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DESIGN FOR REACTOR CORE SAFETY IN NUCLEAR POWER PLANTS

A Safety Guide

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1. INTRODUCTION

1.1 PURPOSE

The purpose of this Safety Guide is to identify certain reactor safety requirements and to provide guidance on reactor core design for the implementation of these requirements. 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 in implementing the Code. It should be noted that the reactor core safety is achieved by a combination of proper design, manufacturing and operation. For the operational aspects reference should be made to Safety Series 50-C-O "Safety in Nuclear Power Plant Operation, including Commissioning and Decommissioning, A Code of Practice" and the associated Guides of this series No. 50.

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: Advanced Gas-Cooled Reactor (AGR), Boiling Water Reactor (BWR), Pressure Tube Heavy Water Reactor (PHWR) (pressure tube and pressure vessel type), Pressure Water Reactor (PWR) fuelled with oxide fuels. It deals with the individual components and systems that make up the core, with information and design provisions for the operation of the core, and handling of the fuel and other core components.

This Guide discusses the reactor vessel internals and the reactivity control and shutdown devices* mounted to the vessel. Possible feedback on requirements for the reactor coolant, the primary cooling

* In this guide the term "device" (shutdown d. or reactivity control d.) is used when mainly the 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 m. or reactivity control m.) is used to address the functional aspect in a more general way.

systems and its pressure boundary including the pressure vessel is addressed only as far as necessary to clarify the interface with Safety Series No. 50-SG-D13 "Reactor Cooling Systems in Nuclear Power Plants" and other guides. For instrumentation and control systems the guidance is mainly limited to functional requirements.

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. It comprises also the moderator, coolant and absorber 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.

- Supporting structures including those that physically support the core within the vessel, the flow guide structure such as a core barrel or the pressure tubes of a PHWR (pressure rube type), guide tubes for reactivity control devices, etc.

- Other internals such as instrumentation tubes, in-core instrumentation for core monitoring, steam separators, and neutron sources which are dealt wi.h to a limited extent in this Guide.

2. SAFETY DESIGN PRINCIPLES

2.1 GENERAL

Section 4 of the Code, entitled "Reactor Core", provides general safety principles for core design which are the basis for the more detailed design requirements 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 bodies. 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 Safety Series 50-SG-D13 "Reactor Cooling Systems in Nuclear Power Plants")
- the reactor containment system (See Safety Series No. 50-SG-D12 "Design of the Reactor Containment System in Nuclear Power Plants")

A more complete discussion of the defence-in-depth concept is given in Safety Series 50-SG-D11 "General Design Safety Principles in Nuclear Power Plants".

Core design can help to achieve these goals by ensuring that radioactive materials are confined within the fuel matrix and the fuel cladding itself to the maximum extent practical. This design process requires an iterative consideration of neutronic, thermohydraulic, mechanical and chemical aspects. The size and number of fuel assemblies, the fuel enrichment, the coolant flow rate, etc. are decided upon by this iterative design process to meet the required heat production, reactivity control and the fuel management scheme. From a safety point of view the design shall be such that the reactor power can be safely controlled and the core can be adequately cooled to maintain the fuel performance parameters within acceptable design limits.

A list of postulated initiating events (PIE) with significant potential consequences shall be established* as a basis for safety design and accident analysis. The consequences shall be analysed with respect to the reactivity variation of the core, core coolability and fuel element and assembly integrity.

Acceptable design limits for the core are closely related to limitation of the release of radioactive material from the fuel elements. For operational states the goal of the design limits shall be to maintain fuel element integrity. For postulated accident conditions the goal of the design limits shall be to ensure that the severity of fuel element damage remains within acceptable values. A basic safety design intent shall be to achieve, as far as practicable, behaviour characteristics of the core which are favourable to safety. Reactor core components and connected structures shall be designed taking into account safety actions during accident conditions, e.g. shutdown and emergency core cooling.

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

2.2 BASIC CONSIDERATIONS FOR NEUTRONIC AND THERMOHYDRAULIC DESIGN OF THE CORE

(1) The combination of inherent reactor neutronic and thermohydraulic characteristics and the control system capability shall be sufficient to adequately regulate the reactor power in a stable manner for all operational states of the plant. Further discussion relating to reactivity coefficients is given in Annex 1.

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

(3) The reactor shall be capable of being shut down and held sub-critical under operational states and accident conditions.

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* See Annex 4 for a typical example.

(4) Assessments of the core power distribution, especially peak channel power and peak linear heat rating, shall be carried out for representative operational states to provide bases for operational limits and conditions and operating procedures for compliance with fuel design limits throughout reactor core life.

