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Design of the Reactor Core for Nuclear Power Plants

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DESIGN OF THE REACTOR CORE FOR NUCLEAR POWER PLANTS

FOREWORD

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CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide was prepared in support of new safety requirements established in the Specific Safety Requirements publication, IAEA Safety Standards Series No. SSR-2/1, Safety of Nuclear Power Plants: Design [1]~~[1]~~, which was published in 2012 and was revised (as Rev. 1) in 2016. The present publication supersedes the Safety Guide on Design of the Reactor Core for Nuclear Power Plants, Safety Standards Series No. NS-G-1.12, published in 2005 in support of the previous Safety Requirements publication (Safety Standards Series No. NS-R-1 (2000)) that was superseded by SSR-2/1 in 2012.

OBJECTIVE

1.2. The objective of this Safety Guide is to provide recommendations for safety in the design of the reactor core for nuclear power plants. This Safety Guide provides recommendations and guidance for the implementation and interpretation of safety requirements stated in the Specific Safety Requirements publication SSR-2/1 Rev. 1 [1]~~[1]~~, applied to the reactor core design.

SCOPE

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

1.4. The reactor core is the central part of a nuclear reactor where nuclear fission occurs. The reactor core consists of four basic systems and components (i.e., fuel[1]~~,~~ coolant~~,~~ moderator and control rods), and additional structures (e.g., reactor internals, core support plates, lower and upper internal structure in LWRs). This Safety Guide addresses the safety aspects of the core design and includes the neutronic; thermal-hydraulic; thermal mechanical; structural mechanical; reactor core control, shutdown and monitoring; and core management aspects for the safe design of the reactor core for nuclear power plants. Specifically, the following structures, systems and components are ~~dealt with~~covered:

(a) Fuel elements[2] that include fuel pellets with or without burnable absorbers in cladding tubes, and generate and transfer heat to the coolant;

(b) Fuel assemblies[3] that include ~~and those~~ a bundle of fuel elements, structures and components (e.g., guide tubes, spacer grids, bottom and top nozzles, fuel channel) ~~-~~that

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[1] In this Safety Guide, the term “fuel” means fuel matrix, elements and/or assemblies unless otherwise specified.

[2] The fuel element is interchangeably called the fuel rod or the fuel pin.

[3] The fuel assembly is interchangeably called the fuel bundle for PHWRs.

~~hold~~ maintain the fuel assemblies ~~and other components~~ in a predetermined geometrical configuration;

(c) Reactor core control, shutdown and monitoring systems, including c~~C~~omponents and structures used for reactivity control and shutdown, comprising the neutron absorbers (solid or liquid), the associated structure and drive mechanism;

(d) Support structures that provide the foundation for the core within the reactor vessel (within the Caladria in PHWRs), the structure for guiding the flow and guide tubes for reactivity control devices;

(e) Coolant;

(f) Moderator; and

~~(g) Reactor core control, shutdown and monitoring systems; and~~

(g~~h~~) Other reactor vessel internals such as steam separators and neutron sources. These are considered only to a limited extent in this Safety Guide.

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

1.6. For innovative fuel materials, such as uranium-nitride fuel or inert matrix fuel, or cladding materials other than zirconium-based alloys, this Safety Guide may be applicable with judgment.

1.7. The design of the reactor core may be interfaced with other reactor systems and other related aspects. In this Safety Guide, the guidance on these interfacing systems and aspects is described mainly for their functional considerations. Corresponding Safety Guides are referenced, as appropriate, in order to clarify their interfaces.

STRUCTURE

1.8. Section 2 describes general considerations for the safe core design based on the principal technical requirements and the general design requirements established in Sections 4 and 5 of Ref. [1]~~[1]~~, respectively. Section 3 describes specific considerations for the safe design of each reactor core component, based on specific design requirements (i.e., Requirements 43 through 46) of Ref. [1]~~[1]~~. Section 4 describes guidance on the qualification testing for the structures, systems and components of the reactor core. Annex 1 describes important items that need to be taken into account for the design of fuel elements, fuel assemblies and the reactivity control assemblies.

2. GENERAL SAFETY CONSIDERATIONS IN THE REACTOR CORE DESIGN

MANAGEMENT SYSTEM

2.1. The design of the reactor core should be conducted taking into account the recommendations of GS-G-3.1 [3] and GS-G-3.5 [4] to meet Requirements 1-3 of SSR-2/1 Rev. 1 [1]~~[1]~~ and GSR Part 2 requirements [2].

DESIGN OBJECTIVES

Fundamental safety functions

2.2. The three fundamental safety functions, described in Requirement 4 of Ref. [1]~~[1]~~, are required to be assured in the design of the reactor core, for operational states and a wide range of accident conditions. The fundamental safety functions are stated as follows for specific application to the design of the reactor core:

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

Adequate design based on concept of defence in depth

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

2.4. Physical barriers considered as part of or affecting reactor core design include the fuel matrix[4], the fuel cladding and the reactor pressure vessel for LWRs (or fuel channels for PHWRs). For normal operation and anticipated operational occurrences, fuel elements and assemblies~~-~~ are required to maintain their structural integrity and a leaktight barrier to prevent fission product transport into the coolant~~-~~ (Requirement 43 of Ref. [1]~~[1]~~). For design basis accidents and design extension conditions without significant fuel degradation, the reactor core is required to be coolable (Requirement 44 of Ref. [1]~~[1]~~).

Proven engineering practices

2.5. The reactor core should be of a design that has been proven either by ~~in~~ equivalent applications based on operational experience or on the results of relevant research programs, or ~~. If not, the new design, whether it applies to a component or a system, is not allowed to be used until it is proven as appropriate~~ according to the design and design verification/validation processes stated in applicable codes and standards (as indicated in paras 4.14 and 4.16 of Ref. [1]~~[1]~~).

Safety assessment in design process

2.6. As indicated in para. 4.17 of Ref. [1]~~[1]~~, the safety assessment is required to be performed as part of the design process, with iteration between the design and confirmatory analyses, and increasing in its scope and level of detail as the design progresses. Guidance on safety assessment methods is described in Ref. [5]~~[5]~~.

Features to facilitate radioactive waste management

2.7. The reactor fuel element and fuel assembly design should account for features that will facilitate ~~the~~ future waste management (including reprocessing when applicable). ~~and reprocessing.~~ Physical conditions of discharged fuel assemblies from the reactor core affect the design of the storage and disposal systems ~~of the~~for used fuel. Guidance to account for the

[4] "Fuel matrix" refers to the structure/microstructure of various types of ceramic fuel pellets.

impact of used fuel conditions on the design of ~~the~~ spent fuel handling and storage systems is described in Refs [6]~~[6]~~ and [7]~~[7]~~.

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

2.8. As indicated in Requirement 14 of Ref. [1]~~[1]~~, the design basis for the reactor core is required to specify the necessary capability, reliability and functionality for all applicable plant states in order to meet the specific acceptance criteria.

Categories of plant states and postulated initiating events

2.9. Plant states[5], as described in Requirement 13 of Ref. [1]~~[1]~~, which are typically considered for the reactor core design are normal operation, anticipated operational occurrences, design basis accidents and design extension conditions without significant fuel degradation. Design extension conditions with core melt are out of the scope of the reactor core design.

2.10. The design process should ~~be performed via~~include ~~analyzing~~ analysis of the effects of postulated initiating events on the reactor core for all applicable plant states. Guidance on the identification of postulated initiating events for all applicable plant states and relevant safety analyses is described in Ref. [5]~~[5]~~.

External hazards

2.11. Consequences of earthquake should be taken into account in the design of the reactor core. Seismic categorization of the structures, systems and components of the reactor core should be determined according to Ref. [8].

Design limits

2.12. Design limits[6] for individual structures, systems and components of the reactor core are required to be specified for all applicable plant states (as indicated in Requirement 15 of Ref. [1]~~[1]~~). The fulfilment of these limits with appropriate provisions will assure that the concept of defence in depth stated in para. 2.4 is successfully implemented with adequate margins[7]. Typical examples of these parameters with quantitative or qualitative limits are described in Section 3.

Safety classification aspects of the reactor core

2.13. The structures, systems and components of the reactor core are required to be classified on the basis of their safety function and their safety significance (see Requirement 22 of Ref. [1]~~[1]~~). The safety classification process is described in Ref. [10]~~[10]~~.

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[5] These four states, normal operation, anticipated operational occurrences, design basis accidents and design extension conditions without significant fuel degradation are called ‘all applicable plant states’ throughout this Safety Guide. ~~Design extension conditions with core melt are out of the scope of the reactor core design.~~

[6] “Design limits” are used interchangeably with the commonly used “safety limits” or “acceptance criteria” defined in Ref. [9].

[7] In the context of this Safety Guide, “margin” refers to the difference between the maximum value of a physical parameter and the acceptance criterion defined for this specific physical parameter.

2.14. Fuel elements and assemblies should be identified as ~~the highest safety class (i.e.,~~ Safety Class 1 in Ref. [10]~~[10]~~, the highest safety class~~)~~. Leaktightness and structural integrity of the fuel elements are required to ~~prevent~~ maintain these barriers to the release of ~~spread of~~ radioactive materials[8]. ~~—~~The structural integrity of fuel assemblies is required to maintain geometry compatible with design basis[9].

2.15. Failure of control rods has the potential to endanger the leaktightness and structural integrity of the fuel element which is a Safety Class 1 barrier; from this perspective, control rods should be identified as a Safety Class 1 component.

2.16. For all ~~the~~ Safety Classes identified according to the method described in Ref. [10]~~[10]~~, corresponding engineering design rules should be specified and applied.

Engineering design rules

2.17. The engineering design rules for the structures, systems and components of the reactor core represent methods to achieve the adequacy of the design and should include the following, as appropriate:

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

Design for reliability

2.18. According to para. 5.37 of Ref.[1]~~[1]~~, fuel elements and assemblies, and components and systems for reactor control and shutdown are required to be designed with high reliability, in consideration of their safety significance. Provision for achieving high reliability in these designs is described in paras 3.27 and 3.88, respectively.

Operational limits and conditions

2.19. As indicated in Requirement 28 of Ref. [1]~~[1]~~, the operational limits and conditions are required to be established to assure that the reactor core operates safely, in accordance with design assumptions and intent. ~~(parameters and components), and to include those limits within which the reactor core has been shown to be safe.~~ Relevant guidance on the operational limits and conditions is described in the Safety Guide NS-G-2.2 [11]~~[11]~~.

DESIGN FOR SAFE OPERATION

2.20. The structures, systems and components of the reactor core should be designed such that their required testing, inspection, repair, replacement, calibration or maintenance is facilitated.

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[8] Cladding leaktightness should be maintained to prevent ~~the release of~~ volatile fission ~~products~~ products release and cladding structural integrity should be maintained to prevent solid product dispersal.

[9] Fuel assembly structural integrity should be maintained to ensure a coolable geometry during accident conditions.

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

(a) A significant change in fuel types (e.g., mixed-oxide fuel);
(b) An increase of the discharge burnup beyond the current design limit;
(c) A significant increase in the duration of a fuel cycle;
(d) An increase in the rated power of the plant; and
(e) A significant change in the operating domain.

2.22. The fuel elements and fuel assemblies~~y~~ should be designed to prevent the potential for fuel failures due to specific operational conditions (e.g., startup rates, degraded coolant ~~chemistries~~chemistry conditions, presence of foreign material, etc.) during operational states.

2.23. The LWR core should be designed such that the consequences of the worst misloaded fuel assembly, if any, remain within nuclear and fuel design limits.

REACTOR CORE SAFETY ANALYSIS

2.24. As indicated in Requirement 42 of Ref. [1]~~[1]~~, ~~the reactor core~~ safety analysis is required to ~~be performed to~~ evaluate and assess ~~the~~ challenges to safety in all applicable plant states, using ~~the~~ deterministic approaches.

