IAEA Safety Standards

for protecting people and the environment

Design of the Reactor Core for Nuclear Power Plants

Safety Guide
No. NS-G-1.12

IAEA
International Atomic Energy Agency

IAEA SAFETY RELATED PUBLICATIONS

IAEA SAFETY STANDARDS

Under the terms of Article III of its Statute, the IAEA is authorized to establish or adopt standards of safety for protection of health and minimization of danger to life and property, and to provide for the application of these standards.

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Safety standards are coded according to their coverage: nuclear safety (NS), radiation safety (RS), transport safety (TS), waste safety (WS) and general safety (GS).

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http://www-ns.iaea.org/standards/

The site provides the texts in English of published and draft safety standards. The texts of safety standards issued in Arabic, Chinese, French, Russian and Spanish, the IAEA Safety Glossary and a status report for safety standards under development are also available. For further information, please contact the IAEA at P.O. Box 100, A-1400 Vienna, Austria.

All users of IAEA safety standards are invited to inform the IAEA of experience in their use (e.g. as a basis for national regulations, for safety reviews and for training courses) for the purpose of ensuring that they continue to meet users' needs. Information may be provided via the IAEA Internet site or by post, as above, or by e-mail to Official.Mail@iaea.org.

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Safety related publications are also issued in the **Technical Reports Series**, the **IAEA-TECDOC Series**, the **Training Course Series** and the **IAEA Services Series**, and as **Practical Radiation Safety Manuals** and **Practical Radiation Technical Manuals**. Security related publications are issued in the **IAEA Nuclear Security Series**.

DESIGN OF THE REACTOR CORE
FOR NUCLEAR POWER PLANTS

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IAEA SAFETY STANDARDS SERIES No. NS-G-1.12

DESIGN OF THE REACTOR CORE FOR NUCLEAR POWER PLANTS

SAFETY GUIDE

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2005

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FOREWORD

by Mohamed ElBaradei
Director General

The IAEA's Statute authorizes the Agency to establish safety standards to protect health and minimize danger to life and property — standards which the IAEA must use in its own operations, and which a State can apply by means of its regulatory provisions for nuclear and radiation safety. A comprehensive body of safety standards under regular review, together with the IAEA's assistance in their application, has become a key element in a global safety regime.

In the mid-1990s, a major overhaul of the IAEA's safety standards programme was initiated, with a revised oversight committee structure and a systematic approach to updating the entire corpus of standards. The new standards that have resulted are of a high calibre and reflect best practices in Member States. With the assistance of the Commission on Safety Standards, the IAEA is working to promote the global acceptance and use of its safety standards.

Safety standards are only effective, however, if they are properly applied in practice. The IAEA's safety services — which range in scope from engineering safety, operational safety, and radiation, transport and waste safety to regulatory matters and safety culture in organizations — assist Member States in applying the standards and appraise their effectiveness. These safety services enable valuable insights to be shared and I continue to urge all Member States to make use of them.

Regulating nuclear and radiation safety is a national responsibility, and many Member States have decided to adopt the IAEA's safety standards for use in their national regulations. For the Contracting 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 designers, manufacturers and operators around the world to enhance nuclear and radiation safety in power generation, medicine, industry, agriculture, research and education.

The IAEA takes seriously the enduring challenge for users and regulators everywhere: that of ensuring a high level of safety in the use of nuclear materials and radiation sources around the world. Their continuing utilization for the benefit of humankind must be managed in a safe manner, and the IAEA safety standards are designed to facilitate the achievement of that goal.

IAEA SAFETY STANDARDS

SAFETY THROUGH INTERNATIONAL STANDARDS

While safety is a national responsibility, international standards and approaches to safety promote consistency, help to provide assurance that nuclear and radiation related technologies are used safely, and facilitate international technical cooperation and trade.

The standards also provide support for States in meeting their international obligations. One general international obligation is that a State must not pursue activities that cause damage in another State. More specific obligations on Contracting States are set out in international safety related conventions. The internationally agreed IAEA safety standards provide the basis for States to demonstrate that they are meeting these obligations.

THE IAEA STANDARDS

The IAEA safety standards have a status derived from the IAEA's Statute, which authorizes the Agency to establish standards of safety for nuclear and radiation related facilities and activities and to provide for their application.

The safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment.

They are issued in the IAEA Safety Standards Series, which has three categories:

Safety Fundamentals
— Presenting the objectives, concepts and principles of protection and safety and providing the basis for the safety requirements.

Safety Requirements
— Establishing the requirements that must be met to ensure the protection of people and the environment, both now and in the future. The requirements, which are expressed as 'shall' statements, are governed by the objectives, concepts and principles of the Safety Fundamentals. If they are not met, measures must be taken to reach or restore the required level of safety. The Safety Requirements use regulatory language to enable them to be incorporated into national laws and regulations.

Safety Guides
— Providing recommendations and guidance on how to comply with the Safety Requirements. Recommendations in the Safety Guides are expressed as 'should' statements. It is recommended to take the measures stated 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. Each Safety Requirements publication is supplemented by a number of Safety Guides, which can be used in developing national regulatory guides.

The IAEA safety standards need to be complemented by industry standards and must be implemented within appropriate national regulatory infrastructures to be fully effective. The IAEA produces a wide range of technical publications to help States in developing these national standards and infrastructures.
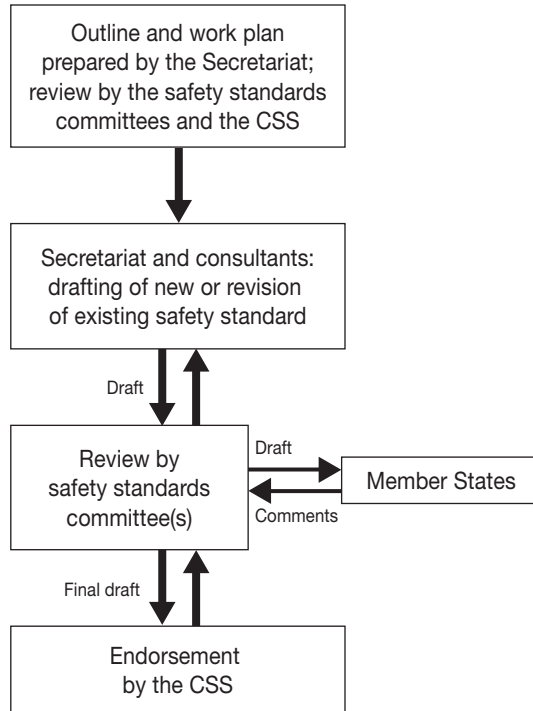
MAIN USERS OF THE STANDARDS

As well as by regulatory bodies and governmental departments, authorities and agencies, the standards are used by authorities and operating organizations in the nuclear industry; by organizations that design, manufacture and apply nuclear and radiation related technologies, including operating organizations of facilities of various types; by users and others involved with radiation and radioactive material in medicine, industry, agriculture, research and education; and by engineers, scientists, technicians and other specialists. The standards are used by the IAEA itself in its safety reviews and for developing education and training courses.

DEVELOPMENT PROCESS FOR THE STANDARDS

The preparation and review of safety standards involves the IAEA Secretariat and four safety standards committees for safety in the areas of 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 entire safety standards programme. All IAEA Member States may nominate experts for the safety standards committees and may provide comments on draft standards. The membership of the CSS is appointed by the Director General and includes senior government officials having responsibility for establishing national standards.

For Safety Fundamentals and Safety Requirements, the drafts endorsed by the Commission are submitted to the IAEA Board of Governors for approval for publication. Safety Guides are published on the approval of the Director General.

Through this process the standards come to represent a consensus view of the IAEA's Member States. 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 standards. Some 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 International

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

Labour Organization, the OECD Nuclear Energy Agency, the Pan American Health Organization and the World Health Organization.

The safety standards are kept up to date: five years after publication they are reviewed to determine whether revision is necessary.

APPLICATION AND SCOPE OF THE STANDARDS

The IAEA Statute makes the safety standards binding on the IAEA in relation to its own operations and on States in relation to operations assisted by the IAEA. Any State wishing to enter into an agreement with the IAEA concerning any form of Agency assistance is required to comply with the requirements of the safety standards that pertain to the activities covered by the agreement.

International conventions also contain similar requirements to those in the safety standards, and make them binding on contracting parties. The Safety Fundamentals were used as the basis for the development of the Convention on Nuclear Safety and the Joint Convention on the Safety of Spent Fuel Management and on the Safety of Radioactive Waste Management. The Safety

Requirements on Preparedness and Response for a Nuclear or Radiological Emergency reflect the obligations on States under the Convention on Early Notification of a Nuclear Accident and the Convention on Assistance in the Case of a Nuclear Accident or Radiological Emergency.

The safety standards, incorporated into national legislation and regulations and supplemented by international conventions and detailed national requirements, establish a basis for protecting people and the environment. However, there will also be special aspects of safety that need to be assessed case by case at the national level. For example, many of the safety standards, particularly those addressing planning or design aspects of safety, are intended to apply primarily to new facilities and activities. The requirements and recommendations specified in the IAEA safety standards might not be fully met at some facilities built to earlier standards. The way in which the safety standards are to be applied to such facilities is a decision for individual States.

INTERPRETATION OF THE TEXT

The safety standards use the form 'shall' in establishing international consensus requirements, responsibilities and obligations. Many requirements are not addressed to a specific party, the implication being that the appropriate party or parties should be responsible for fulfilling them. Recommendations are expressed as 'should' statements, indicating an international consensus that it is necessary to take the measures recommended (or equivalent alternative measures) for complying with the requirements.

Safety related terms are to be interpreted as stated in the IAEA Safety Glossary (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 within the 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 main text (e.g. material that is subsidiary to or separate from the main text, is included in support of statements in the main text, or describes methods of calculation, experimental 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 standard. Material in an appendix has the same status as the main 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. An annex is not an integral part of the main text. Annex material published by the IAEA is not necessarily issued under its authorship; material published in standards that is under other authorship may be presented in annexes. Extraneous material presented in annexes is excerpted and adapted as necessary to be generally useful.

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide was prepared under the IAEA programme for establishing safety standards for nuclear power plants. The basic requirements for the design of safety systems for nuclear power plants are provided in the Safety Requirements publication, Safety Standards Series No. NS-R-1: Safety of Nuclear Power Plants: Design [1], which it supplements. This Safety Guide recommends how to meet the requirements for the design of the reactor core for nuclear power plants.

1.2. The publication is a revision of a previous Safety Guide on Design for Reactor Core Safety in Nuclear Power Plants, issued in 1986 as Safety Series No. 50-SG-D14, which is superseded by the present Safety Guide. This revision takes account of developments in the design of the reactor core since the earlier Safety Guide was issued and includes recommendations and guidance on general and specific design considerations.

OBJECTIVE

1.3. The objective of this Safety Guide is to make recommendations concerning safety features for incorporation into the design of the reactor core for a nuclear power plant. The Safety Requirements publication on Safety of Nuclear Power Plants: Design establishes general safety requirements for design. The present Safety Guide provides recommendations and guidance on the interpretation and implementation of these requirements. It should be noted that nuclear safety is achieved through a combination of proper design, manufacture, construction and operation. For operational aspects, reference should be made to the Safety Requirements publication on Safety of Nuclear Power Plants: Operation [2] and its associated Safety Guides.

