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No. 50-SG-D14

IAEA SAFETY GUIDES

Design for Reactor Core Safety in Nuclear Power Plants

A Safety Guide

INTERNATIONAL ATOMIC ENERGY AGENCY, VIENNA, 1986

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DESIGN FOR REACTOR CORE SAFETY IN NUCLEAR POWER PLANTS

A Safety Guide

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FOREWORD
by the Director General

The demand for energy is continually growing, both in the developed and the developing countries. Traditional sources of energy such as oil and gas will probably be exhausted within a few decades, and present world-wide energy demands are already overstraining present capacity. Of the new sources nuclear energy, with its proven technology, is the most significant single reliable source available for closing the energy gap that is likely, according to the experts, to be upon us by the turn of the century.

During the past 25 years, 19 countries have constructed nuclear power plants. More than 200 power reactors are now in operation, a further 150 are planned, and, in the longer term, nuclear energy is expected to play an increasingly important role in the development of energy programmes throughout the world.

Since its inception the nuclear energy industry has maintained a safety record second to none. Recognizing the importance of this aspect of nuclear power and wishing to ensure the continuation of this record, the International Atomic Energy Agency established a wide-ranging programme to provide the Member States with guidance on the many aspects of safety associated with thermal neutron nuclear power reactors. The programme, at present involving the preparation and publication of about 50 books in the form of Codes of Practice and Safety Guides, has become known as the NUSS programme (the letters being an acronym for Nuclear Safety Standards). The publications are being produced in the Agency's Safety Series and each one will be made available in separate English, French, Russian and Spanish versions. They will be revised as necessary in the light of experience to keep their contents up to date.

The task envisaged in this programme is a considerable and taxing one, entailing numerous meetings for drafting, reviewing, amending, consolidating and approving the documents. The Agency wishes to thank all those Member States that have so generously provided experts and material, and those many individuals, named in the published Lists of Participants, who have given their time and efforts to help in implementing the programme. Sincere gratitude is also expressed to the international organizations that have participated in the work.

The Codes of Practice and Safety Guides are recommendations issued by the Agency for use by Member States in the context of their own nuclear safety requirements. A Member State wishing to enter into an agreement with the Agency for the Agency's assistance in connection with the siting, construction, commissioning, operation or decommissioning of a nuclear power plant will be required to follow those parts of the Codes of Practice and Safety Guides that pertain to the activities covered by the agreement. However, it is recognized that the final decisions and legal responsibilities in any licensing procedures always rest with the Member State.

The NUSS publications presuppose a single national framework within which the various parties, such as the regulatory body, the applicant/licensee and the supplier or manufacturer, perform their tasks. Where more than one Member State is involved, however, it is understood that certain modifications to the procedures described may be necessary in accordance with national practice and with the relevant agreements concluded between the States and between the various organizations concerned.

The Codes and Guides are written in such a form as would enable a Member State, should it so decide, to make the contents of such documents directly applicable to activities under its jurisdiction. Therefore, consistent with accepted practice for codes and guides, and in accordance with a proposal of the Senior Advisory Group, "shall" and "should" are used to distinguish for the potential user between a firm requirement and a desirable option.

The task of ensuring an adequate and safe supply of energy for coming generations, and thereby contributing to their well-being and standard of life, is a matter of concern to us all. It is hoped that the publication presented here, together with the others being produced under the aegis of the NUSS programme, will be of use in this task.

STATEMENT
by the Senior Advisory Group

The Agency's plans for establishing Codes of Practice and Safety Guides for nuclear power plants have been set out in IAEA document GC(XVIII)/526/Mod.1. The programme, referred to as the NUSS programme, deals with radiological safety and is at present limited to land-based stationary plants with thermal neutron reactors designed for the production of power. The present publication is brought out within this framework.

A Senior Advisory Group (SAG), set up by the Director General in September 1974 to implement the programme, selected five topics to be covered by Codes of Practice and drew up a provisional list of subjects for Safety Guides supporting the five Codes. The SAG was entrusted with the task of supervising, reviewing and advising on the project at all stages and approving draft documents for onward transmission to the Director General. One Technical Review Committee (TRC), composed of experts from Member States, was created for each of the topics covered by the Codes of Practice.

In accordance with the procedure outlined in the above-mentioned IAEA document, the Codes of Practice and Safety Guides, which are based on documentation and experience from various national systems and practices, are first drafted by expert working groups consisting of two or three experts from Member States together with Agency staff members. They are then reviewed and revised by the appropriate TRC. In this undertaking use is made of both published and unpublished material, such as answers to questionnaires, submitted by Member States.

The draft documents, as revised by the TRCs, are placed before the SAG. After acceptance by the SAG, English, French, Russian and Spanish versions are sent to Member States for comments. When changes and additions have been made by the TRCs in the light of these comments, and after further review by the SAG, the drafts are transmitted to the Director General, who submits them, as and when appropriate, to the Board of Governors for approval before final publication.

The five Codes of Practice cover the following topics:

Governmental organization for the regulation of nuclear power plants
Safety in nuclear power plant siting
Design for safety of nuclear power plants
Safety in nuclear power plant operation
Quality assurance for safety in nuclear power plants.

These five Codes establish the objectives and minimum requirements that should be fulfilled to provide adequate safety in the operation of nuclear power plants.

The Safety Guides are issued to describe and make available to Member States acceptable methods of implementing specific parts of the relevant Codes of Practice. Methods and solutions varying from those set out in these Guides may be acceptable, if they provide at least comparable assurance that nuclear power plants can be operated without undue risk to the health and safety of the general public and site personnel. Although these Codes of Practice and Safety Guides establish an essential basis for safety, they may not be sufficient or entirely applicable. Other safety documents published by the Agency should be consulted as necessary.

In some cases, in response to particular circumstances, additional requirements may need to be met. Moreover, there will be special aspects which have to be assessed by experts on a case-by-case basis.

Physical security of fissile and radioactive materials and of a nuclear power plant as a whole is mentioned where appropriate but is not treated in detail. Non-radiological aspects of industrial safety and environmental protection are not explicitly considered.

When an *appendix* is included it is considered to be an integral part of the document and to have the same status as that assigned to the main text of the document.

On the other hand *annexes, footnotes, lists of participants* and *bibliographies* are only included to provide information or practical examples that might be helpful to the user. Lists of additional bibliographical material may in some cases be available at the Agency.

A list of relevant *definitions* appears in each book.

These publications are intended for use, as appropriate, by regulatory bodies and others concerned in Member States. To fully comprehend their contents, it is essential that the other relevant Codes of Practice and Safety Guides be taken into account.

NOTE

The following publications of the NUSS programme are referred to in the text of the present Safety Guide:

Safety Series No. 50-C-D
Safety Series No. 50-SG-D3
Safety Series No. 50-SG-D8
Safety Series No. 50-SG-D9
Safety Series No. 50-SG-D10
Safety Series No. 50-SG-D11
Safety Series No. 50-SG-D12
Safety Series No. 50-SG-D13
Safety Series No. 50-SG-O2
Safety Series No. 50-SG-O8
Safety Series No. 50-SG-O10
Safety Series No. 50-C-QA

The titles are given in the List of NUSS Programme Titles printed at the end of this Guide, together with information about their publication date. Instructions on how to order them will be found on the last page of this Guide.

CONTENTS

1. INTRODUCTION

1.1. Purpose

The purpose of this Safety Guide is to identify certain reactor safety requirements and to provide a general design approach to the reactor core and its reactivity control. The Code of Practice on Design for Safety of Nuclear Power Plants (IAEA Safety Series No. 50-C-D), hereinafter referred to as the Code, establishes certain nuclear safety principles which define the minimum safety requirements for a nuclear power plant. Since these requirements are general in nature, more guidance is required to establish specific design requirements. The present Safety Guide aims to provide this additional guidance on the implementation of the Code. It should be noted that reactor core safety is achieved by a combination of proper design, manufacture and operation. For the operational aspects reference should be made to the Code of Practice on Safety in Nuclear Power Plant Operation, Including Commissioning and Decommissioning (IAEA Safety Series No. 50-C-O) and its associated Guides.

1.2. Scope

This Guide covers the neutronic, thermal, hydraulic, mechanical, chemical and irradiation considerations important to the safe design of a nuclear reactor core. The Guide applies to the types of thermal neutron reactor power plants that are now in common use and fuelled with oxide fuels: advanced gas cooled reactor (AGR), boiling water reactor (BWR), pressurized heavy water reactor (PHWR) (pressure tube and pressure vessel type) and pressurized water reactor (PWR). It deals with the individual components and systems that make up the core and associated equipment and with design provisions for the safe operation of the core and safe handling of the fuel and other core components.

The Guide discusses the reactor vessel internals and the reactivity control and shutdown devices[1] mounted on the vessel. Possible effects on requirements for the reactor coolant, the reactor coolant system and its pressure boundary (including the pressure vessel) are considered only as far as necessary to clarify the interface with the Safety Guide on Reactor Coolant and Associated Systems in Nuclear Power Plants (IAEA Safety Series No. 50-SG-D13) and other Guides. In relation to instrumentation and control systems the guidance is mainly limited to functional requirements.

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[1] In this Guide the term device (shutdown device or reactivity control device) is used when mainly a physical part inserted in the core is meant – such as control rods (of any shape, purpose and material), fluid-containing tubes for reactivity control, etc. The term may even embrace the drive mechanism for these parts. In contrast to this, the term 'means' (shutdown means or reactivity control means) is used to denote the functional aspect in a more general way.

1.3. Extent of the reactor core and associated equipment

The following hardware is covered by this Guide:

- The *reactor core* consisting of the fuel assemblies and those structures which hold the fuel assemblies in a predetermined geometrical configuration. The term also comprises the moderator and the coolant in the vicinity of the fuel.
- The *reactivity control and shutdown means*, comprising the neutron absorbers (solid or liquid), associated structure and drive mechanism or relevant fluid system components.
- *Support structures* that provide the foundation for the core within the vessel, the flow guide structure such as a core barrel or the pressure tubes of a PHWR (pressure tube type), guide tubes for reactivity control devices, etc.
- *Other vessel internals* such as instrumentation tubes, in-core instrumentation for core monitoring, steam separators and neutron sources. These are dealt with only to a limited extent in this Guide.

