Interpretations of the “Safety Requirements for Nuclear Power Plants”

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Note:

This is a translation of the document entitled: "Interpretationen zu den Sicherheitsanforderungen an Kernkraftwerke".

In case of discrepancies between the English translation and the German original, the original shall prevail.

Interpretations of the "Safety Requirements for NPPs of 22/11/2012“

The "Safety Requirements for Nuclear Power Plants" contain fundamental and overriding safety-related requirements within the framework of the non-mandatory guidance instruments which serve for putting in concrete terms the precaution in line with the state of the art in science and technology against damage caused by the construction and operation of the plant as stipulated in §7 para 2 no. 3 of the Atomic Energy Act (AtG) as well as the requirements pursuant to §7d AtG.

The "Safety Requirements for Nuclear Power Plants" leave room for interpretation, which may leads to difficulties regarding the design and technical implementation in practice. The explanatory and specifying interpretations are to fill these gaps, thus allowing and achieving the uniform execution of the safety requirements for nuclear power plants.

The "Interpretations of the Safety Requirements" presented herewith comprise the following:

- Requirements for the design and operation of the reactor core
  (Interpretation I-1)
- Requirements for the implementation of the reactor coolant pressure boundary, the external systems as well as the containment
  (Interpretation I-2)
- Requirements for instrumentation and control and for accident overview measuring systems
  (Interpretation I-3)
- Requirements for the electrical energy supply
  (Interpretation I-4)
- Requirements for structures, systems and components
  (Interpretation I-5)
- Requirements for the handling and storage of the fuel assemblies
  (Interpretation I-6)
- Requirements for accident management
  (Interpretation I-7)
- Requirements for radiation protection
  (Interpretation I-8)

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Interpreted number of the "Safety requirements for Nuclear Power Plants"	Associated interpretation with section or number	
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Interpreted number of the "Safety requirements for Nuclear Power Plants"	Associated interpretation with section or number	
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Interpreted number of the "Safety requirements for Nuclear Power Plants"	Associated interpretation with section or number		
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Interpretation I-1: Requirements for the design and operation of the reactor core

Contents

1 Scope of application

This text contains interpretations regarding the requirements for the design, monitoring and operation of the reactor core, amongst other things in particular the reactor-physical design of the control, limitation and shutdown systems.

2 Requirements for the design, monitoring and operation of the reactor core

Interpretation regarding Number 3.2 (2) of the "Safety Requirements for Nuclear Power Plants"

2 (1) The reactor core shall be designed, monitored and operated as follows:

a) The power and the power density as well as the safety variables that are essential for reactivity, power or power density as well as for fuel cooling with regard to the adherence to the safety-related acceptance targets and acceptance criteria on levels of defence 1 to 4a shall be monitored to the extent necessary.

   The temporal and spatial distribution of the monitoring as well as the sensitivity and design of the monitoring equipment must ensure the respective functions of the control, limitation and safety installations.

b) In normal operation, the occurrence of departure from nucleate boiling shall be avoided by maintaining a sufficient safety margin.

c) In normal operation, the power and power density shall be kept stable within permissible limits, also with a view to the effects of xenon redistribution.

d) In normal operation, changes in the reactivity, power or power density must take place in a manner controlled by the control installations with consideration of the reactor-physical feedback properties.

e) On levels of defence 1 to 4a, internal or natural hazards as well as human-induced external hazards must not cause any deformation of the fuel rods, the fuel assembly structure or the control assemblies that would call the possibility of mechanical shutdown (in the case of a large leak inside the containment of a PWR, the possibility of permanent shutdown) into question.

2 (2) Within the framework of the mechanical design of the reactor core, design limits shall be specified for the conditions of normal specified operation (levels of defence 1 and 2) with consideration of the uncertainties of the experimental data base. These design limits shall be specified such that if they are adhered to, no defects need be postulated, neither on the fuel rods, the fuel assembly structure or the control assemblies nor on the associated structural parts.

Instead of design limits, defect probabilities may also be used if these have been backed up experimentally.

2 (3) Within the framework of the mechanical design, it shall be shown when demonstrating fuel rod integrity in normal specified operation (levels of defence 1 und 2)

a) that when furnishing proof conservatively pursuant to Number 3.4 of Annex 5 of the "Safety Requirements for Nuclear Power Plants", no fuel rod will exceed any design limit during its residence time, or

b) that when quantifying the result uncertainties pursuant to Number 3.3 of Annex 5 of the der "Safety Requirements for Nuclear Power Plants" by means of statistical methods, no more than one fuel rod in the core will be expected to be damaged per cycle, with the requirements of Number 3.3 (4) of Annex 5 of the "Safety Requirements for Nuclear Power Plants" applying.

3 Requirements for the reactor-physical design of the reactor core

Interpretation regarding Numbers 3.2 (3) and 3.2 (4) of the "Safety Requirements for Nuclear Power Plants"

3 (1) The reactor core shall be designed such that due to inherent reactor-physical feedback properties

a) an increase in the fuel temperature in the reactor core will have a negative reactivity effect;

b) an increase in the void content in the reactor core will have a negative reactivity effect;

c) an increase in the coolant temperature or a decrease in the coolant density in the reactor core (without or with negligible void formation) will have a negative reactivity effect,

- in the case of a PWR at the latest upon reaching a stationary operating condition with xenon equilibrium at the start of the cycle and
- in the case of a BWR at the latest upon reaching operating temperature.

3 (2) A positive reactivity effect upon coolant temperature increase or coolant density decrease (without or with negligible void formation) prior to reaching the conditions mentioned in Number 3 (1) letter c is permissible if it has been shown that

- in normal operation, stable control of reactor power is possible under these conditions and
- the corresponding safety-related assessment targets and assessment criteria are fulfilled, taking into account the resulting positive reactivity effects in the events considered on levels of defence 2 to 4a.

4 Requirements for the control, limitation and shutdown installations

Interpretation regarding Numbers 3.2 (5), 3.2 (6) and 3.2 (7) of the "Safe-ty Requirements for Nuclear Power Plants"

4 (1) In the design of the reactor power control, limitation and shutdown installations, the mechanical, thermal and chemical impacts and those caused by radiation shall be considered that may occur

a) during normal specified operation of the plant as well as

b) in control, limitation and shutdown installations whose functioning is necessary for controlling events on levels of defence 2 to 4a, upon internal or natural hazards as well as human-induced external hazards, also under the respective event conditions

and which are significant with respect to ensuring the effectiveness and reliability of the installations.

4 (2) The reactor scram system pursuant to Numbers 3.2 (5) and 3.2 (6) of the "Safety Requirements for Nuclear Power Plants"

a) must be automatically triggered by actuations that are formed of different process variables;

b) must, even if it shares components with the control and limitation installations, not be impaired in its own specified function by the functions of the control and limitation installations (neither by a function caused as a result of a malfunction in these installations).

4 (3) When demonstrating the adequate effectiveness of the installations for injecting soluble neutron absorbers into the coolant pursuant to Numbers 3.2 (5) and 3.2 (7) of the "Safety Requirements for Nuclear Power Plants", it shall be shown for the BWR that a shutdown

reactivity value of 5% is reached under normal operating conditions in operating phases A to C.

4 (4) Regarding the prevention of the ejection of control assemblies (see events D3-16 and S3-09 in Annex 2 of the "Safety Requirements for Nuclear Power Plants"), these shall not only have a correspondingly safe design and be subjected to thorough in-process inspections, but independent installations for limiting the ejection length shall also be provided unless it has been shown that the safety-related assessment targets and assessment criteria are fulfilled for the case of the complete ejection of the control assembly with the highest reactivity values.

Regarding the prevention of the dropping out of control assemblies in a BWR (see event S3-10 in Annex 2 of the "Safety Requirements for Nuclear Power Plants"), these shall not only have a correspondingly safe design, be subjected to thorough in-process inspections and dispose of electro-technical interlocks, but independent installations for limiting the drop-out length shall also be provided unless it has been shown that the safety-related assessment targets and assessment criteria are fulfilled for the case of the complete ejection of the control assembly with the highest reactivity values.

Interpretation I-2: Requirements for the design of the reactor coolant pressure boundary, the external systems as well as the containment

Content

1 Scope of application

This text contains interpretations regarding safety-related requirements for the design, manufacture and operation of the reactor coolant pressure boundary, the pressure-retaining walls of components of external systems, and the containment. This text also contains requirements for components with small nominal bore and guard lines for piping as well as additional requirements for components with restricted break postulates.

2 Reactor coolant pressure boundary and external systems

Interpretation regarding Number 3.4 of the "Safety Requirements for Nuclear Power Plants"

2.1 Scope of application

2.1 (1) The following requirements shall be applied to the pressure envelopes of components made of metallic materials and pertaining to the reactor coolant pressure boundary and the external systems of light water reactors.

Note: For components (vessels, heat exchangers, piping, valves, pumps) to which specific requirements apply due to the fact that they retain radioactive materials and which are not allocated to the scope of application referred to above, the requirements of the "Safety Requirements for Nuclear Power Plants", I-5, Section 7, shall apply. There, requirements regarding the function of the components addressed here can also be found.

Interpretation regarding Number 3.1 (4) of the "Safety Requirements for Nuclear Power Plants"

2.1 (2) Regarding the design principles, the same requirements apply to the components of the reactor coolant pressure boundary and the external systems. The higher safety significance of the reactor coolant pressure boundary as part of the barrier concept compared with the external systems shall be taken into account by special requirements for the choice of materials, verification depth and quality assurance as well as by wider scopes of in-service inspections and operational monitoring.

2.1 (3) If components made of non-metallic materials are used, requirements shall be specified that ensure equal reliability.

Interpretation regarding Number 3.4 (3) of the "Safety Requirements for Nuclear Power Plants"

2.1 (4) The following requirements do not apply to components smaller than or equal to a nominal bore of 50. To such components of smaller nominal bore, the requirements pursuant to Section 4 shall be applied.

2.2 Principles of basic safety in design and manufacture

Interpretation regarding Number 3.4 (3) of the "Safety Requirements for Nuclear Power Plants"

2.2 (1) The basic safety of components, which excludes their catastrophic failure due to manufacturing faults, shall be ensured by compliance with the requirements below, taking the service fluid into account:

- use of high-quality materials, especially with regard to toughness and corrosion-resistance,
- conservative stress limitation,
- prevention of peak stresses through optimised design, and
- guarantee of the application of optimised manufacturing and inspection technologies.

This includes the knowledge about and the assessment of any possibly existing fault conditions.

Interpretation regarding Numbers 3.1 (12) and 3.4 (3) of the "Safety Requirements for Nuclear Power Plants"

2.2 (2) Furthermore, all components shall be designed such that the requirements for their construction in a manner that is favourable to stress, suitable with regard to the materials used, the manufacturing process and the prospective functions are fulfilled and that the non-destructive tests during manufacture and at the site of installation as well as the non-destructive in-service tests can be carried out to the extent that is necessary. This applies in particular to weld seams and the carrier material of cladded material areas.

2.3 Design

Interpretation regarding Numbers 3.1 (1), 3.1 (2) and 3.4 of the "Safety Requirements for Nuclear Power Plants"

2.3.1 Principles and toughness

2.3.1 (1) Proof of integrity as part of the design shall be furnished such that the necessary safety margins with respect to the occurrence of postulated failure modes are demonstrated over the entire intended period of operation for all impacts resulting from normal specified operation (levels of defence 1 and 2), events on levels of defence 3 and 4a as well as postulated site-specific external natural hazards or human-induced external hazard. The loads caused by the mechanical and thermal impacts in the components shall be limited such that a safety margin with respect to the occurrence of postulated failure modes is provided for the respective levels of defence pursuant to the "Safety Requirements for Nuclear Power Plants" Number 3.1 (2) a). Any possible ageing-related degradation mechanisms and changes in the material properties through impacts such as temperature and irradiation, which may occur during operation, shall be taken into account. If there are uncertainties regarding the state of knowledge about certain degradation mechanisms, these shall be considered by corresponding safety margins or by conservative verification.

Note: See Appendix 1 of I-2 " Principles of the strength-related design and allocation of load stages to the different levels of defence".

2.3.1 (2) In the design of the components, certain load cases shall be taken as a basis, starting from the impacts. The load cases shall be derived especially from the specified operation of the plant, including inspections, from operating experience and from the events postulated pursuant to the "Safety Requirements for Nuclear Power Plants", Annex 2 and Annex 3 and have to cover the resulting impacts. The load cases and their combinations shall be specified and fully described according to their characteristic and frequency.

Load case combinations shall be postulated if the events or operating phases to be combined may be in a causal relationship or if their simultaneous occurrence has to be assumed on the basis of probability considerations. The hazards resulting from these load cases shall be described component-related, also taking into account the systems engineering of adjacent systems and the time history as well as the load transfer capacity of the support structure. Impacts from component parts shall be considered in the integrity verification (e.g. with regard to dead weight, stability, mechanical impacts, thermal hydraulic conditions) if these may influence the integrity of the pressure envelopes.

Interpretation regarding Annex 5, Number 4 of the "Safety Requirements for Nuclear Power Plants"

2.3.1 (3) The proof of integrity shall be furnished experimentally or by calculations or in combination of these methods, with the applicability of the boundary conditions of the calculation method and the experiment to the boundary conditions of the component or system to be verified having to be demonstrated. Compliance with corresponding assessment criteria shall be demonstrated by validated methods. Here, a safety margin with regard to the failure or the onset of a condition that is to be avoided shall be shown.

Note: For requirements for experimental proof and the validation of methods, see also "Safety Requirements for Nuclear Power Plants", Annex 5.

2.3.1 (4) Regarding events on levels of defence 3 and 4a as well as postulated external site-specific natural hazards or human-induced external hazard for whose control the function of parts of the reactor coolant pressure boundary or the external systems is necessary, the service limits for the active components utilised here shall be specified such that the operability of these components (e.g. pumps, valves) will remain ensured.

2.3.1 (5) If components of the external systems adjacent to the reactor coolant pressure boundary are made operational to control events of levels of defence 3 and 4a or postulated external site-specific natural hazards or human-induced external hazard, then the loads arising in the pressure envelopes of these systems shall be limited such that the necessary reliability of the systems is ensured for the specified operating period and frequency of use.

Interpretation regarding Number 3.1 (4) of the "Safety Requirements for Nuclear Power Plants"

2.3.1 (6) Regarding the components of the external systems, the choice of materials, manufacturing methods and safety demonstration methods shall be harmonised, taking into account the different functional requirements, in a way that equal reliability of the components is achieved. As for the variety of the components, measures shall be determined that ensure reliable quality assurance.

This shall be done for the components by classifying them into inspection and material groups dependent on design data and dimensions with consideration of the materials and stress limits. Here, it may be that components of one system, or possibly parts of a component, will be allocated to different inspection and material groups.

As for the inspection groups for component parts and components of the external systems, specifications of the verification depth regarding the scope of the stress and fatigue analyses as well as the extent of testing (destructive and non-destructive) shall be made dependent on the maximum stress ratio and the choice of material.

2.3.2 Choice of material

Interpretation regarding Number 3.1 (2) of the "Safety Requirements for Nuclear Power Plants"

2.3.2 (1) By the choice of material and by proper shaping, welding and heat treatment it shall be ensured for the components that a sufficiently tough and rigid material condition is achieved and maintained over the intended operating period of the plant in such a way that the loads arising during normal specified operation (levels of defence 1 und 2) and in events on levels of defence 3 and 4a as well as during postulated external site-specific natural hazards or human-induced external hazard can safely be shed.

2.3.2 (2) To prove the specified toughness and rigidity, it shall be shown for all materials by certificates that they have been manufactured according to the specifications. Ferritic steels have to have a sufficiently high level of toughness in the upper shelf range.

As for components of the reactor coolant pressure boundary, the lowest load temperature in loads resulting from stationary operating states of levels of defence 1 and 2 must be high enough above the nil-ductility temperature so that that a specific defined minimum toughness is guaranteed. This applies to the basic material, the weld material, and the heat-affected zone.

Components of the external systems must have a material toughness that complies with the design concept as well as a distinct strain-hardening behaviour.

Note: For ferritic materials, the latter usually demands the use of low- or medium-tough materials with heat-treatment states that are normal in nuclear engineering. Austenitic materials fulfil the latter-mentioned criteria without restrictions.

2.3.2 (3) The materials used must be suitable for welding and, in combination with the design chosen and the processing methods applied, must show sufficient resistance against corrosion and other ageing effects under the operating conditions. The water qualities in normal specified operation (levels of defence 1 and 2) necessary for corrosion resistance shall be specified.

2.3.2 (4) Observing the other requirements for the materials, the choice of the materials in contact with reactor coolant shall be made such that an activation of the materials and their corrosion products will be as slight as possible. In particular, component parts with sealing or sliding function shall show a sufficiently high chemical, mechanical and physical resistance under the conditions of normal specified operation (levels of defence 1 and 2) in order to keep any radiological effects as low as possible and prevent component damage through corrosion.

2.3.3 Construction and layout

Interpretation regarding Numbers 3.1 (2), 3.1 (12) and 3.4 (3) of the "Safety Requirements for Nuclear Power Plants"

2.3.3 (1) Sufficient possibilities for inspections and in-service inspections shall be provided for all pressure-retaining parts of the components.

2.3.3 (2) Sealing connections shall be realised such that the necessary leaktightness is reliably achieved. Their realisation shall be qualified or their suitability shall be demonstrated on the basis of technical experience. They shall be monitored in a suitable manner so that any leakiness that may possibly occur can be detected in such good time that impermissible consequences are avoided.

2.3.3 (3) In outgoing lines, the isolating valve shall be installed as close to the branch-off point as possible.

2.3.3 (4) Assembly parts of isolating valves shall be realised such that they have the load-bearing capacity necessary for ensuring the sealing function.

2.3.3 (5) When routing piping and arranging valves, it shall be ensured that accumulations of condensate in steam-retaining plant components are avoided through draining.

2.3.3 (6) Piping of the external systems connecting to the isolating valves of the reactor coolant pressure boundary and not penetrating the containment must have a further isolating valve on the inside of the containment unless pressure relief into a closed system (e.g. pressure suppression pool, pressure relief tank) is provided for safety-related reasons.

2.3.3 (7) Components of external systems which, by assumption of a single failure in the isolating valve of the reactor coolant pressure boundary, may be subjected to higher pressure or higher temperatures shall be realised such that their integrity will be ensured in these load cases, too.

2.3.3 (8) It shall be ensured by suitable installations that a transgression above the load limits underlying the integrity verification

a) of the main-steam line by steam generator overfeed,

b) of the components of the external systems due to water hammer,

c) of the components of the external systems due to the reaction of radiolysis gases,

d) of the components of the external systems of low-pressure systems connecting to high-pressure systems due to leaks in isolating valves of the higher-pressure system, and

e) of the components of the external systems due to heat influx into enclosed media

will be reliably excluded for the operating conditions and events on levels of defence 1 to 3, and from internal hazards or postulated external site-specific natural hazards. The effectiveness of the installations shall be monitored.

2.3.3 (9) Pressure relief pipes and venturi nozzles in boiling water reactors shall be designed with regard to the outflowing amounts of steam in such a way that for all events on levels of defence 2 and 3 as well as from internal hazards as well as from postulated external site-specific natural hazards, a reliable outflow of the medium (steam, steam/water mixture) into the pressure suppression pool in compliance with the design values is ensured.

It shall be ensured that no leaks in the pressure relief pipes will occur in the gas phase of the pressure suppression pool above the water pool, or that leakages that cannot be excluded will be safely drained away (e.g. by the installation of an outer guard line).

An accumulation of radiolysis gases in the pressure suppression pipes caused by the condensation of possible steam leakages shall be limited by suitable measures (e.g. purging with nitrogen) in such a way that no reactive mixtures can form.

Note: For precautionary measures against plant-internal explosions, see also "Safety Requirements for Nuclear Power Plants", Annex 3, Number 3.2.9.

2.4 Manufacture

Interpretation regarding Numbers 3.1 (1), 3.1 (2), 3.4 (1) and 3.4 (3) of the "Safety Requirements for Nuclear Power Plants"

2.4.1 Principles

2.4.1 (1) The quality characteristics to be adhered to for ensuring integrity shall be determined and taken into account in the planning of the manufacturing process.

2.4.1 (2) The manufacturing process shall be monitored and documented such that any deviations from the quality characteristics determined will be recognised and that it is possible to trace back the deviations with a view to identifying their cause. Any measures taken additionally to comply with the quality characteristics shall be documented.

2.4.2 Accompanying destructive testing

2.4.2 (1) It shall be shown by testing of product forms that the characteristics of the chemical composition, the toughness, the rigidity, the metallic microstructure and the corrosion resistance specified across the wall thickness exist. This shall cover:

a) representatively, the different deformation directions in several sampling locations as well as

b) all formation and heat treatments taking place during the manufacturing process.

2.4.2 (2) For the qualification of the welding methods of for verifying the quality characteristics of component welds, procedure qualification and production tests shall be carried out. Combining the performance of production tests with procedure qualification tests is permissible.

2.4.3 Accompanying non-destructive testing

2.4.3 (1) The volume and surface of all product forms and weld connections including butterings intended for the components shall be tested by non-destructive methods with sufficient detectability of discontinuities.

The testing methods and parameters shall be chosen such that all faults that are clearly below the extent of safety-relevant faults can be detected.

2.4.3 (2) The tests to be carried out to assess the relevant quality condition of the product forms and components shall be performed after the last heat treatment.

2.4.3 (3) Completing the manufacturing process, all components shall be subjected to a pressure test at a defined test pressure above the design pressure (preservice hydrostatic test). Following the pressure test, non-destructive tests shall be carried out to a representative scope.

2.4.3 (4) Within the framework of specified leaktightness requirements, leak tests shall be performed (e.g. overall system, steam generator tubes).

2.4.3 (5) The results of the tests shall be documented such that they can be used for the comparison with in-service inspections.

2.5 Operation

Interpretation regarding Numbers 3.1 (1) and 3.1 (2) of the "Safety Requirements for Nuclear Power Plants"

2.5.1 Principles

2.5.1 (1) A monitoring and inspection concept shall be established by which

- compliance with the boundary conditions of and prerequisites for the design can be verified,
- the modifications of the mode of operation and the intended service life of the plant can be documented
- the feedback from operating experience and its utilisation for ageing management can be ensured.

