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Swiss Federal Nuclear Safety Inspectorate ENSI



Reactor Core, Fuel Assemblies and Control Rods: Design and Operation

Guideline for Swiss Nuclear Installations

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ENSI-G20/e

February 2015 Edition

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1 Introduction

The Swiss Federal Nuclear Safety Inspectorate (ENSI) is the regulatory authority for nuclear safety and security of the nuclear installations in Switzerland. ENSI issues guidelines either in its capacity as a regulatory authority or based on a mandate established by an ordinance. Guidelines are support documents that formalise the implementation of legal requirements and facilitate uniformity of implementation practices. Furthermore, they concretise the state-of-the-art in science and technology. ENSI may allow deviations from the Guidelines in individual cases, provided that the suggested solution ensures at least an equivalent level of nuclear safety or security.

2 Subject and scope

Guideline ENSI-G20 stipulates the requirements for the design and operation of the reactor core, fuel assemblies and control rods of nuclear power plants. It includes corresponding requirements for levels of defence 1 to 4a. Requirements that do not explicitly refer to individual levels of defence are applicable to levels of defence 1 to 3.

In addition to the requirements set out in Guideline ENSI-A01 and the DETEC Ordinance of 17 June 2009 on Hazard Assumptions and the Evaluation of Protection against Incidents in Nuclear Plants (SR 732.112.2), this Guideline also includes corresponding requirements for levels of defence 1 and 2. In addition, this Guideline contains further clarification of the requirements set out in Guideline ENSI-A01 for level of defence 3.

3 Legal basis

This Guideline implements the legal requirements stated in Articles 7 and 10 of the Nuclear Energy Ordinance of 10 December 2004 (NEO, SR 732.11).

4 Safety demonstrations and calculation codes

4.1 General requirements

- a. In order to comply with the fundamental protection objectives, criteria must be defined and limits set for the reactor core, fuel assemblies and control rods with respect to each level of defence unless specific criteria or limits are stipulated in this Guideline.

- b. Compliance with the criteria must be demonstrated by analyses of events and reactor core states. For this purpose, it is compulsory to apply conservative methods or realistic methods with quantified uncertainties.
- c. Conservative initial and boundary conditions must be used.
- d. Where necessary, compliance with the defined criteria must be confirmed by measurements on the reactor core.
- e. Analyses must be performed using harmonised neutronic, thermohydraulic and structural mechanics calculation codes. The calculation codes must allow the characteristics that are relevant to operation and essential for safety to be determined.
- f. The analyses must reflect the current state of the plant or the planned modification and must be consistent with the existing analyses and documentation of the nuclear plant.
- g. ENSI is to be informed about changes in calculation codes with a minor influence on the results of the calculation.
- h. Demonstration methods and calculation codes must correspond to the state-of-the-art in technology and take into account the state-of-the-art in science.
- i. Calculation codes must be created, used and maintained by appropriately qualified or trained personnel and within a certified quality management system (see Guideline ENSI-G07).

4.2 Empirical correlations

- a. It is permissible to use empirical correlations in calculation codes in order to describe physical aspects.
- b. Every correlation must be based on experiments whose measurement results have been adequately reviewed for consistency and quality.
- c. The experiments must be designed to ensure that the operating conditions in the reactor core are adequately covered. In cases where these operating conditions are not replicated, the transferability of the results must be justified.
- d. A correlation's range of validity must be determined.
- e. Extrapolations beyond the range covered by the measurement data are permissible only in justified cases. The additional uncertainties in the extrapolated range must be indicated and taken into account.
- f. The range of the correlation's validity must cover the range of application.

- g. In the case of correlations that are not derived from physical principles, the correlation must be validated based on independent data. This data must not have been used to develop the correlation.
- h. For correlations representing physical relationships (functions or tables), systematic deviations and statistical uncertainties must be determined and taken into account.

4.3 Validation and verification

- a. The applied calculation codes must be verified and validated.
- b. Levels of defence 1 and 2: validation must be performed based on measurement results.
- c. Level of defence 3: validation must be performed based on measurement results wherever possible. If no suitable measurement results are available from incidents or experiments, it is also possible to use results from other calculation codes. The reference codes used should be based on reliable physical calculation methods. Any additional uncertainties that emerge in this process must be taken into account.
- d. Measurement results must cover the range of application with regard to safety-relevant parameters. If this is not the case, justification must be given for the transferability of the measurement results to the range of application and any additional uncertainties must be taken into account.

4.4 Uncertainties

- a. Any uncertainties that have a significant impact on the safety-relevant parameters must be taken into account.
- b. The demonstration methods must take the following uncertainties into account as a minimum requirement:
 - 1. manufacturing tolerances,
 - 2. tolerances of material characteristics,
 - 3. measurement uncertainties,
 - 4. variations in operating parameters,
 - 5. changes brought about by operation,
 - 6. systematic deviations in the calculation models,
 - 7. uncertainties of the calculation models.

