Safety Standards

of the
Nuclear Safety Standards Commission (KTA)

KTA 3101.1 (2022-11)

**Design of Reactor Cores of Pressurized Water and Boiling Water Reactors;
Part 1: Principles of the Thermo-Hydraulic Design**

(Auslegung der Reaktorkerne von Druck- und Siedewasserreaktoren; Teil 1: Grundsätze der thermohydraulischen Auslegung)

The previous versions of this safety standard were issued in 1980-02,2012-11 and 2016-11

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KTA SAFETY STANDARD

November 2022	Design of Reactor Cores of Pressurized Water and Boiling Water Reactors; Part 1: Principles of the Thermo-Hydraulic Design	KTA 3101.1

Previous versions of this safety standard: 1980-02 (BAnz No. 92 of May 20, 1980)
2012-11 (BAnz AT 23.01.2013 B5)
2016-11 (BAnz AT 19.06.2017 B1)

Contents

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Comments by the Editor:

Taking into account the meaning and usage of auxiliary verbs in the German language, in this translation the following agreements are effective:

shall	indicates a mandatory requirement,
shall basically	is used in the case of mandatory requirements to which specific exceptions (and only those!) are permitted. It is a requirement of the KTA that these exceptions - other than those in the case of **shall normally** - are specified in the text of the safety standard,
shall normally	indicates a requirement to which exceptions are allowed. However, exceptions used shall be substantiated during the licensing procedure,
should	indicates a recommendation or an example of good practice,
may	indicates an acceptable or permissible method within the scope of this safety standard.

Basic Principles

(1) The safety standards of the Nuclear Safety Standards Com-mission (KTA) have the task of specifying those safety-related requirements which shall be met with regard to precautions to be taken in accordance with the state of science and technology against damage arising from the construction and operation of the plant (Sec. 7, para. (2), subpara. (3) Atomic Energy Act - AtG) in order to attain the protective goals specified in the AtG, the Radiation Protection Act (StrlSchG) and the Radiation Protection Ordinance (StrlSchV) as well as further detailed in the Safety Requirements for Nuclear Power Plants (SiAnf) and the Interpretations of the SiAnf.

(2) The objective of safety standards series KTA 3101 is to specify the requirements for the reactor core design of pressurized water and boiling water reactors. Safety standards series KTA 3101 is comprised of the following three parts:

Part 1: Principles of the thermo-hydraulic design (*the present safety standard*),

Part 2: Neutron-physical requirements for the design and operation of the reactor core and adjacent systems,

Part 3: Mechanical and thermal design.

(3) The present Part 1 of safety standards series KTA 3101 deals with the precautions mentioned under para. (1) that are particular to the thermo-hydraulic design of the reactor core of nuclear power plants.

(4) In accordance with the Safety Requirements for Nuclear Power Plants (SiAnf), no. 2.3, para 2, it is necessary - on safety levels 1 to 4a - to ensure that the fuel assemblies are cooled and that the heat within the nuclear fuel is transported to the heat sink. In addition, the mechanical, thermal, chemical and radiation induced influences on the barriers or containment devices shall be delimited in such a way, that the barriers and devices remain functional to ensure the radiological safety goals.

(5) In accordance with SiAnf, no. 3.3, para 1, the cooling of the fuel assemblies (heat removal from the reactor core) on safety levels 1 to 4a must be ensured during all phases of operation, so that the verification goals and criteria that have been set up for the individual safety levels for the fuel assemblies and other safety relevant installations are upheld during the entire operating time.

1 Scope

(1) This safety standard applies to stationary nuclear power plants with light-water moderated pressurized water or boiling water reactors. It specifies the thermo-hydraulic requirements for the design and operation of the reactor core as well as the requirements for the associated experiments. In the context of the present safety standard, the reactor core includes: fuel assemblies, fuel assembly channels (BWR), control assemblies, absorber assemblies, in-core instrumentation, neutron sources and coolant inlet orifices.

(2) Requirements for adjacent system are dealt with only insofar as this is necessary with regard to the thermo-hydraulic design and operation of the reactor core.

(3) This safety standard does not specify any requirements for the analysis of loss-of-coolant accidents nor for the design requirements resulting from these accidents.

(4) This safety standard does not specify any requirements for the safety-related design regarding zero-power operation with an open primary circuit (PWR) or with an open water-steam circuit (BWR).

Note:
The respective requirements are specified in safety standards KTA 3301 and KTA 3303.

2 Definitions

2.1 General Definitions

(1) Anticipated transients without scram - ATWS

ATWS is a transient of abnormal operation in conjunction with an assumed failure of the emergency shutdown system.

(2) Initial state power distribution

An initial state power distribution is a measured or calculated distribution which serves as the basis for the analyses of transients and design basis accidents and is characterized by the fact that it represents the most unfavorable initial state for the respective analysis.

(3) Fuel rod cluster

A fuel rod cluster is that part of a fuel assembly which is used as the smallest unit in the design calculations.

(4) Correlation, empirical

An empirical correlation describes an empirically determined connection between the physical parameters of a system.

(5) Film boiling

Film boiling is the boiling process described by a stable steam film existing between fuel rod cladding and cooling liquid.

(6) Technological limit

A technological limit is that value of a physical parameter used for describing the condition of components, systems or of the media contained therein, which, when exceeded, may possibly cause a failure of the respective component or of the respective system.

(7) Most heavily loaded fuel rod, fuel assembly or fuel rod cluster

The most heavily loaded fuel rod, fuel assembly or fuel rod cluster is the one that, with respect to one of its characteristics, exhibits the smallest distance to the corresponding technological limit.

(8) Correlation describing critical boiling conditions

The correlation describing critical boiling conditions specifies the dependency of the critical heat flux or the critical steam quality on the characteristics of the coolant flow and on the geometry of the coolant channel.

(9) Coolant channel

A coolant channel is the technological conglomerate comprising the fuel rod cluster and the corresponding portions of the coolant and of the devices directing the coolant flow.

(10) Power form factor

The power form factor at a certain location in the reactor core is the ratio of the power density at this location to the average power density in the reactor core or in parts of the reactor core.

Note:
Power form factors can also be specified as ratios involving the linear heat rate, heat flux or enthalpy rise.

(11) Verification criterion

A verification criterion is a criterion which, in the course of verification, must be verified as being fulfilled.

(12) Safety Levels 1 through 4a

Cf. safety standard KTA 3103.

(13) Critical boiling condition

A critical boiling condition is one that arises as well at the onset of film boiling (zero departure from nucleate boiling, DNB = 0) as at the onset of dryout.

(14) Tolerance limit, 95 % / 95 % - tolerance limit

The 95 % / 95 % - tolerance limit is a value that overestimates the 95 % quantile with a confidence level (statistical confidence) of 95 %.

Note:
The tolerance limit for the design or for a safety-related verification shall be applied one-sided or two-sided depending on the respective verification criterion.

(15) Critical heat flux (CHF)

The critical heat flux is the heat flux when film boiling sets in or upon the start of dryout.

2.2 PWR-specific Terms

(1) DNB ratio (DNBR)

The DNB ratio is the ratio of critical heat flux to actual heat flux.

(2) $DNBR_{min}$

$DNBR_{min}$ is the smallest value of the ratio of critical heat flux to actual heat flux.

(3) DNBR limit value ($DNBR_{limit}$)

$DNBR_{limit}$ is the smallest value of the DNB ratio where film boiling may be assumed to be impossible with a 95 % / 95 % - tolerance limit.

(4) $DNBR_0$

$DNBR_0$ is the minimum permissible DNB ratio during normal operation.

Note:
The value of $DNBR_0$ will be specified such that when it is maintained during normal operation – in conjunction with other design requirements – the fulfillment of the safety-related requirements at Safety Levels 1 through 4a can be verified.

(5) $\Delta DNBR_{trans}$

$\Delta DNBR_{trans}$ is the margin of the DNB ratio required for achieving $DNBR_{limit}$ at the limiting transient of abnormal operation (Safety Level 2).

2.3 BWR-specific Terms

(1) ASL

ASL (margin to the power at onset of transition boiling) is the ratio of the transition boiling power to the actual operating power.

(2) Dryout

Dryout is the drying up of a heated surface due to a partial or complete vanishing of the wetting liquid film from this heated surface.

(3) Critical steam quality

The critical steam quality is the steam mass ratio at which dryout begins.

(4) MASL

MASL (minimum margin to the power level at onset of transition boiling) is the smallest ratio of the transition boiling power to the actual operating power.

(5) MASL100

MASL100 is that value of MASL where the expected value of the number of fuel rods subject to a critical boiling condition is smaller than 1 fuel rod of the reactor core.

(6) MASL99.9

MASL99.9 is that value of MASL where the expected value of the number of fuel rods subject to critical boiling condition is smaller than 0.1 % of the total number of fuel rods of the reactor core.

(7) MASLperm

$MASL_{perm}$ is the smallest permissible value of MASL during normal operation.