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

(a) ~~The~~ O~~o~~perating state (e.g., ~~the~~ thermal-hydraulic conditions, ~~and the~~ power levels and time in the cycle ~~at subcritical, part load, full load and xenon transient~~);
(b) ~~The~~ T~~t~~emperature coefficient of reactivity for the fuel (Doppler coefficient);
(c) ~~The~~ T~~t~~emperature coefficients of reactivity for the coolant and the moderator;
(d) ~~The~~ V~~v~~oid coefficients of reactivity for the coolant and the moderator;
(e) ~~The~~ R~~r~~ate of change of the concentration of soluble absorber in the moderator and the coolant;
(f) ~~The~~ R~~r~~ate of insertion of positive reactivity caused by the reactivity control device(s) or changes in process parameters;
(g) ~~The~~ R~~r~~ate of insertion of negative reactivity associated with a reactor trip;
(h) Individual channel transient response related to the average thermal power of the core (for BWRs);
(i) ~~The~~ P~~p~~erformance characteristics of safety system equipment, including the changeover from one mode of operation to another (e.g., from the injection mode for emergency core cooling to the recirculation mode); and
(j) ~~The~~ D~~d~~ecay of xenon and other neutron absorbers in the long term core analysis.

~~Adequate~~ Appropriate provisions or margins should be included in the above factors such that the safety analyses remain valid for specific loading patterns or fuel designs.

2.26. Safety analysis for the reactor core should be performed to verify that fuel design limits are not exceeded in all allocable plant states. During accident conditions t~~T~~he effect ~~on core cooling~~ of ~~such~~ fuel behavior ~~conditions~~ on core cooling should be included in the safety analysis, e.g., ~~as~~ ballooning and rupture of the cladding, exothermic metal–water reactions and distortions of fuel elements and fuel assemblies~~should be included in the safety analysis~~.

Accumulation of ~~The~~ hydrogen ~~buildup~~ , as a result of exothermic reaction between the ~~Zircaloy~~ zirconium based alloy cladding and water at high temperature, ~~will~~ may threaten the integrity of the pressure vessel for LWRs (of the pressure tubes for PHWRs) and of the containment and ~~therefore it~~ should ~~also~~ be evaluated.

3. SPECIFIC SAFETY CONSIDERATIONS IN THE REACTOR CORE DESIGN

GENERAL

3.1. This section addresses specific design aspects for the structures, systems and components of the reactor core to meet Requirements 43-46 described in the Specific Safety Requirements publication SSR-2/1 Rev. 1 [1]~~[1]~~. It also includes the interface with core management which strongly influences the core design with regard to the performance of fuel elements and fuel assemblies. ~~as well as the fuel economy.~~ Specific guidance for core management and fuel handling in nuclear power plants is described in Ref. [15].

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

3.3. The reactor core and associated control and protection systems should be designed with ~~appropriate~~ adequate margins to assure that fuel design limits are not exceeded during all applicable plant states. Fuel design limits are described in paras 3.49-3.59.

Fuel type

3.4. ~~The~~ F~~f~~uel elements ~~for use in thermal reactors~~ contains ~~a~~ fissile materials (e.g., ~~uranium~~U-235, Pu-239) that ~~is~~ are highly reactive with thermal neutrons. Fuel pellet materials ~~used in thermal reactors~~[10] should be selected with consideration of the following optimized properties:

(a) Reactivity with thermal neutrons;
(b) Impurities with low thermal neutron absorption properties;
(c) Thermal performance (~~i.e.,~~e.g., high thermal conductivity for operational states and high thermal diffusivity for accident conditions are desirable);
(d) Dimensional stability;
(e) Fission gas retention; and
(f) Pellet-cladding interaction resistance.

~~Examples of the pellet material include:~~

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[10] Examples of the pellet material include: (a) Enriched uranium dioxide (UO_2); (b) Natural uranium dioxide (UO_2) (for use in PHWRs); (c) Mixed oxide (UO_2-PuO_2); (d) Thorium-based fuel (e.g., ThO_2, thorium-blended UO_2, thorium-blended mixed-oxide fuel); (e) Reprocessed uranium dioxide (UO_2); and (f) Doped fuel pellets (e.g., Cr, Al, Si) to improve their performance (for use in LWRs). Burnable absorber material (e.g., Gd, Dy, B and Er) may be used, for example, blended in sintered UO_2 pellets or coated on their surface, to suppress temporarily the excess reactivity resulting from a high concentration of the fissile material in the fuel.

~~(a) Enriched uranium dioxide (UO_2);~~
~~(b) Natural uranium dioxide (UO_2) (for use in PHWRs);~~
~~(c) Mixed oxide (UO_2-PuO_2) ;~~
~~(d) Thorium-based fuel (e.g., ThO_2, thorium-blended UO_2, thorium-blended mixed-oxide fuel);~~
~~(e) Reprocessed uranium dioxide (UO_2); and~~
~~(f) Doped fuel pellets (e.g., Cr, Al, Si) to improve their performance (for use in LWRs).~~

~~Burnable poison material (e.g., Gd, Dy, B and Er) may be used, for example, blended in sintered UO_2 pellets or coated on their surface, to suppress temporarily the excess reactivity resulting from a high concentration of the fissile material in the fuel.~~

3.5. ~~The fissile material is typically fabricated in cylindrical form of sintered pellets, and is loaded in cylindrical cladding tubes that have low neutron absorption properties and high mechanical strength. The c~~Cladding materials[11] should be selected with consideration of the following properties:

(a) Low absorption cross-section for thermal neutrons and high resistance to irradiation conditions;
(b) High thermal conductivity and high melting point;
(c) High corrosion resistance and low hydrogen pick-up;
(d) Low oxidation/hydriding at high temperature conditions;
(e) Adequate breakaway oxidation resistance at ~~high~~ integrated-time~~-integrated~~ temperature conditions[12];
(f) Chemical inertness with the coolant and the fuel at all temperatures;
(g) Adequate mechanical properties, e.g.,~~-~~ high strength and high ductility; and
(h) Low susceptibility to stress corrosion cracking.

~~Zirconium-based alloy materials (e.g., Zircaloy-2, Zircaloy-4, Zirlo and Optimized Zirlo, M5, E110) are typically used for the cladding material. Other innovative cladding materials, e.g., enhanced accident tolerant fuel, with focus on more benign steam reaction and lower hydrogen generation, are under development.~~

Coolant

3.6. In LWRs, the coolant also acts as the moderator. ~~Chemical additives to the coolant may be used as neutron absorbers to provide a second system of control over the core reactivity. An example of this is the boric acid used in PWRs. Other chemical additives (e.g., Zn, H, Li, Cu) can also be used to control the chemistry of the coolant (e.g. to control pH and oxygen content) in order to inhibit corrosion of or crack propagation in core components and reactor internals, and thereby to reduce contamination of the reactor coolant system through crud generation.~~ The design of coolant should account for ~~the effects~~all potential interactions

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[11] Zirconium-based alloy materials (e.g., Zircaloy-2, Zircaloy-4, ZIRLO™ and Optimized ZIRLO™, M5®, E110) are typically used for the cladding material. Other innovative cladding materials, e.g., enhanced accident tolerant fuel, with focus on more benign steam reaction and lower hydrogen generation, are under development.
[12] Integrated-time temperature refers to the assessment of total time achievable at a given cladding temperature without reaching oxidation breakaway (uncontrolled oxidation kinetics).

between chemical conditions[13] in the coolant and ~~of these chemicals on~~ fuel and core components.

For PHWRs, the coolant and the moderator are separated; typically, chemicals are not added to the coolant for controlling reactivity.

3.7. The coolant should be physically and chemically stable with respect both to high temperatures and to nuclear irradiation in order to fulfil its primary function~~:~~ the continuous removal of heat from the core[14]. ~~provided that fuel and core coolable geometry is maintained and that the core is designed to prevent or control flow instabilities and consequent fluctuations in reactivity.~~ The reactor core design should also include the following safety considerations associated with the coolant that affect the fuel and core design: ~~:~~

(a) Ensuring that the reactor coolant system is free of foreign materials prior to the initial start-up of the reactor and following refuelling and maintenance outages for the operating lifetime of the plant~~;~~
(b) Keeping the radionuclide activity in the coolant at an acceptably low level by means of purification systems, corrosion product minimization, ~~and~~ or removal of defective fuel as appropriate;
(c) Monitoring and controlling the effects that the coolant and coolant additives have on reactivity under all plant states; and
(d) Determining and controlling the physical and chemical properties of the coolant in the core.

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

Moderator

3.9. The choice of moderator and the spacing of the fuel elements within it should meet engineering and safety requirements on temperature coefficient of reactivity for the moderator, while aiming at optimizing ~~be based on the need to optimize~~ the neutron economy~~,~~ and hence fuel consumption. ~~, and to meet engineering and safety requirements.~~ The prevalent thermal reactor types use either light water or heavy water as the moderating medium.

Depending on the reactor design, t~~T~~he moderator ~~should be allowed to~~may contain a soluble neutron absorber to maintain adequate shutdown margins during operational states. ~~.~~

3.10. In PHWRs, the reactor core design should assure the effectiveness of the shutdown and hold-down capability of the reactor during an absorber dilution accident (e.g. an in-core break). Means should be provided to prevent the inadvertent removal of such absorber material (e.g. due to chemistry transients) and to ensure that its removal is controlled and

[13] Chemical additives to the coolant may be used as neutron absorbers to provide a second system of control over the core reactivity. An example of this is the boric acid used in PWRs. Other chemical additives (e.g., Zn, H, Li, Cu) can also be used to control the chemistry of the coolant (e.g. to control pH and oxygen content) in order to inhibit corrosion of or crack propagation in core components and reactor internals, and thereby to reduce contamination of the reactor coolant system through crud generation.

[14] In this case, fuel and core coolable geometry should be maintained and the core should be designed to prevent or control flow instabilities and resultant fluctuations in reactivity.

slow. The moderator should provide the capability to remove decay heat without loss of core geometry for ~~operational states and accident conditions~~all applicable plant states. ~~Because of radiolysis of the moderator,~~ Mmeasures should be provided to prevent ~~either hydrogen~~ deflagration or explosion of hydrogen generated by radiolysis in the moderator.

NEUTRONIC DESIGN

Design considerations

3.11. The ~~reactor core design~~ design of the reactor core should assure that the feedback characteristics of the core rapidly compensate for an increase in reactivity. The reactor power should be controlled by a combination of the inherent neutronic characteristics of the reactor core and its thermal-hydraulic characteristics, and the capability of the control and shutdown systems to actuate for all applicable plant states.

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

Nuclear design limits

Nuclear key safety parameters

3.13. Nuclear key safety parameters influencing neutronic core design and fuel management strategies should be established from the ~~deterministic~~ safety analyses that verify the compliance with specific fuel design limits described in paras 3.49-3.59. ~~Adequate~~ Appropriate provisions should also be provided on the nuclear key safety parameters, such that they would remain valid for specific core reload designs. Typical nuclear key safety parameters include:

(a) ~~The~~ Ttemperature coefficients of reactivity for fuel, coolant and moderator;
(b) ~~The~~ Bboron reactivity coefficient and concentration (PWR);
(c) ~~The~~ Sshutdown margin;
(d) ~~The~~ Mmaximum reactivity insertion rates;
(e) ~~The~~ Ccontrol rod and control bank worth;
(f) ~~The~~ Rradial and axial power peaking factors, including allowance for Xe induced oscillation;
(g) ~~The~~ Mmaximum linear heat generation rate; and
(h) ~~The~~ Vvoid coefficient of reactivity.