- c. Uncertainties can be taken into account either conservatively or statistically. It is permissible to combine the two approaches.

5 Reactor core

5.1 Neutronic design

- a. Level of defence 1: the local power density must be limited to the initial values used in the safety cases for deviations from normal operation and design-basis accidents. At the same time, the interaction between control and limitation systems must be taken into account.
- b. Levels of defence 1 and 2: the inherent properties of the reactor core to limit increases in reactivity and power must be guaranteed (inherent safety).
- c. Level of defence 2, as well as accident categories 1 and 2 of level of defence 3: the power and power densities must be limited so that the integrity of the fuel cladding is not affected. At the same time, the inherent properties of the reactor core must be taken into account, as must the effects of the control systems, limitation systems and reactor protection system.
- d. Level of defence 3: the emergency shutdown (scram) and long-term subcriticality of the reactor core must be guaranteed (see also section 5.5).
- e. Accident category 3 of level of defence 3: if fuel failure cannot be ruled out in the event of reactivity initiated accidents, then fuel release into the primary coolant must be limited.
- f. Level of defence 4a: if failure of the scram system is assumed, the pressure in the reactor coolant loop must be limited to the admissible value that is to be specified in accordance with section 4.1, point a. The long-term subcriticality and the coolability of the reactor core must be guaranteed.

5.2 Thermohydraulic design

5.2.1 General requirements

- a. Levels of defence 1 and 2, as well as accident categories 1 and 2 of level of defence 3: safety limits must be defined in order to rule out critical boiling conditions with a sufficiently high confidence level.
- b. Level of defence 1: in order to comply with the safety limits for power density in the event of deviations from normal operation and design-basis accidents

of accident categories 1 and 2, an operating limit must be determined with a sufficient margin from the critical boiling conditions.

- c. Level of defence 2, as well as accident categories 1 and 2 of level of defence 3: for all components of the reactor core, it must be ensured that no location experiences temperatures, pressures or pressure differences that cause unacceptable changes in the properties of the materials used or the safety-relevant function of the components.
- d. Accident category 3 of level of defence 3, as well as level of defence 4a: the coolability of the reactor core must be guaranteed. Unacceptable loads on the components of the reactor coolant loop must be ruled out, as must unacceptable compressive loads on the reactor core and core internals.
- e. Accident category 3 of level of defence 3, as well as level of defence 4a: a self-sustained exothermic zirconium-water reaction must be prevented.
- f. Accident category 3 of level of defence 3: the following additional requirements apply to loss of coolant accidents:
 - 1. The cladding temperature must be limited in order to avoid excess embrittlement and oxidation.
 - 2. Sufficient residual ductility or residual strength of the cladding must be ensured in order to avoid fragmentation.
 - 3. Incident-induced oxidation of the zirconium must be restricted to 1% of the zirconium contained in the fuel rod cladding tubes in order to limit hydrogen generation.
 - 4. Potential fuel release must be limited.

5.2.2 Distribution of mass flow density in the reactor core

- a. The core internals and the reactor core must be designed to ensure that the reactor core does not experience flow redistributions that prevent safe heat removal or lead to unacceptable mechanical loads on the components of the reactor core.
- b. The determination of the flow distribution in the reactor core must take the following aspects into account:
 - 1. the flow distribution prior to entering the core,
 - 2. the core geometry,
 - 3. differences in hydraulic resistance,
 - 4. differences in heating and coolant mixing in the reactor core,
 - 5. local effects, e.g. cross flows.

5.2.3 Flow forces

- a. Hydraulically induced vibrations in the reactor core must be minimised.
- b. Level of defence 1: for boiling water reactors, it must be ensured that the forces resulting from buoyancy and flow do not exceed the weight of the fuel assemblies, the sub-bundles or the individual fuel rods.
- c. Level of defence 1: for pressurised water reactors, it must be ensured that the forces resulting from buoyancy and flow do not exceed the weight of the fuel assemblies, taking into account the force exerted by the hold-down springs.
- d. Levels of defence 2 and 3: it must be proven that short-term fuel assembly lift-off is admissible without impairing control rod insertion.

5.2.4 Stability

- a. Level of defence 1: the thermohydraulic stability of the reactor core must be guaranteed.
- b. The following additional requirements apply to boiling water reactors:
 1. Level of defence 1: precautions must be taken to ensure that, in normal operation, a sufficient margin is maintained from the area in the power-flow map in which undamped power density oscillations can occur.
 2. Levels of defence 2 and 3: appropriate measures must be taken to ensure compliance with the safety-relevant requirements of the respective level of defence in the event of power density oscillations.
 3. The area of the power-flow map in which undamped power density oscillations can occur must be reviewed in the event of modifications that influence the power-flow map. If measurements of stability properties are not performed, this decision must be justified.