2.5.1 (2) The boundary conditions used as a basis for the design of the components with respect to the spatial layout, anchoring, function of support, valves, pumps and internals shall be documented (e.g. in the case of high-temperature systems free paths, dislocations, deflections, play). Upon commissioning and, if necessary, following interventions (e.g. maintenance measures), compliance with these boundary conditions shall be verified. Impermissible deviations from these boundary conditions shall be avoided or registered in such good time that there will be no effects on the integrity of the pressure envelopes.

2.5.1 (3) Operating parameters that are relevant for the integrity of the components shall be monitored (e.g. mechanical and thermal impacts, water quality) and their plausibility assessed, taking the assumed associated systems state into account.

2.5.1 (4) The operating conditions in the operating phases of low-power and shutdown operation (operating phases B to F) and during function tests shall be specified with regard to the impacts influencing the integrity of the components. Compliance with these requirements shall be ensured by operating rules (e.g. temperature, water chemistry). Deviations from these rules shall be avoided or registered in such good time that there will be no effects on the integrity of the pressure envelopes.

2.5.1 (5) The inspection concept must ensure a representative selection of test locations for in-service inspections. Apart from a number of test locations chosen at random, this shall include to an adequate extent in particular components or component areas for which leading loads can be expected on the basis of analyses or operating experience as well as areas in which unusual features regarding manufacture have been observed.

2.5.1 (6) The general condition of the systems and components that are accessible during operation shall be monitored by regular walk-downs. The results shall be documented.

2.5.1 (7) Accumulations of non-condensable gases

a) at high points of cooling loops and

b) in components of low or no flow

shall be registered with regard to possible impacts on the pressure-retaining wall and possible functional disturbances of the system. They shall be assessed with regard to their safety-related effects.

2.5.1 (8) If findings are identified in inspections, the procedure pursuant to Section 7 shall be followed.

2.5.1 (9) An ageing management system shall be installed for the systematic detection, monitoring or prevention of ageing effects on the integrity of the components.

2.5.1 (10) The technical installations and aids as well as the handling procedures for the work to be carried out on pressurised components (e.g. on bolted joints in tests and during cleaning) shall be determined such that adverse effects on the components are avoided or recognised in such good time that there will be no impermissible effects on the integrity of the pressure envelopes.

2.5.2 In-service leak and pressure tests

2.5.2 (1) After each reclosure of a pressurised system, in integral leak test shall be performed at a defined reference state.

2.5.2 (2) In-service leak and pressure tests shall allow obtaining safety-related information comparable to the information obtained from safety-related pressure testing during manufacture.

2.5.2 (3) Following the in-service pressure test, non-destructive tests (e.g. ultrasonic testing) shall be performed in representative locations of the reactor pressure vessel and other components.

2.5.3 Non-destructive in-service tests

2.5.3 (1) Regarding possible degradation mechanisms, the non-destructive in-service tests shall be carried out in a representative manner with qualified methods, with all kinds of welded joints and base material areas having to be included. The choice of test procedures and methods shall be made with consideration of the technological progress and the testing target. The defined test intervals shall be guided by the general technical experience and shall take operating experience into account.

2.5.3 (2) Test procedures and methods shall be chosen such that service-induced faults (e.g. due to fatigue, corrosion) with their possible orientations can be registered and documented. Any indications that were documented during manufacturing and have been left untreated shall be registered and, if necessary, followed up.

2.5.3 (3) For each test procedure, evaluation limits for the identification of indications shall be specified.

2.5.3 (4) For a better reproducibility of the test parameters and the boundary conditions of the test and for a better comparability of the test results as well as for the minimisation of the radiation exposure of the personnel, the in-service inspections shall, as far as possible and pertinent, be mechanised.

3 Additional requirements for components and systems for the limitation of break assumptions

Interpretation regarding Numbers 3.4 (1) and 3.4 (4) of the "Safety Requirements for Nuclear Power Plants"

3.1 Principles

3.1 (1) If limited leak and break postulates are applied within the framework of the design concept to piping systems and components of the reactor coolant pressure boundary or the external systems according to the "Safety Requirements for Nuclear Power Plants", Number 3.4 (4), then these piping systems and components shall be protected by structural installations or uncoupling against postulated site-specific natural hazards or human-induced external hazards, also considering the vibrations induced by these events, in such a way that their integrity will remain intact.

3.1 (2) Additionally to the requirements pursuant to Section 2, an analysis shall be performed that includes all possible impacts from the operating conditions and events on levels of defence 1 to 4a and the postulated site-specific external natural hazards with consideration of the response behaviour of the system. The conservative load assumptions derived shall be used to demonstrate from a fracture-mechanical point of view and postulating faults that the latter cannot not lead to a leak in or break of the component that would call the leak and break postulates used into question. Here, the faults shall be chosen such that under the loads arising, they will behave less favourably with regard to the integrity of the component than faults that may possibly exist in the component or that can clearly be identified.

Note: The leak cross-section and breaks to be assumed for the components of the reactor coolant pressure boundary as well as external systems are described in Annex 2, Appendix 2 of the "Safety Requirements for Nuclear Power Plants".

3.1 (3) In this context, the extent of the postulated faults shall be defined such that can be safely detected with the specified test procedures. The postulated faults shall be assumed to be in the location on the surface and in the orientation for which the highest crack growth potential is identified.

Note: Specific assumptions and procedures for different component groups are indicated in the following Numbers of this Section 3.

3.1 (4) The components affected have to fulfil the requirements pursuant to Section 2. Here, the prerequisites for the use of limited leak and break postulates shall be ensured, i.e. it shall be ensured for operation by the design and manufacture that

- degradation mechanisms such as corrosion and erosion processes, fatigue due to oscillations or dynamic loads as well as service-induced material changes are limited and detectable in such a way that they cannot lead to an impermissible quality impairment and

- the permissible stresses will not be exceeded by excess pressures, additional thermal and mechanical loads nor by malfunctions of the supports, either.

3.2 Fracture resistance of the reactor pressure vessel

3.2 (1) Regarding the reactor pressure vessel, whose integrity is necessary for ensuring all fundamental safety functions pursuant to the "Safety Requirements for Nuclear Power Plants", all changes of the material properties expected over the intended operating lifetime shall be conservatively considered for the demonstration of break preclusion.

3.2 (2) Regarding the areas of the pressure vessel wall that are exposed to neutron radiation, fluence levels shall be limited by design specifications and requirements for the chemical composition of base materials and weld material shall be fulfilled such that the change in the toughness and rigidity characteristics caused by irradiation will remain within permissible limits.

3.2 (3) For the characterisation of the material properties changed due to irradiation, a graded monitoring programme with suspended accelerated irradiation capsules (base materials, weld materials) shall be carried out dependent on the accumulated neutron fluence.

3.2 (4) Regarding postulated surface faults and, if necessary, regarding any manufacturing-induced flaw sizes identified in the volume, it shall be shown that when using fracture-mechanical demonstration methods for all stresses from the relevant loads,

- in the case of operating conditions on levels of defence 1 and 2 there will be no crack initiation and
- in the case of events on levels of defence 3 and 4a as well as in postulated site-specific external natural hazards there will be no unstable crack growth in through-wall direction.

Regarding events on levels of defence 3 and 4a as well as in postulated site-specific external natural hazards, merely a limited, stable crack growth that is of no significance with regard to the wall thickness is permissible in the upper-shelf toughness only.

Furthermore, it shall be shown by calculation that variations in stress on the flaw sizes considered will not cause any crack growth that is significant regarding the wall thickness.

3.3 Break preclusion for piping

If the concept of break preclusion is applied to piping systems pursuant to Section 3.1, it shall be demonstrated that

– postulated faults in the pressure-retaining wall will not show any significant crack growth through the wall under the operating conditions and events assumed on levels of defence 1 and 2, and

– furthermore a postulated through-thickness crack of the pressure-retaining wall will remain stable under loads resulting from events on levels of defence 3 and 4a as well as in the case of postulated site-specific external natural hazards, i.e. that it will show a leak-before-break behaviour. It shall be demonstrated that a sufficient safety margin to critical crack sizes will be maintained, taking into account the loads resulting from the leak case and the grace periods from the detection of the leak until the moment when the system affected is taken out of operation. The size of the postulated cracks shall be chosen such that detection in good time of the leaks caused by the cracks during operation is ensured. Leak detection systems shall be executed to be highly reliable. This shall be ensured in particular by the use of diverse measuring methods.

3.4 Demonstration of fracture resistance of vessels

If the concept of break preclusion is applied to vessels pursuant to Section 3.1, it shall be demonstrated that there will be no unstable crack growth through the wall under operating conditions and events on levels of defence 1 to 4a as well as in the case of postulated site-specific external natural hazards. A limited, stable crack growth is permissible in the upper-shelf toughness only, with a safety margin to critical crack sizes having to be maintained.

3.5 Demonstration of fracture resistance of housings

If the concept of break preclusion is applied to valve housings pursuant to Section 3.1, it shall be demonstrated that there will be no unstable crack growth through the wall under operating conditions and events on levels of defence 1 to 4a as well as in the case of postulated site-specific external natural hazards. A limited, stable crack growth is permissible in the upper-shelf toughness only, with a safety margin to critical crack sizes having to be maintained.

3.6 Precautionary measures regarding the demonstration of leaktightness

Interpretation regarding Annex 3, Number 3.2.4 of the "Safety Requirements for Nuclear Power Plants"

For sections of high-energy piping of the reactor coolant pressure boundary and the external systems between the containment and the outer isolating valve, which in the case of a leak may lead to

– an impermissible pressure increase in the surrounding building structure or

– impermissible impacts on safety-relevant installations (e.g. flooding, jet forces, temperature, humidity) or

- an impermissible release of reactor coolant outside the building structure

but for which the safety demonstration does not consider any consequential damage if a leak occurs in them, the following requirements shall be fulfilled additional to the demonstration of break preclusion pursuant to Section 3.3:

- To avoid peak stresses, the structural design criteria of basic safety shall be implemented in particular.
- The spatial expansion of the affected areas shall be closely restricted.
- Branching-off piping or weld locations are not permitted.
- For the confirmation of the integrity verification, they shall be monitored such that the locally occurring impacts are known.
- For the connecting building isolation valves (in the case of pressurised water reactors) or penetration valves (in the case of boiling water reactors) as well as other isolating valves, break preclusion shall be demonstrated pursuant to Section 3.5.

3.7 Break preclusion for low-energy, rarely stressed or little stressed components

Interpretation regarding Annex 2, Appendix 2 of the "Safety Requirements for Nuclear Power Plants"

If regarding low-energy, rarely stressed or little stressed piping sections, vessels or housings of the external systems with a nominal bore of 50 or greater as they are addressed in the "Safety Requirements for Nuclear Power Plants", Annex 2, in Section 4.2 of Appendix 2, only subcritical cracks are assumed in lien with the criteria mentioned there, the requirements pursuant to Section 3.1 for applying the break preclusion concept shall be fulfilled. Here, the fracture-mechanical analyses pursuant to 3.1 (2) and (3) may be dispensed with.

4 Components of small nominal bore

Interpretation regarding Numbers 3.1 and 3.4 of the "Safety Requirements for Nuclear Power Plants"

4.1 Scope of application

The following requirement apply to the pressure-retaining walls of piping and valves with a nominal bore of 50 or less which are allocated from a systems-related point of view to the reactor coolant pressure boundary or the external systems.

Excepted are steam generator tubes and other heat exchanger tubes.

Other components of small nominal bore (immersion shells, detector assemblies, pressuriser heaters etc.) are not explicitly dealt with. For those, equal reliability shall be demonstrated by design, choice of materials, and testing.

4.2 Design

The dimensioning, routing and support of piping and valves shall be in line with defined written specifications and shall be documented. These specifications have to ensure that

– under operating conditions and events on levels of defence 1 to 3 as well as in the case of postulated site-specific external natural hazards the service limits are kept in order to avoid any impermissible consequences. By making specific requirements for the integrity of the piping under dynamic excitations, especially from the connecting systems and components, a single failure shall be avoided and a systematic failure (e.g. through fatigue, rupture, bending) shall be excluded.

– no failure will occur as a result of internal as well as human-induced external hazards which will call the effectiveness of the measures and installations necessary for the control of the respective event into question.

4.3 Choice of materials and manufacture

4.3 (1) The choice of materials and quality of manufacture have to ensure that possible degradation mechanisms, taking the service fluids and operating conditions into account, will not lead to a systematic failure.

4.3 (2) Prior to their commissioning, the pressure envelopes of the piping and valves have to be subjected to a pressure test at a defined pressure above the design pressure (preservice hydrostatic test).

4.4 Operation

The routing, position and function of supports as well as the integrity of the pressure envelopes shall be examined

– upon commissioning,

– if necessary, following interventions (e.g. maintenance measures), as well as

– to a representative extent by in-service inspections that also include leak tests.

When determining the representative extent, the safety significance shall be taken into account. Impermissible deviations from the documented boundary conditions must be recognised in such good time that systematic effects on the integrity of the pressure envelopes during long-term operation can be avoided and thus the reliability that is necessary for undisturbed operation is maintained.

5 Guard pipes

Interpretation regarding Annex 3, Number 3.2.3 of the "Safety Requirements for Nuclear Power Plants"

For piping sections carrying medium which run along a guard pipe to prevent any impermissible consequential effects from assumed leaks or breaks, the following requirements shall apply:

– The guard pipe shall be designed such that the impacts from the assumed leaks and breaks of the piping sections carrying medium can be shed without global plastic deformations. Its function must be maintained in such cases.

– The design of guard pipes that in the case of a challenge assume the function of the containment must at least correspond to the design conditions of the containment.

6 Containment

Interpretation regarding Number 3.6 of the "Safety Requirements for Nuclear Power Plants"

6.1 Scope of application

The following requirements shall be applied to containments made of steel or reinforced concrete with steel liner, including their penetrations, airlocks and components upstream of the isolation system and, in the case of boiling water reactors, including the pressure suppression system with the associated components for the introduction of released reactor coolant into a water pool.

6.2 Design

6.2.1 Principles

6.2.1 (1) The containment, including its penetrations, isolating valves and air locks, as well as in the case of boiling water reactors the pressure suppression system for pressure limitation shall be designed such that it will withstand the static, dynamic and thermal impacts from operating conditions and events on levels of defence 1 to 4a as well as consequential effects from postulated site-specific natural hazards and human-induced external hazards while keeping to the underlying leak rate. Containments made of steel or the steel liners of containments made of reinforced concrete and pre-stressed concrete shall, if necessary, be protected by building structures in such a way that their leaktightness function will not be impaired upon internal hazards.

Note: Specifications for the determination of differential pressures can be found in the "Safety Requirements for Nuclear Power Plants", Annex 5, Appendix 2. Specifications for the determination of the impacts of jet and reaction forces as well as of flying missiles can be found in the "Safety Requirements for Nuclear Power Plants", Annex 5, Appendix 3.

6.2.1 (2) To ensure the pressure differences, the containment penetrations have to be sufficiently leaktight during normal specified operation (levels of defence 1 and 2) of operating phases A and B as well as in the case of events on level of defence 3, internal hazards, and postulated site-specific natural hazards.

6.2.2 Choice of materials

Interpretation regarding Numbers 3.6 and 3.1 of the "Safety Requirements for Nuclear Power Plants"

6.2.2 (1) Regarding containments made of steel as well as steel liners, the metallic materials, including the filler materials and load-bearing nuts and bolts, shall be chosen such that they will comply with the functional requirements (leaktightness) and withstand the postulated loads (of e.g. mechanical, thermal, chemical kind).

6.2.2 (2) Regarding containments made of steel as well as steel liners, the material characteristics, the intended joining procedures and the quality assurance measures shall be defined such that a quality and testability that is in line with the specifications is achieved.

6.2.2 (3) Regarding containments made of steel, the material characteristics have to ensure that a sufficiently tough material condition is maintained everywhere and under all operating and accidents conditions.

Note: For the steel shell, medium-tough fine-grained structural steel suitable for welding shall preferably be used.

6.2.2 (4) Regarding containments made of reinforced concrete and pre-stressed concrete, the materials (concrete, reinforcement steel and pre-stressing steel) have to fulfil the applicable technical standards or be building-inspectorate-approved for the intended use.

6.2.3 Construction and layout

Interpretation regarding Numbers 3.6 and 3.1 of the "Safety Requirements for Nuclear Power Plants"

6.2.3 (1) The construction and surface condition of the containment made of steel and of steel liners have to allow sufficient and reliable non-destructive testing, especially of the weld seams. Areas which due to the constructive layout of the plant are no longer accessible for in-service inspections shall be realised such that corrosive influences are avoided.

Note: For components made of reinforced concrete and pre-stressed concrete, reference is made to DIN 25449 (Version February 2008).

6.2.3 (2) The design of the containment shall provide for devices for carrying out leak and pressure tests and for the installation of the necessary associated instrumentation.

6.2.3 (3) Airlocks shall be provided in the containment for bringing in and taking out materials and objects and for the entry and exit of persons.

6.2.3 (4) Emergency airlocks shall be arranged in such a way that persons can escape from the containment as quickly as possible and with the lowest possible exposure to radiation. Here, apart from radiation fields and contamination, the fact that escape routes may be blocked e.g. by outflowing media such as water, steam of gas shall also be taken into account.

6.2.3 (5) It shall be ensured through interlocking that in the operating phases in which airlocks are meant to be closed, one interlock gate cannot be opened unless the opposite interlock gate and its associated pressure equalisation device are closed and sealed. Unlocking shall only be permissible in exceptions under permissible safety-related conditions.

6.2.3 (6) Airlocks and dampers necessary for the containment system shall be connected to a leakoff system by which leakages can be pumped back into the containment.

6.2.3 (7) It must be possible to inspect the chambers surrounding the penetrations at the containment's design pressure.

6.2.3 (8) The number of penetrations and their cross-sections shall be kept as low as practically possible

6.2.3 (9) Unless the occurrence of impermissible negative pressures due to events on levels of defence 2 to 4a, internal hazards and postulated site-specific natural hazards as well as human-induced external hazards can be excluded, reliable installations shall be provided to prevent failure due to negative pressure.

6.2.3 (10) Piping in contact with the reactor coolant or the inner atmosphere of the containment and penetrating the latter shall as a rule be fitted with two isolating valves, of which one shall be installed outside as closely as possible to the containment. Exceptions shall be allowed if this is necessary due to the technical character or operating mode of the piping concerned (e.g. valves that need to be opened for accident control) and if the safety-related function of the containment is not impaired. Each single isolating valve must fulfil the specified leaktightness requirements for itself.

Piping that penetrates the containment but is not in contact with the reactor coolant or the inner atmosphere of the containment shall be fitted with at least one isolating valve lying on the outside of the containment.

6.2.3 (11) All penetrations through the containment wall and the airlocks in the containment must fulfil at least the design requirements for the containment itself.

This also applies to piping penetrating the containment wall up to the outer isolating valve, the associated containment isolation devices and, if necessary, the chamber surrounding the penetration.

regarding ventilation ducts, this shall also apply to the duct areas between the isolation valve and the associated containment isolating devices.

6.2.3 (12) If necessary, installations shall be provided to avoid any impermissible pressures transgressions between the isolating valves.

6.2.3 (13) The penetrations shall be designed such that they can shed all forces and moments of the pipe guided through them during normal specified operation (levels of defence 1 and 2) and in events on levels of defence 3 and 4a as well as during postulated site-specific external natural hazards as well as human-induced external hazards. Penetrations which due to high acting loads cannot be rigidly connected to the containment nozzle shall be connected using compensators and shall be chambered.

6.2.3 (14) The closing speed of the containment isolation devices shall be limited such that there will be no impermissible effects.

6.2.3 (15) Between the isolating valves and the containment, piping distances shall preferably be short. In these areas, pipe branches shall as a rule not be permissible. Exceptions shall be justified from a safety-related point of view (e.g. drainage nozzles, test connections).

6.2.3 (16) As for piping that penetrates the containment, the structures within the containment shall be isolated by suitable constructions from mechanical impacts from events on levels of defence 2 to 4a outside the containment as well as from postulated site-specific external natural hazards as well as human-induced external hazards so that a consequential failure inside the containment need not be assumed.

6.2.4 Strength-related design

Interpretation of Numbers 3.6 and 3.1 of the "Safety Requirements for Nuclear Power Plants"

6.2.4 (1) To ensure integrity and the specified leaktightness, the maximum pressures and temperatures that may occur as well as the impacting loads in events on level of defence 3, in internal hazards and in postulated site-specific natural hazards as well as human-induced external hazards shall be applied as a basis. Regarding large dry containments, the integrity demonstration for human-induced external hazards may be limited to the undisturbed areas of the steel shell. When determining the design pressure, safety factors and minus allowances shall be considered for

– model uncertainties and

– the worst-case initial operating condition.

Note: See also Annex 1 to I-2, "Principles of the strength-related design and allocation of load stages to the different levels of defence".

On the effects of leaks and breaks in high-energy piping on the containment, see Section 3.6 and also Annex 3, Number 3.2.4 of the "Safety Requirements for Nuclear Power Plants".

On the determination of differential pressures within the containment was well as jet and reaction forces in connection with leaks in pressure-retaining systems within the containment, see also Annex 5, Appendix 2 as well as 3 of the "Safety Requirements for Nuclear Power Plants".

6.2.4 (2) The containment of a pressurised water reactor shall be designed such that the mass and the energy content of the reactor coolant pressure boundary and of the secondary side of a steam generator can be accommodated up to the secondary-side isolation. In addition, the heat emission of all steam generators to the outflowing reactor coolant shall be considered.

6.2.4 (3) The containment of a boiling water reactor with pressure suppression system shall be designed such that the mass and the energy content of the reactor coolant pressure boundary can be accommodated up to the reactor-side isolation. In the design, the amounts of water and steam in the main-steam or feedwater lines shall also be considered which upon the closing of the valve may flow back or be conveyed back into the containment.

The atmosphere and the water pool in the pressure-suppression pool shall be treated with separate energy balances (imbalance). The condensation effect of the water pool shall be taken into account in connection with pressure reduction. The accident loads shall be taken into account together with their effects such as pressure build-up, pressure relief and pressure decay processes, induced vibrations as well as simultaneous occurrence of such processes regarding the effects on the containment, the pressure suppression and relief system as well as other systems with their maximum consequences.

