# Design of the Pressurized Water Reactor Core

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# SÚJB SAFETY GUIDES

Safe utilisation of nuclear energy and ionising radiation

# Design of the Pressurized Water Reactor Core

## Nuclear Safety

## BN-JB-3.2 (Rev. 0.1)

Státní úřad pro jadernou bezpečnost

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**Nuclear safety**

**Safety Guide DESIGN OF THE PRESSURIZED WATER REACTOR CORE**

**BN-JB-3.2 (Rev. 0.1)**

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# 1. ABBREVIATIONS, DEFINITIONS AND TERMS

## ABBREVIATIONS

| | |
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| RC | Reactor core |
| BEPU | Best Estimate Plus Uncertainties method |
| SG | SÚJB Safety Guide |
| SFSP | Spent Fuel Storage Pool |
| BUC | Burnup Credit |
| DNBR | Departure from nucleate boiling ratio |
| I. O | Primary Circuit |
| IAEA | International Atomic Energy Agency |
| NI | Nuclear installation |
| CCs | Core components |
| KHP | FE cladding leak-tightness control device |
| LaC | Limits and Conditions of Safe Operation |
| LOCA | Loss of Coolant Accident |
| LTA | Lead Test Assembly |
| OS | Cask |
| PBC | Partial Boron Credit |
| PCMI | Pellet – cladding mechanical interaction |
| FE | Fuel element |
| PIIP | Post Irradiation Inspection Programme |
| FA | Fuel assembly |
| RIA | Reactivity Initiated Accident |
| RC SSCs | Reactor core systems, structures and components |
| SÚJB | State Office for Nuclear Safety |

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| WENRA | Western European Nuclear Regulators' Association |

## DEFINITION AND TERMS

For the purposes of this Guide, the following definitions apply:

**Abnormal operation** A state of a nuclear installation deviating from normal operation which is expected to occur which does not cause any significant damage to systems, structures and components important to safety and after which the nuclear installation is capable of normal operation with no repair (after removing the cause of initiating event)

**Safety limit** A limit on a parameter characterising the state of a nuclear installation or any other expression of safety, technical or administrative condition beyond which there is a risk to nuclear safety, radiation protection or technical safety as a result of failure of system, structure or component

**Shutdown margin** An existing (or an instantaneous achievable) amount of reactor subcriticality following compensation for the temperature and power effect of reactivity assuming all full-length rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth that is assumed to be fully withdrawn

**Burnup Credit - BUC** Taking account for the reduction in reactivity of the fuel assemblies placed in the SFSP due to the use of nuclear fuel in the reactor core (nuclear fuel burnup), used in calculating subcriticality for the SFSP

**Physical barriers** A barrier that prevents the release of radioactive material to the environment

**Coolant** Any liquid used to remove heat from the fuel system, reduce the speed of fast neutrons (functioning as a moderator) and ensure a homogeneous dispersion of soluble absorber in the reactor core

**Instrumentation** The device for measuring key safety parameters of the reactor core (e.g. temperature or neutron flux measurement) located inside or outside the reactor vessel

**Nuclear installation** A facility or plant comprising a nuclear reactor using fission chain reaction; a spent nuclear fuel storage facility, and a fresh nuclear fuel

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| | storage facility, if not part of any other nuclear installation |
| **Core components** | RC SSCs that are placed during operation in the reactor core (material clusters, control rods, neutron sources, discrete burnable absorber, hydraulic plugs for guide tubes, etc.) |
| **Acceptance criterion** | A safety, technical or administrative condition or limit value of the quantity determining its acceptance in terms of nuclear safety, radiation protection, technical safety, radiation situation monitoring, emergency management or security |
| **Limits and Conditions** | The Limits and Conditions (LaC) consists of a set of requirements, compliance with which means that the performance of activities is considered safe (a set of uniquely defined conditions, for which it has been proved that the operation of a nuclear installation is safe) |
| **Load follow** | Reactor operation under load following according to the needs of the grid or known power diagram |
| **Leak** | A failure of the integrity of FE cladding resulting in the loss of leak-tightness and the release of radioactive material from the sealed area of FE cladding into the coolant |
| **New types of FA** | Such a type of FA for which a design modification was made, which requires authorisation from the State Office for Nuclear Safety |
| **Cask load** | A load exhibiting the worst characteristics of all relevant variants in terms of the objective pursued (e.g. acceptance criteria) and the parameters of which are subject to safety analyses (i.e. the load is sufficiently conservative to cover the expected real loads) |
| **Reactor shutdown** | Bringing the reactor core to a subcritical state and holding it in this state by the controlled insertion or drop of control rods or by the introduction of soluble absorber |
| **On-line and off-line sipping** | The method for detecting failed FAs. The on-line sipping is carried out before or during FA unloading from the reactor core into the SFSP. The off-line sipping is carried out on the predetermined FAs (suspected of failure, confirmation of tightness in case of non-functionality of the on-line sipping, etc.) in order to confirm or disprove failure |
| **Fuel element** | The nuclear material hermetically sealed within the FE cladding (it contains: FE cladding, fuel pellets, filling gas, springs, plugs, burnable |

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| | absorber, etc.) |
| **Fuel assembly** | A group of fuel elements, which allows safe handling of nuclear fuel and is loaded into the nuclear reactor as a single unit (it contains: FE, top, bottom, spacer and turbulent grids, springs, connecting components, etc.) |
| **Fuel system** | A configuration of fuel assemblies and other components of the reactor core, provided for under the design of a nuclear installation, necessary to control reactivity and to maintain the design structure of fuel assemblies in the reactor core (the fuel system consists of all FA components and core components) |
| **Partial Boron Credit - PBC** | Including the effect of soluble absorber in coolant in the calculations of subcriticality for the SFSP |
| **Pellet - Cladding Interaction - PCI** | A comprehensive fuel pellet-FE cladding interaction where there is direct contact between pellet and FE cladding, which can result in failure of FE cladding |
| **FE cladding** | FE cladding is a hermetically sealed tube of FE filled with inert gas that houses the fuel pellets |
| **FE failure** | Disruption of the leak tightness of FE cladding, which allows radioactive material to escape from the fuel element |
| **Damage to FEs/FAs/CCs** | FEs, FAs or CCs are considered damaged if any design limit is exceeded, unless it is proven otherwise |
| **Design margin** | A difference between the design limit, defined for specific physical parameters, and the maximum/minimum calculated or measured value of the physical parameter after including all uncertainties |
| **Design limit** | An acceptance criterion that is used to assess the capability of a nuclear installation to perform its function envisaged by the design of a nuclear installation; the design limit is particularly the limit set out by legislation or the acceptance criterion derived on its basis, which corresponds to the method for assessing the capability of a nuclear installation to perform its function envisaged by the design of a nuclear installation |
| **Design of the reactor core** | Design of the new reactor core (in the context of the construction of a new NI), design modification of the reactor core or its part |

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| **Applicable operational experience and experimental measurements** | The results and experience gained in any other nuclear or experimental installation, which are used to demonstrate the characteristics of the design of reactor core, see para. (4.6) and (4.11) |
| **Transverse flow** | Coolant flowing among the FAs in the reactor core caused by different local hydraulic resistance or capacity, different location of spacer grids or compensation of uneven flow distribution at reactor core inlet, which mainly affects blanket-free structures of the FA |
| **Mixed core** | Mixed core is such a reactor core, with two or more types of fuel loaded at the same time, which differ in their designs in terms of thermal-hydraulic, thermal-mechanical or neutron-physical properties to change the assumptions for the safety assessment conducted for the current fuel |
| **Current FAs, FEs or CCs** | FEs, FAs or CCs that are operated/located at least for one campaign in the reactor core |
| **FE type** | A type of structural (e.g. spiral FE) or any other design of the fuel element (e.g. pellets without a central hole), which differs in some of the characteristics of the FE |
| **All states of the NI** | The states cover normal and abnormal operation, design basis accidents and design extension conditions without the core melting |
| **Burnable absorber** | A material with a high effective absorption cross-section for the capture of thermal neutrons, which is changed, after neutron absorption, to a material with a very low effective absorption cross-section for thermal neutrons |

# 2. INTRODUCTION

## REASON FOR THE ISSUE

(2.1) The State Office for Nuclear Safety is a central state administration body in the field of utilisation of nuclear energy and ionising radiation.

(2.2) Within the framework of its respective powers and competences, in accordance with the principles of the activities of administration bodies and international practice, the Office issues guides that elaborate requirements for nuclear safety, technical safety, radiation protection, radiation situation monitoring, emergency management, and security.

(2.3) The reason for issuing this Safety Guide - **DESIGN OF THE PRESSURIZED WATER REACTOR CORE** is the requirement for ensuring the safety of fuel system design in accordance with Czech legislation, taking account of the international recommendations defined by WENRA, IAEA, US NRC and major European operators of nuclear power plants. In Czech legislation, the requirements are set out in Act No. 263/2016 Coll., the Atomic Act [1], as well as in the Decree of the State Office for Nuclear Safety on basic design criteria for a nuclear installation No. 329/2017 Coll. [2].

(2.4) This Guide should be used in the preparation of the supporting documents submitted as part of the application for a licence for the siting of a nuclear installation or for the carrying out of design modification of the pressurized water reactor core and associated RC SSCs. This Guide can be reasonably used in the preparation of the supporting documents submitted as part of the application for a licence for the siting of a nuclear installation.

## OBJECTIVE

(2.5) The Safety Guide is particularly intended for the applicant for a licence for the operation or for the carrying out of design modification of the pressurized water reactor core and associated RC SSCs, offering the applicant the possible procedure, compliance with which ensures that its activities in the given area are in compliance with the requirements set out in the Atomic Act, its implementing regulations and IAEA recommendations.

(2.6) The objective of this Safety Guide - **DESIGN OF THE PRESSURIZED WATER REACTOR CORE** is to elaborate the requirements for design or design modification of the reactor core and associated RC SSCs.

## SCOPE OF APPLICATION

(2.7) The Safety Guide focuses on nuclear installations covered by the Convention on Nuclear Safety – nuclear power plants; to a limited extent, its principles and procedures can be applied to other nuclear installations with the use of a graded approach.

## VALIDITY AND EFFECT

(2.8) The Safety Guide or its last revision shall become valid on the date of its publication on www.sujb.cz; the effective date is given on page 2. The Safety Guide shall be revised in the light of new knowledge of science and technology, comments of the public obtained and experience with its use in practice.

# 3. SCOPE AND BASIS

## SCOPE

(3.1) The Safety Guide elaborates the general requirements set out in Section 46 of Act No. 263/2016 Coll. [1], and Decree on basic design criteria for a nuclear installation No. 329/2017 Coll. [2] concerning the design and design modification of the reactor core and core SSCs.

(3.2) This Safety Guide is built on BN JB-1.12 PROPOSED CONTENT OF SAFETY ANALYSIS REPORTS [14], which sets out general requirements for the content of safety analysis reports. It builds further on BN-02.2 SPENT NUCLEAR FUEL STORAGE IN SEPARATE NUCLEAR FACILITIES [15].

(3.3) The design of the reactor core should meet the requirements set out herein as well as in other guides issued by the State Office for Nuclear Safety, see lit. [14] and [16] in order to meet all legislative requirements relating to the design of the reactor core.

(3.4) The verification of the safety limits of reactor core, initiating events, safety analyses of Chapter 15 of the Pre-operation Safety Analysis Report, and design extension conditions are described in lit. [16], [17], [18] and [19].

(3.5) This Safety Guide focuses on design and design modifications of the reactor core, i.e. neutron, thermal-hydraulic, thermal-mechanical and mechanical part of the design, problems related to mixed cores, PBC, BUC and other associated problems with the use of experience from local and world practice.

(3.6) This Safety Guide is primarily intended for fuel assemblies with fuel elements that consist of zirconium-based FE cladding and uranium oxide pellets, with the enrichment below 5% of U235 including uncertainties.

(3.7) For the purposes of the Safety Guide, the systems, structures and components of the reactor core below are used, which are placed inside the reactor pressure vessel. Fission chain reaction occurs in the core. In terms of basic segmentation, the reactor core consists of fuel system (FAs and RC SSCs), coolant and other components and structures (e.g. instrumentation and internals of the reactor pressure vessel). The systems, structures and components of the reactor core include the following:

- The fuel element (FE), which consists of uranium oxide pellets (with or without burnable absorber) placed in FE cladding, filling gas, spacer spring and corresponding terminals;
- The fuel assembly (FA), which consists of fuel elements, structures and components (e.g. guide tubes, spacer grids, top nozzle, bottom nozzle, shroud),which hold the FEs and FAs in the predefined geometry;
- The systems, structures and components of the reactor core (RC SSCs), which consist of components placed during operation in FA guide tubes or transported into the reactor core during control and shutdown of the reactor core (e.g. control rods);
- In-core instrumentation;
- Coolant, which removes heat from the reactor core, functions as a moderator at the same time to reduce the speed of fast neutrons and contains soluble absorber;
- Part of the internals of reactor pressure vessel in direct contact with the FA (support plates, core baffle, etc.).

(3.8) The states of a nuclear installation mean normal and abnormal operation, design basis accidents and design extension conditions without the core melting. The analyses relating to fuel melting in the reactor core and SFSP are not part of this Safety Guide.

## STRUCTURE

(3.9) The basic structure of this Guide corresponds to the usual structure of the safety assessment of the design of reactor core by individual professional disciplines and the fields of assessment, and includes: neutron-physical characteristics, thermal-hydraulic properties, thermal-mechanical properties, mechanical properties of structures, nuclear reactor shutdown systems, control systems and core monitoring systems, and nuclear fuel administration and management. Within the individual disciplines and fields, the assessment discusses design requirements and relevant design criteria. Special attention shall be paid to problems related to mixed cores.

(3.10) The content of this Guide is based on the draft Safety Guide of the IAEA for the design of the reactor core for pressurized water reactors IAEA DS488 lit. [8] and on the document - Operation and Licensing of Mixed Cores in Water Cooled Reactors IAEA TECDOC No. 1720 [9].

# 4. GENERAL REQUIREMENTS FOR DESIGN AND DESIGN MODIFICATION OF THE RECTOR CORE

(4.1) The basic requirements for design and design modification of the reactor core are laid down in Act No. 263/2016 Coll. [1].

(4.2) Further requirements for the design of fuel system are specified in the Decree on basic design criteria for a nuclear installation No. 329/2017 Coll. [2].

(4.3) The requirements for the completeness and comprehensibility of management system documentation are set out in Decree No. 408/2016 Coll., on management system requirements [3].

(4.4) The basic requirements for design of a nuclear installation and design modifications are set out in the IAEA SSR-2/1 (Rev. 1) [7]. Further recommendations for the field of core designing, which serve as a basis for this Safety Guide, are elaborated in the document - WENRA Safety Reference Levels for Existing Reactors [11] (Issue E, G a Q).

(4.5) The requirements for the content of safety documentation for Chapter 4 of the Operational Safety Analysis Report, which serve as a source for this Safety Guide, are set out in BN-JB-1.12 (Chapter 4) lit. [14], US NRC NUREG – 0800 (Chapter 4) lit. [12] and US NRC Regulatory Guide 1.70 lit. [13].

(4.6) **Applicable operational experience and experimental measurements** should be gained for FAs or core components of the same design and properties, which are critical for the transferability of the result in question. The results and experience should be gained in such a nuclear or experimental installation where they are operated and examined under the same conditions as those present/to be present in the licensee's nuclear installation; this is particularly the case with regards to the parameters and mode of operation.

