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

DRAFT SAFETY GUIDE
DS351

New Safety Guide

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CONTENTS

Comment [DW1]:

1. INTRODUCTION

BACKGROUND

~~2.1.~~1.1. This ~~document~~ Safety Guide presents ~~guidance~~ recommendations on the ~~use of the~~ graded approach to application of the safety requirements for research reactors ~~as presented~~ established in the Safety Requirements publication on Safety of Research Reactors[1] [1].

~~2.2.~~1.2. Research ~~r~~Reactors in Member States employ a variety of designs. ~~The o~~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~~ the radioactive material of the core inventory, but also ~~that~~ radioactive material contained in stored spent fuel elements, radioactive waste from 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 materials science, transmutation studies, commercial production of radioisotopes~~ 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.

~~2.3.~~1.3. Because of the wide range of utilization activities, the ~~noted above~~ safety requirements for research reactors may not ~~need to~~be required to be applied to every research reactor in the same way. For example, requirements that are applicable to multipurpose, high power level research reactors ~~may~~ might not be fully applicable to research reactors with very low power and very low associated radiological hazard to the ~~facility staff~~workers, the public and the environment. ~~The Safety Requirements document on the Safety of Research Reactors,~~ Reference~~.~~ [1], which ~~has been developed to~~ appliesy to a wide range of research reactors, includes ~~recommendations~~ information ~~(paras 1.11 to 1.14)~~ ~~for applying~~on application of the safety requirements ~~utilizing~~ in accordance with a graded approach (see paras 1.11 to 1.14 of Ref. [1]).

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[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~~ the~~ 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, but excludes reactors used for the production of electricity, naval propulsion, desalination or district heating. see para 1.7 and footnote 4 of Ref. [1].

~~2.4.~~1.4. The general definition and purpose of the graded approach is ~~taken from~~set out in Ref. [2]. Both parts of ~~T~~the ~~two part~~ definition~~s~~ 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.

(ii) An application of safety requirements that is commensurate with the characteristics of the practice or source and with the magnitude and likelihood of the exposures.

~~2.5.~~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 [2].

~~2.6.~~1.6. ~~The idea of providing~~ ~~G~~guidance has been provided in the past on grading the application of safety requirements ~~and safety guides in IAEA documents is not new~~. There are a number of historical ~~references~~publications, which are now superseded, relat~~ing~~ed to grading[2] and several current ~~contemporary~~ IAEA ~~documents~~ safety standards ~~continue to~~ refer to a graded approach:

– ~~Ref. [3], para. 3.15,~~ Principle 3 of the Fundamental Safety Principles ~~indicates~~ states that "~~S~~safety has to be assessed ~~and periodically reassessed throughout the lifetime of~~for all facilities and activities, consistent with a graded approach" (Ref. [3], para. 3.15).

– ~~Ref. [3], paras 3.22-3.24,~~ Principle 5 of the Fundamental Safety Principles ~~indicates~~ states that "~~T~~the resources devoted to safety by the licensee, and the scope ... have~~are~~ to be commensurate with the magnitude of the ~~potential~~ radiation risks"~~.~~ (Ref. [3], para. 3.24).

[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. [6], para. 2.2~~Requirement 1 of Ref. [4] states that “The government shall establish a national policy and strategy for safety, the implementation of which shall be subject to a graded approach in accordance with national circumstances and with the radiation risks associated with facilities and activities”~~ 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~~.~~erence [5~~4~~], in paras 2.6-2.7, establishes requirements for grading the application of management system requirements and Ref. [6~~5~~], in paras 2.37-2.44, 5.6 and 6.68, ~~discuss the graded approach application to Management Systems~~provides related recommendations. ~~References [54] and [65] have now replaced supersede the QA documentspublications listed in footnote 2.~~

– Requirement 1 of Ref. [7~~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~~ states that “A graded approach shall be used in determining the scope and level of detail of the safety assessment carried out in a particular State for any particular facility or activity, consistent with the magnitude of the possible radiation risks arising from the facility or activity.”

– Ref. [8~~32~~], para. 3.10 ~~discusses~~ states that “in implementing ~~[an]~~the inspection programmes, ~~that~~ the regulatory body should establish ~~, using a~~a graded approach[3] in~~to~~ responding to unforeseen circumstances~~planned situations or events~~.”

OBJECTIVE

~~2.7.~~1.7. The objective of this S~~s~~afety G~~g~~uide is to provide support for the application of the safety requirements for research reactors throughout the various stages of the lifetime of a research reactor~~'s lifetime~~ (site selection and~~,~~ site evaluation, design, construction, commissioning, operation and decommissioning). The relevant safety requirements ~~considered~~ are ~~primarily those~~ established in Ref. [1], ~~with some references to other thematic publications of the IAEA, e.g. on Legal and Governmental Infrastructure~~and also in Ref~~s.~~ [4], [6~~5~~] and [7]~~, Management Systems Refs. [4] and [5] and General Safety Requirements, Ref. [28]~~. ~~It~~ This Safety Guide is intended for ~~the~~ use ~~of~~ by regulatory bodies, operating organizations and other organizations involved in the design, construction and operation of research reactors.

[3] In S~~s~~ome Member States~~, refer to the~~ a graded approach is referred to as ‘proportionality’.

SCOPE

~~2.8.~~1.8. This S~~s~~afety G~~g~~uide presents ~~guidance for~~recommendations on applying a graded approach~~,~~ without compromising safety.

~~2.9.~~1.9. The application of a graded approach ~~throughout~~ to all the important activities[4] ~~in~~ throughout the lifetime of a research reactor ~~facilities~~ is discussed. These activities are identified in sections 3 to 8 of Ref. [1]~~, Chapters 3 to 8~~. A major ~~component~~ aspect of the design activity, as described in ~~Chapter~~ Section 6 of this Safety Guide, involves the grading of specific ~~design~~ requirements~~,~~ for~~applied to the~~ design of s~~S~~tructures, s~~S~~ystems and c~~C~~omponents (SSCs) for particular reactor types, so that safety objectives set out in Ref. [1], para. 2.2~~,~~ are achieved. Recommendations on ~~T~~the application of grading ~~applicable~~ to reactor hardware and equipment (SSC~~'~~s), as opposed to activities in general, ~~is~~ are also ~~discussed~~ provided in~~as part of~~ Section ~~Chapter~~ 6~~ and uses the list of SSCs provided in Ref. [1]~~.

~~2.10.~~1.10. In this S~~s~~afety G~~g~~uide it is considered that all relevant safety requirements ~~must~~ have to be complied with~~,~~ in applications of ~~the~~ a graded approach. The graded approach should be used to determine the appropriate manner to comply with a requirement; it is not used to provide relief from meeting the requirement. ~~To eliminate a requirement, a waiving process, as suggested in Ref. [1] can be used.~~ Th~~e~~is process of waiving a requirement is separate and distinct from the graded approach. Waiving[5] is not discussed in this publication. Reference~~.~~ [7~~28~~], para. 1.5 also ~~notes~~ states that "a graded approach must be ~~used for~~taken to implementation of the safety requirements, to provide flexibility". However, ~~It should though be recognised that~~ [not a should statement] while safety requirements are to be complied with, "the level of effort to be applied in carrying out the necessary safety assessments needs to be commensurate with the ~~potential~~ possible radiation risks, and ~~any~~ their uncertainties~~y~~, associated with the ~~potential radiological hazard of the~~ facility or activity" [7].

STRUCTURE

~~2.11.~~1.11. ~~Chapter 1 outlines the background, the objective, scope and structure of this Guide. Chapter~~ Section 2 provides the description of the general principles of a graded

[4] Activities, in the context of this S~~s~~afety G~~g~~uide, 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. Reference~~.~~ [5~~4~~] uses a more general definition of activities ~~which~~ that encompasses any practice or circumstances in which people may be exposed to radiation~~ sources~~.

