Safety Standards

of the
Nuclear Safety Standards Commission (KTA)

KTA 3101.2 (2012-11)

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

(Auslegung der Reaktorkerne von Druck- und Siedewasserreaktoren; Teil 2: Neutronenphysikalische Anforderungen an Auslegung und Betrieb des Reaktorkerns und der angrenzenden Systeme)

The previous version of this safety standard was issued in 1987-12

If there is any doubt regarding the information contained in this translation, the German wording shall apply.

Editor:
KTA-Geschaeftsstelle c/o Bundesamt fuer Strahlenschutz (BfS)
Willy-Brandt-Str. 5 • 38226 Salzgitter • Germany
Telephone +49(0)30-18333-(0) • Telefax +49(0)30-18333-1625

KTA SAFETY STANDARD

November 2012	**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 3101.2

Previous versions of this safety standard: 1987-12 (BAnz No. 44a of March 4, 1988)

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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 Commission (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 AtG and the Radiological Protection Ordinance (StrlSchV) and further detailed in the "Safety Criteria" and in the "Design Basis Accident Guidelines".

(2) In order to achieve these protective goals, a reactor plant is designed and operated in such a way that any superordinate safety-related requirements specified for the reactor core can be fulfilled. These requirements include the capabilities for shutdown, residual heat removal and retention of activity.

(3) The fulfillment of these superordinate requirements, among other things, is verified by safety-related analyses. These analyses are carried out for stationary conditions during normal operation, on postulated event sequences during specified normal operation (i.e., normal operation and abnormal operation) and for design basis accidents; these analyses are generally assigned to various analysis areas such as nuclear reactor core design, thermo-hydraulic reactor core design and thermo-mechanical reactor core design.

(4) The safety standards series KTA 3101 is comprised of the following three parts:

Part 1: Principles of the thermo-hydraulic design,

Part 2: Neutron-physical requirements for the design and operation of the reactor core and adjacent systems (*the present safety standard*),

Part 3: Mechanical and thermal design (*in preparation*).

The present Part 2 of safety standards series KTA 3101 deals with those precautionary measures referred to under para. (1) for nuclear power plants that are particular to the neutron-physical design of the reactor core.

(5) Some of the analysis areas are interconnected in a sense that the results of a prior analysis is needed as input data for a subsequent analysis; **Figure G-1** presents several examples of the safety-related parameters and shows their typical interconnections. Specific requirements are specified for each individual analysis; in order to meet these requirements, the results of the analyses must meet specified criteria.

(6) Part of the results of the nuclear reactor core analysis are used as input data for subsequent event sequence analyses. The transferred data describe physical situations, however, are dependent on the details of the mathematical models and numerical program codes used in the analyses. Nevertheless, a certain number of safety-related parameters of the reactor core can be used, independently of the models, to describe the basic physical situation. These represent the safety-related properties of the reactor core.

(7) The type of the safety-related parameters depends on the reactor type, on the event sequences on which the design of the overall plant is based, and on the applied analysis method. **Table G-1** lists typical safety-related parameters from the analysis areas of neutron physics and thermo-hydraulics which are representative for light water reactors of the current design.

Reactor power
Power density distribution, Power density
Margin to critical boiling states
Effectiveness of the control rods
Shutdown rate of emergency shutdown system
Reactivity rate during control rod motion
Effectiveness of the boron injection systems
Reactivity rate during boron injection
Shutdown reactivity
Average reactor core burnup, Fuel rod burnup, Local burnup
Reactivity coefficients of - coolant temperature, - coolant density (void coefficient), - fuel temperature (Doppler coefficient), - boron concentration
Decay-ratio (BWR)
Kinetic parameters

Table G-1: Typical safety-related parameters

(8) The nuclear safety-related parameters are calculated with the help of nuclear analysis systems. These analysis systems are suited to determine the following parameters:

a) Multiplication factor, reactivity,

b) Neutron flux density,

c) Neutron current density,

d) Gamma flux density,

e) Reaction rates for neutron capture, neutron scattering and for fission,

f) Thermal energy release rate of (power density),

g) Burnup distribution, and

h) Change of nuclide densities.

KTA 3101.1 Thermo-hydraulic Core Design	***KTA 3101.2 Nuclear Core Design***	***KTA 3101.3 Thermo-mechanical Core Design***	***Evaluated Data as input to other analysis areas***
Departure from nucleate boiling Decay ratio (stability – BWR)	Power density	Fuel rod and fuel temperatures Casing deformations	Thermo-hydraulic reactor core design
Pressure losses Coolant flow distribution Density and coolant temperature distribution inside reactor pressure vessel	Reactivity / Reactivity balance Power density	Permissible fuel assembly / fuel rod / peak pellet burnup Fuel rod and fuel temperatures Permissible power density gradients (PCI – pellet cladding interaction)	Nuclear reactor reactor core design
Flow forces	Neutron flux Fuel assembly / fuel rod / peak pellet burnup Power density Power histories	Fission gas plenum pressure Oxide layer thickness Expansion / reference stress values Casing deformations	Thermo-mechanical reactor core design
Departure from nucleate boiling Pressure losses Density and coolant temperature distribution inside reactor pressure vessel	Reactivity coefficients Effectiveness / speed of shutdown systems Max. absolute / differential reactivity insertion Kinetic parameters Power density Power histories Fuel assembly / fuel rod / peak pellet burnup Decay energy	Oxide layer thickness Fuel rod and fuel temperatures Permissible fuel rod enthalpy values Permissible hydrogen content of cladding tube	Analyses of transients and design basis accidents
Decay ratio (stability – BWR)	Neutron flux Power density	Permissible power density gradients (PCI – pellet cladding interaction)	Instrumentation and control equipment
Flow distribution			Thermo-hydraulics of plant
	Boron-10 content Decay power Neutron flux Nuclide inventory		System engineering
	Nuclide inventory		Radiological protection

Figure G-1: Examples of results from the three analysis areas of reactor core design and their typical interconnections as well as connections to other analysis areas

(9) An analysis system is generated by combining a numerical method with the dataset. Both are always subject to approximating assumptions and, thus, jointly determine the accuracy of the results. In this context, the terms used and their meaning are as follows:

a) Numerical method:
Combination of mathematical models for the solution of the transport equation in a reactor core region with a defined material composition, this combination being supplemented by the mathematical description of the nuclide transformations.

b) Data set:
Set of input data that is independent of a specific application and that, unchanged, remains valid for a larger analysis area, e.g.,

ba) a collection of nuclear physics constants comprising the range of nuclides and reactor core reactions which are important to reactor engineering.