The safety impacts of any major modifications on the reactor core design should be assessed using the nuclear key safety parameters, in order to assure that the specified fuel design limits are not violated. Otherwise, new nuclear key safety parameters should be defined and justified. Such modifications may include:

(a) ~~In preparation for any~~ Mmajor plant design, equipment or operational modifications;
(b) ~~including~~ Mmajor ~~in-core~~ fuel management changes, such as large cycle length extension;
(c) Nnew fuel type introduction (e.g., mixed-oxide fuel or gadolinium fuel)); and~~or~~
~~(a)~~(d) Fuel burnup limit extension. ~~the safety impacts of the modifications on the reactor core design should be assessed using the nuclear key safety parameters, in order to assure that the specified fuel design limits continue to be met under the modifications.~~

Core reactivity characteristics

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

3.15. Nuclear k~~K~~ey ~~reactivity~~ safety parameters such as reactivity coefficients should be evaluated for ~~each core state~~selected core operating conditions (e.g., zero power, full power, beginning of cycle, end of cycle) and for the corresponding ~~strategy for~~ fuel management strategy. Their dependence on the core loading and the burnup of the fuel should be analyzed. ~~Adequately~~ Appropriate provisions~~conservative assumptions~~ should be ~~applied~~ included ~~for~~ in the reactivity coefficients used in the safety analysis for all applicable plant states.

Maximum reactivity worth and reactivity insertion rate

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

(a) Control rod ejection;
(b) Control rod drop;
(c) Boron dilution; and
(d) Uncontrolled bank withdrawal.

These reactivity limits should be determined via safety analys~~i~~es to ensure that fuel design limits described in paras 3.49-3.59 are met. These ~~reactivity insertion~~ analyses should be performed for all fuel types in the core (e.g. UO_2 or mixed-oxide fuel) or a representative core with appropriate provisions and as a function of allowable operating conditions and fuel exposure.

Control of global and local power

3.17. The design should assure that the core power can be ~~-~~controlled globally and locally using the means of reactivity control in such a way that the peak linear heat generation rate of each fuel element does not exceed the specified limits anywhere in the core. Variations in the power distribution caused by local variations in reactivity due to xenon instability or other local effects (e.g., mixed core, crud induced power shifts or axial offset anomalies for PWRs, fuel assembly bow or distortion) should be addressed~~-~~ in the design of the control system. Provisions and uncertainties should be included to account for flux detectors' measurement variations (e.g., due to operability, location, shadowing and ageing).

Power changes can be controlled by movement of the control rods, or other means that may include~~. Additional design features may include~~:

(a) Arranging ~~the~~ control rod banks so as to avoid ~~the~~ large radial and axial distortions of the power distribution (PWR);
(b) Adjusting ~~the~~ boron concentration of the reactor coolant to control reactivity ~~-~~(PWR);
(c) Adjusting ~~the~~ circulation flow rate (BWR); and
(d) Adjusting ~~the~~ levels of light water in liquid zone compartments, and also solid absorber and/or adjuster rods and liquid absorber (PHWR).

Shutdown margin[15]

3.18. The insertion of control rods should provide adequate shutdown margin in all applicable plant states. The specification and monitoring of control rod insertion limits as a function of power level should assure adequate shutdown margin at all times.

3.19. The effects of ~~the~~ depletion of burnable ~~poison~~absorber[16] on the core reactivity should be evaluated to ensure adequate shutdown margin in all ~~the~~ resulting applicable core conditions throughout the fuel cycle.

THERMAL-HYDRAULIC DESIGN

Design considerations

3.20. The thermal-hydraulic design of the reactor core should include ~~appropriate~~ adequate margins ~~and provision~~and provisions to assure that

(a) S~~s~~pecified thermal-hydraulic design limits are not exceeded in operational states (i.e., during normal operation and anticipated operational occurrences);~~,~~ and

~~(a)~~(b) T~~t~~he failure rates of fuel elements during design basis accidents and design extension conditions without significant fuel degradation remain below acceptance levels.

Thermal-hydraulic design limits

3.21. Specific thermal-hydraulic design limits should be established with ~~sufficient~~ adequate margins on predictable parameters, such as the maximum linear heat generation rate, the minimum critical power ratio (for BWRs) or the minimum departure from nucleate boiling ratio (for PWRs) or dryout power ratio (for PHWRs), the peak fuel temperature or enthalpy, and the peak cladding temperature. Uncertainties in the values of process parameters (e.g., reactor power, coolant flow rate, core bypass flow, inlet temperature and pressure, nuclear and engineering hot channel factors), core design parameters, and calculation methods used in the assessment of thermal margin should be accounted for in the design analyses.

3.22. The thermal-hydraulic design should include design analyses that take into account design features of ~~the~~ a fuel assembly including fuel element spacing, ~~the~~ fuel element power, ~~the~~ sizes and shapes of subchannels, spacer and mixing grids (for LWRs), ~~braces (for LWRs),~~ and flow deflectors (for LWRs) or turbulence promoters. In addition, for fuel channel type PHWRs, effects of fuel bundle string, appendages, gaps between fuel elements and ~~the~~ pressure tube, and junctions between neighboring endplates should be addressed in the design analyses.

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15 Shutdown margin is not defined in Ref.[9]~~[9]~~; however, it is generally accepted as the instantaneous amount of reactivity by which a reactor remains subcritical from its present conditions assuming all full-length rod cluster assemblies are fully inserted except for the single rod cluster assembly of highest reactivity worth that is assumed to be fully withdrawn.

16 For PWRs, in order to maintain a negative moderator temperature coefficient, the designer may choose to reduce the required concentration of the burnable absorber in the moderator by adding fixed burnable ~~poison~~ absorber to the fuel pellet or to the fuel assembly in the form of burnable ~~poison~~ absorber rods. Burnable ~~poison~~ absorber may also be used to flatten the power distribution and to reduce variations in reactivity during fuel burnup.

For LWRs, the thermal-hydraulic design should also consider core inlet and outlet coolant temperature and flow distributions. These effects should also be considered in the core monitoring and protection systems.

The design should assure that the minimum ratio of operating power to critical power (i.e., 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) should cover the fact that the critical heat flux correlations have been developed from tests performed at ~~steady state power ratios are maintained within limits established for defined ratios~~steady-state conditions ~~(i.e., 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)~~. ~~in order to provide margins to avoid critical heat flux conditions and related cladding damage. The~~ As a consequence, adequate margin~~s~~ should ~~be~~ be provided ~~sufficient~~ to allow for anticipated operational occurrences. The objective is to avoid potential for cladding failure by using critical heat flux limits as surrogates[17]. ~~These minimum ratios should be established to provide a conservative design basis for operational states for water cooled reactors.~~

3.23. Experiments should be conducted over the range of expected operational conditions to provide data for defining the limiting values of the minimum ratios. Correlations for predicting critical heat flux are continually being revised as a result of additional experimental data, changes in fuel assembly design, and improved calculation techniques involving coolant mixing and the effect of axial power distributions. These correlations may be overly conservative for fast transients (e.g., rod ejection accidents), and therefore, they may be reassessed for these applications.

3.24. Approaches as the following examples should be taken to demonstrate the fulfilment of paras 3.21-3.23:

(a) For departure from nucleate boiling ratio, critical heat flux ratio or critical power ratio correlations, there should be a 95-percent probability at the 95-percent confidence level that the hot element[18] in the core does not experience a departure from nucleate boiling or boiling transition condition during normal operation or anticipated operational occurrences;

(b) The minimum value of departure from nucleate boiling ratio, critical heat flux ratio, or critical power ratio correlations should be established such that the number of fuel elements that experience a departure from nucleate boiling or boiling transition during normal operation or in anticipated operational occurrence conditions does not exceed a limit, e.g., at least one element per thousand in the reactor core; and

(c) For PHWRs, if the maximum fuel cladding temperature remains below a certain limit (e.g., 600 ^{o}C) and the duration of post-dryout operation is limited (e.g., less than 60

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

[18] The hot element is referred to the fuel element with the highest relative power, considering the conservative radial core power distribution.

seconds), it is considered that the fuel deformation is small so that fuel elements are not in contact with the pressure tube and will not cause a failure of the pressure tube.

FUEL ELEMENT AND FUEL ASSEMBLY THERMAL MECHANICAL DESIGN

Design considerations

3.25. The design should assure that the structural integrity of the fuel assemblies (geometry) and the structural integrity of the fuel elements (leaktightness) ~~of the fuel element~~ ~~is~~ are maintained for normal operation and anticipated operational occurrences. For accident conditions (design basis accidents and design extension conditions without significant fuel degradation)~~,~~ only a limited number of fuel failures should be allowed.~~ to occur while maintaining a~~ L~~l~~ong term coolable geometry should be maintained. Under these conditions, the level of radionuclide activity should be assessed to confirm that the permissible limits for the release of fission products are met.

3.26. The design of the fuel elements (with or without burnable absorbers)~~,~~ ~~control devices, burnable poisons~~ and fuel assemblies should address the irradiation and environmental conditions (e.g., temperature, pressure, coolant chemistry, irradiation effects on fuel, cladding and fuel assembly structure; static and dynamic mechanical loads including flow induced vibration; and changes in the chemical characteristics of the constituent materials). The design should assure that these items can withstand handling loads in transport, storage, installation and refuelling operation. Important items that are typically considered for the design of fuel elements, fuel assemblies, ~~and~~ the reactivity control assemblies, burnable absorbers and neutron source assemblies are described in Annex I.

3.27. The design should assure that fuel elements and fuel assemblies are reliable throughout their lifetime including manufacturing, transportation, handling, in-core operation, interim storage, and disposal, where applicable. Several key contributors throughout their lifetime should be addressed; important key contributors to fuel reliability ~~as identified by the Institute of Nuclear Power Operations (INPO)~~ include:

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

Thermal and burnup effects on fuel elements

3.28. In all operational states, the design should assure that the peak fuel temperature is lower than the fuel melting temperature[19] by an~~ sufficient~~ adequate margin to prevent melting of the fuel, when appropriate ~~allowances~~ provisions ~~for~~ and uncertainties are ~~applied~~considered. In design basis accidents (e.g., reactivity initiated accidents) and design extension conditions without significant fuel degradation, limited volume of fuel melting can be allowable locally. In the assessment of the peak fuel temperatures during operational states, the following burnup dependent phenomena should be addressed: the changes in fuel thermal

[19] Due to irradiation effects mentioned above, the fuel melting temperature varies as a function of fuel burnup and should be determined using representative irradiated fuel samples.

conductivity/diffusivity and in pellet-cladding gap thermal conductance, fuel densification, fuel swelling, accumulation of fission products in the fuel pellets, fission gas release~~s~~ in the free volumes of the fuel elements and any other changes in the pellet~~s'~~ microstructure.

3.29. The design should assure that cladding strains are limited. Limits for cladding long term deformation and corrosion/hydriding, therefore, should be specified for different operational states and accident conditions. The design of fuel pellets and fuel cladding should account for the changes in the mechanical properties (strength, creep and stress relaxation) as a function of irradiation and temperature. Straining of the cladding is caused by fuel element internal gas overpressure or by fuel gaseous swelling or fuel thermal expansion as a consequence of fuel burnup or local power increases.

Solid and gaseous fission products affect the thermal mechanical behavior of fuel elements during normal operation and anticipated operational occurrences~~, and also are used for specifying the initial conditions for accident analyses~~. The production and release rates of those fission products within each fuel element depend largely on its power history during in-core residence. Fission products ~~releases~~ will have various impacts on the fuel element~~s~~ design and accident analyses which need to be evaluated such as:

(a) ~~(a)~~ Increase of the fuel element internal pressure;
~~(a)~~(b) Degradation of the thermal conductivity of the pellets;
(c~~b~~) Degradation of the thermal conductance of the pellet-to-cladding gap;
(d~~c~~) Stress corrosion of the inner side of the cladding during power transients; and
(e~~d~~) Gaseous fission products induced swelling of the fuel stack.