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

approach and its application. ~~Chapters~~ Sections 3 to 8 ~~discuss~~ provide recommendations on the application of a graded approach to the following six activities:

(a) Regulatory ~~S~~supervision;

(b) Management and v~~V~~erification of ~~S~~safety;

(c) Site ~~E~~evaluation;

(d) Design;

(d) Operation; ~~and~~

(e) Decommissioning.

~~Chapters~~ Sections 3 to 8 have titles identical to the corresponding ~~chapter~~sections of Ref. [1].

~~2.12.~~1.12. Each ~~chapter~~ section of this ~~publication~~ Safety Guide begins with a brief description of the relevant safety requirements ~~as specified~~established in Ref. [1] and, in some areas, ~~augmented with~~a summary of additional requirements ~~contained~~ established in other IAEA Safety Requirements 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~~ in all ~~the~~ stages ~~during the various stages~~ of the lifetime of a research reactor~~'s lifetime,~~ (see para. 1.7~~8~~).

–2.2. During the lifetime of a research reactor, any grading that is performed should ~~ensure~~ be such that safety functions and ~~Operating~~ operational l~~L~~imits and c~~C~~onditions are preserved and that there are no ~~negative effects~~undue radiological hazards ~~to~~ on ~~the facility staff~~workers, the public~~,~~ or the environment.

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

~~Some of the activities listed in para. 1.112 have are subject to safety requirements that are identified to be general safety requirements. Hence an initial preliminary step in the grading process is to identify whether the features of a specific research reactor require necessitate consideration within theof the general safety requirements.~~

–2.4. ~~No~~ This Safety Guide does not recommend the use of a quantitative ranking procedure ~~for the application of the graded approach to~~in grading the safety requirements~~ is suggested~~. The application of the graded approach will determine~~s~~ the appropriate effort to be expended and appropriate manner ~~needed~~ of~~to~~ complying with a requirement, in accordance~~ing~~ ~~to~~with the attributes of the facility.

–2.5. The application of grading presented in this Safety Guide begins with ~~a~~ categorization of the facility in accordance with its potential hazard ~~categorization~~ (Step 1)~~, which determines the baseline of the potential radiological hazard~~. ~~With~~ In this step, a facility can ~~be~~ initially be categorized into a range from facilities posing the highest risk to those posing the ~~least~~ lowest risk. This categorization serves to provide an initial ~~screening~~grading of~~, 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: Categorization of the Facility in Accordance with ~~its~~ Potential Hazards ~~Categorization~~

–2.6. ~~Perform a q~~Qualitative categorization of the facility~~ hazard,~~ should be performed ~~based~~ on the basis of the potential radiological hazard, using a ~~ranking~~ multi-category system similar to that set out in Ref. [~~1~~19], para. 1.11:

(i) Facilities with ~~O~~off-site radiological hazard potential;

(ii) Facilities with ~~O~~on-site radiological hazard potential only; ~~and~~

(iii) Facilities with ~~N~~no radiological hazard potential beyond the research reactor hall and associated beam ~~line~~ tubes -or connected experimental facility areas.

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

(a) The ~~R~~reactor power~~,~~ (for pulsed reactors, energy deposition is typically used, while~~and~~ for accelerator driven subcritical systems, thermal power is typically~~would be~~ used);

(b) The ~~R~~radiological source term;

(c) The ~~A~~amount and enrichment of fissile and fissionable material;

(d) Spent fuel storage areas, high pressure systems, heating systems~~,~~ and the storage of flammables, which may affect the safety of the reactor;

(e) The type of fuel and its chemical composition;

(f) The type and mass of moderator, reflector and coolant;

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

(h) The ~~Q~~quality of the containment structure or other means of confinement;

(i) The u~~U~~tilization of the reactor (experimental devices, tests, radio~~-~~isotope production, reactor physics experiments);

(j) The ~~L~~location of the site,~~;~~ including the potential for ~~occurrence of~~ external hazards (including those due to the proximity of other nuclear facilities) and the characteristics ~~zation for~~ of airborne and liquid releases of radioactive material~~s~~; ~~and~~

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

Step 2: Analysis and Grading

–2.8. ~~With~~ In this step, the level of detail at which requirements~~d~~ are applied to~~for grading~~ activities and/or SSCs is ~~chosen to be~~ determined, in accordance~~commensurate~~ with the~~ir relative~~ importance to safety of the activity or SSC. The level of detail ~~would specify~~covers, for example, the rigour of the analysis to be ~~required~~conducted, the frequency of activities such as testing and preventive maintenance, the ~~depth~~ stringency of required approvals and the degree of oversight of activities~~y oversight level~~.

–2.9. The appropriateness of applying a graded approach should be ~~D~~determined through analysis for each of the major activities and SSCs ~~defined~~ set out in~~by~~ ~~Chapter~~Sections 3 to 8 of Ref. [1]~~ the appropriateness of applying a graded approach~~. The ~~grading~~ application of grading should be commensurate with the ~~characteristics of safety requirements~~importance to safety of the activities and SSCs and with the magnitude ~~and likelihood~~ of the associated radiological risks.

2.10. ~~Identify a list of~~The safety functions[6] ~~associated with~~performed by each item important to safety[7] should be identified (see Ref. [~~12~~10], para. 2.11(b)). A starting point for assessing the importance to safety of activities and SSCs is ~~the performance~~conduct of ~~the~~ a safety ~~analysis~~assessment [8].

2.11. Paragraphs~~.~~ 6.12 and 6.13 of Ref. [1] state~~s~~ that all ~~the~~ SSCs (including software for instrumentation and control) that are important to safety ~~shall~~ are required first to be ~~first~~ identified and then to be 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] performed by the SSC~~s~~ and on the consequences of ~~its~~ the SSC's failure to perform its function. Analytical techniques together with engineering judgement~~,~~ (see para. 2.3~~2~~)~~,~~ ~~are~~ should be used to evaluate these consequences. The basis ~~of~~ for the safety classification of ~~the~~ SSCs, including software, should be stated and the~~ir~~ engineering design ~~requirements~~ rules applied should be commensurate with their safety class~~ification~~. ~~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~~application of management system~~s~~ requirements ~~should~~ shall be graded so as to deploy appropriate resources, on the basis of the consideration of:

~~3.1.~~ “The significance and complexity of each product or activity;

~~3.2.~~ “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;

~~3.3.~~ “The possible consequences if a product fails or an activity is carried out incorrectly” (Ref. [5], para. 2.6).

[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 [2].

[8] Guidance on this subject is provided in Ref. [11~~7~~].

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

~~–~~2.13. “Grading of the application of management system requirements ~~should~~ shall be applied to the products and activities of each process” (Ref. [5], para. 2.7). Where these activities involve modifications or experiments, further categorization ~~is suggested~~should be carried out ~~,~~ (see para. 7.50 of this Safety Guide).

3. REGULATORY SUPERVISION

~~4.1.~~3.1. The requirements for the legal~~islative~~ and regulatory infrastructure for a broad range of ~~nuclear~~ facilities and activities are ~~presented~~ established in Ref. [4~~6~~]. Additional guidance is provided in the associated S~~s~~afety G~~g~~uides, Refs~~.~~ ~~[5] and~~ [12~~3~~] 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. In E~~e~~ach ~~Member~~ State, ~~should identify~~ the requirements and recommendations that are applicable for the regulatory supervision of its nuclear programme should be identified. For the purpose of this ~~publication~~Safety Guide, the applicable safety requirements are those for the regulatory supervision of research reactors that are ~~presented~~ established in section 3 of Ref. [1], ~~Chapter 3~~ and include the following:

(a) Legal infrastructure;

(b) The ~~R~~regulatory body;

(c) The ~~L~~licensing process;

(d) The programme for ~~I~~inspection and enforcement ~~programme~~.

APPLICATION OF GRADING TO LEGAL INFRASTRUCTURE

~~4.2.~~3.2. The requirements for the legal infrastructure are established in Ref. [1]:~~,~~ para. 3.2 and in Ref. [4]: Requirement 3 on establishment of a regulatory body and Requirement 4 on independence of the regulatory body. ~~. 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 the~~i~~se requirements ~~should not~~cannot be graded.