These nuclear physics constants include:

- nuclide cross sections,
- energy distribution of fission neutrons and the primary gamma radiation,
- nuclide decay constants,
- fission product yields,
- neutron and gamma yields,
- energy release from nuclear reactions.

bb) material properties, e.g., the variables of state of water.

(10) The values of the safety-related parameters of the reactor core depend on the design, the burnup condition of the reactor core and its current operating state. Thus, fulfillment of the requirements cannot be ensured alone by the reactor core design; it also requires considering requirements to be met by the adjacent systems and by plant operation.

(11) The present safety standard, therefore, also includes requirements to be met by the adjacent systems insofar as these requirements must be applied in relation to the design and operation of the reactor core. Those properties of the adjacent systems which have a significant influence on the result of safety analyses are referred to, herein, as safety-related parameters of the adjacent systems. The applicable values of these parameters depend on the actual operating conditions of the respective systems.

1 Scope

(1) This safety standard applies to stationary nuclear power plants with light water moderated pressurized or boiling water reactors (PWR or BWR). It contains requirements for the nuclear design and the operation of the reactor core. Requirements for adjacent systems are included insofar as they are necessary as based on the design and operation of the reactor core.

(2) The adjacent systems referred to under para. (1) include:

a) Systems needed for monitoring and limiting the reactor power and power density,

b) Systems needed for the reactivity control, for the shutdown as well as for monitoring and maintaining subcriticality during specified normal operation (e.g., control rods, boron injection systems, coolant recirculation pumps in a BWR, residual heat removal systems in a PWR), and

c) Systems needed for maintaining subcriticality after design basis accidents (e.g., boron injection systems, residual heat removal systems in a PWR).

2 Definitions

(1) Absorber, burnable

Burnable absorbers are such nuclides added to the fuel or to the structural parts of the fuel assembly that have a high neutron absorption capacity, however, whose binding capacity for reactivity changes with time due to nuclide conversion during power operation.

(2) Shutdown rate of the emergency shutdown system

The shutdown rate of the emergency shutdown system is the rate of decrease in reactivity caused by the forced insertion or falling-in of control rods after activation of emergency shutdown.

(3) Shutdown rate of the boron injection system

The shutdown rate of the boron injection system is the rate of decrease in reactivity caused by the increase in the concentration of boron in the reactor core after activation of boron injection.

(4) Shutdown reactivity

The shutdown reactivity is the reactivity of the reactor after it has been brought into a subcritical operational state by the designated shutdown systems.

Note:
The shutdown reactivity is dependent on the condition of the reactor after shutdown.

(5) Boron concentration

The boron concentration denotes the relative concentration of boron dissolved in the coolant. If the proportion of boron-10 deviates from the natural isotopic composition, this must be accounted for.

(6) Calibration error of a power density monitoring signal

The calibration error of a power density monitoring signal is the relative deviation of the actual value of the signal from its value as specified for an undisturbed power distribution.

Note:
The calibration error of a power density monitoring signal can be caused by

a) changes
   - of the ratio between measurand and power density,
   - of the undisturbed power distribution with burnup and the operational control rod position,
   - of the measurement sensor burnup with regard to the previous calibration,

b) tolerances of the calibration equipment and instrumentation (e.g., adjustment accuracy).

(7) Reactor core monitoring zone

A reactor core monitoring zone is a reactor core area in which the power density is being monitored and in which a uniform value applies to the permissible maximum power density.

(8) Power density monitoring signal

A power density monitoring signal is a signal which is created on the basis of the display signals of the inner or the outer measurement sensors of the reactor core instrumentation, or on the basis of both display signals, and which is representative of the maximum power density or the power density change in the respective reactor core monitoring zone.

(9) Net effectiveness of the emergency shutdown system

The net effectiveness of the emergency shutdown system is the effectiveness of the emergency shutdown system left in case of failure of that particular component of the emergency shutdown system which would result in the maximum possible loss of effectiveness of this system.

Note:
Cf. definition (16) "Effectiveness of the emergency shutdown system".

(10) Net effectiveness of a boron injection system

The net effectiveness of the boron injection system is the effectiveness of a boron injection system left in case of failure of that particular component of the boron injection system which would result in the maximum possible loss of effectiveness of this system.

Note:
Cf. definition (17) "Effectiveness of a boron injection system".

(11) Reactivity coefficient

The reactivity coefficient of a condition parameter is the partial differential quotient describing the change of reactivity as a function of this particular condition parameter.

(12) Tracking error of a monitoring signal

The tracking error of a monitoring signal is that deviation of the monitoring signal from its desired value which must be assumed to occur in the case of a disturbance of the power distribution that must be assumed.

Note:
The tracking error of a monitoring signal depends on

- the number, positioning and calibration of the measurement sensors,
- the way in which the individual measurement sensor signals are combined to form the monitoring signal,
- the kind of disturbance of the power distribution that must be assumed.

(13) Parameter validation

A parameter validation is considered to be the process of verifying that the characteristics of a numerical model with regard to its intended application represents the real conditions (e.g., the physical or chemical conditions or processes) with sufficient accuracy.

(14) Model verification

A model verification is considered to be the process of verifying that the implemented model corresponds to the conceptual description of this model.

(15) Effectiveness of the emergency shutdown system

The effectiveness of the emergency shutdown system is the difference in reactivity between an initial critical state during normal operation (with the control rods in their operational position) and the respective final state (control rods in their final position after an emergency shutdown).

(16) Effectiveness of a boron injection system

The effectiveness of a boron injection system is the difference in reactivity between an initial operating state before the boron injection system is activated and the respective final state.

