Safety Standards

of the
Nuclear Safety Standards Commission (KTA)

KTA 3107 (2014-11)

Nuclear Criticality Safety Requirements during Refueling

(Anforderungen an die Kritikalitätssicherheit beim Brennelementwechsel)

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

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PLEASE NOTE: Only the original German version of the present safety standard represents the joint resolution of the 35-member Nuclear Safety Standards Commission (Kerntechnischer Ausschuss, KTA). The German version was made public in Bundesanzeiger (BAnz) of January 15, 2015.
Copies of the German versions of KTA safety standards may be mail-ordered through Wolters Kluwer Deutschland GmbH (info@wolterskluwer.de). Downloads of the English translations are available at the KTA website: 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 the present 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 Requirements for Nuclear Power Plants (SiAnf) and the SiAnf-Interpretations.

(2) The present safety standard details the required precautionary measures with respect to maintaining criticality safety during refueling. It is specified under Sec. 3.10 para. (1) and (2) of SiAnf that, during all operating phases, the control of reactivity shall be ensured regarding fuel handling and fuel storage, and that such measures and equipment for handling and storage of the fuel elements shall be provided such that a criticality event does not need to be assumed to occur in the storage facilities even under design basis accident conditions.

(3) In this context, the present safety standard adopts the following safety approach which is based on the defense-in-depth concept:

a) A strict application of the single event criterion to every step of planning and executing the refueling process with the result that an incorrect positioning could only happen if at least two mutually independent event sequences would occur which themselves, however, are not expected to occur during specified normal planning and executing of the refueling process.

b) Creating specific instructions regarding the refueling process (e.g., the exclusive use of absolute coordinates when shuffling fuel assemblies) with the goal of ensuring that any incorrect positioning of a fuel assembly occurring despite the provisions specified under item a) is detected and corrected before the end of the refueling process.

c) With the additional requirement that subcriticality has to be ensured at all times during the refueling process even in case of a postulated incorrect positioning of a fuel assembly, criticality safety is normally ensured above and beyond the specific design basis of strictly applying the single event criterion to every step of planning and executing the refueling process.

d) Additional monitoring measures during or close to the end of fuel loading (e.g. final core check) are required to ensure – in conjunction with the requirements under items a) and b) – that a plant start-up with an incorrect positioning of fuel assemblies is prevented.

(4) The approach described under para. (3) is meant to ensure that the reactor is operated exactly with that reactor core that is specified in the permit for plant start-up after refueling.

(5) The present safety standard supplements the requirements specified in safety standard KTA 3602. While KTA 3602 focusses mainly on the handling and storage of fuel assemblies outside of the reactor pressure vessel, the present safety standard specifies the requirements pertaining to maintaining criticality safety during the refueling process of shuffling fuel assemblies and associated items between the reactor pressure vessel and the fuel pool.

1 Scope

(1) This safety standard applies to the planning and execution of the handling of fuel assemblies and associated items during the refueling process in nuclear power plants with light water reactors.

(2) This safety standard specifies the requirements regarding the maintaining of criticality safety during refueling in the time span between opening and closing of the reactor pressure vessel.

(3) The handling of fuel assemblies and associated items during the refueling process comprises the following tasks:

a) unloading fuel assemblies from the reactor, loading fuel assemblies into the reactor and, if applicable, shuffling of fuel assemblies within the reactor,

b) shuffling and exchange of the associated items, and

c) monitoring the fuel loading process and functional checks,

d) additional tasks relevant to the planning and execution such as

   da) inspection of fuel assemblies or associated items,

   db) repair of fuel assemblies or associated items,

   dc) measures particular to pressurized water reactor plants in conjunction with the use of demineralized water (e.g., minimizing the radiation exposure from a release of aerosols).

Note:

Except when "refueling in accordance with Section 1 para. (2)" is explicitly used, the term "refueling" is used in the following – for the sake of simplicity – implies the "handling of fuel assemblies and associated items during refueling".

2 Definitions

(1) Fuel loading schedule (for one core loading)

The fuel loading schedule (for one core loading) describes the position and orientation of the fuel assemblies and associated items in the reactor core for one cycle (i.e., one operating period).

Note:

In practice, the term "fuel loading schedule" is often used for the entirety of safety-related certifying documents of the cycle-specific core loading.

(2) Benchmark

Benchmarks are the experiments and measurements performed on reference assemblies to validate an analysis model or simulation model for a specific case of application.

(3) Benchmark, theoretical

Theoretical benchmarks are reference solutions of hypothetical arrangements that are calculated exclusively for the comparison of different analysis or simulation models as well as for the sensitivity analyses.

(4) Bias (systematic deviation)

Bias (systematic deviation) is defined as the deviation, $b_N(\theta) = E[\theta] - \theta_W$, of the expected value, $E[\theta]$, of an estimation function, $\theta = \theta(X_1, \ldots, X_N)$, based on a number, N, of observations with the results $(X_1, \ldots, X_N)$ of a parameter θ, from its true value, θ_W.

