IAEA Safety Standards

for protecting people and the environment

Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors

Specific Safety Guide

No. SSG-22

IAEA

International Atomic Energy Agency

IAEA SAFETY STANDARDS AND RELATED PUBLICATIONS

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USE OF A GRADED APPROACH IN THE APPLICATION OF THE SAFETY REQUIREMENTS FOR RESEARCH REACTORS

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

IAEA SAFETY STANDARDS SERIES No. SSG-22

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SPECIFIC SAFETY GUIDE

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FOREWORD

by Yukiya Amano
Director General

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

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

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

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

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

NOTE BY THE SECRETARIAT

The IAEA safety standards reflect an international consensus on what constitutes a high level of safety for protecting people and the environment from harmful effects of ionizing radiation. The process of developing, reviewing and establishing the IAEA standards involves the IAEA Secretariat and all Member States, many of which are represented on the four IAEA safety standards committees and the IAEA Commission on Safety Standards.

The IAEA standards, as a key element of the global safety regime, are kept under regular review by the Secretariat, the safety standards committees and the Commission on Safety Standards. The Secretariat gathers information on experience in the application of the IAEA standards and information gained from the follow-up of events for the purpose of ensuring that the standards continue to meet users' needs. The present publication reflects feedback and experience accumulated until 2010 and it has been subject to the rigorous review process for standards.

Lessons that may be learned from studying the accident at the Fukushima Daiichi nuclear power plant in Japan following the disastrous earthquake and tsunami of 11 March 2011 will be reflected in this IAEA safety standard as revised and issued in the future.

THE IAEA SAFETY STANDARDS

BACKGROUND

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

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

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

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

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

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

THE IAEA SAFETY STANDARDS

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

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

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

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

Safety Fundamentals

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

Safety Requirements

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

Safety Guides

Safety Guides provide recommendations and guidance on how to comply with the safety requirements, indicating an international consensus that it is necessary to take the measures recommended (or equivalent alternative measures). The Safety

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

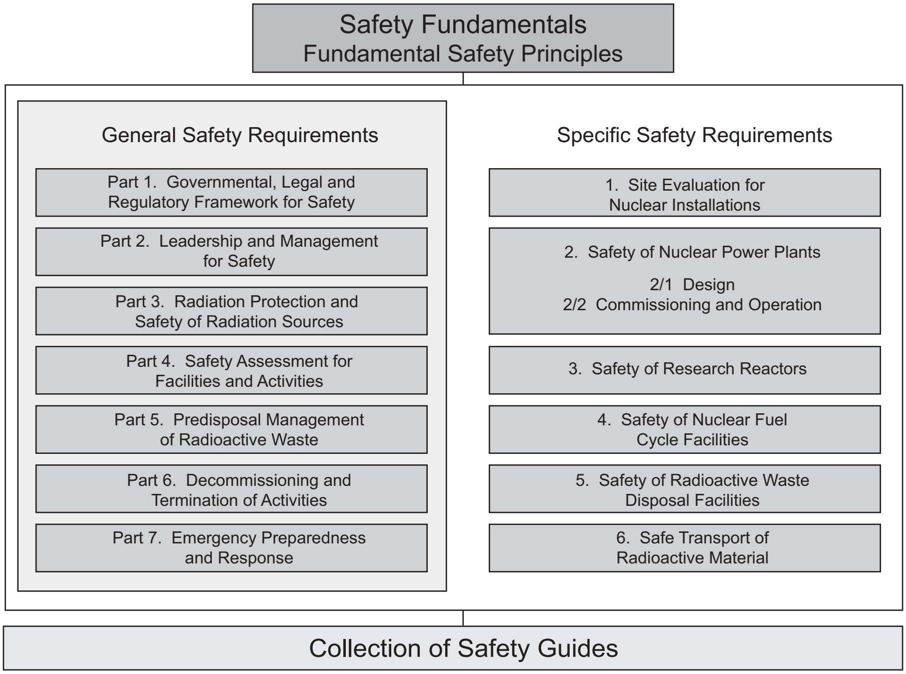

FIG. 1. The long term structure of the IAEA Safety Standards Series.

Guides present international good practices, and increasingly they reflect best practices, to help users striving to achieve high levels of safety. The recommendations provided in Safety Guides are expressed as ‘should’ statements.

APPLICATION OF THE IAEA SAFETY STANDARDS

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

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

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

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

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

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

DEVELOPMENT PROCESS FOR THE IAEA SAFETY STANDARDS

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

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

A management system has been established for the processes of planning, developing, reviewing, revising and establishing the IAEA safety standards. It articulates the mandate of the IAEA, the vision for the future application of the

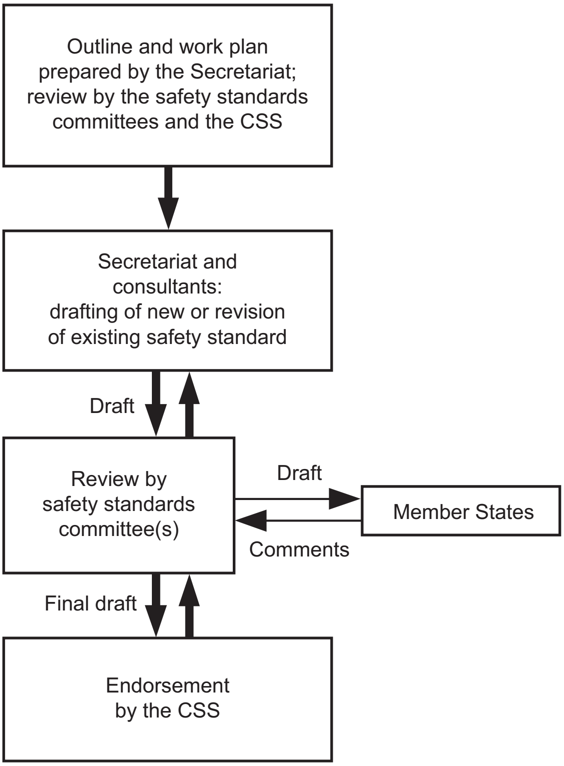

FIG. 2. The process for developing a new safety standard or revising an existing standard.

safety standards, policies and strategies, and corresponding functions and responsibilities.

INTERACTION WITH OTHER INTERNATIONAL ORGANIZATIONS

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

INTERPRETATION OF THE TEXT

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

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

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

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

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide presents recommendations on the graded approach to application of the safety requirements for research reactors established in the Safety Requirements publication on Safety of Research Reactors[1] [1].

1.2. Research reactors in Member States employ a variety of designs. Operating power levels vary significantly, ranging from a few watts to over a hundred megawatts in a few cases. The inventory of radioactive material may also have a broad range, including not only the radioactive material of the core inventory, but also radioactive material contained in stored spent fuel elements, radioactive waste from radioisotope production and various types of active experimental facility. Utilization of research reactors covers a wide range of activities such as: core physics experiments, training, target material irradiation for materials science, transmutation studies, commercial production of radioisotopes, neutron activation analysis, experiments involving high pressure and temperature loops for fuel and material testing, cold and hot neutron sources, neutron scattering research, and neutron and gamma radiography. These uses call for a variety of different design features and operational regimes. Therefore, site evaluation, design and operating characteristics of research reactors vary significantly.

1.3. Owing to the wide range of utilization activities, the safety requirements for research reactors may not be required to be applied to every research reactor in the same way. For example, the way in which requirements are demonstrated to be met for a multipurpose, high power level research reactor might be very different from the way in which the requirements are demonstrated to be met for a research reactor with very low power and very low associated radiological hazard to facility staff, the public and the environment. Reference [1], which applies to a wide range of research reactors, includes information on the application of the safety requirements in accordance with a graded approach (see paras 1.11–1.14 of Ref. [1]).

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[1] "A research reactor is a nuclear reactor used mainly for the generation and utilization of the neutron flux and ionizing radiation for research and other purposes" (footnote 4 of Ref. [1]). In the context of this Safety Guide, the term 'research reactor' also includes associated experimental devices and critical assemblies, but excludes reactors used for the production of electricity, naval propulsion, desalination or district heating (see para. 1.7 and footnote 4 of Ref. [1]).

1.4. The general definition and purpose of the graded approach is set out in Ref. [2]. Both parts of the definition are applicable to the safety requirements of Ref. [1]:

(i) "For a system of control, such as a regulatory system or a safety system, a process or method in which the stringency of the control measures and conditions to be applied is commensurate, to the extent practicable, with the likelihood and possible consequences of, and the level of risk associated with, a loss of control" [2].
(ii) "An application of safety requirements that is commensurate with the characteristics of the practice or source and with the magnitude and likelihood of the exposures" [2].

1.5. The graded approach in general is a structured method by means of which the stringency of application of requirements is varied in accordance with the circumstances and the regulatory and management systems used. For example, a method in which:

(i) "The significance and complexity of a product or service are determined" [2];
(ii) "The potential impacts of a product or service on health, safety, security, the environment, and the achieving of quality and the organization's objectives are determined" [2];
(iii) "The consequences if a product fails or if a service is carried out incorrectly are taken into account" [2].

1.6. Guidance has been provided in the past on grading the application of safety requirements. There are a number of historical publications, which are now superseded, relating to grading[2] and several current IAEA safety standards refer to a graded approach:

— Principle 3 of the Fundamental Safety Principles states that "Safety has to be assessed for all facilities and activities, consistent with a graded approach" (para. 3.15 of Ref. [3]).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Functions and Component Classification for BWR, PWR and PTR, IAEA Safety Series No. 50-SG-D1, IAEA, Vienna (1979); INTERNATIONAL ATOMIC ENERGY AGENCY, Grading of Quality Assurance Requirements, Technical Reports Series No. 328, IAEA, Vienna (1991).

— Principle 5 of the Fundamental Safety Principles states that "The resources devoted to safety by the licensee, and the scope...have to be commensurate with the magnitude of the radiation risks" (para. 3.24 of Ref. [3]).
— Requirement 1 of Ref. [4] states that:

"The government shall establish a national policy and strategy for safety, the implementation of which shall be subject to a graded approach in accordance with national circumstances and with the radiation risks associated with facilities and activities."

— Paragraphs 2.6 and 2.7 of Ref. [5] establish requirements for grading the application of management system requirements, and paras 2.37–2.44, 5.6 and 6.68 of Ref. [6] provide related recommendations.
— Requirement 1 of Ref. [7] states that:

"A graded approach shall be used in determining the scope and level of detail of the safety assessment carried out in a particular State for any particular facility or activity, consistent with the magnitude of the possible radiation risks arising from the facility or activity."

— Paragraph 3.10 of Ref. [8] states that "in implementing the inspection programme, the regulatory body should establish a graded approach in responding to unforeseen circumstances."[3]

OBJECTIVE

1.7. The objective of this Safety Guide is to provide support for the application of the safety requirements for research reactors throughout the various stages of the lifetime of a research reactor (site selection and site evaluation, design, construction, commissioning, operation and decommissioning). The relevant safety requirements are established in Ref. [1], and also in Refs [4, 5, 7]. This Safety Guide is intended for use by regulatory bodies, operating organizations and other organizations involved in the design, construction and operation of research reactors.

[3] In some Member States, a graded approach is referred to as 'proportionality'.

SCOPE

1.8. This Safety Guide presents recommendations on applying a graded approach without compromising safety.

1.9. The application of a graded approach to all of the important activities[4] throughout the lifetime of a research reactor is discussed. These activities are identified in sections 3–8 of Ref. [1]. A major aspect of the design activity, as described in Section 6 of this Safety Guide, involves the grading of specific requirements for design of structures, systems and components (SSCs) for particular reactor types, so that the safety objectives set out in para. 2.2 of Ref. [1] are achieved. Recommendations on the application of grading to reactor hardware and equipment (SSCs), as opposed to activities in general, are also provided in Section 6.

1.10. In this Safety Guide, it is considered that all relevant safety requirements have to be complied with in applications of a graded approach. The graded approach should be used to determine the appropriate manner to comply with a requirement; it is not used to provide relief from meeting the requirement. To eliminate a requirement for the purposes of identifying all relevant safety requirements, a waiving[5] process, as suggested in para 1.10 of Ref. [1], can be used.

STRUCTURE

1.11. Section 2 provides the description of the basic elements of a graded approach and its application. The remaining sections provide recommendations on the application of a graded approach to regulatory supervision (Section 3); management and verification of safety (Section 4); site evaluation (Section 5); design (Section 6); operation (Section 7); and decommissioning (Section 8). Sections 3–8 have titles identical to the corresponding sections of Ref. [1].

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[4] 'Activities', in the context of this Safety Guide, include all of the stages needed to achieve the purpose for which the nuclear research reactor was designed and constructed or modified (see footnote 2 of Ref. [1]). Reference [5] uses a more general definition of activities that encompasses "any…practices or circumstances in which people may be exposed to radiation".

[5] 'Waiving' is sometimes called 'grading to zero', implying complete elimination of a requirement.

1.12. Each section of this Safety Guide begins with a brief description of the relevant safety requirements established in Ref. [1] and, in some areas, a summary of additional requirements established in other IAEA Safety Requirements publications. The descriptions are followed by a discussion of grading in the application of the requirements.

