DS351 Draft 2
Date: 2009/08/18

IAEA SAFETY STANDARDS

for protecting people and the environment

Status: For approval for submission to MS

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

DRAFT SAFETY GUIDE
DS351

New Safety Guide

IAEA
International Atomic Energy Agency

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. This document presents guidance on the use of the graded application of the safety requirements for research reactors as presented in *Safety of Research Reactors*[1].

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

1.3. Because of the wide range of utilization activities noted above it is clear that the safety requirements for research reactors should not be applied to every research reactor in the same way. Thus, requirements that are applicable to multipurpose, high power level research reactors may not be fully applicable to research reactors with very low power and very low associated radiological hazard to facility staff, the public and the environment. The Safety Requirements document on the Safety of Research Reactors, Ref. [1], which has been developed to apply to a wide range of research reactors, includes recommendations (paras 1.11 to 1.14) for applying the safety requirements utilizing a graded approach.

1.4. The idea of providing guidance on grading the application of safety requirements and safety guides in IAEA documents is not new. There are a number of historical references related to grading[2]. Ref. [2] provides a detailed definitions of the graded approach:

"(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

[1] A research reactor (as defined in Ref. [1], NS-R-4, footnote 4) is a nuclear reactor used mainly for the generation and utilization of neutron flux and ionizing radiation for research and other purposes. In the context of this Safety Guide publication, the term research reactor also includes associated experimental devices (defined in NS-R-4, footnote 5) and critical assemblies.

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

commensurate, to the extent practicable, with the likelihood and possible consequences of, and the level of risk associated with, a loss of control.

(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.

1.5. Contemporary IAEA documents continue to refer to a graded approach:

– Ref. [3], para. 3.15, Principle 3 of the Fundamental Safety Principles indicates safety has to be assessed and periodically reassessed throughout the lifetime of facilities and activities, consistent with a graded approach.

– Ref. [3] paras 3.22-3.24, Principle 5 of the Fundamental Safety Principles indicate that the resources devoted to safety by the licensee and the scope are to be commensurate with the magnitude of the potential radiation risks.

– Ref. [4] para. 2.2 notes that grading is to be executed in accordance with national circumstances and risks associated with facilities, as part of the national policy and strategy for safety.

– Ref. [4] paras 2.6-2.7 and Ref. [5] paras 2.37-2.44, 5.6 and 6.68 discuss the graded approach application to Management Systems. References [4] and [5] have now replaced the QA documents listed in footnote 2.

– Ref. [28], Section 1.3 and Chapter 3 note the special attention to be given for safety assessment with regard to the application of a graded approach.

– Ref. [32], para 3.10 discusses inspection programs of the regulatory body should establish a graded approach in responding to unplanned situations or events.

1.6. A graded approach[3] is helpful to users of the IAEA publications on research reactors and relevant thematic documents in deciding whether the application of safety requirements may be suitably gradable for a particular installation, taking into account the available national infrastructure, the safety requirements for Systems, Structures and Components (SSCs), and activities within the facility.

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 a research reactor's lifetime (site selection, site evaluation, design, construction, commissioning, operation

[3] Some Member States refer to the graded approach as 'proportionality'.

((which includes commissioning, normal operation, utilization and modification) and decommissioning). The requirements considered are primarily those in Ref. [1], with some references to other thematic publications of the IAEA (e.g. on Legal and Governmental Infrastructure Ref. [6], Management Systems Refs. [4] and [5] and General Safety Requirements, Ref. [28]). It is intended for the use of regulatory bodies, operating organizations and other organizations involved in the design, construction and operation of research reactors.

SCOPE

1.8. This safety guide presents guidance for applying a graded approach, without compromising safety.

1.9. The application of a graded approach throughout all the important activities[4] in the lifetime of research reactor facilities is discussed. These activities are identified in Ref. [1], Chapters 3 to 8. A major component of the design activity, Chapter 6 of this Guide, involves grading of specific design requirements, applied to the design of SSCs for particular reactor types so that safety objectives Ref. [1], para. 2.2, are achieved. The application of grading applicable to reactor hardware and equipment (SSC's), as opposed to activities in general, is also discussed as part of Chapter 6 and uses the list of SSCs provided in Ref. [1].

1.10. In this safety guide it is considered that all relevant safety requirements must be complied with, in all applications of the 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, a waiving process, as suggested in Ref. [1] can be used. This process is separate and distinct from the graded approach. Waiving[5] is not discussed in this publication. Ref. [28], para. 1.5 also notes that a graded approach must be used for implementation of the safety requirements to provide flexibility. It should though be recognised that while safety requirements are to be complied with, the level of effort applied in carrying out the necessary safety assessments needs to be commensurate with the potential radiation risks and any uncertainty associated with the potential radiological hazard of the facility or activity.

[4] Activities, in the context of this safety guide, include all the stages needed to achieve the purpose for which the nuclear research reactor was designed and constructed or modified, see Ref. [1], footnote 2. Ref. [4] uses a more general definition of activities which encompasses any practice or circumstances in which people may be exposed radiation sources.

[5] Waiving is sometimes called grading to zero, implying complete elimination of a requirement. Ref. [1] para. 1.14 implies that some selected factors, which may be contributors to various requirements, may be waived, so that the concept of a graded approach is still being applied.

STRUCTURE

1.11. Chapter 1 outlines the background, the objective, scope and structure of this Guide. Chapter 2 provides the description of the general principles of a graded approach and its application. Chapters 3 to 8 discuss the application of a graded approach to the following six activities:

(a) Regulatory Supervision;

(b) Management and Verification of Safety;

(c) Site Evaluation;

(d) Design;

(d) Operation; and

(e) Decommissioning.

Chapters 3 to 8 have titles identical to the corresponding chapters of Ref. [1].

1.12. Each chapter of this publication begins with a brief description of the safety requirements as specified in Ref. [1] and, in some areas, augmented with additional requirements contained in other IAEA publications. The descriptions are followed by a discussion of the use 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 to all the stages during the various stages of a research reactor's lifetime, see para 1.7.

2.2. During the lifetime of a research reactor any grading that is performed should ensure that safety functions and Operating Limits and Conditions are preserved and there are no negative effects on the facility staff, the public, or the environment.

2.3. A general example of a graded approach from Ref. [2] is a structured method by means of which the stringency of application of requirements is varied in accordance with the circumstances, the regulatory systems used, the management systems used. For example, a method in which:

(1) The significance and complexity of a product or service are determined;

(2) The potential impacts of the product or service on health, safety, security, the environment, and the achieving of quality and the organization's objectives are determined;

(3) The consequences if a product fails or if a service is carried out incorrectly are taken into account.

2.4. The grading of activities should be based on engineering judgment, safety analyses, and regulatory requirements. Other guiding elements are the complexity and the maturity level of the technology, operating experience associated with the activities and the lifecycle stage of the facility.

2.5. The safety requirements should be applied in such a way that the level of analysis, documentation and actions are commensurate with the potential hazard associated with the facility, without adversely affecting safety. The factors to be considered are as follows, see Ref. [1] and Ref. [9]:

(a) The reactor power;

(b) The radiological source term;

(c) The amount and enrichment of fissile 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 elements;

(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, reactor physics experiments);

(j) Location of the site; including potential for occurrence of external hazards and characterization for airborne and liquid releases of radioactive materials; and

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

DESCRIPTION OF THE APPLICATION FOR A GRADED APPROACH

2.6. A number of the activities in para. 1.11 have safety requirements that are identified to be general requirements. Hence an initial step in the grading process is to identify whether features of a specific research reactor require consideration within the general requirements.

2.7. No quantitative ranking procedure for the application of the graded approach to the safety requirements is suggested. The application of the graded approach determines the appropriate effort and manner needed to comply with a requirement, according to the attributes of the facility.

2.8. The application of grading presented in this Safety Guide begins with a facility categorization (Step 1) which determines the baseline potential radiological risk. With this step a facility can be initially categorized into a range from the highest to the least risk. This categorization serves to provide an initial screening, at the facility level. The next step (Step 2) is analysis and grading of activities and/or SSCs important to safety. This second step provides a more detailed level of grading to be applied to the particular characteristics of the facility.

Step 1: Facility Categorization

2.9. Perform a qualitative categorization of the facility, based on the potential radiological hazard, using the multi-category ranking from Ref. [11], para. 1.11:

(i) Off-site radiological hazard potential;

(ii) On-site radiological hazard potential only;

(iii) No radiological hazard potential beyond the reactor hall; and

(iv) No radiological hazard potential.

Step 2: Analysis and Grading

2.10. With this step the level of detail required for grading activities and/or SSCs is chosen to be commensurate with their relative importance to safety. The level of detail would specify, for example, the rigour of analysis required, the frequency of activities such as testing and preventive maintenance, the depth of required approvals and the activity oversight level.

2.11. Determine through analysis for each of the major activities and SSCs defined by Chapters 3 to 8 of Ref. [1] the appropriateness of applying a graded approach. The grading application should be commensurate with the characteristics of safety requirements of the activities and SSCs and with the magnitude and likelihood of the radiological risk.

2.12. Identify a list of safety functions[6] associated with each item important to safety[7] see Ref. [12] para. 2.11(b). A starting point for assessing the importance to safety of activities and SSCs is the performance of the safety analysis[8].

2.13. Paras. 6.12 and 6.13 of Ref. [1] states that all the SSCs (including software for instrumentation and control) that are important to safety shall be first identified and then classified according to their function and significance for safety. The classification of SSCs including software, in a research reactor facility should be based on the safety function(s)[9] formed by the SSCs and on the consequences of its failure to perform its function. Analytical techniques together with engineering judgment are used to evaluate these consequences. The basis of the safety classification of the SSCs, including software, should be stated and their design requirements should be commensurate with their classification. The safety functions that each SSC fulfils should be identified. A selected list of safety functions with the associated list of items important to safety for research reactors is provided in Annex 1 of Ref. [1].

2.14. The application of management system requirements should 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.

---

[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 site personnel or members of the public. Items important to safety include:

- Those SSCs whose malfunction or failure could lead to undue radiation exposure of site personnel or members of the public;
- Those SSCs that prevent anticipated operational occurrences from leading to accident conditions;
- Those features that are provided to mitigate the consequences of malfunction or failure of SSCs.

[8] Guidance on this subject is provided in Ref. [7].

[9] The safety functions are essential characteristic functions associated with SSCs for ensuring the safety of the reactor and one of the key elements in grading the application of requirements to SSCs, see also [2]. Some safety functions may not be relevant for some types of research reactor.

2.15. Grading of the application of management system requirements should be applied to the products and activities of each process. Where these activities involve modifications or experiments further categorization is suggested, see para. 7.52.