Note:
The value of $MASL_{perm}$ will be specified such that when it is maintained during normal operation – in conjunction with other design requirements – the fulfillment of the safety-related requirements at Safety Levels 1 through 4a can be verified.

(8) $\Delta MASL_{trans}$

$\Delta MASL_{trans}$ is the margin of MASL required for achieving $MASL_{99.9}$ at the limiting transient of abnormal operation (Safety Level 2).

(9) Boiling length

The boiling length is that length of the fuel rod region where nucleate boiling occurs.

(10) Power level at onset of transition boiling

Power level at onset of transition boiling is the power at which dryout sets in.

Note:
The power level at onset of transition boiling is the power of a single fuel assembly.

3 Safety-Related Requirements for the Thermo-Hydraulic Design of Reactor Cores

3.1 Basics

(1) The safety-related requirements of the present safety standard apply to normal operation (Safety Level 1), to abnormal operation (Safety Level 2), to design basis accidents (Safety Level 3), as well as to the very seldom events to be considered in connection with the thermo-hydraulic design (anticipated transients without scram – ATWS – Safety Level 4a). Insofar as different requirements apply at the individual safety levels, these are specified in the present safety standard.

(2) The events to be considered for the thermo-hydraulic reactor core design and their allocation to the individual safety levels are specified in Appendix A.

(3) The safety levels by themselves represent a graded safety concept (defense-in-depth concept) where the individual events to be considered are allocated to these levels in accordance with their probability of occurrence.

(4) At all safety levels, requirements from other analysis areas (e.g., neutron-physical and mechanical design) and from superordinate technical standards shall be taken into account.

(5) The geometry of the fuel assemblies and the coolant flow guides shall be specified such that the graded requirements within the Safety Levels 1 through 4a fulfill the superordinate protective goals of

a) controlling the reactivity,

b) cooling the reactor core, and

c) retaining the radioactive substances.

Any uncertainties in the mathematical models as well the operational variations and uncertainties of the parameters in context with the safety-related verifications shall be taken into account (cf. Section 4.3)

(6) The pressure loads of the pressure retaining boundary shall at all safety levels be limited to the permissible values.

Note:

The permissible pressure load values are a result of the mechanical load capacity of the pressure retaining boundary (cf. safety standards KTA 3201.2). In safety standard KTA 3201.2 the events to be assumed are correlated to various service limit levels at each of which different requirements apply.

(7) The pressure loads of the reactor core and of the reactor pressure vessel internals at Safety Levels 1, 2 and 3 shall be limited to be in accordance with the thermo-mechanical design requirements.

(8) At Safety Level 4a, the specific requirements regarding the thermo-hydraulic design of the reactor core pertain solely to anticipated transients without scram (ATWS).

(9) The further requirements specified in the present safety standard are correlated according to the respective safety levels. The strictest requirements apply to normal operation (Safety Level 1). Each superordinate safety level includes the requirements of the following higher numbered safety level (cf. **Figure 3-1**).

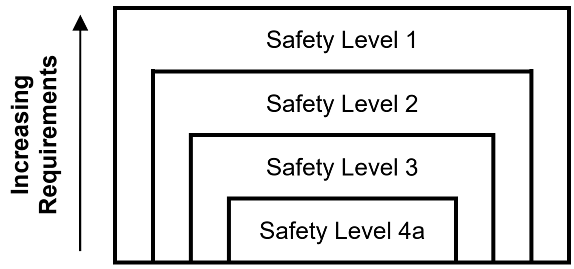


Figure 3-1: Safety levels

3.2 Safety Level 1

(1) The thermo-hydraulic stability of the reactor core shall be ensured. The design of the reactor core – in case of a BWR, in combination with the operational characteristics – shall be such that there is always a sufficient margin to the operational region in which undamped power density oscillations could occur.

(2) In the interaction with the controlling and limiting devices (limitation of process variables), the maximum values of the local power density and the minimum margins to critical boiling states shall be limited to those values that are used as input values for verifying the mitigation of the conditions of abnormal operation and design basis accidents.

(3) It shall be ensured with sufficient statistical certainty that critical boiling states will not occur. In the case of BWR, this requirement is considered as being fulfilled if the limit value for $MASL_{100}$ is maintained. In the case of PWR, no individual verification is required, provided, the transient margin $\Delta DNBR_{trans}$ that follows from the design at Safety Level 2 also covers the requirements of Safety Level 1.

Note:

Usually no individual verification is required in the case of PWR since $\Delta DNBR_{trans}$ sufficiently exceeds the variance of the DNB correlation.

(4) Any lift-off of the fuel assemblies from the lower core grid (core plate) caused by upward flow forces shall be prevented.

3.3 Safety Level 2

(1) It shall be ensured that the fuel assemblies will retain their unrestricted reusability.

(2) An unrestricted reusability of the fuel rods inside the reactor core after a transient at Safety Level 2 can be verified either by item a) or item b) as follows:

a) Verifying that the expected value of the number of fuel rods reaching the critical boiling state is smaller than 0.1 % of the total number of fuel rods in the reactor core.

   aa) In the case of BWR, this requirement corresponds to meeting the $MASL_{99,9}$ limit value.

   bb) In the case of PWR, this requirement is considered fulfilled if it is verified for the most heavily loaded fuel rod in the hot channel that film boiling will not occur with a 95 % / 95 % - tolerance limit.

   Note:

   This simplified verification in case of a PWR is based on the fact that, because of the heterogeneity of the power distribution, the other fuel rods of the reactor core will display a smaller probability of film boiling than the hot-channel rod.

b) Verifying that the material dependent temperature-over-time criteria of the fuel rod cladding are fulfilled and that no center fuel melt will occur. This is the case if the expected value for the number of fuel rods exceeding these criteria is smaller than 1 fuel rod of the reactor core.

(3) It shall be ensured for all components of the reactor core that the temperatures and pressures or pressure differences are nowhere higher than the values from where on the characteristics of the applied materials or the safety-related function of the components would be impermissibly altered.

3.4 Safety Level 3

(1) Any self-sustaining exothermal zirconium-water reaction shall be prevented.

(2) The power and power densities in the interaction with the reactor protection system shall be limited such that fuel rod failures are either prevented or that the radiological effects of these failures are limited to permissible values.

(3) In order to prevent any exceedance of the radiological limit values, the expected value for the number of failed fuel rods shall be limited taking the respective uncertainties into account.

(4) The requirement under para. (3) is fulfilled if it is shown that the superposition of probabilities for fuel rod failures due to

a) critical boiling states or exceedance of material dependent temperature-over-time criteria of the fuel rod cladding,

b) center fuel melting, and

c) exceedance of the limit values for fast enthalpy insertion

will lead to an expected value for the number of failed fuel rods that is smaller than the permissible number of failed fuel rods of the reactor core.

(5) Insofar as fuel rod failures cannot be completely prevented, any sequential failures shall be taken into account.

3.5 Safety Level 4a
(very seldom, postulated events; here, only ATWS)

(1) The capability for cooling the reactor core and for maintaining the long-term subcriticality shall be ensured.

(2) The initial state for the ATWS analysis shall be that of full operational power at xenon equilibrium for the most unfavorable point in time of the fuel cycle.

4 Requirements Regarding Methods Applied to the Thermo-Hydraulic Design of Reactor Cores

4.1 Essential Interconnections of the Thermo-Hydraulic Design of Reactor Cores with other Analysis Fields

(1) The interconnections of the thermo-hydraulic design of reactor cores with other analysis fields are presented graphically in **Figure 4-1**. The thermo-hydraulic design is closely interconnected with the fields of nuclear design, of mechanical and thermo-mechanical design, and the design of plant thermo-hydraulics and instrumentation and control. These latter fields furnish the input data needed to perform the thermo-hydraulic design. The results calculated with these input data must then fulfill the requirements specified for the thermo-hydraulic design and must stay within the specified limits. On the other hand, the results of the thermo-hydraulic core design are dynamically fed back to the fields of nuclear design, of mechanical and thermo-mechanical design, and the design of plant thermo-hydraulics and instrumentation and control as the respective input data needed in these fields.

(2) In order to meet the safety-related requirements of Section 3, the power density distribution, coolant throughput through the reactor core, coolant throughput distribution, inlet temperature and system pressure shall be dimensioned within the reactor plant's operational boundary defined by the reactor protection system, the limitation of process variables, the protective limitations, the controls and the operating manual in order to comply with the respective technological boundary. In this context, the interconnections presented in **Figure 4-1** shall be taken into consideration.

(3) All in all, the contributing analysis fields taken together make up an analysis system in which each one of the fields must be well tuned to the other.

(4) In the following sections the general requirements are dealt with from the point of view of

a) Coupled analyses (cf. Section 4.2),

b) Accounting for uncertainties in the reactor core design (cf. Section 4.3)

c) Simplifications and approximations (cf. Section 4.4).