2. SAFETY DESIGN PRINCIPLES

2.1. General

Section 4 of the Code provides general safety principles for core design which are the basis for the more detailed design requirements contained in this Safety Guide.

As stated in the Code, the safety goals for the design of nuclear power plants are to contain and control all sources of radioactivity on the plant site, to ensure the safety of site personnel and the public, and to keep radiation exposure as low as reasonably achievable and within limits specified by the regulatory body. To achieve these goals a defence-in-depth approach is adopted whereby a series of barriers is introduced to impede the escape of radioactivity. The barriers are the following:

- the fuel matrix
- the fuel cladding
- the reactor coolant system pressure boundary (see IAEA Safety Series No. 50-SG-D13)
- the reactor containment system (see the Safety Guide on Design of the Reactor Containment Systems in Nuclear Power Plants (IAEA Safety Series No. 50-SG-D12)).

A more complete discussion of the defence-in-depth concept is given in the Safety Guide on General Design Safety Principles in Nuclear Power Plants (IAEA Safety Series No. 50-SG-D11).

Core design has a significant effect on achieving these goals by ensuring that radioactive materials are confined within the fuel matrix and the fuel cladding itself to the maximum extent practical. The design process requires an iterative consideration of neutronic, thermohydraulic, mechanical and chemical aspects. The specification of the basic features of the core design will depend on the status of the project. If a new reactor design concept is involved, there would be considerable interaction between the various aspects during the specification of the conceptual design of the plant. On the other hand, if the project represents a relatively small change from reactors previously designed, the basic requirements may be established readily on the basis of past experience. In any case, before the detailed analyses of a core design can begin, preliminary selection of a number of key factors is required. These include the size of the core, the number and the design of fuel assemblies, the required operating conditions, the material to be used in the core hardware, the required heat production, the reactivity control, the fuel management scheme, the heat rating of the fuel, the coolant flow rate across the fuel, the neutron flux peaking, etc. Some of these factors may well need to be changed as a result of the analyses and hence an iterative process takes place in order to satisfy the various requirements. From a safety point of view the design shall be such that the reactor power can be safely controlled and the core adequately cooled to maintain the fuel parameters within the limits specified for operational states and accident conditions.

A list of postulated initiating events (PIEs) shall be established[2] as a basis for reactor core safety design and analysis. The consequences shall be analysed with respect to core coolability, the integrity of the fuel and equipment associated with the core, and reactivity variations of the core.

A main goal of reactor core safety design is to limit the release of radioactive material from the fuel elements. For operational states the aim shall be to maintain fuel element integrity; for PIEs leading to accident conditions the aim shall be to ensure that the severity of fuel element damage remains acceptable. A basic safety design intent shall be to achieve, as far as practicable, reactivity behaviour characteristics of the core which are favourable to safety. Reactor core components and associated structures shall be designed taking into account safety functions to be achieved during and following accident conditions, e.g. reactor shutdown, emergency core cooling and long term stable cooling.

Regardless of the details of the approach taken, there are several basic design principles which are important for the achievement of the overall goals. These are discussed in the following sections.

2.2. Basic considerations for neutronic and thermohydraulic design

(1) The combination of inherent reactor neutronic characteristics, thermohydraulic characteristics and the control system capability shall be

[2] See Annex IV for examples of PIEs.

sufficient to provide adequate regulation of the reactor power for all operational states. (Information on inherent neutronic characteristics and reactivity coefficients is given in Annex I.)

(2) The reactor shall be capable of being shut down and held subcritical under operational states and accident conditions.

(3) Adequate provision for cooling the core under operational states and accident conditions shall be made and cooling effectiveness shall be proven by analysis or experiment.

(4) Assessments of the core power distribution, especially peak channel power and peak linear heat rating, shall be performed on an iterative basis during the design for representative operational states to provide bases for: a) operational limits and conditions; and b) operating procedures which would ensure compliance with design limits, including core design parameters, throughout reactor core life.

(5) Appropriate instrumentation and control means shall be provided so that parameters indicative of core conditions including fuel element integrity can be monitored and core conditions adjusted safely to ensure that design limits are not exceeded during operational states.

(6) The design of the core should take into account the fact that it is desirable to achieve a low demand on the control system for maintaining axial, radial and local power distributions within the limits stipulated for normal operation.

(7) Analytical models, data and computer codes used in the neutronic and thermohydraulic design of the core shall be based on adequate experiments or measurements applicable to the conditions expected.

(8) Thermohydraulic design limits on such parameters as minimum critical power ratio, minimum departure from nucleate boiling ratio (see Section 3.1.2.1), local cladding temperature and fuel temperature shall be set such that sufficient margins exist during operational states to keep fuel failures to an acceptably low value.

(9) Appropriate monitoring instrumentation shall be provided for assessing the state of the core during accident conditions.

2.3. Basic considerations for mechanical design

(1) The fuel elements and assemblies shall be designed to ensure that the cladding remains leaktight for operational states, as far as is practicable.

(2) Structural integrity shall be ensured as far as necessary so that the core can be safely controlled, shut down and cooled under operational states and accident conditions.

(3) All core and associated components shall be designed to be compatible with each other under the effects of irradiation, chemical and physical

processes and static and dynamic mechanical loads, including thermal stress, existing during operational states and accident conditions.

(4) Means shall be provided for safe handling of core components to ensure their integrity during transport, storage, installation and refuelling operations (see also the Safety Guide on Fuel Handling and Storage Systems in Nuclear Power Plants (IAEA Safety Series No. 50-SG-D10)).

(5) Means, preferably physical, shall be provided to inhibit the incorrect location in the core of any components important to safety, e.g. fuel assemblies and reactivity control or shutdown devices.

(6) Uncontrolled movement of reactivity control devices shall be prevented.

(7) High quality design and fabrication shall be ensured by the establishment and implementation of satisfactory quality assurance procedures (see also the Code of Practice on Quality Assurance for Safety in Nuclear Power Plants (IAEA Safety Series No. 50-C-QA) and its associated Guides).

(8) The design of the core, other reactor internals and the reactor cooling system shall minimize the chance of any obstruction of the coolant flow which could lead to core damage during any operational state.

3. CORE DESIGN REQUIREMENTS

In order to meet the design principles set out in Section 2 of this Guide, it is necessary to study the implications and limitations that these may impose on the design of the reactor core components. This is done in the present section. A subsection on core management is included since the fuel rating history, which is important in establishing the integrity of the fuel elements throughout their life, is clearly influenced by the strategies adopted in the fuel cycle.

Some hardware items may also perform safety functions within the scope of other Guides. Design of such hardware shall take into account the requirements and recommendations of this Guide as well as other applicable Guides such as IAEA Safety Series Nos 50-SG-D8 (Safety-Related Instrumentation and Control Systems for Nuclear Power Plants), 50-SG-D10 and 50-SG-D13.

3.1. Fuel elements and assemblies

3.1.1. Fuel element design requirements

The Code of Practice (Section 4.2) states: "The design of fuel elements shall be such that they will satisfactorily withstand their intended exposure in the reactor core despite all processes of deterioration that can occur."

It also requires that "Allowance shall be made for uncertainties in data, calculations and fabrication".

The following sections give the fuel element design requirements and considerations that are needed to meet this objective. They apply to fuel consisting of uranium oxides or a mixture of uranium and plutonium oxides.

3.1.1.1. Thermal effects

The evaluation of fuel temperatures in operational states should take account of changes in pellet thermal conductance and pellet–cladding gap thermal conductance due to effects such as oxide densification.

It is common practice to limit the fuel temperature during operational states to a level that is below the melting point. However, more restrictive operating limits may be imposed in view of effects of accident conditions such as a loss of coolant accident.

The strength and the corrosion behaviour of the cladding are very temperature dependent. Limits for stress, long term deformation and corrosion may therefore be specified for operational states. Under accident conditions the cladding temperature in water cooled reactors shall be limited to control ballooning and the zirconium–steam reaction. These effects shall not prevent safe shutdown and the maintenance of a shutdown condition.

3.1.1.2. Fission product effects

The design of fuel elements shall take into account the effects of solid and gaseous fission products during in-core residence. Fission gas migration from the fuel pellet and its effect on internal pressure and thermal conductance across the pellet-to-cladding interface shall be considered. The corrosive effects of fission products on the cladding shall also be considered in the design (see Section 3.1.1.6). Swelling of the fuel material due to fission products alters material properties such as thermal conductivity and causes dimensional changes; the design shall take these changes into account.

In safety analysis the consequence of reactor depressurization shall be considered in order to ensure that the gas pressure within the fuel element does not impose unacceptable loadings on the cladding. The likelihood of a cladding failure can be reduced by limiting the release of gas from the fuel matrix, providing a free volume within the fuel element to accommodate the gas and ensuring that the cladding strength will remain adequate during such an event.

3.1.1.3. Irradiation effects

The effects of irradiation, particularly by fast neutrons, on metallurgical properties such as tensile strength, ductility and creep behaviour, and on the geometrical stability of all materials, shall be considered in the design.

The burnup of uranium-235 and the production of plutonium result in changes in power distribution within the core and the fuel assemblies and in changes in the reactivity and reactivity coefficients of the core; these shall be taken into account in the core and fuel design.

3.1.1.4. Power variation effects

Local or global power variations during power transients caused by control device movements or other reactivity effects may lead to stresses on fuel pellets and cladding, i.e. a pellet–cladding interaction (see Section 3.1.1.6).

The effect of anticipated power transients on local heat rates shall be studied. To ensure acceptable fuel integrity, stresses and working cycles of cladding materials should be accommodated by the design with allowance for control system actions.

3.1.1.5. Mechanical effects on fuel elements

The fuel cladding can be designed to be collapsible or free-standing when subjected to coolant operating pressure. Collapsible claddings are rapidly pressed on to the fuel by the external pressure, and the outer cooler region of the fuel pellet supports the cladding throughout its life. The diametral gap between a collapsible cladding and fuel pellets shall be limited so that longitudinal ridges cannot form in the cladding.

Free-standing claddings can undergo a long term deformation (creep deformation) under external pressure, resulting in a decrease in the diametral gap between cladding and fuel.

Some cladding that is initially free-standing will eventually collapse and be supported by the pellets. In other cases, particularly with a low pressure coolant or pre-pressurized fuel elements, cladding collapse does not occur.