3. REGULATORY SUPERVISION

3.1. The requirements for the legislative and regulatory infrastructure for a broad range of nuclear facilities and activities are presented in Ref. [6]. Additional guidance is provided in the associated safety guides, Refs. [5] and [13] to [15]. Because of the broad range of applicability of the requirements and recommendations in these publications, not all will apply to the nuclear activities in all Member States. Each Member State should identify the requirements and recommendations that are applicable for the regulatory supervision of its nuclear programme. For the purpose of this publication, the applicable safety requirements are those for the regulatory supervision of research reactors that are presented in Ref. [1], Chapter 3 and include the:

(a) Legal infrastructure;

(b) Regulatory body;

(c) Licensing process;

(d) Inspection and enforcement programme.

APPLICATION OF GRADING

Legal infrastructure

3.2. The requirements for the legal infrastructure are established in Ref. [1], para. 3.2. The key legal requirement is that: "This legislation shall provide for the establishment and maintenance of a regulatory body 'which shall be effectively independent of organizations or bodies charged with the promotion of nuclear technologies or responsible for facilities or activities".

Application of this requirement should not be graded.

Regulatory body

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

3.4. The regulatory body should have the authority and a sufficient number of experienced staff and resources to develop and issue regulations, review and assess safety related information (e.g. from the Safety Analysis Report (SAR)), issue licenses, perform compliance inspections, take enforcement actions and provide information to other competent authorities and the public. External experts or advisory committees may assist the regulatory body in these activities[10].

3.5. Typical examples of regulatory body functions and activities that are gradable are; number of staff required, in-house technical support resources, type and frequency of compliance inspections, content and detail of licenses, and extent and depth of detail required in a facility SAR.

Licensing process

3.6. The licensing process is often performed in steps for various stages of the reactor lifetime as described in Ref. [1], paras 3.4 and 3.5 and the Appendix of Ref. [14]. For a research reactor, these stages are site approval, authorization of construction[11], authorization of commissioning, authorization of initial and routine operation and modifications, and authorization of decommissioning.

3.7. At each of these stages, regulatory evaluations are usually made and license authorizations or approvals 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 reactor. This control is accomplished by using the following processes:

(a) clearly defined lines of authority for authorizations to proceed,

(b) review and assessment of documents,

(c) issuance of permits and licenses, for the various stages,

(d) hold points for inspections, review and assessment,

(e) review, assessment and approval of OLCs,

(f) commissioning authorization,

(g) operating license,

(h) licensing of operational personnel,

[10] The IAEA provides safety review services that are available to Member States, Regulatory Bodies and Operating Organizations.

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

(i) decommissioning license.

3.9. The steps in the licensing process apply to all research reactors during all stages of their lifetime, including all proposed experiments and modifications. However, each step in the licensing process should be subject to grading, Ref. [30] paras 2.17 and 2.41 to 2.45. For example, in general there will be fewer inspections and hold points for a research reactor, new experiment or modification for a facility with a power level <100 kW, compared to those for a research reactor with a power level >5 MW.

Inspection and enforcement

3.10. The requirements for inspection and enforcement are presented in Ref. [1], paras 3.14 to 3.16. For inspections, Ref. [1] states, “The regulatory body shall establish a planned and systematic inspection programme”. The scope of this programme and frequency of inspection shall be proportionate to the potential risk posed by the research reactor and particular situations such as organizational changes or personnel turnover. Ref. [32], para. 3.14 recommends that "inspections by the regulatory body should be concentrated on areas of safety significance" and in para 3.10, that the regulatory body should use a pre-established graded approach in responding to unforeseen circumstances

3.11. Enforcement actions should also be graded since the severity and impact on safety of non-compliance with the requirements of an authorization may vary, Ref. [31], page 40. For example there may be a severe non-compliance, but on an issue of low significance. Regulatory bodies should use the graded approach that allocates resources and applies enforcement actions or methods commensurate with the seriousness of the non-compliance, escalating them as needed 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 evaluation, Ref. [5], para. 6.68.

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

(a) The safety significance or seriousness of the deficiency or violation;

(b) The complexity of the remedial, corrective or preventive action needed;

(c) The frequency of this or other violations or the degree of recidivism;

(d) Who identified and reported the non-compliance, i.e., whether the non-compliance was self-reported or identified by an inspector; and

(e) The need for consistency and transparency in the treatment of operators and licences.

4. MANAGEMENT AND VERIFICATION OF SAFETY

4.1. Ref. [1], Chapter 4 "Management System[12] and Verification of Safety" addresses the elements to be considered, the responsibilities of the operating organization and the interaction with the Regulatory Body. Guidance for establishing, assessing the performance and improving a management system is also provided in Refs. [4] and [5]. The elements of Management of Safety for an operating organization include:

(a) The establishment and implementation of safety policies and ensuring that safety-related issues are given the highest priority;

(b) Clearly defining responsibilities and accountabilities with corresponding lines of authority and communication;

(c) Ensuring that the operating organization has sufficient staff with appropriate education and training at all levels;

(d) Developing and strictly adhering to sound procedures for all activities that may affect safety, and ensuring managers and supervisors promote and support good safety practices while correcting poor practices;

(e) Reviewing, monitoring and auditing all safety-related matters on a regular basis and implementing appropriate corrective actions where necessary; and

(f) A commitment to safety culture on the basis of a documented statement of safety policy and safety objectives which is prepared and disseminated and is understood by all staff.

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

4.3. Such assessments should include consideration of the modifications and their cumulative effects, changes to procedures, ageing, operating experience, lessons learnt from

[12] In NS-R-4 the term 'Quality Assurance' was used. Subsequent to NS-R-4, Safety Guides Refs. [4] and [5] were issued which adopted the term management systems 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.

similar reactors, technical developments, radiation protection, site evaluation and re-evaluation, physical security, and emergency planning. Some specific requirements on these topics for research reactors are presented in paras 4.14 to 4.16 (for general purpose and scope) and in paras 7.108 to 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, items such as (a) through (f), in para. 4.1, is possible while still meeting the requirement that they be comprehensive. For example in item (c) grading is clearly essential in defining the staffing levels required for operations and maintenance. Staff education and training requirements should be based on the operating schedule and the complexity of the facility. The latter is determined in particular by the reactor power level, extent of isotope production and scope of experimental facilities. In addition, grading is possible in the depth and frequency of regulatory reviews, monitoring and auditing of all safety related matters

4.5. The extent of the detailed management system for a particular research reactor and experimental facilities will depend on the potential hazard of the reactor and the experimental facilities and the requirements of the regulatory body. Guidance for the preparation and implementation of a graded management system is provided in Ref. [4] paras 2.6 and 2.7 which note that grading of the application of management system requirements shall be applied to the products and activities of each process and that the grading should deploy appropriate resources, by considering;

- the 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; and
- the possible consequences, if an activity is carried out incorrectly.

4.6. The requirements of management systems should be graded to use appropriate resources, based on the significance and complexity of the SSC or activity, the hazards associated with the SSCs and activities, and the consequences if an SSC fails, or an activity is performed incorrectly. Items[13] that should be graded include:

(a) Type and content of training;

(b) Amount of detail and degree of review and approval of operating procedures;

[13] See para. 210 of Safety Guide Q1 and para. 206 of Safety Guide Q13 of Safety Series No. 50-C/SQ, IAEA, Vienna (1996).

(c) Need for and detail of inspection plans;

(d) Degree of in-process reviews and controls;

(e) Type of safety assessment;

(f) Records to be generated and retained;

(g) Level and detail of operating procedures;

(h) Reporting level and authorities of non-conformances and corrective actions;

(i) Testing, surveillance, maintenance and inspection activities;

(j) Equipment to be included in plant configuration control;

(k) Control applied to the storage and records of spare parts;

(l) Need to analyze events and equipment failure data.

4.7. Ref. [5] paras 2.37 to 2.44 also discuss the need for management system activity grading. A detailed example of where grading should be applied for the specific item (f) above (document and record management system) is reproduced from Ref. [5] in Section 7.49 of this guide.

APPLICATION OF GRADING TO THE VERIFICATION OF SAFETY

4.8. Grading is possible in the frequency and scope of self-assessments[14] and peer reviews. The frequency and scope of safety assessments and peer reviews should be graded based on the complexity and potential risk of the facility and whether they have a safety function and the importance of the safety function of the activity or SSC being assessed. A discussion on safety classification may be found in paras 6.13 and 6.14 of this publication.

4.9. Grading is possible in the number, size, composition and frequency of meetings of reactor advisory groups or safety committees. The safety committee shall advise the operating organization on relevant aspects of the safety of the reactor, the safety of its utilization, and on the safety assessment of design, commissioning and relevant operational issues and modifications. A safety committee shall also advise the reactor manager. This is discussed in Ref. [1], para. 4.15. It is acceptable to have one safety committee advising the operating

[14] 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 based on the results of re-qualification examinations. In some instances operating organizations prepare an annual report on the general performance of the reactor facility, which is a good practice. Safety committees can perform an assessment based on the report.

organization and the reactor manager. The safety committee should be independent from 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 impact of normal and accidental releases of radioactive material" (para. 5.1, Ref. [1]). Accordingly, it is necessary to assess those characteristics of the site that may affect the safety of the research reactor, to determine if there are site deficiencies and if they can be mitigated by appropriate design features, site protection measures and administrative procedures. For a graded approach the scope and depth of site evaluation studies and evaluations should be commensurate with the facility risk, Ref. [34] para 6.10. 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 in the design of a particular SSC that is readily accommodated in the overall design may allow simplification of site evaluation.

APPLICATION OF GRADING

5.2. Grading should be applied when assessing the aspects mentioned in para. 5.1, above. Ref. [16] para. 6.6 develops the basis for applying a graded approach to the various site related investigations and decisions in accordance with the potential hazard of the research reactor facility.

5.3. The site investigations should be graded, provided that an adequate level of conservatism in the design and siting criteria are provided, to compensate for reduced site hazards analysis, site investigation campaigns and simplified analysis methods, Refs. [33], [35] and [36].

6. DESIGN

6.1. Chapter 6 of Ref. [1] discusses design under the three categories below;

Philosophy of design

Paras 6.5 to 6.12 discuss the use of grading in the application of the philosophy of design listed in Ref. [1] paras 6.1 to 6.11.

General requirements for design

Paras 6.10 to 6.49 discuss the use of grading in the application of the general requirements listed in Ref. [1] paras 6.12 to 6.78.

Specific requirements for design

Paras 6.50 to 6.82 discuss the use of grading in the application of the specific requirements listed in Ref. [1] paras 6.79 to 6.171.