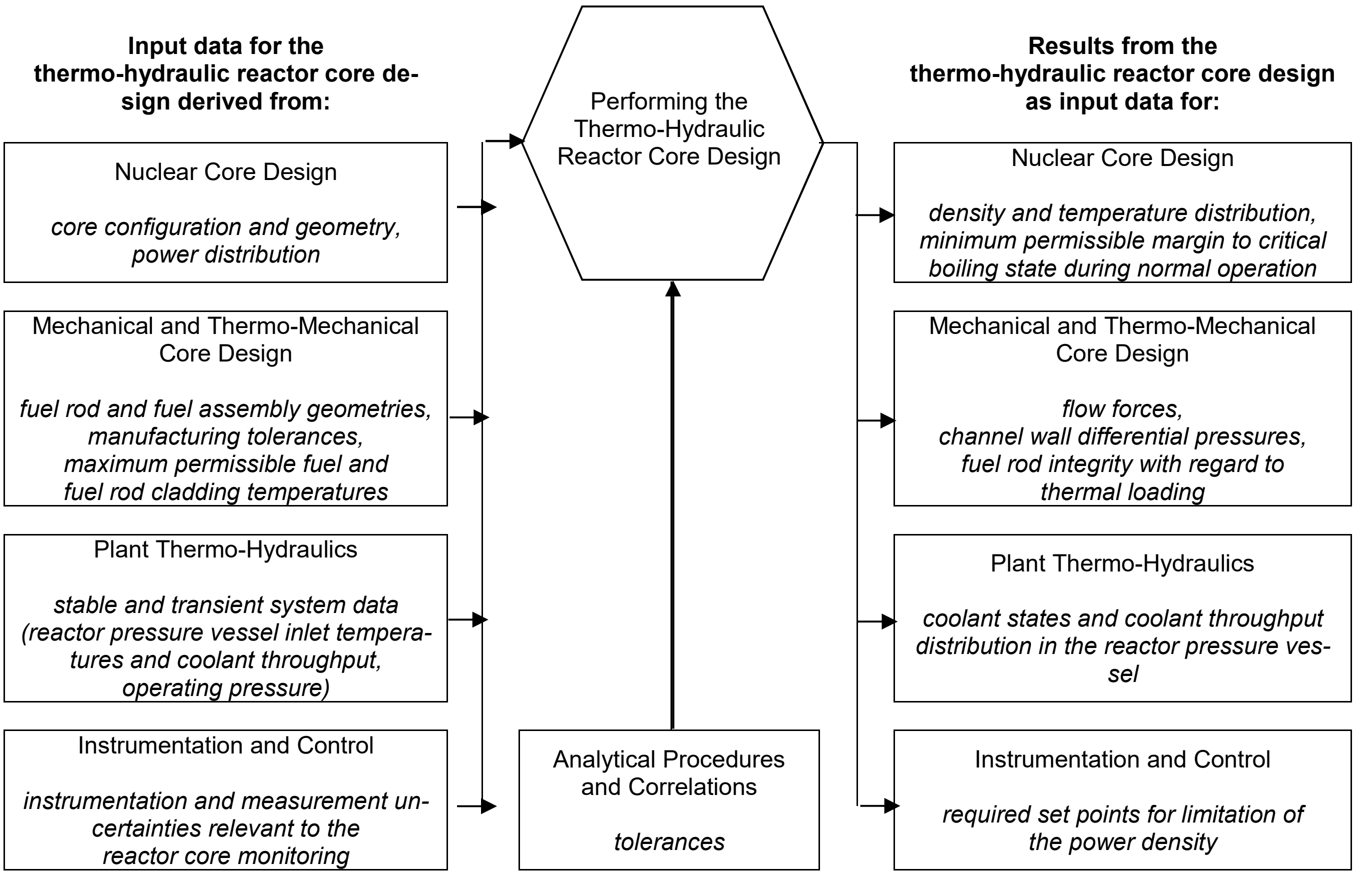


Figure 4-1: Interconnections of the thermo-hydraulic design of reactor cores with other analysis fields

4.2 Coupled Analyses

(1) The behavior and functioning of technical systems can be described by closed mathematical solutions or by coupled calculation models. Coupled calculation models allow interconnecting clearly defined models for different parts of a technical system by a mutual interchange of the input and output parameters between the individual models. Two models are considered as being coupled if the input parameters of the one model are the output parameters of the other and vice versa. Because each one influences the other it may be necessary to involve an iteration algorithm, especially for the calculation of a steady state. The coupling of models may be carried out directly at the level of the program code or by superordinate procedures.

(2) A typical example of how a coupled program system can be constructed is shown in **Figure 4-2** indicating the individual calculation models and their essential coupling parameters.

(3) If a coupled program system is used for the safety analyses, it must be sufficiently flexible to allow performing realistic analyses (so-called “best estimate analyses”) as well as conservative analyses. In this context, it shall offer sufficient possibilities to alter not only the input parameters of the coupled models but also the coupling parameters themselves, either directly or indirectly and, thereby, making it possible to examine or cover the uncertainties of the parameters and models. It shall, furthermore, allow presetting the failures of system components or of control signals that must be assumed to occur.

(4) The coupled design programs shall basically be verified and validated in the same way as the other reactor core design programs(cf. Section 4.5). However, since it usually will not be possible to check the entire scope of application of individual models of a coupled program system with the integral program tests and calculation checks, it will be necessary to additionally validate the individual models separately. Integral program tests shall be performed to check at least a certain number of model couplings and thus, taken together, to check all models in their interaction within the respectively coupled models.

Note:

Of major importance in this context are calculation checks of those experiments in which the interaction of several models is a significant factor.

4.3.1 Basics

(1) The reactor core design shall fulfill the associated verification criteria. In this context, the following uncertainties shall be taken into account at the Safety Levels 1, 2 and 3:

a) fabrication uncertainties,

b) measurement uncertainties,

c) fluctuations of operating parameters,

d) systematic deviations, and

e) uncertainties of the calculation models.

Note:

Requirements regarding Safety Level 4a are specified under Section 4.3.4.

(2) All those uncertainties shall be taken into account that have a significant influence on the safety-related parameters.

Note:

The uncertainties accounted for in the reactor core design shall also cover the uncertainties of the reactor core monitoring.

(3) Systematic and statistical errors of the correlations of physical interdependencies (either in functional or tabular form) shall be determined. These errors shall be introduced into the calculations either directly or in the form of appropriate margins (cf. Section 6).

(4) The uncertainties may be accounted for either by treating them globally or statistically or by combining both design methods.

4.3 Accounting for Uncertainties in the Reactor Core Design

4.3.2 Global treatment of uncertainties

(1) The uncertainties shall be accounted for globally either by choosing conservative initial states and boundary conditions or by choosing conservative calculation models.

(2) In the case of the global treatment of uncertainties by choosing conservative initial states and boundary conditions, the associated verification criteria shall be fulfilled even for the most unfavorable, nevertheless possible, combination of the essential influencing parameters.

Notes:

(1) The procedure specified under para. (2) leads to safety margins that will be all the larger the higher the number of influencing parameters in their most unfavorable combination are accounted for. The influencing parameters are, therefore, selected such that a conservative result is achieved.

(2) If several influencing parameters (initial states and boundary conditions) are available, a conservative result is usually achieved if the 95 % quantile is used for the distribution of the influencing parameters. If the distribution is not known, a technologically well substantiated maximum deviation may be used.

(3) It is permissible to account for uncertainties of influencing parameters by applying conservative calculation models. It shall be shown for the conservative calculation models that the safety-related parameters are conservatively calculated with regard to the associated verification criterion.

4.3.3 Statistical treatment of uncertainties

4.3.3.1 Basics

(1) The application of statistical design methods is an alternative to the method of accounting for the influencing parameters as specified under Section 4.3.1. Based on the uncertainties of the influencing parameters, of the boundary conditions and if the calculation models, these statistical design methods help to determine the uncertainties of the results.

(2) The statistical treatment of uncertainties demands that realistic (best estimate) calculation models are available with regard to the variation of the influencing parameters.

Note:

Realistic (best estimate) calculation models are those that reproduce the mean values of the results gained from experiments.

(3) The statistical treatment of uncertainties may be calculated by varying the influencing parameters (Monte-Carlo simulation method) or by performing error calculations with partial derivatives of the influencing parameters (Gaussian procedure in accordance with DIN 1319-4) or by applying correction terms. In the case of correction terms, these shall be determined by comparing the results from measurements and calculations.

(4) The statistical treatment of uncertainties delivers quantitative data with respect to the effects that uncertainties of relevant influencing parameters have on the results of calculations. The uncertainty range of the results is an indication of whether or not the associated verification criteria are being fulfilled.