APPLICATION OF GRADING TO THE REGULATORY BODY

~~4.3.~~3.3. A graded approach should be applied in~~when~~ establishing the regulatory body and determining aspects of its organizational framework, based on the potential hazards of all the facilities and activities under its supervision or oversight.

~~4.4.~~3.4. The regulatory body ~~should~~ is required to be provided with ~~adequate~~ sufficient authority and power and sufficient number of experienced staff and financial resources to discharge its assigned responsibilities~~,~~ (Ref. [4~~6~~] para 2.8~~2~~), ~~(e.g.~~The responsibilities of the regulatory body include ~~develop~~ establishing ~~and issue~~ regulations, review and assessment of safety related information (e.g. from the s~~S~~afety a~~A~~nalysis r~~R~~eport ~~(SAR)~~), issuing~~e~~ ~~licens~~licences, performing compliance inspections, taking~~e~~ enforcement actions and providing~~e~~ information to other competent authorities and the public~~)~~. External experts, technical ~~safety~~ support organizations ~~(TSO)~~ or advisory committees may assist the regulatory body in these activities[10].

~~4.5.~~3.5. Examples of the regulatory organization, associated activities and requirements that ~~are~~ can be graded~~able~~ are:~~;~~ ~~staff~~ requirements for staffing, resources for in-house technical support ~~resources~~, compliance inspections, the content and detail of licenc~~s~~es, regulations and guides, and the detail required of the licensee for ~~facility safety~~ submissions of documentation on safety of the facility, including the ~~SAR~~safety analysis report.

APPLICATION OF GRADING TO THE LICENSING PROCESS

~~4.6.~~3.6. The licensing process is often performed in steps for the various stages of the lifetime of a research reactor, ~~lifetime~~ as described in Ref. [1], paras 3.4 and 3.5 and ~~the Appendix of~~ Ref. [154]. 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.

~~4.7.~~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.

~~4.8.~~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.~~ [15~~30~~]. This control is accomplished by ~~using~~ means of the following:

(a) ~~c~~Clearly defined lines of authority for authorizations to proceed;~~,~~

(b) R~~r~~eview and assessment of all safety ~~-~~relevant documents, particularly the ~~SAR~~safety analysis report;~~,~~

[10] The IAEA provides safety review services ~~that are available to Member States~~governments of Member States, R~~r~~egulatory b~~B~~odies and o~~O~~perating o~~O~~rganizations.

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

(c) I~~i~~ssue~~ance~~ of permits and licen~~s~~ces, for the various stages;~~,~~

(d) H~~h~~old points for inspections, review and assessment;~~,~~

(e) R~~r~~eview, assessment and approval of o~~O~~perational ~~L~~limits and c~~C~~onditions ~~(OLCs)~~,

(f) C~~e~~ommissioning authorization;~~,~~

(g) O~~o~~perating licen~~s~~ce;~~,~~

(h) L~~l~~icensing of operating~~onal~~ personnel;~~,~~

(i) D~~d~~ecommissioning ~~licens~~licence.

~~4.9.~~3.9. The steps in the licensing process apply to all research reactors, including all proposed experiments and design modifications, ~~during~~ at all stages of the reactor lifetime. However, at each step in the licensing process, ~~should be subject to~~a graded~~ing~~ approach should be taken by the regulatory body in determining the scope, extent, level of detail and effort that should be used, depending on the magnitude of the potential risks~~,~~ (see Ref. [15~~30 draft~~], paras 2.19 (h)~~7~~ and 2.46~~1~~ to 2.450). For example, in general there will be fewer inspections and hold points for a research reactor~~,~~ with a power level less than ~~<~~100 kW, compared to those for a research reactor with a power level greater than ~~>~~5 MW

~~4.10.~~3.10. ~~Specific to the licensing of research reactor decommissioning, d~~Detailed recommendations on the application of ~~the~~ a graded approach ~~for~~ to the regulatory review of a~~the~~ ~~decommissioning~~ safety assessment in support of the licence for decommissioning of a research reactor ~~is~~ are provided in Ref. [16~~39~~], paras 5.6 to 5.8.

APPLICATION OF GRADING TO INSPECTION AND ENFORCEMENT

~~4.11.~~3.11. The requirements for inspection and enforcement are ~~presented~~ established 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~~commensurate with the potential ~~risk~~ hazard posed by the research reactor" and particular situations such as organizational changes or personnel turnover. Ref~~.~~erence [8~~32~~]~~,~~ ~~para. 3.14~~ recommends in para. 3.14 that "~~"~~inspections by the regulatory body should be concentrated on areas of safety significance"~~"~~ and in para 3.10~~,~~ that "the regulatory body should ~~use a pre-~~establish~~ed~~ a graded approach in responding to unforeseen circumstances".

~~4.12.~~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~~.[12] Regulatory bodies should ~~use the graded approach that~~ allocates resources and appl~~ies~~y enforcement actions or methods in a manner commensurate with the seriousness of the non-compliance, ~~escalating~~ increasing them as ~~needed~~ necessary to bring about compliance with requirements. A graded approach should be applied with respect to the corrective action process for non-conformances, to ensure that problems of the highest significance are afforded the most evaluation~~,~~ (see Ref. [6~~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) The need for 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 reported by an operator or identified by an inspector;

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

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

3.14. In contrast to the factors of para 3.13 however, an enforcement action based on a conscious decision by an individual to violate a regulatory requirement should not be graded. This is necessary to hold regulatory compliance in the highest regard, in order to protect the integrity of the regulatory system.

~~4.13.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) The need for 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 by an operator or identified by an inspector;~~

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

~~[12] An IAEA TECDOC on regulatory oversight of management systems is in preparation.~~

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

4. MANAGEMENT AND VERIFICATION OF SAFETY

~~5.1.~~4.1. Reference~~s.~~ [1], ~~Chapter~~Section 4 ~~"Management System~~[13] ~~and Verification of Safety"~~ ~~addresses~~ establishes the requirements on the elements of the management and verification of safety to be considered, the responsibilities of the operating organization and the interaction with the r~~R~~egulatory b~~B~~ody. Requirements are also established in Ref. [5] and ~~Further~~ recommendations and guidance for the management system and verification of safety ~~is~~ are also provided in Refs~~.~~ ~~[4],~~ [6~~5~~], [17] and [~~3~~18~~7~~]. ~~The elements of~~For m~~M~~anagement of ~~S~~safety, at a minimum, "~~for an~~the operating organization shall~~include, but are not limited to~~:

(a) "~~The e~~Establish~~ment~~ and implement~~ation of~~ safety policies and ensur~~eing~~ that safety~~-related issues~~ matters are given the highest priority;

(b) "Clearly defin~~eing~~ responsibilities and accountabilities with corresponding lines of authority and communication;

(c) "Ensur~~eing~~ that ~~the operating organization~~it has sufficient staff with appropriate education and training at all levels;

(d) "Develop~~ing~~ and strictly adher~~eing~~ to sound procedures for all activities that may affect safety, ~~and~~ ensuring that managers and supervisors promote and support good safety practices while correcting poor safety practices;

(e) "Review~~ing~~, monitor~~ing~~ and audit~~ing~~ all safety ~~-~~related matters on a regular basis, ~~and~~ implementing appropriate corrective actions where necessary; ~~and~~

(f) "Be~~A~~ committ~~ed~~~~ment~~ 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." (Ref. [1], para. 4.1).

~~5.2.~~4.2. The management system should provide for a process ~~of~~ for assessment and verification of safety, including ~~a~~ periodic safety review at an interval specified by the regulatory body. The basis for the assessment should include~~s [should statement??]~~, inter-alia, data derived from the ~~SAR~~safety analysis report and other information (e.g.~~,~~ the

~~[13] 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.~~

operational limits and conditions, the radiation protection programme, the emergency plan, operating procedures and training documentation).