The consequences of significant cladding deformation (e.g., cladding ballooning) during operational states should be evaluated in accident analyses to determine the potential for cladding failure (e.g., burst or rupture) and any resulting release of fission products from the fuel.

Effects of irradiation on fuel assembly structures

3.30. The design should assure that the dimensional changes of LWR fuel assembly structures ~~is~~ are ~~constrained~~limited, so that contacts or interactions between the fuel elements and the fuel assembly components (top and bottom fuel assembly nozzles) ~~is~~ are avoided, and that fuel element bow and assembly bow, as well as control rods swelling and any potential interaction with the assembly guide tubes do not ~~impede maintaining the structural~~affect the structural integrity of fuel assemblies ~~and~~ or ~~performing~~ the performance of ~~the safety functions of~~ control rod safety function~~s~~s.

Grid spring~~s~~ relaxation under irradiation should be assessed to limit the ~~risk~~ potential for ~~end of life~~ grid-to-fuel element fretting. In the dimensional stability analyses, the effects of irradiation, in particular, the effects of fast neutrons on fuel assembly components and control devices, on their ~~metallurgical~~ mechanical properties such as tensile strength, ductility, growth or creep/relaxation should be taken into account. The effect of irradiation on ~~the~~ buckling resistance of the spacer grids should be considered when assessing seismic events or loss-of-coolant accidents.

3.31. For PHWRs, the design should assure that the fuel cavity length in the fuel channel is sufficient to accommodate the irradiation and thermal effects on the fuel string in the fuel channel for all applicable plant states.

Effects of variations in power levels

3.32. For all operational states, the design should assure that the fuel elements withstand thermal mechanical loads during local and global power transients (due to fuel assembly shuffling, to movements of control devices or to other causes of reactivity changes).

Mechanical effects in fuel elements

3.33. The design should include analyses to assure that stressing and straining of the fuel cladding due to mechanical loads[20] (e.g., coolant pressure, seismic loads) meet fuel design limits.

~~Due to the effect of external coolant pressure, temperature and irradiation, the cladding creeps down, leading to a decrease in the radial gap between the cladding and the fuel pellets. The gap closure kinetics, which is important when evaluating pellet-cladding interaction, depends on various parameters such as fuel densification properties, fuel gaseous swelling, fuel pellet cracking and repositioning of fragments, cladding creep behaviors at low stresses, initial rod internal pressure, fission gas release to the free volumes, and operating parameters including power history and coolant pressure.~~

3.34. Stress corrosion cracking induced by pellet-cladding interaction in the presence of fission products should be minimized.

3.35. ~~The design should assure that the fuel cladding withstands~~ S~~s~~tress concentration~~s~~ ~~due to local deformation conditions including~~ caused by ~~unloaded~~ missing pellets, axial gaps between fuel pellets, missing pellet surface or fuel pellet chips trapped in the gap~~s~~ ~~and non-uniform fuel densification~~cannot be explicitly considered in the fuel element design, and hence those anomalies should be avoided.

Effects of burnable ~~poison~~ absorber in the fuel

3.36. The design should include analyses to demonstrate that the fuel element can accommodate the effects of any in-fuel burnable ~~poisons~~ absorbers on fuel pellets thermal, mechanical, chemical, and microstructural properties and on fuel element behavior. ~~The amount and the kinetics of volatile fission products releases to the free volumes of the fuel elements might be affected by the presence of burnable poisons in the fuel pellets.~~

Corrosion and hydriding

3.37. The cladding design should ensure that ~~the appropriate~~ hydrogen pickup ~~fraction~~ correlation[21] is adequately specified for each cladding type. ~~In-reactor corrosion reduces the bearing thickness of the cladding but the hydriding of the cladding, which is a consequence of~~

[20] Due to the effect of external coolant pressure, temperature and irradiation, the cladding creeps down, leading to a decrease in the radial gap between the cladding and the fuel pellets. The gap closure kinetics, which is important when evaluating pellet-cladding interaction, depends on various parameters such as fuel densification properties, fuel gaseous swelling, fuel pellet cracking and repositioning of fragments, cladding creep behavior at low stress, initial rod internal pressure, fission gas release to the free volumes, and operating parameters including power history and coolant pressure.

[21] In-reactor corrosion reduces the bearing thickness of the cladding but hydriding of the cladding, which is a consequence of corrosion mechanism, can be more detrimental because it degrades the mechanical properties of the cladding. As a result, some fuel design limits such as those for reactivity initiated accident and loss of coolant accident are now expressed as a function of cladding pre-transient hydrogen content rather than the amount of corrosion or the burnup levels.

~~the corrosion mechanism, can be more detrimental because it degrades the mechanical properties of the cladding. As a result, some fuel design limits, such as those for reactivity initiated accident and loss of coolant accident are now expressed as a function of the cladding pre-transient hydrogen content rather than the amount of corrosion or the burnup levels.~~

3.38. Fuel elements and fuel assemblies should be designed to be compatible with the coolant environment[22] in all operational states, including shutdown and ~~refuelling~~refueling. ~~. Corrosion and hydriding depend strongly on the material performances and on the operating conditions, such as the temperature, the coolant chemistry and the linear heat generation rates (governing, for a given discharge burnup, the allowable irradiation time). These environmental conditions should be considered. In order not to degrade the corrosion performance of the materials, appropriate water chemistry should be implemented (e.g., by maintaining a low oxygen content and the appropriate pH level).~~

3.39. For PHWRs, ~~The~~ initial hydrogen content ~~of~~ in ~~zirconium alloy~~the fuel element ~~cladding~~ should be limited to reduce the likelihood of fuel defects being caused by hydrogen induced embrittlement of the cladding. ~~To achieve this, the moisture content inside the fuel element should be controlled during the manufacturing process.~~

Crud

3.40. The design analyses should account for the degradation in the fuel element heat transfer due to the formation of deposits on the surface of the cladding via corrosion products coming from the reactor coolant system or other chemical changes. In case boron is trapped in the crud layer in PWRs, its potential impact on the neutronic performance of the core should be assessed and accounted for in the core design analyses.

Hydraulic effects in fuel assemblies

3.41. Hydraulic effects should be addressed primarily in the thermal-hydraulic design of the fuel assembly, and in the evaluation of localized corrosion, erosion, flow-induced vibration and fretting of the fuel assembly. ~~Hydraulic effects can be generated due to fuel element spacing, fuel element power, sub-channel sizes and shapes, grids (for LWRs), spacers, braces (for LWRs), flow deflectors (for LWRs) or turbulence promoters. Additional sources of hydraulic effects include: appendages (i.e., spacers, bearing pads and critical heat flux enhancement pads) and endplates (for PHWRs); and cross-flows induced by mixed cores or fuel elements/assembly bows, baffles jets and premature grid-to-rod springs relaxation (for PWRs).~~ Hydraulic effects on the fuel assembly design should be demonstrated via fuel assembly endurance tests performed in qualified out-of-reactor loops, using full scale fuel assembly mock-ups with prototypical test conditions (e.g., pressure, temperature, cross-flows and end-of-life grid spring relaxation).

Consideration of mechanical safety in the design

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

(a) Fuel handling and loading;

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[22] ~~. Corrosion~~Corrosion and hydriding depend strongly on the material performances and on the operating conditions, such as the temperature, the coolant chemistry and the linear heat generation rates (governing, for a given discharge burnup, the allowable irradiation time). These environmental conditions should be considered. In order not to degrade the corrosion performance of the materials, appropriate water chemistry should be implemented (e.g., by maintaining a low oxygen content and the appropriate pH level).

(b) Power variations;
(c) Hold down loads for PWRs (which should balance the hydrodynamic lift-off forces and the geometrical changes of the core cavity and of the fuel assemblies under irradiation);
(d) Temperature gradients;
(e) Hydraulic forces, including cross-flows between distorted fuel assemblies or mixed fuel assembly concepts;
(f) Irradiation effects (e.g. irradiation induced ~~-~~growth and swelling);
(g) Fuel element vibration and fretting wear (grid-to-~~rod~~ fuel element fretting for LWRs, between spacers for PHWRs) ~~-~~induced by coolant ~~-~~flow;
(h) Creep deformation of the fuel assembly structure (that may lead to fuel assembly distortion);
(i) Safe shutdown earthquake loading conventionally combined with loss-of-coolant accidents; and
(j) Postulated initiating events (i.e., anticipated operational occurrences and design basis accidents) and design extension conditions without significant fuel degradation.

3.43. For all applicable plant states, ~~operational states,~~ the following mechanical safety aspects should be addressed in the design of fuel elements and fuel assemblies:

(a) The clearance within and adjacent to the fuel assembly should provide space to allow for irradiation induced growth and ~~swelling~~ bowing (LWRs) and bulging of the fuel channel[23] (~~for~~ B~~L~~WRs);
(b) Bowing of fuel elements or distorsion of ~~fuel elements~~and assemblies should be limited so that thermal-hydraulic behaviour,~~,~~ -power distribution, fuel performance, and fuel handling are not adversely affected;
(c) Strain induced fatigue should not cause the failure of a fuel assembly;
(d) Fuel assembly distortion as a result of mechanical and hydraulic hold down forces and in-core cross-flows should be limited to a level which does not impact the local critical heat flux margins. Also, the fuel assembly distortion should not impair the insertion of the reactivity control cluster assembly (e.g., drop time in PWRs) to ensure safe reactor shutdown during all applicable plant states (for LWRs); and
(e) Vibration and fretting damage should not affect the overall performance and functions of the fuel assembly and the support structure.

3.44. For accident conditions ~~ranging from~~ (design basis accidents and ~~to~~ design extension conditions without significant fuel degradation), the design should prevent any interaction between ~~the~~ fuel elements or~~,~~ fuel assemblies~~, fuel string (for PHWRs)~~ and fuel assembly support structures ~~should be designed to ensure that any interactive effects between these do~~that ~~not~~would ~~prevent~~ impede safety systems from performing their functions as specified in the safety analysis. In particular, the following should be assured: ~~by ensuring:~~

(a) Proper~~The proper~~ functioning of the components of safety systems (e.g., shutdown devices and their guide tubes for PWRs) ~~is maintained~~; and
(b) Proper c~~C~~ooling of the core ~~is not impeded.; and~~

[23] In BWRs, the pressure difference between the inside and outside of the boundary of the fuel channel ~~box~~ may induce ~~swelling~~ bowing and bulging of the fuel channel ~~box~~. This deformation, as well as fuel cladding bowing, may consequently increase the local flux peaking ~~factor~~and may cause friction affecting control rods movement.

~~Mechanical, thermal, or irradiation damage does not affect the integrity of the pressure boundary of the reactor coolant system.~~

Fuel pellet-cladding interaction

3.45. The design should assure that no fuel cladding failure takes place due to ~~excessive~~ pellet-cladding mechanical interaction[24] during normal operation and anticipated operational occurrences. ~~The cladding creepdown and fuel pellet thermal expansion and gaseous swelling will lead to strain-driven pellet-cladding mechanical interaction during all applicable plant states. The failure mode via pellet-cladding mechanical interaction is presented by cladding ductility exhaustion.~~ Fuel element design analysis and p~~P~~lant specific guidelines for power changes during normal operation should be provided to prevent excessive pellet-cladding mechanical interaction.

