Safety Standards

of the
Nuclear Safety Standards Commission (KTA)

KTA 3101.3 (2015-11)

Design of Reactor Cores of Pressurized Water and Boiling Water Reactors;
Part 3: Mechanical and Thermal Design

(Auslegung der Reaktorkerne von Druck- und Siedewasser-reaktoren; Teil 3: Mechanische und thermische Auslegung

Please note:
This translation includes the correction published in BAnz of March 10th, 2017.

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KTA SAFETY STANDARD

2015-11	Design of Reactor Cores of Pressurized Water and Boiling Water Reactors; Part 3: Mechanical and Thermal Design	KTA 3101.3

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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 Bundesanzeiger BAnz of January 1st 2016; a correction was published in BAnz of March 10th 2017. Copies may be ordered through the Carl Heymanns Verlag KG, Luxemburger Str. 449, D-50939 Koeln (Telefax +49-221-94373-603).

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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, the 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 objective to specify safety-related requirements, compliance of which provides the necessary precautions in accordance with the state of the art in science and technology against damage arising from the construction and operation of the facility (Sec. 7 para. 2 subpara. 3 Atomic Energy Act - AtG) in order to achieve the fundamental safety functions specified in the Atomic Energy Act and the Radiological Protection Ordinance (StrlSchV) and further detailed in the Safety Requirements for Nuclear Power Plants as well as in the Interpretations on the Safety Requirements for Nuclear Power Plants.

(2) The tasks of the KTA 3103 series are to set the requirements for the design of reactor cores of pressurized and boiling water reactors. The KTA 3103 series are subdivided into three parts:

Part 1: Principles of Thermohydraulic Design,

Part 2: Neutron-Physical Requirements for Design and Operation of the Reactor Core and Adjacent Systems and

Part 3: Mechanical and Thermal Design

(3) This part of the KTA 3103 series details the precautions as per (1) for nuclear power plants with respect to the mechanical, thermal and thermomechanical design.

Note:

In the following, the aspects of the mechanical, thermal and thermomechanical design are comprised to call it thermomechanical design.

1 Scope

(1) This safety standard applies to stationary nuclear power plants with pressurized or boiling water reactors. It deals with the requirements for the thermomechanical design of core components for specified normal operation, accidents, very rare events (anticipated transient without scram / ATWS), internal events, external events (design basis earthquake) as well as for emergency cases (explosion blast wave and aircraft crash). In addition, this standard covers the requirements for operational planning and operation arising out of design.

(2) The scope of this safety standard also covers requirements for the manufacture and transportation of non-irradiated core components as well as the manageability and storage suitability of the core components in nuclear power plants.

(3) This safety standard does not cover requirements for the

a) thermo-hydraulic and nuclear design of the reactor core laid down in KTA 3101.3 and 3101.2, respectively,

b) mechanical and thermal design of pressurized walls of the core instrumentation laid down in KTA 3201.2,

c) design of load attaching points of core components laid down in KTA 3905,

d) nuclear criticality safety in case of storage and handling laid down in KTA 3602,

e) nuclear criticality safety in case of refuelling laid down in KTA 3107,

f) core components during temporary storage in power-plant external facilities and final disposal.

(4) This safety standard does neither deal with neutron sources, poisoning elements (absorber elements), dummy elements, neutron-absorbing internals nor with core instrumentamentation.

2 Definitions

(1) Absorber assembly (PWR)

Absorber assemblies are used in the first PWR cycle for compensation of excessive reactivity and for maintaining a negative moderator-temperature coefficient in the reactor core.

(2) Safety-related requirements

Safety-related requirements are requirements to detail the acceptance targets of the Safety Requirements for Nuclear Power Plants referred to core components.

(3) Acceptance criterion

An acceptance criterion is a criterion the fulfilment of which has to be demonstrated in the course of the safety demonstration.

(4) Acceptance target

An acceptance target is a safety-related objective of the safety demonstration which is reached by meeting acceptance criteria.

(5) Component part

Component part is a part of a component defined separately according to structural or functional aspects.

(6) Fuel assembly

A fuel assembly consists of parts, e.g. the spacers and the fuel rods. All parts of the fuel assembly except for the fuel rods form the fuel assembly structure.

(7) Fuel rod

A fuel rod is a gas-tight metal tube closed on both ends and filled with nuclear fuel.

(8) Fuel rod cladding

The term fuel rod cladding in the following means the fuel rod cladding tube including the end plugs and all welds present.

(9) Flow restrictor assembly

The flow restrictor assembly consists of an end piece with flow restrictor plugs attached which protrude into the guide tubes of a PWR fuel assembly for the purpose of limiting the coolant flow rate.

(10) Event, external

An external event means external forces or fluids with physical or chemical influences, or a combination thereof, acting upon components or component parts.

(11) Event, internal

An internal event means component-internal induced forces or fluids with physical or chemical influences, or a combination thereof, acting upon components or component parts.

(12) Fretting

Fretting means material wear occurring at the contact surface between two parts under load at relative movement.

(13) Manufacture

Manufacture is the entirety of all fabrication and testing steps necessary for the implementation of a design into a product.

(14) Core component

A Core component is a component part or component of which the reactor core is composed. These include: fuel assemblies, control assemblies, flow restrictors, poisoning and dummy elements, fuel assembly cassettes and cassette fasteners, neutron sources, neutron-absorbing devices of the fuel assemblies and detector assemblies.

(15) Component

A component is a part of a system, in the present case of the reactor core, defined in terms of structural or functional criteria. The components of the reactor core are designated as core components.

(16) Storage suitability

For the purpose of this safety standard core components can be stored in a suitable manner if they can be stored without limitation at the intended locations in the nuclear power plant for further use. Intended locations are e.g. the incoming goods reception store (for core components without nuclear fuel), the dry storage site (for fuel assemblies prior to irradiation) and the wet storage site.

Note:

This safety standard does not consider criticality and radiation protection aspects.

(17) Storage

For the purpose of this safety standard, the storage of core components comprises the provision of components in the nuclear power plant prior to their use as well as the storage between periods of use or upon final unloading up to the transportation for conditioning or to a separate temporary storage facility or final repository.

(18) Level of defence

A level of defence covers a category of plant conditions with defined boundary conditions of similar type:

a) Level of defence 1: normal operation
b) Level of defence 2: abnormal operation
c) Level of defence 3: accident
d) Level of defence 4a: very rare events.

(19) Control assembly

A control assembly consists of the control assembly structure (load bearing structure) as well as of absorber-containing parts.

Note:

In the case of BWR's the total control assembly is also called control rod.

(20) Validation

Validation is a process to demonstrate that the properties of a model reproduce the real conditions to be modelled (e.g. physical or chemical conditions/occurrences) with sufficient precision with respect to the intended use of the model.

(21) Verification

Verification is a process to demonstrate that the implemented design model correctly reproduces the conceptual model description (given specification).

(22) Failure limit

Failure limit is a quantitative criterion where, when being exceeded, the component does no more satisfy the conditions required (e.g. leak-tightness, functional capability).

3 Basic requirements for the thermomechanical design of core components

3.1 General

(1) Based on the safety concept of the Safety Requirements for Nuclear Power Plants and the fundamental safety functions and safety objectives formulated therein, this safety standard concretizes the specific acceptance targets and criteria for the levels of defence covered by this safety standard (see **Figure 3.1-1**).

(2) The requirements formulated in this safety standard apply to normal operation (level of defence 1), anomalous operation (level of defence 2), accidents (level of defence 3), the very rare events to be considered here (anticipated transient without scram / ATWS on level of defence 4a) as well as internal events and external events (design basis earthquake) and for emergency cases (explosion blast wave and aircraft crash). As far as different requirements have to be satisfied on the specific levels of defence, this will be indicated in this standard.

(3) For internal events and external events as well as for emergency cases that have not been categorized in specific levels of defence in overriding regulations, this safety standard lays down the same requirements as for level of defence 3.

(4) The core components shall be designed and operated such that, graded to the pertinent safety requirements of levels of defence 1 to 4a, the fundamental safety functions

- cooling of fuel assemblies (K),
- reactivity control (R),
- confinement of radioactive materials (B)

as well as the fundamental radiological safety objective

- limitation of radiation exposure (S)

are satisfied.

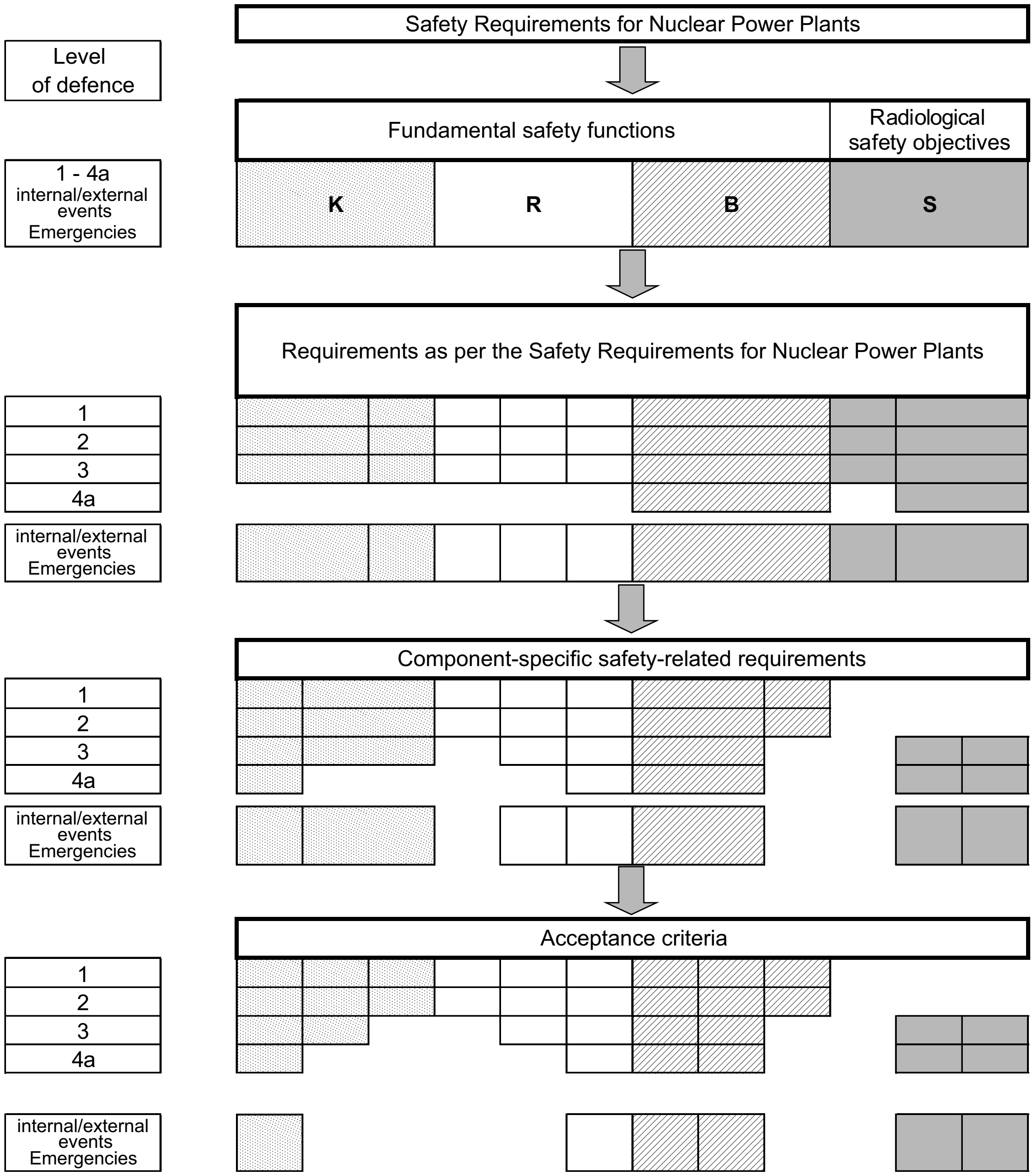
(5) The design and operation of core components shall be such that the mechanical, thermal, chemical and radiation-induced loadings resulting from external and internal events to be expected can be safely transferred.

(6) The loadings are subject to the operating conditions at level of defence 1 and the postulated events at levels of defence 2 to 4a. The safety-related requirements are derived from the fundamental safety functions and the radiological safety objective and may differ at the 4 levels of defence.

(7) In section 3.2 the component-specific safety-related requirements derived for core components from the radiological safety objective and the fundamental safety functions are laid down and are staggered at levels of defence. From these requirements acceptance criteria are derived in Section 4, the observance of which – with respect to each level of defence – ensures the fulfilment of the acceptance targets and thus of the safety-related requirements.

Note:

Annex A contains a table comparing the safety-related requirements for core components with overriding requirements (safety objective, fundamental safety functions) and acceptance targets laid down in the Safety Requirements for Nuclear Power Plants.



K - Cooling of fuel assemblies
R - Reactivity control
B - Confinement of radiological materials
S - Compliance with fundamental radiological safety objectives

Figure 3.1-1 Schematic representation of the safety concept for the levels of defence covered by this safety standard

N o t e :

This is a schematic representation of the hierarchical connections. Boxes left blank indicate that not all combinations have requirements.

3.2 Safety-related requirements

3.2.1 Fuel assemblies (including fuel assembly channel in the case of BWR)

3.2.1.1 Level of defence 1 (specified normal operation) and level of defence 2 (anomalous operation)

(1) The safety-related requirements for levels of defence 1 and 2 can be comprised as follows:

a) The condition and operation of the fuel assemblies shall be such that the fuel assembly geometry (shape and position) required for reactivity control and the required material properties of the fuel assemblies are adhered to.

b) The condition and operation of the fuel assemblies shall be such that the allowable values of power and power density are adhered to.

N o t e :

As regards the fulfillment of this requirement see also sections 3.2 and 3.3 of KTA 3101.2.

c) The condition and operation of the fuel assemblies shall be such that the geometry (shape and position) required for heat removal and the required material properties of the fuel assemblies are adhered to.

d) The condition and operation of the fuel assemblies shall be such that the requirements laid down in KTA 3101.1 for critical boiling conditions/heat flux densities are adhered to.

Note:

See sections 3.2 and 3.3 of KTA 3101.1 as regards the fulfilment of the requirements for critical boiling conditions/heat flux densities.

e) The condition and operation of the fuel assemblies shall be such that tightness of the fuel elements is ensured.

Note:

As experience has shown fuel rod leakage cannot be completely excluded even if the design has been made most carefully. In case of such events it has to be laid down for each individual case how to proceed further. The plant shall be designed such that in case of fuel rod leakage with minor release into the primary circuit the allowable release rate of radioactive products to the environment is not exceeded.

f) Utilization of suitable materials to minimize radiation exposure (ALARA).

(2) Feedback effects onto adjacent core components, RPV internals and other plant systems are permitted as long as their safety-related requirements are met.

(3) The safety-related requirements for levels of defence 1 and 2 including the fundamental safety functions are shown in **Annex A**, Table A 1-1.

3.2.1.2 Level of defence 3 (accidents)

(1) The safety-related requirements for level of defence 3 can be comprised as follows:

a) The design of the fuel assemblies shall be such that the geometry (shape and position) required for reactivity and power density control and the required material properties are adhered to.
b) The design of the fuel assemblies shall be such that the geometry (shape and position) required for heat removal and the required material properties or the fuel elements are adhered to.
c) The design of the fuel assemblies shall be such that the event-related requirements for the tightness of the fuel rods is ensured.

Note:

The event-specific requirements are concretized in Section 4.

(2) Feedback effects onto adjacent core components, RPV internals and other plant systems are permitted as long as their safety-related requirements are met.

(3) The safety-related requirements for level of defence 3 including the fundamental safety functions are shown in **Annex A**, Table A 1-2.

3.2.1.3 Level of defence 4a

(1) The safety-related requirements for level of defence 4a can be comprised as follows:

a) The design of the fuel assemblies shall be such that the geometry (shape and position) required for reactivity and power density control and the required material properties are adhered to.
b) The design of the fuel assemblies shall be such that the geometry (shape and position) required for heat removal and the required material properties of the fuel elements are adhered to.

(2) Feedback effects onto adjacent core components, RPV internals and other plant systems are permitted as long as their safety-related requirements are met.

(3) The safety-related requirements for level of defence 4a including the fundamental safety functions are shown in **Annex A**, Table A 1-3.

3.2.2 Control assemblies

3.2.2.1 Level of defence 1 (specified normal operation) and level of defence 2 (anomalous operation)

(1) The safety-related requirements for levels of defence 1 and 2 can be comprised as follows:

a) The design and operation of the control assemblies shall be such that the geometry and shape of the control assemblies required for power control and shutdown including the quantity, geometry (shape and position) and the material properties of the absorber material satisfy the requirements of KTA 3101.2 and KTA 3103.
b) The design and operation of the control assemblies shall be such that, in consideration of their dead weight and the loadings occurring, they can be inserted at sufficient rate into the reactor core in accordance with the requirements of KTA 3101.2 and KTA 3103.
c) The condition and operation of the control assemblies shall be such that the geometry (shape and position) required for heat removal from the fuel assemblies and the required material properties are adhered to.
d) Utilization of suitable materials to keep radiation exposure as low as reasonably acceptable (ALARA).
e) The condition and operation of the control assemblies shall be such that no inadmissible radiation exposure is caused by the control assemblies.

Notes:

(1) The admissible radiation exposure values can be taken from the Radiation Protection Ordinance and – where available – from the limit values authorized for the plant by the licensing authority.

(2) Radiation exposure may arise e.g. from the selection of material or, in case of BWR, due to tritium loading in case of leakage of the absorber cover of a control rod with boron as absorber material.

(2) Feedback effects onto adjacent core components, RPV internals and other plant systems are permitted as long as their safety-related requirements are met.

(3) The safety-related requirements for levels of defence 1 and 2 including the fundamental safety functions are shown in **Annex A**, Table A 2-1.

3.2.2.2 Level of defence 3 (accidents)

(1) The safety-related requirements for level of defence 3 can be comprised as follows:

a) The design of the control assemblies shall be such that the geometry and shape of the control assemblies required for shutdown including the quantity, geometry (shape and position) and the material properties of the absorber material satisfy the requirements of KTA 3101.2 and KTA 3103.
b) The design of the control assemblies shall be such that, in consideration of their dead weight and the loadings occurring, they can be inserted at sufficient rate into the reactor core in accordance with the requirements of KTA 3101.2 and KTA 3103.
c) The design of the control assemblies shall be such that the geometry (shape and position) required for heat removal and the required material properties are adhered to.
d) The design of the control assemblies shall be such that no inadmissible radiation exposure is caused by the control assemblies.