SCOPE

1.4. In Safety of Nuclear Power Plants: Design [1] it is stated that:

"It is expected that this publication will be used primarily for land based stationary nuclear power plants with water cooled reactors designed for

electricity generation or for other heat production applications (such as district heating or desalination). It is recognized that in the case of other reactor types, including innovative developments in future systems, some of the requirements may not be applicable, or may need some judgement in their interpretation."

1.5. This Safety Guide covers the neutronic, thermal, hydraulic, mechanical, chemical and irradiation considerations that are important for the safe design of the core of a nuclear power reactor. This Safety Guide further deals with the individual systems and components that make up the core, the equipment associated with the core and the design provisions for the safe operation of the core. The safe handling of the fuel and other core components is discussed in Ref. [3].

1.6. This Safety Guide covers the internals of the reactor vessel and the devices[1] mounted on the reactor vessel for reactivity control and shutdown. Interactions of these internals and devices with the reactor coolant and the reactor coolant system components, including its pressure boundary (i.e. including the pressure vessel or the pressure tubes), are considered only to the extent necessary to clarify the interface with the Safety Guide on the Design of the Reactor Coolant System and Associated Systems in Nuclear Power Plants [4] and other Safety Guides. The guidance in relation to instrumentation and control systems is limited mainly to functional considerations.

1.7. The following structures, systems and components in the reactor vessel are considered in this Safety Guide:

— The *fuel assemblies* and those structures that hold the fuel assemblies and other components in a predetermined geometrical configuration. The moderator and the coolant within the core are also considered.
— The *components and structures* used for reactivity control and shutdown, comprising the neutron absorbers (solid or liquid), the associated structure and drive mechanism, and related components of the fluid system.

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[1] In this Safety Guide the term 'device' (e.g. shutdown device or reactivity control device) is used to designate a part (regardless of its shape, material or purpose) that is inserted into the core (e.g. a control rod or a tube containing fluid for reactivity control). The term 'device' may include the drive mechanism for the parts. By contrast, the term 'means' (e.g. means of shutdown or means of reactivity control) is used to denote more broadly the functional aspect of the component or system.

— The *support structures* that provide the foundation for the core within the reactor vessel, the structure for guiding the flow, such as the core barrel or the pressure tubes of a pressurized heavy water reactor (PHWR, pressure tube type), and guide tubes for reactivity control devices.
— Other *reactor vessel internals* such as instrumentation tubes, in-core instrumentation for core monitoring, steam separators and neutron sources. These are dealt with to a limited extent only in this Safety Guide.

STRUCTURE

1.8.   Section 2 describes basic considerations for safe core design in addition to the requirements established in Ref. [1]. Section 3 discusses detailed considerations in safe design for each reactor core component and the related design for safe operation. Section 4 describes qualification and testing. Section 5 addresses quality assurance in design. Some specific technical aspects are discussed in Appendices I to V.

2. GENERAL SAFETY CONSIDERATIONS IN DESIGN

GENERAL

2.1.   The safety objectives and concepts, as presented in Ref. [5] and reproduced in Ref. [1], Section 2, are as follows:

"General Nuclear Safety Objective: To protect individuals, society and the environment from harm by establishing and maintaining in nuclear installations effective defences against radiological hazards.

"Radiation Protection Objective: To ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accidents.

"Technical Safety Objective: To take all reasonably practicable measures to prevent accidents in nuclear installations and to mitigate their consequences

should they occur; to ensure with a high level of confidence that, for all possible accidents taken into account in the design of the installation, including those of very low probability, any radiological consequences would be minor and below prescribed limits; and to ensure that the likelihood of accidents with serious radiological consequences is extremely low.

"Safety Objectives require that nuclear installations are designed and operated so as to keep all sources of radiation exposure under strict technical and administrative control. However, the Radiation Protection Objective does not preclude limited exposure of people or the release of legally authorized quantities of radioactive materials to the environment from installations in operational states. Such exposures and releases, however, must be strictly controlled and must be in compliance with operational limits and radiation protection standards."

The requirements for the safety concept of defence in depth are also established in Ref. [1] (para. 4.1). In summary, application of the concept of defence in depth in the design of a nuclear power plant results in a number of levels of defence (inherent features, equipment and procedures) aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails.

2.2. Three types of event are required to be prevented to the extent practicable in the design of the core (Ref. [1], para. 4.2) to ensure that defence in depth is maintained:

"(1) challenges to the integrity of physical barriers;
(2) failure of a barrier when challenged;
(3) failure of a barrier as a consequence of failure of another barrier."

For example, the plant should be designed in such a way that: the fuel pellets do not release an inordinate amount of radioactive fission products; the fuel does not challenge the integrity of the fuel cladding; the integrity of the fuel cladding is maintained under all operating conditions and under transient conditions as far as practicable; and failure of fuel cladding does not propagate and result in failure of the reactor vessel or of pressure tubes. Any resulting effects of core behaviour on other barriers should also be considered to ensure that the core design fully meets the intent of the requirements [1].

2.3. The three fundamental safety functions should be considered in the design of the core, for operational states and a wide range of accident conditions:

(1) Control of the reactivity;
(2) Removal of heat from the core;
(3) Confinement of radioactive material and control of operational discharges, as well as limitation of accidental radioactive releases.

The performance of these functions should be ensured by means of appropriate design of the core and its associated systems to ensure that they are capable and robust (Ref. [1], para. 4.6).

2.4. The core design should be such as to minimize the adverse effects of a wide range of postulated initiating events (see para. 4.7 and Appendix I of Ref. [1]; see also under the heading Safety Analysis in Section 3 of this Safety Guide).

2.5. In summary, the following should be taken into account in considering general design features: appropriate safety margins; safety standards on radiation protection [6–8]; static and dynamic loadings; the maximum rate and amount of addition of reactivity; post-accident recriticality; and inspection and testing throughout the plant's operating lifetime (Ref. [1], paras 6.1–6.5).

2.6. Structures, systems and components of the core should be designed, fabricated, erected, constructed, tested and inspected in accordance with appropriate national or international engineering codes and standards or practices (Ref. [1], para. 3.6).

2.7. The core design needs to be reviewed and, if necessary, modified accordingly when a significant configuration change occurs during the plant's operating lifetime, as a result of, for example:

   — The use of mixed uranium oxide and plutonium oxide (mixed oxide) fuel;
   — An increase in burnup for a discharged fuel assembly;
   — An increase in the duration of a fuel cycle;
   — An increase in the rated power of the plant.

2.8. For operational states (normal operation and anticipated operational occurrences), it is required to maintain the integrity of fuel elements; for design basis accidents it is required to ensure that any damage to fuel elements is kept to a minimum (Ref. [1], paras 6.1 and 6.35). Components of the reactor core and its associated structures should be designed with account taken of the safety functions to be achieved during and following design basis accidents (e.g. shutdown of the reactor, emergency core cooling, long term stable cooling and

reactivity control, and confinement of radioactive material). The approach to mitigating beyond design basis accidents is outlined in Ref. [1], para. 5.31.

2.9. Suitable provision is required to be made in the design and layout of the structures, systems and components of the reactor core to ensure that radiation doses to the public and to site personnel in all operational states, including maintenance, testing and inspection and decommissioning, do not exceed authorized limits and are as low as reasonably achievable [1, 5–7].

2.10. Additional requirements for core design are given in Ref. [1], paras 6.6–6.20 under the headings fuel elements and assemblies, control of the reactor core and reactor shutdown. The following paragraphs give related guidance on design.

NEUTRONIC DESIGN

2.11. The design of the reactor core should be such 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, its thermal-hydraulic characteristics and the capability of the control and shutdown systems to actuate for all operational states and in design basis accident conditions. When rapid acting control or shutdown systems are necessary, their capabilities (e.g. speed and reliability) should be fully justified. (Further information on inherent neutronic characteristics and reactivity coefficients is given in Appendix I.)

2.12. In accordance with para. 4.6 of Ref. [1], the maximum insertion rate for positive reactivity in operational states and in design basis accidents should be limited in such a way that the means of reduction of reactor power described in para. 2.11 are effective in maintaining core coolability, minimizing damage to the core and preventing failure of the pressure boundary for the reactor coolant. The design basis for fuel elements should be adequate to prevent undesired consequences of reactivity initiated accidents (e.g. by means of limits on maximum fuel enthalpy or rise in fuel enthalpy).

2.13. At least two independent and diverse shutdown systems should be provided (see paras 6.3–6.20 of Ref. [1] and the discussion in Section 3 on means of reactor shutdown for further information).

2.14. Calculation of the core power distribution should be performed in the design for representative operational states to provide information for use in determining: (a) operational limits and conditions; (b) action set points for safety protection systems; (c) operating procedures that will ensure compliance with design limits, including core design parameters, throughout the service life of the reactor core.

2.15. Reactivity control devices should be used to maintain the reactor in a subcritical condition, with account taken of possible design basis accidents and their consequences. Adequate provision should be made 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, for example when the reactor vessel is open for maintenance or refuelling.

2.16. The shutdown systems should be testable, as far as practicable, during operation in order to provide assurance that the systems are available on demand.

THERMAL-HYDRAULIC DESIGN

2.17. Thermal-hydraulic design limits on parameters such as the maximum linear heat generation rate, the minimum critical power ratio, the minimum departure from nucleate boiling ratio, the peak fuel temperature and the cladding temperature should be set in such a way that there are sufficient margins in operational states to keep the failure rates of fuel elements under design basis accident conditions to an acceptably low level.

2.18. Suitable means of instrumentation and control should be provided so that parameters indicative of the core conditions (e.g. the rate of coolant flow, the coolant temperature and the neutron flux) can be monitored and adjusted safely to ensure that the design limits are not exceeded for all operational states, including refuelling.

2.19. Suitable instrumentation for monitoring is required to be provided for assessing the status of the core and associated features under accident conditions (see Ref. [1], para. 6.68).

MECHANICAL DESIGN

2.20. The fuel elements, control devices, burnable poisons and fuel assemblies should be designed to ensure that the cladding remains leaktight for all operational states throughout its lifetime.

2.21. In the design of the fuel elements, control devices, burnable poisons and fuel assemblies, account should be taken of the effects of temperature, pressure, irradiation, fission products, static and dynamic mechanical loads, including seismic loads, flow induced vibration and changes in the chemical characteristics of the constituent materials.

2.22. Means should be provided for safe handling of core components such as the fuel assemblies, control and shutdown devices and core support structures to ensure their integrity in transport, storage, installation and refuelling operations (see Ref. [3]).

2.23. The structural integrity of the core should be ensured so that the core can be safely controlled, shut down and cooled under operational states and in design basis accident conditions. Static and dynamic mechanical loads, including thermal stress, acting in operational states and in design basis accident conditions should be considered.

2.24. The fuel assembly, other reactor vessel internals and the reactor cooling system should be designed to minimize the chance of any obstruction of the coolant flow due to the release of loose parts, so as to prevent core damage in any operational state and in design basis accident conditions.