Stressing and straining of the cladding can be caused by fuel swelling or fuel thermal expansion due to an increase of local power or internal gas pressure and should be limited.

The allowable length of unsupported plenum or allowable size of axial gaps between fuel pellets, resulting from densification of the fuel, shall be determined for each design.

A discussion of cladding stress and strain due to pellet expansion and cracking is included in Annex II. Mechanical loads imposed on the fuel element by the fuel assembly are discussed in Section 3.1.2.

3.1.1.6. Pellet–cladding interaction

Pellet–cladding interaction is a particularly important consideration for fuel with zirconium alloy cladding because it has been the cause of fuel defects. The stress-corrosion cracking induced by the pellet-cladding interaction in the presence of fission products should be minimized. A discussion of pellet-cladding interaction control for zirconium alloy and steel-clad fuel is included in Annex II.

3.1.1.7. Effects of burnable poison in fuel elements

Where burnable poisons are mixed in the fuel to compensate for reactivity changes, they shall not affect the integrity of fuel elements. Due consideration shall be given to the change in thermal properties of fuel and to chemical, mechanical and metallurgical effects on both the fuel material and the cladding. Consideration should be given to the possibility that adding burnable poison may increase the release of volatile fission products from the fuel matrix. The effect of the burnable poison on the fuel and moderator temperature coefficients of reactivity and the effect on local power peaking factors shall also be taken into account.

3.1.1.8. Fluid environment of the fuel elements

Fuel elements and fuel assemblies shall be designed to be compatible with the normal fluid environment to which they are exposed during all modes of operation, including shutdown and refuelling. Environmental conditions include, but are not limited to, local boiling, pressure, temperature and chemical effects.

Changes caused by the environment which could increase the resistance to heat flow from the fuel element shall be considered in evaluating thermal characteristics. These include oxidation or other chemical changes (corrosion) at the external surface of the cladding and the deposition of matter (crud) on the surface of the cladding. The range of environmental conditions within which the fuel will operate under normal operating conditions should be considered in the definition of suitably conservative design parameters for surface oxidation and crud buildup. The design parameters used should be based on actual experience or experiments appropriate to operating conditions.

3.1.1.9. Hydrogen embrittlement

The hydrogen content of zirconium alloy cladding shall be limited in order to reduce the likelihood of fuel defects during operation, resulting from hydrogen embrittlement. On this account the moisture content in the free space of fuel elements shall be controlled.

3.1.2. Mechanical safety design requirements for fuel assemblies

3.1.2.1. Thermohydraulic effects within fuel assemblies

Fluid flow past the fuel elements is the mechanism by which the energy generated in the fuel element is transported from the core region. Steady state fuel assembly temperatures shall be limited so that there is no cladding degradation in the event of anticipated operational occurences. The designer shall take into account effects from element spacing, element power, subchannel sizes and shapes, grids, spacers, braces, flow deflectors or turbulence promoters. These effects are primarily thermohydraulic but potentially include localized corrosion, erosion and fretting. For water cooled reactors heat transfer coefficients drop if the surfaces become dry and fuel cladding temperatures can rise appreciably. Conditions are then termed 'critical'. The normal approach is therefore to ensure that the surfaces are always kept wet during steady state conditions. Critical heat flux conditions are avoided by maintaining local steady state power levels such that certain ratios or margins to critical heat flux conditions exist. The margins shall be sufficient to allow for anticipated operational occurrences[3]. 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. For operational states these ratios constitute a conservative basis for water cooled reactors.

Because of the importance of localized effects caused, for example, by fuel element spacers, the critical heat flux (CHF) and the critical power ratio (CPR) depend on detailed fuel assembly design. For this reason the CHF or CPR is usually determined experimentally over the range of conditions expected in actual operation. The test results are then analysed and converted into correlations for use in fuel assembly design and safety analyses.

3.1.2.2. Mechanical effects

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

- fuelling and refuelling
- power variation
- holddown loads for PWRs
- temperature gradients
- hydraulic forces
- bowing due to irradiation
- vibration and fretting induced by coolant flow
- external events such as earthquakes
- PIEs such as a LOCA

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[3] The objective of this requirement is to avoid cladding failures caused by high clad temperatures. Therefore in some Member States critical heat flux conditions during transients can be tolerated if it is shown by other methods that clad temperatures are not exceeding the acceptable limits.

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

(a) The clearance within and adjacent to the fuel assembly shall provide space for irradiation growth and swelling;
(b) Fuel element bowing shall be limited so that thermohydraulic behaviour and fuel performance are not significantly affected;
(c) The fuel assembly shall not fail because of strain fatigue;
(d) The fuel assembly shall withstand the holddown mechanical and hydraulic forces without unacceptable deformations;
(e) The fuel assembly and support structure functions shall not be affected unacceptably by vibration or fretting damage;
(f) The fuel assembly shall withstand the irradiation and shall be compatible with the coolant chemistry.

In the event of PIEs including earthquakes, explosions and equipment failures, the design of the fuel elements, fuel assemblies and fuel assembly support structures shall be such as to ensure that interactive or consequential effects from these components will not

- prevent functioning of safety system components, e.g. shutdown devices and their guide tubes
- impede cooling of the fuel
- damage unacceptably the reactor coolant system pressure boundary mechanically or thermally

to the extent that safety systems cannot perform their functions as claimed in the transient and accident analysis (see Section 3.9).

3.2. Coolant

The coolant is the fluid which transports the heat from the reactor core region. The heat transfer at the surface of a fuel element is a function of a number of variables, including fluid velocity, flow pattern, thermodynamic properties, etc., and is usually expressed in terms of empirically derived heat transfer correlations. Safety considerations associated with the coolant shall include:

(1) Ensuring that the coolant system is free from foreign objects and debris prior to initial reactor startup and maintaining it in that condition;
(2) Keeping the coolant radioactivity at an acceptably low level by the use of purification systems and by removal of defective fuel as appropriate (see also IAEA Safety Series No. 50-SG-D13, Section 4.5);

(3) Taking into account effects on reactivity of the coolant and coolant additives[4] in determining the capabilities of the reactor control and shutdown systems for operational states and accident conditions;
(4) Determining and controlling the chemical and physical characteristics of the coolant in the core to ensure compatibility with other components of the reactor core and minimize corrosion and contamination of the reactor coolant system;
(5) Ensuring a sufficient supply of coolant for operational states and accident conditions in order to meet specified fuel integrity criteria, including decay heat removal at shutdown (see also IAEA Safety Series No. 50-SG-D13, Sections 3.1, 4.3 and 4.5);
(6) Taking secondary effects of additives into account, e.g. chemical, physical and irradiation effects;
(7) Ensuring that where boiling occurs or can occur in operational states the core design shall prevent or control instabilities of flow and consequent fluctuations in reactivity;
(8) Ensuring that the core internals are so designed as to distribute the coolant in the appropriate proportions to the fuel assemblies and associated core structures so that the required cooling is provided.

3.2.1. Light water

In a pressurized water reactor (PWR), the water is maintained in a bulk subcooled state during normal operation. In a boiling water reactor (BWR), water enters the core subcooled but leaves it as a two-phase mixture of water and saturated steam.

The effects of coolant density changes (including fluid phase changes) on core reactivity shall also be considered in core design. In PWRs and BWRs the coolant also acts as the moderator; therefore any change in density will have an effect on core power, locally and globally. In order that the power coefficient remain negative, burnable poisons may be used. This can reduce the need for a soluble absorber, e.g. boric acid in the coolant.

Additives can be used to control the chemistry of the coolant, inhibit corrosion or reduce the contamination of the reactor coolant system. Additives can also be used as neutron absorbers to help control core reactivity; an example of this is the boron salts used in PWRs. Whenever additives are used their effects on core components shall be accounted for in core design.

Radiolysis of the coolant requires measures to control corrosion and prevent explosion as discussed in more detail in Section 4.4.2.1 of the Safety Guide on

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

Design Aspects of Radiation Protection for Nuclear Power Plants (IAEA Safety Series No. 50-SG-D9).

3.2.2. Heavy water

The relevant characteristics of heavy water are mostly similar to those of light water and the factors considered in Section 3.2.1 apply though it should be noted that coolant and moderator in some reactor design are separated. The use of additives in the coolant, whether for chemical purposes (e.g. pH control, oxygen control) or for reactivity holddown purposes, could affect the neutron absorption in the coolant or moderator. Any such effect shall be considered in the design of the reactor control and shutdown systems for operational states and accident conditions.

Radioactivation shall also be considered. Radioactive corrosion products and ^{16}N are found in light water and heavy water but tritium (^{3}H) builds up to a larger extent in heavy water. Therefore, in heavy water reactor coolant or moderator systems provisions shall be made to prevent or control the release of tritiated heavy water.

3.2.3. Carbon dioxide

Because of its low density and low neutron absorption, changes in carbon dioxide temperature and pressure have a negligible effect on reactivity.

3.3. Moderator

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

PWR BWR	– light water moderated
PHWR (pressure tube type) PHWR (pressure vessel type)	– heavy water moderated
AGR	– graphite moderated

3.3.1. Light water

Light water is used as a moderator and as a coolant in both pressurized water reactors and boiling water reactors; the two functions are not physically separated. The considerations regarding additives, reactivity characteristics, radiation effects, etc., discussed in Sections 3.2 and 3.2.1 are therefore applicable.

3.3.2. Heavy water

In reactors of the pressure tube type cooled and moderated by heavy water, the moderator is physically separated from the coolant by a calandria tube and a pressure tube. At times the moderator may contain a soluble neutron absorber either for reactivity control or for reactivity holddown after a shutdown. The moderator also serves to cool various reactor structures, e.g. the calandria vessel itself, instrumentation support structures, reactivity control devices and their guide tubes. Although highly unlikely, it is conceivable that the pressure tube and calandria tube might rupture, allowing a jet of heavy water coolant to be injected into the moderator region, i.e. some moderator will be displaced by coolant. If this occurs and the moderator contains absorber and the coolant does not, then it is possible that the reactivity of the core will increase. The shutdown system shall be designed to provide means to maintain the shutdown condition should such an accident occur. The effects of moderator flow and temperature, e.g. hydraulic forces, differential temperature, shall be considered in the design of the reactor structures.

