IAEA Safety Standards
for protecting people and the environment

Design of the Reactor Core for Nuclear Power Plants

Specific Safety Guide
No. SSG-52

IAEA
International Atomic Energy Agency

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

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DESIGN OF THE REACTOR CORE FOR NUCLEAR POWER PLANTS

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The Agency's Statute was approved on 23 October 1956 by the Conference on the Statute of the IAEA held at United Nations Headquarters, New York; it entered into force on 29 July 1957. The Headquarters of the Agency are situated in Vienna. Its principal objective is "to accelerate and enlarge the contribution of atomic energy to peace, health and prosperity throughout the world''.

IAEA SAFETY STANDARDS SERIES No. SSG-52

DESIGN OF THE REACTOR CORE FOR NUCLEAR POWER PLANTS

SPECIFIC SAFETY GUIDE

INTERNATIONAL ATOMIC ENERGY AGENCY
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FOREWORD

The IAEA's Statute authorizes the Agency to "establish or adopt… standards of safety for protection of health and minimization of danger to life and property" — standards that the IAEA must use in its own operations, and which States can apply by means of their regulatory provisions for nuclear and radiation safety. The IAEA does this in consultation with the competent organs of the United Nations and with the specialized agencies concerned. A comprehensive set of high quality standards under regular review is a key element of a stable and sustainable global safety regime, as is the IAEA's assistance in their application.

The IAEA commenced its safety standards programme in 1958. The emphasis placed on quality, fitness for purpose and continuous improvement has led to the widespread use of the IAEA standards throughout the world. The Safety Standards Series now includes unified Fundamental Safety Principles, which represent an international consensus on what must constitute a high level of protection and safety. With the strong support of the Commission on Safety Standards, the IAEA is working to promote the global acceptance and use of its standards.

Standards are only effective if they are properly applied in practice. The IAEA's safety services encompass design, siting and engineering safety, operational safety, radiation safety, safe transport of radioactive material and safe management of radioactive waste, as well as governmental organization, regulatory matters and safety culture in organizations. These safety services assist Member States in the application of the standards and enable valuable experience and insights to be shared.

Regulating safety is a national responsibility, and many States have decided to adopt the IAEA's standards for use in their national regulations. For parties to the various international safety conventions, IAEA standards provide a consistent, reliable means of ensuring the effective fulfilment of obligations under the conventions. The standards are also applied by regulatory bodies and operators around the world to enhance safety in nuclear power generation and in nuclear applications in medicine, industry, agriculture and research.

Safety is not an end in itself but a prerequisite for the purpose of the protection of people in all States and of the environment — now and in the future. The risks associated with ionizing radiation must be assessed and controlled without unduly limiting the contribution of nuclear energy to equitable and sustainable development. Governments, regulatory bodies and operators everywhere must ensure that nuclear material and radiation sources are used beneficially, safely and ethically. The IAEA safety standards are designed to facilitate this, and I encourage all Member States to make use of them.

THE IAEA SAFETY STANDARDS

BACKGROUND

Radioactivity is a natural phenomenon and natural sources of radiation are features of the environment. Radiation and radioactive substances have many beneficial applications, ranging from power generation to uses in medicine, industry and agriculture. The radiation risks to workers and the public and to the environment that may arise from these applications have to be assessed and, if necessary, controlled.

Activities such as the medical uses of radiation, the operation of nuclear installations, the production, transport and use of radioactive material, and the management of radioactive waste must therefore be subject to standards of safety.

Regulating safety is a national responsibility. However, radiation risks may transcend national borders, and international cooperation serves to promote and enhance safety globally by exchanging experience and by improving capabilities to control hazards, to prevent accidents, to respond to emergencies and to mitigate any harmful consequences.

States have an obligation of diligence and duty of care, and are expected to fulfil their national and international undertakings and obligations.

International safety standards provide support for States in meeting their obligations under general principles of international law, such as those relating to environmental protection. International safety standards also promote and assure confidence in safety and facilitate international commerce and trade.

A global nuclear safety regime is in place and is being continuously improved. IAEA safety standards, which support the implementation of binding international instruments and national safety infrastructures, are a cornerstone of this global regime. The IAEA safety standards constitute a useful tool for contracting parties to assess their performance under these international conventions.

THE IAEA SAFETY STANDARDS

The status of the IAEA safety standards derives from the IAEA's Statute, which authorizes the IAEA to establish or adopt, in consultation and, where appropriate, in collaboration with the competent organs of the United Nations and with the specialized agencies concerned, standards of safety for protection of health and minimization of danger to life and property, and to provide for their application.

With a view to ensuring the protection of people and the environment from harmful effects of ionizing radiation, the IAEA safety standards establish fundamental safety principles, requirements and measures to control the radiation exposure of people and the release of radioactive material to the environment, to restrict the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation, and to mitigate the consequences of such events if they were to occur. The standards apply to facilities and activities that give rise to radiation risks, including nuclear installations, the use of radiation and radioactive sources, the transport of radioactive material and the management of radioactive waste.

Safety measures and security measures[1] have in common the aim of protecting human life and health and the environment. Safety measures and security measures must be designed and implemented in an integrated manner so that security measures do not compromise safety and safety measures do not compromise security.

The IAEA safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. They are issued in the IAEA Safety Standards Series, which has three categories (see Fig. 1).

Safety Fundamentals

Safety Fundamentals present the fundamental safety objective and principles of protection and safety, and provide the basis for the safety requirements.

Safety Requirements

An integrated and consistent set of Safety Requirements establishes the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements are governed by the objective and principles of the Safety Fundamentals. If the requirements are not met, measures must be taken to reach or restore the required level of safety. The format and style of the requirements facilitate their use for the establishment, in a harmonized manner, of a national regulatory framework. Requirements, including numbered 'overarching' requirements, are expressed as 'shall' statements. Many requirements are not addressed to a specific party, the implication being that the appropriate parties are responsible for fulfilling them.

Safety Guides

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it

[1] See also publications issued in the IAEA Nuclear Security Series.

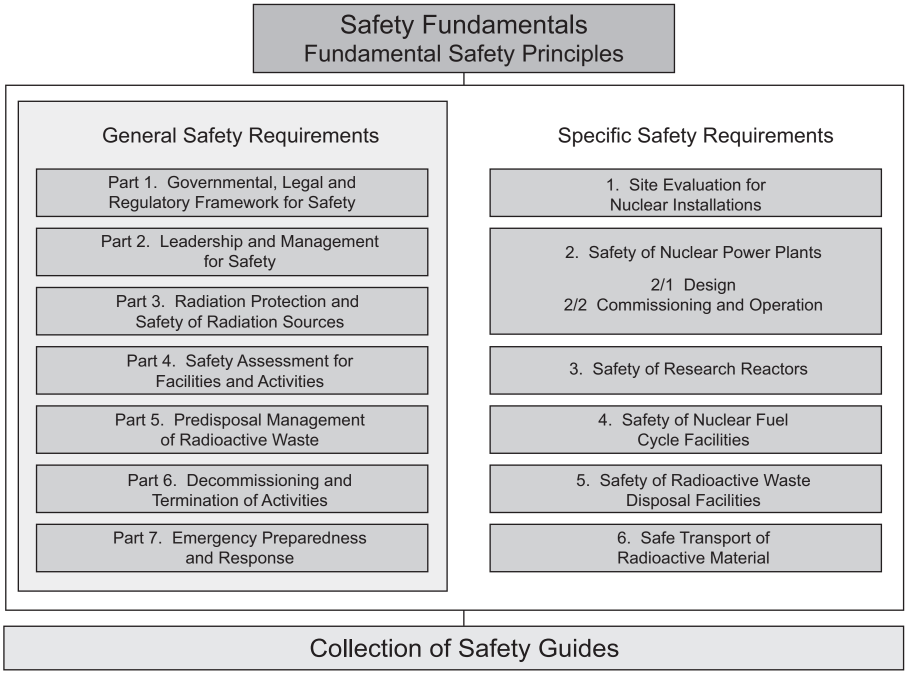

FIG. 1. The long term structure of the IAEA Safety Standards Series.

is necessary to take the measures recommended (or equivalent alternative measures). The Safety Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as ‘should’ statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

The principal users of safety standards in IAEA Member States are regulatory bodies and other relevant national authorities. The IAEA safety standards are also used by co-sponsoring organizations and by many organizations that design, construct and operate nuclear facilities, as well as organizations involved in the use of radiation and radioactive sources.

The IAEA safety standards are applicable, as relevant, throughout the entire lifetime of all facilities and activities — existing and new — utilized for peaceful purposes and to protective actions to reduce existing radiation risks. They can be used by States as a reference for their national regulations in respect of facilities and activities.

The IAEA's Statute makes the safety standards binding on the IAEA in relation to its own operations and also on States in relation to IAEA assisted operations.

The IAEA safety standards also form the basis for the IAEA's safety review services, and they are used by the IAEA in support of competence building, including the development of educational curricula and training courses.

International conventions contain requirements similar to those in the IAEA safety standards and make them binding on contracting parties. The IAEA safety standards, supplemented by international conventions, industry standards and detailed national requirements, establish a consistent basis for protecting people and the environment. There will also be some special aspects of safety that need to be assessed at the national level. For example, many of the IAEA safety standards, in particular those addressing aspects of safety in planning or design, are intended to apply primarily to new facilities and activities. The requirements established in the IAEA safety standards might not be fully met at some existing facilities that were built to earlier standards. The way in which IAEA safety standards are to be applied to such facilities is a decision for individual States.

The scientific considerations underlying the IAEA safety standards provide an objective basis for decisions concerning safety; however, decision makers must also make informed judgements and must determine how best to balance the benefits of an action or an activity against the associated radiation risks and any other detrimental impacts to which it gives rise.

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

The preparation and review of the safety standards involves the IAEA Secretariat and five safety standards committees, for emergency preparedness and response (EPReSC) (as of 2016), nuclear safety (NUSSC), radiation safety (RASSC), the safety of radioactive waste (WASSC) and the safe transport of radioactive material (TRANSSC), and a Commission on Safety Standards (CSS) which oversees the IAEA safety standards programme (see Fig. 2).

All IAEA Member States may nominate experts for the safety standards committees and may provide comments on draft standards. The membership of the Commission on Safety Standards is appointed by the Director General and includes senior governmental officials having responsibility for establishing national standards.

A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards.

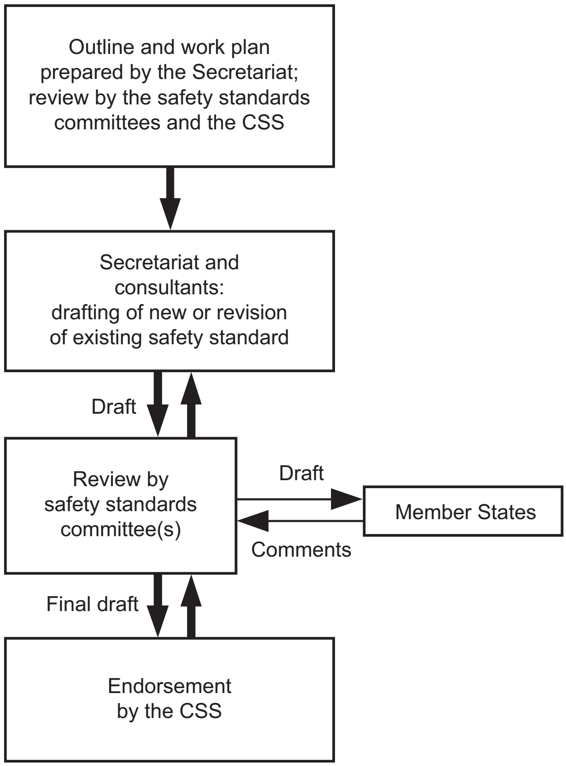

FIG. 2. The process for developing a new safety standard or revising an existing standard.

It articulates the mandate of the IAEA, the vision for the future application of the safety standards, policies and strategies, and corresponding functions and responsibilities.

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

The findings of the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR) and the recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP), are taken into account in developing the IAEA safety standards. Some safety standards are developed in cooperation with other bodies in the United Nations system or other specialized agencies, including the Food and Agriculture Organization of the United Nations, the United Nations Environment Programme, the International Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization.

INTERPRETATION OF THE TEXT

Safety related terms are to be understood as defined in the IAEA Safety Glossary (see http://www-ns.iaea.org/standards/safety-glossary.htm). Otherwise, words are used with the spellings and meanings assigned to them in the latest edition of The Concise Oxford Dictionary. For Safety Guides, the English version of the text is the authoritative version.

The background and context of each standard in the IAEA Safety Standards Series and its objective, scope and structure are explained in Section 1, Introduction, of each publication.

Material for which there is no appropriate place in the body text (e.g. material that is subsidiary to or separate from the body text, is included in support of statements in the body text, or describes methods of calculation, procedures or limits and conditions) may be presented in appendices or annexes.

An appendix, if included, is considered to form an integral part of the safety standard. Material in an appendix has the same status as the body text, and the IAEA assumes authorship of it. Annexes and footnotes to the main text, if included, are used to provide practical examples or additional information or explanation. Annexes and footnotes are not integral parts of the main text. Annex material published by the IAEA is not necessarily issued under its authorship; material under other authorship may be presented in annexes to the safety standards. Extraneous material presented in annexes is excerpted and adapted as necessary to be generally useful.

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide provides recommendations on the design of the reactor core to meet the requirements established in IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design [1]. This publication is a revision of IAEA Safety Standards Series No. NS-G-1.12[1], which it supersedes.

OBJECTIVE

1.2. The objective of this Safety Guide is to provide recommendations on meeting the safety requirements established in SSR-2/1 (Rev. 1) [1] for the design of the reactor core for nuclear power plants.

SCOPE

1.3. This Safety Guide is applicable primarily to land based stationary nuclear power plants with water cooled reactors for electricity generation or for other heat production (such as district heating or desalination). All recommendations are applicable to light water reactors (i.e. pressurized water reactors and boiling water reactors) and are generally applicable to pressurized heavy water reactors unless otherwise specified. This Safety Guide may also be applied, with judgement, to other reactor types (e.g. gas cooled reactors, floating reactors, small and modular reactors, innovative reactors) to contribute to the interpretation of the requirements that have to be considered in developing the design of the reactor core.

