

Safety Standards

of the
Nuclear Safety Standards Commission (KTA)

KTA 3101.3 (2022-11)

**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)

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

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KTA SAFETY STANDARD

2022-11

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 the Federal Gazette (Bundesanzeiger) on July 25, 2023. Copies of the German versions of the KTA safety standards may be mail-ordered through the Wolters Kluwer Deutschland GmbH (info@wolterskluwer.de). Downloads of the English translations are available at the KTA website (<http://www.kta-gs.de>).

All questions regarding this English translation should please be directed to the KTA office:

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

(2) The 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 instrumentation.

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.

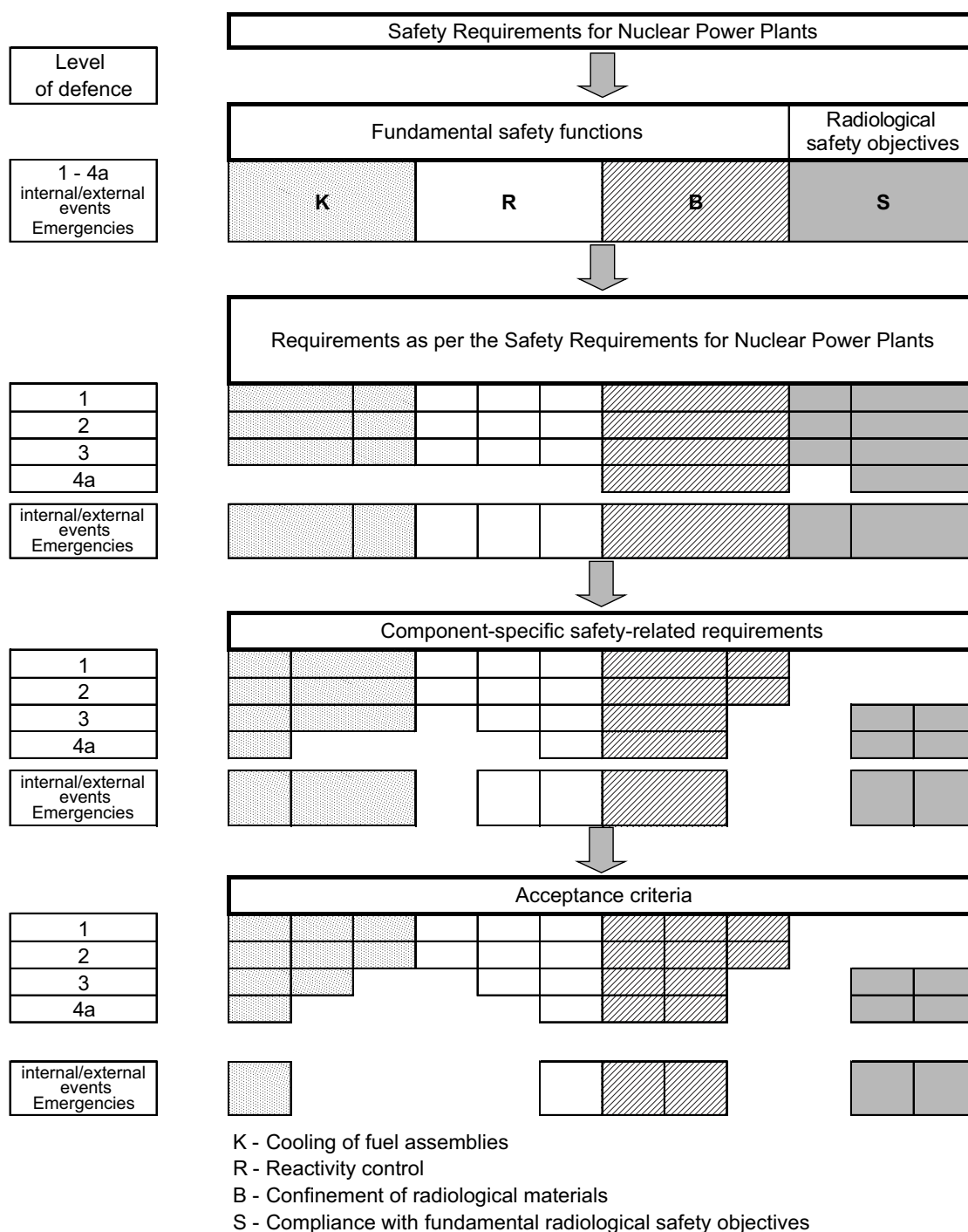


Figure 3.1-1 Schematic representation of the safety concept for the levels of defence covered by this safety standard

Note:

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.

Note:

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:

- 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.
- 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.
- 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:

- 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.
- 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:

- 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.
- 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.
- 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.
- Utilization of suitable materials to keep radiation exposure as low as reasonably acceptable (ALARA).
- 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:

- 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.
- 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.
- 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-para. (1) to (7).

4.1.2 Utilization of suitable materials

(1) The materials used shall be suited. The specification and the manufacturing process shall normally ensure, that the material properties under operating conditions – taking into account the expected variance for each relevant property – are reproducible. For this all essential influences on the materials, 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.

Notes:

(1) Reproducibility means, that the used materials behave in such a way, that - under defined operating conditions - the relevant properties remain within safety-wise acceptable bandwidths.

(2) 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 unstable 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.

Notes:

(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).

Note:

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.

Note to (3) and (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. UO₂ 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.