2.25. The core and its associated components should be designed to be compatible under the effects of irradiation and chemical and physical processes.

2.26. The uncontrolled movement of reactivity control devices should be prevented.

SAFETY CLASSIFICATION ASPECTS OF CORE DESIGN

2.27. In the light of the Safety Requirements (Ref. [1], paras 5.1–5.3), and the central importance of the reactor core to safety, it is expected that all

components of the reactor core and its associated structures should be classified appropriately according to their importance to safety.

2.28. Two safety barriers (the fuel itself and its cladding) are located inside the core, and the performance of two fundamental safety functions (reactivity control and core cooling) depends on the maintenance of the correct core configuration and the accessibility of devices. Furthermore, if an accident that is disruptive of the core occurred, it would have the potential to challenge the containment barrier. Proper safety classification and consequently appropriate analysis, judgement and verification should therefore be performed to ensure the integrity of the core and to maintain the performance of the fundamental safety functions of reactivity control and core cooling.

2.29. The consequences of the failure of each system and component, including the consequences of its possible effects on other systems and components, should be analysed carefully to determine an appropriate classification.

3. SPECIFIC SAFETY CONSIDERATIONS IN DESIGN

GENERAL

3.1. This section addresses further design considerations for the core components in meeting the objectives and requirements for design mentioned in Section 2. It also covers core management, which strongly influences the core design with regard to the integrity of fuel elements as well as the fuel economy.

3.2. Some devices in a core may also perform safety functions that are within the scope of other Safety Guides. In the design of such hardware, account should be taken of the recommendations and guidance of the present Safety Guide and other relevant Safety Guides, such as Refs [3, 4, 9].

FUEL ELEMENTS AND ASSEMBLIES

3.3. Considerations in this subsection apply to both uranium fuel and mixed oxide fuel.

Thermal and burnup effects

3.4.   In the evaluation of the temperatures of fuel pellets in operational states, account should be taken of the changes in the thermal conductivity of the pellets and in the thermal conductance of the gap between pellet and cladding due to burnup dependent effects such as oxide densification, swelling, accumulation of fission products and other changes in the microstructure of pellets. In determining the melting temperature of fuel, the changes in the composition and microstructure of the fuel due to burnup effects should be taken into account. In all operational states, the peak fuel temperature should be lower than the fuel melting temperature by a sufficient margin to prevent melting of the fuel, with allowance made for uncertainties.

3.5.   In the design of fuel, account should be taken of changes in mechanical properties (strength, creep and stress relaxation) and changes in the corrosion related behaviour of the cladding with temperature. Limits for stress, long term deformation and corrosion may therefore be specified for different operational states. Stressing and straining of the cladding can be caused by the swelling or thermal expansion of the fuel due to an increase in the local power or the internal gas pressure, and the stresses and strains should be limited.

3.6.   In the design of fuel elements, account should be taken of the effects of solid and gaseous fission products, the release rates of which depend largely on the power history during their in-core residence. The effects of gaseous fission products on the internal pressure of a fuel element and the thermal conductance of the pellet-to-cladding gap should be considered. The corrosive effects of fission products on the cladding should also be considered in the design. Swelling of the fuel material as a consequence of the formation of fission products causes changes in its material properties, such as thermal conductivity, and dimensional changes, and these changes should be taken into account in the design.

3.7.   The consequences of reactor depressurization events (in normal operation and following anticipated operational transients such as those initiated by the automatic depressurization system) should be considered in safety analyses in terms of the potential for failure of the cladding and a resulting release of fission products from the fuel.

Effects of irradiation

3.8.   The effects of irradiation, in particular the effects of fast neutrons on fuel assemblies (including control devices and burnable poisons), on metallurgical properties such as the tensile strength of the cladding, ductility and creep behaviour, fuel densification and swelling (in radial and axial directions), and on the geometrical stability of all materials should be considered in the design.

Effects of variations in power levels

3.9.   Account should be taken in the design of the effects on the integrity of the fuel cladding of local and global power transients due to fuel shuffling, movements of control devices or other reactivity changes. One possible effect of these power variations is cracking of the cladding due to pellet–cladding interactions (see Appendix II).

3.10. The power distribution in the core and the fuel assemblies changes during the fuel cycle owing to the burnup of fuel. Accordingly, the excess reactivity of the core and the reactivity coefficients of the core also change. These phenomena should be taken into account in the design of the core and the fuel.

3.11. The effects of anticipated power transients on the peak heating rates should be taken into account in design of the core and the fuel.

Mechanical effects in fuel elements

3.12. Stress corrosion cracking induced by pellet–cladding interactions in the presence of fission products should be minimized. The control of pellet–cladding interactions is described in Appendix II. Considerations of mechanical safety in the design of fuel assemblies are discussed later in this section.

3.13. The fuel cladding can be designed to be collapsible or free standing when subjected to the coolant operating pressure. Free standing cladding can undergo long term deformation (creep deformation) under external pressure, leading to a decrease in the radial gap between the cladding and fuel. Some cladding that is initially free standing will eventually collapse and be supported by the pellets. Collapsible cladding is rapidly pressed onto the fuel pellets by the external pressure and the outer cooler region of the fuel pellet supports the cladding throughout its lifetime. The radial gap between collapsible cladding

and fuel pellets should be limited to prevent the formation of excessive longitudinal ridges in the cladding.

3.14. Stressing and straining of the cladding can be caused by the swelling or thermal expansion of the fuel due to an increase in local power or by an increase in the internal gas pressure. Such stressing and straining of the cladding should be limited so that the safe performance of the fuel is not compromised.

3.15. The mechanical loading of the cladding due to the length of unsupported plenum, axial gaps between fuel pellets, fuel densification or other causes should be considered in the design of the fuel.

Effects of burnable poison in the fuel

3.16. Due consideration should be given to the potential adverse effects of burnable poisons in the fuel on its thermal properties, and on the chemical, mechanical and metallurgical properties of the fuel and cladding material. The possibility should be considered that the release of volatile fission products from the fuel pellet may increase owing to the addition of burnable poisons. The effects of the burnable poisons on the core reactivity, on the temperature coefficients of reactivity of the fuel and the moderator and on local power peaking factors should be taken into account.

Corrosion and hydriding of fuel elements

3.17. Fuel assemblies should be designed to be compatible with the coolant environment in all operational states, including shutdown and refuelling. Corrosion and hydriding depend strongly on the properties of the cladding material and on the temperature, the presence of oxides, and stresses and strains. The environmental conditions for liquid coolant, such as conditions of water purity, local boiling, pressure, temperature and fluid chemistry should be taken into account. In practice, corrosion is controlled by means of appropriate water chemistry (i.e. by maintaining a low oxygen content and the appropriate pH).

3.18. Oxidation or other chemical changes tending to lead to the formation of deposits on the surface of the cladding may affect the transfer of heat from the fuel element and so should be taken into consideration in the thermal-hydraulic analysis. In determining the ranges of the parameters for coolant design for operational states, the consequential effects on the surface oxidation

and the buildup of deposits should be taken into account. High burnup fuel (see Appendix IV) necessitates additional considerations in the design to limit surface oxidation in normal operation. The design parameters used should be based on operating experience and/or experiments that are appropriate for the operating conditions.

3.19. The hydrogen content of zirconium alloy cladding should be limited to reduce the likelihood of fuel defects being caused by the embrittlement of cladding during operation. For this purpose, the moisture content in the free space within a fuel element should be controlled.

Thermal-hydraulic effects in fuel assemblies

3.20. Effects that depend on the fuel element spacing, the fuel element power, the sizes and shapes of subchannels, grids, spacers, braces, flow deflectors or turbulence promoters should be taken into account in the design. These effects are primarily thermal-hydraulic but they may potentially include localized corrosion, erosion, flow induced vibration and fretting. Steady state power should be maintained at levels that allow for certain ratios or margins to avoid critical heat flux conditions. The margins should be sufficient to allow for anticipated operational occurrences.[2] The ratios between critical and actual parameters may be expressed as a minimum critical heat flux ratio, a minimum departure from nucleate boiling ratio, a minimum critical channel power ratio or a minimum critical power ratio. These ratios lead to a conservative design basis for operational states for water cooled reactors.

3.21. The critical heat flux (CHF) and the critical power ratio (CPR) are influenced by the detailed design of the fuel assembly and by coolant conditions (e.g. by local effects due to fuel element spacers and by local subcooling and/or coolant quality). Experiments should be conducted over the range of expected operational conditions to provide data for defining the limiting values of CHF, the departure from nucleate boiling ratio or CPR in the design and for the purposes of safety analysis.

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[2] The objective of this recommendation is to avoid cladding failures caused by high cladding temperatures. In some designs, critical heat flux conditions during transients can be tolerated if it can be shown by other methods that the cladding temperatures do not exceed the acceptable limits.

Considerations of mechanical safety in the design

3.22. The fuel assembly is subjected to mechanical stresses as a result of:

— Fuel handling and loading;
— Power variations;
— Holddown loads for PWRs;
— Temperature gradients;
— Hydraulic forces, including cross-flows between open fuel assemblies;
— Irradiation (e.g. radiation induced growth and swelling);
— Vibration and fretting induced by coolant flow;
— Creep deformation;
— External events such as earthquakes;
— Postulated initiating events such as a loss of coolant accident.

3.23. For operational states, the design considerations for the fuel assembly (which may contain housings for control devices, flux monitors and burnable poison rods) include the following:

(a) The clearance within and adjacent to the fuel assembly should provide space to allow for irradiation growth and swelling[3];
(b) Bowing of fuel elements should be limited so that thermal-hydraulic behaviour and fuel performance are not significantly affected;
(c) Strain fatigue should not be able to cause the failure of a fuel assembly;
(d) The fuel assembly should be able to withstand the mechanical and hydraulic holddown forces without unacceptable deformation;
(e) The performance of the functions of the fuel assembly and the support structure should not be unacceptably affected by damage due to vibration or fretting;
(f) The fuel assembly should be able to withstand irradiation and its materials should be compatible with the chemical properties of the coolant;
(g) Any deformation of the fuel element or the fuel assembly should not affect the capability for the insertion of control rods for the safe shutdown of the reactor.

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[3] In BWRs, the pressure difference between the inside and outside of the boundary of the channel box may induce swelling of the box. This deformation, as well as fuel bowing, may consequently increase the local flux peaking factor.

3.24. In plant states ranging from anticipated operational occurrences to design basis accident conditions, the fuel elements, fuel assemblies and fuel assembly support structures should be designed to ensure that any interactive or consequential effects of these components will not prevent safety systems from performing their functions as claimed in the safety analysis by:

— Preventing the functioning of components of safety systems (e.g. shutdown devices and their guide tubes);
— Impeding cooling of the core;
— Causing unacceptable mechanical or thermal damage to the pressure boundary of the reactor coolant system.