(2) Feedback effects onto adjacent core components, RPV internals and other plant systems are permitted as long as their safety-related requirements are met.

(3) The safety-related requirements for level of defence 3 including the fundamental safety functions are shown in **Annex A**, Table A 2-2.

3.2.2.3 Level of defence 4a

(1) The safety-related requirements for level of defence 4a (ATWS) can be comprised as follows:

a) Prior to the transient, the control assemblies in the core shall remain effective during and after the transient (no redistribution of absorber material).

b) In the case of BWR, it shall additionally be possible to insert a sufficient number of control assemblies by electric motors.

   Note:

   For the PWR there are no requirements for the maneuverability of control assemblies in the case of ATWS.

(2) Feedback effects onto adjacent core components, RPV internals and other plant systems are permitted as long as their safety-related requirements are met.

(3) The safety-related requirements for level of defence 4a including the fundamental safety functions are shown in **Annex A**, Table A 2-3.

3.2.3 Flow restrictor assemblies

(1) As regards the flow restrictor assemblies inserted in the reactor core for harmonization of the thermo-hydraulic conditions the following safety-related requirements are to be met:

a) The requirements for fuel element cooling laid down in KTA 3101.1 shall be met.

b) The flow restrictor assemblies shall be designed such that the geometry (shape and position) required for heat removal and the required material properties are adhered to.

(2) Feedback effects onto adjacent core components, RPV internals and other plant systems are permitted as long as their safety-related requirements are met.

4 Acceptance criteria

4.1 Requirements applying to all components

4.1.1 Design principles

(1) The acceptance criteria to be met for fulfilment of the acceptance targets are derived from the safety-related requirements and consider the actual design of the component.

Note:

The design comprises all product properties arising from manufacture.

(2) On levels of defence 1 to 4a as well as for internal events and external events (design basis earthquake) and emergency cases (explosion blast wave and aircraft crash) and over the entire operating time, the acceptance criteria shall consider the relevant influences and significant effects, such as loadings due to external and internal mechanical, thermal, chemical and radiation-induced effects, changes in material properties, geometry changes as well as the boundary conditions arising from the functional requirements.

(3) For internal events and external events as well as for emergency cases not classified into levels of defence in the overriding regulations, the same acceptance criteria as for level of defence 3 are used in this safety standard.

(4) Loadings and influences arising from transport, handling and storage shall be considered.

(5) To ensure that the requirements of Section 3 are met, Section 4 lays down suitable acceptance criteria for all relevant loadings for the pertinent components.

(6) These component-specific acceptance criteria are laid down with such a distance to the failure limits that uncertainties arising from the determination of the failure limits or the limitation of the failure-free area have been considered.

(7) Feedback from experience gained e.g. from operation or experiments shall be considered.

(8) The component-specific acceptance criteria hereafter refer to the current state of design and construction. In the case of other design types, the acceptance criteria shall be determined analogously to sub-paras. (1) to (7).

4.1.2 Utilization of suitable materials

(1) The materials used shall be suited. All essential influences on the material properties, e.g. chemical, mechanical, thermal and radiation-induced influences shall be considered.

Note:

The materials used for core components will be subject to chemical, mechanical, thermal and radiation-induced influences when being used in the reactor core. In addition, influences arising from fabrication processes, e.g. during forming or welding and heat treatment may be possible. Moreover, the ductility of the materials may be influenced by e.g. irradiation or hydrogen absorption as well as, in the case of control assemblies, the absorber material may be influenced by neutron absorption or possible contact with the cooling fluid.

(2) The materials shall be selected in due consideration of the stresses occurring and the environmental conditions such that their function is not impaired by stress corrosion-cracking.

Note:

The function may be impaired directly at the pertinent component or even at other components (e.g. in case of exfoliation of large parts from the fracture surface).

(3) The materials shall be selected such that radiation exposure is kept as low as reasonably acceptable (ALARA), among other things, by limited activation.

4.1.3 Compatibility

(1) The design and construction of the core components shall be compatible with the individual parts of the core component, to other core components and adjacent components and systems.

(2) The core components shall be designed such that interaction of core component parts among each other and of the core component with other core components does not impair their respective functions and properties on the pertinent levels of defence.

(3) Especially interactions between the fuel assemblies among each other and with other core components which are caused by static or dynamic loadings and resulting vibrations or deformations shall not impair the functions of the core components as a whole.

(4) In particular, the following combinations shall be considered:

a) parts of a core component among each other (e.g. sufficient free space between fuel rod and fuel assembly structure),

b) core components with adjacent core components such as

   ba) fuel assemblies among each other,

   bb) fuel assemblies with fuel assembly channels and fuel element channels among each other (BWR)

   bc) fuel assemblies with control assemblies and flow restrictor assemblies (PWR),

   bd) fuel assembly channels with the control assemblies (BWR),

   be) fuel assemblies with core instrumentation, and

c) core components with power plant systems (e.g. RPV internals, control rod drive mechanism, handling and storage facilities).

Note:

An essential aspect of core component compatibility is the limitation of deformations (e.g. of fuel assemblies in PWR and fuel assembly channels in BWR) to avoid damage during operation and handling. Strong deformations of fuel assemblies in PWR and fuel assembly channels in BWR may affect the manoeuvrability of the control assemblies such that the safety-related task of the shutdown system to KTA 3102 and KTA 3102.2 is not satisfied. Essential parameters for the determination of the deformation behaviour of core components (fuel assemblies in PWR and fuel assembly channels in BWR) are their lateral stiffness and the creep behaviour of the materials used.

Within the reactor core the fuel assemblies are supported in RPV internals and guided (in case of BWR) or restrained (in case of PWR). Besides the geometric compatibility to be presumed thermo-hydraulic and mechanical compatibility are considered in case of interacting loadings.

The instrumentation lances required for operational monitoring are positioned within the fuel assembly (in case of PWR in the guide tubes) or between the fuel assemblies at the crossing point of 4 core cells (in case of BWR). In the top end pieces of PWR fuel assemblies which do not contain control assemblies, usually flow restrictor assemblies are installed to harmonize the thermo-hydraulic conditions. For these core components the same compatibility requirements are to be met.

In case of geometry changes of the fuel assembly due to radiation-induced growth or creep of structural elements, individual functions (e.g. covering of the finger springs by the fuel assembly channel or reduction of flow cross-sections in cooling-water channels in BWR) and fuel assembly compatibility (e.g. axial covering of spacers of adjacent fuel elements in PWR, axial and radial clearances) may be impaired.

(5) The deformation of fuel assemblies shall be limited such that the requirements of KTA 3103 as regards control rod movement are met.

(6) The deformation of fuel assemblies shall not cause damage on fuel assemblies or on adjacent components.

(7) As regards the proof of fulfilment of the requirements set in (5) and (6), the deformation behaviour of the fuel assemblies during operation shall be assessed within the design stage. In this case, the relevant structural and material-specific properties (e.g. stiffness of the fuel assembly and its structure and the creep behaviour of the fuel assembly structure) and relevant boundary conditions during operation (e.g. fuel assemblies on adjacent core positions) shall inter alia be taken into account.

Note:

The fulfilment of the requirements set in (5) and (6) may e.g. be demonstrated in case of design changes by a comparative assessment of the deformation behaviour of fuel elements with new design compared to a design of proven performance during operation.

(8) The compatibility of the fuel assemblies with the handling and storage facilities shall also be ensured where deformations due to operation shall be taken into account.

(9) KTA 3101.1 contains requirements for thermo-hydraulic compatibility.

(10) The materials shall be compatible with adjacent components in due consideration of electrochemical interactions.

4.1.4 Accessibility for inspection

(1) Core components shall be designed such that they can be inspected even in irradiated condition. This includes inspections with remote-controlled devices.

(2) Where it can be expected that bolted joints have to be loosened, suitable designs shall be provided to prevent sticking of the threaded or bolting system.

Note:

Suitable designs e.g. are rolled threads or thread systems with tip or flank clearance as well as with rounded edges.

4.1.5 Avoidance of loose parts

Core component parts and joints (e.g. bolted joints) shall be constructed such that loose parts in the reactor are safely prevented (e.g. by secured bolted joints).

4.2 Fuel assembly structure including fuel assembly channel in case of BWR

4.2.1 General

(1) The component-specific acceptance criteria described hereafter apply independently of the plant type (PWR/BWR). The acceptance criteria relating to the BWR fuel assembly channel are specifically addressed.

(2) Design aspects for fuel rods which due to interactions between fuel assembly structural parts and fuel rods require an integral verification procedure are also dealt with in this section 4.2.

4.2.2 Acceptance criteria for the fuel assembly structure on levels of defence 1 and 2

4.2.2.1 Limitation of stresses and strains

(1) The stresses in the fuel assembly structural parts, the fuel assembly channels and their connections (e.g. bolted joints, welded joints, brazed joints, positive connections) shall be limited.

Note:

Fuel assembly structural parts and their connections will be loaded during handling by their dead weights and inertial forces action upon them.

When fuel assemblies are used in the reactor (operation), the structural parts and their connections will be affected by forces arising from dead weight, hydrostatic buoyancy and flow forces. In the case of PWR additional hold-down forces occur. These forces will cause tensile, pressure and bending loads.

In addition, forces will arise among structural parts and between structural parts and fuel rods due to differential thermal strain and differential radiation-induced thermal growth.

The compression condition of the pressure springs will change due to thermal and radiation-induced growth during various operating conditions and during the operating time of the fuel assembly. In addition, the spring force will be relaxed e.g. due to neutron irradiation.

(2) In this case, the material-specific limits shall not be exceeded as regards:

a) the equivalent stresses in fuel assembly structural parts, fuel assembly channels and connections,

b) the shear stresses in connections,

c) the surface pressure in bolted joints between head or nut and bearing surface, and

d) the shear stresses in pressure springs.

(3) The influence of geometry changes arising during the use in the reactor (i.e. creep deformations) on the stresses arising in the component shall be evaluated and be considered, where necessary.

(4) If deformation limits have to be observed for functional and compatibility reasons, a strain analysis shall be performed.

(5) In the verification procedure the safety factors given in **Annex B** shall be taken into account.

4.2.2.2 Geometric stability

(1) The structural parts as well fuel rods shall not buckle under axial pressure loading. Therefore, the pressure forces shall be limited or the component be designed such that such instable conditions cannot occur.

(2) The verification procedure shall consider geometric and structural imperfections for the purpose of DIN EN 1993-1-1. Geometry changes arising from reactor operation shall be evaluated and be considered, where necessary.

Note:

Annex B shows possible verification procedures for considering imperfections and geometry changes.

(3) In the verification procedure the safety factors given in **Annex B** shall be taken into account.

4.2.2.3 Fatigue

(1) Alternating thermal or mechanical loadings may also lead to a progressing damage process of the material even if they are below the static mechanical strength of the material. The loadings in the fuel assembly structural parts and fuel assembly channels shall be limited such that no fatigue damage occurs.

Note:

By changes of the operating conditions (e.g. during start-up and shutdown or capacity changes) and due to flow-induced vibrations the fuel assembly structural parts and bolted joints will be subjected to cyclic stresses which may contribute to fatigue.

During normal operation alternating loadings arise, e.g. due to

- flow-induced vibration excitations of the fuel assemblies,
- loadings from load-following operation and
- loadings from reactor cyclic operational mode (e.g. day/night cycles, weekend cycles).

Due to the low loadings at high-cycle conditions or the low number of load cycles at higher loadings and in due consideration of operational experience the influence of the fluid on fatigue need not be expected for these components.

(2) In case of proofs to (1) the procedures and safety factors mentioned under **Annex B 2.4** shall be considered.

4.2.2.4 Limitation of corrosion

Corrosion shall be limited such that a sufficient thickness of the metallic structure is available to bear the mechanical loads.

4.2.2.5 Safeguarding of sufficient ductility

(1) It shall be ensured that the materials used have sufficient ductility. Here, operational effects such as neutron embrittlement or embrittlement due to hydrogen absorption as well as influence on such effects by the stress condition prevailing (see B 2.3 and B 2.5) shall be considered for the specific materials.

(2) The absorption of hydrogen shall be limited to material-specific values.

Note:

A portion of the hydrogen originating during the corrosion process will be absorbed by the structural parts. A too high hydrogen content will cause a reduction of ductility.

4.2.2.6 Safeguarding of the fuel assembly position

(1) The axial and radial positions of the fuel rods in the fuel assembly and of the fuel assemblies in the core shall be ensured.

(2) For PWR fuel assemblies on level of defence 1 it shall be proved that the resultant of all forces acting axially on the fuel assembly effect a contact force on the lower core support plate. In the case of transients on level of defence 2 lift-off of the fuel assemblies from the lower core support plate shall not occur. A short-time lift-off on level of defence 2 is permitted if it is proved that

a) no inadmissible loadings are applied on fuel assemblies, other core components and RPV internals,

b) during lift-off the compatibility with adjacent fuel assemblies, other core components and RPV internals is maintained, and

c) no fuel assembly (fuel assembly grab) leaves the cylindrical part of the centring pin of the lower core support plate and each fuel element returns to its initial position.

(3) For BWR fuel assemblies it shall be proved that the resultant of all forces acting axially on the fuel element (where provided also for the fuel assembly bundle or partial bundle) effect a contact force on the respective contact area.

(4) The fuel rods shall be stored in the fuel assembly such that they are maintained in their axial and radial positions under the loadings occurring (e.g. dead weight, flow forces, vibration) so that the boundary conditions for neutron-physical and thermo-hydraulic design are satisfied and changes in length relative to the fuel assembly structure can be balanced.

4.2.2.7 Bearing of the fuel rod

The fuel rods shall be retained in the fuel assembly such that no cladding tube wall thickness reduction occurs (e.g. by fretting wear between fuel rod and spacer) as a result of which the requirements for ensuring the fuel rod tightness cannot be satisfied anymore.

Note:

The fuel rods are kept by the spacers within the fuel assembly structure in a geometrical position optimized for operation. To be capable of ensuring differential thermal and neutron-induced growth between fuel assembly structure and fuel rod without exceeding the allowable stresses, the fuel rods are usually retained in the spacer on spring elements. Due to the interaction between fuel rod and spacer cell, material abrasion (fretting) may occur due to vibrations.

The verification to (1) may be effected to show that sufficient residual spring force of the spring elements in the spacer is still available at the end of service life time. It is however also possible to prove by means of experiments that if gaps between fuel rod and spacer occur even in the course of service life, fretting will not cause inadmissible cladding tube wall thickness reduction according to (1).

4.2.2.8 PWR-specific requirements

For PWR fuel assemblies it shall be proved that in case of control assembly fall in no inadmissible loadings on the fuel assembly structure occur. The loadings shall be considered in the stress calculation.

Note:

This is proved if the kinetic energy of the control assembly, when striking the fuel assembly top end piece, is absorbed by the spring under the control assembly spider such that there is no hard contact between the fuel assembly top end piece and the fuel assembly bolt, and the fuel assembly structure is capable to transfer the maximum spring force during fall in process.

4.2.3 Acceptance criteria for the fuel assembly structure on levels of defence 3 and 4a

4.2.3.1 Limitation of deformations

Permanent deformations of the spacers and guide tubes (PWR) and of fuel assembly channels (BWR) shall be limited such that the possibility of shutting down and cooling the reactor is ensured. Possible deformations due to specified normal operation shall be assessed and be taken into account, where necessary.

N o t e s :

(1) For loss-of-coolant accidents a leakage cross-section of 0.1 F for PWR and a leakage cross-section of 2 F for BWR shall be used as load assumption according to the Safety Requirements for Nuclear Power Plants.

(2) Permanent deformation of guide tubes and spacers may be caused by plastic deformation (excess of yield limit) or stability failure. Depending on the construction and the properties of the materials used one of the two mechanisms will govern the design.

4.2.3.2 Limitation of stresses

(1) On level of defence 3 the primary stresses in the fuel assembly structural parts, the fuel rods, the fuel assembly channels and their connections (e.g. bolted joints, welded joints, brazed joints, positive connections) shall be limited.

(2) In this case, the material-specific limits shall not be exceeded as regards the:

a) stress intensities in fuel assembly structural parts, fuel rods, fuel assembly channels and connections, and
b) shear stresses in threaded joints.

(3) The influence of pre-deformations from specified normal operation (e.g. creep deformations) on the stresses arising in the component shall be evaluated and be considered, where necessary.

(4) In the verification procedure the safety factors given in **Annex B** shall be taken into account.

4.2.4 Component-specific requirements for transport, handling and storage

(1) The fuel assemblies and fuel assembly channels (BWR) shall be compatible with the transport, handling and storage facilities.

(2) The quality proved during manufacture of the fuel assemblies and fuel assembly channels shall not be impaired by the loadings arising from transport, handling and storage in a manner that could lead to restrictions for further utilization.

(3) Here, it shall be proved that

a) the stresses and strains occurring during transport, handling and storage are limited to allowable values specific to the materials. In the verification procedure the safety factors given in **Annex B** for levels of defence 1 and 2 shall be taken into account.
b) the pellet column in the fuel rod is not displaced,
c) the fuel rods in the fuel assembly are axially displaced neither individually nor as bundle,
d) the spring elements are not loaded due to the mass effect of the fuel rods which upon transport cause the specified minimum spring force to be less than required,
e) no damage is caused on the spacers by the fuel rod mass effects, i.e. the transport loadings shall always be less than the loadings which the spacers can withstand,
f) surface damage (fretting) on fuel assembly components, e.g. on fuel rod cladding tubes due to the relative movement between fuel rods and spacer bearing points is limited to the values specified in the manufacturer's documents.

(4) In the case of MOX fuel assemblies, the consequential effects of thermal output during transport and insertion into the wet storage facility shall be taken into account.

(5) The verifications may be made by calculation or by qualification of the transportation process.

4.3 Fuel rod

4.3.1 General

(1) The component-specific acceptance criteria described hereafter apply independently of the plant type (PWR/BWR).

(2) Design aspects for fuel rods which due to interactions between fuel assembly structural parts and fuel rods require an integral verification procedure are also dealt with in section 4.2.

4.3.2 Acceptance criteria for the fuel rod on levels of defence 1 and 2

4.3.2.1 Limitation of stresses and strains in the fuel rod cladding tube

(1) Stresses shall basically be limited to the allowable values specific to the materials. The safety factors given in **Annex B** shall be considered. Exceptions are permitted as per (3).