1.4. The reactor core is the central part of a nuclear reactor where nuclear fission occurs. The reactor core consists of four basic systems and components (i.e. the fuel (including fuel rods and the fuel assembly structure), the coolant, the moderator and the control rods), as well as additional structures (e.g. reactor pressure vessel internals, core support plates, and the lower and upper internal structure in light water reactors). This Safety Guide addresses the safety aspects

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[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of the Reactor Core for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.12, IAEA, Vienna (2005).

of the core design and includes neutronic aspects, thermohydraulic aspects, thermomechanical aspects, structural mechanical aspects, aspects relating to reactor core control, shutdown and monitoring, and core management aspects for the safe design of the reactor core for nuclear power plants. Specifically, the following structures, systems and components (SSCs) are covered:

(a) Fuel rods, containing fuel pellets with or without burnable absorbers in cladding tubes, which generate and transfer heat to the coolant.
(b) Fuel assemblies, comprising bundles of fuel rods, along with structures and components (e.g. guide tubes, spacer grids, bottom and top nozzles, fuel channels) that maintain the fuel rods and fuel assemblies in a predetermined geometrical configuration.
(c) The reactor core control system, the shutdown system and the monitoring system, including components and equipment used for reactivity control and shutdown, comprising neutron absorbers (solid or liquid), the associated structure and the drive mechanism.
(d) Support structures that provide the foundation for the core within the reactor vessel (within the calandria for pressurized heavy water reactors), the structure for guiding the flow (for pressurized water reactors) and the guide tubes for reactivity control devices (for pressurized heavy water reactors).
(e) The coolant.
(f) The moderator.
(g) Other core components such as steam separators (for boiling water reactors) and neutron sources. These are considered only to a limited extent in this Safety Guide.

1.5. This Safety Guide is intended mainly for NPPs that use natural and enriched UO_2 fuels and plutonium-blended UO_2 fuel (mixed oxide fuel) with zirconium based alloy cladding. Unless otherwise specified, all recommendations apply to these fuel types.

1.6. For innovative fuel materials, such as uranium nitride fuel or inert matrix fuel, or cladding materials other than zirconium based alloys, this Safety Guide can be applied with judgement.

1.7. The design of the reactor core may interface with the design of other reactor systems and other related aspects. In this Safety Guide, recommendations on these interfacing systems and aspects are provided mainly to identify their functional interface. The relevant Safety Guides are referenced, as appropriate, in order to clarify the interfaces.

1.8. The terms used in this Safety Guide are to be understood as defined in the IAEA Safety Glossary [2]. Explanations of additional technical terminology are provided in Annex I.

STRUCTURE

1.9. Section 2 describes general considerations for safe core design based on requirements for the management of safety, principal technical requirements and general design requirements established in sections 3, 4 and 5 of SSR-2/1 (Rev. 1) [1], respectively. Section 3 describes specific considerations for the safe design of fuel rods, fuel assemblies, core structures and core components, and the core control system and the reactor shutdown system based on specific design requirements (i.e. Requirements 43–46) of SSR-2/1 (Rev. 1) [1]. Section 4 provides recommendations on the qualification and testing of the SSCs of the reactor core.

1.10. Annex I provides supplementary technical information for clarification of the terminology used in this Safety Guide, additional background information and examples supporting specified design recommendations. Annex II describes important items that need to be addressed within the design of the fuel rod, fuel assembly, reactivity control assembly, neutron source assembly and hydraulic plug assembly.

2. GENERAL SAFETY CONSIDERATIONS IN THE DESIGN OF THE REACTOR CORE

MANAGEMENT SYSTEM

2.1. The design of the reactor core should take into account the recommendations of IAEA Safety Standards Series Nos GS-G-3.1, Application of the Management System for Facilities and Activities [3], and GS-G-3.5, The Management System for Nuclear Installations [4] to meet Requirements 1–3 of SSR-2/1 (Rev. 1) [1], and the requirements of IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [5].

DESIGN OBJECTIVES

Fundamental safety functions

2.2. The three fundamental safety functions, described in Requirement 4 of SSR-2/1 (Rev. 1) [1], are required to be ensured in the design of the reactor core for operational states and for a wide range of accident conditions. The fundamental safety functions as they apply specifically to the design of the reactor core are as follows:

(a) Control of reactivity;
(b) Removal of heat from the reactor core;
(c) Confinement of radioactive material.

Adequate design based on the concept of defence in depth

2.3. Adequate design (i.e. capable, reliable and robust design) of the reactor core, based on the concept of defence in depth, will enable achievement of the fundamental safety functions, together with provision for associated reactor safety features.

2.4. Physical barriers considered as part of, or affecting the design of, the reactor core include the fuel matrix, the fuel cladding and the boundary of the reactor coolant system. For normal operation and anticipated operational occurrences, fuel rods are required to be designed such that their structural integrity and a leaktight barrier are maintained to prevent the transport of fission products into the coolant (see Requirement 43 of SSR-2/1 (Rev. 1) [1]).

2.5. For design basis accidents, fuel cladding failures should be kept to a minimum. Components of the reactor core and its associated structures should be designed with account taken of the safety functions to be achieved. In addition, the reactor core is required to be designed to maintain a configuration such that it can be shut down and remains coolable for design basis accidents and design extension conditions without significant fuel degradation (see Requirement 44 of SSR-2/1 (Rev. 1) [1]).

Proven engineering practices

2.6. The reactor core should be of a design that has been proven either in equivalent applications, by means of operating experience or the results of relevant research programmes, or, as appropriate, in accordance with the design,

design verification and validation processes stated in applicable codes and standards (in accordance with paras 4.14 and 4.16 of SSR-2/1 (Rev. 1) [1]).

Safety assessment in the design process

2.7. Paragraph. 4.17 of SSR-2/1 (Rev. 1) [1] states (footnote omitted):

> "The safety assessments shall be commenced at an early point in the design process, with iterations between design activities and confirmatory analytical activities, and shall increase in scope and level of detail as the design programme progresses."

Recommendations on safety assessment methods are provided in IAEA Safety Standards Series No. SSG-2 (Rev. 1), Deterministic Safety Analysis for Nuclear Power Plants [6].

Features to facilitate radioactive waste management

2.8. The design of fuel rods and fuel assemblies should provide features that will facilitate future waste management (including reprocessing when applicable). The physical condition of discharged fuel assemblies from the reactor core will influence the design of the storage and disposal systems for the used fuel. Recommendations on taking into account the impact of the condition of used fuel on the design of spent fuel handling and storage systems are provided in IAEA Safety Standards Series Nos SSG-63, Design of Fuel Handling and Storage Systems for Nuclear Power Plants [7], and SSG-15 (Rev. 1), Storage of Spent Nuclear Fuel [8].

DESIGN BASIS FOR STRUCTURES, SYSTEMS AND COMPONENTS OF THE REACTOR CORE

2.9. In accordance with Requirement 14 of SSR-2/1 (Rev. 1) [1], the design basis for the reactor core is required to specify the necessary capability, reliability and functionality for all applicable plant states (see para. 2.10) in order to meet the specific acceptance criteria.

Plant states and postulated initiating events

2.10. As stated in Requirement 13 of SSR-2/1 (Rev. 1) [1], plant states are required to be identified and grouped into categories. The plant states typically considered for the design of the reactor core are normal operation, anticipated operational occurrences, design basis accidents and design extension conditions without significant fuel degradation. These four states are referred to as ‘all applicable plant states’ throughout this Safety Guide. Accidents with significant core melting are outside the scope of the design of the reactor core.

2.11. The design process should include an analysis of the effects of postulated initiating events on the reactor core for all applicable plant states. Recommendations on the identification of the postulated initiating events for all applicable plant states and relevant safety analyses are provided in SSG-2 (Rev. 1) [6].

External hazards

2.12. The consequences of earthquakes should be taken into account in the design of the reactor core. Seismic categorization of the SSCs of the reactor core should be determined in accordance with IAEA Safety Standards Series No. NS-G-1.6, Seismic Design and Qualification for Nuclear Power Plants [9].

Design limits

2.13. Design limits on relevant physical parameters for individual SSCs of the reactor core are required to be specified for all applicable plant states, in accordance with Requirement 15 of SSR-2/1 (Rev. 1) [1]. Adherence to these limits with appropriate provisions will ensure that the concept of defence in depth, as stated in paras 2.4 and 2.5, is successfully applied with adequate margins. Typical examples of relevant parameters with quantitative or qualitative limits are provided in paras 3.33 and 3.65–3.76.

Safety classification aspects of the reactor core

2.14. The SSCs of the reactor core are required to be classified on the basis of their function and their safety significance (see Requirement 22 of SSR-2/1 (Rev. 1) [1]). The safety classification process is described in IAEA Safety Standards Series No. SSG-30, Safety Classification of Structures, Systems and Components in Nuclear Power Plants [10].

2.15. Fuel rods and fuel assemblies should be classified in safety class 1, the highest safety class, since they are essential for achieving the three fundamental safety functions in para. 2.2.

2.16. The failure of control rods has the potential to endanger the control of reactivity in the core and the integrity of the fuel rods, which are safety class 1 barriers; from this perspective, control rods should be classified in safety class 1.

2.17. For all safety classes identified in accordance with the method described in SSG-30 [10], corresponding engineering design rules should be specified and applied.

Engineering design rules

2.18. The engineering design rules for the SSCs of the reactor core represent methods to achieve the adequacy of the design and should include the following, as appropriate:

(a) The use of applicable codes and standards, and proven engineering practices;
(b) Conservative safety assessment;
(c) Specific design analyses for reliability;
(d) Qualification and testing;
(e) Operational limits and conditions.

Design for reliability

2.19. In accordance with para. 5.37 of SSR-2/1 (Rev. 1) [1], fuel rods and assemblies, and components and systems for reactor control and shutdown are required to be designed with high reliability, in consideration of their safety significance. Provisions for achieving high reliability in these designs are set out in paras 3.39 and 3.112 of this Safety Guide, respectively.

Operational limits and conditions

2.20. In accordance with Requirement 28 of SSR-2/1 (Rev. 1) [1], operational limits and conditions are required to be established in order to ensure that the reactor core operates safely in accordance with design assumptions and intent. Relevant guidance on the operational limits and conditions is provided in IAEA Safety Standards Series No. NS-G-2.2, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants [11].

DESIGN FOR SAFE OPERATION

2.21. The SSCs of the reactor core should be designed such that their required testing, inspection, repair, replacement, calibration or maintenance is facilitated.

2.22. The design of the reactor core should be reviewed and modified when a significant configuration change occurs during the operating lifetime of the plant, as a result of, for example:

(a) Major modifications to the plant design or to equipment, or operational modifications, such as the following:
   (i) Replacement of the steam generator (not for boiling water reactors);
   (ii) An increase in the rated power of the plant;
   (iii) A significant change in the operating domain.

(b) A new fuel type or a significant change in fuel type (e.g. the introduction of mixed oxide or gadolinium fuel, new design of the fuel rods or the fuel assembly with modified geometrical or thermohydraulic characteristics).

(c) An increase of the fuel discharge burnup beyond the design limit.

(d) Major fuel management changes such as a large extension to the length of the reloading cycle.

2.23. Fuel rods and fuel assemblies should be designed to prevent the potential for fuel failures due to specific operational conditions (e.g. startup rates, degraded coolant chemistry conditions or the presence of foreign materials) during operational states.

REACTOR CORE SAFETY ANALYSIS

2.24. In accordance with Requirement 42 of SSR-2/1 (Rev. 1) [1], safety analysis is required to be conducted to evaluate and assess challenges to safety in all applicable plant states using deterministic approaches and including uncertainties to the extent possible.

2.25. The following major factors should be taken into account in the safety analysis for the reactor core:

(a) Initial operating conditions (e.g. global and local thermohydraulic conditions, power levels, power distributions and time in the reloading cycle);

(b) Reactivity feedback;

(c) Rate of change of the concentration of soluble absorber in the moderator and the coolant;
(d) Position or rate of insertion of positive (or negative) reactivity regulated by the reactivity control device(s), or caused by changes in process parameters;
(e) Rate of insertion of negative reactivity associated with a reactor trip;
(f) The response of individual channels to transients in relation to the average thermal power of the core (for boiling water reactors);
(g) 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);
(h) The decay of xenon and other neutron absorbers in the analysis of the long term behaviour of the core;
(i) The activity inventory of the core.

Appropriate provisions or margins should be included in the above factors such that the safety analysis remains valid for specific loading patterns or fuel designs. Recommendations on methods of safety analysis are provided in SSG-2 (Rev. 1) [6].

2.26. Safety analysis for the reactor core should be performed to verify that fuel design limits are not exceeded in all applicable plant states. For accident conditions, the effect of fuel behaviour on core cooling should be included in the safety analysis (e.g. ballooning and rupture of the cladding, exothermic metal–water reactions, distortions of fuel rods and fuel assemblies). The effects of hydrogen accumulation (as a result of a metal–water reaction between the zirconium based alloy cladding and water at high temperature) on the boundary of the reactor coolant system should be evaluated.

2.27. Systematic, complete, qualified and up to date documentation of the state of the SSCs of the plant and the reactor core should be maintained to ensure that the safety analysis is performed using the actual plant and core configuration.

3. SPECIFIC SAFETY CONSIDERATIONS IN THE DESIGN OF THE REACTOR CORE

GENERAL

3.1. This section addresses specific design aspects for the SSCs of the reactor core for meeting Requirements 43–46 established in SSR-2/1 (Rev. 1) [1]. It also addresses the interface with core management, which strongly influences the core design with regard to the performance of fuel rods and fuel assemblies. Specific guidance is provided in IAEA Safety Standards Series No. NS-G-2.5, Core Management and Fuel Handling for Nuclear Power Plants [12].

3.2. The design of the reactor core, in combination with the design of reactor cooling systems, and the reactor control and reactor protection systems, should enable the fulfilment, at all times, of the fundamental safety functions (para. 2.2) for all applicable plant states (i.e. normal operation, anticipated operational occurrences, design basis accidents and design extension conditions without significant fuel degradation).

3.3. The reactor core and associated control and protection systems should be designed with adequate margins to ensure that fuel design limits are not exceeded for all applicable plant states. Fuel design limits are described in paras 3.65–3.76.

Fuel type

3.4. Fuel rods contain fissile materials (e.g. ^{235}U, ^{239}Pu) that are highly reactive with thermal neutrons. In selecting the fuel pellet materials, the following properties should be optimized (examples of pellet materials are provided in Annex I):

(a) Reactivity with thermal neutrons;
(b) Impurities with low thermal neutron absorption properties;
(c) Thermal performance (e.g. high thermal conductivity is desirable for operational states while high thermal diffusivity is desirable for accident conditions);
(d) Dimensional stability;
(e) Fission gas retention;
(f) Resistance to pellet–cladding interaction.

3.5. Cladding materials should be selected with consideration of the following properties (examples of cladding materials are provided in Annex I):

(a) Low absorption cross-section for thermal neutrons;
(b) High resistance to irradiation conditions;
(c) High thermal conductivity and high melting point;
(d) High corrosion resistance and low hydrogen pick-up;
(e) Low oxidation and low hydriding in high temperature conditions;
(f) Adequate resistance to breakaway oxidation at high integrated-time temperature conditions;
(g) Adequate mechanical properties (e.g. high strength, high ductility, low creep rate in normal operation, high relaxation rate in transients);
(h) Low susceptibility to stress corrosion cracking;
(i) Adequate resistance to hydrogen assisted cracking and hydride related cracking in normal operation and for fuel storage.