COOLANT

3.25. The coolant should be physically and chemically stable with respect both to high temperatures and to nuclear irradiation in order to fulfil its primary function: the continuous removal of heat from the core. Safety considerations associated with the coolant should include:

(a) Ensuring that the coolant system is free of foreign objects and debris prior to the initial startup of the reactor and for the operating lifetime of the plant;
(b) Keeping the activity of the coolant at an acceptably low level by means of purification systems and the removal of defective fuel as appropriate;
(c) Taking into account the effects on reactivity of the coolant and coolant additives[4] and in particular the effects in determining the capabilities of the reactor control system and shutdown systems for operational states and design basis accidents;
(d) Determining and controlling the physical and chemical properties of the coolant in the core to ensure compatibility with other components of the reactor core, and minimizing corrosion and contamination of the reactor coolant system;
(e) Ensuring a sufficient supply of coolant for operational states and in design basis accidents;

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[4] It is general practice for some reactor types to ensure that coolant additives do not cause the power coefficient of reactivity to become positive. (For further discussion of the coefficients of reactivity, see Appendix I.)

(f)   Ensuring that the core is designed to prevent or control flow instabilities and consequent fluctuations in reactivity.

Light water reactors

3.26. In PWRs and BWRs the coolant also acts as the moderator. The effects of changes in coolant density (including fluid phase changes) on core reactivity and core power, both locally and globally, should therefore also be taken into consideration in the design of the core.

3.27. Chemical additives to the coolant are used as neutron absorbers to provide a second system of control over the core reactivity; an example of this is the boric acid used in PWRs. Other additives are used to control the chemistry of the coolant (e.g. for the control of the pH and the oxygen content) so as to inhibit the corrosion of core components and reactor internals or to reduce the contamination of the reactor coolant system. Whenever additives are used, their effects on core components should be taken into account in the core design.

3.28. Means of controlling corrosion products and hydrogen resulting from radiolysis of the coolant (see Ref. [4]) should be provided in the design.

Heavy water reactors

3.29. The relevant properties of heavy water are mainly similar to those of light water. The factors considered in relation to LWRs in paras 3.26–3.28 should also be applied in relation to heavy water. It should be noted that for pressure tube type HWR designs, the coolant and the moderator are separated; chemicals are not usually added to the coolant for controlling the reactivity.

3.30. Radioactivation should also be considered. Tritium (^{3}H) builds up to a greater extent in HWRs. Provision should therefore be made in coolant systems and moderator systems for HWRs to prevent or control the release of tritiated heavy water from the closed water circulation systems. Consideration should be given in the design to restricting the tritium concentration in air to ensure that radiation doses to site personnel for all operational states, including maintenance and inspection and decommissioning, do not exceed authorized limits and are as low as reasonably achievable [1, 5–7].

MODERATOR

3.31. The choice of moderator and the spacing of the fuel within it is based on the needs to optimize the neutron economy, and hence fuel consumption, and to meet engineering requirements. The main reactor types use either light water or heavy water as the moderating medium:

— PWR: light water moderated;
— BWR: light water moderated;
— PHWR (pressure tube type):   heavy water moderated;
— PHWR (pressure vessel type): heavy water moderated.

Light water reactors

3.32. Light water is used as the moderator and the coolant in both PWRs and BWRs; the two functions are not physically separated. The considerations regarding coolant, as discussed in paras 3.26–3.29, are therefore applicable.

Heavy water reactors

3.33. In PHWR reactors, the moderator is physically separated from the coolant by a calandria tube and a pressure tube. The moderator may at times contain small amounts of a soluble neutron absorber for the compensation of reactivity during operation or larger amounts to ensure subcriticality (usually during the initial approach to equilibrium fuelling conditions). The design should be such as to ensure the effectiveness of the shutdown and holddown capability of the reactor during an absorber dilution accident (e.g. an in-core break). Means should be provided to prevent the inadvertent removal of such absorber material (e.g. due to chemistry transients) and to ensure that its controlled removal is slow.

3.34. The long term control of pH and oxidation potential is necessary to ensure the long term holddown of reactivity by the second shutdown system in some PHWRs.

3.35. Because of radiolysis of the moderator, measures should be provided to control corrosion and to prevent hydrogen deflagration and explosion, as discussed in Ref. [4].

3.36. To apply the concept of defence in depth, for operational states as well as certain design basis accident conditions and severe accident conditions, the

moderator in PHWRs (for both pressure tube and pressure vessel type HWRs) should provide the capability to remove decay heat without loss of core geometry.

CORE REACTIVITY CHARACTERISTICS AND MEANS OF CONTROL OF REACTIVITY

3.37. Core reactivity characteristics in normal operation and in limiting conditions for the purposes of accident analysis and the means of control of reactivity for normal operation are discussed in this section.

Core reactivity characteristics

3.38. On the basis of the geometry and the fuel composition of the reactor core, the nuclear evaluations for design provide 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 and at shutdown conditions. Inherent neutronic characteristics are represented by reactivity coefficients and parameters as discussed in Appendix I.

3.39. In para. 2.11 recommendations are made concerning the necessary response of the core to an inadvertent increase in reactivity. The reason for the recommendations is the need to limit the resulting incremental addition of energy to the fuel so as to prevent the propagation of an accident. This limit is needed to ensure that the requirement established in para. 4.2 of Ref. [1] is met. Adequately conservative assumptions should be made for reactivity coefficients in the analysis of all design basis accidents and anticipated operational occurrences.

3.40. Key reactivity parameters such as reactivity coefficients should be evaluated for each core state and for the corresponding strategy for fuel management (see Appendix III). Their dependence on the core loading and the burnup of fuel should be taken into account.

Types and efficiency of means of control of reactivity

3.41. 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 (such as those associated with normal power transients, changes in xenon concentrations,

effects relating to temperature coefficients, the rate of flow of coolant or changes in temperature, the depletion of fuel and of burnable poison, and cumulative poisoning by fission products) so as to keep the process parameters within specified operating limits.

3.42. The means of control of reactivity and the means of shutting down the reactor may use common devices provided that the shutdown capability is maintained with an adequate margin at all times (see Ref. [1], para. 6.13).

3.43. The means of control of reactivity used for regulating the core reactivity and the power distribution for different reactor types include the following:

— Use of solid neutron absorber rods and blades (for PWRs, BWRs and PHWRs);
— Use of soluble absorber in the moderator or coolant (for PWRs and PHWRs);
— Control of the coolant flow (moderator density) (for BWRs);
— Use of fuel with distributed or discrete burnable poison;
— Control of the moderator temperature (for pressure vessel type PHWRs);
— Control of the moderator height (for older pressure tube type PHWRs);
— Use of liquid absorber in tubes (for PHWRs);
— Use of a batch refuelling and loading pattern;
— Use of on-load refuelling (for pressure tube and pressure vessel type PHWRs).

3.44. The efficiency of the means of reactivity control can be represented by reactivity coefficients and/or the integral worth of absorbers, which will depend on the design of the reactor core and the associated fuel management. The effectiveness of the reactivity control devices such as neutron absorber rods should be checked by direct measurement.

3.45. In balance with the need for efficiency of the means of control of reactivity for normal operation, the most adverse conditions of the corresponding equipment should be investigated and taken into account in the accident analysis (e.g. the consequences of the integral and differential worth of the control banks for rod withdrawal accidents and, for a PWR, the differential worth of soluble boron for an inadvertent dilution accident).

Maximum reactivity worth and reactivity insertion rate

3.46. 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 credible abnormal withdrawal of the devices does not cause the specified fuel limits to be exceeded.

3.47. The maximum reactivity worth of the reactivity control devices should be limited, or interlock systems should be provided, so that for relevant design basis accidents, such as a control rod ejection for PWRs or a control rod drop for BWRs, the resultant power transient does not exceed specified limits. These limits should be chosen so as to ensure that the following are restricted to acceptably low levels:

(1) Damage to the fuel pellets and cladding which could give rise to the release of radioactive material into the coolant;
(2) The risk of a molten fuel–coolant interaction which could damage the core structure and impede the insertion of the shutdown devices.

In assessing these reactivity accidents, the effect of the fuel burnup and the fuel type (e.g. uranium dioxide or mixed oxide fuel) should also be considered.

3.48. With regard to the soluble absorber, the reactivity control system should be designed to prevent any decrease in the concentration of absorber in the core, which could cause the 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 Ref. [4]). The concentrations of the soluble absorber in all storage tanks should be monitored. Whenever enriched ^{10}B is used, adequate monitoring should be provided.

3.49. A detailed functional analysis of the control systems should be performed to identify the possibilities for giving rise to the inadvertent dilution of boron in operation and in shutdown conditions, and to ensure the adequacy of preventive measures. Such preventive measures are: 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.

3.50. For PWRs the consequences of local dilution of boron in the core should be evaluated as a reactivity insertion event that occurs owing to reflux

condensation in a steam generator under particular conditions of a small break loss of coolant accident.

Control of global and local power

3.51. The core power should be controlled globally and locally using the means of reactivity control in such a way that the peak linear heat rate of each fuel element and channel does not exceed the design limits anywhere in the core. Variations in the power distribution caused by local variations in reactivity due to xenon instability, changes in coolant conditions and changes in the positions and characteristics of flux detectors should be taken into account in the design of the control system. Further information is given in the section on core management in Section 3 and in Appendix III.

3.52. In PWRs, rapid changes in the power level are automatically controlled by movement of the control rods. The control rod banks should be so arranged as to avoid large radial and axial distortions of the power distribution. This may be done, for example, by locating the control banks symmetrically for radial symmetry of the power distribution, or by partially inserting and overlapping the control rod banks so as to avoid axial power distortions, which may cause adverse xenon oscillations. Xenon oscillations can also be controlled by manually or automatically adjusting the boron concentration of the reactor coolant. The long term reactivity compensation for burnup of the fuel is achieved by reducing the boron concentration in the coolant.

3.53. For BWRs, rapid changes in the power level are controlled either by adjusting the speed of the recirculation pumps or by moving control rods. In this regard, the operating conditions for BWRs should provide sufficient margin against thermal-hydraulic instabilities. A reactor core calculation analysis should be made to demonstrate the thermal-hydraulic stabilities. The long term reactivity compensation for fuel burnup is achieved by continuously withdrawing the control rods in a predetermined sequence.

3.54. For some PHWRs, rapid changes of the local and global power levels are normally controlled by adjusting the levels of light water in liquid zone compartments. Solid absorber and/or adjuster rods and liquid absorber for addition to the moderator are also available for control purposes. Control of the moderator level was used in older designs of reactor. Long term reactivity compensation for fuel burnup is achieved by means of on-power refuelling.

3.55. The insertion of control rods should be such that an adequate shutdown margin remains in all operational states. In the specification and monitoring of insertion limits for the control rods, depending on the power level, the shutdown margin should be guaranteed.

Effects of burnable poison

3.56. The effects of the burnup of burnable poison on the core reactivity should be evaluated and then it should be ensured that the shutdown margin is adequate in all the resulting core conditions throughout the fuel cycle.

3.57. For PWRs, in order to keep the temperature coefficient of the moderator negative, the designer may choose to reduce the amount of absorber in the moderator and to restore the absorption effect by adding burnable poison to the fuel. Burnable poison may also be used to flatten the power distribution and to reduce variations in reactivity during fuel burnup.

Irradiation effects

3.58. Effects of irradiation such as burnup, changes in physical properties, production of gas and swelling of absorber materials, and the contamination of reactor coolant should be taken into account in the design of reactivity control systems.