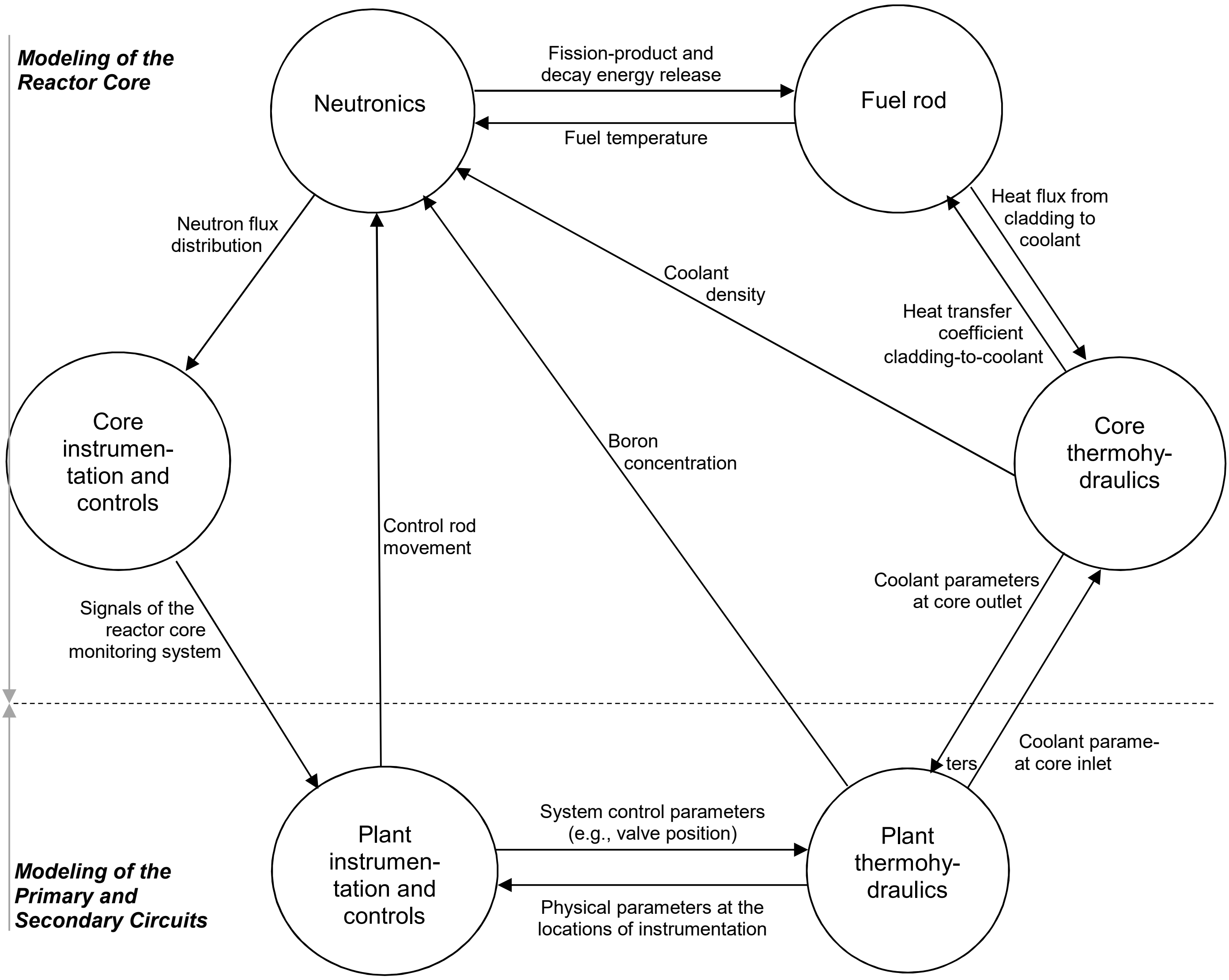


Figure 4-2: Example of a coupled program system

4.3.3.2 Gaussian procedure (root-mean-square method – RMS)

(1) The application of the Gaussian procedure in accordance with DIN 1319-4, Sec. 7.1, demands that the following requirements are fulfilled to a sufficient degree:

a) The input parameters are independent of each other.

b) The variable input parameters are each normally distributed.

c) Within the uncertainty band widths, the output parameters are linearly dependent on the input parameters.

(2) If the expanded Gaussian procedure in accordance with DIN 1319-4, Sec. 7.2, is applied, the input parameters do not need to be independent of each other.

4.3.3.3 Monte-Carlo simulation method

(1) A Monte-Carlo simulation method accounts for the uncertainties of influencing parameters by a random choice from a sufficiently large number of combinations of these influencing parameters.

(2) Depending on the problem in question, a tolerance limit shall be specified and well substantiated.

Note:
Commonly used is the 95 % / 95 % - tolerance limit.

(3) The number of calculations (size of random sample) shall be specified on the basis of the previously specified tolerance limits.

(4) The procedure for quantifying the uncertainties of the output parameters shall be as follows:

a) The influencing parameters whose distribution functions have an essential influence on the results (uncertainties of the input parameters) shall be accounted for within the framework of the Monte-Carlo simulation method. These include:

  aa) Distribution functions of the plant parameters

    aaa) The distributions of the measured fabrication parameters shall be covered by the distribution functions assumed in the analysis.

    aab) Fluctuations of the operational parameters during normal operation (power, coolant throughput, temperature, pressure) shall be quantified by suitable distribution functions. The distribution functions shall be checked on the basis of measurement values for the operational parameters.

aac) Possible cut-off points of the distribution functions shall be well substantiated on the basis of the technological conditions involving the fabrication and operational parameters.

ab) Distribution functions of the calculation model parameters

aba) The distribution functions of the modeling parameters used in the analyses shall be derived in a qualified manner from the validation, e.g., of the measurement values from experiments or of the measurements carried out on nuclear reactors.

abb) Possible cut-off points of the distribution functions shall be well substantiated.

b) It is permissible to replace the distribution functions by conservative global values.

c) Any possible interdependencies of uncertainties of the input parameters specified under item a) shall be quantified.

d) A random sample of value combinations shall be generated in accordance with the distribution functions specified under item a) and, if applicable, with the interdependencies under item c). In this context, all input parameters connected with uncertainties shall be varied simultaneously.

e) Each value combination from the random sample created as specified under item d) shall be subjected to an individual calculation run. The final result is a random sample of calculation results from which the calculation uncertainty can be determined.

4.3.4 Special aspects of the core design regarding Safety Level 4a

(1) The analysis of transients involving a possible failure of the emergency shutdown system shall basically be performed with realistic (so-called “best estimate”) calculation models. This means, in particular, that it is permissible

a) to use realistic values for the realistic initial states and boundary conditions,

b) to assume that all measures and devices are available that are not considered to have failed on account of the presumed event, and

c) to include alterations of the operational parameters and conditions by control processes in the modelling.

(2) The analysis of transients involving a possible failure of the emergency shutdown system shall, in addition to para. (1), take the following into account:

a) The initial state to be assumed shall be the stable state of full operational power at xenon equilibrium for the most unfavorable point in time of the fuel cycle.

   Note:
   The state of full operational power and xenon equilibrium is the most unfavorable initial state.

b) If a shutdown of the coolant pumps may be assumed within the short-term range (i.e., in the time range before reaching the pressure maximum) then the corresponding actuation controls shall be designed to comply with Category 1 or 2 in accordance with RSK Guidelines, Sec. 7.3.

4.4 Simplifications and Approximations

Simplifications and approximation within the calculation models and procedures are permissible, e.g., the agglomeration of fuel rods as fuel rod groups.

Note:
The permissibility of the simplifications is verified by checking the validity and accuracy of the calculation systems.

4.5 Checking the Validity and Accuracy

4.5.1 Basics

(1) The applied calculation systems shall be validated and verified. They shall be documented in accordance with safety standard KTA 3101.2, Sec. 7.4.

Note:
The terms verification and validation are defined in safety standard KTA 3101.2.

(2) The validation procedure is dependent on the accuracy required for the results.

(3) There are two forms of validation, the validation of the entire calculation system for the overall scope of application (integral validation) and the validation of individual components of the calculation system (partial validation). In addition to the integral validation of the calculation system, the scope of application should be verified by a partial validation of the individual components.

Note:
The partial and integral validations supplement each other and are usually combined. Applying an integral validation procedure alone would not prevent possible error compensations. Thus, a smaller range of extrapolation within the scope of application would result. On the other hand, applying partial validation procedures alone would make it difficult to verify the overall calculation system by these individual validation steps.

(4) The results of the numerical program codes shall be comprehensible and, as far as possible, shall have been compared to the results of experiments, plant transients or to the results of previously validated numerical program codes.

(5) In the course of validation, the systematic deviations and statistical uncertainties of the respective calculation system shall be determined. The verified systematic deviations may be compensated for by applying corresponding correction factors to the results.

Note:
Regarding the determination of uncertainties, cf. Section 4.3.

4.5.2 Validation procedure

(1) The calculation systems shall be validated by comparing the calculated results with the results from

a) operational measurements (e.g., startup measurements, regular measurements during operation, special measurements),

b) experiments,

c) evaluations of actual transients, or

d) other calculation systems (bench marks or recognized reference solutions).

(2) The measurement results of para. (1), items a) and b), shall normally cover the entire operating range of the reactor plant with regard to the essential parameters. In those cases where the original reactor conditions have not been properly modeled, the transfer of the experimental data to actual reactor conditions shall be well substantiated.