With reactors of the pressure vessel type, cooled and moderated by heavy water, the moderator is separated from the coolant in the core region by the cooling channels. However, the coolant and moderator circuits may be separated or connected, depending on the operational status of the plant (e.g. power operation or residual heat removal). In the power operation mode the moderator is connected to the coolant by a small number of pressure compensating holes such that it can be operated at the same pressure but at a lower temperature than the coolant. The moderator temperature level is maintained by a separate high pressure cooling system. In residual heat removal operation the coolant and moderator systems are interconnected, so that there are no differences in pressure, temperature or liquid poison concentration.

A high specific tritium activity can build up in heavy water moderators. Therefore, the design of the moderator system shall take into account the possibility of a release of tritiated heavy water should a major breach occur in the moderator system.

Radiolysis of the moderator requires measures to control corrosion and prevent explosions, as discussed in detail in IAEA Safety Series NO. 50-SG-D13.

Under certain accident conditions, the moderator in a pressure tube reactor provides storage capacity for decay heat.

3.3.3. Graphite

The moderator adopted for advanced gas cooled reactors is graphite. In these reactors the graphite core is composed of bricks with a keying system which maintains the lattice alignment. The core assembly is provided with a restraint structure which maintains the external configuration. The safety issues with this moderator are as follows.

(1) Entry of the shutdown devices into the core and maintenance of the shutdown condition shall not be impeded. In order to confirm that this condition holds, an assessment shall be made of the ability of the graphite to hold the core in a stable position without failure due to the effects of:

- temperature
- corrosion
- fast neutron damage
- irradiation
- dimensional changes.

Postulated earthquake conditions impose limits on deformation and strength characteristics and these shall be taken into consideration.

For the initial core, the temperature coefficient of the moderator has a value near zero, typically slightly negative so that the cold core condition is the most reactive. For the equilibrium core, plutonium has built up to some extent in most of the fuel channels and the moderator temperature coefficient is positive. The most reactive condition for the equilibrium core is therefore assumed to be associated with the moderator at its hot operating temperature, even for a reactor maintained in the shutdown condition. During transient conditions, however, the significance of the positive temperature coefficient is limited by the slow response time of the moderator temperature compared with that of the fuel, for which the temperature coefficient is negative.

(2) The release of radioactive materials into the coolant circuit should be kept as low as practical. In order to achieve this, the impurities within the graphite should be limited (particularly Mg, Cl and B).

(3) The integrity of the moderator should be assured over the reactor design lifetime. Methane shall be added to the CO_2 coolant to inhibit corrosion, but secondary effects shall be taken into account and CO level shall be controlled.

(4) The design of the distribution holes drilled through the graphite bricks and of the interbrick passages should be chosen to limit the peak graphite temperatures by providing an adequate distribution of the coolant to all bricks, not only for conditions in the initial core but for all anticipated behaviour of brick shrinkage or growth throughout the life of the core.

3.4. Reactivity control means

This section discusses the reactivity control means for normal operation referred to in Section 4.3 of the Code. The control of reactivity for reactor shutdown is considered in Section 3.6 of the present Guide.

Reactivity control means shall be designed to enable power and power distribution to be regulated safely. This includes compensating for reactivity changes (such as those associated with xenon concentration changes, coolant temperature change, burnup of fuel and burnable poison, anticipated operational transients) in order to keep the reactor process variables within specified operating limits.

The instrumentation and control systems used shall meet the requirements of IAEA Safety Series No. 50-SG-D8.

3.4.1. Types of reactivity control means

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

- moderator temperature (pressure vessel type PHWR)
- moderator height (pressure tube type PHWR)
- coolant flow (moderator density) (BWR)
- soluble absorber in the moderator or coolant (PWR, PHWR)
- solid neutron absorber rods or liquid absorber in tubes (PWR, BWR, AGR, PHWR)
- fuel with distributed or discrete burnable poison
- fuel assembly axial movement
- refuelling and loading pattern.

3.4.2. Maximum reactivity worth and reactivity insertion rate

The arrangement, grouping, speed of withdrawal and withdrawal sequence of the reactivity control devices, used together with an interlock system, shall be designed to ensure that any credible abnormal withdrawal of the devices does not cause the specified fuel conditions to be exceeded. The maximum reactivity worth of the reactivity control devices shall be limited, or interlock systems shall be provided, so that for an accident condition such as 'control rod ejection' in PWRs or 'control rod drop' in BWRs the resultant power transient does not exceed specified limits. These limits shall be chosen so as to ensure acceptably low levels of:

(a) Damage to fuel and cladding which could produce releases of radioactivity into the coolant circuits; and

(b) Risk of a molten fuel–coolant interaction which could damage the core structure and prevent successful insertion of the shutdown devices.

If necessary, the maximum reactivity worth of the control devices shall be evaluated for each refuelled core.

For soluble absorber, the control system shall be designed so that any depletion of absorber concentration in the core does not cause the specified fuel conditions to be exceeded. All portions of systems that contain boric acid shall be designed to prevent precipitation, e.g. by heating of the components containing the boric acid solution (see IAEA Safety Series No. 50-SG-D13, Section 4.5).

3.4.3. Control of global and local power

Core power shall be controlled globally and locally by reactivity control means in such a way that the peak linear rating of the fuel and channel power will not exceed design limits anywhere in the core. The control system design shall take into account variations in power distribution caused by local variations in reactivity due to xenon instability, changes of coolant conditions, changes in the position of the in-core detectors and changes in the characteristics of the in-core detectors themselves. Further information is given in Section 3.8 and Annex III.

3.4.4. Effect of burnable poison

The reactivity increase caused by burnup of burnable poison in the core shall be evaluated and accommodated by other reactivity control means.

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

3.4.5. Irradiation effects

Effects of irradiation such as burnup, changes in physical properties, gas production and contamination of liquid loop boundary shall be taken into account in the design of reactivity control means.

3.5. Core monitoring system

Instrumentation shall be provided to monitor core parameters such as core power (level, distribution and temporal variation), the physical states of the coolant and moderator and the status of reactivity control means, so that any necessary corrective action can be taken. The level of fission product radioactivity in the coolant shall be monitored to verify that design limits are not exceeded. Some designs use systems that can indicate the location of failed fuel assemblies during power operation. Failed fuel location monitors are particularly effective for reactor types that employ on-load refuelling since there is the possibility of removing defective fuel more easily and thereby keeping radioactivity levels in the coolant low. Another advantage of a failed fuel monitor is that it can give early warning of coolant flow blockage or other physical damage.

The accuracy, speed of response, range and reliability of all monitoring systems shall be adequate for the performance of the functions for which they are intended (see the Safety Guide on Protection Systems and Related Features in Nuclear Power Plants (IAEA Safety Series No. 50-SG-D3) and also IAEA Safety Series No. 50-SG-D8).[5] The design shall also incorporate facilities that allow for continuous or periodic testing of monitoring systems, as required.

Guidance on accident monitoring can be found in IAEA Safety Series No. 50-SG-D8, Section 4.9.3.

In the case of large cores it may be necessary to monitor the spatial power distribution, for example by in-core neutron detectors or gamma thermometers. Measurements of local power at different positions in the core have the purpose of ensuring adequate safety margins and providing data for optimum utilization of the fuel. In this case detectors shall be distributed in such a way as to reduce as far as practicable the possibility that a local excessive power density buildup will go undetected.

Many parameters such as:

- neutron flux
- coolant temperatures
- water level
- system pressure
- radioactivity in the coolant

are monitored at various locations for safety purposes.

Other safety related parameters are derived from the monitored parameters. Examples are:

- neutron flux doubling time
- neutron flux rate of change
- flux difference across the core
- reactivity
- subcooling across the core.

The selection of parameters to be monitored depends on the reactor type. The necessary redundancy, diversity and independence of the signals and their transmission paths shall be ensured by the design according to the requirements of IAEA Safety Series Nos 50-SG-D3 and 50-SG-D8.

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 a particular flux range.

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[5] Further guidance is given in the publications on In-Core Instrumentation for Neutron Fluence Rate (Flux) Measurements in Power Reactors, IEC Publication 568, Geneva (1977) and on In-Core Temperature or Primary Envelope Temperature Measurements in Nuclear Power Reactors: Characteristics and Test Methods, IEC Publication 737, Geneva (1982).

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

- the sources function properly for their planned lifetime, and
- the sources are compatible with the fuel assemblies and the fuel assembly support structures.

Analysis of neutronic and acoustic noise may provide useful information on loose parts or incipient mechanical failure of core components.

3.6. Reactor shutdown means

This section deals with the means (see Section 4.4 of the Code) of rendering the reactor subcritical in operational states or accident conditions and of maintaining it in that state.

Means shall be provided to ensure that the reactor can be rendered subcritical and held in this state, assuming the most reactive core conditions when one of the shutdown devices that have the maximum effect on core reactivity cannot be inserted into the core (one rod stuck). For operational states and accident conditions, specified fuel and reactor coolant system pressure boundary conditions shall not be exceeded.

As required by the Code, the means of shutting down the reactor shall consist of two diverse systems, each being able to perform its function assuming a single failure. At least one of the systems shall be, on its own, capable of rendering the reactor subcritical by an adequate margin from operational states and accident conditions with a response such that in combination with the performance of other systems no unacceptable fuel damage occurs. At least one of these systems shall be, on its own, capable of rendering the core subcritical from normal operating conditions, and shall provide adequate long term holddown following the reactor trip, even in the most reactive condition of the core.

In meeting the long term holddown requirements, deliberate actions that increase reactivity during the shutdown state, such as absorber movement for maintenance and refuelling actions, shall be identified to ensure that the most reactive condition is taken into account.

The design of the shutdown systems shall recognize the importance of reactor shutdown following anticipated operational occurrences and during accident conditions. The necessary reliability shall therefore be incorporated in the design of the equipment to effect shutdown for all postulated initiating events so as to meet the safety requirements. The designs shall incorporate the necessary independence from plant process and control systems and protection from the consequential effects of the postulated initiating events such that the shutdown will be performed as required.

The means of shutdown shall be designed fail-safe as far as practical and shall be engineered to the high reliability required for such safety systems. If operation of the holddown system is manual or partly manual, the necessary prerequisites for manual operation shall be met (see IAEA Safety Series No. 50-SG-D3, Section 7.3.2).

