DS351 Draft 6
Date: 2010/09/13

IAEA SAFETY STANDARDS

for protecting people and the environment

Status: Member States comments addressed, changes shown highlighted

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 1
BACKGROUND 1
OBJECTIVE 3
SCOPE 3
STRUCTURE 4

2. BASIC ELEMENTS OF THE APPROACH TO GRADING 4
GENERAL CONSIDERATIONS REGARDING THE CONCEPT OF GRADING 4
DESCRIPTION OF THE APPLICATION OF A GRADED APPROACH 5

3. REGULATORY SUPERVISION 8
APPLICATION OF GRADING TO LEGAL INFRASTRUCTURE 8
APPLICATION OF GRADING TO REGULATORY BODY 8
APPLICATION OF GRADING TO LICENSING PROCESS 9
APPLICATION OF GRADING TO INSPECTION AND ENFORCEMENT 10

4. MANAGEMENT AND VERIFICATION OF SAFETY 11
APPLICATION OF GRADING TO THE MANAGEMENT OF SAFETY 12
APPLICATION OF GRADING TO THE VERIFICATION OF SAFETY 13

5. SITE EVALUATION 14
APPLICATION OF GRADING 14

6. DESIGN 16
APPLICATION OF GRADING 16

7. OPERATION 33
GENERAL 33
APPLICATION OF GRADING TO ORGANIZATIONAL PROVISIONS 33
APPLICATION OF GRADING TO TRAINING, RETRAINING AND QUALIFICATION 35
APPLICATION OF GRADING TO OPERATIONAL LIMITS AND CONDITIONS 35
APPLICATION OF GRADING TO COMMISSIONING 37
APPLICATION OF GRADING TO OPERATING PROCEDURES 38
APPLICATION OF GRADING TO INSPECTION, PERIODIC TESTING AND MAINTENANCE 39
APPLICATION OF GRADING TO CORE MANAGEMENT AND FUEL HANDLING 40
APPLICATION OF GRADING TO FIRE SAFETY 41
APPLICATION OF GRADING TO EMERGENCY PLANNING 41
APPLICATION OF GRADING TO RECORDS AND REPORTS 43
APPLICATION OF GRADING TO REACTOR UTILIZATION AND MODIFICATION 43
APPLICATION OF GRADING TO RADIATION PROTECTION 44
APPLICATION OF GRADING TO SAFETY ASSESSMENTS 45
APPLICATION OF GRADING TO AGEING RELATED ASPECTS 47
APPLICATION OF GRADING TO EXTENDED SHUTDOWN 47

8. DECOMMISSIONING 48
APPLICATION OF GRADING TO DECOMMISSIOING 48

REFERENCES 51

ANNEX I: EXAMPLE OF STEPS IN THE GRADED APPROACH FOR PACKAGING OF RADIOACTIVE MATERIALS 54

CONTRIBUTORS TO DRAFTING AND REVIEW 57

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 and gamma 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 safety requirements for research reactors may not be applied to every research reactor in the same way. For example 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 general definition and purpose of the graded approach is taken from Ref. [2]. The two part definitions are both applicable to the safety requirements of Ref. [1]:

(i) *For a system of control, such as a regulatory system or a safety system, a process or method in which the stringency of the control measures and conditions to be applied is commensurate, to the extent practicable, with the likelihood and possible consequences of, and the level of risk associated with, a loss of control.*

---

[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.

(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. The Ref. [2] definition further notes that the graded approach in general is a structured method by means of which the stringency of application of requirements is varied in accordance with the circumstances, and the regulatory and management systems used. For example, a method in which:

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

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

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

1.6. 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] and 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 indicates 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. [6], 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], para 1.3 and Chapter 3 note the special attention to be given for safety assessment with regard to the application of a graded approach.

[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).*

– Ref. [32], para 3.10 discusses inspection programs that the regulatory body should establish, using a graded approach[3] to respond to unplanned situations or events.

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 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 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

---

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

[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.

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.

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 1.8.

[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.

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 that there are no negative effects on the facility staff, the public, or the environment.

The grading of activities should be based on safety analyses, regulatory requirements and engineering judgement. Engineering judgement implies that account is taken of the safety functions of SSCs and the consequences of failure to perform these functions and the judgement is documented. 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.

DESCRIPTION OF THE APPLICATION OF A GRADED APPROACH

2.3. Some of the activities in para. 1.12 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.

2.4. 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.