(5) Adequate provision for cooling the core under operational states and accident conditions shall be made and its effectiveness proven by analysis and experiments to meet specified fuel integrity criteria.

(6) The design of the core should sufficiently minimise the demand made on the control system for maintaining flux shapes and levels within stipulated limits in all operational states.

(7) Thermohydraulic design limits on such parameters as Minimum Critical Power Ratio, Minimum Departure from Nucleate Boiling Ratio (see subsection 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 a minimum.

(8) The heat transfer correlations and other data used in the thermohydraulic design shall be based on adequate and reliable data applicable to the conditions expected in actual operation.

(9) Design provisions shall minimize the chance of any obstruction of the coolant flow during operational states which could lead to core damage.

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

2.3 BASIC CONSIDERATIONS FOR MECHANICAL DESIGN OF CORE COMPONENTS

(1) The fuel shall be designed so as to achieve leak-tight operation of fuel elements for all 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 all operational states and accident conditions.

(3) All core and associated components shall be designed to be compatible with each other under the environment of irradiation, chemical and physical process, static and dynamic mechanical loads 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 IAEA Safety Series No. 50-SG-D10, "Fuel Handling and Storage Systems in Nuclear Power Plants")

(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 devices should be prevented.

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

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. In this section the components are considered individually, beginning with the fuel progressing to the core support structure and finishing with the transient and accident analysis. A section 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 chosen 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 requirements and recommendations of this guide as well as other applicable guides such as:

SG-D8 on safety-related I&C systems
SG-D10 on fuel handling systems
SG-D13 on reactor cooling systems

3.1 FUEL ELEMENTS AND ASSEMBLY

3.1.1 Fuel Element Design Requirements

The Code of Practice 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."

The following subsections give the design requirements and considerations for fuel elements that are needed to meet this objective. These subsections 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 fuel temperature during operational states to a level that is below its melting point. However, there may be more restrictive operating limits imposed due to the consideration of postulated accidents such as loss-of-coolant accidents.

The strength and corrosion behaviour of the cladding is very temperature-dependent. Limits for stress, long-term deformation and corrosion may therefore be specified for operational states. For accident conditions, cladding temperature and the oxidation of zircalloy cladding shall be limited; firstly, to control ballooning and maintain a coolable geometry, and secondly, to limit a zirconium-steam reaction for zirconium alloy cladding. These effects shall not prevent shutdown or the maintenance of a shutdown condition.

The design shall include margins to allow for fabrication, calculation and other uncertainties.

3.1.1.2 Fission Product Effects

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 paragraph 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 consequences of reactor depressurization shall be considered, to ensure that the gas pressure within the fuel element does not cause unacceptable failure of the cladding. The likelihood of 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 U 235 and the production of plutonium result in changes in power distribution within the core and the fuel assemblies and in changes of 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. pellet-cladding interaction (see paragraph 3.1.1.6).

The effect of anticipated power transients on local heat rates shall be studied. To ensure good fuel integrity, stresses and working cycles of cladding materials should be accommodated by the design taking into account 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 can not form in the cladding.

Free-standing claddings can undergo a long-term deformation (creep deformation) under external pressure resulting in decrease of 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 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, fuel thermal expansion due to increase of local power or internal gas pressure and should be limited.

The allowable length of axial gaps between fuel pellets, caused by 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 2. Mechanical loads imposed on the fuel element by the fuel assembly are discussed in sub-section 3.1.2.

3.1.1.6 Pellet-cladding interaction

Pellet-cladding interaction is a particularly important consideration for fuel cladded by zirconium alloys, because it has been the cause of fuel defects. The stress-corrosion cracking induced by pellet-cladding interaction in the presence of fission products should be minimized. A discussion of pellet-cladding interaction control for zirconium alloy and steel-cladded fuel is included in Annex 2.

3.1.1.7 Effects of Burnable Poisons 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, should 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 parameters include pressure, temperature and chemical composition.

3.1.2 Mechanical Safety Design Requirements of 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 if anticipated operational occurrences arise. 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 and erosion. 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. These ratios can 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 which is currently used 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 operating conditions expected in actual operation. The test results are then analysed and converted into correlations for use in safety analyses.

3.1.2.2 **Mechanical Effects**

The fuel assembly is subjected to mechanical stresses as a result of phenomena including the following:

- fuelling and refuelling procedures
- power variation
- hold-down loads for PWR
- hydraulic forces
- bowing due to irradiation
- vibration and fretting induced by coolant flow
- vibration induced by external events such as earthquakes
- accident events such as LOCA.