2. BASIC ELEMENTS OF THE APPROACH TO GRADING

GENERAL CONSIDERATIONS REGARDING THE CONCEPT OF GRADING

2.1. A graded approach is applicable in all stages of the lifetime of a research reactor (see para. 1.7).

2.2. During the lifetime of a research reactor, any grading that is performed should be such that safety functions and operational limits and conditions are preserved, and such that there are no undue radiological hazards to workers, the public or the environment.

2.3. The grading of activities should be based on safety analyses, regulatory requirements and engineering judgement. Engineering judgement implies that account is taken of the safety functions of SSCs and the consequences of failure to perform these functions, and implies that the judgement is documented. Other elements to be considered in grading are the complexity and the maturity of the technology, operating experience associated with the activities and the stage in the lifetime of the facility.

DESCRIPTION OF THE APPLICATION OF A GRADED APPROACH

2.4. This Safety Guide does not recommend the use of a quantitative ranking procedure in grading the safety requirements. The application of the graded approach will determine the appropriate effort to be expended and appropriate manner of complying with a requirement, in accordance with the attributes of the facility.

2.5. The application of grading presented in this Safety Guide begins with categorization of the facility in accordance with its potential hazard (Step 1). In this step, a facility can initially be categorized into a range from facilities posing the highest risk to those posing the lowest risk. This categorization serves to provide an initial grading of the facility. The next step (Step 2) is analysis and grading of activities and/or SSCs important to safety. This second step provides more detailed grading to be applied to the particular characteristics of the facility.

Step 1: Categorization of the facility in accordance with potential hazards

2.6. Qualitative categorization of the facility should be performed on the basis of the potential radiological hazard, using a multi-category system similar to that set out in para. 1.11 of Ref. [9]:

(a) Facilities with off-site radiological hazard potential;
(b) Facilities with on-site radiological hazard potential only;
(c) Facilities with no radiological hazard potential beyond the research reactor hall and associated beam tubes or connected experimental facility areas.

2.7. The individual characteristics, or attributes, to be considered in deriving the category of the facility in accordance with its hazard are typically as follows (see para 1.14 of Ref. [1]):

(a) The reactor power (for pulsed reactors, energy deposition is typically used, while for accelerator driven subcritical systems, thermal power is typically used);
(b) The radiological source term;
(c) The amount and enrichment of fissile material and fissionable material;
(d) Spent fuel storage areas, high pressure systems, heating systems and the storage of flammables, which may affect the safety of the reactor;
(e) The type of fuel and its chemical composition;
(f) The type and mass of moderator, reflector and coolant;
(g) The amount of reactivity that can be introduced and its rate of introduction, reactivity control, and inherent and engineered safety features;
(h) The quality of the containment structure or other means of confinement;
(i) The utilization of the reactor (experimental devices, tests, radioisotope production, reactor physics experiments);
(j) The location of the site, including the potential for external hazards (including those due to the proximity of other nuclear facilities) and the characteristics of airborne and liquid releases of radioactive material;

(k) Proximity to population groups and the feasibility of implementing emergency plans.

Step 2: Analysis and grading

2.8. In this step, the level of detail at which requirements are applied to activities and/or SSCs is determined, in accordance with the importance to safety of the activity or SSC. The level of detail covers, for example, the rigour of the analysis to be conducted, the frequency of activities such as testing and preventive maintenance, the stringency of required approvals and the degree of oversight of activities.

2.9. The appropriateness of applying a graded approach should be determined through analysis for each of the major activities and SSCs set out in sections 3–8 of Ref. [1]. The application of grading should be commensurate with the importance to safety of the activities and SSCs, and with the magnitude of the associated radiological risks.

2.10. The safety functions[6] performed by each item important to safety[7] should be identified (see para. 2.11(b) of Ref. [10]). A starting point for assessing the importance to safety of activities and SSCs is conducting a safety assessment[8].

2.11. Paragraphs 6.12 and 6.13 of Ref. [1] state that all SSCs (including software for instrumentation and control) that are important to safety are required first to be identified and then to be classified according to their function and significance for safety. The classification of SSCs, including software, in a research reactor

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[6] See annex I of Ref. [1].

[7] An item important to safety is an "item that is part of a safety group and/or whose malfunction or failure could lead to radiation exposure of the site personnel or members of the public…Items important to safety include:
- — Those structures, systems and components whose malfunction or failure could lead to undue radiation exposure of site personnel or members of the public;
- — Those structures, systems and components that prevent anticipated operational occurrences from leading to accident conditions;
- — Those features that are provided to mitigate the consequences of malfunction or failure of structures, systems and components" [2].

[8] Guidance on this subject is provided in Ref. [11].

facility should be based on the safety function(s)[9] performed by the SSC and on the consequences of the SSC's failure to perform its function. Analytical techniques together with engineering judgement (see para. 2.3) should be used to evaluate these consequences. The basis for the safety classification of SSCs, including software, should be stated and the engineering design rules applied should be commensurate with their safety class.

2.12. "The application of management system requirements shall be graded so as to deploy appropriate resources, on the basis of the consideration of:

- The significance and complexity of each product or activity;
- The hazards and the magnitude of the potential impact (risks) associated with the safety, health, environmental, security, quality and economic elements of each product or activity;
- The possible consequences if a product fails or an activity is carried out incorrectly" (para. 2.6 of Ref. [5]).

2.13. "Grading of the application of management system requirements shall be applied to the products and activities of each process" (para. 2.7 of Ref. [5]). Where these activities involve modifications or experiments, further categorization should be carried out (see para. 7.49 of this Safety Guide).

3. REGULATORY SUPERVISION

3.1. The requirements for the legal and regulatory infrastructure for a broad range of facilities and activities are established in Ref. [4]. Additional guidance is provided in the associated Safety Guides (Refs [12–15]). Owing to the broad applicability of the requirements and recommendations in these publications, not all will apply to the nuclear activities in all States. In each State, the requirements and recommendations that are applicable for the regulatory supervision of its nuclear programme should be identified. For the purpose of this Safety Guide, the

[9] The safety functions are essential, characteristic functions associated with SSCs for ensuring the safety of the reactor and are one of the key elements in grading the application of requirements to SSCs. Some safety functions may not be relevant for some types of research reactors.

applicable safety requirements are those for the regulatory supervision of research reactors that are established in section 3 of Ref. [1], and include the following:

— Legal infrastructure;
— The regulatory body;
— The licensing process;
— The programme for inspection and enforcement.

APPLICATION OF GRADING TO LEGAL INFRASTRUCTURE

3.2. The requirements for the legal infrastructure are established in para. 3.2 of Ref. [1], and in Requirement 3 on establishment of a regulatory body and Requirement 4 on independence of the regulatory body in Ref. [4]. The application of these requirements cannot be graded.

APPLICATION OF GRADING TO THE REGULATORY BODY

3.3. A graded approach should be applied in establishing the regulatory body and determining aspects of its organizational framework, based on the potential hazards of all of the facilities and activities under its supervision or oversight.

3.4. The regulatory body is required to be provided with sufficient authority and power, and a sufficient number of experienced staff and financial resources to discharge its assigned responsibilities (para 2.8 of Ref. [4]). The responsibilities of the regulatory body include establishing regulations, review and assessment of safety related information (e.g. from the safety analysis report), issuing licences, performing compliance inspections, taking enforcement actions and providing information to other competent authorities and the public. External experts, technical support organizations or advisory committees may assist the regulatory body in these activities[10].

3.5. Examples of the regulatory organization, associated activities and requirements that can be graded are: requirements for staffing, resources for in-house technical support, compliance inspections, the content and detail of

[10] The IAEA provides safety review services to governments of Member States, regulatory bodies and operating organizations.

licences, regulations and guides, and the detail required of the licensee for submissions of documentation on safety of the facility, including the safety analysis report.

APPLICATION OF GRADING TO THE LICENSING PROCESS

3.6. The licensing process is often performed in steps for the various stages of the lifetime of a research reactor, as described in paras 3.4 and 3.5 of Ref. [1], and in Ref. [15]. For a research reactor, these stages are site approval, authorization of construction[11], authorization of commissioning, authorization of initial and routine operation, and all proposed modifications, and authorization of decommissioning.

3.7. At each of these stages, regulatory evaluations are usually made and authorizations or approvals are issued. In some cases, the stages may be combined, depending on the nature of the facility and relevant laws and regulations. This practice is consistent with the concept of the graded approach.

3.8. The licensing process should be used by the regulatory body to exercise control during all stages of the lifetime of the research reactor [15]. This control is accomplished by means of the following:

- Clearly defined lines of authority for authorizations to proceed;
- Review and assessment of all safety relevant documents, particularly the safety analysis report;
- Issue of permits and licences, for the various stages;
- Hold points for inspections, review and assessment;
- Review, assessment and approval of operational limits and conditions;
- Commissioning authorization;
- Operating licence;
- Licensing of operating personnel;
- Decommissioning licence.

3.9. The steps in the licensing process apply to all research reactors, including all proposed experiments and design modifications, at all stages of the reactor lifetime. However, at each step in the licensing process, a graded approach should

[11] In some States, design and manufacturing activities are included in the licensing process.

be taken by the regulatory body in determining the scope, extent, level of detail and effort that should be used, depending on the magnitude of the potential risks (see paras 2.19(h) and 2.46–2.50 of Ref. [15]). For example, in general, there will be fewer inspections and hold points for a research reactor with a power level less than 100 kW, compared to those for a research reactor with a power level greater than 5 MW.

3.10. Detailed recommendations on the application of a graded approach to the regulatory review of the safety assessment in support of the licence for decommissioning of a research reactor are provided in paras 5.6–5.8 of Ref. [16].

APPLICATION OF GRADING TO INSPECTION AND ENFORCEMENT

3.11. The requirements for inspection and enforcement are established in paras 3.14–3.16 of Ref. [1]. For inspections, Ref. [1] states that "The regulatory body shall establish a planned and systematic inspection programme. The scope of this programme and frequency of inspections shall be commensurate with the potential hazard posed by the research reactor" and particular situations such as organizational changes or personnel turnover. Reference [8] recommends in para. 3.14 that "Inspections by the regulatory body should be concentrated on areas of safety significance" and in para 3.10 that "the regulatory body should establish a graded approach in responding to unforeseen circumstances."

3.12. Enforcement actions should also be graded, since the severity and impact on safety of non-compliance with the requirements of an authorization may vary. Regulatory bodies should allocate resources and apply enforcement actions or methods in a manner commensurate with the seriousness of the non-compliance, increasing them as necessary to bring about compliance with requirements. A graded approach should be applied with respect to the corrective action process for non-conformances, to ensure that problems of the highest significance are afforded the most critical evaluation (see para. 6.68 of Ref. [6]).

3.13. Some of the factors to consider in the grading of enforcement actions are:

— The safety significance or seriousness of the deficiency or violation;
— The need for timeliness of corrective actions to restore compliance;
— The frequency of this or other violations, or the degree of recidivism;

— Who identified and reported the non-compliance, i.e. whether the non-compliance was reported by an operator or identified by an inspector;
— The need for consistency and transparency in the treatment of operators and licences;
— The complexity of the remedial, corrective or preventive action needed.

3.14. In contrast to the factors in para 3.13, enforcement actions for violation of a regulatory requirement should not be graded. This is necessary to hold regulatory compliance in the highest regard.

4. MANAGEMENT AND VERIFICATION OF SAFETY

4.1. Section 4 of Ref. [1] establishes the requirements on the elements of the management and verification of safety to be considered, the responsibilities of the operating organization and the interaction with the regulatory body. Requirements are also established in Ref. [5], and recommendations and guidance for the management system and verification of safety are also provided in Refs [6, 17, 18]. For management of safety, at a minimum:

> "the operating organization shall:
>
> (a) Establish and implement safety policies and ensure that safety matters are given the highest priority;
> (b) Clearly define responsibilities and accountabilities with corresponding lines of authority and communication;
> (c) Ensure that it has sufficient staff with appropriate education and training at all levels;
> (d) Develop and strictly adhere to sound procedures for all activities that may affect safety, ensuring that managers and supervisors promote and support good safety practices while correcting poor safety practices;
> (e) Review, monitor and audit all safety related matters on a regular basis, implementing appropriate corrective actions where necessary;
> (f) Be committed to safety culture on the basis of a statement of safety policy and safety objectives which is prepared and disseminated and is understood by all staff." (para. 4.1 of Ref. [1]).

4.2. The management system should provide for a process for assessment and verification of safety, including periodic safety review at an interval specified by the regulatory body. The basis for the assessment should include, inter alia, data derived from the safety analysis report and other information (e.g. the operational limits and conditions, the radiation protection programme, the emergency plan, operating procedures and training documentation).

4.3. Such assessments should include consideration of modifications to SSCs and their cumulative effects. Additional safety related aspects that should be included in the assessment are changes to procedures, radiation protection measures, regulations and standards; ageing effects; operating experience; lessons learned from similar reactors; technological developments; site re-evaluation; physical protection; and emergency planning. Requirements on assessment and verification of safety for research reactors are established in paras 4.14–4.16 (for general purpose and scope) and in paras 7.108–7.110 (for operational issues) of Ref. [1].