2.5. The application of grading presented in this Safety Guide begins with a facility hazard categorization (Step 1), which determines the baseline of the potential radiological hazard. 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 Hazard Categorization

2.6. Perform a qualitative categorization of the facility hazard, based on the potential radiological hazard, using ranking system similar to Ref. [11], para. 1.11:

(i) Off-site radiological hazard potential;

(ii) On-site radiological hazard potential only; and

(iii) No radiological hazard potential beyond the research reactor hall and associated beam line or connected experimental facility areas.

2.7. The individual characteristics, or attributes, to be considered in deriving a hazard categorization would typically be as follows, see Ref. [1], para 1.14:

(a) Reactor power, (for pulsed reactors, energy deposition and for accelerator driven subcritical systems, thermal power would be used);

(b) Radiological source term;

(c) 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) Type of fuel and its chemical composition;

(f) Type and mass of moderator, reflector and coolant;

(g) Amount of reactivity that can be introduced and its rate of introduction, reactivity control, and inherent and engineered safety features;

(h) Quality of the containment structure or other means of confinement;

(i) Utilization of the reactor (experimental devices, tests, radio-isotope production, reactor physics experiments);

(j) Location of the site; including potential for occurrence of external hazards (including those due to the proximity of other nuclear facilities) and characterization for airborne and liquid releases of radioactive materials; and

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

Step 2: Analysis and Grading

2.8. 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.

2.9. 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.

2.10. 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].

2.11. 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, (para. 2.2), 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].

2.12. The resources applied for management systems should be graded, on the basis 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.

2.13. 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. 7.50.

[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. Some safety functions may not be relevant for some types of research reactor.

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 TO 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".

The application of this requirement should not be graded.

APPLICATION OF GRADING TO 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 be provided with adequate authority and power and sufficient number of experienced staff and financial resources to discharge its assigned responsibilities, Ref. [6] para 2.2, (e.g. 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, technical safety organizations (TSO) or advisory committees may assist the regulatory body in these activities[10].

3.5. Examples of the regulatory organization, associated activities and requirements that are gradable are; staff requirements, in-house technical support resources, compliance inspections, content and detail of licenses, regulations and guides and the detail required of the licensee for facility safety submissions, including the SAR .

APPLICATION OF GRADING TO LICENSING PROCESS

3.6. The licensing process is often performed in steps for various stages of the research 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 all proposed modifications, and authorization of decommissioning.

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

3.8. The licensing process should be used by the regulatory body to exercise control during all stages of the lifetime of the research reactor, Ref. [30]. This control is accomplished by using the following:

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

(b) review and assessment of all safety-relevant documents, particularly the SAR,

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

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

(e) review, assessment and approval of Operational Limits and Conditions (OLCs),

(f) commissioning authorization,

(g) operating license,

(h) licensing of operational personnel,

(i) decommissioning license.

[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.

3.9. The steps in the licensing process apply to all research reactors, including all proposed experiments and design modifications, during all stages of the reactor lifetime. However, each step in the licensing process should be subject to grading by the regulatory body in determining the scope, extent, level of detail and effort that should be used, depending on the magnitude of the potential risks, Ref. [30 draft], 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, with a power level <100 kW, compared to those for a research reactor with a power level >5 MW

3.10. Specific to the licensing of research reactor decommissioning, detailed recommendations on the application of the graded approach for the regulatory review of a decommissioning safety assessment is provided in Ref. [39], paras 5.6 to 5.8.

APPLICATION OF GRADING TO INSPECTION AND ENFORCEMENT

3.11. 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.

3.12. Enforcement actions should also be graded since the severity and impact on safety of non-compliance with the requirements of an authorization may vary, Ref. [31], page 40. 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.

3.13. 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) Timeliness of corrective actions to restore compliance;

(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;

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

(f) The complexity of the remedial, corrective or preventive action needed.

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. Further guidance for the management system and verification of safety is also provided in Refs. [4], [5] and [37]. The elements of Management of Safety for an operating organization include, but are not limited to:

(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

[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.