(4.7) The design of the reactor core should determine the values of key parameters and design limits, for which the magnitude of margins should be evaluated. Where any parameter is not defined by the design, such margins should be set to ensure the fulfilment of design limits in all states.

(4.8) The design of the reactor core and associated SSCs should be such so as not to cause failure of any of the design limits for fuel (new and current type of fuel) and shall not exceed the design limits for the NI. The new type of fuel should be compatible with the current type of fuel placed in the core from thermal-hydraulic, mechanical, thermal-mechanical and neutron-physical points of view.

(4.9) The reactor core should be designed so as to ensure the performance of relevant safety functions in all states, compliance with the relevant design limits and compliance with the acceptance criteria. Protection, control and monitoring systems of the core should be designed and configured so as not to exceed the set design limits of the fuel system in all states. The margins should be properly justified, documented and their possible utilization should be controlled. For key safety parameters, 15% margin is recommended, unless different magnitude of the margin is proved sufficient.

(4.10) The mixed core is characterized in that there are two and more types of fuel operated in the core at once, which have such an impact on the current safety assessment that results in the change in the assumptions for safety analyses. They are the changes with impacts in any of the following areas:

a) Neutron-physical – material composition and geometry of the fissile material and burnable absorber;
b) Thermal-hydraulic – change in the size of pressure loss (total and local), thermal-hydraulic properties (DNBR);
c) Mechanical – change in static and dynamic characteristics of fuel design (deformations, vibrations, load during LOCA and earthquake);
d) Thermal-mechanical – default values of the enthalpy of pellets, initial quantity of fissile substances, FE cladding material.

(4.11) If the **applicable operational experience and experimental measurements** in the operation of the mixed core cannot be demonstrated (i.e. co-location of the new and operated type of nuclear fuel in the core), the LTA programme should be proposed. The LTA programme should establish the requirements for measuring the predetermined parameters and there should be a limit in time for how long this programme will be implemented (however, the minimum of three years is recommended). In the LTA programme, the number of FAs should be determined, which will be loaded into the reactor under the programme (the maximum of 12 fuel assemblies is recommended). This minimum number should be properly justified, supported by analyses and in accordance with the specific objective of the LTA programme. The requirements for the LTA programme and its evaluation are set out in para. (6.1) - (6.24).

(4.12) The operator should ensure that the nuclear material is efficiently used (minimization of parasitic capture and release of neutrons) and at the same time, that the radiation load applied to the reactor pressure vessel is minimized, while maintaining sufficient margins according to para. (4.7) and (4.9).

**Principles of defence-in-depth**

(4.13) The design of the reactor core should be robust (it considers all events that may occur during operation of the NI), reliable and the margins should be defined for the design. The three fundamental safety functions will be performed by complying with these requirements for cask loads.

(4.14) The design of the reactor core should be such as to fulfil the three fundamental safety functions in all states. The fundamental safety functions are as follows:

- Ensuring the safe shutdown of the nuclear reactor and holding it in a subcritical state on a long-term basis (including secondary criticality) and ensuring a sufficient subcriticality in the SFSP and in other nuclear fuel handling and storage systems;
- Removing heat from the reactor core and in the long term, removing residual heat from the reactor core, spent fuel storage pool and other handling and storage systems;
- Preventing the release of radioactive material into the environment and its limitation to the lowest possible level when released.

(4.15) The design of the reactor core should ensure the following preventions:

- Damages to the integrity of physical barriers;
- Failure of a barrier when damaged;
- Failure of a barrier as a consequence of failure of another barrier.

(4.16) The design of the reactor core should consider the following physical barriers:

- Fuel pellets;
- FE cladding;
- Primary circuit, and
- Containment,

that prevent the release of radioactive material to the environment.

(4.17) The fuel pellets and PE cladding represent the first two barriers, which should be able of preventing the release of radioactive material into the coolant in normal and abnormal operation. For design basis accidents and design extension conditions without the core melting, subcriticality and ability of long-term cooling of the core and the SFSP should be ensured.

(4.18) The individual RC SSCs should be divided into safety classes by their function and relevance in accordance with the requirements set out in lit. [1] and Annex 1 lit. [2].

(4.19) The FA and its individual components are significant contributors to the performance of all three fundamental safety functions. The FE cladding should be classified in the safety class in accordance with lit. [1], [2], because:

- The tightness and structural integrity of the FE provide the first and second barriers of defence-in-depth in operational states of the NI, i.e. prevention of the release of radioactive material into the environment;
- Maintaining the geometry of the FA and FE bundle facilitates the reactor scram and to maintain the sufficient cooling of the core in emergency conditions of the NI, to maintain the integrity of physical barriers and the limited rate of their failure in emergency conditions, thus ensuring the

performance of fundamental safety functions. In addition, it facilitates the handling of the FA as well as the remedy of an accident in the NI.

(4.20) The control rods, as a part of the shutdown system, should be classified in the safety class in accordance with lit. [1], [2], because their failure can lead to the failure of reactor scram system and indirectly to the failure of FE integrity and structural geometry of the FA.

(4.21) Justification for the classification of individual RC SSCs should be described in detail in accordance with the rules set out in lit. [1] and [2].

**Engineering practices in ensuring safety**

(4.22) The design of the reactor core should demonstrate a sufficient safety in all states. Sufficient safety can be demonstrated by calculations, on the basis of documented operating experience, experimental measurement or by using a combination of the procedures above. Analyses and calculations should be carried out by using such computer codes that were assessed in the Expert Assessment Committee in accordance with the requirements of the State Office for Nuclear Safety, or the verification of accuracy and adequacy of the codes used should be provided on the basis of the calculations of benchmark tasks. Operating experience and experimental measurement should be applicable.

(4.23) The design and design modifications of the reactor core should be assessed by using the safety assessment that will still be developed and improved in line with the new knowledge gained from both operating experience and research. The assessment procedure is described in detail in Chapter 5 hereof.

(4.24) The design of the reactor core should take account of the conditions for nuclear fuel management and storage set out by the manufacturer of the FA and the core components for all types of the FA and the core components operated jointly in the reactor. The design shall include requirements for storage and handling of all FAs and core components that will be stored together.

## GENERAL REQUIREMENTS FOR THE DESIGN OF SYSTEMS, STRUCTURES AND COMPONENTS OF THE CORE

(4.25) The RC SSCs, whether as a whole or individually, should be designed so as to be operable, reliable and capable of performing their functions in all states. Therefore, the design of the RC SSCs should meet the specific design requirements referred to in para. (3.8) - (4.38).

**External effects**

(4.26) The design of the RC SSCs shall include seismic qualification in accordance with the requirements referred to in Chapter 3.9 of lit. [14].

**Design limits**

(4.27) The design limits for individual RC SSCs should be set for all operational states of the NI. The limits should be set so as to ensure the performance of the safety functions concerned.

(4.28) The design limits should be set so as to meet the requirements on defence-in-depth referred to in para. (4.13) - (4.19).

**Engineering rules for the design**

(4.29) The engineering rules for the design of the RC SSCs are the methods and procedures achieving the appropriate proven and repeatable design in accordance with the specification, while meeting the safety requirements. The rules shall include the following:

- The application of appropriate technical standards;
- The application of verified computer codes and engineering approaches;
- The conservative or BEPU approach to safety assessment;
- The specific analyses for the verification of design reliability (e.g. experimentally or by operating experience);
- Verification and testing;
- Compliance with the operational limits and conditions;
- Taking account of operating experience.

**Design reliability**

(4.30) The RC SSCs should be designed so as to meet the strict requirements on reliability, having regard to their safety significance. The calculations, production and all associated activities should be controlled so as to achieve a high level of quality and reliability of the entire system/plant. The level of reliability should be appropriately demonstrated (e.g. experimentally or by operating experience).

(4.31) The specific requirements on design reliability are described in para. (5.28).

**Limits and conditions of safe operation**

(4.32) The LaC shall include the conditions for safe operation of all RC SSCs in the reactor and their safe storage in the SFSP for normal and abnormal operation.

(4.33) The LaC should be proposed as to meet the relevant design criteria for all RC SSCs in all states. For more detailed specification see lit.[2] and [6].

**Requirements for the design of the RC SSCs**

(4.34) The design of the RC SSCs should be prepared so as to facilitate the testing, inspections, repairs, handling, calibrations or maintenance of the FEs, FAs and core components.

(4.35) The design of the RC SSCs should further ensure that the core components, FEs, and FAs withstand handling during transport, storage, assembly and reload of the FAs without causing damage.

(4.36) The design of the reactor core should be reassessed in respect of each design modification of the reactor core and associated systems, e.g.:

- The significant modification of the FA or FE design or the change to the fuel type (e.g. MOX);
- The increase in the burnup above the actual limit (based on the results of research);
- The significant extension of the design duration of fuel campaign (e.g. changeover from 12- to 18-month campaign, etc.);
- The increase in thermal output of the reactor.

(4.37) The design of the RC SSCs should be such as to prevent damage to the FEs, FAs and core components during operation under non-standard conditions (e.g. during physical tests, degrading chemical conditions).

(4.38) The design of the RC SSCs should be such as to ensure the drop of control rods into the core up to the lower limit position at the required time and to prevent the deceleration or stuck of the control rod inside the guide tubes. Failure to perform this safety function may be the result of bowing of the FA. The capability of the drop of control rods and its duration (time of drop) should be periodically tested.

## DESIGN ANALYSES OF THE CORE

(4.39) The analyses should be carried out for all states of the NI. A conservative approach should be used for the analyses and evaluations, unless different approaches are proposed by the applicant for specific cases.

(4.40) The following major factors should be considered in the analyses:

- The expected extent of operational states;
- The temperature coefficient of reactivity for the fuel (Doppler coefficient);
- The temperature coefficient of reactivity for the coolant;
- The void coefficient of reactivity for the coolant and the moderator;
- The rate and efficiency of change of the concentration of soluble absorber in the coolant and the moderator;
- The rate and magnitude of insertion of positive reactivity caused by the movement of control rods or changes in process parameters;
- The rate and magnitude of insertion of negative reactivity associated with a reactor trip (the rate of insertion of override reactivity);
- The performance characteristics of safety system equipment including the changeover from one mode of operation to another (e.g. from the injection mode for emergency core cooling to the recirculation mode);

- The decay of xenon and other neutron absorbers in the long-term core operation analysis.

The above mentioned factors should be set out so as to cover all events that can occur during operation of the NI for all FA types.

(4.41) Analyses should demonstrate for all states of the NI that the fuel design limits applicable to the specified states of the NI are not exceeded.

(4.42) The specific effects of fuel behaviour such as ballooning, rupture of the FE cladding, and twisting and bowing of FEs and FAs should be included in the analyses.

(4.43) The evaluation of the core characteristics should clearly define which analyses or parameters are applicable to the cask load and can be considered as generic and which analyses should be repeated for each specific load. This scope should be defined for all areas of the design of the reactor core (neutron-physical, thermal-hydraulic, thermal-mechanical, and mechanical). All analyses should be properly documented, independently verified and carried out by qualified personnel, all in accordance with the requirements set out in lit. [3] and [4].

(4.44) The computer codes on the same methodological base (using the similar mathematical base of calculation) should be used for each of the areas of the design core analyses (in particular for neutron-physical, thermal-hydraulic, thermal-mechanical and mechanical behaviour). The sensitivity analysis should be conducted using the methodology same as that conducted for the current design of the reactor core/nuclear installation.

# 5. SPECIFIC REQUIREMENTS FOR THE DESIGN OF THE REACTOR CORE AND ASSOCIATED CRITERIA AND LIMITS

## GENERAL

(5.1) The design of the reactor core should be such so as not to exceed any of the design limit for the FE and the FA (the new and current FA or FE) and not to compromise nuclear safety of the NI (the margins should be sufficient so as not to compromise nuclear safety of the NI). The design of the reactor core should be compatible with the current FAs and FEs in the core from neutron-physical, thermal-hydraulic, mechanical, and thermal-mechanical points of view.

(5.2) The specific design limits should be set taking account of the adequate margins, specifically for the evaluable parameters such as in particular:

- The maximum local and mean linear or overall performance of the FE;
- The minimum DNBR;
- The maximum temperature and enthalpy of the fuel pellet;
- The maximum temperature of FE cladding.

The design analyses should adequately take account of the uncertainties in the values of process parameters (e.g. reactor power, coolant flow through the core, flow through the core bypasses, inlet coolant temperature and pressure, nuclear properties, uncertainties in the correlation of DNBR and engineering coefficients of uncertainty in the hot channel), neutron-physical properties of the reactor core and the calculation methods used in the evaluation of thermal-hydraulic design limits.

(5.3) The design limits should be established with the comprehensive and conservative inclusion of all chemical, physical, hydraulic and mechanical factors affecting the relevant degradation mechanism of the FE cladding as well as the dimensional tolerances of the FE. If the given damage/degradation mechanism of the FE cladding and the relevant limit depend on FE burnup, the experimental analysis should include the effect of irradiation on the properties of the FE cladding and the FE as to ensure the representativeness of experimental results.

## NEUTRON-PHYSICAL CHARACTERISTICS OF THE REACTOR CORE

### Design requirements

(5.4) The design of the reactor core should be such that the final effect of feedback characteristics of the core is negative (i.e. that the final effect of reactivity compensates for a rapid increase in reactor power) in all operational states with a critical reactor. The reactor power should be controlled by a combination of the inherent (neutron-physical and thermal-hydraulic characteristics) and active interventions of the protection and control systems of the nuclear reactor and reactor core to adequately actuate for all operational states of the NI.

(5.5) The relevant neutron-physical parameters, such as reactivity, reactivity coefficients, efficiency of the control rods and power distribution, should be analysed with respect to all the types of FA used. Demonstration of compliance for all types of FA can be derived from the calculations of a separate FA in the infinite slab geometry.

### Design criteria and limits

(5.6) The design of the reactor core should set the key safety parameters characterizing the neutron-physical properties of the reactor core. The set of key safety parameters should be defined in line with the assumptions for safety analyses that demonstrate compliance with the specific design criteria described in para. (5.47) - (5.56). The set of key safety parameters should be defined as to be maintained for the designs of all envisaged specific loads. The typical key safety parameters include in particular the following:

- The temperature coefficient of reactivity for the fuel and for the moderator;
- The coolant density coefficient of reactivity;
- The boron coefficient of reactivity and the boric acid concentration;
- The shutdown margin;
- The maximum insertions of reactivity by control rods;
- The efficiency (insertion of negative reactivity) of the control rods;
- The radial and axial power peaking factors (coefficients of power distortions), including the allowance induced by xenon oscillations, where relevant;
- The relative maximum linear heat rate;
- The void coefficient of reactivity.

(5.7) For each fundamental design modification with a potential impact on the core design, the validity of the set of key safety parameters should be verified. If there is a change in the value of any of the parameters, the level of influence of the change should be assessed. In case of a significant deviation from the original values of key safety parameters, new key safety parameters should be established, evaluated and verified. Fundamental design modifications mean the following modifications:

- The design modification to the NI, SSC or operation of the NI;
- The significant modifications to the control of fuel campaigns, e.g. significant extension of the fuel campaign;
- The introduction of the new type of nuclear fuel;
- The increase in the burnup limit of nuclear fuel.