3 Safety-Related Requirements for the Neutron-Physical Design and the Operation of Reactor Cores

(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 (anticipated transients without scram – ATWS – Safety Level 4a). Insofar as different requirements apply at the individual safety levels, these are specified in this safety standard.

(2) The reactor core shall be designed such that, graded according to the specifics of Safety Levels 1 though 4a, the reactivity control, the reactor core cooling and the retention of radioactive substances are ensured. This requirement leads to the subsequent requirements regarding function and effectiveness of the adjacent systems.

(3) At all safety levels, requirements from other analysis areas (e.g., thermo-hydraulic and mechanical design) and the requirements from superordinate technical standards shall be taken into account.

(4) In the following, the necessary requirements are grouped according to safety levels. The safety levels by themselves represent a graded safety concept (defense-in-depth concept) where the individual events to be considered are allocated to a specific safety level according to their probability of occurrence.

3.1 Safety Level 1 (normal operation)

It shall be ensured that the inherent properties of the reactor core for limiting any reactivity and power increases are not impaired. The local power density, in its interaction with the controlling and limiting devices (limitation of process variables), 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. The shutdown capability of the control elements and a long-term subcriticality shall be ensured.

3.2 Safety Level 2 (abnormal operation)

The permissible values of the local power density in its interaction of the limiting and reactor protection devices shall not be exceeded, in order to ensure the unrestricted continuing use of the fuel assemblies. Otherwise, the same requirements apply as for Safety Level 1 (cf. Section 3.1)

3.3 Safety Level 3 (design basis accidents)

(1) The power and local power density in their interaction with the inherent properties of the reactor core and the reactor protection system, shall basically be limited such that any damage to fuel rods is prevented. If, in individual cases, this requirement cannot be fulfilled, it shall be verified that the capability for cooling the reactor core is ensured, that impermissible loads of the pressure retaining boundary are prevented and that radiological effects stay below permissible limit values. The requirements of superordinate technical standards with regard to the permissible extent of damages shall be taken into consideration.

(2) The emergency shutdown and sustained long-term subcriticality shall be ensured. A short recriticality is permissible, provided, the requirements under para. (1) continue to be fulfilled.

3.4 Safety Level 4a (very seldom postulated events)

In case a failure of the emergency shutdown system must be assumed, the pressure within the pressure retaining boundary shall be limited to permissible values and the long-term subcriticality and capability for cooling the reactor core shall be ensured.

4 Safety-Related Parameters, Requirements and Measures

4.1 Safety-Related Parameters

(1) The safety-related parameters shall be specified for each reactor as a function of the reactor type, the overall plant design and the analysis methods.

(2) The permissible value ranges of the safety-related parameters during normal operation (Safety Level 1) shall be determined by analyzing representative event sequences. Criteria for the permissible values ranges are:

a) that the loading limits for reactor core components specified in other analysis areas are not exceeded,

b) that those initial operating states of the reactor core are not exceeded that are assumed or have been proven as permissible in the analyses of the event sequences for abnormal operation (Safety Level 2), for design basis accidents (Safety Level 3), or for the very seldom postulated events to be considered (ATWS, Safety Level 4a),

c) that the inherent safety of the reactor core is maintained,

   N o t e :

   The inherent safety of reactor cores in light water reactors is understood as being its capability

   - of limiting, by means of prompt feedback properties of the reactor core, any fast uncontrolled power increase due to design basis accidents, and effecting this faster than the time needed for detecting these accidents and for initiating protective actions, and
   - of automatically effecting a shutdown or a reduction of the fission-product energy to permissible values in the case of design basis accidents involving a pressure drop and void formation, even without the initiation of protective actions.

d) that those initial operating states of the adjacent systems are not exceeded and their effectiveness maintained that must be assumed or have been proven as permissible in the analyses of event sequences for abnormal operation (Safety Level 2), for design basis accidents (Safety Level 3), or for the very seldom postulated events to be considered (ATWS, Safety Level 4a).

N o t e :

For the initial operating states to be assumed for the reactor core and for the adjacent systems, the permissible ranges of values depend on:

- the kind of the operating state or event analyzed at the respective Safety Levels.
- the specific safety-related requirements or permissible sequential effects at the respective Safety Levels.
- the applied analysis method. and
- the numerical uncertainties to be taken into account (Safety Levels 1, 2 and 3)

The permissible range of values for a given parameter shall be the one determined from the joint range of values from all relevant analyses that is verified as being permissible.

(3) The analyses for determining the permissible ranges of values of the safety-related parameters may, alternatively, be carried out in exemplary fashion, provided, it is ensured that the condition parameters for the event sequences to be taken into account are chosen conservatively enough to cover all initial operating states to be assumed.

4.2 Safety-Related Requirements and Measures

(1) The reactor core and the adjacent systems shall be designed and operated in such a way that, during normal operation, the safety-related parameter values will stay within the respective ranges that have been verified as being permissible.

(2) Safety-related parameters shall be monitored to the extent necessary with regard to their maintaining the permissible value range during operation and inservice inspections. An instrumentation shall be provided for the reactor core and the adjacent systems that is suited to detect the safety-related parameters themselves or the allocated measurands. Examples for the correspondence between measurands and safety-related parameters are presented in **Table 4-1**.

(3) The following measures shall be provided with regard to maintaining the permissible value ranges of safety-related parameters as specified under para. (2):

a) manual measures in combination with operating instructions (including inservice inspections),

b) automatic control devices,

c) automatic limiting devices, or

d) emergency shutdown.

(4) The manual measures under para. (3), item a) are permissible as sole measures, provided,

a) it is signaled to the operator if the respective limit values have been exceeded, and

b) there is sufficient time available for the initiation of countermeasures considering the respective event sequences.

(5) The kind of these measures as well as the permissible time lag until countermeasures become effective shall be specified on the basis of the respective event sequences.