~~5.3.~~4.3. Such assessments should include consideration of ~~SSC~~ modifications to SSCs and their cumulative effects. Additional~~ly~~ safety related aspects ~~to~~ that should be included in the assessment are changes to procedures, radiation protection measures, regulations and standards; ageing effects; operating experience;~~;~~ lessons learnt from similar reactors; ~~technical~~ technological developments~~[22]~~; site re-evaluation; physical protection; and emergency planning. ~~Some specific~~ ~~r~~Requirements on ~~these topics~~assessment and verification of ~~safety for~~safety for research reactors are ~~presented~~ established 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[14]

~~5.4.~~4.4. Grading of the scope and content of activities making up the elements of management of safety, such as items (a) ~~through~~ to (f)~~,~~ in para. 4.1, is possible while still meeting the requirement that they be comprehensive. For example, in item (c) of para. 4.1, grading is clearly essential in ~~defining~~ specifying the staffing levels required for operations and maintenance. Requirements for ~~S~~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~~ on the basis of the research reactor power level, the extent of isotope production and the scope of experimental facilities. In addition, grading can be applied to determining ~~is possible in~~ the depth, frequency and type of safety assessments, in-service inspections and auditing of all safety related matters.

~~5.5.~~4.5. The ~~extent level of the detailed~~ complexity of the management system for a particular research reactor and associated experimental facilities ~~will depend on~~should be commensurate with the potential hazard of the reactor and the experimental facilities and the requirements of the regulatory body. ~~Guidance~~ Requirements for the preparation and implementation of a graded management system ~~is provided~~are established in Ref. [54], paras 2.6 and 2.7, which ~~note~~ states that grading of the application of management system requirements ~~shall~~ is required to be applied to the products and activities of each process and

[14] In ~~NS-R-4~~ Ref. [1] the term '~~Q~~quality a~~A~~ssurance' was used. ~~Subsequent to NS-R-4, Safety Guides~~ References~~.~~ [45] and [~~5~~6] were issued later, and ~~which~~ adopted the term 'management system'~~s~~ 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.

that the grading ~~should~~ is required to be such as to deploy appropriate resources, ~~by~~ on the basis of considerationing of:;

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

~~5.6.Application of tThe requirements of the management systems should is required to be graded so as to use deploy 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 iincorrectly.~~

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

(a) Type and content of training;

(b) Level of detail and degree of review and approval of instructions and procedures;

(c) Level of detail of testing, surveillance, maintenance and inspection activities;

(d) Extent of operational safety reviews and reporting, analysis and corrective actions ~~procedures of~~ for non-conformances and system and equipment failures;

(e) The type and frequency of safety assessments;

4.7. Reference. [65], paras 2.37 to 2.44 also discuss the need for grading of management system controls~~activity grading~~. A detailed example of ~~where~~ the application of grading to the document and records management system (i.e.~~should be applied for the specific~~ item (f) ~~above~~ of para. 4.6) ~~(document and record management system~~) is reproduced from Ref. [56] in para. 7.46 of this Safety gGuide.

APPLICATION OF GRADING TO THE VERIFICATION OF SAFETY

4.8. Grading ~~is possible in~~can be applied to the frequency and scope of self-assessments[15] and peer reviews. The frequency and scope of safety assessments and peer reviews should be graded ~~based~~ on the basis of the complexity and potential risk of the facility, ~~and~~ whether the~~y~~ activity or SSC has~~ve~~ a safety function and the ~~importance~~ safety significance ~~of the safety function~~ of the activity or SSC being assessed.

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

5. SITE EVALUATION

~~6.1.~~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~~ consequences 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 in the site 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~~ for the design of a particular SSC that is readily accommodated in the overall design may ~~allow~~ permit simplification of the site evaluation.

[15] 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 basis of results of re-qualification examinations. ~~In some instances, the operating organizations prepares an annual report on the general performance of the reactor facility, which is a good practice. Safety committees can perform an assessment on the basised onf the report.~~

APPLICATION OF GRADING TO SITE EVALUATION

~~6.2.~~5.2. Grading ~~should~~ can be applied when assessing the aspects mentioned in para. 5.1~~, above~~. Reference~~.~~ [19~~6~~], 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~~ factors to be considered in site evaluation are the influences of:~~;~~

- ~~p~~Potential external events of natural origin, such as seismic and volcanic events;
- ~~Ssite m~~Meteorological and hydrological characteristics of the site that may influenc~~eing the extent of potential public doses~~ exposure of the public and environmental contamination from facility releases;
- P~~p~~otential human induced events associated with the ~~particular~~ site[16];
- P~~p~~opulation density and population distribution ~~and~~;
- O~~o~~ther characteristics of the site that could affect important safety requirements, ~~aspects~~ such as the ultimate heat sink capability.

~~6.3.~~5.3. The site evaluations should be graded, provided that there is an adequate level of conservatism in the design and siting criteria ~~are is provided~~, to compensate for ~~reduced~~ a simplified site hazard~~s~~ analysis ~~or, site evaluation campaigns studies~~ and ~~for~~ simplified analysis methods.

~~6.4.~~5.4. Reference~~.~~ [20~~33~~], ~~paras 6.8 to 6.10~~Section 10 provides ~~information~~ recommendations on applying a graded approach with respect to seismic hazard~~s~~ ~~for site~~ evaluation. The grading ~~should~~ can be based upon the ~~facility~~ complexity of the installation and the potential radiological hazards, including hazards due to ~~the presence of~~ other materials present. A seismic hazard assessment following a graded approach~~,~~ should initially apply a conservative screening process ~~based on the assumption~~ in which it is assumed that the ~~complete~~ entire radioactive inventory of the installation is released by an accident initiated by a seismic event. If such a release ~~indicates~~ would lead to no unacceptable consequences for ~~facility staff~~workers, the public~~, 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 a release~~s~~ would be significant, a seismic hazard

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

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.~~

~~6.5.~~5.5. Reference~~.~~ [21~~36~~], ~~chapter~~section 7 provides similar ~~information~~ recommendations to those in Ref. [20]~~ para 5.4~~, ~~on~~ for application of a graded approach with respect to volcanic hazards ~~for~~ in site evaluation. A volcanic hazard assessment following a graded approach~~,~~ should initially apply a conservative screening process ~~based on the assumption~~in which it is assumed that the ~~complete~~ entire radioactive inventory of the installation is released by an accident initiated by a volcanic event. If such a release ~~indicates~~ would lead to no unacceptable consequences for ~~facility staff, and/or~~workers, the public or the environment, the installation may be screened out from further volcanic hazard assessment. If the results of the conservative screening process show~~s~~ that the potential consequences of ~~the~~ such a release are significant, a more detailed volcanic hazard assessment ~~is then required~~should be performed, and the grading process outlined in Ref. [21~~36~~] ~~paras 7.9 to 7.13~~ should then be used to ~~allocate~~ categorize the ~~facility~~ installation ~~into a defined~~for the purposes of volcanic ~~risk category~~hazard assessment.

~~6.6.~~5.6. Recommendations on application of a graded approach ~~Site grading categorization is discussed~~ with respect to meteorological and hydrological hazards in site evaluation are provided in Ref. [22~~35~~]~~, para.s 1.13~~ and ~~chapter~~section 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, the ~~facilities~~ installation should be graded on the basis of ~~their~~ its complexity, the potential radiological hazards and hazards due to other materials present. If the results of a conservative screening process, similar to that in described in paras 5.4 and 5.5, shows that the consequences of a potential release~~s~~ are significant, a detailed meteorological and hydrological hazard assessment for the ~~facility~~ installations should be carried out, in accordance with the grading process outlined in Ref. [21~~35~~]~~, paras 10.5 to 10.11~~.

