss at upper bundle plate and spacers
32	ZFFJ0/ZFBJ0	0.4	2.5	1	Uniform	Correction factor for form loss at downcomer cross connections
33	ZFFJ0/ZFBJ0	0.8	1.25	1	Uniform	Correction factor for form loss in surge line

Heat transfer						
34	IHTCI0	1 or 2				Selection of correlation (parameter 35)
35	OHWFB	0.65	1.3	1	Uniform	Correction factor for film boiling, modified Dougall-Rohsenow-correlation (50 %)
		0.75	1.25	1	Polygonal	Correction factor for film boiling, Condie-Bengston (50 %)
36	ICHFI0	0 or 4				Selection of correlation (parameter 37)
37	OTRNB	0.7	1.3	1	Uniform	Correction factor for critical heat flux, minimum value (50 %)
		0.7	1.3	1	Uniform	Correction factor for critical heat flux, Biasi correlation (50 %)
38	OHWFC	0.85	1.15	1	Uniform	Correction factor for single phase forced convection to water (Dittus-Boelter)
39	OHWNC	0.85	1.15	1	Uniform	Correction factor for single phase natural convection to water (Mc Adams)
40	IHTC30	1 or 2				Selection of correlation (parameter 41)
41	OHVFC	0.8	1.2	1	Uniform	Correction factor for single phase forced convection to steam (Dittus-Boelter II, 50 %)
		0.85	1.25	1	Uniform	Correction factor for single phase forced convection to steam (Mc Eligot, 50 %)
42	OHWNB	0.8	1.2	1	Uniform	Correction factor for nucleate boiling (modified Chen correlation)
43	OHWPB	0.75	1.25	1	Uniform	Correction factor for pool film boiling at natural convection (Bromley correlation)
44	OTMFB	0.9	1.28	1	Uniform	Correction factor for minimum film boiling temperature (Groeneveld-Stewart correlation)
45	HECU/HTCLO	20	100	50	Uniform	Accumulator heat transfer coefficient (W/m^2 K)
Convergence criteria, heater power						
46	EPS	10^{-4}	10^{-2}	10^{-3}	Triangular	Convergence criterion (upper local relative error)
47	QROD0/00	0.99	1.01	1	Uniform	Correction factor for heater power (nominal: 10 MW maximum power)
48	CLIMX	0.1	1	0.2	Uniform	Correction factor for lower local absolute error of void fraction (factor 1: $5x10^{-4}$)