<table>
<tr><th rowspan="2">No.</th><th colspan="2">Possible Measurands</th><th rowspan="2">Safety-Related Parameters</th></tr>
<tr><th>Pressurized Water Reactor</th><th>Boiling Water Reactor</th></tr>
<tr><td rowspan="2">1</td><td>Enthalpy rise of the cooling circuits,</td><td></td><td rowspan="2">Reactor power</td></tr>
<tr><td colspan="2">Heat balance,
Neutron flux,
Gamma flux</td></tr>
<tr><td rowspan="2">2</td><td>Neutron flux distribution inside and outside of the reactor core,
Coolant temperature inside the reactor core,
Enthalpy rise of the cooling circuits</td><td>Neutron flux distribution inside the reactor core,</td><td rowspan="2">Power density distribution,
Power density</td></tr>
<tr><td colspan="2">Gamma flux distribution inside the reactor core,
Control rod positions</td></tr>
<tr><td rowspan="3">3</td><td colspan="2">as under Nos. 1 and 2, and in addition:</td><td rowspan="3">Margin to critical boiling states</td></tr>
<tr><td>Coolant pressure,
Coolant temperature,</td><td>Reactor pressure,
Core inlet subcooling,</td></tr>
<tr><td colspan="2">Speed of coolant recirculation pumps,
Pressure difference</td></tr>
<tr><td>4</td><td>Immersion depth of the control rods</td><td>(covered by No. 9)</td><td>Effectiveness of the control rods</td></tr>
<tr><td>5</td><td>Control rod drop time</td><td>Fast-insertion time of the control rods</td><td>Shutdown rate of the emergency shutdown system</td></tr>
<tr><td>6</td><td>Moving speed of control rods</td><td>Differential control rod effectiveness,
Moving speed of control rods,
Number and position of control rods which can be withdrawn at the same time</td><td>Maximum reactivity rate of moving control rods</td></tr>
<tr><td>7</td><td colspan="2">Level and boron concentration (including boron-10 content) in storage tanks</td><td>Effectiveness of the boron injection systems</td></tr>
<tr><td>8</td><td colspan="2">Rated flow of injection pumps,
Concentration of injected boron (including boron-10 content)</td><td>Reactivity rate during boron injection</td></tr>
<tr><td rowspan="2">9</td><td colspan="2">Control rod position,
Neutron flux,
Coolant temperature,</td><td rowspan="2">Shutdown reactivity</td></tr>
<tr><td>Boron concentration of the coolant (including boron-10 content)</td><td>Critical control rod configurations for the cold critical reactor (taking control rod burnup into account)</td></tr>
<tr><td>10</td><td colspan="2">Time integral of power</td><td>Average reactor core burnup</td></tr>
<tr><td>11</td><td colspan="2">Time integral of power,
Power density distribution</td><td>Fuel assembly burnup</td></tr>
</table>

Table 4-1: Example for the correspondence between possible measurands and safety-related parameters of the reactor core and adjacent systems

5 Limiting and Monitoring the Power Density

5.1 Limiting the Power Density

(1) The power density shall be limited in such a way that

a) the requirements in accordance with safety standard KTA 3101.1 for specified normal operation (i.e., Safety Levels 1 and 2) are met,

b) the safety-related boundary conditions of the mechanical design of the fuel rods are adhered to during specified normal operation,

c) the initial values of the power density which have been verified to be permissible for normal operation are also, in the sense of Section 4.2, para. (1), adhered to during abnormal operation events and design basis accidents.

d) for those events at Safety Levels 2, 3 and 4a which could lead to a power density increase or to an impairment of the cooling, the cooling conditions as well as the fuel and cladding conditions which have been verified to be permissible in accordance with safety standard KTA 3101.1 are not exceeded.

(2) The required limit values of the power density shall be determined by analyzing representative event sequences as specified under Section 4.1.

(3) The most restrictive limit value determined by the analyses for a reactor core monitoring zone shall be the required limit value for the power density in the respective reactor core monitoring zone.

Note:
The limit values of the power density may differ in different regions of the reactor core and they reflect possible influences from the type of fuel assembly, the burnup and the local thermo-hydraulic conditions (pressure, temperature, void content, mass flow density of the coolant).

5.2 Instrumentation of the Reactor Core

The instrumentation of the reactor core serves the purpose of monitoring the power density distribution (conformance of the actual with the expected power density distribution) as well as of monitoring the power density itself.

5.2.1 Monitoring the power density distribution

(1) Insofar as necessary for meeting the requirements specified under Section 5.1, a continuously or intermittently displaying reactor core instrumentation shall be provided for monitoring of the power density distribution.

Note:
Examples of an intermittently displaying instrumentation are the aeroball system and the traversing in-core probe system ("tip system") with the associated evaluation devices.

(2) The number and locations of the measurement sensors shall be sufficient to be able to detect any significant deviations between the actual and the anticipated power density distribution. In particular, the measuring system shall be capable of detecting any azimuthal asymmetries of the power density distribution as well as local differences of the axial power density distributions.

5.2.2 Monitoring the power density

(1) Insofar as necessary for meeting the requirements specified under Section 5.1, a continuously displaying instrumentation for the reactor core and cooling circuits shall be provided for monitoring the local power density.

Note:
The following measurement sensors may be used for a continuous monitoring of the power density:

a) Neutron flux or gamma flux measurement sensors inside the reactor core (in-core instrumentation),

b) Neutron flux or gamma flux measurement sensors outside of the reactor core (ex-core instrumentation),

c) Temperature sensors inside the reactor core and the cooling circuits.

(2) Number and locations of the measurement sensors, their calibration and the kind of signal forming shall be selected such that an impermissible increase of local power density in the sense of Section 5.1 can be detected in the individual reactor core monitoring zones.

(3) The signals of these measurement sensors may be used separately or in combination with each other for monitoring the power density. If determining local variations of the power density requires supplementing the displays of the measurement sensors by calculated data concerning the behavior of the power density distribution, the requirements of Section 7 shall be applied to these analytical procedures. The calculated data may be obtained from previous exemplary calculations or from a concurrently performed calculation.