Note:

Generally, neither the expected value of the estimation function, θ, nor the true value, θ_W, are known. Thus, the bias can usually only be estimated – cf. definition (5) of "Bias, empirical".

(5) Bias, empirical

In the present safety standard, "empirical bias" of a parameter θ is understood to be the estimation $b_n(\theta) = \Phi(b_1,\ldots,b_n)$ of the bias defined under definition (4). The empirical bias is the result of applying a statistical procedure, Φ (e.g., generation of the arithmetic or weighted average value), to the observed differences, $b_i = \theta_i - \theta_i^R$ with i = 1, …,n , between the results, θ_i, obtained from the n number of reference measurements or theoretical benchmarks for the parameter, θ, and the reference solution, θ_i^R, given for these reference measurements or theoretical benchmarks.

Notes:

(1) Typical examples of empirical bias values are the estimated value of the systematic deviation of an analysis system, or the estimated value of the systematic deviation of a measurement procedure.

(2) A given reference solution for a particular reference measurement or benchmark is, generally, not identical with the true value, θ_w, of the respective parameter, θ, for this reference measurement or benchmark. This is due to the reference solution being itself the – possibly methodically best – estimate of the parameter, θ; the true value, θ_w, generally, remains unknown. Therefore, a bias may be attached to a reference solution. This leads to the necessity that (e.g., for the validation of an analysis system) a larger number, n, of reference measurements or benchmarks should be evaluated, provided, the benchmark and case of application are not identical.

(3) The necessity of applying the statistical procedure, Φ, for determining the empirical bias implies that the statistical uncertainty of the empirical bias calculated by this procedure must be accounted for.

(6) Double event principle

According to the double event principle it would take at least two mutually independent, simultaneous and, during a specified normal refueling process not to be expected events for an inadmissible condition to occur (e.g., an incorrect positioning of a fuel assembly in the reactor)

Note:

This principle is often referred to as “double accident principle” even though, in the practice of criticality safety design it is not only applied to accidents but also to single failures (which, e.g., may lead to a redundant and diverse design of system components of safety equipment that must serve to ensure criticality safety during specified normal operation or design basis accidents). Thus, the term “double event principle” better describes its application in the practice of criticality safety design than would the term “double accident principle”; moreover, it is in conformance with the term “double contingency principle” used in Anglo-Saxon standards and safety guidelines.

(7) Three-quarter core loading (or ¾-core loading)

The term “three-quarter core loading” is a condition during fuel loading of a boiling water reactor core that – with the exception of the unaltered control rod cells – is characterized by having all control rods installed and fully inserted and each control rod cell having exactly three out of four fuel assembly positions loaded with fuel assemblies in accordance with the fuel loading schedule.

(8) Control rod cell, unaltered

The unaltered control rod cell is a control rod cell in a boiling water reactor core the fuel loading of which is not changed from the previous cycle when going to the following cycle.

(9) Single event criterion

The single event criterion requires that a single deviation from the specified normal refueling process will not lead to an inadmissible condition (e.g., an incorrect positioning of a fuel assembly in the reactor).

Notes:

(1) To fulfill the single event criterion, the double event principle is often applied.

(2) Examples for events that must fulfill the single event criterion are the failure of system components, the failure of a safety-related measure, but also human errors such as not abiding to administrative instructions.

(3) The term “single event criterion” is different from the term “single failure”, the latter being defined in Appendix 4 of SiAnf.

(10) Unloading schedule (unloading of a PWR reactor core)

The unloading schedule specifies the sequence by which fuel assemblies are unloaded from their fuel assembly positions in a pressurized water reactor core.

(11) Event sequence

An event sequence starts with an initiating event and, following the principle of causality, continues with the sequential events in chronological order.

Note:

In the present safety standard, the term “event sequence” is also used for the special case that an initiating event does not lead to any sequential events.

(12) Misloading

A misloading is the situation where at least one fuel assembly or one associated item was inserted at a location into which it should not have been placed in accordance with the applicable requirements under Section 3.

(13) Incorrect positioning

An incorrect positioning is the shuffling of a fuel element or an associated item into a location into which it should not have been shuffled in accordance with the applicable requirements under Section 3.

(14) Qualified person

A qualified person is a person who has sufficient knowledge in a specific field of expertise, i.e., who has sufficient knowledge and skills to perform the specified tasks in a professional and proper way.

Note:

The person does not need to have a complete overview of the entire associated field of expertise.

(15) Step sequence plan

A step sequence plan is plan for a specific shuffling action, e.g., the unloading of a reactor core, that specifies the sequence for shuffling the fuel assemblies or associated items from their individual position of origin to the goal position.

(16) Safety distance to criticality, Δk_S

The term “safety distance to criticality, Δk_S” is understood as being the minimum distance of the neutron multiplication factor to the point of criticality that must be maintained to ensure criticality safety.

(17) Control rod cell (BWR)

A control rod cell is the square arrangement of those four fuel assemblies in the reactor pressure vessel that are adjacent to a control rod.

Note:

The control rod cell is often termed “core cell”.