N o t e :

Stress loadings arise e.g. due to the pressure difference between cooling fluid pressure and internal fuel rod pressure in the cladding tube as well as due to thermal or bend loadings.

In case of fast increase of fuel rod output the loadings will not be limited by the stress criteria, but the strain criteria as per 4.3.2.1(3).

(2) Among the operational effects to be considered as per 4.1.1 (2) for the specific materials are, inter alia, wall thickness reduction due to corrosion and radiation-induced hardening.

(3) In case of fast increase of fuel rod output due to events on level of defence 2 where no stress reduction by relaxation processes occurs, excess of the elasticity limit is permitted, however the total strain (elastic and plastic) of the cladding tube in tangential direction shall be limited.

(4) In case of strains with low strain rates and a low stress level below the yield limit (e.g. due to long-term effects), the plastic equivalent strain of the cladding tube in the tensile range shall be limited.

N o t e t o (3) a n d (4) :

At the beginning of operation in the reactor, in warm condition an operating clearance is generated in a fuel rod at reactor nominal output in consideration of the elastic upsetting of the cladding tube due to the coolant pressure and the different thermal strains of cladding tube and fuel. The fuel rod is set under pressure stress due to the difference between the coolant pressure and the fuel rod internal pressure and creeps onto the fuel. Upon resting of the cladding tube on the fuel, the cladding tube will be slowly elongated due to fuel swelling in which case the cladding tube may be elongated to exceed the initial diameter in case of high burn-up rate.

In addition, interaction between fuel and cladding tube may be effected by rapid increase in power output.

As regards the design of fuel rods the interaction is subdivided as follows:

a) rapid increase of fuel rod heat generation rate (see (3)):

In case of rapid increase of fuel rod heat generation rate, the cladding tube may be elongated due to fuel thermal expansion depending on burn-up condition and the level of heat generation rate increase or the final rate reached. For this case, the total strain in tangential direction (elastic and plastic) caused by the rapid increase in heat generation is relevant.

b) Long-term interaction by fuel swelling (see (4))

Upon creeping of the cladding tube onto the fuel, the cladding tube will creep at low strain rate and due to the stress relaxation at low stress level. For this case, creep ductility is relevant.

4.3.2.2 Safeguarding of sufficient ductility

(1) It shall be ensured that cladding tube material has sufficient ductility. Here, operational effects such as neutron embrittlement, embrittlement due to hydrogen absorption, hydride

direction as well as influence on such effects by the stress condition prevailing shall be considered for the specific material.

Note:

This may be verified e.g. by experiments investigating the plastic strain that can be obtained by this material.

(2) The absorption of hydrogen shall be limited to material-specific values.

Note:

A portion of the hydrogen originating during the corrosion process will be absorbed by the structural parts. A too high hydrogen content will cause a reduction of ductility.

4.3.2.3 Geometric stability

(1) Elastic buckling and plastic deformations under external pressure shall be excluded. This does not apply to creeping at low strain rates and low stress level as per 4.3.2.1 (4).

Note:

The cladding tubes of fuel rods are generally set under external pressure due to the pressure difference between the coolant pressure and the rod internal pressure. A cladding tube under external pressure may be subject to elastic buckling or, if the stresses exceed the yield strength, be subject to plastic deformation.

(2) In case of proofs to (1) the procedures and safety factors mentioned under **Annex B** shall be considered.

4.3.2.4 Fatigue

(1) Alternating thermal or mechanical loadings may also lead to a progressing damage process of the material even if they are below the static mechanical strength of the material. The loadings in the fuel rod cladding tubes shall be limited such that no fatigue damage of the cladding tubes occurs.

(2) In case of proofs to (1) the procedures and safety factors mentioned under **Annex B 2.4** shall be considered.

Note:

During normal operation alternating loadings arise, e.g. due to

- flow-induced vibration excitations of the fuel rods,
- load-following operation,
- reactor cyclic operational mode (e.g. day/night cycles, weekend cycles).

Due to the low loadings at high-cycle conditions or the low number of load cycles at higher loadings and in due consideration of operational experience the influence of the fluid on fatigue need not be expected for these components.

4.3.2.5 Limitation of loadings due to chemical-mechanical interaction (PCI) and stress corrosion cracking (SCC)

(1) Depending on

a) the rate of power output changes,

b) level of steps (ramps) and final power output rate,

c) holding time of final power output rate and

d) conditioning prior to increase of power output

an increase in power output may lead to pellet clad interaction (PCI) and stress corrosion cracking (SCC). The loadings resulting here from shall be limited such that the tightness of the fuel rod is ensured.

(2) The pertinent criteria shall be derived using experimental results (e.g. ramp tests in test reactors; operational experience).

Note:

When deriving the criteria, empiric or analytical methods may be applied.

(3) The proof of observance of the criteria may be made on level of defence 1 based on the regulations for reactor mode of operation as well as on level of defence 1 and 2 in due consideration of the facilities for limitation of power and power density.

4.3.2.6 Limitation of corrosion

Corrosion shall be limited such that

a) the temperature at the metal/metal oxide interface is within a range where an uncontrolled increase of the corrosion rate can be prevented, and

b) a sufficient thickness of the cladding tube wall is provided to bear the mechanical loads.

Note:

Under operating conditions, a reaction may be caused between the fuel rod cladding tube material and the surrounding water or water/steam atmosphere along with the formation of an oxide layer on the cladding tube outer surface (cladding tube corrosion). This may influence the temperature conditions of the fuel rod system (degradation of heat transfer from fuel via the cladding to the coolant) and lead to a wall thickness reduction. To avoid an uncontrolled temperature rise (and thus an increased corrosion rate) the corrosion will be limited (e.g. by limiting the oxide layer thickness). The wall thickness reduction caused along with corrosion has been taken into account in 4.3.2.1. In conjunction with cladding tube corrosion, hydrogen is generated which is partly absorbed by the cladding tube. By limiting the corrosion the hydrogen content in the cladding tube is also limited (see also 4.3.2.2).

For corrosion various forms exist, e.g. uniform and nodular corrosion as well as shadow corrosion. Treatment of corrosion effects may specifically be made from case to case.

4.3.2.7 Limitation of internal pressure

Internal pressure rise due to fission gas release within the fuel rod shall be limited such that no self-reinforcing thermal feedback occurs.

Note:

Pressure stresses caused by the coolant pressure in the cladding tube are partly compensated by the fuel rod internal pressure. At the beginning of the usage time, an internal pressure from the helium filling resulting from manufacture as well as from the temperature and volume ratios inside the fuel rod is generated. During the course of burn-off, inter alia, gaseous fission products will be generated a portion of which (depending on the operation history of the respective fuel rod) is released into the free spaces of the fuel rod thus causing an increase of the fuel rod internal pressure. Where the acceptance criteria to (1) is satisfied, it will be ensured that even in case of internal pressure the thermal stability of the fuel rod system is guaranteed: Possible creeping of the cladding tube under internal pressure to the outside may cause heat transfer degradation and consequently a fuel temperature rise and thus increase the fission gas release rate, which then again would further increase the internal pressure (thermal feedback).

Verification methods well established up to now are e.g. an appropriate internal pressure limitation, the imitation of the cladding tube strain rate based on experiments (ROPE) or verification by comparison of the calculated fuel swelling rate with the cladding tube strain rate.

4.3.2.8 Limitation of fuel temperature

(1) The maximum fuel temperature shall always remain below the melting temperature.

(2) With this verification, the most unfavourable operating conditions of level of defence 2 including the maximum possible overload at the location of highest fuel power shall be assumed.

Note:

The intention is to exclude displacements of molten fuel inside the fuel rods. In addition, it is inter alia avoided by this stipulation that the fuel volume increase caused by melting leads to additional cladding tube strain.

The melting temperature inter alia depends on the fuel composition (e.g. UO_2 or MOX, additives like gadolinium) and on burn-up.

4.3.3 Acceptance criteria for the fuel rod on level of defence 3

4.3.3.1 Limitation of deformation of fuel rod cladding tubes

The cladding tube strain of fuel rods shall be limited such that a free flow field is obtained to ensure sufficient cooling of the fuel rods.

Note:

Sufficiently free flow field areas may be determined by tests (e.g. REBEKA, see Wiehr [4]).

4.3.3.2 Tightness of fuel rods

(1) The fuel rod tightness shall basically be ensured.

(2) Deviating here from, the requirements of 4.3.3.4 (4) apply in case of loss-of-coolant accidents and those of 4.3.3.5 in case of reactivity accidents.

Note:

The requirements with regard to critical boiling conditions are laid down in KTA 3101.1.

4.3.3.3 Limitation of fuel temperature

The maximum fuel temperature shall basically be below the melting temperature. Partial fuel melting is permitted if the retention function of the fuel rod cladding tubes is not impaired and large-area displacements of the fuel are excluded.

4.3.3.4 Specific requirements in case of loss-of-coolant accidents

(1) A maximum cladding temperature of 1200 °C shall be adhered to.

Note:

The adherence to the maximum cladding tube temperature of 1200 °C prevents a self-sustaining exothermal zirconium-water reaction and counteracts excessive cladding tube embrittlement.

(2) To limit the hydrogen quantity generated it shall be ensured that not more than 1% of the zirconium contained in the reactor core oxidizes.

(3) For all fuel rods it shall be proved that during rewetting of the cladding tube (quenching) sufficient residual ductility (e.g. by an appropriate criterion for limiting the cladding tube oxidation depth) or sufficient residual strength is obtained so that no fragmentation of the cladding tube occurs in the course of the event.

Note:

The verification of residual ductility is based on the measurement of the deformability of specimens upon high-temperature oxidation, e.g. by ring pressure tests. The transition to pure brittle behaviour of the specimen is defined by the coincidence of tensile strength and yield strength (zero ductility), see e.g. CSNI technical opinion papers no. 13 [5].

(4) In case of loss-of-coolant accidents with leakage cross-sections exceeding 0.1 F, the number of burst fuel rods shall be limited to 10% of all rods in the core. In case of loss-of-coolant accidents with leakage cross-sections less than or equal to 0.1 F, the number of burst fuel rods shall be limited to 1% of all rods in the core.

(5) In case of cladding tube damage, the consequences of bursting shall additionally be taken into account:

a) In consideration of a possible local ductility reduction at the location of bursting by absorption of hydrogen on the cladding tube inside it shall be verified that the requirement of (3) is met.

b) A possible fuel discharge especially in case of high fuel burn-up shall be limited such that nuclear criticality safety and cooling capability is ensured and the radiological safety objective is satisfied.

Note:

In case of fuel rods with high burn-up increased fuel discharge can be expected in case of bursting (see e.g. [6]).

As a rule, the assumptions made in the existing radiological verifications are limiting with regard to the maximum total fuel quantity released.

Where the total quantity is limited to 0.1% of the solid material of fuel rods (the assumed extent of damage is 10% of the core inventory), it can be assumed that if this limit for fuel release is adhered to, the criticality safety and core cooling capability are ensured.

4.3.3.5 Specific requirements in case of reactivity accidents

In case of rapid reactivity changes it shall be ensured that the fuel remains within the cladding tube.

Note:

A prevalent acceptance criteria is the tightness of the fuel rod. Fuel rod tightness is ensured if the maximum enthalpy release in the fuel (radially averaged over the pellet cross-section) remains below a cladding tube defect limit which depends on the material condition or fuel burn-up.

4.3.4 Acceptance criteria for fuel rods on level of defence 4a

(1) It shall be shown that

a) a maximum cladding tube temperature of 1200 °C is adhered to and

b) in case of cladding tube rewetting (quenching) a residual ductility (e.g. by an appropriate criterion for limiting the cladding tube oxidation depth) or sufficient residual strength is obtained so that no fragmentation of the cladding tube occurs in the course of the event.

Note:

The reactor core cooling capability is maintained in case of an anticipated transient without scram (ATWS) if it can be proved that no fuel rod damage occurs which may block the cooling channels.

The cooling capability can be ensured by exclusion of cladding tube fragmentation. This is ensured if, comparable to the loss-of-coolant accident procedure, a maximum cladding tube temperature of 1200 °C is maintained and residual ductility or sufficient residual strength of the cladding tube is obtained.

In case of an anticipated transient without scram (ATWS), no such high pressure differentials over the cladding tube wall thickness occur so that buckling or bursting need not be expected.

(2) Prevalent to the criteria (1) a) and b) the cooling capability is ensured if it is proved that no critical boiling conditions as per KTA 3101.1 occur.

(3) The requirement for long-term sub-criticality is met due to the fact that no cladding tube fragmentation as per (1) occurs.

Note:

In case of an anticipated transient without scram (ATWS), the long-term sub-criticality of the reactor core is ensured if the core geometry upon ATWS permits that in the course of the event the neutron absorbers inserted in the core (control assemblies in BWR, boron in PWR) also remain sufficiently effective locally.

4.4 Control assemblies

4.4.1 General

(1) The component-specific acceptance criteria described hereafter apply independently of the plant type (PWR/BWR).

Note:

Control assemblies with their drives are usually attached (in case of PWR on the outside on the PWR closure head and in case of BWR on the outside on the RPV bottom head) by means of drive shafts (PWR) or hollow pistons (BWR). In a PWR the absorber rods of the control assemblies are moved within the fuel rod guide tubes of the fuel assemblies and in a BWR the absorber-containing parts of the control assemblies (cross-shaped arranged control assembly blades) are moved between the 4 fuel assemblies of a core cell.

(2) The control assemblies shall be designed such that the safety-related task of the shutdown system which they are part of, is satisfied in accordance with KTA 3103 and KTA 3101.2.

(3) The design shall be such that absorber material can be inserted in the reactor core in a controlled manner and to the required extent by the control assemblies.

(4) Here, the neutron-absorbing effect of the absorber material and its distribution in the control assemblies as well as the required periods for obtaining reactivity changes shall be considered.

Note:

The requirements for the neutron-physical properties of the control assemblies are dealt with in KTA 3101.2 and KTA 3103.

(5) Washable or non-wear resistant absorber materials or absorbers which may react chemically with the coolant shall be coated.

(6) If control assemblies are used, where the absorber material releases radionuclides (e.g. tritium) into the coolant, it shall be proved that the radiological limit values are adhered to.

(7) The control assemblies shall be designed such that an unobstructed control assembly path at operating temperature is available to meet the requirements of KTA 3103, section 4.2.

Note:

Prerequisite to a sufficient freedom of movement of the control assemblies is the compatibility of the control assemblies with the fuel assemblies and fuel assembly channels (in case of BWR) as well as with RPV internals and control assembly drives, in which case, besides the geometric and thermo-hydraulic compatibility, also the mechanical compatibility regarding interacting loadings is considered.

(8) The design of the control assemblies shall ensure that the dead weight of the control assemblies does not negatively affect the shutdown rate required by KTA 3103.

(9) Due to the acceleration and slow-down of the control assemblies no inadmissible loadings shall be applied on the control assemblies themselves, the RPV internals and control assembly drives.

4.4.2 Acceptance criteria for the control assemblies on levels of defence 1 and 2

4.4.2.1 Use of suitable absorber material

In addition to the general requirements as per 4.1.2 the absorber material shall be selected such that the neutron-physical effects required by KTA 3103 are ensured over the total service time.

4.4.2.2 Limitation of the absorber material temperature

The temperature of the absorber material shall always remain below the melting temperature.

Note:

To exclude displacement of absorber material due to absorber melting, the melting point of the absorber shall not be attained. In addition, additional strain of the cladding usually surrounding the absorber material caused by an extension of the absorber volume during melting or damage to the cladding by molten absorber material will be avoided.

4.4.2.3 Limitation of stresses in the absorber cladding

The stresses in the absorber cladding shall be limited to material-specific values, in which case the safety factors given in **Annex B** shall be considered.

Note:

Within the course of operating time of the fuel assemblies, stress loadings on the absorber cladding will occur e.g. due to the fact that during movement of the fuel assemblies, especially in case of reactor scram, inertia forces will act upon the total of control assemblies and thus also on the absorber cladding. In addition, an internal pressure will be built up in the absorber cladding due to neutron absorption if certain absorber materials such as boron carbide (B_4C) are used.

4.4.2.4 Fatigue of absorber cladding, control assembly structural parts and connections

(1) Alternating thermal or mechanical loadings may lead to a progressing damage process in the material even when below the static mechanical strength of the material. Loadings in structural parts shall be limited such that no failure due to fatigue damage occurs.

(2) In case of proofs to (1) the procedures and safety factors of **Annex B 2.4** shall be taken into account.

Note:

When moving control assemblies, the absorber cladding, the control assembly structural parts and the bolted connections will be loaded by dynamic alternating stresses resulting e.g. from inertial forces and possibly contributing to fatigue damage.

(3) For control assemblies made of austenitic steel 1.4541 it can be assumed that the fluid influence on fatigue strength is not effective if according to [1] at least one of the three governing factors of influence, i.e. the strain amplitude ε_a, average temperature T and strain rate $\dot{\varepsilon}$ exceeds or is less than the threshold values defined in equation 4.4.2.4-1. This is exactly the case if, during the respective cycle (or the transient observed), the strain amplitude ($\varepsilon_a \leq 0.1\%$) or the average temperature ($T \leq 100$ °C) is less than the prescribed limit values or with regard to the strain rate ($\dot{\varepsilon} > 10\%/s$) the threshold value is exceeded. In such case, the factor F_{en} used in the fatigue analysis for consideration of the fluid influence shall be equal to 1.0.

Note:

As regards the loading situation, it can as a rule be assumed for the control assemblies made of the material 1.4541 that the governing threshold values are effective and the factor F_{en} can be set equal to 1.0.