APPLICATION OF GRADING TO THE MANAGEMENT OF SAFETY

4.4. Grading of the scope and content of activities making up the elements of management of safety[12], such as items (a)–(f) in para. 4.1, is possible while still meeting the requirement that they be comprehensive. For example, in item (c) of para. 4.1, grading is clearly essential in specifying the staffing levels required for operations and maintenance. Requirements for staff education and training should be based on the operating schedule and the complexity of the facility. The latter is determined, in particular, on the basis of the research reactor power level, the extent of isotope production and the scope of experimental facilities. In addition, grading can be applied to determine the depth, frequency and type of safety assessments, in-service inspections and auditing of all safety related matters.

[12] In Ref. [1], the term 'quality assurance' was used. References [5, 6] were issued later, and adopted the term 'management system' instead. The term 'management system' reflects and includes the initial concept of 'quality control' (controlling the quality of products) and its evolution through 'quality assurance' (the system to ensure the quality of products) and 'quality management' (the system to manage quality). The management system is a set of interrelated or interacting elements that establishes policies and objectives, and which enables those objectives to be achieved in a safe, efficient and effective manner.

4.5. The complexity of the management system for a particular research reactor and associated experimental facilities should be commensurate with the potential hazard of the reactor and the experimental facilities, and the requirements of the regulatory body. Requirements for the preparation and implementation of a graded management system are established in paras 2.6 and 2.7 of Ref. [5], which state that grading of the application of management system requirements is required to be applied to the products and activities of each process and that the grading is required to be such as to deploy appropriate resources, on the basis of consideration of:

— The safety significance and complexity of each activity;
— The hazards and the magnitude of the potential impact (risks) associated with the safety, health, environmental, security, quality and economic elements of each activity;
— The possible consequences if an activity is carried out incorrectly.

4.6. In general, application of the management system requirements should be most stringent for items, services or processes with the highest grade; for the lowest grade, application of the management system requirements may be the least stringent. The following are examples of detailed elements of the management system where grading can be applied (see para. 2.41 of Ref. [17]):

— Type and content of training;
— Level of detail and degree of review and approval of instructions and procedures;
— Level of detail of testing, surveillance, maintenance and inspection activities;
— Extent of operational safety reviews and reporting, analysis and corrective action for non-conformances and system and equipment failures;
— The type and frequency of safety assessments.

4.7. Paragraphs 2.37–2.44 of Ref. [6] also discuss the need for grading of management system controls. A detailed example of the application of grading to the document and records management system is reproduced from Ref. [6] in para. 7.46 of this Safety Guide.

APPLICATION OF GRADING TO THE VERIFICATION OF SAFETY

4.8. Grading can be applied to the frequency and scope of self-assessments[13] and peer reviews. The frequency and scope of safety assessments and peer reviews should be graded on the basis of the complexity and potential risk of the facility, whether the activity or SSC has a safety function and the safety significance of the activity or SSC being assessed.

4.9. Grading can be applied to the number, size, composition and frequency of meetings of reactor advisory groups or safety committees. The safety committee is required to advise the operating organization of relevant aspects of the safety of the reactor and the safety of its utilization, and on the safety assessment of design, commissioning and relevant operational issues and modifications. A safety committee should also advise the reactor manager. This is discussed in para. 4.15 of Ref. [1]. To facilitate safety committee assessment, the operating organization should, as part of good practice, prepare an annual review report on the safety performance of the reactor facility and hold scheduled safety review meetings at suitable intervals. It is acceptable to have one safety committee advising both the operating organization and the reactor manager. The safety committee is required to be independent of the reactor management.

5. SITE EVALUATION

5.1. "The main safety objective in evaluating the site for a research reactor is the protection of the public and the environment against the radiological consequences of normal and accidental releases of radioactive material" (para. 5.1 of Ref. [1]). Accordingly, it is necessary to assess those characteristics of the site that may affect the safety of the research reactor, to determine whether there are deficiencies in the site and if they can be mitigated by appropriate design features, site protection measures and administrative procedures. For a

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[13] Self-assessments are frequently performed as part of routine activities. For example, during periodic maintenance of safety related SSCs, an evaluation is made of the performance of the SSCs and an assessment can be made concerning the lifetime and continued availability of the SSCs; during other activities (e.g. retraining), an assessment can be made of the continued competence of the staff on the basis of results of re-qualification examinations.

graded approach, the scope and depth of site evaluation studies and evaluations should be commensurate with the facility's radiological risk. The scope and detail of the site investigation may also be reduced if the operating organization proposes to adopt conservative parameters for design purposes, which may be a preferred approach for research reactors. For example, a conservative assumption for the design of a particular SSC that is readily accommodated in the overall design may permit simplification of the site evaluation.

APPLICATION OF GRADING TO SITE EVALUATION

5.2. Grading can be applied when assessing the aspects mentioned in para. 5.1. Paragraphs 2.4–2.13 and para. 6.6 of Ref. [19] develop the basis for applying a graded approach to the various site related evaluations and decisions, in accordance with the radiological hazard of the research reactor facility. The main factors to be considered in site evaluation are the influences of:

- Potential external events of natural origin, such as seismic and volcanic events;
- Meteorological and hydrological characteristics of the site that may influence exposure of the public and environmental contamination from facility releases;
- Potential human induced events associated with the site[14];
- Population density and population distribution;
- Other characteristics of the site that could affect important safety requirements, such as the ultimate heat sink capability.

5.3. The site evaluations should be graded, provided that there is an adequate level of conservatism in the design and siting criteria, to compensate for a simplified site hazard analysis and simplified analysis methods.

5.4. Section 10 of Ref. [20] provides recommendations on applying a graded approach with respect to seismic hazard evaluation. The grading can be based upon the complexity of the installation and the potential radiological hazards, including hazards due to other materials present. A seismic hazard assessment following a graded approach should initially apply a conservative screening process in which it is assumed that the entire radioactive inventory of the

[14] These external events may be due to the proximity of other nuclear facilities, local industries or road transport and air traffic routes.

installation is released by an accident initiated by a seismic event. If such a release would not lead to unacceptable consequences for workers, the public or the environment, the installation may be screened out from further seismic hazard assessment. If the results of the conservative screening process show that the potential consequences of such a release would be significant, a seismic hazard evaluation should be performed.

5.5. Section 7 of Ref. [21] provides recommendations similar to those in Ref. [20] for application of a graded approach with respect to volcanic hazards in site evaluation. A volcanic hazard assessment following a graded approach should initially apply a conservative screening process in which it is assumed that the entire radioactive inventory of the installation is released by an accident initiated by a volcanic event. If such a release would not lead to unacceptable consequences for workers, the public or the environment, the installation may be screened out from further volcanic hazard assessment. If the results of the conservative screening process show that the potential consequences of such a release are significant, a more detailed volcanic hazard assessment should be performed, and the grading process outlined in Ref. [21] should then be used to categorize the installation for the purposes of volcanic hazard assessment.

5.6. Recommendations on application of a graded approach with respect to meteorological and hydrological hazards in site evaluation are provided in Ref. [22]. For the purpose of the evaluation of meteorological and hydrological hazards, including flooding, the installation should be graded on the basis of its complexity, the potential radiological hazards and hazards due to other materials present. If the results of a conservative screening process, similar to that described in paras 5.4 and 5.5, show that the consequences of a potential release are significant, a detailed meteorological and hydrological hazard assessment for the installations should be carried out, in accordance with the grading process outlined in Ref. [21].

5.7. Paragraphs 2.14–2.21 of Ref. [19] provide criteria for use in applying a graded approach in the assessment of hazards associated with human induced events; similarly, paras 2.26–2.28 of Ref. [19] provide criteria with regard to population density and population distribution factors; and paras 3.52–3.55 of Ref. [19] establish important requirements with regard to other external events.

6. DESIGN

6.1. Section 6 of Ref. [1] discusses design under the three categories below:

— Philosophy of design: Paragraphs 6.2–6.8 of this Safety Guide provide recommendations on grading the application of requirements relating to the philosophy of design established in paras 6.1–6.11 of Ref. [1].
— General requirements for design: Paragraphs 6.9–6.37 of this Safety Guide provide recommendations on grading the application of the general design requirements established in paras 6.12–6.78 of Ref. [1].
— Specific requirements for design: Paragraphs 6.38–6.71 of this Safety Guide provide recommendations on grading the application of the specific design requirements established in paras 6.79–6.171 of Ref. [1].

APPLICATION OF GRADING TO DESIGN

Philosophy of design

Defence in depth

6.2. Paragraphs 2.6 and 6.6 of Ref. [1] describe the five levels of defence in depth that are aimed at preventing deviations in normal operation, and controlling them if they occur, and at preventing accidents and mitigating their radiological consequences should they occur, as follows:

(a) First level: to prevent deviations from normal operation and to prevent system failures.
(b) Second level: to control (by detection and intervention) deviations from operational states so as to prevent anticipated operational occurrences from escalating to accident conditions.
(c) Third level: to provide engineered safety features or inherent safety features, to prevent an escalation of a design basis accident and to achieve a controlled state and then a safe shutdown state following an initiating event. One barrier for the confinement of radioactive material is required to be maintained.
(d) Fourth level: to address beyond design basis accidents and to ensure that radioactive releases are kept as low as practicable. The objective is the protection of the confinement function.

(e) Fifth level: to mitigate the radiological consequences of potential releases of radioactive material that may result from accident conditions.

6.3. Defence in depth is an important design principle that must be applied to the design of a research reactor of any type or power level.

6.4. Defence in depth is required to be applied with account taken of the graded approach and the fact that many low power research reactors do not qualify for the fourth or fifth level of defence in depth (see para. 2.6 of Ref. [1]). In addition, the defence in depth concept is required to be applied in the design to provide graded protection against various reactor transients, including transients resulting from equipment failure and human error and from internal or external events that could lead to a design basis accident (see para. 6.6 of Ref. [1]).

Safety functions

6.5. Requirements for the design of safety systems are established in para. 6.10 of Ref. [1]:

> "In the design of the safety systems, including engineered safety features, that are used to achieve the three basic safety functions — shutting down the reactor, cooling, in particular the reactor core, and confining radioactive material — the single failure criterion shall be applied, high reliability shall be ensured and provisions shall be included to facilitate regular inspection, testing and maintenance."

6.6. The application of grading to the three basic safety functions is discussed in the following:

(a) Shutting down the reactor:
   (i) In general, the capability to shut down the reactor when necessary cannot be graded, although the size of the subcriticality margin available and the speed of response required of the shutdown system may vary according to the reactor design.
   (ii) Some research reactors may have inherent self-limiting power levels and/or systems that physically limit the amount of positive reactivity that can be inserted into the core. This property can be used for grading the design of the shutdown system.

(b) Cooling, in particular reactor core cooling:
   (i) In general, this basic safety function cannot be graded, although the extent of the requirements on the cooling system will vary according to the reactor design. For example, a forced convection cooling system to remove fission heat may be necessary in one facility, whereas in other facilities, such as critical assemblies, fission heat can be adequately removed by natural convection cooling.
   (ii) Similarly, for decay heat removal following shutdown, the extent of the cooling system requirements will vary according to the reactor design and a forced convection cooling system may be necessary in one facility, whereas in low power facilities decay heat can be adequately removed by natural convection cooling, depending on the reactor design.
   (iii) Some facilities may need an emergency core cooling system to prevent damage to the fuel in the event of a loss of flow or loss of coolant accident; other facilities may not need an emergency core cooling system. The need for an emergency cooling system should be defined in the design stage, and emergency operating procedures should be established, as necessary, taking into consideration the timescale needed for safe removal of the decay heat.
(c) Confining radioactive material:
   (i) The basic safety function of requiring radioactive material to be confined is not gradable. However, the way in which the confinement function is implemented may be graded (see the description of the fourth level of defence in depth in para. 6.2).

Acceptance criteria and design rules

6.7. Basic acceptance criteria[15] are defined by the regulatory body. Specific acceptance criteria can be defined by the designer in advance of the final design and agreed by the regulatory body (see section 4 of Ref. [23]). In principle, such specific acceptance criteria are not graded, as they are fixed by the specific facility characteristics. However, the way that such acceptance criteria are met by the design can be graded, as indicated in the following.

6.8. For the design of SSCs, acceptance criteria may be set in the form of engineering design rules. These rules include regulatory requirements as well as

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[15] Acceptance criteria are specified bounds on the value of a functional indicator or condition indicator used to assess the ability of an SSC to perform its design function [2].

requirements established in relevant codes and standards, which can be graded on a case by case basis. This is discussed in paras 6.9 and 6.10.

General requirements for design

Classification of SSCs

6.9. The requirements for classification of SSCs are established in paras 6.12 and 6.13 of Ref. [1]. The method for determining the safety significance of SSCs should be based on deterministic methods, complemented by probabilistic methods and engineering judgement (see para 2.3).