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 SSC modifications and their cumulative effects. Additionally safety related aspects to be included are changes to procedures, radiation protection, regulations and standards; ageing effects; operating experience;; lessons learnt from similar reactors; technical developments site re-evaluation; physical protection; 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, 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 research reactor power level, extent of isotope production and scope of experimental facilities. In addition, grading is possible in the depth, frequency and type of safety assessments, in-service inspections 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 safety significance and complexity of each activity;
- the hazards and the magnitude of the potential impact (risks) associated with the safety, health, environmental, security, quality and economic elements of each activity; 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 that should be graded include:

(a) Type and content of training;

(b) Level and detail of operating procedures and their associated extent of review and approval;

(c) Need for and detail of inspection plans;

(d) Depth of operational safety reviews and controls;

(e) Type and frequency of safety assessments;

(f) Records to be generated and retained;

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

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

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

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

(k) 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 para. 7.46 of this guide.

APPLICATION OF GRADING TO THE VERIFICATION OF SAFETY

4.8. Grading is possible in the frequency and scope of self-assessments[13] 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.

4.9. Grading is possible in the number, size, composition and frequency of meetings of reactor advisory groups or safety committees. The safety committee should advise the operating organization on relevant aspects of the safety of the reactor, the safety of its

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

utilization, and on the safety assessment of design, commissioning and relevant operational issues and modifications. A safety committee should also advise the reactor manager. This is discussed in Ref. [1], para. 4.15. It is acceptable to have one safety committee advising the operating 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 radiological risk. The scope and detail of the site investigation may also be reduced if the operating organization proposes to adopt conservative parameters for design purposes, which may be a preferred approach for research reactors. For example, a conservative assumption 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], paras 2.4 to 2.13 and para. 6.6 develop the basis for applying a graded approach to the various site related evaluations and decisions, in accordance with the radiological hazard of the research reactor facility. The main siting characteristics to be considered are the influences of;

- potential external events of natural origin, such as seismic and volcanic events;
- site meteorological and hydrological characteristics influencing the extent of potential public doses and environmental contamination from facility releases;
- potential human induced events associated with the particular site[14];
- population density and population distribution and;

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

– other characteristics of the site such as ultimate heat sink capability.

5.3. The site evaluations 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 evaluation campaigns and simplified analysis methods.

5.4. Ref. [33], paras 6.8 to 6.10 provides information on a graded approach with respect to seismic hazards for site evaluation. The grading should be based upon the facility complexity and potential radiological hazards including hazards due to the presence of other materials. A seismic hazard assessment following a graded approach, should initially apply a conservative screening process based on the assumption that the complete radioactive inventory of the installation is released by an accident initiated by a seismic event. If such a release indicates no unacceptable consequences for facility staff, and/or the environment the installation may be screened out from further seismic hazard assessment. If the results of the conservative screening process show that the potential consequences of such releases would be significant, a seismic hazard evaluation should be performed. Ref. [34], paras 10.1 to 10.10 provides guidance on methodologies to be used for seismic hazard assessments, based on a graded approach.

5.5. Ref. [36] chapter 7 provides similar information to para 5.4, on a graded approach with respect to volcanic hazards for site evaluation. A volcanic hazard assessment following a graded approach, should initially apply a conservative screening process based on the assumption that the complete radioactive inventory of the installation is released by an accident initiated by a volcanic event. If such a release indicates no unacceptable consequences for facility staff, and/or the environment the installation may be screened out from further volcanic hazard assessment. If the conservative screening process shows that the consequences of the release are significant, a more detailed volcanic hazard assessment is then required, the grading process outlined in Ref. [36] paras 7.9 to 7.13 should then be used to allocate the facility into a defined volcanic risk category.

5.6. Site grading categorization is discussed with respect to meteorological and hydrological hazards in Ref. [35], paras 1.13 and chapter 10, where specific guidelines for site meteorological and hydrological hazard analysis requirements are provided. For the purpose of the evaluation of meteorological and hydrological hazards, facilities should be graded on the basis of their complexity, potential radiological hazards and hazards due to other materials present. If the results of a conservative screening process, similar to that in para 5.4 and 5.5, shows that the consequences of potential releases are significant, a detailed meteorological and hydrological hazard assessment for the facility should be carried out, in accordance with the grading process outlined in Ref. [35] paras 10.5 to10.11.

5.7. Ref. [16] paras 2.14 to 2.21 discusses graded approach criteria for hazard assessment due to human induced events and similarly paras 2.26 to 2.28, with regard to population density and population distribution factors and paras 3.52 to 3.55, with regard to other site characteristics such as heat sink capability.