(4) The design of the systems for monitoring and limiting the power density shall be based on the power density redistributions which can occur as a result of

a) the size and design of the reactor core,

b) the reactor core management (load changes, load ramps),

c) the failures of adjacent systems to be assumed (e.g., failure of the power distribution control system),

d) the incorrect control rod positions to be assumed.

Note:
The requirements specified under paras. (1) through (4) can be fulfilled, e.g., by generating a monitoring signal from the indications of the measurement sensors of each reactor core monitoring zone which, in all power density redistributions that may lead to an increase in power density in the respective reactor core monitoring zone, is either

a) proportional to the maximum power density in the respective reactor core monitoring zone, or

b) proportional to the increase in maximum power density in the respective reactor core monitoring zone above a specified reference level.

(5) A specification of the incorrect control rod positions to be assumed during abnormal operation shall take

a) the control rod movement limitations and control rod movement interlocks, as well as

b) the design of the systems for the actuation and the position monitoring of the control rods.

into account.

(6) With regard to Safety Level 1, the instrumentation and control system shall be able to detect in which area of the operational performance chart the reactor is being operated. Insofar as undamped power oscillations can occur at Safety Level 2, the instrumentation and control system shall be sufficiently fast to be able to promptly initiate counter measures.

(7) Any increase of the power density which may lead to impermissible value levels shall be identified by more than one measurement sensor (information redundancy).

(8) The temporal behavior and tracking accuracy shall be taken into account in the monitoring and evaluation of the measurement values.

5.2.3 Measurement and response errors

(1) The uncertainties of the calibration of signals for the identification of the power density and of the thermo-hydraulic operating state shall be specified (calibration errors).

(2) Tracking errors which may lead to an underestimation of the power density with respect to the increases in power density to be assumed shall be determined by analytical or experimental methods.

(3) Calibration and tracking errors shall be combined to form a resulting response error for the power density limitation and shall be taken into account when specifying the response values.

5.3 Identification of the Thermo-Hydraulic Operating States

If the limit values of the local power density that must be adhered to are dependent on the thermo-hydraulic cooling conditions, an instrumentation shall be provided for the reactor core and the cooling circuits which can identify the cooling conditions inside the reactor core.

N o t e :

Measurands which may be used for this purpose are specified in safety standard KTA 3101.1, Sec. 5.8.2.

5.4 Devices and Measures for Limiting the Power Density

The reactor core shall be designed and operated in such a way that the power density is limited to the permissible levels specified under Section 5.1. Insofar as necessary, devices and measures for limiting the power density shall be provided as specified under Section 4.2, para. (3), for example

a) manual measures in combination with operating instructions,

b) automatic control devices,

c) control rod movement limitations,

d) automatic limitations of the integral power and local power density, or

e) reactor scram.

N o t e :

The necessary measures and equipment for limiting the power density depend on

a) the size and design of the reactor core,

b) the thermo-hydraulic design of the cooling circuits,

c) the intended load operation program (constant load, load changes, load ramps),

d) the control rod movement program, and

e) the margin between the operationally possible power density levels and the limit values verified as permissible by corresponding analyses.

6 Systems for Reactivity Control and Shutdown

6.1 General Requirements

(1) Systems shall be provided for the reactivity control and shutdown; these systems include control rods, boron injection systems and, in the case of a BWR, the pump speed controls.

N o t e s :

(1) With regard to their task objectives, the shutdown systems are summarily considered as shutdown facilities.

(2) The shutdown facilities of a PWR are

a) the control rod system, possibly in combination with a supporting earthquake-safe boron injection system (e.g., extra boron injection system), and

b) the entirety of the other boron injection systems (e.g., volumetric and chemical control system, accumulator and borated water storage tanks together with the associated injection pumps).

(3) The shutdown facilities of a BWR are

a) the control rod system with a hydraulically controlled forced insertion, and

b) the control rod system with electro-mechanical insertion and, additionally, the boron injection system.

(4) In both reactor types the control rod system, in the case of the BWR with a hydraulically controlled forced insertion, also serves as emergency shutdown system.

(2) The shutdown systems shall be designed such that the reactor core can be transferred to an operating state of subcriticality from any state of specified normal operation and that it can be permanently maintained at subcriticality.

N o t e :

Safety-related parameters of a shutdown system are:

a) its effectiveness and net effectiveness,

b) its shutdown rate, as well as

c) the maximum possible positive reactivity rate which could be caused by faulty operation of the reactivity control elements.

(3) The technical requirements concerning the shutdown systems including the requirements for inservice inspections are specified in safety standard KTA 3103.

(4) The reactivity-related requirements for these systems result from the interaction of the following nuclear reactor core design parameters:

a) excess reactivity to ensure a specified length of the fuel cycle,

b) reactivity feedback from power changes, from the coolant temperature, coolant density, fuel temperature, steam bubble content (void coefficient) and boron concentration,

c) binding of reactivity due to burnable absorbers, and

d) binding of reactivity due to the neutron poisons created during reactor operation (e.g., xenon-135 as the most significant contributor).

(5) A sufficient effectiveness of the shutdown systems shall be verified for each fuel cycle. The validity of analytical verifications shall be spot checked by representative measurements.

N o t e :

Typical representative measurements are:

a) PWR: Determination of the boron equivalents of the control rod groups,

b) BWR: Determination of the differential effectiveness of the control rods.

(6) For specified normal operation (i.e., Safety Level 1 and 2) of the control rod system the verification of sufficient effectiveness shall be performed conservatively based on the net effectiveness of the control rod system.

(7) When applying the single failure criterion to a shutdown system, a single failure shall be assumed at Safety Level 3. This failure shall be assumed for those components, the failure of which would lead to the largest reduction of shutdown rate or shutdown effectiveness. In this context, the assumption of ineffectiveness for the most effective control rod or control assembly may be considered as being a single failure.

(8) The required value for the long-term maintainable net shutdown reactivity shall be at least 0.3 %. In the case of an analytical verification with established analytical design procedures, the required value shall not be less than 1 %. If necessary, larger values shall be applied.

Note:

The values above are considered as minimum requirements. They are specified under the assumption that the insecurities of established analytical design procedures are smaller than 0.7 %. If this should not apply, the required value for the analytical verification shall be adjusted accordingly.