(3) The measurement results shall be selected taking primarily the following criteria into account:

a) documentation of the measurements,

b) quality of the measurements and error analysis, and

c) transferability of the measurement conditions to the scope of application of the calculation system required for the design.

(4) When applying correlations and tables in calculation systems, the requirements specified under Section 6 shall be fulfilled.

4.5.3 Safety levels

(1) The validation of the calculation systems used for the verifications at Safety Levels 1 and 2 shall normally be primarily based on measurement results specified under Section 4.5.2, para. (1), items a) and b). Insofar as possible, the actual transients (cf. Section 4.5.2, para. (1), item c)) shall be included in the validation procedure.

(2) The validation of the calculation systems used for the verifications at Safety Level 3 shall be based on measurement results specified under Section 4.5.2, para. (1), items b), c) and d).

(3) As far as possible, the same models shall be applied at Safety Level 4a that have been applied at Safety Levels 1, 2 and 3 and have been validated for partial aspects of the occurring physical procedures (partial validation). If this is not possible, the models shall be constructed based on the current state of knowledge and shall, separately, be well substantiated.

5 Special Requirements Regarding the Thermo-Hydraulic Design of Reactor Cores

5.1 Stability of the Boiling Water Reactor

(1) Measures shall be taken in the thermo-hydraulic design of a BWR reactor core that will ensure that a sufficient margin is maintained during normal operation to the range where undampened power density oscillations could occur.

Note:
In heated closed channels in which the coolant boils and is therefore present in two phases, a high power and low coolant flow may lead to thermo-hydraulic instabilities depending on the ratio of the one-phase to the two-phase pressure loss. This results in a cyclic change of boiling length and coolant throughput. In a BWR, whose fuel assemblies form parallel, heated and closed channels, any in-phase thermo-hydraulic oscillation may, due to the neutron-physical feedback, lead to global or regional oscillations of the neutron flux and, therefore, of the power density. These oscillations may develop in a specific range of the coolant-flow-vs.-power diagram (operating characteristic). This range depends on the core configuration (especially, on the one-phase and two-phase pressure losses in the fuel assemblies) and on the operating condition (e.g., control rod position, xenon distribution, burnup).

(2) Suitable measures shall be provided to ensure that the respective safety-related requirements specified under Section 3 are fulfilled even with regard to events at higher safety levels with possibly occurring undamped power density oscillations.

(3) The thermo-hydraulic stability of the reactor core during normal operation shall be verified either by analysis with validated numerical program codes or by direct measurements.

Note:
The validation of a numerical program code is always dependent on the corresponding scope of application. Any significant alteration of core components can be enough of an influence on the scope of application of a numerical program code to require expanding the validation, possibly, even by a measurement of the stability behavior.

5.2 Thermo-Hydraulic Compatibility of Core Component

The core components shall be designed such that, with regard to their hydraulic resistance, no flow redistribution will occur within the core that might prevent a safe heat removal or might lead to impermissible mechanical loads.

Note:
When applying various types of fuel assemblies (e.g., mixed core) it is important to observe that they are thermo-hydraulically compatible.

5.3 Initial State Power Distribution

(1) The initial state power distribution shall be determined for all events at Safety Levels 2, 3 and 4a that must be assumed to occur. In this context, that initial state of Safety Level 1 shall be taken as the basis that is most unfavorable for the analysis. Furthermore, this analysis shallc be based either

a) on three-dimensional power density distributions, or

b) on simplified radial power form factors including their respective axial distributions.

(2) The following influencing factors shall normally be taken into account when determining the initial state power distribution:

a) power density limitation,

b) fuel enrichment,

c) burn-up,

d) temperature within the fuel

e) xenon,

f) pressure, temperature and voids in the coolant,

g) control assemblies,

h) other kinds of neutron absorbers, and

i) structural materials.

5.4 Coolant Throughput Distribution in the Reactor Core

5.4.1 Basics

(1) The coolant throughput distribution prior to core entry, the coolant throughput through the reactor core, the reactor core geometry, different hydraulic resistances, local variations of the heating and mixing of the coolant in the reactor core shall all be determined together with their effect on the coolant throughput distribution in the reactor core.

Note:
The unevenness of the coolant supply at the inlet nozzles (PWR) or those caused by the recirculation pumps (BWR) is evened out to a large extent by hydraulic equalizing sections and series resistances upstream of the core inlet. However, individual reactor pressure vessel internals (for instance lower core grid for BWR) can result in adverse effects, for example through vortex formation and non-stationary flow conditions. An evening-out of the flow within the reactor core depends on the structural design of the reactor core, e.g., an open reactor core geometry or closed parallel coolant channels, and on the subsequent hydraulic resistances in the reactor core, e.g., coolant inlet flow limiters, rod friction, spacer grids.

(2) The local coolant temperatures and steam qualities within a fuel assembly are also influenced by the unevenness of heating due to the nuclear power density distribution, and by the possibly increased flow turbulence caused by the spacer design. It shall be taken into account that the local coolant temperatures and steam qualities in the fuel assembly will affect the local coolant throughputs via acceleration, friction and cross flow effects.

5.4.2 Coolant throughput distribution prior to core entry

The coolant throughput distribution prior to core entry shall be determined taking all relevant hydraulic series resistances and potential influences from the reactor pressure vessel internals into account.

5.4.3 Core coolant throughput and coolant bypass

(1) The coolant throughput through the reactor pressure vessel shall be broken down into the portion that actively contributes to cooling the fuel assemblies (core coolant throughput) and that bypasses the core. The coolant bypass flows through design-related gaps (structural clearances) or openings in the reactor pressure vessel internals, particularly, through the

holes in the fuel assembly components, and serves to cool the internals of the reactor pressure vessel and reactor core. In this context, particular attention shall be paid to the temperature dependency of the effects that structural clearances have on the coolant bypass.

(2) The distribution of the core coolant throughput shall be evaluated with regard to the minimum, average and maximum coolant throughputs through the coolant channels. The respective variables of state shall be specified together with the coolant throughput.

(3) Suitable bypasses shall be provided in the thermo-hydraulic design of the reactor core that, in addition to the cooling of the fuel assemblies, will ensure the necessary cooling of the other core components (e.g., instrumentation lance, control assemblies, neutron sources).

5.4.4 Coolant throughput through fuel assemblies or fuel rod clusters

In addition to the reactor core geometry, the analysis of the coolant throughput through individual fuel assemblies or fuel rod clusters shall take into consideration the hydraulic feedback effects stemming from upstream or downstream internals of the reactor pressure vessel and reactor core.

Note:
These effects stem from the coolant supply to the fuel assembly, the dimensioning of the coolant inlet flow limiter, the frictional and distributing effects of the fuel rods, the spacers, the fuel channels if applicable, as well as the coolant outflow behavior at the fuel assembly outlets.

5.4.5 Coolant displacement due to unequal heating

The effects on the local coolant throughput due to a spatially varying coolant density (including steam quality) shall be accounted for, globally, for various regions in the reactor core and, locally, for the individual fuel assembly.

5.4.6 Cross circulation of the coolant

If it is assumed in the analysis that cross circulation of the coolant will reduce a locally increased heating in the fuel assembly, it is necessary to verify this assumption by experimental data.

5.5 Pressure Differences In The Reactor Core

(1) The pressure differences within the reactor core shall be determined taking the following effects into account:

a) fuel assembly bottom end piece (considering the coolant inlet orifice, if any),
b) friction,
c) change in velocity,
d) geodetic height differences,
e) spacers as well as further components within the fuel assembly, and
f) fuel assembly head piece.

(2) The coolant throughputs through the fuel assembly associated with the pressure differences and the coolant conditions shall also be specified.

(3) If water/steam mixtures must be assumed as flowing through the reactor core internals, the changed pressure differences – with regard to those of single phase flow – shall be treated analogously.

(4) The requirements specified under Section 5.4 shall be applied to coolant throughputs, and those specified under Section 6 to the treatment of experimental data.

5.6 Resulting Forces Inside the Reactor Core

(1) In case of PWR, the upward flow forces that must be considered in the mechanical design of the fuel assembly hold-down equipment shall be determined.

(2) In case of BWR, it shall be verified that at Safety Level 1 the forces resulting from buoyancy and flow do not exceed the weight of the fuel assemblies or, if applicable, of the partial fuel element bundles.

Note:
Certain BWR fuel assembly constructions have free-standing partial fuel element bundles.

(3) The mechanical design of the BWR fuel assembly casing demands that the pressure differences through the casing wall be determined.

5.7 Heat Transfer to the Coolant

5.7.1 Extent of analysis

(1) The verification that the associated safety-related requirements are met shall extend to, at least, the most heavily loaded fuel assemblies or the most heavily loaded fuel rod clusters in the reactor core.

(2) The necessary analyses for this verification require the determination of the power density distribution, of the coolant throughput through the reactor core and of the pressure differences, and are also – as shown in **Figure 5-1** – influenced by:

a) operational fluctuations of process variables,
b) measurement tolerances of process variable,
c) fabrication tolerances,
d) tolerances of the numerical methods.