(4) Where, on account of the loading situation, the effectiveness of the threshold values cannot be verified, the fluid influence shall be accounted for in the fatigue analysis according to the procedure described in [1]. It shall be shown that the cumulative usage factor does not exceed the value of 1.0.

$$F_{en,i} = \exp\left(-T^* \cdot O^* \cdot \dot{\varepsilon}^*\right), \qquad (4.4.2.4\text{-}1)$$

with $F_{en,\,i} = 1$ for $\varepsilon_a \leq 0{,}1$ %
(forged, rolled and cast stainless austenitic steels)

$T^* = 0$	($T \leq 100$ °C)
$T^* = (T-100)/250$	(100 °C < T < 325 °C)
$\dot{\varepsilon}^* = 0$	($\dot{\varepsilon} > 10$ %/s)

$\dot{\varepsilon}^* = \ln(\dot{\varepsilon}/10)$ (0.0004 ≤ $\dot{\varepsilon}$ ≤ 10 %/s)

$\dot{\varepsilon}^* = \ln(0.0004/10)$ ($\dot{\varepsilon}$ < 0.0004 %/s)

O* = 0.29 (< 0,1 ppm dissolved oxygen in the reactor water)

(all forged, rolled and cast stainless austenitic steels and heat treatments as well as stainless austenitic weld metals)

O* = 0.29 (≥ 0.1 ppm dissolved oxygen in the reactor water)

(sensitized high-carbon forged, rolled and cast stainless austenitic steels)

O* = 0.14 (≥ 0.1 ppm dissolved oxygen in the reactor water)

(all forged and rolled stainless austenitic steels except for sensitized high-carbon stainless austenitic steels)

4.4.2.5 Geometric stability of the absorber cladding

(1) Elastic buckling and plastic deformation under external pressure shall be excluded.

N o t e :

In general, the absorber cladding of the control assemblies (in case of a PWR the absorber cladding tube, in case of a BWR the absorber-containing parts) is under external pressure due to the pressure difference between coolant pressure and internal pressure (in case of PWR control assemblies over the total service time, in case of BWR control assemblies an internal pressure will be built up in the course of service time which can reach the value of the internal overpressure). A cladding under external pressure may be subject to elastic buckling, or if the stresses exceed the yield strength, be subject to plastic deformation.

(2) In case of proofs to (1) the procedures and safety factors of **Annex B** shall be taken into account.

4.4.2.6 Limitation of plastic strain of the absorber cladding

Where the cladding tube is strained due to the absorber, e.g. by swelling or thermal expansion, the plastic strain intensity of the absorber cladding in the tensile range shall be limited to the material-specific allowable values.

N o t e :

At the beginning of operation in the reactor an operating clearance is generated in the control assemblies between absorber and absorber cladding taking into account the elastic contraction of the absorber cladding due to the coolant pressure and the different thermal strains of absorber cladding and absorber. The absorber cladding tube then will be under pressure stresses due to external pressure and will creep onto the absorber. At the same time, the absorber is subject to radiation-induced swelling during its operation in the reactor. When the gap between absorber and absorber cladding has been closed due to these two effects, the absorber cladding will be slowly expanded due to absorber swelling in which case the absorber cladding can be expanded beyond its initial position. By complying with the requirement under (1) it is achieved that the plastic strain intensity of the absorber cladding thus caused does not exceed the allowable value verified for the material used.

4.4.2.7 Limitation of stresses and strains in structural parts and their connections

(1) The stresses in structural parts of the control assemblies and their connections (e.g. welded joints, bolted joints, brazed joints, and positive connections) shall be limited.

(2) In this case, the material-specific limits shall not be exceeded as regards

a) the equivalent stresses in control assembly structural parts and connections,

b) the shear stresses in connections,

c) the surface pressure in bolted joints between head or nut and bearing surface, and

d) the shear stresses in pressure springs.

(3) If specific deformation limits have to be observed for functional capability and compatibility reasons, a strain analysis shall be performed.

(4) In the verification procedure the safety factors given in **Annex B** shall be taken into account.

N o t e :

The control assembly structural parts and their connections will be loaded by inertial forces due to movement of the control assemblies during handling as well as in the reactor core, especially in case of reactor scram.

In a PWR the absorber rods of the control assemblies are moved within the fuel rod guide tubes of the fuel assemblies and in a BWR the absorber-containing parts of the control assemblies (cross-shaped arranged control assembly blades) are moved between the 4 fuel assemblies of a core cell. To limit, in the case of reactor scram, the loadings on adjacent components, especially on fuel assemblies, and on the control assemblies themselves, the control assemblies will be slowed down prior to complete insertion into the reactor core.

This is typically achieved in a PWR prior to the control assembly structure contacting the fuel assembly top end plate by hydraulic friction when the absorber rods enter the lower part of the guide tubes with reduced inside diameter. The velocity is reduced such that the contact velocities to be verified as being permitted for the adjacent components and the control assemblies are not exceeded.

In a BWR the slow down process is typically achieved by means of spring supports arranged in the control assembly drive.

4.4.2.8 Safeguarding of sufficient ductility

It shall be ensured that the materials used for the absorber cladding and for the control assembly structural parts have sufficient ductility. Here, operational effects such as neutron irradiation or embrittlement due to hydrogen absorption as well as influence on such effects by the stress condition prevailing (see B 2.3 and B 2.5) shall be considered for the specific materials.

4.4.3 Acceptance criteria for control assemblies on levels of defence 3 and 4a

4.4.3.1 Limitation of absorber material temperature

It shall be excluded that the possibility of shutting down and sufficiently cooling the reactor core due to displaced absorber material is impaired. This is generally proved if the temperature of the absorber material always remains below the melting temperature.

N o t e :

To exclude the displacement of absorber material outside the control assembly or, if enveloped by cladding, within the control assembly, due to melting of the absorber, the melting point of the absorber shall not be reached. In addition, it is avoided in case of absorber temperature being less than the melting temperature, that the cladding usually enveloping the absorber is additionally strained due to volume increase of the absorber during melting, or that the cladding is damaged by molten absorber material.

4.4.3.2 Limitation of stresses in absorber cladding, structural parts and connections

(1) On level of defence 3 as well as for BWR control assemblies also on level of defence 4 the primary stresses in the absorber cladding, in the control assembly structural parts and their connections (bolted joints, welded joints, brazed joints, positive connections) shall be limited.

(2) In this case, the material-specific limits shall not be exceeded as regards

a) the stress intensities in the absorber cladding, in fuel assembly structural parts and their connections, and

b) the shear stresses in connections.

(3) In the verification procedure, the safety factors given in **Annex B** for level of defence 3 shall be considered for level of defence 3 and for BWR control assemblies also on level of defence 4a.

N o t e :

As regards the analysis of events on level of defence 4a (events with postulated failure of the reactor scram system, ATWS) it shall be ensured in accordance with KTA 3103 that besides the scram system another shutdown system is provided to be capable of rendering the reactor sub-critical and maintaining it sub-critical for a long period of time. To this end, the control assemblies in a BWR are used with an electro-mechanical drive which is diverse to the scram system. In case of a PWR, several boron injection systems may be utilized alternatively or in combination to fulfil this function.

4.4.4 Component-specific requirements for transport, handling and storage

(1) The control assemblies shall be compatible with the transport, handling and storage facilities.

(2) The quality proved during manufacture of the control assemblies shall not be impaired by the loadings arising from transport, handling and storage in a manner that could lead to restrictions for further utilization.

(3) Here, the following shall be proved:

a) The stresses and strains occurring during transport, handling and storage are limited to allowable values specific to the materials. In the verification procedure, the safety factors given in **Annex B** for levels of defence 1 and 2 shall be taken into account.

b) Surface damage on control assemblies, e.g. due to the relative movement between control assemblies and bearing points is limited to the allowable value.

(4) The verifications may be made by calculation or by qualification of the transportation process.

4.5 Flow restrictor assemblies

(1) For flow restrictor assemblies it shall be proved that the function is maintained during the total service time and no inadmissible feedback on other components occurs.

(2) On level of defence it shall be proved that the resultant of all forces acting axially upon the flow restrictor assembly does not effect a contact force on the fuel assembly top end piece.

(3) The stress calculation including the verifications regarding transport, handling and storage shall be made analogously to the structural design of the control assemblies.

N o t e :

Due to the low loadings at high-cycle conditions or the low number of load cycles at higher loadings and in due consideration of operational experience, no influence of the fluid on fatigue need be expected for these components.

(4) The geometric shape shall be such that the requirements of KTA 3101.1 are met.

5 Further general requirements

5.1 Requirements for safety demonstration of the design

5.1.1 General

(1) The safety-related objective of core component design is to determine all loadings to be considered on the pertinent levels of defence and to select the type of construction (e.g. by appropriate dimensioning and materials) such that the requirements of Section 4 will be satisfied over the total intended service time.

(2) In this case, all relevant effects as per 4.1.1 (2) shall be considered.

(3) The safety demonstration shall show that

a) all loadings to be assumed at the postulated load cases on the pertinent levels of defence are transferred such that the acceptance criteria to Section 4 are adhered to,

b) the core components can safely be handled, transported and stored,

c) the compatibility of the core components among each other and with the adjacent systems is satisfied,

d) the materials used for the respective components withstand the chemical, mechanical, thermal, and radiation-induced loadings to be expected during the service time,

e) assumptions and boundary conditions which the verifications of the adjacent systems and areas under analysis on the pertinent levels of defence were based on, are taken into account, such as

   ea) nuclear criticality safety during handling, transport and storage,

   eb) neutron-physical design of the reactor core,

   ec) thermo-hydraulic design of the reactor core and the plant,

   ed) behaviour of the shutdown systems,

f) interactions between the thermo-hydraulic and neutron-physical design and the thermo-mechanical design (e.g. in case of fuel assembly deformations) have been evaluated and be taken into account, where necessary,

g) assumptions and boundary conditions resulting from the manufacture of the core components (such as machining tolerances, quality of weld seams, etc.) are taken into account, and

h) the experience gained from operation is considered.

5.1.2 Methodology of safety demonstration

(1) The verifications shall be made using appropriate procedures.

(2) For a verification

a) deterministic analyses,

b) statistical analyses,

c) engineering assessment,

d) results obtained from the evaluation of tests and experiments,

e) results obtained from the evaluation of operating experience

may be applied individually or as combination of a) to e).

5.1.3 Extent and depth of safety demonstration

(1) The design verifications shall deal with at least the following items:

a) proof of compliance with the acceptance criteria for core components on all levels of defence,

b) verification of the design principles used.

Note:

Such principles are e.g.:

- description and validation of the analysis models and material laws,
- description and results of component part tests and material tests on irradiated and non-irradiated material.

(2) The verifications shall contain the boundary conditions to be observed. At least the following shall be shown:

a) assumed load cases,

b) assumed loadings,

c) geometry and material data used, and

d) compatibility data from adjacent systems and core components.

(3) In addition, the following shall be shown:

a) the models and material properties used,

b) the calculation methodology in case of calculations,

c) the boundary conditions of the tests in case of test-based safety demonstration, and

d) knowledge gained from operational experience as far as used for safety demonstration.

(4) The degree of detailing of the analysis methods applied as well as of the modelling used are based on the tasks and the required verification exactness.

(5) Experimental safety demonstrations shall be performed such that transformation to real conditions is possible. The design, performance and evaluation of experimental studies shall consider the following parameters:

a) influence of the model scale,

b) differences in dimensions of actual component and test component,

c) differences in the governing material properties, and

d) number of tests.

5.1.4 Uncertainties during the safety demonstration process

(1) During the safety demonstration process on levels of defence 1 to 3 the relevant loadings and boundary conditions shall be selected such that uncertainties in the use of the methods to 5.1.2 (2) are considered such that a conservative result is obtained with regard to the acceptance target.

(2) In this case, inter alia the uncertainties due to:

a) fabrication tolerances, geometry and material properties,

b) calculation models and

c) variation of operating conditions

shall be considered.

(3) For the analyses the following shall be performed:

a) to identify the parameters (initial and boundary conditions as well as model parameters) and models which may significantly influence the uncertainties of results,

b) to quantify according to the current state of knowledge the uncertainty range of identified parameters, in case of statistical procedures along with the distribution of these parameters, as well as

c) to determine and consider dependencies and interactions between individual parameters.

(4) Uncertainties may be covered within a deterministic analysis by

a) the use of standardized proven procedures or data from which the uncertainty or a safe distance to the acceptance criteria can be derived,

b) using the results with additional safety margins which are to be derived from the validation of the analysis procedure, experimental results or operational experience,

c) a combination of the most unfavourable values of the uncertainty range of the individual parameters referred to the acceptance criteria, or

Note:

Provided that the result is a monotonously increasing or decreasing function of the initial parameters, the boundary values of the initial parameters provide the extreme values.

d) the use of sufficiently conservative individual parameters or models for which it has been proved in a comparable case that the uncertainties regarding the pertinent acceptance criteria are covered.

(5) The observance of statistical acceptance criteria shall be proved with a statistical certainty of at least 95%.

(6) If during the determination of the total uncertainty statistical methods (e.g. to DIN ISO 16269-6) are applied, the tolerance limit of the result value biased towards the acceptance criteria shall be determined in which case a probability of at least 95% with a statistical certainty of 95% shall be proved as regards the observance of the acceptance criteria.

5.1.5 Validation of analysis procedures

(1) The analysis procedures used (e.g. calculation programs) shall be validated.

(2) In validation, differentiation is made between validation of the total calculation system used for the pertinent range of application (integral validation) and validation of individual components of the calculation system (partial validation). Beside the integral validation of the calculation system, the range of application should be verified by partial validation of the individual components.

Note:

Partial and integral validations are complementary and as a rule are being combined. Where integral methods are applied alone, error compensation cannot be excluded. Therefore, extrapolability in the range of application shall be considered to be possible to a less degree. On the other hand, the proof of complete coverage of the total system by individual validation steps may be difficult in cases where only partial validation methods are applied.

(3) The models used in analysis procedures (e.g. calculation methods) shall describe the real conditions and processes to be modelled with sufficient accuracy with regard to the acceptance target.

(4) For the purpose of validation, the results obtained from analysis procedures with reference solutions gained from results of experiments, materials testing, standard problems or the results of other validated analytical procedures shall be compared. The degree of conformity (systematic and non-systematic deviations shall be determined. Systematic deviations shall be assessed separately.

(5) Reference solutions should cover the range of application with regard to the essential parameters. In cases where no reproduction of actual reactor conditions is made, the transferability of the reference solutions on reactor conditions shall be justified.

(6) The range of validity of a calculation model follows from the validation of the calculation model and shall be indicated. The calculation model may be applied beyond the verified range of application (by extrapolation) if the admissibility of the extrapolations is justified.

(7) Validation on levels of defence 1 and 2 is to be performed as realistically as possible using actually occurring events or loadings.

(8) For the validation on level of defence 3, references shall be made to events on levels of defence 1 and 2 where possible. In addition, the validation shall be made using representative test.

(9) As far as possible, models shall be used on level of defence 4a which are also used for verifications on level of defence 3. If this is not possible, the models shall be established to the state of knowledge.

5.1.6 Verification

Calculation methods and models used for safety demonstration shall have been verified.

5.1.7 Documentation

The proofs required under 5.1.1 to 5.1.6 shall be documented in a verifiable manner so that they can be reviewed by experts.

5.2 Requirements for manufacture

(1) Manufacture shall only be made based on previously approved documents (design-review documents).

(2) The design review documents shall at least consist of parts lists, drawings and specifications.

(3) The requirements for manufacture shall be clearly laid down in the design-review documents. This covers, inter alia, the design including tolerance range, the materials used, the material properties, the surface condition, the fabrication and test procedures, the extent of quality tests as well as the type and extent of the quality certifications to be issued. In addition, it shall be laid down whether the tests and inspections are to be performed by the fabrication department or a fabrication-independent quality department.

(4) The manufacturing-relevant properties on which design is based shall cover the properties laid down in the design-review documents.

5.3 Requirements for transportation

(1) The transportation means, routes and packaging shall be selected such that the quality documented by the quality certificates as well as the functional capability of the core components is maintained.

Note:

During the transport of nuclear fuel further requirements have to be satisfied regarding packaging and containers due to other rules and specifications.

(2) It shall be ensured that the loadings occurring during transport are within the loadings assumed in the design.

(3) The limitation of loadings shall be proved by appropriate certificates, e.g. by qualification of the transportation process or by acceleration monitoring.

(4) In individual cases, loadings may occur during transport (e.g. accelerations) which may affect the product quality or the usability of core components. Where these loadings have not been covered by the design, the usability shall be evaluated separately (see 4.2.4 and 4.4.4).

5.4 Requirements for operational planning and operation

(1) The operational planning and operation of core components shall be such that

a) the boundary conditions on which thermomechanical design proofs of core components on levels of defence 1 to 4a are based, such as

   aa) operational parameters of the reactor (inter alia, pressure and temperature of the primary coolant as well as coolant chemistry),

   ab) power limits,

   ac) power ramps or load step changes, power history,

   ad) time limits for service life,

   ae) limitation of load case frequencies

   are adhered to, and

b) the experience gained from operation of core components is considered.

(2) The operational planning and operation of the core components shall be such that the conditions of specified normal operation which were taken as a basis for the initial conditions for anomalous operation and accidents during design, are adhered to.

Annex A

Tabulated overview of safety-related requirements

A 1 Fuel assemblies

Level of defence 1 (Specified normal operation) and 2 (Anomalous operation)	
Fundamental safety function R: Reactivity control	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)	**Requirements for fuel assemblies**
- Reactivity changes shall be restricted to admissible values. - It shall be possible to shut down the reactor core and keep it subcritical in the long term. Safety-related acceptance target: Power adaptation or reactor shutdown	- The condition and operation of the fuel assemblies shall be such that the fuel assembly geometry (shape and position) required for reactivity control and the required material properties of the fuel assemblies are adhered to.
Fundamental safety function K: Cooling of fuel assemblies	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)	**Requirements for fuel assemblies**
- Heat transfer from fuel to heat sink shall be ensured. Safety-related acceptance target: Unrestricted continued usability of fuel assemblies	- The condition and operation of the fuel assemblies shall be such that the allowable values of power and power density are adhered to. - The condition and operation of the fuel assemblies shall be such that the geometry (shape and position) required for heat removal and the required material properties of the fuel assemblies are adhered to. - The condition and operation of the fuel assemblies shall be such that the requirements laid down in KTA 3101.1 for critical boiling conditions/heat flux densities are adhered to.
Fundamental safety function B: Confinement of radioactive materials	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)	**Requirements for fuel assemblies**
- The mechanical, thermal, chemical and radiation-induced impacts resulting on the different levels of defence for the barriers or retention functions shall be limited such that their effectiveness regarding the achievement of the radiological safety objectives according to para. 2.5 is maintained. Safety-related acceptance target: Maintenance of barrier integrity	- The condition and operation of the fuel assemblies shall be such that tightness of the fuel assemblies is ensured.
Radiological safety objective S: Limitation of radiation exposure	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.5 (1)	**Requirements for fuel assemblies**
- Radiation exposure of the personnel shall be kept as low as achievable for all activities, even below the limits of the Radiation Protection Ordinance, taking into account all circumstances of each individual case. - Any radiation exposure or contamination of man and the environment by direct radiation from the plant as well as by the discharge of radioactive materials shall be kept as low as achievable, even below the limits of the Radiation Protection Ordinance, taking into account all circumstances of each individual case.	- Utilization of suitable materials to minimize radiation exposure.