~~6.7.~~5.7. Reference~~.~~ [19~~6~~] paras 2.14 to 2.21 ~~discusses~~ provides ~~graded approach~~ criteria for use in applying a graded approach in ~~hazard~~ the assessment of hazards ~~due to~~associated with human induced events; ~~and~~ similarly paras 2.26 to 2.28 of Ref. [19]~~,~~ provide criteria with regard to population density and population distribution factors; and paras 3.52 to 3.55 of Ref. [19]~~,~~ establish important requirements with regard to other external events. ~~other site characteristics such as those that could affect the ultimate heat sink capability.~~

6. DESIGN

~~7.1.~~6.1. ~~Chapter~~ Section 6 of Ref. [1] discusses design under the three categories below;

Philosophy of design

Paragraphs 6.2 to 6.8 of this Safety Guide ~~discuss the use of~~provide recommendations on grading ~~in~~ the application of requirements relating to the philosophy of design, ~~listed~~ established in Ref. [1], paras 6.1 to 6.11.

General requirements for design

Paragraphs 6.9 to 6.37 of this Safety Guide provide recommendations on ~~discuss the use of~~ grading ~~in~~ the application of the general design requirements ~~listed~~ established in Ref. [1], paras 6.12 to 6.78.

Specific requirements for design

Paragraphs 6.38 to 6.71 of this Safety Guide provide recommendations on ~~discuss the use of~~ grading ~~in~~ the application of the specific design requirements ~~listed~~ established in Ref. [1], paras 6.79 to 6.171.

APPLICATION OF GRADING TO DESIGN

Philosophy of design

Defence in depth

~~7.2.~~6.2. Paragraphs 2.6 and 6.6 of Ref. [1] describes the five levels of defence in depth ~~(DiD)~~ that are aimed at~~to~~ preventing deviations in normal operation and controlling them ~~should~~ if they occur ~~from operational states~~ and at ~~to~~ preventing accidents ~~conditions~~ and mitigating~~e~~ their radiological consequences should such ~~conditions~~ they occur, as follows:

– ~~FIRST LEVEL~~First level: To ~~P~~prevent deviations from normal operation~~s~~ and to prevent system failures.

– ~~SECOND LEVEL~~Second level: To control (by detection and intervention) deviations from operational states so as to prevent anticipated operational occurrences from escalating to accident conditions.

– ~~THIRD LEVEL~~Third level: To provide ~~for E~~engineered s~~S~~afety f~~F~~eatures ~~(ESF)~~ or inherent safety features, to prevent an escalation ~~from~~ of a d~~D~~esign b~~B~~asis a~~A~~ccident~~s~~ and to achieve a ~~stable and acceptable~~controlled state and then a safe shutdown state following an initiating event. One barrier for the confinement of radioactive material is required to be maintained.

– ~~FOURTH LEVEL~~Fourth level: To address beyond design basis accidents and to ensure that radioactive releases are kept as low as practicable. The objective is the protection of the confinement function.

– ~~FIFTH LEVEL~~Fifth level: To ~~M~~mitigate~~ion~~ ~~of~~ the radiological consequences ~~from~~ of potential releases of radioactive material that may result from accident conditions.

~~7.3.~~6.3. ~~DiD~~ Defence in depth is an important design principle that ~~should~~ is required to be applied in the design of a research reactor of any type or power level.

~~7.4.~~6.4. Defence in depth ~~DiD should~~ is required to be applied with account taken of the graded approach ~~(see para. 2.6 of Ref. [1]) recognizing~~ and the fact that many low power research reactors do not qualify for the fourth or fifth level of defence in depth (see para. 2.6 of Ref. [1]). In addition, the defence in depth ~~DiD~~ concept ~~should~~ is required to be applied in the design to provide graded protection against various reactor transients, including transients resulting from equipment failure and human error and from internal or external events that could lead to a d~~D~~esign b~~B~~asis a~~A~~ccident ~~(DBA),~~ (see para. 6.6~~.~~ of Ref. [1]).

Safety ~~F~~functions

~~7.5.~~6.5. Requirements for the design of safety systems are ~~presented~~ established in para. 6.10 of Ref. [1]: “In the design of the safety systems, including engineered safety features, that are used to achieve the three basic safety functions – shutting down the reactor, cooling, in particular the reactor core, and confining radioactive material –~~;~~ the single failure criterion shall be applied, high reliability shall be ensured and provisions shall be included to facilitate regular inspection, testing and maintenance.”

~~7.6.~~6.6. The application of grading to the three basic safety functions ~~are~~ is discussed ~~below~~in the following ~~with respect to grading~~:

a) Shutting down the reactor~~Function~~

(i) In general, the ~~basic safety function of requiring the reactor~~capability to shut down the reactor when ~~required~~necessary~~,~~ ~~is not gradable~~cannot be graded, although the ~~extent~~ size of the sub-criticality margin available and the ~~required~~ speed of response required of the shutdown system may vary according to the reactor design.

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

b) ~~Core~~ Cooling, in particular ~~the~~ reactor core cooling

(i) In general, this basic safety function ~~is not gradable~~cannot be graded, although the extent of the requirements on 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~~ necessary in one facility, whereas in other facilities, such as critical assemblies, ~~all~~ fission heat ~~may might~~can be adequately removed by natural convection cooling.

(ii) Similarly for decay heat removal following shutdown, the extent of the cooling system requirements will vary according to the reactor design and a forced convection cooling system may be ~~needed~~ necessary in one facility, whereas in low power facilities decay heat ~~may might~~can be adequately removed by natural convection cooling~~.~~, depending on the reactor design .

(iii)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; other~~s~~ facilities may not need an ~~ECCS~~emergency core cooling system.

c) Confining radioactive material

(i) Systems for confining radioactive material ~~may~~ can be graded (see the description of the fourth level of defence in depth in para. 6.2~~, (Level 4~~).

Acceptance c~~C~~riteria[17] *and ~~D~~design ~~R~~rules*

~~7.7.~~6.7. Basic acceptance criteria are defined by the regulatory body. Specific acceptance criteria ~~may~~ can be defined by the designer in advance of the final design and agreed by the regulatory body~~,~~ (see Ref. [23~~17~~], ~~Chapters~~Section 4). In principle such ~~they~~ specific acceptance criteria are not graded, ~~being~~ as they are fixed by the specific facility characteristics. However, the way ~~they~~ that such acceptance criteria are met ~~in~~ by the design ~~is~~ can be grad~~able~~ed, as indicated ~~below~~in the following.

~~7.8.~~6.8. For the design of SSCs, acceptance criteria may be ~~used~~ set in the form of engineering design rules. These rules include regulatory requirements as well as requirements established in relevant codes and standards, which ~~may~~ can be graded on a case by case basis. This is discussed in paras 6.9 and 6.10.

General requirements for design

Classification of SSCs

[17] Acceptance c~~C~~riteria~~:~~ are ~~S~~specified bounds on the value of a functional indicator or condition indicator used to assess the ability of a structure, system or component to perform its design function [2].

~~7.9.~~6.9. The requirements for classification of SSCs are ~~presented~~ established in paras 6.12 ~~to~~ and 6.13 of Ref. [1]. The method for ~~grading~~ determining the safety significance of SSCs~~,~~ should be based on deterministic methods, complemented by probabilistic methods and engineering judgement (see para 2.3~~2)~~ and also Ref. [24~~46~~]).

Codes and s~~S~~tandards

~~7.10.~~6.10. The requirements for codes and standards are ~~presented~~ established in paras 6.14 to 6.15 of Ref. [1]. Codes and standards have been developed ~~which~~ that provide guidance in the design of SSCs. These codes and standards ~~may~~ could be regulatory, international[18], national, or even local[19]. They ~~may~~ can be highly specialized (e.g.~~,~~ an industrial code for the design of a pump, or a code for the design of a pump in a nuclear application)~~;~~ or can be based on ~~the~~ management system procedures and/or performance requirements ~~because of its~~relating to the application of the component (e.g.~~,~~ an electronic component used in the protection system of a research reactor).

~~7.11.~~6.11. The codes and standards used in the design of SSCs should be appropriately selected using a graded approach that tak~~esing~~ into account the safety classification of SSCs [24] and the potential ~~radiation~~ radiological hazard ~~of~~ associated with the research reactor.