During ~~rapid~~ design basis accidents that lead to rapid power transients (e.g., reactivity initiated accident), fuel cladding can fail due to excess pellet-cladding mechanical interaction ~~at high burnup where excess hydrides~~combined with cladding embrittlement due to in-reactor hydriding at high burnup. ~~exist in the cladding.~~ ~~Failure conditions~~Fuel failures corresponding to this failure mode ~~are expected~~should be considered in safety analysis. ~~to be different from those for operational transients.~~

3.46. The design should assure that ~~no~~ likelihood of stress corrosion cracking in the fuel cladding ~~takes place~~is minimized during normal operation and anticipated operational occurrences. Stress corrosion cracking in the fuel cladding occurs when the stresses on the inner surface of the cladding (as a result of pellet-cladding interaction) reach a certain limit under a corrosive environment. After a power reduction, the thermal contraction of the fuel pellets causes re-opening the pellet-cladding gap (or the gaps between the pellets fragments). If the reduced power operation is maintained long enough (i.e., extended reduced power operation), the fuel cladding will creep down and close the gaps again. The fuel element is then considered as re-conditioned at this lower power level. When the reactor core goes back to full power later on, tensile stresses will appear in the cladding. Those residual stresses will increase the susceptibility to stress corrosion cracking driven by pellet-cladding interaction under corrosive fission product environments in the fuel element.

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

(a) Reduce tensile stresses in the fuel cladding by restricting rates of power change (allowing for the cladding stresses to relax) or by delaying the time at which the pellet-cladding gap closes (this can be achieved by increasing the initial fill gas pressure in the fuel element or by optimizing the creep properties of the cladding);

(b) Reduce corrosive effects of the fission products (e.g., iodine, cadmium, caesium) generated by the pellet by using a liner (for BWRs) or a graphite coating (for PHWRs) that is less susceptible to the corrosive effects on the inner surface of the cladding. This liner can also smooth down the stress concentration;

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[24] Cladding~~The cladding~~ creepdown and fuel pellet thermal expansion and gaseous swelling will lead to enhanced strain-driven pellet-cladding mechanical interaction during all applicable plant states. ~~The failure~~Failure mode via pellet-cladding mechanical interaction is ~~presented~~ by cladding ductility exhaustion.

(c) Reduce the availability of corrosive fission products at the pellet-cladding interface by using additive fuels which are able to better retain the corrosive fission gas products within the fuel matrix; and

(d) Reduce local power peaking factors (and thus local linear heat generation rates changes) through core design techniques.

3.47. The power-ramp failure threshold should be established by means of in-pile power ramp tests, for each type of fuel or cladding. The database should cover the entire burnup range[25]. Fuel performance codes can be used to analyze and interpret the power-ramp database and define a failure threshold, where the evaluation parameter used to define this threshold is usually the maximum cladding stress but strain energy density can also be used. These same fuel performance codes can be used to assess risk factors that cause this type of stress corrosion cracking of fuel elements in the reactor core and to define adequate guidelines.

Fuel design limits

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

Design limits for operational states

3.49. For normal operation and for anticipated operational occurrences, the design of fuel elements should address at least the following limits:

(a) No melting occurs in any location within the fuel pellets;

(b) No cladding overheating occurs (e.g., by ensuring no departure from nucleate boiling in PWR, critical power ratio below limits in BWRs, no dryout conditions for PHWRs);

(c) Fuel element cladding does not collapse (LWR fuel only);

(d) Rod internal pressure does not increase to the extent that cladding deformations caused by it would negatively affect the heat transfer between fuel pellets and coolant (i.e., fuel pellet-cladding gap reopening by cladding lift-off);

(e) Fuel cladding corrosion and hydriding do not exceed specified allowable limits; and

(f) Cladding stress and strain remain below specified allowable limits.

3.50. Components of fuel elements and~~,~~ fuel assembly ~~and control rods~~ for LWRs should be designed to maintain low deformation and growth so that:

(a) No geometrical interaction between fuel elements and fuel assembly top and bottom nozzles occurs[26] (for LWRs). No geometrical interaction between the fuel bundle string and the shield plugs occurs (for PHWRs);

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[25] It is recognized that the power-ramp failure threshold is the lowest within a burnup range called the "critical burnup range". ~~As expected, for~~ For fuel burnup values below this "critical burnup range", the pellet-cladding gap remains open, so that the power change has to be larger to reach the same level of stress in the cladding as compared to a closed gap condition. For fuel burnup values above this "critical burnup range", experience shows that the pellet cladding interfacial material compound generated by irradiation is such that the stress concentration on the cladding inner surface is reduced, making stress corrosion cracking in the cladding unlikely. Since the "critical burnup range" depends on pellet-cladding gap closure kinetics, it is dependent upon the specific material properties of the cladding type.

[26] To avoid fuel elements and fuel assemblies bow.

(b) No abnormal local power peaking occurs in the fuel elements;
(c) No degradation of the critical heat flux performance of the fuel assembly occurs;
(d) Reactor scram or other movement of control rods is not impeded; and
(e) Handling of fuel assemblies is not hampered.

3.51. To maintain low probability of fuel cladding failure occurrences caused by pellet-cladding mechanical interaction, possibly assisted by stress corrosion cracking, appropriate operational limits on power changes and power ramp rates of change should be determined such that the power-ramp failure thresholds are not exceeded.

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

3.53. Discharge burnup limits, which depend on the fuel elements and fuel assembly performance and on the fuel management approach, should be assessed and justified accordingly.

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

3.54. For design basis accidents and design extension conditions without significant fuel degradation, fuel element design should be such that:

(a) The number of fuel element failures does not exceed a certain percentage of the total number of fuel elements in the reactor core, which limits the radiological consequences of each accident under consideration to within the onsite and offsite release requirements;
(b) In determining the total number of fuel element failures, all known potential failure mechanisms are accounted for. Chemical reactions including oxidation and hydriding, cladding ballooning or collapse of the cladding, or damage to the cladding caused by an increase in the fuel enthalpy are some of the failure mechanisms to be considered;
(c) Limits employed in assessing the risk of loss of cladding integrity are based on experimental studies. In determining the limits, chemical, physical, hydraulic and mechanical factors affecting the failure mechanisms as well as the dimensional tolerances of the fuel elements, are comprehensively and conservatively evaluated. When fuel failure mechanisms and fuel failure limits are burnup dependent, irradiation effects on cladding and fuel properties should be considered in the experiment and incorporated in the analyses to ensure the application of the experimental results is comprehensive; and
(d) Fuel failures under reactivity initiated accidents are considered to occur if the radial average enthalpy or enthalpy rise of a fuel element at any axial location, calculated with validated tools, exceeds a certain value to be determined based on representative experimental tests results by appropriately adjusting test conditions to represent in-reactor conditions (test parameters to be accounted for include: coolant temperature, coolant pressure, coolant flow rate, reactivity insertion kinetics, rod internal pressure, etc.). Since cladding mechanical resistance changes with irradiation and may vary from one cladding type to another, the reactivity initiated accident failure limit is expected to be dependent on the cladding material.

3.55. Core coolability should not be endangered due to, for example,

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

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

3.56. To ensure that the structural integrity of the fuel elements is preserved, the following fuel design limits should be determined and justified:

(a) Peak cladding temperature during ~~the accidental transient~~accident conditions should not exceed a level where ~~the oxidation~~ cladding oxidation ~~of the cladding in consequence of a metal-water reaction~~causes excessive cladding embrittlement or accelerates uncontrollably.~~ (e.g., 1,204 °C for loss-of-coolant accidents or other justified value for more rapid transients than loss-of-coolant accidents).~~ In addition, for LWRs, effects of fuel fragmentation and~~,~~ relocation ~~and dispersal~~ on peak cladding temperature should be assessed as appropriate. Possible effects of fuel particles dispersal on dose consequences and core coolability should also be addressed;
(b) The cladding high temperature oxidation during ~~an accidental transient~~accident conditions should not be to such a degree that it cannot withstand accident induced loadings (e.g., loss-of-coolant accident quenching). The assessment should take into account both pre-transient in-reactor cladding oxidation and transient oxidation (outer side oxidation and possibly inner-side oxidation), as well as chemical interactions between fuel pellets and cladding material. The hydrogen absorbed by the cladding during normal operation and during ~~the accidental transient~~accident conditons should not cause the cladding mechanical properties to deteriorate excessively. The effect of the absorbed hydrogen on cladding resistance and ductility should be determined experimentally;
(c) The allowable enthalpy or enthalpy rise for reactivity initiated accidents should be limited to ~~a~~ values~~,~~ ~~taking into account~~which account for ~~the~~ initial fuel element conditions (e.g., pre-transient hydrogen content of the cladding, fuel burnup, etc.); and ~~(e.g., for reactivity initiated accident transients); and~~
(d) Fuel elements should withstand loadings resulting from post-transient fuel assembly handling, interim storage, transport and long-term storage.

3.57. For LWRs, the amount of hydrogen generated by the chemical reaction between the coolant and the cladding during a loss-of-coolant accident should not exceed a fraction, e.g., 1%, of the amount of hydrogen that would be generated assuming all the claddings surrounding the fuel pellets, excluding the cladding surrounding the plenum volume, in the whole reactor core reacted with the coolant.

3.58. Dispersal of molten fuel particles in case of fuel failure during a reactivity initiated accident transient should be prevented. For this purpose, the radial average enthalpy at any axial location of any fuel element should not exceed a certain value based on, e.g., the analysis of ~~deduced from~~ a prototypical database.

3.59. Structural deformations of fuel elements, fuel assemblies, control rods or reactor internals should remain limited to avoid any impairment of control rods movements in the reactor. In addition, melting temperatures should not be exceeded in control rods at any time or in any location.

CORE STRUCTURES AND COMPONENTS MECHANICAL DESIGN

Design considerations

3.60. The reactor core structures and components should be designed to maintain their structural integrity for all applicable plant states, under various damage mechanisms caused by, for example: vibration (mechanical or flow induced) and fatigue; debris; thermal, chemical, hydraulic and irradiation effects (including radiation induced growth); and seismic motions. Of particular concern are: damage to shutdown and reactivity holddown systems; and damage to the reactor coolant pressure boundary. The effects of high pressures, high temperatures, temperature variations and the temperature distribution, corrosion, radiation absorption rates and the lifetime radiation exposure on physical dimensions, mechanical loads and material properties should be addressed.

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

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

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

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

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

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

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

Design limits for core structure and components

3.68. The design should meet limits specified in the applicable codes and standards that are selected according to safety classification in paras 2.13-2.15.

REACTOR CORE CONTROL, SHUTDOWN AND MONITORING SYSTEMS

Reactor core control system

3.69. This section describes important considerations in the core design to accommodate the control system for maintaining the shapes, levels and stability of the neutron flux within specified limits in all applicable plant states, in order to fulfill Requirement 45 of Ref. [1]~~[1]~~. To achieve the objectives for reactor core control, as stated in para. 6.4 of Ref. [1]~~[1]~~, adequate means of detecting the neutron flux distributions in the reactor core and their changes are required for the purpose of ensuring that there are no regions of the core in which specified limits could be exceeded.

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

Reactivity control devices

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

(a) Normal power manoeuvers;
(b) Changes in xenon concentrations;
(c) Effects relating to temperature coefficients;
(d) ~~The~~ R~~r~~ate of flow of coolant or changes in coolant or moderator temperature;
(e) ~~The~~ D~~d~~epletion of fuel and of burnable ~~poison~~absorber; and
(f) Cumulative ~~poisoning~~ neutron absorbing by fission products.

3.72. Reactivity control devices should be used to maintain the reactor in a subcritical condition, with consideration given to design basis accidents and their consequences[27]. ~~Adequate~~ ~~P~~Pprovision should be ~~made~~ included in the design to maintain subcriticality for plant states in which normal ~~-~~shutdown, ~~-~~fuel cooling or the integrity of the primary cooling system is temporarily disabled[28]~~, for example when the reactor vessel is open for maintenance or refuelling (in PWRs).~~.