6.2.4 (4) The anchoring and supports of the safety and relief valves, pressure suppression pipes and vent pipes necessary in a boiling water reactor in the pressure suppression pool area of the containment shall be designed such that they will reliably shed the loads from operating conditions and events on levels of defence 1 to 4a , from internal hazards and from postulated site-specific external natural hazards (fluid-dynamic loads, jet and reaction forces) as well as human-induced external hazards. Furthermore, structural or functional installations shall be provided so that the integrity of the containment structure will not be impaired by jet and impulse forces of the vent pipes.

6.2.4 (5) To ensure stability and integrity, especially regarding the leaktightness of the containment and its components, a confirmation concept shall be applied that considers the following principles:

a) The load cases and their combinations to be allocated to the different levels of defence according to the events shall be unambiguously clarified (e.g. in a load case catalogue including the kind, intensity, frequency, time history and course of the impact). regarding the load case combinations, partial loads that may happen at the same time shall be considered as occurring simultaneously. Moreover, impacts from assembly shall be considered.

b) The impacts resulting from the load cases shall be described component-specific (e.g. in design data sheets).

c) The stresses caused by the loads shall be limited such that for each level of defence and postulated site-specific external natural hazard, a sufficient safety margin to the postulated failure mode is ensured.

d) To maintain the leaktightness function in a challenge, proof shall be furnished of the dimensional stability and the deformation limit.

6.2.4 (6) For a containment made of reinforced concrete and prestressed concrete, a steel liner shall be provided. The latter shall be anchored in the concrete structure in such a way that its leaktightness functions will be maintained under all loads resulting from operating conditions and events on levels of defence 1 to 4a, internal hazards as well as postulated site-specific natural hazards and human-induced external hazards. Pressure-retaining steel components of the penetrations and connections to the liner shall be executed and anchored such that in the case of the above-mentioned hazards, they will bear the stresses resulting from the arising forces from pressure and temperature impacts, piping reactions, and other loads.

A containment made of reinforced concrete and prestressed concrete shall fulfil the applicable technical standards.

Note: For components made of reinforced concrete and prestressed concrete, reference is made to DIN 25449 (version February 2008).

6.3 Manufacture

Interpretation relating to Numbers 3.6, 3.1 (1) and 3.1 (2) of the "Safety Requirements for Nuclear Power Plants"

6.3.1 Principles

6.3.1 (1) The quality characteristics to be adhered to for ensuring containment integrity shall be defined and complied with during the manufacturing process.

6.3.1 (2) The manufacturing process shall be monitored and documented such that any deviations from the quality characteristics determined will reliably be recognised and that it is possible to trace back the deviations with a view to identifying their cause. Any measures taken additionally to comply with the quality characteristics shall be documented.

6.3.1 (3) For the qualification of the welding methods of for verifying the quality characteristics of component welds, procedure qualification and production tests shall be carried out. Combining the performance of production tests with procedure qualification tests is permissible.

6.3.2 Accompanying non-destructive testing for components made of steel

6.3.2 (1) It shall be shown by suitable tests on product forms that the component has the specified toughness, ductility, strength and structural characteristics across the wall thickness.

6.3.2 (2) Requirements for the kind and scope of the inspections to be performed shall be defined with consideration of the loads. The mechanical-technological characteristics shall be demonstrated on each product form (component test). These tests shall cover

a) in a representative manner the different deformation directions during the manufacturing process in several sampling locations as well as

b) all heat treatments carried out during the manufacturing process.

6.3.3 Accompanying non-destructive testing, leak and pressure tests

6.3.3 (1) The volume and the surfaces of all product forms and weld seams shall be subjected to non-destructive tests with sufficient fault recognition.

The testing methods and parameters shall be chosen such that all faults that are clearly below the extent of safety-relevant faults can be detected.

6.3.3 (2) The containment and its penetrations as well as chambers shall be subjected to a pressure test prior to first commissioning. After the pressure test, representative non-destructive tests shall be performed.

6.3.3 (3) The leaktightness of the containment shall be demonstrated by way of an integral leak test before starting power operation for the first time.

6.3.3 (4) Starting from a pressureless condition of the containment, the first leak test shall be carried out with ever increasing pressure steps at the overpressure intended for the regularly recurring pressure tests (see Number 6.4.3 (1) below) and at design pressure.

6.4 Operation

Interpretation regarding Numbers 3.6 and 3.1 (2) of the "Safety Requirements for Nuclear Power Plants"

6.4.1 Principles

6.4.1 (1) Operating data that are relevant for the function of the containment shall be monitored. This concerns,

a) for large dry containments, the subatmospheric pressure system,

b) for containments with pressure suppression system, apart from the maintaining of subatmospheric pressure in the drywell also the effectiveness of the separation between drywell and pressure suppression pool,

c) for containments made of reinforced concrete and prestressed concrete, suitable measures for an assessment of whether prestress is maintained, and

d) the effectiveness of inertisation if inertisation or partial inertisation is operationally intended.

Measures that are intended for indicating a functional impairment of the containment shall either be designed redundantly or readings from diverse systems shall be used.

6.4.1 (2) If seals or sealing elements are used that are made of materials that may lose their effectiveness due to the impacting ambient conditions, loads or loading frequency, precautions shall be taken to control their ageing.

6.4.1 (3) Regarding work processes in the containment, cleanliness conditions shall be defined. In particular, the entry of corrosive products into areas of the containment that are not accessible for regular testing shall be avoided.

6.4.2 Non-destructive in-service inspections

6.4.2 (1) The walls and components of the containment shall be inspected regularly in representative locations with regard to their general condition as well as any mechanical and corrosive degradation. Here, especially the transitions between the steel shell or steel liner to the concrete and the elastic sealing's of these transitions shall be part of these inspections.

6.4.2 (2) The non-destructive in-service inspections shall be carried out and assessed with regard to possible degradation mechanisms in a representative manner by means of qualified methods, taking all kinds of welded joints into account.

The defined inspection intervals shall be guided by technical experience and shall take operating experience into account.

6.4.2 (3) Test procedures and methods shall be chosen such that service-induced faults (e.g. due to fatigue, corrosion) with their possible orientations can be registered and documented. Any indications that were documented during manufacturing and have been left untreated shall be registered and, if necessary, followed up.

6.4.2 (4) For each test procedure, evaluation limits for the identification of indications shall be specified.

6.4.3 Recurring function and leak tests

6.4.3 (1) To demonstrate the required leaktightness of the containment during the intended service life of the plant, regular recurrent leak tests shall be performed.

The regular recurrent leak tests shall be performed at pressures at which the measured leak rates can be reproducible and at which an adequate conclusion can be drawn with regard to the leak rate under design conditions.

6.4.3 (2) The recurrent leak tests shall be performed at the end of a shutdown phase once all maintenance and repair work that may alter the leaktightness of the containment has been finished.

6.4.3 (3) The leaktightness of the components connected to the leakage exhaust system as well as of the system itself shall be quantitatively determined in a joint measurement at the start and at the end of an interruption of power operation (operating phases C, D and E).

6.4.3 (4) The reliability of containment isolation with the required leaktightness shall be demonstrated for the conditions given in a loss-of-coolant accident with the highest pressure-build-up and the highest temperatures inside the containment.

6.4.3 (5) The functional condition and leaktightness of airlocks, isolating installations and swing check valves as well as the manipulating speed of valves for isolating the containment shall be regularly checked.

6.4.3 (6) The housings of the pipe penetrations of the containment, the airlocks, cable penetrations and access panels shall be regularly and after in-service maintenance measures tested for their leaktightness.

6.4.3 (7) Assembly openings and backup penetrations shall be examined for their leaktightness after use.

6.4.3 (8) In a boiling water reactor with pressure suppression system, the permissible leak rate between drywell and pressure suppression pool shall be determined before the leak test and verified by measurement.

7 Handling of crack indications on components and piping

Interpretation regarding Number 3.4 of the "Safety Requirements for Nuclear Power Plants"

7 (1) The following criteria apply to components and piping of the reactor coolant pressure boundary and the external systems, but not to heat exchanger tubes.

7 (2) If a crack indication is found during an in-service or incident-related inspection and if this crack indication exceeds the evaluation limit, then it shall be designated as a finding. If any results from previous examinations of the inspected area are documented, these shall be used for comparison.

7 (3) If a finding has been identified for the first time or if a change during operation cannot be excluded on the basis of the comparison made with the results of previous examinations, supplementary

examinations shall be performed in order to be able to draw conclusions as to the kind, location and size of the finding. The thus determined finding shall be conservatively attributed a type of defect and a corresponding dimension with regard to the acceptance target. The underlying cause of the type of defect and the degradation mechanism shall be determined. Here, the possibility of a systematic cause of defect shall be investigated in particular. If necessary, additional analysis methods shall be used for the root cause analysis.

7 (4) An advanced analysis shall show

- to what extent the fulfilment of the requirements for the design was affected by the defect and
- what options are available for preventing the causes in future.

To clarify whether there is a systematic cause of defect, verification tests shall be carried out on similar components or areas of components on which the identified or assumed cause of defect might also have an effect.

7 (5) Any identified defects that might impair the fulfilment of the requirements of the design concept under the specified impacts from operating states and events on levels of defence 1 to 4a and from postulated site-specific natural hazards or human-induced external hazards must not be left unrectified. Any identified defects for which an operationally-induced cause cannot be excluded shall as a rule not be left unrectified.

7 (6) If in justified exceptions the defects are left unrectified, these defects shall be assessed. The possible development of the defect shall be determined conservatively for the specified operating lifetime. Under these conditions it shall be shown that considering all impacts from operating states and events on levels of defence 1 to 4a and from postulated site-specific natural hazards or human-induced external hazards, any consequences that are impermissible from a safety-related point of view can be excluded. To back up the predicted development of the defect, verification tests shall be carried out. Their scope and date shall be specified in accordance with their safety significance.

7 (7) It shall furthermore be shown that

- any defects left such unrectified are not related to a systematic cause of defect that may lead to further defects and would hence require a large number of monitoring and control measures,
- there is no accumulation of defects which – each taken for its own or together with the other ones - will lead to a safety-significant impermissible impairment of the integrity of the respective affected component nor of the reliability of the affected systems.

7 (8) It shall be examined whether the kind and extent, circumstance and date of the discovery or the frequency of the occurrence of defects allow conclusions as to any gaps or inadequacies in the system- and component-specific requirements (e.g. specifications, testing manual); if necessary, the corresponding gaps shall be closed and inadequacies remedied. New insights from the root cause analysis shall be adapted into the technical documents (e.g. regarding the specified impacts, water chemistry, and vibrations) and considered in ageing management. Also, where required, corresponding measures shall be taken on the affected components or with regard to their mode of operation.

Annex 1: Principles of the strength-related design and allocation of load stages to the different levels of defence

Interpretation regarding Numbers 3.4 (1), (2) and 3.6 (1), Annex 2, Appendix 1 of the "Safety Requirements for Nuclear Power Plants"

Note: The following are explanations of the requirements made in Sections 2.3.1 and 6.2.4 of Interpretation I-2 for the integrity, with consideration of the principles also underlying the technical rules. These explanations have to be added by the matrix from the "Safety Requirements for Nuclear Power Plants", Annex 2, Appendix 1 "Principal classification of load levels according to levels of defence and hazards affecting more than one redundancy". In this Appendix, load levels – as defined in the KTA Nuclear Safety Standards – are allocated to the levels of defence as well as to natural hazards and human-induced external hazards defined in the "Safety Requirements for Nuclear Power Plants".

1 Reactor coolant pressure boundary and external systems

(1) For the components of the reactor coolant pressure boundary and the external systems, precautions shall be taken against failure due to the following mechanisms:

a) plastic instability,

b) global deformation,

c) progressing deformation,

d) fatigue,

e) unstable crack propagation,

f) elastic instability.

(2) The safety margins necessary in this context for the loads resulting from the impacts shall be defined for the different levels of defence as follows:

a) The load limits of levels of defence 1 and 2 must ensure that the loads establish the balance to the impacts in such a way that there will be no global plastic deformations, no elastic instability, no fracture due to progressing deformation and no failure to fatigue in the course.

In this connection, the safety margins shall be chosen such that if there are loads resulting from inner pressure, weight, fluid dynamics and other quasi-static impacts, the load-carrying cross-section shall remain within the range of elastic material behaviour excepting locally limited areas. In case of additionally acting stationary and varying impacts from operating states on levels of defence 1 and 2, the safety margins shall be defined such that furthermore, a failure due to progressing deformation and fatigue shall not be postulated, either.

b) The load limits of levels of defence 3 and 4a as well as those relating to internal hazards and postulated site-specific natural hazards or human-induced external hazards have to ensure that the loads will establish the balance to the impacts in such a way that a failure through plastic or elastic instability or die to unstable crack propagation is excluded.

In this connection, the safety margins shall be chosen such that if there are loads resulting from inner pressure, weight, fluid dynamics and other additional loads that are characteristically equal as a result of external hazards, plastic deformation will remain limited. The verification concerning the exclusion of failure due to unstable crack propagation must additionally contain the impacts from the restrained thermal expansion.

Furthermore, the load limits shall be chosen such that plastic deformation remains confined to areas of geometrical discontinuities. For geometrically simple components (e.g. piping), plastic deformation of the entire cross-section under dynamic loads shall only be permissible if the occurring elongation will remain clearly below the uniform elongation of the material. Here, influences of multiaxiality that may lead to a restriction of workability and other effects that might increase the elongation occurring shall be taken into account.

(3) When determining the load limits or the impacts, the safety allowances according to the "Safety Requirements for Nuclear Power Plants" Number 3.4 (2) shall be applied. In addition, Annex 5, "Requirements for verification and documentation", Number 3.2.1 shall be considered.

2 Containment

(1) Regarding the steel containment and its components, precaution against failure shall be taken by way of the following measures:

a) elastic and plastic buckling,

b) global deformation,

c) local deformation or progressive deformation,

d) fatigue.

(2) The safety margins necessary in this context for the loads resulting from the impacts shall be defined for the different levels of defence as follows:

a) The load limits for the operating states and events on levels of defence 1 to 4a and the postulated site-specific external natural hazards must ensure that the leaktightness function is maintained. Here, the safety margins shall be chosen such that regarding all static and dynamic loads there will be no elastic or plastic buckling and that regarding all static loads the load-bearing cross-sections will remain within the range of elastic material behaviour, with the exception of locally limited areas. In case of loads changing over time (specified load collective), the safety margins shall be defined such that fatigue failure need not be postulated.

b) For local, singular loads (e.g. pressure test), the safety margins shall be chosen such that an plastic deformation will be restricted to partial areas of the cross-section. The permissible extent of deformation shall be defined specific to the component and material.

(3) Following the occurrence of events on levels of defence 3 and 4a or external hazards, areas for which plastic deformation has been calculated shall be examined by means of a qualified inspection. Comprehensible assessment criteria shall be defined for the inspection.

(4) For a containment made of reinforced concrete and prestressed concrete, the principles of the design, the safety coefficients to be applied as well as the limit conditions of the load-bearing capacity and fitness for use according to DIN 25449 Issue 2008-02 shall be followed.

3 Load limits for safety systems

Regarding safety systems including the containment, the load limits to be applied in connection with the impacts resulting from the events that these systems are meant to control shall be compatible with the respective requirements for their reliability and the function of active components.

Note Corresponding remarks are contained in the footnotes of the tables in Annex 2, Appendix 1 of the "Safety Requirements for Nuclear Power Plants".

Interpretation I-3: Requirements for instrumentation and control and for accident instrumentation

Contents

1 Scope of application

The following interpretations apply to instrumentation and control equipment with safety-significant instrumentation and control functions on levels of defence 1 to 4.

2 Categorisation

Interpretation regarding Numbers 3.1 (4) and 3.7 (10) of the "Safety Requirements for Nuclear Power Plants"

The instrumentation and control functions, including instrumentation and control functions of the accident instrumentation, shall be classified in accordance with their safety significance into different categories for which graded requirements apply.

Category A

The instrumentation and control functions of category A comprise all functions that are necessary to control events on level of defence 3.

Category B

The instrumentation and control functions of category B comprise all functions that are necessary to control events on level of defence 2 and to prevent the occurrence of events on level of defence 3.

Category C

The instrumentation and control functions of category C comprise all other safety-relevant functions.

Not categorised are instrumentation and control functions that have no safety-relevant functions.

3 Design requirements

3.1 Instrumentation and control equipment including accident instrumentation performing instrumentation and control functions of categories A to C

Interpretation regarding Number 3.7 in connection with 3.1 (1) and 3.1 (2) of the "Safety Requirements for Nuclear Power Plants"

3.1 (1) Instrumentation and control equipment intended to fulfil instrumentation and control functions that belong to different categories shall be planned, designed and operated according to the requirements for instrumentation and control equipment that follow from the instrumentation and control functions of the category with the highest safety-significance.

3.1 (2) The applied hardware shall be tested in qualification tests or shall have operating experience concerning its usage or the expected operational conditions. This hardware shall be maintenance-free during power operation.

The applied software shall be tested for suitability.

3.1 (3) Lines and cables, including fibre optics cables, shall be routed separately for redundant trains and, if necessary, also be protected against impacts of internal and natural hazards as well as human-induced external hazards.

3.1 (4) The instrumentation and control equipment shall be designed, installed, shielded and protected such that any unacceptable influencing of the signals by plant-internal as well as external sources of disturbance is prevented.

3.1 (5) Measures and equipment shall be provided that allow checking of the operability of the instrumentation and control equipment and their interaction with the active and passive components of the safety system and the monitoring of the conditions of these safety-related components.

3.1. (6) Checkback signals from active equipment (e.g. actuators) which co-determine the functional sequence of the instrumentation and control equipment performing instrumentation and control functions of categories A to C shall preferably be derived from the process variable or picked up from the actuator itself. A reliable coupling between the position indicator and the actuator shall be ensured.

3.1 (7) Instrumentation and control equipment performing instrumentation and control functions of categories A and B shall be designed and operated such that their operability is ensured irrespective of the kind and extent of the temporal change of their input signals. The associated monitoring and alarm systems shall be designed such that a signal surge can be processed without any loss of safety-relevant information.

3.1 (8) The instrumentation and control equipment shall be designed such that any necessary adaptations to regularly recurring conditions of normal operation (e.g. stretch-out operation) can be carried out easily and reliably.

Interpretation regarding Numbers 3.1 (6), 3.1 (7) and 3.7 of the "Safety Requirements for Nuclear Power Plants"

3.1 (9) The instrumentation and control equipment shall not unacceptably impair the fulfilment of the requirements for independence and the control of failure combinations on which the procedural design of the plant is based.

Interpretation regarding Numbers 3.1 (2), 3.7 (3) and 3.7 (8) of the "Safety Requirements for Nuclear Power Plants"

3.1 (10) For instrumentation and control equipment designed for operating under accident conditions the robustness against accident conditions shall be demonstrated.

Interpretation regarding Number 3.7 and regarding Annex 4 Numbers 4 (6) and 4 (7) of the "Safety Requirements for Nuclear Power Plants"

3.1 (11) As a prevention against operating errors, technical provisions shall be given preference over organisational measures.

3.1 (12) The instrumentation and control equipment shall be designed such that the options for intervention necessary for the control of events and for the performance of accident management measures are provided. The options for intervention shall be designed such that they will not unacceptably impair the operability of the instrumentation and control equipment in the control of the events on levels of defence 2 and 3. The options for intervention shall be protected against operating errors.

3.2 Instrumentation and control equipment performing instrumentation and control functions of category A

Interpretation regarding Number 3.7 of the "Safety Requirements for Nuclear Power Plants"

Interpretation regarding Numbers 3.1 (6) and especially 3.7 (3) of the "Safety Requirements for Nuclear Power Plants"

3.2 (1) Failure-initiating evens within and outside the safety system shall be considered in the design of the instrumentation and control equipment performing instrumentation and control functions of category A.

Interpretation regarding Number 3.7 (3) and regarding Annex 4 Number 4 of the "Safety Requirements for Nuclear Power Plants"

3.2 (2) Changes in the standby positions of components of the safety system may only be made if corresponding clearance conditions are fulfilled. These changes shall be cancelled automatically or by technical provisions or organisational measures if the clearance conditions are no longer fulfilled. In the state of the system which is required from a safety-related point of view, this equipment shall be protected against unauthorised intervention.

Interpretation regarding Numbers 3.1 (2) and 3.7 (3) of the Safety Requirements for Nuclear Power Plants"

3.2 (3) If distinct standby positions of actuators during normal operation are prescribed for components of the safety system, any departure from this standby position shall be signalled. Actuators that do not signal their position shall be secured against leaving the standby position.

Interpretation regarding Numbers 3.1 (3), 3.1 (6), 3.1 (7), 3.7 (3) and regarding Annex 4 Number 2 of the "Safety Requirements for Nuclear Power Plants"

3.2 (4) A failure in the instrumentation and control equipment of the safety system may at the most have effects on the function of the redundant safety system train affected.

The instrumentation and control equipment that is necessary for the operability of the system following the onset of events on level of defence 3 shall be designed such that they can withstand the respective most adverse ambient and accident conditions that may arise in the associated installation area.

The instrumentation and control equipment shall be designed such that any inadvertent actuation of protective actions with consideration of Number 3.2 (11) is prevented if this may lead to beyond-design-basis plant states.

Interpretation regarding Numbers 3.1 (2), 3.1 (3), 3.1 (12) and 3.7 (3) of the "Safety Requirements for Nuclear Power Plants"

3.2 (5) The instrumentation and control equipment performing instrumentation and control functions of category A shall generally be designed to be self-monitoring. Their functions and characteristics that are not covered by self-monitoring shall be subjected to regular and complete inspection. The inspection cycles shall be determined on the basis of reliability considerations. These inspections shall be easy to perform at specially provided interfaces with the help of inspection aids.

Interventions for inspection purposes and manual actions shall be determined such that necessary safety functions will neither be hindered nor that the reliability of their actuation will be significantly reduced.

3.2 (6) Self-monitoring shall be designed such that it will not impair the function of the instrumentation and control equipment performing instrumentation and control functions of category A. The regular inspections shall be planned and carried out such that there will be no simultaneous inspection of necessary redundant instrumentation and control equipment.

Interpretation regarding Number 3.7 (3) of the "Safety Requirements for Nuclear Power Plants"

3.2 (7) The instrumentation and control equipment performing instrumentation and control functions of category A shall only be used for tasks within the safety system. If installations that perform instrumentation and control functions of category A are also used for instrumentation and control functions of lower categories, the associated instrumentation and control equipment shall be designed such that the required reliability of the installations that perform instrumentation and control functions of category A is maintained.