Codes and standards

6.10. The requirements for codes and standards are established in paras 6.14 and 6.15 of Ref. [1]. Codes and standards have been developed that provide guidance in the design of SSCs. These codes and standards could be regulatory, international (such as the IAEA safety standards), national or even local[16]. They can be highly specialized (e.g. an industrial code for the design of a pump, or a code for the design of a pump in a nuclear application) or can be based on management system procedures and/or performance requirements relating to the application of the component (e.g. an electronic component used in the protection system of a research reactor).

6.11. The codes and standards used in the design of SSCs should be appropriately selected using a graded approach that takes into account the safety classification of SSCs and the potential radiological hazard associated with the research reactor.

Design basis

6.12. The requirements for the design basis are established in paras 6.16–6.34 of Ref. [1]. Challenges that the research reactor might be expected to face during its operational lifetime are required to be taken into consideration in the design. These challenges are represented by a list of selected postulated initiating events, an example of which is included in the appendix of Ref. [1]. Design requirements will be supported by design limit specifications for all relevant parameters for all operational states and design basis accidents. The requirements and limitations

[16] Some States have codes that are applied nationally (national codes), whereas others may have some local codes with jurisdiction limited to particular provinces, cities or towns.

then form the basis of a practicable set of operating limits and conditions for reactor operation.

6.13. The classification of the SSCs, based on their importance to safety, should be utilized to establish the design requirements for withstanding challenges stemming from the postulated initiating events without exceeding authorized limits. Paragraph 6.17 of Ref. [1] states that "It shall be shown that the set of postulated initiating events selected covers all credible accidents that may affect the safety of the research reactor. In particular the [design basis accidents] shall be identified." Application of the requirement to identify the postulated initiating events and the design basis accidents for research reactors cannot be graded. The postulated initiating events and design basis accidents should be identified using current safety standards and feedback of operational experience. However, the extent of the postulated initiating events and the design basis accidents considered for the reactor can be graded.

6.14. Development of the design basis can be graded in the sense that a different set of applicable design basis accidents, based on the specific hazards posed, will be considered in the design basis for each reactor. Higher power research reactors with significant in-core experimental facilities such as loops will require a greater number of higher safety class SSCs.

Design for reliability

6.15. The requirements for design for reliability are established in paras 6.35–6.43 of Ref. [1]. Design for reliability requires application of the principles of redundancy, diversity, independence and fail-safe design. These measures should be applied using a graded approach to ensure the required reliability of SSCs in accordance with the safety function to be performed by each SSC. In the design of a research reactor, the required reliability of SSCs can be related to the expected utilization of the facility, and grading can be employed to achieve operational reliability. Where automatic or passive performance of a safety function is required or an inherent safety feature is used, a minimum requirement for the reliability of the associated SSC should be established and maintained. Depending on the type of the research reactor, performance of one or more of the following safety functions may need to be automatic: reactor shutdown, initiation of emergency core cooling, and isolation of radioactive material by the containment or other means of confinement.

Design for commissioning

6.16. The requirements for design for commissioning are established in para. 6.44 of Ref. [1] which states that "The design shall include design features as necessary to facilitate the commissioning process for the reactor." The design basis of the reactor provides information on the tests and measurements that should be employed in the commissioning process. This information should be used to anticipate difficulties in carrying out the tests and measurements, and to provide for such testing and measurement in the design.

6.17. Grading can be applied in the selection of design features to facilitate commissioning, in accordance with the importance to safety of the system to be commissioned and the associated difficulties of conducting the commissioning tests and measurements.

Provision for inspection, testing and maintenance

6.18. The requirements established in paras 6.45–6.47 of Ref. [1] include a requirement to make provisions to facilitate in-service inspection for determining the conditions of SSCs subject to corrosion, erosion, fatigue or other ageing effects.

6.19. Where the performance of inspection, testing and maintenance takes place in controlled areas, it is required to ensure that occupational doses to workers will be below the authorized limits (para. 7.93 of Ref. [1]). This cannot be graded.

6.20. Grading can be applied to the inventory of spare parts and components on the basis of the ease of procurement of such components from vendors, and budget rules and considerations (see para. 2.44 of Ref. [6]). Controls for the procurement process that can be graded are:

— Expectations of suppliers for assessment, evaluation and qualification;
— Scope and level of detail of the procurement specification;
— Need for and scope of supplier quality plans;
— Extent of inspection, surveillance and audit activities for suppliers;
— Scope of documents to be submitted by the supplier and approved by the organization;
— Records to be provided or stored and preserved.

Most consideration should be given to components of systems important to safety having a high obsolescence rate (such as computerized systems or instrumentation and control systems).

6.21. Grading can be applied in determining the provision for inspection, testing and maintenance in two steps:

(a) First, the types and frequencies of the required inspections, tests and maintenance operations should be determined, with account taken of the importance to safety of the SSC and its required reliability, and all of the effects that may cause progressive deterioration of the SSC.
(b) Second, the provisions to be included in the design to facilitate the performance of these inspections, tests and maintenance operations should be specified, with account taken of the frequency, the radiation protection implications and the complexity of the inspection, test or maintenance operation. These provisions include accessibility, shielding, remote handling and in situ inspection, self-testing circuits in electrical and electronic systems, and provisions for easy decontamination and for non-destructive testing.

Design for emergency planning

6.22. The requirements for design for emergency planning are established in paras 6.48 and 6.49 of Ref. [1].

6.23. These specific design features include alarm systems, communication and public address systems, illuminated escape routes, designated assembly areas, on-site and off-site surveillance systems with provision for remote readout, and other means to facilitate early assessment of the situation and efficient response. While all of these features should be considered in the design, grading can be applied to many of the features. For example:

— The number and type of escape routes should be based on the layout and size of the facility, and the potential hazards in various zones.
— The gathering places should be in the most convenient location while still remaining safe for persons attending.
— On-site and off-site monitoring can be performed by utilizing personnel with portable devices or technology using fixed devices with remote readout.
— The scope and frequency of emergency drills can be graded.

— Safety analysis should be performed to determine the need for a supplementary control room and the degree of automatic and/or manual control necessary can be graded.

Design for decommissioning

6.24. The requirements for design for decommissioning are established in paras 6.50 and 6.51 of Ref. [1]: "attention shall be directed to keeping the radiation exposure of personnel and of the public…as low as reasonably achievable and to ensuring adequate protection of the environment from undue radioactive contamination" arising from decommissioning activities.

6.25. Grading can be applied in the selection of the design features to meet the requirements for radiation protection of workers, the public and the environment. For example:

- — Low power level research reactors with small cores that could be easily removed and packaged may require minimal special provisions for removal and packaging of the core. The need for disposal facilities for high level radioactive waste will, therefore, be minimal.
- — Higher power level, pool type research reactors that allow for easy access and underwater handling of the core components may require design provisions for disassembling the reactor under the water. Radioactive waste disposal facilities will be an important consideration.

Design for radiation protection

6.26. The requirements for design for radiation protection are established in paras 6.52–6.59 of Ref. [1]. The primary objective in the design for radiation protection is:

> "To ensure that in all operational states radiation exposure within the installation or due to any planned release of radioactive material from the installation is kept below prescribed limits and as low as reasonably achievable, and to ensure mitigation of the radiological consequences of any accidents."[17]

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[17] From IAEA Safety Series No. 110, The Safety of Nuclear Installations (1993), and reproduced in para. 2.2 of Ref. [1].

6.27. Grading can be applied in the choice of radiation protection design features used to satisfy the requirements of paras 6.52–6.59 of Ref. [1], including their effective placement in the facility. In general, the scope of radiation protection design provisions for a high power level multipurpose facility will be more extensive and more complex than those for a small research reactor with limited utilization possibilities and low potential for significant exposure (see also para. 6.57 of this Safety Guide).

Human factors and ergonomic considerations

6.28. The requirements for the human factors and ergonomic considerations are established in paras 6.61–6.64 of Ref. [1]. Grading can be applied to human factors and ergonomics by giving particular consideration to the following:

— The design of the control room, reactor systems and experiments;
— The design of control room displays and audible signals for parameters important to safety;
— The need to rely on administrative controls and procedures for safety, in order to achieve flexibility in certain activities;
— Operating procedures;
— Determination of whether responses to system alarms necessitate an operator response or have to be automatic, by considering available time constraints, expected physical and environmental conditions and possible psychological pressure on the operator;
— The need for interlocks and hierarchical access controls (e.g. keys and passwords);
— The determination of minimum staffing levels for reactor operators and facility support personnel that have to be present on the site at particular times, such as when fuel is in the reactor.

Provision for utilization and modification

6.29. The requirements for design for utilization and modification are established in paras 6.65–6.67 of Ref. [1]. Research reactors are flexible in nature and are used for a variety of purposes.

6.30. The main precautions concerning provisions for utilization and modification taken in design are:

- It is required to ensure that the reactor configuration is known at all times and that it is appropriately assessed and authorized.
- New utilization and modification projects, including experiments having an impact on safety, are required to be subject to safety analyses and to procedures for design, construction, commissioning and decommissioning that are equivalent to those used for the research reactor itself.
- Where experimental devices penetrate the reactor vessel or reactor core boundaries, they are required to be designed to preserve the means of confinement and reactor shielding. Protection systems for experiments are required to be designed to protect the experiment and the reactor.

6.31. It is, therefore, necessary that these aspects of utilization are taken into account or analysed at the design stage and that appropriate provisions are made in the design to ensure safety. Such provisions and the design of modifications and experimental facilities should be subject to grading in the same way that grading is applied in the design of other SSCs, i.e. on the basis of their importance to safety, their complexity, their maturity, and the scope of analysis and of commissioning tests necessary to verify their acceptability.

Selection and ageing of materials

6.32. Requirements on the selection and ageing of materials are established in paras 6.68–6.70 of Ref. [1]. Ageing management in the design involves the use of proven durable materials with sufficient design margins and provisions for testing, inspection and replacement. The extent to which these measures are utilized in the design can be graded, on the basis of the safety significance of the SSCs and their ease of replacement.[18]

6.33. In applying a graded approach, consideration should be given to the utilization and anticipated lifetime of the reactor facility. Facilities with a long expected lifetime (e.g. 30–40 years) will need to include provision for ageing management in the design of SSCs, and also to provide for the knowledge management necessary to support this aspect (see also para. 7.66 of this Safety Guide).

[18] Proper selection of equipment and materials and design principles will help to reduce the need to replace SSCs due to obsolescence.

Provision for extended shutdown

6.34. Requirements on provision for extended shutdown are established in para. 6.71 of Ref. [1]. These provisions will depend on the anticipated duration of the extended shutdown. A graded approach can be used in designing such provisions. For all SSCs that are important to safety and which could suffer some degradation during the extended shutdown period, provision should be made for inspection, testing, maintaining, dismounting and disassembling during the shutdown period. It may be more convenient to remove equipment than to implement a preservation programme with the equipment in place; this decision is usually linked to the future of the research reactor.

6.35. Research reactor designs normally include facilities necessary to ensure safety during shutdown of the facility and these facilities might be used during extended shutdown. The design of such facilities can be graded.[19]

Safety analysis

6.36. The requirements for safety analysis are established in paras 6.72–6.78 of Ref. [1] and related recommendations are provided in Ref. [11]. Safety analysis is required to include analyses of the response of the reactor to a wide range of postulated initiating events. The list of postulated initiating events is required to be complete, covering all credible accidents. Safety analysis is a fundamental part of the design process, and is the basis for determining the safety significance of the SSCs. It is also the basis for demonstrating the safety of the proposed design in support of an application for a licence, and should be used to confirm and validate that grading of application of the requirements has been performed in a consistent and balanced way.

6.37. Grading can be applied to the scope and depth of the safety analysis (see section 1.3 and annex I of Ref. [23] and paras 3.1–3.7 of Ref. [7]). The selection of analysis methods can be graded. The use of enveloping events can also be graded. For example:

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[19] For example, some system requirements will be different during reactor operation and during shutdown. A graded approach may allow a reduced need for some operating equipment (e.g. ventilation, cooling and water purification systems). Provision could be made in the design to take account of prolonged shutdown states. Such situations often occur frequently in research reactors, as many are kept in extended shutdown conditions during holiday seasons owing to lack of continuous utilization. Provisions to maintain subcriticality may also allow some grading of the operational limits and conditions.

- The analysis required for a small facility with a relatively small number of SSCs and applicable postulated initiating events would be much simpler than that for a large and complex facility. A low power reactor having a limited hazard potential may require less analytical detail than a higher power level research reactor.
- Analysis may demonstrate that for some identified postulated initiating events there can be no release of radioactive material from the core, which would eliminate the need for extensive engineered safety features and analysis of their failure.
- The presence of passive or inherent safety features and/or the absence of in-core experiments may also be reflected in the grading of the scope and depth of the safety analysis.
- The use of conservative methods and criteria is a means of simplifying the safety analysis. Facilities with small potential hazard may use conservative criteria in safety analysis, with low impact on the facility design and operation or cost.
- The process of preparation of the safety analysis report allows for the definition and refinement of the postulated initiating events and engineered safety features, and should be graded.