(5.8) The design of the reactor core shall include the calculations of stationary and non-stationary spatial distributions of neutron flux and of the heat power, neutron-physical characteristics and the efficiency of the means of reactivity control for all states of the NI, unless demonstrated for some selected states of the NI that such effects do not occur or are verified otherwise. The calculations should be carried out for all envisaged operational states of the NI. The calculated distributions of the power may be applied to design basis accidents and design extension conditions without the core melting. This distribution should be respected in the analyses for the thermal-mechanical behaviour of the FE and the margin for the failure to achieve critical conditions for the transfer of heat and demonstrations of compliance with the other relevant design criteria.

(5.9) Key safety parameters such as reactivity coefficients should be calculated for the selected operational states (e.g. the zero power, nominal power, beginning of life, end of life) and for the corresponding strategy for fuel load management (campaigns). Their dependence on the fuel charge and the burnup of fuel should be analysed. In all considered states of the NI, the adequate conservative approaches should be used in the application of the coefficients of reactivity.

(5.10) The rate and amount of reactivity insertion from the individual means of control (the control rods and/or the soluble absorber) should be limited, or the adequate protections and limitations should be set to ensure that the final power modification does not exceed the specific limits for the major initiating events associated with the reactivity insertion such as:

- The ejection of the most effective control rod;
- The drop of the control rod;
- The uncontrolled entry of clean condensate;
- The uncontrolled withdrawal of one group of the control rods.

The limitation on the rate and amount of reactivity insertion should be established on the basis of the analyses demonstrating compliance with the design criteria of fuel described in para. (5.48) - (5.56). The analyses should be carried out for all types of the fuel used in the core, or for the cask fuel load in the core, for all considered operational states of the NI, and the extent of burnup and with the adequate conservative conditions and assumptions.

(5.11) The design of the reactor core should facilitate the monitoring and control of the overall and local power by using the means for power monitoring and reactivity control so as to ensure that the limits of local linear and post-element power are not exceeded anywhere in the core and on any of the FEs. The core control and monitoring systems should be designed as to be capable of taking account of the change in power distribution caused by local changes in reactivity as a result of, for example, the use of mixed core, the occurrence of the anomalies in axial offset, bowing or distortion of the FE. The calculation models and their uncertainties, used in the design of core control and monitoring systems, should also consider the variability in the measurement of neutron flux detectors (for example caused by the location, reduced operational ability, shielding or ageing).

(5.12) The maximum allowable insertion of the control rods should ensure the adequate shutdown margin, i.e. safe reactor shutdown, in all states. The determination and monitoring of the limits for the insertion of control rods (e.g. depending on the power level or burnup) should ensure the adequate shutdown margin throughout the operation.

(5.13) The calculations of shutdown margin shall include the operational effects of the core reactivity in all states of the reactor core that may occur during the fuel campaign, e.g. the effects of the loss of burnable absorber, boron burning in boric acid, the effects of spatial power distribution, etc.

## THERMAL-HYDRAULIC PROPERTIES OF THE CORE

### General requirements

(5.14) The fundamental objective of thermal and hydraulic design of the reactor core should be the ensuring of such transfer of heat from the core to ensure the removal of heat from the core in all states.

**Design requirements**

(5.15) The thermal-hydraulic design of the reactor core shall include the adequate margins and measures in order to ensure that:

- The specific thermal-hydraulic design limits are not exceeded in all operational states, i.e. during normal operation and during abnormal operation;
- The portion of failed FEs (determined by means of the DNB occurrence) during the design basis accident and design extension conditions without the core melting remains below the level of adequate acceptance criteria.

(5.16) The statistical methods used in the thermal-hydraulic calculations shall include the effects of the bowing of FEs and FAs.

**Design criteria and limits**

(5.17) The analyses should take account of the design limits for the minimum and maximum coolant flow in the core and the limits should be respected, or a detailed analysis should be presented to demonstrate that the design limits of the reactor core are not exceeded.

(5.18) The analyses should take account of the limits for hydraulic-dynamic stability of the flow through the FAs and the core components. The limits should be respected, or a detailed analysis should be presented to demonstrate that the design limits of the reactor core are not exceeded.

(5.19) The analyses of the thermal-hydraulic design of the reactor core should consider all specific design elements of the FAs and the associated production and operation deviations which in particular are the rod pitch, the power of the FEs, the shape and size of the sub-channels, spacer grids and mixing grids including their acceptable operational deformations.

(5.20) The thermal-hydraulic design of the reactor core should respect the spatial distribution of inlet and outlet coolant temperature distribution and the flow distribution in the core. These parameters should be respected in designing the nuclear reactor shutdown systems, and the core control and monitoring systems.

The analyses shall include the impact assessment of the breakdown of coolant temperatures in the loops and its method of measuring. This impact shall be further included in designing the nuclear reactor shutdown systems, and the core control and monitoring systems, and in proposing the changes in their setting.

The design of the reactor core should ensure that the determination of the minimum critical and operating power ratio (i.e. determination of the minimum

DNBR achieved) takes account of the fact that the correlations of critical heat flux were determined on the basis of the tests performed in stationary conditions. Consequently, the margin established for the assessment of non-stationary initiating events in normal and abnormal operation should be used and clearly demonstrated. Since the correlation of critical heat flux was measured under steady state stationary conditions, the margin should be adequate enough to avoid damage to the FE even under non-stationary conditions of normal and abnormal operation.

In certain designs (unless the criterion governing the failure to achieve the DNBR limit is used), the critical heat flux achieved on the number of FEs greater than the number allowed in the conditions of design basis accidents and the design extension conditions without the core melting can be acceptable even in the case where any other acceptable analytical method is used in the thermal-hydraulic design to determine the number failed FEs.

(5.21) The DNBR should be also calculated for the cask load to demonstrate the sufficient safety of the core. The values of the DNBR should be calculated for different pressures and plotted in the reactor power-inlet temperature diagram, showing the DNB curves and the saturations for different pressure values.

(5.22) The experiments to establish the correlations of critical heat flux should be executed for a sufficiently wide range of the envisaged operating conditions and with the sufficient number of the points measured to ensure that the data obtained in such a way for the determination of the limit values of minimum critical heat flux can be statistically evaluated in accordance with the world practice.

(5.23) In order to demonstrate compliance with para. (5.2) - (5.20), it is possible to use approaches, the examples of which are below:

- For assessment of the DNBR or the correlationsof critical heat flux it should be ensured with a 95% probability at the 95% level of probability that the “hot” fuel element in the core does not achieve the conditions of boiling crisis under no conditions of normal or abnormal operation;
- The limit (minimum) value of the DNBR, or the correlationsof critical heat flux should be determined to ensure that the number of FEs achieving the boiling crisis during normal or abnormal operation does not exceed the limit value, e.g. expressed by the proportion of the number of the FEs occurring in the DNB conditions per 1000 FEs located in the core.

(5.24) The impact assessment of hydraulic loads for the behaviour of the FEs and the FAs should be primarily a part of the thermal-hydraulic part of the design of the FA, while including it in the assessment of the acceptability of localized corrosion, surface erosion, coolant flow-induced vibrations and fretting of the FEs.

(5.25) The integrity of the FEs due to the hydraulic-dynamic effects should be demonstrated by means of the tests performed on the qualified hydraulic loops, using the full-dimension dummy fuel assemblies and in the prototype test conditions (e.g. pressure, temperature and transverse flow). The hydraulic tests should be particularly performed for the following:

- The flow-induced fretting wear test – the test should confirm that the design limits for damage to the FE due to fretting are not exceeded;
- The flow-induced vibration test – the test should demonstrate that the vibration amplitude of the frequency of the FA is not exceeded for different rates of coolant flows around the FA to the extent exceeding the design limit for damage to the FE due to fretting;
- The pressure drop test – the test should demonstrate that the flow of coolant does not change in the way exceeding the design limits of the FA.

The testing methodology and the selection of test conditions should ensure, with reserve, compliance with the adequate design limits of the FA.

## THERMAL-MECHANICAL PROPERTIES OF THE CORE

### Design requirements

(5.26) The design of the reactor core should ensure that all design limits for the FAs and the FEs (structural integrity, geometry, maximum allowable stress, etc.) are respected during normal and abnormal operation and that thus they are not exceeded and that there is no failure of the FEs (loss of the FE cladding). In the accident conditions (design basis accidents and design extension conditions without the core melting), the number of failed FEs should comply with the acceptance criteria for radiological consequences. It is also necessary to ensure such FA geometry that could be cooled on a long-term basis and that does not prevent the drop of mechanical control rods. At the same time, the activity level of radionuclides should be assessed under the above mentioned conditions to demonstrate compliance with the allowable limits of the release of radioactive material into the environment and that they are not exceeded.

(5.27) The design of the FE (with or without burnable absorber) and the FA should respect the particular conditions of the working environment of the NI in question (e.g. temperature, pressure and chemistry of the coolant, the effects of irradiation on the fuel, the microstructure of the FE and FA materials; the static and dynamic loading s including coolant flow-induced vibrations; the changes in the chemical and physical properties of structural materials).

The important effects that shall be included in the design of the FE, FA in terms of irradiation and working environment are described in Chapter 9, together with those for the control rods, neutron sources and hydraulic plugs for the guide tubes of the core components.

(5.28) The design of the reactor core should ensure the reliable functioning of the core components, the FEs and the FAs throughout their life cycles including the fabrication, transport, handling in the NI, operation in the core, storage and placement in the storage facility, where applicable. The design of the core components, the FEs and the FAs should define the key ways of ensuring the reliability, while the following issues are particularly important:

- The oversight of the fabrication (fabrication processes);
- The suppression of the effect of debris (preventing the presence of foreign materials);
- The monitoring and control of power changes of the core in order to reduce the pellet-FE cladding interaction;
- The control of the formation of deposits and corrosion layers (chemistry of primary circuit);
- The prevention of FE cladding failure due to the vibrations of the FE in the spacer grid (Grid-to-Rod fretting);
- The monitoring of fuel state and the inspection procedures.

(5.29) The design of the FA and the NI should ensure that no damage to the FA due to the mechanical load induced particularly by the following effects is done in the conditions of normal and abnormal operation:

- All handling of FA and FE including loading;
- The intentional and unintentional changes in power level;
- The contact forces on the FA (which compensate for the hydraulic-dynamic lift forces and the changes in the geometry of the reactor internals and the FAs; due to irradiation and thermal expansion);
- The temperature gradients;
- The hydraulic forces including transverse flow among the deformed FAs or the different FAs in the mixed core;
- The irradiation effects on the materials (e.g. radiation induced growth and swelling);
- The hydraulic-dynamic vibrations of the FE and the vibration fretting of the FE cladding (Grid-to-Rod fretting);
- The non-elastic deformations of the FA skeleton that may lead to excessive deformation of the FA – bowing and twisting and bowing).

(5.30) In the conditions of normal and abnormal operation, the design of the reactor core should ensure that the maximum temperature in the fuel pellets is lower than the fuel melting temperature by a sufficient margin, with allowance made for the relevant uncertainties and the design properties of the NI concerned. The melting temperature should be defined as the function depending on the burnup and chemical composition of the pellet, which should be adequately supported by the results obtained from experiments.

(5.31) The design of the reactor core should ensure that the stress and deformation in the FE cladding are limited to such an extent to prevent the fuel design limits from exceeding for the states concerned. For operational and emergency conditions of the core, specific design limits should be defined for the stress in the FE cladding, its permanent deformation, corrosion and hydriding.

The consequences of excessive deformations of the FE cladding (e.g. ballooning) should be assessed in the safety analyses for emergency conditions in order to sufficiently conservatively identify their impact on heat removal, FE failure (e.g. rupture and bursting) and the subsequent release of fission products from the FE.

(5.32) The design of the reactor core should ensure that the changes in the geometry of the FA and the FE are limited to an acceptable level so as to avoid the contact or the mechanical interaction between the FE and the top part of the FA (unacceptable buckling stress of the FE) and between the FE and the bottom part of the FA, that the bowings of the FE and/or the FA are limited to an acceptable level and that the deformations of the control rods and their other possible interactions with the guide tubes of the FA do not affect the structural integrity of the FA and/or the safety function of the control rods.

The design of the reactor core shall include an assessment of the relaxation of the springs of spacer grids during FA irradiation to prevent the possibility of grid-to-rod-fretting. The analyses of geometrical stability shall include the effect of irradiation on the FE, FA and core components, on mechanical properties of the materials used such as the tensile strength, ductility, radiation induced growth, creep, radiation hardening and relaxation. The effect of irradiation on the resistance of spacer grids to compression in the horizontal direction should be respected in an assessment of the seismic events or LOCA (analysis of the mechanical response of the FA to seismic/LOCA load – evidence of maintaining the coolable geometry of the FE and the FA).

(5.33) The irradiation induced growth of the FE and the FA should be limited in order to avoid contact of the FE with the FA top and bottom nozzle. The bowingand twisting and bowing of the FE and the FA may result in power anomalies, in particular where there is a change in the rod pitch. These effects should be analysed or supported by experimental data.

(5.34) For conditions of normal and abnormal operation of the NI, the design of the reactor core should ensure that no damage is done to the FE due to the thermal-mechanical load during local and global power transients (e.g. due to FE shuffling in the core, movements of control rods or other reasons for reactivity changes) and the changes in power distribution.

(5.35) The design of the FE shall include analyses demonstrating that the design limits for the deformation or stress in the FE cladding due to the mechanical load (e.g. coolant pressure, seismic loads). Mechanical and thermal-mechanical analyses of the behaviour of the FE, FA and the core components should also take into account the effects of irradiation, chemical interactions and other changes in the properties of the materials used.

(5.36) The design of the reactor core should consider stress corrosion cracking of the FE cladding induced by complex pellet-FE cladding interaction in the presence of aggressive gaseous fission products and criteria should be set to avoid damage to the FE due to this mechanism. The criteria should be laid down in order:

- To reduce tensile stresses in the FE cladding by limiting the rate of change in power (providing sufficient time for stress relaxation in the FE cladding) or by delaying the closing of the gap between the FE cladding and the pellet (this can be achieved by increasing the initial filling pressure in the FE cladding or by optimising the creep behaviour of the FE cladding);
- To limit the corrosive action of fission products (e.g. iodine, cadmium, caesium) formed in the pellet by using the protective layer on the inner side of the FE cladding. This layer may reduce the size of stress concentration in the FE cladding;
- To reduce the quantity of corrosive fission products at the pellet-FE cladding interface by using additives to the fuel material that can contribute to lower release of corrosive gaseous fission products from the crystal lattice of the fuel;
- In the neutron-physical part of the design of the reactor core, to reduce the value of the coefficients of local power unbalance (thus the potential changes in local density of heat generation).

(5.37) The concentration of stress in the FE cladding caused by missing pellets, axial gaps in the column of pellets, disrupted surfaces of the pellets or fuel pellet fragments in the fuel-FE cladding gap should be explicitly modelled (included in uncertainties or margins), or these anomalies should be demonstrably excluded (e.g. by inspections during the production process).

(5.38) The design of the FE shall include analyses demonstrating that no damage is done to the FE due to the use of integral burnable absorber in the fuel pellets (different thermal, mechanical, chemical and microstructural properties) and their effect on the integral behaviour of the FE.

(5.39) The design of the FE should ensure that an adequately validated model is used for zirconium alloys in the FE cladding to determine the content of absorbed hydrogen and oxygen in the FE cladding. The effect of absorbed hydrogen and oxygen on the behaviour of the FE cladding should be considered in setting the design criteria of the FE for the conditions of operational states (hydrogen pick-up) to be able to express the specific design criteria of the FE for emergency conditions (e.g. for RIA and LOCA) as a function of hydrogen concentration in the FE cladding before the beginning of the transient.