Interpretation regarding Numbers 3.1 (2) and 3.7 (3) der "Safety Requirements for Nuclear Power Plants"

3.2 (8) Instrumentation and control equipment performing instrumentation and control functions of category A shall be constructed such that the necessary verifications for the qualification of the instrumentation and control equipment of the safety system can be provided reliably.

3.2 (9) Items of Instrumentation and control equipment can be divided into three types. First, there is non-programmable equipment, consisting of discrete, non-programmable components (the application function is realised by wiring). Furthermore, there is programmable equipment, consisting of at least one discrete programmable component (the application function is realised by wiring or by electronic component functions). Finally, there is computer-based equipment, consisting of at last one processor (the application function is saved in the memory element).

3.2 (10) Diverse instrumentation and control equipment is understood as instrumentation and control equipment that differs in type or principle of operation. Dissimilarity is a subsumable concept of diversity that relates to computer-based or programmable systems.

Instrumentation and control equipment that is sufficiently dissimilar or disparate to other instrumentation and control equipment regarding its hardware, software, development tools, development teams, manufacture, testing and maintenance are referred to as dissimilar instrumentation and control equipment with the result that a systematic failure of instrumentation and control equipment that is dissimilar to each other such systems need no longer be postulated. Regarding the control of systematic failures, instrumentation and control equipment that consists of non-programmable equipment shall be considered as diverse to those instrumentation and control equipment that consists of computer-based or programmable equipment.

Note: The term "dissimilar instrumentation and control equipment" is introduced to express by assessing different aspects that whenever similar technologies are employed, two systems are sufficiently unlike each other. The assessment may also include the admissibility of the similarity of individual aspects.

Interpretation regarding Numbers 3.1 (5), 3.7 (3) and 3.7 (4) of the "Safety Requirements for Nuclear Power Plants"

3.2 (11) The design of the instrumentation and control equipment performing instrumentation and control functions of category A shall include provisions against systematic failures of the instrumentation and control installations to reduce their probability in such a way that a systematic failure on level of defence 3 need no longer be postulated.

If this verification cannot be provided according to the state of the art in science and technology for computer-based or programmable instrumentation and control equipment, provisions shall be taken such that a systematic failure of hardware and software on level of defence 3 is controlled.

If computer-based or programmable instrumentation and control equipment is used, diverse instrumentation and control equipment shall generally be employed with consideration of the following conditions.

There are no requirements regarding the application of diverse equipment if an active systematic failure for the corresponding instrumentation and control function is safety-oriented.

If computer-based or programmable instrumentation and control equipment is used, two-fold or three-fold diverse instrumentation and control equipment shall be provided for protective actions that are not safety-oriented for every plant state, depending on the effects of passive or active systematic failures in the instrumentation and control equipment performing instrumentation and control functions of category A. At least two-fold diverse equipment shall be used

– if the accident will be controlled with the help of the safety equipment still available or

– if each of the two diverse instrumentation and control equipment will trigger the necessary protective action each by itself.

If with the use of computer-based or programmable instrumentation and control equipment one of the two conditions mentioned above for the employment of a two-fold diverse equipment do not apply, three-fold diverse instrumentation and control equipment shall be provided.

Interpretation regarding Numbers 3.1 (3), 3.1 (7), 3.7 (3) and regarding Annex 4 Number 2 of the "Safety Requirements for Nuclear Power Plants"

3.2 (12) The design of the instrumentation and control equipment performing instrumentation and control functions of category A shall generally be such that they fulfil their functions in case of demand with consideration of the following assumptions: There is

a) a random failure caused by a single failure

b) and a systematic failure (systematic failure of the hardware or systematic software failure); this does not apply if the condition of Number 3.2 (11) is fulfilled,

c) and consequential failures

d) and a maintenance case.

During a maintenance case, the simultaneous occurrence of the systematic failure and the random failure need not be assumed for a time span of 100 h.

In the case of computer-based and programmable instrumentation and control equipment with a sufficiently high degree of self-monitoring and demonstrated maintenance times of less than 8 h, the simultaneous occurrence of the systematic failure and a random failure or the maintenance case need not be assumed.

Note: Regarding failure due to single failure and unavailability due to maintenance, further requirements are specified in Annex 4 of the "Safety Requirements for Nuclear Power Plants".

Interpretation regarding Numbers 3.1 (3) and 3.7 of the "Safety Requirements for Nuclear Power Plants"

3.2 (13) Protective equipment of components and auxiliary equipment shall be designed such that if a component is challenged by the instrumentation and control equipment of the safety system, the protective equipment will generally not become effective unless the possibly resulting consequential damage will impair the safety of the plant to a greater extent than the failure of the component.

Note: Protective equipment of components and auxiliary equipment is the equipment (s. 3.2(9)) of equipment unit protection.

The protective equipment shall be designed such that the priority of the instrumentation and control functions of category A over the protective equipment is ensured.

If in a piece of protective equipment it is necessary that the latter shall have priority over the instrumentation and control functions of category A, the same requirements need to be made for the protective equipment as for the instrumentation and control equipment performing instrumentation and control functions of category A.

The requirements for the instrumentation and control equipment performing instrumentation and control functions of category A need not be made for the protective equipment if it can be shown that defects in the protective equipment are so unlikely that an inadvertent actuation by this equipment need no longer be assumed.

Interpretation regarding Numbers 3.1 (2) and 3.7 (3) of the "Safety Requirements for Nuclear Power Plants"

3.2 (14) In the control room and in the supplementary control room, the protective actions and measures triggered by the instrumentation and control functions of category A shall be represented to the extent that is necessary for the defined tasks of the control room and the supplementary control room. Here, the protective actions and measures triggered by the instrumentation and control functions of category A together with their effects on the process shall be represented in the control room and in the supplementary control room in such a way that it is possible for the personnel to assess the plant state reliably and timely.

3.3 Instrumentation and control equipment performing instrumentation and control functions of category B

Interpretation regarding Number 3.7 (2) of the "Safety Requirements for Nuclear Power Plants"

The instrumentation and control equipment performing instrumentation and control functions of category B shall be designed such that it will fulfil their tasks even if in case of a challenge an additional random failure with resulting consequential failures occur.

3.4 Requirements for the accident instrumentation

Interpretation regarding Numbers 3.1 (2), 3.1 (4) and 3.7 (8) of the "Safety Requirements for Nuclear Power Plants"

3.4.1 General criteria for the accident instrumentation

The design concept and the safety-relevant individual features of the accident instrumentation shall be documented in a verifiable manner.

3.4.2 Design of the accident instrumentation

3.4.2.1 Accident overview measuring systems

3.4.2.1 (1) The accident monitoring system shall be designed such that the data necessary for the assessment of the plant's safety, the effectiveness of the safety system and for the decision about accident management measures before, during and after the onset of an event on levels of defence 3 and 4a, during internal or natural hazards, and in human-induced external hazards are indicated reliably and with sufficient exactness.

It shall be considered in the design of the accident monitoring system that the data generated before, during and after the onset of an event sequence or plant state that may lead to an increased release of radioactive materials into the environment of the power plant (levels of defence 4b or 4c) are needed for the decision about accident management measures. Under the postulated ambient conditions, these data shall be indicated with the requisite exactness.

3.4.2.1 (2) The accident overview monitoring system shall be designed such that the measured variables that are essential for an assessment of the plant state and the radiological consequences for the environment before, during and after the onset of an event on levels of defence 3 and 4a, during internal or natural hazards, and in human-induced external hazards are recorded.

3.4.2.1 (3) Wide-range monitoring shall be provided for those measured variables that characterise the representative event sequences and derived plant states of levels of defence 4b and 4c (see "Interpretations regarding the "Safety Requirements for Nuclear Power Plants", I-7: Requirements for accident management").

3.4.2.2 Accident recording

3.4.2.2 (1) The accident recording system shall be designed such that the measured variables that are recorded before, during and after the onset of

- an event on levels of defence 3 and 4a, during internal or natural hazards, and in human-induced external hazards or
- an event that may lead to an increased release of radioactive materials into the environment of the power plant (levels of defence 4b or 4c)

are documented clearly and in the correct chronology.

3.4.2.2 (2) The accident recording system shall be designed such that for each measured variable recorded by the accident overview monitoring system, the time reference can be determined from the associated documentation to such an exact degree that it is possible to allocate the recording times of data from other information sources.

3.4.2.2 (3) It shall be specified which equipment of the accident recording system, which has to be operational during plant operating phases B to F.

For the recording and storage of accident sequence data, at least two data storages that are diverse if possible shall be used as a precaution against a systematic failure. The failure of one data storage shall be signalled.

3.4.2.2 (4) The accident records shall be kept safely secured. It shall be ensured that these secured data can be neither altered nor erased.

3.4.2.2 (5) The documentation facilities shall be clearly structured and marked unambiguously.

3.5 Instrumentation and control equipment performing instrumentation and control functions in human-induced external hazards and on levels of defence 4b or 4c

Interpretation regarding Numbers 3.1 (10) and 3.7 of the "Safety Requirements for Nuclear Power Plants"

This instrumentation and control equipment shall be designed such that it will fulfil its tasks with the reliability required for these levels of defence under the ambient conditions postulated for the respective tasks. All instrumentation and control equipment that will contribute to the compliance with the fundamental safety functions may be used for accident management measures.

3.6 Requirement specification for instrumentation and control equipment performing instrumentation and control functions of categories A to C

Interpretation regarding Numbers 3.1 (2), 3.1 (4) and 3.7 of the "Safety Requirements for Nuclear Power Plants"

3.6 (1) The requirements for the instrumentation and control functions shall be documented in a requirement specification structured in a clear manner of presentation.

The requirement specification for the instrumentation and control functions of categories A, B and C shall indicate at least:

– tasks,

– categories of the instrumentation and control functions,

– actuation criteria,

– input signals,

– signal processing,

– control of the actuators,

– signals/indicators,

– ambient conditions,

– requirements for data recording,

– interfaces to other instrumentation and control functions,

– reaction times, and

– fundamental safety functions.

3.6 (2) The tasks of the instrumentation and control functions used on levels of defence 2, 3 and 4a, during internal or natural hazards as well as during human-induced external hazards shall be determined on the basis of an analysis of the event sequences that comprise the postulated events on levels of defence 2, 3 and 4a and the internal or natural hazards as well as the human-induced external hazards.

3.6. (3) The requirement specification for the instrumentation and control functions of categories A and B shall be structured such that the process-based formulation of the tasks is divided into clearly separated subtasks. These subtasks shall be represented as instrumentation and control functions.

The subtasks of the computer-based and programmable instrumentation and control equipment performing instrumentation and control functions of category A shall be designed such that the instrumentation and control functions derived from them have a small functional range.

The entirety of all instrumentation and control functions of categories A, B and C shall be documented in a clearly structured manner.

3.6 (4) It shall be demonstrated that compliance with the fundamental safety functions is ensured with the help of the requisite instrumentation and control functions according to the requirement specification in all postulated events and event sequences.

3.6 (5) The safety-relevant functions of the process management and information installations shall be defined in the requirement specification.

3.7 Acquisition of process variables

Interpretation regarding Numbers 3.1 (2) and 3.7 (8) of the "Safety Requirements for Nuclear Power Plants"

3.7 (1) The necessary process variables for the postulated events on levels of defence 2 to 4a as well as the accident management measures shall be acquired.

3.7 (2) For each event on level of defence 3 to be controlled by the instrumentation and control equipment performing instrumentation and control functions of category A, at least two different actuation criteria to be deduced from physically different process variables shall be applied generally. If this is not technically feasible, other measures and equipment shall be provided to achieve a high degree of reliability.

3.8 Redundancy and independence

Interpretation regarding Numbers 3.1 (3), 3.1 (7) and 3.7 of the "Safety Requirements for Nuclear Power Plants"

3.8 (1) The instrumentation and control equipment shall be structured such that the prescribed redundancy in the active equipment of the safety system is maintained.

3.8 (2) Redundant instrumentation and control equipment performing instrumentation and control functions of categories A and B shall be designed to be independent of each other to an extent that a plant-internal failure-initiating event will not lead to the loss of several redundant system trains.

If individual redundant system trains of instrumentation and control equipment performing instrumentation and control functions of category A fail due to human-induced external hazards, the remaining redundant system trains shall suffice for controlling this event.

3.8 (3) As a protection against failure-initiating events affecting several redundant system trains within the instrumentation and control equipment and the plant as a whole, redundant system trains of the instrumentation and control equipment performing instrumentation and control functions of categories A or B shall be arranged physically separate.

3.8 (4) Connections of the instrumentation and control equipment performing instrumentation and control functions of categories A and B with uncategorised equipment or with data processing or data transmission equipment of category C shall be minimised, taking technical and operational needs into account. They shall be implemented to be non-interacting.

3.8 (5) The instrumentation and control equipment performing instrumentation and control functions of categories A to C shall be designed independent of each other such that in case of a failure-initiating event in the equipment performing instrumentation and control functions categorised as being of lesser safety relevance, the instrumentation and control functions categorised as being of higher safety relevance are maintained.

3.8 (6) The instrumentation and control equipment performing instrumentation and control functions of categories A to C shall be designed such that the output signals of instrumentation and control functions categorised as being of higher safety relevance will have priority over the output signals of instrumentation and control functions categorised as being of lesser safety relevance.

3.9 Robustness

Interpretation regarding Numbers 3.1 (2) and 3.7 of the "Safety Requirements for Nuclear Power Plants"

3.9 (1) It shall be defined for of instrumentation and control equipment performing functions of category A to C which electrical, electromagnetic, thermal, mechanical and radiation- as well as humidity-induced impacts shall be controlled in such a way that the postulated operational and accident conditions are reliably covered.

3.9 (2) The operational reliability of the instrumentation and control equipment performing functions of category A to C shall not be unacceptably impaired by manual operation and maintenance.

3.9 (3) The instrumentation and control equipment necessary for the execution of the measures within the framework of internal accident management shall be designed such that they will not lose their necessary functions as a result of the consequences of the postulated event sequences or plant states.

3.9 (4) The instrumentation and control installations with functions of category A to C shall be designed such that there are safety margins with respect to ageing effects.

3.9 (5) Plant-specific voltage tolerances shall be considered in the design of the instrumentation and control equipment performing functions of category A to C.

3.9 (6) The instrumentation and control equipment performing functions of category A to C shall be structured in a fault-tolerant manner.

The instrumentation and control equipment performing functions of category A to C shall be designed such that their behaviour upon failure is defined and safety-directed.

3.9 (7) The instrumentation and control equipment performing functions of category A to C shall be designed such that if possible no maintenance work has to be carried out during power operation.

3.10 Electrical power supply of the instrumentation and control equipment performing functions of category A to C

Interpretation regarding Numbers 3.1 (2), 3.1 (3), 3.1 (6) and 3.7 of the "Safety Requirements for Nuclear Power Plants"

3.10 (1) The instrumentation and control equipment performing functions of category A to B as well as the necessary instrumentation and control functions of category C shall be supplied by uninterrupted emergency power supply facilities with energy storage. Assuming that the electrical power demand of one redundant system train will only be covered by the energy storage belonging to that redundant system train, the capacity of the energy storage shall be dimensioned such that the supply will be maintained for at least 2 h without falling below the permissible minimum voltage level. The instrumentation and control equipment and its electrical power supply shall be designed such that following a complete loss of voltage or a drop below the minimum voltage, the instrumentation and control equipment shall be operable again as soon as the voltage has returned.

3.10 (2) In the design of the electrical power supply of the instrumentation and control equipment performing functions of category A to C, the same failure combinations shall be postulated as in the design of the instrumentation and control equipment to be supplied (see for category A Number 3.2 (12) and see for category B: Section 3.3).

3.10 (3) The design of the feeding generating units, the distribution grids and the instrumentation and control equipment shall be synchronised such that the loads postulated for the instrumentation and control equipment and the static and dynamic limits of the permissible supply voltages specified for the instrumentation and control equipment will not be exceeded.

3.10 (4) Failures of the electrical power supply for the instrumentation and control equipment performing functions of category A to C shall be registered and reported by means of monitoring equipment.

4 Qualification

Interpretation regarding Numbers 3.1 (2), 3.1 (3) and 3.7 of the "Safety Requirements for Nuclear Power Plants"

4.1 Qualification of hard- and software of the instrumentation and control equipment for instrumentation and control functions of categories A to C

4.1. (1) In all phases of development, manufacturing, commissioning and operation of the instrumentation and control equipment performing functions of category A to C, administrative, constructive and analytical measures, including practical examinations within the scope of quality assurance, shall be carried out and documented.

4.1 (2) The assessment of the instrumentation and control equipment performing functions of category A to C shall be carried out during the manufacture and assembly process when integrating system components. The system specification and implementation of the individual system components shall be examined whether they meet the relevant instrumentation and control requirements.

4.1 (3) The instrumentation and control equipment performing functions of category A to C shall be comprehensively tested under plant and ambient conditions that are as realistic as possible regarding their capability of controlling all postulated event sequences.

4.1 (4) After assembly in the plant or after any modifications to the instrumentation and control equipment performing functions of category A to C, a commissioning test shall be carried out.

4.1 (5) The information systems shall be qualified according to their safety-significance.

4.2 Qualification of the hardware

4.2 (1) For instrumentation and control equipment performing functions of category A and B, hardware shall be used that is reliable, type-tested or has been proven by operating experience for the postulated ambient conditions. This hardware shall be maintenance-free during operation.

4.2 (2) For instrumentation and control equipment performing functions of category C, hardware shall be used that is reliable and suitable for the postulated ambient conditions.

4.2 (3) The plant-specific suitability shall be demonstrated by a comparison of the characteristics of the hardware of instrumentation and control equipment performing the criteria specified for the case of operation.

4.3 Qualification of the software

4.3.1 Software for instrumentation and control functions of categories A to C

4.3.1 (1) The software shall be developed in verifiable steps according to a phase model.

4.3.1 (2) The software architecture of instrumentation and control equipment shall be designed such that the functions of the application software and the system software are implemented in individual software units and that the application software is separated from the system software.

Note: The system software includes e.g. the operating system and, in case of multi-computer systems, the software for the communication between the different computers.

4.3.1 (3) The software shall be designed such that there will be no unacceptable feedback effects of instrumentation and control equipment with instrumentation and control functions being of a category of lesser safety relevance on the instrumentation and control equipment with instrumentation and control functions being of a category of higher safety relevance.

4.3.1 (4) The software shall be designed such that it runs as required irrespective of the kind and scope of any changes in the time of the input signals.

4.3.2 Software for instrumentation and control functions of category A

4.3.2.1 Principles

4.3.2.1 (1) The development and qualification of the software for instrumentation and control functions of category A shall be such that a consistent demonstration of the correct operation of the software is ensured. Its design and implementation shall be carried out with formalised and computer-based design and test methods according to the state of the art in science and technology.

4.3.2.1 (2) The software for instrumentation and control functions of category A shall be simply structured.

4.3.2.1 (3) The functional scope of the software for instrumentation and control functions of category A shall be limited to the extent necessary for the respective function.

4.3.2.1 (4) The software for instrumentation and control functions of category A shall be designed to be robust. Self-monitoring of the instrumentation and control functions of category A shall be provided.

4.3.2.2 Quality assurance

4.3.2.2 (1) The software shall be programmed consistently with computer-based tools according to a phase model.

4.3.2.2 (2) The software shall be built up from clearly delimited units with little functional scope. These software units shall be programmed such that they are restricted to essential statements and interfaces and shall be integrated in a clear program structure.

4.3.2.2 (3) The results of the individual development phases of the software shall be duly verified on the requirements, using formal analysis methods and additional tests. For this purpose, tests shall be carried out at defined milestones.

4.3.2.2 (4) Following the installation of the software on the computers, the design behaviour of the hardware- and software system shall be validated. If the validation is carried out in several steps, the individual validation steps shall overlap.

4.3.2.2 (5) The organisation and administration of software development and quality assurance shall be such that it is ensured that the software is programmed and used according to complete development, testing, maintenance and quality assurance plans. The independence of construction from quality assurance shall be maintained throughout. Complete development, quality assurance and user documentation shall be provided.

4.3.2.2 (6) Processes and measures shall be applied that ensure the consistent configurations of the software (configuration management).

4.3.2.3 Use of pre-developed software

4.3.2.3 (1) The use of pre-developed software, unless designed in accordance with the requirements of Sections 4.3.2.1 and 4.3.2.2, shall be restricted to indispensable component parts, and software modifications shall be avoided. These component parts shall be subjected to examinations and tests that are equal in scope and depth to those required in Sections 4.3.2.1 and 4.3.2.2.

4.3.2.3 (2) For the assessment of equality, the following shall be considered:

- references about the software producer,
- the development, user and quality assurance documentation of the software,
- the results of independent assessments (certificates) of the software,
- the operating experience with the software, taking user profiles into account, and
- additional software tests.

4.3.3 Software for instrumentation and control functions of category B

4.3.3.1 Principles

4.3.3.1 (1) Descriptions and computer-based test methods shall be used for the development and qualification of the software for instrumentation and control functions of category B that support the demonstration of its correct operation.

4.3.3.1 (2) The software for instrumentation and control functions of category B shall be designed to be robust. Self-monitoring of the instrumentation and control functions of category B shall be provided.

4.3.3.2 Quality assurance

4.3.3.2 (1) The software shall be programmed largely with computer-based tools according to a phase model.

4.3.3.2 (2) The software shall be built up from units with clearly delimited functional scope. These software units shall be programmed such that they are restricted to essential statements and interfaces and shall be integrated in a clear program structure.

4.3.3.2 (3) The individual phases of software development shall be subjected to documented examinations. All safety-relevant program parts shall be tested by a combination of test procedures with the aim of achieving complete functional overlap.

4.3.3.2 (4) The design behaviour of the hardware and software system shall be validated.

4.3.3.2 (5) The organisation and administration of software development and quality assurance shall be such that it is ensured that the software is programmed and used according to complete development, testing, maintenance and quality assurance plans. The independence of construction from quality assurance shall be maintained throughout. Complete development, quality assurance and user documentation shall be provided.

4.3.3.2 (6) The consistent configuration of the programs shall be ensured (configuration management).

4.3.3.3 Use of pre-developed software

4.3.3.3 (1) The use of pre-developed software, unless designed in accordance with the requirements of Sections 4.3.3.1 and 4.3.3.2, shall be restricted to indispensable component parts, and software modifications shall be avoided. These component parts shall be subjected to examinations and tests that are equal in scope and depth to those required in Sections 4.3.3.1 and 4.3.3.2.