Specific requirements for design

The reactor core and reactivity control system

6.38. The requirements for the reactor core and reactivity control system are established in paras 6.79–6.89 of Ref. [1]. The design requirements relating to the design of the reactor core as a whole and of its individual components (i.e. the fuel assemblies, the reactivity control mechanisms, reflectors, experimental devices, cooling channels and structural parts) are concerned with ensuring that the reactor can be shut down, cooled and maintained subcritical with an adequate shutdown margin for all operational states and design basis accidents.

6.39. A graded approach should be applied in the design of the core by considering the conditions that the core components will need to withstand in the course of their intended service lives in the core. The effects of these conditions, such as integrated neutron flux, thermal and mechanical stresses and chemical compatibility on various materials and fuel assembly types, are generally well known. The extent of analyses and experiments necessary to demonstrate the acceptability of a particular design could be substantially smaller than that necessary for reactors that make use of new types of fuel assemblies and/or novel experimental set-ups. A similar situation can be found in relation to the reactor power: for smaller reactor powers for which it is demonstrated that there is a smaller risk potential, substantially less extensive analysis and simplified conservative criteria might be adequate.

The reactor shutdown system

6.40. The requirements for the reactor shutdown system are established in paras 6.90–6.94 of Ref. [1]. The reactor shutdown system fulfils a crucial safety function for research reactors of all types and sizes. Therefore, the design requirements established in paras 6.90–6.94 of Ref. [1] cannot be graded.

6.41. However, grading can be applied in determining how many redundant shutdown channels are necessary and the extent of instrumentation required for monitoring the state of the shutdown system (see section 3 of Ref. [23]).

6.42. The need for a second, independent shutdown system is required to be considered for research reactors in which experiments with major safety significance are conducted that could affect, in the event of an accident, the first shutdown system, unless inherent self-limiting properties of the design of the core or fuel would prevent a damaging reactivity excursion under all foreseeable reactor states.

The reactor protection system

6.43. The requirements for the reactor protection system are established in paras 6.95–6.105 of Ref. [1]. The reactor protection system is required to be capable of automatically initiating the required protective actions for the full range of postulated initiating events to terminate the event safely. Consequently, the system has to be reliable, and redundancy and independence are required to be applied in its design, to ensure that no single failure or common cause failure in the system could result in the loss of automatic protective actions. If there is a high level of confidence that there are no identified postulated initiating events requiring automatic shutdown, manual operator action could be considered to be sufficiently reliable, as explained in para. 6.96 of Ref. [1].

6.44. The reactor protection system can be graded in the sense that two different research reactors may face different postulated initiating events, or may cope with them in different ways, so that their respective protection systems may differ in the extent of protective actions included in their designs. For example:

— At sites that could be affected by significant seismic events, a seismic sensor may be required to shut down the reactor, while at other sites such protection would not be necessary.
— Initiation of emergency core cooling may be necessary for certain reactors, while in others it would not be necessary (see para. 6.6).

The reactor coolant system and related systems

6.45. The requirements for the reactor coolant system and related systems are established in paras 6.106–6.119 of Ref. [1]. Cooling is one of the basic safety functions. The coolant system is required to be designed to provide adequate cooling to the reactor with an acceptable and demonstrated margin. Adequate cooling is required not only during normal operation at the authorized power levels, but also after shutdown, under a range of anticipated operational occurrences and design basis accidents that involve loss of flow or loss of coolant transients. Grading can be applied to the design of the cooling system. The coolant system can range from the provision of forced cooling with emergency electrical power being available to power some or all of the main coolant pumps, to no emergency power for any of the coolant pumps, to a system where natural convection cooling is used for both heat removal under full power operation as well as decay heat removal. Cooling by natural convection might be adequate for some small research reactors.

Means of confinement

6.46. The requirements for the means of confinement are established in paras 6.120–6.130 of Ref. [1]. Confinement is one of the basic safety functions, as discussed in para. 6.6 of this Safety Guide. Means of confinement are required to be provided to prevent or mitigate an unplanned release of radioactive material in operational states or in design basis accidents. The basic design requirement is to ensure that a release to the environment does not exceed acceptable limits for all accidents taken into account in the design. The results of safety analysis should be used to determine how and to what extent the design of the means of confinement can be graded, e.g. whether volatile fission product (e.g. iodine) traps are necessary in the event of a release of fission products from the reactor.

Experimental devices

6.47. The requirements for experimental devices are established in paras 6.131–6.135 of Ref. [1]. Experimental devices in a research reactor facility can have a significant effect on the safety of the reactor by affecting reactivity, cooling capacity and radiation exposure. In addition, failure of an experimental device may affect the integrity of the reactor.

6.48. Grading can be applied to the alarm and trip signals of experiments interconnecting with the reactor protection system, and/or the control signals of the experiment interconnecting with the reactor instrumentation and control system. Grading can also be applied to the monitoring of the experimental devices from the control room(s).

6.49. Grading can be applied to the design, analysis and the authorization process, in accordance with the type and magnitude of the anticipated hazards, with account taken of the complexity of the experiment and the familiarity of reactor personnel with the experiment.

Instrumentation and control

6.50. The requirements for instrumentation and control are established in paras 6.136–6.144 of Ref. [1]. The basic design requirements for instrumentation and control in this respect are to include sufficient instrumentation in the design to monitor safety related reactor parameters, with reliability commensurate with the importance to safety of the system. The grading of the instrumentation and control system should be based on careful definition of the design basis. Due consideration should be given to the maintainability of the system and its associated cost.

6.51. A graded approach should be taken in determining the types of measurement, locations of measurement and number of measurements to be taken of reactor parameters, such as temperature, pressure, flow, pool/tank water level, gamma radiation, neutron flux and water chemistry parameters. Technical specifications covering all operational states and accident conditions should provide the basis for grading of the design of the instrumentation and control system. A typical example is the measurement of pressure drop across the core. This is a measurement performed in many reactors in order to detect reduced flow through the core (due to either a bypass or a blockage); this measurement is generally not necessary in a critical assembly or in a reactor operating in a natural convection cooling mode.

6.52. Another means of grading instrumentation and control systems is by means of the choice of the level of redundancy. Triple and quadruple channel redundancy is often used for research reactors that need to operate continuously, in order to minimize spurious trips and to allow for testing and/or maintenance of instrumentation and control equipment during operation at power. For research reactors that operate for only a few hours per week or less frequently, such as critical assemblies, a lower level, i.e. two channel (one-out-of-two), redundancy can be applied, thus reducing design and operational complexity as well as costs.

6.53. The level of reliability as well as the accuracy required for measurement of relevant parameters will depend on the importance to safety of the instrumentation and control equipment.

6.54. The degree of automation required for the control system, including the extent of manual control provided, can be graded.

6.55. The instrumentation and control system is required to monitor reactor parameters and to allow for an appropriate response to anticipated operational occurrences and design basis accidents. If analysis shows that, in some situations, the main control room cannot be occupied, then a supplementary control room, separated from and functionally independent[20] of the main control room, is required to be provided in the design. The design and equipment of this supplementary control room can also be graded in accordance with the reactor characteristics and foreseen accident conditions. If the need for a supplementary control room is confirmed, there should be an analysis of its operational requirements and, in particular, the parameters to be monitored and controlled, and the actions necessary to maintain the reactor in a safe shutdown state. Typical features that can be included in the supplementary control room, depending on requirements, are: radiation monitors, fire detection systems and actuators of fire extinguishers, communication means, features for control of the ventilation system, scram and/or safe shutdown features, features for operation of experimental devices, and features for operation of cooling systems.

6.56. A complex and costly human machine interface might not be justified for a low power level research reactor facility.

Radiation protection systems

6.57. The requirements for radiation protection systems are established in paras 6.145–6.148 of Ref. [1]. To achieve the basic requirement established in para. 2.2 of Ref. [1], as discussed in paras 6.26 and 6.27 of this Safety Guide, a

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[20] This means that equipment and features in the supplementary control room should not be functionally dependent on any equipment or features in the main control room.

wide range of radiation protection systems are required to be provided in the design "to ensure adequate monitoring for radiation protection purposes in operational states, [design basis accidents] and, as practicable, [beyond design basis accidents]" (para. 6.145 of Ref. [1]). Paragraph 6.145 of Ref. [1] lists the radiation protection systems used in research reactor facilities and the purposes they serve. All of these systems are likely to be required for research reactors. Grading can be applied in determining the degree to which these systems will provide adequate monitoring for a specific facility.

6.58. Examples of considerations in the grading of radiation monitoring are provided in the following:

— A wide distribution of fixed instrumentation and numerous portable instruments should be employed in a high power level facility.
— A research reactor with various experimental devices, such as beam tubes and neutron guides, neutron activation analysis and radioisotope production facilities, should employ neutron and gamma monitors for the beam tubes and neutron guides and instruments, gamma monitors in the neutron activation analysis facility and in the radioisotope production handling systems and equipment for monitoring of contamination.
— A low power reactor without beam tubes used only for teaching purposes would need only limited and basic monitoring equipment, such as gamma monitors at the open pool end or in the control console and contamination monitors.
— For high power level reactors, supplementary monitoring displays outside the control room should be employed for displaying and recording radiation conditions at specific locations in the facility for operational states and accident conditions (wide range monitoring). Such additional radiation monitoring locations might not be necessary for very low power level facilities (less than 50 kW).

Fuel handling and storage system

6.59. The requirements for the fuel handling and storage system are established in paras 6.149–6.154 of Ref. [1]. The aim of these requirements is to ensure safety in the handling and storage of fresh and irradiated fuel and experimental devices. The main concerns are the prevention of inadvertent criticality and fuel damage from mechanical impacts, corrosion or other chemical damage events.

6.60. The application of the requirements to different reactors can be graded in several aspects, in accordance with the design of the reactor and its utilization programme. For example:

- — Some reactors may need an irradiated fuel storage pool, separate from the reactor pool.
- — Some research reactors may use different types of fuel assemblies for research or testing purposes and may need special storage places for temporary storage of these assemblies.
- — The need for decay heat removal may differ for different reactors, leading to different provisions in the design for decay heat removal.

Process support systems (electrical power, cooling water, process air, heating, ventilation and air conditioning, building service systems)

6.61. The requirements for electrical power supply systems are established in paras 6.155–6.161 of Ref. [1]. The basis for the design of the normal electrical power supply systems is determined by the systems and equipment included in the design that require electrical power during reactor operation and shutdown.

6.62. Consideration is required to be given to the need for an emergency electrical power system to back up the off-site power supply system. Grading can be applied in the design of the power supply system and the emergency power supply system. Considerations relevant for grading include the type and number of safety functions and engineering safety features for which emergency power would be required. The reliability requirements might be different for different reactors, for the various utilization programmes of a particular reactor and for the needs of experimental devices. In a graded approach, the number, size and reliability of any necessary emergency power supply systems should be considered:

- — A reactor may or may not need forced convection cooling after shutdown. The emergency power supply needs and the length of time after shutdown that cooling will be required will determine the specifications of the emergency power supply system. Depending on the reactor power, the power density and the duty cycle, this time could be hours, days or weeks, and this will provide input into determining the necessary reliability of the emergency power supply system. Reliability requirements, in general, may call for a degree of redundancy and separation in the design, which will depend on the design basis accidents for the facility.

— The reactor power will determine the extent of cooling water requirements for power operation and decay heat removal.

Radioactive waste systems

6.63. The requirements for radioactive waste systems are established in paras 6.162–6.166 of Ref. [1]. Radioactive waste (in solid, liquid and gaseous forms) is generated from fuel, neutron and gamma irradiation of reactor core components and coolants, in-core experiments and irradiation facilities, and also from maintenance and operational activities[21].

6.64. A graded approach can be applied to the handling, processing, storage, transport and disposal of radioactive waste, and for control and monitoring of solid, liquid and gaseous effluent discharges, and grading can be related to the types and quantities of radioactive waste generated in the specific reactor facility. Reference [24] provides information on grading of performance standards for package designs for the safe transport of radioactive material, and the appendix of Ref. [25] provides detailed examples of grading for all aspects of transport of radioactive material. A detailed example of the application of the graded approach to the packaging of radioactive material is provided in the Annex to this Safety Guide, taken from the appendix of Ref. [25].

6.65. Grading can be applied on the basis of safety analysis and in accordance with regulatory requirements, including the requirement for application of defence in depth in the design for different types and quantities of radioactive waste, for example:

— Retention tanks may or may not be required to detain radioactive effluents for decay before their removal or release.
— A spill of an amount of heavy water from a heavy water reactor may involve a significant release of tritiated water. For this reason, as well as for economic reasons, a high degree of leaktightness is required in heavy water reactors.

[21] Solids: devices and irradiation targets; replaced components from the reactor systems; irradiated control rods; consumables such as ventilation systems filters; irradiated samples; ionic resins; paper, gloves and plastics used during operations; metallic capsules used during irradiation; water filters. Liquids: primary system coolant; water from dehumidifiers; water used for cleaning and decontamination activities; waste from laundry operations; drainage from hot cells and laboratories; lubricants used in machinery in active zones. Gases: from the reactor tank or pool; from the cooling systems and from irradiation facilities; gases produced by active material created during reactor operation; fission product noble gases; tritium.