APPLICATION OF GRADING

The Basic Philosophy

Defence in depth

6.2. Paras 2.6 and 6.6 of Ref. [1] describes five levels of defence in depth (DiD) to prevent deviations and control them should they occur from operational states and to prevent accident conditions and mitigate their radiological consequences should such conditions occur as follows:

– FIRST LEVEL: Prevent deviations from normal operations and to prevent system failures.
– SECOND LEVEL: To control (by detection and intervention) deviations from operational states as to prevent anticipated operational occurrences from escalating to accident conditions.
– THIRD LEVEL: To provide for Engineered Safety Features (ESF) or inherent safety features, to prevent an escalation from Design Basis Accidents and to achieve a stable and acceptable state following an initiating event. One barrier for the confinement of active material is maintained.
– FOURTH LEVEL: To address beyond design basis accidents to ensure radioactive releases are kept as low as practicable. The objective is the protection of the confinement function.
– FIFTH LEVEL: Mitigation of the radiological consequences from potential releases of radioactive material.

6.3. DiD is an important design principle that should be applied in the design of a research reactor of any type or power level.

6.4. DiD should be subjected to grading (see para. 2.6 of Ref. [1]) in the sense that level 5 and in some cases aspects of level 4 may be met by inherent characteristics of the reactor, instead of through engineering safety features. In addition, the protection against various reactor transients provided by the applications of the DiD concept should be graded, based on the potential hazard of the reactor facility (see para. 6.6. of Ref. [1]). For example:

For level 4

(1) The system for confining radioactive material to prevent or mitigate its unplanned release to protect people and environment should be graded in relation to the potential radioactive releases. The reactor building for a critical assembly, with a minimal source term for instance, may require only conventional industrial standard construction while an intermediate power level reactor with potential for a substantial source term may require a confinement system with clean up capability and controlled release of effluents.

(2) Research reactors with higher source terms (from radioisotope production or higher power level) may require, inter alia, more complex confinement systems.

For level 5

While applying the DiD concept to emergency preparedness and planning, the extent of the emergency planning for a particular reactor may affect the reactor building alone, or it may be extended to the on-site zone, or further extended to off-site zones, depending on the potential for radioactive releases from the reactor that may have operated beyond the OLCs.

Safety Functions

6.5. Requirements for the design of safety systems are presented 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 design should ensure effective means to perform safety functions under normal operation, during anticipated operational occurrences and in accidents, relying on deterministic assumptions and without explicit probabilistic criteria. The design should limit the influence of failures of basic safety functions caused by human error, external events and currently conducted or foreseen future experimental arrangements.

6.7. The three basic safety functions are discussed below with respect to grading:

Shutdown Function

(1) In general, this basic safety function is not gradable, although the extent of the sub-criticality margin available and the required speed of response required may vary according to the reactor design.

(2) Some research reactors may have inherent self-limiting power levels and/or systems which physically limit the amount of positive reactivity that can be inserted in the core. This property may be used for grading the shutdown system design.

Core Cooling

(1) In general, this basic safety function is not gradable, although the extent of the cooling requirements will vary according to the reactor design. For example a forced convection cooling system to remove fission heat may be needed in one facility, in other facilities all fission heat may be adequately removed by natural convection cooling;

(2) decay heat following shutdown may be removed by forced convection cooling or natural convection cooling;

(3) some facilities may need an emergency core cooling system (ECCS) to prevent damage to the fuel in the event of a loss of flow or loss of coolant accident; others may not need an ECCS.

Confining radioactive material

Systems for confining radioactive material may be graded. This has been discussed in para. 6.4, (Level 4).

Acceptance Criteria[15] *and Design Requirements*

6.8. Basic acceptance criteria are defined by the regulatory body. Specific acceptance criteria may be defined by the designer in advance of final design and agreed by the regulatory body, Ref. [17] Chapter 4. In principle they are not graded, being fixed by the specific facility characteristics. However, the way they are met in the design is gradable as indicated in this section.

6.9. For the design of SSCs, acceptance criteria may be used in the form of engineering design requirements. These rules include regulatory requirements as well as requirements in relevant codes and standards, which may be graded on a case by case basis. This is discussed in paras 6.10 and 6.11.

General requirements for design

Classification of SSCs

6.10. The requirements for classification of SSCs are presented in paras 6.12 to 6.13 of Ref. [1], including that "appropriate design interfaces shall be considered between SSCs of

[15] Acceptance Criteria: Specified bounds on the value of a *functional* or *condition indicator* used to assess the ability of a s*tructure, system or component* to perform its *design* function.

different classes to ensure that any failure of an item of a lower safety class will not cause failure of an item of a higher safety class".

6.11. The design of a research reactor can be graded based on the classification of the reactor's SSCs in accordance with their safety significance, regulatory requirements and if, appropriate, the design maturity and its complexity. Design maturity may be reflected by a very lengthy (>20 years) and successful operating history with a high duty factor for individual reactors of unique designs. It may also be reflected in shorter periods of operation (few years each) but for numerous reactors of a similar generic type.

Codes and Standards

6.12. The requirements for codes and standards are presented in paras 6.14 to 6.15 of Ref. [1]. Codes and standards have been developed which provide guidance in the design of SSCs. These codes and standards may be regulatory, international[16], national, or even local[17]. They may 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 based on the management system procedures and/or performance requirements because of its application (e.g., an electronic component in the protection system of a research reactor).

6.13. The codes and standards used in the design of SSCs should be in accordance with the safety classification of the SSCs and the potential radiation hazard of the reactor.

Design Basis

6.14. The requirements for the design basis are presented in paras 6.16 to 6.34 of Ref. [1]. Potential challenges that the reactor may face during its operational lifetime should be taken into consideration in the design. These challenges are represented by the Postulated Initiating Events (PIEs), a selected list of events, which is included as an Appendix of Ref. [1].

6.15. The classification of the SSCs, based on importance to safety, should be utilized to establish the design requirements for the PIEs, without exceeding authorized limits. Ref. [1] para. [2.17] requires: "It shall be shown that the set of postulated initiating events selected covers all credible accidents that may affect the safety of the research reactors. In particular the Design Basis Accidents (DBAs) shall be identified". The need to carry out such a process applies for all research reactors, i.e. it is not gradable. The identification of the DBAs is gradable.

[16] Such as the IAEA Safety Standards.

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

6.16. The requirements established in Ref. [1] should be analyzed while developing the design basis for a specific reactor. As a result of the analysis, a unique design basis will be established for each specific research reactor. Grading exists in the development of the design basis in the sense that the design basis for reactors posing different potential radiation hazards will have a different set of applicable DBAs based on the specific design. The higher power reactors with significant in-core experimental facilities such as loops will require a greater number of high importance SSCs. An example of the type of grading applicable for a design basis requirement is provided in Annex 1.

Design for Reliability

6.17. The requirements for design for reliability are presented in paras 6.35-6.43 of Ref. [1]. Design for reliability may require the use of redundancy, fail-safe criteria, diversity and independence. These measures should be used in a graded way to ensure the required reliability of SSCs in accordance with the safety function to be performed by the SSCs.

In the design of a reactor, the reliability of SSCs may be related to the expected utilization of the facility and grading may be employed to achieve operational reliability. Where an automatic safety function is required, a minimum reliability requirement should be established and maintained. Depending on the type of the reactor, one or more of the following safety functions may be needed to be automatic: reactor shutdown, emergency core cooling initiation, and confinement/containment isolation.

Design for Commissioning

6.18. The requirements for the design for commissioning are presented 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 them in the design.

6.19. Grading may be applied in the selection of features to be included in the design, in accordance with the importance to safety of the considered system and the associated difficulties of conducting the commissioning tests and measurements.

6.20. While conducting commissioning tests it may be necessary to operate with transition cores. (See also Ref. [1], paras 6.44, 6.81). The approach taken to reach an equilibrium core configuration may require the use of fuels with different fissile content, or non-standard reactivity control mechanisms ("fuel or control rod followers"). These devices might be needed to reduce neutron flux peaking or to ensure the adequate cooling of these cores. To ensure fulfilment of the safety functions, one solution or another (grading) should be applied according to the operating organization needs.

6.21. For reactors expecting to reach a significant burnup during their operation, it might be necessary to utilize different types of fuels, burnable poisons or varied core geometry to reach the equilibrium core. The grading here would take account of the balance between the added complexity of using more than one type of fuel or burnable poisons and commissioning and operational requirements.

Provision for inspection, testing and maintenance

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

6.23. Where the performance of inspection, testing and maintenance takes place in radiation areas it is necessary to ensure that occupational doses to workers are within the authorized limits. This is not gradable.

6.24. The inventory of spare parts and components is gradable based on the ease of procurement from vendors and budget rules and considerations, see Ref. [5], para. 2.44. Gradable procurement process items 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 attention should be given to components of systems important to safety having high obsolescence rate (such as computerized systems or I&C systems).

6.25. Most attention should be given to components of systems important to safety having high obsolescence rate (such as computerized systems or I&C systems).

6.26. Grading may be applied in the design stage in two steps:

(1) Firstly, determine the types and frequencies of the required inspections, tests and maintenance operations taking into account the importance to safety of the SSC and its required reliability and all the effects that may cause progressive deterioration of the system.

(2) Secondly, specify the provisions that should be included in the design to facilitate the performance of these inspections, tests and maintenance operations taking into account the frequency, the radiological implications and the complexity of the inspection, test and maintenance. 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.

6.27. The difficulties encountered will vary from one reactor type to another. For example, the core support of a critical assembly could be easily inspected visually without major problems. On the other hand, the core support of a multipurpose research reactor, with a complex plenum and cooling system, may require a remote surveillance system for inspection and provision of a program of exposure of surveillance coupons to comply with dose constraints and limits as applicable. The design provisions may be much more complicated in the second case but the design should ensure that the required programme can be carried out.

Design for emergency planning

6.28. The requirements for the design for providing easier emergency plan implementation are presented in paras 6.48 to 6.49 of Ref. [1].

6.29. These specific design features include alarm systems, communication and public-address systems, illuminated escape routes, designated gathering places, on- and off-site surveillance systems with provision for remote readout, and other means to facilitate early assessment of the situation and efficient response. While the design should consider all of these features, grading is possible in many of the features. For example:

(a) The number and type of escape routes should be based on the layout, size of the facility, and potential hazards in various zones;

(b) The gathering places should be in the most convenient location while still remaining safe for persons attending;

(c) On and off site monitoring can be performed by utilizing personnel with portable devices or technology using fixed devices with remote readout;

(d) Scope and frequency of emergency drills.