(5.40) The FEs and the FAs should be designed to be compatible with the coolant (chemical) environment in all operating modes including shutdown and refuelling.

(5.41) The design analyses of the FE shall include degradation of the conditions of heat transfer in the FE due to the formation of deposits on the surface of the FE cladding, caused by the transport of corrosion products from the primary circuit or other chemical processes, unless the formation of such deposits can be demonstrably excluded. For pressurized water reactor, where the soluble absorber is captured in the layers of deposits on the surface of the FE, this effect (axial offset anomaly, crud induced power shift) shall be included in the neutron-physical part of the design of the reactor core.

(5.42) The design of the NI should further ensure that no such damage is done to the FA (e.g. due to the load of force caused by design seismic event combined with the load in large LOCA) in the conditions of design basis accidents and design extension conditions without the core melting, which would prevent the performance of fundamental safety functions (safe shutdown, maintaining the coolable geometry of the FA), see para. (5.44).

(5.43) For conditions of normal and abnormal operation of the NI, compliance with the following requirements should be demonstrated in the design of the FE and the FA:

- The gaps and tolerances inside the FAs and between the adjacent FAs should be large enough to provide sufficient space for radiation induced growth and deformation (bowing, distortion) of the FE and the FA;
- Any bowing and growth of the FE as well as any deformation of the FA should be limited to an acceptable level to prevent the adverse effects of the thermal-hydraulic behaviour, the power distribution in the core, the thermal-mechanical behaviour of the fuel, and fuel handling not envisaged by the design;
- Material fatigue induced by cyclic stress-strain loading should not be able to cause the failure of the FA;
- Any deformation of the FA, due to the mechanical and hydraulic hold-down forces and the transverse flow in the core, should be limited to such an extent so as not to affect the local margins for critical heat flux set out by the design. Furthermore, any deformation of the FA should not affect the capability for the insertion or drop of control rods (e.g. drop time) to preclude the safe shutdown of the reactor in all states;
- Damage due to vibration and fretting should not affect the overall behaviour and function of the FE, the FA and the support structures.

(5.44) For accident conditions (design basis accidents and design extension conditions without the core melting), the design of the reactor core should ensure that any interactions between the FE or the FA and the support structures of the FA will not prevent safety systems from performing their functions as claimed in the safety analyses. In particular, the following should be ensured:

- The design function of the components of safety systems (e.g. control devices of nuclear reactor shutdown systems and their guide tubes);
- Design cooling of the core (even for long-term cooling of the core).

(5.45) The design of the FE should ensure that any excessive failure of the FE cladding due to the PCI is prevented in the conditions of normal and abnormal operation, unless otherwise demonstrated by the design.

During rapid design basis accidents, which lead to rapid changes in the power level (e.g. reactivity initiated accidents), the FE can failed due to the PCI in combination with the embrittlement of the FE cladding material caused by hydriding in the reactor at high burnups. Failure of the FE induced by this mechanism should be reflected and assessed in the safety analyses.

(5.46) The criterion for the failure of the FE cladding during power ramps should be supported by one of the following approaches or their combination, taking account of the characteristics of the specific type of the FE, in particular fuel and FE cladding material:

- In-core tests, the so-called in-pile power-ramp tests;
- In-core test modelling; or
- Experiments conducted on the irradiated FE cladding.

The experimental database should sufficiently cover the design scope of burnup and possible power ramps. The design scope should cover all the critical areas, in particular the area of burnup 25 – 40 MWd/kgU. The number of measuring points should facilitate sufficient statistical processing in the overall design scope.

Where the criterion for the failure of the FE is verified by using the thermal-mechanical computer code (e.g. in the form of maximum allowable stress or density of deformation energy), it is necessary to derive this criterion from experimental data using the same code, or to demonstrate that any other computer code is applicable to verify this criterion.

**Design criteria and fuel limits**

(5.47) The design limits of the fuel should be specified taking into account all physical, chemical and mechanical processes, which affect the behaviour of the FE and the FA in all states.

(5.48) For conditions of normal and abnormal operation of the NI, the design of the FE shall include at the least the following limits:

- Preventing the melting anywhere in the fuel pellet;
- Preventing the overheating of the FE cladding (boiling crisis);
- Preventing the collapse of the FE cladding (due to external coolant overpressure);

- Preventing the internal pressure in the FE from exceeding the value that would induce deformation of the FE cladding degrading the conditions of heat transfer between the FE and the coolant;
- Preventing the corrosion and hydriding of the FE cladding from exceeding the allowed limits;
- Preventing the fretting and scratching induced failure of the FE cladding from exceeding the specified limits;
- Preventing the stress and deformation of the FE cladding from exceeding the specified limits.

(5.49) The components of the FE and the FA should be designed as to exhibit deformations and radiation induced growth only to such an extent as to prevent violation of the design limits for the FE by preventing the excessive:

- Interaction of force between the FE terminals and the FA top nozzle (in order to prevent the bowing of the FE due to the excessive buckling stress);
- Disruption of the local power distribution in the FE;
- Reduction in local margins for the critical heat flux in the FA;
- Disruption of the mobility of control rods and their capability for the drop into the core;
- Disruption of the possibility of FA handling.

(5.50) The FA, core components and the reactor internals should be designed to minimize the risk/danger of the capture of foreign objects in the FA, thus preventing damage to the FE in operational states.

(5.51) The design limits for fuel burnup, resulting from the thermal-mechanical behaviour of the FE and the FA, should not be exceeded in the design of the reactor core.

(5.52) For design basis accidents and design extension conditions without the core melting, the design of the FE should ensure that:

- The number of failed FEs does not exceed a certain percentage of the total number of the FEs in the core as to limit the radiological consequences of each accident considered in the design of the NI and to maintain the limits for radiological consequences;
- In determining the total number of failed FEs, all known damage mechanisms of the FE cladding should be considered. In particular, chemical reactions including oxidation and hydriding, ballooning or collapse of the FE cladding, or failure of the FE cladding due to mechanical stress or melting, etc.;
- The criteria used for the failure of the FE cladding should be based on the results of experimental studies;
- If any key parameter (radially averaged enthalpy of the FE, stress, deformation, temperature, strain energy density, etc.) exceeds the certain limit value (cladding burst limit) in any axial coordinate, specified on the basis of the representative experimental results, obtained under the conditions

corresponding to the in-core conditions (the experimental parameters shall include at least: coolant temperature, coolant pressure, coolant flow rate, kinetics of reactivity insertion and internal pressure in the FE), the failure of the FE should be considered. Since the resistance of the FE cladding changes with the irradiation effects and depends on the type of FE cladding alloy, this limit is also expected to depend on the type and material of the FE cladding.

(5.53) The removal of heat from the core should not be compromised, for example, by:

- The effects of ballooning or bursting of the FE cladding (e.g. during LOCA);
- Excessive deformation of the components of the FA or the reactor internals (e.g. during seismic event);
- Blocking the coolant flow.

The design of the FE and of the reactor core should sufficiently prevent the failure of the primary circuit pressure boundary during the RIA due to the thermal explosion of the FE, or the failure of the fuel system and the primary internals limiting the capability of removing heat from the core. In general, this is ensured by the limits for maximum key parameters (e.g. pellet enthalpy) and for its allowed increase.

(5.54) To ensure the structural integrity of the FE, the following design limits should be specified and justified:

- The maximum local temperature of the FE cladding and the equivalent quantity of the oxidized FE cladding material in emergence conditions should not exceed such a value where the oxidation of the FE cladding causes the excessive embrittlement of the FE cladding material or is uncontrollably accelerated (exothermic reaction). The effects of the fragmentation of fuel pellets and of the relocation of fragments inside the FE on maximum local temperature of the FE cladding should be properly assessed. The potential effects of the fuel particles dispersed in the primary circuit from failed FEs on the radiation consequences of accidents and on the core coolability should be properly assessed;
- In accident conditions, the FE cladding should not be damaged to the extent that it had failed to withstand mechanical stress during accident (e.g. load induced by LB LOCA). The assessment should take into consideration the rate of oxidation of the FE cladding both before the transition and during the transition (from the outside of the FE cladding and if necessary, from the inside of the FE cladding for failed FEs); it shall further include chemical interactions between the pellet and the FE cladding. Both hydrogen and oxygen, absorbed in the FE cladding during normal and abnormal operation and in accident conditions, should not cause such a degradation of the mechanical properties of the FE cladding (embrittlement) as to be able to withstand the loads caused by handling, transport and storage of the FAs. The effect of oxygen and hydrogen absorbed in the FE cladding on its strength and ductility should be determined;

- The criteria for the failure of the FE and damage to the FE and FA for the RIA should be set as to respect the initial state of the FE before the event (e.g. hydrogen and oxygen content prior to the disruption of normal operation, FE burnup).

(5.55) During LOCA, the amount of hydrogen formed as a result of a coolant-FE cladding reaction should not exceed the limit (e.g. 1% ratio) to the total amount of hydrogen formed as a result of a coolant-FE cladding reaction. It is the value ensuring that the concentration of hydrogen achieved does not result in its burning (explosion).

(5.56) Any structural deformations of the FE, FA, core components or the reactor internals should not degrade to mobility of control rods beyond the scope expected in the safety analyses. The melting temperature of control rods, taking into account the eutectic reactions, should not be exceeded anywhere in the core in all states.

## MECHANICAL PROPERTIES OF THE CORE

### Design requirements

(5.57) The core components and the associated reactor internals should be designed as to maintain their structural integrity in all states, considered in the design. The structural integrity should be demonstrated for all relevant damage mechanisms of the core components and of the reactor internals, e.g. vibrations (induced mechanically or by coolant flow) and fatigue, thermal, chemical, hydraulic and radiation effects (including radiation induced growth) and seismic loads, by either calculations or experiments. Special emphasis should be placed on demonstration of non-failure of the components designed for reactivity control and safe shutdown of the reactor, heat removal capability as well as integrity of the primary circuit pressure boundary. The demonstration should take into account the effects of high pressures, high temperatures, temperature changes and temperature distribution, corrosion, radiation load and overall radiation exposure throughout the service life of the components on their physical dimensions, load of force and material properties.

(5.58) The design of the control rods and other core components should ensure that they withstand all handling operations during refuelling, transport and storage without being damaged.

(5.59) The design of the core components and the load-bearing reactor internals should facilitate the necessary inspection of the core components and the reactor internals (e.g. visual).

(5.60) The load-bearing and supporting reactor internals should be designed as to withstand the static and dynamic load including their load in relocation and handling of the FAs.

(5.61) The design of the load-bearing structures in the reactor should demonstrate the margins to the limit for temperature stress in all states, including the effect of gamma heating on their cooling and temperature responses. Similarly, the analyses shall include chemical effects of the coolant on the components, in particular corrosion, hydriding, stress corrosion cracking and formation of deposits.

(5.62) The structures of the FA and the guide tubes for the control/shutdown devices and for the in-core instrumentation should be designed as to ensure that the devices and instrumentation are accurately located and that the possibilities for inadvertent operator actions are minimized (it can be demonstrated by safety analyses or provided for administratively in the operating documentation by setting the sufficient inspection), by power loads and hydraulic forces due to coolant flow, in all states. The design of the NI should facilitate maintenance of the devices and instrumentation. The design should respect the possibility that the hydraulic-dynamic vibrations of these components, instrumentation or their guide tubes may result in fretting, wear and their consequent malfunction/failure in long-term operation. The dimensional stability of the guide structures should be demonstrated throughout their design life.

(5.63) The design of the core components and of the reactor internals should facilitate the replacement of the control rods and in-core instrumentation whenever necessary without causing damage to the other core components and reactor internals, unacceptable insertion of reactivity and/or undue radiation exposures to humans.

(5.64) Depending on the particular reactor type, various other internals may be installed within the reactor vessel. These include, for example, thermal shields, core baffles and reflector. These internals should be so designed that their mechanical performance does not jeopardize the performance of any associated safety functions of the core throughout their service life.

(5.65) The design of neutron sources should be such as to ensure that:

- The sources produce the long-term stable and sufficient neutron flux, which ensures more precise detection of potential changes at the low level of neutron flux even in the case of long-term reactor shutdown;
- The sources are mechanically stable and suitable for the use in the conditions of normal and abnormal operation.

**Design criteria and limits**

(5.66) The design of the core components and of the reactor internals should meet all requirements and criteria defined in the technical standards, referred to in the management system programme, which apply to the relevant safety class, referred to in para. (4.18) - (4.21).

## REACTOR SHUTDOWN SYSTEMCORE CONTROL, SHUTDOWN AND MONITORING SYSTEMS

**Design requirements**

(5.67) The reactor core control and monitoring systems and the way of processing and interpreting the measured values should be designed to be able to identify changes in neutron flux distribution in the core with sufficient accuracy, thus preventing the fuel design limits from exceeding.

(5.68) The parameters sufficiently characterizing the state of the core should be determined in the design of the reactor core and their appropriate measurement should be ensured. The most important parameters include the following:

- The power of the reactor;

- The spatial density distribution of the neutron flux and associated power peaking factors;
- The inlet and outlet coolant temperature and the temperature distribution in the loops of the primary circuit;
- The coolant flow in the core;
- The pressure of primary circuit;
- The position of the control rods in the core;
- The concentration of soluble absorber and the content of B-10.

Other parameters may be derived from the measured parameters. Examples are:

- The neutron flux doubling time (or the period);
- The neutron power rate;
- The current reactivity value;
- Nonuniformity of Axial and radial neutron flux;
- Thermal-hydraulic core parameters (e.g. DNBR).

(5.69) The design of the FA and of the reactor core should facilitate and ensure the use of the necessary instrumentation and detectors for the monitoring of core parameters such as the core power (level, distribution and time dependent variations), the conditions and physical parameters of the coolant (pressure and temperature) and the efficiency of the means of shutdown of the reactor (e.g. the reactivity insertion rate of the control rods compared with their reactivity insertion limits). The instrumentation used and the way of processing and interpreting the measured values should be such so that any necessary corrective action can be taken in case of risk to the specified limit values. The instrumentation used and its location should facilitate the monitoring of these parameters in the entire range of their anticipated values and in all states including refuelling.

(5.70) The design of the reactor shutdown systemcore control, shutdown and monitoring systems should ensure that all power transients which may lead to the violation of fuel design limits during normal and abnormal operation are reliably and timely detected and suppressed.

(5.71) The means of control of reactivity should be designed to enable the required power levels and the power distribution to be maintained with a sufficient margin within safe operating limits, including changes associated with the compensation for reactivity such as:

- The controlled power transients;
- The changes in xenon concentrations;
- The changes in feedback coefficients;
- The changes in the rate of coolant flow or temperature;
- The depletion of fuel and of burnable absorber.

(5.72) The control rods should be designed to be capable of rendering the reactor subcritical even in the conditions of design basis accidents and the design extension conditions without the core melting.

(5.73) The use of the control rods or of the reactivity control systems during normal and abnormal operation should not reduce their operational ability and capability of ensuring the performance of safety functions where reactor trip is necessary.

(5.74) The maximum amount and rate of positive reactivity insertion in all states should be limited and/or sufficiently compensated in order to prevent failure of the pressure boundary of the primary circuit and as a result thereof, damage to the core and core coolability during subsequent event associated with coolant leakage from the primary circuit.

(5.75) The spatial location, grouping, and the speed of withdrawal and withdrawal sequence of the control rods including associated settings in the control and shutdown systems should be designed to prevent the design limits of the core from exceeding during any uncontrolled withdrawal of the control rods.