Coolant

3.6. In light water reactors, the coolant also acts as the moderator. The choice of coolant should take into account interactions between the coolant and fuel and core components in all chemical conditions (see Annex I for supplementary information). For pressurized heavy water reactors, the coolant and the moderator are separated; typically, chemicals are not added to the coolant for controlling reactivity.

3.7. The coolant should be physically and chemically stable with respect both to high temperatures and to irradiation in order to fulfil its primary function, namely the continuous removal of heat from the core.

3.8. The reactor core should be designed to prevent or control flow instabilities and the resultant fluctuations in core reactivity or power.

3.9. The reactor fuel and core design should include the following safety considerations associated with the coolant:

(a) Ensuring that the coolant system is free of foreign materials prior to the initial startup of the reactor and following refuelling and maintenance outages, for the operating lifetime of the plant;
(b) Maintaining the radionuclide activity in the coolant as low as reasonably achievable by means of purification systems, corrosion product minimization or removal of defective fuel as appropriate;

(c) Monitoring and controlling the effects that the coolant and coolant additives have on reactivity in all plant states;
(d) Determining and controlling the physical and chemical properties of the coolant in the core;
(e) Ensuring that the chemical composition of the coolant is compatible with the materials that are present in the primary circuit (e.g. to avoid crud formation on fuel rods, and to minimize corrosion and the generation of radioactive products).

3.10. The design should take into account the effect of changes in coolant density (including fluid phase changes) on core reactivity and core power, both locally and globally.

Moderator

3.11. The choice of moderator and of the spacing of the fuel rods and fuel assemblies within it should meet engineering and safety requirements with respect to reactivity feedback due to changes in moderator temperature, density or void fraction, while also optimizing the neutron economy and, hence, fuel consumption. The prevalent thermal reactor types use either light water or heavy water as the moderating medium.

3.12. Depending on the reactor design, the moderator could contain a soluble neutron absorber, such as boron in pressurized water reactors, to maintain adequate shutdown margins in operational states and, by means of controlled dilution, to compensate the decrease in core reactivity throughout the whole reloading cycle.

3.13. For pressurized heavy water reactors, the reactor core design should ensure the effectiveness of the shutdown system of the reactor in an accident involving dilution of the absorber. Means should be provided to prevent the inadvertent removal of such absorber material (e.g. due to chemistry transients) and to ensure that its removal is controlled and slow.

3.14. For pressurized heavy water reactors, the moderator should provide the capability to remove decay heat without loss of core geometry in accident conditions.

3.15. For pressurized heavy water reactors, measures should be provided to prevent deflagration or explosion of hydrogen generated by radiolysis in the moderator.

NEUTRONIC DESIGN

Design considerations

3.16. The design of the reactor core should ensure that the feedback characteristics of the core rapidly compensate for an increase in reactivity. The reactor power should be controlled by a combination of the inherent neutronic characteristics of the reactor core (see Annex I for supplementary information) and its thermohydraulic characteristics, and the capability of the control system and the shutdown system to actuate in all applicable plant states.

3.17. The design should ensure that power changes that could result in conditions exceeding fuel design limits for normal operation and anticipated operational occurrences will be reliably and readily detected and suppressed.

Nuclear design limits

Nuclear key safety parameters

3.18. Nuclear key safety parameters influencing the neutronic design of the core and fuel management strategies should be established from the safety analyses that verify compliance with the specific fuel design limits described in paras 3.65–3.76. Appropriate provision should also be made for the nuclear key safety parameters, such that they will remain valid for specific core reload designs and throughout the reloading cycle. Typical nuclear key safety parameters include the following:

(a) The temperature coefficients of reactivity for the fuel and the moderator;
(b) The boron reactivity coefficient and concentration (for pressurized water reactors);
(c) The shutdown margin;
(d) The maximum reactivity insertion rate;
(e) The control rod worth and control bank worth;
(f) The radial and axial power peaking factors, including allowance for xenon induced oscillation;
(g) The maximum linear heat generation rate;
(h) The void coefficient of reactivity.

3.19. The safety impacts of any major modifications (see para. 2.22) to the reactor core design should be assessed using the nuclear key safety parameters in order to ensure that the specified fuel design limits are not violated. Otherwise, new nuclear key safety parameters should be defined and justified.

Core reactivity characteristics

3.20. On the basis of the geometry and the fuel composition of the reactor core, the design should include evaluations of the core to determine steady state spatial distributions of neutron flux and of the power, core neutronic characteristics and the efficiency of the means of reactivity control for normal operation of the plant at power, in shutdown conditions and in accident conditions.

3.21. Nuclear key safety parameters, such as reactivity coefficients, should be evaluated for selected core operating conditions (e.g. zero power, full power, beginning of cycle, end of cycle and at key points relating to poison burnout) and for the corresponding fuel management strategy. The dependence of such nuclear key safety parameters on the core loading and on the burnup of the fuel should be analysed. Appropriate margins should be included in the reactivity coefficients or within the modelling approaches used to evaluate reactivity feedback in the safety analysis for all applicable plant states.

Maximum reactivity worth and reactivity insertion rate

3.22. The maximum reactivity worth of the reactivity control devices (e.g. control rods and/or chemical and volume control systems) should be limited, or interlock systems should be provided, so that any resultant power variations do not exceed specified limits for relevant reactivity insertion transients and accidents, such as the following:

(a) Control rod ejection;
(b) Control rod drop;
(c) Boron dilution;
(d) Uncontrolled withdrawal of control banks.

Such reactivity limits should be determined via safety analyses to ensure that the fuel design limits described in paras 3.65–3.76 are not exceeded. These analyses should be performed for all fuel types in the core (e.g. UO_2 or mixed oxide fuel) or a representative core with appropriate margins, and for all allowable operating conditions and fuel burnup values.

Control of global and local power

3.23. The design should ensure that the core power can be controlled globally and locally using the means of reactivity control (see Annex I for supplementary information) in such a way that the peak linear heat generation rate of each fuel rod does not exceed the specified limits anywhere in the core. Variations in the power distribution (e.g. caused by effects such as xenon instability) or other local effects (e.g. in a mixed core, crud induced power shifts or axial offset anomalies for pressurized water reactors, fuel assembly bowing or distortion) should be addressed in the design of the control system. Provisions should be included to take into account measurement variations between flux detectors (e.g. due to operability, location, shadowing or ageing).

Shutdown margin

3.24. The insertion of control rods should provide an adequate shutdown margin in all applicable plant states (see Annex I for supplementary information). The specification and monitoring of control rod insertion limits as a function of power level should ensure an adequate shutdown margin at all times to ensure satisfactory tolerance to faults.

3.25. The effects of depletion of burnable absorber on the core reactivity should be evaluated to ensure an adequate shutdown margin in all resulting applicable core conditions throughout the operating cycle. (Examples of the use of burnable absorbers in pressurized water reactors are provided in Annex I.)

THERMOHYDRAULIC DESIGN

Design considerations

3.26. The thermohydraulic design of the reactor core should include adequate margins and provisions to ensure the following:

(a) Specified thermohydraulic design limits are not exceeded in operational states (i.e. in normal operation and anticipated operational occurrences);
(b) The failure rates of fuel rods in design basis accidents and design extension conditions without significant fuel degradation remain within acceptance levels;
(c) Minimum and maximum values of core flow rate are consistent with thermohydraulic design limits and mechanical design limits.

Thermohydraulic design limits

3.27. Specific thermohydraulic design limits should be established with adequate margins for predictable parameters, such as the maximum linear heat generation rate, the minimum critical power ratio (for boiling water reactors), the minimum departure from nucleate boiling ratio (for pressurized water reactors) or the dryout power ratio (for pressurized heavy water reactors), the peak fuel temperature or enthalpy, and the peak cladding temperature. Uncertainties in the values of process parameters (e.g. reactor power, coolant flow rate, core bypass flow, inlet temperature and pressure, and power peaking factors), core design parameters and calculation methods used in the assessment of the thermal margins should be addressed in the design analyses.

3.28. The thermohydraulic design should include design analyses that take into account design features of the fuel assembly, including the fuel rod spacing, the fuel rod power, the sizes and shapes of subchannels, spacer and mixing grids (for light water reactors), and flow deflectors (for light water reactors) or turbulence promoters. In addition, for fuel channel type pressurized heavy water reactors, the effects of fuel bundle string, appendages, gaps between fuel rods and the pressure tube, anticipated change in shape of the pressure tube with reactor ageing, and junctions between neighbouring end-plates should be addressed in the design analyses.

3.29. For light water reactors, the thermohydraulic design should also consider core inlet and outlet coolant temperatures and flow distributions. These effects should also be considered in the core monitoring and protection systems.

3.30. The design should ensure that the minimum ratio of operating power to critical power (i.e. a minimum critical heat flux ratio, a minimum departure from the nucleate boiling ratio, a minimum critical channel power ratio or a minimum critical power ratio) takes into account that critical heat flux correlations have been developed from representative tests performed at steady state conditions. As a consequence, adequate margins or provisions should be added to the minimum ratio to take into account additional factors not considered in the correlation itself, such as the following:

(a) The thermohydraulic response to anticipated operational occurrences;
(b) Impacts resulting from the chosen loading pattern;
(c) Impacts resulting from the potential presence of crud in the core.

In addition, uncertainties, such as plant operational uncertainties and code uncertainties, should be adequately taken into account in the safety analysis.

3.31. Critical heat flux limits should be applied in the safety analysis to ensure that the potential for cladding failure is avoided. In some reactor designs, critical heat flux conditions during transients can be tolerated if it can be shown, using suitable analytical methods, that the cladding temperatures will not exceed the fuel failure limits.

3.32. Experiments should be conducted on representative fuel assembly designs over the range of expected operational states, including various axial heat flux profiles, to identify the limiting values of the minimum ratios. Correlations for predicting critical heat flux are continually being generated as a result of additional experimental data, changes in fuel assembly design and improved calculation techniques involving coolant mixing and the effect of axial power distributions. The impact of any change in an established correlation used in thermohydraulic design should be evaluated. For fast transients (e.g. rod ejection accidents), the correlations used may be reassessed as steady state conditions may not be sufficiently representative.

3.33. Approaches, such as those in the following examples, should be taken to demonstrate the fulfilment of the recommendations in paras 3.27–3.32:

(a) For pressurized water reactors, the limiting (minimum) value of departure from nucleate boiling ratio should be established such that the hot rod in the core does not experience any heat transfer deterioration during normal operation or anticipated operational occurrences with a 95% probability at the 95% confidence level.
(b) For boiling water reactors and for some pressurized water reactors that do not comply with the recommendation in para. 3.33(a), the limiting (minimum) value of critical power ratio, the critical heat flux ratio or the departure from the nucleate boiling ratio should be established such that the number of fuel rods that experience heat transfer deterioration does not exceed a very small fraction (e.g. at most, 0.1%) of the total number of fuel rods in the core.
(c) For pressurized heavy water reactors, if the maximum fuel cladding temperature remains below a certain limit (e.g. 600ºC) and the duration of post-dryout operation is limited (e.g. less than 60 s), it is considered that the fuel deformation is small, so that fuel rods are not in contact with the pressure tube and will not cause a failure of the pressure tube.

THERMOMECHANICAL DESIGN OF FUEL RODS AND FUEL ASSEMBLIES

Design considerations

3.34. The design should ensure that the structural integrity of fuel assemblies (i.e. their geometry) and of fuel rods (i.e. their leaktightness) is maintained for normal operation and anticipated operational occurrences. For accident conditions (design basis accidents and design extension conditions without significant fuel degradation), the design should ensure no fuel rod failures where this is reasonably practicable; otherwise, only a limited number of fuel rod failures should be allowed. The allowable number of failed fuel rods may depend on the frequency and nature of the event. A coolable geometry of the core should be ensured by design for design basis accidents and design extension conditions without significant fuel degradation. In accident conditions, the level of radionuclide activity should be assessed to confirm that the dose limits for workers are not exceeded.

3.35. For accident conditions involving the ballooning and rupture of the cladding, the dispersal of fuel fragments in the coolant should be prevented.

3.36. The design of fuel rods (with or without burnable absorbers) and fuel assemblies should take into account the irradiation conditions and the environmental conditions (e.g. temperature; pressure; coolant chemistry; irradiation effects on fuel, cladding and fuel assemblies; static and dynamic mechanical loads, including flow induced vibration; and changes in the chemical characteristics of the constituent materials).

3.37. Fuel rods and fuel assemblies should be designed to withstand handling loads during transport, storage, installation and refuelling operations.

3.38. Annex II describes important aspects that are typically taken into account in the design of fuel rods and fuel assemblies, and of reactivity control assemblies, neutron source assemblies and hydraulic plug assemblies, including the irradiation and environmental conditions.

3.39. The design should ensure that fuel rods and fuel assemblies are reliable throughout their lifetime, including during manufacturing, transport, handling, in-core operation, storage and disposal, where applicable. Key contributors to fuel reliability should be addressed; important key contributors include the following:

(a) Oversight of fuel fabrication;
(b) Debris mitigation (exclusion of foreign materials);
(c) Control of in-reactor power changes to limit excessive pellet–cladding interaction;
(d) Control of crud and corrosion;
(e) Prevention of grid-to-rod fretting (for light water reactors);
(f) Fuel surveillance and inspection practices.

Thermal and burnup effects on fuel rods

3.40. In operational states, the design should ensure that the peak fuel temperature is lower than the fuel melting temperature by an adequate margin to prevent melting of the fuel, when appropriate provisions and uncertainties are considered. For design basis accidents (e.g. reactivity initiated accidents) and for design extension conditions without significant fuel degradation, incipient fuel melting can be allowed (e.g. fuel centreline melting limited to a small fraction of fuel pellet volume). The design and safety assessments should take into account the effects of fuel burnup on the fuel rod and fuel assembly properties (see Annex I for supplementary information).

3.41. Straining of the cladding is caused by overpressure of internal gases in the fuel rod or by gaseous swelling of the fuel or thermal expansion of the fuel as a consequence of fuel burnup or local power increases. The design should ensure that cladding stresses and strains are limited. Limits for cladding stress, accumulated cladding strain, and cladding corrosion and hydriding should be specified for all applicable plant states and should be applied throughout the reloading cycle.

3.42. For accident conditions, cladding deformation should be evaluated to determine the potential for cladding failure (e.g. burst or rupture) and any resulting release of fission products from the fuel.

Effects of irradiation on fuel assembly structures

3.43. The design should ensure that the dimensional changes of light water reactor fuel assembly structures are minimized so that contacts or interactions between the fuel rods and fuel assembly components (top and bottom nozzles of the fuel assembly) are precluded, and that bowing of the fuel rods and fuel assemblies, as well as swelling of the control rods and any potential interaction with the fuel assembly guide tubes, will not affect the structural integrity or the thermohydraulic performance of the fuel assemblies or the safety functions of the control rods.