Table A 1-1: Safety requirements for fuel assemblies, Defence levels 1 and 2

Level of defence 3: Accidents	
Fundamental safety function R: Reactivity control	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)	**Requirements for fuel assemblies**
- Reactivity changes shall be restricted to admissible values. - It shall be possible to shut down the reactor core and keep it subcritical in the long term. Safety-related acceptance target: Reactor shutdown	- The design of the fuel assemblies shall be such that the geometry (shape and position) required for reactivity and power density control and the required material properties are adhered to.
Fundamental safety function K: Cooling of fuel assemblies	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)	**Requirements for fuel assemblies**
- Ensuring the heat removal from the fuel rods, fuel assemblies and the core (heat transfer from fuel to heat sink shall be ensured; Safety requirements for nuclear power plants no. 2.3(2)) Safety-related acceptance target: Possibility of shutdown and coolability of the reactor core	- The design of the fuel assemblies shall be such that the geometry (shape and position) required for heat removal and the required material properties of the fuel assemblies are adhered to.
Fundamental safety function B: Confinement of radioactive materials	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)	**Requirements for fuel assemblies**
- The mechanical, thermal, chemical and radiation-induced impacts resulting on the different levels of defence for the barriers or retention functions shall be limited such that their effectiveness regarding the achievement of the radiological safety objectives according to para. 2.5 is maintained. Safety-related acceptance target: Maintenance of barrier integrity	- The design of the fuel assemblies shall be such that the event-related requirements for the tightness of the fuel rods is ensured.
Radiological safety objective S: Limitation of radiation exposure	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.5 (1)	**Requirements for fuel assemblies**
- The on-site and off-side radiological consequences shall be kept as low as possible, taking into account all circumstances of each individual case.	Covered by the requirements under fundamental safety function B “Confinement of radioactive materials“.

Table A 1-2: Safety-related requirements for fuel assemblies, Level of defence 3

<table>
<tr><td colspan="2">Level of defence 4a: Very rare events</td></tr>
<tr><td colspan="2">Fundamental safety function R: Reactivity control</td></tr>
<tr><td>Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)</td><td>Requirements for fuel assemblies</td></tr>
<tr><td>- Reactivity changes shall be restricted to admissible values.
- It shall be possible to shut down the reactor core and keep it subcritical in the long term.

Safety-related acceptance target:
Reactor shutdown</td><td>- The design of the fuel assemblies shall be such that the geometry (shape and position) required for reactivity and power density control and the required material properties are adhered to.</td></tr>
<tr><td colspan="2">Fundamental safety function K: Cooling of fuel assemblies</td></tr>
<tr><td>Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)</td><td>Requirements for fuel assemblies</td></tr>
<tr><td>- The heat transfer from fuel to heat sink shall be ensured.

Safety-related acceptance target:
Possibility of shutdown and coolability of the reactor core</td><td>- The design of the fuel assemblies shall be such that the geometry (shape and position) required for heat removal and the required material properties of the fuel assemblies are adhered to.</td></tr>
<tr><td colspan="2">Fundamental safety function B: Confinement of radiological materials</td></tr>
<tr><td>Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)</td><td>Requirements for fuel assemblies</td></tr>
<tr><td>- The mechanical, thermal, chemical and radiation-induced impacts resulting on the different levels of defence for the barriers or retention functions shall be limited such that their effectiveness regarding the achievement of the radiological safety objectives according to para. 2.5 is maintained.

Safety-related acceptance target:
Maintenance of barrier integrity</td><td>none [1]</td></tr>
<tr><td colspan="2">Radiological safety objective S: Limitation of radiation exposure</td></tr>
<tr><td>Requirements as per Safety Requirements for Nuclear Power Plants no. 2.5 (1)</td><td>Requirements for fuel assemblies</td></tr>
<tr><td>- The on-site and off-side radiological consequences shall be kept as low as possible, taking into account all circumstances of each individual case.</td><td>none [1]</td></tr>
<tr><td colspan="2">[1] There is no specific radiological safety objective. The general radiological safety objectives as per the Safety Requirements for Nuclear Power Plants no. 2.5 always apply.</td></tr>
</table>

Table A 1-3: Safety-related requirements for fuel assemblies, Level of defence 4a

A 2 Control assemblies

Level of defence 1 (Specified normal operation) and 2 (Anomalous operation)	
Fundamental safety function R: Reactivity control	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)	**Requirements for control assemblies**
- Reactivity changes shall be restricted to admissible values. - It shall be possible to shut down the reactor core and keep it subcritical in the long term. Safety-related acceptance target: Power adaptation or reactor shutdown	- The condition and operation of the control assemblies shall be such that the geometry and shape of the control assemblies required for power control and shutdown including the quantity, geometry (shape and position) and the material properties of the absorber material satisfy the requirements of KTA 3101.2 and KTA 3103. - The condition and operation of the control assemblies shall be such that, in consideration of their dead weight and the loadings occurring, they can be inserted at sufficient rate into the reactor core in accordance with the requirements of KTA 3101.2 and KTA 3103.
Fundamental safety function K: Cooling of fuel assemblies	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)	**Requirements for control assemblies**
- The heat transfer from fuel to heat sink shall be ensured. Safety-related acceptance target: Unrestricted continued usability of fuel assemblies	- The design of the control assemblies shall be such that the geometry (shape and position) required for heat removal and the required material properties are adhered to.
Fundamental safety function B: Confinement of radiological materials	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)	**Requirements for control assemblies**
- The mechanical, thermal, chemical and radiation-induced impacts resulting on the different levels of defence for the barriers or retention functions shall be limited such that their effectiveness regarding the achievement of the radiological safety objectives according to para. 2.5 is maintained. Safety-related acceptance target: Maintenance of barrier integrity	- The condition of the control assemblies shall be such that no inadmissible radiation exposure is caused by the control assemblies.
Radiological safety objective S: Limitation of radiation exposure	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.5 (1)	**Requirements for control assemblies**
- Radiation exposure of the personnel shall be kept as low as achievable for all activities, even below the limits of the Radiation Protection Ordinance, taking into account all circumstances of each individual case. - Any radiation exposure or contamination of man and the environment by direct radiation from the plant as well as by the discharge of radioactive materials shall be kept as low as achievable, even below the limits of the Radiation Protection Ordinance, taking into account all circumstances of each individual case.	- The condition and operation of the control assemblies shall be such that no inadmissible radiation exposure is caused by the control assemblies.

Table A 2-1: Safety-related requirements for control assemblies, Levels of defence 1 and 2

<table>
<tr><th colspan="2">Level of defence 3: Accidents</th></tr>
<tr><th colspan="2">Fundamental safety function R: Reactivity control</th></tr>
<tr><th>Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)</th><th>Requirements for control assemblies</th></tr>
<tr><td>- Reactivity changes shall be restricted to admissible values.
- It shall be possible to shut down the reactor core and keep it subcritical in the long term..

Safety-related acceptance target:
Reactor shutdown</td><td>- The design of the control assemblies shall be such that the geometry and shape of the control assemblies required for shutdown including the quantity, geometry (shape and position) and the material properties of the absorber material satisfy the requirements of KTA 3101.2 and KTA 3103.
- The design of the control assemblies shall be such that, in consideration of their dead weight and the loadings occurring, they can be inserted at sufficient rate into the reactor core in accordance with the requirements of KTA 3101.2 and KTA 3103.</td></tr>
<tr><th colspan="2">Fundamental safety function K: Cooling of fuel assemblies</th></tr>
<tr><th>Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)</th><th>Requirements for control assemblies</th></tr>
<tr><td>- The heat transfer from fuel to heat sink shall be ensured.

Safety-related acceptance target:
Possibility of shutdown and coolability of the reactor core</td><td>- The design of the control assemblies shall be such that the geometry (shape and position) required for heat removal and the required material properties are adhered to.</td></tr>
<tr><th colspan="2">Fundamental safety function B: Confinement of radiological materials</th></tr>
<tr><th>Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)</th><th>Requirements for control assemblies</th></tr>
<tr><td>- The mechanical, thermal, chemical and radiation-induced impacts resulting on the different levels of defence for the barriers or retention functions shall be limited such that their effectiveness regarding the achievement of the radiological safety objectives according to para. 2.5 is maintained.

Safety-related acceptance target:
Maintenance of barrier integrity</td><td>- The design of the control assemblies shall be such that no inadmissible radiation exposure is caused by the control assemblies.</td></tr>
<tr><th colspan="2">Radiological safety objective S: Limitation of radiation exposure</th></tr>
<tr><th>Requirements as per Safety Requirements for Nuclear Power Plants no. 2.5 (1)</th><th>Requirements for control assemblies</th></tr>
<tr><td>- The on-site and off-side radiological consequences shall be kept as low as possible, taking into account all circumstances of each individual case.</td><td>- The design of the control assemblies shall be such that no inadmissible radiation exposure is caused by the control assemblies.</td></tr>
</table>

Table A 2-2: Safety-related requirements for control assemblies, Level of defence 3

Level of defence 4a: Very rare events	
Fundamental safety function R: Reactivity control	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)	**Requirements for control** assemblies
- Reactivity changes shall be restricted to admissible values. - It shall be possible to shut down the reactor core and keep it subcritical in the long term. Safety-related acceptance target: Reactor shutdown	- The design of the control assemblies shall be such that the geometry and shape of the control assemblies required for shutdown including the quantity, geometry (shape and position) and the material properties of the absorber material satisfy the requirements of KTA 3101.2 and KTA 3103. - The design of the control assemblies shall be such that, in consideration of their dead weight and the loadings occurring, they can be inserted at sufficient rate into the reactor core in accordance with the requirements of KTA 3101.2 and KTA 3103.
Fundamental safety function K: Cooling of fuel assemblies	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)	**Requirements for control assemblies**
- The heat transfer from fuel to heat sink shall be ensured. Safety-related acceptance target: Possibility of shutdown and coolability of the reactor core	- The design of the control assemblies shall be such that the geometry (shape and position) required for heat removal and the required material properties are adhered to.
Fundamental safety function B: Confinement of radiological materials	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.3 (2)	**Requirements for control assemblies**
- The mechanical, thermal, chemical and radiation-induced impacts resulting on the different levels of defence for the barriers or retention functions shall be limited such that their effectiveness regarding the achievement of the radiological safety objectives according to para. 2.5 is maintained. Safety-related acceptance target: Maintenance of barrier integrity	none [1]
Radiological safety objective S: Limitation of radiation exposure	
Requirements as per Safety Requirements for Nuclear Power Plants no. 2.5 (1)	**Requirements for control assemblies**
- The on-site and off-side radiological consequences shall be kept as low as possible, taking into account all circumstances of each individual case.	none [1]
[1] There is no specific radiological safety objective. The general radiological safety objectives as per the Safety Requirements for Nuclear Power Plants no. 2.5 always apply.	

Table A 2-3: Safety-related requirements for control assemblies, Level of defence 4a

Annex B

Requirements regarding analytical and experimental verification of strength

B 1 General stipulations

(1) The verification of strength includes the stress, fatigue, strain and buckling resistance analysis.

(2) The determination of stress/strain loadings may be effected by way of analytical or experimental analysis or by a combination of both methods.

N o t e :

The component-specific requirements to be met are laid down in the main part of this Safety Standard.

This Annex provides methods by which a proof of strength can be rendered. It specifies the corresponding requirements given in the main part of this Safety Standard and the safety factors required for the respective method.

B 2 Analytical verification of strength

B 2.1 General

(1) The verification may be rendered by elastic or limit analysis on all levels of defence, and additionally on level of defence 3 by plastic analysis.

N o t e :

The prerequisite for an elastic analysis is the classification of the stresses occurring into the classes given in section B 2.2.1.1.

(2) In an elastic analysis, the stresses and in a plastic analysis the stresses and plastic strains are subject to evaluation.

(3) Analyses regarding fasteners are dealt with in Section B 2.7.

B 2.2 Stress analysis

B 2.2.1 Elastic analysis

B 2.2.1.1 Classification of stresses

B 2.2.1.1.1 General

(1) For the analysis stresses shall be classified in dependence of the cause of stress and its effect on the mechanical behaviour of the structure into primary stresses, secondary stresses and peak stresses and be limited in different ways with regard to their classification.

(2) Where the classification into the aforementioned stress categories is unclear, the effect of plastic deformation on the load-bearing behaviour shall govern the classification where an increase of the intended loading is assumed.

B 2.2.1.1.2 Primary stresses

(1) Primary stresses are stresses which satisfy the laws of equilibrium of external forces and moments (loads).

(2) Regarding the mechanical behaviour of a structure the basic characteristic of this stress is that in case of (an inadmissibly high) increment of external loads the distortions upon full plasticisation of the section considerably increase without being self-limiting.

(3) Regarding primary stresses distinction shall be made between membrane stresses (P_m) and bending stresses (P_b) with respect to their distribution across the cross-section governing the load-bearing behaviour. Here, membrane stresses are defined as the average value of the stresses distributed over the wall thickness. Bending stresses are defined as the portion of the stresses distributed across the wall thickness that can be altered linearly. In case of asymmetric cross-sections, the stresses shall be weighted in accordance with the respective surfaces.

(4) Where at stressed regions (discontinuities) local membrane stresses occur, these stresses may be assessed separately, i.e. by a limit analysis.

B 2.2.1.1.3 Secondary stresses

(1) The basic characteristics of secondary stresses are that they are self-limiting. These stresses can cause local plastic deformation and slight distortions. Failure due to a single load application need not be expected.

(2) Examples of secondary stresses (Q) are stresses developed under mechanical or thermal loads due to

a) geometric discontinuities,

b) different elastic constants (i.e. modulus of elasticity) as well as

c) different thermal expansions or solid-state thresholds.

(3) For the elastic analysis only stresses that are distributed linearly across the cross-section are considered to be secondary stresses.

B 2.2.1.1.4 Peak stresses

Peak stress is that increment of stress which is additive to the sum of primary and secondary stresses. Since peak stress occurs only in locally limited areas, it cannot cause any noticeable distortion and is objectionable only in conjunction with primary and secondary stresses as a possible source of fatigue (see B 2.4) or brittle fracture (see B 2.5).

B 2.2.1.2 Superposition and evaluation of stresses

B 2.2.1.2.1 General

(1) The following stresses acting simultaneously shall be added separately to obtain summed-up stresses:

a) primary membrane stresses,

b) primary membrane plus bending stresses and

c) primary membrane plus bending stresses and secondary stresses.

(2) From these summed-up stresses the stress intensities required for the evaluation of stresses in the following sections are derived.

B 2.2.1.2.2 Stress intensities

(1) Having chosen a three-dimensional set of coordinates the stress intensities shall be obtained from the summed-up primary and secondary stresses using an appropriate strength theory.

(2) The stress intensities shall be limited to material-specific admissible values.

(3) The safety factors for determination of allowable stresses specified in **Tables B 4-1** and **B 4-2** for different components and materials apply only when using the stress theory of von Mises or the Tresca`s maximum shear stress theory.

N o t e :

Prerequisite to the application of these stress theories is a ductile behavior of the material. This is satisfied by compliance with the requirements regarding ductility specified in Section 4.

B 2.2.2 Plastic analysis

The performance of plastic analysis shall be made according to the methodology laid down in KTA 3204.

B 2.2.2.1 Classification of stresses

The primary stresses to clause B 2.2.1.1.2 determined on level of defence 3 by plastic analysis shall be limited according to **Table B 4-2**. In addition, a deformation analysis to clause B 2.3 shall be performed with regard to local failure.

B 2.2.2.2 Stress intensities

(1) Having chosen a three-dimensional set of coordinates the stress intensities shall be obtained from the summed-up primary stresses using an appropriate strength theory.

(2) The stress intensities shall be limited to material-specific admissible values.

(3) The safety factors for determination of allowable stresses specified in **Table B 4-2** for different components and materials apply only when using the stress theory of von Mises or the Tresca`s maximum shear stress theory.

Note:

The strain hardening of the material may be considered for both the effective monotonous stress-strain curve at loading temperature and for any approach of the effective stress-strain curve where, in the latter case, the approach shall show, at any point, less stresses for equal strains than the effective curve.

B 2.2.3 Limit analysis

Limit analyses shall be performed according to Annex B of KTA 3204.

B 2.3 Deformation analysis

(1) A deformation analysis shall be performed if specified strain limits (elastic or plastic) are to be adhered to for functional or compatibility reasons.

(2) If a plastic strain analysis to clause B 2.2.2 is performed, a deformation analysis with regard to local failure shall also be performed.

Note:

Further details regarding the performance of deformation analysis with regard to local failure can be found i.e. in ASME BPVC VIII, Division 2, Chapter 5.3.

B 2.4 Fatigue analysis

B 2.4.1 General stipulations

(1) A fatigue analysis shall be made to avoid fatigue failure due to cyclic loading. Hereinafter distinction is made between elastic fatigue analysis and simplified elastic-plastic fatigue analysis.

Note:

In practice, at first an elastic fatigue analysis is performed; a simplified elastic-plastic fatigue analysis may be performed if necessary. The general elastic-plastic fatigue analysis is not dealt with in this Safety Standard because experience shows that it need not be performed for core components.

(2) The methodological approach shall be based on KTA 3204.

Note:

The fatigue analysis procedure shown in KTA 3204, clause 6.2.4.2.3 and in the text hereafter does not consider the following factors of influence:

a) high-cycle loadings due to vibration excitation in combination with loadings in the endurance limit range (e.g. due to thermal transients),

b) the possible reduction of the endurance limit in the ultra-high cycle range ($N > 2 \cdot 10^7$),

c) the influence of radiation (especial neutron irradiation),

d) the influence of strain hardening in case of austenitic materials subject to the fluid conditions,

e) long-term effect of hydrogen.

According to the current state of knowledge, a significant influence of these factors on fatigue damage is not assumed. These factors of influence are objects of research.

These factors of influence will be taken into consideration, where required, if secured experimental studies are available.

(3) The elastic fatigue analysis is only permitted if the equivalent stress range resulting from primary and secondary stresses does not exceed a value of $3 \cdot S_m$ for steels, zirconium-base alloys and nickel-base alloys as well as the value of $4 \cdot S_m$ for cast steel.