Design ~~B~~basis

~~7.12.~~6.12. The requirements for the design basis are ~~presented~~ established in paras 6.16 to 6.34 of Ref. [1]. ~~Potential c~~Challenges that the research reactor ~~may~~ might be expected to face during its operational lifetime ~~should~~ are required to be taken into consideration in the design. These challenges are represented by ~~the~~ a list of selected p~~P~~ostulated ~~I~~initiating e~~E~~vents ~~(PIEs), a selected list of events~~, an example of which is included ~~as an~~in the a~~A~~ppendix ~~in~~ of Ref. [1]. Design requirements will be supported by design limit specifications for all relevant parameters for all operational states and design basis accidents. The requirements and limitations then form the basis of a practicable set of operating limits and conditions for reactor operation. ~~Application of tThe requirement for technical specifications is not gradablecannot be graded, but the level and of detail needed necessary will can be gradedable.~~

~~7.13.~~6.13. The classification of the SSCs, based on their importance to safety, should be utilized to establish the design requirements so that challenges stemming from~~for~~ the postulated initiating events~~PIEs,~~ without exceeding authorized limits. Reference~~.~~ [1] para. ~~[~~6.17~~]~~ ~~requires~~states that~~:~~ "~~"~~It shall be shown that the set of postulated initiating events

[18] Such as the IAEA s~~S~~afety ~~S~~standards.

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

selected covers all credible accidents that may affect the safety of the research reactors. In particular the [d~~D~~esign b~~B~~asis a~~A~~ccidents] ~~(DBAs)~~ shall be identified~~"~~." Application of ~~T~~the requirement to identify the postulated initiating events ~~PIEs~~ and the design basis accidents ~~DBAs~~ for research reactors ~~is not gradable~~cannot be graded. The postulated initiating events ~~PIEs~~ and design basis accidents ~~DBAs~~ should be identified using current safety standards and feedback of operational experience ~~feedback., but~~ However, the extent of the postulated initiating events ~~PIEs~~ and the design basis accidents considered for the reactor ~~DBAs is~~ can be ~~howeve~~graded~~able~~.

~~7.14.~~6.14. ~~The requirements established in Ref. [1] should be analyszed 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~~ D~~d~~evelopment of the design basis can be graded in the sense that ~~the design basis for reactors posing different potential hazards will have~~ a different set of applicable ~~DBAs~~ design basis accidents, based on the specific hazards posed, will be considered in the design basis for each reactor ~~.~~ ~~The h~~Higher power research reactors with significant in-core experimental facilities such as loops will require a greater number of higher safety class ~~importance~~ SSCs.

Design for ~~R~~reliability

~~7.15.~~6.15. The requirements for design for reliability are ~~presented~~ established in paras 6.35-6.43 of Ref. [1]. Design for reliability ~~may~~ requires application of the ~~use~~ principles of redundancy, diversity, independence and fail-safe ~~criteria~~design. These measures should be ~~used~~ applied using~~in~~ a graded ~~way~~ approach to ensure the required reliability of SSCs in accordance with the safety function to be performed by ~~the~~ each SSC~~s~~. In the design of a research reactor, the required reliability of SSCs ~~may~~ can be related to the expected utilization of the facility, and grading ~~may~~ can be employed to achieve operational reliability. Where ~~an~~ automatic or passive ~~/ inherent~~ performance of a safety function is required or an inherent safety feature is used, a minimum ~~reliability~~ requirement for the reliability of the associated SSC should be established and maintained. Depending on the type of the research reactor, performance of one or more of the following safety functions may ~~be~~ need~~ed~~ to be automatic: reactor shutdown, initiation of emergency core cooling ~~initiation~~, and isolation of radioactive material by the containment or other means of confinement~~/containment isolation.~~

Design for c~~C~~ommissioning

~~7.16.~~6.16. The requirements for ~~the~~ design for commissioning are ~~presented~~ established in para. 6.44 of Ref. [1] which states that "~~"~~The design shall include design features as necessary to facilitate the commissioning process for the reactor~~"~~." The design basis of the reactor provides information on the tests and measurements that should be employed in the commissioning process. This information should be used to anticipate difficulties in carrying out the tests and measurements and to provide for ~~them~~ such testing and measurement in the design.

~~7.17.~~6.17. Grading ~~may~~ can be applied in the selection of design features to facilitate commissioning~~to be included in the design~~, in accordance with the importance to safety of the ~~considered~~ system to be commissioned and the associated difficulties of conducting the commissioning tests and measurements.

Provision for inspection, testing and maintenance

~~7.18.~~6.18. The requirements ~~presented~~ established in paras 6.45 to 6.47 of Ref. [1] include a requirement to make provisions to facilitate in-service inspection for determining the conditions of SSCs subject to corrosion, erosion, fatigue or other ageing effects.

~~7.19.~~6.19. Where the performance of inspection, testing and maintenance takes place in ~~radiation~~ controlled areas, it is ~~necessary~~ required to ensure that occupational doses to workers ~~are~~ will be below~~within~~ the authorized limits (Ref. [1], para. 7.93). This ~~is not gradable~~cannot be graded.

~~7.20.~~6.20. Grading can be applied to ~~T~~the inventory of spare parts and components ~~is gradable based~~ on the basis of the ease of procurement of such components from vendors, and budget rules and considerations, see Ref. [6~~5~~], para. 2.44. Controls for the ~~Gradable~~ procurement process that can be graded ~~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~~ consideration should be given to components of systems important to safety having a high obsolescence rate (such as computerized systems or ~~I&C~~instrumentation and control systems).

~~7.21.~~6.21. Grading ~~may~~ can be applied in determining the provision for inspection, testing and maintenance~~ the design stage~~ in two steps:

(1) First~~ly~~, ~~determine~~ the types and frequencies of the required inspections, tests and maintenance operations should be determined, with~~taking into~~ account taken of the importance to safety of the SSC and its required reliability and all the effects that may cause progressive deterioration of the ~~system~~SSC.

(2) Second~~ly~~, ~~specify~~ the provisions ~~that should~~to be included in the design to facilitate the performance of these inspections, tests and maintenance operations should be specified, with~~taking into~~ account taken of the frequency, the ~~radiological~~ radiation protection implications and the complexity of the inspection, test ~~and~~ or maintenance operation. These provisions include accessibility, shielding, remote handling and in-situ inspection, self-testing circuits in electrical and electronic systems, and provisions for easy decontamination and for non-destructive testing.

Design for emergency planning

~~7.22.~~6.22. The requirements for ~~the~~ design for emergency planning~~ implementation~~ are ~~presented~~ established in paras 6.48 to 6.49 of Ref. [1].

~~7.23.~~6.23. These specific design features include alarm systems, communication and public-address systems, illuminated escape routes, designated gathering places, on-site and off-site surveillance systems with provision for remote readout, and other means to facilitate early assessment of the situation and efficient response. While ~~the design should consider~~ all of these features should be considered in the design, grading can be applied to ~~is possible in~~ many of the features. For example:

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

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

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

(d) The ~~S~~scope and frequency of emergency drills can be graded;~~.~~

(e) Safety analysis should be performed to determine t~~T~~he need for a supplementary control room~~, if justified by the safety analysis,~~ and the degree of automatic and/or manual control ~~needed~~necessary can be graded.

Design for d~~D~~ecommissioning

~~7.24.~~6.24. The requirements for ~~the~~ design for decommissioning are ~~presented~~ established in paras 6.50 to 6.51 of Ref. [1]. "Attention ~~should~~ shall be given to keeping ~~doses to~~the radiation exposure of personnel and ~~to~~ of the public ~~to acceptable levels~~as low as reasonably achievable and to ensuring adequate protection of the environment from undue radioactive contamination" arising from ~~the~~ decommissioning activities.

~~7.25.~~6.25. Grading ~~may~~ can be applied in the selection of the design features to meet the requirements for radiation protection ~~goals~~of workers, the public and the environment. 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 disposal facilities 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 disposal facilities will be an important consideration.