(18) Validation (of an analysis model or simulation model)

Validation is the process of determining the degree of accuracy with which an analysis or simulation model – from the view point of the intended application of this model – describes the reality, or sufficiently represents the reality, or virtually performs the simulated function.

N o t e s :

(1) In the case of an analysis model meant to determine the values of continuous parameters, θ, the degree of accuracy is given by the empirical bias, $b_n(\theta) = \Phi(b_1,\ldots,b_n)$. Since $b_n(\theta)$ is a statistical value, the degree of accuracy is, likewise, a statistical value.

(2) In the case of simulation models intended for the virtual performance of certain functions, the degree of accuracy of the function can be expressed as the percentage of that portion of the function that was performed without error.

(19) Verification (of an analysis or simulation model)

Verification is the process of demonstrating that the analysis or simulation model performs exactly as expected according to the conceptual description and specification presented by the developer of the model.

(20) Associated items

The term “associated items” pertains to the core internals and other components like, e.g., control assemblies, flow restrictor assemblies, poison and dummy assemblies, fuel channels and fuel channel mountings, neutron sources, neutron absorbing inserts in fuel assemblies, and the detector assemblies. Associated items of fuel assemblies are employed inside the reactor core or for the handling of fuel assemblies.

3 General Requirements

(1) Every one of the work steps, tools and auxiliary devices necessary for refueling in accordance with Section 1 para. (2) shall be specified in a written plant regulation.

N o t e :

Appendix 3 Sec. 2 para. (6) of SiAnf specifies requirements for administrative measures that can also be applied analogously to refueling procedures; these are accounted for in the present safety standard.

(2) The actual fuel loading of the reactor and of the storage pools for fuel assemblies and associated items shall be recorded in arrangement diagrams that indicate the identification markings and locations of the fuel assemblies and associated items. Any other materials stored in the storage racks shall also be indicated in the arrangement diagrams.

(3) It shall be ensured that the fuel loading schedule intended for the reactor fulfills the safety-related requirements in accordance with the respective Secs. 3 of safety standards KTA 3101.1, KTA 3101.2 and KTA 3101.3 (the latter in preparation). If, during refueling, a deviation from the fuel loading schedule becomes necessary, the requirements specified under Section 6.3.1 paras. (8) and (9) shall be met.

(4) It shall be ensured that, during refueling, the criticality safety is upheld both in the reactor as specified under Section 5 as well as in the fuel storage pools in accordance with Sec. 4.6.2 of safety standard KTA 3602.

(5) When handling and shuffling fuel assemblies and associated items in the fuel storage pool in preparation for refueling in accordance with Section 1 para. (2) as well as during and after completion of this refueling, the requirements in accordance with Sec. 4.4.1 paras. (6) and (9) of safety standard KTA 3602 shall be fulfilled.

N o t e s :

(1) Sec. 4.4.1 para. (6) of KTA 3602 specifies that without a special safety certification no more than one fuel assembly at a time may be handled in the fuel storage pool. However, it is admissible to simultaneously handle multiple fuel assemblies in the inspection and repair equipment as well as, in the case of BWR fuel assemblies, in the channel stripping machine.

(2) Sec. 4.4.1 para. (9) of KTA 3602 specifies the requirements that must be met in case of a multi-zone fuel storage pool that make it impossible for a fuel assembly to be incorrectly positioned in a zone different from the operative zone.

(6) The handling of fuel assemblies and associated items in reactor pressure vessels and fuel storage pools requires a written order and a step sequence plan released for execution that shall fulfill the requirement specified under Section 4 paras. (3) through (9). In the case of multiple step sequence plans, the conditions for the sequential application of the different step sequence plans shall be precisely described.

(7) The handling operations shall be performed by a qualified person who shall be supervised by a qualified responsible person on-site (on-site supervisor).

(8) It shall be ensured that every individual handling and shuffling of a fuel assembly or associated item meets the single event criterion such that, in case of a single deviation from the specified normal refueling process, this will not lead to an inadmissible condition (e.g., incorrect positioning of a fuel assembly in the reactor).

N o t e :

This means that every step of the planning and execution of the refueling process is subject to the single event criterion.

(9) The arrangement diagrams specified under para. (2) shall be updated in a timely manner following the step sequence plan specified under para. (6) and shall be documented in writing.

4 Planning the Refueling Process

(1) Only qualified persons may perform the planning of refueling and of the refueling process. Parts of the planning are, among others:

a) ensuring that the requirements specified under Section 3 paras. (3) through (5) and (8) are met,

b) checking the actuality of the arrangement diagrams specified under Section 3 para. (2),

c) developing, modifying and checking the test sequence plans, and

d) specifying boundary conditions to be observed during fuel loading and shuffling of the fuel assemblies (in case of PWRs, e.g., the fuel unloading and loading schedule; in case of BWRs, e.g., the planned three-quarter core loading).

(2) Prior to planning the refueling process, the consistency between the actual fuel loading and the arrangement diagrams of the fuel storage pools shall be ensured.