(3) The numerical methods used for the thermal design shall normally cover all technical aspects indicated in **Figure 5-1**.

5.7.2 Operational fluctuations and measurement tolerances of process variables

The operational fluctuations of process variables, together with their associated measurement tolerances, shall be taken into account.

5.7.3 Fabrication tolerances

Fabrication tolerances shall be accounted for either in the form of input values of the numerical methods or as safety margins added to the results. The following influencing factors shall, in particular, be taken into account:

a) geometrical tolerances in the reactor core, especially those of the fuel assemblies and fuel assembly casings, and
b) tolerances with respect to fuel density and enrichment or isotope composition.

5.7.4 Tolerances of the numerical methods and computer codes

The tolerances of the numerical methods and computer codes used shall be substantiated (cf. Sections 4.5 and 6).

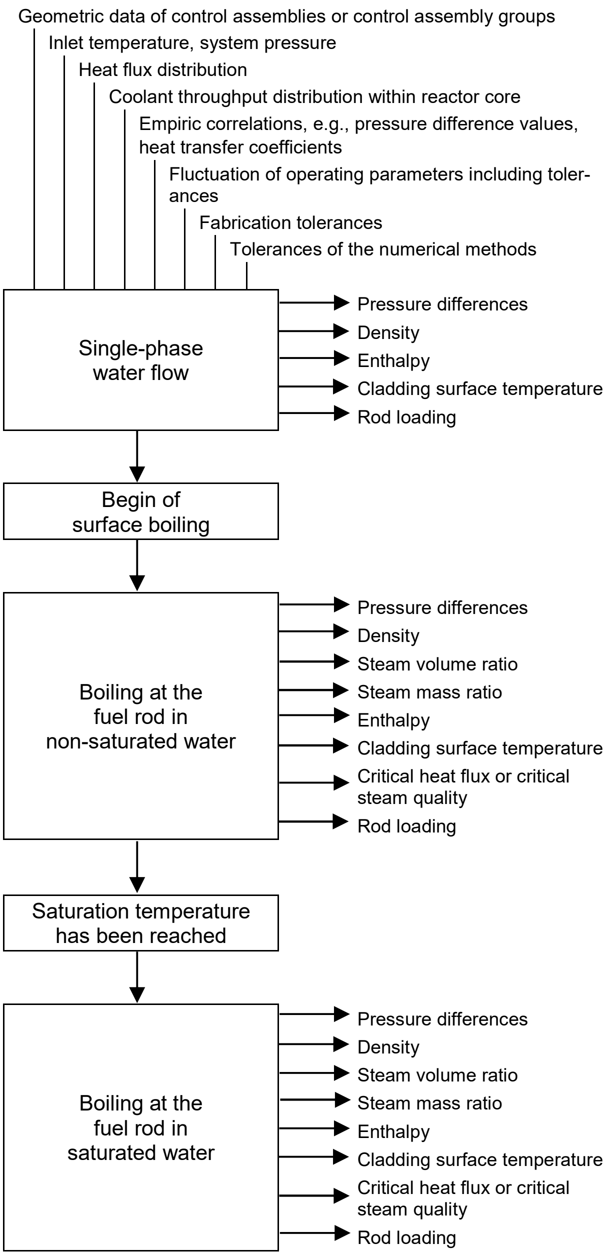


Figure 5-1: Requirements regarding treatment of the heat transfer to the coolant in the reactor cores of pressurized water and boiling water reactors

5.8 Adjacent Systems and Components

5.8.1 Total coolant throughput and design of the coolant pumps

(1) The design of the coolant pumps shall be based on a total coolant throughput comprising the coolant throughput through the reactor core as specified in Section 5.4.3 plus the bypass throughput resulting from the structural design and the cooling of the reactor pressure vessel internals.

(2) The design of the coolant pumps shall be based on the pump head to be calculated from the core pressure differences specified in Section 5.5 and the pressure differences of the other components along the coolant flow path.

(3) The moments of inertia of the pump impellers, including all necessary additional moments of inertia, shall be dimensioned such that, during coast down of the coolant pumps, the requirements specified under Section 5.8.2, paras. (5) and (6), are met.

5.8.2 Protection of the reactor core against impermissible operating conditions

(1) It is assumed for the thermo-hydraulic core design that all relevant operating parameters are maintained within specified ranges. The design of the systems for protecting the reactor core from impermissible operating conditions shall ensure that the coordination between controls (manual or automatic), limitations and reactor protection will maintain these operating parameters within the ranges on which the safety-related analyses were based.

Note:
The respective ranges are determined by the thermo-hydraulic design, by the overall plant design, and by the capabilities of the instrumentation and control system.

(2) The following operating parameters are important to the thermo-hydraulic design:

a) coolant pressure,
b) coolant temperature,
c) coolant throughput,
d) integral thermal power of the reactor,
e) control assembly positions,

Note:
The insertion depths of the control rods are important boundary conditions for the PWR, especially with regard to the analyses regarding control assembly ejection and erroneous control assembly group withdrawal.

f) power density distribution, and
g) DNBR (PWR) or MASL (BWR) – see **Figure 5-2**.

(3) The values of these operating parameters shall be determined from suitable measurements or derived from measurement values of validated models. The associated uncertainties shall be taken into account.

Note:
Uncertainties are caused, e.g., by measurement tolerances, by the extent to which the measurements of process variables are performed and by the accuracy of the empiric correlations, methods, measurement procedures and fabrication tolerances used for deriving the respective values from models.

(4) During normal operation, the values of these operating parameters shall be limited such that the margins required to meet the safety-related requirements at Safety Levels 2, 3 and 4a are maintained.

(5) The following requirements apply to the limitation of DNBR as specified under para. (4):

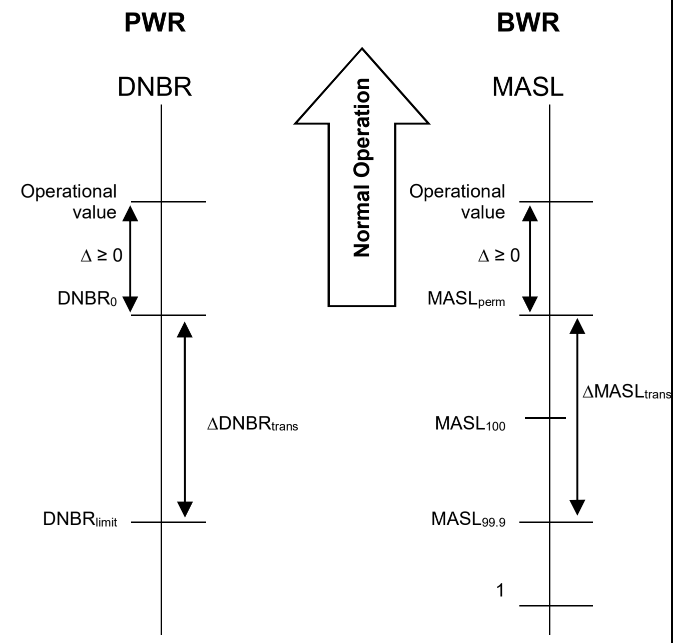
a) The minimum value of the DNB ratio permissible during normal operation (i.e. $DNBR_0$) at which the safety-related requirements at Safety Levels 2, 3 and 4a are fulfilled shall be determined. In this context, the uncertainties at Safety Levels 2 and 3 shall be taken into account (cf. Section 4.3). The $DNBR_0$ may be determined generically (encompassing all cycles) or specifically for the individual cycle (advance computation or by process computer – cf. item b)).

b) If the value of $DNBR_0$ is determined by the process computer on the basis of the actual variables of state and a subsequent simulation of the limiting transient or transients, the margins used to account for uncertainties shall, in particular, take deviations of the variables of state into account that are due to changes of normal operation between the points in time at which these calculations are performed.

c) It shall be ensured by proper surveillance and control measures that the minimum value of the DNB ratio permissible during normal operation is maintained at all times. These measures include:

ca) Setting fixed limit values for the power density, cb) Dynamic limitations of the power density (e.g., via a DNB computing module) or

cc) Administrative surveillance of the minimum DNB ratio determined by the process computer.



DNBR operation ≥ $DNBR_0$ = $DNBR_{limit}$ + $\Delta DNBR_{trans}$

$DNBR_0$: minimum permissible DNBR during normal operation

MASL operation ≥ $MASL_{perm}$ = max($MASL_{100}$,$MASL_{99.9}$+$\Delta MASL_{trans}$)

$MASL_{perm}$: minimum permissible MASL during normal operation

N o t e s :

(1) This figure only accounts for those cases where the limit values for DNBR or $MASL_{99.9}$, respectively, are the verification goal (Safety Levels 1 and 2). Meeting the verification goal in the analyses for Safety Levels 3 and 4a may require accepting higher values than $DNBR_0$ or $MASL_{perm}$.

(2) The respective DNBR and MASL values are dependent on the actual operating conditions, and in particular, on the type of fuel assemblies and on the power distribution. With regard to BWR, special attention must be paid to the fact that the coolant throughput is regulated during operation and that the values of MASL are dependent of this throughput.