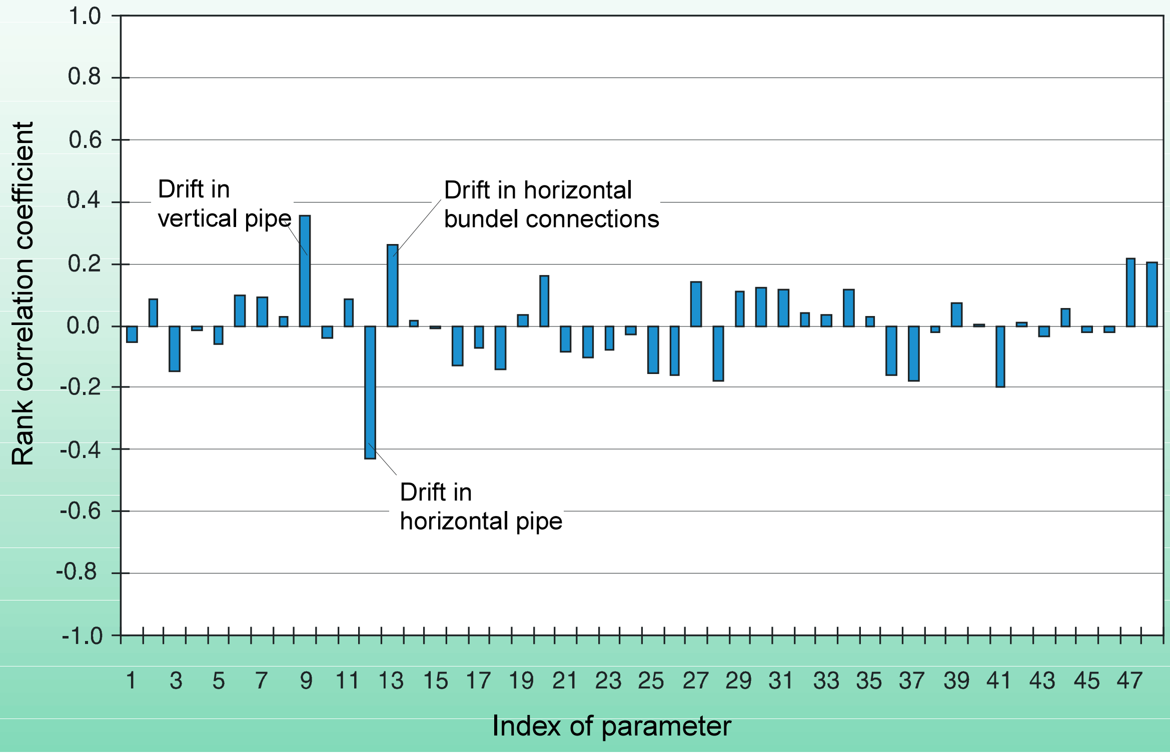
RESULTS

A total number of 100 ATHLET calculations was performed. According to Wilks' formula a minimum of 93 runs are required to establish two-sided tolerance limits with 95% probability and 95% confidence



Calculated uncertainty range and best estimate reference calculation compared with measured minimum and maximum values of rod clad temperature at level 8 on experiment LSTF-SB-CL-18