Design for r~~R~~adiation p~~P~~rotection

~~7.26.~~6.26. The requirements for ~~the~~ design for radiation protection are ~~presented~~ established 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~~"~~."[20]

~~7.27.~~6.27. Grading ~~may~~ can be applied in the choice of ~~the~~ radiation protection design features ~~for the SSCs employed~~ used to satisfy the requirements of paras 6.52 to 6.59 of Ref. [1], including their effective placement in the facility~~, by considering their feasibility ease of use and their effectiveness~~. In general, the scope of radiation protection design provisions ~~included in~~for a high power level multi-purpose facility will be more extensive and more complex than ~~that~~those for~~in~~ a small research reactor with limited utilization possibilities and low potential for significant exposure. (See also para. 6.57~~6.57~~ of this ~~publication~~Safety Guide).

Human factors and ergonomic considerations

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

6.28. The requirements for the human factors and ergonomic considerations are ~~presented~~ established in paras 6.61 to 6.64 of Ref. [1]. Grading can be applied to ~~applied to human factors, such as operator response requirements, by Tthe use of ergonomic principles and by giving due consideration to human factor principles and the 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 application of a gradeding approach~~ human factors and ergonomics ~~considerations~~by giving particular consideration to the following: ~~are the frequency of usage of a system, and such pertinent human aspects such as the need~~

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

~~7.28. for written procedures [?unclear??] writing, fatigue and working in demanding conditions. For Ssome facilities, will have licenslicence requirements for will specify minimum staffing levels for reactor operators and facility support personnel (e.g. radiation protection personnel and maintenance personnel) that who must be present on the site at particular times, such as at all times when fuel is in the reactor.~~

Provision for utilization and modification

~~7.29.~~6.29. The requirements for ~~the~~ design for utilization and modification are ~~presented~~ established in paras 6.65 to 6.67 of Ref. [1]. ~~"~~Research reactors are flexible in nature ~~and they may can be in various different states"~~ and ~~they~~ are used for a variety of purposes.

~~7.30.~~6.30. The main precautions concerning provisions for utilization and modification taken in design are:

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

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

(c) ~~that Experiments and modifications they should be performed within the authorized operating envelope or, if not, are given explicit consideration should be given to ensure that appropriate safety measures are in place.~~ Where experimental devices penetrate the reactor vessel or reactor core boundaries they are required to be designed to preserve the means of confinement and reactor shielding. Protection systems for experiments are required to be designed to protect the experiment and the reactor.

~~7.31.~~6.31. It is therefore necessary that these aspects of utilization are taken into account or analys~~z~~ed at the design stage and that appropriate provisions are made in the design to ensure safety. Such provisions and the design of modifications and experimental facilities should be subject to grading in the same way that grading is applied in the design of other SSCs, i.e. on the basis of their importance to safety, their complexity, their maturity, and the scope of analysis and of commissioning tests ~~needed~~ necessary to verify their acceptability.

Selection and ageing of materials

~~7.32.~~6.32. Requirements on ~~T~~the selection and ageing of materials ~~is discussed~~are established in paras 6.68 to 6.70 of Ref. [1]. Ageing management in the design ~~focuses~~ involves the use of~~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~~can be graded, ~~based~~ on the basis of the safety significance of the SSCs and their ease of replacement.[21]

~~7.33.~~6.33. In applying a g~~G~~raded~~ing~~ approach~~, should~~ consideration should be given to the utilization and anticipated lifetime of the reactor facility. Facilities with a long expected lifetime (e.g.~~,~~ 30 to ~~-~~40 years) will need to include provision for ageing management in the design of SSCs, and also to provide for the knowledge management ~~needed~~ necessary to support this aspect (see also para. 7.66 of this Safety Guide). ~~Particularly important material~~

[21] Proper selection of equipment and materials and design principles ~~may be used to~~will help to reduce the need~~s~~ to ~~update~~ replace SSCs~~them~~ due to ~~high rate of~~ obsolescence.

~~ageing concerns are corrosion in reactor tanks and vessels, where leak detection can be difficult and repair or replacement may might not be practical. Similarly, the management of corrosion of inaccessible primary coolant piping and associated components are is of key importance for reactor longevity. An important knowledge management area, which supplementsing the original appropriate selection of materials selection and ageing the management of ageing related effects, is the need in recent years for improved human resources management to address the ageing of research reactor personnelnuclear workforce. Other key knowledge management areas are configuration management, document control, and programmes for feedback of operating experience programmes.~~

Provision for extended shutdown

~~7.34.~~6.34. Requirements on ~~P~~provision for extended shutdown ~~is discussed~~ are established in para. 6.71 of Ref. [1]. These provisions will depend on the anticipated duration of the extended shutdown. A graded approach ~~is~~ can be used in designing such provisions. For ~~A~~all SSCs that are important to safety and which could suffer some degradation during the extended shutdown period~~, should include~~ provision~~s~~ should be made for inspection, testing, maintaining, dismounting, and disassembling during the shutdown period. It may be more convenient to remove equipment than to implement a preservation programme with the equipment in place; this decision is usually linked to the future of the research reactor.

~~7.35.~~6.35. Research reactor designs normally include facilities necessary to ensure safety during shutdown of the facility and these facilities ~~may~~ might be used during extended shutdown ~~conditions~~. The design of such~~se~~ facilities ~~may~~ can be graded~~ during design~~.[22]

Safety Analysis

~~7.36.~~6.36. The requirements for safety analysis are ~~specified~~ established in paras 6.72 to 6.78 of Ref. [1] and related recommendations are provided in Ref. [~~7~~11]. Safety analysis is required to~~ and~~ include analyse~~i~~s of the response of the reactor to a wide range of ~~PIEs~~postulated initiating events. The ~~Completeness of the PIEs~~list of postulated initiating events is required to be complete, ~~which should enveloped by the analyzed events~~covering all credible accidents.~~, and the conservatism of the assumptions on the effectiveness of preventive and mitigatoryive features should is required. to be demonstrated. The s~~Safety

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

analysis is a fundamental part of the design process, and is the basis for determining the safety ~~importance~~ significance of the SSCs~~ and the extent that to which features to control the potential hazards can be graded~~. It is also the basis for demonstrating the ~~licensability~~ safety of the proposed design in support of an application for a licence, and should be used to confirm and validate that ~~the~~ grading of application of the requirements has been performed in a consistent and balanced way.

~~7.37.~~6.37. Grading ~~may~~ can be applied to the scope and depth of the safety analysis, (see ~~Ref. [17],~~ Section 1.3 and Annex I of Ref. [23] and ~~Ref. [28]~~ paras 3.1 to 3.7 of Ref. [7]).~~ The applicability of the analysis methods needs tois required to be justifiedverified, but the effort for such justification verification~~The selection of analysis methods ~~may~~ can be graded. The use of enveloping events ~~may~~ can also be graded. For example:

(a) The analysis required for a small facility with a relatively small number of SSCs and applicable ~~PIEs~~ postulated initiating events would be much simpler than that for a large and complex facility. A low -power reactor having a limited hazard potential may require less analytical detail than a higher power level research reactor.

(b) Analysis may demonstrate that for some identified ~~PIEs~~ postulated initiating events there can be no release of radioactive material~~s~~ from the core, which would eliminat~~eing~~ the need for extensive e~~E~~ngineered s~~S~~afety f~~F~~eatures ~~(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) The use of ~~C~~conservative methods and criteria ~~are~~ is a means of simplifying the safety analysis. Facilities ~~of~~ with small potential hazard may use conservative criteria in safety analysis, with low impact on the facility design and operation or cost.

(e) The process of ~~development~~ preparation of the safety analysis report allows for the definition and refinement of the ~~PIEs~~ postulated initiating events and ~~ESFs~~engineered safety features, and ~~is an important element to~~ should be graded ~~during the design phase~~.

Specific requirements for design

The reactor core and reactivity control system

~~7.38.~~6.38. The requirements for the reactor core and reactivity control system are ~~specified~~ established in paras 6.79 to 6.89 of Ref. [1]. The design requirements relating~~ed~~ to the design of the reactor core as a whole and of its individual components~~,~~ (i.e. the fuel assemblies, the reactivity control ~~system~~mechanisms, reflectors, experimental devices, cooling channels and structural parts) are concerned with ensuring that the reactor can be shut down, cooled and ~~held~~ maintained sub-critical with an adequate shutdown margin for all operational states and ~~DBAs~~design basis accidents.