REACTOR SHUTDOWN SYSTEMS

3.59. This section deals with the means of bringing the reactor to a subcritical state in operational states and in design basis accidents and of maintaining it in this state.

3.60. Means are required to be provided to ensure that the reactor can be rendered subcritical and held in this state, on the assumption for a single failure that the most reactive core conditions arise when the shutdown device that has the highest reactivity worth cannot be inserted into the core (the assumption of one shutdown device stuck)[5] (Ref. [1], para. 6.13).

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[5] For some designs, e.g. PHWRs, the unavailability of two shutdown rod devices with the highest reactivity worth is assumed for this case.

3.61. "The means for shutting down the reactor shall consist of at least two different systems to provide diversity. At least one of the two systems shall be, on its own, capable of quickly rendering the nuclear reactor subcritical by an adequate margin from operational states and in design basis accidents, on the assumption of a single failure. Exceptionally, a transient recriticality may be permitted provided that the specified fuel and component limits are not exceeded" (Ref. [1], paras 6.14 and 6.15).

The specified fuel and component limits should be provided as a set of design limits for operational states and for design basis accidents. The design of this system should be fail-safe. (See the subsection on the rate of shutdown for the necessary rate.)

3.62. "At least one of these two systems shall be, on its own, capable… of maintaining the reactor subcritical by an adequate margin and with high reliability, even for the most reactive conditions of the core" (Ref. [1], para. 6.16). One of these systems should be, on its own, capable of maintaining the reactor in a subcritical state for any core coolant temperature.

3.63. In meeting the long term reactivity holddown requirements, deliberate actions that increase reactivity in the shutdown state, such as the movement of absorbers for maintenance purposes, the dilution of the boron content and refuelling actions, should be identified for the purpose of ensuring that the most reactive condition is taken into account.

3.64. The shutting down of the reactor following anticipated operational occurrences and in design basis accidents is important, and its importance should be recognized in the design of the shutdown systems. For operational states and design basis accidents, the design limits such as the specified fuel limits and the limits for the pressure boundary of the reactor coolant system for each plant state or condition should not be exceeded. The necessary reliability should be ensured through the design of the equipment. The design should ensure the necessary independence from plant processes and control systems.

3.65. If the operation of the reactivity holddown system is manual or partly manual, the necessary prerequisites for manual operation should be met [9].

3.66. Part of the means of shutdown may be used for the purposes of reactivity control and flux shaping in normal operation. Such a use of the means of shutdown in normal operation should not be able to jeopardize the functioning of the shutdown system.

Different means of shutdown

3.67. Various means of introducing negative reactivity into the reactor core are used for different reactor types, including:

— Injection of boron into the moderator;
— Injection of gadolinium into the moderator;
— Dumping of the moderator;
— Insertion of boron and Ag–In–Cd and hafnium in stainless steel rods, tubes or cruciform plates;
— Insertion of cadmium sandwiched in steel tubes;
— Insertion of hafnium and steel rods in zirconium alloy guide tubes;
— Insertion of liquid absorber in tubes.

3.68. The efficiency of these neutron absorbing materials will depend on the fuel composition, the design of the reactor core and core management. For instance, the reactivity worth of thermal neutron absorbers such as boron or Ag–In–Cd is lower for mixed oxide fuel than it is for uranium fuel. A change in the core loading scheme from 'out-in' to 'in-out' would affect the shutdown margin.

3.69. Table 1 gives examples of means of shutdown used in different types of reactor, illustrating the diversity of means.

TABLE 1. MEANS OF SHUTDOWN

Reactor type	Fast shutdown system	Diverse shutdown system
BWR	B_4C in steel tubes/Hf plates	Boron solution injected into moderator/coolant
PWR	Ag–In–Cd in steel tubes/B_4C in steel tubes	Boron injected into moderator/coolant
PHWR	Cadmium sandwiched in steel tubes	Gadolinium injected into moderator; dumping of moderator; liquid absorber in tubes
PHWR (pressure vessel type)	Hf and steel rods in zirconium alloy guide tubes	Boron injected into moderator

Reliability

3.70. A high reliability of shutdown should be achieved by using a combination of measures such as:

(a)   Adopting systems that are as simple as possible.
(b)   Using a fail-safe design as far as practicable[6].
(c)   Giving consideration to the possible modes of failure and adopting redundancy in the activation of shutdown systems (e.g. sensors or actuation devices). Provision for diversity may be made, for example, by using two different physical trip parameters for each accident as far as practicable.
(d)   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, including common cause failure.
(e)   Ensuring easy entry of the means of shutdown into the core, with account taken of the in-core environmental effects of operational states and accident conditions within the design basis.
(f)   Designing to facilitate maintenance, in-service inspection and operational testability.
(g)   Providing means for performing comprehensive testing during commissioning and outages for maintenance.
(h)   Testing of the actuation process (or of partial rod insertion, if feasible) during operation.
(i)   Selecting equipment of proven design.

3.71. A reliability analysis of shutdown systems should be performed to quantify the effectiveness of the design.

3.72. The capability and reliability of the shutdown systems should be assessed as part of the safety analysis. In parallel with efforts to raise the reliability of the fast shutdown system to a very high level, anticipated operational transients without fast shutdown should also be analysed to confirm that the consequences for the

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[6] The simplest common form of design for fail-safe shutdown allows the shutdown decices 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).

fuel and the pressure boundary are acceptable and that a subcritical condition can be maintained in the long term[7]. For designs that incorporate at least two fully capable, diverse and independent shutdown systems, these events should be analysed on the assumption that one of the fast shutdown systems fails.

Effectiveness of shutdown and reactivity holddown

3.73. The design should be such as to ensure the capability of the shutdown and reactivity holddown systems to render and maintain the reactor subcritical by an adequate margin even in the most reactive core conditions. This should apply for the entire range of operating conditions and core configurations that may occur throughout the intended operation cycle, anticipated operational occurrences and design basis accidents. This should be demonstrated:

— In design, by means of calculation;
— During commissioning and immediately after refuelling, by means of appropriate neutronic and process measurements to confirm the calculations for a given core loading;
— During reactor operation, by means of measurements and calculations covering the actual and anticipated reactor core conditions.

3.74. These analyses should cover the most reactive core conditions, on the assumption of the failure of shutdown device(s), as outlined in para. 3.62. In addition, reactivity holddown should be maintained if a single random failure occurs in the shutdown system. In the resulting subcriticality margin, account should be taken of the uncertainties in the calculations, the possible deviations in core loading during refuelling and the available monitoring ranges with their associated uncertainties. The necessary absolute value of the shutdown margin may depend on the reactor core design and on the way in which the safety demonstration is performed.

3.75. The number and the reactivity worth of shutdown rods needed in the systems is largely determined by the following factors, which should all be considered:

(a) The core size.

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[7] In some States an analysis of anticipated operational transients without scram is not required if two diverse and independent shutdown systems are provided, each of which is capable of quickly rendering the nuclear reactor subcritical from all operational states and accident conditions within the design basis.

(b)   The fuel type and the core loading scheme.

(c)   The necessary margin of subcriticality.

(d)   The assumption of the failure of shutdown device(s) as outlined in para. 3.61.

(e)   The uncertainties associated with the calculations. These may be estimated by the comparison of calculations with measurements made in experimental and prototype reactors and during the commissioning of the reactor.

(f)   Shutdown device 'shadowing'. 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.

(g)   The most reactive core conditions after shutdown. These are the result of a number of parameters such as:

—   The most reactive core configuration (and where appropriate the corresponding boron concentration) that will occur during the intended fuel cycle, including refuelling;

—   The most reactive credible combination of fuel and moderator temperatures;

—   The amount of positive reactivity insertion resulting in design basis accident conditions;

—   The amount of xenon as a function of time after shutdown;

—   The burnup of the absorber.

Rate of shutdown

3.76. The rate of shutdown should be adequate to render the reactor sufficiently subcritical that the specified design limits of the fuel and of the pressure boundary are not exceeded.

3.77. In designing for the rate of shutdown, the response time of the protection systems and the associated safety actuation systems (the means of shutdown) should be taken into account.

3.78. In evaluating the rate of shutdown, the following factors should be considered:

(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 corresponding to different reactor core designs. The rate of shutdown is sensitive to:

—   the distance of the shutdown devices from the active region of the core prior to insertion;

> — the locations of the injection nozzles for the soluble absorber, which should be such that the absorber may be quickly dispersed in the active region of the core.

(d) 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 and 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 give the necessary insertion speed:
> — gravity drop of shutdown rods into the core,
> — hydraulic or pneumatic pressure drive of shutdown rods into the core,
> — hydraulic or pneumatic pressure injection of soluble neutron absorber.

3.79. Means of checking the insertion speed of shutdown devices should be provided. The insertion time should be checked regularly.

Environmental considerations

3.80. The following environmental effects should be considered in the design of shutdown systems:

(a) *Irradiation effects.* Depletion of the absorber (e.g. boron) and swelling and heating of materials due to neutron and gamma absorption.
(b) *Chemical effects.* Chemical effects such as corrosion of the shutdown devices. The transport of activated corrosion products through the reactor coolant system and moderator system should also be considered.
(c) *Changes in structural dimensions.* Dimensional changes and movements of internal core structures due to temperature changes, irradiation effects or external events such as earthquakes should not prevent the insertion of the shutdown devices.

REACTOR CORE AND ASSOCIATED STRUCTURES

3.81. This section concerns the structures that form and support the reactor core assembly and those that are central to the performance and the safety of the reactor core.

3.82. Possible damage mechanisms that could affect the core and its associated structures and should be considered in the design include: vibration, both

transmitted structurally and induced by coolant flow; fatigue; other mechanical effects such as internal missiles; thermal, chemical, hydraulic and irradiation effects; and seismic motions. Of particular concern are: damage to shutdown and holddown systems; insufficient capability for cooling the fuel; damage to fuel; 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 also be considered.

3.83. The expected radiation heating of the structures should be calculated and proper cooling should be provided. There should be adequate safety margins for the thermal stresses generated in operational states and in design basis accidents. The chemical effects of the coolant and the moderator on the structures should be considered.

3.84. Provisions for the necessary inspection of the core components and associated structures should be included in the design.

Reactor coolant pressure boundary

3.85. The fuel assemblies and other core components in PWRs and BWRs and in pressure vessel PHWRs should be arranged so as to ensure that there is a low neutron flux at the reactor coolant pressure boundary (i.e. the wall of the reactor vessel). Where necessary, the neutron flux at locations on the reactor coolant pressure boundary that are vulnerable to neutron irradiation embrittlement should be monitored.

Reactor core support structures

3.86. The reactor core support structures comprise tube sheets, a core barrel, support keys and other structures, depending on the reactor core design, which maintain the fuel assembly support structures in the desired geometrical position against the reactor coolant pressure boundary. These support structures should be designed to remain intact and capable of performing their functions throughout the lifetime of the reactor for operational states and design basis accidents. Mechanical loads such as those induced by hydraulic forces and by normal refuelling and postulated abnormal refuelling should be considered. Seismic loads should be taken into account.