(3) Step sequence plans shall be developed for the refueling based on

a) the actual arrangement diagrams,

b) the intended fuel loading schedule, and

c) the boundary conditions specified under para. (1) item d).

(4) With respect to the sequence of the steps to be performed during refueling, the step sequence plan shall be structured such that criticality safety is ensured at every step as specified under Section 3 para. (4).

(5) When creating a step sequence plan for a refueling in a BWR power plant, the requirements specified under Section 6.3.2 paras. (1), (2) and (3) shall additionally be observed.

(6) A step sequence plan shall

a) unambiguously identify the campaign to which it applies (e.g., “Refueling <calendar year>” or “Refueling after Cycle No. <n>”),

b) in its title unambiguously and summarily describe the task (e.g., “Unloading of reactor into wet storage facility” or “New fuel loading of reactor for Cycle No. <n+1>”),

c) have an unambiguous identification marking,

d) show the date of its preparation,

e) unambiguously specify the sequence of the required shuffling steps of fuel assemblies and associated items,

f) specify the following data unambiguously and unmistakably for each shuffling step:

 fa) identification of the fuel assembly or associated item to be shuffled,

 fb) position of origin where the shuffling shall begin by specifying the plant component where this position is located (e.g., “RE” for “reactor”) and the coordinates of this position,

 fc) goal position where the shuffling shall end by specifying the plant component where this position is located (e.g., “NL” for “wet storage facility””) and the coordinates of this position, and

 fd) if necessary, the change of the orientation of the fuel assembly or the associated item;

g) contain an additional entry field for each shuffling step for documenting the completion of the shuffling step as specified under Section 6.3.1 para. (6),

h) clearly specify the interruptions of the fuel loading of a BWR core that are required on account of Section 6.3.2 paras. (4) or (6) for the performance of a function and subcriticality test (FUP) or a shutdown-safety test (AST), and identify the actions, tests and releases required before the fuel loading of the core may be continued.

(7) The initial positions and the goal positions shall be identified by the absolute coordinates of the individual component. It is not admissible to describe the locations of the initial or goal positions of a shuffling step in terms of their position relative to an initial and goal position of a prior shuffling step.

(8) The visual design of the step sequence plan shall be such that clarity and good legibility are ensured regarding its use on the refueling machine.

(9) After completion of the step sequence plan, it shall be signed by the person who prepared it. Before its application the step sequence plan shall be checked by a person who was not involved in its preparation. The correctness of the step sequence plan shall be documented by a signature and the date when it was checked.

(10) Instead of manually producing the step procedure plans and arrangement diagrams and manually documenting the updated fuel loading of the reactor and of the fuel storage pools for fuel assemblies and associated items it is preferable to apply a computer program system to these tasks. This computer program system shall be verified and validated before it is first applied to the respective functions.

5 Criticality Safety in the Reactor During Refueling

5.1 Basic Requirements

(1) To ensure criticality safety, the required safety distance to criticality shall not fall below $\Delta k_S = 0.003$.

(2) It shall be demonstrated that this safety distance, Δk_S, is upheld both during the specified normal refueling process as well as in case of a single deviation from the specified normal refueling process (single event criterion); this demonstration shall be performed considering the plant-specific requirements and boundary conditions as specified under Section 5.2.

(3) When performing the demonstration that the safety distance, Δk_S, is upheld, the distance to the criticality value, Δk_{eff}, shall fulfill the inequality (5-1) under consideration of all uncertainties, Δk_U, specified under Section 5.3:

$$\Delta k_{eff} \geq \Delta k_S + \Delta k_U \qquad (5\text{-}1)$$

In this context it shall be assumed that the event sequence that was assumed in the design of refueling process but was never expected to occur during specified normal refueling, that this event sequence leading to the smallest distance to criticality does occur.

N o t e :

During specified normal refueling and complete unloading of the reactor into the wet storage facility and the subsequent new fuel loading of the reactor from the wet storage facility, the most reactive configuration is, generally, the completed and newly refueled reactor. However, this is not necessarily the case if the reactor is not completely unloaded but fuel assemblies are shuffled within the reactor: In this case arrangements of fuel assemblies are feasible that have a higher criticality than the completed and newly refueled reactor. With respect to identifying the configuration with the highest reactivity during refueling, in case of a PWR core it is also important to consider whether the presence of control assemblies in the core is considered as specified in Section 5.2.1 para. (2) item b).

(4) If the safety distance, Δk_S, is demonstrated analytically based on proven computational design procedures, the value of Δk_{eff} shall not be smaller than 0.01.

5.2 Plant-specific Requirements and Specifications

5.2.1 Pressurized water reactor plants

(1) Concentration of Boron-10 in the coolant

Regarding the concentration of Boron-10 in the water within the reactor well the following requirements and specifications apply to the demonstration that the requirement specified under Section 5.1 para. (3) is met:

a) The minimum Boron-10 concentration in the water of the reactor well shall be determined that is required to satisfy the requirement for adhering to the safety distance to criticality, Δk_S, as specified under Section 5.1 para. (3).

b) It shall be examined whether the occurrence of an event sequence must be assumed that would lead to a reduction of the Boron concentration in the water of the reactor well. If this should be the case the one sequence of all possible sequences shall be assumed that would lead to the largest reactivity increase.

