

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

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

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

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PLEASE NOTE: Only the original German version of this safety standard represents the joint resolution of the 35-member Nuclear Safety Standards Commission (Kerntechnischer Ausschuss, KTA). The German version was made public in the Federal Gazette (Bundesanzeiger) on July 25, 2023. Copies of the German versions of the KTA safety standards may be mail-ordered through the Wolters Kluwer Deutschland GmbH (info@wolterskluwer.de). Downloads of the English translations are available at the KTA website (<http://www.kta-gs.de>).

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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 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) DNB 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) $MASL_{perm}$

$MASL_{perm}$ 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**).

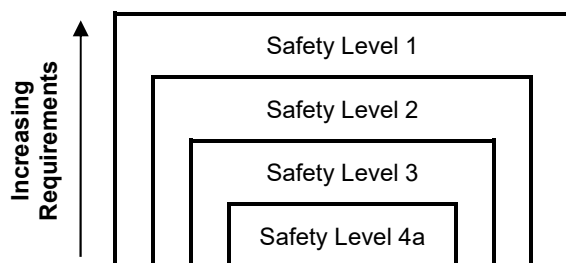


Figure 3-1: Safety levels

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.