(5.76) With regard to the soluble absorber, the reactivity control system should be designed to prevent any unforeseeable decrease in the concentration of absorber in the coolant, which could cause the fuel design limits to be exceeded. The individual parts of those systems should be designed to ensure an even concentration of the soluble absorber in its entirety (e.g. by heating and mixing). The concentrations of the soluble absorber in all storage tanks should be monitored.

(5.77) The possibilities of unintentional dilution of boron acid should be identify in operation and in shutdown states by a detailed functional analysis of the core control systems and associated operating conditions.

(5.78) The effectiveness of the control rods (mechanical regulation) should be checked by direct measurement.

(5.79) The control rods design should take into account their wear and the irradiation effect in operation, e.g.: absorber burnup, changes in physical properties and production of fission gaseous products.

The following effects should be considered in the design of the mechanical control system of the core:

- Irradiation effects of the control rods – depletion of the absorber or swelling and heating of materials due to neutron and gamma absorption. Following the impacts of these effects, the mechanical regulation devices should be, in accordance with the design, renewed or their location in the core should be changed;
- Chemical effects – e.g. corrosion of the cladding of the control rods, movement of radioactive corrosion products in the primary coolant;

- Deformation and displacement of the structural members – taking into account movements of reactor internals due to thermal expansion, radiation load, mechanical load or external events such as earthquake.

(5.80) The reactor shutdown system in all states should ensure that the design limits specified for the shutdown margin are not exceeded, as provided for in para. (5.12) and (5.13). The design of the individual devices should ensure the reliability of the entire system as well as the independence of control and shutdown functions.

(5.81) The efficiency of the control rods, the reactivity insertion rate and the shutdown margin should be set as to comply with the specified design criteria for the fuel system. For detailed description of the requirements see para. (5.82) - (5.91).

(5.82) The rate of shutdown should be adequate to ensure that the design limits of the FE, FA and of the primary circuit pressure boundary are not exceeded.

(5.83) In designing and calculating the rate of shutdown, the following factors should be considered:

- The response time of the instrumentation to initiate the shutdown;
- The response time of the actuation mechanism of the regulation devices;
- The location of the regulation devices;
- The axial profile of the absorber in the control rods;
- The insertion speed of the absorber into the core, e.g. the gravity drop time of control rods or the dosing efficiency of soluble absorber.

(5.84) The verification of the adequacy of the design of nuclear reactor shutdown systems shall include an assessment of the event including failure resulting in limitation of their efficiency for reactor shutdown. Failure in the drop of the most efficient control rod, i.e. its jamming in the upper limit position, can be considered as sufficiently representative event. The design of the reactor shutdown system should prevent its common cause failure.

(5.85) To ensure a high reliability, the design of the nuclear reactor shutdown systems including their control devices (regulation devices) should apply any of the below engineering approaches or their reasonable combination:

- Adopting systems that are simple and facilitate simple operation;
- Selecting components and equipment of proven design;
- Using a fail-safe design as far as practicable;
- Giving consideration to the possible modes of failure and adopting redundancy in the initiation of shutdown system (e.g. measurements, sensors). Provision for diversity in the initiation of shutdown should be made, for example, by using two different and physically independent parameters to initiate the reactor shutdown system. Those parameters should be determined for all the events considered, where practicable;

- Functionally isolating and physically separating the shutdown systems (this includes the separation of control and shutdown functions) for all the events considered including adjustable mechanisms of single failures and common cause failures, as far as practicable;
- Ensuring easy entry of the control rods into the core in the ambient conditions corresponding to normal and abnormal operation, design basis accidents and design extension conditions without the core melting.
- Designing to facilitate repairs, in-service inspections and testing;
- Providing means for performing comprehensive testing during commissioning and regular outages for refuelling or maintenance;
- Testing of the initiating mechanisms (or of partial control rod insertion, if feasible) during operation.

In case of the design change to the existing equipment, it is not necessary to submit a detailed description of the design. A detailed description of this system should be given in the relevant chapter of the safety analysis report, to which the applicant may refer.

(5.86) The design of the reactor shutdown system should take into account the wear of individual components and shall include the calculations of the service life of regulation devices (e.g. determination of the number of trips). The calculations of the service life shall include not only the wear due to movements of control rods in the core (in particular drops of the control rods in the reactor trip) and radiation load (swelling, change in the chemical properties), as well as depletion of the absorber (burnup).

(5.87) The reactor shutdown system should be designed to be capable of compensating for envisaged uncontrolled insertions of positive reactivity, which may result in a secondary criticality, during fuel handling or during routine and non-routine activities in shutdown state in normal and abnormal operation. The design of the reactor shutdown systems and of the systems ensuring subcriticality in the SFSP should assess all conditions and activities, which may result in a decrease in the level of subcriticality in shutdown condition (e.g. handling of absorbers, depletion in the content of absorber, fuel handling and entry of clean condensate). These events should be subsequently analysed for the conditions with the lowest achievable shutdown margin.

The design of the reactor shutdown systems should define the number of control devices and their efficiency (i.e. reactivity stored in the individual control devices).

(5.88) The efficiency of the reactor shutdown systems and subcriticality maintenance should be demonstrated for the conditions of cask charge.

(5.89) A part of the control devices of the reactor shutdown systems can be used for the purposes of reactivity control and spatial power distribution, however without compromising the operational ability of the reactor shutdown system in all states.

(5.90) The reactor shutdown systems should be testable in operation (to the practicable extent), thus demonstrating their capability of performing their safety functions.

(5.91) The reactor shutdown systems should be physically separated from the control systems as to virtually exclude the possibility of common cause failure and as to prevent any failure of the control systems from compromising the operational ability of the shutdown systems.

**Operational limits for control systems and setting for the reactor shutdown systems**

(5.92) The design of the reactor core should set the relevant operational limits and for cases where they are exceeded, set the associated setpoints for interventions of the control systems and for activation of the activities, setting of alarms or action levels of the nuclear reactor shutdown system.

The setting of operational limits and associated setpoints of the control and shutdown systems should take into account the effects of fuel burnup, the mutual effect of adjacent control rods (reduction in the efficiency by the so-called shadowing effect), coolant streaming in the circulation loops and temperature distribution in the coolant at the outlet of the FA.

(5.93) The setting of operational limits and associated setpoints of the control systems should take into account the ageing effects of the reactor coolant system (e.g. clogging of steam generator tubes).

(5.94) The setpoints should be adjusted as to facilitate reliable control of the reactor and its shutdown throughout its operation. The automatic initiation of the core control systems and of the nuclear reactor shutdown systems should be set with such a reserve as to prevent fuel damage even in the course of transient, or to mitigate the extent of fuel damage at early stages of the accident to the level envisaged in the safety analyses.

(5.95) The core instrumentation and its operation, operational limits and procedures should be defined as to eliminate excessive values of the overall efficiency of the mechanical regulation devices (for initiating events of the type of withdrawal of the cluster or uncontrolled withdrawal of clusters) and the positive reactivity insertion rate. The efficiency of those actions should be checked. Where possible, an alarm should be set to warn of failure or imminent failure of those actions.

(5.96) The design limits, uncertainties, operational limits, instrumentation requirements and setpoints should be adequately reflected in the Limits and Conditions for safe operation of the NI.

**Instrumentation and in-service inspection of the core**

(5.97) The core instrumentation should be designed to provide sufficient information about the behaviour of the core to relevant control and shutdown systems, in particular spatially and time sufficiently detailed information about power distribution in the core. The design of the core shall include implementation of detectors and devices to sufficiently monitor the size and changes in the overall performance of the core and the local values of power distribution in the core in order to ensure that the design limits (e.g. power distribution, heat rate, neutron rate) are not exceeded in the entire range of design power ramps. Depending on the evaluation of the rapidity of the variation in a particular parameter, specified by the design of the reactor core, the actuation of the relevant function of the control systems may be manual or automatic.

(5.98) The amount of radionuclides in the coolant should be monitored and continuously evaluated during operation in order to provide a systematic check of the integrity of the FE cladding and an indication of failure in the integrity of the FE cladding.

(5.99) The accuracy in measurement, speed of response, range of measurement and reliability of all monitoring systems should be adequate for performing their intended functions. The monitoring system should be designed to allow the periodic or continuous testing.

(5.100) The instrumentation intended for post-accident monitoring should be qualified for the appropriate environmental conditions, thus being capable of monitoring the situation (e.g. reactivity, level in the reactor, temperatures) following the accident.

(5.101) The power and its spatial distribution should be monitored by means of ex-core or in-core instrumentation or their combination (e.g. neutron ionization chambers, fission chambers, self-powered detectors, local coolant temperature measurements). Measurements of the local power at different positions in the core should be designed to be able to ensure that there are adequate safety margins maintained for the fuel design limits, taking into account the changes in spatial power distribution in the core as a result of core management and fuel burnup. The power distribution in the core should be continuously or at least periodically monitored. The in-core detectors should be systematically distributed in the core reliably to detect local changes in power density. Both ex-core and in-core detectors should be periodically calibrated.

(5.102) The core monitoring system should be designed as to maintain the core parameters in accordance with the operational limits.

(5.103) During reactor shutdown, the number and type of instruments or combinations of neutron flux measurement and additional neutron sources in the core should be available to reliably monitor the neutron flux in the core and the development of residual heat (e.g. by using special detectors with an adequate sensitivity), in all states and throughout the operation when the FAs are in the reactor pressure vessel, also during handling of the FAs in the reactor and in all reactor start-up phases.

(5.104) The capability of the drop of control rods and its time of drop should be periodically tested. The design of the reactor trip system shall include the programme and equipment for drop time testing of the control rods. The time development of the results of measurements should be periodically evaluated.

(5.105) Where several neutron flux monitoring systems are used for different ranges, the systems should be logically interconnected by appropriate interlocks and overlap settings in order to monitor the core in the given range with the required accuracy and to avoid undue reactor shutdown.

## FUEL AND CORE MANAGEMENT

### Design requirements

(5.106) The main objective of fuel system and core management is to ensure safe, reliable and efficient utilization of the FE, FA and core component in the nuclear reactor.

(5.107) The fuel loadings should be designed in accordance with the requirements for controlling the core reactivity and the power distribution in the core so as not to achieve or exceed the fuel design limits.

The fuel system and core management shall include the following:

- The means such as particularly the computer codes should ensure fulfilment of the following functions of the fuel system and core management: design and safety assessment of the loading (specification of the FAs loaded into the core, arrangement of the FAs in order to ensure optimum burnup and required neutron fluxes, etc.), order of the range of FAs and core components, reactivity control and core operation monitoring;
- The strategy for core operation to facilitate the maximum operational flexibility with the optimum utilization of nuclear fuel and maintaining the margins to the design limits for fuel management.

(5.108) The strategy for the fulfilment of fuel system and core management functions should provide information about:

- The flow diagram of core loading including enrichment and location of fission material and burnable absorbers at the level of the FE, position of the FAs and the core components in the core and their orientation in the core for each fuel campaign;
- The plan (schedule) for fuel handling (the instruction set of FA handling operations, etc.);

- The configuration and limitation for the means for core management and reactor shutdown (including setting of the core control and monitoring systems, setting of the shutdown system, setting of the core monitoring system);
- The location and handling of the core components.

(5.109) The charge burnup and the physical parameters of the core are the parameters entering the safety analyses, monitoring and shutdown system setting, and operating procedures. The physical parameters include: conditions for the performance of physical testing (e.g. critical concentration of soluble absorber and critical positions of the control rods), kinetic parameters of the core, fuel temperature coefficients of reactivity, moderator temperature coefficients of reactivity, efficiency of the individual control rods and of the whole groups of control rods, and power peaking factors.

It is necessary to continuously or periodically verify the course of loading burnup and the validity of expected physical parameters within the limits of measurement of the monitored core parameters.

(5.110) In designing the core loading, it should be evaluated and confirmed that the established approaches to the fuel utilization and to the setting of operational limits for the core are respected. Otherwise, the operational limits could be exceeded or violated.

(5.111) In designing the fuel loading pattern and for their realistic analyses, it is desirable to preferably use multidimensional and global computer codes for all states of the NI. The BEPU calculations and the relevant methodologies shall include a sensitivity analysis.

(5.112) The analyses of the design of fuel loading and associated calculations should cover all usual operational events and operational states during campaign, e.g.:

- Normal and abnormal operation including representative power distribution;
- Load follow and other cyclic power ramps;
- Approach to criticality;
- Unit restart and operation at different power levels;
- Refuelling;
- Reactor shutdown.

Once the design loading has changed or the characteristics of the FE/FA have changed (e.g. enrichment, FE dimensions, location of the FE or change in the FE cladding material), new calculations should be executed and documented.

(5.113) The analyses of the design of the core shall include the limiting power distribution in any FAs and any axial position in the core, i.e. in terms of both post-element power distribution and local linear power. The analyses should respect the uncertainties including the impacts of the change in the geometry of the FA due to the operation on neutron and thermal-hydraulic behaviour and calculations (e.g. change in the fuel to moderator ratio due to the bowing or distortion of the FA). The analyses of the core design shall further include radial power distribution within the FA (including consideration of the distortion) and the deformations in the axial power distribution caused by spacer grids and other components to ensure that the calculations properly identify the most loaded places and determine the local power levels.

(5.114) The safety assessment should describe every initiating event that could cause unexpected criticality in fuel handling during refuelling.

(5.115) The accuracy in fuel loading in accordance with the design should be verified during start-up testing by using the in-core instrumentation for power distribution in the core (for more detailed requirements see Chapter 6 of this Safety Guide).

(5.116) The loading should be designed to ensure that even in the case of incorrect position of the FA, the design limits of the core are not exceeded for the given initiating event.

**Design criteria and limits**

(5.117) The cask charge shall include a set of key safety parameters to be used for the verification of the acceptability of the particular design of fuel loading in operation.

(5.118) The analysis of the design of core loading should verify that the charge in question meets all specified fuel design limits in all states, or verify compliance with all key safety parameters, see para. (5.6) and (5.117).

**Specific cases of core operation**

(5.119) For different operational states, the anticipated power distribution in the core and the anticipated history of temperatures should be pre-analysed in order to determine the impact on operation (e.g. load follow, power ramps, reactor start-up and refuelling). Evaluating these parameters should identify the effects of associated power and temperature cycles on thermal-mechanical behaviour of the FE (e.g. the increase in pressure in the FE due to increased leak of gaseous fission products into the gap between the pellet and the FE cladding, and fatigue of the FE cladding material).

(5.120) The design of the reactor core should ensure sufficient capability of reactivity control in load follow and other power ramps as to ensure the balance between core power and turbo-generator power, and sufficient stability in core power distribution.

(5.121) The operational limits of the core should adequately reflect the local deviations in power level caused in load follow.

(5.122) The design of the NI should facilitate detection and identification of damaged

FAs and FEs. In addition, the procedures for the identification of the cause of damage to the FA and the FE should be defined. The appropriate instrumentation should be designed for identifying the causes of damage to the FA and the FE. The licensee is obliged to identify the cause of damage to the FA and the FE, and should define the corrective actions.

(5.123) The adequate procedures should be defined for handling of damaged FAs and operational limits for the core with damaged FAs.

(5.124) The eventual operation of the core with damaged FEs should be kept within the limits of radiochemical requirements, which are given by the activity limits for radionuclides in the coolant referred to in the LaP.

(5.125) The design of the reactor core and the planning of operation of the NI should set the adequate procedures and limits for the operation of the core with damaged FAs in order to minimize the operation with leaking FEs, thus minimizing the individual doses to NI personnel.