A portion of the shutdown means may be used for the purposes of reactivity control and flux shaping during normal operation (see Section 3.4). Such use during normal operation shall not jeopardize the function of the shutdown system. For a more detailed discussion, see Section 7.8.4 of IAEA Safety Series No. 50-SG-D3.

3.6.1. Types of shutdown means

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

- boron injection into moderator
- gadolinium injection into moderator
- nitrogen injection
- moderator dump
- boron and cadmium in stainless steel rods, tubes, or cruciforms
- hafnium and steel rods in Zircaloy guide tubes
- boron glass bead injection
- liquid absorber in tubes.

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

3.6.2. Reliability

High reliability of shutdown shall be achieved by using a combination of measures such as:

(1) Adopting systems that are as simple as possible;
(2) Using a fail-safe design as far as practicable[6];
(3) Giving consideration to modes of failure and adopting redundancy and diversity in the initiating mechanisms (e.g. sensors, actuation devices that detect and respond to the need for a reactor trip);
(4) Functionally isolating and physically separating the shutdown systems (this includes separation of control and shutdown functions) as far as

[6] The simplest form of fail-safe design which is often used is that the shutdown devices are held above the core by active means. Providing that the shutdown device guide structures are not obstructed, the devices drop into the core by gravity in the event of de-energization of the active holding means, e.g. loss of current through a holding electromagnet.

TABLE I. SHUTDOWN MEANS

Reactor type	Primary system	Secondary system
BWR	B_4C in steel tubes	Boron solution injected into moderator/coolant
PWR	Ag-In-Cd in steel tubes or B_4C in steel tubes	Boron injected into moderator/coolant
PHWR	Cadmium sandwiched in steel tubes	Gadolinum injected into moderator; moderator dump; liquid absorber in tubes
PHWR (pressure-vessel type)	Hafnium and steel rods in Zircaloy guide tubes	Boron injected into moderator
AGR	Boron steel rods + stainless steel rods	Nitrogen injection into coolant within the core and boron glass beads injection into the core

practicable, to cater for credible modes of failure, including common cause[7];

(5) Ensuring easy entry of shutdown means into the core taking into account the in-core environmental effects of operational states and accident conditions;
(6) Designing to facilitate maintenance, in-service inspection, and operational testing;
(7) Selecting equipment of proven design and high reliability;
(8) Providing means for performing comprehensive testing during manufacture, installation and commissioning.

3.6.3. Shutdown and holddown effectiveness

The design shall ensure the capability of the shutdown and holddown systems to render and hold the reactor subcritical by an adequate margin even in the most reactive core conditions. This shall hold for the whole range of operating conditions and core configurations that occur throughout the intended fuel cycle and for anticipated operational occurrences and accident conditions, so that accep-

[7] Some Member States require that there be two independent and different shutdown systems, each of which is adequate when acting alone.

table fuel cooling and radioactivity release criteria can be met. It shall be possible to demonstrate this

- during design by calculation;
- during commissioning by appropriate neutronic and process measurements to confirm the calculations for start of life;
- during reactor operation by measurements and calculations covering the existing and anticipated reactor conditions.

These analyses and measurements shall cover the most reactive core conditions, with the assumption that one shutdown device of the highest reactivity worth cannot be inserted into the core. In addition, holddown shall be achieved if a single random failure occurs in the shutdown system. However, there is considerable variation among Member States on what subcriticality margin is accepted as adequate[8].

The number and reactivity worth of shutdown devices required in the systems is largely determined by:

(1) The required subcriticality margin;
(2) The deterministic requirement to calculate shutdown reactivity with the assumption that the device with the greatest negative reactivity worth shall be regarded as not inserted;
(3) The uncertainties associated with the calculations. These may be estimated from comparison of calculations with measurements made in experimental reactors and prototype reactors and during reactor commissioning.
(4) The distribution of the shutdown devices within the core. This may influence the reactivity worth of that shutdown device which was to be disregarded in the calculation of the subcriticality margin. The distribution may also influence the reactivity worth of installed (fresh) fuel assemblies.
(5) The most reactive core conditions after shutdown. This is the result of a number of factors such as:
    - the most reactive core configuration (and where appropriate, the corresponding boron concentration) that will occur during the intended fuel cycle, including refuelling ahead of schedule
    - the most reactive credible combination of fuel and moderator temperatures
    - the rate of reactivity insertion taking accident conditions into account
    - the amount of xenon as a function of time after shutdown
    - absorber burnup.

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[8] In many cases, a subcriticality margin of 1% is specified. For the AGR it is 2% with three rods in the array assumed not to be inserted. In France the margin is 10% when the reactor vessel is opened for fuelling.

3.6.4. Rate of shutdown

The rate of shutdown for at least one of the systems shall be adequate to render the reactor sufficiently subcritical in time to prevent fuel damage and to maintain the pressure boundary integrity in all anticipated operational occurrences. The shutdown system shall be designed to shut the reactor down under accident conditions so as to keep fuel and core damage to a practical minimum and prevent the failure of the reactor coolant system pressure boundary.

For the design basis (see Section 2.1), the course of the postulated initiating events to be considered in detail, the response of the protection system and the associated safety actuation systems (shutdown means) shall be established in defining the shutdown rate requirement. The selection of variables for the sensing of these postulated initiating events shall meet the requirements of Sections 7.7 and 7.13 of IAEA Safety Series No. 50-SG-D3.

The rate of shutdown is dependent on the following:

(1) Ability of the instrumentation to recognize and respond to the need for a reactor trip. This requires a choice of instrumentation to adequately cover the range of postulated initiating events.

(2) Response time of the actuation mechanism of the shutdown means. This may govern the choice of mechanism, though the response times are usually relatively short compared with other time factors.

(3) Location of the shutdown devices. The rate is sensitive to:
- the distance of the shutdown devices from the core prior to insertion (see Section 3.6.5); and
- the location of absorber injection nozzles which shall be such that the absorber may be quickly dispersed in the active region of the core.

(4) Ease of entry of the shutdown devices into the core. This may be achieved by the use of guide tubes or other structural means (see Section 3.7) to facilitate device access and the possible incorporation of flexible couplings to reduce rigidity over the length of the devices.

(5) Insertion speed of the shutdown means. One or more of the following may be used to provide the required speed:
- gravity drop of shutdown devices into the core;
- gravity drop of shutdown devices into the core initiated by a spring;
- hydraulic or pneumatic pressure drive of shutdown devices into the core;
- hydraulic or pneumatic pressure injection of soluble absorber.

Design provisions shall be made for testing to check the speed of insertion of shutdown devices. Ease of insertion can be checked by monitoring with appropriate sensors on the shutdown devices (weight sensors in the suspension chain are used in AGRs).

The capability of the shutdown systems shall be assessed as part of the safety analysis described in Section 3.9.

3.6.5. Environmental considerations

In order that the integrity of shutdown systems not be jeopardized during reactor life, the effects of their environment inside the reactor shall be considered. The effects considered shall include:

(1) *Irradiation effects.* If devices used for shutdown are poised in a high neutron flux or are used for reactivity control, the effects of absorber (e.g. boron) depletion shall be considered in their design. Depletion of boron is accompanied by helium production. If helium buildup in devices containing boron can result in swelling, it shall be ensured that the performance of the devices is not thereby impaired.

(2) *Temperature effects.* The effects of heating of shutdown devices as a result of neutron or gamma absorption shall be considered.

(3) *Chemical effects.* The effects of chemicals in the external fluid environment, i.e. coolant or moderator, on corrosion rates and the physical integrity of shutdown devices, as well as the transport of activated corrosion products throughout the reactor coolant and moderator system shall be considered.

(4) *Structural dimension changes.* Dimensional changes and movements of internal core structures due to temperature changes, irradiation or external events such as earthquakes shall not prevent entry of a sufficient number of the shutdown means into the core (see Section 3.7).

3.7. Core and associated structures

The scope of this section includes the structures which form and support the reactor core assembly and which are closely associated with the performance and the safety of the reactor core.

The core and associated structures shall be designed such that their integrity is maintained during operational states and accident conditions to the extent that the required safety functions can be performed.

Possible damage mechanisms which could affect the core and associated structures and need to 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 fuel coolability, damage to fuel, and damage to the reactor coolant pressure boundary. The effects of pressure, temperature, temperature variation and distribution, corrosion, radiation dose rates and lifetime dose on dimensional changes, mechanical loads and material properties shall be considered.

The radiation heating of the structures shall be calculated and proper cooling shall be provided. Proper allowance shall be made for thermal stresses during operational states and accident conditions. Chemical effects of coolant or moderator on the structures shall be considered.

Provisions for the necessary inspections and replacements of the core and associated structures shall be included in the design.

3.7.1. Reactor coolant pressure boundary

Certain aspects of the reactor coolant pressure boundary design are also related to the core structure design. The core portion of the reactor coolant pressure boundary can be either:

(a) A pressure vessel surrounding the total core, including the fuel assemblies, their support structures, and the moderator and reactor coolant; or
(b) An assembly of individual pressure tubes each forming a fuel channel. The low pressure liquid moderator surrounds the pressure tubes.

The pressure vessel of AGRs, LWRs and PHWRs is a large thick walled structure surrounding the core with penetrations to accommodate the reactor coolant, the instrumentation, and the reactivity control and shutdown devices which reside in the high pressure reactor coolant region. The core assembly and other components are arranged so as to reduce the neutron flux at the pressure vessel wall.

The pressure boundary of pressure tube PHWRs is made up of a number of thin walled cylindrical tubes with no wall penetrations, since the instrumentation and reactivity control and shutdown devices reside in the low pressure moderator region. The reactor coolant pressure boundary is inside the active core and subject to the neutron and gamma fluxes of the core centre.

Pressure tubes and vessels shall meet the support structure configuration requirements of this section and the pressure boundary design requirements of IAEA Safety Series No. 50-SG-D13. Pressure tubes shall also meet the fuel assembly support structure requirements of Section 3.7.3.

3.7.2. Reactor core assembly support structures

The reactor core assembly support structures comprise tube sheets, a core barrel, graphite keying system, etc., depending on the reactor design, and hold the fuel assembly support structures in the desired geometrical relationship with the reactor coolant pressure boundary. These support components shall be designed to remain intact to the extent necessary to perform their functions throughout the life of the reactor for operational states and accident conditions. Mechanical loads such as those induced by hydraulic forces and by normal and postulated abnormal refuelling shall be considered. Seismic conditions, as specified, shall be taken into account.