LSTF-SB-CL-18 experiment, first heat-up

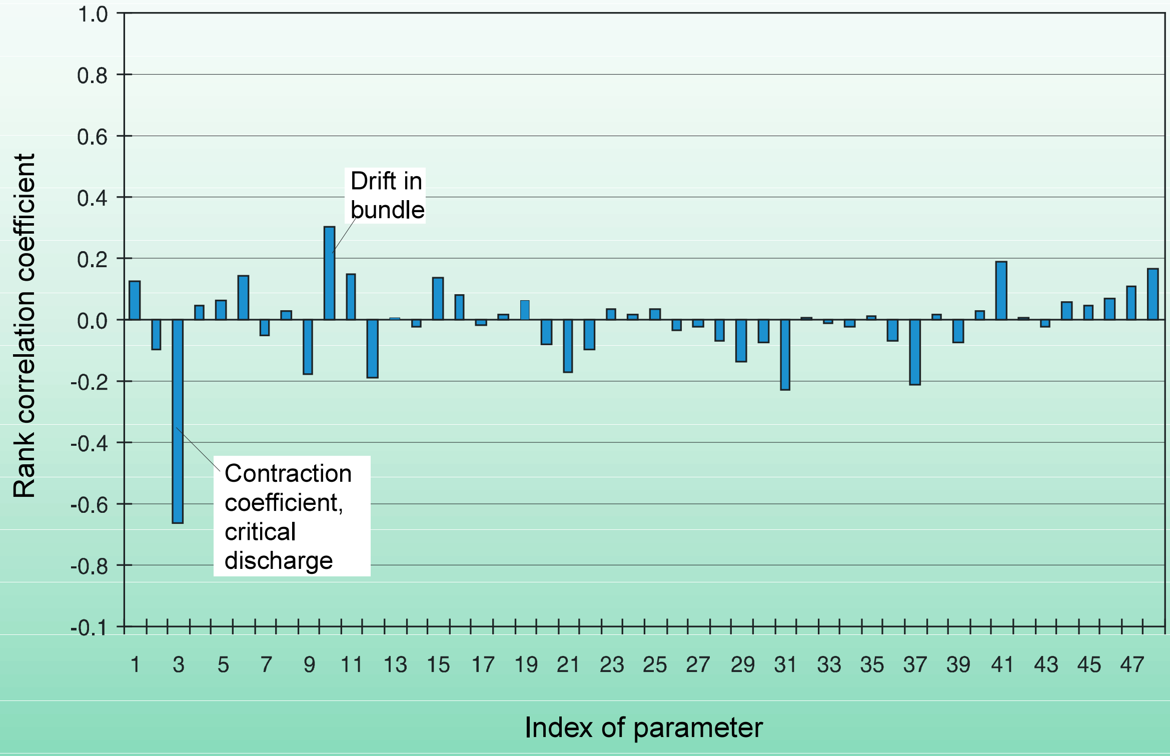


Sensitivity measures of the first peak clad temperature with respect to the selected 48 uncertain input parameters (rank correlation coefficient) for the LSTF experiment

According to these quantities, the most important three parameters are: **Drift in horizontal pipe, drift in vert pipe, and drift in horizontal connections of the heater rod bundle.**

An increased drift in the horizontal bundle connections (decreased water droplet transport to the hot bundle regions) and increased drift in the vertical pipe (impedes loop seal clearance) tend to increase clad temperature, whereas increased drift i horizontal pipe impedes loop seal filling and results in decreased clad temperature.

LSTF-SB-CL-18 experiment, second heat-up



Sensitivity measures of the second peak clad temperature with respect to the selected 48 uncertain input parameters (rank correlation coefficient) for the LSTF experiment

The Spearman Rank Correlation Coefficient shows the top ranking of the parameter uncertainties: **Contraction coefficient and vertical drift in the heater rod bundle**. An increased contraction coefficient will lead to an earlier accumulator injection, and consequently, tends to decrease the clad temperature. A higher drift in the bundle results in increased clad temperature in the upper bundle region.

5. Application of GRS method to a German Reference Reactor

Description of the accident scenario

A 5% break in the cold leg of a German PWR of 1300 MW electric power is investigated. Like for the LSTF experiment, a loss of off-site power at scram is assumed. The high pressure injection system is assumed to fail (this assumption is beyond design basis). All eight accumulators are available, four are connected to each of the four hot legs and four to each of the cold legs. The accumulator system is specified to initiate coolant injection into the primary system below a pressure of 2.6 MPa. After about 500 seconds the injection is into the hot legs only because the cold leg accumulators will be closed. The low pressure injection system is activated at 1.06 MPa.

Uncertain parameters

All parameters identified as potentially important are included in the uncertainty analysis. For this analysis a total of 45 potentially important uncertain parameters are identified. Included are 38 model parameters, 2 uncertainties of bypass flow cross sections in the reactor vessel (between upper downcomer and upper plenum, as well as upper downcomer and upper head), 4 uncertainties of reactor plant conditions, and 1 uncertainty of the numerical solution procedure.

Model uncertainties

For this reactor application 34 parameters are characterising computer code model uncertainties by uncertain corrective multipliers. Four additional model uncertainties are expressed by sets of alternative model formulations, i.e. two from wall heat transfer, and two from hydrodynamics (drift, pressure drop). The quantification of model uncertainties is based on the experience gained from ATHLET validation.