Fuel assembly support structures

3.87. The fuel assembly support structures should be designed to hold the fuel assembly in the desired geometrical position for operational states and design basis accidents.

Guide structures for shutdown and reactivity control devices

3.88. The guide structures for the shutdown and reactivity control devices should be designed to perform their functions under operational states and in design basis accidents.

3.89. Since these guide structures are in close proximity to the fuel assemblies or fuel channels in PWRs and BWRs, the possibility of their physical interaction and damage during operation and shutdown and in design basis accidents should be given due consideration in the design. The potential for fatigue due to high neutron flux irradiation and/or gamma heating should also be considered. In the case of shutdown and reactivity control devices immersed in a bulk moderator (e.g. for PHWRs), the effects of hydraulic forces on these structures should be considered and their maximum allowable distortion should not be exceeded.

3.90. The design should facilitate the replacement of the reactivity control and shutdown devices whenever necessary without causing damage to the reactor core, unacceptable insertion of reactivity or undue radiation exposures.

Support structures for in-core instrumentation

3.91. The structures and guide tubes containing instrumentation for the detection of accidents and the mitigation of their consequences, within and in close proximity to the core, should be designed to perform their functions in all operational states and in design basis accidents. Consideration should be given to the possibility that the flow induced vibration of these structures and guide tubes may result in fretting and consequent failure in long term operation.

3.92. The structures and tubes should be so designed that the instrumentation is accurately located and cannot be moved by inadvertent operator actions, strains on equipment, forces due to coolant flow or movements of bulk moderator, in operational states or in design basis accidents. The design should facilitate the replacement of the instrumentation whenever necessary.

Other reactor vessel internals

3.93. Depending on the reactor type, various other structures may be installed within the reactor vessel. These include feedwater spargers, steam separators, steam dryers, core baffles, reflectors and thermal shields. The functions of these internals 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 so designed that their mechanical performance does not jeopardize the performance of any associated safety functions throughout their service life.

CORE MANAGEMENT

3.94. The objective of core management is to fulfil the requirements for the safety of the reactor core and the economic utilization of the nuclear fuel [10].

3.95. A fuel cycle should be selected with appropriate levels of enrichment and appropriate means of controlling the core reactivity and the power distribution so as to extract energy from the fuel in the most economic manner within the safety limitations. Various means that are available for achieving this objective of core management are given in Appendix III of this Safety Guide.

Safety limitations

3.96. The specified design limits for normal operation should be taken into account in the design for core management.

3.97. For operational states the goal is that no cladding failures should occur. However, certain conditions (e.g. manufacturing defects in fuel elements, wear due to debris fretting or unexpected transients in operational states) may make it extremely difficult to meet this no-failure goal. In practice some fuel cladding failures can be accepted in operational states since the concentration of radioactive material in the reactor coolant systems can be reduced by the reactor coolant cleanup function. Cleanup of the reactor coolant and other necessary means should be provided in the design to ensure that releases of radioactive material to the environment remain within authorized limits.

3.98. For design basis accident conditions, the permissible degree of fuel failure depends on the likelihood of the conditions arising and the expected radiological consequences. To limit the number of acceptable fuel failures under design basis

accident conditions, operational limits should be placed on the fuel that are more restrictive than those resulting from normal operational demands. These may be, for example, to keep the minimum departure from nucleate boiling ratio (in PWRs) or the minimum critical power ratio (in BWRs) for normal operation at a value high enough to avoid the cladding becoming dry during anticipated operational occurrences. In PHWRs, operational limits should be placed on the fuel, in conjunction with the performance of safety systems, limiting the maximum fuel temperature and oxidation as well as the maximum rate of power change in order to prevent cladding failure.

Design information for reactor operation

3.99. To achieve the desired core reactivity and power distribution for reactor operation, the core management programme should provide the operating organization with the following information:

— The pattern of fuel assemblies in each fuel cycle;
— The schedule for the subsequent unloading and loading of fuel assemblies;
— The configurations of reactivity control and shutdown devices;
— The fuel assemblies to be shuffled;
— Burnable poisons and other core components to be removed, inserted or adjusted.

Further guidance on fuel handling is provided in Ref. [10].

Reactor core calculational analysis

3.100. In many cases safety parameters, such as the fuel and cladding temperatures and the peak linear heat rate, are not directly measurable and their values are not available to the reactor operators. A calculational analysis of the reactor core conditions should therefore be carried out so as to specify such parameter values in the operating procedures. Sufficient instrumentation should be provided to be able to verify the results of the analysis by measurement.

3.101. Analytical methods and associated computer codes should be verified and validated by comparison with one or more of the following:

— Measurements made in experimental reactors;
— Measurements made in prototype reactors;
— In-core measurements of fuel elements and assemblies made under simulated conditions;

— Operating data for cores of similar design;
— Measurements made during commissioning;
— Measurement made during the operation of reactors: at different power levels after each refuelling and at different times during the cycle;
— Post-irradiation measurements made on fuel elements and assemblies to evaluate fine structure and burnup effects;
— Benchmark calculations made with other validated codes.

3.102. A reactor core analysis should be carried out at appropriate times to ensure throughout the reactor's operating lifetime that the operational strategy and the limitations on operation do not violate the design limits.

3.103. The analysis should cover typical cases from the entire operating cycle for the following reactor core conditions:

— Full power, including representative power distributions;
— Load following;
— Approach to criticality and power operation;
— Power cycling;
— Startup;
— Refuelling;
— Shutdown;
— Anticipated operational occurrences;
— Operation at the thermal-hydraulic stability boundary for BWRs.

Whenever the management of fuel in the core is changed or any characteristics of the fuel elements (such as the fuel enrichment or the fuel cladding material) are changed, the analysis should be updated.

3.104. To derive peak channel power and peak linear power rates for normal full power operation, steady state power distributions should be calculated for each assembly location and axially along the fuel assemblies. Allowance should be made for the effects of changes in the geometry of the assembly on neutronic and thermal-hydraulic effects (e.g. changes in the moderator thickness due to bowing of the assembly). To identify hot spots, the radial power distribution within a fuel assembly and the axial power distortion due to spacers, grids and other components should be superimposed.

3.105. The power and temperature distributions throughout the lifetime of a fuel element should be predicted by the methods of para. 3.104 to assess the

behaviour of the fuel in the core and to demonstrate the continuing integrity of the fuel elements.

3.106. When fuel assemblies of different types are loaded into the core (a so-called mixed core), their mechanical and thermal-hydraulic compatibility (e.g. in terms of the pressure drop characteristics through the fuel assemblies), as well as their compatibility in terms of the nuclear characteristics of the core, should be analysed.

3.107. The effects of operating conditions such as load following, power cycling, reactor startup and refuelling should, whenever necessary, be superimposed onto the rating and temperature histories to evaluate the effects of thermal cycling, such as the buildup of pressure due to fission gas and fuel cladding fatigue.

Refuelling

3.108. Further information on fuel handling and storage systems is given in Refs [3, 10].

3.109. For on-power refuelling, the integrity of the pressure boundary of the reactor coolant system should be maintained. It should be ensured that the effects of the refuelling operation on the neutronic behaviour of the core remain within the capability for control of the reactor control systems.

3.110. Administrative control measures may be used to ensure that the individual fuel assemblies are loaded into their intended positions in the core. For this purpose the following provisions may be considered:

— Reactivity monitors to verify the correct fissile enrichment of the fuel assemblies (PWRs and BWRs);
— Mechanical means to prevent the inadvertent loading of fuel assemblies (PWRs and BWRs).

In PHWRs utilizing on-power refuelling, administrative control measures normally specify that no refuelling should take place while the reactor is shut down, as an added protection against inadvertent criticality during refuelling.

3.111. Measurements of the in-core flux distribution may be made to provide an ultimate verification of the fuel loading pattern.

CORE MONITORING SYSTEM

3.112.   Instrumentation should be provided 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, temperature) and the expected efficiency of the means of shutdown of the reactor (e.g. the insertion rate of the absorber devices compared with their insertion limits), so that any necessary corrective action can be taken. In addition, the activity levels in the coolant should be monitored to verify that design limits are not exceeded. Control systems may be used to ensure the necessary variation of core parameters and to maintain them within defined operating ranges. Depending on the rapidity of the variation in a parameter, the actuation of the monitoring systems may be automatic or manual.

3.113.   The selection of parameters to be monitored will depend on the reactor type.

3.114.   For reactor types that employ on-load refuelling, a system indicating the location of a failed fuel assembly may enable the defective fuel assembly to be removed more easily, thereby keeping the activity levels in the coolant low.

3.115.   The accuracy, speed of response, range and reliability of all monitoring systems should be adequate for performing their intended functions (see Ref. [9]). The design should allow the continuous or periodic testing of monitoring systems.

3.116.   Guidance on post-accident monitoring is provided in Ref. [9]. If core monitoring is needed in accident conditions within the design basis, the instrumentation to be used should be qualified for the environmental conditions to be expected following the accident.

3.117.   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 there are adequate safety margins. The in-core flux distribution should be regularly monitored. Detectors should be distributed in the core in such a way as reliably to detect local changes in power density. Both ex-core and in-core neutron detectors should be periodically calibrated. (These recommendations apply for PWRs, BWRs and PHWRs.)

3.118.   The following are examples of parameters to be measured for the purposes of core monitoring:

— Neutron flux;
— Coolant temperature;
— Coolant flow rate;
— Water level;
— System pressure;
— Activity in the coolant;
— Control rod insertion position;
— Concentration of soluble boron (for a PWR).

Such parameters may be monitored for safety purposes, at various locations if possible.

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

— The neutron flux doubling time,
— The rate of change of neutron flux,
— Axial and radial neutron flux imbalances,
— The reactivity balance,
— Thermal-hydraulic core parameters (e.g. the departure from nucleate boiling ratio or the critical power ratio).

3.120. During reactor shutdown, a minimum set of instruments should be available (i.e. neutron flux detectors with an adequate sensitivity). At least one means of shutdown should be available to ensure a safe response following an inadvertent criticality.

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

3.122. During 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 needed 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 the neutron sources should be such as to ensure that:

— The sources function properly for their planned lifetime,

— The sources are compatible with the fuel assemblies and the fuel assembly support structures.

3.123. Analysis of neutronic and acoustic noise may provide useful information on loose parts or the incipient mechanical failure of core components or core internals, or the malfunctioning of the measurement equipment.

3.124. A computerized core monitoring system may be used to ensure that the status of the core is within the operating limits assumed in the safety analysis. Qualification of the system should be ensured where it is coupled to a protection system (see Ref. [9]).

SAFETY ANALYSIS

3.125. Postulated initiating events (including credible combinations of events such as equipment failure, operator errors and external natural and human induced events) should be analysed in accordance with Ref. [11].

Postulated initiating events

3.126. Postulated initiating events and event sequences vary for different designs of reactor core; the response of the reactor core will also vary widely depending on reactivity coefficients (see Appendix I) and the systems involved.

