RSK RECOMMENDATION

Requirements for LOCA safety analyses

20/21 July 2005 (385th meeting)

1 Advisory request

At its 363rd meeting on 04/05 June 2003, GRS informed the RSK about the reliability of the results of thermohydraulic calculations within the framework of the discussions on probabilistic methods and procedures as well as their application for safety-related issues in the field of nuclear technology in Germany and abroad. The RSK requested the Committee on PLANT AND SYSTEMS ENGINEERING to discuss this and to prepare recommendations for the safety analyses methods against the background of the advanced state of knowledge.

2 Course of discussions

At its 18th meeting on 23.05.2003, GRS informed the Committee on PLANT AND SYSTEMS ENGINEERING about the treatment of uncertainties of computer code results on loss-of-coolant accidents (LOCAs) within the framework of its discussions on thermal power increase of the Grafenrheinfeld nuclear power plant (KKG) [1]. Following the explanations of GRS, the committee concluded that approaches practiced today with realistic (best estimate) computer codes and conservative initial and boundary conditions but without consideration of model uncertainties do not necessarily have to lead to conservative results for LOCA analyses. The committee referred to the draft of the KTA Basic Standard 6 in which the use of best estimate calculations was proposed for the determination of uncertainties.

At its 363rd meeting on 04/05 June 2003, the RSK heard a report of GRS on the determination of the reliability of thermohydraulic calculation results [2]. The RSK stated that, according to its opinion the uncertainties of the models computer codes should be quantified. Further discussions should take place in the RSK Committee on PLANT AND SYSTEMS ENGINEERING which discussed the topic at its 23rd meeting on 18.12.2003 and heard the relevant reports of the manufacturer [3] and of GRS [4]. The committee prepared a draft of this statement at its 24th, 25th, 28th, 30th and 31st meetings on 05.02.2004, on 30.03.2004, on 08.09.2004, on 09.12.2004 and on 03.02.2005 and adopted it at its 32nd meeting on 30.03.2005.

3 Assessment criteria

The aim of LOCA analyses is to demonstrate that the respective necessary precautions to prevent damage are taken. The necessary precautions shall be deemed to be taken if the demonstration criteria defined in the nuclear rules and regulations are fulfilled (22.1.1 in the RSK guidelines [7]).

Assessment criterion for an analysis is the fulfilment of the demonstration criteria with a high degree of safety. In this respect, the state of the art in science and technology is also defined by international approaches, e. g. in the USA [10,11,12] and in publications of the IAEA [8,9].

4 Requirements for LOCA analyses (emergency cooling analyses)

4.1 Background

The conservative “assumptions for the emergency core cooling calculations” listed in 22.1.3 of the RSK guidelines for pressurized water Reactors [7] are applied less and less since the use of experimentally verified models, as recommended in the RSK guidelines, proceeded. For years, realistic (best estimate) computer codes are used in LOCA analyses for the description of physical processes. These computer codes were and will be further improved and validated by means of comprehensive experiments. The aim of using these computer codes is to obtain results that are physically realistic as far as possible.

Despite validation of the calculation models, an uncertainty band remains due to model uncertainties and uncertainties of the plant and fuel rod parameters, e. g. scattering of measurement values, simplifications in the models and due to variation and imprecise knowledge of initial and boundary conditions and also because models are developed under consideration of experiments by which the complex behaviour of a reactor plant during accidents can only be determined approximately. It is therefore not obvious with which probability and statistical safety the result is below the demonstration criterion.

Further, it was postulated that the uncertainties of the analyses are covered by conservative boundary conditions and assumptions. However, a quantitative verification is not given as long as this coverage is not confirmed by uncertainty analyses. An uncertainty analysis performed by GRS [1,2] on the large double-ended break in the cold leg of the main coolant line gave indications that calculation results with an upper uncertainty band of 95 % probability (quantile) and a statistical safety of 95 % are not necessarily covered by one calculation which is based on the conservative initial and boundary conditions of the rules and regulations. This was primarily due to the fact that the conservative initial and boundary conditions for the emergency core cooling calculations do not cover the uncertainties associated with the models used.

From the point of view of the RSK, demonstration on the basis of best-estimate analyses therefore requires a quantification of the model uncertainties and of the data uncertainties of the plant and fuel rod parameters.

In the following sections, requirements on the performance of LOCA analyses are formulated (emergency cooling analyses for demonstrating the effectiveness of the emergency cooling systems). Distinction is made between demonstration by means of a realistic approach with uncertainty analysis and demonstration by a simplified procedure compared to it. In the first case, the uncertainties associated with the determination of the results are explicitly shown for the demonstration and considered in the evaluation of the results. In the second case, these uncertainties have to be covered conservatively with regard to the objective of the demonstration by a simplified approach.

4.2 Deterministic requirements

In the following, postulates on the scenario are defined deterministically by which the effectiveness of the emergency cooling system is to be demonstrated and which have a conservative effect regarding the fulfilment of demonstration criteria.

Within the framework of an emergency cooling analysis, e. g. system losses are not treated probabilistically. Consequently, for the analyses, the most unfavourable combinations is defined from:

- a single failure to be postulated,
- a loss due to maintenance work,
- break location,
- size and type of the break
  - double-ended break from 1F to 2F
  - small leak,
- the loss of offsite power,
- nominal thermal power (in case of incidents, the most unfavourable values have to be considered which can occur during specified normal operation under consideration of the limitation systems in the integral power and power density. Measurement and calibration errors can be considered statistically), and
- the cycle time.

For the determination of the most unfavourable conditions, sensitivity analyses may be required. Further relevant assumptions of the RSK guidelines, 22.1.3 [7] that are not considered in the computer code by experimentally verified models, are to be considered for the analysis according to the state of the art in science and technology. These are (1) 8. (pump behaviour), (1) 10. (flow rate reduction), (2) (long-term prevention of steam release from the pressure boundary) and (3) (suction head of the RHR pumps after switching to sump operation).