5.3 Safety-relevant characteristics

- a. Safety-relevant characteristics must be specified for each reactor core in accordance with the reactor type, the design of the overall plant and the analytical method used.
- b. Admissible ranges of values must be determined for all safety-relevant characteristics. Uncertainties and deviations must be taken into account.
- c. Level of defence 1: the reactor core and neighbouring systems must be designed and operated to ensure compliance with the admissible ranges of values for the safety-relevant characteristics.

- d. Levels of defence 2 and 3: suitable measures must be taken to comply with the admissible ranges of values. The type of measures and the permissible time for the measures to take effect must be defined on the basis of the respective event sequences.

5.4 Instrumentation

- a. Instrumentation with sufficient spatial and temporal resolution must be provided to allow monitoring of the safety-relevant characteristics of the reactor core and to allow verification of the core calculations.
- b. The safety-relevant characteristics must be recorded either directly or indirectly and monitored for compliance with their admissible ranges of values. It must be possible to identify trends and anomalous behaviour.
- c. The number and position of detectors are to be chosen so that significant differences between readings and calculated values can be recorded.
- d. Safety-relevant parameters are to be logged and must be retrievable in a timely manner.

5.5 Systems for reactivity control and shutdown

- a. The required minimum effectiveness of the shutdown systems (shutdown margin) is to be specified on a plant-specific basis.
- b. Sufficient effectiveness and speed of the shutdown systems must be demonstrated for each operating cycle.
- c. After core loading modifications that exceed minor changes, measurements must be performed to confirm that the shutdown margin satisfies the requirements.
- d. Components of the shutdown systems may also be used for operational control functions, although the effectiveness required for shutdown must be retained (see also section 7.1.1, point c).

5.5.1 Control rod system

- a. The control rod system must be shown to be sufficiently effective without the most reactive control rod.
- b. The control rod system must bring the reactor core into a hot shutdown state within a sufficiently short time and hold the reactor core in a subcritical state in the long term after a scram – if necessary, in combination with other shutdown systems (boric acid injection systems).

- c. The necessary effectiveness and speeds are to be determined through conservative analyses of postulated event sequences at the respective levels of defence.

5.5.1.1 Pressurised water reactors

- a. The control rod system and the reactor core are to be designed so that, following reactor shutdown, the shutdown margin complies with the value specified in accordance with section 5.5, point a, until long-term subcriticality is ensured by the boric acid injection systems.
- b. Accident category 3 of level of defence 3: temporary recriticality is allowed as long as the requirements of section 5.2.1, point d, are met.

5.5.1.2 Boiling water reactors

- a. The control rod system and the reactor core are to be designed so that, following shutdown, the shutdown margin in the most unfavourable reactor state complies with the value specified in accordance with section 5.5, point a.
- b. Locking systems that prevent start-up and core loading must be provided, as must scram triggering systems, as a precaution against inadvertent initiation of criticality and inadvertent power increases.

5.5.2 Boric acid injection systems

5.5.2.1 Pressurised water reactors

- a. The necessary effectiveness and shutdown speeds of the boric acid injection systems must be determined by analysing the event sequences at the respective levels of defence that put the highest demands on the performance of the individual boric acid injection systems.
- b. Acting in addition to the scram system, the boric acid injection systems must compensate the reactivity addition at the required speed based on the event sequence analyses to ensure that the reactor core remains subcritical in the long term and achieves the necessary shutdown margin.
- c. Levels of defence 1 and 2: boric acid injection systems alone must be capable of bringing the reactor core into a subcritical state from any postulated initial state, though the shutdown margin in accordance with section 5.5, point a, must be ensured in the long term.
- d. Levels of defence 3 and 4a: boric acid injection systems must ensure long-term subcriticality.

5.5.2.2 Boiling water reactors

Level of defence 4a: at level of defence 4a, a boric acid injection system must safely shut down the reactor core and keep it in a subcritical state in the long term.

6 Fuel assemblies

6.1 Safety-relevant requirements

- a. The properties and operation of the fuel assemblies must be designed so that the following requirements are met:
 1. Levels of defence 1 to 4a: the fuel assemblies must have the necessary geometric (shape and position) and material properties to control reactivity.
 2. Levels of defence 1 to 4a: the admissible power levels must be observed.
 3. Levels of defence 1 to 4a: the fuel assemblies must have the necessary geometric (shape and position) and material properties to ensure heat removal in accordance with the requirements.
 4. The requirements with regard to preventing critical heat flux specified in section 5.2.1 must be met.
 5. Levels of defence 1 and 2, as well as accident categories 1 and 2 of level of defence 3: the leaktightness of the fuel rods must be guaranteed (see Articles 9 and 10 of the DETEC Ordinance on Hazard Assumptions and the Evaluation of Protection against Incidents in Nuclear Plants).
 6. Accident category 3 of level of defence 3: in cases where the leaktightness of the fuel rods cannot be guaranteed, fuel release into the primary coolant must be limited.
- b. Levels of defence 1 to 4a: interactions of the fuel assemblies with the core internals and neighbouring systems must be taken into account.