For operational states the design requirements of 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 hold-down 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.

For postulated initiating events including earthquakes, explosions, equipment failures, the design of the fuel elements, fuel assemblies and fuel assembly support structures shall ensure that interactive or consequential effects from these components will not

- prevent functioning of safety system components, e.g. shut-down 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 credited in the transient and accident analysis (ref. 3.9).

Changes caused by the environment, which could increase the resistance to heat flow from the fuel element, shall be considered in evaluating thermal characteristics. This includes 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 in which the fuel will operate over its design lifetime under normal operaing conditions should be considered when defining suitably conservative design parameters for surface oxidation and crud buildup. The design parameters used should be based on actual experience and experimental data appropriate to operating conditions.

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 are 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 start up and maintaining it in that condition

(2) Keeping the coolant radioactivity at an acceptably low level by the use of purification systems and removal of defective fuel as appropriate (see also SG-D13, section 4.5)

(3) Effects on reactivity of the coolant and coolant additives* shall be taken into account in determining the capabilities of the reactor control and shutdown systems for operational states and accident conditions

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* It is general practice in some reactor types to ensure that coolant additives do not cause the power coefficient of reactivity to be positive. (For further discussion of the reactivity coefficients see Annex 1).

(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 minimise corrosion

(5) Ensuring a sufficient supply of coolant for all operational states and accident conditions in order to meet specified fuel integrity criteria including decay heat removal at shutdown (see also SG-D13, section 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 consequential fluctuations in reactivity

(8) Ensuring that the core internals shall be 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 subcooled state during normal operation. In a Boiling Water Reactor (BWR), water enters the core subcooled, but leaves the core 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 overall. In order that the power coefficient remains negative, burnable poisons may be mixed in the fuel to reduce the need for boron in the coolant.

Additives can be used to control the chemistry of the coolant and inhibit corrosion. 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 SG-D9, section 4.4.2.1.

3.2.2 **Heavy Water**

The relevant characteristics of heavy water are mostly similar to those of light water, and the factors considered in sub-section 3.2.1 apply, if the fact that coolant and moderator in some reactor designs are separated, is taken into account. The use of additives in the coolant, whether for chemical (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 all operational states and accident conditions.

Radioactivation shall also be considered. In addition to the radioactive corrosion products and ^{16}N that are found in H_2O, tritium (^3H) builds up in D_2O to a larger extent than in light water. Therefore, in a D_2O reactor coolant system provisions shall be made to prevent or control the release of tritiated heavy water from the system.

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 purpose of the moderating material in the core is to reduce the energy of fast neutrons produced in the fission processes to the thermal energy range, so that they will cause further fissions, thus continuing the heat-producing process. The choice of moderator and spacing of fuel assemblies within it are based on optimizing the neutron economy, and hence fuel consumption, in conjunction with the engineering requirements. The main reactor types use either light water, heavy water or graphite as the moderating medium, e.g.:

PWR	Pressurized Water Reactor)	light water
BWR	Boiling Water Reactor)	moderated
PHWR	Pressure tube heavy water reactor)	heavy water
PHWR	Pressure vessel heavy water reactor)	moderated
GCR	Advanced gas cooled reactor	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, without physically separating the two functions. The considerations regarding additives, reactivity characteristics, and radiation effects discussed in section 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 D_2O, 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 control or for reactivity holddown after a shutdown. The moderator also serves to cool various reactor structures, e.g. calandria vessel itself, reactivity control devices and their guide tubes, instrumentation support structures. Although highly unlikely, it is conceivable that the pressure tube and calandria tube might rupture, allowing a jet of D_2O coolant to be injected into the moderator region; that is, some moderator will be displaced by coolant. If this occurs and the moderator contains absorber (and, of course, the coolant does not), 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 D_2O, the moderator is separated from the coolant in the core region by the cooling channels. However, the circuits of the coolant and the moderator are separated or connected, depending on plant status (e.g. power operation or residual heat removal operation). In the power operation mode the moderator is connected to the coolant only by some pressure compensating holes such that it can be operated at the same pressure but at a lower temperature level than the coolant. The coolant temperature level is maintained by 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 activity of tritium can build up in D_2O 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 more detail in SG-D13.

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

3.3.3 Graphite

The moderator adopted for advanced gas cooled reactors is graphite. In advanced gas cooled reactors (AGRs) 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:

(1) Shutdown

Entry of the shutdown devices into the core and maintaining the shutdown condition shall not be impeded. In order to confirm this an assessment shall be made of the capacity 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 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) Circuit radioactivity

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

In addition 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.