3.44. Relaxation of grid springs under irradiation should be assessed to limit the potential for grid-to-rod fretting (for light water reactors). In the dimensional stability analyses for fuel assembly components and control devices, the effects of irradiation, and in particular the effects of fast neutrons, on mechanical properties, such as tensile strength, ductility, growth, creep or relaxation, should be taken into account. The effect of irradiation on buckling resistance of the spacer grids should be considered when assessing seismic events or loss of coolant accidents.

3.45. For pressurized heavy water reactors, the design should ensure that the length of the cavity in the fuel channel is sufficient to accommodate the irradiation and thermal effects on the fuel bundle string in the fuel channel for all applicable plant states.

Effects of variations in power levels

3.46. For operational states, fuel rods should be designed to withstand thermomechanical loads during local and global power transients (e.g. loads due to fuel assembly shuffling, movements of control devices, load following, flexible operation or other causes of reactivity changes).

Mechanical effects in fuel rods

3.47. The design should include analyses to ensure that straining of the fuel cladding due to mechanical loads (e.g. coolant pressure, seismic loads) meets fuel design limits. The analyses should take into account radial gap closure kinetics, which depend on various parameters such as fuel densification, fuel swelling, fuel pellet cracking, fragmentation and its radial relocation within the fuel rod after a power change, cladding creep behaviour at low stress, initial internal pressure of the fuel rods, release of fission gases to the free volumes, and operating parameters including power history and coolant pressure.

3.48. Stress corrosion cracking induced by pellet–cladding interaction in the presence of corrosive fission products should be prevented (see Annex I for supplementary information).

3.49. Stress concentration in the cladding, caused by missing pellets, axial gaps between fuel pellets, missing pellet surfaces or fuel pellet chips trapped in the gap, cannot be explicitly considered in the fuel rod design and, hence, those anomalies should be avoided to the extent possible.

Effects of burnable absorber in the fuel

3.50. The design should include analyses to demonstrate that the fuel rod can accommodate the effects of any in-fuel burnable absorbers on the thermal, mechanical, chemical and microstructural properties of the fuel pellets, and on the behaviour of the fuel rods.

Corrosion and hydriding

3.51. Hydrogen pick-up correlations should be determined as a function of the corrosion of the cladding in normal operation for each cladding type, so that appropriate fuel design limits, such as for reactivity initiated accidents and loss of coolant accidents, can be expressed as a function of the pre-transient hydrogen content of the cladding (see Annex I for supplementary information).

3.52. Fuel rods and fuel assemblies should be designed to be compatible with the coolant environment in operational states, including shutdown and refuelling (see Annex I for supplementary information).

3.53. For pressurized heavy water reactors, the initial hydrogen content in the fuel rods should be limited to reduce the likelihood of fuel defects being caused by hydrogen induced embrittlement of the cladding.

Crud

3.54. The design analyses should take into account the degradation of the heat transfer from fuel rods due to the formation of deposits on the surface of the cladding via corrosion products coming from the reactor coolant system or other chemical changes. For pressurized water reactors, in the event that boron is trapped in the crud layer, its potential impact on the neutronic performance of the core should be assessed and addressed in the core design analyses.

Hydraulic effects in fuel assemblies

3.55. Hydraulic effects should be addressed primarily in the thermohydraulic design of the fuel assembly and in the evaluation of aspects such as localized corrosion, erosion, flow induced vibration, grid-to-rod fretting, fuel assembly lift-off and fuel assembly distortion. Hydraulic effects on the fuel assembly design should be characterized by means of fuel assembly endurance tests performed in qualified out-of-reactor loops using full scale fuel assembly mock-ups with prototypical test conditions (e.g. pressure, temperature, cross-flows and end-of-life grid spring relaxation).

Considerations of mechanical safety in the design

3.56. The fuel assembly should be designed to withstand mechanical stresses as a result of the following:

(a) Fuel handling and loading;
(b) Power variations;
(c) Hold-down loads for pressurized water reactors (which should balance the hydrodynamic lift-off forces and the geometrical changes of the core cavity and of the fuel assemblies under irradiation);
(d) Temperature gradients;
(e) Hydraulic forces, including cross-flows between distorted fuel assemblies or in mixed fuel core configurations (i.e. cores with different types of fuel);
(f) Irradiation effects (e.g. irradiation induced growth and swelling);
(g) Vibration and fretting wear of fuel rods (grid-to-rod fretting for light water reactors, wear between spacers for pressurized heavy water reactors) induced by coolant flow;
(h) Creep deformation of the fuel assembly structure (which could lead to distortion of fuel assemblies);
(i) Seismic loading at the level of the safety shutdown earthquake, typically combined with the loading due to loss of coolant accidents;
(j) Postulated initiating events (i.e. anticipated operational occurrences and design basis accidents) and design extension conditions without significant fuel degradation.

3.57. For all applicable plant states, the following mechanical safety aspects should be addressed in the design of fuel rods and fuel assemblies:

(a) The clearance within and adjacent to the fuel assembly should provide space to allow for irradiation induced growth and bowing (for light water reactors) and bulging of the fuel channel (for boiling water reactors; see Annex I for supplementary information).

(b) Bowing of fuel rods or distortion of assemblies should be limited, so that thermohydraulic behaviour, power distribution, fuel performance and fuel handling are not adversely affected.
(c) Fatigue should not cause the failure of any component of the fuel assembly.
(d) Fuel assembly distortion as a result of mechanical and hydraulic hold-down forces and in-core cross-flows should be limited to a level that does not impact the local critical heat flux margins. In addition, the fuel assembly distortion should not impair the insertion of the reactivity control assembly (e.g. there should be no increase of drop time in pressurized water reactors) to ensure safe reactor shutdown for all applicable plant states (for light water reactors).
(e) Vibration and fretting damage should not affect the overall performance of the fuel assembly and its support structure.
(f) Hydraulic and mechanical loads (including those resulting from a safety shutdown earthquake) should not cause the failure of any component of the fuel assembly.

3.58. For accident conditions (design basis accidents and design extension conditions without significant fuel degradation), the design should prevent any interaction between fuel rods or fuel assemblies and fuel assembly support structures that would impede safety systems from performing their functions as specified in the safety analysis. In particular, the following should be ensured:

(a) Proper functioning of the components of safety systems (e.g. shutdown devices and their guide tubes for pressurized water reactors);
(b) Proper cooling of the core.

Fuel pellet–cladding interaction

3.59. The design should ensure that no fuel cladding failure takes place due to pellet–cladding mechanical interaction in normal operation and anticipated operational occurrences (see Annex I for supplementary information). The design of the fuel rods and plant specific guidelines for power changes in normal operation and anticipated operational occurrences should ensure that excessive pellet–cladding mechanical interaction is prevented.

3.60. In design basis accidents that lead to rapid power transients (e.g. a reactivity initiated accident), the fuel cladding can fail due to excessive pellet–cladding mechanical interaction combined with cladding embrittlement due to in-reactor hydriding at high burnup levels. Fuel failures corresponding to this failure mode should be considered in safety analysis.

3.61. The design should ensure that the likelihood of stress corrosion cracking in the fuel cladding is minimized in normal operation and anticipated operational occurrences (see Annex I for supplementary information).

3.62. Stress corrosion cracking of the fuel cladding should be prevented by implementing adequate design methods such as those given in the following examples:

(a) Reduce tensile stresses in the fuel cladding by restricting rates of power change (allowing for the cladding stresses to relax) or by delaying the time at which the pellet–cladding gap closes (this can be achieved by increasing the initial fill gas pressure in the fuel rod or by optimizing the creep properties of the cladding).
(b) Reduce the corrosive effects of the fission products (e.g. iodine, cadmium, caesium) generated by the pellet by using a liner (for boiling water reactors) or a graphite coating (for pressurized heavy water reactors) that is less susceptible to the corrosive effects on the inner surface of the cladding. This liner can also even out local stress concentrations in the cladding.
(c) Reduce the availability of corrosive fission products at the pellet–cladding interface by using additive fuels that are able to better retain the corrosive fission gas products within the fuel matrix.
(d) Reduce local power peaking factors (and, thus, changes in local linear heat generation rates) through core design techniques.

3.63. The power-ramp failure threshold should be established, if applicable, in test reactors by means of power-ramp tests for each type of fuel or cladding. The data collected should cover the entire burnup range (see Annex I for supplementary information).

3.64. Fuel performance analysis codes can be used to analyse and interpret the data from power-ramp tests and to determine a failure threshold. The parameter used to define this threshold is usually the maximum cladding stress but the strain energy density can also be used. These same fuel performance analysis codes can be used to assess risk factors that cause this type of stress corrosion cracking of fuel rods in the reactor core and to define adequate guidelines to avoid it.

Fuel design limits

3.65. Fuel design limits should be established based on all physical, chemical and mechanical phenomena that affect the performance of fuel rods and fuel assemblies for all applicable plant states.

Design limits for operational states

3.66. For normal operation and for anticipated operational occurrences, the design of fuel rods should address at least the following limitations throughout the whole reloading cycle:

(a) No melting occurs in any location within the fuel pellets;
(b) No cladding overheating occurs (e.g. no departure from nucleate boiling for pressurized water reactors, critical power ratio below limits for boiling water reactors and no dryout condition for pressurized heavy water reactors);
(c) Fuel cladding does not collapse (light water reactor fuel only);
(d) The internal pressure of the fuel rods does not increase to the extent that cladding deformations caused by it would negatively affect the heat transfer between the fuel pellets and the coolant (i.e. there is no reopening of the fuel pellet–cladding gap by cladding lift-off);
(e) Fuel cladding corrosion and hydriding do not exceed specified limits;
(f) Cladding stress and strain remain below specified limits;
(g) Reduction of the cladding wall thickness (e.g. through wear or erosion) does not exceed specified limits.

3.67. Components of fuel rods and fuel assemblies for light water reactors should be designed to maintain low deformation and growth so that the following are ensured:

(a) No geometrical interaction between the fuel rods and fuel assembly top and bottom nozzles occurs (in order to avoid bowing of fuel rods and fuel assemblies for light water reactors). No geometrical interaction between the fuel bundle string and the shield plugs occurs (for pressurized heavy water reactors).
(b) No abnormal local power peaking occurs in the fuel rods.
(c) No degradation of the critical heat flux performance of the fuel assembly occurs.
(d) Reactor scram or other movement of control rods is not impeded.
(e) The handling of fuel assemblies is not hampered.

3.68. To prevent fuel cladding failure caused by pellet–cladding mechanical interaction, possibly assisted by stress corrosion cracking, appropriate operating limits on power changes and power-ramp rates of change should be determined such that the power-ramp failure thresholds are not exceeded.

3.69. The fuel assembly, other reactor vessel internals and the reactor cooling system should be designed to minimize the risk of any obstruction of the coolant flow due to a release of loose parts or debris, so as to prevent fuel damage in operational states.

3.70. Fuel discharge burnup limits, which depend on the performance of the fuel rods and fuel assembly, and on the fuel management approach, should be assessed and justified accordingly.

Design limits for design basis accidents and design extension conditions without significant fuel degradation

3.71. For design basis accidents and design extension conditions without significant fuel degradation, the following should be ensured:

(a) For accident sequences in which some fuel rod failures cannot reasonably be avoided, the number of fuel rod failures should not exceed a small percentage of the total number of fuel rods in the reactor core to minimize the radiological consequences of the accident under consideration.
(b) In determining the total number of fuel rod failures, all known potential failure mechanisms should be evaluated. Chemical reactions, including oxidation and hydriding, cladding ballooning or collapse of the cladding, or damage to the cladding caused by an increase in the fuel enthalpy, are some of the failure mechanisms that should be considered.
(c) Limits applied in assessing the risk for loss of cladding integrity should be based on experimental studies. In determining the limits, chemical, physical, hydraulic and mechanical factors affecting the failure mechanisms, as well as the dimensional tolerances of the fuel rods, should be comprehensively and conservatively evaluated. When fuel failure mechanisms and fuel failure limits are burnup dependent, irradiation effects on cladding and fuel properties should be considered in the experimental studies and should be incorporated into the analyses to ensure that the application of the experimental results is comprehensive.
(d) Fuel failure is considered to occur if the radial average enthalpy of a fuel rod at any axial location, calculated with validated tools, exceeds a certain value to be determined based on representative experimental results by appropriately adjusting test conditions to represent in-reactor conditions (test parameters to take into account include the coolant temperature, coolant pressure, coolant flow rate, reactivity insertion kinetics and fuel rod internal pressure). Since the mechanical resistance of the cladding changes with irradiation and may vary from one cladding type to another,

the reactivity initiated accident failure limit is expected to be dependent on the fuel burnup and on the cladding material.

3.72. The ability to cool the core should not be endangered in the event of the following:

(a) Excessive ballooning or bursting of the fuel rods (e.g. in a loss of coolant accident);
(b) Significant deformation of fuel assembly components or reactor internals (e.g. in a seismic event);
(c) Flow blockage or other consequences of fuel dispersal and fuel coolant interaction as a result of fuel cladding failure (e.g. in a reactivity initiated accident).

The design of fuel rods should also be adequate to prevent undesired consequences of reactivity initiated accidents that may cause damage to the reactor coolant pressure boundary or damage that impairs the capability to cool the core. This is generally ensured by means of limits on the maximum fuel enthalpy and on the allowable increase in fuel enthalpy.

3.73. To ensure that the structural integrity of the fuel rods is preserved, the following design limits should be defined and justified:

(a) The peak cladding temperature in accident conditions should not exceed a level at which cladding oxidation causes excessive cladding embrittlement or accelerates uncontrollably. In addition, for light water reactors, effects on the peak cladding temperature due to fuel fragmentation and its axial relocation within the ballooned area of the fuel rod should be assessed as appropriate. Possible effects of the dispersal of fuel particles on doses to workers and on core coolability should also be addressed.
(b) The total oxidation of the cladding should remain below limits such that the cladding can still withstand accident induced loadings (e.g. in the quenching phase of a loss of coolant accident). Such limits should be determined by experiments that take into account pre-transient in-reactor cladding oxidation and transient oxidation (outer side oxidation and possibly inner side oxidation), pre-transient and transient hydrogen absorption, as well as chemical interactions between the fuel pellets and cladding material.
(c) The allowable enthalpy rise for reactivity initiated accidents should be limited to values that take into account initial fuel rod conditions (e.g. pre-transient hydrogen content of the cladding and fuel burnup).

(d) If applicable, fuel centreline melting should be limited to a small fraction of fuel pellet volume.
(e) Fuel rods should be designed to withstand loadings resulting from post-transient fuel assembly handling, storage and transport to a reprocessing or disposal facility.

3.74. For light water reactors, the amount of hydrogen generated by the chemical reaction between the coolant and the cladding during a loss of coolant accident should not exceed a fraction (e.g. 1%) of the amount of hydrogen that would be generated under the assumption that all claddings surrounding the fuel pellets in the reactor core (excluding the cladding surrounding the plenum volume) react with the coolant.

3.75. In the event that fuel cladding failures during a reactivity initiated accident cannot be prevented, the dispersal of molten fuel particles should not challenge the ability to cool the core.