(9) Components of the shutdown systems may also be employed for operational control tasks. In this case, safety-related precautionary measures during operation shall ensure that the effectiveness required for this component during shutdown is maintained at every operatiing state.

6.2 Control Rod System

6.2.1 General requirements

(1) The control rod system in a PWR or BWR is used for an emergency shutdown. Emergency shutdowns are initiated automatically by the reactor protection system. A manual initiation shall also be provided for.

(2) The control rod system – under consideration of the inherent properties of the reactor core and in conjunction with other engineered safety features – has the task of ensuring that the safety-related requirements allocated to the respective Safety Levels are fulfilled. This means that it shall

a) transfer the reactor to its subcritical hot condition in a sufficiently short time, and,

b) after an emergency shutdown, maintain the reactor permanently in the state of subcriticality – possibly in conjunction with further shutdown systems (e.g., boron injection system).

(3) The shutdown effectiveness and shutdown rate required for the fulfillment of these tasks shall be determined by analyzing representative event sequences to be assumed for the respective Safety Levels.

(4) The analyses to be carried out may be limited to event sequences involving the highest requirements regarding effectiveness, shutdown rate and reactivity release (cf. safety standard KTA 3101.1, Appendix A).

Note:

Examples for such event sequences are

a) regarding required effectiveness:
events involving reactor core subcooling (PWR) and shutdown to the operating state ‘cold, xenon-free’ (BWR),

b) regarding required shutdown rate:
the simultaneous failure of all coolant recirculation pumps (PWR), failure of the main heat sink (BWR),

c) regarding largest possible reactivity release:
erroneous initiation of control rod retraction caused by a malfunction of the controls (startup design basis accident) or falling out (BWR) or retraction of one control rod (PWR).

(5) If components of the control rod system are also used for operational control tasks, the requirements specified under Section 6.2.4 shall be taken into account.

6.2.2 Special requirements for a PWR

(1) The emergency shutdown system and the reactor core shall be designed such that, following a shutdown as a result of events during specified normal operation (i.e., Safety Levels 1 and 2) and until achieving long-term subcriticality by the boron injection systems, the amount of net shutdown reactivity will not fall below the value specified under Section 6.1, para. (8).

(2) Following a design-basis-accident-related shutdown (i.e., Safety Level 3), the same requirements as specified under para. (1) shall be applied; in this context, a temporary recriticality and associated renewed increase of the power density are permissible as long as the requirements of Section 3.3 are met.

6.2.3 Special requirements for a BWR

(1) The emergency shutdown system and the reactor core shall be designed such that, following a shutdown as a result of events during specified normal operation (i.e., Safety Levels 1 and 2) and following a design-basis-accident-related shutdown (i.e., Safety Level 3), the amount of net shutdown reactivity for the operating state ‘zero load, xenon-free’ at a temperature leading to the highest reactivity will not fall below the value specified under Section 6.1, para. (8).

(2) To ensure against unintentional criticality and unintentional power increases, startup and loading interlocks as well as emergency shutdown actuations shall be provided.

6.2.4 Safety-related conditions for operation

It shall be ensured that the permissible value ranges for the safety-related parameters of the emergency shutdown system as verified by analyses (cf. Section 6.1) are not exceeded during operation. In this context, additional measures in addition to the design of the reactor core and the emergency control system shall be provided such as

a) in the case of a PWR: limitation of the permissible immersion depth of control rods by means of operating instructions or automatic limitation systems,

b) limitation of the withdrawal rate of control rod banks, limitation of the number of control rods which can be withdrawn at the same time.

6.3 Boron Injection Systems

Boron injection systems shall be provided. Their tasks and requirements depend on the specific design characteristics of the PWR or BWR.

6.3.1 Boron injection systems of a PWR

6.3.1.1 Tasks

The boron injection systems of a PWR shall fulfill the following tasks:

a) supplementing the emergency shutdown system for maintaining the subcritical condition of the reactor, and

Note:

Boron injection systems shall be provided if the effectiveness of the emergency shutdown system is not sufficient to transfer the subcritical reactor into the subcritical operating state ‘cold, xenon-free’. The task of such boron injection systems, in conjunction with the inherent properties of the reactor core and, if applicable, other systems, is to maintain subcriticality of the reactor even in its most reactive operating state after an emergency shutdown

b) functioning as a second shutdown system independent of the emergency shutdown system

   ba) for all those conditions of specified normal operation (i.e., Safety Levels 1 and 2) that do not require fast reactivity changes, and

   bb) for those events of Safety Levels 3 and 4a for which it must be assumed that the emergency shutdown system will not be available.

6.3.1.2 Requirements

(1) The required effectiveness and shutdown rates of the boron injection systems shall be determined on the basis of analyses of those event sequences to be assumed at the respective Safety Levels (cf. safety standard KTA 3101.1, Appendix A) which demand the highest effectiveness of the boron injection systems. In this context the following shall be taken into account:

a) the design of the boron injection systems with regard to the storage tank volumes, to the flow rate of the injection system, to the extraction rate by the volume control system,

b) the flow and mixing conditions in the cooling circuit,

c) the operating state of the boron injection systems with regard to the stocked amount of the solution available, the boron concentration and the boron-10 content of the stock solution,

d) the design and the burnup state of the reactor core (boron effectiveness),

e) the operating state of the reactor core in the initial and in the shutdown state, and

f) the boron concentration in the initial operating state.

(2) The effectiveness or net effectiveness of the boron injection system used for the task specified under Section 6.3.1.1, item a), shall be such that, regarding

a) the reactivity insertion during transfer from the operating state 'zero load, hot' to the state 'zero load, cold',

b) the reactivity insertion during transfer to the operating state 'xenon-free', based on the respective maximum xenon concentration to be assumed,

c) the reactivity change due to the decay of other reactivity-effective isotopes (e.g., neptunium-239),

are compensated by the associated boron injection system as quickly as required on the basis of the event sequence analyses and such that the reactor will remain in the subcritical operating state and reach the necessary shutdown reactivity.