The design of the distribution holes drilled through the graphite bricks and 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 subsection discusses the means for reactivity control for normal operation, referred to in section 4.3 of the Code. The control of reactivity for reactor shutdown is addressed 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 Safety Series No. 50-SG-D8, "Safety-Related Instrumentation and Control Systems.

3.4.1 Types

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

- moderator temperature (PWR, 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, GCR, PHWR)
- fuel with distributed or discrete burnable poison
- fuel assembly repositioning

3.4.2 Maximum Reactivity Worth and Reactivity Insertion Rate

The arrangement, grouping, speed of withdrawal and withdrawal sequence of the reactivity control devices, combined with adoption of interlock systems shall be designed to ensure that any credible abnormal withdrawal of the reactivity control devices does not cause the conditions specified for the fuel to be exceeded. The maximum reactivity worth of the reactivity control devices shall be limited, or interlock systems shall be provided so that for a postulated accident condition such as "control rod ejection" for PWR or "control rod drop" for BWR 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 circuit and

(b) risk of a molten fuel-coolant interaction which could damage the core structure and prevent successful insertion of the shut-down 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 conditions specified for the fuel to be exceeded. All portions of systems that contain boric acid shall be designed to prevent precipitation, e.g. by heating the components containing boric acid solution (see Safety Guide 50-SG-D13, section 4.5).

3.4.3 Control of Overall and Local Power

Core power shall be controlled overall and locally by reactivity control means in such a way that peak linear rating of the fuel and channel power will not exceed design limits anywhere in the core. 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 subsection 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 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 reactivity and power variations during fuel burnup.

3.4.5 Irradiation Effects

The reactivity variations, physical property changes, gas production, activation in liquid loops, etc. of reactivity control means due to irradiation shall be taken into account in their design.

3.5 CORE MONITORING SYSTEM

Instrumentation shall be provided to monitor core parameters such as level, distribution and temporal variation of core power, physical states of the coolant and moderator, 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 the latter offer the possibility to remove defected fuel more easily and thereby keep radioactivity levels in the coolant low. Another advantage of a failed-fuel monitor is that it can give an 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 to perform the functions for which the various monitoring systems are intended (see the Safety Guide on Protection Systems and Related Features in Nuclear Power Plants, IAEA Safety Series No. 50-SG-D3, and SG-D8). The design shall also incorporate facilities that allow for continuous or periodic testing of monitoring systems, as required.

For guidance on accident monitoring reference should be made to SG-D8, subsection 4.9.3.

In the case of large cores it may be necessary to monitor the spatial power distribution by incore neutron detectors. Measurements of local power at different positions in the core have the purpose to ensure adequate safety margins and to provide data for optimum utilisation of the fuel. In this case the position of the detectors shall be distributed 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 location for safety purposes.

Other safety-related parameters are derived from the monitored parameters, for example:

- 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 safety significance of the use of the monitors

- in circuits of the reactor protection system
- in circuits of the reactor control system
- for display to the operators

shall be assessed. The necessary redundancy, diversity and independence of the signals and their transmission shall be ensured by the design according to the requirements of SG-D3 and 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.

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. In some reactor types a neutron source may also 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 neutron sources function properly for their planned lifetime, and
- the sources are compatible with the fuel assemblies and the fuel assembly support structures.

3.6 REACTOR SHUTDOWN MEANS

This sub-section deals with the means (refer to section 4.4 of the Code) of rendering the reactor subcritical in operational states or under accident conditions, and maintaining it in that state

Means shall be provided which 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 conditions of the fuel and the reactor coolant system pressure boundary 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 longterm 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 shutdown means shall therefore incorporate the necessary reliability in the design of the equipment to effect shutdown for all postulated initiating events as necessary to meet plant safety requirements. The designs shall incorporate the necessary independence from plant process and control systems and protection from the consequential effects of the initiating events such that the shutdown will be performed as required.

The means of shutdown shall be designed using a fail-safe philosophy as far as practical, and shall be engineered to the high reliability required for those safety systems. If operation of the holddown system is manual or partly manual, the necessary prerequisites for manual operation shall be met (see SG-D3, section 7.3.2).

For the purposes of reactivity control and flux shaping during normal operation (see section 3.4) a portion of the shutdown means may be used. In this case, it shall be ensured that none of the functions of the control system will jeopardize the function of the shutdown system. For a more detailed discussion,see subsection 7.8.4 of SG-D3.