Scaling effects

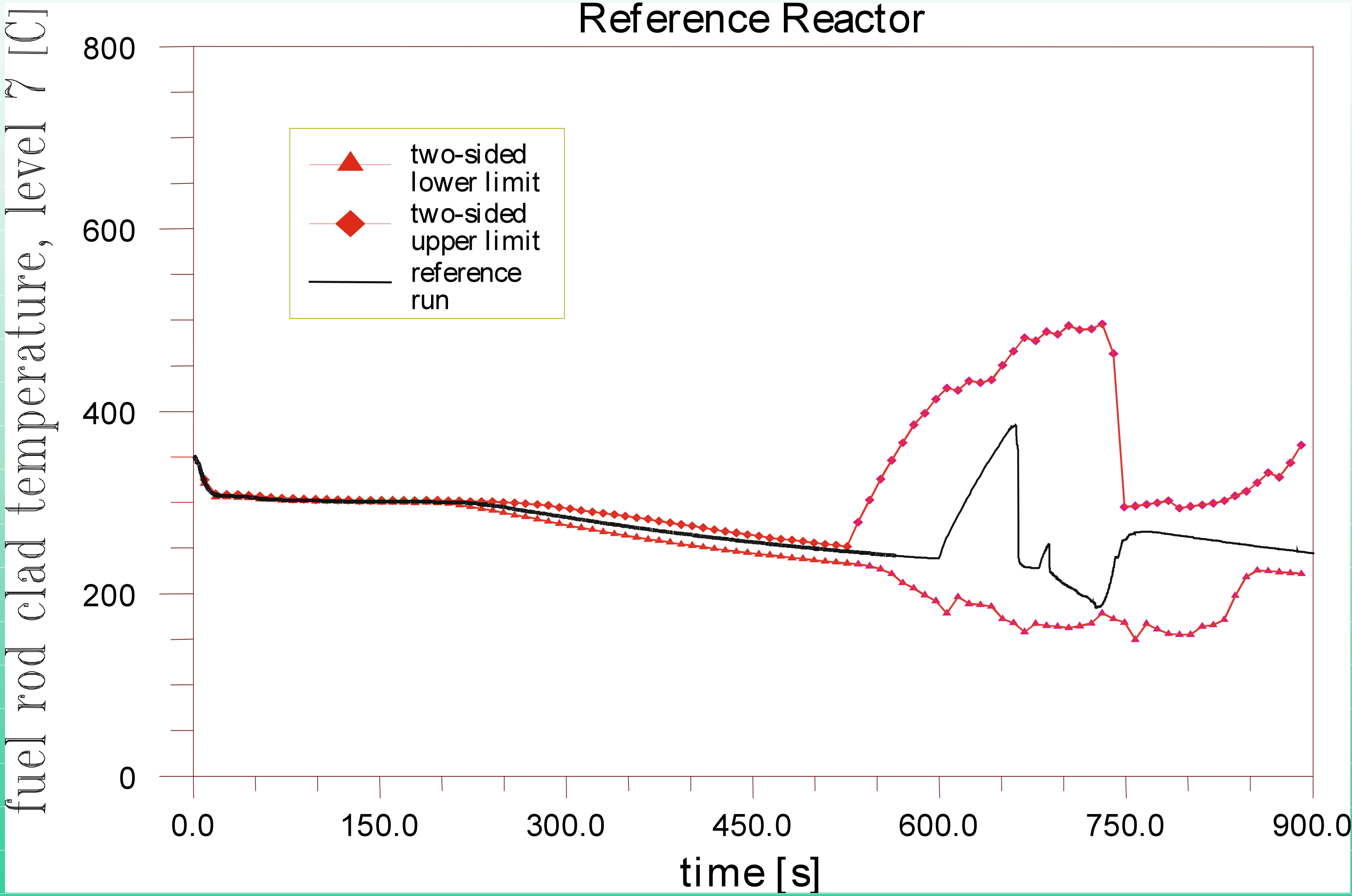
Possible contributions to the uncertainty of reactor calculations may come from scaling effects. Several tests of the German „Upper Plenum Test Facility" (UPTF) and UPTF-TRAM (Transient and Accident Management) tests in 1:1 scale were investigated by comparisons with ATHLET code calculations or with results of small scale facilities. It turned out that no additional uncertain model parameter has to be introduced to account for scaling effects.

Reactor plant conditions

- In order to account for uncertainties in reactor plant conditions, the uncertainties in core **power [100% - 106%], decay heat power [DIN ± 10%], fuel-clad gap conductance correction factor [0.885 - 1.63], and the temperature of the cooling water in the accumulators [30 - 40°C]** are included. For gap conductance a normal distribution is specified, for the other parameters a uniform distribution.