Figure 5-2: Schematic of the relationships between the various values of DNBR and MASL during normal and abnormal operation

(6) The limitation of MASL as specified under para. (4) shall meet the following requirements:

a) The MASL value permissible during normal operation (i.e., $MASL_{perm}$) shall be the maximum value of

aa) $MASL_{100}$ and

ab) those MASL values for which the safety-related requirements at Safety Levels 2, 3 and 4a are fulfilled.

In this context, the uncertainties at Safety Levels 1, 2 and 3 shall be taken into account (cf. Section 4.3). The $MASL_{perm}$ may be determined generically (encompassing all cycles) or specifically for the individual cycle (advance computation or by process computer – cf. item b)). If necessary, the values shall be differentiated according to the type of fuel assembly.

b) If the value of $MASL_{perm}$ is determined by the process computer on the basis of the actual variables of state and a subsequent simulation of the limiting transient or transients, the margins used to account for uncertainties shall, in particular, take deviations of the variables of state into account that are due to changes of normal operation between the points in time at which these calculations are performed.

c) It shall be ensured by proper surveillance and control measures that the MASL value permissible during normal operation is maintained at all times. These measures include:

ca) Setting fixed limit values for the power density,

cb) Dynamic limitations of the power density (e.g., via an MASL computing module) or

cc) Administrative surveillance of the actual MASL value determined by the process computer.

(7) The safety analyses provide certain guidelines for setting the limit values of process variables and (if applicable) of the protective limitations as well as of the reactor protection. If the mitigation of a transient at Safety Level 2 depends on the function of a protective limitation, the effectiveness of the protective limitation shall be ensured by adjusting its set point to react sufficiently in advance of the reactor protection set point.

6 Requirements Regarding Empirical Correlations

6.1 Basics

(1) Empirical correlations establish the relationship between physical input and output parameters derived from physical experiments. These correlations may be in the form of mathematical functions or tables.

(2) In conjunction with the thermo-hydraulic design of reactor cores, empirical correlations are used in particular for the calculation of the following physical parameters:

a) Critical heat flux,

b) Critical steam quality (BWR),

c) Steam bubble content,

d) Pressure loss due to friction (one- and two-phase flow),

e) Material properties (e.g., water/steam table), and

f) Heat transfer coefficients.

(3) The creation of an empirical correlation is comprised of the following steps:

a) Performing the experiment (setup, execution, data acquisition, data analysis); cf. Section 6.2,

b) Development of the correlation (mathematical function or table); cf. Section 6.3,

c) Specifying the scope of application; cf. Section 6.4,

d) Validation; cf. Section 6.5.

(4) Each step shall be properly documented.

6.2 Performing the Experiment

(1) Basis for any correlation are physical experiments.

(2) The physical experiments shall basically be planned such that the actual conditions inside the reactor are adequately reproduced. In those cases, where these actual conditions are not reproduced, the transferability of the results shall be well substantiated.

N o t e s :

(1) Not every method applied to the thermo-hydraulic design of reactor cores can be validated in the respective reactor plant. That is

why a number of problems either have to be explored in experiments that adequately reproduce the reactor conditions or have to be solved by transferring experimental results described in literature to the conditions of the respective reactor core.

(2) The experimental verification of the applicability of the methods used can be performed at the following verification levels depending on the individual problem area:

a) Simple models that have been publicized as having been verified by experimental data.
b) Measurements of the system behavior by experiments performed on original components or on life-size or small-scale replicates of the original components.
c) Measurements of process variables in the respective reactor plant.

(3) All measurement values shall be checked with regard to their consistency and quality.

6.3 Development of the Correlation

(1) An adequate functional form shall be chosen for the correlation of the measurement data.

Note:

The usual approach is one of the following:

a) The basic functional relationship is derived from physical considerations; the coefficients are determined from the measurement data.
b) A mathematical function is chosen that replicates the measurement data with as few parameters as possible.
c) An interpolation curve is adapted that is based on tabular data.

(2) Based on the chosen form for the correlation, the possibly necessary coefficients shall be determined.

Note:

Usually, the coefficients are determined by value minimization procedures.

(3) The uncertainties of the correlations (mathematical functions or tables) for the physical relationships shall be determined. These uncertainties shall be accounted for in the design either directly or in the form of safety margins.

Note:

Uncertainties can also be caused by systematic errors (of the measurement values) and by scaling effects (if the experiments were not performed on full scale components).

6.4 Specifying the Scope of Application

The scope of application of the respective correlation shall be determined on the basis of the underlying measurement data. Extrapolations are permissible, provided, the uncertainty values for the extrapolated ranges are well substantiated.

6.5 Validation

(1) All empirical correlations shall be validated. The validation procedure shall demonstrate the robustness of the correlation within its scope of application.

(2) In those cases where the assumed functionality of the empirical correlation was not derived from underlying physical laws, the validation of the correlation shall be performed based on independent data, i.e., these same data shall not have been used in the development of the correlations.

Note:

This can be achieved. e.g., by a prior allocation of the experimental data. Possible criteria for this allocation may be a random selection from the available data or creating new data by performing additional experiments with a comparable test setup.

(3) Suitable statistical methods shall be applied to verify that the correlation replicates the validation data with sufficient accuracy or, at least, conservatively.

Note:

For example, the methods in accordance with DIN ISO 5479 can be applied to check whether or not the measurement data are normally distributed around the correlation. In this case the t-distribution may be applied. In the case of other distribution, other methods can be applied.

Appendix A

Representative Events Regarding the Design of Reactor Cores

A.1 Representative Events Regarding the Design of PWR Reactor Cores		
No.	Event (Condition)	Remarks
	Safety Level 1 (Normal Operation)	
D1	**Normal Operation** (Power operation, cycling operation, startup and shutdown, cooling during shutdown, refueling, fuel store cooling, maintenance situations, pressure tests)	Normal operation is included here because it defines the initial conditions for all events.
	Safety Level 2 (Abnormal Operation)	
D2.1	**Reduced Heat Removal by the Main-Steam and Feedwater System**	
D2.1.1	Load shedding to auxiliary power	Relevant for the requirements of the limitation devices
D2.1.2	Turbine trip without opening of the turbine bypass (e.g., due to loss of the condenser vacuum)	
D2.1.3	Unintentional closing of individual main-steam isolation valves	
D2.1.4	Emergency power condition (lasting for a short time, ≤ 10 hours)	
D2.1.5	Failure of a main feedwater pump	Relevant for power limitations, covered by D.21.1.
D2.2	**Reduction of the Coolant Throughput in the Reactor Coolant System**	
D2.2.1	Failure of all reactor coolant pumps	Covered by D2.1.4.
D2.2.2	Failure of one reactor coolant pump	
D2.3	**Faulty Change of the Reactivity and of the Power Distribution**	
D2.3.1	Erroneous withdrawal of control assemblies or control assembly groups	
D2.3.2	Coldwater injection into the reactor coolant system from directly connected systems (e.g., circumventing the recuperative heat exchanger of the volume control system)	
D2.3.3	Worst fuel loading error involving the fuel assembly with the highest reactivity	Only relevant during refueling (not during startup).
D2.4	**Leakage of Primary Coolant / Reduction of Coolant Inventory**	
D2.4.1	Erroneous opening of a pressurizer relief valve	Limiting pressure gradient regarding the DNB-design.
	Safety Level 3 (Design Basis Accidents)	
D3.1	**Increased Heat Removal by the Main-Steam and Feedwater System**	
D3.1.1	Leakage or rupture of main-steam pipe inside the containment vessel	Representative event regarding re-criticality.
D3.2	**Reduced Heat Removal by the Main-Steam and Feedwater System**	
D3.2.1	Failure of all plant-operational feedwater supplies	Representative event regarding requirements for the emergency feedwater supply.
D3.2.2	Reduction of the core coolant flow rate due to fracture of a main coolant pump shaft or due to freezing up of a main coolant pump	