3.76. Structural deformations of fuel rods, fuel assemblies, control rods and reactor internals should remain limited, so as to avoid any impairment of the movement of control rods in the reactor. In addition, melting temperatures should not be exceeded in the control rods at any time or in any location.

MECHANICAL DESIGN OF CORE STRUCTURES AND COMPONENTS

Design considerations

3.77. The reactor core structures and components should be designed to maintain their structural integrity for all applicable plant states, under various damage mechanisms caused by, for example: vibration (mechanical vibration or flow induced vibration) and fatigue; debris effects; thermal, hydraulic and mechanical loads (e.g. loss of coolant accidents and seismic events); and chemical and irradiation effects (including radiation induced growth). (See Annex I for supplementary information.)

3.78. Of particular concern are damage to reactivity control devices and shutdown devices, and damage to the reactor coolant pressure boundary. The effects of high pressures, high temperatures, temperature variations and the temperature distribution, corrosion, radiation absorption rates and the lifetime radiation exposure on physical dimensions, mechanical loads and material properties should be addressed.

3.79. In addition, solid reactivity control devices should also be designed to withstand handling loads during refuelling operations, transport and storage.

3.80. Important items that are typically addressed within the design of the reactivity control assembly, neutron source assembly and hydraulic plug assembly are described in Annex II.

3.81. The design of the support structures in the core should provide adequate safety margins for thermal stresses generated in all applicable plant states and should take into account additional effects induced by gamma heating on their cooling and thermal responses. The chemical effects of the coolant and the moderator on these structures, which include corrosion, hydriding, stress corrosion and crud buildup, should also be addressed.

3.82. Provision for the inspection of the core components and associated structures should be included in the design of the fuel assembly, control rods and guide structures, and the fuel assembly support structures.

3.83. In light water reactors, the core support structures comprise tube sheets, a core barrel and support keys, which maintain the fuel assembly support structures in the desired geometrical position within the core cavity. These core support structures and fuel assembly support structures should be designed to withstand static and dynamic loads including those induced by refuelling and fuel handling.

3.84. The structures and guide tubes for the shutdown and reactivity control devices, and for instrumentation should be designed such that these devices and instrumentation cannot be moved by inadvertent operator actions, strains on equipment, hydraulic forces due to coolant flow, or movements of bulk moderator for all applicable plant states. The design should facilitate the replacement of these devices and instrumentation. The design should consider the possibility that flow induced vibration of these devices, instruments or their guide tubes may result in fretting, wear and consequent failure in long term operation. The need for dimensional stability of the guide structures over their lifetime should also be addressed in the design.

3.85. In the case of shutdown and reactivity control devices immersed in a bulk moderator (e.g. for pressurized heavy water reactors), the design should be able to accommodate the effects of hydraulic forces on these structures.

3.86. The design should facilitate the replacement of the reactivity control and shutdown devices without causing damage to other reactor core components, unacceptable insertion of reactivity, or excessive radiation exposures of workers.

3.87. Depending on the reactor type, various other structures might be installed within the reactor vessel. These include, for example, feedwater spargers, steam separators, steam dryers, core baffles, reflectors and thermal shields. The functions of these internal structures include flow distribution for the reactor coolant, separation of steam and moisture, and protection of the reactor vessel from the effects of gamma radiation heating and neutron irradiation. These structures should be designed in accordance with paras 3.77–3.81, so that their mechanical performance does not jeopardize the performance of any reactor core safety functions throughout their service life.

Design limits for the mechanical design of core structures and components

3.88. The design of core structures and components should meet limits specified in the applicable codes and standards that are selected in accordance with the safety class (see paras 2.15–2.17).

REACTOR CORE CONTROL, SHUTDOWN AND MONITORING SYSTEMS

Reactor core control system

3.89. This section describes important considerations for the control system for maintaining the shapes, levels and stability of the neutron flux within specified limits in all applicable plant states, in order to meet Requirement 45 of SSR-2/1 (Rev. 1) [1].

3.90. Paragraph 6.4 of SSR-2/1 (Rev. 1) [1] states that “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.”

3.91. The core design should allow for the installation of the necessary instrumentation and detectors for monitoring the core parameters, such as the core power (level, distribution and time dependent variation), the conditions and physical properties of the coolant and moderator (flow rate and temperature), and the expected effectiveness of the means of reactor shutdown (e.g. the insertion rate of the absorber devices compared with their insertion limits), so that any necessary corrective action can be taken. The instrumentation should monitor relevant parameters over their expected ranges for all applicable plant states including during refuelling.

Reactivity control devices

3.92. The means of control of reactivity should be designed to enable the power level and the power distribution to be maintained within safe operating limits. This includes compensating for changes in reactivity to keep the process parameters within specified operating limits, such as those associated with:

(a) Normal power manoeuvres;
(b) Changes in xenon concentration;
(c) Effects relating to temperature coefficients;
(d) Rate of flow of coolant, or changes in coolant (or moderator) temperature and density;
(e) Depletion of fuel and of burnable absorber;
(f) Cumulative neutron absorption by fission products.

3.93. Reactivity control devices should be capable of maintaining the reactor in a subcritical condition, with consideration given to design basis accidents and their consequences. Provisions should be included in the design to maintain subcriticality for plant states in which normal shutdown, fuel cooling or the integrity of the primary cooling system is temporarily disabled (e.g. when the reactor vessel is open for maintenance or refuelling in light water reactors).

3.94. The types of reactivity control device used for regulating the core reactivity and the power distribution for different reactor designs are described in Annex I.

3.95. The use of control rods or systems as the means of reactivity control for normal operation should not adversely affect their capability and efficiency required to execute fast reactor shutdown.

3.96. The maximum degree of positive reactivity and its rate of increase by insertion in all applicable plant states are required to 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 (see para. 6.6 of SSR-2/1 (Rev. 1) [1]).

3.97. The arrangement, grouping, speed of withdrawal and withdrawal sequence of the reactivity control devices, used in conjunction with an interlock system, should be designed to ensure that any abnormal withdrawal of the devices does not cause the specified fuel limits to be exceeded. Such abnormal withdrawal of the reactivity control devices should be addressed in the safety assessment.

3.98. Reactivity control systems using a soluble absorber should be designed to prevent any unanticipated decrease in the concentration of absorber in the core that could cause specified fuel limits to be exceeded. Those parts of systems that contain soluble absorbers, such as boric acid, should be designed to prevent precipitation (e.g. by heating of the components; see IAEA Safety Standards Series No. SSG-56, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants [13]). The concentrations of the soluble absorber in all storage tanks should be monitored. Whenever enriched boron (^{10}B) is used, appropriate monitoring should be provided.

3.99. A detailed functional analysis of the alignments and operational conditions of the control systems should be performed to identify any potential for inadvertent dilution of boron in operation and in shutdown conditions, and to ensure the adequacy of preventive and recovery measures. Such preventive measures may include permanent administrative locking (of valves or parts of circuits), active isolation actions, interlocks of external injection systems, monitoring of boron concentrations in connected vessels or piping systems, and interlocks for starting recirculation pumps.

3.100. The effectiveness of reactivity control devices, such as neutron absorber rods, should be verified by direct measurement.

3.101. Paragraph 6.5 of SSR-2/1 (Rev. 1) [1] states that "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."

3.102. In particular, the following environmental effects should be addressed in the design of control systems:

(a) Irradiation effects such as depletion of the absorber material or swelling and heating of materials due to neutron and gamma absorption. Control rods should be replaced or exchanged accordingly.

(b) Chemical effects such as corrosion of the reactivity control devices. The transport of activated corrosion products through the reactor coolant system and moderator system should also be addressed.
(c) Changes in structural dimensions, such as dimensional changes or movements of internal core structures due to temperature changes, irradiation effects or external events such as earthquakes, should not prevent the insertion of the reactivity control devices.

Reactor shutdown system

3.103. This section describes important considerations for systems designed to bring the reactor to a subcritical state from all applicable plant states, and to maintain it in this state, in accordance with Requirement 46 of SSR-2/1 (Rev. 1) [1]. Requirement 61 of SSR-2/1 (Rev. 1) [1] for the protection system also applies to the reactor shutdown system.

3.104. The reactor shutdown system should ensure that, for all applicable plant states, design limits for the shutdown margin (paras 3.24 and 3.25) are not exceeded. The necessary reliability should be ensured through the design of the equipment. In particular, the design should ensure the necessary independence between plant processes and control and protection systems.

3.105. Paragraph 6.7 of SSR-2/1 (Rev. 1) [1] states that "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." Recommendations on the rate of shutdown are provided in paras 3.106–3.108.

3.106. The rate of shutdown should be adequate to render the reactor subcritical with an adequate margin, so that the specified design limits on fuel and on the reactor system pressure boundary are met.

3.107. In designing for or evaluating the rate of shutdown, the following factors should be addressed:

(a) The response time of the instrumentation to initiate the shutdown.
(b) The response time of the actuation mechanism of the means of shutdown.
(c) The location of the shutdown devices (depending on the chosen reactor core design).
(d) The ease of entry of the shutdown devices into the core. This can be achieved by the use of guide tubes or other structural means to facilitate the insertion

of devices, including the possible incorporation of flexible couplings to reduce rigidity over the length of the devices.

(e) The insertion speed of the shutdown devices. One or more of the following can be used to deliver the necessary insertion speed:
   (i) Gravity drop of shutdown rods into the core;
   (ii) Hydraulic or pneumatic pressure drive of shutdown rods into the core;
   (iii) Hydraulic or pneumatic pressure injection of soluble neutron absorber.

3.108. Means of checking the insertion speed of shutdown devices should be provided. The insertion time should be checked regularly (typically, at the beginning of each cycle) and possibly during the cycle if the margins to the limits are not sufficient.

3.109. Paragraph 6.8 of SSR-2/1 (Rev. 1) [1] states:

> "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."

Generally, in assessing the failure of a control rod to insert, it should be assumed that the most reactive core conditions arise when the shutdown device that has the highest reactivity worth cannot be inserted into the core (i.e. the assumption that one shutdown device is stuck).

Different means of shutdown

3.110. Paragraph 6.9 of SSR-2/1 (Rev. 1) [1] states that "The means for shutting down the reactor shall consist of at least two diverse and independent systems." In addition, para. 6.10 of SSR-2/1 (Rev. 1) [1] states "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."

3.111. Table 1 provides typical examples that illustrate the diversity of the means of shutdown for three different reactor types (boiling water reactors, pressurized water reactors and pressurized heavy water reactors).

TABLE 1. MEANS OF SHUTDOWN FOR DIFFERENT REACTOR TYPES

Reactor type	Fast shutdown system	Diverse shutdown system
Boiling water reactor	B_4C in steel tubes or hafnium plates (or a hybrid design)	Boron solution injected into the moderator or coolant
Pressurized water reactor	Ag–In–Cd in steel tubes or B_4C in steel tubes, hafnium rods	Boron solution injected into the moderator or coolant
Pressurized heavy water reactor	Cadmium rods sandwiched and sealed between stainless steel tubes moving in zirconium alloy guide tubes	Gadolinium solution injected into low pressure moderator*

* This shutdown system can also act as another fast shutdown system.

Reliability

3.112. The design should include the following measures to achieve a high reliability of shutdown by means of each the following measures, or a combination of these as appropriate:

(a) Adopting systems with uncomplicated design and simple operation, and with automatic activation.
(b) Selecting equipment of proven design.
(c) Using a fail-safe design as far as practicable (see Annex I for supplementary information).
(d) Giving consideration to the possible modes of failure and adopting redundancy in the activation of the shutdown systems (e.g. sensors). Provision for diversity may be made, for example, by using two different and independent physical trip parameters for each accident condition as far as practicable.
(e) Functionally isolating and physically separating the shutdown systems (this includes the separation of control and shutdown functions) as far as practicable, on the assumption of credible modes of failure and including common cause failure.
(f) Ensuring ease of entry of the means of shutdown into the core, with consideration of the in-core environmental conditions for operational states and accident conditions within the design basis.

(g) Designing to facilitate maintenance, in-service inspection and operational testability.
(h) Providing means for performing comprehensive testing during commissioning and periodic refuelling or maintenance outages.
(i) Testing of the actuation mechanism (or of partial rod insertion, if feasible) during operation.
(j) Designing to function under extreme conditions (e.g. earthquakes).

3.113. In the design of shutdown systems, as stated in para. 6.5 of SSR-2/1 (Rev. 1) [1], wear out of the control rod cladding and the effects of irradiation, such as burnup, changes in physical properties and production of helium gas are required to be taken into account. The recommendations in para. 3.102 are also applicable to the design of shutdown systems. Specific recommendations for diverse shutdown systems injecting neutron absorbers into the reactor coolant system are provided in SSG-56 [13].

Effectiveness of the shutdown system

3.114. Paragraph 6.11 of SSR-2/1 (Rev. 1) [1] states:

> "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."

The requirements for long term shutdown and deliberate actions that increase reactivity in the shutdown state (e.g. the movement of absorbers for maintenance purposes, the dilution of the boron content and refuelling actions) should be identified and evaluated to ensure that the most reactive condition is addressed in the criticality analysis.

3.115. The design should determine the number and the reactivity worth of shutdown rods by considering various factors. Important factors to be taken into account include:

(a) The core size.
(b) The fuel type and the core loading scheme.
(c) The required margin of subcriticality.
(d) Assumptions relating to failure of a shutdown device or devices.
(e) Uncertainties associated with the calculations.
(f) Shutdown device shadowing (see Annex I for supplementary information).

(g) The most reactive core conditions after shutdown. These are associated with a number of parameters such as:
    (i) The most reactive core configuration (and, where appropriate, the corresponding boron concentration) that will occur during the whole reloading cycle, including during refuelling;
    (ii) The most reactive credible combination of fuel and moderator temperatures;
    (iii) The amount of positive reactivity insertion resulting in design basis accident conditions;
    (iv) The amount of xenon as a function of time after shutdown;
    (v) Burnup of the absorber.

3.116. The effectiveness of the shutdown system should be demonstrated:

(a) In the design, by means of calculation;
(b) During commissioning and prior to startup after each refuelling, by means of appropriate neutronic and process measurements to confirm the calculations for the given core loading;
(c) During reactor operation, by means of measurements and calculations covering the actual and anticipated reactor core conditions.

These analyses should cover the most reactive core conditions, and should include the assumption of the failure of the shutdown device(s). In addition, the shutdown margin should be maintained if a single random failure occurs in the shutdown system.

3.117. If the operation of the reactor shutdown system is manual or partly manual, the necessary prerequisites for manual operation should be met (see IAEA Safety Standards Series No. SSG-39, Design of Instrumentation and Control Systems for Nuclear Power Plants [14]).

3.118. Part of the means of shutdown may be used for the purposes of reactivity control and flux shaping in normal operation. Such use should not jeopardize the functioning of the shutdown system under any condition in all applicable plant states.