- **Realistic initial and boundary conditions** are used in the uncertainty and sensitivity analysis of the reference reactor. If specific conditions are not exactly known, they are considered uncertain. **The single failure criterion, however, is taken into account in a deterministic way, it is not treated as uncertainty**. This is a superior principle of safety analysis (redundance). The probability of system failures is part of probabilistic safety analyses, not of demonstrating the effectiveness of emergency core cooling systems. For design basis accidents the cooling system effectiveness has to be proven by deterministic safety analyses with regard to the available systems. The uncertainty analysis of such deterministic calculations permits a quantitative probabilistic statement about the margin between the tolerance limits of the calculation results and the acceptance limits. In the present investigation, however, **the single failure of one high pressure system, and the unavailability of a second high pressure system due to preventive maintenance are exceeded by the assumption of a complete failure of the high pressure injection system. High pressure system failures are the worst unavailabilities in small break loss of coolant accidents.**

RESULTS



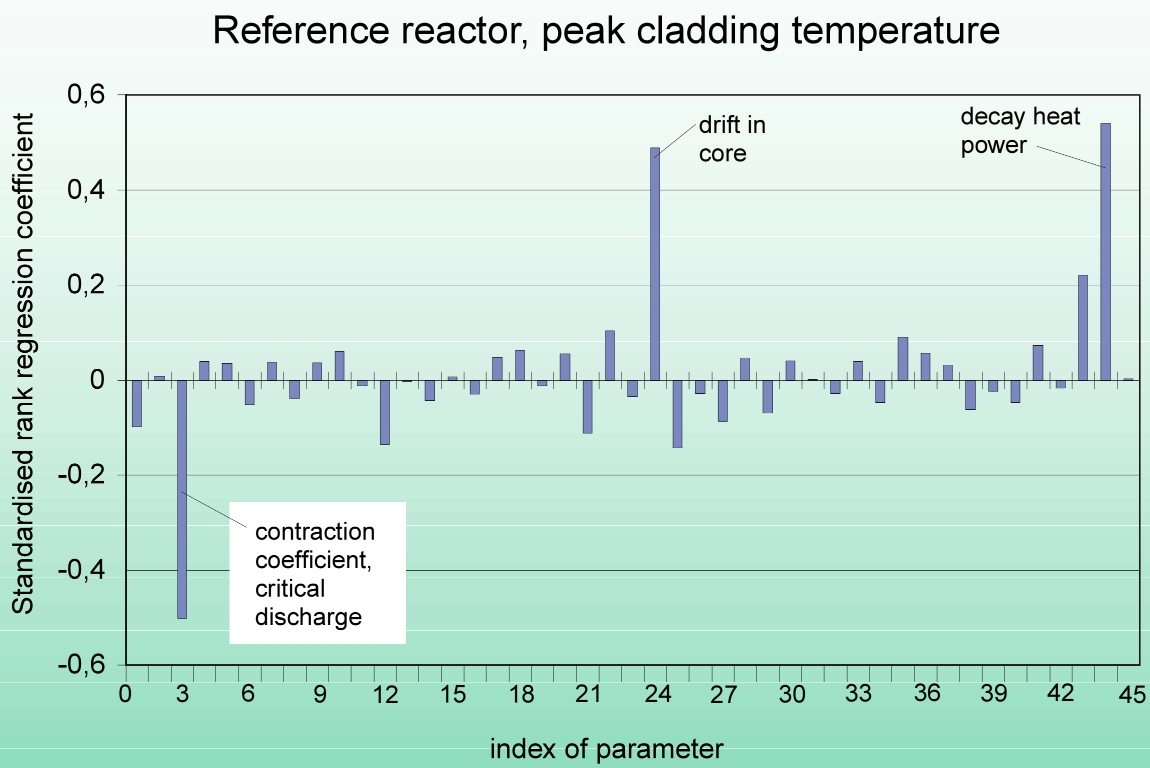
Calculated uncertainty range of rod clad temperature and best estimate calculation for reference reactor

Results

- The highest calculated clad temperatures are of special interest. The peak clad temperature is calculated in the upper part of the core (level 7). The previous figure shows the uncertainty range for the clad temperature of the rods at level 7. At least 95% of the combined influence of all identified uncertainties are within this range at a confidence level of 90% (77 calculation runs were performed).