(2) Control assemblies

a) When demonstrating that the requirement specified under Section 5.1 para. (3) is met it shall basically be assumed that there are no control assemblies present in the reactor core.

b) However, the presence of control assemblies in the reactor core may be taken into account, provided, for each of these individual control assemblies

 ba) it is ensured by proper measures fulfilling the single event criterion that the control assembly is truly present and correctly positioned in the reactor core, and

 bb) the control assembly’s absorber content is verifiably known or is conservatively estimated, and

 bc) the control assembly’s reactivity absorption is verifiably known or is conservatively estimated under consideration of the minimum Boron-10 concentration in the water of the reactor well as specified under para. (1) item a).

c) If the presence of control assemblies in the reactor core is taken into account it shall however be assumed that the control assembly with the highest reactivity absorption is absent. No further failure is required to be assumed for this postulated event.

(3) Postulated event of an incorrect positioned fuel assembly in the reactor core

When demonstrating that the requirement specified under Section 5.1 para. (3) is met the incorrect positioning of one fuel assembly shall be assumed. The incorrect position assumed shall be the one that would lead to the highest value of k_{eff}. No additional failure is required to be assumed for this postulated event.

5.2.2 Boiling water reactor plants

(1) When demonstrating that the requirement specified under Section 5.1 para. (3) is met the following postulated events shall be considered:

a) It shall be assumed that one control rod is absent in one otherwise completely loaded control rod cell. The control rod cell assumed in this context shall be the one in which the absence of one control rod would lead to the highest reactivity change. No additional failure is required to be assumed for this postulated event.

b) One incorrectly positioned fuel assembly shall be assumed for the procedure of completing the fuel loading of one control rod cell. The incorrect position assumed shall be the one that would lead to the highest value of k_{eff}. No additional failure is required to be assumed for this postulated event.

(2) During fuel loading of a BWR, while meeting the requirements specified under Section 6.3.2 para. (6),

a) the requirements specified under Section 5.1 para. (3) may be waived for the duration of performing a function and sub-criticality test (FUP), and

b) the requirements specified under Section 5.1 paras. (1) and (3) may be waived for the duration of performing a shutdown-safety test (AST).

Notes:

(1) During a function and sub-criticality test (FUP) the control rod of a finalized and fully loaded control rod cell is moved through its entire length while monitoring the neutron flux. Primary objectives of the FUP is to check the functionality of the control rod drives and to check for the freedom of movement of the control rods.
When performing the FUP it shall also be ensured that a shutdown margin of at least one control rod is available for the actual loading configuration.

(2) A shutdown-safety test (AST) is usually performed if, during fuel loading, a rectangular arrangement of 2-by-3 completed and fully loaded control rod cells was created of which none of the cells had previously been covered by an AST. To be covered by an AST means – depending on the plant –

a) that the quadratic arrangement of 2-by-2 control rod cells contains the two diagonally neighboring fully inserted control rods, or

b) the entire arrangement of 2-by-2 control rod cells.

During a shutdown-safety test (AST) the most reactivity-effective control rod of the arrangement is fully withdrawn while monitoring the neutron flux. Subsequently, the diagonally neighboring control rod is withdrawn to an extent that a calculated reactivity of 1 % is reached. During the AST it shall be ensured for the fully withdrawn control rod that a shutdown margin is available of more than the reactivity equivalent of this control rod.

(3) The zero power tests are used to demonstrate that a sufficient net shutdown margin is available.

(3) The type and procedure of the tests and checks specified und para. (2) shall be described and specified in a written plant regulation. The results of the tests and checks specified under para. (2) shall be documented.

5.3 Requirements for the Analytic Determination of Criticality Safety

5.3.1 Requirements for the nuclear analysis system

(1) The nuclear analysis systems used for demonstrating criticality safety shall

a) be able to calculate the relevant safety-related parameters for the cold reactor core that are defined by the arrangements of the fuel assemblies and associated items occurring during the refueling process,

b) be able to describe the neutron-physical effects of the influencing parameters specified in Section A.1 of Appendix A, and

c) be validated at least for calculating the relevant safety-related parameters according to the requirements and specifications under Section A.2 of Appendix A.

Note:

Herein, a “nuclear analysis system” is understood to comprise the entirety of program components necessary to demonstrate the criticality safety for the case of application.

(2) The analysis of the relevant safety-related parameters and the descriptions of neutron-physics effects specified in para. (1) item b) may be based on approximations and simplifications, provided, they are validated as specified in para. (1) item c). Typical approximations and simplifications are, e.g.,

a) approximations and simplifications when describing the material distribution inside the reactor,

b) approximation methods for calculating k_{eff},

c) discretization of the continuous neutron spectrum, and

d) analyzing partial areas of the reactor.

(3) An analysis system may be validated by comparison with another analysis system, provided, the latter has been validated for the case of application.