Buildings and structures

6.66. The requirements for buildings and structures are established in paras 6.167–6.169 of Ref. [1]. The requirements relating to the design of buildings and structures will depend on their intended safety function and their importance to safety.

6.67. The design basis for buildings and structures can be graded by examining their safety function. For example, the reactor building can constitute a confinement barrier and could be designed accordingly. However, different reactor buildings may require different degrees of leaktightness, which is required to be determined in accordance with the safety analysis of the reactor.

6.68. Careful design of buildings and structures will help in the application of grading to other systems (or to avoiding costly refurbishment later). For example:

- — Separation of areas according to their potential hazard can minimize the need for radioactive waste handling; contribute to design for radiation protection, design for emergency preparedness and response, and design for fire protection; and help to reduce operational costs.
- — Good site evaluation will help to reduce unnecessary conservatism in engineering requirements for buildings and structures in relation to protection against external events, which may have a high impact in relation to the total cost of the reactor facility (see section 2.2.1 of Ref. [23]).

Auxiliary systems

6.69. The requirements for auxiliary systems are established in paras 6.170 and 6.171 of Ref. [1]. Auxiliary systems may affect reactor safety in a number of ways and should be classified and treated in the design accordingly.

6.70. Those auxiliary systems that are not important to safety can be designed to standards commensurate with good industrial practice.

7. OPERATION

GENERAL

7.1. Operation includes all activities performed to achieve the purpose for which the research reactor was designed and constructed or modified. Section 7 of Ref. [1] includes 15 operational topics, and recommendations on application of a graded approach to 14 of these are provided in this section (application of a graded approach to the requirements for physical protection is not within the scope of this Safety Guide).

APPLICATION OF GRADING TO ORGANIZATIONAL PROVISIONS

7.2. The organizational requirements for a research reactor are established in paras 7.1–7.26 of Ref. [1]. Recommendations on meeting these requirements are provided in Ref. [10].

7.3. The general responsibilities and functions of the operating organization cannot be graded. The general responsibilities and functions of the operating organization of a low power research reactor are comparable with those for a high power level, multipurpose research reactor. For example, the direct responsibility and the necessary authority for the safe operation of the reactor are required to be assigned to the reactor manager. This responsibility cannot be graded. However, the manner in which the associated functions are performed can be graded in accordance with their safety significance, maturity and complexity[22].

7.4. Grading may lead to a different organizational structure for research reactors with different hazard potentials. For similar reactors belonging to different operating organizations, different operational structures that have the same functionality can be established. For example:

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[22] A reactor manager of a large research reactor may have direct authority over a technical support group, a safety analysis group, a training group and a quality assurance group, for example. Smaller organizations may have similar groups not under the direct authority of the reactor manager. In either case, the reactor manager should always be kept informed and is required to be the person responsible for the implementation of all of the relevant programmes and projects, and for the safe operation of the reactor.

(a) A research reactor in a State with a limited nuclear programme may need a large and complete in-house capability (such as a technical support group, expertise in quality control, a large inventory of spare components, expertise in isotope production and maintenance personnel). A similar research reactor in a State with a large infrastructure and nuclear programme may not need such a large in-house capability because support could be easily obtained from external organizations.
(b) Grading can be applied, inter alia, in the following areas:
   (i) The number and duties of operating personnel. For reactors with a low potential radiological hazard, an individual could be assigned multiple duties. However, Ref. [1] requires that duties, responsibilities, experience and lines of communication be documented; application of this requirement cannot be graded.
   (ii) Membership of and frequency of meetings of the safety committee(s) (see para. 4.9 of this Safety Guide).
   (iii) Preparation and periodic updating of the safety analysis report (see discussion of the licensing process in para. 3.6 of Ref. [1]).
   (iv) Training, retraining and qualification programmes (see paras 7.5–7.7 of this Safety Guide).
   (v) Operating procedures (see paras 7.21–7.25 of this Safety Guide).
   (vi) Maintenance, periodic testing and inspection programmes (see paras 7.26–7.33 of this Safety Guide).
   (vii) Emergency planning and procedures (see para. 7.41–7.44 of this Safety Guide).
   (viii) The radiation protection programme (see paras 7.50–7.55 of this Safety Guide).
   (ix) The management system (see para. 4.1 of this Safety Guide).

APPLICATION OF GRADING TO TRAINING, RETRAINING AND QUALIFICATION

7.5. Requirements for training, retraining and qualification for research reactor staff and other personnel, such as experimenters, are established in paras 7.27 and 7.28 of Ref. [1]. Recommendations on meeting these requirements are provided in Ref. [10].

7.6. Training, retraining and qualification requirements for research reactor staff and other personnel, such as experimenters, should be consistent with the complexity of the design, the hazard potential, the planned utilization of the facility, the available infrastructure and other functions that might be assigned to staff and other personnel. The required levels of education and operational experience (e.g. the minimum number of hours of operation per year) for the various staffing positions and the contents and the duration of training can be graded in accordance with the above criteria (see para. 1.10 of Ref. [10]).

7.7. Provisions should be put in place for the assessment of the training needs and their fulfilment, including retraining, qualification and operational experience of the staff. Relevant staff positions to be assessed include the reactor manager, shift supervisors, reactor operators, radiation protection staff, maintenance personnel and quality assurance personnel. The requirement that there be adequate training and that it be implemented cannot be graded. The nature and details of the training can be graded (see para. 5.13 of Ref. [10]). Reauthorization after absences can be approached in a graded manner, with retraining, requalification and examinations commensurate with the duration of the absence, the complexity of the facility, and the changes to the facility and its operation during the absence of the individual.

APPLICATION OF GRADING TO OPERATIONAL LIMITS AND CONDITIONS

7.8. The requirements for research reactor operational limits and conditions[23] are established in paras 7.29–7.41 of Ref [1]. Recommendations for the preparation and implementation of operational limits and conditions are provided in Ref. [26].

General

7.9. Since the operational limits and conditions are based on the reactor design and on the information from the safety analysis report concerning conduct of operations, grading will have been employed, as discussed in Sections 3 and 6 of this Safety Guide.

[23] The operational limits and conditions are a set of rules setting forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the regulatory body for safe operation of an authorized facility.

Safety limits

7.10. The requirement to set safety limits and corresponding operational limits to protect the integrity of physical barriers cannot be graded. However, the depth of analysis used to establish the limits can be graded.

Safety system settings

7.11. For each safety limit, at least one safety system is required to be put in place to monitor parameters and to provide a signal to accomplish an action (e.g. to shut down the reactor) to prevent the parameter from approaching the safety limit. The safety system setting should be at an acceptable safety margin from the safety limit. For safety actions of particular importance, such as neutronic trips (scrams), redundant systems should be employed. The analysis performed to establish a suitable safety margin can be graded, together with the level of redundancy.

7.12. Another possibility for grading, which is related to the redundancy and diversity of instruments, lies in the selection of the types and varieties of safety system settings relating to the safety limits and the operational limits and conditions. For example, in a low power reactor, the coolant outlet temperature could be selected as the parameter relating to the fuel temperature for which a safety system setting is defined, while in a higher power reactor, to prevent the safety limits from being approached, a complex system of variables should have defined safety system settings, such as the coolant outlet temperature, the inlet temperature, the coolant flow rate, the differential pressure across the core and the primary pump discharge pressure, as well as parameters from experimental facilities.

Limiting conditions for safe operation

7.13. Limiting conditions for safe operation are operational constraints and administrative limitations on parameters and equipment that are established to provide acceptable margins between normal operating values and the safety system settings during startup, operation, shutting down and shutdown. Appendix I of Ref. [26] provides a list of operational parameters and equipment to be considered in establishing limiting conditions for safe operation, and recommends the selection of only the appropriate items, in accordance with the type of reactor and conditions of operation. Grading can also be applied in the type of analysis performed in establishing a limiting condition for safe operation, on the basis of the selection from the list in appendix I of Ref. [26] in accordance with the type of reactor and conditions of operation.

Requirements for inspection, periodic testing and maintenance

7.14. In order to ensure that safety limits and limiting conditions for safe operation are met, the relevant SSCs are required to be maintained, monitored, inspected, checked, calibrated and tested in accordance with an approved surveillance programme (see paras 7.56–7.64 of Ref. [1]). Surveillance requirements can specify the frequency and scope of inspections and acceptance criteria for each SSC. Grading should be used in establishing these requirements on the basis of the importance to safety and the necessary reliability of each SSC. Additional information is provided in paras 7.26–7.33.

Administrative requirements

7.15. Administrative requirements include those for the organizational structure and responsibilities, minimum staffing, training and retraining, review and audit procedures, records and reports, and event investigation and follow-up (see para. 7.38 of Ref. [1]). The grading that may be possible in relation to some of these activities is discussed in paras 7.3 and 7.4 of this Safety Guide.

7.16. The requirement for action after a violation cannot be graded. The nature of the action can be graded depending on the severity of the violation, i.e. whether operational limits and conditions have been exceeded.

APPLICATION OF GRADING TO COMMISSIONING

7.17. The safety requirements for commissioning of research reactors are established in paras 7.42–7.50 of Ref. [1]. Recommendations for commissioning of research reactors are provided in Ref. [27].

7.18. The commissioning process itself cannot be graded, in the sense that all SSCs, activities and experiments are required to be commissioned. However, grading can be applied to the commissioning programme in the areas of:

- — Organizational structure;
- — Preparation of procedures;
- — Number of hold points and tests;

— Documentation;
— Reporting.

7.19. While grading can be applied to the number of hold points imposed, there should always be a hold point for tests prior to fuel loading (pre-operational tests). A graded approach to testing should be adopted (see para. A.2 of the appendix of Ref. [27]). The extent and type of tests to be performed should be determined on the basis of the importance to safety of each item and the overall hazard potential of the reactor.

7.20. The principles applied in commissioning for the initial approach to criticality, reactivity device calibrations, neutron flux measurements, determination of core excess reactivity and shutdown margins, power raising tests and testing of the containment system or other means of confinement are similar for all research reactors.

APPLICATION OF GRADING TO OPERATING PROCEDURES

7.21. The requirements for the operating procedures for research reactors are established in paras 7.51–7.55 of Ref. [1]. Recommendations for the preparation of operating procedures are provided in Ref. [26]. Appendix II of Ref. [26] presents an indicative list of operating procedures for research reactors.

7.22. For all research reactors, grading will have been employed in the design and construction of the reactor and in the preparation of the safety analysis report and the operational limits and conditions. In addition, grading will have been employed in the preparation and implementation of the management system that governs the format, development, initial and periodic review, control and training on the use and implementation of operating procedures.

7.23. The list of operating procedures presented in appendix II of Ref. [26] should be graded for applicability to a specific research reactor. Consequently, the number of operating procedures developed will depend upon the research reactor and will be less for simpler reactors with low potential hazard.

7.24. Grading can be applied to staff training in the use of the operating procedures. However, all personnel using the operating procedures are required to be thoroughly familiar with them and proficient in their use.

7.25. While all operating procedures are required to be prepared, reviewed and submitted for approval on the basis of criteria established by the operating organization and regulatory requirements, operating procedures can be graded on the basis of their importance to safety. Several examples are:

(a) The procedure for regeneration of an ion exchange system for producing the de-mineralized water inventory in a storage tank will be of low safety significance and will involve mature and non-complex technology. The implications for safety of an error in the regeneration process are low. Consequently, the operating procedure governing this application can be simplified.
(b) In contrast, an operating procedure developed for an application in which an error could cause a violation of the operational limits and conditions should be more detailed. An example is the procedure for regeneration of an ion exchange system for the primary cooling water purification system. While it involves the same basic technology as the example in item (a) above, the safety implications of an error in this application could be much more significant (e.g. if resin were allowed to enter the primary cooling water and, hence, the reactor core). Design features and/or procedural arrangements should, therefore, take into account the greater hazard associated with operation of this system, and the development, review and approval of operating procedures governing such safety significant activities should follow a stringent process.
(c) Procedures required for changes in reactor utilization, special fuel tests, experiments and other special applications are often complex and infrequently used. Since these activities will often impact safety, the development, review and approval of procedures for these activities should follow the same process as that for other procedures governing safety significant activities.

APPLICATION OF GRADING TO INSPECTION, PERIODIC TESTING AND MAINTENANCE

7.26. The requirements for maintenance, periodic testing and inspection of research reactors are established in paras 7.56–7.64 of Ref. [1]. Recommendations for maintenance, periodic testing and inspection are provided in Ref. [28].

7.27. Grading can be applied to the frequency of maintenance, periodic testing and inspection of individual SSCs, and is required to be adjusted on the basis of experience and the importance to safety of the SSC concerned.

7.28. In developing the procedures for maintenance, periodic testing and inspection, consideration should be given to the importance to safety of the SSC concerned, to the complexity of the maintenance, testing or inspection activity, and to the experience of the staff and their familiarity with the systems. Grading of procedures is discussed in paras 7.21–7.25 of this Safety Guide.

7.29. The period for which an SSC is permitted to be out of service while reactor operation continues is usually stated in the operational limits and conditions for the research reactor and can be graded. For example, no outage time whatsoever might be acceptable for automatic shutdown systems, while outage times of up to several days might be acceptable for other systems (e.g. for a purification system monitoring the primary coolant pH). The allowed outage time will depend on the extent to which safety is impacted, or the ease of applying compensatory measures.