3.6.1 Types

Various means of introducing absorbing materials into the reactor core are adopted for different reactor types, including

- B injection into moderator
- Gd injection into moderator
- N_2 injection
- moderator dump
- B, Cd in stainless steel rods, tubes, or cruciforms
- Hf and steel rods in Zry -4 guide tubes
- B glass bead injection
- liquid absorber in tubes.

Table I shows examples of shutdown means used in different reactor types illustrating the incorporation of diversity:

TABLE I Shutdown Means

Reactor Type	Primary System	Secondary System
BWR	B_4C in steel tubes	B solution injected into moderator/coolant
PWR	Ag-In-Cd in steel tubes or B_4C in steel tubes	B injected in moderator/coolant
PHWR	Cd sandwiched in steel tubes	Gd injected in moderator, moderator dump, liquid absorber in tubes
PHWR (pressure-vessel type)	Hf and steel rods in Zry-4 guide tubes	B injected in moderator
AGR	B steel rods + stainless steel rods	N_2 injection into coolant and B glass beads injection into the core

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 and that use components of established high reliability

2) Using a fail-safe philosophy as far as practicable*

3) Giving consideration to modes of failure and adopting redundancy and diversity in the initiating mechanisms (e.g. sensors, actuation devices that detect and respons to the need for a reactor trip)

4) Functionally isolating and physically separating the shutdown systems including separation of control and shutdown functions, as far as practicable, to cater for credible modes of failure, including common cause**.

5) Ensuring easy entry of shutdown means into the core taking into account the incore 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.

8) Performing comprehensive testing during manufacture, installation, and commissioning.

3.6.3 Shutdown and Holddown Effectiveness

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 shall be demonstrated:

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* The simplest form of fail-safe philosophy used in many designs is that the shutdown devices are held above the core by active means. Providing that the shutdown device channels are not obstructed, the devices drop into the core by gravity in the event of failure (de-energization) of the active holding means, e.g. loss of current through a holding electromagnet.

**Some Member States require that there be two independent, different, shutdown systems each of which is adequate acting alone.

(1) during the design, to cover the whole range of operating conditions and core configurations that occur throughout the intended fuel cycle;

(2) at reactor commissioning, by appropriate neutronic and process measurements to confirm the calculations for start of life;

(3) during reactor operation, by measurements and calculations to cover present and anticipated reactor conditions;

(4) during fuel changes (see SG-010 for core management), to ensure that reactivity shutdown margins are maintained using;

(5) during anticipated operational occurrences and accident conditions, to meet acceptable fuel cooling and radioactivity release criteria

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*.

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

(1) 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

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* 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.

-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 burn-up.

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

(3) The required subcriticality margin.

(4) The greatest negative reactivity worth which is unavailable on the assumption of a single missing shutdown device.

(5) The distribution of the shutdown devices within the core. This may influence the reactivity worth of the shutdown device not to be considered due to the single failure assumption and of recently installed (fresh) fuel assemblies.

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 postulated 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 be established in defining the shutdown rate requirement. The selection of variables for the sensing of these PIEs shall meet the requirements of SG-D3, section 7.7 and 7.13.

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 governs the choice of instrumentation to cover adequately 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 factors.

(3) Location of the shutdown devices. The rate is sensitive to:
- the distance of the poised shutdown devices from the core (see section 3.6.5); and
- the location of soluble absorber injection nozzles 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 sub-section 3.7) to facilitate device access, and the possible incorporation of rod 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 devices into the core;
- gravity drop of devices into the core initiated by a spring
- hydraulic or pneumatic pressure drive;
- hydraulic or pneumatic pressure injection of soluble absorber

Ease and speed of insertion can be checked by monitoring with appropriate sensors on the reactivity control 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 sub-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 when inside the reactor shall be considered. Such considerations 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 their performance is not thereby impaired.

(2) Temperature effects - The effects of heating of shutdown devices due to 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 physical integrity of shutdown devices, and 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 sub-section includes the structures which form and support the reactor core assembly and which are closely associated to 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 during and following accident conditions to the extent that their 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; seismic motions. Of particular concern are: damage to shutdown and holddown systems, insufficient fuel coolability, damage to fuel, and damage to the 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 all operational states and accident conditions. Chemical effects of coolant or moderator on the structures shall be considered. Some Member States require in the regulations that the structural failure of any single core component shall not prevent the reactor from being safely shut down nor result in the fuel exceeding its design limits.