4.3.3.3 (2) For the assessment of equality, the following shall be considered:

– references about the software producer,

– the development, user and quality assurance documentation of the software,

– the results of independent assessments (certificates) of the software,

– the operating experience with the software, taking user profiles into account, and

– additional software tests.

4.3.4 Software for instrumentation and control functions of category C

4.3.4.1 Principle

The software for instrumentation and control functions of category C shall be qualified according to the recognised state of the art.

4.3.4.2 Quality assurance

4.3.4.2 (1) The development steps of software programming shall be shown individually. Computer-based tools shall be used in essential development steps if possible.

4.3.4.2 (2) Reaching the phase objectives shall be demonstrated by testing and documented.

4.3.4.2 (3) The design behaviour of the safety-relevant functions of the hardware and software system shall be validated.

4.3.4.2 (4) The software shall be programmed according to a quality assurance plan in line with the generally acknowledged technical rules. A complete development, quality assurance and user documentation shall be provided.

4.3.4.3 Use of pre-developed software

Any pre-developed software used shall be certified or shall be proven in operation. The characteristics necessary for assessing usability shall be documented.

5 Maintenance and modification

Interpretation regarding Numbers 3.1 (2), 3.1 (12), 3.7 and regarding Annex 4 Numbers 3 and 4 of the "Safety Requirements for Nuclear Power Plants"

5 (1) The operability of the instrumentation and control equipment performing instrumentation and control functions of categories A to C shall be verified by tests throughout during the operating lifetime of the plant. These tests shall cover all safety-relevant equipment.

5 (2) The instrumentation and control equipment performing instrumentation and control functions of categories A to C shall be designed such that any changes caused by tests shall be reversed after the tests. Tests may be carried out automatically or manually.

5 (3) It shall be possible to monitor tests on instrumentation and control equipment performing instrumentation and control functions of categories A to C from central locations.

5 (4) If modifications are applied to the instrumentation and control equipment performing instrumentation and control functions of categories A to C, quality standards shall be applied that are at least equal to those applied to the manufacture of the instrumentation and control equipment.

5 (5) If modifications are applied to the instrumentation and control equipment performing instrumentation and control functions of categories A to C, it shall be ensured that the modified parts fulfil their functions and that they will interact as required with the unmodified parts.

5 (6) Any modifications to the software of the instrumentation and control equipment performing instrumentation and control functions of categories A to C shall be carried out in compliance with the quality requirements according to Section 4. Any software modifications and the associated necessary interventions in the instrumentation and control equipment shall be carried out such that the requirements of the "Safety Requirements for Nuclear Power Plants", Annex 4 will be fulfilled. All software interventions shall be documented.

5 (7) Any modifications of the parameter data and the software of the instrumentation and control equipment performing instrumentation and control functions of categories A to C shall be carried out in such a way that they can be traced back.

6 Specific requirements for the documentation relating to instrumentation and control equipment of categories A to C including accident instrumentation

Interpretation regarding Numbers 3.1 (2) and 3.7 and regarding Annex 5 Number 7 of the "Safety Requirements for Nuclear Power Plants"

6 (1) The plant-specific configuration of the hardware and the software of instrumentation and control equipment performing instrumentation and control functions of categories A to C shall be documented regarding its current status and modifications that shall be carried out throughout its entire operating lifetime.

6 (2) The maintenance processes and interventions in the instrumentation and control equipment performing instrumentation and control functions of categories A to C shall be documented.

6 (3) The operating experience from the maintenance of the instrumentation and control installations with instrumentation and control functions of categories A to C shall be documented and systematically evaluated according to the safety-significance of the instrumentation and control equipment.

Interpretation I-4: Requirements for the electrical energy supply

Contents

1 Scope of application

This guidance text contains interpretations relating to the requirements for the electrical power supply of the nuclear power plant.

2 Requirements for the electrical power supply

Interpretation regarding Number 3.9 (1) of the "Safety Requirements for Nuclear Power Plants"

2 (1) The design of the electrical power supply equipment and the design of the connected consumers shall be geared to each other in such a way that the design based loads are not exceeded.
For the design of the electrical power supply short-circuits and ground-faults as well as disconnections in all phases (symmetrical fault) as well as those that only affect one or two phases (asymmetrical fault) shall be taken into account for all operating states.

2 (2) Protection against internal and external electrical impacts shall be designed such that the electrical equipment of the electrical power supply supplying the consumers and equipment executing functions on levels of defence 1 to 4, during internal and natural hazards and as well as in case of human-induced external hazards will not be unacceptably impaired. For levels of defence 4b and 4c, graded requirements apply to the design of this protection in accordance with Number 2.1 (13) of the "Safety Requirements for Nuclear Power Plants".

Interpretation regarding Number 3.9 (2) of the "Safety Requirements for Nuclear Power Plants"

2 (3) The following supply options shall be provided for the electrical power supply of the consumers in a nuclear power plant executing functions on levels of defence 1 to 4a, during internal and natural hazards as well as in case of human-induced external hazards:

a) A main generator that shall keep up the electrical power supply for the functions on levels of defence 1 and 2 even if there are disturbances within the main grid or if there is a failure within the main grid connection

b) A main grid connection that in case of the unavailability of the main generator shall ensure the electrical power supply for the functions on levels of defence 1 to 4a.

c) A standby grid connection that shall ensure the electrical power supply for the functions on levels of defence 1 and 2 which are necessary in case that neither the main generator nor the main grid are available, including shutdown and residual-heat removal via the main heat sink as well as for the functions on levels of defence 3 and 4a.

d) Emergency power generating units within the grounds of the power plant that shall ensure the electrical power supply of the respective necessary emergency power consumers upon failure or unavailability of the supply options mentioned in a) to c), during internal or natural hazards as well as in case of human-induced external hazards.

e) An electrical power supply option (e.g. the emergency power grid connection) that is independent of the supply options mentioned in a) to d) and shall provide at least the electrical power needed for removing the residual-heat with one redundant residual-heat removal train.

External grid connections shall be monitored with suitable equipment with regard to their availability and operability.

2 (4) The electrical power supply option according to Number 2 (3) letter e) as well as the main and standby grid connections shall furthermore be designed such that each by itself shall be capable of ensuring the electrical power supply of the installations provided for accident management measures (levels of defence 4b and 4c).

Moreover, it shall be possible even within accident management measures to switch back the electrical power supply to the main or standby grid connections once they have become available again.

2 (5) Switchover from the main grid connection to the standby grid connection shall be automatic if the electrical power supply from the main generator is not available, the conditions for the power supply from the main grid connection cannot be maintained and the standby grid is available.

2 (6) The start-up and connection of the emergency power generating units shall be automatic in case of demand so that no manual actions will be necessary within 30 minutes. The control of the emergency power generating units shall be designed such that manual start-up and connection of the operable emergency power generating units is possible if needed.

Interpretation regarding Number 3.9 (4) paragraph 1 of the "Safety Requirements for Nuclear Power Plants"

2 (7) In order to carry out the accident management measures on levels of defence 4b and 4c, the necessary electrical power supply shall be provided. This also applies in case of a failure of the non-battery-buffered electrical power supply (i.e. loss of the complete AC power supply with the exception of the AC power supply facilities that are battery-supplied via inverters). In such a case, the electrical power supply shall be ensured for a period of 10 hours even without any external support (i.e. no delivery of operating media such as fuel, lubricants or spare parts) to control the plant state without interruption (e. g. with the help of the accident instrumentation and the safety lighting), to carry out accident management measures, and to be able to establish a plant-internal or -external electrical power supply (e. g. by repair measures or connections with external power plants).

In this context, the capacity of the electrical energy storages shall be matched with the time necessary until the readiness of other power sources (e.g. mobile diesel generators or additional energy storages) in such a way that there will be no unacceptable voltage conditions or interruptions for the above-mentioned period of 10 hours after the onset of the event.

When determining the discharge times of the energy storages, the characteristic plant states of level of defence 4b shall be considered in such an event. Here, graded requirements apply according to Number 2.1 (13) of the "Safety Requirements for Nuclear Power Plants".

Due to the potential destruction of the infrastructure outside the plant, the equipment for the accident management measures required in this case shall be provided within the site. This equipment may include mobile diesel generators or additional energy storages.

Interpretation regarding Number 3.9 (4) paragraph 2 of the "Safety Requirements for Nuclear Power Plants"

2 (8) The re-establishment of the electrical power supply after a failure of the non-battery-buffered electrical power supply shall be understood as the switching back to the main grid or the standby grid, the return to service of the emergency power generating units or the establishment of a supply via another supply option according to Number 2 (3) letter e), such as the emergency power grid connection.

Interpretation regarding Number 3.9 (4) paragraph 3 of the "Safety Requirements for Nuclear Power Plants"

2 (9) In case of a longer lasting unavailability of the grid connections or all external grids, caused by a destruction of the infrastructure across the region, a re-establishment of the electrical power supply from the grid cannot always be guaranteed. If in such a case no power supply option according to Number 2 (3) letter b) or c) is available, an electrical power supply of the necessary equipment shall be provided as a substitute measure by means of a connection via the emergency power grid connection or via another adequate connection according to Number 2 (3) letter e) to external grid substations or power plants. When planning these substitute measures, a supply from external power sources or grids that will be available in time before the 3 days have run out shall be considered. These considerations should also include existing connections to the neighbouring unit or neighbouring power plants, gas turbines in the area close to the plant, as well as mobile diesel generators. Also, the planning should take into account that if needed, the technical conditions for receiving this power as well as a legally binding

warranty about the supply of the electrical power allowing for a priority supply exist.

The external connection points to be provided shall be prepared and hence be available at short notice for the connection to external grids or power plants as well as to mobile electrical power generating units.

Interpretation regarding Number 3.9 (4) paragraph 4 of the "Safety Requirements for Nuclear Power Plants"

2 (10) The electrical power to be provided for the accident management measures shall be sufficient for the residual-heat removal during the plant state described in the 3rd paragraph of Number 3.9 (4) of the "Safety Requirements for Nuclear Power Plants" and during the characteristic plant states of levels of defence 4b and 4c.

To provide this power, a combination of several individual substitute measures or equipment (e.g. connection of 2 mobile diesel generators) is also permissible.

Interpretation I-5 Requirements for structures, systems and components

Contents

1 Scope of application

This guidance text contains interpretations relating to the requirements for installations of the nuclear power plant, i.a. with respect to impacts resulting from a human-induced external hazard, the prevention of multiple failures of safety systems, the application of the single-failure concept as well as precautionary measures in connection with special events.

2 Requirements for the consideration of the vibrations induced by the impact of an aircraft

Interpretation regarding Annex 3 Number 4.2.2.1 of the "Safety Requirements for Nuclear Power Plants"

The vibrations induced by the impact of an aircraft may be considered by means of a calculatory verification on the basis of the building response spectra determined for the impact of an aircraft or by means of a simplified method using conservative equivalent static loads. If the simplified method is applied, the equivalent loads (acceleration in horizontal and vertical direction in the frequency range up to 16 Hz) as well as the permissible component loads shall be determined such that a level of safety is achieved that corresponds to the verification by means of building response spectra.

3 Requirements for the prevention of multiple failures of safety equipment

Interpretation regarding Numbers 3.1 (3) and 3.1 (5) of the "Safety Requirements for Nuclear Power Plants"

3 (1) Suitable precautions shall be taken against safety equipment failures due to a common cause, applying Numbers 3.1 (3) and 3.1 (5) of the "Safety Requirements for Nuclear Power Plants".

3 (2) The necessary degree of diversity shall be determined according to the safety significance of the safety equipment. When determining the degree of diversity, operating experience shall be taken into account. In the determination of the necessary degree of diversity, the requisite auxiliary systems (e.g. electrical energy supply, cooling, lubricant supply, control) shall also be taken into account.

3 (3) If common equipment or procedures such as test equipment, inspection documents, supply or auxiliary systems are used for several redundant system trains, it shall be ensured that possible failure mechanisms in this equipment and the connecting parts or human errors such as operating or maintenance errors will not lead to any effects that will affect other redundant system trains.

3 (4) Maintenance measures shall be organised and devised such that possible human errors will remain limited to one redundant system train. Furthermore, suitable quality assurance measures shall prevent that auxiliary and operating supplies (e.g. fuels, lubricants) are used erroneously or as a result of the installation of unsuitable components (e.g. seals, electrical parts) in several redundant system trains, for example by the lagged installation of new parts in individual redundant system trains. The quality assurance measures shall also be applied to operating supplies, module assemblies and spare parts in stock.

3 (5) The in-service inspections of redundant equipment shall be devised by suitable measures, e.g. lagged inspection of the redundant system trains, in such a way that defects affecting several redundant system trains can be identified and rectified in good time

3 (6) Modifications of redundant equipment or of the operating mode or operating conditions shall if possible not be carried out at the same time in order to avoid defects in several redundant system trains.

4 Requirements for the application of the single-failure concept

4.1 Requirements for the installations for fuel pool cooling

Interpretation regarding Annex 4, Number 2.3 (2) of the "Safety Requirements for Nuclear Power Plants"

regarding the installations for fuel pool cooling, the requirements of Number 2.3 (2) in Annex 4 of the "Safety Requirements for Nuclear Power Plants" also apply to the operating phases A to D.

Note: Operating experience has shown that it is possible to make at least one pool cooling system train available within 10 hours if sufficient maintenance resources (sufficient and qualified maintenance personnel, stocks of spare parts, etc.) are provided at the plant.

4.2 Application of the single failure concept to functions needed in a human-induced external hazard

Interpretation regarding Annex 4, Number 2.4 (2) of the "Safety Requirements for Nuclear Power Plants"

Functions that are demanded within the first 30 minutes and whose effectiveness for achieving and maintaining a controlled plant state is necessary during the 10 hours of self-supporting mode must be ensured even with consideration of a single failure.

4.3 Application of the single-failure concept to the design of precautionary measures

Interpretation regarding Annex 3, Number 2 (3) of the "Safety Requirements for Nuclear Power Plants"

4.3 (1) In the design of precautionary measures, it shall be ensured in analogy to the requirements of Number 1 (3), Annex 4 of the "Safety Requirements for Nuclear Power Plants" that the effectiveness of a precautionary measure will be independent of the random failure of any individual part of the technical equipment or the presence of a maintenance case.

4.3 (2) If the precautionary measure is based wholly or in part in administrative measures, the effectiveness of these precautionary measures shall be demonstrated by the fulfilment of the requirements according to Annex 3 Number 2 (6) of the "Safety Requirements for Nuclear Power Plants".

5 Event-specific requirements relating to event B3-01

Interpretation regarding event B3-01 "Longer-term failure (>30 min) of two trains of the spent fuel pool cooling system" of Annex 2 of the "Safety Requirements for Nuclear Power Plants"

A system train shall be understood to be a complete residual-heat removal chain of the spent fuel pool. Regarding event B3-01, it shall be demonstrated for all operating phases that for compliance with the protection goal "fuel cooling", a limitation of the pool water temperature to levels below the design temperature of the pool is achieved to ensure pool integrity.

The event sets in with the failure of an operational system train during the unavailability of a second train due to maintenance measures.

Note See also Section 4.1 of Interpretation I-5.

6 Requirements for precautionary measures in the case of special events

6.1 Entry of demineralised water or low-borated coolant into the reactor core

Interpretation regarding events D3-19 and D3-20 "Entry of demineralised water or low-borated coolant into the reactor core" of Annex 2 of the "Safety Requirements for Nuclear Power Plants"

6.1 (1) Measures and installations shall be provided that ensure that any reactivity changes due to an entry of demineralised water or low-borated coolant into the reactor core will be limited to levels at which in the case of an initially critical reactor the safety-related acceptance target for the reactivity accident according to the "Safety Requirements for Nuclear Power Plants" Annex 2, Table 3.1b and in the case of an initially subcritical reactor the required amount of shutdown reactivity according to the "Safety Requirements for Nuclear Power Plants" Annex 2, Table 3.1a, is maintained.

6.1 (2) Possible sources of an entry of demineralised water, the potentially entering amounts of demineralised water and the possible effects on the reactor core shall be analysed for all operating phases. Here, the following sources of demineralised water shall be considered in particular:

a) external sources of demineralised water:
   - all systems containing demineralised water that are connected to the reactor coolant system,
   - heat exchanger leakages (steam generators, after-coolers) and
   - low-borated media in adjacent systems and vessels.

b) internal sources of demineralised water:
   - deboration of the coolant in the case of a "small leak" (reflux condenser mode) and
   - cooldown during natural recirculation mode and simultaneous secondary-side steam generator isolation.

6.1 (3) The analysis of possible entry paths for demineralised water shall also consider human errors (e.g. leaving the valve of the system preventing the entry of demineralised water erroneously open, connection of pumps e.g. following reflux condenser mode, failure to carry out checks of the boric-acid concentration in vessels) unless these can be excluded through preventive measures.

6.1 (4) Impermissible injections of demineralised water from external sources shall be prevented e.g. by the following measures and installations:

a) reliable closing and sealing of all valves via which demineralised water may inadvertently reach into the reactor coolant system,

b) monitoring of the boron concentration in adjacent systems and components,

c) direct (and not calculated) automatic continuous monitoring of the boron injection concentration,

d) sampling from vessels (e.g. refuelling water storage tank) to ensure the correct boric-acid concentration,

e) requirements in the operating procedures.

6.2 Leaks in the reactor cavity or the setdown pool with opened pool hatch

Interpretation regarding event B3-03 "Leaks in the reactor cavity or the setdown pool with opened pool hatch" of Annex 2 of the "Safety Requirements for Nuclear Power Plants"

Possible precautionary measures for this event are e.g.:

a) travel-dependent interlocks for hoisting gear,

b) travel path limiters for suspended loads that if dropped may lead to the initiation of non-controllable events,

c) prevention of inadvertently open vales or apertures (e.g. drains) by interlocking or administrative measures.

7 Requirements for structures, systems and components

7.1 General requirements for specific installations

Interpretation regarding Number 3.1 of the "Safety Requirements for Nuclear Power Plants"

7.1 (1) The design of the structures, systems and components shall be based on load cases postulated as a result of defined hazards. The load cases shall be derived in particular from the specified normal operation of the plant including the inspections, from operating experience and from the postulated events, internal and natural hazards as well as human-induced external hazards according to the

"Safety Requirements for Nuclear Power Plants", Annex 2 and Annex 3 and must cover the resulting impacts. The load cases and their combinations shall be specified and fully described according to their characteristics and frequencies.

Load case combinations shall always be postulated if the events or operating phases may have a causal relation or if their simultaneous occurrence must be assumed on the basis of probability considerations. The impacts resulting from these load cases shall be described component-specifically with consideration of the systems engineering also of adjacent systems and of the temporal distribution as well as of the load shedding of the supporting structure.

7.1 (2) All relevant impacts from internal or natural hazards and from human-induced external hazards on the safety-significant installations with the resulting mechanical, chemical, radiological and thermal impacts, corrosion and erosion shall be considered in the design, construction, calculation and maintenance.

Depending on the ambient conditions, the surfaces of metallic materials must be sufficiently protected against corrosion and must be easy to decontaminate. To prevent corrosion, the surfaces of austenitic materials shall be protected from contact with ferritic materials or chloride-containing agents from the construction and operation of the plant.

7.1 (3) Piping areas in which pressure may build up between closed valves as a result of the heat-up of the medium shall be secured against overpressure failure by suitable installations or measures.

7.1 (4) Boundary conditions that result from the requirements of radiation protection, e.g. for the planning of maintenance measures, shall be taken into account.

7.1 (5) All components shall be identification-marked systematically.

7.2 Requirements for building structures

Interpretation regarding Number 3.5 of the "Safety Requirements for Nuclear Power Plants"

7.2 (1) In the case of the postulated impacts, building structures have to remain in a fit-for-use or at least load-bearing condition to the extent required by their safety significance. Additional to the maintenance of the load-bearing capacity, necessary deformation limits and crack width limits have shall be kept so that the structures can fulfil safety-related functions.

7.2 (2) The design, function and layout of the building structures shall consider as decisive design requirement the maintenance of the functional performance of safety-relevant installations to control events on levels of defence 2 to 4a and during internal and natural hazards as well as during human-induced external hazards.

7.2 (3) As a basis for the structural design, all impacts on the building structures shall be described and quantified in such a way that they can be used as unambiguous requirement for the design and construction of the building structures, including the anchoring structures for components. The design shall take possible impacts such as subsidence or the caving-in of mines into account.

Note: See also Number 4.2 as well as Annex 3, Number 3.1 of the "Safety Requirements for Nuclear Power Plants".

7.2 (4) The building structure interaction loads of the plant components shall be transmitted safely into the building structure via anchoring and mounting structures and shall be shed by the former. The building structure interaction loads of the plant components shall be indicated for the interfaces between anchorage and component.

7.2 (5) The mutual influence of buildings shall be limited such that the installations housed in them or the buildings themselves will fulfil their safety-related functions.

7.2 (6) Subsidence of the building structures must not lead to the condition that the fitness for use of the building structures or the functions of safety-relevant installations will be impaired. Differential subsidence shall be considered in the routing of cables and piping between building structures.

7.2 (7) Safety-relevant building structures shall be protected by corresponding sealing measures against water entering from outside. For this purpose, watertight structures or external structural waterproofing shall be provided. The structural waterproofing shall be designed in particular against impacts resulting from groundwater, flooding, earthquakes as well as plant-internal events including ionising radiation.

7.2 (8) For the retention of radioactively contaminated liquids, no credit must be taken on levels of defence 1 and 2 of any external structural waterproofing. As for events on level of defence 3, the existence of functioning external structural waterproofing may be taken into account regarding the leaking of radioactively contaminated liquids as a supplement to the internal retaining functions.

7.2 (9) Regarding dimensions and choice of material, the building structures shall be designed such that they ensure a shielding effect that is in line with the requirements of the radiation protection ordinance.

7.2 (10) Surfaces of rooms in which contamination needs to be reckoned with shall be designed such that they are easy to decontaminate.

7.2 (11) In the rooms where components that contain radioactive waters are installed, floor drains shall be provided.

7.2 (12) The building structures shall meet the criteria and requirements placed on them over the entire period of their use.

7.2 (13) Inspection and monitoring measures shall be provided, at least in the form of regular walkdowns and visual checks of the component part surfaces and anchorages. The results shall be documented. A report about the condition of the building structures shall be prepared every ten years.