3.7.3. Fuel assembly support structures

The fuel assembly support structures shall be designed to hold the fuel assembly in the desired geometry for operational states and accident conditions.

In pressure tube type HWRs, the fuel assembly support structures are the pressure tubes themselves because they contain the fuel assemblies (bundles) in the reactor core.

Design considerations relating to the pressure tubes and their connection to the end fittings including closure plugs are covered in IAEA Safety Series No. 50-SG-D13. The following additional factors shall be considered:

(1) Irradiation and creep of the pressure tubes which result in diameter and length changes and possible effects on fuel cooling.
(2) Fretting effects on the pressure tube and sliding wear effects due to fuel assembly movements during refuelling.

3.7.4. Shutdown and reactivity control device guide structures

The structures that guide the shutdown devices shall be designed to perform their required functions under operational states and accident conditions. The structures that guide the reactivity control devices that are used only for reactor control and are not required for shutdown shall be designed to perform satisfactorily at least for operational states.

Since shutdown and reactivity control device guide structures are in close proximity to the fuel assemblies or fuel channels, the possibility of physical interaction and damage during operation and shutdown, and in accident conditions shall be carefully considered during design. In the case of shutdown and reactivity control devices immersed in a bulk moderator, the effects of forces due to flow currents shall be considered and the maximum allowable distortion shall not be exceeded. For graphite moderated reactors the maximum distortion due to fast neutron damage shall be evaluated and the design arranged to accommodate it.

The design shall allow for removal of reactivity control and shutdown devices which have become damaged or separated from the drive mechanisms, so that there will be no danger of damage to the reactor core, no unacceptable reactivity effects, and no personnel radiation exposure greater than that allowed by the regulations.

3.7.5. In-core instrumentation support structures

The structures and guide tubes containing instrumentation within the core, and in close proximity to the core, shall be so designed that they perform their functions during all operational states and accident conditions.

The structure and the tubes shall be designed so that the detector can be located with the required accuracy and so that it will not be moved from its location inadvertently by operator action, equipment strain, coolant flow forces or bulk moderator movements, during operational states and accident conditions. The design shall facilitate replacement of the detector as necessary.

3.7.6. Other vessel internals

Depending upon the reactor type, various other structures will be installed within the pressure vessel. These include feedwater spargers, steam separators, steam dryers, core baffles, reflectors and thermal shields. The functions of these other internals include reactor coolant flow distribution, steam moisture separation, protection of the pressure vessel from both the heating effects of gamma radiation and the effects of neutron irradiation.

These structures shall be designed so that their performance for the necessary period of time in the imposed environmental conditions is acceptable to any associated equipment which is important to safety.

3.7.7. Decommissioning considerations

When a nuclear power plant is to be decommissioned at the end of its normal life the radioactive fuel and reactivity control and shutdown devices can be disposed of in the same way as during the life of the reactor. This leaves the core structure, the support structures and the moderator to be considered. These shall be designed so as to facilitate disposal and to ensure that the radiation exposures of the general public and decommissioning personnel are kept as low as reasonably achievable and do not exceed prescribed limits.

3.8. Core management

There is a close relationship between certain aspects of reactor safety and the economic utilization of the fuel. The essential objective of core management is to ensure safe operation of the fuel in the reactor, taking into account the restraints imposed by the design of the fuel and of the entire plant.

The fuel performance objective is to choose a fuel cycle with appropriate enrichments and means to control reactivity and power distribution so that energy can be extracted from the fuel in the most economic manner within the design limitations. These limitations are set with due reference to the safety limitations associated with operational states and accident conditions (see Section 3.9). The various means available for achieving this performance objective are given in Annex III together with the implications of shutdown requirements.

3.8.1. Safety limitations

The design for core management shall take into account the specified design limits for normal operation.

For operational states the goal is that no cladding failures should occur. However, certain conditions, e.g. fuel element manufacturing defects or unexpected operational transients during operational states, may make it extremely difficult to meet this no-failure goal. In practice some fuel cladding failures can be accepted during operational states since the concentration of radioactive material in the reactor coolant systems will be reduced by the reactor coolant cleanup function. This function and other design provisions shall ensure that releases to the environment remain within prescribed limits.

For accident conditions the permissible degree of fuel failure depends upon the likelihood of the conditions and the associated radiological consequences. In some instances the number of acceptable fuel failures under accident conditions could require that operational limits be placed on the fuel which are more restrictive than those resulting from normal operational demands, e.g. fixing the minimum departure from nucleate boiling ratio for normal operation at a value low enough to avoid the cladding becoming dry during accident conditions in water cooled reactors.

3.8.2. Design information for reactor operation

In order to achieve the desired core reactivity and flux distribution for reactor operation, the core management programme shall provide the reactor operators as appropriate to the reactor design with: the pattern of fuel assemblies to be loaded for the initial core; the subsequent schedule for unloading and loading of fuel assemblies; the fuel assemblies to be shuffled; and the configurations of reactivity control and shutdown devices, burnable poisons, flux shaping absorbers and other core components to be removed, inserted or adjusted.

A description of the refuelling operations and of the evaluation needed to confirm that the safety requirements are met are set out in IAEA Safety Series No. 50-SG-O10.

3.8.3. Reactor core analysis

In many cases the safety parameters affecting fuel utilization, such as fuel and cladding temperatures and peak linear heat rating, are not directly measurable and available to the reactor operator. This requires that an analysis of reactor conditions be carried out in order to specify reactor operating procedures which will ensure compliance with fuel design limits expressed in terms of measurable parameters. Sufficient instrumentation shall be provided so that the analysis can be adequately supported by measurements.

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

- measurements in experimental reactors
- measurements in prototype reactors
- in-pile measurements on prototype assemblies under simulated conditions

- operating data gathered from reactor cores of similar design
- measurements made during the commissioning of reactors
- post-irradiation measurements on fuel elements and assemblies to evaluate the fine structure and burnup effects
- calculations by other codes which have been verified.

A reactor analysis shall be carried out by the designer at appropriate stages to ensure that the reactor operational strategy and limitations are sufficient to meet the design requirements before commissioning and throughout reactor life.

The analysis should therefore cover cases typical of the whole fuel cycle for the following reactor conditions:

- full power, including representative power distributions
- load following
- approach to criticality and power operation
- power cycling
- startup
- refuelling
- shutdown (decay heat removal)
- anticipated operational occurrences.

In order to derive peak channel power and peak linear power rates for normal full power operation, steady state power distributions shall be calculated for each assembly location and axially along the fuel assemblies. In order to identify hot spots, the radial power distribution within a fuel assembly and axial effects resulting from spacers, grids and other components should then be superimposed. By means of a series of such calculations, the power and temperature as a function of enrichment and burnup throughout the life of a fuel element should be determined and separately assessed to establish that the integrity of the fuel is not impaired by the effects listed in Section 3.1.

The effects of operating conditions such as load following, power cycling, reactor startup and refuelling shall, where necessary, be imposed onto the rating and temperature histories described in this section, so that the effects of thermal cycling on such parameters as fission gas pressure and fuel cladding fatigue can be evaluated.

3.8.4. Fuel handling systems

In order to prevent an unacceptable release of radioactivity during refuelling, the refuelling systems shall be designed to prevent unacceptable handling stresses on the fuel and the inadvertent dropping onto the core of heavy objects such as spent fuel casks or cranes. The refuelling systems shall also be designed to prevent an unacceptable release of radioactive material during the transit of failed fuel.

For on-load refuelling, the integrity of the reactor coolant system pressure boundary shall be maintained at all times. The effects of the refuelling operation on the neutronic behaviour of the reactor shall be within the capability of the reactor control systems.

Further information on fuel handling and storage systems is given in IAEA Safety Series Nos 50-SG-D10 and 50-SG-O10.

It is important to ensure that the fuel assemblies be loaded into the intended positions in the core. This requires administrative controls to make certain that individual assemblies are clearly identified and that correct loading into the core is verified.

In addition, the following measures may be taken:

- Reactivity monitors may be used to detect the enrichment level and check against that required.
- Mechanical means may be employed to prevent the entry of high enrichment assemblies into regions where lower enrichments are required.

An ultimate verification of the fuel loading pattern is provided by measurements of the in-core flux distribution.

3.9. Transient and accident analysis

The analysis of postulated initiating events relative to nuclear power plant and core behaviour (including credible combinations of events such as equipment failures, operational errors, external natural and man-induced events) shall be carried out in accordance with the Safety Guide on General Design Safety Principles for Nuclear Power Plants (IAEA Safety Series No. 50-SG-D11). The results shall be taken into account in the design of the core.

3.9.1. Postulated initiating events

The postulated initiating events (see Annex IV) and event sequences vary for different reactor designs and the reactor response to them also varies widely (as can be understood from Annex I, where coefficients of reactivity are considered). The postulated initiating events considered shall include failure of a shutdown system as discussed in Section 6.3.2.2(6) of IAEA Safety Series No. 50-SG-D11. The specified limits for core design corresponding to the various event sequences shall be consistent with the likelihood of the occurrence and the radiological consequences associated with each event.

3.9.2. Analysis

Studies shall be carried out to investigate the transient behaviour for postulated initiating events and event sequences in order to establish that the subsequent fuel conditions do not exceed allowable limits. These evaluations should use either a conservative bounding approach for important parameters or a realistic (best estimate) approach including evaluation of uncertainties. In a best estimate analysis, it is general practice to study the sensitivity of the results to variations in various parameters. (See IAEA Safety Series No. 50-SG-D11 for a general description of nuclear safety principles.)

Some event sequences, such as relatively slow changes in coolant flow rate or moderator density, can be analysed with steady state methods. Postulated initiating events which involve more rapid changes in parameters and which require action by reactor protection systems require more sophisticated transient analysis methods. Frequently, the complexity of the postulated initiating event and the subsequent safety actions will require a step-by-step analysis in which separate models are used for various components or parts of the reactor and the input for one model is given by the output of another.