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

~~(a) PWR~~

- ~~Use of solid neutron absorber rods;~~
- ~~Use of soluble absorber in the moderator or coolant;~~
- ~~Use of fuel with distributed or discrete burnable poison; and~~
- ~~Use of a batch refuelling and loading pattern.~~

~~(b) BWR~~

- ~~Use of solid neutron absorber blades;~~
- ~~Control of the coolant flow (moderator density);~~
- ~~Use of fuel with distributed or discrete burnable poison; and~~
- ~~Use of a batch refuelling and loading pattern.~~

~~(c) PHWR~~

- ~~Use of solid neutron absorber rods;~~
- ~~Use of soluble absorber in the moderator;~~
- ~~Control of the moderator temperature;~~
- ~~Control of the moderator height (for older pressure tube type PHWRs);~~
- ~~Use of liquid absorber in tubes; and~~
- ~~Use of on-power refuelling.~~

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[27] The types of reactivity control devices used for regulating the core reactivity and the power distribution for different reactor designs include the following:

(a) PWR

- Use of solid neutron absorber rods;
- Use of soluble absorber in the moderator or coolant;
- Use of fuel with distributed or discrete burnable absorber; and
- Use of a batch refuelling and loading pattern.

(b) BWR

- Use of solid neutron absorber blades;
- Control of the coolant flow (moderator density);
- Use of fuel with distributed or discrete burnable absorber; and
- Use of a batch refuelling and loading pattern.

(c) PHWR

- Use of solid neutron absorber rods;
- Use of soluble absorber in the moderator;
- Control of the moderator temperature;
- Control of the moderator height (for older pressure tube type PHWRs);
- Use of liquid absorber in tubes; and
- Use of on-power refuelling.

[28] An example includes the situation when the reactor vessel is open for maintenance or refueling in LWRs.

The design of the solid reactivity control devices should address the irradiation and environmental conditions (e.g., coolant chemistry, irradiation effects; static and dynamic mechanical loads including flow induced vibration; and changes in the chemical characteristics of the constituent materials). The design should assure that these items can withstand handling loads during refueling operations, transport and storage. Important items that are typically considered for the design of the reactivity control devices are described in Annex I.

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

3.74. The maximum degree of positive reactivity and its rate of increase in all applicable plant states are required to be either limited and/or compensated to prevent any resultant failure of the pressure boundary of the reactor coolant systems, to maintain the capability for cooling and to prevent any significant damage to the reactor core (para. 6.6 of Ref. [1]~~[1]~~).

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

3.76. Reactivity control systems using a soluble absorber should be designed to prevent any unanticipated decrease in the concentration of absorber in the core, which could cause specified fuel limits to be exceeded. Those parts of systems that contain soluble absorbers such as boric acid should be designed to prevent precipitation (e.g. by heating of the components) (see Ref. [12]~~[13]~~). The concentrations of the soluble absorber in all storage tanks should be monitored. Whenever enriched B-10 is used, appropriate monitoring should be provided.

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

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

3.79. The design ~~–~~of reactivity control devices, as stated in para. 6.5 of Ref. [1]~~[1]~~, is required to account for wear out and for the effects of irradiation, such as burnup, changes in physical properties and production of ~~fission~~ gases.
In particular, the following environmental effects should be addressed in the design of control systems:

(a) Irradiation effects – Depletion of the absorber material ~~(e.g., boron) and~~ or swelling and heating of materials due to neutron and gamma absorption. Control rods should be replaced or exchanged accordingly;

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

Reactor shutdown system

3.80. This section describes important considerations for systems designed to transition the reactor to a subcritical state from operational states and design extension conditions without significant fuel degradation, and to maintain it in this state, according to Requirement 46 of Ref. [1]~~[1]~~. Together with this safety requirement, Requirement 61 of Ref. [1]~~[1]~~ for the protection system applies to the reactor shutdown system.

3.81. The reactor shutdown system should assure that for all applicable plant states, design limits for shutdown margin (paras 3.18-3.19) are not ~~be~~ exceeded. The necessary reliability should be ensured through the design of the equipment. In particular, t~~T~~he design should ensure ~~the~~ necessary independence ~~from~~ between plant processes, ~~and~~ control and protection systems.

3.82. As stated in para.6.7 of Ref. [1]~~[1]~~, the effectiveness, speed of action and shutdown margin of the means of shutdown of the reactor are required to be such that fuel design limits are met. Guidance on the rate of shutdown is described below in the following three paragraphs.

3.83. The rate of shutdown should be adequate to render the reactor subcritical with a ~~sufficient~~ adequate margin within a specified time so that the specified design limits on fuel and on the reactor system pressure boundary are met.

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

(a) ~~The~~ R~~r~~esponse time of the instrumentation to initiate the shutdown;
(b) ~~The~~ R~~r~~esponse time of the actuation mechanism of the means of shutdown;
(c) ~~The~~ L~~l~~ocation of the shutdown devices for the chosen reactor core designs;
(d) Ease of entry of the shutdown devices into the core. This can be achieved by the use of guide tubes or other structural means to facilitate the insertion of devices including the possible incorporation of flexible couplings to reduce rigidity over the length of the devices; and
(e) ~~The~~ I~~i~~nsertion speed of the shutdown devices. One or more of the following can be used to deliver the necessary insertion speed:
- Gravity drop of shutdown rods into the core,
- Hydraulic or pneumatic pressure drive of shutdown rods into the core, and
- Hydraulic or pneumatic pressure injection of soluble neutron absorber.

3.85. Means of checking the insertion speed of shutdown devices should be provided. The insertion time should be checked regularly, for example, at beginning of each cycle and possibly during the cycle if the margin to the limit is not sufficient.

3.86. As stated in para. 6.8 of Ref. [1]~~[1]~~, in judging the adequacy of the means of shutdown of the reactor, consideration is required to be given to failures arising anywhere in the plant that could render part of the means of shutdown inoperative (such as failure of a control rod to insert) or that could result in a common cause failure. Generally, the characterization of the failure of a control rod to insert should assume that the most reactive core conditions arise when the shutdown device that has the highest reactivity worth cannot be inserted into the core, i.e., the assumption that one shutdown device is stuck.

Different means of shutdown

3.87. As indicated in paras 6.9-6.10 of Ref. [1]~~[1]~~, at least two independent and diverse shutdown systems are required to be provided, and at least one of two different shutdown systems is required to be capable, on its own, of maintaining the reactor in a subcritical state with an adequate margin and with high reliability even for the most reactive conditions of the reactor core for anticipated core coolant temperatures.

TABLE 1~~TABLE 1~~ provides typical examples that illustrate the diversity of means of shutdown that may be used.

TABLE 1. MEANS OF SHUTDOWN

Reactor type	Fast shutdown system	Diverse shutdown system
BWR	B_4C in steel tubes/Hafnium plates (or a hybrid design)	Boron solution injected into moderator/coolant
PWR	Ag–In–Cd in steel tubes/B_4C in steel tubes, Hafnium rods	Boron solution injected into moderator/coolant
PHWR	Hafnium in zirconium alloy guide tubes	Gadolinium solution injected into low pressure moderator (Note 1)

Note 1: This shutdown system can also act as another fast shutdown system.

Reliability

3.88. The design should include the following measures to achieve a high reliability of shutdown individually or in combination as appropriate:

(a) Adopting systems with uncomplicated design and simple operation;
(b) Selecting equipment of proven design;
(c) Using a fail-safe design as far as practicable[29];

[29] The simplest common form of design for fail-safe shutdown allows the shutdown devices to be held above the core by active means. Provided that the guide structures for the shutdown devices are not obstructed, the

(d) Giving consideration to the possible modes of failure and adopting redundancy in the activation of shutdown systems (e.g., sensors). Provision for diversity may be made, for example, by using two different and independent physical trip parameters for each accident as far as practicable;
(e) Functionally isolating and physically separating the shutdown systems (this includes the separation of control and shutdown functions) as far as practicable, on the assumption of credible modes of failure and including common cause failure;
(f) Ensuring easy entry of the means of shutdown into the core, with consideration of the in-core environmental effects of operational states and accident conditions within the design basis;
(g) Designing to facilitate maintenance, in-service inspection and operational testability;
(h) Providing means for performing comprehensive testing during commissioning and periodic refuelling or maintenance outages;
(i) Testing of the actuation mechanism (or of partial rod insertion, if feasible) during operation; and
(j) Designing to function under extreme conditions (e.g., seismic).

3.89. The design of shutdown systems, as stated in para. 6.5 of Ref. [1]~~[1]~~, is required to account for wear out of the control rod cladding and for the effects of irradiation, such as burnup, changes in physical properties and production of helium gases. The bullet items in para. 3.79 are also applicable to the design of shutdown systems. Specific recommendations for diverse shutdown systems injecting ~~poisons~~ neutron absorbers to the reactor coolant system are described in Ref. [12]~~[13]~~.

Effectiveness of shutdown and reactivity holddown

3.90. As indicated in para. 6.11 of Ref. [1]~~[1]~~, the means of shutdown is required to be adequate to prevent any foreseeable increase in reactivity leading to unintentional criticality during the shutdown, or during refuelling operations or other routine or non-routine operations in the shutdown state. The long term reactivity holddown requirements, deliberate actions that increase reactivity in the shutdown state (e.g., the movement of absorbers for maintenance purposes, the dilution of the boron content and refuelling actions) should be identified and evaluated to ensure that the most reactive condition is addressed in the criticality analysis.

3.91. The design should determine the number and the reactivity worth of shutdown rods by considering various factors. Important factors to be accounted for include:

(a) ~~The~~ C~~c~~ore size;
(b) ~~The~~ F~~f~~uel type and the core loading scheme;
(c) ~~The~~ N~~n~~ecessary margin of subcriticality;
(d) ~~The~~ A~~a~~ssumptions related to failure of shutdown device(s);
(e) ~~The~~ U~~u~~ncertainties associated with the calculations;
(f) Shutdown device shadowing[30]; and

devices will drop into the core under gravity in the event of a de-energization of the active means of holding them (e.g. a loss of current through a holding electromagnet). This does not apply to BWRs.

[30] The overall reactivity worth of the shutdown devices is a function of the spacing between the devices, as well as of their locations in the reactor. When two devices are close together, their worth is less than the sum of their individual worths.

(g) The most reactive core conditions after shutdown. These are the result of a number of parameters such as:
- The most reactive core configuration (and where appropriate the corresponding boron concentration) that will occur during the intended fuel cycle, including during refuelling;
- The most reactive credible combination of fuel and moderator temperatures;
- ~~The~~ A~~a~~mount of positive reactivity insertion resulting in DBA conditions;
- ~~The~~ A~~a~~mount of xenon as a function of time after shutdown; and
- ~~The~~ B~~b~~urnup of the absorber.

3.92. The effectiveness of shutdown and reactivity holddown should be demonstrated via:

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

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

3.93. If the operation of the reactivity holddown is manual or partly manual, the necessary prerequisites for manual operation should be met (see Ref. [13]~~[14]~~).

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

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

Separation of protection systems from control systems

3.96. As stated in Requirement 64 of Ref. [1]~~[1]~~, protection systems are required to be physically and functionally separated from control systems to avoid failures of control systems causing failures in the protection system. Guidance on separation of the protection system from other systems is described in Ref. [13]~~[14]~~.

Partial trip system

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

Operational limits and setpoints

Operational limits for control system

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

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

3.99. Determination of the operational limits and setpoints should include effects of the ageing of the reactor coolant system (e.g., steam generator tube plugging in PWRs, increase of the diameter of the pressure tube in PHWRs).

Setpoints for reactor core protection

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

3.101. Equipment performance, operational limits~~,~~ and procedures should be defined ~~designed~~ to ~~restrict disallowed~~prevent excessive rod worths or reactivity insertion rates. ~~and their~~ Their ~~performance~~ capability should be demonstrated. Where feasible, an alarm should be installed to function when any such limit or restriction is violated or is about to be violated.

3.102. The design limits, uncertainties, operational limits, instrument requirements, and setpoints should be translated into the technical specifications for the facility.