5.3.2 Requirements for demonstrating that criticality safety is ensured

(1) When demonstrating that the acceptance criterion for criticality-safety defined by inequality (5-1) is met for the case of application specified under Section 5.2, the term Δk_U of inequality (5-1) shall basically be quantified under consideration of the requirements specified in Section A.2.2 of Appendix A. The term Δk_U is made up of the following components:

a) the approximated empiric bias of the applied analysis system determined by validating the applied analysis system with respect to the case of application,

b) the uncertainties of the calculated neutron multiplication factors caused by uncertainties (variants and correlations) of the nuclear data for the case of application and for the benchmarks used for validating the applied analysis system,

c) the uncertainties of the calculated neutron multiplication factor for the case of application resulting from the uncertainties (fabrication tolerances or parameter-dependent biases, variances and correlations) of the parameters characterizing the case of application (cf. Section A.1 Appendix A), and

d) the uncertainties of the neutron multiplication factors resulting from the uncertainties (fabrication tolerances or parameter-dependent biases, variances and correlations) characterizing the parameters used for evaluating the benchmarks that in turn are used to validate the applied analysis system.

Depending on the available validation basis, the term Δk_U may be quantified either integrally (cf. para. (2)) or by determining the individual contributions.

(2) A separate analysis of the individual components of the term Δk_U of inequality (5-1) as specified in para. (1) items a) through d) may be waived, provided, the case of application and the benchmark system used as basis for the validation are directly comparable, e.g., due to the critical measurements performed in the reactor as specified in Section A.2.1 para. (4) items a) through e) of Appendix A.

Note:

If a direct comparability is given, the sum, Δk_U, of the individual components specified under para. (1) items a) through d) can be directly calculated from the difference between the computed value of the neutron multiplication factor, k_{eff}, and the expected value $E[k_{eff}] = 1$.

(3) The term Δk_U of inequality (5-1), in so far as it is a statistical value, shall be expressed as a one-sided 95 % / 95 %-tolerance boundary.

(4) Approximation methods may be applied to quantifying the components specified in para. (1) items a) through d) of the term Δk_U of inequality (5-1), provided, it is demonstrated that these approximation methods do not lead to an underestimation of the term Δk_U for the case of application.

6 Execution of Refueling

6.1 General Requirements

(1) It shall be ensured that the requirements specified under Section 3 para. (4) – ensured criticality safety – is fulfilled.

(2) The neutron count rate in the reactor shall be monitored by measurements during the fuel loading and the shuffling procedures.

Note:

In the case of the fuel loading of a PWR, the neutron count rate can be predicted in good approximation when based on the step sequence plan. Lower limit alarm thresholds for the predicted count rates (failure of the detector or the measurement electronics) and upper limit values (unexpected increase of the neutron flux) can be defined as dependent on the loading condition and loading progress. The boron concentration to be adjusted as specified under Section 6.2 para. (1) can, therefore, be reliably monitored taking the sensitivity of the measured neutron count rate to changes into account.

(3) Prior to loading the reactor, the actual fuel loading of the fuel storage pools for fuel assemblies and associated items shall be checked regarding whether they are in conformance with the arrangement diagrams. The results of this check shall be documented. If necessary, the arrangement diagrams or the loading diagram of the fuel storage pool shall be updated in accordance with the results of the checks, and the updates shall be documented.

6.2 Preparations for Refueling

(1) In case of pressurized water reactors, it shall be ensured prior to opening the connection between the reactor well and the fuel storage pool that the Boron-10 concentration of the water in the reactor well and the fuel storage pool is at least high enough to fulfill the requirement specified under Section 3 para. (4).

(2) The additional requirements in accordance with safety standard KTA 3602 Sec. 4.4.3 (Refueling) shall be met.

6.3 Refueling

6.3.1 General requirements

(1) The unloading and loading of fuel assemblies and associated items of the reactor core may only be performed by qualified persons.

(2) The handling of fuel assemblies and associated items shall be performed using a refueling machine that is in accordance with safety standard KTA 3902. Handling tasks that cannot be performed with the refueling machine shall be carried out with lifting equipment that is in accordance with safety standard KTA 3902.

(3) The travel paths of the refueling machine shall be kept clear of obstacles while the machine is in operation.

(4) In the operating area of the refueling machine no other tasks may be carried out that might endanger the work procedure.

(5) During work breaks, changes of shifts or personnel as well as during interruptions of handling tasks the grippers and load-bearing equipment shall be driven load-free into a safe position.

(6) The handling of fuel assemblies and associated items shall be performed following a step sequence plan that shall meet the requirements specified under Section 4 paras. (3) through (9). Each completed handling operation shall be documented in the step sequence plan by the date, time and personal signature of the qualified responsible person on-site (on-site supervisor).

Note:

Modern refueling machines can accept the step sequence plan in electronic (digital) form. This plan is visualized on the monitor of the refueling machine. The shuffling procedure can be executed in fail-safe manner and in accordance with this step sequence plan by the controls of the refueling machine.