(3) It shall be ensured by boron injection systems that the following requirements are fulfilled:

a) For the task specified under Section 6.3.1.1, item a), the boron injection systems, in combination with the control rod system, shall be able to establish the long-term subcriticality. The long-term subcriticality for the operating state 'zero load, xenon-free' at the coolant temperature to be assumed as leading to the highest reactivity shall be at least equal to the value specified under Section 6.1, para. (8).

b) For the task specified under Section 6.3.1.1, item ba), the boron injection systems shall be designed such that it itself can transfer the reactor from any initial operating state to be assumed for specified normal operation (i.e., Safety Levels 1 and 2) to the state of subcriticality and maintain at least the value specified under Section 6.1, para. (8) over a long time period. For the task specified under Section 6.3.1.1, item bb), the boron injection systems shall be designed such that the long-term subcriticality is ensured taking all event sequences that must be assumed into account.

This requires that the calculations are performed by a program system that has been validated by experiment and, this also requires that the neutron flux and boron concentration are monitored. If any one of these requirements is not fulfilled, the boron injection system shall be designed such that at least a calculated value of 5 % for the shutdown reactivity is achieved.

6.3.1.3 Special requirements

(1) As a protective measure against unintentional criticality, the boron concentration in the primary coolant system and in the relevant storage tanks shall be monitored, and any dilution shall be prevented by the measures specified under Section 4.2 para. (3).

(2) It shall be ensured during plant operation that the amount and concentration of the boron stock solutions is maintained within the permissible value ranges verified by analyses.

6.3.2 Boron injection system of a BWR

(1) A boron injection system shall be provided that shall be designed such that the reactor can be safely shutdown and permanently maintained at subcriticality from any initial state to be assumed for normal operation (i.e., Safety Level 1).

(2) The value of the long-term subcriticality to be ensured for the operating state 'zero-power, xenon-free' shall be at least 5 % at the coolant temperature to be assumed as leading to the highest criticality.

Note:

Applying Section 6.3.1.2, para. (3) in a general sense, it is assumed that the respective criteria (program system validated by experiment; monitoring neutron flux; monitoring boron concentration) will not all be fully satisfied.

7 Requirements for Nuclear Analysis Systems

7.1 General Requirements

(1) The nuclear analysis systems comprise the entirety of program codes used for designing the reactor core. These include, in particular, the program codes for the

a) nuclear design of the fuel assemblies,

b) steady-state reactor core design, and

c) analysis of the reactor core transients.

(2) Nuclear analysis systems shall be capable of determining the operationally relevant and the essential safety-related parameters of the reactor core, insofar as they are based on the design of the reactor core; they shall also be capable of determining the measurands required for the validation of the analysis systems. Furthermore, the nuclear analysis systems shall supply the input data required for the other analysis areas.

(3) The nuclear analysis systems shall be capable of describing the following physical processes and parameters as a function of location, time and burnup:

a) Neutron transport;

b) Reaction rates and power density distribution;

c) Feedback of the changes of states of the coolant, fuel and control rod positions on the reactivity and on the flux density distribution; and

d) Changes of the nuclide inventory.

(4) The following simplifications and approximations of the models for analyzing these processes and parameters are, among others, permissible:

a) Splitting up the overall problem into partial problems;

b) Simplified representation of the geometric and physical structure of the reactor core;

c) Discretization of the continuous neutron or gamma spectrum;

d) Partitioning the neutron flux distribution into a limited number of time intervals of constant fluxes;

e) Approximation procedures for solving the neutron transport equation.

N o t e :

The permissibility of the simplifications and approximations quoted shall be verified by checking the validity and accuracy, cf. Section 7.3.

(5) The sensitivity of the results with regard to the applied simplifications of the models shall always be examined as soon as more powerful numerical methods become available.

(6) When describing partial physical aspects within the analysis system, it is permissible to use correlations which have been derived from experiments, provided, the experiments are representative for the intended scope of application of these correlations.

7.2 System Description and Boundary Conditions

(1) Performing nuclear calculations requires a detailed knowledge of the systems to be described as well as of the additional boundary conditions.

(2) The system description shall take the following aspects into account:

a) Reactor design:
number, size and arrangement of the fuel assemblies, control assemblies and further core internals;

b) Plant parameters:
thermal power, system pressure, inlet temperature and coolant flow rate (pressure/temperature versus load diagram, circulation control characteristics);

c) Structure of the fuel assemblies:
geometric arrangement and composition of the fuel, of the burnable absorbers, of the moderator and of the structural parts within the fuel assemblies;

d) Geometry and composition of the control rods;

e) Type and composition of the neutron absorbers dissolved in the moderator;

f) Geometry and composition of the materials outside of the reactor core (reflector).

(3) The following variable influences shall be taken into account:

a) Change in fuel composition, i.e., spatial and temporal changes of the nuclide densities as a function of the burnup, power and power history;

b) Spatial and temporal changes of

   ba) fuel temperature as function of the burnup and power, and

   bb) moderator temperature and density as function of power and coolant flow rate,

c) Changes in the boron concentration as a function of burnup and power (PWR); and

d) Control rod positioning.

(4) The following shall additionally be taken into account in the case of the analysis of transients:

a) Influences of the plant behavior including those of the systems regarding controls, limitation and protection;

b) Kinetic parameters of the core inventory; and

c) Decay power.

7.3 Checking the Validity and Accuracy

7.3.1 General

(1) The nuclear analysis systems used shall have been verified and validated.

(2) The validation procedure depends on the accuracies required of the results.

(3) The validation procedure shall discern between a validation of the overall analysis system used for the respective scope (integral validation) and a validation of individual modules of the analysis system (partial validation). In addition to an integral validation of the nuclear analysis system, the respective scope should be verified by a partial validation of the individual modules.

N o t e :

The partial and integral validation supplement each other and are usually used in combination. When applying an integral validation procedure alone, it is possible that various errors might compensate each other. This would limit the possibilities for an interpolation within the scope of application. On the other hand, the sole use of the partial validation procedure with its individual validation steps might make it difficult to validate the overall analysis system.