3.119. The shutdown system should be testable, as far as practicable, during operation in order to provide assurance that the system is available on demand.

Separation of protection systems from control systems

3.120. As stated in Requirement 64 of SSR-2/1 (Rev. 1) [1], protection systems are required to be physically and functionally separated from control systems to avoid failures of control systems causing failures in the protection system. Guidance on separation of the protection system from other systems is provided in SSG-39 [14].

Partial trip system

3.121. In some reactor designs, when measured core parameters (e.g. temperatures, pressures, levels, flows and flux) exceed certain plant design limits, a partial trip system can be activated for protection of the reactor. If applicable, the design should ensure that a partial trip triggered by any anticipated operational occurrence transient does not allow specified fuel design limits to be exceeded.

Operating limits and set points

Operating limits for the control system

3.122. The design should include operating limits and associated set points for actions, alarms or reactor trip to ensure that the operating power distributions remain within the design power distributions.

3.123. Limits and set points should consider the impacts of fuel burnup, shadowing effects and coolant stratification (coolant temperature distribution).

3.124. Determination of the operating limits and set points should include effects of the ageing of the reactor coolant system (e.g. steam generator tube plugging in pressurized water reactors, and an increase of the diameter of the pressure tube in pressurized heavy water reactors).

Set points for reactor core protection

3.125. Set points should be established and used to control or shut down the reactor at any time during operation. The automatic initiation of control and protection systems during a reactor transient should prevent damage to the fuel and, in the early stages of a reactor accident, should minimize the extent of damage to the fuel.

3.126. Equipment performance requirements and operating limits, and procedures should be defined to prevent excessive control rod worth or reactivity insertion rates. Their capability should be demonstrated. Where feasible, an alarm should be installed to function when any such limit or restriction is violated or is about to be violated.

3.127. The design limits, uncertainties, operating limits, instrument requirements and set points should be stated in technical specifications to be used by facility operators.

Core monitoring system

3.128. In accordance with Requirement 59 of SSR-2/1 (Rev. 1) [1], core monitoring instrumentation is required to be provided to support the reactor protection and control systems, as well as to supply sufficiently detailed and timely information on the local heat generation conditions prevailing in the core. The core design should accommodate the detectors and devices for monitoring the magnitude and changes of core power, as well as the local distribution of heat generation in the core, in order to enable any required modification of core parameters (e.g. the insertion position of control rods, neutron flux, reactor coolant temperature and pressure) within their defined operating ranges. The speed of the variation in a parameter should determine whether the actuation of the reactor control systems is to be automatic or manual.

3.129. In addition, radionuclide activity levels in the coolant should be monitored to assess the integrity of the fuel system during operation and to verify that design limits or operational limits are not exceeded.

3.130. Appropriate parameters should be selected for core monitoring. This selection will depend on the reactor type. The following are examples of parameters to be measured for the purposes of core monitoring:

(a) Spatial distribution of the neutron flux and related power distribution peaking factors;
(b) Pressure of the reactor coolant system;
(c) Coolant temperature (e.g. inlet temperature and outlet temperature);
(d) Speed of the reactor coolant pump;
(e) Water level (for light water reactors);
(f) Radionuclide activity in the coolant (see Annex I for supplementary information);
(g) Insertion position of the control rods;

(h) Concentration of soluble boron or ^{10}B content when enriched boron is used (for a pressurized water reactor).

3.131. Other safety related parameters may be derived from the measured parameters. Examples of such safety related parameters include the following:

(a) Neutron flux doubling time;
(b) Rate of change of the neutron flux;
(c) Axial and radial neutron flux imbalances;
(d) Reactivity balance;
(e) Thermohydraulic core parameters (e.g. core thermal power, linear heat generation rate, reactor coolant flow rate, the departure from nucleate boiling ratio or the critical power ratio).

3.132. The accuracy, speed of response, range and reliability of all monitoring systems should be adequate for performing their intended functions (see SSG-39 [14]). The design of the monitoring system should provide for the continuous or adequate periodic testing of these systems.

3.133. Guidance on post-accident monitoring is provided in SSG-39 [14]. If core monitoring is necessary in accident conditions, for example, to monitor system temperatures, the reactor vessel water level or reactivity, the instrumentation to be used should be qualified to withstand the environmental conditions to be expected during and following the accident.

3.134. The spatial power distribution should be monitored by means of ex-core or in-core instrumentation (such as neutron detectors and gamma thermometers). Measurements of the local power at different positions in the core should be performed to ensure that adequate safety margins are maintained considering the impact of the spatial power distribution changes due to core control effects and core burnup effects. The in-core power distribution should be monitored routinely. Detectors should be adequately distributed in the core to reliably detect the local changes in power density. Both ex-core and in-core neutron detectors should be calibrated periodically.

3.135. A computerized core monitoring system should be used to ensure that the status of the core is within the operational limits assumed in the safety analysis. Qualification of the system should be ensured to a level consistent with the safety category of the functions performed, wherever it is coupled to a protection system (see SSG-39 [14]).

3.136. During reactor shutdown, a minimum set of instruments or combination of instruments and neutron sources should be available to monitor neutron flux and heat generation distribution (e.g. using flux detectors with an adequate sensitivity) whenever fuel assemblies are present in the reactor vessel, including during fuel loading and startup phases.

3.137. During reactor startup in some reactors, a combination of interlocks on flux monitoring systems and reactivity control devices is used to ensure that the most appropriate monitors are used for particular flux ranges and to avoid undue reactor trips. The design of such interlock systems should be consistent with the design of the reactor protection system.

3.138. During reactor startup, and especially during the first startup, the neutron flux is very low relative to that in full power operation, so more sensitive neutron detectors may be necessary temporarily to monitor the neutron flux. A neutron source may be necessary to increase the flux to a level that is within the range of the startup neutron flux monitors. The design of such neutron sources should ensure that:

(a) The sources function properly to provide sufficient signals from the neutron flux monitors over their planned lifetime;
(b) The sources are compatible with the fuel assemblies and the fuel assembly support structures.

CORE MANAGEMENT

Design considerations

3.139. The primary objective of core management is to ensure the safe, reliable and optimum use of the fuel in the reactor, while remaining within operational limits and conditions.

3.140. Each reloading cycle should be designed with appropriate means of controlling the core reactivity and the power distribution to address fuel design limits.

3.141. While the details of core management depend on the reactor type, in all cases, the core management programme should provide the following:

(a) Means to perform core management functions effectively throughout the reloading cycle so as to ensure that core parameters remain within core management design limits. Core management functions include core design (specification of loading and shuffle patterns of fuel assemblies to provide optimum fuel burnup and desired fluxes), procuring fuel assemblies, reactivity determinations and core performance monitoring.
(b) Core operating strategies that permit operating flexibility and good fuel utilization while remaining within core management design limits.

Core design

3.142. To achieve the desired core reactivity and power distribution for reactor operation, the operating organization should be provided with the following information:

(a) Loading patterns (including enrichment and configuration of fuel rods) and orientation of fuel assemblies in each reloading cycle (for light water reactors);
(b) Schedule for the subsequent unloading and loading of fuel assemblies;
(c) Configurations of reactivity control and shutdown devices;
(d) Burnable absorbers and other core components to be discharged, inserted or adjusted.

3.143. Parameters associated with depletion of fuel and burnable absorber, and other reactor physics parameters are provided as inputs to safety analyses, plant monitoring and protection systems, and operator guidance. Thus, these parameters should be analysed based on predetermined plant operating objectives and resultant plans. These reactor physics parameters include the reactor startup conditions (e.g. critical boron concentrations and control rod positions, reactor kinetics, fuel temperature coefficients, moderator temperature coefficients, control rod and control bank worths, and power peaking factors).

3.144. Unplanned power manoeuvring during flexible operation may alter the power and burnup profile across the core. As such, predictions of parameters associated with depletion of fuel and burnable absorber and other nuclear parameters should be continuously or periodically examined and evaluated, using relevant monitoring parameters.

3.145. The design of the reactor core should include analyses to demonstrate that the fuel management strategy and the established limitations on operation are such that the nuclear design limits and, hence, the fuel design limits will be met during the whole reloading cycle.

3.146. Multidimensional and multiscale physics codes and system thermohydraulic codes should preferentially be used for realistic analysis of the reactor core for all applicable plant states. Uncertainties should be adequately incorporated into the analyses (see SSG-2 (Rev. 1) [6] for details).

3.147. The reactor core analysis should be performed based on typical cases covering the whole reloading cycle for the following reactor core conditions:

(a) Full power, including representative power distributions;
(b) Load following (as applicable);
(c) Approach to criticality and power operation;
(d) Power cycling;
(e) Startup;
(f) Refuelling;
(g) Shutdown;
(h) Anticipated operational occurrences;
(i) Operation at the thermohydraulic stability boundary (for boiling water reactors).

Whenever the management of fuel in the core is changed or any characteristics of the fuel rods (such as the fuel enrichment, fuel rod dimensions, fuel rod configuration or the fuel cladding material) are changed, a new reactor core analysis should be performed and documented.

3.148. The reactor core analysis should include analyses of the performance of the fuel rods based on average and local power levels, and axial temperature distributions to demonstrate that the respective thermal and mechanical fuel design limits are met for all operational states. For light water reactors, the reactor core analysis should include analyses of peak channel power and peak linear power rates for normal full power operation and steady state radial power distribution at each fuel assembly location, and axial power distributions in each fuel assembly. Allowance should be made to take into account the effects of changes in the geometry of the fuel assembly on its neutronic and thermohydraulic performance (e.g. changes in the moderator gap thickness due to bowing of the assembly). The reactor core analysis should also include the radial power distribution within a fuel assembly and the axial power distortion due to spacers, grids and other components in order to identify hot spots and to evaluate the local power levels.

Refuelling

3.149. For on-power refuelling in pressurized heavy water reactors, the effects of the refuelling operation on the neutronic behaviour of the core should be demonstrated to remain within the control capability of the reactor control systems.

3.150. Safety assessment should address any event that could cause inadvertent criticality during core loading or unloading and during handling phases.

3.151. The fuel loading sequence should be monitored through the use of in-core (for boiling water reactors) or ex-core flux distribution measurements, or by means of special administrative measures. The fuel loading pattern after reloading should be validated through the measurement of the flux distribution.

3.152. For light water reactors, the reactor core should be designed such that the consequences of the worst misloaded fuel assembly, if any, remain within nuclear design limits and fuel design limits. If a misloaded fuel assembly can be prevented by special measures and equipment, the effectiveness and reliability of these precautionary measures should be demonstrated. Computational analyses should be performed if it cannot be demonstrated that the specified precautionary measures are sufficient.

Core management design limits

3.153. The reactor core analysis should verify that the core fuel loading pattern will meet the fuel design limits for all applicable plant states.

3.154. For practical reasons and simplicity, for light water reactors, a system that develops and monitors the nuclear key safety parameters (see para. 3.130) can be used to verify the suitability of the reload core design.

Special core configurations

Mixed core

3.155. When fuel assemblies of different types are loaded into the core (a ‘mixed core’), all fuel assembly types should meet the fuel design limits for all applicable plant states. An assessment should be performed for the initial loading and subsequent reloading of mixed cores. It should include the dimensional, mechanical and thermohydraulic response of the various fuel types (e.g. in terms of pressure drop characteristics through the fuel assembly or assemblies and flow rate), the compatibility of each fuel assembly with the neutronic and thermohydraulic characteristics of the original core and with the related safety analyses. The critical heat flux or critical power correlation used in the core monitoring system should be valid for all fuel assembly types present in the mixed core.

3.156. Relevant nuclear key safety parameters, such as reactivity, reactivity coefficients, control rod worth and power distributions, should be evaluated for the different fuel assembly designs. The evaluation of the compatibility of fuel assemblies may be developed based on calculations for single type fuel assemblies with appropriate provisions to cover all fuel assemblies. The combined effects on the related core-wide parameters should be evaluated.

Mixed oxide fuel core

3.157. The design of a mixed oxide core should include analyses to ensure that the nuclear design limits (for both initial loading and subsequent reloading) and the fuel design limits are met for all applicable plant states. In the analyses, the following considerations should be addressed:

(a) The properties of mixed oxide fuel (see Annex I for supplementary information) are somewhat different from those of UO_2 fuel and this should be incorporated into computer codes and models used for the fuel design and safety analyses.
(b) In a mixed oxide core, the control rod worth and absorber worth are reduced as a result of neutron spectrum hardening owing to the higher thermal absorption cross-sections of plutonium compared with uranium, and, as a result, the reactor shutdown margin can be reduced. To compensate for the reduced shutdown margin, additional control rods should be available for insertion or additional absorption capability of the absorbing materials (e.g. an increase in ^{10}B enrichment) should be implemented.
(c) The kinetic parameters for mixed oxide fuel, namely the total fraction of delayed neutrons and the prompt neutron lifetime, are lower than those for UO_2 fuel. The lower delayed neutron fraction of mixed oxide fuel can result in a prompt critical reactor condition with a smaller reactivity insertion; thus, there is less time for control rod insertion or boron system injection to provide reactivity control. This should be addressed in the core design and

safety analyses for all applicable plant states (e.g. reactivity initiated events as anticipated operational occurrences and design basis accidents).

(d) The fission cross-sections in mixed oxide fuel are larger than those in UO_2 fuel, and this can result in steep flux gradients between adjacent mixed oxide and UO_2 fuel rods. This effect can be reduced by varying the plutonium content and by adjusting the loading pattern in the core design. Other consequences of the differences in cross-section between plutonium and uranium are changes in the moderator temperature coefficient, the fuel temperature coefficient and the coefficient of reactivity for coolant voids. The core design and safety analyses should evaluate the effects of these changes in reactivity coefficients.

Load following and power manoeuvring

3.158. The effects of operating conditions, such as load following (see Annex I for supplementary information), power cycling, reactor startup and refuelling manoeuvring, should, whenever specified, be superimposed onto the power level distributions and temperature histories to evaluate the potential effects of thermal cycling on fuel rod thermomechanical responses, such as the buildup of pressure due to fission gas release to the pellet–cladding gap and due to fuel cladding fatigue.

3.159. Once the extent of the desired flexibility is determined, an in depth evaluation of impacts on the design and operation of the nuclear power plant (i.e. requirements on the safety analysis and the operational limits and conditions) should be performed. Based on this evaluation, additional specifications for qualification and implementation can be developed.

3.160. To ensure the control of core reactivity with load following and power manoeuvring, the core and generator power balance and the reactor stability should be maintained.

3.161. The operational limits should be adjusted to cover perturbations due to load following operation (see Annex I for supplementary information).

Reactor operation with leaking fuel rods

3.162. Fuel rod failures can affect ease of access, work scheduling and worker dose for plant operations personnel. Operation of a reactor core with defective fuel rods should stay within the radiochemical requirements (see Annex I for supplementary information) as defined by the limit on coolant radionuclide activity included in the technical specifications for the plant.