6. DESIGN

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

Philosophy of design

Paras 6.2 to 6.8 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 6.9 to 6.37 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 6.38 to 6.71 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

Philosophy of design

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 applied with account taken of the graded approach (see para. 2.6 of Ref. [1]) recognizing that many low power research reactors do not qualify for the forth or fifth level of defence in depth. In addition the DiD concept should be applied in the design to provide graded protection against various reactor transients, including transients resulting from equipment failure and human error and from internal or external events that could lead to a Design Basis Accident (DBA), see para. 6.6. of Ref. [1]).

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

Shutdown Function

(1) In general, the basic safety function of requiring the reactor to shut down when required, is not gradable, although the extent of the sub-criticality margin available and the required speed of response required of the shutdown system 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

system 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 see para. 6.2, (Level 4).

Acceptance Criteria[15] and Design Rules

6.7. 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 below.

6.8. For the design of SSCs, acceptance criteria may be used in the form of engineering design rules. 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 6.9 and 6.10.

General requirements for design

Classification of SSCs

6.9. The requirements for classification of SSCs are presented in paras 6.12 to 6.13 of Ref. [1]. The method for grading the safety significance of SSCs, should be based on deterministic methods, complemented by probabilistic methods and engineering judgement (see para 2.2) and also Ref. [46].

Codes and Standards

6.10. 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.

[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.

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).

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

Design Basis

6.12. The requirements for the design basis are presented in paras 6.16 to 6.34 of Ref. [1]. Potential challenges that the research 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, an example of which is included as an Appendix in Ref. [1]. Design requirements will be supported by technical specifications. The requirement for technical specifications is not gradable, but the level and detail needed will be gradable

6.13. 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. [6.17] 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 requirement to identify the PIEs and the DBAs for research reactors is not gradable. The PIEs and DBAs should be identified using current safety standards and operational experience feedback, but the extent of the PIEs and the DBAs is however gradable.

6.14. The requirements established in Ref. [1] should be analyzed while developing the design basis for a specific research 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 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.

[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.

Design for Reliability

6.15. 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, diversity, independence and fail-safe criteria. 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 research 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 or passive / inherent safety function is required, a minimum reliability requirement should be established and maintained. Depending on the type of the research 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

6.16. 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.

6.17. 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.

Provision for inspection, testing and maintenance

6.18. 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.

6.19. 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.

6.20. 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).

6.21. 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.

Design for emergency planning

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

6.23. 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.

(e) The need for a supplementary control room, if justified by the safety analysis, and the degree of automatic and/or manual control needed.

Design for Decommissioning

6.24. 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.

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

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

(2) Higher power level, pool type research 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.

Design for Radiation Protection

6.26. 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]

6.27. 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 research reactor with limited utilization possibilities and low potential for significant exposure. (See also 6.57 of this publication).

Human factors and ergonomic considerations

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

6.28. 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 such as procedure writing, fatigue and working in demanding 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, such as at all times when fuel is in the reactor.

Provision for utilization and modification

6.29. 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.

6.30. 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 research 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.

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

Selection and ageing of materials

6.32. 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, 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]

6.33. Grading should 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 SSCs, and also for the knowledge management needed to support this aspect.

Particularly important material ageing concerns are corrosion in reactor tanks and vessels, where leak detection can be difficult and repair or replacement may not be practical. Similarly, corrosion of inaccessible primary coolant piping and associated components are of key importance for reactor longevity. An important knowledge management area, supplementing the original material selection and ageing management, is the need in recent years for improved human resource management to address the ageing nuclear workforce. Other key knowledge management areas are configuration management, document control, and operating experience programmes.

Provision for extended shutdown

6.34. 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.

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

Safety Analysis

---

[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.

[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.

6.36. 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 completeness of the PIEs, which are enveloped by the analyzed events and the conservatism of the assumptions on the effectiveness of preventive and mitigative features should be demonstrated. The safety analysis is a fundamental part of the design process, and is the basis for determining the safety importance of the SSCs and the extent that the potential hazards can be graded. It is also the basis for demonstrating the licensability of the proposed design and should confirm and validate that the grading of the requirements has been performed in a consistent and balanced way.

6.37. 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 effort for 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 facility. A low-power reactor having a limited hazard potential may require less analytical detail than a higher power level research 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 design

The reactor core and reactivity control system

6.38. 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.