(5.126) The FAs containing damaged FEs can be repaired or reconstructed by replacement of these FEs for spare FEs, or dummy FEs or by leaving empty spaces. Leaving empty spaces in the FAs should be limited in order to maintain the operational limits.

(5.127) The effect of repaired FE on the design of the reactor core should be adequately evaluated.

**The fuel design and fuel management effects on transport, storage and deposition**

(5.128) Other requirements should be added in the fuel design limits described in para. (5.48) - (5.56), the fulfilment of which ensures that the FEs and the FAs remains undamaged after the fuel campaign following their removal from the core to the SFSP. These requirements apply to handling, storage and deposition of FE and FA. The key safety parameters, which are monitored in the period of operation in the core and can substantially affect the behaviour of the FE and the FA after operation (irradiation in the reactor), include the following characteristics, which should be appropriately assessed:

- Increased oxidation and hydriding of the FE cladding and exceeding of the acceptance criteria, defined in the design of the FE and the FA, may result in a delayed hydrogen cracking of the FE cladding in handling or storage, or in damage to the FE during accident in transport, which should be limited and included in the design or eliminated.
- Excessive FE-to-grid fretting – in most cases, it is impossible to detect excessive (above the limit considered in the design analyses) local damage to the FE cladding until the wear passes through the whole wall and the FE cladding fails. For some FEs damaged by fretting the local reduction in wall thickness can occur and may result in damage to the FE cladding, e.g. due to

long-term creep in storage or stress as a result of an accident in transport, which should be limited and included in the design or eliminated.

- Achievable burnup – fuel design, design of fuel charges and final fuel discharge burnup affecting its isotopic composition and internal pressure in the FE. The high burnups achieved may increase the requirements for fuel shielding and cooling in subsequent handling and storage of the fuel.

## CALCULATIONS OF SUBCRITICALITY IN FUEL STORAGE IN SFSP AND CASKS

### General requirements

(5.129) The design of the NI and of the cask should be  to meet the requirements for subcriticality for the cask charge in transfer, transport, storage and disposal of fuel including the adequate uncertainties and margins for all the types of FA used.

(5.130) The design of the NI and of the cask should meet the requirements referred to in para. (5.129) to ensure that the value of effective multiplication factor is not exceeded:

- 0.95 for all states; and
- 0.98 for the conditions of optimum moderation for the casks in dry storage.

(5.131) The value of effective multiplication factor should be determined for the cask charge using a conservative approach, i.e. for fresh fuel. In calculating the value of multiplication factor, the sensitivity analysis should be conducted to determine the values of all uncertainties (in particular structural, calculation, methodology, etc.).

### Specific requirements for wet storage in storage grids

(5.132) The design of the storage grids shall include analyses for fresh fuel determining the value of subcriticality for the group, or shall use any of the methods referred to in para. (5.133) - (5.136). The analyses should be particularly conducted for the following conditions:

- Nuclear fuel flooding with clean water with maximum density;
- Drop of the FAs in handling of FAs (e.g. on the top parts of other FAs stored in the SFSP).

(5.133) For analyses of subcriticality in the SFSP, it is possible to use the BUC, PBC methods or to include the content of burnable absorbers, where the initial conditions include the minimum concentration of soluble absorber in the coolant of the SFSP (i.e. PBC), or the minimum burnup of the FAs stored in the SFSP (i.e. BUC). If any of the three methods is used, it is necessary to define the inspections and administrative measures to ensure that the limits specified on the basis of the assumptions of these calculations (minimum concentration of soluble absorber or minimum burnup of the FA) are not exceed. If the above mentioned subcriticality analyses reflect the amount of absorber in the structures of the SFSP, it is also necessary to define the inspections and administrative measures to ensure that the concentration of absorber in the given material (structural material or coolant) does not drop below the value used in calculations taking into account all uncertainties and margins.

(5.134) In justified cases of safety assessment, the BUC method using the change in isotopic composition during fuel burnup, which results in reduction of the reactivity of the FA, can allow to reflect the real content of fissile isotopes, the increase in the concentration of actinides, the increase in the concentration of fission products and the reduction in the concentration of burnable absorbers in individual FAs.

(5.135) The PBC method uses the minimum concentrations of soluble absorber in the coolant of the SFSP. This value should be determined on the basis of including particularly:

- The uncertainties in the measurement of concentration of soluble absorber in the coolant;
- Transport delay in reducing the concentration of soluble absorber;
- The maximum rate of the reduction in the concentration of soluble absorber;
- The uncertainty reflecting the absence of operator action in uncontrolled reduction in the concentration of soluble absorber;
- The uncertainties in determination of the value of shutdown concentration of soluble absorber.

(5.136) The method including the content of burnable absorbers uses the content of the burnable absorbers integrated in the fuel, either as part of the pellet or of the cladding. This method shall include a sensitivity analysis that particularly determines the uncertainties in the calculation, concentration, isotopic composition, amount and structural dimensions.

**Specific requirements for dry storage in casks**

(5.137) The analyses of subcriticality for the casks should be conducted for the conditions of design basis accidents and the conditions of optimum moderation. In addition, the conditions of optimum moderation should be defined.

(5.138) The analyses of subcriticality should be conducted for transport casks, in particular for the storage containers for fresh nuclear fuel and for the storage casks for irradiated nuclear fuel.

(5.139) The analyses of subcriticality shall include events associated with the drop of FAs on the cask, flooding with clean water with the maximum density and flooding with water foam.

# 6. VERIFICATION AND TESTING

## GENERAL

(6.1) The safe operation of the core, fuel system (FE, FA and core components), core control and monitoring systems throughout their design life requires a robust programme of verification, inspection and testing of the design processes and the design assessment of equipment. It can be achieved in the way described below.

(6.2) In case of operation of the new type of FA or of the new/modified type of FA (complete replacement of the entire core or the mixed core), it is required to propose the procedure for monitoring and testing of this core and/or fuel. This programme shall particularly cover the parameters that cannot be reliably demonstrated otherwise than by direct measurement on the systems, structures and components of the core, having regard to the practical feasibility of the measurement.

(6.3) The scope and procedure of monitoring/testing should be defined in the core/fuel monitoring and testing programmes for individual stages (the programme of core monitoring, physical tests, PIIP). The Post Irradiation Inspection Programme (PIIP) should identify the parameters that can be measured and implemented on the given type of the FA, e.g. visual inspections for seal-free structures of the FA.

## DESIGN VERIFICATION

(6.4) The verification programme should demonstrate the capability of the core/fuel of reliably performing its function at a relevant time interval, placing emphasis on the functional and safety assumptions considered, applicable in the given operating conditions (e.g. conditions associated with pressure, temperature, radiation load, mechanical load and vibrations). Those operating conditions shall include the deviations of parameters anticipated in all states.

(6.5) The characteristics of specified postulated initiating events may preclude performance of real tests at both first start and restart of the given system, which should demonstrate that the given systems, structures and components of the core are capable of performing their specified safety functions in the situation concerned, e.g. in case of earthquake. In such case, it is necessary to schedule and perform the required verification tests on the given systems, structures and components of the core prior to their installation in the NI.

(6.6) The methods for verification shall include the following:

- The type tests performed on the representative systems, structures and components of the core to be supplied;
- The tests performed on the systems, structures and components of the core supplied;
- The use of **applicable operational experience and experimental measurements**;
- The analyses based on available and usable test data;
- The direct measurement on the work in operation (where possible);
- Any combination of the methods mentioned above.

(6.7) The verification of the design of the fuel system can be based on properly documented operating experience with the same or similar fuel but the transferability of such experience should be demonstrated. The field of validity of such transferable operating experience should be clearly defined and the relevant records of the measurement in operation should be evaluated. The maximum burnup achieved should be stated, and the performance history and the measured characteristics of the behaviour of the FA should be compared with the defined design criteria for damage mechanisms and parameters such as radiation induced growth, bowing, distortion, friction and drop time of the control devices of mechanical regulation, fretting, oxidation, hydriding and formation of deposits.

(6.8) The scope of and method for measuring the physical start-up tests to verify the neutron-physical characteristics of the core should be defined prior to the physical start-up of the nuclear reactor with the new type of the core or fuel, either in case of first start-up or following refuelling. These tests shall mainly include measurement of key safety parameters of the core, which can be reliably verified in the NI (critical concentration of $H_3BO_3$, neutron characteristics of the control rods, power distribution, reactivity coefficients and effects, etc.). The results of measurement obtained should be compared with the results obtained analytically and with the values of the relevant design criteria. The physical start-up tests should be designed and performed in accordance with lit. [2], using lit. [20].

(6.9) The physical start-up tests should verify the validity of the pre-calculated neutron-physical characteristics of the core and of the safety assessment of the core design, which include a determination of the most important parameters and their comparison with the values of cask charge. The neutron-physical characteristics of the core should be calculated prior to the first core load.

## INSPECTIONS

(6.10) The systems of the NI should be designed as to be able to identify each FA and the core component, and to ensure its correct location and orientation (turning) in the core. After the first core load and after every refuelling, the correct position and orientation of each FA and the core component should be verified (for core components, administratively or visually).

(6.11) The instrumentation for measurement and inspections of the fuel system should be maintained in operable condition. The design of the instrumentation shall include the service life, mode of maintenance and handling of instrumentation. The design of the instrumentation should be designed in accordance with the latest knowledge of science and technology at the time of design.

(6.12) The instrumentation for measurement and inspections of the fuel system should be periodically checked to ensure that the accuracy and application of measurement are adequate or to modernize the instrumentation and bring it into line with the latest knowledge of science and technology.

(6.13) The instrumentation for measurement and inspections of the fuel and the fuel measurement and inspection programmes should be designed as to provide information about the current condition of the fuel. The condition of the fuel should be supported by calculations, measurements or their combination.

(6.14) The fuel monitoring and testing programme should take into account the design limits for the FA and the FE, and define the procedures for their verification, in particular:

- The rate of oxidation and hydriding of the FE cladding and of the structural members of the FA;
- The deformation of the FA and of the FE;
- The presence of foreign objects and failure of the integrity of the FA and the FE.

(6.15) Prior to the loading of the FAs into the casks, they should be checked for tightness.

(6.16) Prior to the loading of the FAs into the casks, the following should be particularly determined by calculations (taking into account the real performance history) or by measurements or by their combination for each FA:

- The size of burnup determined in accordance with the performance history of the FA;
- The inventory of the amount of fissile materials;
- The internal pressure in the FE;
- The quantity of heat power of the FA.

## PROTOTYPE AND LEAD TEST ASSEMBLY TESTING

(6.17) In testing the prototypes and the LTA in the core, a schedule should be drawn up for operation and a monitoring and testing programme for the given FA/core component. The prototype means a FA or core component which has not been operated in the same type of the NI yet, i.e. **applicable operational experience and experimental measurements** with its behaviour is not available.

(6.18) The FA/core component monitoring and testing programme should be drawn up in the extent to check the behaviour of the FA/core component in operation in the core. An example of such programme for seal-free structures of the FA is provided in Annex 2 in para. (9.5). The programme shall specify, among others, what measurements will be performed and what methods will be used to measure the individual acceptance criteria and to meet the objectives of testing. In addition, the expected results of these measurements should be determined and compared with the values measured.

(6.19) The FA/core component monitoring and testing programme should be continuously evaluated after each fuel campaign. The evaluation of the programme shall particularly include the comparison of the expected (calculated) values and the measured values of the monitored parameters and their comparison with relevant acceptance criterion. After completion of the programme, the overall evaluation of the results of the programme should be prepared, including but not limited to what are the final margins for acceptance criteria; the representativeness and scope of applicability of the results obtained; the method for transferring the conclusions of the programme into the conditions of future use of the FA/core components, their design, fabrication, quality assurance, operation, storage and deposition. The summary evaluation of the FA/core component monitoring and testing programme (LTA, PIIP, etc.) should be performed immediately after completion of the programme. The evaluation shall include the comparison of the expected values and the pattern of the monitored parameters with the measured data and the criterion values; the representativeness and scope of applicability of the results obtained in comparison with the real operating conditions; the conclusions regarding the test results and compliance with the objectives of the programme; the feedback from operating experience and the subsequent corrective and safety measures including recommendations for continued operation or testing.

(6.20) Prior to each first core load for the new type of the FA/core component, the given prototype or the selected group of the LTA/CCs should be subject to the measurement of the initial condition to the extent necessary to meet the objectives of testing (the so-called pre-characterization, i.e. measuring the mass, geometry, as specified in the design documentation and the documentation submitted after transport to the NI) and neutron-physical, thermal-hydraulic and mechanical conditions should be defined, in which the relevant prototype or the LTA/CCs will be operated during the next campaign. After each campaign, the actual working conditions for the LTA/CCs should be evaluated and compared with the expected working conditions. In case of major differences, the validity of the assumptions of the testing programme should be re-evaluated or the relevant time schedule for operation should be modified in order to meet the objective of the monitoring and testing programme.

(6.21) The design of the prototype of the FA/core component should provide for monitoring, testing and inspection in order to demonstrate that the core, the FA, the core components, the reactor internals, and the control and shutdown systems are capable of performing their design and in particular safety functions throughout their service life. For more detailed information and recommendations for monitoring, inspection and testing see [10].

(6.22) Testing and measuring the prototypes of the FA/core components outside the reactor, e.g. on test stands, loops, in the SFSP and in hot cells in order to determine their actual characteristics (mechanical, thermal-hydraulic, etc.) should be performed whenever practicable. In particular, it includes the following types of measurement:

- Load tests of the spacer grids;
- Mechanical and functional measurement of the mobility of control rods;
- Structural mechanical tests of the FA (lateral, axial and torsional rigidity, measurement of frequencies, amplitude and characteristics of vibration damping);
- Measuring of hydraulic characteristics of the FA, hydraulic resistance, total hydraulic loss and local resistance, verification of the validity of the correlations of critical heat flux, uplift forces, hydraulic-dynamic vibrations and fretting of the control rods, FA vibrations, FE fretting (while respecting the expected relaxation of contact springs in the spacer grid), wear and assessment of the overall life of the FA/core component, seismic resistance, resistance to transverse flow, etc.

(6.23) During irradiation tests in the form of the LTS programmes in a reactor, the burnup achieved should be in accordance with the design value of burnup, fluency or performance history for the new type of the FE, the FA or the core components. The following parameters should be measured:

- Radiation induced growth and changes in the outer diameter of the FE, core components;
- Oxidation and hydriding of the FE, spacer grids, guide tubes (e.g. FA seal);
- Grid-to-rod fretting;
- Growth, bowing and turn of the FA;
- Axial growth and bowing of the FE;
- External signs of mechanical interaction between the pellets and the FE cladding (ridging, bambooning);
- (Un)leakage and failureof the FE cladding;
- Relaxation of the hold-down springs of the FA and of the contact springs of the spacer grids;
- Fretting and failure of guide tubes of the FA.

(6.24) Where it is not possible to perform the testing of the new FA/core component in the reactor or on the test stand, special attention should be paid to analytical methods for evaluating the relevant design criteria and subsequent fuel monitoring and inspection plans as to ensure compliance with the functional and safety requirements for the FA/core components.

(6.25) The core, where the test LTAs are located, is considered to be a mixed core and so the mixed core requirements are applicable. .

# 7. REQUIREMENTS FOR THE MIXED CORE

## ANALYSES OF THE DESIGN CHARACTERISTICS OF THE MIXED CORE

(7.1) All analyses should be conducted for all the types of fuel used in the core, although the suppliers of fuel are different, unless otherwise demonstrated (e.g. by specifying the method on the basis of more recent and more precise experiments) and justified. The analyses should be conducted in accordance with para. (4.44).