Provisions for the necessary inspections or 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 associated with the core structure design. The core portion of the reactor coolant pressure boundary can be either:

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

AGRs, light water reactors and PHWR's of the pressure vessel type are type a). The pressure vessel is a large thick-walled structure surrounding the core with penetrations to accomodate the reactor coolant, instrumentation, and 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.

In PHWRs of the pressure tube type (type b)), the pressure boundary comprises a number of thin walled cylindrical tubes with no wall penetrations, since the instrumentation, 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 SG-D13. Pressure tubes shall also meet the fuel assembly support structure requirements of sub-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 their desired geometrical relationship with the reactor coolant pressure boundary. These support components shall be designed to remain intact to the degree necessary to perform their functions throughout the life of the reactor for all operational states and accident conditions. Mechanical loads such as those induced by normal and postulated abnormal refuelling, including hydraulic forces, 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 all operational states and accident conditions.

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

Design considerations for the pressure tubes and their connection to the end fittings including closure plugs, are covered in SG-D13. The following additional factors shall be considered:

(1) The 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 used for reactor control only shall be designed to perform satisfactorily for all 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, shutdown, and under postulated accident conditions shall be carefully addressed 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 accomodate 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 regulation.

3.7.5 Core Instrumentation Support Structures

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

If accurate knowledge of the location of a detector such as a neutron detector or thermocouple is necessary for its proper functioning, the structure shall be designed so that the detector can be located with the desired 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 detector replacement, as necessary.

3.7.6 Steam Separator

In BWRs the two-phase mixture of saturated water and steam is separated by a steam separator located above the fuel assemblies in the upper part of the pressure vessel. The steam separator shall be designed to remain intact for all operational states and accident conditions. To avoid damage of the fuel assemblies, other core components, and coolant pumps by debris from the steam separator, the mechanical loads caused by the steam flow shall not damage the separator under any operational state.

The steam separator shall be designed to allow periodic inspections to be undertaken to check its integrity.

3.7.7 Other Vessel Internals

Depending upon the reactor type there are other structures installed within the pressure vessel. These include feedwater spargers, steam dryers, core barrel, 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.

3.7.8 Decommissioning Considerations

When a nuclear 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.

The core structure, moderator and support structures shall be designed so as to facilitate disposal; and to ensure that the exposures to radiation 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 aspects of reactor safety and the economic utilization of the fuel. The 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 plant.

The fuel performance objective is to choose a fuel cycle with 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 fuel design limitations. These limitations are set with due reference to the safety limitations associated with operational states and accident conditions. (See sub-section 3.9). The various means available to achieve this performance objective are given in Annex 3 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 having considered the factors discussed in section 3.1.

For operational states the goal is that no cladding failures occur. To achieve this goal limits as appropriate to reactor type, e.g. linear heat rating, margin to critical heat flux conditions, cladding temperature are established. However, there are certain conditions, e.g. fuel element manufacturing defects, unexpected operational transients or manoevers, which may make it extremely difficult to meet this no failure goal during operational states. In practice some fuel cladding failures can be accepted during operational states provided that releases of radioactive material to the reactor coolant systems and from there to the environment remain within prescribed limits.

For accident conditions the permissible degree of fuel failure depends upon the likelihood of such conditions and the associated radiological consequences. In some instances these limits 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. choosing the minimum departure from nucleate boiling ratio for normal operation to avoid the cladding becoming dry during an accident transient in water cooled reactors.

3.8.2 Design Information for Reactor Operation

In order to achieve the desired core reactivity and flux distribution for reactors with enriched fuel, the core management programme shall provide the reactor operators with: the pattern of fuel assemblies to be loaded for the initial core; the subsequent schedule for unloading and loading of fuel assemblies, and in some designs 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. For reactors fuelled on-load with natural uranium the fuelling is more flexible but a programme with a similar list of contents is still required (see Annex 3).

The description of the refuelling operations and the evaluation to confirm that the safety requirements are met are set out in SG-010.

3.8.3 Reactor Core Analysis

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

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

- measurements in experimental reactors
- measurements in prototype reactors
- in-pile measurements on prototype assemblies under simulated conditions
- normal reactor operating data
- measurements during 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.

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

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

- full power, including representative power distributions
- load following
- 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, steady state power distributions shall be caluclated for each assembly location and axially along the fuel assemblies. In order to identify hot spots in the fuel cladding, factors should then be superimposed to allow for the radial power distribution within the fuel assembly, as a function of fuel irradiation and enrichment; and to allow for axial effects resulting from such items as spacers, grids and braces and other structural components. Using a series of such calculations, the power and temperature history 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 start-up and refuelling shall, where necessary, be imposed onto the rating and temperature histories described in this sub-section, in order to evaluate the effects of thermal cycling on such parameters as fission gas pressure and fuel cladding fatigue.