6.39. 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 and mechanical stresses and chemical compatibility 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.

The reactor shutdown system

6.40. 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.

6.41. 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.

6.42. A second and diverse shutdown system should be considered for research 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

6.43. 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.

6.44. Grading may be possible in the reactor protection system in the sense that two different research 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 sites which could be impacted by significant seismic events, 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 6.6 and 6.59).

The reactor coolant systems and related systems

6.45. 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 research reactors.

Means of confinement

6.46. 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. 6.6 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 (DBA and BDBA). 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. It is the safety analysis which identifies how and to what extent the confinement design should be graded by considering the potential release from the reactor will determine the confinement design and the need for volatile fission product (e.g. iodine) traps. An example of the use of these considerations as a basis for grading is presented in para. 6.4 of this publication.

Experimental devices

6.47. 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.

6.48. 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.

6.49. 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

6.50. 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.

6.51. 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, neutron flux and water chemistry. 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.

6.52. Another means of grading I&C systems is the choice of the level of redundancy. Triple and quadruple channel redundancy is often used for research reactors that need to operate continuously, in order to minimize spurious trips and to allow for testing and/or maintenance on power of I&C equipment. For research reactors that operate for only a few hours per week, or less frequently such as critical assemblies for example, a lower level (two channel, one-out-of-two) redundancy may be selected, thus reducing design and operational complexity as well as costs.

6.53. 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.

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

6.55. 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 and, in particular, the parameters to be supervised and the actions required to maintain the reactor in a safe shutdown state. Typical features that may be included according to documented requirements are: radiation monitors, fire detection and actuators of extinguishers, communication means, ventilation system control, scram and/or safe shutdown features, operation of experimental devices, and operation of cooling systems.

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

Radiation protection systems

6.57. 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 6.26 to 6.27 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 and accident conditions (Design Basis Accidents, DBAs, and Beyond Design Basis Accidents, BDBAs". Para. 6.145 of Ref. [1] lists the radiation protection

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

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.

6.58. For example:

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

(b) A research 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.

(c) 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.

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

Fuel handling and storage system

6.59. 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 and experimental devices. 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 security may be equally applicable to many research reactors, the only difference being that of scale.

6.60. 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 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;

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

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

6.61. 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.

6.62. 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.

Radioactive waste systems

6.63. The requirements for the radioactive waste systems are specified in paras 6.162 to 6.166 of Ref. [1]. Radioactive materials (in solid, liquid and gaseous forms) are generated from fuel, neutron and gamma irradiation of reactor core components and coolants, in-core experiments and irradiation facilities, and also from operational waste[22].

6.64. The specific requirements for handling, storage, transport and disposal of radioactive waste and the control of solid, liquid and gaseous effluent discharges are all gradable and should be related to the type and quantities of radioactive waste generated in the specific reactor facility. Ref. [44] provides information on grading of performance standards for transport regulations and Ref. [45] Appendix provides detailed examples of grading for all aspects of transport. A detailed example of the use of the graded approach for the packaging of radioactive material is given in Annex 1, taken from Ref. [45] Appendix.

6.65. Grading considerations should be compatible with safety analysis and regulatory requirements, including the application of defence in depth design requirements for different types and quantities of radioactive waste, 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 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.

Buildings and structures

6.66. 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.

6.67. 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.

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

[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 active material created during reactor operation; fission product noble gases; tritium.

(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

6.69. 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.

6.70. 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 fourteen of these, (omitting physical protection as this is out of scope) are discussed in this chapter.

APPLICATION OF GRADING TO 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].

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 reactors with a low potential radiological hazard, 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. Training, re-training and qualification program (see paras 7.5 to 7.7);

v. Procedures (see paras 7.21 to 7.25);

vi. Maintenance, periodic testing and inspection program (see paras 7.26 to 7.33);

vii. Emergency plan and procedures (see para. 7.41 to 7.44);

viii. Radiation protection program (see paras. 7.51 to 7.56); and

ix. Management system (see para. 4.1).

[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.

APPLICATION OF GRADING TO TRAINING, RETRAINING AND QUALIFICATION

7.5. Training, retraining and qualification requirements for research reactor staff and other personnel such as experimenters are presented in paras 7.27 to 7.28 of Ref. [1]. Guidance on meeting these requirements is presented in Ref. [12].