(7.2) The design of the mixed core shall include calculations of pressure gradient redistribution between the new and the current type of the FA, which is used for determination of the force applied to the hold-down spring of the FA. The hold-down forces of springs should be designed as to prevent the lifting of the FA and at the same time, not to cause damage to the geometrical stability of the FA (loss of stability in buckling stress of the FA).

(7.3) The design of the mixed core should prevent excessive deformation of FA geometry and should not exceed the predefined design difference in hydraulic resistance of the FAs of different types (in particular of the new type of the FA adjacent to the current type of the FA).

(7.4) The design of the new type of the FA shall include evaluation of the dependence of the vibrations of individual components on the burnup, vibration intensity and the rate of fretting having regard to the conditions in the mixed core.

(7.5) The cask load for the mixed core shall include calculations of coolant flow around the individual FEs. The flow should be characterized by the position of the FA in the core and by the distribution of the rate of coolant flow in the core. The calculations of flows should cover the worst possible variants of the distribution of coolant flow in the core and around the FE (ensuring a sufficient conservatism).

(7.6) The cask charge shall include calculations to determine the magnitude of the excitation forces applied to the individual types of the FAs located together in the core. The results of the calculations, executed for all types of the FAs located in the core, should be confirmed by experimental measurements of the frequencies of the FAs.

(7.7) The different number of turbulent and spacer grids in the new type of the FA compared to the current FA causes a change in the magnitude and distribution of hydraulic resistance, resulting in a change in the distribution of the transverse flow of the coolant in the core and of the flow in the FA. A large difference in hydraulic resistance may cause significant reduction in flow rate, in particular local flows in the individual types of the FA, thus reducing the value of the DNBR. The rate of flow and of transverse flow should be experimentally verified, analysed and evaluated. For the purposes of DNBR calculation, in the absence of experimental data for local values of the flow through the mixed core, the rate of flows should be determined using a conservative approach or calculated using a validated computer code. The impact of flow redistribution on the DNBR should be assessed, while this assessment shall include requirements for the mode of reactor operation (in this case, usually the so-called correction factors for element performance limits) ensuring compliance with the DNBR criteria for the mixed core.

(7.8) If the types of the FA concurrently operated in the mixed core have different structural rigidity, the design shall include calculations of the forces applied to the FA during design basis accidents and shall determine the rate of deformation of the geometrical stability of the FA and demonstrate the performance of fundamental safety functions. Combining the bowings of the individual types of FA should not limit the capability of dropping the control rods or cause jamming the control rods in the guide tube, the so-called IRI (Incomplete Rod Insertion), which would result in failure of the limit for shutdown margin. The analyses or experimental data should demonstrate that the geometrical stability of the FA is not challenged and that the fundamental safety functions are performed.

(7.9) The hold-down forces of springs of the FA should be determined to push sufficiently the FA of all fuel types located in the core against the support plate in order to provide protection against lifting of the FA in all states. At the same time, the forces should not cause a bowing of the FA of all fuel types located in the core resulting in failure of the fundamental safety functions.

(7.10) For safety analyses for the LOCA, it is necessary to include non-homogeneities due to the mixed core for the worst configuration.

## NEUTRON-PHYSICAL CHARACTERISTICS OF THE MIXED CORE

(7.11) The design of the reactor core should reflect all geometrical, material and operational changes associated with the implementation of the new fuel system. For example, change in the structure of the FA or the FE may lead to a change in the water-uranium ratio, changing (distorting) the neutron flux at the interface of different fuel types.

(7.12) The calculations should adequately include all non-homogeneities at the boundaries of individual fuel types, which may result in an unexpected change in the neutron flux density and the power distribution in the core, or in a loss of the validity of the calculation methods used. The design of the reactor core should adequately take into account the effects of the use of different types and location of burnable absorbers, structural materials and different thermal-hydraulic behaviour of the channels on the distribution of temperatures. In particular, the impacts on design uncertainties of critical parameters should be taken into account.

(7.13) If burnable absorbers are used in the design of the fuel system, the pattern of their burnup and the effect of this burnup on the adjacent FA and the entire core should be modelled. At the same time, the effect of high gradient of the neutron flux caused by the presence of burnable absorber should be included in the uncertainties of in-core measurement.

(7.14) The effects of the mixed core described in para.(7.11) - (7.13) should be demonstrably taken into account in the models and uncertainties in the imbalance coefficients of power distribution monitoring systems.

(7.15) If the design of the new type of the FA differs from other types of fuel in column height of the pellets in the FE and in the size of axial blanket (column of uranium oxide pellets made of naturally occurring uranium), changing the assumptions for safety analyses, the new FAs should be loaded gradually by lower number of the FAs in the core in order to minimize the axial non-homogeneities caused by loading such different FAs. The loading and refuelling programme for the new type of the FA should be drawn up and justified before the intended loading.

(7.16) In calculations in the context of the analyses, the relevant computer codes and the proven libraries reflecting the new types of the FA should be used. The analyses shall include evaluation of the uncertainties in calculations, the computer code methods used and the sensitivity analysis. The computer codes and the libraries used should be properly supported by valid experimental data and verified by benchmark tasks or should be assessed by the Expert Assessment Committee.

(7.17) The design of the mixed core shall include the analyses conducted for all states of the NI, which determine the maximum values of key safety parameters depending on the type of fuel and if appropriate, limit the value of global and local density of heat generation in the core.

## THERMAL-HYDRAULIC PROPERTIES OF THE MIXED CORE

(7.18) The thermal-hydraulic characteristics of the new type of the FA or the FE should be designed as close to the current FAs and FEs as possible.

(7.19) The overall hydraulic resistance of new FAs should minimize the adverse impact on the input parameters for the calculations of safety assessment of the current FAs. Any change in the hydraulic resistance of the new FA compared to the current FA, which deteriorates the input parameters for the calculations of safety analyses, should be analysed in detail and assessed.

(7.20) The uplift forces should be calculated for the new type of FA and FE for the most severe hydraulic conditions, i.e. overriding the operating speed of the main circulation pumps in hot condition at nominal power and for cold conditions at zero power.

(7.21) Should the new type of the FA have the hydraulic resistance different than the current FA, the thermal-hydraulic analyses should reflect the redistribution of coolant flow in the core (calculations of the DNBR, coolant rate vectors and distribution of coolant enthalpy).

(7.22) The analysis of seal-free structures of the FA shall include calculations of transverse flow among the individual FAs of different types. The analyses should demonstrate sufficient cooling of the individual FEs and the mechanical stress on the FEs and the FAs should not increase to the level exceeding the design limits. The calculations of transverse flow should be supported by experimental data obtained from testing of the new dummy FA.

(7.23) The effect of hydraulic loads according to para. (5.24) should be evaluated for all types of the FAs located in the core. The methodology for testing and the selection of test conditions should ensure compliance with the relevant design limits for each FA located in the core with a reserve ensuring that the design limits are not exceeded for all FAs located together in the core.

## NUCLEAR REACTOR SHUTDOWN SYSTEMS AND CORE CONTROL AND MONITORING SYSTEMS

(7.24) The core monitoring and control systems should be set to be capable of measuring and identifying any changes in important parameters for all fuel types located in the core to ensure that the fuel design limits are not exceeded. In addition, the effect of different types of the FA and the FE on the accuracy in measurement and on the capability of identifying any change in important parameters should be evaluated.

## DESIGN VERIFICATION

(7.25) In case of the first application of the mixed core, the current programme of physical start-up testing should be re-evaluated and it should be assessed whether the current scope is sufficient or requires extension.

# 8. CONCLUSION

(8.1) This Safety Guide summarizes and elaborates the requirements laid down in Czech legislation, WENRA and IAEA documents (for comparison see Annex 3), world practice of the regulators (e.g. US NRC, STUK, ÚJD, etc.).

# 9. ANNEXES

## ANNEX 1

## Points to be covered in the design of the FE, FA, control rods, neutron sources, hydraulic plugs and other core components (e.g. material cluster)

(9.1) The design of the FE shall include the analyses for the following:

**FE cladding**

- The FE fretting and vibrations (FE-to-grid fretting);
- The mechanical properties of the FE cladding after irradiation;
- The material and chemical properties;
- The corrosion stress;
- The cyclical stress and material fatigue;
- The geometrical and chemical stability of the FE cladding after irradiation.

**Material of pellets, control rods and burnable absorbers**

- The geometrical stability of irradiated pellets;
- The temperature compaction of the pellets (kinetics and densification rate);
- The chemical interactions between the FE cladding and the pellets;
- The amount of gaseous fission productions and their distribution in the pellet;
- The kinetics of the release of gaseous fission products;
- Smooth swelling;
- The thermal-mechanical properties after irradiation;
- The microstructural changes as a function of irradiation.

**FE properties**

- Temperatures and temperature distribution in the pellet and in the FE cladding;
- The kinetics and gaps between the pellet and the FE cladding;
- The effects of irradiation on the behaviour of the FE (e.g. rearrangement of the fragments of pellets, cracking of pellets, swelling of solid and gaseous fission products, release of gaseous fission products and increase in pressure in the FE, deterioration in heat transfer in the FE cladding);
- The bowing of the FE;
- The growth of the FE.

The analyses should be conducted using validated analytical models and/or experimental data (obtained from research facilities or from operating experience, e.g. LTA). The models should facilitate burnup simulation.

(9.2) The design of the control rods shall include the analyses for the following:

- The internal pressure and associated stress in the FE cladding in normal and abnormal operation and in the conditions of design basis accidents;
- Thermal expansion and swelling due to irradiation;
- The changes in absorbing material and FE cladding due to irradiation;
- The effects of fretting on resistance of the FE cladding.

(9.3) The design of the neutron sources shall include the analyses for the following:

- The effects of irradiation;
- The efficiency including the effects of shadowing by peripheral FAs;
- External hazards (e.g. earthquake).

(9.4) The design of the hydraulic plugs and of the material clusters shall include the analyses for the following:

- The interactions with guide tubes due to thermal expansion or swelling from irradiation;
- The effect of fretting on resistance of the guide tubes.

# ANNEX 2

(9.5) **Monitoring and testing programme**

- The comparison of predicted and measured performance of the FE/FA during the previous campaign for the start, middle and end of the campaign.
- For control rod drop test, recording the course and total time of drop for all control rods, thus verifying the performance of safety function of the FA and the control rods, the proper functioning of the position indicator and the geometrical stability of guide tubes.
- The coolant gamma-spectrometry analysis during the campaign and after unit shutdown in order to detect the presence of leaking FEs.
- The friction test of control rods in dismantling and reassembly of the reactor.
- The on-line sipping, or the off-line or FE cladding leak-tightness control to detect leaking FAs, which can be further checked or tested (e.g. ultrasound testing).
- Recording the forces applied by the fuel charging machine in handling of the FAs and the core components in order to protect the FAs and the core components against damage in handling (fretting, jamming, rupture, etc.) and in order to detect any excessive deformation of the FAs and the core components.
- The verification of compliance of the design assumptions with the actual behaviour of the FA and the FE in the core. This verification can be made, for example, by:
    - Inspecting the leaking FAs in order to determine the causes and extent of FE failure (fretting, manufacturing defects, etc.) by visual inspection using a camera system, ultrasound or any other available method for detecting the position and identifying the failure mechanism for the leaking FE.
    - Identifying the FAs that are representative for all types of the FAs operated in the core, by subsequently measuring their geometry after removal from the core. In particular, it includes measurement of twist, bowinging and axial growth of the FE and the FA.
    - Inspecting the FAs to detect the presence and rate of hydriding and oxidation in the FE cladding by visual inspection or by using other available methods.
    - Testing the connection of suspension/inserted rod with the control devices.

# ANNEX 3

## Table for comparison with world practice

## Table 1 Comparison with the reference levels: WENRA Reactor Safety Reference Levels – E, G and Q areas

| **WENRA Reactor Safety Reference Levels**<br>**E, G and Q areas** | **Implementing paragraphs of this Guide** |
|---|---|
| **Issue E: Design Basis Envelope for Existing Reactors** | |
| **E1 Objective** | |
| **E1.1** The design basis shall have as an objective the prevention or, if this fails, the mitigation of consequences resulting from anticipated operational occurrences and design basis accidents. Design provisions shall be made to ensure that potential radiation doses to the public and the site personnel do not exceed prescribed limits and are as low as reasonably achievable. | **(4.13)** |
| **E2 Safety strategy** | |
| **E2.1** Defence-in-depth shall be applied in order to prevent, or if prevention fails, to mitigate harmful radioactive releases. | **(4.14)** |
| **E2.2** The defence-in-depth concept shall be applied to provide several levels of defence including a design that provides a series of physical barriers to prevent uncontrolled releases of radioactive material to the environment, as well as a combination of safety features that contribute to the effectiveness of the barriers. The design shall prevent as far as practicable:<br>• challenges to the integrity of the barriers;<br>• failure of a barrier when challenged;<br>• failure of a barrier as consequence of failure of another barrier. | **(4.14)**<br>**(4.15)** |
| **E3. Safety functions** | |
| **E3.1** During normal operation, anticipated operational occurrences and design basis accidents, the plant shall be able to fulfil the fundamental safety functions:<br>• control of reactivity,<br>• removal of heat from the reactor core and from the spent fuel, and<br>• confinement of radioactive material. | **(4.14)** |

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| **E4. Establishment of the design basis** | |
| **E4.1** The design basis shall specify the capabilities of the plant to cope with a specified range of plant states within the defined radiation protection requirements. Therefore, the design basis shall include the specification for normal operation, anticipated operational occurrences and design basis accidents from Postulated Initiating Events (PIEs), the safety classification, important assumptions and, in some cases, the particular methods of analysis. | **(4.21)** <br> **(4.22)** |
| **E4.2** A list of PIEs shall be established to cover all events that could affect the safety of the plant. From this list, a set of anticipated operational occurrences and design basis accidents shall be selected using deterministic or probabilistic methods or a combination of both, as well as engineering judgement. The resulting design basis events shall be used to set the boundary conditions according to which the structures, systems and components important to safety shall be designed, in order to demonstrate that the necessary safety functions are accomplished and the safety objectives met. | **(4.22)** <br> **(4.23)** <br> **(4.25)** |
| **E4.3** The design basis shall be systematically defined and documented to reflect the actual plant. | **(4.23)** |
| **E5. Set of design basis events** | |
| **E5.1** Internal events such as loss of coolant accidents, equipment failures, maloperation and internal hazards, and their consequential events, shall be taken into account in the design of the plant. The list of events shall be plant specific and take account of relevant experience and analysis from other plants. | **(4.13)** <br> **(4.23)** <br> **(4.29)** |
| **E7.2** Criteria for protection of the fuel rod integrity, including fuel temperature, Departure from Nucleate Boiling (DNB), and cladding temperature, shall be specified. In addition, criteria shall be specified for the maximum allowable fuel damage during any design basis accident. | **(4.7)** |
| **E8. Demonstration of reasonable conservatism and safety margins** | |
| **E8.1** The initial and boundary conditions shall be specified with conservatism. | **(4.40)** |
| **E8.4** A stuck control rod shall be considered as an additional aggravating failure in the analysis of design basis accidents. | **(5.84)** |
| **E8.7** The safety analysis shall: <br> (a) rely on methods, assumptions or arguments which are justified and conservative; <br> (b) provide assurance that uncertainties and their impact have been given adequate consideration; <br> (c) give evidence that adequate margins have been included when defining the design basis to ensure that all the design basis events are covered; <br> (d) be auditable and reproducible. | **(4.39)(4.40)** <br> **(4.41)** <br> **(4.42)** <br> **(4.43)** |