Methods of analysis

3.127. The specified limits for core design for the various sequences of initiating events should be consistent with the likelihood of the occurrence of each event and its associated radiological consequences. Evaluations to establish that the subsequent fuel conditions do not exceed acceptable limits should use either a conservative approach for the relevant parameters or a realistic (best estimate) approach that includes the evaluation of uncertainties. In a best estimate analysis, it is general practice to investigate the sensitivity of the results to variations in relevant parameters.

3.128. All the computer codes used in the safety analysis should be verified and validated. The methods and/or models used in the computer code for calculations should be adequate for the purpose. (For further guidance, see under Assessment of the Computer Codes Used in Ref. [11].)

3.129. With regard to the results of calculations made using the computer codes, it should be confirmed that the predictions of the code have been validated by adequate means; that is, by comparison with experimental data or with benchmark calculations made with other validated codes, or by the combination of both.

3.130. Detailed core analyses such as neutronic, thermal-hydraulic and mechanical analyses should be performed.

3.131. The following major factors should be considered in the detailed core analyses:

— The operating state (e.g. the thermal-hydraulic conditions: subcritical, part load, full load, xenon history);
— The temperature coefficient of reactivity for the fuel;
— The temperature coefficients of reactivity for the coolant and the moderator;
— The void coefficients of reactivity for the coolant and the moderator;
— The rate of change of the concentration of soluble absorber in the moderator and the coolant;
— The rate of insertion of positive reactivity caused by the reactivity control device or changes in process parameters;
— The rate of insertion of negative reactivity associated with a reactor trip;
— Individual channel transient response related to the average thermal power of the core;
— The performance characteristics of safety system equipment including the changeover from one mode of operation to another (e.g. from the injection mode for emergency core cooling to the recirculation mode);
— The decay of xenon and other neutron absorbers in the long term core analysis.

3.132. Areas of uncertainty should be dealt with by making conservative assumptions in the analysis or by adding a margin for uncertainty to the input parameters used. These uncertainties include both random and systematic components to cover statistical and physical uncertainties. Systematic uncertainties should be added as a bias in the analysis; random uncertainties may be combined statistically.

3.133. Safety analysis for the core is used to check that the specified fuel limits are not exceeded. The effect on core cooling of such conditions as ballooning and rupture of the cladding, exothermic metal–water reactions and distortions of fuel elements should be included in the analysis. The formation of hydrogen as a result of a metal–water reaction can threaten the integrity of the containment and therefore should also be taken into account (see Ref. [12]).

3.134. Analyses may lead to operational restrictions to ensure that the fuel design limits are not exceeded.

3.135. The identification and ranking of the key physical phenomena and systems that determine the plant response should be performed. This information may be used to evaluate the modelling capability of the associated codes.

4. QUALIFICATION AND TESTING

GENERAL

4.1. The safety considerations for the core design should be taken into account throughout the lifetime of the structures, systems and components of the core. This can be achieved as discussed below.

EQUIPMENT QUALIFICATION

4.2. A qualification programme should confirm the capability of the reactor core equipment to perform its function, for the relevant time period, with account taken of the appropriate functional and safety considerations under given environmental conditions (e.g. conditions of pressure, temperature, radiation levels, mechanical loading and vibration). These environmental conditions should include the variations expected in normal operation, anticipated operational occurrences and design basis accidents.

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

4.4. Methods of qualification may include:

(a) The performance of a type test on equipment representative of that to be supplied;
(b) The performance of a test on the equipment supplied;
(c) The use of pertinent past experience;

(d)   Analysis based on available and applicable test data;
(e)   Any combination of the above methods.

PROVISION FOR INSPECTION AND TESTING

4.5.   As stated in para. 2.9, suitable provision is required to be made in the design and layout of the structures, systems and components of the reactor core to ensure that radiation doses to the public and to all personnel in all operational states, including testing and inspection, do not exceed authorized limits and are as low as reasonably achievable [1, 5–7].

4.6.   Provisions should be made in the design for in-service testing and inspection to ensure that the core and associated structures and the reactivity control and shutdown systems will perform their intended functions throughout their lifetime. Further guidance on in-service inspections is provided in Ref. [13].

4.7.   A system should be designed to allow the identification of each assembly as well as its orientation within the core. Provision should also be made for inspecting each fuel assembly before and after irradiation to detect any possible damage.

5. QUALITY ASSURANCE IN DESIGN

5.1.   High quality design and fabrication should be ensured for fuel and core components by means of the establishment and application of satisfactory quality assurance procedures (see also Ref. [14])[8].

5.2.   A high level of quality assurance should be applied in the development and assessment of computer codes and associated methods for safety analysis.

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[8] The IAEA is revising the requirements and guidance in the subject area of quality assurance as established in Safety Series No. 50-C/SG-Q (1996) in new safety standards on management systems for the safety of nuclear facilities and activities involving the use of ionizing radiation. The term 'management system' has been adopted in the revised standards instead of the terms 'quality assurance' and 'quality assurance programme'. The new standards will integrate all the aspects of managing a nuclear facility, including the safety, health, environmental and quality requirements, into one coherent system.

Appendix I

REACTIVITY COEFFICIENTS

I.1. One important feature of the behaviour of the reactor core in any transient condition is the rate at which the transient progresses. This rate depends on the combined effects of the core nuclear characteristics discussed in this appendix. The factors of importance are:

— The temperature coefficient of reactivity for the fuel,
— The temperature coefficient of reactivity for the coolant,
— The temperature coefficient of reactivity for the moderator,
— The coolant density coefficient of reactivity,
— The delayed neutron fraction,
— The prompt neutron lifetime,
— The effects of power redistribution on reactivity (e.g. the xenon efficiency and the moderator density).

In PWRs and BWRs, the coolant and the moderator are one and the same and thus the corresponding coefficients of reactivity are the same.

I.2. The power coefficient of reactivity is commonly used to define the change in reactivity with unit power.

I.3. The nuclear characteristics of the core determine the inherent feedback effects on reactivity during a power rise due to the increase in temperature of the fuel, the coolant and the moderator.

I.4. Owing to the increase in the neutron resonance absorption cross-section of ^{238}U with temperature (the Doppler effect), the temperature coefficient of the fuel is normally negative.

I.5. For reactors using soluble boron in the coolant–moderator, the temperature coefficient of reactivity for the coolant–moderator is influenced by the boron concentration. At high boron concentrations this coefficient attains positive values. However, a slightly positive temperature coefficient of reactivity for the coolant–moderator is acceptable if the overall feedback effect on reactivity with temperature is sufficiently negative to limit the power increase to acceptable values. A negative temperature coefficient of reactivity for the coolant–moderator in the power range can be achieved by reducing the

boron concentration if necessary, by the use of a fixed or separated burnable absorber and/or the insertion of control devices[9].

I.6.   The transient behaviour of the core depends on the type and design of reactor, and therefore the signs of the various coefficients of reactivity vary from one type of reactor to another. Table 2 illustrates these variations.

TABLE 2. REACTIVITY COEFFICIENTS FOR POWER OPERATION

| Parameter | Reactivity coefficient | | | |
| | PWR | BWR | HWR | |
			Pressure tube reactor	Pressure vessel reactor
Coolant temperature	–	–	+	+
Coolant density[10]	+	+	–	–
Moderator temperature	–[11]	–	~0[11]	–
Fuel temperature	–	–	–	–
Power	–	–	~0[11]	~0[11]

Note: Reactivity will increase (+) or decrease (–) during power operation when the relevant parameter increases.

[9] In certain types of reactor, moderator and coolant are separated in such a manner that a decrease in density by voiding in the coolant may lead to a positive reactivity effect.

[10] For BWRs, the coolant density coefficient of reactivity is expressed as a void coefficient of reactivity.

[11] In several States the temperature coefficient of reactivity for the moderator and/or coolant for PWRs is allowed to be slightly positive.

Appendix II

FUEL PELLET–CLADDING INTERACTION

II.1. Fuel pellet–cladding interaction, which is stress corrosion cracking caused when the fuel pellet expands and stresses the cladding in the presence of a corroding agent, should be taken into consideration.

II.2. A combination of the following can lead to stress corrosion cracking in cladding:

— High tensile stress, uniform or local, perhaps caused by a crack opening in the pellet as it expands;
— A certain concentration of corrosive species, such as iodine, cadmium, caesium or other fission products;
— Long exposure.

II.3. Fuel failure can occur when, under long exposure to radiation, the fuel power is increased at a fast rate to a high power level. This is because the high tensile stress in the cladding due to expansion of the pellet can cause cracking of cladding that is already corroded.

II.4. Several approaches may be considered for limiting failures due to stress corrosion cracking. For example:

— Tensile stresses may be lowered by other means, such as limiting the rate of change in power or pre-pressurization of the fuel element;
— A fission product barrier may be placed at the inner surface of the cladding;
— The fission products may be immobilized by means of an additive;
— Local power peaking may be reduced by the appropriate overall design of the core.

II.5. There is an extensive database on operating experience, prototype testing and out-of-reactor testing. However, the phenomenon of stress corrosion cracking is only partially understood. At present, therefore, extensive judgement and the use of the available data or the results of testing on prototype fuel are needed in the design of fuel elements to confirm that the fuel design and the associated operating limits are adequate to prevent failure caused by stress corrosion cracking.

Appendix III

DESIGN CONSIDERATIONS FOR CORE MANAGEMENT

POWER SHAPING

III.1. The fuel cycle adopted for a particular reactor type presents numerous options, within the relevant design constraints, to make the most efficient use of the fuel. The following parameters relating to fuel management should be taken into account:

(1) *Core radial power shaping*. The flattening of the radial power distribution can be achieved by a combination of the following:
   — the radial distribution of reactivity control devices,
   — the relative movement of the reactivity control devices,
   — radial variations in fuel enrichment or burnup,
   — radial shuffling of fuel assemblies over their lifetime in the core,
   — radial distribution of fuel assemblies containing burnable poison in the fuel.
(2) *Assembly-to-assembly power variation*. This variation is largely a function of fuel enrichment and irradiation in the fuel assembly. The variation can be reduced by the use of:
   — burnable poisons within the fuel assemblies;
   — variations in fuel enrichment in a chequered pattern with the positions of the reactivity control devices taken into account;
   — the choice of the refuelling sequence, in particular for reactors with on-load refuelling.
(3) *Axial power shaping*. Although the means of reactivity control are utilized mainly to control the core reactivity and the radial power distribution, they are in some cases used to limit the peak axial power rating or the cladding temperature. Axial variation of the fuel enrichment or the content of burnable poison may be used to limit axial power peaking. On-power refuelling and adjuster rods are also used in PHWRs.
(4) *Variations within an assembly*. Within a fuel assembly, variation of the fuel enrichment or of the content of burnable poison may be used to optimize the power rating variations throughout the lifetime of the assembly.

FUEL DISCHARGE BURNUP

III.2. In enriched uranium reactors the maximum fuel discharge burnup is chosen on the basis of an engineering evaluation of fuel behaviour and the integrity of the fuel cladding (see Appendix IV, paras IV.2 and IV.3). The evaluation should be based on operating experience and in-pile and out-of-pile testing.

III.3. For natural uranium reactors with on-load refuelling, the maximum fuel discharge burnup is usually much lower than for enriched uranium reactors.