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

(5) When validating the analysis system, all systematic deviations and statistical uncertainties shall be determined. The verified systematic deviations may be corrected by applying corresponding correction factors to the results.

N o t e :

The statistical uncertainties may be determined analogous to the methods specified in safety standard KTA 3101.1, Sec. 4.3.

7.3.2 Validation procedure

(1) The nuclear analysis system shall be validated by comparing the calculated results with the results from

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

b) post-irradiation examinations,

c) experiments,

d) evaluations of actual transients, or

e) other nuclear analysis systems (benchmarks or reference results).

   N o t e :

   Reference results are results from calculation systems that either have already been validated or that represent the physical situations to be calculated by more realistic models.

(2) The results of the measurements under para. (1), items a), b) and c) shall normally cover the entire range of the reactor plant operation with regard to the essential parameters. In those cases where the original reactor conditions were not reproduced (modeled), the transferability of experimentally determined results to the reactor conditions shall be well substantiated.

(3) The selection of measurement results shall, in particular, take the following criteria into account:

a) Documentation of the measurements;

b) Measurement quality and error consideration;

c) Transferability of the measurement conditions to the scope of application of the analysis system that it must cover with regard to the design.

(4) When applying correlations and tables in nuclear analysis systems, the parameter limits inherent to the experiments shall be observed. In those exceptional cases where extrapolations are necessary, the extrapolation shall be well substantiated.

(5) **Table 7-1** lists examples of the measurements for the validation of nuclear analysis systems.

7.3.3 Safety levels

(1) The validation of the analysis systems applied to the verification at Safety Levels 1 and 2 shall normally be based primarily on the measurement results specified under Section 7.3.2, para. (1), items a), b) and c). Insofar as possible, actual transients (cf. Section 7.3.2, para. (1), item d)) shall be included in the validation.

(2) The validation of analysis systems applied to the verification at Safety Level 3 shall be based on the measurement results specified under Section 7.3.2, para. (1), items c), d) and e).

(3) The models used for verification at Safety Level 4a shall, as far as possible, be the same as those applied at Safety Levels 1, 2 and 3 and that are validated for partial aspects of the occurring physical processes (partial validation). If this is not possible, the models used shall be constructed based on current knowledge and shall, individually, be well substantiated.

7.4 Requirements Regarding Documentation

Reports shall be prepared concerning the analysis system. These reports shall

a) describe the analysis system with regard to the **numerical method**s, the data sets and the verification and validation procedure, and

b) specify the scope of application of the analysis system and quantify the accuracy of the results.

I. Validation of Nuclear Analysis Systems at a Power Reactor	
Operational Measurements and Evaluation of Actual Transients	***Reference to Safety-Related Parameters***
Generation of critical states at zero load and a xenon-free reactor core while varying - coolant temperature, - boron concentration (PWR), - control rod position in order to determine the boron effectiveness (PWR), the integral and differential effectiveness of control rod banks and single rods, the isothermal temperature coefficients.	Effectiveness of the control rods and the boron injection systems, Shutdown reactivity, Reactivity coefficients
Generation of critical states in a PWR at zero load after a preceding stationary power operation while varying the boron concentration in order to determine the reactivity equivalents of the power and the xenon concentration.	Shutdown reactivity, Reactivity coefficients
Variation of the control rod position, coolant temperature and boron concentration (PWR) or coolant flow rate (BWR) during stationary power operation in order to determine the differential control rod effectiveness, the coolant temperature coefficient, the boron effectiveness and the circulation control characteristic.	Reactivity rate during control rod movements, Reactivity coefficients
Evaluation of neutron flux (or gamma flux) sensitive detector signals of the incore instrumentation together with characteristic coolant data (pressure, temperature) during stationary power operation for various control rod positions and during local xenon transients.	Power density distribution, Power density, Departure from nucleate boiling, Average and local burn-up
Measurement of the stability behavior in case of a BWR	Decay ratio
Evaluation of characteristic data of the reactor core in the case of planned or unplanned transients, for example - reactor emergency shutdown, - failure of coolant recirculation pumps, - reactor core subcooling, - load rejection (load shedding), - erroneous movement of control rods.	Speed of emergency shutdown, Reactor power, Reactivity coefficients
II. Validation of Nuclear Calculation Systems at a Critical or Subcritical Core Assemblies	
Measurement of - the microscopic flux distribution and the reaction rate distribution, - the macroscopic flux density distribution, - the kinetic parameters, - the reactivity.	
III. Validation of Nuclear Calculation Systems by Means of Measurements of Irradiated Fuel	
Gamma scanning, Isotope analysis.	

Table 7-1: Examples of measurements for the validation of nuclear analysis systems

Appendix A

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 – AtG) of December 23, 1959, revised version of July 15, 1985 (BGBl. I, p. 1565), most recently changed by Article 5 Paragraph 6 of the Act of February 24, 2012 (BGBl. I, p. 212)
StrlSchV		Ordinance on the protection from damage by ionizing radiation (Radiological Protection Ordinance – StrlSchV) of July 20, 2001 (BGBl. I, p. 1714; 2002 I, p. 1459), most recently changed by Article 5 Paragraph 7 of the Act of February 24, 2012 (BGBl. I, p. 212)
Safety Criteria	(1977-10)	Safety criteria for nuclear power plants of October 21, 1977 (BAnz. No. 206 of November 3, 1977)
Design Basis Accident Guidelines	(1983-10)	Guidelines for the assessment of the design of nuclear power plants with pressurized water reactors against design basis accidents as defined in Sec. 28, para. 3 StrlSchV (Design Basis Accident Guidelines) of October 18, 1983 (Addendum to BAnz No. 245 of December 31, 1983)
KTA 3101.1	(2012-11)	Design of reactor cores of pressurized water and boiling water reactors; Part 1: Principles of thermo-hydraulic design
KTA 3103	(1984-03)	Shutdown systems for light water reactors *(Revision draft of this safety stand is in preparation)*
KTA 3501	(1985-06)	Reactor protection system and monitoring equipment of the safety system