Core monitoring system

3.103. ~~As indicated in Requirement 59 of Ref. [1], the provision of instrumentation systems is required to apply core monitoring in full to the nuclear core protection and control systems. The core design should accommodate the detectors and devices for adequate monitoring of changes in core reactivity in order to enable any required modification of core parameters, while ensuring that they are maintained within defined operating ranges. The rapidity of the variation in a parameter should determine whether the actuation of the reactor control systems is automatic or manual.~~ As established by Requirement 59 of Ref. [1], core monitoring instrumentation is required to support reactor protection and control systems, as well as to supply sufficiently detailed and timely information on the local heat generation conditions prevailing in the core. The core design should accommodate the detectors and devices for monitoring the magnitude and changes of core power, as well as the local distribution of heat generation in the core, in order to enable any required modification of core parameters within their defined operating ranges. The rapidity of the variation in a parameter should determine whether the actuation of the reactor control systems is automatic or manual.

In addition, the radionuclide activity levels in the coolant should be monitored to assess the integrity of the fuel system during operation and to verify that design limits are not exceeded. ~~For PHWRs that employ on-power refuelling, for example, a failed fuel detection and location system can enable the defective fuel assembly to be removed more easily, thereby keeping the radionuclide activity levels in the coolant low.~~

3.104. The core monitoring parameters such as the following examples to be measured should be adequately selected, which will depend on the reactor type. The following are examples of parameters to be measured for the purposes of core monitoring:

(a) ~~Neutron flux~~Spatial distribution of the neutron flux and related power distribution peaking factors;
(b) Coolant temperature;
(c) Reactor c~~C~~oolant flow rate;
(d) Water level (for a BWR);
(e) System pressure;
(f) Radionuclide activity in the coolant[31];
(g) Control rod insertion position; and
(h) Concentration of soluble boron and B-10 content (for a PWR).

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

(a) ~~The ~~N~~n~~eutron flux doubling time;~~,~~
(b) ~~The ~~R~~r~~ate of change of neutron flux;~~,~~
(c) Axial and radial neutron flux imbalances~~,~~;
(d) ~~The ~~R~~r~~eactivity balance;~~,~~ and
(e) Thermal-hydraulic core parameters (e.g., the departure from nucleate boiling ratio or the critical power ratio).

3.105. The accuracy, speed of response, range and reliability of all monitoring systems should be adequate for performing their intended functions (see Ref. [13]~~[14]~~). The monitoring system design should provide for the continuous or adequate periodic testing of these systems.

3.106. Guidance on post-accident monitoring is provided in Ref. [13]~~[14]~~. If core monitoring is needed in accident conditions, for example, to monitor system temperatures, reactor vessel water level, or reactivity changes, the instrumentation to be used should be qualified to withstand the environmental conditions to be expected following the accident.

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

3.108. A computerized core monitoring system ~~may~~ should be used to ensure that the status of the core is within the operational limits assumed in the safety analysis. Qualification of the system should be ensured wherever it is coupled to a protection system (see Ref. [13]~~[14]~~).

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[31] Reactor coolant activity is measured by the device belonging to the primary coolant makeup and water cleaning system; for details see Ref. [12].

3.109. During reactor shutdown, a minimum set of instruments or combination of instruments and neutron sources should be available to monitor neutron flux and heat generation distribution ~~reactivity~~ (e.g., using ~~neutron~~ flux detectors with an adequate sensitivity) whenever fuel assemblies are present in the reactor vessel, including fuel loading and start-up phases. At least one means of shutdown should be available to assure core subcriticality under cold conditions.

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

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

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

CORE MANAGEMENT

Design considerations

3.112. The primary objective of core management should be to ensure the safe, reliable and optimum use of the nuclear fuel in the reactor, while remaining within design limits imposed by the design of fuel elements, fuel assemblies, thermal-hydraulics and neutronics.

3.113. ~~A f~~Fuel cycles should be developed with ~~appropriate levels of enrichment and~~ appropriate means of controlling the core reactivity and the power distribution ~~so as to extract energy from the fuel in an economic and reliable manner within~~to address fuel design limits.

While the details of core management depend on the reactor type, in all cases the core management program should provide:

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

Core Design

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

(a) ~~The~~ Loading patterns (including enrichment and configuration of fuel elements) and orientation of fuel assemblies in each fuel cycle (for LWRs);
(b) ~~The~~ S~~s~~chedule for the subsequent unloading and loading of fuel assemblies;
(c) ~~The~~ C~~c~~onfigurations of reactivity control and shutdown devices; and
~~(d) The fuel assemblies to be shuffled; and~~
~~(e)~~(d) Burnable ~~poisons~~ absorbers and other core components to be removed, inserted or adjusted.

3.115. Core-reload depletions and reactor physics parameters are provided as input to safety analyses, plant monitoring and protection systems, and operator guidance; therefore, these parameters should be analyzed based on pre-determined plant operational objectives and resultant plans. These reactor physics parameters include: reactor start-up conditions (e.g., critical boron and rod position), reactor kinetics, fuel temperature coefficients, moderator temperature coefficients, control rod and bank worths, and power peaking factors.

Unplanned power maneuvering during flexible operation may alter the power and burnup profile across the core relative to those predictions. As such, core-reload depletions and reactor physics parameters predictions should be continuously or periodically examined and evaluated, using relevant monitoring parameters.

3.116. The design of t~~T~~he reactor core design should include analyses to demonstrate that the fuel management strategy and the established limitations on operation do not change in any manner that would cause nuclear design limits to be violated throughout the reactor operating cycle or lifetime.

3.117. In reactor core analyses, multi-dimensional and multi-scale physics codes and system thermal-hydraulic codes are preferentially used for realistic analysis of the reactor core for all applicable plant states. Uncertainties should be incorporated in the analyses.

3.118. The reactor core analysis should be performed based on typical cases covering the entire operating cycle for the following reactor core conditions such as:

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

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

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

Refuelling

3.120. For on-power refuelling in PHWRs, the effects of the refuelling operation on the neutronic behavior of the core should be demonstrated to remain within the control capability of the reactor control systems.

3.120a.Safety assessment should address any event that may cause inadvertent criticality during core loading or unloading and during handling phases.

3.121. The fuel loading pattern should be validated through the use of in-core flux distribution measurements.

Core management design limits

3.122. The reactor core ~~design~~ analysies should verify that the core fuel loading pattern will meet nuclear and fuel design limits for all applicable plant states.

3.123. For practical reasons and simplicity, for LWRs, a system that develops and monitors the nuclear key safety parameters can be used to verify the suitability of the reload core design.

Special core configurations

Mixed Core

3.124. When fuel assemblies of different types are loaded into the core (a so-called mixed core), the fuel assembly types in the mixed core should be assessed in such a manner to demonstrate that the mixed core meets nuclear design limits for both the initial and subsequent reload mixed cores, and that the fuel assemblies in the mixed core meet fuel design limits for all applicable plant states. These assessments include: dimensional, mechanical and thermal-hydraulic response~~s~~ of the fuel~~s~~ types (e.g. in terms of the pressure drop characteristics through the fuel assembly(s)), compatibility with the nuclear characteristics of the original core and with related safety analysis.

3.125. Relevant nuclear parameters such as reactivity, reactivity coefficients, control rod worth and power distributions should be evaluated for the different fuel assembly designs. The compatibility evaluation may be developed based on single fuel assembly calculations in infinite medium.

Mixed-Oxide Fuel Core

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

(a) The mixed-oxide fuel properties[32] are somewhat different from the UO_2 fuel and this should be incorporated in ~~analyses tools~~computer codes and models used for the fuel design and safety analyses;

(b) In the mixed-oxide core, control rod and absorber worth~~s~~ are reduced as a result of neutron spectrum hardening due to the higher thermal absorption cross sections of plutonium compared with uranium, and as a result, the reactor shut-down margin can be reduced. To compensate for the reduced shutdown margin, additional control rods or absorption capability of the ~~poison~~ absorbing materials (e.g., B-10 enrichment increase) should be implemented;

(c) The kinetic parameters for mixed-oxide fuel, namely, the total fraction of delayed neutrons and the prompt neutron lifetime are ~~slightly~~ lower than those for UO_2 fuel. The lower delayed neutron fraction of mixed-oxide fuel can result in a prompt critical reactor condition with a smaller reactivity insertion; thus, there is less time for control rod insertion or boron system injection to provide reactor control. This should be addressed in the core design and safety analyses for all applicable plant states (e.g., reactivity initiated accident transients); and

(d) The fission cross sections in mixed-oxide fuel are larger than those in UO_2 fuel, and this can result in steep flux gradients between adjacent mixed-oxide and UO_2 fuel elements. This effect can be reduced with ~~enrichment~~ variations of the plutonium content and core design pattern adjustments. Another consequence of the differences in cross sections between plutonium and uranium are the changes in the moderator temperature coefficient, the fuel temperature coefficient, and the coefficient of reactivity for coolant voids. Core design and safety analyses should evaluate the effects of these changes in reactivity coefficients.~~;~~

Load following and power maneuvering

3.127. The effects of flexible operating conditions such as load following[33], power cycling, reactor startup, and refuelling manoeuvers should, whenever necessary, be superimposed onto the power level distributions and temperature histories to evaluate the potential effects of thermal cycling on fuel element thermal mechanical responses such as the buildup of pressure due to fission gas release to the pellet-cladding gap and fuel cladding fatigue.

3.128. Once the extent of the desired operational flexibility is determined, in-depth evaluation of impacts on the nuclear power plants design and operation (i.e. requirements on the safety

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[32] Isotopic composition and Pu content in mixed-oxide fuel depend strongly on the discharge burnup of spent fuel assemblies from which plutonium has been extracted. The ratio of fissile isotopes for the plutonium also varies; this will affect the characteristics of the reactor core. In addition, the Pu vector (Pu-238, Pu-239, Pu-240, Pu-241 and Am-241) should be incorporated in the mixed-oxide core design, recognizing that there are changes affecting reactivity and key neutronics parameters as a function of the start-up time after mixed-oxide fuel fabrication. These features should be taken into account in core design and safety analyses.

[33] Load following means that the reactor will be operated so that electricity generation will match a varying electrical demand. Load following implies operation with power manoeuvers at levels less than the rated thermal power, so the total amount of electrical energy output is less than if the unit operates at a relatively constant base load. Load following operation may require increased maintenance and monitoring and may complicate the reliability and aging assessments of some structures, systems and components.

analysis and the operational limits and conditions) should be performed. Based on this evaluation, additional specifications for qualification and implementation can be developed.

3.129. To assure the control of core reactivity with load following and power maneuvering, the core and generator power balance and the ~~-~~reactor stability should be maintained.

3.130. ~~During load following operation, power density redistributions are caused promptly by control rod movement, but then enable inherent subsequent redistribution processes via feedback effects linked with the reactor coolant conditions and the xenon distributions. This generates power density distribution changes characterized by higher peak power densities (and/or lower departure from nucleate boiling ratios) compared with initial unperturbed conditions. Therefore, the~~ The operational limits should be adjusted to cover ~~amplitude of such~~ perturbations[34] due to load following operation. ~~should be limited, and preferably conducted at lower power levels in order to avoid any violation of the operational limits and conditions which could occur at higher power levels.~~

Reactor operation with leaking fuel elements

3.131. Fuel failures can affect ease of access, work scheduling and worker dose for plant operations personnel. Reactor core with failed fuel elements should stay within the radiochemical technical specifications[35] as defined by the limit on coolant radionuclide activity included in the Technical Specifications document. The core design and operations program should also establish procedures and limits for operating the core with defective fuel assemblies while assuring ~~without giving plant personnel significant~~ radiation dose limits are met~~s~~ for plant personnel.

In BWRs and PHWRs[36], fission product release from failed fuel and subsequent secondary ~~oxidation~~ hydriding of the cladding can be minimized by reducing the power level[37] of failed rods. ~~In BWRs, it is often possible to locate the region or regions in the core that contain~~

[34] During load following operation, power density redistributions are caused promptly by control rod movement, but then enable inherent subsequent redistribution processes via feedback effects linked with the reactor coolant conditions and the xenon distributions. This generates power density distribution changes characterized by higher peak power densities (and/or lower departure from nucleate boiling ratios) compared with initial unperturbed conditions.