3.163. The core design and operations programme should establish procedures and limits for operating the core with defective fuel assemblies while ensuring that dose constraints for workers are not exceeded. In light water reactors, the reactors should be shut down if the operating radiochemical limits are exceeded, and all defective fuel assemblies should be replaced in accordance with procedures during the outage. In pressurized heavy water reactors, fission product release from defective fuel and, subsequent, secondary hydriding of the cladding can be minimized by reducing the power level of defective fuel rods. (See Annex I for supplementary information.)

Core redesign after fuel assembly repair

3.164. In light water reactors, fuel assemblies containing damaged and leaking fuel rods may be repaired and reconstituted with replacement rods, dummy rods or vacancies. The use of vacancies should be limited so that design limits are met.

3.165. The impact of a reconstituted fuel assembly on the design of the reactor core should be assessed.

Impact of fuel design and core management on fuel handling, transport, storage, reprocessing and disposal

3.166. Design limits are determined, based on the concept of defence in depth, to meet safety requirements for all applicable plant states. The fuel design limits described in paras 3.65–3.76 should be extended to ensure that the fuel rods and fuel assemblies remain intact (when applicable) or do not degrade further (in the case of leaking fuel rods) in the back end of the nuclear fuel cycle after the assemblies are discharged from the core. The back end of the fuel cycle includes handling, transport, storage, reprocessing and disposal. The following fuel performance parameters are among those that may have an impact on the post-irradiation behaviour of the fuel rods and the fuel assemblies:

(a) Internal pressure of fuel rods at the end of life

    Even though fuel rods can withstand some extent of overpressurization exceeding the normal coolant pressure without failure in normal operation, the handling of such highly pressurized used fuel rods might not be acceptable when coolant counter-pressure is diminished (e.g. in spent fuel storage facilities). This is particularly relevant for mixed oxide fuels that remain at a higher temperature for a longer period of time and continue to release helium gas from the fuel material.

(b) Massive cladding hydriding and cladding mechanical properties

Localized hydriding (e.g. due to spalling of the corrosion layer or due to axial pellet–pellet gaps) might not take place during normal operation or be of consequence in accident conditions, but such a condition may lead to delayed hydride cracking of zirconium based alloy cladding in post-irradiation handling or storage, or undesired failures in the event of a transport accident.

(c) Fretting wear

Localized wear (i.e. grid-to-rod fretting wear for light water reactors, spacer-to-spacer fretting wear for pressurized heavy water reactors) is usually undetected unless it wears completely through the cladding wall and creates a leakage pathway. Some fuel rods affected by excessive wear may exhibit localized weakness that may lead to long term creep failures or other mechanical failures in the event of a transport accident.

(d) Fuel discharge burnup

Fuel design, core management and the resultant fuel discharge burnup affect the fuel isotopic vector, which, in turn, will impact the economy of fuel reprocessing or disposal. High fuel discharge burnups degrade fuel isotopic compositions and, therefore, affect reactivity. In mixed oxide fuel, the plutonium content should be adjusted to maintain parity with the reactivity of the UO_2 fuel present in the core, up to the anticipated fuel discharge burnup level.

(e) Other aspects

New fuel rod designs or new fuel assembly designs, proposed by the fuel vendors to address other in-reactor issues (e.g. stress corrosion cracking of fuel cladding, fission gas release, fuel assembly distortion and fuel performance in accident conditions), should remain compatible with requirements relating to the back end of the nuclear fuel cycle.

4. QUALIFICATION AND TESTING

GENERAL

4.1. The safe operation of the reactor core throughout the lifetime of the SSCs of the reactor core, including the fuel rods and assemblies, core components and control systems, necessitates a robust programme for qualification, inspection and testing of the equipment design and analysis process. This can be achieved as described in this section of the Safety Guide.

DESIGN QUALIFICATION

4.2. A qualification programme should confirm the capability of the reactor core SSCs to perform their function for the relevant time period, with account taken of the appropriate functional and safety considerations under prescribed environmental conditions (e.g. pressure, temperature, radiation levels, mechanical loading and vibration). These environmental conditions should include the variations expected in normal operation, anticipated operational occurrences, design basis accidents and design extension conditions without significant fuel degradation.

4.3. The characteristics of certain postulated initiating events may preclude the performance of realistic commissioning tests and recurrent tests that could confirm that SSCs would perform their intended safety functions when called upon to do so, for example in the case of an earthquake. For the SSCs concerned and the events considered, a suitable qualification programme should be planned and performed prior to their installation.

4.4. Methods of qualification should include:

(a) Performance of a type test on the SSCs representative of those to be supplied;
(b) Performance of a test on the SSCs supplied;
(c) Use of pertinent past experience;
(d) Analysis based on available and applicable test data;
(e) Any combination of the above methods.

4.5. Design qualification may be established through operating experience with fuel systems of the same or similar design. The basis for the previous experience should be identified and the performance record should be evaluated.

The maximum burnup and operating experience of the core at power should be referenced, and the performance of the fuel assemblies should be compared against design criteria identified for phenomena such as fretting wear, oxidation, hydriding, crud buildup and bowing of fuel assemblies.

INSPECTION

4.6. A system should be designed to allow the identification of each fuel assembly and to ensure its proper orientation within the core. Following initial loading of the fuel and any reloading, the locations and orientation of each fuel assembly should be inspected to verify correct location and positioning.

TESTING INCLUDING PROTOTYPE ASSEMBLIES AND LEAD USE ASSEMBLIES

4.7. Provisions should be made in the design for in-service testing and inspection to ensure that the core and its associated structures and the reactivity control and shutdown systems will perform their intended functions throughout their lifetime. Further guidance on in-service inspection is provided in IAEA Safety Standards Series No. NS-G-2.6, Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants [15].

4.8. Out-of-reactor tests should be performed on fuel assembly prototypes, when practical, to determine the characteristics of a new design. The following out-of-reactor tests are generally performed for this purpose:

(a) For light water reactors:
   - (i) Spacer grid tests (including pressure drop tests, crush strength tests and other structural tests such as seismic resistance tests);
   - (ii) Control rod structural and performance tests;
   - (iii) Fuel assembly structural tests (lateral, axial and torsional stiffness, and frequency and damping);
   - (iv) Fuel assembly hydraulic flow tests, including the determination of pressure drop and fuel assembly lift-off force, control rod vibration and wear, fuel assembly vibration, grid-to-rod fretting (with account taken of the relaxation of spacer grid springs), and evaluations of the wear and the lifetime of fuel assemblies;
   - (v) Fuel assembly thermohydraulic tests, including the determination of critical heat flux correlations.

(b) For pressurized heavy water reactors:
   (i) Fuel bundle string pressure drop tests;
   (ii) Cross-flow endurance tests;
   (iii) Mechanical endurance tests;
   (iv) Bundle impact tests;
   (v) Bundle strength tests;
   (vi) Wear tests;
   (vii) Seismic qualification tests;
   (viii) Wash-in and wash-out tests (where applicable);
   (ix) Critical heat flux tests.

4.9. In-reactor testing of design features through irradiation of fuel rods or fuel assemblies in materials test reactors or through irradiation of lead use fuel assemblies in power reactors should be used to justify the specified maximum burnup limit for a new design. The following phenomena may be tested in this manner:

(a) Fuel and burnable absorber rod growth;
(b) Fuel rod bowing;
(c) Fuel rod, spacer grid and fuel channel (if present) oxidation and hydride levels;
(d) Fuel rod fretting and spacer (for pressurized heavy water reactors) fretting;
(e) Fuel assembly growth;
(f) Fuel assembly bowing;
(g) Fuel channel (for boiling water reactors) wear and distortion;
(h) Fuel rod ridging (i.e. pellet–cladding interaction);
(i) Fuel rod integrity;
(j) Hold-down spring relaxation (for pressurized water reactors);
(k) Spacer grid spring relaxation (for light water reactors);
(l) Control rod and guide tube wear (for pressurized water reactors).

4.10. In cases where in-reactor testing of a new fuel assembly design or a new design feature cannot be performed, special attention should be given to analytical evaluations and to augmented inspection or surveillance plans to validate the fuel design capability and performance features.

REFERENCES

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Nuclear Power Plants: Design, IAEA Safety Standards Series No. SSR-2/1 (Rev. 1), IAEA, Vienna (2016).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Glossary: Terminology Used in Nuclear Safety and Radiation Protection, 2018 Edition, IAEA, Vienna (2019).

[3] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

[4] INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Nuclear Installations, IAEA Safety Standards Series No. GS-G-3.5, IAEA, Vienna (2009).

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Leadership and Management for Safety, IAEA Safety Standards Series No. GSR Part 2, IAEA, Vienna (2016).

[6] INTERNATIONAL ATOMIC ENERGY AGENCY, Deterministic Safety Analysis for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-2 (Rev. 1), IAEA, Vienna (2019).

[7] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Fuel Handling and Storage Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-63, IAEA, Vienna (in preparation).

[8] INTERNATIONAL ATOMIC ENERGY AGENCY, Storage of Spent Nuclear Fuel, IAEA Safety Standards Series No. SSG-15 (Rev. 1), IAEA, Vienna (in preparation).

[9] INTERNATIONAL ATOMIC ENERGY AGENCY, Seismic Design and Qualification for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-1.6, IAEA, Vienna (2003). (A revision of this publication is in preparation.)

[10] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Classification of Structures, Systems and Components in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-30, IAEA, Vienna (2014).

[11] INTERNATIONAL ATOMIC ENERGY AGENCY, Operational Limits and Conditions and Operating Procedures for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-2.2, IAEA, Vienna (2000). (A revision of this publication is in preparation.)

[12] INTERNATIONAL ATOMIC ENERGY AGENCY, Core Management and Fuel Handling for Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-2.5, IAEA, Vienna (2002). (A revision of this publication is in preparation.)

[13] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-56, IAEA, Vienna (in preparation).

[14] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of Instrumentation and Control Systems for Nuclear Power Plants, IAEA Safety Standards Series No. SSG-39, IAEA, Vienna (2016).

[15] INTERNATIONAL ATOMIC ENERGY AGENCY, Maintenance, Surveillance and In-service Inspection in Nuclear Power Plants, IAEA Safety Standards Series No. NS-G-2.6, IAEA, Vienna (2002). (A revision of this publication is in preparation.)

Annex I

SUPPLEMENTARY TECHNICAL INFORMATION

I–1. Table I–1 provides supplementary technical information to clarify the meaning of terms that are not defined in the IAEA Safety Glossary [I–1] but that are used in this Safety Guide, and to provide additional background or supporting examples for specified design recommendations provided in this Safety Guide.

TABLE I–1. SUPPLEMENTARY TECHNICAL INFORMATION TO CLARIFY TERMINOLOGY AND PROVIDE ADDITIONAL BACKGROUND

Topic	Clarification	Relevant paragraphs in the Safety Guide
Burnable absorber	For light water reactors, in order to maintain a negative moderator temperature coefficient, the designer could choose to add a fixed burnable absorber to the fuel pellets or to the fuel assembly in the form of burnable absorber rods to reduce the required concentration of the burnable absorber in the moderator. A burnable absorber could also be used to flatten the power distribution and to reduce variations in reactivity during fuel burnup.	3.25
Cladding	Leaktightness of the cladding is necessary to prevent the release of volatile fission products, and structural integrity of the cladding is necessary to maintain a coolable geometry and to permit discharge of components from the core using normal refuelling equipment.	2.4
	Zirconium based alloy materials (e.g. zircaloy-2, zircaloy-4, ZIRLO and Optimized ZIRLO, M5, E110) are typically used for the cladding material. Other innovative cladding materials are under development for use in applications, such as enhanced accident tolerant fuel, with a focus on more benign steam reactions and lower hydrogen generation.	3.5
	Integrated-time temperature refers to the assessment of total time achievable at a given cladding temperature without reaching breakaway oxidation (uncontrolled oxidation kinetics).	3.5

TABLE I–1. SUPPLEMENTARY TECHNICAL INFORMATION TO CLARIFY TERMINOLOGY AND PROVIDE ADDITIONAL BACKGROUND (cont.)

Topic	Clarification	Relevant paragraphs in the Safety Guide
Cladding	In-reactor corrosion reduces the bearing thickness of the cladding but the hydriding of the cladding, which is a consequence of the corrosion mechanism, can be more detrimental than the corrosion itself because it degrades the mechanical properties of the cladding. As a result, some fuel design limits, such as those for a reactivity initiated accident and a loss of coolant accident, are now expressed as a function of the pre-transient hydrogen content of the cladding rather than the amount of corrosion or the burnup levels.	3.51
	Corrosion and hydriding strongly depend on the material performance and on operating conditions, such as temperature, coolant chemistry and linear heat generation rates (governing, for a given fuel discharge burnup, the allowable irradiation time). These environmental conditions need to be considered. In order not to degrade the corrosion performance of the materials, appropriate water chemistry needs to be implemented (e.g. by maintaining a low oxygen content and the appropriate pH level).	3.52

TABLE I–1. SUPPLEMENTARY TECHNICAL INFORMATION TO CLARIFY TERMINOLOGY AND PROVIDE ADDITIONAL BACKGROUND (cont.)

Topic	Clarification	Relevant paragraphs in the Safety Guide
Control	The types of reactivity control devices used for regulating the core reactivity and the power distribution for different reactor designs are listed in the following. — For pressurized water reactors: • The use of solid neutron absorber rods; • The use of soluble absorber in the moderator or coolant; • The use of fuel with a distributed or discrete burnable absorber; • The use of a batch refuelling and loading pattern. — For boiling water reactors: • The use of solid neutron absorber blades; • Control of the coolant flow (moderator density); • The use of fuel with a distributed or discrete burnable absorber; • The use of a batch refuelling and loading pattern. — For pressurized heavy water reactors: • The use of solid neutron absorber rods; • The use of a soluble absorber in the moderator; • Control of the moderator temperature; • Control of the moderator height (for older pressure tube type pressurized heavy water reactors); • The use of liquid absorber in tubes; • The use of at-power refuelling.	3.23, 3.94

TABLE I–1. SUPPLEMENTARY TECHNICAL INFORMATION TO CLARIFY TERMINOLOGY AND PROVIDE ADDITIONAL BACKGROUND (cont.)

Topic	Clarification	Relevant paragraphs in the Safety Guide
Coolant	Chemical additives to the coolant (e.g. boric acid for pressurized water reactors) could be used as neutron absorbers to provide a second system of control over core reactivity. Other chemical additives (e.g. Zn, H, Li, Cu) can also be used to control the chemistry of the coolant (e.g. to control pH and oxygen content) in order to inhibit corrosion of, or crack propagation in, core components and reactor internals, and, thereby, to reduce contamination of the reactor coolant system through crud generation.	3.6
	Reactor coolant activity is measured by a device belonging to the primary coolant makeup and water cleaning system (for details, see IAEA Safety Standards Series No. SSG-56, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants [I–2]).	3.130(f)
Core components	Core components are the elements of a reactor core, other than fuel assemblies, that are used to provide structural support for the core construction, or the tools, devices or other items that are inserted into the reactor core for core monitoring, flow control or other technological purposes and are treated as core elements. Examples of core components are reactivity control devices or shutdown devices, neutron sources, dummy fuel, fuel channels, instrumentation, flow restrictors and burnable absorbers [I–1].	3.77

TABLE I–1. SUPPLEMENTARY TECHNICAL INFORMATION TO CLARIFY TERMINOLOGY AND PROVIDE ADDITIONAL BACKGROUND (cont.)