7.6. Training, retraining and qualification requirements for research reactor staff and other personnel such as experimenters should be consistent with the complexity of the design, the hazard potential, the planned utilization of the facility, the available infrastructure and other functions that might be assigned to staff and other personnel. The 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. 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.

APPLICATION OF GRADING TO OPERATIONAL LIMITS AND CONDITIONS

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].

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.

[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.

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

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 SSCs. 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.

APPLICATION OF GRADING TO 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].

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.

APPLICATION OF GRADING TO 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.

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) 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.

APPLICATION OF GRADING TO 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].

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].

APPLICATION OF GRADING TO 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].

7.35. Research reactors with a low potential radiological hazard, having power ratings up to several tens of kilowatts and critical assemblies may need a less comprehensive core management and fuel handling programme. Low power reactors require infrequent core adjustments to compensate for burnup. They operate with substantial margins to thermal limits, allowing the consideration of a broad envelope of acceptable fuel loading patterns in the initial safety analysis in lieu of core specific calculations. While all recommendations in this Safety Guide should be considered, some 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], paras 1.11 and 2.2.

7.36. 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 7.47 to 7.50.

APPLICATION OF GRADING TO FIRE SAFETY

7.37. 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] and Ref. [42].

7.38. 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.

7.39. Grading the operational fire protection may be facilitated by provisions incorporated into the design corresponding to the fire hazard analysis required to be performed in [42] and to be periodically updated in [21] as well as by siting considerations.

7.40. Since fire safety assessment and analysis 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.

APPLICATION OF GRADING TO EMERGENCY PLANNING

7.41. 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]. Detailed approaches for a wide range of emergencies, suitable for the application of grading, are discussed in Ref. [38].

7.42. 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 and effects on the environment for credible accident scenarios are negligible and that emergency preparedness may be focused on on-site response. An understanding of the nature and magnitude of the potential hazard posed by an individual research reactor is required for preparing an appropriate emergency plan.

7.43. In conformance with the concept of a graded approach, Ref. [18], paras 3.6 to 3.7 utilize a nuclear and radiation emergency categorization scheme which provides a basis for developing optimized arrangements for preparedness and response. This scheme 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.

7.44. The magnitude of the potential source term and the engineered safety features are the most important factors affecting the grading of the emergency plan. Grading 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.

APPLICATION OF GRADING TO RECORDS AND REPORTS

7.45. 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] paras 5.21 and 5.22 and Ref. [5], para. 5.35 to 5.49 and Annexes I, II and III.

7.46. 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.

APPLICATION OF GRADING TO REACTOR UTILIZATION AND MODIFICATION

7.47. 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].

7.48. 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.

7.49. 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.

7.50. 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.

APPLICATION OF GRADING TO RADIATION PROTECTION

7.51. 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].

7.52. 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.

7.53. 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. For example a densely populated site will generally require a more extensive environmental monitoring programme.

7.54. It should be noted that a critical assembly may present a higher hazard of external radiation exposure for operating personnel than a higher power research reactor, but the latter may have a higher potential hazard for personnel contamination causing internal exposure. Also because critical assemblies are sometimes located within conventional industrial standard buildings, critical assembly reactivity accidents could result in a higher potential hazard, for contamination outside the building, compared to larger source term higher power reactors that have a containment structure.

7.55. Working areas within a research reactor should be 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 should be subjected to classification (grading) according to measures or expected radiological level, Ref. [19], paras. 5.44 to 5.46 and 5.48.

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.

7.56. 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.

APPLICATION OF GRADING TO SAFETY ASSESSMENTS

7.57. The requirements for safety assessments are presented in para. 7.108 of Ref. [1]. Guidance for performing safety assessments is presented in Ref. [23].

7.58. 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.

7.59. 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.

7.60. 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.

7.61. 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.

7.62. The graded approach should also be applied to the requirements for updating safety assessments, Ref. [28], para. 5.10. The frequency and depth of safety assessments should be graded depending upon on the number and extent of modifications for reactor systems as well as experimental facilities, changes to procedures, compliance monitoring of OLCs, 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. A maximum time between periodic safety assessments is though suggested to be no more than 10 years, regardless of reactor type or usage. With regard to reactors with more than 20 years of operation more emphasis on safety assessments of component ageing would be expected, particularly with regard to control systems and safety-related passive components in difficult-to-inspect and repair locations (e.g. inaccessible coolant piping and reactor tanks/vessels).