6.2 Design

The following design requirements must be met in order to comply with the safety-relevant requirements specified in section 6.1.

6.2.1 General design requirements

- a. The design must take account of the loads, as well as changes in material properties and in geometry, that occur during transport, handling, operation and storage, e.g. as a result of external and internal mechanical, thermal, chemical and radiation-induced effects.
- b. The process of selecting materials and manufacturing processes must take account of the stresses and ambient conditions that occur in order to prevent malfunctions due to stress corrosion cracking.
- c. Activation of materials that pass into the coolant must be limited by appropriate selection of materials. This applies in particular to the cobalt content.
- d. Compatibility:
 1. The components of the fuel assembly must be compatible. In particular, it must be ensured that there is sufficient clearance between the fuel rod and the fuel assembly structure.
 2. The fuel assembly's geometric, mechanical, nuclear and thermohydraulic compatibility with the other assemblies of the reactor core, the core internals and the handling and storage equipment must be guaranteed.
 3. Taking account of electrochemical interactions, materials must be compatible with those of the neighbouring components.
- e. It must be possible to monitor the fuel assemblies with sufficient accuracy by core monitoring.
- f. It must be possible to inspect and repair fuel assemblies.
- g. The maximum burnup of the fuel assemblies must be defined.

6.2.2 Fuel rod-specific design requirements

6.2.2.1 Levels of defence 1 and 2, as well as accident categories 1 and 2 of level of defence 3

Fuel rods must be designed to ensure that the following objectives are met:

- a. avoiding fuel melting;
- b. ensuring that the fuel is at the specified position inside the cladding tube;
- c. limiting stresses and strains in the cladding tube to the admissible values for the specific material, taking account of loads such as the following:
 1. rapid increases in power (total cladding hoop strain),

2. long-term interactions due to fuel swelling (residual cladding hoop strain,
 3. pressure differences across the cladding tube due to excess external pressure (residual cladding hoop strain),
 4. thermal loads,
 5. bending loads;
- d. limiting the hoop stress in the cladding tube (taking account of hydride orientation);
 - e. limiting cyclical mechanical and thermal loads in order to avoid fatigue;
 - f. limiting the internal pressure in the fuel rod due to fission gas release and helium production in order to avoid inadmissible thermal feedback;
 - g. avoiding elastic buckling and plastic deformation due to excess external pressure;
 - h. ensuring sufficient ductility;
 - i. limiting corrosion;
 - j. limiting hydrogen pickup;
 - k. avoiding fuel failure due to mechanical-chemical interactions between the fuel and the cladding tube.

6.2.2.2 Accident category 3 of level of defence 3 and level of defence 4a

- a. Fuel rods must be designed to ensure that the following objectives are met (see section 5.1, point e, and section 5.2.1, point f):
 1. limiting deformation;
 2. avoiding fuel melting as a general rule.
- b. The following additional requirements apply to loss of coolant accidents and accidents of level of defence 4a:
 1. limiting the fuel cladding temperature in order to avoid excess embrittlement and oxidation;
 2. ensuring sufficient residual ductility or residual strength of the cladding tubes in order to avoid fragmentation.

6.2.3 Design requirements specific to the fuel assembly structure (including fuel channel in boiling water reactors)

6.2.3.1 Levels of defence 1 and 2, as well as accident categories 1 and 2 of level of defence 3

The fuel assembly structure must be designed such that the following objectives are met:

- a. ensuring that the positions of the fuel rod in the fuel assembly and of the fuel assembly in the reactor core comply with the specifications;
- b. ensuring that fuel rods are supported in such a way as to prevent fretting;
- c. limiting fretting on structural components;
- d. limiting stresses and strains on structural components, on joints and (in boiling water reactors) on the fuel channel due to factors such as:
 1. dead load,
 2. axial compression,
 3. forces of inertia,
 4. hydrostatic buoyancy and flow forces,
 5. hold-down forces exerted by the hold-down springs (pressurised water reactors),
 6. differential thermal expansion,
 7. differential radiation-induced growth,
 8. changes in operational states,
 9. flow-induced oscillations,
 10. control rod drop into the reactor (pressurised water reactors).
- e. ensuring that the hold-down spring force meets the requirements (pressurised water reactors);
- f. limiting cyclical mechanical and thermal loads in order to avoid fatigue;
- g. ensuring sufficient ductility;
- h. limiting corrosion;
- i. limiting hydrogen pickup in the fuel assembly structure;
- j. limiting differential growth of the various fuel assembly components;
- k. limiting fretting at contact points of the fuel assembly structure;
- l. limiting compressive stresses on the fuel rod that originate from the spacer;
- m. ensuring a clear path for the control rods;

- n. avoiding loose material originating from the fuel assembly.