(4) The simplified elastic-plastic fatigue analysis may be used for load cycles where the equivalent stress range resulting from all primary and secondary stresses exceeds the limit value of $3 \cdot S_m$ for parts made of steel, zirconium-base alloys and nickel-base alloys as well as the value of $4 \cdot S_m$ for parts made of cast steel, however, these limit values are adhered to by the equivalent stress range resulting from primary and secondary stresses due to mechanical loads. The influences of plasticisation are considered by using the factor K_e according to clause B 2.4.3.

(5) If a simplified elastic-plastic fatigue analysis is needed, it shall be demonstrated that no failure due to ratcheting (progressive distortion) occurs.

(6) Linear-elastic stresses shall be used for the elastic and for the simplified elastic-plastic fatigue analysis. All simultaneously acting stresses from all stress categories shall be considered to ensure fatigue resistance.

(7) The basis for fatigue evaluation are design fatigue curves to be determined for the specific materials, which create a correlation between equivalent stress ranges and allowable numbers of load cycles.

(8) In the fatigue analysis the loadings for all relevant load cases shall be superpositioned.

Note:

Load cases (load-time functions) may be comprised to form service loading combinations.

B 2.4.2 Equivalent stress ranges

(1) The following equivalent stress ranges shall be determined based on different stress categories:

$S_{n,PQ}$ - using the simultaneously acting stresses from the primary and secondary stress categories

$S_{n,PQF}$ - using the simultaneously acting stresses from all stress categories.

(2) The following two paragraphs describe calculation procedures based on principal stresses. Alternatively, the equivalent stress ranges $S_{n,\,PQ}$ and $S_{n,\,PQF}$ may also be determined using the stress tensors in any system of coordinates.

(3) For those cases, where the directions of principal stresses at the respective point in the structure will not change during the stress cycle, at first the principal stresses $\sigma_1(t)$, $\sigma_2(t)$ and $\sigma_3(t)$ shall be determined over the time of the stress cycle. From the three principal stresses the three time-dependent principal stress differences shall then be formed

$$\begin{aligned} S_{1,2}(t) &= \sigma_1(t) - \sigma_2(t) \\ S_{2,3}(t) &= \sigma_2(t) - \sigma_3(t) \\ S_{3,1}(t) &= \sigma_3(t) - \sigma_1(t) \end{aligned} \qquad \text{(B 2.4.2-1)}$$

Finally, the procedure to a) or b) shall be followed.

a) The equivalent stress ranges $S_{n,\,PQ}$ or $S_{n,\,PQF}$ to the von Mises theory shall be determined to the following rule in the stress cycle over all points in time t_i und t_j:

$$\max_{t_i,t_j} \sqrt{\frac{\left(S_{1,2}(t_i)-S_{1,2}(t_j)\right)^2+\left(S_{2,3}(t_i)-S_{2,3}(t_j)\right)^2+\left(S_{3,1}(t_i)-S_{3,1}(t_j)\right)^2}{2}}$$

(B 2.4.2-2)

b) The equivalent stress ranges $S_{n,\,PQ}$ or $S_{n,\,PQF}$ to the Tresca`s maximum shear stress theory shall be determined to the following rule in the stress cycle over all points in time t_i and t_j:

$$\max_{t_i,t_j} \left\{|\,S_{1,2}(t_i)-S_{1,2}(t_j)\,|,|\,S_{2,3}(t_i)-S_{2,3}(t_j)\,|,|\,S_{3,1}(t_i)-S_{3,1}(t_j)\,|\right\}$$

(B 2.4.2-3)

(4) For those cases, where the directions of principal stresses at the respective point in the structure will change during the stress cycle, at first the differences of normal and shear stress components shall be determined in the stress cycle over all points in time t_i and t_j:

$$\begin{aligned}
\Delta\sigma_X(t_i,t_j) &= \sigma_X(t_i)-\sigma_X(t_j)\\
\Delta\sigma_Y(t_i,t_j) &= \sigma_Y(t_i)-\sigma_Y(t_j)\\
\Delta\sigma_Z(t_i,t_j) &= \sigma_Z(t_i)-\sigma_Z(t_j)\\
\Delta\sigma_{XY}(t_i,t_j) &= \sigma_{XY}(t_i)-\sigma_{XY}(t_j)\\
\Delta\sigma_{YZ}(t_i,t_j) &= \sigma_{YZ}(t_i)-\sigma_{YZ}(t_j)\\
\Delta\sigma_{XZ}(t_i,t_j) &= \sigma_{XZ}(t_i)-\sigma_{XZ}(t_j)
\end{aligned} \qquad \text{(B 2.4.2-4)}$$

From the six time-dependent differences of stress components the principal stresses $\sigma_1(t_i, t_j)$ $\sigma_2(t_i, t_j)$ and $\sigma_3(t_i, t_j)$ shall be determined which depend on the points in time t_i and t_j.

From the three principal stresses the three possible differences shall be determined which also depend on the points in time t_i and t_j:

$$\begin{aligned}
S_{1,2}(t_i,t_j) &= \sigma_1(t_i,t_j)-\sigma_2(t_i,t_j)\\
S_{2,3}(t_i,t_j) &= \sigma_2(t_i,t_j)-\sigma_3(t_i,t_j)\\
S_{3,1}(t_i,t_j) &= \sigma_3(t_i,t_j)-\sigma_1(t_i,t_j)
\end{aligned} \qquad \text{(B 2.4.2-5)}$$

Note:

(1) The direction of the principal stresses $\sigma_1(t_i,t_j)$, $\sigma_2(t_i,t_j)$ and $\sigma_3(t_i,t_j)$ will change with the points in time t_i and t_j. The principal stresses shall maintain their identity during rotation.

(2) The largest equivalent stress range will be formed by comparison of the stress condition at the point in time t_i with each stress condition at the point in time t_j.

Finally, the procedures to a) an b) shall be followed.

a) The equivalent stress ranges $S_{n,\,PQ}$ and $S_{n,\,PQF}$ to the von Mises theory shall be determined to the following rule in the stress cycle over all points in time t_i und t_j:

$$\max_{t_i,t_j} \sqrt{\frac{S_{1,2}^2(t_i,t_j)+S_{2,3}^2(t_i,t_j)+S_{3,1}^2(t_i,t_j)}{2}} \qquad \text{(B 2.4.2-6)}$$

b) The equivalent stress ranges $S_{n,\,PQ}$ and $S_{n,\,PQF}$ to the Tresca`s maximum shear stress theory shall be determined to the following rule in the stress cycle over all points in time t_i and t_j:

$$\max_{t_i,t_j} \left\{|\,S_{1,2}(t_i,t_j)\,|,|\,S_{2,3}(t_i,t_j)|,|\,S_{3,1}(t_i,t_j)\,|\right\} \qquad \text{(B 2.4.2-7)}$$

(5) In case of bolted joints the equivalent stress ranges shall be multiplied with a factor of 4 for stress raisers in the notch root of the thread if no more favourable values can be verified.

B 2.4.3 Determination of the allowable number of cycles

(1) The material specific design fatigue curves create a correlation between equivalent stress ranges and allowable numbers of load cycles.

Note:

The material-specific fatigue curves (so-called Wöhler curves or S/N curves) were established on the basis of uni-axial low- strain fatigue tests where the strains were multiplied with the modulus of elasticity so that stress units (fictitious values in the plastic range) can be plotted on the ordinate.

Fatigue curve examples for common materials can be found in Section B 5.

For fuel assembly structural parts made of zirconium alloys, e.g. the correlation to O'Donnell Langer (see Figures B 5-4 and B 5-5) is used (factors of influence: factor 2 for the stress amplitude, factor 20 to load cycles).

(2) From the equivalent stress range $S_{n,\,PQF}$ the reference value S_{alt} is determined as follows:

$$S_{alt} = x \cdot K_e \cdot S_{n,PQF} \qquad \text{(B 2.4.3-1)}$$

x = 1 or 0.5 depending on the definition of the ordinate used in the fatigue curve (half or full equivalent stress range).

$$K_e = 1.0 \text{ for } S_{n,PQ} \le 3 \cdot S_m \qquad \text{(B 2.4.3-2)}$$

$$K_e = 1.0 + \frac{(1-n)}{n \cdot (m-1)} \cdot \left(\frac{S_{n,PQ}}{3 \cdot S_m} - 1\right)$$

$$\text{for } 3 \cdot S_m < S_{n,PQ} < m \cdot 3 \cdot S_m \qquad \text{(B 2.4.3-3)}$$

$$K_e = \frac{1}{n} \text{ for } S_{n,PQ} \ge m \cdot 3 \cdot S_m \qquad \text{(B 2.4.3-4)}$$

where m and n are material parameters, which shall be taken from the following Table:

Type of material	m	n	T_{max} (°C)
Martensitic stainless steel	2.0	0.2	370
Austenitic stainless steel	1.7	0.3	425
Zirconium based alloy	1.7	0.3	425
Nickel based alloy	1.7	0.3	425

Table B 2-1: Material parameter

(3) In lieu of these K_e values other values may be used, which have been proved by experiments or calculation or have been taken from literature. Where the maximum temperature shown in the table is exceeded, the m and n values shall be adapted accordingly. Their applicability shall be verified.

Note:

As regards K_e =1.0 an elastic fatigue analysis is concerned and regarding K_e > 1 a simplified elastic-plastic fatigue analysis (necessity of verification of strength against failure by progressive distortion (ratcheting)) is concerned.

The literature referenced in [2] contains a proposal for the determination of K_e values.

(4) Where a simplified elastic-plastic fatigue analysis is performed, the material used shall show an elastic ratio less than 0.8.

(5) The number of load cycles N_k allowable for the service loading combination k is obtained from the design fatigue curve depending on the determined value S_{alt}.

Note:

Where the maximum allowable equivalent stress range is exceeded, the allowable number of load cycles cannot be determined using the given design fatigue curve.

B 2.4.4 Cumulative usage factor

(1) If n_k is the number of load cycles within the service loading combination k, which is specified or the value to be assumed, the cumulative usage factor D is calculated as follows:

$$D = \sum_k \frac{n_k}{N_k} \qquad \text{(B 2.4.4-1)}$$

(2) The cumulative usage factor shall be limited to values equal to or less than 1.

(3) The influence of the fluid is addressed in section 4 for each specific component.

B 2.4.5 Analysis of progressive distortion (ratcheting)

(1) The equivalent stress range $S_{n,\,PQ}$ shall be used for assessment of progressive distortion (ratcheting).

(2) Where the equivalent stress range $S_{n,\,PQ}$ exceeds the value of $3 \cdot S_m$, it shall be proved that the distortions developing as a result of stress ratchet due to load cycles remain within acceptable limits, i.e. that functionality and compatibility remain unchanged.

(3) The same service loading combinations used in fatigue analysis shall be taken into consideration in the proof of progressive distortion limitation.

Note:

The proof of progressive distortion limitation may be effected following the stipulations in KTA 3201.2 section 7.13.

B 2.5 Assessment of brittle fracture risk

(1) A brittle fracture risk may exist in the case of higher strain rates, low temperatures or multi-axial stress conditions including residual stresses.

(2) It shall be ensured that during the whole service life of the component the required deformability is assured (especially in the case of plastic deformations).

Note:

Further details regarding the performance of brittle fracture analysis are given e.g. in KTA 3201.2.

B 2.6 Stability analysis

B 2.6.1 General

(1) Where under the effect of compressive loading a sudden deformation without considerable increase in load may be expected, a stability analysis shall be performed.

Note:

A stability failure on plates, shells or bar-shaped structures may occur e.g. due to buckling.

(2) Requirements regarding buckling of cylindric components and bar-shaped structures are given in the following sections.

B 2.6.2 Buckling of pressurized cylindrical components

(1) In the case of external pressure the pressure load Δp shall be less than the critical elastic buckling pressure $p_{krit,el}$ and be less than the critical plastic deformation pressure $p_{krit,pl}$ taking into account a component-specific safety factor. The pressure load Δp is the difference between external and internal pressure.

(2) The critical elastic buckling pressure shall be calculated by means of the following equation:

$$p_{krit,el} = \frac{E}{4 \cdot \left(1 - \nu^2\right)} \cdot \left(\frac{h_{min}}{r_{H,mittel}}\right)^3 \qquad \text{(B 2.6.2-1)}$$

where:

$p_{krit,el}$: critical elastic buckling pressure (N/mm^2)

E : modulus of elasticity (N/mm^2)

ν : Poisson`s ratio

h_{min} : local minimum wall thickness taking the service life into account (mm)

$r_{H,mittel}$: mean radius of the hollow cylinder wall (mm)

(3) The required safety factor S_{el} is shown in **Tables B 4-1** and **B 4-2** depending on the specific component.

(4) The following condition shall be satisfied:

$$\Delta p \le \frac{p_{krit,el}}{S_{el}} \qquad \text{(B 2.6.2-2)}$$

(5) The critical plastic deformation pressure $p_{krit,pl}$ shall be calculated by means of the following equation:

$$p_{krit,pl} = 2 \cdot \frac{h_{min} \cdot R_{p0.2T}}{\left(D_a - h_{min}\right)} \qquad \text{(B 2.6.2-3)}$$

where:

$p_{krit,pl}$: critical plastic deformation pressure (N/mm^2)

h_{min} : local minimum wall thickness taking the service life into account (mm)

$R_{p0.2T}$: 0.2 % proof stress of the cladding tube material at operating temperature (N/mm^2)

D_a : outside diameter of the cladding tube (mm)

(6) The required safety factor S_{pl} is shown in **Tables B 4-1** and **B 4-2** depending on the specific component.

(7) The following condition shall be satisfied:

$$\Delta p \le \frac{p_{krit,pl}}{S_{pl}} \qquad \text{(B 2.6.2-4)}$$

B 2.6.3 Buckling of bar-shaped structures

(1) Axial compressive loadings can result in buckling of component parts. The compressive stresses σ_{Druck} shall be limited or the component part must be designed such that these unstable conditions will not occur.

(2) The critical buckling stress σ_{knick} shall be determined for each individual component part and under consideration of the stress state.

Note:

In the case of purely elastic buckling the critical buckling stress can be determined by means of the Euler hyperbola provided that the slenderness ratio exceeds the limiting slenderness ratio value. Otherwise appropriate approximation methods are e.g. the theory of Engesser and von Kármán or the Tangent-Modulus Theory.

(3) The influence of geometric and structural imperfections (according to DIN EN 1993-1-1) shall be taken into account if they cause higher stresses. This may be done

a) by explicit consideration of the imperfections when deriving the critical buckling stress or

b) by appropriate selection of the safety factors S_k

Note:

The component-specific safety factors S_k shown in **Tables B 4-1** and **B 4-2** are appropriate for proofs not considering imperfections corresponding to the theory (e.g. to Euler or Tangent-Modulus).

(4) The influence of permanent geometry changes (e.g. creep deformations) occurring during reactor operation on the stability of the component part shall be evaluated and be taken into account, if necessary.

Note:

In case of major geometry changes, there is no stability problem in its strict sense. The proof of stability can be replaced by a stress analysis using the second order theory as well as by a proof that the resulting deformation does not inadmissibly affect the function of the component part.

(5) The following condition shall be satisfied:

$$\sigma_{Druck} \leq \frac{\sigma_{knick}}{S_k} \qquad \text{(B 2.6.3-1)}$$

(6) The component-specific safety factor S_k against buckling shown in **Tables B 4-1** and **B 4-2** shall be used only if no imperfections have been considered in the determination of the critical buckling stress (e.g. Euler or Tangent-Modulus).

B 2.7 Verifications for fasteners

B 2.7.1 General

In the text hereafter, analytical proofs for bolted and welded joints are dealt with. For other types of joints suitable methods, e.g. experimental verifications, shall be used.

B 2.7.2 Bolted joints

(1) For bolted joints, the location of highest loading shall be determined in which case the various failure mechanisms are to be observed.

Note:

Failure mechanisms are e.g.

- shearing-off of bolt head at too low head height,
- thread-stripping at too little thread engagement length,
- tearing-off or shearing-off of bolt shank at too little shank thickness and
- rupture in in the free loaded thread cross-section.

(2) The stress limits applying to bolted joints are given in **Tables B 4-1** and **B 4-2** for the levels of defence 1 to 3. As regards the stresses during assembly, the values of level of defence 1 shall be used.

(3) Further information on the design and construction of bolted joints can be found in VDI 2230.

(4) For bolted joints a fatigue analysis to B 2.4 shall be performed.

B 2.7.3 Welded joints

(1) For welded joints, a stress analysis to B 2.2 shall be performed where the allowable primary stress intensities for welded joints are to be formed from the allowable primary stress intensity of the base material multiplied with the weld factors v and v_1 for the type of loading and v_2 for the weld quality. The weld factors shall be taken from **Table B 4-1**.

(2) For welded joints a fatigue analysis to B 2.4 shall be performed.

B 3 Design by experimental analysis

(1) The stipulations of sub-clause 5.1.3 (5) apply to experimental proofs.

(2) The transferability of the results on real conditions in accordance with the stipulations of sub-clauses 5.1.3 (5) b) to d) is deemed to have been satisfied if it is verified by a test on a prototype or model in due consideration of its scale that the specified loads do not exceed

a) on levels of defence 1 and 2: 44 % and

b) on levels of defence 3 and 4: 90 %

of the rupture or collapse load determined in the test or of the test load obtained.

Note:

The consideration of the stipulations under a) and b) shall ensure that the loads determined by the test represent a conservative proof of the load-carrying capacity of the real structure under the specified loads.