Design for Decommissioning

6.30. The requirements for the design for decommissioning are presented in paras 6.50 to 6.51 of Ref. [1]. Attention should be given to keeping doses to personnel and to the public to acceptable levels and to ensuring adequate protection of the environment from undue radioactive contamination arising from the decommissioning activities.

6.31. The design features that help to achieve this goal are the use of materials in and around the core that minimize activation, and the inclusion in the layout of the facility of easy access routes and manoeuvring possibilities to facilitate the dismantling and removal of activated components, using remote handling where necessary.

6.32. Grading may be applied in the selection of the design features to meet the radiation protection goals. For example:

(1) Low power level reactors with small cores that could be easily removed and packaged may require minimal special provisions for removal and packaging of the core. Therefore the need for high-level radioactive waste facilities will be minimal.

(2) Higher power level, pool type reactors that allow for easy access and under-water handling of the core components may require design provisions for disassembling the reactor under the water. Radioactive waste facilities will be an important consideration.

(3) In any case provision must be made for handling and display of low-level waste and non-active debris from decommissioning and for storage of relatively high-level waste.

Design for Radiation Protection

6.33. The requirements for the design for radiation protection are presented in paras 6.52 to 6.59 of Ref. [1] and 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".[18]

6.34. Numerous means may be employed in the design to achieve this objective, including minimization of sources, shielding, provisions for remote handling, filtered ventilation, monitoring instrumentation for radiation and airborne contamination, provisions for ease of decontamination, and by establishing zones within the facility (classified in accordance with their potential for hazard) and access control. These and other design aspects are described in Ref. [19].

[18] Safety Series Number 110, The Safety of Nuclear Installations, (1993).

6.35. Grading may be applied in the choice of the design features for the SSCs employed to satisfy the requirements of paras 6.52 to 6.59 of Ref. [1] including their placement in the facility, by considering their feasibility and their effectiveness. In general, the scope of radiation protection design provisions included in a high power level multi-purpose facility will be more extensive and more complex than in a small reactor with limited utilization possibilities and low potential for significant exposure. (See also 6.69 of this publication).

Human factors and ergonomic considerations

6.36. The requirements for the human factors and ergonomic considerations are presented in paras 6.61 to 6.64 of Ref. [1]. The use of ergonomic principles and due consideration to human factor principles and human machine interface in the design of the main control room, experimental and other reactor systems allow grading of human factors, such as operator response requirements. Additional factors that should be taken into account in the grading considerations are the frequency of usage of a system and such pertinent human aspects as fatigue and working in stressful conditions. Some facilities will have license requirements for minimum staffing levels for reactor operators and facility support personnel (e.g. radiation protection and maintenance personnel) that must be present on the site, either at all times if fuel is in the reactor, or if the reactor is operating at power.

Provision for utilization and modification

6.37. The requirements for the design for utilization and modification are presented in paras 6.65 to 6.70 of Ref. [1]. “Research reactors are flexible in nature and they may be in various different states” and they are used for a variety of purposes.

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

(a) to ensure that each configuration of the reactor is known at all times and appropriately assessed and authorized;

(b) that new utilization and modification projects, including experiments, having a impact on safety should be subject to safety analyses and to procedures for design, construction, commissioning and decommissioning that are equivalent to those used for the reactor itself;

(c) that they should be within the authorized operating envelope or, if not, are given explicit consideration to ensure that appropriate safety measures are in place.

6.39. It is therefore necessary that these aspects of utilization are taken into account or analyzed at the design stage and appropriate provisions 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. their importance to safety, their complexity, their maturity, and the scope of analysis and of commissioning tests needed to verify their acceptability.

6.40. The expected utilization should be carefully determined at the design stage to provide sufficient flexibility for envisaged future use while avoiding unnecessary features which increase complexity. For example, provisions for future modifications may be made at a lower cost and less complexity at the design stage.

Selection and ageing of materials

6.41. The selection and ageing of materials is discussed in paras 6.68 to 6.70 of Ref. [1]. Ageing management in the design focuses on proven durable materials with sufficient design margins and provisions for testing and inspection and replacement. The extent to which these measures are utilized in the design is gradable based on the safety significance of the SSCs and the ease of replacement.[19]

6.42. Grading should also consider 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 the SSCs, as well as in provisions for knowledge management.

Provision for extended shutdown

6.43. Provision for extended shutdown is discussed in para. 6.71 of Ref. [1]. These provisions will depend on the anticipated duration of the extended shutdown. A graded approach is used in designing such provisions. All SSCs that are important to safety and which could suffer some degradation during the extended shutdown period should include provisions 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 is usually linked to the future of the research reactor.

---

[19] Proper selection of equipment and materials and design principles may be used to reduce the needs to update them due to high rate of obsolescence.

6.44. Research reactor designs normally include facilities necessary to ensure safety during shut down of the facility and these facilities may be used during extended shutdown conditions. These facilities may be graded during design.[20]

6.45. Personnel doses associated with these activities, the need to dispose of deteriorated equipment as radioactive waste, the complexity of the involved activities and the difficulties that may arise in the course of performing some activities are additional criteria to be used in these considerations. For example:

(a) Irradiated fuel assemblies from a critical assembly may be only slightly radioactive and storage requires only a safe and secure place. The fuel may be temporarily stored in a dry room and/or container.

(b) Some fuel assemblies may require wet storage with provisions for maintaining water quality and active or passive systems for removing decay heat.

(c) Similar considerations may apply when removing experiments from the reactor tank or core.

Safety Analysis

6.46. The requirements for safety analysis are specified in paras 6.72 to 6.78 of Ref. [1] and [7] and include analysis of the response of the reactor to a wide range of PIEs. The safety analysis is a fundamental part of the design process, and is the basis for determining the safety importance of the SSCs. It is also the basis for demonstrating the licensability of the proposed design.

6.47. Grading may be applied to the scope and depth of the safety analysis, Ref. [17], Section 1.3 and Annex I and Ref. [28] paras 3.1 to 3.7. The applicability of the analysis methods needs to be justified, but the extent of such justification may be graded. The use of enveloping events may also be graded. For example:

(a) The analysis required for a small facility with a relatively small number of SSCs and applicable PIEs would be much simpler than that for a large and complex

[20] For example some system requirements will be different during reactor operation and during shutdown. A graded approach may allow for use of the system reducing the extent of use of operating equipment (e.g. ventilation, cooling and water purification systems). Provisions could be taken during design to account for prolonged shutdown states. These situations often occur frequently in RRs as many are kept in extended shutdown conditions during holiday seasons due to lack of continuous utilization. Provisions to maintain subcriticality may also allow some grading of the OLCs.

facility. A low-power reactor having a limited hazard potential requires less analytical detail than a higher power level reactor.

(b) Analysis may demonstrate that for some identified PIEs there can be no release of radioactive materials from the core, eliminating the need for extensive Engineered Safety Features (ESFs) and analysis of their failure.

(c) 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.

(d) Conservative methods and criteria are a means of simplifying the safety analysis. Facilities of small potential hazard may use conservative criteria with low impact on the facility design and operation or cost.

(e) The process of development of the safety analysis report allows for the definition and refinement of the PIEs and ESFs, and is an important element to grade during the design phase.

Specific Requirements for the Design

The reactor core and reactivity control system

6.48. The requirements for the reactor core and reactivity control system are specified in paras 6.79 to 6.89 of Ref. [1]. The design requirements related to the design of the reactor core as a whole and of its individual components, (i.e. the fuel assemblies, the reactivity control system, reflectors, experimental devices, cooling channels and structural parts) are concerned with ensuring that the reactor can be shut down, cooled and held sub-critical with an adequate shutdown margin for all operational states and DBAs.

6.49. The graded approach should be applied in the design of the core by considering the effects that these components must meet in the course of their intended life in the core. The effects such as integrated neutron flux, thermal, mechanical and chemical stresses on various materials and fuel assembly types are generally well known. The extent of analyses and experiments needed to demonstrate the acceptability of a particular design may be substantially smaller than that which is required for reactors which make use of new types of fuel assemblies, and/or novel experimental setups. A similar situation may be found in relation to the reactor power; smaller reactor powers shown to present smaller risk potential may need substantially less extensive analysis, and simplified conservative criteria.

6.50. Grading can be used in the design of the reactivity control system. Different reactivity mechanisms can be used according to the reactor design (e.g. burnable poisons in fuel assemblies and soluble neutron absorbers).

The reactor shutdown system

6.51. The requirements for the reactor shutdown system are specified in paras 6.90 to 6.94 of Ref. [1]. The reactor shutdown system fulfils a crucial safety function for all research reactor types and sizes. Therefore, all the design requirements specified in paras 6.90 to 6.94 of Ref. [1] should be fully met.

6.52. Grading may be applied in deciding how many shutdown channels are needed and the extent of instrumentation required for monitoring the state of the shutdown system, Ref. [17] Chapter 3.

6.53. A second and diverse shutdown system should be considered for reactors conducting experiments with major safety significance that could affect, in the event of an accident, the first shutdown system, unless inherent self limiting properties of the core/fuel design prevents a damaging reactivity excursion under all foreseeable reactor states.

The reactor protection system

6.54. The requirements for the reactor protection system are presented in paras 6.95 to 6.105 of Ref. [1]. The reactor protection system is required to automatically initiate the required protective actions for the full range of identified PIEs to terminate the event safely. Consequently, the system has to be reliable, utilizing, as required, redundancy and independence 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 are no identified PIEs requiring automatic shutdown, manual operator action could be considered sufficiently reliable, as explained in para. 6.96 of Ref. [1]. A high level of confidence in this determination is required.

6.55. Grading may be possible in the reactor protection system in the sense that two different reactors may face different PIEs, 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:

(a) at seismically active sites, a seismic sensor may be required to shutdown the reactor, while at other sites, such protection is not needed;

(b) initiation of emergency core cooling may be needed in certain reactors while in others it is not needed (see paras 6.7 and 6.59).

The reactor coolant systems and related systems

6.56. The requirements for the reactor coolant systems and related systems are specified in paras 6.106 to 6.119 of Ref. [1]. Cooling is one of the basic safety functions discussed in Sec. 6.65 of this publication. The coolant system is required 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, but also, after shutdown, under a range of anticipated operational occurrences, postulated accidents and Design Basis Accidents (DBAs) that involve loss of flow and loss of coolant transients. Grading can be used in the design of the cooling system. This 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 circulation is adequate for both heat removal under full power operation as well as decay heat removal for some small reactors.

Means of confinement

6.57. The requirements for the means of confinement are specified in paras 6.120-6.130 of Ref. [1]. Confinement is one of the basic safety functions discussed in para. 6.7 of this publication. Means of confinement are provided to prevent or mitigate an unplanned release of radioactive material in operational states or in accident conditions. The basic design requirement is to ensure that the release to the environment does not exceed acceptable limits for all accidents taken into account in the design. Consequently, the potential release from the reactor will determine the design criteria for the confinement. An example of the use of these considerations as a basis for grading is presented in para. 6.4 of this publication.