6.2.3.2 Accident category 3 of level of defence 3 and level of defence 4a

The fuel assembly structure must be designed to ensure that the following objectives are met:

- a. limiting deformation of the spacers, the guide tubes (pressurised water reactors) and the fuel channels (boiling water reactors);
- b. limiting horizontal and vertical displacement.

6.3 Manufacturing

- a. The fuel assemblies must be procured in accordance with the procedures set out in the operating organisation's management system.
- b. The operating organisation must ensure that manufacturing of the fuel assemblies complies with the applicable Swiss statutory provisions and regulatory and technical requirements.
- c. The operating organisation must check that the fuel assembly supplier's quality management system is suitable for ensuring that the fuel assemblies are manufactured in accordance with the design.
- d. Once delivery is completed, a summary report must be submitted to ENSI for each reload and for each set of lead assemblies. The report must contain the following:
 - 1. an audit plan,
 - 2. a list of the new and requalified processes,
 - 3. design modifications,
 - 4. a summary of deviation reports, including justification for acceptance of the deviation,
 - 5. findings and measures derived from the findings,
 - 6. a summary assessment.

6.4 Operation

- a. The required primary coolant water quality must be specified and monitored, in order to minimise radiation exposure and maintain the important safety-relevant properties of the fuel assemblies and other core components.
- b. Following design-basis accidents of categories 1 and 2, it must be checked that the fuel assemblies are suitable for further operation.

6.4.1 Lead assemblies

- a. Lead assemblies must be used in order to evaluate the plant-specific performance of new types of fuel assemblies and fuel assemblies that have undergone significant modifications.
- b. In the absence of operating experience under comparable conditions in other plants, lead assemblies must be used for at least two cycles prior to the use of a larger number of fuel assemblies. If operating experience from other plants is available, a lead time of one cycle is sufficient.
- c. The number of lead assemblies must be limited, taking into account the importance of the modification.
- d. In addition to the use of lead assemblies, a dedicated inspection programme must be planned. ENSI must be informed of changes to the inspection plan.
- e. The approved maximum burnup limit may be slightly exceeded by the burnup of lead assemblies and individual fuel rods operated within the scope of irradiation programmes, provided that it can be demonstrated that the fuel assemblies or fuel rods in question perform in accordance with the design and are suitable for final disposal. Approval is required if the burnup limit is exceeded in this way.

6.4.2 Operation monitoring and operating experience

- a. The cladding tube integrity must be monitored based on radiochemical data and routinely assessed by means of trend analyses.
- b. Radiological limits must be specified for power operation following fuel cladding failures.
- c. Procedures and methods must be established for determining the type and severity of fuel cladding failures, their position and their possible causes, as well as the necessary actions.
- d. Plant-specific and international operating experience with the respective fuel assemblies must be collected and evaluated.

6.4.3 Inspections

- a. Regular inspections must be conducted in order to demonstrate that the fuel assemblies are operated within the design limits. ENSI must be informed of changes to the inspection plan.
- b. If a condition outside the design limits is identified, the fuel assembly must be repaired or brought into a condition that complies with the safety guidelines for further operation or for storage (see also section 6.5.4).

- c. Inspections must be carried out and evaluated by appropriately qualified or trained personnel and in accordance with the requirements of the management system.
- d. Qualified inspection equipment and tools must be used.
- e. During inspections, account must be taken of the radiation protection guidelines.
- f. Radiation protection measures must be planned to cover the eventuality of fuel cladding failures that already exist or might suddenly occur.
- g. Criticality safety ($k_{\text{eff}} < 0.95$) must be demonstrated during inspection activities.
- h. The results of the inspection must be documented and assessed by the operating organisation.

6.5 Handling and storage

- a. Fuel assemblies, control rods and core internals must only be handled using qualified equipment and systems provided for this purpose and in accordance with quality-assured work instructions.
- b. Fuel assemblies must be handled by appropriately trained and certified personnel.
- c. Fuel assemblies, control rods and core internals must only be stored in dedicated storage facilities.
- d. Radiation protection measures must be adopted, especially with regard to the shielding, building ventilation and containment isolation.
- e. Measures must be adopted and equipment provided to ensure that foreign materials do not enter the open reactor pressure vessel or the fuel storage pool. Foreign material must be retrieved, or it must be demonstrated that its presence in the system does not cause any safety problems.

6.5.1 Refuelling

- a. A step-by-step plan for shuffling, loading and unloading fuel assemblies in the reactor core and in the fuel storage pool must be prepared. This plan must cover every movement and position of the fuel assemblies, control rods and core internals. The completion of each individual step must be logged.
- b. For each step in the step-by-step plan, compliance with the required subcriticality in the reactor core and fuel storage pool must be proved.