The major factors influencing these assessments include:

- operating state (e.g. subcritical, part load, full load)
- fuel temperature coefficient of reactivity
- coolant and moderator temperature coefficients of reactivity
- coolant and moderator void coefficients of reactivity
- rate of change of soluble absorber concentration in moderator and coolant
- positive reactivity injection rate caused by reactivity control device or process parameter changes
- negative reactivity insertion rate associated with reactor trip
- individual channel transient response related to the core average thermal power
- the performance characteristics of safety system equipment including the changeover from one mode of operation to another, e.g. from the emergency core cooling injection mode to the recirculation mode.

Areas of uncertainty should be handled by using conservative assumptions in the analysis or by adding a margin for uncertainty (usually 2 sigma, i.e. twice the standard deviation) to the input parameters used. These uncertainties include both random and systematic components to cover probabilistic, statistical and physical uncertainties.

Core transient and accident analysis is used to determine whether fuel element integrity will remain within acceptable limits. All analysis ultimately involves some thermal analysis of individual fuel assemblies and fuel elements. The fuel design shall be shown to be such that the occurrence of an anticipated operational transient will not require the imposition of additional restrictions on the use of fuel that was in the reactor during the transient. For accident conditions where some fuel element damage may be allowed, the analysis should assume that this damage occurs during the transient. The effect on core cooling of such conditions as ballooning and cladding rupture, exothermic metal–water reactions and fuel element distortions should be included in the analysis. The formation of hydrogen as a result of a metal–water reaction should also be taken into account (see the Safety Guide on Design of the Reactor Containment Systems in Nuclear Power Plants (IAEA Safety Series No. 50-SG-D12)).

The analysis may lead to operational restrictions in order to ensure that the design limits for fuel are satisfied.

The methods that are used should be verified against experiments to the extent practicable. For analysis of transient behaviour, and particularly of the more severe transients, directly applicable experimental data may not be available. In these cases a comparison with results obtained from different computer models and codes which have been verified for less severe transients may be required for validation.

4. QUALIFICATION AND TESTING

The safety principles established for the core design shall be fulfilled throughout the life of the core structures and components. This objective can be achieved by applying the principles discussed below.

4.1. Equipment qualification

A qualification programme shall confirm the capability of the reactor core equipment to meet, for the relevant time period, the appropriate functional and safety requirements while subject to given environmental conditions (e.g. pressure, temperature, radiation, mechanical loading, vibration). These environmental conditions shall include the variations expected during normal operation, anticipated operational occurrences and accident conditions.

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

Methods of qualification may include:

(1) Performance of a type test on equipment representative of that to be supplied
(2) Performance of a test on supplied equipment
(3) Application of pertinent past experience
(4) Analysis based on available test data or extrapolation of such data
(5) Any combination of the above methods.

4.2. Provision for inspection and testing

As specified in Section 2.9 of the Code, structures, systems and components important to safety shall be designed to accommodate testing, inspection or monitoring for functional capability during their life without undue radiation exposure of site personnel.

Design provisions shall be made for in-service testing and inspection to ensure that the core assembly and the reactivity control and shutdown system equipment will meet their intended functions during their lifetime. The objectives and the methods of in-service inspections are covered in more detail in the Safety Guide on In-Service Inspection for Nuclear Power Plants (IAEA Safety Series No. 50-SG-O2). Guidance and recommendations on in-service monitoring and testing are given in the Safety Guide on Surveillance of Items Important to Safety in Nuclear Power Plants (IAEA Safety Series No. 50-SG-O8).

For reactivity control systems, due attention shall also be given to IAEA Safety Series Nos 50-SG-D3 and 50-SG-D8.

For fuel assemblies of some reactor designs (e.g. LWRs), a unique system should be designed so as to follow identification of each assembly as well as of its orientation within the core. There shall also be provisions for inspecting each fuel assembly in order to detect any possible transportation damage before insertion into the core.

5. QUALITY ASSURANCE IN DESIGN, MANUFACTURE AND OPERATION

The establishment and implementation of satisfactory quality assurance practices for the design, manufacture, installation and operation of the reactor core is essential for the safe operation of the nuclear power plant. More comprehensive information can be found in the Code of Practice on Quality Assurance for Safety in Nuclear Power Plants (IAEA Safety Series No. 50-C-QA) and its associated Safety Guides.

Annex I

REACTIVITY COEFFICIENTS

One important feature of the behaviour of a reactor in any transient condition is the rate at which the transient progresses. This rate depends on a number of nuclear characteristics which are considered in this Annex. It is the combined effect of all of the reactivity coefficients and the rate at which the variables causing the transient are changing that determines the severity of the transient. The factors of importance are:

- fuel temperature coefficient of reactivity
- coolant temperature coefficient of reactivity
- moderator temperature coefficient of reactivity
- coolant density coefficient of reactivity
- delayed neutron fraction
- prompt neutron lifetime
- effects of power redistribution.

The power coefficient of reactivity is a combination of the first four items in this list.

The transient behaviour of a reactor depends on the reactor type and design. For example in standard designs, a reduction in coolant density in a PHWR, a coolant void collapse in a BWR, or a cooldown of the coolant in a PWR all result in reactivity transients.

The signs of the various coefficients of reactivity vary from one reactor type to another. Consequently, safety related considerations are very much dependent on the reactor type. Table A.I. shows whether reactivity will increase (+) or decrease (–) during power operation when the relevant parameter increases.

TABLE A.I. REACTIVITY COEFFICIENTS

Parameter	Reactivity coefficient				
	Pressurized water reactor	Boiling water reactor	Advanced gas cooled reactor	Heavy water reactor: Pressure tube	Heavy water reactor: Pressure vessel
Coolant temperature	–	–	~0	+	+
Coolant density	+	+	~0	–	–
Moderator temperature	–	–	+	~0	–
Fuel temperature	–	–	–	–	–
Power	–	–	–	~0	~0

Annex II

PELLET–CLADDING INTERACTION

A-II.1. Zirconium-alloy cladding

For zirconium-alloy clad fuel the 'pellet–cladding interaction', i.e. stress corrosion cracking caused when the pellet expands and stresses the cladding in the presence of corroding agent, should be considered.

Stress corrosion cracking in zirconium-alloy clad fuel requires all of the following:

– high tensile stress, uniform or local, perhaps caused by a crack opening in the pellet as it expands
– susceptibility to stress corrosion cracking
– a certain concentration of corrodents possibly iodine, cadium, caesium, or other fission products
– a relatively long exposure.

When fuel has received sufficient irradiation to create fission products that act as corrodents, and the cladding has also developed increased corrosion cracking susceptibility as a result of a fast neutron dose or other factors, fuel failures can occur. Under certain operating conditions failures are possible if the fuel power is increased at a fast rate to a high power level, because pellet expansion can create a high tensile stress in the cladding.

To eliminate stress corrosion cracking failures several approaches can be considered. For exemple:

– Local stresses can be lowered by a pellet–cladding interface lubricant
– Tensile stresses can be lowered by other means, such as a slow change of power or pre-pressurization of the fuel element
– A fission product barrier can be placed at the inner surface of the cladding
– The fission products can be absorbed by an additive
– The rate of rise of power can be controlled to a lower limit
– Local power peaking can be reduced by proper overall core design
– The linear heat rating of the fuel can be reduced.

Thus the designer has several possible ways of avoiding stress corrosion cracking. A considerable database on operating experience, prototype testing and out-of-reactor testing is already in existence. The phenomenon of stress corrosion cracking, however, is only partly understood; therefore, at present, the fuel element design requires extensive judgement and the use of available data or testing on prototype fuel to confirm that the fuel performance is adequate to prevent failure by this mechanism.

The possibility of stress corrosion cracking induced by the pellet–cladding interaction should be minimized in the design process by using the suggested methods or appropriate operating procedures.

A-II.2. Steel cladding

For AGR cladding the pellet–cladding interaction mechanism which requires consideration is exhaustion of cladding ductility as a result of plastic strain. This becomes important because of the reduction in ductility caused by the action of thermal neutrons on the boron and nickel in the steel cladding and it results in helium gas bubbles at grain boundaries.

Plastic strain may occur as thermal creep strain when the reactor is at power or as yield strain when the reactor is being shut down, since the thermal expansion coefficient of stainless steel is larger than that of the fuel pellets, and the cladding operating temperature is very power dependent. Repeated power cycles may produce a 'ratcheting' effect, causing a gradually increasing cladding plastic strain; alternatively, a single large increase in fuel element rating could conceivably cause sufficient strain for failure.

Cladding damage resulting from the pellet–cladding interaction due to radial strains can be limited by the use of hollow fuel pellets and careful control of cladding temperature changes to minimize differential expansions and contractions. Differential axial strains at shutdowns may be controlled by the provision of circumferential 'antistacking' grooves on some of the pellets. Before operation, controlled collapse is applied; this anchors the cladding to the fuel pellets at predeterminated axial locations and prevents the formation of gaps between pellets which would be large enough for unacceptable degrees of collapse to take place. Since permanent changes in the overall length of the fuel elements can also be caused, due consideration shall be given to preventing unacceptable interactions between the antistacking grooves and support grids for the fuel elements.

Annex III

DESIGN CONSIDERATIONS FOR CORE MANAGEMENT

A-III.1. Power shaping

The fuel cycle adopted for a particular reactor type employs numerous options, within the relevant design constraints, to make the most efficient use of fuel. Certain factors are important for economic reactor design and for establishing the ratio of peak-to-mean rating. They also affect fuel rod rating and temperature histories throughout the life of the fuel. Among these factors are:

(1) *Radial power shaping.* The flattening of the radial power distribution may be achieved by a combination of the following:
- radial distribution of reactivity control devices
- relative movement of the reactivity control devices
- radial variation in fuel enrichment or burnup
- radial shuffling of fuel assemblies during their life in the core
- radial distribution of assemblies containing burnable poison elements.

(2) *Assembly-to-assembly variation.* This variation is largely a function of assembly irradiation. The variation may be reduced by the use of:
- burnable poisons within the fuel assemblies
- enrichment variation on a checkerboard pattern according to reactivity control device positions
- choice of the refuelling sequence, particularly for on-load refuelling reactors.

(3) *Axial power shaping.* Although reactivity control means are used mainly to control core reactivity and radial power distribution, they are used in some cases to limit the peak axial rating or cladding temperature. Axial variation in fuel enrichment or burnable poison content may be used for axial shaping.

(4) *Variations within an assembly.* A variation in enrichment or burnable poison from fuel element to element may be used to optimize the rating variations throughout the life of the assembly.