7.3 Requirements for the design of reactor pressure vessel internals

Interpretation regarding Number 3.1 of the "Safety Requirements for Nuclear Power Plants"

Note: In the following, reactor pressure vessel internals are understood in particular to be:

- in a PWR:
  - upper and lower core structure
- in a BWR:
  - core shroud,
  - upper and lower core structure,
  - control rod guide tubes,
  - steam-water separator,
  - steam dryer,
  - feedwater sparger.

7.3 (1) The design of the reactor pressure vessel internals shall consider all mechanical, thermal, chemical and radiation-induced impacts that can occur during normal specified operation of the plant as well as during events on levels of defence 2 to 4a and during internal or natural hazards and human-induced external hazards.

7.3 (2) The reactor pressure vessel internals shall be designed such that during events on levels of defence 2 to 4a and during internal or natural hazards and human-induced external hazards, fulfilment of the safety-related acceptance targets and acceptance criteria for these levels of defence is ensured.

It shall be ensured in particular that in consequence of events on level of defence 3 and during internal or natural hazards and human-induced external hazards, the option for mechanical shutdown (in the case of a large leak accident in a PWR, the option for permanent shutdown) and the coolability of the reactor core will be maintained.

Note: Regarding the verification scope concerning leaks larger than 0.1 F, see "Safety Requirements for Nuclear Power Plants", Annex 2, events D3-24 and S3-19 as well as Appendix 2.

7.3 (3) The reactor pressure vessel internals must withstand all loads arising during normal specified operation (level of defence 1) during their entire service life in such a way that the it is ensured that the normal operating conditions of the reactor core will be maintained.

7.3 (4) Suitable measures and installations shall be provided to prevent that reactivity control or fuel cooling will be impaired by impurities or loose parts in the coolant.

7.3 (5) Measures and installations shall be provided for operational vibration and loose-parts monitoring to an extent that is justified from a safety-related point of view.

7.3 (6) Inspections of the reactor pressure vessel internals shall be provided with regard to the occurrence of any damage and the maintenance of the operability of the internals in accordance with the requirements.

7.4 Requirements for the emergency core cooling and residual-heat removal systems

Interpretation regarding Number 3.3 of the "Safety Requirements for Nuclear Power Plants"

General requirements

7.4 (1) In a PWR, it must be possible to flood the cavity around the reactor pressure vessel at least up to the upper core edge in the event of a leak in the reactor pressure vessel.

Requirements for ensuring emergency coolant supplies

7.4 (2) In a PWR, the amounts of emergency coolant supplies shall be such

a) that in case of demand, coolant make-up by high-pressure injection will be possible for as long as it takes until the as the primary system has reached a pressure level through corresponding measures (e.g. secondary-side primary system cooldown, successive switch-off of individual safety injection pumps in the case of small leaks) that allows coolant make-up through low-pressure injection (from refuelling water storage tanks or the reactor building sump);

b) that following the injection of the emergency coolant supplies, event in the event of the most unfavourable leakage with consideration of dead volumes in the containment, assured suction of the low-pressure reverse flow from the containment sump will be possible and long-term heat removal will be ensured.

7.4 (3) In a BWR, the amounts of emergency coolant supplies shall be such that coolant make-up will always be sufficient and assured suction of the low-pressure reverse flow from the containment sump with consideration of dead volumes will be possible and long-term heat removal will be ensured.

7.4 (4) As regards leaks in the emergency core cooling and residual-heat removal system (PWR and BWR) in any location outside the containment, the water supply for emergency core cooling must remain sufficient.

Note: If preventive measures according to the requirements of the "Safety Requirements for Nuclear Power Plants", Annex 2, Number 2, have been realised in the emergency core cooling and residual-heat removal system (PWR and BWR) in any location outside the containment, the failure of the latter need not be postulated.

Requirements for the emergency core cooling systems and the functionality of the containment

7.4 (5) Regarding PWRs, the characteristic for the high-pressure injection system shall be defined such that the core can be kept covered through coolant injection even under a maximum primary-side saturation pressure to be postulated after reactor trip as a result of a reliable secondary-side heat removal.

7.4 (6) It must be possible to maintain the active components of the residual-heat removal systems that are essential for their effectiveness during a long-term period of residual-heat removal operation.

7.4 (7) It shall be ensured by the design of the containment and its internals that in the event of a loss-of-coolant accident, the coolant flowing out from the break location will reach the containment sump (PWR, BWR) or the pressure suppression pool (BWR) in sufficient amounts to ensure cavitation-free operation of the residual-heat removal pumps.

7.4 (8) The emergency core cooling system shall be designed such that in the event of a loss-of-coolant accident, a long-term containment temperature and pressure increase after the replenishment of the core in "sump operation mode" is prevented.

Requirements for secondary-side heat removal

7.4 (9) The water supply for emergency injection shall be calculated conservatively with regard to the accidents to be postulated.

The water supply must be sufficient for removing the decay heat over 10 hours (human-induced external hazards), including the removal of the stored heat. Additional amounts of water needed for the cooling of rooms and components shall be taken into account in the determination of the water supply. It must be possible to provide in time a sufficient water supply for subsequent event control including cooldown.

7.5 Requirements for ventilation systems

Interpretation regarding Number 3.1 of the "Safety Requirements for Nuclear Power Plants"

Note: Requirements that concern radiology are contained in Interpretation I-8 "Requirements for radiation protection".

The ventilation systems also fulfil important fire protection functions. This includes the separation of fire sections by fire dampers and the smoke exhaust. Corresponding requirements can be found in Annex 3, Number 3.2.1 of the "Safety Requirements for Nuclear Power Plants".

7.5 (1) The nuclear power plant shall dispose of reliable and effective ventilation systems for the following rooms:

– Rooms in which the limits specified for the room air conditions (e.g. pressure, temperature, humidity) for normal specified operation (levels of defence 1 und 2), for accident control, for internal and natural hazards as well as human-induced external hazards must not be transgressed and where this is not possible without ventilation systems.

– Rooms housing safety-relevant installations for accident control that need to be air-cooled when challenged or which have to have air guided to them for the operation of diesel generator sets.

– Rooms in which the air is replaced by an inert gas or in which certain room air conditions have to be maintained for reasons of occupational health and the ability of persons to act.

7.5 (2) The ventilation systems shall be designed and harmonised with the properties of the other systems such that on levels of defence 1 to 4a, the respective limits specified as permissible for the room air conditions are adhered to in internal and natural hazards as well as in human-induced external hazards.

7.6 Requirements for the pressure limitation and pressure relief of the reactor coolant system and the main-steam system

Interpretation regarding Number 3.4 (5a) of the "Safety Requirements for Nuclear Power Plants"

General requirements for the pressure limitation and pressure relief of the reactor coolant system and the main-steam system

7.6 (1) The installations for pressure limitation shall ensure that in events on levels of defence 2 to 4a, in internal and natural hazards as well as in human-induced external hazards, the maximum permissible loads to be postulated in these cases for the systems and components to be secured according to the "Safety Requirements for Nuclear Power Plants" Annex 2, Appendix 1 will not be exceeded.

7.6 (2) Under the postulated conditions of levels of defence 2 to 4a and under the conditions of internal and natural hazards as well as of human-induced external hazards, the installations for pressure limitation shall open and close reliably.

7.6 (3) In events on level of defence 2, the response pressure of the reactor coolant system overpressure protection must not be reached.

7.6 (4) The states of matter of the medium to be removed that may result from the events to be controlled by the pressure limitation systems shall be considered.

7.6 (5) The valves shall be qualified in terms of the respective expected bypass conditions (e.g. states of matter, phase mixtures).

7.6 (6) Reliable pressure relief installations shall be provided for boiling water reactors and for the secondary side of the pressurised water reactors.

7.6 (7) Pressure limitation systems shall be regularly subjected to function testing. The aim of the organisation and time management of the inspection shall be that the reliable operation of the installation over the entire maintenance interval is achieved.

7.6 (8) The response of the pressure limitation and pressure relief installations of activity-retaining systems must not lead to a release of radioactive materials into the building atmosphere.

Specific requirements for primary-side pressure limitation in pressurised water reactors

7.6 (9) Relief valves shall be fitted with an upstream isolating vale that will close automatically if the relief valve remains inadvertently in open position. In order to exclude the inadvertent isolation of the installations for pressure limitation, installations shall be provided that will assume the pressure limitation function in the event of an inadvertent isolation irrespective of the relief valves (and their excitation).

7.6 (10) To ensure brittle fracture resistance, the response pressure of the installations for reactor coolant system pressure limitation shall be adapted to the temperature level of the system to be protected.

7.7 Requirements for the pressure suppression system (BWR)

Interpretation regarding Number 3.4 (5a) of the "Safety Requirements for Nuclear Power Plants"

7.7 (1) The design of the pressure suppression system shall consider all loads resulting from levels of defence 1 to 4a, from internal and natural hazards was well as from human-induced external hazards (especially also the dynamic loads). The containment system, consisting of drywell and pressure suppression pool, shall be designed such that the function of the pressure suppression pool regarding pressure reduction and relief is ensured without consideration of the spray system of the pressure suppression pool. Leak-tight isolation between drywell and pressure suppression pool shall be ensured.

7.7 (2) No components must be installed within the pressure suppression pool whose failure may impair the operability of the pressure suppression system.

7.7 (3) After successful equalisation of pressure, the isolation devices in the connections between pressure suppression pool and drywell shall close automatically and reliably and shall be sufficiently leak-tight. Their leak-tightness shall be verifiable. The isolation devices provided for the equalisation of pressure following loss-of-coolant accidents may respond in operational pressure equalisation processes.

7.7 (4) Condensation and clearing processes in the pressure suppression pool must not cause any impermissible impacts.

7.7 (5) It shall be demonstrated that in the event of a loss-of-coolant accident, no negative pressure can establish itself in the drywell compared to the pressure suppression pool that will its function or the steel liner and its anchorage at risk.

7.7 (6) Any responses during loss-of-coolant accidents and blow-off processes must not induce any impermissible building vibrations.

7.8 Requirements for support structures, mounting supports and platforms

Interpretation regarding numbers 3.1 and 3.5 of the "Safety Requirements for Nuclear Power Plants"

Note: The pieces of equipment considered here include supports, suspensions, cable racks, pipe whip restraints, crane runways, platforms and protective structures with safety relevance.

7.8 (1) Support structures, mounting supports and platforms with safety relevance shall be able to transfer the specified loads to the load-bearing structure.

7.8 (2) The collective of all impacts and the resulting loads on the safety-relevant support structures, mounting supports and platforms shall be registered in their entirety and considered in the design of these pieces of equipment. This may include: dead weight, working loads, lifting gear loads, settlement of buildings, test loads, construction loads, internal and natural hazards and human-induced external hazards (especially induced vibrations, impact loads, impacts from anticipated operational occurrences and from incidents).

7.8 (3) Moveable parts of safety-relevant support structures (for example spring hangers, snubbers and dashpots, dampers) shall be subjected to in-service inspections. Rigid components shall be subjected to regular visual inspection, and non-destructive tests shall be carried out if necessary.

7.8 (4) Temporarily erected platforms and support structures for or in the vicinity of safety-relevant installations that have to be provided according to the plant operating procedures in the corresponding operating condition shall be secured such that they will not lose their stability as a result of operating states and events on levels of defence 1 to 4a, during internal or natural hazards as well as under human-induced external hazard conditions or that the loss of stability will not lead to any impermissible hazards.

7.8 (5) The possible crash of component parts during the assembly and disassembly of the temporary equipment as well as the possible crash of parts resting on it with the consequence of a possible risk to safety-relevant installations shall be taken into account.

7.9 Requirements for valves

Interpretation regarding number 3.1 of the "Safety Requirements for Nuclear Power Plants"

7.9 (1) If valves form part of the reactor coolant pressure boundary, of the pressure envelope of the external systems of the containment system, the "Safety Requirements for Nuclear Power Plants" Numbers 3.4 and 3.6 as well as the requirements of Interpretation I-2 "Requirements for the implementation of the reactor coolant pressure boundary, the external systems as well as the containment" shall be fulfilled.

7.9 (2) All parameters that are relevant for the design function of safety-relevant valves, such as loads, stresses, friction and material properties, shall be considered in the design of the drive, the valve body and the parts in the load path in such a manner that even in a combination of the ranges of variation of the individual parameters, functioning is ensured with a sufficient safety margin. Here, the holding device and bearing shall also be considered.

7.9 (3) The operability of valves which in events on levels of defence 2 to 4a in the case of a leak have to open or close against the maximum possible differential pressure prevailing under the respective conditions shall be verified.

7.9 (4) In the case of an de-energising failure, the integrity of valves of the containment shall be maintained. Any further-going safety-related requirements (e.g. for operability) shall be specified case by case.

7.9 (5) It shall be ensured by suitable measures, such as by safety valves or constructive implementation, no impermissible pressure can build up in the housings of safety-relevant isolating valves.

7.10 Requirements for safety-relevant pumps

Interpretation regarding Number 3.1 of the "Safety Requirements for Nuclear Power Plants"

Note: Regarding pump housings that are part of the reactor coolant pressure boundary or which for other reasons are allocated to the scope of Interpretation I-2 "Requirements for the implementation of the reactor coolant pressure boundary, the external systems as well as the containment", the requirements referred to therein shall apply.

7.10 (1) Pumps have to fulfil their function reliably under the conditions of the levels of defence to which they are assigned and, in line with their tasks, also during all internal or natural hazards as well as during the human-induced external hazards specified for the respective functions.

7.10 (2) The drive units shall be suitable for the ambient conditions. They shall have the necessary motor power as well as the torque necessary upon start-up and during maximum power operation. The vibration transfer from the pump to other components shall be considered. The drive units shall be supported and fixed correspondingly.

If steam turbines or diesel engines are used as drive units, the requirements for these components shall be considered.

7.10 (3) Both Gears and clutch shall transmit the necessary torques reliably. Both gears and clutch, including their cooling and lubrication, shall fulfil their function under the expected operating and ambient conditions.

7.10 (4) Pumps shall be fitted with devices with whose help relevant operating parameters, e.g. pressures, flow rates, temperatures and vibrations, can be monitored.

7.10 (5) Radiation protection requirements, such as leak-tightness to the outside and ease of decontamination, shall be considered.

7.11 Requirements for safety-relevant heat exchangers

Interpretation regarding Number 3.1 of the "Safety Requirements for Nuclear Power Plants"

Note: If the heat exchangers are part of the reactor coolant pressure boundary or of the pressure envelope of the external systems, the requirements of Interpretation I-2 "Requirements for the implementation of the reactor coolant pressure boundary, the external systems as well as the containment" shall also apply.

7.11 (1) Heat exchangers shall fulfil the safety-related requirements regarding energy transfer and barrier and retaining function under all specified boundary conditions. Here, not only the conditions of normal specified operation (levels of defence 1 and 2), of incidents, internal and natural hazards as well as human-induced external hazards shall be considered but also special boundary conditions in connection with maintenance measures (e.g. heat input with cooling water side isolated).

7.11 (2) The relevant mechanical and thermal loads, especially fast mechanical and thermal as well as cyclical loads, shall be considered in the design of heat exchangers.

7.11 (3) It shall be ensured that no medium or foreign matter can accumulate in the heat exchangers that will impermissibly impair the essential safety-related heat transport or the integrity of the heat exchanger surface area. Here, the special conditions of incidents, internal and natural hazards as well as human-induced external hazards shall also be considered.

7.11 (4) A monitoring programme for safety-relevant heat exchangers shall be provided to ensure the parameter values that are essential for the energy transfer. Continuous monitoring of the relevant values and the triggering of an alarm if safety-relevant design parameters like flow rates and heat transfer rate are not adhered to shall be especially provided for those heat exchangers that are at risk of possible discontinuous impacts (e.g. foreign-body input, discontinuous dirtying effects). Here, accidental impacts, internal and natural hazards as well as human-induced external hazards shall also be considered (e.g. entry of insulating material during loss-of-coolant accidents).

7.11 (5) Heat exchangers which besides the function of energy transfer have an important retaining function shall be monitored for leakages between the circuits. The permissible leakage amounts shall be specified in the operating procedures.

7.11 (6) The condition of the heat exchanger tubes shall be examined to the necessary extent as part of the maintenance programme with consideration of any relevant damage mechanisms.

7.12 Requirements for piping and vessels

Interpretation regarding Number 3.1 of the "Safety Requirements for Nuclear Power Plants"

Note: If piping or vessels are part of the reactor coolant pressure boundary or of the pressure envelope of the external systems or of the containment system, the requirements of Interpretation I-2 "Requirements for the implementation of the reactor coolant pressure boundary, the external systems as well as the containment" shall also apply.

7.12 (1) Piping and vessels shall reliably fulfil the safety-related requirements regarding the confinement of radioactive materials and with respect to their functions as pressurised components under all specified boundary conditions.

7.12 (2) Piping and vessels in which an inner pressure may build up (for example due to heated-up medium confined within) protected against impermissible inner pressures.

7.12 (3) besides the loads arising from inner pressures and static loads, dynamic loads such as forced oscillations and thermal expansions shall be considered in the design, assembly and installation.

7.12 (4) The boundary conditions, such as accessibility, that ensue from the execution of maintenance measures shall be considered.

7.12 (5) Safety-relevant piping and vessels shall as a rule be routed, arranged and equipped in such a way that their design filling, deaerating and draining is possible and that hence no water hammer can occur. If this cannot be excluded with sufficient reliability, these impacts shall be considered in the design.

7.12 (6) Buried safety-relevant piping or vessels must not lose their integrity e.g. due to corrosion or loads arising due to subsidence. Their location shall be documented.

7.12 (7) Any transport of fuel assembly transport or storage casks within the plant shall be along short and safe paths and without any unnecessary stops on a specified transport route. Any transports over safety-relevant pieces of equipment shall be avoided.

7.12 (8) The transport paths of the fuel assembly transport and storage casks shall be devised such that the casks will not be subjected to any impermissible impacts.

7.12 (9) Regarding the design of the inner and outer surfaces, requirements for ease of contamination, corrosion and wear protection shall be considered.

7.13 Requirements for electrical drives

Interpretation regarding Number 3.1 of the "Safety Requirements for Nuclear Power Plants"

7.13 (1) The electrical drives fulfilling functions on levels of defence 1 to 4a, during internal and natural hazards as well as in human-induced external hazards shall also fulfil their functions u der the postulated ambient conditions, process-based loads, and electrical conditions. These requirements shall also apply to those drives that fulfil functions in the installations provided for accident management.

Note: In 7.13 (1), the installations that are provided, i.e. pre-planned, for accident management according to the "Safety Requirements for Nuclear Power Plants", Number 3.1(10) are mentioned. It is assumed that regarding the pre-planned installations, which are also mentioned in the emergency manual, knowledge exists about the ambient conditions likely to prevail.

7.13 (2) The protective devices of the electrical drives shall be aligned with the drives to be protected and the electrical energy supply in such a way that the components are safely protected and a sufficient safety margin exists to the most adverse operating modes of the electrical energy supply. Any response of protective devices shall be signalled.

7.13 (3) Devices of equipment unit protection shall be designed such that the equipment unit protection will as a rule not come into operation when electrical drives are challenged by the safety system.

7.13 (4) As regards the electrical drives of valves, output, moment and power reductions due to self-heating, increased ambient temperature and voltage drop upstream of the drive itself shall be considered.

8 Other requirements

8.1 Requirements for escape and rescue routes and for alarming

Interpretation regarding Number 3.8 of the "Safety Requirements for Nuclear Power Plants"

8.1 (1) Escape and rescue routes shall be provided along which persons can quickly and safely get outside or be rescued from outside in the event of danger.

8.1 (2) The layout, dimensions and implementation of the escape and rescue routes shall be guided by the purpose, furnishing and floor area of the rooms as well as by the number of persons usually present in the rooms.

8.1 (3) As a rule, redundant alarm systems shall be installed which give optical or acoustic signals. Redundant alarm systems may be dispensed with in sub-areas if a single failure will not prevent the signalling of the alarm. Signalling must be within the buildings and on the plant premises.

8.1 (4) The personnel shall regularly be instructed about the meaning of the alarm signals, about how to behave upon perceiving a signal, and how to use rescue and personal protection gear.

8.1 (5) Alarm and rescue exercises shall be carried out at regular intervals. External rescue organisations shall be involved in the exercises at appropriate intervals or when necessary.

8.1 (6) Communication facilities shall be provided at the plant in sufficient numbers for informing the main control room about a plant hazard state as well as for initiating rescue operations.

8.1 (7) Plant- and accident-specific criteria for the kind and tripping moment of the specified alarms – also including automatically tripped alarms as the case may be – shall be established and the necessary actions of the personnel shall be planned. These actions shall be exercised at least at intervals of six months.

8.1 (8) It shall be ensured by measures and installations that the personnel will have sufficient time for escaping upon a response of the safety valves inside the containment (especially before the response of the rupture disks of the pressuriser relief tank) or that sufficient protection under the then occurring conditions is guaranteed.

8.2 Requirements for the design of the working environment and working appliances

Interpretation regarding Number 3.1 (13) of the "Safety Requirements for Nuclear Power Plants"

8.2 (1) The ergonomic design of the working environment and working appliances shall be demonstrated with suitable assessment methods. Verification shall be repeated at regular intervals. Simulations, experiments, experience with similar situations or established scientific assessment systems may be used for the assessment.

8.2 (2) The internal communication of the nuclear power plant and the communication to the outside that is necessary for the safety of normal specified operation (levels of defence 1 and 2), for the control of events on level of defence 3, levels of defence 4b and 4c and in the event of internal and natural hazards and human-induced hazard conditions shall be ensured at any time.

8.2 (3) Essential functional plant changes as well as ergonomic modifications in the control room should be reviewed prior to the implementation of the modification, e.g. by means of a simulator. Prior to taking up work on the changed system and especially in the control room, the personnel shall be instructed to the necessary extent.

Interpretation I-6: Requirements for the handling and storage of the fuel assemblies

Content

1 Scope of application

This text contains interpretations regarding the requirements for handling and storage of fuel assemblies.

2 Requirements for reactivity control in connection with the handling and storage of fuel assemblies

Interpretation regarding Numbers 3.10 (1) and 3.10 (2) of the "Safety Requirements for Nuclear Power Plants"

2 (1) Fuel assemblies may only be stored in the positions or areas earmarked for them in the storage facilities.