Topic	Clarification	Relevant paragraphs in the Safety Guide
Defective fuel	The iodine spiking phenomenon after plant transients has received particular attention in safety evaluations. For particular pre-accident conditions, the occurrence of this phenomenon may increase the radiological consequences of the postulated accident. One approach in safety evaluation is to specify a limit to the amount of iodine activity allowed in the reactor coolant after plant transients. The behaviour of leaking fuel rods in design basis accidents (e.g. loss of coolant accidents, reactivity initiated accidents and steam generator tube rupture) may be specific and may need to be assessed individually. For example, margins for a loss of coolant accident might not be affected by the presence of leaking fuel because conservative assumptions are specified as requirements for the evaluation of radiological consequences. Design limits for reactivity initiated accidents might not be affected by the presence of a limited number of leaking fuel rods, although it is recognized that a leaking fuel rod has a lower capability to withstand reactivity initiated accident loadings and, consequently, has a higher probability to cause limited fuel coolant interaction.	3.162
	In boiling water reactors, it is often possible to locate the region or regions in the core that contain defective fuel by using the flux tilting method. Once those regions have been identified, it is possible to reduce the power of fuel assemblies in those regions through the selective placement of control blades. In pressurized heavy water reactors, defective fuel assemblies can be detected and located by means of tracing fission product elements and delayed neutrons. Operation of a reactor with a defective fuel core could be continued, at a reduced operating power level, without significant iodine spikes until the defective fuel assemblies are discharged from the reactor. The effectiveness of the mitigation provided by power suppression is greatest when applied to relatively small, localized defects. For this reason, detection and suppression need to be undertaken as early as possible once the presence of a leaking fuel rod is indicated in a core.	3.163

TABLE I–1. SUPPLEMENTARY TECHNICAL INFORMATION TO CLARIFY TERMINOLOGY AND PROVIDE ADDITIONAL BACKGROUND (cont.)

Topic	Clarification	Relevant paragraphs in the Safety Guide
Fuel	‘Fuel’ means the fuel matrix, fuel rods and/or fuel assemblies, unless otherwise specified.	1.4
	The ‘fuel rod’ refers to the fuel element, fuel pin or any structure containing fuel pellets.	1.4
	The ‘fuel assembly’ is also called the fuel bundle for pressurized heavy water reactors.	1.4
	Innovative fuel material (e.g. enhanced accident tolerant fuel with a focus on a more benign steam reaction and lower hydrogen generation) are under development.	1.6
	‘Fuel matrix’ refers to the structure and microstructure of various types of ceramic fuel pellet.	2.4
	Examples of the materials used in fuel pellets include: — Enriched UO_2; — Natural UO_2 (for use in pressurized heavy water reactors); — Mixed oxide (UO_2–PuO_2); — Thorium based fuel (e.g. ThO_2, thorium blended UO_2, thorium blended mixed oxide fuel); — Reprocessed UO_2; — Doped fuel pellets (e.g. Cr, Al, Si) to improve their performance (for use in light water reactors). Burnable absorber material (e.g. Gd, Dy, B, Er) is blended in sintered UO_2 pellets or coated on their surface, for example, to temporarily suppress the excess reactivity resulting from a high concentration of fissile material in the fuel.	3.4

TABLE I–1. SUPPLEMENTARY TECHNICAL INFORMATION TO CLARIFY TERMINOLOGY AND PROVIDE ADDITIONAL BACKGROUND (cont.)

Topic	Clarification	Relevant paragraphs in the Safety Guide
Fuel	The term ‘hot rod’ refers to the fuel rod with the highest relative power, considering the conservative radial core power distribution.	3.33(a)
	In the assessment of the peak fuel temperatures for operational states, the following burnup dependent phenomena need to be addressed: changes in fuel thermal conductivity and diffusivity and in the thermal conductance of the pellet–cladding gap, fuel densification, fuel swelling, accumulation of fission products in the fuel pellets, fission gas release in the free volumes of the fuel rods and any other changes in the pellet microstructure. Owing to irradiation effects, the fuel melting temperature varies as a function of fuel burnup and, thus, needs to be determined using representative irradiated fuel samples.	3.40
	Isotopic composition and plutonium content in mixed oxide fuel strongly depend on the fuel discharge burnup of spent fuel assemblies from which plutonium has been extracted. The ratio of fissile isotopes for the plutonium also varies; this will affect the characteristics of the reactor core. In addition, the Pu vector (^{238}Pu, ^{239}Pu, ^{240}Pu, ^{241}Pu, ^{241}Am) needs to be incorporated into the design of mixed oxide cores, recognizing that there are changes that affect reactivity and the nuclear key safety parameters as a function of the startup time after fabrication of the mixed oxide fuel. These features need to be taken into account in core design analyses and safety analyses.	3.157(a)
Fuel channel	In boiling water reactors, the pressure difference between the inside and the outside of the boundary of the fuel channel could induce bowing and bulging of the fuel channel. This deformation, as well as fuel cladding bowing, could, consequently, increase the local flux peaking and cause friction affecting the movement of the reactivity control assembly.	3.57(a)

TABLE I–1. SUPPLEMENTARY TECHNICAL INFORMATION TO CLARIFY TERMINOLOGY AND PROVIDE ADDITIONAL BACKGROUND (cont.)

Topic	Clarification	Relevant paragraphs in the Safety Guide
Load following	The term ‘load following’ means operation of the reactor so that electricity generation will match a varying electricity demand. Load following implies operation with power manoeuvres at levels less than the rated thermal power, so that the total amount of electrical energy output is less than if the unit were operated at a relatively constant base load. Load following operation could require increased maintenance and monitoring, and may complicate the reliability and ageing assessments of some structures, systems and components.	3.158
	During load following operation, power density redistributions are caused promptly by control rod movement, but they then enable inherent subsequent redistribution processes via feedback effects linked to the reactor coolant conditions and the xenon distribution. This generates changes in the power density distribution characterized by higher peak power densities (and/or a lower departure from nucleate boiling ratios) compared to initial unperturbed conditions.	3.161
Margin	In the context of this Safety Guide, ‘margin’ refers to the difference between the design limit defined for a specific physical parameter and the extremum (minimum or maximum) value of this physical parameter.	2.13
	The term ‘design limits’ is used in this Safety Guide to cover the commonly used terms ‘safety limits’, ‘operational limits and conditions’ and ‘acceptance criteria’ as defined in the IAEA Safety Glossary [I–1].	

TABLE I–1. SUPPLEMENTARY TECHNICAL INFORMATION TO CLARIFY TERMINOLOGY AND PROVIDE ADDITIONAL BACKGROUND (cont.)

Topic	Clarification	Relevant paragraphs in the Safety Guide
Margin	The term ‘shutdown margin’ means the instantaneous amount of reactivity by which a reactor remains subcritical from its present condition, assuming all full length control rod cluster assemblies (for pressurized water reactors) or full length control rods (for boiling water reactors and pressurized heavy water reactors) are fully inserted, except for the one assembly or rod exhibiting the highest reactivity worth that is assumed to be fully withdrawn.	3.24
Pellet–cladding interaction	Cladding creepdown and fuel pellet thermal expansion and gaseous swelling will lead to a strain driven pellet–cladding mechanical interaction in all applicable plant states. The failure mode via pellet–cladding mechanical interaction is by ductility exhaustion of the cladding.	3.59
	Stress corrosion cracking in the fuel cladding occurs when the stresses on the inner surface of the cladding (as a result of pellet–cladding interaction) reach a certain limit in a corrosive environment. After a power reduction, the thermal contraction of the fuel pellets causes reopening of the pellet–cladding gap (or the gaps between the pellet fragments). If operation at reduced power is maintained long enough (i.e. extended reduced power operation), the fuel cladding will creep down and close the gaps again. The fuel rod is then considered ‘re-conditioned’ at this lower power level. When the reactor core returns to full power at a later time, tensile stresses will appear in the cladding. These residual stresses will increase the susceptibility to stress corrosion cracking driven by the pellet–cladding interaction under the corrosive fission product environment in the fuel rod.	3.48, 3.61

TABLE I–1. SUPPLEMENTARY TECHNICAL INFORMATION TO CLARIFY TERMINOLOGY AND PROVIDE ADDITIONAL BACKGROUND (cont.)

Topic	Clarification	Relevant paragraphs in the Safety Guide
Pellet–cladding interaction	The power-ramp failure threshold is the lower bound of the power-ramp induced fuel failures within a burnup range called the ‘critical burnup range’. For fuel burnup values below the critical burnup range, the pellet–cladding gap remains open, so that the power change would have to be larger to reach the same level of stress in the cladding compared to the situation in which the pellet–cladding gap is closed. For fuel burnup values above the critical burnup range, experience shows that the material compound in the pellet–cladding interface generated by irradiation is such that the stress concentration on the inner surface of the cladding is reduced, making stress corrosion cracking in the cladding unlikely. Since the critical burnup range depends on the gap closure kinetics of the pellet–cladding gap, it is, therefore, dependent on the specific material properties of the cladding type and fuel rod design.	3.63
Reactivity feedback	The inherent neutronic characteristics of the reactor core are typically characterized by the following types of reactivity feedback or coefficient: — Reactivity feedback due to fuel temperature changes (i.e. the temperature coefficient of reactivity for the fuel or the Doppler coefficient); — Reactivity feedback due to coolant or moderator temperature changes, including related coolant or moderator density (i.e. temperature coefficients of reactivity for the coolant and the moderator); — Reactivity feedback due to changes in the void fraction in the coolant or moderator (i.e. void coefficients of reactivity for the coolant and the moderator); — Reactivity feedback due to changes of boron concentration in the coolant or moderator (i.e. boron coefficients of reactivity for the coolant and the moderator);	3.16

TABLE I–1. SUPPLEMENTARY TECHNICAL INFORMATION TO CLARIFY TERMINOLOGY AND PROVIDE ADDITIONAL BACKGROUND (cont.)

Topic	Clarification	Relevant paragraphs in the Safety Guide
Reactivity feedback	— The delayed neutron fraction and the prompt neutron lifetime; — The effects of power redistribution on reactivity (e.g. the xenon efficiency and the moderator density); — Decay of xenon and other neutron absorbers in analysis of the core in the long term.	
Shutdown	The simplest common form of design for fail-safe shutdown allows the shutdown devices to be held above the core by active means. Provided that the guide structures for the shutdown devices are not obstructed, the devices will drop into the core under gravity in the event of a de-energization of the active means of holding them (e.g. a loss of current through a holding electromagnet). This does not apply for boiling water reactors.	3.112(c)
	The overall reactivity worth of the shutdown devices is a function of the spacing between the devices, as well as of their locations in the reactor. When two devices are close together, their worth is less than the sum of their individual worths.	3.115(f)

REFERENCES TO ANNEX I

[I–1] INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Glossary: Terminology Used in Nuclear Safety and Radiation Protection, 2018 Edition, IAEA, Vienna (2019).

[I–2] INTERNATIONAL ATOMIC ENERGY AGENCY, Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants, IAEA Safety Standards Series No. SSG-56, IAEA, Vienna (in preparation).

Annex II

ASPECTS TO BE ADDRESSED IN THE DESIGN OF THE FUEL ROD, FUEL ASSEMBLY, REACTIVITY CONTROL ASSEMBLY, NEUTRON SOURCE ASSEMBLY AND HYDRAULIC PLUG ASSEMBLY

FUEL ROD

II–1. The design of the fuel rod needs to address the aspects described in the following:

(a) Cladding:
- — Fuel rod vibration and wear (i.e. grid-to-rod fretting wear for light water reactors, spacer-to-spacer fretting wear for pressurized heavy water reactors);
- — Evolution of the mechanical properties of the cladding with irradiation (displacement and pressure driven loadings);
- — Materials and chemical evaluation;
- — Stress corrosion;
- — Cycling and fatigue;
- — Geometrical and chemical stability of the cladding under irradiation.

(b) Fuel material (including burnable absorbers):
- — Dimensional stability of the fuel under irradiation;
- — Fuel densification (kinetics and amplitude);
- — Potential for chemical interaction with the cladding and the coolant;
- — Fission gas generation and distribution within the fuel pellets;
- — Fission gas release kinetics;
- — Gaseous swelling;
- — Thermomechanical properties under irradiation;
- — Microstructure changes as a function of irradiation.

(c) Fuel rod performance:
- — Pellet and cladding temperatures and temperature distributions;
- — Fuel–cladding gap closure kinetics and amplitude (to address issues relating to pellet–cladding interactions);
- — Irradiation effects on fuel rod behaviour (e.g. fuel restructuring, cracking of fuel pellets, solid and gaseous fission product swelling, fission gas release and increases in internal pressure of fuel rods, degradation of thermal conductivity of fuel rods);
- — Fuel rod bowing;
- — Fuel rod growth.

Fuel rod performance is demonstrated using validated analytical models and/or representative experimental data collected either in test programmes or from other nuclear power plants (lead use fuel rods or lead use fuel assemblies). The models used are, generally, burnup dependent.

FUEL ASSEMBLY

II–2. Fuel assembly components (e.g. top and bottom nozzles, guide tubes, spacers, mixing grids, grid springs, connections and fuel assembly hold-down systems for pressurized water reactors) need to be designed to withstand the following conditions and loads:

- Core restraint system loads;
- Hydrodynamic loads;
- Thermohydraulic limits (e.g. critical heat flux);
- Accident loads (e.g. loss of coolant accident) and seismic loads;
- Handling and shipping loads;
- Fuel assembly bowing.

REACTIVITY CONTROL ASSEMBLY

II–3. The design of the reactivity control assembly needs to address the following aspects:

- Rod internal pressure and related cladding stresses during normal, transient and accident conditions;
- Thermal expansion and irradiation induced swelling;
- Evolution of absorber materials and the cladding under irradiation;
- The effect of fretting wear on cladding resistance.

NEUTRON SOURCE ASSEMBLY

II–4. The design of the neutron source assembly needs to address the following aspects:

(a) Irradiation effects;
(b) Efficiency to take into account burnup shadowing effects of peripheral fuel assemblies;

(c) External events such as earthquakes.

HYDRAULIC PLUG ASSEMBLY

II–5. The design of the hydraulic plug assembly needs to address the following aspects:

(a) Interaction with guide tubes due to thermal expansion or irradiation induced swelling;
(b) Impact on coolant bypass flow (for pressurized water reactors);
(c) The effect of fretting wear on guide tube resistance.

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