B 4 Tables of safety factors

B 4-1 Table of safety factors and allowable stresses for levels of defence 1 and 2

<table>
<tr><td>Fundamental safety functions</td><td colspan="5">Reactivity control, cooling of fuel assemblies, confinement of radioactive materials, limitation of radiation exposure</td></tr>
<tr><td>Levels of defence</td><td colspan="5">Levels of defence 1 and 2</td></tr>
<tr><td>Components</td><td>Fuel rod</td><td>Fuel assembly structure [3)]</td><td>Absorber rod cladding tube (PWR)</td><td>Control assembly structure (PWR + BWR) and absorber-containing parts (BWR)</td><td>Flow restrictor assembly</td></tr>
<tr><td>Elastic buckling of cylindrical parts</td><td colspan="3">$S_{el} = 3$ for $U \geq 0.015$
$S_{el} = 2 \cdot 100 \cdot U$ for $0.01 \leq U < 0.015$
$S_{el} = 2$ for $U < 0.01$
Here U is the out-of-roundness of the tube.
The out-of-roundness shall be determined as follows:
$U = 2 \cdot \frac{d_{i,max} - d_{i,min}}{d_{i,max} + d_{i,min}}$
with:
U : out-of-roundness
$d_{i,max}$: maximum inside diameter
$d_{i,min}$: minimum inside diameter</td><td>Irrelevant for the control assembly structure (PWR).
The proof of BWR control assembly shall be performed depending on the geometry and the boundary conditions.</td><td>See control assembly structure.</td></tr>
<tr><td>Plastic deformation (plastic buckling)</td><td>$S_{pl} = 1.1$</td><td>irrelevant</td><td>$S_{pl} = 1.5$</td><td>Irrelevant for the control assembly structure (PWR).
The proof of BWR control assembly shall be performed depending on the geometry and the boundary conditions.</td><td>irrelevant</td></tr>
</table>

Table B 4-1: Table of safety factors and allowable stresses for levels of defence 1 and 2 (continued on next pages)

Components	Fuel rod	Fuel assembly structure [3)]		Absorber rod cladding tube (PWR)	Control assembly structure (PWR + BWR) and absorber-containing parts (BWR)	Flow restrictor assembly
Elastic buckling according to Euler / plastic buckling according to Engesser and von Kármán	S_k =1.5	S_k = 1.5		irrelevant	irrelevant	irrelevant
Allowable stresses for the stress categories [1)]		Austenitic steel (forgings or precision casting) Nickel based alloy	Zirconium based alloy			
P_m	S_1 = Min $\{0.9\ R_{p0.2T};0.5\ R_{mT}\}$	S_1 = Min $\{R_{p0.2RT}/1.5;$ $R_{p0,2T}/1.1;$ $R_{mRT}/3.0\ ;\ R_{mT}/2.7\}$ Precision casting: S_1 = Min $\{R_{p0.2T}/2;$ $R_{mRT}/4.0\ ;\ R_{mT}/3.6\}$	S_1 = Min $\{0.7\ R_{p0.2T};\ 0.5\ R_{mT}\}$	S_1 = Min $\{0.9\ R_{p0.2T};0.5\ R_{mT}\}$	S_1 = Min $\{R_{p0.2RT}/1.5\ ;\ R_{p0.2T}/1.1;$ $R_{mRT}/3.0\ ;\ R_{mT}/2.7\}$	See control assembly structure.
P_m+P_b	S_2 = Min $\{1.35\ R_{p0.2T};0.7\ R_{mT}\}$	S_2 = 1.5 S_1	S_2 = Min $\{1.0\ R_{p0.2T};\ 0.7\ R_{mT}\}$	S_2 = Min $\{1.35\ R_{p0.2T};0.7\ R_{mT}\}$	S_2 = 1.5 S_1	See control assembly structure.
$P_m + P_b + Q$	S_3 = Min $\{2.7\ R_{p0.2T};1.0\ R_{mT}\}$	S_3 = 3.0 S_1 Precision casting: S_3 = 4.0 S_1	S_3 = Min $\{2.1\ R_{p0.2T};\ 1.0\ R_{mT}\}$	S_3 = Min $\{2.7\ R_{p0.2T};1.0\ R_{mT}\}$	S_3 = 3.0 S_1	See control assembly structure.
Welded joints	For static loading: $S_1^* = S_1 \cdot v \cdot v_2$ $S_2^* = S_2 \cdot v \cdot v_2$ For dynamic loading: $S_1^* = S_1 \cdot v_1 \cdot v_2$ $S_2^* = S_2 \cdot v_1 \cdot v_2$ with: v : Weld factor for static loading (see Table 5-1 of KTA 3905) v_1 : Weld factor for dynamic loading (see Niemann [3]) = 0.1 up to 1.0 depending on weld geometry and type of loading v_2 : Weld factor for weld quality = 1.0 for proved weld quality = 0.5 without proof of weld quality					

Table B 4-1: Table of safety factors and allowable stresses for levels of defence 1 and 2 (continued)

Components	Fuel rod	Fuel assembly structure [3]	Absorber rod cladding tube (PWR)	Control assembly structure (PWR + BWR) and absorber-containing parts (BWR)	Flow restrictor assembly
Stress intensities in bolted joints	irrelevant	$S = 1.0\ R_{p0.2T}$	$S = 1.0\ R_{p0.2T}$	$S = 1.0\ R_{p0.2T}$	See control assembly structure.
Shear stresses in bolted joints	irrelevant	$S = 0.6\ R_{p0.2T}$	$S = 0.6\ R_{p0.2T}$	$S = 0.6\ R_{p0.2T}$	See control assembly structure.
Surface pressure	irrelevant	$S = 1.5\ R_{p0.2T}$	$S = 1.5\ R_{p0.2T}$	$S = 1.5\ R_{p0.2T}$	See control assembly structure.
Shear stresses in compression spring [4]	$S = 0.56\ R_{mT}$	$S = 0.56\ R_{mT}$	irrelevant	$S = 0.56\ R_{mT}$	See control assembly structure.
Fatigue evaluation	D [2] ≤ 1	D [2] ≤ 1	D [2] ≤ 1	D [2] ≤ 1	See control assembly structure.

1) Index 1 = stress intensity due to primary membrane stresses
Index 2 = stress intensity due to primary membrane and bending stresses
Index 3 = stress intensity due to primary and secondary membrane and bending stresses

S_m is used in Section B 2.4.
$S_m = S_3 / 3$ for all materials considered except austenitic precision casting
$S_m = S_3 / 4$ for austenitic precision casting

2) cumulative usage factor $D = \Sigma_k$ (number of load cycles to be assumed n_k / allowable number of load cycles N_k) of the service loading combination k

3) including fuel assembly channel (BWR)

4) stress (uncorrected in the case of static loading and corrected in the case of dynamic loading according to DIN EN 13906-1) at maximum spring deflection structurally possible

Table B 4-1: Table of safety factors and allowable stresses for levels of defence 1 and 2 (continued)

B 4-2 Table of safety factors and allowable stresses for level of defence 3

Fundamental safety functions	Reactivity control, cooling of fuel assemblies, confinement of radioactive materials, limitation of radiation exposure					
Levels of defence	Level of defence 3					
Components	Fuel rod	Fuel assembly structure [3]		Absorber rod cladding tube (PWR)	Control assembly structure (PWR + BWR) and absorber-containing parts (BWR)	Flow restrictor assembly
Elastic buckling	irrelevant	irrelevant		irrelevant	irrelevant	irrelevant
Plastic buckling	irrelevant	irrelevant		irrelevant	irrelevant	irrelevant
Elastic buckling according to Euler / plastic buckling according to Engesser and von Kármán	$S_k > 1.1$	$S_k > 1.1$		irrelevant	irrelevant	irrelevant
Allowable stresses for the stress categories [1]		Austenitic steel (forgings or precision casting) Nickel based alloy	Zirconium based alloy			
Stress intensities [1] P_m $P_m + P_b$	$S'_1 =$ $\text{Min}\,\{R_{p0.2T}\,,\,0.7\,R_{mT}\}$ $S'_2 =$ $\text{Min}\,\{1.5\,R_{p0.2T};\,R_{mT}\}$	$S'_1 =$ $\text{Min}\,\{2.4\,S_1;\,0.7\,R_{mT}\}$ $S'_2 =$ $\text{Min}\,\{3.6\,S_1;\,R_{mT}\}$	$S'_1 = 0.7\,R_{mT}$ $S'_2 = 1.0\,R_{mT}$	$S'_1 =$ $\text{Min}\,\{R_{p0.2T}\,;\,0.7\,R_{mT}\}$ $S'_2 =$ $\text{Min}\,\{1.5\,R_{p0.2T};\,R_{mT}\}$	$S'_1 =$ $\text{Min}\,\{R_{p0.2T};\,0.7\,R_{mT}\}$ $S'_2 =$ $\text{Min}\,\{1.5\,R_{p0.2T};\,R_{mT}\}$	irrelevant
Welded joints	For static loading: $S_1^{*)} = S_1' \cdot v \cdot v_2$ $S_2^{*)} = S_2' \cdot v \cdot v_2$ For dynamic loading: $S_1^{*)} = S_1' \cdot v_1 \cdot v_2$ $S_2^{*)} = S_2' \cdot v_1 \cdot v_2$ with: v : Weld factor for static loading (see Table 5-1 of KTA 3905) v_1 : Weld factor for dynamic loading (see Niemann [3]) = 0.1 up to 1.0 depending on weld geometry and type of loading v_2 : Weld factor for weld quality = 1.0 for proved weld quality = 0.5 without proof of weld quality					irrelevant

Table B 4-2: Safety factors and allowable stresses for level of defence 3 (continued next page)

Components	Fuel rod	Fuel assembly structure [2]	Absorber rod cladding tube (PWR)	Control assembly structure (PWR + BWR) and absorber-containing parts (BWR)	Flow restrictor assembly
Stress intensities in bolted joints	irrelevant	S = 1.1 $R_{p0.2T}$	irrelevant	S = 1.1 $R_{p0.2T}$	S = 1.1 $R_{p0.2T}$
Shear stresses in bolted joints	irrelevant	S = 0.7 $R_{p0.2T}$	irrelevant	S = 0.7 $R_{p0.2T}$	S = 0,7 $R_{p0,2T}$
Surface pressure	irrelevant	irrelevant	irrelevant	irrelevant	irrelevant
Shear stresses in compression spring	irrelevant	irrelevant	irrelevant	irrelevant	irrelevant

[1] Index 1 = stress intensity due to primary membrane stresses
Index 2 = stress intensity due to primary membrane and bending stresses

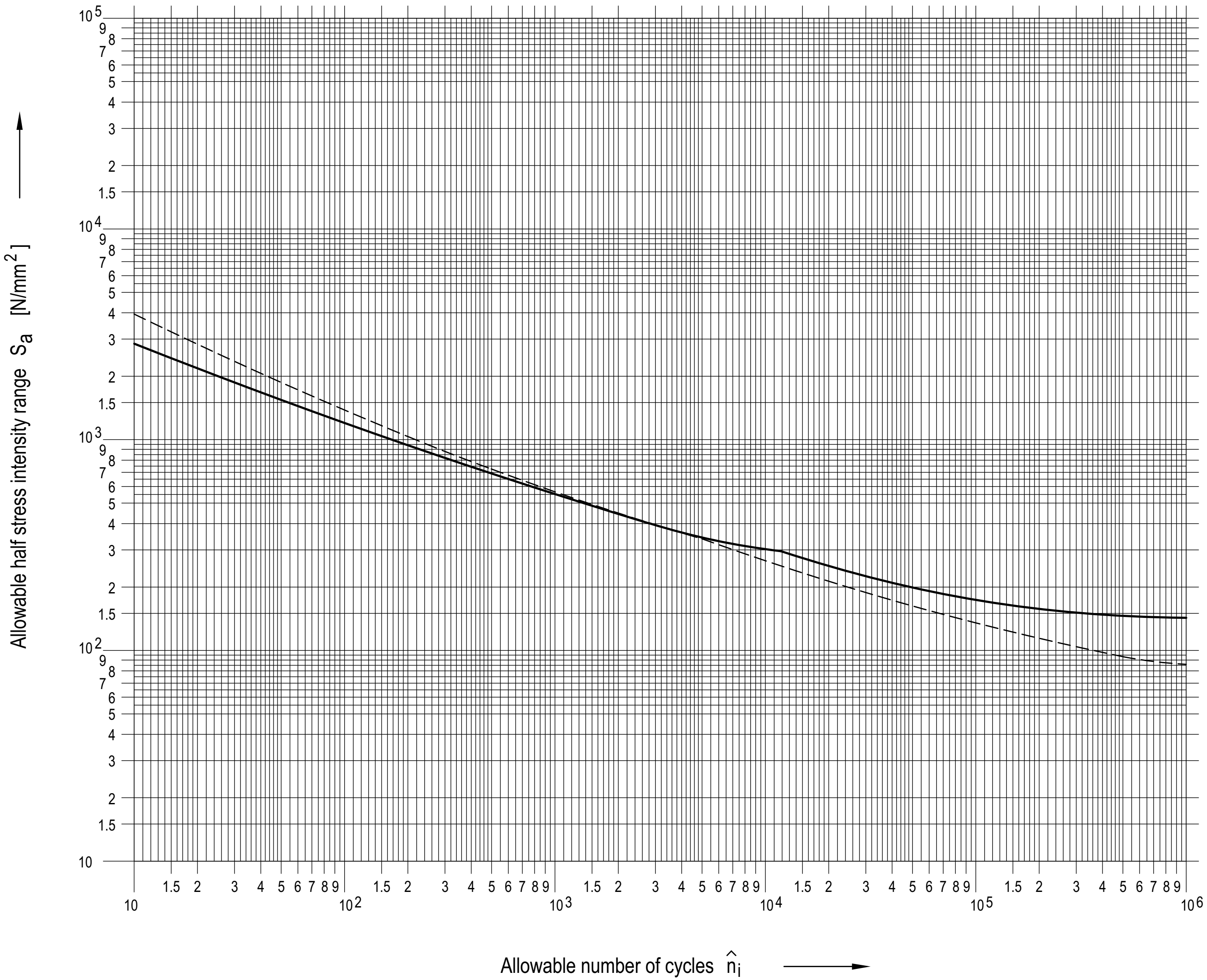
[2] including fuel assembly channel (BWR)

Table B 4-2: Safety factors and allowable stresses for level of defence 3 (continued)

B 5 Design fatigue curves

The design fatigue curves shown in **Figure B 5-1** to **Figure B 5-3** were taken over from KTA 3201.2. The exact values to be used for the relationship between S_a and $\hat{n}_i$ are given in Table 7.8-2 of KTA 3201.2. The design fatigue curves shown in **Figure B 5-4** and **Figure B 5-5** were taken over from [7]. The exact values to be used for the relationship between S_a and $\hat{n}_i$ are given in Section B 5.6.

B 5.1 Design fatigue curves for ferritic steels



– – – $R_m \leq 550$ N/mm^2

—— $R_m = 790$ to 900 N/mm^2

$E = 2.07 \cdot 10^5$ N/mm^2

Values for tensile strengths between 550 N/mm^2 and 790 N/mm^2 may be subject to straight interpolation.

Where the calculated stress intensity range is based on strains with an elastic modulus $E_T \neq E$ the calculated stress intensity range shall be multiplied with the ratio E/E_T.