2 (2) The incorrect positioning of a fuel assembly in the spent fuel pool or the incorrect loading into the reactor core shall be prevented by

- the quality-assured planning of the reshuffling processes,
- quality-assured measures during the reshuffling processes,
- refuelling machine control systems in keeping with the requirements,
- the creation of conditions for the operation of the handling equipment that are in keeping with the tasks, and
- reliable communication between all involved.

It shall be ensured that the required subcriticality might only be violated if at least two errors or erroneous actions occur that are independent of each other, have a simultaneous effect, and are not expected during normal specified operation (levels of defence 1 and 2).

3 Requirements for the cooling of the fuel assemblies in the spent fuel pool

Interpretation regarding Number 3.10 (3) of the "Safety Requirements for Nuclear Power Plants"

3 (1) The spent fuel pool shall be designed such that any coolant losses from the pool that will lead to a failure to meet the safety-related acceptance targets and acceptance criteria of the protection goal "fuel cooling" of level of defence 3, see Annex 2 of the "Safety Requirements for Nuclear Power Plants", Table 3.2, are excluded. This shall also apply to internal or natural hazards as well as to human-induced hazard conditions.

3 (2) Installations shall be provided that will remove the residual heat reliably and in keeping with the requirements from the spent fuel pool under operating conditions and in events on levels of defence 1 to 3, during internal or natural hazards as well as under human-induced hazard conditions, also considering all operational circumstances of refuelling, if necessary also the simultaneous need for cooling the fuel assemblies in the reactor core and during maintenance measures.

The respective permissible temperature limits must not be exceeded, not even if the spent fuel pool is occupied to its permissible maximum including a fully unloaded core.

3 (3) It shall be possible to monitor the coolant level and the coolant temperature in the spent fuel pool from the control room.

3 (4) A systems-related connection between the residual-heat removal system or the core cooling system with the spent fuel pool cooling system shall only be permissible if disturbances in the pool cooling equipment will demonstrably not lead to any noteworthy impairment of the reliability of the residual-heat removal and emergency core

cooling system. The valves to be operated for a switch-over to spent fuel pool cooling shall be located outside the containment unless otherwise feasible or if this is detrimental from a safety-related point of view.

3 (5) If the systems of the residual-heat removal system or the core cooling system are connected with the spent fuel pool cooling system, an additional cooling train for spent fuel cooling shall be available which can on its own cool the spent fuel pool after a loss-of-coolant accident in the reactor coolant system. If possible and expedient, this train shall have no active equipment inside the containment. If possible and expedient, valves that have to be operated to take this train into operation shall be located outside the containment.

3 (6) The spent fuel pool shall have such sufficient storage capacities that it is guaranteed at any time that the reactor core can be fully accommodated in it with the corresponding requisite number of vacant positions in the spent fuel pool. Setdown positions that can be inserted in the spent fuel pool at short notice may also be taken into account.

4 Requirements for radiation protection during the handling and storage of fuel assemblies

Interpretation regarding Number 3.11 (4) of the "Safety Requirements for Nuclear Power Plants"

4 (1) Equipment shall be provided for the inspection of fuel assemblies and for the verification of compliance with the radiological requirements in case of any operationally induced fuel rod damage.

If defective fuel rods are temporarily stored in the spent fuel pool, it shall be ensured that there will be no noteworthy additional contamination of the coolant.

Interpretation I-7: Requirements for accident management

Content

1 Scope of application

This text contains fundamental requirements for the planning and safety demonstration of measures and equipment as well as the staffing-related and organisational prerequisites of internal accident management on levels of defence 4b and 4c.

2 Plant states on levels of defence 4b and 4c

Interpretations regarding Numbers 2.1 (9), 2.1 (10) and 2.1 (13) of the "Safety Requirements for Nuclear Power Plants"

2 (1) Level of defence 4b is characterised by beyond-design-basis plant states that are governed by failures of safety equipment to an extent that the effectiveness of safety equipment that is necessary for the control of accidents as per design is no longer given (multiple failures of safety equipment). Beyond-design-basis plant states of level of defence 4b are e.g.:

PWR:

- total loss of steam generator feed water supply with a tendency to the complete dryout of the secondary side of the steam generators,
- loss of coolant with small leak cross-section with a tendency to coolant pressure remaining above the pressure head of the high-pressure injection pumps,
- double-ended break of a steam generator tube and rise of the main-steam pressure with a tendency to open the main-steam safety valve.

BWR:

- loss of coolant with subsequent overfeeding of a main-steam line and the possibility of water hammer downstream the steam line isolation valves,
- transients with a tendency of the reactor pressure vessel water level falling below the core bottom.

PWR and BWR:

- loss of the entire AC power supply – unless battery-supported – for a period of up to 2 hours.

2 (2) Accident management measures on level of defence 4c shall consider phenomena of accidents with severe fuel assembly damage. Such phenomena are characterised e.g. by

- reactor pressure vessel failure at low pressure,
- hydrogen release,
- containment pressure build-up.

3 Requirements for the effectiveness of accident management measures

Interpretations regarding Numbers 2.1 (9), 2.1 (10) and 2.1 (13) of the "Safety Requirements for Nuclear Power Plants"

3 (1) The effectiveness of the accident management measures on level of defence 4c shall be such that the requirements made in the "Safety Requirements for Nuclear Power Plants", Number 2.1 (3b) can be fulfilled.

3 (2) For the beyond-design-basis plant states on level of defence 4b and phenomena during accidents with severe fuel assembly damage on level of defence 4c, criteria shall be defined for the selection,

preparation, execution and control of the effectiveness of the accident management measures. The selection of the emergency measures to be applied in case of an accident shall be guided by the respective plant state.

3 (3) Emergency measures shall be planned such that the times necessary for their preparation and execution are shorter than the times assumed to be available from the event sequences that have been determined as being representative of the beyond-design-basis plant states and on which the planning has therefore been based. This is to achieve that any necessary action steps may be repeated if need be.

3 (4) The equipment provided for accident management measures including the necessary supply systems shall be designed such that regarding the postulated event sequences or plant states, the emergency measures

- will be effective under the expected conditions on levels of defence 4b and 4c, and
- are as easily manageable as possible under the special condition of the severe accident situation.

3 (5) There are no requirements regarding the application of the principles of redundancy, diversity, segregation and physical separation of the equipment provided to accident management measures. Neither a single failure nor a maintenance case need be assumed (degree of redundancy n+0).

Interpretations regarding Number 3.1 (10) of the "Safety Requirements for Nuclear Power Plants"

3 (6) Accident management measures shall be guided by the possibilities provided by the plant concepts.

3 (7) In the case of multi-unit plants, available equipment of the respective other unit may also be used for accident management as far as this will not impair the safety of the latter.

Interpretations regarding Number 3.7 (9) of the "Safety Requirements for Nuclear Power Plants"

3 (8) Any interventions in equipment fulfilling instrumentation and control function on levels of defence 1 to 4a shall be carefully planned in advance if possible and be laid down in written procedures.

4 Requirements for severe accident management guidelines

Interpretations regarding Numbers 2.1 (9), 2.1 (10) and 2.1 (13) of the "Safety Requirements for Nuclear Power Plants"

4 (1) The equipment considered in connection with severe accident management guidelines may also be used outside their design scope if it appears possible that the objectives indicated in Number 2.1 (3b) of the "Safety Requirements for Nuclear Power Plants" may be reached in this way.

4 (2) The severe accident management guidelines may also consider measures for the repair of equipment.

4 (3) The objective of severe accident management guidelines is to provide a plant-specific guideline for the support of the work of the emergency response staff in the use of available systems, components, resources, structural conditions etc. for mitigation of the consequences of accidents with severe fuel assembly damage.

4 (4) In this context, the priority objectives of the severe accident management guidelines are:

- to stop the core meltdown process,
- to maintain the barriers for activity retention that still exist,
- to limit the release of fission products, and
- to reach a plant state that is controllable in the long term.

4 (5) Severe accident management guidelines shall be applicable to a broad spectrum of plant states.

4 (6) The severe accident management guidelines shall

- include criteria for the assessment of the plant state by the emergency response staff, with the possibility of verifying fulfilment of these by means of the available plant instrumentation or other sources,
- indicate alternative options for obtaining information in case of a partial or total accidental failure of the instrumentation,
- compare positive and negative consequences of the actions in question with regard to their effects,
- if necessary, contain decision aids for the successful execution of prepared severe accident management guidelines (e.g. diagrams of the amounts of water needed for core cooling, water level diagrams).

4 (7) For the selection and execution of a suitable severe accident management guidelines it is necessary to have a reliable diagnosis of the plant state, based on information about the reliability of systems, utilisable coolant inventories and electricity supply options as well as about the condition of the reactor core and the containment. The diagnosis shall consider the design ranges of the measuring systems and the instrumentation to be used.

5 Requirements regarding internal and natural hazards as well as human-induced external hazards

Interpretations regarding Numbers 3.1 (11) of the "Safety Requirements for Nuclear Power Plants"

5 (1) The "Safety Requirements for Nuclear Power Plants", Number 3.1 (11), demand that the measures and equipment of accident management that may be needed in the case of internal and natural hazards as well as in the case of human-induced external hazards shall also remain effective even in these cases. However, this requirement concerns only those measures and equipment provided for accident management that are necessary for ensuring the fundamental safety functions on the subsequent levels of defence 4b and 4c in the event of a multiple failure of necessary safety equipment with the consequence of a beyond-design-basis plant state.

5 (2) In case of a malfunction of safety equipment – which cannot be excluded – according to the fundamental assumptions in Number 4.3 (2) of the "Safety Requirements for Nuclear Power Plants", beyond-design-basis plant state may develop that make the use of measures and equipment of accident management necessary. According to Number 4.3 (1) of the "Safety Requirements for Nuclear Power Plants", such beyond-design-basis plant states shall be determined by general consideration of the results of deterministic and probabilistic safety analyses, operating experience, safety research, and international recommendations.

6 Requirements for filtered containment venting

Interpretations regarding Number 3.6 (8) of the "Safety Requirements for Nuclear Power Plants"

To reduce the radioactive aerosols released through the filtered venting of the containment, the pressure relief line shall be installed in an area of the containment in which as low airborne aerosol concentrations as possible are to be expected during the accident sequence. In the case of the BWR, this would be the gas space of the pressure suppression pool and in a PWR the peripheral area of the containment (service compartments).

7 Requirements for the probe sampling system

Interpretations regarding Number 3.7 (8) of the "Safety Requirements for Nuclear Power Plants"

Equipment shall be available for probe sampling from the containment atmosphere and for coolant sampling in order to obtain information about the radioactive materials released into the containment during plant states on levels of defence 4b and 4c.

8 Requirements for the emergency organisation

Interpretations regarding Numbers 3.8 (5) and 3.8 (7) of the "Safety Requirements for Nuclear Power Plants"

8 (1) Detailed requirements regarding levels of defence 4b and 4c for the

- emergency organisation,
- qualification and training of the intended personnel,
- alarming and communication (internal and external),
- technical equipment and rooms provided for the emergency organisation, and
- organisation and execution of emergency exercises

are described in the "Basic recommendations for the planning of emergency control measures by the licenses of nuclear power plants", recommendation by SSK and RSK (2010). A further-reaching explanation is not necessary.

Interpretations regarding Numbers 6 (1) to 6 (3) of the "Safety Requirements for Nuclear Power Plants"

8 (2) Requirements for

- the documentation related to accident management,
- the qualification of personnel, instruction and exercising, and
- ensuring effective accident management

are also contained in the "Basic recommendations for the planning of emergency control measures by the licenses of nuclear power plants", recommendation by SSK and RSK (2010). A further-reaching explanation is not necessary.

9 Requirements for analyses of the effectiveness of accident management measures

Interpretations regarding Annex 5, Number 3.2.1 (6) of the "Safety Requirements for Nuclear Power Plants"

9 (1) The analyses of the effectiveness of preventive and mitigative accident management measures shall be performed with suitable methods for the underlying representative event sequences or plant states.

9 (2) The feasibility of accident management measures shall be demonstrated and documented.

9 (3) The analyses of the effectiveness of preventive and mitigative accident management measures for the retention or the long-term re-establishment of the fundamental safety function "confinement of the radioactive materials" shall demonstrate that the radiological safety objectives mentioned in Number 2.5 (1) of "Safety Requirements for Nuclear Power Plants" are achieved.

Analyses of the effectiveness of preventive accident management measures

9 (4) The analyses of the effectiveness of preventive accident management measures for the retention or long-term re-establishment of the protection goal "reactivity control" shall demonstrate that the retention or re-establishment of the necessary subcriticality of k_{eff} < 0.999 of the fuel assemblies in the reactor core and/or the fuel assemblies in the spent fuel pool is reached.

9 (5) The analyses of the effectiveness of preventive accident management measures for the retention or long-term re-establishment of the protection goal "fuel cooling" for fuel assemblies in the reactor core and in the fuel pool shall demonstrate that severe core damage is prevented.

9 (6) For these analyses, the criterion

- maximum fuel cladding temperature < 1200°C

may be used for fuel assemblies in the reactor core and the criterion

- complete water coverage of the fuel assemblies

may be used for fuel assemblies in the spent fuel pool.

9 (7) As an alternative to Number 9 (6), the effectiveness of the measures and equipment of preventive accident management may be

demonstrated by the fulfilment of the requirements of Number 2.2 (6) of the "Safety Requirements for Nuclear Power Plants".

Analyses of the effectiveness of mitigative accident management measures

9 (8) For analyses of the effectiveness of mitigative accident management measures, events with severe fuel assembly damage in the reactor core or in the spent fuel pool shall be used as a basis.

9 (9) The analyses of the effectiveness of mitigative accident management measures shall demonstrate the maintenance of the function of the containment as a barrier or of retention functions for the case of the storage of spent fuel assemblies outside the containment.

9 (10) The analyses of the effectiveness of mitigative accident management measures shall demonstrate

- that combustion processes of gases (H_2, CO) in the containment are either prevented or that these will not put containment integrity at risk,
- that combustion processes of gases (H_2, CO) (in the case of a spent fuel pool located outside the containment) within the surrounding structural shell (reactor building) are either prevented or that the structural shell is not put containment integrity at risk.

9 (11) The analyses of the effectiveness of mitigative accident management measures shall furthermore demonstrate

- that an overpressure failure of the containment by a continuously rising pressure is prevented,
- that containment venting can be initiated at a pressure that is less than the containment test pressure,

- that effective containment venting (pressure reduction) is possible,
- that the requirements for the filters in the pressure relief line of the containment venting system are fulfilled,
- that low pressure failure of the containment due to venting is prevented,
- that the measure of containment venting may be interrupted or repeated, if necessary,
- that combustion processes of gases (H_2, CO) in the containment venting system are prevented until the gases are discharged into the environment or that they do not impair the function of the containment venting system.

Analyses of the effectiveness of severe accident management guidelines

9 (12) The appropriateness of the measures and the equipment provided in the severe accident management guidelines to comply with the fundamental safety functions shall be demonstrated.

Interpretation I-8: Requirements for radiation protection

Content

1 Scope of application

This text contains interpretations regarding the requirements for radiation protection in connection with the operation of nuclear power plants. This text contains requirements for the occupational radiation protection and for the measures and installations in the area of radiation protection. On the one hand, they specify the requirements of the Atomic Energy Act and the Radiation Protection Ordinance as mandatory legal basis and on the other hand affect the contents of relevant guidelines, recommendations and technical rules, which in turn have a higher degree of detail.

2 Interpretations of safety requirements

2.1 Interpretations regarding safety requirement Number 2.5 (1)

2.1.1 General interpretations regarding organisational radiation protection and the radiation protection of the personnel

2.1.1 (1) The experience with the operation of the plant shall be regularly evaluated with respect to possible ways of further reducing the radiation exposure of the personnel, the population, and the environment. Besides the experience with the operation of the licensee's own plant, available operating experience with similar domestic and foreign plants shall also be considered.

2.1.1 (2) In addition to suitable working processes, permanent installations for the confinement of radioactive materials as well as for the limitation and reduction of direct radiation, contamination and airborne activity shall be provided to reduce radiation exposure.

If necessary, mobile equipment, such as mobile shields, exhausts or decontamination equipment shall also be used.

Personal protective equipment (e.g. respiratory protection, protective clothing) shall be used if the necessary protective effect cannot be achieved by the above-mentioned structural and technical means, taking all circumstances of the individual case into account.

2.1.2 Interpretations regarding organisational radiation protection and the radiation protection of the personnel on levels of defence 3 und 4

2.1.2 (1) All actions for the control, mitigation of the effects or elimination of the consequences of event sequences and plant states on levels of defence 3 and 4a shall be planned with participation of the radiation protection personnel. The radiation protection personnel shall be involved in the performance of the actions. The radiation protection officer shall be involved appropriately in the planning of actions in connection with accident management on levels of defence 4b and 4c. The radiation protection personnel shall be involved in the performance of actions in connection with accident management on levels of defence 4b and 4c in an appropriate manner.

2.1.2 (2) The radiation protection requirements that are relevant for the planning, performance and follow-up of work in connection with the operation of the plant shall in principle also be applied to actions for the control, mitigation of the effects or elimination of the consequences of event sequences and plant states on levels of defence 3 and 4. Exceptions shall be justified. On levels of defence 4b and 4c in particular, deviations from individual requirements may be permissible if these deviations are described in operational documents and are justified.

2.1.3 Interpretations regarding the control of the activity and the activity flow on levels of defence 1 and 2

2.1.3 (1) In order to comply with the radiological protection goals according to Number 2.5 (1) of the "Safety Requirements for Nuclear Power Plants" for levels of defence 1 and 2, the sources of ionising radiation that are related to the operation of the plant shall be identified in the design of the plant and be kept under control during the operation of the plant by measures and installations in accordance with the requirements of the radiation protection ordinance.

2.1.3 (2) The entry of corrosion products that can be or have been activated into the reactor coolant shall be kept as low as achievable by the choice of materials as well as by the chemical operating mode of the coolant, taking all circumstances of the individual case into account.

Note: A major reduction of the local dose rate can be achieved especially by minimising the cobalt content and by avoiding the use of cobalt-based alloys.

2.1.3 (3) The entry of nuclear fuel and fission products as well as of spallings of the oxide layer of the fuel cladding into the coolant shall be kept as low as achievable by quality assurance measures during the manufacturing and handling of the fuel assemblies as well as by the operating mode, taking all circumstances of the individual case into account.

2.1.3 (4) Measures and installations shall be provided that allow the detection of fuel rod damage. When deciding about the continued operation of the plant with defective fuel rods, the radiation exposure of the operating and maintenance personnel during operation as well as during subsequent refuelling outages, which is caused by the defective fuel rods, shall be taken into account.

2.1.3 (5) Clean-up systems for the reactor coolant system and for the spent fuel pool shall be installed which must be operated on demand and which must be effective against dissolved as well as undissolved impurities.

2.1.3 (6) Systems that contain media contaminated with radioactivity shall be sealed off such that the spreading of radioactive materials is prevented.

2.1.3 (7) The effectiveness of barriers and the retaining functions of the systems that contain radioactive medium shall be monitored. For this purpose, values shall be defined for maximum permissible leakages in terms of the respective system and respective medium.

2.1.3 (8) Waters that are contaminated with radioactivity (e.g. coolant system waters, sump waters, laboratory waters or rinse waters) shall be collected by their origin, treated, and conditioned. If the re-use of the waters inside the plant is out of the question, they shall be discharged under controlled conditions.

2.1.3 (9) Exhaust gases that come from nuclear systems and are contaminated with radioactivity shall as a rule be collected and treated according to their contamination by installations for activity retention or delay. Exceptions shall be justified. In the case of delay, appropriate delay times shall be observed so that the discharge of short-lived radioactive noble gases will not contribute considerably to radiation exposure.

2.1.4 Interpretations regarding the control of the activity and the activity flow on level of defence level of defence 3

2.1.4 (1) When designing the plant, the potential sources of ionising radiation that may arise due to events on level of defence 3 shall be identified so that measures and installations can be provided for level of defence 3 for the control of these sources to ensure compliance with the radiological protection goals according to Number 2.5 (1) of the "Safety Requirements for Nuclear Power Plants".

2.1.5 General interpretations regarding radiation and activity monitoring in the plant

2.1.5 (1) The following shall be provided for radiation and activity monitoring in the plant:

1. equipment for monitoring radioactive materials that may be released or discharged with the air or waste water;
2. fixed installations for measuring the concentration of radioactive materials in circuits in which corresponding monitoring is necessary to detect any possible released radioactive materials;
3. fixed installations for measuring the concentration of radioactive materials in the room air of groups of rooms or rooms in which corresponding monitoring is necessary for the protection of individuals or for the early detection of any possible released radioactive materials;
4. fixed installations for measuring local dose rates;
5. equipment for measuring personal dose rates, the local dose rate and the room air concentration at workplaces as well as for measuring the contamination of individuals and objects;
6. suitable laboratory equipment for evaluating analyses of radioactive samples.

Further criteria for radiation and activity monitoring installations and equipment that fulfil the functions of accident overview measuring systems result from the interpretation of the "Safety Requirements for Nuclear Power Plants" I-3 "Requirements for instrumentation and control and accident overview monitoring systems" and the interpretation of the "Safety Requirements for Nuclear Power Plants" I-4 "Requirements for the electrical energy supply”.

2.1.5 (2) The results of the measurements of the radiation and activity monitoring equipment in the plant shall be documented and kept in accordance with the relevant legal or official requirements and in line with the stipulations of the relevant safety standards. If for the purpose of preserving evidence the safekeeping of samples should be necessary, suitable provisions shall be taken.

2.2 Interpretations regarding safety requirement Number 2.5 (2)

2.2.1 Interpretation regarding the control of the activity and the activity flow on level of defence level of defence 3

2.2.1 (1) Waters contaminated with radioactivity that arise due to events on level of defence 3 shall be collected within the plant. Corresponding measures shall be provided and equipment shall exist so that any waters contaminated with radioactivity arising as a result of events on level of defence 3 will not reach uncontrolled into the environment of the plant. Any waters released inside the containment, e.g. due to events involving a loss of coolant shall as far as possible be confined within the containment and the systems necessary for core cooling until they can be treated further. The necessary conditioning and discharge during the long-term phase shall take place according to a concept that tales the radiological aspects into account.

2.2.2 Interpretations regarding structural and technical radiation protection on level of defence 3

2.2.2 (1) If any equipment needs to be operated for controlling events on level of defence 3, access to this equipment shall be as unobstructed as possible.