- c. In pressurised water reactors, the minimum required boron concentration in the reactor core and fuel storage pool must be ensured and monitored in order to maintain subcriticality.
- d. In boiling water reactors, it must be ensured that all control rods are inserted, except when control rods are withdrawn individually for testing. Multiple control rods may be withdrawn or removed if the fuel assemblies of the affected control cells are unloaded and no fuel assemblies are loaded into the reactor core.
- e. During refuelling, it must be guaranteed that the neutron flux in the reactor core is monitored at all times.
- f. Effective and reliable measures and equipment must be provided to prevent handling errors and incorrect positioning of fuel assemblies.
- g. Before the reactor pressure vessel is closed, the position and orientation of the fuel assemblies, control rods and core internals must be checked and documented.

6.5.2 Dry storage of non-irradiated fuel assemblies

- a. Levels of defence 1 and 2: to ensure subcriticality, the calculated neutron multiplication factor k_{eff} must be less than 0.95 including all uncertainties and tolerances. Limiting moderation and reflection conditions as well as limiting assembly type with regard to k_{eff} must be considered throughout fuel storage. At the same time, technical and operational procedures must be taken into account.
- b. Level of defence 3: over and above the requirements of levels of defence 1 and 2, the calculated k_{eff} value must be less than 0.98 including all uncertainties and tolerances, provided a conservative moderator density is assumed that yields the highest k_{eff} value (optimum moderation).
- c. Level of defence 4a: the maximum calculated k_{eff} value must be less than 1.0 including all uncertainties and tolerances. Realistic boundary conditions may be taken into account.
- d. Cooling must be guaranteed for fuel assemblies with non-negligible thermal output (MOX).
- e. In principle, it is only permissible to store fuel assemblies in the specified condition. If fuel assemblies must be stored in a condition that does not correspond to the design, confinement of radioactive materials must be ensured.

6.5.3 Wet storage of fuel assemblies

- a. Levels of defence 1 and 2: the calculated k_{eff} value must be less than 0.95 including all uncertainties and tolerances. Limiting moderation and reflection conditions as well as limiting assembly type with regard to k_{eff} must be considered throughout fuel storage. At the same time, technical and operational procedures must be taken into account. If soluble boron in the coolant water is taken into account, then adequate monitoring of the boron concentration must be ensured and it must additionally be shown that the k_{eff} value remains below 1.0 if the boron is disregarded.
- b. Level of defence 3: the requirements under point a apply, taking into account any accidental changes in parameters that affect nuclear safety.
- c. Level of defence 4a: the maximum calculated k_{eff} value must be less than 1.0 including all uncertainties and tolerances. Realistic boundary conditions may be taken into account.
- d. If fuel burnup is taken into account in the criticality analyses, then the requirements of standard DIN-25471 or ANSI/ANS-8.27 must be met.
- e. With regard to the criticality safety of fuel assemblies in the transport casks located in the fuel storage pool, it is necessary to prove compliance with the requirements of the applicable set of regulations for the transport of dangerous goods (IAEA Safety Standard SSR-6).
- f. The limits for the pool water temperature must not be exceeded even when the fuel storage pool is full, including complete unloading of the reactor core. In each case, the water temperature is to be calculated by assuming the least favourable conditions with regard to decay heat and cooling conditions.
- g. Adequate water quality must be ensured in order to limit the corrosion of fuel assemblies, core internals and storage components.
- h. In the event that defective fuel rods are stored temporarily, it must be ensured that there is no significant additional contamination of the coolant water or storage installations.
- i. Severely damaged fuel assemblies or fuel rods must be stored in quivers for containment purposes.
- j. The free capacity of the fuel storage pool must always allow unloading of the entire reactor core.

6.5.4 Modification work on fuel assemblies

- a. The work instructions for modifying fuel assemblies must include, as a minimum requirement, the intended procedures, applicable safety regulations and checks following completion of these procedures.

- b. Radiation protection measures must be planned when spent fuel assemblies are to be modified.
- c. Criticality safety ($k_{\text{eff}} < 0.95$) must be demonstrated during these modifications.
- d. The modification procedures and the results of the checks must be documented.
- e. After modification work is completed, it must be demonstrated that the fuel assembly is in a condition that corresponds to the design or meets the safety requirements.
- f. Annex 2 sets out the reporting and approval obligations in the event of modification work on intact and defective fuel assemblies.

6.6 Disposal

- a. All fuel assemblies must be suitable for disposal, and a corresponding disposal concept must be provided (interim storage facilities, deep repositories).
- b. In accordance with the disposal concept, the following information must be specified as a minimum requirement:
 1. the intended package design (manufacturer, model) for transport or dual purpose (transport/storage) and the minimum required decay time in the operational fuel storage pools or the wet storage facility before transfer to the interim storage facility;
 2. the relevant nuclide inventory at the time of unloading from the reactor core, the earliest possible commencement of interim storage, and placement in the deep geological repository;
 3. development of the post-irradiation heat output for the maximum fuel assembly burnup over the period from unloading from the reactor core to placement in the deep geological repository;
 4. proof of the integrity of the fuel cladding tubes during interim storage;
 5. proof of suitability for final disposal.