3.8.4 Fuel Handling Systems

In order to prevent unacceptable release of radioactivity during refuelling, the refuelling systems shall be designed to prevent the 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 unacceptable release of radioactive material from failed fuel during transit.

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

Further information on fuel handling and storage systems is given in SG-D10 and SG-O10.

For reactors using enriched fuel, it is important to ensure that the fuel assemblies be loaded into the intended positions in the core. This requires that administrative controls ensure that individual assemblies are clearly identified, and that correct loading into the core is verified.

In addition, the following measures may be taken:

- Use of reactivity monitors - detecting the enrichment level and checking against that required.
- Mechanical means - preventing the entry of high enrichment assemblies into regions where lower enrichments are required.

An ultimate verification of fuel loading pattern is realized by measuring in core flux distribution.

3.9 TRANSIENT AND ACCIDENT ANALYSIS

The analysis of events relative to nuclear plant and core behaviour including credible combinations of such events (operating mode, equipment failures, natural phenomena, man-induced events) shall be carried out in accordance with 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 (examples given in Annex 4) vary for different reactor designs, and the reactor response to them also varies widely (as can be understood from Annex 1 where the coefficients of reactivity are addressed). These shall include the necessary considerations of failure of a shutdown system* as discussed in SG-D11 Section 7.3.2.2(6). The specified limits for core design corresponding to the various faults and fault sequences shall be consistent with the likelihood of the occurrence and the radiological consequences associated with them.

* A fault sequence comprising a transient without operation of the shutoff rod system is called an anticipated transient without scram (ATWS) for some current designs of light water reactors.

3.9.2 Analysis

The basic design parameters are first established to meet the performance requirements of the nuclear power plant. These parameters are then refined, taking into consideration potential transients and accidents. The safety questions associated with transients and accidents are addressed through a combination of operating limits and safety features designed to anticipate and mitigate transients and accidents. These include emergency shutdown and cooling systems, containment designs, and siting policies, all of them consistent with some agreed level of acceptable risk for plant personnel and the general public, usually expressed as a regulation by the licensing body. A general description of nuclear safety principles can be found in SG-D11. That Guide also gives detailed requirements for the safety analysis itself.

For anticipated operational occurrences and postulated accidents, analytical techniques or methods can vary to fit the particulars of the event being analysed. Some events, such as relatively slow changes in coolant flow rate or moderator density, can be analysed with steady-state methods. 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 event will require a step-by-step analysis, using separate models for various components or parts of the reactor and defining the input for one model by the output of the other.

Studies shall be carried out to investigate the transient behaviour following such postulated initiating events and shutdown sequences, 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.

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 changeover from one mode of operation to another, e.g. from ECC injection mode 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 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 occurence 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 postulated accidents where some fuel element damage may be allowed, the analysis should allow for damage during the transient. The effect on core cooling of such conditions as ballooning and rupture of cladding, exothermic metal-water reactions and fuel element distortions should be included in the analysis.

The analysis carried out may lead to operational restrictions being identified and imposed in order to satisfy the design limits for fuel life.

The methods used should be verified against experiments to the extent practicable to demonstrate their suitability for analysing the event in question. For analysis of transient behaviour, particularly of the more severe transients, directly applicable experimental data may not be available. In these cases the comparison of results using 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 set 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 time period required, its functional and safety requirements while subject to the environmental conditions (e.g. pressure, temperature, radiation, vibration). These environmental conditions shall include the variations expected during normal operation, anticipated operational occurrences and accident conditions.

The characteristics of certain PIEs may preclude the performance of realistic commissioning and recurrent tests which could confirm that the equipment would preform their safety function when called upon to do so, e.g. during an earthquake. For such equipment, a suitable qualification programme shall be foreseen and performed on such items prior to installation.

Methods of qualification can be:

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 extrapolating such data
5) Any combination of the above methods.

4.2 Provision for Inspection and Testing

As specified in section 2.9 of the Design Code of Practice, structures, systems and components important to safety shall be designed to accomodate 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. Objectives and 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 Safety Guides on Protection System and Related Features in Nuclear Power Plants (IAEA Safety Series No. 50-SG-D3) and on Safety-Related Instrumentation and Control Systems (IAEA Safety Series NO. 50-SG-D8).