Experimental devices

6.58. The requirements for experimental devices are specified in paras 6.131 to 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.

6.59. 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 I&C system should be subject to grading. The monitoring of the experimental devices from the control room(s) is also subject to grading.

6.60. Experimental devices should therefore be designed taking all the anticipated hazards into account, and an analysis should be performed to demonstrate the adequacy of the design. Limiting conditions for safe operation should be prepared for each device and incorporated into the facility's OLCs.

6.61. Grading should be applied to the design, analysis, and authorization process, in accordance with the types and magnitudes of the anticipated hazards, taking into account the complexity of the experiment and familiarity (based on experience) with its performance.

Instrumentation and control

6.62. The requirements for the instrumentation and control (I&C) are specified in paras 6.136 to 6.144 of Ref. [1]. The basic (I&C) design requirements in this respect are to include in the design sufficient instrumentation for the purpose, with reliability commensurate with the importance to safety of the system. The grading of the I&C systems should be based on a careful definition of the Design Basis. Due consideration should be given to the maintainability of the system and its associated cost.

6.63. Grading should be performed in determining the types, places, and number of measurements taken of reactor parameters such as temperature, pressure, flow, pool/tank water level, gamma radiation and neutron flux. System requirement specifications covering all operational states and accident conditions, should provide for grading of the I&C systems. A typical example is the measurement of pressure drop across the core. This is a safety measurement implemented in many reactors in order to detect reduced flow through the core (either due to a by pass or to a blockage): this measurement in general is not needed in a critical assembly or in a reactor operating in a natural convection cooling mode.

6.64. Redundancy is another means of grading I&C systems. Two-out-of-three redundancy is often used in reactors that need to operate continuously to minimize spurious trips and to allow for I & C testing and/or maintenance on power. On the other hand, a reactor operated a few hours per week or intermittently may not need such features because spurious shutdowns may be less of a problem and so a one-out-of-two redundancy may be selected. Excessive redundancy increases cost and complexity.

6.65. The level of reliability as well as the accuracy required for measurements of the relevant parameters will depend on the importance to safety and process requirements.

6.66. The degree of automation required for control system actions can be graded.

6.67. The I&C system should monitor reactor parameters and allow for appropriate response for anticipated operational occurrences and DBAs. If analysis shows that in some situations the main control room can not be occupied, then a supplementary control room, separated and functionally independent[21] from the main control room, should be provided in the design. The design and equipment of this secondary control room is also gradable according to the reactor characteristics and foreseen accident conditions. If the need for an emergency control room is confirmed, then there should be an analysis of its operational requirements. Typical features that may be included according to documented requirements are: radiation monitors, fire detection and actuators of extinguishers, communication means, ventilation control, scram and/or safe shutdown features, operation of experimental devices, operation of cooling or water systems.

[21] This means that this suplementary control room should not be slave to the main control room for any of its equipment and features.

6.68. A complex and costly human machine interface in a low power level research reactor facility may not be justified.

Radiation protection systems

6.69. The requirements for the radiation protection systems are specified in paras 6.145 - 6.148 of Ref. [1]. To achieve the basic requirement of para. 2.2 of Ref.[1] as discussed in paras 6.33 to 6.35 of this publication a wide range of radiation protection systems are provided in the design "to ensure adequate monitoring for radiation protection purposes in operational states, DBAs and, as practicable, Beyond Design Basis Accidents (BDBAs)". (para. 6.145 in Ref [1]). Para. 6.145 of Ref. [1] lists the radiation protection systems used in research reactor facilities and the purposes they serve. All these systems are likely to be required for research reactors. Grading may be applied in determining the level of adequacy for a specific facility.

6.70. For example:

(a) A high power level facility should require a wide distribution of fixed instrumentation and numerous portable instruments.

(b) A reactor with various experimental devices: beam tubes and neutron guides, neutron activation analysis (NAA), and radioisotope production (RIP) facilities should require neutron and gamma monitors for the beam tubes and neutron guides and instruments, gamma monitors in the NAA facility and in the RIP handling systems as well as equipment for contamination monitoring. A low power reactor without beam tubes used only for teaching purposes would need only limited and basic equipment such as gamma monitors at the open pool end or in the control console and contamination monitors.

(c) For high power level reactors supplementary monitoring displays outside the control room should be required for displaying and recording radiation conditions at specific locations in the facility for normal operational and accident conditions (large range monitoring).

(d) Such additional radiation monitoring locations may not be required for very low power level facilities (< 50 kW).

Fuel handling and storage system

6.71. The requirements for the fuel handling and storage system are specified 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. The main concerns are the prevention of inadvertent criticality and fuel damage from mechanical impacts, corrosion or other chemical damage events. Requirements related to the prevention of damage and to ensuring physical security may be equally applicable to many research reactors, the only difference being that of scale.

6.72. The application of the requirements to different reactors may be graded in several aspects according to the design and utilization program. For example:

(a) Some reactors may need an irradiated fuel storage pool, separate from the reactor pool;

(b) Some low power level reactors may accommodate irradiated fuel in the reactor pool;

(c) Some research reactors may use different types of fuel assemblies for research or testing purposes and may require special storage places for temporary storage of these assemblies;

(d) Requirements for decay heat removal may vary requiring different provisions in the design for decay heat removal;

(e) Corrosion control will depend on the fuel design, storage conditions and residence times;

(f) Criticality concerns will vary amongst the various types and quantities of reactor fuels, but adequate care should be taken to ensure sub-criticality in any foreseen situation.

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

6.73. The requirements for the electric power systems are specified in paras 6.155 to 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.

6.74. Grading may be applied in the design of the power supply system and the emergency power supply. Considerations relevant for grading include: the type and number of safety functions, and ESFs, for which emergency power is required. The reliability requirements may be different for different reactors, for the utilization programmes for the same reactor and for the needs of experimental devices. Consideration should be given to the need for emergency power supplies to back up the offsite power supply system. Grading would consider the number, size, and reliability of any necessary emergency power supply systems. Examples would include the control system, protection system, monitoring system and decay heat removal.

(1) A reactor may or may not need forced circulation cooling after shutdown. The emergency power supply requirement and the time needed after shutdown to operate this system determine the specifications of the emergency power supply system. Depending on the reactor power, power density, duty cycle this time could be hours, days, or weeks, giving rise to reliability considerations. Reliability requirements, in general, may call for a degree of redundancy and separation in the design that depends on the DBAs postulated for the facility.

(2) The reactor power will determine the extent of process water requirements for power operation and decay heat removal. Similarly, the use of process air may be required for some special air-cooling functions, for experimental devices, or to operate pneumatic valves for safety or control purposes.

(3) Ventilation system operation is normally required during reactor operation or shutdown. Air conditioning requirements may be vital for computerized controls and to prevent humidity and corrosion problems in process areas, depending on not only the reactor but also on local climatic conditions. Similarly the building service requirements such as cranes, may be essential for certain critical operations, although these may not be required frequently.

Radioactive waste systems

6.75. The requirements for the radioactive waste systems are specified in paras 6.162 to 6.166 of Ref. [1]. Radioactive materials are generated in the research reactor by fission and by activation of reactor internals, experiments and operational waste[22].

[22] 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 cooling; water from the dehumidifiers; water used for cleaning and decontamination activities; waste from laundry operations; drainage from hot cells and laboratories; lubricants used in machinery from active zones. GASEOUS: from the reactor tank or pool; from the cooling systems and from irradiation facilities; gases produced by the radioactive material created during the reactor operation; noble gases; tritium.

6.76. The grading of radioactive waste systems should be related to the type and quantities of radioactive waste generated in the specific reactor facility.

6.77. The design should aim to minimise the generation of radioactive waste, to provide for its containment and for its appropriate safe handling and removal, taking into account all stages in the lifetime of the facility, and under all operational states including decommissioning.

6.78. Such grading considerations should be compatible with the safety analysis and regulatory requirements, for example:

(a) retention tanks may or may not be required to detain radioactive effluents for decay before their removal or release;

(b) a spill of water from the pool or tank of a research reactor may lead to only low level surface contamination, provided proper water chemistry and purity has been maintained;

(c) a spill of a similar amount of heavy water from a heavy water reactor may involve a significant release of tritiated water. For this, as well as for economic reasons, a high degree of leak-tightness is required in heavy water reactors.

6.79. On the design of radioactive waste systems for research reactors, the requirement to handle, store, transport and dispose of radioactive waste and the control of solid, liquid and gaseous effluent discharges are not gradable.

Buildings and structures

6.80. The requirements for the buildings and structures are specified in paras 6.167 to 6.169 of Ref. [1]. The requirements related to the design of buildings and structures depend on their intended safety functions and their importance to safety.

6.81. The design basis for buildings and structures may be graded by examining their safety function. For example, the reactor building may be required to act as a confinement barrier and designed accordingly. However, different reactors may require different degrees of leak-tightness, which should be determined in accordance with the reactor's safety analysis.

6.82. Careful design of building and structures should help in the application of grading in other systems (or avoid costly refurbishment later). For example:

(a) Separation of areas according their potential hazard and the use of adequate structural material can simplify (and consequently reduce the grade required of) other SSC or activities such as: radioactive waste, design for radiation protection, design for emergency, fire protection as well as operational costs.

(b) The architecture of the building should facilitate the provision of the control room and, where appropriate, an emergency control centre.

(c) Good site evaluation helps to reduce unnecessary conservatism in engineering requirements for building and structures in relation to the protection against external events, Ref. [17] para. 2.2.1 which may have a high impact in relation to the total cost of the reactor facility.

Auxiliary systems

6.83. The requirements for the auxiliary systems are specified in paras 6.170 to 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.

6.84. For example, a leak from the demineralizer system may have the potential to drain the pool and thus jeopardize the reactor. Adequate measures should be employed to prevent such occurrences.

6.85. Those auxiliary systems that do not have an effect on nuclear safety may 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. Ref. [1], Chapter 7 includes fifteen operational topics and the grading aspects of each of these are discussed in this chapter.

ORGANIZATIONAL PROVISIONS

7.2. The organizational requirements for a research reactor are presented in paras 7.1 to 7.26 of Ref. [1]. Guidance on meeting these requirements is presented in Ref. [15].

Application of grading

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 to those at a high power level, multi-purpose research reactor. For example, the direct responsibility and the necessary authority for the safe operation of the reactor should be assigned to the reactor manager. This responsibility should not be graded. However, the manner in which the associated functions are performed may be graded in accordance with their safety significance, maturity and complexity[23].