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| **E9. Design safety functions** | |
| **E9.1** The fail-safe principle shall be considered in the design of systems and components important to safety. | **(5.84)** |
| **E9.2** A failure in a system intended for normal operation shall not affect a safety function. | **(5.85)**<br>**(5.84)**<br>**(5.91)** |
| **E9.4** The reliability of the systems shall be achieved by an appropriate choice of measures including the use of proven components, redundancy, diversity, physical and functional separation and isolation. | **(5.85)**<br>**(5.84)**<br>**(5.91)** |
| **E9.8** Sub-criticality shall be ensured and sustained:<br>• in the reactor after planned reactor shutdown during normal operation and after anticipated operational occurrences, as long as needed;<br>• in the reactor, after a transient period (if any) following a design basis accident[33];<br>• for fuel storage during normal operation, anticipated operational occurrences, and design basis accidents. | **(5.88)**<br>**00** |
| **E9.9** Means for removing residual heat from the core after shutdown and from spent fuel storage, during and after anticipated operational occurrences and design basis accidents, shall be provided taking into account the assumptions of a single failure and the loss of off-site power. | **(4.14)** |
| **E10. Instrumentation and control systems** | |
| **E10.1** Instrumentation shall be provided for measuring all the main variables that can affect the fission process, the integrity of the reactor core, the reactor cooling systems, the containment, and the state of the spent fuel storage. Instrumentation shall also be provided for obtaining any information on the plant necessary for its reliable and safe operation, and for determining the status of the plant in design basis accidents. Provision shall be made for automatic recording of measurements of any derived parameters that are important to safety. | **(5.68)** |
| **E10.2** Instrumentation shall be adequate for measuring plant parameters and shall be environmentally qualified for the plant states concerned. | **(5.99)** |
| **E10.7** Redundancy and independence designed into the protection system shall be sufficient at least to ensure that:<br>• no single failure results in loss of protection function; and<br>• the removal from service of any component or channel does not result in loss of the necessary minimum redundancy. | **(5.91)** |
| **E10.8** The design shall permit all aspects of functionality of the protection system, from the sensor to the input signal to the final actuator, to be tested in operation. | **(5.90)** |

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| Exceptions shall be justified. | **(5.85)** |
| **E10.9** The design of the reactor protection system shall minimize the likelihood that operator action could defeat the effectiveness of the protection system in normal operation and anticipated operational occurrences. Furthermore, the reactor protection system shall not prevent operators from taking correct actions if necessary in design basis accidents. | **(5.62)** |
| **Issue G: Safety Classification of Structures, Systems and Components** | |
| **G1. Objective** | |
| **G1.1** All SSCs important to safety shall be identified and classified on the basis of their importance for safety. | **(4.18)**<br>**(4.21)** |
| **G2. Classification process** | |
| **G2.1** The classification of SSCs shall be primarily based on deterministic methods, complemented where appropriate by probabilistic methods and engineering judgment. | **(4.21)** |
| **G2.2** The classification shall identify for each safety class:<br>• The appropriate codes and standards in design, manufacturing, construction and inspection;<br>• Need for emergency power supply, qualification to environmental conditions;<br>• The availability or unavailability status of systems serving the safety functions to be considered in deterministic safety analysis;<br>• The applicable quality requirements. | **(4.29)** |
| **Issue Q: Plant Modifications** | |
| **Q1. Purpose and scope** | |
| **Q1.1** The licensee shall ensure that no modification to a nuclear power plant, whatever the reason for it, degrades the plant's ability to be operated safely. | **(5.7)** |
| **Q2. Procedure for dealing with plant modifications** | |
| **Q2.1** The licensee shall establish a process to ensure that all permanent and temporary modifications are properly designed, reviewed, controlled, and implemented, and that all relevant safety requirements are met. | **(2.4)**<br>**(4.23)**<br>**(5.7)** |
| **Q2.2** For modifications to SSC, this process shall include the following:<br>• Reason and justification for modification; | **(3.1)** |

| | |
|---|---|
| • Design;<br>• Safety assessment;<br>• Updating plant documentation and training;<br>• Fabrication, installation and testing; and<br>• Commissioning the modification. | **(5.7)** |
| **Q3. Requirements on safety assessment and review of modifications** | |
| **Q3.1** An initial safety assessment shall be carried out to determine any consequences for safety. | **(4.39)**<br>**(6.4)** |
| **Q3.3** Comprehensive safety assessments shall demonstrate all applicable safety aspects are considered and that the system specifications and the relevant safety requirements are met. | **(4.22)**<br>**(6.6)** |
| **Q4. Implementation of modifications** | |
| **Q4.1** Implementation and testing of plant modifications shall be performed in accordance with the applicable work control and plant testing procedures. | **(6.1)**<br>**(6.2)**<br>**(6.3)** |
| **Q4.2** The impact upon procedure training, and provisions for plant simulators shall be assessed and any appropriate revisions incorporated. | **(6.7)** |
| **Q4.3** Before commissioning modified plant or putting plant back into operation after modification, personnel shall have been trained, as appropriate, and all relevant documents necessary for plant operation shall have been updated. | **(6.5)** |

## Table 2 Comparison with the requirements of SSR-2/1 (Rev. 1)

| **IAEA Safety Standards**<br>**For protecting people and the environment**<br>**Safety of Nuclear Power Plants: Design** | **Implementing paragraphs of this Guide** |
|---|---|
| **Requirement 4: Fundamental safety functions**<br>**Fulfilment of the following fundamental safety functions for a nuclear power plant shall be ensured for all plant states: (i) control of reactivity; (ii) removal of heat from the reactor and from the fuel store; and (iii) confinement of radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.** | **(4.14)** |
| 4.1 A systematic approach shall be taken to identifying those items important to safety that are necessary to fulfil the fundamental safety functions and to identifying the inherent features that are contributing to fulfilling, or that are affecting, the fundamental safety functions for all plant states. | **(4.18)** |
| 4.2. Means of monitoring the status of the plant shall be provided for ensuring that the required safety functions are fulfilled. | **(5.70)**<br>**(5.68)** |
| **Requirement 7: Application of defence in depth**<br>**The design of a nuclear power plant shall incorporate defence in depth. The levels of defence in depth shall be independent as far as is practicable.** | **(4.13)** |
| 4.9. The defence in depth concept shall be applied to provide several levels of defence that are aimed at preventing consequences of accidents that could lead to harmful effects on people and the environment, and ensuring that appropriate measures are taken for the protection of people and the environment and for the mitigation of consequences in the event that prevention fails. | **(4.17)** |
| 4.13A. The levels of defence in depth shall be independent as far as practicable to avoid the failure of one level reducing the effectiveness of other levels. In particular, safety features for design extension conditions (especially features for mitigating the consequences of accidents involving the melting of fuel) shall as far as is practicable be independent of safety systems. | **(4.15)** |
| **Requirement 9: Proven engineering practices** | |
| 4.15. National and international codes and standards that are used as design rules for items important to safety shall be identified and evaluated to determine their applicability, adequacy and sufficiency, and shall be supplemented or modified as | **(4.22)** |

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| necessary to ensure that the quality of the design is commensurate with the associated safety function. | |
| 4.16. Where an unproven design or feature is introduced or where there is a departure from an established engineering practice, safety shall be demonstrated by means of appropriate supporting research programmes, performance tests with specific acceptance criteria or the examination of operating experience from other relevant applications. The new design or feature or new practice shall also be adequately tested to the extent practicable before being brought into service, and shall be monitored in service to verify that the behaviour of the plant is as expected. | **(6.1) - (6.8)** |
| **Requirement 43: Performance of fuel elements and assemblies**<br><br>**Fuel elements and assemblies for the nuclear power plant shall be designed to maintain their structural integrity, and to withstand satisfactorily the anticipated radiation levels and other conditions in the reactor core, in combination with all the processes of deterioration that could occur in operational states.** | **(5.29)** |
| 6.1. The processes of deterioration to be considered shall include those arising from:<br>• Differential expansion and deformation;<br>• External pressure of the coolant;<br>• Additional internal pressure due to fission products and the build-up of helium in fuel elements;<br>• Irradiation of fuel and other materials in the fuel assembly;<br>• Variations in pressure and temperature resulting from variations in power demand;<br>• Chemical effects;<br>• Static and dynamic loading, including flow induced vibrations and mechanical vibrations;<br>• Variations in performance in relation to heat transfer that could result from distortion or chemical effects.<br><br>Allowance shall be made for uncertainties in data, in calculations and in manufacture. | **(5.29)**<br><br>**(5.35)**<br><br>**(5.79)** |
| 6.2. Fuel design limits shall include limits on the permissible leakage of fission products from the fuel in anticipated operational occurrences so that the fuel remains suitable for continued use. | **(5.125)** |
| 6.3. Fuel elements and fuel assemblies shall be capable of withstanding the loads and stresses associated with fuel handling. | **(4.35)**<br><br>**(9.5)** |
| **Requirement 44: Structural capability of the reactor core** | **(5.57)** |

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| **The fuel elements and fuel assemblies and their supporting structures for the nuclear power plant shall be designed so that, in operational states and in accident conditions other than severe accidents, a geometry that allows for adequate cooling is maintained and the insertion of control rods is not impeded.** | |
| **Requirement 45: Control of the reactor core**<br><br>**Distributions of neutron flux that can arise in any state of the reactor core in the nuclear power plant, including states arising after shutdown and during or after refuelling, and states arising from anticipated operational occurrences and from accident conditions not involving degradation of the reactor core, shall be inherently stable. The demands made on the control system for maintaining the shapes, levels and stability of the neutron flux within specified design limits in all operational states shall be minimized.** | **(5.67)**<br><br>**(5.101)**<br><br>**(5.103)** |
| 6.4. Adequate means of detecting the neutron flux distributions in the reactor core and their changes shall be provided for the purpose of ensuring that there are no regions of the core in which the design limits could be exceeded. | **(5.67)**<br><br>**(5.97)**<br><br>**(7.24)** |
| 6.5. In the design of reactivity control devices, due account shall be taken of wear out and of the effects of irradiation, such as burnup, changes in physical properties and production of gas. | **(5.79)** |
| 6.6. The maximum degree of positive reactivity and its rate of increase by insertion in operational states and accident conditions not involving degradation of the reactor core shall be limited or compensated for, to prevent any resultant failure of the pressure boundary of the reactor coolant systems, to maintain the capability for cooling and to prevent any significant damage to the reactor core. | **(5.74)** |
| **Requirement 46: Reactor Shutdown**<br><br>**Means shall be provided to ensure that there is a capability to shut down the reactor of the nuclear power plant in operational states and in accident conditions, and that the shutdown condition can be maintained even for the most reactive conditions of the reactor core.** | **(5.80)**<br><br>**(5.82)** |
| 6.7. The effectiveness, speed of action and shutdown margin of the means of shutdown of the reactor shall be such that the specified design limits for fuel are not exceeded. | **(5.87)** |
| 6.8. In judging the adequacy of the means of shutdown of the reactor, consideration shall be given to failures arising anywhere in the plant that could render part of the means of shutdown inoperative (such as failure of a control rod to insert) or that could result in a common cause failure. | **(5.85)**<br><br>**(5.86)** |

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| 6.9. The means for shutting down the reactor shall consist of at least two diverse and independent systems. | **(5.85)** |
| 6.10. At least one of the two different shutdown systems shall be capable, on its own, of maintaining the reactor subcritical by an adequate margin and with high reliability, even for the most reactive conditions of the reactor core. | **(5.85)** |
| 6.11. The means of shutdown shall be adequate to prevent any foreseeable increase in reactivity leading to unintentional criticality during the shutdown, or during refuelling operations or other routine or non-routine operations in the shutdown state. | **(5.87)** |
| 6.12. Instrumentation shall be provided and tests shall be specified for ensuring that the means of shutdown are always in the state stipulated for a given plant state. | **(5.90)** |
| **Requirement 80: Fuel handling and storage systems**<br><br>**Fuel handling and storage systems shall be provided at the nuclear power plant to ensure that the integrity and properties of the fuel are maintained at all times during fuel handling and storage.** | (5.29) |
| 6.64. The design of the plant shall incorporate appropriate features to facilitate the lifting, movement and handling of fresh fuel and spent fuel. | **(4.24)** |
| 6.65. The design of the plant shall be such as to prevent any significant damage to items important to safety during the transfer of fuel or casks, or in the event of fuel or casks being dropped. | **(4.35)** |
| 57 6.66. The fuel handling and storage systems for irradiated and non-irradiated fuel shall be designed:<br>a) To prevent criticality by a specified margin, by physical means or by means of physical processes, and preferably by use of geometrically safe configurations, even under conditions of optimum moderation;<br>b) To permit inspection of the fuel;<br>c) To permit maintenance, periodic inspection and testing of components important to safety;<br>d) To prevent damage to the fuel;<br>e) To prevent the droPEEing of fuel in transit;<br>f) To provide for the identification of individual fuel assemblies;<br>g) To provide proper means for meeting the relevant requirements for radiation protection;<br>h) To ensure that adequate operating procedures and a system of accounting for, and control of, nuclear fuel can be implemented to prevent any loss of, or loss of control over, nuclear fuel. | **(4.14)**<br><br>**(5.28)**<br><br>**(5.133)** |

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| 6.67. In addition, the fuel handling and storage systems for irradiated fuel shall be designed:<br><br>a) To permit adequate removal of heat from the fuel in operational states and in accident conditions;<br>b) To prevent the dropping of spent fuel in transit;<br>c) To avoid causing unacceptable handling stresses on fuel elements or fuel assemblies;<br>d) To prevent the potentially damaging dropping of heavy objects such as spent fuel casks, cranes or other objects onto the fuel;<br>e) To permit safe keeping of suspect or damaged fuel elements or fuel assemblies;<br>f) To control levels of soluble absorber if this is used for criticality safety;<br>g) To facilitate maintenance and future decommissioning of fuel handling and storage facilities;<br>h) To facilitate decontamination of fuel handling and storage areas and equipment when necessary;<br>i) To accommodate, with adequate margins, all the fuel removed from the reactor in accordance with the strategy for core management that is foreseen and the amount of fuel in the full reactor core;<br>j) To facilitate the removal of fuel from storage and its preparation for off-site transport. | **(4.14)**<br><br>**(4.24)**<br><br>**(5.29)**<br><br>**(5.132)** |
| 6.68A. The design shall include the following:<br><br>a) Means for monitoring and controlling the water temperature for operational states and for accident conditions that are of relevance for the spent fuel pool;<br>b) Means for monitoring and controlling the water level for operational states and for accident conditions that are of relevance for the spent fuel pool;<br>c) Means for monitoring and controlling the activity in water and in air for operational states and means for monitoring the activity in water and in air for accident conditions that are of relevance for the spent fuel pool;<br>d) Means for monitoring and controlling the water chemistry for operational states. | **(5.87)**<br><br>**(5.128)**<br><br>**(5.133)** |

# 10.LITERATURE


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[19] BN – JB – 2.8, DETERMINISTIC ANALYSES OF EVENTS UNDER SHUTDOWN CONDITIONS OF NUCLEAR INSTALLATIONS AND IN IRRADIATED NUCLEAR FUEL STORAGE POOLS (in preparation)

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