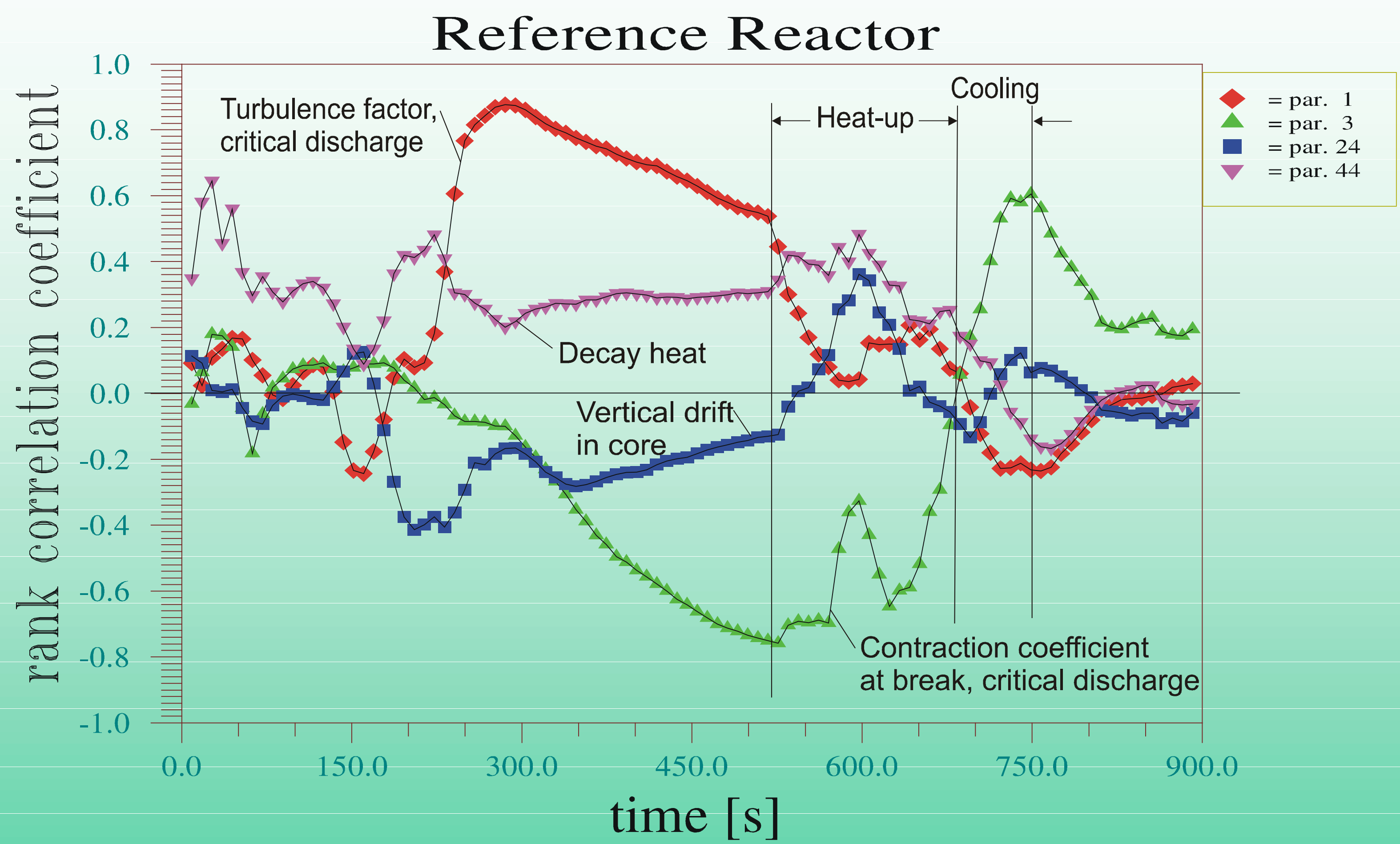
- A comparison of this calculation with LSTF results show differences at about 120 to 160 s. While LSTF shows a first heat-up during this time span, the reactor calculation does not show an increase in clad temperature. **This difference is mainly due to different decay power curves.** The maximum power in LSTF is only 14% of the scaled power under normal operating conditions. Therefore, this highest possible power is kept for 35 s after the scram signal, and is decreased subsequently to compensate for the lower initial power. Comparisons in the LOBI experimental facility, where the full scaled initial power was available, running experiments with reactor typical decay heat immediately after scram, show nearly no first heat-up compared with a power curve similar to LSTF.