APPLICATION OF GRADING TO AGEING RELATED ASPECTS

The requirements for ageing related aspects are presented in para. 7.109 of Ref. [1]. Guidance on ageing related aspects of research reactors is presented in Ref. [24].

7.63. 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 in-service inspection and the ageing management programmes during the operating life of the facility.

Grading may be applied in determining the appropriate frequency of inspections, in selecting detection methods as well as in establishing ageing prevention and mitigation measures, which may be based on the estimated lifetime of the SSCs, their complexity and ease of replacement. In most research reactors, it is feasible to inspect most SSC's periodically and replace the components, if needed.

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

APPLICATION OF GRADING TO EXTENDED SHUTDOWN

7.64. 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. 6.34 to 6.35 of this publication and Ref. [25].

7.65. 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.

7.66. 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

The requirements for decommissioning are presented in paras 8.1 to 8.8 of Ref. [1]. Further guidance can be found in Ref. [26], under revision as DS 402. Ref. [39] is intended to assist regulators, operators and supporting technical specialists in the application of a graded approach to the development and review of safety assessments for decommissioning activities. Ref. [40] Chapter 3 discusses the graded approach requirements applicable to the development of the decommissioning plan and similarly Ref. [41], to obtain release of sites from regulatory control.

APPLICATION OF GRADING TO DECOMMISSIOING

8.1. Decommissioning requirements are applicable to every research reactor. The range of decommissioning activities for which a safety assessment is required is broad, and the scope, extent and level of detail of safety assessment should be commensurate with the types of hazards and their potential consequences. A graded approach should therefore be applied to the development and review of safety assessments. The effort associated with fulfilling them (e.g., in the preparation and review of the plans and procedures) may be graded. The grading of the plan and procedures 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 see Ref. [40], Chapter 3.

8.2. 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. The retention of facility knowledge, due to retirement and loss of experienced personnel when the reactor is permanently shutdown, are also very important to manage, in order to facilitate efficient and safe decommissioning operations.

8.3. 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) Research 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 research 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.

8.4 The operating organization should define the boundary conditions of the decommissioning plan by considering the most significant criteria, such as the resources available for decommissioning, the implementation period and final decommissioning end state required, (such as immediate dismantling, deferred dismantling, entombment or site release from regulatory control). With these conditions defined, the degree of grading possible e.g., based on the present state of the installation and possible future uses of the decommissioned installation or site, can be established.

8.5 Regulatory review of the decommissioning plan should adopt a graded approach and account should be taken of the following, Ref. [39] paras 5.6 to5.8:

(a) All relevant safety requirements and criteria derived from national legal and regulatory frameworks;

(b) The potential (e.g. in terms of likelihood and magnitude of consequence) for the proposed decommissioning activities to lead to an uncontrolled or accidental release of radioactivity (e.g. in working premises, on the site, off the site or at nearby facilities);

(c) The safety assessment’s estimates of radioactive release and dose to workers arising from planned decommissioning activities;

(d) The complexity and novelty of the proposed decommissioning activities;

(e) Operator aspects (e.g. the operator's, or the contractor's, past performance and relevant experience, both in decommissioning and introducing safety assessments for decommissioning; the complexity of the organization);

(f) Relevant incidents and events at other facilities or at similar facilities during decommissioning;

(g) The scope of the decommissioning activities being assessed (e.g. a stage of a larger project; a single large project; or a proposal leading to the final release of the facility from regulatory control);

(h) Technical or safety related concerns of other competent authorities (e.g. authorities having oversight over physical protection, security or non radiological hazards).

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, Governmental, Legal and Regulatory Framework, 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). Currently under revision as DS-402.

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 revision 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, GS-G-1.3, IAEA, Vienna (2002).

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

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

35. INTERNATIONAL ATOMIC ENERGY AGENCY, Meteorological and Hydrological 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).

37. INTERNATIONAL ATOMIC ENERGY AGENCY, Management of Operational Safety in Nuclear Power Plants, INSAG-13, IAEA, Vienna (1999).

38. INTERNATIONAL ATOMIC ENERGY AGENCY, Method for the Development of Emergency Response Preparedness for Nuclear or Radiological Accidents, TECDOC-953, IAEA, Vienna (1997).

39. INTERNATIONAL ATOMIC ENERGY AGENCY, Safety Assessment for the Decommissioning of Facilities using Radioactive Material, Safety Guide, WS-G-5.2, IAEA, Vienna (2008).

40. INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL ATOMIC ENERGY AGENCY, Decommissioning of Facilities Using Radioactive Material, Safety Requirements, WS-R-5, IAEA, Vienna (2006).

41. INTERNATIONAL ATOMIC ENERGY AGENCY, INTERNATIONAL ATOMIC ENERGY AGENCY, Release of Sites from Regulatory Control on Termination of Practices, Safety Guide, WS-G-5.1, IAEA, Vienna (2006).

42. INTERNATIONAL ATOMIC ENERGY, Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants, Safety Standards Series No. NS-G-1.7, IAEA, Vienna (2004).

43. INTERNATIONAL ATOMIC ENERGY, Protection against Internal Fires and Explosions in the Design of Nuclear Power Plants, Safety Standards Series No. NS-G-1.7, IAEA, Vienna (2004).

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

45. INTERNATIONAL ATOMIC ENERGY, The Management System for the Transport of Radioactive Waste, Safety Standards Series No. TS-G-1.4, IAEA, Vienna (2008).

46. INTERNATIONAL ATOMIC ENERGY, Safety Classification of Structures, Systems and Components in Nuclear Power Plants, Safety Guide DS 367 , (in preparation)

ANNEX I:
EXAMPLE OF STEPS IN THE GRADED APPROACH FOR PACKAGING OF RADIOACTIVE MATERIALS

(Reproduced from Ref [45] Appendix)

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

(1) Identification of the package type according to the Transport Regulations (Ref. [44]);

(2) Classification of the package by the development of a list of the packaging components and software to be used in its design, fabrication, use, inspection or testing, and assignment of a quality category (grade) to each (Table 2);

(3) Specification of the management controls required and assignment of a quality category (grade) to each (Table 3).

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

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

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

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

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

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

Note: Items whose failure does not impact safety or quality of the product or service need not be included in this graded system. Examples of such non-grade items may include some software that facilitates routine operation, handling and/or use of the package or packaging.

TABLE 3. GRADED MANAGEMENT CONTROLS

Graded management controls	**Quality categories**		
	Grade 1	**Grade 2**	**Grade 3**
The design is based on the most stringent industry codes or standards, and the design verification is accomplished by prototype testing or formal design review.	**X**		
The suppliers and subtier suppliers have a management system based on applicable criteria established in an acceptable national or international standard.	**X**		
The manufacturing planning specifies complete traceability of raw materials and the use of certified welders and processes.	**X**		

Graded management controls	**Quality categories**		
	Grade 1	**Grade 2**	**Grade 3**
The procurement documentation for materials for services specifies that only suppliers from qualified vendor lists are used.	**X**	**X**	
A comprehensive programme for specifying commercial grade items and controlling counterfeit parts is required.	**X**	**X**	
The verification planning (testing and inspection) requires the use of qualified inspectors (i.e. individuals performing non-destructive examinations such as radiography and ultrasonic testing are qualified in accordance with recommended practices described in appropriate national or international standards).	**X**	**X**	
Only qualified auditors and lead auditors perform audits.	**X**	**X**	
Comprehensive design, fabrication and assembly records; results of reviews, inspections, tests and audits; results of the monitoring of work performance and materials analyses; and results of maintenance, modification and repair activities are maintained.	**X**	**X**	
The design is based on the most stringent industry codes and standards, but design verification can be achieved by the use of calculations or computer codes.		**X**	
The manufacturing planning need not require traceability of materials, and only specified welds are done by qualified welders.		**X**	
Only the lead auditor need meet certain qualification requirements.		**X**	
Verification activities still require use of independent inspectors qualified to appropriate codes, standards or other industry specifications.		**X**	**X**
The procurement of materials need not be from a qualified vendor list.			**X**
Items are purchased from a catalogue of 'off the shelf' items.			**X**
When the item is received, the material is identified and checked for damage.			**X**
Self-assessments rather than independent assessments are the primary method for assessing and verifying performance.			**X**
Records are maintained in temporary files for a specific retention period (e.g. six months) after shipment.			**X**

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, Argentina
Sharma, S.K.	Bhabha Atomic Research Centre, India
Shim, Sang.	Canadian Nuclear Safety Commission, Canada
Shokr, A	International Atomic Energy Agency
Winfield, D.J.	International Atomic Energy Agency Consultant