2.2.2 (2) Regarding equipment that is expected to need maintenance or repair in connection with the long-term control of events on level of defence 3, measures and installations for shielding shall be provided for the maintenance case. Room shall be provided for any possible dismantling aids, or these shall be installed on site.

2.2.2 (3) Areas shall be provided on the plant premises that are suitable for the assembly of the personnel as well as for measuring the contamination of the personnel in case of any events on level of defence 3. These areas shall for this purpose be sufficiently protected from any possible increased radiation exposure and contamination.

2.2.3 Interpretations regarding structural and technical radiation protection on level of defence 4

2.2.3 (1) Equipment shall be arranged and if necessary, shielded such that the feasibility of pre-planned manual actions for level of defence 4 is ensured.

2.2.3 (2) Areas shall be provided on the plant premises that are suitable for the assembly of the personnel as well as for measuring the contamination of the personnel in case of any event sequences and plant states on level of defence 4. These areas shall for this purpose be sufficiently protected from any possible increased radiation exposure and contamination.

2.2.3 (3) Areas shall be provided on the plant premises that are suitable for the stay of the persons of the plant's internal emergency organisation tasked with controlling the emergency situation in case of any event sequences and plant states on level of defence 4.

2.2.4 General interpretations regarding ventilation systems

2.2.4 (1) The nuclear power plant shall be equipped with reliable ventilation systems for the following rooms:

a) rooms in which it cannot be ensured without ventilation systems that the amount of radioactive materials to be discharged into the environment with the exhaust air to fulfil the relevant criteria according to Number 2.5 (1) of the "Safety Requirements for Nuclear Power Plants" will be kept low;

b) rooms in which the activity concentration has to be kept low for reasons of occupational radiation protection and where this cannot be ensured without ventilation systems.

Note: Further, safety-related requirements for ventilation systems are contained in the interpretation of the "Safety Requirements for Nuclear Power Plants" I-5 "Requirements for structures, systems and components"

2.2.4 (2) If the concentration of radioactive materials in the air of certain rooms may become so high that permissible limits will be exceeded in connection with the discharge of radioactive materials with the exhaust air, air filter systems shall be provided for the corresponding ventilation systems. It shall be permissible to realise ventilation systems in such a manner that the exhaust air will only be guided through filter systems if necessary. The air filter systems shall be sufficiently reliable and constituted such that they will have the necessary separation efficiency under the respective conditions of operation. The equipment necessary for inspecting their condition shall be provided.

2.2.5 Interpretations regarding ventilation systems on level of defence 3

2.2.5 (1) The ventilation systems shall be designed, constituted and aligned with the properties of the other systems in such a manner that on level of defence 3, the associated permissible limits for the activity concentration in the room air and for the discharge or possible release of radioactive materials will not be exceeded. Air recirculation systems shall be combined in a suitable manner with ventilation systems so that the relevant criteria according to Number 2.5 (1) of the Safety Requirements for Nuclear Power Plants" will be fulfilled. The ventilation systems and filter systems provided for use during or after events on level of defence 3 shall be connected to the emergency power supply system.

2.2.5 (2) Measures and installations shall be provided for preventing the escape of high levels of activity from the containment. If the containment is ventilated in closed condition, automatic ventilation isolation of the containment, responding when high activity levels are measured in the containment, shall be provided.

2.2.5 (3) Filter systems that are provided for use during or after events on level of defence 3 for cleaning the exhaust air and thus for limiting the consequences of the event shall be designed such that the separation efficiencies for suspended matter, organically bound iodine and elementary iodine that are to be considered according to the relevant safety-related standards will not be less when the filter systems are used on level of defence 3.

2.2.5 (4) Filter systems which on levels of defence 1 and 2 are permanently or intermittently perfused by exhaust air or which must be used during or after events on level of defence 3 shall be installed and operated such that it can be excluded that separation levels that have been applied as a basis when planning technical protection measures against events on level of defence 3 will not fall below a minimum separation efficiency.

The influence of the loading with contaminants during use on levels of defence 1 and 2 shall be particularly monitored, and an adequate safety margin to the minimum separation efficiency shall be ensured by a timely exchange of the filter material.

2.2.5 (5) Filter systems according to Number 2.2.5 (3) shall be designed with respect to their active components (e.g. ventilator and secondary heater) with a redundancy of 3 x 100 % or 4 x 50 % and with respect to their passive components with a redundancy of 2 x 100 % (to be optionally connected or switched over). The filter systems shall be equipped with installations that prevent deviations below the dew point in the filter inlet air and condensate accumulations or limit those to an extent that can demonstrably not lead to a deviation below the required degrees of separation efficiency. The filter inlet air conditions occurring during or after events on level of defence 3 shall be specified. If redundant filters are installed in one room, it shall be ensured that

a) the redundant filters cannot fail simultaneously due to an event on level of defence 3 for whose control they are needed, and

b) a redundant filter system cannot fail due to the failure of another filter system in an event on level of defence 3 for whose control it is needed.

2.3 Interpretations regarding safety requirement Number 3.7 (8) for radiation and activity monitoring in the plant on levels of defence 3 and 4

2.3 (1) For monitoring the activity in the containment atmosphere during event sequences and plant states on level of defence 3 and 4, high-dose-rate measuring devices and sampling equipment shall be provided which

– supply the necessary information for the initiation of emergency measures and for the support of disaster response measures

– allow especially during plant states on level of defence 4c an estimate of the airborne activity inside the containment to facilitate a prognosis of the extent of the activity released from the containment if filtered containment venting is intended or if containment integrity is at risk.

Note: Emergency measures shall be implemented regarding the taking of samples for the purpose of diagnosing the conditions inside the containment, which provides information that can be considered in connection with the assessment of the radiological consequences of filtered containment venting (see interpretation I-7 „Requirements for accident management", subsections 3.1(2) and 3.3(6)).

2.3 (2) For assessing the radiological consequences of event sequences and plant states on levels of defence 4b and 4c, the local dose rate shall be monitored in suitable locations within the plant (e.g. in the reactor building and in boiling water reactors additionally in the turbine building) so that the necessary information for initiating emergency measures and for the support of disaster response measures can be provided.

2.3 (3) In event sequences and plant states on level of defence 4 in which the release of airborne radioactive materials into the environment cannot otherwise be determined, it shall be ensured in accordance with the relevant safety-related standards that the release can be estimated with the help of the measuring results of immission monitoring in combination with the survey of the dispersion conditions.

2.4 Interpretations regarding safety requirement Number 3.11 (1)

2.4.1 General interpretation regarding the radiation and activity monitoring of systems

2.4.1 (1) For the early detection of any possible releases of radioactive materials, the concentration of radioactive materials shall be monitored in all systems acting as barriers against the escape of radioactive materials.

2.4.2 Interpretations regarding the radiation and activity monitoring of systems on levels of defence 1 and 2

2.4.2 (1) Monitoring shall be devised such that any impermissible changes in the activity concentration in systems, especially the passing-over of activity into systems or system areas which by design contain no radioactive materials, will be reliably recognised.

2.4.2 (2) The activity concentration shall be monitored by continuous measurement by means of permanently installed measuring instruments and by regular sampling. If the activity concentration in the reactor coolant system, in the systems immediately connected to it or in the pool cooling or clean-up system circuits is not measured continuously but determined by sampling, samples shall be taken with sufficient frequency.

2.4.2 (3) If the equipment continuously measuring the activity concentration to monitor systems yields results showing that specified threshold values are exceeded, an alarm shall be signalled in the control room.

2.4.2 (4) Determination of the activity concentration in the circuits by sampling shall also be carried out if there are indications of an increased activity concentration.

2.4.3 Interpretations regarding the radiation and activity monitoring of systems on levels of defence 3 to 4b

2.4.3 (1) Monitoring shall be devised such that the entry of radioactive materials due to event sequences on levels of defence 3 to 4b into systems that by design contain no radioactive materials is recognised so that measures necessary for limiting a possible release caused by this circumstance can be initiated and that information that may perhaps be needed for the support of disaster response measures is available.

2.4.3 (2) As for systems that according to Number 2.4.2 (2) have to be monitored by continuous measurement, this shall also be ensured on level of defence 3.

2.4.4 General interpretation regarding the monitoring of room air activity

2.4.4 (1) Rooms or groups of rooms of the controlled area that are entered regularly by operating personnel and in which the contamination of the room air may be higher shall be continuously monitored for the radionuclide groups (noble gases, suspended matter, gaseous iodine) that may occur. For this purpose, monitoring equipment shall be permanently installed that will signal an alarm if threshold values are exceeded.

The system to be permanently installed indicates the passability of the monitored areas, the plant state, and the integrity of the systems.

2.4.5 Interpretation regarding the monitoring of room air activity on levels of defence 1 and 2

2.4.5 (1) The system shall be designed such that under operating conditions on levels of defence 1 and 2,

a) increased activity concentrations in the room air are detected,

b) the affected buildings or groups of rooms can be identified, and

c) leaks in activity-retaining systems are detected (leakage monitoring).

2.4.6 Interpretation regarding the monitoring of room air activity on levels of defence 3 and 4

2.4.6 (1) The system shall be designed such that in events of level of defence 3, any activity releases into the room air can be detected and located.

2.4.7 General interpretations regarding the monitoring of the local dose rate (fixed system)

2.4.7 (1) A fixed system shall be provided for the monitoring of local dose rates in controlled areas that will signal an alarm if threshold values are exceeded.

2.4.7 (2) The measured values of this fixed system shall be indicated on site and in the control room and shall be recorded. The measured values shall be monitored for any transgressions of warning thresholds. Any such transgression shall be signalled optically and acoustically on site and in the control room.

2.4.8 Interpretations regarding the monitoring of the local dose rate (fixed system) on levels of defence 1 and 2

2.4.8 (1) Fixed local dose rate measuring devices of this system shall be installed in those areas of the plant where changes in the local dose rate are expected and individuals have to be warned.

2.4.9 Interpretations regarding the monitoring of the local dose rate (fixed system) on level of defence 3

2.4.9 (1) The system shall be designed such that indications of the accessibility of monitored areas can be given in events on level of defence 3.

2.4.9 (2) In order to be able to assess the radiological consequences of events on level of defence 3, the local dose rate shall be monitored in a suitable location within the plant (e.g. in the reactor building and in the case of boiling water reactors additionally in the turbine building).

2.4.10 Interpretations regarding the monitoring of workplaces and other measuring and monitoring tasks on levels of defence 1 and 2

2.4.10 (1) For the measuring tasks intended for the protection of individuals carrying out jobs in a controlled area, mobile measuring devices or equipment shall be provided in suitable locations so that samples can be taken and evaluated:

a) local dose rate measuring devices for

   aa) gamma and beta radiation,

   ab) neutron radiation,

b) equipment for the nuclide-specific recording of contamination e.g. by sampling and laboratory evaluation,

c) measuring equipment for the determination of surface contamination,

d) equipment for the determination of the activity concentration in the room air.

2.5 Interpretations regarding safety requirement Number 3.11 (2)

2.5.1 Interpretations regarding facilities for waste water treatment on levels of defence 1 and 2

2.5.1 (1) The facilities for waste water treatment and their storage capacities shall be designed such that the waters arising during operation on levels of defence 1 and 2 in areas with open radioactive materials can be received and treated if necessary.

2.5.1 (2) It shall be ensured by suitable measures and installations that water which is contaminated with radioactivity cannot enter into the soil and hence possibly into the groundwater, nor uncontrolled into a non-activity-retaining system or the surface water.

2.5.2 Interpretations regarding the monitoring of the discharge of airborne radioactive materials on levels of defence 1 and 2

2.5.2 (1) The equipment for the monitoring of the discharge of airborne radioactive noble gases and the monitoring of suspended radioactive matter and radioactive iodine with the exhaust air through the stack shall be so implemented such that monitoring continues to be ensured even in the case of a failure of a measuring device.

2.5.3 Interpretations regarding the monitoring of the discharge of radioactive materials with water on levels of defence 1 and 2 as well as in events on level of defence 3

2.5.3 (1) The discharge of radioactive materials with water shall be monitored. It shall be shown by way of the monitoring that the licensed limits for the discharges are observed.

Note: According to Numbers 2.1.3 (8) and 2.2.1 (1), the waste waters from controlled areas shall be collected for all operating conditions on levels of defence 1 and 2 and in all events on level of defence 3 and shall only be discharged under controlled conditions if it is ensured that specified activity concentration limits in the transfer vessel are not exceeded.

2.5.3 (2) The discharge of waste water from controlled areas shall be monitored by means of fixed activity measuring points and shall be automatically interrupted in time if specified limits are exceeded.

2.5.3 (3) The discharge of radioactive materials via other systems that can retain activity, e.g. the nuclear service water system or the turbine building waste water system, shall be monitored and quantitatively determined if specified activity limits are exceeded.

2.5.3 (4) The water flowing back into the receiving water shall be monitored continuously.

2.6 Interpretations regarding safety requirement Number 3.11 (4)

2.6.1 Interpretations regarding structural and technical radiation protection on levels of defence 1 and 2

2.6.1 (1) When structuring the layout of the plant and when designing and arranging in particular those components that retain activity, it shall be considered that it may become necessary to exchange them during the operating life of a nuclear power plant. Hence, to reduce radiation exposure, taking all circumstances of each individual case into account, provisions shall be made for being able to exchange components undismantled and with as little radiation exposure as achievable. The measures and installations necessary to fulfil these requirements must not be opposed to safety-related requirements; for example, the possibility of carrying out in-service inspections of these components shall not be limited.

2.6.1 (2) When structuring the layout of the plant and when designing and arranging the components, sufficient accessibility, shielding of the access and transport paths as well as suitable provisions for the execution of decontamination work, also on vessels and piping systems (e.g. through mechanical cleaning and flushing) shall be ensured.

2.6.2 Interpretations regarding ventilation systems on levels of defence 1 and 2

2.6.2 (1) The ventilation systems shall be designed such and must be constituted and aligned with the characteristics of the other installations in such a way that the associated permissible values on levels of defence 1 and 2 for the activity concentration in the room air and for the discharge of radioactive materials are not exceeded. Air recirculation systems shall be combined with exhaust air systems in a suitable manner so that the relevant criteria according Number 2.5 (1) of the "Safety Requirements for Nuclear Power Plants" are fulfilled. Exhaust air systems fulfilling functions to maintain subatmospheric pressure in events on level of defence 2 shall be connected to the emergency power supply.

2.6.2 (2) In rooms that are connected to the ventilation systems, an uncontrolled escape of activity into the environment shall be prevented by maintaining subatmospheric pressure and a correspondingly directed flow or by closing suitable isolating valves. The measures necessary for this purpose shall be operable from the control room.

2.6.2 (3) To prevent radioactive materials from being carried off by the room air, the air in the controlled area shall on levels of defence 1 and 2 as a rule be guided such and the groups of rooms be sealed against each other and against the atmosphere in such a way that the air from rooms or groups of rooms with lower risk of contamination shall be guided towards rooms or groups of rooms with higher risk of contamination. Exceptions shall be justified.

2.6.2 (4) The exhaust air shall be monitored and, if necessary, discharged via filters. Exhaust air that arises on levels of defence 1 and 2 as a result of the need for maintaining subatmospheric pressure in containment areas in which components retaining reactor coolant are accommodated shall continuously cleaned by means of HEPA filters and monitored for iodine sorption filters.

2.6.3 Interpretations regarding other activity-retaining systems on levels of defence 1 and 2

2.6.3 (1) Activity-retaining systems shall be deaerated in a controlled manner. If any noteworthy accumulations of fission and radiolysis gases may occur in systems, these shall be connected to the exhaust gas system. The remaining systems shall be connected to the exhaust air system.

2.6.3 (2) Installations for pressure limitation on systems that contain contaminated media and the receiving systems or room areas shall be designed such that in case of a response, the media escaped into the receiving systems or room areas can be discharged in a controlled manner.

2.7 Interpretations regarding safety requirement Number 3.11 (5) for the collection, handling and storage of radioactive waste and radioactive materials to be re-used in a non-hazardous manner on levels of defence 1 and 2

2.7 (1) Radioactive waste and radioactive materials to be re-used in a non-hazardous manner shall as a rule be collected and stored separately in accordance with their intended further handling. Exceptions shall be justified.

In particular, materials that according to the provisions of the Radiation Protection Ordinance are intended for clearance or which have been cleared shall be collected and stored separate from other radioactive materials to prevent contamination.

2.8 Interpretations regarding ventilation systems of level of defence 4 with respect to safety requirement Number 3.11 (6)

2.8 (1) Ventilation systems that are intended for use in connection with measures on level of defence 4 shall be designed such that they fulfil the functions required for the execution of the measures.

2.8 (2) Filter systems that are intended for being used for filtered containment venting during plant states on level of defence 4c shall be designed such that they will not fall below the following separation efficiencies:

- suspended matter: $\eta = 99.9\ \%$
- elementary iodine: $\eta = 90\ \%$

The functional condition of the filter systems for the underlying representative event sequences and phenomena according to Section 2 of the interpretation of the "Safety Requirements for Nuclear Power Plants" I-7 "Requirements for accident management" shall be ensured.

2.9 Interpretations regarding safety requirement Number 5 (1)

2.9.1 Interpretation regarding the calculation of the radiological consequences for demonstrating the limitation of radiation exposure on levels of defence 1 and 2

2.9.1 (1) For planning purposes, assumptions, parameters and calculation models which will yield a conservative result for the calculated radiation exposure shall be used for calculating the radiation exposure through direct radiation in the environment of the plant.

Note: For planning purposes, the general administrative procedures decreed by the provisions of the Radiation Protection ordinance for the calculation of the radiation exposure due to the discharge of radioactive materials during the operation of the plant on levels of defence 1 und 2 apply.

2.9.2 Interpretation regarding the calculation of the radiological consequences for demonstrating the limitation of radiation exposure on level of defence 3

2.9.2 (1) The possible radiological consequences shall be calculated for the events on level of defence 3 for which according to Annex 2 of the "Safety Requirements for Nuclear Power Plants": "Events to be considered" compliance with the radiological protection goals has to be demonstrated and which according to Number 2 (4) of Annex 2 of the "Safety Requirements for Nuclear Power Plants": "Events to be considered" have been determined to be representative for this verification. In the calculation, those assumptions, parameters and calculation models shall as a rule be used as a basis which are defined in the relevant Incident Calculation Basis.

2.9.2 (2) Other parameters and calculation models may be used if this is justified by the design characteristics of the nuclear power plant, the characteristics of the respective site, or the release and dispersion conditions. Deviations from the Incident Calculation Bases shall be justified in detail; here, it shall be shown that the other parameters and calculation models correspond better to the actual conditions of the respective individual case.

2.9.2 (3) Assumptions, parameters and calculation models shall be used for the calculation by which the expected radiation exposure in the environment of the plant will be determined in a manner that is sufficiently conservative for planning purposes.

2.9.2 (4) For this purpose, proven assumptions regarding the initial conditions and characteristics of the plant (e.g. regarding activity content, leak rates, efficiency of clean-up and retention systems), the activity release into the enclosing systems, the deposition processes on internals and the temporal distribution of leak and outflow rates for the enclosing systems as well as realistic assumptions, calculation models and parameters regarding the event sequence and the release and dispersion of radioactive materials shall be applied as a basis, using – as far as possible – observed frequency distributions.

If simplified calculation methods are used, the assumptions, calculation models and parameters shall be specified such that a conservative overall result in accordance with Number 2.9.2 (3) will be achieved.

Alternatively, if realistic assumptions, calculation methods and parameters are used, verification that at least 95% of the scatter range of the expected radiation exposure, quantifying the uncertainties, is covered shall be permissible. This verification that at least 95% of the scatter range is covered shall also be permissible for interim results of substeps of the analysis (e.g. for the calculation of the activity release a n d / o r the calculation of the dispersion of radioactive materials), if it can be demonstrated that a conservative overall result will be achieved for the expected radiation exposure by the chosen combination of realistic and conservative substeps.

Note: For a verification with quantification of the uncertainty of the results, the criteria according to the "Safety Requirements for Nuclear Power Plants" Annex 5 "Requirements for verification and documentation " in Section 3.3 apply; for a conservative verification, the criteria in Section 3.4 apply.

2.9.2 (5) Parameters for the calculation of the activity release, whose values can scatter strongly, shall be

– estimated conservatively or

- chosen such that the nuclide-specific activity concentration and contamination in the environment of the plant that is to be calculated will cover at least 95% of the scatter range or
- determined by means of their observed frequency distributions, taking the subsequent conditions into account.

2.9.2 (5a) Verified distribution functions of the parameters shall be available; this also includes the acquisition of the measured values in a representative temporal distribution.

2.9.2 (5b) The parameter values used for the calculation of the activity release shall cover 95% of the distribution of the measured values.

2.9.2 (5c) If suitable meteorological data are available for the respective site, the statistical calculation method may also be used in the determination of the dispersion parameters.

2.9.2 (6) When calculating the possible radiological consequences of events on level of defence 3, releases of radioactive materials via the exhaust air path shall be considered.

2.9.2 (7) Radiation exposure via the exposure paths of external irradiation, inhalation and ingestion shall be determined. When calculating the radiation exposure, the restricted uses following the onset of an event on level of defence 3 shall be taken into account.

Note: Assumptions about the consumption behaviour and about restricted uses are specified in the Incident Calculation Bases (see Number 2.9.2 (1)).

2.9.2 (8) The calculation of the radiological consequences of events on level of defence 3 may be carried out with consideration of the operational equipment contributing to damage mitigation if this equipment has been manufactured and is operated according to valid codes and guidelines, if it has suitable quality features regarding its design and proven operating experience, and if it is not impaired in its functional performance by the consequences of the respective event.

Note: In accordance with the Incident Calculation Bases, it can be assumed when calculating the possible radiological consequences of events on level of defence 3 that the first actuation of the reactor protection system or the first actuation of reactor scram will become effective unless this itself is impaired by the respective event.

2.9.2 (9) A single failure going beyond the application of the single-failure concept according to the "Safety Requirements for Nuclear Power Plants", Annex 4 "Fundamental principles for the application of the single-failure criterion and for maintenance", Chapter 1 and 2, that would aggravate the respective boundary conditions in the analysis regarding the demonstration of the limitation of the radiation exposure of the population need not be postulated.

Note: Regarding the demonstration of the limitation of the radiation exposure of the population on level of defence 3 in accordance with the requirements of the radiation Protection Ordinance, further boundaries are specified in Annex 5, Number 3.2.4 of the "Safety Requirements for Nuclear Power Plants".