- A heat-up is calculated at 500 s during core uncovery. The calculated upper tolerance limit of the maximum clad temperature at level 7 is 495°C. The maximum clad temperature at level 7 in the reference calculation (using nominal values) is 400 °C. After 750 s the uncertainty range decreases when the rods are cooled due to accumulator water injection. The earliest start of accumulator injection is at 540 s.



Sensitivity measures of the peak clad temperature with respect to the selected 45 uncertain input parameters (standardised rank regression coefficient) for reference reactor

The main contributions to uncertainty of the peak clad temperature come from the decay heat power (parameter 44), the vertical drift model in the core (parameter 24), and the contraction coefficient of the critical discharge model (parameter 3). Increasing the contraction coefficient tends to decrease the peak clad temperature and vice versa. An increased drift in the core and an increased decay heat power results in increased clad temperature at higher elevations.



Time dependent sensitivity measures of rod clad temperature for reference reactor (heat-up: 530-680 s, cooling: 680-750 s)

In addition to those mentioned for the peak clad temperature, the turbulence factor for evaporation in the critical discharg model (parameter 1) shows up to be an important uncertainty contributor in the time between 200 and 500 s. Between 40 and 800 s the contraction coefficient at the break is a major uncertainty contributor

Analysis of Loviisa-1 Transient, VVER-440, Load drop of one turbo-generator at nominal power (proposals done at VALCO – Workshop Helsinki, 25. – 26. April 2002)

For the uncertainty and sensitivity analysis it is necessary to identify the most relevant physical phenomena determining the transient evolution, this will lead to the identification of the relevant physical model parameters including their status of knowledge i.e. the range of parameter values and the subjective probability density distribution. In addition, it is important to define the main results of the calculation for which the sensitivity should be determined. As a basis of discussion it is proposed to define tables for each topic. The available results are referenced by the figures in the publications of Annuals of Nuclear Energy.

The uncertainty and sensitivity analysis consists of the following steps: Design an experiment and define the list of uncertain factors including the state of knowledge by the range of parameters and the corresponding subjective probability density functions. These steps will be supported by Tables 1, 2 and 3. On this basis by Monte Carlo sampling an input vector matrix will be defined for simulation by the computer model. The results of the calculation represent a sample of the output distribution. that will be evaluated by the uncertainty and sensitivity analysis. This evaluation will be guided by Table 4.

Following tables are proposed:

Table 1: Event sequence of transient

Table 2: Relevant physical phenomena of the transient

Table 3: List of uncertain factors or uncertain model parameters of transient analysis

Table 4: Parameters of the calculation results for which the uncertainty and sensitivity should be determined

Table 5: Most relevant deviations between calculations and measurement

Event sequence of the transient (table 1)

Nr	Time	Event	Consequences
1	0 s	Load drop of one turbo-generator	Pressure increase on secondary side
2	0 – 100 s	Step-wise insertion of control rod group 6	Reduction of neutron power from nominal to about 60 % with an intermediate hold at 50 s
3	20 - 100 s	After the increase of primary pressure the pressure control system activates heater switch-off and pressurizer spray valve opening	These actions lead to a decrease of primary pressure
4	20 – 100 s	Actions of volume control system by letdown opening	These actions lead to a decrease of water level in pressurizer
5	100 s	After the decrease of primary pressure the heaters of pressurizer are activated by the pressure control system	Increase of pressure, this is also affected by volume control means
6	100 s	Actions of volume control system by make-up injection	Increase of water level, which also contributes to increase of pressure

Relevant physical phenomena of the transient (table 2)