7 Control rods

7.1 Design

7.1.1 General requirements

- a. Controls rods must be designed to reliably guarantee their operation during shutdown of the reactor core, to maintain subcriticality, and for the purposes of operational control.
- b. The design must take account of the loads, as well as changes in material properties and in geometry that occur during operation, e.g. as a result of external and internal mechanical, thermal, chemical and radiation-induced effects.
- c. If the reactor is controlled using the control rods, it must be ensured that
 1. a sufficient reserve of effectiveness is available at all times for safety-relevant requirements (see also section 5.5, point d),
 2. safety-relevant requirements take precedence over operational requirements and
 3. the safety-related function cannot be impaired.
- d. It must be possible to inspect control rods.
- e. Activation of materials that pass into the coolant must be limited by appropriate selection of materials. This applies in particular to the cobalt content.
- f. Evidence must be provided of a clear path for the control rods (compatibility).

7.1.2 Mechanical design

- a. The mechanical design of the control rods must take account of plant-specific requirements in terms of the path, speed, shutdown time and number of notches.
- b. The mechanical design must take account of the following loads:
 1. dynamic forces, especially during notchwise travel, during a scram and in the event of earthquakes,
 2. friction-related compressive and tensile stresses during the movement of the control rods,
 3. pressure build-up due to gaseous reaction products (excess internal pressure),

4. swelling expansion of the absorber,
 5. differences between the internal control rod pressure and the coolant pressure,
 6. bending vibrations due to coolant flow,
 7. thermal loads.
- c. The stresses resulting from possible combinations of individual loads must not exceed the material-specific limits.
 - d. The strain on the components under internal pressure must be limited so that the wall integrity is guaranteed.
 - e. For cyclical loads, the fatigue limits must be determined and used to derive the admissible number of load cycles.
 - f. Account must be taken of interactions with neighbouring plant sections. An estimate must be made of mechanical wear for the intended operation.
 - g. The mechanical lifetime must be determined.

7.1.3 Thermal design

- a. The admissible internal pressure must not be exceeded.
- b. Level of defence 1: in the pressurised water reactor, coolant boiling at the surface of the control rod or in the coolant channel between the guide tube and the absorber rod must be ruled out.
- c. Melting of the absorber must be ruled out. If this cannot be demonstrated for accident category 3 of level of defence 3, it must be proven that the absorber material remains in the control rod in accordance with the design.

7.1.4 Nuclear design

- a. The absorber material must meet the requirements for the control and shutdown systems (section 5.5).
- b. The effectiveness of the absorber material must be proven, taking account of burnup and the planned operation of the plant. The nuclear lifetime must be determined.
- c. When control rods are replaced, their effectiveness must be consistent with the safety analyses of the plant for the planned operating life.

7.2 Manufacturing

- a. The control rods must be procured in accordance with the procedures set out in the operating organisation's management system.
- b. The operating organisation must ensure that manufacturing of the control rods complies with the applicable Swiss statutory provisions and regulatory and technical requirements.
- c. The operating organisation must check that the supplier's quality management system is suitable for ensuring that the control rods are manufactured in accordance with the design.
- d. Once delivery is completed, a summary report with the following content must be submitted to ENSI for each campaign:
 1. an audit plan,
 2. a list of the new and requalified processes,
 3. design modifications,
 4. a summary of deviation reports, including justification for acceptance of the deviation,
 5. findings and measures derived from the findings,
 6. a summary assessment.

7.3 Operation

- a. Absorber burnup and the mechanical lifetime must be monitored.
- b. Plant-specific and international operating experience with the respective operating rods must be collected and evaluated.
- c. The use and replacement of control rods must be planned in accordance with the operating lifetime on which the design is based and in accordance with operating experience.

7.3.1 Lead assemblies

- a. Lead assemblies must be used in order to evaluate the plant-specific performance of new types of control rods and control rods that have undergone significant modifications.
- b. In the absence of operating experience under comparable conditions in other plants, lead assemblies must be used for at least two cycles prior to the use of a larger number of control rods. If operating experience from other plants is available, a lead time of one cycle is sufficient.

- c. The number of lead assemblies must be limited, taking into account the importance of the modification.
- d. In addition to using lead assemblies, a dedicated inspection programme must be planned. ENSI must be informed of changes to the inspection plan.

7.3.2 Inspections

- a. Regular inspections must be carried out in order to check that the control rods are suitable for further operation. ENSI must be informed of changes to the inspection plan.
- b. If a condition outside the design limits is identified, the relevant control rod must be replaced or repaired. If operation is to continue with an affected control rod for a limited period of time, evidence must be provided that the intended protective function is always fulfilled.
- c. Inspections must be carried out and evaluated by appropriately qualified or trained personnel and in accordance with the requirements of the management system.
- d. Qualified inspection equipment and tools must be used.
- e. During inspections, account must be taken of the radiation protection guidelines.
- f. The results of the inspection must be documented and assessed by the operating organisation.