Figure B 5-1: Design fatigue curves for ferritic steels

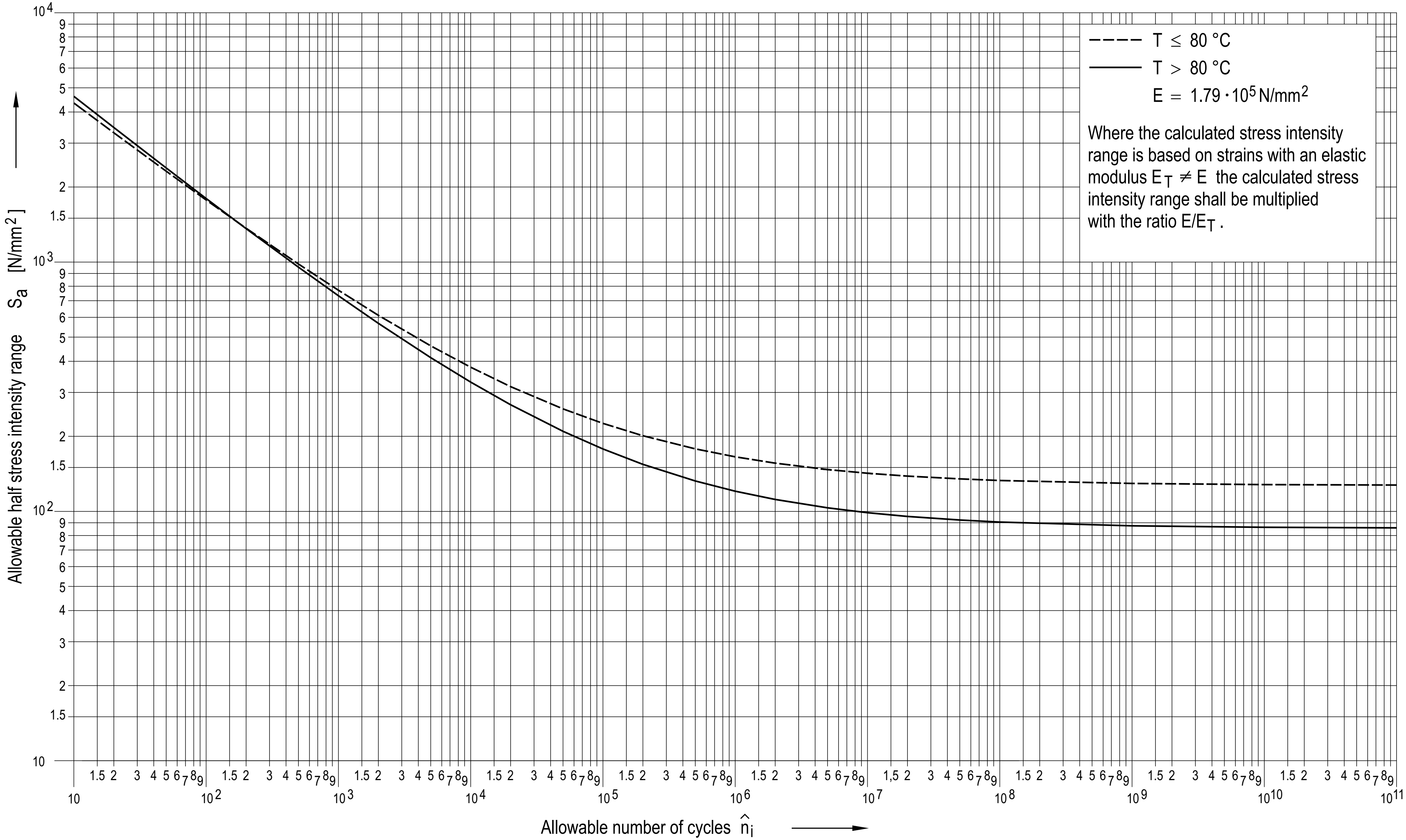


Figure B 5-2: Design fatigue curves for the austenitic steels 1.4550 and 1.4541

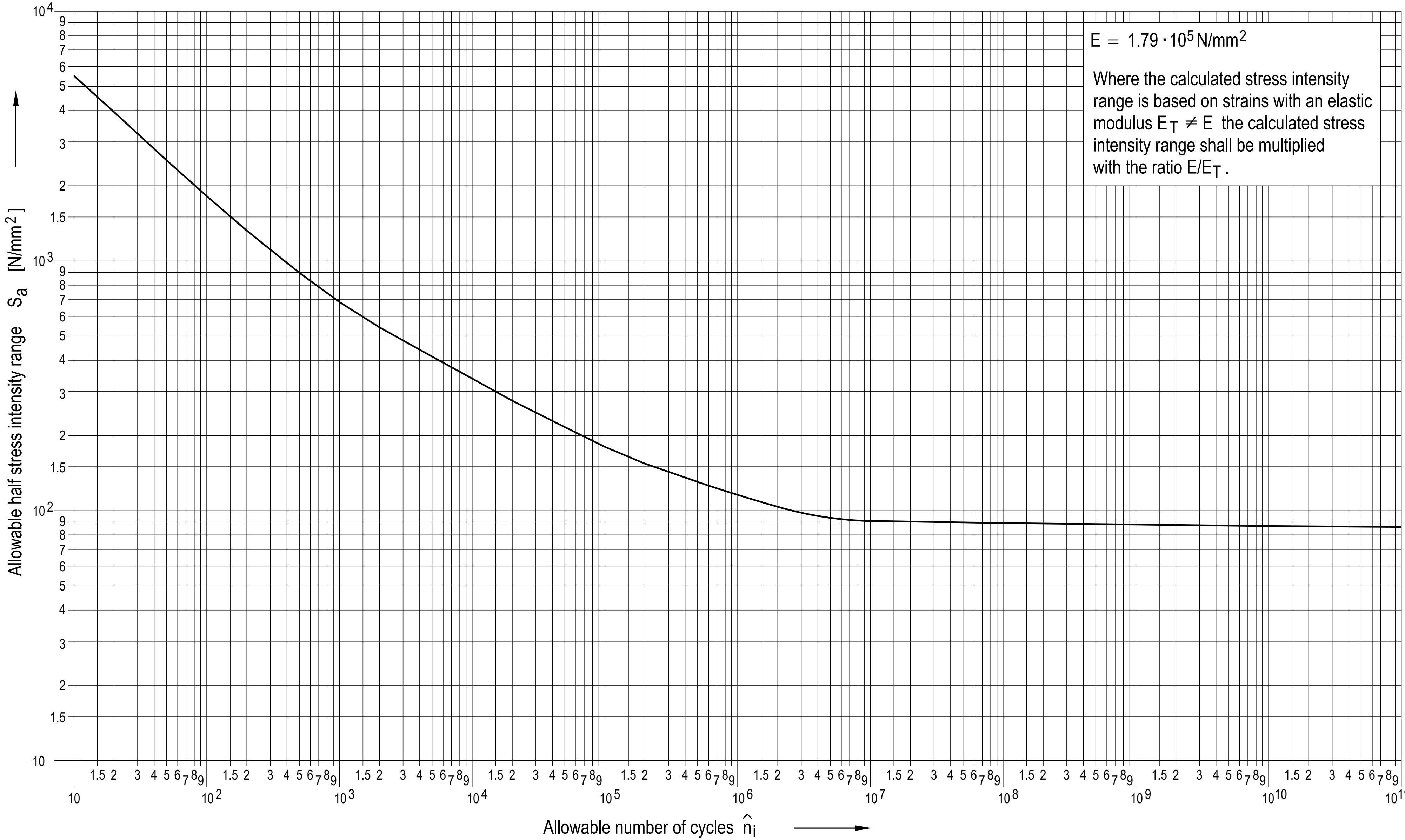


Figure B 5-3: Design fatigue curve for austenitic steels except the steels 1.4550 and 1.4541

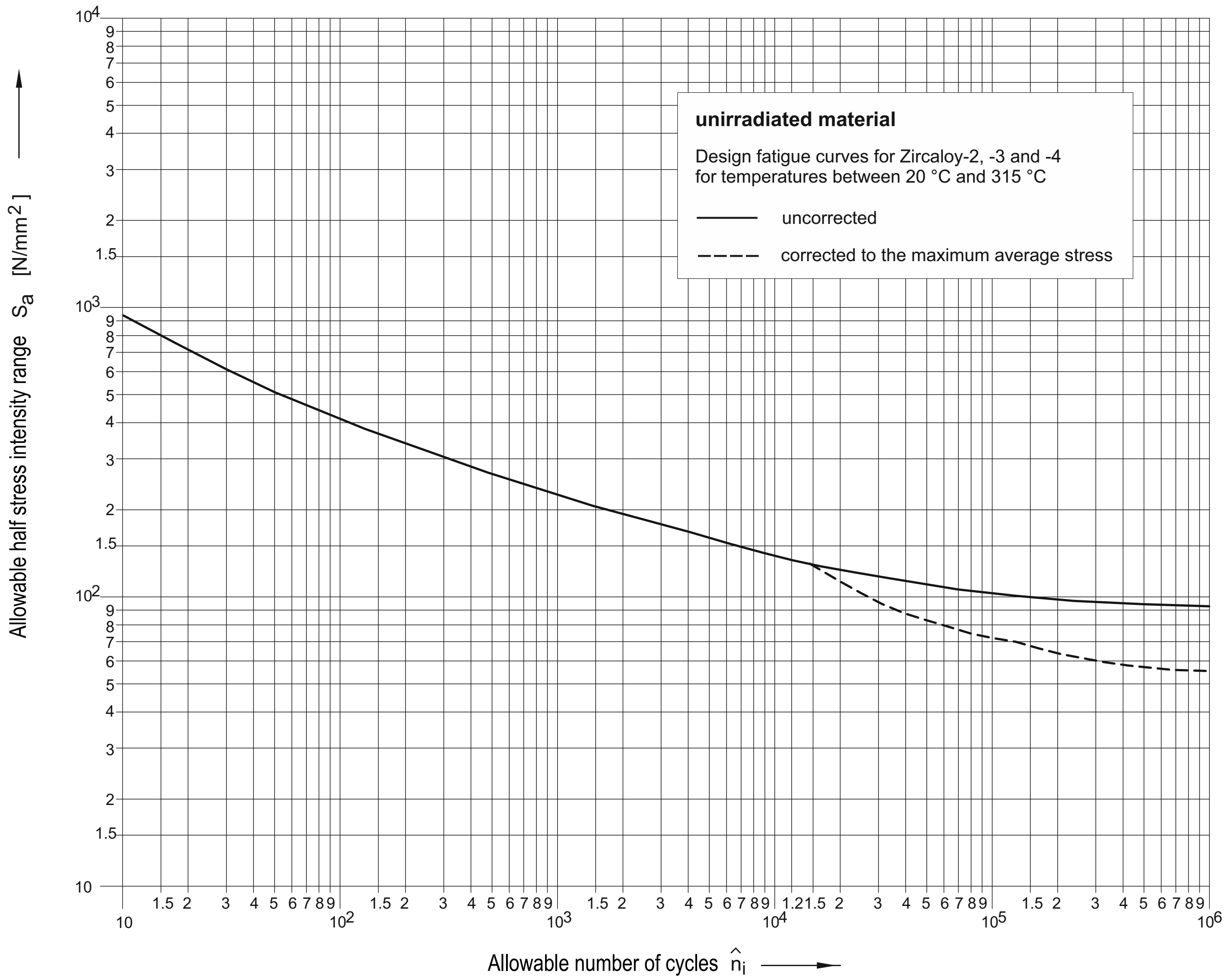


Figure B 5-4: Design fatigue curves for unirradiated Zircaloy-2, 3 and 4 for temperatures between 20 °C and 315 °C

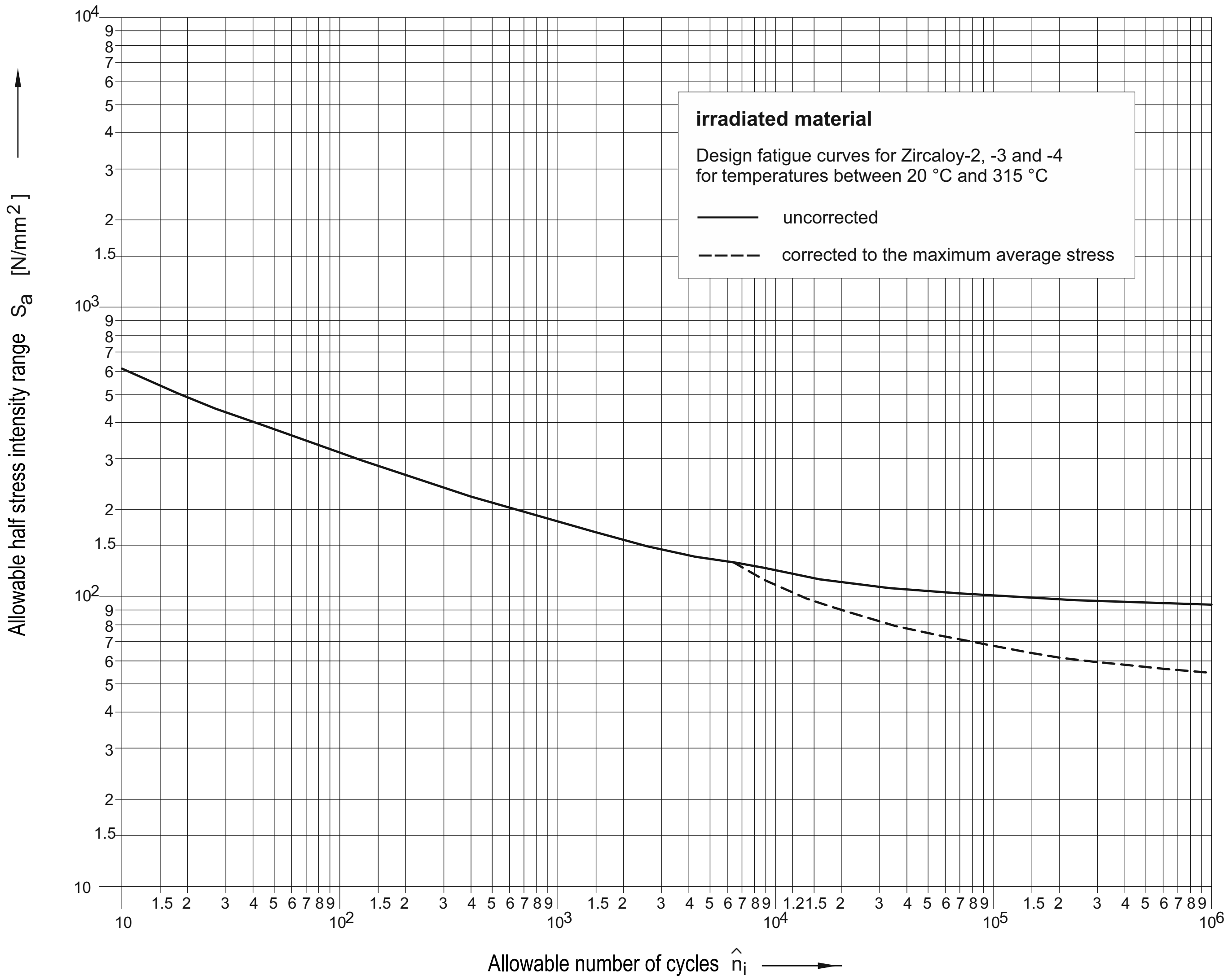


Figure B 5-5: Design fatigue curves for irradiated [1)] Zircaloy- 2, 3 and 4 for temperatures between 20 °C and 315 °C

1) The irradiation was performed under PWR conditions up to a fast neutron fluence of $5.5 \cdot 10^{21}$ n/cm².

B 5.6 Table of values and calculation instruction for the design fatigue curves for unirradiated and irradiated Zircaloy- 2, 3 and 4

		Allowable half stress intensity range S_a in N/mm² at allowable number of load cycles $\hat{n}$																	
		$1{\cdot}10^1$	$2{\cdot}10^1$	$5{\cdot}10^1$	$1{\cdot}10^2$	$2{\cdot}10^2$	$5{\cdot}10^2$	$1{\cdot}10^3$	$2{\cdot}10^3$	$5{\cdot}10^3$	$6.8{\cdot}10^3$	$1{\cdot}10^4$	$1.5{\cdot}10^4$	$2{\cdot}10^4$	$5{\cdot}10^4$	$1{\cdot}10^5$	$2{\cdot}10^5$	$5{\cdot}10^5$	$1{\cdot}10^6$
Figure 5-4	uncorrected	949.0	719.0	516.0	418.0	342.6	269.6	228.0	195.4	162.1	—	140.8	—	126.9	112.2	104.3	99.1	95.1	93.8
	corrected to the maximum average stress	949.0	719.0	516.0	418.0	342.6	269.6	228.0	195.4	162.1	—	140.8	130.0	113.4	84.2	72.4	64.5	58.1	55.3
Figure 5-5	uncorrected	613.0	490.0	379.5	316.0	263.5	213.0	181.5	157.3	135.6	—	122.2	—	112.5	104.6	100.7	98.1	95.6	94
	corrected to the maximum average stress	613.0	490.0	379.5	316.0	263.5	213.0	181.5	157.3	135.6	129.1	110.9	—	90.2	74.8	67.6	62.0	57.5	54.0

(1) The points of support at $\hat{n} = 6.8{\cdot}10^3$ and $\hat{n} = 1.5{\cdot}10^4$ were added for a more exact representation of the curve.

(2) Linear interpolation is permitted in case of a double-log representation (in the double-log diagram: straight lines between the points of support). Where for a given value S_a = S the pertinent number of load cycles $\hat{n}$ is to be determined, this shall be done by means of the adjacent points of support $S_j < S < S_i$ and $\hat{n}_j > \hat{n} > \hat{n}_i$ as follows:

$$\hat{n} / \hat{n}_i = \left(\hat{n}_j / \hat{n}_i\right)^{\log \frac{S_i}{S} / \log \frac{S_i}{S_j}}$$

Example: Given: Zircaloy 4, unirradiated (Figure 5-4), $S_a = S = 180$ N/mm²

from which follows: $S_i = 195.4$ N/mm², $S_j = 162.1$ N/mm², $\hat{n}_i = 2{\cdot}10^3$, $\hat{n}_j = 5{\cdot}10^3$

$$\hat{n}/2000 = (5000/2000)^{\log\frac{195.4}{180} / \log\frac{195.4}{162.1}}$$

$$\hat{n} = 2991$$

Table B-1: Table of allowable half stress intensity range values S_a for the design fatigue curves shown in **Figure B 5-4** (unirradiated Zircaloy-2, 3 and 4) and in **Figure B 5-5** (irradiated [1]) Zircaloy-2, 3 and 4)

[1]) The irradiation was performed under PWR conditions up to a fast neutron fluence of $5.5 \cdot 10^{21}$ n/cm².

Annex C

Regulations referred to in this Safety Standard

(The references exclusively refer to the version given in this annex. Quotations of regulations referred to therein refer to the version available when the individual reference below was established or issued.)

AtG		Act on the Peaceful Utilization of Atomic Energy and the Protection against its Hazards (Atomic Energy Act) of December 23, 1959 (BGbl. I, p. 814) as Amended and Promulgated on July 15, 1985 (BGBl. I, p. 1565), last Amendment by article 307 of the ordinance dated 31st August 2015 (BGBl. I 2015, no. 35, p. 1474)
StrlSchV		Ordinance on the Protection against Damage and Injuries Caused by Ionizing Radiation (Radiation Protection Ordinance) dated 20th July 2001 (BGBl. I 2001, No. 38, p. 1714), at last amended by article 5 of the ordinance dated 11th December 2014 (BGBl. I p. 2010)
SiAnf	(2015-03)	Safety Requirements for Nuclear Power Plants (SiAnf) as Promulgated on March 3rd 2015 (BAnz AT 30.03.2015 B2)
Interpretations	(2015-03)	Interpretations of the Safety Requirements for Nuclear Power Plants of November 22nd 2012, as Amended on March 3rd 2015 (BAnz. AT 30.03.2015 B3)
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 3103	(2015-11)	Shutdown Systems for Light Water Reactors
KTA 3107	(2014-11)	Nuclear Criticality Safety Requirements during Refuelling
KTA 3201.2	(2013-11)	Components of the Reactor Coolant Pressure Boundary of Light Water Reactors; Part 2: Design and Analysis
KTA 3204	(2008-11)	Reactor Pressure Vessel Internals
KTA 3602	(2003-11)	Storage and Handling of Fuel Assemblies and Associated Items in Nuclear Power Plants with Light Water Reactors
KTA 3905	(2012-11)	Load Attaching Points on Loads in Nuclear Power Plants
DIN EN 1993-1-1	(2012-12)	Eurocode 3: Design of steel structures - Part 1-1: General rules and rules for buildings; German version EN 1993-1-1:2005 + AC:2009
DIN EN 13906-1	(2013-11)	Cylindrical helical springs made from round wire and bar - Calculation and design - Part 1: Compression springs; German version EN 13906-1:2013
DIN ISO 16269-6	(2009-10)	Statistical interpretation of data - Part 6: Determination of statistical tolerance intervals (ISO 16269-6:2005); Text in German and English
VDI-2230 Sheet 1	(2003-02)	Systematic calculation of high duty bolted joints; joints with one cylindrical bolt
ASME Code VIII	(2010)	2010 ASME Boiler and Pressure Vessel Code, Section VIII, Division 2, Alternative Rules, Rules for the Construction of Pressure Vessels

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(Stress intensification factor K_e for the determination of plastic strains obtained from elastic design), Technische Überwachung 35 (1994), No. 6, pp. 268 - 278)

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(Machine Elements. Vol. 1: Design and analysis of connections, bearings, shafts; Springer Verlag, Berlin/Göttingen/Heidelberg 2005, 4th edition)

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[5] CSNI Technical Opinion papers No. 13, LOCA criteria Basis and Test Methodology, ISBN 978-92-64-99154-5, OECD 2011

[6] Fuel Fragmentation, Relocation, and Dispersal During the Loss of Coolant Accident (NUREG-2121), prepared by Patrick A.C. Reynaud; Office of Nuclear Regulatory Research

[7] W.J.O’Donnel and B.F.Langer: Fatigue design basis for Zircaloy Components, Nuclear Science and Engineering: 20, 1-12 (1964)