D3.3	**Faulty Change of the Reactivity and Power Distribution**	
D3.3.1	Erroneous withdrawal of control assemblies or control assembly groups upon failure of the first response level of the load limiting device	
D3.3.2	Unintentional reactivity insertion (e.g., injection of demineralized water in conjunction with a failure of limitations and superordinate safety measures)	
D3.3.3	Ejection of the most effective control assembly	
D3.4	**Leakage of Primary Coolant / Reduction of Coolant Inventory**	
D3.4.1	Small leak inside the containment vessel (pipes of the pressure retaining boundary, small crack openings, open pressure relief trains, reflux condenser)	In case of the reflux condenser, reactivity insertion is caused by input of demineralized water.
D3.4.2	Medium and large leak in the coolant pipes of the pressure retaining boundary (dependent on the rupture preclusion quality [0.1 A, 2 A] and on the verification goal)	Certain verification goals require the assumption of a 2 A (double-ended) rupture.
D3.4.3	Rupture of a steam generator tube (lasting for a short time, ≤ 2 A) with steam release through the roof	
D3.4.4	Leak of the emergency cooling system at any place outside the containment vessel within the reactor building annulus during emergency cooling operation	
D3.5	**External Events**	
D3.5.1	Earthquakes (including sequential failures)	
	Safety Level 4a (Special, Very Seldom Events)	
D4.1	**Operational Transients with an Assumed Failure of the Reactor Trip System (ATWS)**	
D4.1.1	Failure of the ultimate heat sink (e.g., due to loss of the condenser vacuum or to closing of the main steam valves) with the station service power supply in working order	Closing of the main steam valves also involves closing of the main steam bypass valves.
D4.1.2	Failure of the ultimate heat sink in conjunction with a failed station service power supply	
D4.1.3	Maximum increase of steam removal (e.g., due to opening of individual main steam bypass valves or main steam safety valves)	An erroneous opening of all main steam valves or of all safety valves is not considered as being an operational transient.
D4.1.4	Failure of the entire feedwater supply	
D4.1.5	Maximum reduction of the core coolant flow rate	
D4.1.6	Maximum increase of reactivity due to withdrawal of control assemblies or control assembly groups starting out from the operational states “Full Power” and “Hot Subcritical”	Error of the operating controls.
D4.1.7	Pressure relief due to an unintentional opening of the pressurizer safety valve	
D4.1.8	Maximum drop of the core inlet temperature due to a failure of an active component of the feedwater supply system	
D4.2	**External Events**	
D4.2.1	Airplane crash	
D4.2.2	Plant-external explosion, plant-external fire	

A.2 Representative Events Regarding the Design of BWR Reactor Cores		
No.	Event (Condition)	Remarks
	Safety Level 1 (Normal Operation)	
S1	**Normal Operation** (Power operation, cycling operation, startup and shutdown, cooling during shutdown, refueling, fuel store cooling, maintenance situations, pressure tests)	Normal operation is included here because it defines the initial conditions for all events.
	Safety Level 2 (Abnormal Operation)	
S2.1	**Increase of Reactor Pressure**	
S2.1.1	Turbine trip without opening of the turbine bypass (e.g., due to loss of the condenser vacuum)	
S2.1.2	Unintentional closure of all penetrating-pipe valves	
S2.2	**Increase of the Coolant Throughput in the Reactor Coolant System**	
S2.2.1	Malfunction of the controls for increasing the coolant throughput in the reactor coolant system	
S2.3	**Reduction of the Coolant Throughput in the Reactor Coolant System**	
S2.3.1	Failure of a number or all of the recirculation pumps	The stability of the final state shall be maintained.
S2.4	**Reduced Heat Removal by the Main-Steam and Feedwater System**	
S2.4.1	Emergency power condition (lasting for a short time, ≤ 10 hours)	Covered by S2.1.1.
S2.4.2	Failure of one or all main feedwater pumps	
S2.5	**Increase of the Coolant Inventory of the Reactor Coolant System**	
S2.5.1	Malfunction of the feedwater controls leading to an increase of the feedwater throughput	
S2.6	**Faulty Change of the Reactivity and of the Power Distribution**	
S2.6.1	Unintentional withdrawal of control assemblies or control assembly groups	
S2.6.2	Coldwater injection into the reactor coolant system from directly connected systems (e.g., erroneous injection from the water make-up systems or failure of the high-pressure feedwater heater)	
S2.6.3	Unintentional insertion of all control assemblies at high power	
S2.6.4	Worst fuel loading error involving the fuel assembly with the highest reactivity	Only relevant during refueling (not during startup).
	Safety Level 3 (Design Basis Accidents)	
S3.1	**Increase of Reactor Pressure**	
S3.1.1	Turbine trip without opening of the turbine bypass (e.g., due to loss of the condenser vacuum) in conjunction with a failure of the first response level of the reactor protection system	
S3.1.2	Unintentional closure of all penetrating-pipe valves in conjunction with a failure of the first response level of the reactor protection system	
S3.2	**Reduction of the Coolant Throughput in the Reactor Coolant System**	
S3.2.1	Malfunction of the controls for increasing the coolant throughput in the reactor coolant system in conjunction with a failure of the first response level of the reactor protection system	
S3.3	**Faulty Change of the Reactivity and of the Power Distribution**	

S3.3.1	Unintentional withdrawal of control assemblies or control assembly groups in conjunction with a failure of the first response level of the reactor protection system	
S3.3.2	Coldwater injection into the reactor coolant system from directly connected systems (e.g., erroneous injection from the water make-up systems or failure of the high-pressure feedwater heater) in conjunction with a failure of the first response level of the reactor protection system	
S3.3.3	Drop-out of the most effective control assembly	The maximum drop length is limited by the latch mechanism.
S3.4	**Leakage of Primary Coolant / Reduction of Coolant Inventory**	
S3.4.1	Small leak inside the containment vessel (pipes of the pressure retaining boundary, small crack openings)	
S3.4.2	Medium and large leak in the coolant pipes of the pressure retaining boundary (depending on the rupture preclusion quality [0.1 A, 2 A] and on the verification goal)	Certain verification goals require the assumption of a 2 A (double-ended) rupture.
S3.4.3	80 cm² leak in the base plate of the reactor pressure vessel	
S3.5	**External Events**	
S3.5.1	Earthquakes (including sequential failures)	
	Safety Level 4a (Special, Very Seldom Events)	
S4.1	**Operational Transients with an Assumed Failure of the Reactor Trip System (ATWS)**	Regarding ATWS, it is assumed that the counter-nut backup of the control assemblies is effective.
S4.1.1	Failure of the ultimate heat sink (e.g., due to loss of the condenser vacuum or to closing of the main steam valves) with the station service power supply in working order	
S4.1.2	Failure of the ultimate heat sink in conjunction with a failed station service power supply	
S4.1.3	Maximum increase of steam removal (e.g., due to opening of the steam bypass or of the main steam station or the safety and relief valves)	
S4.1.4	Failure of the entire feedwater supply	
S4.1.5	Maximum increase of reactivity due to withdrawal of control assemblies or control assembly groups starting out from the operational states “Full Power” and “Hot Subcritical”	
S4.1.6	Maximum drop of the feedwater temperature	
S4.1.7	Closure of pipe penetrations with the station service power supply in working order	
S4.1.8	Closure of pipe penetrations in conjunction with a failed station service power supply	
S4.1.9	Maximum increase of feedwater throughput	
S4.1.10	Startup of coolant recirculation pumps at the maximum control response	
S4.2	**External Events**	
S4.2.1	Airplane crash	
S4.2.2	Plant-external explosion, plant-external fire	

Appendix B

Regulations Referred to in this Safety Standard

(Regulations referred to in this safety standard are valid only in the versions cited below. Regulations which are referred to within these regulations are valid only in the version that was valid when the latter regulations were established or issued.)

AtG		Act on the Peaceful Utilization of Atomic Energy and the Protection against its Hazards (Atomic Energy Act) Atomic Energy Act in the version promulgated on July 15, 1985 (BGBl. I, p. 1565), most recently changed by article 1 of the act dated December 4, 2022 (BGBl. I, p. 2153)
StrlSchG		Act on the Protection against the Harmful Effect of Ionising Radiation (Radiation Protection Act - StrlSchG) Radiation Protection Act of June 27, 2017 (BGBl. I, p. 1966), most recently changed by the promulgation of January 3, 2022 (BGBl. I, p. 15)
StrlSchV		Ordinance on the Protection against the Harmful Effects of Ionising Radiation (Radiation Protection Ordinance - StrlSchV) Radiation Protection Ordinance of November 29, 2018 (BGBl. I, p. 2034, 2036), most recently changed by article 1 of the ordinance dated October, 2021 (BGBl. I p. 4645)
SiAnf	(2015-03)	Safety Requirements for Nuclear Power Plants (SiAnf) of November 22, 2012, amended version of March 3, 2015 (BAnz AT 30.03.2015 B2), most recently changed as promulgated by BMUV on February 25, 2022 (BAnz AT 15.03.2022 B3)
Interpret of SiAnf	(2015-03)	Interpretations of the safety requirements for nuclear power plants of November 22, 2012, of November 29, 2013 (BAnz AT 10.12.2013 B4), changed on March 3, 2015 (BAnz AT of March 30, 2015 B3)
KTA 3101.2	(2012-12)	Design of reactor cores of pressurized water and boiling water reactors; Part 2: Neutron-physical requirements for the design and operation of the reactor core and adjacent systems
KTA 3101.3	(2022-11)	Design of reactor cores of pressurized water and boiling water reactors; Part 3: Mechanical and thermal design
KTA 3301	(2015-11)	Residual heat removal systems of light water reactors
KTA 3303	(2015-11)	Heat removal systems for fuel assembly storage pools in nuclear power plants with light water reactors
DIN 1319-4	(1999-02)	Fundamentals of metrology - Part 4: Evaluation of measurements; uncertainty of measurement
DIN ISO 5479	(2004-01)	Statistical interpretation of data - Tests for departure from the normal distribution (ISO 5479:1997)