For fuel elements of some reactor designs (e.g. LWRs), a unique identification system should be designed such as to allow an identification of each element as well as of its orientation within the core. Provisions shall also be present to inspect each fuel element for detecting any possible transportation damage before its insertion into the core.

5. QUALITY ASSURANCE IN DESIGN, MANUFACTURE AND OPERATION

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 is provided by the Quality Assurance Code of Practice and Safety Guides No. 50-SG-QA1 to 11. For the topics covered by these documents see the List of NUSS Programme Titles printed at the end of this Guide.

ANNEX 1

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 addressed 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 reactors of the 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 concerns are very much dependent on the reactor type. Table A1 shows while at power operation whether reactivity will increase (+) or decrease (-) when the identified parameter increases.

Table A.1

Reactivity Coefficients

Parameter	Reactivity Coefficient				
	Pressurized Water Reactor	Boiling Water Reactor	Advanced Gas-cooled Reactor	Heavy Water Reactor	
				Pressure Tube	Pressure Vessel
Coolant Temperature	-	-	~0	+	+
Coolant Density	+	+	~0	-	-
Moderator Temperature	-	-	+	~0	-
Fuel Temperature	-	-	-	-	-
Power	-	-	-	~0	~0

ANNEX 2

PELLET-CLADDING INTERACTION

A2-1 Zirconium-Alloy-Cladding

For Zirconium-alloy clad fuel "pellet-cladding interaction", i.e. stress corrosion cracking caused when the pellet expands and stresses the cladding in the presence of corroding agent requires to be considered.

Stress corrosion cracking in Zirconium alloy cladd 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, cadmium, cesium, 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 due to 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 example:

- Local stresses can be lowered by a cladding pellet interface lubricant
- Tensile stresses can be lowered by other means, such as a slow change of power or pre-pressurisation of the fuel element
- A fission product barrier can be put between the cladding and the fission products
- The fission products can be absorbed by an additive
- The rate of rise of power can be control led to a tighter limit

- Local power peaking can be reduced by proper overall core design
- The linear heat rating of the fuel can be reduced by using a greater number of fuel elements.

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

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

A2-2 Steel Cladding

For AGR cladding pellet-cladding interaction mechanism which requires consideration is exhaustion of cladding ductility due to plastic strain. This becomes important because of the reduction in ductility caused by action of thermal neutrons on the boron and nickel in the steel clad, 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 clad plastic strain over life; alternatively a single large increase in fuel element rating could conceivably cause sufficient strain for failure.

Cladding damage from PCI due to radial strains can be limited by the use of hollow fuel pellets and careful control of cladding temperature changes to minimise differential expansions and contractions. Differential axial strains at shut downs may be controlled by the provision of circumferential "anti-stacking" grooves on some of the pellets; before operation controlled collapse is applied which thus anchors the cladding to the fuel pellets at pre-determined axial locations and avoids 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 avoid unacceptable interactions between the anti-stacking grooves and support grids for the fuel elements.

ANNEX 3

DESIGN CONSIDERATIONS FOR CORE MANAGEMENT

A3-1 POWER SHAPING

The fuel cycle adopted for a reactor type uses numerous options, within the design constraints of the reactor type, to make the most efficient use of fuel. Certain factors are important in economic reactor design and in establishing the ratio of peak-to-mean ratings. They also affect fuel rod rating histories 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

(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 content of burnable poison may be used for axial shaping.

(4) Variations within assembly - An enrichment variation or burnable poison variation 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 be the largest value possible, based on 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.

A3-2 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 ability to override 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 [illegible]s not economic to provide sufficient excess reactivity within the core [illegible]on override under all conditions. Therefore, booster or adjuster rods a[illegible]ovided 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 is compensated by boron or gadolinium dissolved in the moderator.

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 shutdown the reactor and keep it held down with an appropriate reactivity margin during the reactor's most reactive state.

Annex 4

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 offsite power
- loss of electric load

Component Malfunction

- inadvertant 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 increase of reactor coolant flow during normal operation
- steam line isolation;
- malfunction of feedwater system during normal operation
- reduction in coolant flow
- 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 SG-D11, section 7.

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]

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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.

Ballooning

The result of internal over pressure 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 absorption and used to control reactivity.

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.

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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.

*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.

*Operational States

The states defined under Normal Operation and Anticipated Operational Occurrences (see Normal Operation and Anticipated Operational Occurrences).

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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.

* 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 of which 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.

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

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 (see Anticipated Operational Occurrences and Accident Conditions).

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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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