[35] The iodine spiking phenomenon after plant transients has received particular attention in safety evaluations. For particular pre-accident conditions, its occurrence may increase the radiological consequences of the postulated accident. One approach is to specify a limit to the amount of iodine activity allowed in the reactor coolant after plant transients. Behaviour of leaking fuel elements during design basis accidents, e.g., loss-of-coolant accidents, reactivity initiated accidents and steam generator tube rupture, may be specific and should be assessed. Loss-of-coolant accident margins are not affected by the presence of leaking fuel because conservative assumptions are specified as requirements for radiological consequence evaluation. Reactivity initiated accident design limits should not be affected in case of the presence of leaking fuel, although it is recognized that leaking fuel has lower capability in withstanding ~~a reactivity~~reactivity initiated accident and consequently a higher probability to cause fuel coolant interaction.

[36] In BWRs, it is often possible to locate the region or regions in the core that contain failed fuel by using the flux tilting method. Once those regions have been identified, it is possible to reduce the power of fuel assemblies in those regions through selective placement of control blades. In PHWRs, defective fuel assemblies can be detected and located by means of tracing fission products elements and delayed neutrons. With a reduced operating power level, the reactor operation with defective fuel core may be feasible without significant iodine spikes until defective fuel assemblies are discharged from the reactor.

[37] The mitigation effectiveness provided by power suppression is greatest when applied to relatively small, tight defects. For this reason, detection and suppression should be undertaken as early as possible once the presence of a leaking fuel element is indicated in a core.

~~failed fuel by using the flux tilting method. Once those regions have been identified, it is possible to reduce the power of fuel assemblies in those regions through selective placement of control blades. In PHWRs, defective fuel assemblies can be detected and located by means of tracing fission products elements and delayed neutrons. With a reduced operating power level, the reactor operation with defective fuel core may be continued operated without significant iodine spikes until defective fuel assemblies are discharged from the reactor.~~

Core re-design after fuel assembly repair

3.132. In LWRs, the fuel assemblies containing damaged and leaking fuel elements may be repaired and reconstituted with replacement rods, ~~solid~~ dummy rods or vacancies. The use of ~~filler rods or~~ vacancies should be limited ~~(e.g., to one or a few per assembly and to one or a few assemblies in the core)~~ so that design limits are met.

3.133. The impact of reconstituted fuel assembly on ~~the~~ the design of reactor core ~~design analyses~~ should be assessed.

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

3.134. Design limits are determined, based on the concept of defence in depth, to fulfil safety requirements for all applicable plant states. Fuel design limits described in paras 3.49-3.59 should be extended to assure that the fuel elements and fuel assemblies remain intact in the back-end phases after the fuel assemblies are discharged from the core: handling, shipment, storage, reprocessing and disposal. The following key in-reactor safety parameters are among those that may have an impact on the post irradiation behavior~~s~~ of the fuel elements and the fuel assemblies:

(a) End-of-life fuel element internal pressure.

Even though ~~The rod~~ fuel elements ~~internal pressure~~ can withstand some extent of over-pressurization exceeding the normal coolant pressure without failure in normal operation, such highly pressurized used fuel elements should not be acceptable to handle when coolant counter-pressure is diminished (e.g., in spent fuel storage facilities).~~.~~ ~~However, the pressure build up in the fuel elements should be limited to avoid fuel handling accidents.~~ This is particularly relevant for mixed-oxide fuels which remain hot for a longer period of time and continue to release helium gases.

(b) Massive cladding hydriding and cladding mechanical properties.

Localized hydriding (e.g., due to corrosion layer spalling or due to axial pellet-pellet gaps), may not impact normal operation or be of consequence in ~~accidental transients~~accident conditions, but such a condition may lead to delayed hydrided cladding cracking in post-irradiation handling or storage, or undesired failures in the event of a shipment accident.

(c) Grid-to-~~rod~~ fuel element fretting wear.

Localized wear is usually undetected unless it wears out the complete ~~reaches 100% of the~~ cladding wall thickness and creates a leakage in the fuel cladding. Some fuel elements affected by excessive wear may exhibit localized weakness that may lead to long term creep failures or undesired failures in the event of shipment accidents.

(d) Discharge burnup.

Fuel design, core management and the resultant discharge burnup affect the fuel isotopic vector degradation[38], which in turn will impact the economy of fuel reprocessing.

(e) Others.

For new fuel element or new fuel assembly designs proposed by the fuel vendors to address in-reactor issues (e.g., stress corrosion cracking of fuel cladding~~-~~, fission gas release ~~retention~~ and fuel assembly distortion) should remain compatible with industrial requirements for reprocessing.

4. QUALIFICATION AND TESTING

GENERAL

4.1. Safe operation of the reactor core design throughout the lifetime of the structures, systems and components of the reactor core, including the fuel elements and assemblies, core components, and control systems requires a robust program for qualification, inspection, and testing of the equipment design and analysis process. This can be achieved as described below.

DESIGN QUALIFICATION

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

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

4.4. Methods of qualification should ~~be adequate, which may~~ include:

(a) ~~The~~ P~~p~~erformance of a type test on structures, system and components representative of that to be supplied;
(b) ~~The~~ P~~p~~erformance of a test on the structures, system and components supplied;
(c) ~~The~~ U~~u~~se of pertinent past experience;
(d) Analysis based on available and applicable test data; and
(e) Any combination of the above methods.

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[38] High discharge burnups degrade spent fuel isotopic compositions and therefore its energetic quality. As a result Pu content in mixed-oxide fuel has to be increased to maintain parity with UO_2 enrichment.

4.5. Design qualification may be established through operating experience with fuel systems of the same or similar design. The ~~bases~~ basis for the previous experience should be identified and the performance record should be evaluated. The maximum burnup and core power operating experience should be referenced and the fuel assembly performance should be compared against design criteria identified for phenomena such as fretting wear, oxidation, hydriding, and crud buildup.

INSPECTION

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

TESTING INCLUDING PROTOTYPE AND LEAD USE ASSEMBLIES

4.7. Provisions should be made in the design for in-service testing and inspection to ensure that the core and associated structures and the reactivity control and shutdown systems will perform their intended functions throughout their lifetime. Further guidance on in-service inspections is provided in Ref. [14]~~[15]~~.

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

LWRs

(a) Spacer grid structural tests;
(b) Control rod structural and performance tests;
(c) Fuel assembly structural tests (lateral, axial and torsional stiffness, frequency, and damping) ; and
(d) Fuel assembly hydraulic flow tests, including lift force determination, control rod vibration and wear, assembly vibration, fuel element fretting (accounting for spacer grid spring relaxation), and assembly wear and lifetime evaluations.

PHWRs

(a) Fuel bundle string pressure drop tests;
(b) Cross-flow endurance tests;
(c) Mechanical endurance tests;
(d) Bundle impact tests;
(e) Bundle strength tests;
(f) Wear tests;
(g) Seismic qualification tests;
(h) Wash-in and wash-out tests (where applicable); and
(i) Critical heat flux tests.

4.9. In-reactor testing of design features through irradiations in materials test reactors or through lead-use assembly irradiation should be used to ~~determine the maximum burnup or fluence experience that should be demonstrated with such tests to~~ justify the specified maximum burnup or fluence limit for a new design. The following phenomena may be tested in this manner:

(a) Fuel and burnable ~~poison~~ absorber rod growth;
(b) Fuel element bowing;
(c) Fuel element, spacer grid, and fuel channel ~~box~~ (~~for BWRs~~if present) oxidation and hydride levels;
(d) Fuel element fretting, and spacer (for PHWRs) fretting
(e) Fuel assembly growth;
(f) Fuel assembly bowing;
(g) ~~Channel box~~Fuel channel (for BWRs) wear and distortion;
(h) Fuel element ridging, i.e., pellet-cladding interaction;
(i) Crud formation;
(j) Fuel element integrity;
(k) Holddown spring relaxation (for ~~LWRs~~PWRs);
(l) Spacer grid spring relaxation (for LWRs); and
(m) Guide tube wear characteristics (for ~~LWRs~~PWRs).

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

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ANNEX I.

ITEMS TO BE TAKEN INTO ACCOUNT FOR THE DESIGN OF FUEL ELEMENTS, FUEL ASSEMBLIES AND THE REACTIVITY CONTROL ASSEMBLIES, NEUTRON SOURCE ASSENBLIES AND HYDRALUC PLUG ASSENBLIES

FUEL ELEMENT

I-1. During the design of fuel elements, the following items need to be taken into account:

Cladding

(a) Fuel element vibration and wear (grid-to-rod fretting wear);
(b) Cladding mechanical properties evolution with irradiation (displacement and pressure driven loadings);
(c) Materials and chemical evaluation;
(d) Stress corrosion;
(e) Cycling and fatigue; and
~~(f) Element bowing; and~~
(f~~g~~) Geometrical and chemical stability of the cladding under irradiation.

Fuel material (including burnable absorbers)

(a) Dimensional stability of the fuel under irradiation conditions;
(b) Fuel densification (kinetics and amplitude);
(c) Potential for chemical interaction with the cladding and the coolant;
(d) Fission gas generation and distribution within the fuel pellets;
(e) Fission gas release kinetics;
(f) Gaseous swelling;
(g) Thermal mechanical properties under irradiation; and
(h) Microstructure changes as a function of irradiation.

Fuel element performance

(a) Pellet and cladding temperatures and temperatures distributions;
(b) Fuel-clad gap closure kinetics and amplitude (to address pellet-cladding interaction issue);
(c) Irradiation effects on fuel element behaviour (fuel restructuration, fuel pellets cracking patterns, solid and gaseous fission product swelling, fission gas release and rod internal pressure increases, fuel element thermal conductivity degradation, etc.); and
(d) Fuel element bowing; and
(e~~d~~) Fuel element growth.

Fuel element performance is demonstrated using validated analytical models and/or representative experimental data collected either in test programs or from commercial power plants (lead tests fuel elements or lead test fuel assemblies). The models are usually burnup dependent.

FUEL ASSEMBLY

I-2. Fuel assembly components (i.e., top and bottom nozzles, guides tubes, spacers, mixing grids, grid springs, connections and fuel assembly holddown system) are designed to withstand the following conditions and loadings:

(a) Core restraint system loadings;
(b) Hydrodynamic loadings;
(c) Accident loadings (e.g., seismic, loss-of-coolant accident); ~~and~~
(d) Handling and shipping loadings; ~~.~~ and
(e) Fuel assembly bow.

REACTIVITY CONTROL ASSEMBLY

I-3. During the design of the rod cluster control assembly, the following items need to be taken into account:

(a) Internal pressure and related cladding stresses during normal, transient, and accident conditions;
(b) Thermal expansion and irradiation induced swelling;
(c) Evolution under irradiation of absorber materials and the cladding; and
(d) Fretting wear effect on cladding resistance.

NEUTRON SOURCE ASSEMBLY

I-4. During the design of the neutron source assembly, the following items need to be taken into account:

(a) Irradiation effects;
(b) Efficiency to account for burnup shadowing effects of peripheral fuel assemblies; and
(c) External events such as earthquakes.

~~(f) not finished~~HYDRAULIC PLUG ASSEMBLY

I-5. During the design of the hydraulic plug assembly, the following items need to be taken into account:

(a) Interaction with guide tubes due to thermal expansion or irradiation induced swelling;
(b) Impact on coolant by-pass flow (for PWRs); and
(c) Fretting wear effect on guide tube resistance.

~~ABBREBIATIONS~~

ABBREVIATIONS

BWR	Boiling water reactor.
LWR	Light water reactor.
PHWR	Pressurized heavy water reactor.
PWR	Pressurized water reactor.

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