In enriched uranium reactors the discharge irradiation is chosen to have the largest value allowable on the basis of an engineering evaluation of fuel element integrity. The evaluation is based on operating experience, fuel element behaviour in test loops and consideration of the likely fuel life history resulting from the intended fuel cycle.

In natural uranium reactors with on-load refuelling the discharge irradiation is dependent on the core size and design as well as on the reactivity reserve for refuelling and load change flexibility; it may vary radially according to the fuel management scheme adopted.

The consequences of fuel misloading should be assessed for the fuel cycle adopted and any necessary remedial action should be identified.

A-III.2. Core reactivity level and shutdown

Reactivity in excess of that required for criticality at nominal full power is needed in order to provide reactivity for reactor control, power shaping and re-establishing full power from lower power levels or shutdown. This re-establishment requires the possibility of overriding xenon absorption. For enriched reactors the fuel is of sufficiently high enrichment to enable full power operation at all times for the design fuel cycle. In certain cases, such as when the cycle is extended before the next refuelling is carried out, full power may not be attainable. For natural uranium reactors it is not economic to provide sufficient excess reactivity within the core for xenon override under all conditions. Therefore, booster or adjuster rods are provided for xenon override for a limited time after the reactor shutdown.

The initial core of natural uranium reactors with on-load fuelling will have excess reactivity until the average irradiation of the fuel in the core approaches one half of the average discharge irradiation for equilibrium refuelling conditions. This excess reactivity may be compensated by depleted uranium fuel, by boron or gadolinium dissolved in the moderator or by a combination of these means.

For batch refuelled reactor designs with enriched fuel the most reactive state between refuellings must be identified, taking into account variations in fuel enrichment levels throughout the core during a fuel cycle.

The shutdown systems for any reactor type must be designed to shut down the reactor and keep it held down with an appropriate reactivity margin during the reactor's most reactive state.

Annex IV

EXAMPLES OF POSTULATED INITIATING EVENTS WHICH CAN INFLUENCE THE CORE DESIGN

Examples of PIEs which may influence the core design are:

Abnormal electrical conditions

- loss of off-site power
- loss of electric power

Component malfunction

- inadvertent cooldown of the reactor coolant system
- inadvertent withdrawal of a reactivity control or shutdown device
- inadvertent reduction of the boron concentration in the moderator/coolant system
- spurious reactor trips
- main coolant pump trip including pump seizure
- ejection of a reactivity control or shutdown device
- feedwater pipe break
- steam pipe break
- reactor coolant system pipe break
- inadvertent withdrawal of a reactivity control or shutdown device
- steam line isolation
- malfunction of feedwater system
- decrease in reactor coolant pressure
- increase in reactor coolant pressure
- abnormal conditions during refuelling

External events

- earthquake
- explosion
- aircraft crash.

For a more complete discussion of PIEs refer to Section 4.4 and Appendix A of IAEA Safety Series No. 50-SG-D11.

DEFINITIONS

The following definitions are intended for use in the NUSS Programme and may not necessarily conform to definitions adopted elsewhere for international use. Items marked with an asterisk have been taken from the list of definitions included in the approved Codes of Practice published under the NUSS Programme.

***Acceptable Limits**

Limits acceptable to the regulatory body.

***Accident Conditions**

Substantial deviations from Operational States which are expected to be infrequent and which could lead to release of unacceptable quantities of radioactive materials if the relevant engineered safety features did not function as per design intent.[1]

***Anticipated Operational Occurrences**

All operational processes deviating from Normal Operation which are expected to occur once or several times during the operating life of the plant and which, in view of appropriate design provisions, do not cause any significant damage to items important to safety nor lead to Accident Conditions.[2]

Ballooning

The result of internal overpressure in a fuel element which stresses the Cladding beyond its elastic limit and causes it to expand excessively.

Burnable Poison

Neutron absorbing material with particular capability of being depleted by neutron absorbtion and used to control reactivity.

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[1] A substantial deviation may be a major fuel failure, a loss of coolant accident (LOCA), etc. Examples of engineered safety features are: an emergency core cooling system (ECCS) and containment.

[2] Examples of Anticipated Operational Occurrences are loss of normal electric power and faults such as a turbine trip, malfunction of individual items of a normally running plant, failure to function of individual items of control equipment, loss of power to main coolant pump.

Cladding (material)

An external layer of material applied directly to nuclear fuel or other material that provides protection from a chemically reactive environment and containment of radioactive products produced during the irradiation of the composite. It may also provide structural support.[3]

Fuel Assembly

A grouping of Fuel Elements which is not taken apart during the charging and discharging of a reactor core.

Fuel Element

The smallest structurally discrete part of a reactor which has fuel as its principal constituent.

***Normal Operation**

Operation of a nuclear power plant within specified Operational Limits and Conditions including shutdown, power operation, shutting down, starting up, maintenance, testing and refuelling (see Operational States).

***Operation[4]**

All activities performed to achieve, in a safe manner, the purpose for which the plant was constructed, including maintenance, refuelling, in-service inspection and other associated activities.

***Operational Limits and Conditions**

A set of rules which set forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of the nuclear power plant.

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[3] In the context of this Guide the cladding consists of a tube which surrounds the fuel and which, together with the end cups or plugs, also provides structural support.

[4] The terms Siting, Construction, Commissioning, Operation and Decommissioning are used to delineate the five major stages of the licensing process. Several of the stages may coexist, for example, Construction and Commissioning, or Commissioning and Operation.

***Operational States**

The states defined under Normal Operation and Anticipated Operational Occurrences.

***Prescribed Limits**[5]

Limits established or accepted by the regulatory body.

***Protection System** (revised March 1980)

A system which encompasses all those electrical and mechanical devices and circuitry, from and including the sensors up to the input terminals of the safety actuation systems and the safety system support features, involved in generating the signals associated with the protective tasks.

***Quality Assurance**

Planned and systematic actions necessary to provide adequate confidence that an item or facility will perform satisfactorily in service.

Reactivity

A parameter, ρ, giving the deviation from criticality of a nuclear chain-reacting medium such that positive values correspond to a supercritical state and negative values to a subcritical state.

Quantitatively,

$$\rho = 1 - \frac{1}{k_{eff}}$$

where k_{eff} is the effective multiplication factor. The reactivity is expressed in terms of many different units, such as dollar, cent, inhour, nile and pcm.

Safety Systems

Systems important to safety provided to ensure, in any condition, the safe shutdown of the reactor and the heat removal from the core, and/or to limit the consequences of Anticipated Operational Occurrences and Accident Conditions.[6]

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[5] The term 'authorized limits' is sometimes used for this term in other IAEA documents.

[6] Safety Systems consist of the Protection System, the safety actuation systems and the safety system support features. Components of Safety Systems may be provided solely to perform safety functions or may perform safety functions in some plant Operational States and non-safety functions in other plant Operational States.

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LIST OF NUSS PROGRAMME TITLES

For the Safety Guides no plans exist to fill the gaps in the sequence of numbers

Safety Series No.	Title	Publication date of English version
	1. Governmental organization	
Code of Practice		
50-C-G	Governmental organization for the regulation of nuclear power plants	Published 1978
Safety Guides		
50-SG-G1	Qualifications and training of staff of the regulatory body for nuclear power plants	Published 1979
50-SG-G2	Information to be submitted in support of licensing applications for nuclear power plants	Published 1979
50-SG-G3	Conduct of regulatory review and assessment during the licensing process for nuclear power plants	Published 1980
50-SG-G4	Inspection and enforcement by the regulatory body for nuclear power plants	Published 1980
50-SG-G6	Preparedness of public authorities for emergencies at nuclear power plants	Published 1982
50-SG-G8	Licences for nuclear power plants: content, format and legal considerations	Published 1982
50-SG-G9	Regulations and guides for nuclear power plants	Published 1984

Safety Series No.	Title	Publication date of English version
	2. **Siting**	
Code of Practice		
50-C-S	Safety in nuclear power plant siting	Published 1978
Safety Guides		
50-SG-S1	Earthquakes and associated topics in relation to nuclear power plant siting	Published 1979
50-SG-S2	Seismic analysis and testing of nuclear power plants	Published 1979
50-SG-S3	Atmospheric dispersion in nuclear power plant siting	Published 1980
50-SG-S4	Site selection and evaluation for nuclear power plants with respect to population distribution	Published 1980
50-SG-S5	External man-induced events in relation to nuclear power plant siting	Published 1981
50-SG-S6	Hydrological dispersion of radioactive material in relation to nuclear power plant siting	Published 1985
50-SG-S7	Nuclear power plant siting: hydrogeologic aspects	Published 1984
50-SG-S8	Safety aspects of the foundations of nuclear power plants	Published 1986
50-SG-S9	Site survey for nuclear power plants	Published 1984
50-SG-S10A	Design basis flood for nuclear power plants on river sites	Published 1983
50-SG-S10B	Design basis flood for nuclear power plants on coastal sites	Published 1983

Safety Series No.	Title	Publication date of English version
50-SG-S11A	Extreme meteorological events in nuclear power plant siting, excluding tropical cyclones	Published 1981
50-SG-S11B	Design basis tropical cyclone for nuclear power plants	Published 1984

3. Design

Code of Practice

50-C-D	Design for safety of nuclear power plants	Published 1978

Safety Guides

50-SG-D1	Safety functions and component classification for BWR, PWR and PTR	Published 1979
50-SG-D2	Fire protection in nuclear power plants	Published 1979
50-SG-D3	Protection system and related features in nuclear power plants	Published 1980
50-SG-D4	Protection against internally generated missiles and their secondary effects in nuclear power plants	Published 1980
50-SG-D5	External man-induced events in relation to nuclear power plant design	Published 1982
50-SG-D6	Ultimate heat sink and directly associated heat transport systems for nuclear power plants	Published 1981
50-SG-D7	Emergency power systems at nuclear power plants	Published 1982

Safety Series No.	Title	Publication date of English version
50-SG-D8	Safety-related instrumentation and control systems for nuclear power plants	Published 1984
50-SG-D9	Design aspects of radiation protection for nuclear power plants	Published 1985
50-SG-D10	Fuel handling and storage systems in nuclear power plants	Published 1984
50-SG-D11	General design safety principles for nuclear power plants	Published 1986
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