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, grading may permit different operational structures while maintaining the same functionality of those structures. For example:

(a) A research reactor in a Member State (MS) with a limited nuclear programme may need a large and complete in-house capability (such as a technical support group, quality control, a large inventory of spare components, expertise in isotope production, and maintenance personnel). A similar research reactor in a MS with a large infrastructure and nuclear programme may not need such a large in-house capability because support could be easily obtained.

(b) Grading should be applied, inter-alia, in the following areas:

  i. Number and duties of operating personnel. For a low potential hazard reactor, an individual may be assigned multiple duties. However, Ref. [1] requires that duties, responsibilities, experience and lines of communication be documented; this requirement is not gradable;

  ii. Membership of and meeting frequency of a safety committee(s) (see para 4.9 above);

  iii. Production and periodic updating of the Safety Analysis Report (see discussion of the licensing process in para. 3.6 to 3.16 of [1]);

  iv. Procedures (see paras 7.22 to 7.26);

  v. Radiation protection program (see para. 7.54);

  vi. Emergency plan and procedures (see para. 7.44 to 7.47);

  vii. Training, re-training and qualification program (see paras 7.5 to 7.8);

[23] A reactor manager of a large research reactor may have under her/his direct authority the Technical Support Group, a Safety Analysis Group, a Training Group, and a QA 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 be the person responsible for the implementation of all the relevant programs and projects and the safe operation of the reactor.

viii. Management system (see para. 4.1);

ix. Maintenance, periodic testing and inspection program (see paras 7.26 to 7.34 of this publication).

TRAINING, RETRAINING AND QUALIFICATION OF OPERATING PERSONNEL

7.5. The training, retraining and qualification requirements for research reactor operating personnel are presented in paras 7.11 to 7.28 of Ref. [1]. Guidance on meeting these requirements is presented in Ref. [12].

Application of grading

7.6. Training, retraining and qualification requirements for the staff of the research reactor 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 the operating personnel. The educational level, experience and operational requirements (such as minimum operational activity per year) for the various reactor positions and the contents and duration of training may be graded in accordance with the above criteria, Ref. [12], para 1.10.

7.7. The structure of the operating organization may affect the way of implementing the programme but should not affect its functionality. It may be appropriate to supplement in-house capabilities with external provisions.

Provisions should be put in place for the assessment of the training needs and their fulfilment, including retraining, qualification, and operational experience (such as minimum operational activity per year) 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 staff. The requirement that there be adequate training and that it be implemented is not gradable. The nature and details of the training is gradable Ref. [12] para. 5.13. Reauthorization after absences may 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.

7.8. The requirements for research reactor OLCs[24] are presented in paras 7.29 to 7.41 of Ref [1]. Guidance for the preparation and implementation of OLCs is presented in Ref. [9].

Application of grading

General

7.9. Since the OLCs are based on the reactor design and on the information from the SAR concerning conduct of operations, grading will have already taken place as discussed in other sections of this publication.

Safety Limits

7.10. The need for establishing 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 limit may be gradable.

Safety System Settings

7.11. For each safety limit, there should be at least one safety system instrument used to monitor parameters and cause an action (e.g., shut down the reactor) to preclude approaching the safety limit. The set point should be established to provide an acceptable safety margin between the point of the action and the safety limit. For safety actions of particular importance, such as neutronic trips (scrams) redundant systems should be employed. The analysis to establish a suitable safety margin may be graded along with the level of redundancy.

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

Limiting Conditions for Safe Operation

[24] The OLCs are a set of operating rules, which normally include limits on operational parameters and safety system settings to ensure that safety limits are not violated.

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 the normal operating values and safety system settings during start-up, operation, shutting down and shutdown. Appendix I of Ref. [9] provides a list of operational parameters and equipment to be considered in establishing limiting conditions for safe operation. Appendix I recommends selecting only the appropriate items, in accordance with the type of reactor and conditions of operation. Grading should also be applied in the type of analysis performed in establishing a limiting condition for safe operation, based on the selection in accordance with the type of reactor and conditions of operation.

Requirements for maintenance, periodic testing and inspection

7.14. In order to ensure that safety limits and limiting conditions for safe operation are met, the relevant SSCs should be maintained, monitored, inspected, checked, calibrated and tested in accordance with an approved surveillance programme. Surveillance requirements specify the frequency, scope and acceptance criteria for each SSC. Grading should be used in establishing these requirements based on the importance to safety and reliability of the SSC. Additional information is provided in paras 7.26 to 7.33.

Administrative Requirements

7.15. Administrative requirements include those for the organizational structure and responsibilities, minimum staffing, training and retraining, safety review and verification, procedures, records and reports, and event investigation and follow up. The grading which may be possible in some of these activities is discussed in paras 7.3 and 7.4.

7.16. The requirement for action after a violation is not gradable. The nature of the action is gradable depending on the severity of the violation, i.e. whether a safety limit or a LCO has been exceeded.

COMMISSIONING

7.17. The safety requirements for commissioning a research reactor are presented in paras 4.5, and 7.42 to 7.50 of Ref. [1]. Guidance for research reactor commissioning is presented in Ref. [8].

Application of grading

7.18. The commissioning process itself cannot be graded in that all SSCs, activities and experiments should be commissioned. However, grading may be applied to the commissioning programme in:

(a) organizational structure;

(b) preparation of procedures;

(c) number of hold points and tests;

(d) documentation;

(e) reporting.

7.19. While grading may be applied in the number of hold points required there should always be a hold point for tests prior to fuel loading (pre-operational tests). A graded approach to testing should be adopted (Ref. [8] Appendix A.2). The extent and type of tests to be performed being determined on the basis of their importance to safety of each item and the overall hazard potential of the reactor.

7.20. The principles for the initial approach to criticality, reactivity device calibrations, neutron flux measurements, determination of core excess reactivity and shut-down margins, for power raising tests and containment/confinement system testing should be similar for all research reactors.

OPERATING PROCEDURES

7.21. The requirements for research reactor operating procedures (OPs) are presented in paras 7.51 to 7.55 of Ref. [1]. Guidance for preparation of OPs is presented in Ref. [9]. Appendix II of Ref. [9] presents an extensive list of OPs for a research reactor.

Application of grading

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 OLCs. In addition, grading will have been employed in preparation and implementation of the management system programme which governs the format, development, initial and periodic review, control, training on the use and implementation of procedures.

7.23. The list of operating procedures presented in Appendix II of Ref. [9] should be graded for applicability to a specific reactor. Consequently, the number of procedures depends upon the research reactor and will be smaller for simpler reactors with low potential hazard.

7.24. Grading should be applied to the staff training in the use of the procedures. However personnel using the procedures should be thoroughly familiar with them and proficient in their use.

7.25. While all procedures should be prepared, reviewed and approved based on criteria established by the operating organization and regulatory requirements, operating procedures may be graded based on 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 safety implications of an error in the regeneration process are low. Consequently, the procedure itself may be simplified.

(b) By contrast, an operating procedure that is developed for an application in which an error has the potential for safety significance and causing a violation of the OLCs would be more detailed. An example would be the procedure for regeneration of an ion-exchange system for the primary cooling water purification system. While it may involve the same basic technology as above, the safety implications of an error could be much more significant (e.g., an error which allowed resin to enter the primary cooling water and hence into the reactor core). Design features and/or procedural arrangements should therefore take into account the greater hazard from mis-operation of this system.

(c) While the principles commissioning procedures will be similar for all research reactors, specific procedures will vary between reactors. Similarly normal operational procedures and emergency procedures will vary.

(d) Procedures required for reactor utilization changes, special fuel tests, experiments and other special applications are often complex and infrequently used. Since these activities will often impact safety, development, review and approval of procedures for these activities should follow the same course as that for other procedures of safety significance.

INSPECTION, PERIODIC TESTING AND MAINTENANCE

7.26. The requirements for research reactor maintenance, periodic testing and inspection are presented in paras 7.56 to 7.64 of Ref. [1]. Guidance for maintenance, periodic testing and inspection is presented in Ref. [10].

Application of grading

7.27. Grading can be applied to the frequency of maintenance, periodic testing and inspection based on experience and on 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 equipment to be maintained, to the complexity of the maintenance operation and to the experience of the maintenance staff and their familiarity with the systems to be maintained. Grading of procedures was discussed in paras 7.21 to 7.25.

7.29. The period that a SSC may be out of service while reactor operation continues is usually stated in the OLCs for the research reactor and may be graded. As a result any outage time may not be acceptable for automatic shutdown systems, while outage times up to days may be acceptable for other systems (e.g., for purification system monitoring the primary coolant pH). The allowable outage time will depend on the extent to which safety is impacted, or the ease in applying compensatory measures.

7.30. In a similar way, the frequency for periodic testing may be graded. A balance is necessary between the improvement in unrevealed fault detection due to more frequent testing and the risk that testing may be performed incorrectly and leave the SSC in a degraded state. The testing frequency could also be increased to the point where test failures cause more frequent failures so it should be recognized there is always an optimum test frequency. This also applies also for periodic maintenance.

7.31. At times it may become necessary to perform maintenance, periodic testing and inspection in radiation areas or on components which are radioactive. While the procedure for the inspection, periodic testing and maintenance may have been graded, controls should be in place to ensure that radiation exposures of workers are within the prescribed limits. The radiation protection control measures may be graded based on the potential for occupational exposure.

7.32. When maintenance, periodic testing and inspection of an SSC is uncomplicated and operating experience indicates a high reliability a review and re-grading of the activity leading to a change in the procedure may 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 some maintenance, periodic testing and inspection activities for grading purposes, it may be concluded that the required activities are highly specialized involving complex and sophisticated techniques. Such activities are often performed by contracted, external experts. This should be carefully considered by the Operating Organization to ensure that external support is secured and that resources will be available throughout the operational life of the facility. The use of external contractors for performance of maintenance, periodic testing and inspection is discussed in Ref. [10].

CORE MANAGEMENT AND FUEL HANDLING

7.34. The requirements for core management and fuel handling are presented in paras 7.65 to 7.70 of Ref [1]. Guidance for core management and fuel handling is presented in Ref. [11].

Application of grading

7.35. Low risk research reactors having a power rating of 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 may 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 a particular research reactor Ref. [11], para. 1.10 and 2.2.

7.36. Future core and fuel handling requirements should be anticipated as far as possible during reactor design. These requirements should be analyzed in the Safety Analysis Report, confirmed during the commissioning of the reactor and incorporated into the OLCs. Implementing a change, via a change control process, at some later date may be graded by relying on the work already performed.