7.30. In a similar way, the frequency for periodic testing can be graded. A balance should be sought between the improvement in detection of faults owing to more frequent testing and the risk that testing could be performed incorrectly and leave the SSC in a degraded state. The testing frequency could also be increased to the point where testing causes more frequent failures of SSCs, and so it should be recognized that there is always an optimum test frequency. This consideration also applies for periodic maintenance. The frequency of replacement of SSCs subject to ageing degradation due to, for example, existence of high radiation fields, can be graded based on the feedback of operating experience, including that from other reactors, and on the bases of the results of research and development.

7.31. At times, it may become necessary to perform maintenance, periodic testing or inspection in controlled areas or on components that are radioactive. Although the procedure for the maintenance, periodic testing or inspection activity may have been graded, controls should be put in place to ensure that the radiation exposure of workers is within the prescribed limits. The radiation protection measures can be graded on the basis of the potential for occupational exposure.

7.32. When maintenance, periodic testing or inspection of an SSC is uncomplicated and operating experience indicates a high reliability of the SSC, a review and regrading of the maintenance, periodic testing or inspection activity leading to a change in the procedure might be justified. However, a change in the procedure should be subjected to the established preparation, review and approval process.

7.33. In weighing the importance to safety, maturity and complexity of maintenance, periodic testing and inspection activities for grading purposes, it might be concluded that some activities are highly specialized and involve complex and sophisticated techniques. Such activities are often performed by contracted, external experts. Such outsourcing should be carefully considered by the operating organization to ensure that external support is secured and that resources will be available throughout the operating lifetime of the facility. The use of external contractors for the performance of maintenance, periodic testing and inspection is discussed in Ref. [28].

APPLICATION OF GRADING TO CORE MANAGEMENT AND FUEL HANDLING

7.34. The requirements for core management and fuel handling are established in paras 7.65–7.70 of Ref. [1]. Recommendations for core management and fuel handling are provided in Ref. [9].

7.35. Research reactors with a low potential radiological hazard, having power ratings up to several tens of kilowatts, and critical assemblies may need a less comprehensive core management and fuel handling programme. Low power reactors require infrequent core adjustments to compensate for burnup. They operate with substantial margins to thermal limits, allowing the consideration of a broad envelope of acceptable fuel loading patterns in the initial safety analysis in lieu of core specific calculations. While all recommendations in this Safety Guide should be considered, some might not apply to these low power level reactors. For these reasons, the requirements for core management and fuel handling should be graded for applicability to the particular research reactor (see paras 1.11 and 2.2 of Ref. [9]).

7.36. Reference [29] presents a method for determining the safety significance of modifications to a research reactor and this method is applicable to core management and fuel handling. Grading of the analysis and verification associated with the proposed core management and fuel handling activities may be possible on the basis of their safety significance (see also paras 7.47–7.49 of this Safety Guide).

APPLICATION OF GRADING TO FIRE SAFETY

7.37. The requirements for fire safety are established in paras 6.22–6.25 and 7.71 of Ref. [1]. Recommendations for fire safety are provided in Refs [30, 31].

7.38. The potential fire hazards should be discussed in the safety analysis report and an indication should be provided of their relative importance (i.e. in terms of likelihood and consequences) in the facility. This information can serve as a basis for grading the implementation of the fire prevention and protection measures.

7.39. Grading of the measures for fire protection might be facilitated by provisions incorporated into the design in accordance with the fire hazard analysis, which is required to be performed [1], and which should be periodically reviewed and updated [30], as well as by siting considerations.

7.40. Since techniques for fire safety assessment and analysis are well understood, the amount of analysis needed to determine how best to apply the available resources can be graded. The analysis should employ techniques that have proven adequate in similar facilities elsewhere.

APPLICATION OF GRADING TO EMERGENCY PLANNING

7.41. The requirements for emergency planning are established in paras 6.20 and 7.72–7.78 of Ref. [1]. Further requirements for emergency planning and response are established in Ref. [32]. Detailed approaches for a wide range of emergencies, suitable for the application of grading, are discussed in Ref. [33].

7.42. The emergency plan and its implementing procedures are required to be based on the accidents analysed in the safety analysis report (design basis accidents) as well as those additionally postulated for the purposes of emergency planning (beyond design basis accidents). These analyses will allow the development of a source term for use in emergency planning. For some research reactors, it may be possible to demonstrate that health effects in the population and effects on the environment for credible accident scenarios are negligible and that emergency preparedness may be focused on on-site response. An understanding of the nature and magnitude of the potential hazard posed by each research reactor is required for preparing an appropriate emergency plan.

7.43. In accordance with the concept of a graded approach, paras 3.6 and 3.7 of Ref. [32] establish a categorization scheme for nuclear and radiation related threats which is required to be used, and which provides a basis for developing optimized arrangements for preparedness and response. This scheme requires that an urgent protective action planning zone be specified. The threat categories are:

— Category I: Facilities for which on-site events are postulated that could give rise to severe deterministic health effects off the site, or for which such events have occurred in similar facilities.
— Category II: Facilities, such as some types of research reactor, for which on-site events are postulated that could give rise to doses to people off the site that warrant urgent protective action in accordance with international standards, or for which such events have occurred in similar facilities.
— Category III: Facilities, such as industrial irradiation facilities, for which on-site events are postulated that could give rise to doses or contamination that warrant urgent protective action on the site, or for which such events have occurred in similar facilities.

Most research reactor facilities are in threat category II or III. This grading may lead to an urgent protective action planning zone as small as the reactor building itself or large enough to extend off the site.

7.44. The magnitude of the potential source term and the engineered safety features are the most important factors affecting the grading of the emergency plan. Grading can be applied, inter alia, in the following areas:

— The organization needed to carry out the emergency response.
— The urgent protective action planning zone.
— The identification and classification of the hazard.
— Notification requirements for informing the authorities.
— The amount, nature and storage location of equipment needed to survey and monitor people and the environment in the event of an emergency.
— The number, identity, training of and agreements with off-site agencies (e.g. police, fire services, medical treatment and medical transport) that will help in an emergency. Although the emergency might not have an off-site impact, it is generally prudent to establish contact with off-site agencies to ensure their agreement if a request for assistance is issued.
— The timescales envisaged for the various phases of the response to an emergency.
— The types and the extent of exercises and drills.
— The nature and amount of other resources needed for preparedness for and response to an emergency.

— The facility’s proximity to populated areas, which can significantly increase or decrease the scope and the content of the emergency planning.

APPLICATION OF GRADING TO RECORDS AND REPORTS

7.45. The requirements for records and reports are established in paras 7.81–7.84 of Ref. [1]. Requirements for the control of records are established in paras 5.21 and 5.22 of Ref. [5], and recommendations are provided in paras 5.35–5.49 and annexes I–III of Ref. [6].

7.46. Consistent with the purpose for which reports are prepared and records are kept, para. 2.44 of Ref. [6] lists specific examples of where a graded approach could be applied to controls for the records management process:

“— Preparation of documents and records;
— Need for and extent of validation;
— Degree of review and the individuals involved;
— Level of approval to which documents are subjected;
— Need for distribution lists;
— Types of document that can be supplemented by temporary documents;
— Need to archive superseded documents;
— Need to categorize, register, index, retrieve and store document records;
— Retention time of records;
— Responsibilities for the disposal of records;
— Types of storage medium, in accordance with the specified length of time of storage.”

APPLICATION OF GRADING TO REACTOR UTILIZATION AND MODIFICATION

7.47. The requirements for reactor utilization and modification are established in paras 7.85–7.92 of Ref. [1]. Recommendations for reactor utilization and modification are provided in Ref. [29].

7.48. The operating organization is required to establish criteria for categorizing a proposed experiment or modification in accordance with its importance to safety. The resulting categorization should then be used to determine the types and extent of the analysis and approvals to be applied to the proposal.

7.49. In cases where an experiment or modification was not anticipated in the design, its safety significance should be determined. Paragraph 1.11 of Ref. [9] and annex I of Ref. [29] provide guidance for categorization for the treatment of modifications according to their hazard potential using a four category system:

(a) Changes that could have major safety significance;
(b) Changes that could have a significant effect on safety;
(c) Changes with apparently minor effects on safety;
(d) Changes having no effect on safety.

Alternatively, a two category system can be used. The first category is the category for which the modification or experiment is submitted to the regulatory body for review and approval. It includes modifications or experiments that:

— Involve changes in the approved operational limits and conditions; or
— Affect items of major importance to safety; or
— Entail hazards different in nature or more likely to occur than those previously considered.

The second category requires local review and approval of the modification or experiment, with notification to the regulatory body for information.

APPLICATION OF GRADING TO RADIATION PROTECTION

7.50. The requirements for radiation protection are established in paras 7.93–7.107 of Ref. [1] and in Ref. [34]. Recommendations for radiation protection in the design and operation of research reactors are provided in Ref. [35].

7.51. While the content of the radiation protection programme will depend on the design, power level and utilization of the particular research reactor, many aspects of the programme will be similar for all research reactors.

7.52. The application of grading to the radiation protection programme should be consistent with the reactor's design and with its utilization (see paras 1.5 and 1.9 of Ref. [35]). The environmental monitoring programme will also depend on the location of the reactor. For example, a densely populated area will generally require a more extensive environmental monitoring programme.

7.53. A critical assembly could present a higher risk of external radiation exposure of operating personnel than a higher power research reactor but the latter could present a higher risk of contamination of personnel causing internal radiation exposure. In addition, because critical assemblies are sometimes located within conventional industrial standard buildings, reactivity accidents involving a critical assembly could result in a higher risk of contamination outside the building, compared with higher power reactors with a larger source term that have a containment structure.

7.54. Working areas within a research reactor should be classified (graded) into supervised areas and controlled areas, according to the magnitudes of the expected exposures, the likelihood and magnitude of potential exposures, and the nature and extent of the required radiological protection measures. Controlled areas themselves should be subject to classification (grading) according to the protective measures employed or the expected radiological level (see paras 5.44–5.46 and 5.48 of Ref. [35]):

— For a high power research reactor, it may be necessary to further grade the controlled areas into different levels, for example, levels I, II and III. Specific procedures may be prescribed for work in level II controlled areas (in addition to those procedures prescribed for level I areas), which may require, in some cases, the use of protective garments, equipment or tools. Level III controlled areas will normally be closed by a physical barrier (e.g. an airlock door) that is opened only by authorized workers. Furthermore, opening a door to a level III controlled area during reactor operation could be set to result in automatic shutdown of the reactor.
— For a low power research reactor, level II controlled areas and level III controlled areas may not be needed.

7.55. Reference [35] provides general recommendations concerning the nature and scope of an operational radiation protection programme. The application of these general recommendations can be graded to determine the nature and scope of the elements of the specific operational radiation protection programme.

APPLICATION OF GRADING TO SAFETY ASSESSMENTS

7.56. The requirements for safety assessment are established in para. 7.108 of Ref. [1]. Recommendations for performing safety assessments are provided in Ref. [11].

7.57. Sections 4 and 7 of Ref. [1] establish the requirements for management and verification of safety and for safety assessment throughout all the stages of the lifetime of the reactor. Recommendations on grading in the management and verification of safety have been provided in Section 4 of this Safety Guide.

7.58. Paragraphs 3.1–3.7 of Ref. [7] establish general requirements for the application of the graded approach for the safety assessment of facilities and activities. The main factor to be taken into consideration in the application of the graded approach is that the safety assessment is required to be consistent with the magnitudes of the possible radiation risks arising from the facility or activity.

7.59. The application of a graded approach will vary according to the stage of the safety assessment as the possible radiation risks arising from the facility are clarified. At the design concept stage, for example, the safety case will focus on a statement of intent and principles. As the maturity of the facility progresses into the operational stage, much more detail and analysis should be provided in the safety assessment. The safety assessment for the decommissioning stage should contain significantly less detail and analysis than that for the operational stage. The scope and level of detail of the safety assessment and the resources deployed to produce it should be adjusted accordingly.

7.60. The main factors influencing the radiation risk and, thus, the level of detail of a safety assessment at the operational stage are: the predicted or historical radioactive releases and radiation exposure of workers and the public; the consequences of anticipated operational occurrences and accidents with respect to damage to SSCs and radiation exposure of workers and the public; and the potential consequences (in terms of radiation exposure and damage to SSCs) of low probability events with potentially high consequences.

7.61. The graded approach should also be applied to the requirements for updating the safety assessment (see para. 5.10 of Ref. [7]). The frequency at which the safety assessment is updated and the level of detail of the safety assessment should be graded on the basis of the number and extent of modifications to the reactor systems and experimental facilities; changes to procedures; results of compliance monitoring of operational limits and conditions; modifications of safety significance; evidence of component ageing; feedback from operating experience and from events; changes in site conditions; and new regulatory requirements. In addition, grading could depend on the experience gained in similar facilities. Typically, for a reactor with more than 5–10 years of demonstrated operational maturity, a periodic review of the safety assessment for the overall facility every 5 years would be appropriate. However, it is suggested that the maximum time between periodic reviews of the safety assessment should be no more than 10 years, irrespective of the type or utilization of the research reactor. With regard to reactors with more than 20 years of operation, more emphasis should be placed on safety assessment of component ageing, particularly with regard to control systems and safety related passive components in locations in which it may be difficult to inspect and to carry out repairs (e.g. inaccessible coolant piping and reactor tanks or vessels).