CORE EXCESS REACTIVITY

III.4. Core excess reactivity is needed in order to operate the reactor until the end of a fuel cycle and/or to adjust the power level of the reactor during a fuel cycle. In enriched uranium reactors, the fuel enrichment should be chosen so that the core will have sufficient excess reactivity to enable full power operation at all times during the planned fuel cycle. In LWRs, boron is used to absorb the excess core reactivity; its concentration can be reduced for overriding a xenon transient and for re-establishing the core power levels. In natural uranium reactors it is not economical to provide sufficient excess reactivity within the core for xenon override under all conditions. Booster or adjuster rods are therefore provided for xenon override for a limited time after the shutdown of the reactor.

III.5. The initial core of a natural uranium reactor with on-load fuelling has higher excess reactivity than the equilibrium core. This excess reactivity may be compensated for by depleted uranium fuel, by boron or gadolinium dissolved in the moderator, or by a combination of these means.

III.6. For batch refuelled designs of reactor core with enriched fuel, the most reactive core state should be identified, with account taken of variations in fuel enrichment levels and the depletion of burnable poison throughout the core during a fuel cycle.

III.7. The management of the core should not interfere with the capability of the shutdown systems to fulfil their specified safety functions.

KEY PARAMETERS FOR CORE DESIGN

III.8. Key parameters influencing core design and fuel management strategies include:

— For PWRs and PHWRs, the temperature coefficients of reactivity for fuel and moderator, the boron efficiency, the shutdown margin, the maximum reactivity insertion rate and the reactivity worth, the radial and axial power peaking factors, the maximum linear heat generation rate and the minimum departure from nucleate boiling ratio;
— For BWRs, the fuel temperature coefficient of reactivity, the void coefficient of reactivity, the shutdown margin, the maximum reactivity insertion rate and the reactivity worth, the maximum linear heat generation rate and the minimum critical power ratio.

Appendix IV

HIGH BURNUP FUEL CORES

IV.1. The economic optimization of the nuclear fuel cycle leads to core management strategies with increased fuel burnup. A high burnup range may be defined as a level higher than 50 GW.d/t average burnup for a discharged fuel assembly. Appendix IV describes aspects of core design and fuel element behaviour relevant to high burnup conditions for uranium oxide fuel.

IV.2. A high burnup of the fuel increases the changes to the microstructure of the fuel pellet and the porosity at the periphery of a pellet. These phenomena, which may affect the physical properties of the pellet such as heat conductivity, melting point, density, fission gas retention and fission gas release, should be taken into account.

IV.3. High burnup also increases the thickness of the oxidation layer of the cladding and its hydriding, which accumulate continuously during irradiation. Under high oxidation conditions, spalling of the oxide layer may occur. The hydriding affects the mechanical properties of the cladding. The rate of oxidation and the rate of hydriding depend strongly on the cladding material and also on its manufacturing process. The investigation of new alloys is aimed at improving the behaviour of the cladding by using, for example, a low tin content, surface layers of low tin content or materials containing niobium.

IV.4. In the design of a core for high burnup fuel, the enrichment of the fresh fuel is higher than that for a conventional core. This higher enrichment increases the excess reactivity of the core and the maximum enthalpy in the core coolant, thus reducing the margin to the thermal-hydraulic limits of the core parameters. An increase in the quantity of burnable poison is a possible measure for countering the high excess reactivity of the core.

IV.5. Owing to the extended irradiation time, the high burnup increases the release of fission gas and thus increases the buildup of pressure in a fuel element. This buildup of pressure can be moderated by providing a sufficient volume of gas plenum in the fuel element.[12]

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[12] Research is in progress on the possible benefits of reducing the release of fission gas by changing the microstructure of the fuel.

IV.6.   In the design of a high burnup core, the increase of the loads on the fuel assemblies, and their deformation, which may affect the ease of insertion of the control rods, should be taken into account.

IV.7.   Because of the higher excess reactivity of the core in a high burnup core, a higher concentration of soluble boron in the coolant may be needed. Such a higher concentration may cause problems relating to the coolant chemistry such as increased corrosion in the primary circuit. Enriched ^{10}B may be used to reduce the concentration of soluble boron.

IV.8.   In setting the maximum value for the fuel enrichment, a specified margin to prevent criticality in the storage of dry fuel and in the storage pool should be incorporated.

IV.9.   The changes in the properties of the fuel pellets and the cladding in conditions of high fuel burnup may affect the behaviour of fuel elements in accident conditions such as reactivity initiated accidents or loss of coolant accidents.

IV.10. For reactivity initiated accident conditions, the results of recent experiments indicate a reduction of the fuel failure limits in terms of the average peak enthalpy or the enthalpy rise for high burnup conditions in comparison with values for fresh or low burnup fuel. These analyses are not yet definitive as the experimental programmes are still in progress. Of specific concern are the contribution of the fission gas release to the swelling of pellets and the dynamic mechanical loading of the cladding during a fast power pulse, as well as the possible embrittlement of the cladding material.

IV.11. For loss of coolant accident conditions, the mechanical properties of cladding materials will determine the deformation of fuel elements and the failure behaviour resulting from the thermal loads. Research programmes have been initiated to measure these mechanical parameters of cladding materials for the oxidation conditions observed during operation.

Appendix V

MIXED OXIDE FUEL CORES

V.1.   The use of mixed oxide fuel in LWRs should be subject to the design considerations in this Safety Guide for uranium oxide fuel (UO_2 fuel) slightly enriched with [235]U. Appendix V summarizes the relevant differences between these two types, with account taken of the fact that mixed oxide fuel assemblies have in practice a reactivity worth that allows the same cycle length as for a UO_2 core loading.

V.2.   The material properties of mixed oxide and UO_2 fuel, such as density, heat conductivity and specific heat capacity, are not significantly different. However, the melting temperature of mixed oxide fuel is slightly lower than that of UO_2 fuel.

V.3.   The kinetic parameters for mixed oxide fuel, namely the total fraction of delayed neutrons and the prompt neutron lifetime, are slightly lower than those for UO_2 fuel.

V.4.   The microstructure of the mixed oxide fuel pellet is characterized by agglomerates containing plutonium within the uranium oxide pellet matrix. The size of the plutonium agglomerates depends on the specific fabrication process. The fissions in these agglomerates result in very high local burnup and consequently lead to changes in the microstructure and porosity in the vicinity. In UO_2 fuel, the microstructural changes start at the periphery with high local burnup leading to the so-called rim effect. Changes to the microstructure affect the physical properties of the fuel pellet and the characteristics of fission gas retention.

V.5.   For normal operational conditions it has been observed that the release of fission gas in UO_2 and mixed oxide fuel depends mainly on the power history of the fuel. For accident conditions, such as the fast power pulse of a reactivity initiated accident, some test results indicate that the dynamic fission gas release may cause higher release rates for mixed oxide fuel than for UO_2 fuel.

V.6.   The local power peaking in mixed oxide fuel elements in the direct neighbourhood of uranium fuel assemblies is a specific aspect of core loadings using fuel assemblies of both types. Such mixed oxide fuel elements are affected by the particular neutron energy spectrum of the UO_2 fuel assembly, which is higher in the thermal energy range, leading to a high fission rate in the

mixed oxide fuel element. This interference effect may be compensated for by reducing the content of fissile Pu in the peripheral fuel elements of the mixed oxide fuel assembly.

V.7. The reactivity efficiency in mixed oxide fuel of thermal neutron absorbers (e.g. control rod absorbers, fixed or soluble boron and xenon) is reduced in comparison with that of UO_2 fuel. This reduction in reactivity efficiency is caused by the shift of the neutron energy spectrum to the epithermal range in the mixed oxide fuel. If necessary, the use of a higher boron concentration or enriched ^{10}B or an increase in the number of control devices should be considered for reactivity compensation.

V.8. Reactivity feedback effects in mixed oxide fuel are also affected by the change in the neutron energy spectrum, but the effects of reactivity feedback on the reactivity behaviour of the core are within the usual range of changes resulting from burnup or variations in core loading for a UO_2 core.

V.9. For the characterization of mixed oxide fuel, the total content of plutonium and the detailed isotopic composition need to be known. Because the isotopic composition of plutonium varies strongly depending on the history of the spent fuel assembly from which the plutonium has been extracted, the ratio of fissile isotopes for the plutonium also varies; this will complicate the characteristics of the mixed oxide fuel. This is different from UO_2 fuel, which is fully characterized by the enrichment of ^{235}U.

V.10. The maximum plutonium content in a fuel is selected so that the reactivity behaviour of the total core is consistent with the maximum enrichment of UO_2 fuel. The maximum plutonium content should maintain a negative void reactivity coefficient.

V.11. One possible provision for limiting the changes in the nuclear characteristics of the core due to the introduction of mixed oxide fuel and its possible effects on systems and components is the specification of the maximum number of mixed oxide fuel assemblies to be loaded into the core. Full mixed oxide loading of the core may be considered as an option, but this would necessitate further adaptation of the reactor.

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GLOSSARY

plant states.

Operational states			Accident conditions		
				Beyond design basis accidents	
Normal operation	Anticipated operational occurrences	(a)	Design basis accidents	(b)	Severe accidents
				Accident management	
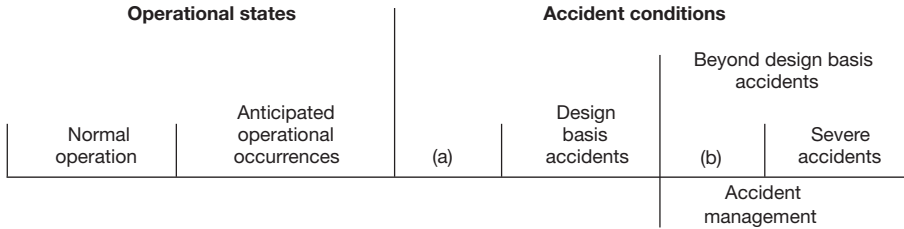
(a) Accident conditions which are not explicitly considered design basis accidents but which are encompassed by them.
(b) Beyond design basis accidents without significant core degradation.

accident conditions.

Deviations from normal operation more severe than anticipated operational occurrences, including design basis accidents and severe accidents.

accident management.

The taking of a set of actions during the evolution of a beyond design basis accident:

— to prevent the escalation of the event into a severe accident;
— to mitigate the consequences of a severe accident; and
— to achieve a long term safe stable state.

anticipated operational occurrence.

An operational process deviating from normal operation which is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions.

design basis accident.

Accident conditions against which a nuclear power plant is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.

normal operation.
> Operation within specified operational limits and conditions.

operational states.
> States defined under normal operation and anticipated operational occurrences.

severe accident.
> Accident conditions more severe than a design basis accident and involving significant core degradation.

protection system. System which monitors the operation of a reactor and which, on sensing an abnormal condition, automatically initiates actions to prevent an unsafe or potentially unsafe condition.

safety function. A specific purpose that must be accomplished for safety.

safety system. A system important to safety, provided to ensure the safe shutdown of the reactor or the residual heat removal from the core, or to limit the consequences of anticipated operational occurrences and design basis accidents.

single failure. A failure which results in the loss of capability of a component to perform its intended safety function(s), and any consequential failure(s) which result from it.

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