7.37. In other cases the core management and fuel handling requirements may not have been anticipated, requiring a determination of the importance to safety of the proposed new core management and fuel handling. In this case, the core management and fuel handling should be treated as a reactor modification.

7.38. Ref. [20] 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. Based on the safety significance, grading in the analysis and verification associated with the proposed core management and fuel handling activities may be possible. See also Reactor Utilization and Modification in paras 7.49 to 7.52.

FIRE SAFETY

7.39. The requirements for fire safety are presented in paras 6.22 to 6.25 and 7.71 of Ref. [1]. Guidance for fire safety is presented in Ref. [21].

Application of Grading

7.40. Since fire protection is important to safety, all the requirements are safety significant. However, in the SAR the potential fire hazards should be discussed and indication given of their relative importance (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.

7.41. Grading the operational fire protection may be facilitated by provisions incorporated into the design and by siting considerations.

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

EMERGENCY PLANNING

7.43. The requirements for emergency planning are presented in paras 6.20 and 7.72 to 7.78 of Ref. [1]. Guidance for emergency planning and response is presented in Ref. [22].

Application of grading

7.44. The emergency plan and its implementing procedures are based on the DBA analyzed in the SAR as well as those additionally postulated for the purposes of emergency planning (BDBA). These analyses will allow the development of a source term to be used for emergency planning. For some research reactors, it may be demonstrated that health effects in the population 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 an individual research reactor is required for preparing an appropriate emergency plan.

7.45. In conformance with the concept of a graded approach, Ref. [18] defines the concept of nuclear and radiation emergency categorization (paras 3.6 to 3.7) which provides a basis for developing optimized arrangements for preparedness and response. This concept requires that an emergency planning zone be considered. The 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 reactors, 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 that warrant or contamination that warrants urgent protective action on the site, or for which such events have occurred in similar facilities.

Most research reactor facilities fit into Category II or III. This grading may lead to an emergency planning zone as small as the reactor building itself or large enough to extend off-site.

7.46. 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 may be possible, inter-alia, in the following areas:

(a) the organization needed to carry out the emergency plan;

(b) the emergency planning zone;

(c) the identification and categorization of the emergency;

(d) notification requirements for informing authorities;

(e) the amount, nature and storage location of the equipment needed to survey and monitor people and the environment during the emergency;

(f) the number, identity, training of and agreements with off-site agencies (police, fire service, medical transport) that are involved. Although the emergency may not have an off site impact, it is generally prudent to establish contact with off site authorities (e.g. police, fire services, medical transport, medical treatment) to ensure their concurrence upon request;

(g) the time scale envisaged for going through the various phases of the emergency;

(h) the types and the extent of the exercises and drills;

(i) the nature and amount of other resources needed to handle the emergency situation; and

(j) the facility's proximity to populated areas can significantly increase or decrease the grading in scope and the content of the emergency planning.

RECORDS AND REPORTS

7.47. The requirements for records and reports are presented in paras 7.81 to 7.84 of Ref. [1]. Guidance for maintenance of records and preparation of reports is presented in Ref. [4] para. 5.21 and 5.22 and Ref. [5], para. 5.35 to 5.49 and Annexes I, II and III.

Application of grading

7.48. Consistent with the purpose for which reports are prepared and records are kept, Ref. [5], para 2.44 lists specific examples of where a graded approach for a records management system can be applied:

– 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.

REACTOR UTILIZATION AND MODIFICATION

7.49. The requirements for reactor utilization and modification are presented in paras 7.85-7.92 of Ref. [1]. Guidance for reactor utilization and modification is presented in Ref. [20].

Application of grading

7.50. The operating organization should develop criteria for classifying a proposed experiment or modification in accordance with its importance to safety. The resulting classification should then determine the types and extent of the analysis and approvals to be applied to the proposal.

7.51. So far as possible, future utilization or modification requirements should have been anticipated during reactor design analyzed in the Safety Analysis Report, confirmed during the commissioning of the reactor and incorporated into the OLCs. Implementation at some later date may be graded relying on the work already performed.

7.52. In other cases an experimental or modification requirement may not have been anticipated, requiring a determination of its safety significance. Ref. [11], para. 1.11 and Ref. [20], Annex I provides guidance for categorization for the treatment of modifications, according to their hazard potential using a four category system:

(i) Changes that could have major safety significance;

(ii) Changes that could have a significant effect on safety;

(iii) Changes with apparently minor effects on safety;

(iv) Changes having no effect on safety.

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

(i) Involve changes in the approved operational limits and conditions; or

(ii) Affect items of major importance to safety; or

(iii) 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.

RADIATION PROTECTION

7.53. The requirements for radiation protection are presented in paras. 7.93 to 7.107 of Ref. [1] and in the Basic Safety Standards, Ref. [27]. Guidance for radiation protection is presented in Ref. [19].

Application of Grading

7.54. While the content of the radiation protection program at a research reactor will depend on its design, power level and utilization, many aspects of the program will be similar at all research reactors.

7.55. The application of grading to the radiation protection programme should be consistent with the reactor's design and with its utilization Ref. [20], paras 1.5 and 1.9. The environmental monitoring programme will depend also on the location of the reactor (in a densely populated site the environmental monitoring programme will generally be more extensive).

7.56. It should be noted that a critical assembly may present a higher hazard of external radiation exposure for operating personnel than a high power reactor, dedicated to radioisotope production, but the latter may have a higher hazard of contamination causing internal exposure.

7.57. Working areas within a reactor are classified (graded) into supervised and controlled areas according to the magnitudes of the expected normal exposures, the likelihood and magnitude of potential exposures, nature and extent of the required radiological protection procedures. Controlled areas themselves are subjected to classification (grading) according to measures or expected radiological level, Ref. [19] paras. 1.7 to 1.9. For a high power research reactor, it may be necessary to further grade the controlled area into different levels, for example, controlled area levels I, II and III. Residence at controlled area level II may require specific procedures (in addition to those required for area level I) that in same cases require the use of protective garment, equipment, or tools. Controlled area level III will normally be closed by a physical barrier (e.g. an airlock door) that is opened only by authorized workers. Furthermore, opening of this door during reactor operation may result in an automatic reactor shutdown action. For low power research reactors, controlled area levels III or II, may not be needed.

7.58. Ref. [19] provides general recommendations concerning the nature and scope of an operational radiation protection programme. The application of these general recommendations may be graded based on the above assessment to determine the nature and scope of the elements of the specific operational radiation protection programme.

SAFETY ASSESSMENTS AND AGEING RELATED ASPECTS

Safety Assessments

7.59. The requirements for safety assessments are presented in paras. 7.108 to 7.110 of Ref. [1]. Guidance for performing safety assessments is presented in Refs. [23].

7.60. Chapters 4 and 7 of Ref. [1] discuss the requirements for management and verification of safety and discuss safety assessment throughout all the stages in the lifetime of the reactor. Grading in the management and verification of safety has been discussed in Chapter 4 of this publication.

Application of grading

7.61. Ref. [28] paras 3.1 to 3.7 specifies 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 has to be consistent with the magnitude of the possible radiation risk arising from the facility or activity.

7.62. The application of a graded approach should vary according to the stage of the safety assessment as the facility potential radiation risks 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 required. The decommissioning stage should require significantly less detail and analysis than the operational stage. The scope, level and detail of the safety assessment and the resources required to produce it should be adjusted accordingly.

7.63. The main factors influencing the radiation risk and thus the level of detail used for a safety assessment at the operational stage would be the; predicted or historical operational releases and doses to on-site staff and public; consequences of anticipated operational occurrences and accidents with respect to facility SSCs and doses to staff and public, and potential consequences (dose and SSC damage), from low probability events with potentially high consequences.

7.64. The graded approach should also be applied to the requirements for updating safety assessments, Ref. [28], para. 5.10. The frequency of safety assessments should depend on the number and extent of modifications, changes to procedures, modifications of safety significance, evidence of component ageing, developments from operating experience and historical unplanned incident experience, changes in site conditions and new requirements from regulatory concerns. In addition, grading could depend on the experience gained in similar facilities. Typically, for a reactor with more than 5 to 10 years of demonstrated operational maturity, a periodic safety assessment for the overall facility every 5 years would be appropriate. With regard to reactors with more than 20 years of operation more emphasis on safety assessment of component ageing would be expected, particularly with regard to control systems.

Ageing related aspects

The requirements for ageing related aspects are presented in paras. 7.108 to 7.110 of Ref. [1]. Guidance on ageing related aspects of research reactors is presented in Ref. [24].

Application of grading

7.65. While selection of materials and the effects of the operating environment on their properties have to be accounted for in the design of all research reactors, the use of a graded approach can be made in developing the material surveillance and the ageing management programmes during the operating life of the facility. In most research reactors, it is feasible to inspect the materials periodically and replace the components, if needed.

Extended shutdown

7.66. The requirements for the safety of a research reactor in extended shutdown are presented in paras 6.71 and 7.111 to 7.112 of Ref. [1]. Further information on extended shutdown is provided in para. 6.43 to 6.45 of this publication and Ref. [25].

Application of grading

7.67. The operating staff of a reactor in extended shutdown may be smaller in number than that for an operating reactor. However, a large reduction in the overall reactor facility staff level may be inappropriate. Concerns such as the loss of operating experience and knowledge of the facility which will be necessary for the restart of the facility may mitigate against a large reduction in staff.

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 the extended shutdown, and the extent of relief from the normal operating regime.

8. DECOMMISSIONING

8.1. The requirements for decommissioning are presented in paras 8.1 to 8.8 of Ref. [1]. Further guidance can be found in Ref. [26].

APPLICATION OF GRADING

8.2. The requirements related to decommissioning are applicable to every research reactor. However, the effort associated with fulfilling them (e.g., in the preparation and review of the plans and procedures) may be graded. The grading may include 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 protection measures and the scope of surveillance activities during decommissioning.

8.3. Decommissioning of a research reactor facility can be graded based on the activity and type of the radioactive materials and sources in the facility, the degree of complexity of dismantling operations, the availability of experienced personnel and of proven techniques and the means to employ them.

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

(a) critical assemblies may not represent a substantial concern from the radiological or radioactive waste point of view although it would be necessary to monitor for activation products before commencing disassembly, and the activities would be conducted without special tools or highly qualified personnel. In many cases, the building and other installations may be used for different purposes.

(b) reactors of low power may have some radiological concerns, that could be easily handled by the competent radiation officers of the operating organizations; a predisposal management plan should be elaborated, usually a small number of high activity level components are found (such as the core support, nuclear detectors, control rods and experimental devices from the core). The buildings should be assessed; sometimes walls and ventilation systems are contaminated as well as the floors. In some cases appropriate decontamination of the tank would allow to release it for other uses.