APPLICATION OF GRADING TO AGEING RELATED ASPECTS

7.62. The requirements for ageing related aspects are established in para. 7.109 of Ref. [1]. Recommendations on ageing management for research reactors are provided in Ref. [36].

7.63. While selection of materials and the effects of the operating environment on their properties should be taken into account in the design of all research reactors, the grading can be applied to development of the ageing management programme, including in-service inspection, throughout the operating lifetime of the facility.

7.64. Grading can be applied in determining the appropriate frequency of inspections, in selecting detection methods, as well as in establishing measures for prevention and mitigation of ageing effects, which could be based on the estimated service lives of the SSCs, their complexity and their ease of replacement. In most research reactors, it is feasible to inspect most SSCs periodically and to replace components if necessary. Particularly important material ageing concerns are corrosion in reactor tanks and vessels, where leak detection can be difficult and repair or replacement might not be practicable. Similarly, the management of corrosion of inaccessible primary coolant piping and associated components is of key importance for reactor longevity. An important knowledge management area, which supplements the appropriate selection of materials and the management of ageing related effects, is the need for human resources management to address the ageing of research reactor personnel. Other key knowledge management areas are configuration management, document control and programmes for feedback of operating experience.

7.65. Grading may also be applicable to the resources necessary to implement the ageing management programme. While a dedicated organizational unit may be needed to implement such a programme for higher power research reactors, the ageing management activities for research reactors having a low power might be performed by the maintenance personnel of the facility.

APPLICATION OF GRADING TO EXTENDED SHUTDOWN

7.66. The requirements for the safety of a research reactor in extended shutdown are established in paras 6.71, 7.111 and 7.112 of Ref. [1]. Further information on extended shutdown is provided in paras 6.34 and 6.35 of this Safety Guide and in Ref. [37].

7.67. The operating staff of a research reactor in extended shutdown could be smaller in number than that for an operating reactor. However, a large reduction in the overall numbers of personnel might be inappropriate. Concerns such as a possible loss of the operating experience and knowledge of the facility that will be necessary for restart may preclude a large reduction in personnel.

7.68. A graded approach should be applied to the scope and details of the activities, the measures to be implemented, the level of reviews, the frequency and extent of maintenance, testing and inspection activities during an extended shutdown, and the extent of relief from requirements that apply during the normal operating regime.

8. DECOMMISSIONING

8.1. The requirements for decommissioning are established in paras 8.1–8.8 of Ref. [1]. Recommendations are provided in Ref. [38]. Reference [16] is intended to assist the regulatory body, the operating organization and technical support organizations in the application of a graded approach to the development and review of safety assessments for decommissioning activities. Section 3 of Ref. [39] establishes the requirements for application of a graded approach to the development of the decommissioning plan and Ref. [40] provides recommendations on application of the graded approach in the release of sites from regulatory control.

APPLICATION OF GRADING TO DECOMMISSIOING

8.2. Decommissioning requirements are applicable to every research reactor. The range of decommissioning activities for which a safety assessment is required is broad, and the scope, extent and level of detail of the safety assessment should be commensurate with the types of hazard and their potential consequences. A graded approach should, therefore, be applied to the development and review of safety assessments. The effort associated with meeting the requirements (e.g. for the preparation and review of decommissioning plans and procedures) can be graded. A graded approach can be applied to the scope of the required analyses and investigations, the number and variety of procedures to be prepared, the scope and depth of the reviews, the controls imposed, the number and types of approvals needed, the extent of measures for radiation protection and the scope of surveillance activities during decommissioning (see section 3 of Ref. [39]).

8.3. Decommissioning of a research reactor facility can be graded on the basis of the activities and types of radioactive material and sources in the facility, the complexity of dismantling operations, and the availability of experienced personnel and of proven techniques and the means to employ them. Knowledge of the facility, which might be lost when the reactor is permanently shut down owing to the retirement or departure of experienced personnel, should be managed, in order to facilitate efficient and safe decommissioning operations.

8.4. Decommissioning should be graded according to the type of facility and the utilization programme implemented. For example:

- Critical assemblies might not represent a substantial concern from a radiation protection or radioactive waste management point of view. A critical assembly should, however, be monitored for activation products before commencing disassembly, although the dismantling activities can generally be conducted without the need for special tools or highly qualified personnel. In many cases, the building and other installations housing a critical assembly may later be used for other purposes.
- Research reactors of low power may have some radiological concerns, which can easily be dealt with by the radiation protection officers of the operating organization. A waste management plan should be elaborated, and usually a small number of high activity level components are found (such as core support structures, neutron detectors, control rods and experimental devices from the core). The buildings should be assessed: sometimes the walls and ventilation systems are contaminated as well as the

floors. In some cases, appropriate decontamination of the reactor tank will allow its release for other uses.
- In higher power research reactors, the secondary cooling system, the process air system, radiation protection equipment and instrumentation and control systems, for example, are usually not contaminated and can either be disposed of or be recycled for other uses. Ventilation and radiation monitoring systems are kept in operation throughout decommissioning.
- Release of the site of a research reactor from regulatory control might not be appropriate, as in some cases the State may intend to continue using the existing infrastructure for other purposes, such as for storage of radioactive sources, for storage of radioactive waste or as a gamma irradiation facility.

8.5. The operating organization should determine the scope of the decommissioning plan by considering the most significant criteria, such as the resources available for decommissioning, the time period to decommissioning and the required end state (such as full or partial decontamination and/or dismantling or release of the site from regulatory control). The scope of the decommissioning plan can then be graded, e.g. on the basis of the current status of the installation and its possible future use.

8.6. Regulatory review of the safety assessment for decommissioning should follow a graded approach and account should be taken of the following (see paras 3.1–3.5 and 5.6–5.8 of Ref. [16]):

- All relevant safety requirements and criteria derived from national legal and regulatory frameworks;
- The potential (e.g. in terms of likelihood and magnitude of consequence) for the proposed decommissioning activities to lead to an uncontrolled or accidental radioactive release (e.g. in working premises, on the site, off the site or at nearby facilities);
- The safety assessment's estimates of radioactive release and dose to workers arising from planned decommissioning activities;
- The complexity and novelty of the proposed decommissioning activities;
- Operator aspects (e.g. the operator's — or the contractor's — past performance and relevant experience, both in decommissioning and in producing safety assessments for decommissioning; the complexity of the organization);
- Relevant incidents and events at other facilities or at similar facilities during decommissioning;

— The scope of the decommissioning activities being assessed (e.g. a stage of a larger project, a single large project, a proposal leading to the final release of the facility from regulatory control);
— Technical or safety related concerns of other competent authorities (e.g. authorities having oversight over physical protection, security or non-radiological hazards).

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Annex

EXAMPLE OF STEPS IN THE GRADED APPROACH FOR PACKAGING OF RADIOACTIVE MATERIAL

The following is reproduced from paras A.6–A.10, and tables 2 and 3 of the appendix to IAEA Safety Standards Series No. TS-G-1.4[1]:

"A.6. Organizations involved in the design and manufacture of packagings typically use a component based graded approach and qualitative expressions of risk based on the safety consequences of failure of the packaging component. Logical steps in the graded approach are:

(1) Identification of the package type according to the Transport Regulations [1];
(2) Classification of the package by the development of a list of the packaging components and software to be used in its design, fabrication, use, inspection or testing, and assignment of a quality category (grade) to each (Table 2);
(e) Specification of the management controls required and assignment of a quality category (grade) to each (Table 3).

"A.7. Many quality requirements are specified by the codes or standards for design, fabrication, inspection and testing that are determined in the initial stages of the package design. These codes, for example, often impose controls on the procurement, receipt, storage and use of the package materials.

"A.8. Quality codes and standards may vary between different components of a single container type, and between similar components of containers of different types. The container materials, for example, may include bulk material such as metal plate, sheet, castings, weld metal and forgings. Items fabricated by subtier vendors (seals, bolts, pressure relief valves, rupture discs, special closure assemblies, etc.) may also be included. Typically, traceability of material, control of chemical and physical properties of the material, and isolation of the material from non-conforming material are used to ensure proper fabrication. When applicable, subtier vendors should control the quality of the component materials used.

[1] INTERNATIONAL ATOMIC ENERGY, The Management System for the Safe Transport of Radioactive Material, IAEA Safety Standards Series No. TS-G-1.4, IAEA, Vienna (2008).

"TABLE 2. EXAMPLES OF QUALITY CATEGORIES BASED ON CONSEQUENCES OF FAILURE

Quality category	Safety classification	Consequences of failure
Grade 1	Safety class — critical to safe operation	Grade 1 items are those directly affecting package leaktightness or shielding, or, for packages of fissile material, those directly affecting geometry and thus criticality control. Examples include the primary and secondary containment vessels, outer and inner O-rings on the vessels, and lead shield, as well as software used in their design, fabrication, use, inspection or testing.
Grade 2	Safety significant — major impact on safety	Grade 2 items are systems, structures or components whose failure could indirectly affect safety in combination with a secondary event or failure. Examples include impact absorbers that provide impact protection between the primary and secondary containment system during an accident, and software used in their design, fabrication, use, inspection or testing.
Grade 3	Production support — minor impact on safety	Grade 3 items are those affecting systems, structures or components whose malfunction would not affect the effectiveness of the packaging and so would be unlikely to affect safety. Examples include devices that indicate tampering, such as security lock wires and seals, and package identification plates.

"**Note:** Items whose failure does not impact the safety or quality of the product or service need not be included in this graded system. An example of such non-graded items is software that facilitates routine operation, handling and/or use of the package or packaging.

“TABLE 3. GRADED MANAGEMENT CONTROLS

Graded management controls	Quality categories		
	Grade 1	Grade 2	Grade 3
The design is based on the most stringent industry codes or standards, and the design verification is accomplished by prototype testing or formal design review.	×		
The suppliers and subtier suppliers have a management system based on applicable criteria established in an acceptable national or international standard.	×		
The manufacturing planning specifies complete traceability of raw materials and the use of certified welders and processes.	×		
The procurement documentation for materials for services specifies that only suppliers from qualified vendor lists are used.	×	×	
A comprehensive programme for specifying commercial grade items and controlling counterfeit parts is required.	×	×	
Verification planning (testing and inspection) requires the use of qualified inspectors (i.e. individuals performing non-destructive examinations such as radiography and ultrasonic testing are qualified in accordance with recommended practices described in appropriate national or international standards).	×	×	
Only qualified auditors and lead auditors perform audits.	×	×	
Comprehensive design, fabrication and assembly records, results of reviews, inspections, tests and audits, results of the monitoring of work performance and materials analyses, and results of maintenance, modification and repair activities are maintained.	×	×	

“TABLE 3. GRADED MANAGEMENT CONTROLS (cont.)

Graded management controls	Quality categories		
	Grade 1	Grade 2	Grade 3
The design is based on the most stringent industry codes and standards, but design verification can be achieved by the use of calculations or computer codes.		×	
The manufacturing planning need not require traceability of materials, and only specified welds are done by qualified welders.		×	
Only the lead auditor need meet certain qualification requirements.		×	
Verification activities still require use of independent inspectors qualified to appropriate codes, standards or other industry specifications.		×	×
The procurement of materials need not be from a qualified vendor list.			×
Items are purchased from a catalogue of ‘off the shelf’ items.			×
When the item is received, the material is identified and checked for damage.			×
Self-assessments rather than independent assessments are the primary method of assessing and verifying performance.			×
Records are maintained in temporary files for a specific retention period (e.g. six months) after shipment.			×

“A.9. Fabrication requirements may also vary between different components of a single type of container and between similar components of containers of different categories, according to the materials of construction. For example, welds that attach or join components should be in the same quality category as the higher level component unless a lower classification can be justified. Welds that join a component (such as a cylinder longitudinal seam weld) should be in the same quality category as the component of which they are a part. Many requirements for processes (e.g. welding and heat treating) are defined within the code used for construction. However, for some special processes (e.g. pouring of gamma shielding material), no specific code exists, and approved procedures are needed to perform the task. Each procedure should be qualified to ensure its conformance to requirements.

“A.10. Since there may be no manufacturer available with an approved management system for Grade 1 component materials such as foam, honeycomb or wood (used in impact limiters), concrete or lead (used in shielding), and polymers (used in seals), packaging or cask vendors may be allowed to use the manufacturer’s management system to procure Grade 1 components. This will place more responsibility on the designers to specify the most important properties and characteristics of materials, and on the manufacturers to comply with these specifications.

“[1] INTERNATIONAL ATOMIC ENERGY, Regulations for the Safe Transport of Radioactive Material, 2005 Edition, IAEA Safety Standards Series No. TS-R-1, IAEA, Vienna (2005).”

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Director General

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