4.3 Best estimate approach

An analysis under consideration of the best estimate approach is to be performed with a best estimate computer code. In addition, such safety analysis requires the quantification of the uncertainty of the calculation result, i. e. the reliability of the calculation result.

The initial and boundary conditions and parameter values required for the performance of a LOCA analysis are to be classified according to type and method of their determination, i. e. according to requirements that can be defined deterministically and those that can be treated statistically.

4.3.1 Computer codes

A best estimate code model must be able to describe the relevant physical processes during the LOCAs to be analysed as realistically as possible. This is primarily to be demonstrated within the frame of the validation of the computer code by means of relevant experiments and knowledge about the behaviour of the reactor plant.

Validation is a process where a comparison of results of computer codes (with the models contained in them) with experimental measurement results is performed and documented. Experiments that have already been used for the development of the calculation models shall not be used for the validation. The suitability of a calculation program must have been demonstrated to a sufficient number of experiments for the scope of application. In this respect, scaling effects have to be considered that occur due to the fact that most of the experimental data used within the frame of validation were measured at smaller experimental facilities that the reactor plant. The validation process must be documented in a comprehensible and traceable form.

4.3.2 Best estimate analysis and treatment of uncertain parameters

A best estimate analysis shall

- use best estimate computer codes that include assumptions for the model parameters that are as realistic as possible,

and has to

- consider the requirements and boundary conditions presented in 4.2, and
- quantify the uncertainty of the calculation results.

For a best estimate analysis, the relevant uncertainties of the models and the input data have to be identified and quantified.

The term “uncertainty” or “reliability” of best estimate calculation results covers different contributions to the total uncertainty of calculation result.

In this respect, the uncertainty analysis shall cover the following uncertainties that are due to imprecise knowledge of input parameters:

- Uncertainties of individual calculation models: Uncertainties of the different models in the computer codes are to be determined on the basis of experimental results.

- Imprecise knowledge of the plant states: Statistically to be considered are, e. g., initial values for pressures, temperatures, mass flows, fuel rod characteristics, decay heat and measurement and calibration errors. Parameters for which scores can be defined which safely lead to conservative results can be considered in the calculation with the respective score.

Further model uncertainties can also be due to the simulation of 3D effects with 1D calculation models. These uncertainties cannot be covered by statistical uncertainty analyses and can only be reduced by further development of the models (increased application of 3D models).

Probability distributions and their band widths are based on experimental data, the experience of the experts and sensitivity analyses performed. They are determined on the basis of data obtained within the frame of code validation or by means of plant and fuel rod data. In this respect, the data for the distribution obtained from experiments or measurements shall be representative for the analysed plant.

The scatter range of the experimental data is considered in the statistical uncertainty analysis in so far that it is considered in the determination of the statistical distribution of the uncertain parameters. The respective findings must be implemented in a comprehensible and traceable manner.

For the analysis, the parameter values are randomly selected according to the respective probability distributions. This selection must consider potential parameter correlations. The dependency of different parameters on each other can be expressed deterministically by functional relationships or by different statistical correlations and conditional distributions. The values of the parameters identified are to be varied. With each value set obtained for the different parameters, a calculation has to be performed with a computer code.

4.3.3 Determination of the total uncertainty of the calculation results

Following the international approach, the RSK states that it has to be demonstrated within the frame of a best estimate analysis that the quantitative demonstration criteria are fulfilled with a high degree of certainty. A high degree of reliability of the results is given if the averaged result is below the demonstration criteria with a probability of at least 95 % and a statistical certainty of 95 %.

The RSK is of the opinion that with the deterministic requirements (see 4.2.) and by quantification of the uncertainties of calculation results with these tolerance limits it is ensured with sufficient reliability that the requirements for LOCA safety analyses are fulfilled.

4.4 Simplified approach

A LOCA safety analysis according to a simplified approach without quantification of the result uncertainties is permissible if influencing parameters have been identified with which a conservative result can be obtained, i. e. the tolerance limits of the uncertainty analysis according to 4.3.3 are covered.

To demonstrate this, applicable generic studies can be referred to.

5 Conclusions and recommendation

This recommendation sets up requirements on LOCA safety analyses with a best estimate approach, including uncertainty analysis, and safety analyses with comparably simplified approach. Both approaches use best estimate computer codes. There are differences regarding the treatment of uncertain model and plant parameters. The best estimate approach explicitly quantifies the uncertainties of the calculation results due to it; with the simplified approach, these uncertainties are covered by selection of conservative values for sensitive influencing parameters. The RSK recommends that future LOCA analyses using best estimate computer codes are performed in accordance with this recommendation.

Documents used for discussions and further literature

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H. Glaeser, GRS, 363. Sitzung der Reaktor-Sicherheitskommission, Köln,
04. und 05. Juni 2003, Folienkopien

[3] Anwendung realistischer Methoden für Störfallanalysen
U. Stoll, 18.12.2003, Framatome ANP, Folienkopien

[4] Anforderungen an realistische („best-estimate“) Störfallanalysen für LWR
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[5] S.S. Wilks
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[7] RSK-Leitlinien für Druckwasserreaktoren, 3. Ausgabe, Oktober 1981 mit Änderungen vom 15.11.1996

[8] Safety margins of operating reactors, Analysis of uncertainties and implications for decision making;
IAEA-TECDOC-1332, January 2003

[9] Safety Report Series No. 23: Accident Analysis for Nuclear Power Plants; International Atomic Energy Agency, Vienna, 2002

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[12] Muftuoglu, K., Ohkawa, K., Frepoli, C., Nissley, M.
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