Nr	Reference to Table 1	Time-period	Event or Physical phenomena	Affected parameters
1	1	0 -20 s	Load drop of one turbo-generator	Pressure increase on secondary side
2		0 -20 s	Reduction of heat transfer between primary and secondary side	Increase of primary pressure Increase of cold leg temperatures
3		0 - 20 s	Heat-up of primary circuit and volume expansion of primary coolant	Increase of primary pressure and activation of pressure control in the pressurizer
4	2	0 – 100 s	Step-wise Insertion of control rod group 6 leading to a power reduction	Reduction of nuclear power generation and consequently decrease of coolant temperatures and decrease of water level
5	3	20 – 100 s	Pressure control in the pressurizer to reduce the pressure (energy balance and condensation)	Primary pressure and water level in the pressurizer
6		100 – 700 s	Long term pressure control in the pressurizer to increase and stabilize the pressure	Primary pressure and water level in the pressurizer

List of uncertain factors or uncertain model parameters of transient analysis (table 3)

Nr	Reference to Table 2	Models or parameters
1	1	Time point of load drop is fixed
2	2	Time function of secondary pressure. In ATHLET this time-function is generated by a GCSM model using feed water flow and steam extraction to obtain good agreement with measured values. What time-functions are acceptable in comparison to measurement? Compare time-functions of ATHLET and SMABRE. What affects the heat-transfer from primary to secondary side? Water level, nodalization other model parameters (HTC)?
3	3	Heat-transfer from primary to secondary side Volume of primary circuit Total mass flow (value and time-function) Primary pressure control
4	4	What are reasonable control rod insertion programmes that are acceptable with measurement? Which models were already used by participants (FZR)?
5	5	Pressure control system (time-point of switching-off heaters, efficiency of spray system) Volume control system (time-point of letdown opening, capacity of letdown system, duration of operating letdown system)
6	6	Time of starting make-up system, capacity of make-up system. What variants of make-up flow are reasonable? Time of operating heaters

Parameters of results for which the uncertainty and sensitivity should be determined (table 4

Nr	Parameter of result	Single valued	Index valued
1	Neutron power		X
2	Difference of neutron power level during power reduction between 40 and 60 s	X	
3	Steam header pressure of secondary side		X
4	Coolant temperatures in six cold legs		X
5	Maximum temperature in cold legs at about 10s	X	
6	Coolant temperature in six cold legs at the end of transient	X	
7	Pressure in primary circuit		X
8	Maximum pressure in primary circuit at about 10 s	X	
9	Value of primary pressure at about 400s	X	
10	Water level in pressurizer		X
11	Maximum of water level at about 10 s	X	
12	Minimum of water level at about 200 s	X	
13	Axial power density distribution		x
14	Detector readings at specified locations		x

Most relevant deviations between calculations and measurement (table 5)

Nr	Time	Parameter
1	50 s	Neutron power level between 40 and 60 s
2	0 -100 s	Position of control rod group 6
3	20 s	Coolant temperatures in six cold legs
4	700 s end of transient	Coolant temperature in six cold legs
5	200 -700 s	Water level in the steam-generator
6	20 s	Maximum value of primary pressure
7	100 – 400 s	Pressure increase in primary circuit due to pressure control
8	100 – 700 s	Pressurizer water level

This table is given to remind the most relevant deviations from measurement seen in the available calculations. It will not be used within the uncertainty and sensitivity analysis.

Conclusions on uncertainty evaluations

- **A challenge** in performing uncertainty analyses is **the specification of probability distributions of input parameters** as expression of the state of knowledge. **The specifications of ranges and probability distributions of uncertain input parameters may have a big influence** on the uncertainty of code results, and, thus, on the quantification of the prediction capability. Current activities in the frame of prediction capabilities of best estimate computer codes are **emphasising these specifications**.

- Investigations are underway to **transform data measured in experiments and post test calculations into thermal-hydraulic model parameters with uncertainties**. It is effective to concentrate on those uncertainties showing the highest sensitivity measures. **The state of knowledge about these uncertain input parameters has to be improved**, and suitable experimental as well as analytical information has to be selected. **This is a general experience applying different uncertainty methods**.