(c) In higher power reactors the secondary cooling system, process air system, radiation protection equipment, instrumentation and control systems for example are usually not contaminated and could be either disposed or prepared for other uses. Ventilation and radiation monitoring systems are kept working for their use during decommissioning activities.

(d) Release from regulatory control may not be appropriate in some cases as the country may need the infrastructure of a research reactor for other uses, such as: storage of radioactive sources, radioactive waste installation, or a gamma irradiation facility.

8.5. The operating organization should select a decommissioning option by considering a wide range of issues, including the resources available at the time of implementing the decommissioning. These options present opportunities for grading (e.g., based on the present state of the installation and possible future uses of the decommissioned installation or site).

8.6. The regulatory review of the decommissioning plan should follow a graded approach Ref. [29], para. 3.11 and consider the phases in the fuel storage facility lifetime.

REFERENCES

1. INTERNATIONAL ATOMIC ENERGY AGENCY, Safety of Research Reactors, Safety Requirements No. NS-R-4, IAEA, Vienna (2005).

2. INTERNATIONAL ATOMIC ENERGY AGENCY, IAEA Safety Glossary, Terminology used in Nuclear Safety and Radiation Protection, 2007 Edition, Vienna (2007).

3. INTERNATIONAL ATOMIC ENERGY AGENCY, Fundamental Safety Principles, Safety Standards Series No. SF-1, IAEA, Vienna (2006).

4. INTERNATIONAL ATOMIC ENERGY AGENCY, The Management System for Facilities and Activities, Safety Requirements, GS-R-3, IAEA, Vienna (2006)

5. INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, GS-G-3.1, IAEA, Vienna (2006).

6. INTERNATIONAL ATOMIC ENERGY AGENCY, Legal and Governmental Infrastructure for Nuclear, Radiation, Radioactive Waste and Transport Safety – Requirements, GS-R-1, IAEA, Vienna, 2000. (DS415 is the current revision of GS-R-1).

7. INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report, Safety Series No. 35-G1, IAEA, Vienna (1994). (DS396 is the current revision of 35-G1).

8. INTERNATIONAL ATOMIC ENERGY AGENCY, Commissioning of Research Reactors, Safety Standards Series, NS-G-4.1, Vienna (2006).

9. INTERNATIONAL ATOMIC ENERGY AGENCY, Operational Limits and Conditions for Research Reactors, NS-G-4.4, IAEA, Vienna (2008).

10. INTERNATIONAL ATOMIC ENERGY AGENCY, Maintenance, Periodic Testing and Inspection of Research Reactors, Safety Series, NS-G-4.2, IAEA, Vienna (2006).

11. INTERNATIONAL ATOMIC ENERGY AGENCY, Core Management and Fuel Handling for Research Reactors, Safety Standards Series, NS-G-4-3, IAEA, Vienna (2008).

12. INTERNATIONAL ATOMIC ENERGY AGENCY, The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors, Safety Standards Series, NS-G-4.5, Vienna (2008).

13. INTERNATIONAL ATOMIC ENERGY AGENCY, Review and Assessment of Nuclear Facilities by the Regulatory Body, GS-G-1.2, IAEA, Vienna (2002).

14. INTERNATIONAL ATOMIC ENERGY AGENCY, Documentation for Use in Regulating Nuclear Facilities, GS-G-1.4, IAEA, Vienna (2002).

15. INTERNATIONAL ATOMIC ENERGY AGENCY, Organization and Staffing of the Regulatory Body for Nuclear Facilities, GS-G-1.1, IAEA, Vienna (2002).

16. INTERNATIONAL ATOMIC ENERGY AGENCY, Site Evaluation for Nuclear Installations, Safety Standards Series, NS-R-3, IAEA, Vienna (2003).

17. INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Analysis for Research Reactors, Safety Report Series No. 55, IAEA, Vienna (2008).

18. INTERNATIONAL ATOMIC ENERGY AGENCY, Preparedness and Response for Nuclear or Radiological Emergency, Safety Standards Series No. GS-R-2, IAEA, Vienna (2002).

19. INTERNATIONAL ATOMIC ENERGY AGENCY, Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors, NS-G-4.6, Vienna (2008).

20. INTERNATIONAL ATOMIC ENERGY AGENCY, Safety in the Utilization and Modification of Research Reactors, Safety Series No. 35-G2, IAEA, Vienna (1994). (DS397 is being revised to supersede 35-G2).

21. INTERNATIONAL ATOMIC ENERGY AGENCY, Fire Safety in Operation of Nuclear Power Plants, Safety Standards Series No. NS-G-2.1, IAEA, Vienna (2000).

22. INTERNATIONAL ATOMIC ENERGY AGENCY, Preparedness and Response for a Nuclear or Radiological Emergency, Safety Standards Series No. GS-R-2, FAO, IAEA, ILO, OECD/NEA, PAHO, OCHA, WHO, Vienna (2002).

23. INTERNATIONAL ATOMIC ENERGY AGENCY, Guidelines for the Review of Research Reactor Safety, IAEA Services Series No. 1, IAEA, Vienna, December 1997.

24. INTERNATIONAL ATOMIC ENERGY AGENCY, Ageing Management for Research Reactors, Safety Standards Series, (DS 412, in preparation).

25. INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Considerations for Research Reactors in Extended Shutdown, IAEA-TECDOC-1387, January 2004.

26. INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Nuclear Power Plants and Research Reactors, Safety Standards Series No. WS-G-2.1, IAEA, Vienna (1999).

27. INTERNATIONAL ATOMIC ENERGY AGENCY, International Basic Safety Standards for Protection against Ionizing Radiation and for the Safety of Radiation Sources, Safety Series No. 115, FAO, IAEA, ILO, OECD/NEA, PAHO, WHO, Vienna (1996). (Under preparation)

28. INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for Facilities and Activities, General Safety Requirements Part 4, No. GSR Part 4, Vienna (2009).

29. INTERNATIONAL ATOMIC ENERGY AGENCY, Storage of Spent Fuel, Safety Standards Series, (DS 371 for CSS endorsement).

30. INTERNATIONAL ATOMIC ENERGY AGENCY, Licensing Process for Nuclear Installations, (In preparation, DS 416, Draft 2 is the current draft).

31. INTERNATIONAL ATOMIC ENERGY AGENCY, Regulatory Oversight of Management Systems, Safety Reports Series, (Draft 3.5).

32. INTERNATIONAL ATOMIC ENERGY AGENCY, Regulatory Inspection of Nuclear Facilities and Enforcement by the Regulatory Body, Safety Guide No.GS-G-1.3, IAEA, Vienna (2002).

33. INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of Seismic Hazards for Nuclear Installations (DS 422 is the current revision of NS-G-3.3).

34. INTERNATIONAL ATOMIC ENERGY AGENCY, Evaluation of Seismic Safety for Nuclear Installations, Safety Guide, NS-G-2.13, IAEA, Vienna (2009).

35. INTERNATIONAL ATOMIC ENERGY AGENCY, Hydrological and Meteorological Hazards in Site Evaluation for Nuclear Installations, (DS 417 in preparation).

36. INTERNATIONAL ATOMIC ENERGY AGENCY, Volcanic Hazards in Site Evaluation for Nuclear Installations, (DS 405 in preparation).

ANNEX I:
EXAMPLE OF DESIGN BASIS GRADING

I.1. Design basis requirements for the electrical power supply, the PIE, “loss of off-site electrical power supply” are analysed for different research reactor types, to indicate where grading may be used to determine the general and specific requirements. A reactor scram/trip is assumed to occur upon loss of electrical power.

General requirements:

(a) For a low power reactor in which core cooling for decay heat after shutdown is adequately provided by natural convection, an emergency electrical power supply is not required for the primary cooling system.

(b) For a reactor, in which forced cooling for removal of decay heat is required, a back-up electrical power supply will be needed to provide adequate availability for forced cooling, in the event of loss of of-site electrical power. Depending upon the decay heat there may be a need, determined by analysis, to also supply flywheel run-down capability on the main coolant pumps.

Specific requirements:

(c) Grading may also be applied to the specific design and regulatory requirements for the type, capacity and response time of a backup power system for supplying forced cooling during shutdown upon loss of off-site power. Back up pumps for cooling may be AC or DC powered. A back-up electrical power system in the event of loss of off-site power may range from (i) one or more diesel generators to supply back up AC power to loads that may be interrupted for the short time (typically for up to about the 10 sec period that diesels require to startup and supply full load). (ii) motor generator sets with inverters for AC and DC power requirements that cannot be interrupted on a loss of off-site power (iii) DC battery power, or UPS systems, supplying DC loads that cannot be interrupted and require the highest reliability, such as essential I&C, neutron flux monitoring, fire protection systems, emergency communication systems and emergency lighting.

NOTE: Consideration should also be given to the historical reliability performance of the off-site power supply in designing the facility emergency power supplies.

I.2. In some cases, a generic design may require the application of additional codes and standards relevant to the environment in which the reactor is required to operate (e.g. a university environment in a built up and highly populated area). A higher grade of SSC may be required and would likely require additional safety measures to be in place compared to a similar reactor (with equivalent source term) in a more remote area.

I.3. The potential effects on the reactor of other installations on the site may allow for grading. Lower potential effects imply lower provisions against such effects in design.

I.4. Reactors with high pressure loop systems or experiments may require additional ESFs.

I.5. Based on the safety analysis of the reactor, the design basis for the reactor building may or may not require a containment or means of confinement as an ESF. If needed, the design basis will have to specify in detail the requirements for the containment or confinement. If not, the design basis may specify a more conventional industrial type building.

I.6. Analysis of site related external event may (or may not) indicate that the design should include a seismic detection/triggering system that will actuate the shutdown system of the reactor if a specific threshold value is exceeded.

I.7. Fire protection measures may be graded based on the results obtained from a fire hazard analysis. House keeping is also a way of grading fire protection. Good house keeping rules and implementation will generally reduce fire loads.

CONTRIBUTORS TO DRAFTING AND REVIEW

Abou Yehia, H.	International Atomic Energy Agency
Alcalá Ruiz, F.	Private Consultant
Boado Magán, H.	International Atomic Energy Agency
Di Meglio, Francis	Private Consultant
Hargitai, T.	International Atomic Energy Agency
Litai, Dan	Private Consultant
Pape, R.P.	Private Consultant
Sajaroff, P	ARN, Agentina
Sharma, S.K.	Bhabha Atomic Research Centre, India
Shim, Sang.	Canadian Nuclear Safety Commission, Canada
Shokr, A	International Atomic Energy Agency
Winfield, D.J.	Private Consultant