(7) Permissible deviations from the step sequence plan in the case of handling problems shall be regulated in advance by an operating instruction. This operating instruction shall meet the requirements specified under Section 3 para. (8) (single event criterion) and under Section 5.1 paras. (1), (2) and (3) (criticality safety). At completion of the core fuel loading, this loading shall be in accordance with the fuel loading schedule. The deviation of the step sequence shall be documented (in the step sequence plan) by the date, time and personal signature of the qualified responsible person on-site (on-site supervisor).

(8) The procedure that must be followed in case of deviations from the fuel loading schedule necessitated by the unavailability of fuel assemblies or associated items (e.g., due to damages to fuel assemblies or associated items) and that, with regard to criticality safety must be compatible with the original fuel loading schedule, shall be regulated in advance by an operating instruction. This operating instruction shall meet the requirements specified under Section 3 para. (8) (single event criterion) and under Section 5.1 paras. (1), (2) and (3) (criticality safety). The operating instruction shall regulate the responsibilities for the preparation and documentation of the changed fuel loading schedule and step sequence plan.

Note:

A compatible fuel loading schedule may, e.g., be one in which one reactive assembly is replaced by a less reactive one.

(9) If changes of the fuel loading schedule are necessary that do not meet the requirements specified under para. (8), then

a) the fuel loading of the reactor core shall be aborted,

b) the reason for the changes shall be documented,

c) the arrangement diagrams specified under Section 3 para. (9) shall be updated, and

d) the further procedure for the fuel loading of the reactor core shall be planned as specified in the present safety standard.

(10) The fuel loading of the reactor shall be checked for its conformity with the valid fuel loading schedule regarding position and orientation of the fuel assemblies and, in case of a PWR, of the associated items, and the result of this check shall be documented (final core check).

6.3.2 Specific requirements for boiling water reactor plants

(1) Prior to starting the refueling, the control rod drives shall be electrically deactivated. The activation of individual drives for performing specific tasks (e.g., sub-criticality test (FUP) or shutdown-safety test (AST)) shall be regulated in an operating instruction.

(2) Before dismantling a control rod drive or a control rod, or before retracting a control rod, the fuel assemblies of the associated control rod cell shall basically all be removed. Exceptions of this requirement are specified under para. (4) and (7).

(3) An operating instruction shall be created that meets the single event criterion and ensures that the control rod itself cannot be moved when dismantling the control rod drive. If this is impossible, the control rod cell shall be prepared by fuel assembly unloading such that the requirements regarding criticality safety specified under Section 5.1 para. (1) are met – even under the assumption of a complete withdrawal of the control rod of the associated control rod cell.

(4) When unloading the entire reactor core, it is admissible in deviation from para. (2) to withdraw the control rods already when a configuration of the core is reached for which it has been demonstrated as specified under Section 5.1 paras. (1), (2) and (3) – e.g., checkerboard pattern – that subcriticality is maintained without control rods.

(5) Prior to loading a fuel assembly into a control rod cell, it shall be ensured that the control rod of this cell has been installed and is fully inserted.

(6) When installing a control rod, the identity of the control rod shall be checked, and the result of this identity check documented.

Note:

In the case of the BWR the control rod identity cannot be determined during the freedom-of-movement check specified under Section 6.4 para. (1). That is why the identity check is performed during installation of the control rod.

(7) When executing a sub-criticality test (FUP) or a shutdown-safety test (AST) the requirement under para. (2) is waived. However, its shall be ensured by control measures that meet the single event criterion that

a) the individual control rod cell or multiple control rod cells to be tested are properly loaded with fuel assemblies, i.e., that there is no incorrect positioning of the fuel assemblies,

b) all control assemblies are fully inserted,

c) the reactor protection system is operational, and

d) the fast shutdown system is operational.

6.4 Measures Prior to Closing the Reactor Pressure Vessel

(1) The freedom of movement of the control assemblies shall be checked and documented

a) in a PWR – after the upper core structure has been installed and the control assembly drive rods have been coupled,

b) in a BWR – unless this was already performed during the sub-criticality test (FUP) before completion of fuel loading into the reactor core.

(2) When using demineralized water for cleaning the walls of the reactor well (PWR) the requirements under Section 5.1 paras. (1), (2) and (3) (criticality safety) shall be met.

Appendix A

Performing the analytic demonstration of criticality safety for light water reactors during refueling

A.1 Influencing Parameters
(details regarding the requirement specified under Section 5.3.1 para. (1) item b))

The nuclear analysis system used for demonstrating criticality safety of light water reactors during refueling shall be able to describe the neutron-physical effects of the following influencing parameters:

1) Materials, dimensions and structural design of fuel assemblies:
   a) Geometric structure of fuel assemblies:
   dimensions and arrangement of the fuel elements, guide thimbles, water pipes, water channels (BWR), fuel channels (BWR) and, if applicable, other structural parts
   b) Nuclear fuel data:
   nuclide inventory of new fuel assemblies taking the possibly present burnable neutron absorbers into account, nuclide inventory of irradiated fuel assemblies, and spatial distribution of the respective nuclide inventories
   c) Materials of the fuel rod cladding, of the guide thimbles, water pipes, water channels (BWR), fuel channels (BWR) and, if applicable, other structural parts
2) Materials, dimensions and structural design of control assemblies
3) Materials, dimensions, structural design and arrangement of the other possibly employed absorber rods
4) Materials, dimensions, structural design and arrangement of associated items in so far as these have any influence on the neutron flux distribution
5) Isotopic composition and concentration of the Boron dissolved in the moderator (PWR)
6) Materials, dimensions and arrangement of the reflector surrounding the reactor core
7) Changes of the arrangement of fuel assemblies, control assemblies and associated items in the reactor core occurring during the refueling process, and the neutron-physical effects on the moderating, absorbing and reflecting conditions caused by these changes
8) Control rod positions (BWR)
9) Changes of the moderator temperature occurring during refueling
10) Changes of nuclide inventory during plant shutdown caused by radioactive decay

A.2 Validation of a Nuclear Analysis System Applied to the Analytic Verification of Criticality Safety of Light Water Reactor Cores during Refueling
(details regarding the requirement specified under Section 5.3.1 para. (1) item c))

A.2.1 Criteria for the Choice of Benchmarks for Validating a Nuclear Analysis System

(1) The validation of a nuclear analysis system shall normally be based on the comparisons of calculations with the measurement results from

a) experiments and reference measurements as specified under para. (3),
b) critical zero-load measurements as specified under para. (4)
c) post-irradiation measurements.

(2) When choosing the experimental arrangements and reference measurements specified under para. (3) for the validation of the analyses performed for the case of application, their neutron-physical similarity to the case of application is of major importance. If the similarity is only slight or if no representative measurement values specified under para. (1) exist for the case of application then

a) comparative sensitivity analyses shall be performed between the case of application and available benchmarks regarding the influencing parameters specified under Section A.1 Nos. 1) through 10), or
b) theoretical benchmarks shall be applied that are based on the comparative sensitivity analyses.

Note:

The term “neutron-physical similarity” is defined in DIN 25478 Supplement 1.

(3) Among the experimental arrangements and reference measurements that can be used for validating the analyses are, particularly, those critical arrangements

a) for which not only the neutron multiplication was measured but for which additional measurements of the microscopic flux and reaction rate distribution as well as of the macroscopic flux density distribution were performed,
b) with which the reactivity values of irradiated nuclear fuels relative to the corresponding new nuclear fuels were measured,
c) with which the reactivity values of burnable neutron absorbers in new nuclear fuels were measured,
d) with which the reactivity values of those neutron absorbers in irradiated nuclear fuels were measured that were created by irradiation and that contribute significantly to the reactivity absorption of the irradiated nuclear fuel,

e) that use solid neutron absorbers in the form of rods or plates within or between the fuel rod grids,

f) in which the moderator contains dissolved Boron (PWR),

g) in which the fuel rod grid is enclosed by a solid reflector (in particular steel as the reflector material in combination with the moderator between the fuel rod grids and the reflector).

(4) Among the critical zero-load measurements on the cold and Xenon-free reactor core are, particularly, the following:

a) measurements of critical control rod positions,

b) measurements of differential effectiveness of the control rod,

c) measurements of the differential effectiveness of one control rod group

d) measurements of temperature coefficients,

e) measurements of critical Boron concentrations of the core with various control rod configurations (PWR).

A.2.2 Statistical Correlation of Benchmark Results

The validation results from different benchmark experiments shall be statistically correlated if they are dependent on mutual influencing parameters subject to uncertainties.

Note:
Details are specified in DIN 25478 Supplement 1.

Appendix B

Regulations Referred to in the Present Safety Standard

(Regulations referred to in the present 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 of the Act of August 28, 2013 (BGBl. I 2013, No. 52, p. 3313)
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 of the Act of February 24, 2012 (BGBl. I, p. 212)
SiAnf	(2012-11)	Safety requirements for nuclear power plants of November 22, 2012 (BAnz of January 24, 2013)
SiAnf- Interpretations	(2013-11)	Interpretations of the “Safety requirements for nuclear power plants of November 22, 2012” of November 29, 2013 (BAnz of December 10, 2013)
KTA 3101.1	(2012-11)	Design of reactor cores of pressurized water and boiling water reactors; Part 1: Principles of thermohydraulic design
KTA 3101.2	(2012-11)	Design of reactor cores of pressurized water and boiling water reactors; Part 2: Neutron-physical requirements for design and operation of the reactor core and adjacent systems
KTA 3101.3 (draft)	(2014-11)	Design of reactor cores of pressurized water and boiling water reactors; Part 3: Mechanical and thermal design – Draft Standard
KTA 3602	(2003-11)	Storage and handling of fuel assemblies and associated items in nuclear power plants with light water reactors
KTA 3902	(2012-11)	Design of lifting equipment in nuclear power plants
DIN 25478, Supplement 1	(2012-09)	Application of computer codes for the assessment of criticality safety - Supplement 1: Explanations