8 List of references

DIN 25471, Kritikalitätssicherheit unter Anrechnung des Brennelementabbrands bei der Lagerung und Handhabung von Brennelementen in Brennelementlagerbecken von Kernkraftwerken mit Leichtwasserreaktoren, May 2009

ANSI/ANS-8.27, Burnup Credit for LWR Fuel, 2008

IAEA Safety Standard SSR-6, Regulations for the Safe Transport of Radioactive Material, 2012

This Guideline was approved by ENSI on 29 January 2015.

The Director General of ENSI: signed: H. Wanner

Annex 1: Terms and definitions (as per the ENSI Glossary)

Shutdown margin

The shutdown margin is the reactivity of the reactor core once it has been placed in a subcritical state by means of a shutdown using the systems provided for this purpose.

Shutdown system

A shutdown system is a system that is capable of bringing the reactor core into a subcritical state and keeping it in that state.

Fuel assembly

The fuel assembly consists of components such as the spacers and the fuel rods. The fuel assembly structure is made up of all of the components of the fuel assembly with the exception of the fuel rods.

Fuel rod

The fuel rod is a gas-tight tube filled with nuclear fuel and sealed at both ends.

Empirical correlation

Empirical correlations are derived from experiments and describe a relationship between physical input and output quantities. They can take the form of functions or tables.

k_{eff}

The neutron multiplication factor k_{eff} gives the calculated ratio of neutron production to neutron loss due to absorption and leakage in a defined system. If the neutron source and neutron sink are in equilibrium, a self-sustained process of neutron production begins and the system is said to be in a critical state. Otherwise, the state is said to be either subcritical ($k_{\text{eff}} < 1$) or supercritical ($k_{\text{eff}} > 1$).

Criterion

A criterion is a mathematical expression (equation, inequality) that clearly states the relation between a safety-relevant characteristic, the uncertainties to be considered, and the limit. It has a defined range of validity.

Critical boiling condition

A critical boiling condition is present both at the onset of departure from nucleate boiling (DNB) and on dryout of the heating surfaces.

Reactivity-initiated accident (RIA)

A reactivity-initiated accident is an accident caused by the inadvertent addition of reactivity to the reactor core.

Reactor core

The reactor core includes the reactor's fuel assemblies, control rods and neutron measurement systems together with the detector tubes.

Level of defence 4a

Level of defence 4a is the part of level of defence 4 that deal with preventive accident management. The aim of level of defence 4a is to maintain control of beyond-design-basis events without core meltdown or, in storage, fuel assembly meltdown.

Safety-relevant characteristics

Safety-relevant characteristics are physical quantities that unambiguously characterise the safety-relevant status of a system.

Safety-relevant parameters

Safety-relevant parameters are quantities, variables or boundary conditions that have a significant impact on the safety-relevant characteristics or the results of analyses.

Control rod

A control rod acts as a neutron absorber both for shutting down the reactor core and for controlling the power. It consists of the control rod structure and the absorber.

Validation

Validation is the process of demonstrating that the properties of a calculation model reproduce the real conditions in question with sufficient accuracy for the model's intended use.

Verification

Verification is the process of demonstrating that the implemented calculation model is an accurate representation of its conceptual description (specification).

Annex 2: Modification work on fuel assemblies

Category	Type of modification work	Reporting obligations	Approval obligations
I	Modification work on defective fuel assemblies containing failed fuel rods, from which activity has been released	<ul style="list-style-type: none"> • Incident (section 5.1.1.2, point e, of Guideline ENSI-B03) 	<ul style="list-style-type: none"> • Repair • Reuse
II	Modification work on defective fuel assemblies, from which no activity has been released	<ul style="list-style-type: none"> • Incident (section 5.1.1.2, point e, of Guideline ENSI-B03) 	<ul style="list-style-type: none"> • Reuse
III	Modification work on intact fuel assemblies		<ul style="list-style-type: none"> • Reuse

Examples

Repair work as a result of fuel rod failure and with release of activity falls within **Category I**. Removal from the fuel channel or withdrawal of sub-bundles in boiling water reactor fuel assemblies within the scope of a visual inspection of fuel assemblies containing failed fuel rods is not considered to be part of the repair work.

Repair work as a result of fuel assembly defects without any release of activity falls within **Category II**.

Modification work on intact fuel assemblies within the scope of inspections in accordance with the inspection programme (e.g. dismantling and reassembly) falls within **Category III**.

Preventive replacement of fuel assembly components on the basis of non-reportable findings falls within **Category III**.

Replacement of fuel assembly components, including fuel rods, for external inspections (e.g. investigations in the hot cells) falls within **Category III**.

Replacement of fuel rods within the scope of a second-life fuel rod programme falls within **Category III**.

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