~~7.39.~~6.39. ~~The~~ A graded approach should be applied in the design of the core by considering the ~~effects~~ conditions that ~~these~~ the core components ~~must~~ will need to ~~meet~~ withstand in the course of their intended service liv~~f~~es in the core. The effects of these conditions, such as integrated neutron flux, thermal and mechanical stresses and chemical compatibility, on various materials and fuel assembly types are generally well known. The extent of analyses and experiments ~~needed~~ necessary to demonstrate the acceptability of a particular design ~~may~~ could be substantially smaller than that ~~which is required~~necessary for reactors ~~which~~ that make use of new types of fuel assemblies~~,~~ and/or novel experimental setups. A similar situation ~~may~~ can be found in relation to the reactor power:~~;~~ for smaller reactor powers ~~shown to~~for which it is demonstrated that there is a ~~present~~ smaller risk potential~~, may need~~ substantially less extensive analysis~~,~~ and simplified conservative criteria might be adequate.

The reactor shutdown system

~~7.40.~~6.40. The requirements for the reactor shutdown system are ~~specified~~ established in paras 6.90 to 6.94 of Ref. [1]. The reactor shutdown system fulfils a crucial safety function for ~~all~~ research reactors of all types and sizes. Therefore, ~~all~~ the design requirements established ~~specified~~ in paras 6.90 to 6.94 of Ref. [1] ~~should be fully met~~cannot be graded.

~~7.41.~~6.41. However, ~~G~~grading ~~may~~ can be applied in ~~deciding~~ determining how many redundant shutdown channels are ~~needed~~ necessary and the extent of instrumentation required for monitoring the state of the shutdown system~~,~~ (see Ref. [~~17~~23] ~~Chapter~~Section 3).

~~7.42.~~6.42. The need for ~~A~~a second, ~~and diverse~~ independent shutdown system ~~should~~ is required to be considered for research reactors ~~conducting~~ in which experiments with major safety significance are conducted that could affect, in the event of an accident, the first shutdown system, unless inherent self limiting properties of the design of the core~~/~~ or fuel ~~design~~ would prevent~~s~~ a damaging reactivity excursion under all foreseeable reactor states.

The reactor protection system

~~7.43.~~6.43. The requirements for the reactor protection system are ~~presented~~ established in paras 6.95 to 6.105 of Ref. [1]. The reactor protection system is required to be capable of automatically initiating~~e~~ the required protective actions for the full range of ~~identified PIEs~~ postulated initiating events to terminate the event safely. Consequently, the system has to be reliable, ~~utilizing, as required,~~and redundancy and independence are required to be applied in its design, to ensure that no single failure or common cause failure in the system could result in the loss of automatic protective actions. If there is a high level of confidence that there are no identified postulated initiating events ~~PIEs~~ requiring automatic shutdown, manual operator action could be considered to be sufficiently reliable, as explained in para. 6.96 of Ref. [1]. ~~There should be Aa high level of confidence in the correctness of this determination is required.~~

~~7.44.~~6.44. ~~Grading may be possible applied in~~ T~~t~~he reactor protection system can be graded in the sense that two different research reactors may face different postulated initiating events~~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) ~~a~~At sites ~~which~~ that could be ~~impacted~~ affected by significant seismic events, a seismic sensor may be required to shutdown the reactor, while at other sites~~,~~ such protection ~~is~~ would not be ~~needed~~necessary;

(b) I~~i~~nitiation of emergency core cooling may be ~~needed~~ necessary ~~in~~ for certain reactors, while in others it ~~is~~ would not ~~needed~~ be necessary (see para.~~s~~ 6.6~~6.6 and 6.59~~).

The reactor coolant system~~s~~ and related systems

~~7.45.~~6.45. The requirements for the reactor coolant system~~s~~ and related systems are ~~specified~~ established in paras 6.106 to 6.119 of Ref. [1]. Cooling is one of the basic safety functions~~, as discussed in Seepara. 6.65 of this publicationSafety Guide~~. The coolant system is required to be designed to provide adequate cooling to the reactor with an acceptable and demonstrated margin. Adequate cooling is required not only during normal operation at the authorized power levels, but also~~,~~ after shutdown, under a range of anticipated operational occurrences~~, postulated accidents~~ and d~~D~~esign B~~b~~asis A~~a~~ccidents ~~(DBAs)~~ that involve loss of flow ~~and~~ or loss of coolant transients. Grading can be ~~used in~~applied to the design of the cooling system. ~~This~~ The coolant system can range from the provision of forced cooling with emergency electrical power being available to power some or all of the main coolant pumps, to no emergency power for any of the coolant pumps, to a system where natural ~~circulation~~ convection cooling is ~~adequate~~ used for both heat removal under full power operation as well as decay heat removal. Cooling by natural convection might be adequate for some small research reactors.

Means of confinement

~~7.46.~~6.46. The requirements for the means of confinement are ~~specified~~ established in paras 6.120 to ~~-~~6.130 of Ref. [1]. Confinement is one of the basic safety functions, as discussed in para. 6.6~~6.6~~ of this ~~publication~~Safety Guide. Means of confinement are required to be provided to prevent or mitigate an unplanned release of radioactive material in operational states or in ~~accident conditions (DBA~~ design basis accidents~~and BDBA)~~. The basic design requirement is to ensure that a ~~the~~ release to the environment does not exceed acceptable limits for all accidents taken into account in the design. ~~It is the~~ The results of safety analysis should be used to determine~~which identifies~~ how and to what extent the ~~confinement~~ design of the means of confinement ~~should~~ can be graded, e.g.~~ by considering the potential release from the reactor will determine the confinement design and the need for~~ whether volatile fission product (e.g. iodine) traps are necessary in the event of a release of fission products from the reactor. ~~An example of the use of these such considerations as a basis for grading is presented in para. 6.4 of this publicationSafety Guide.~~

Experimental devices

~~7.47.~~6.47. The requirements for experimental devices are ~~specified~~ established 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.

~~7.48.~~6.48. Grading can be applied to ~~T~~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~~instrumentation and control system~~ should be subject to grading~~. Grading can also be applied to ~~T~~the monitoring of the experimental devices from the control room(s)~~ is also subject to grading~~.

~~7.49.~~6.49. Grading can be applied to the design, analysis~~,~~ and the authorization process, in accordance with the type~~s~~ and magnitude~~s~~ of the anticipated hazards, ~~taking into~~ with account taken of the complexity of the experiment and the familiarity of reactor personnel ~~(based on experience)~~ with ~~its performance~~the experiment.

Instrumentation and control

~~7.50.~~6.50. The requirements for ~~the~~ instrumentation and control ~~(I&C)~~ are ~~specified~~ established in paras 6.136 to 6.144 of Ref. [1]. The basic ~~(I&C)~~ design requirements for instrumentation and control in this respect are to include in the design sufficient instrumentation ~~for the purpose~~to monitor safety related reactor parameters, with reliability commensurate with the importance to safety of the system. The grading of the instrumentation and control ~~I&C~~ system~~s~~ should be based on ~~a~~ careful definition of the d~~D~~esign b~~B~~asis. Due consideration should be given to the maintainability of the system and its associated cost.

~~7.51.~~6.51. A ~~G~~graded~~ing~~ approach should be ~~performed~~ taken in determining the types of measurement, ~~places~~locations of measurement~~,~~ and number of measurements to be taken of reactor parameters, such as temperature, pressure, flow, pool/tank water level, gamma radiation, neutron flux and water chemistry. ~~System requirement~~Technical specifications covering all operational states and accident conditions~~,~~ should provide ~~for~~ the basis for grading of the design of the instrumentation and control ~~I&C~~ system~~s~~. A typical example is the measurement of pressure drop across the core. This is a ~~safety~~ measurement ~~implemented~~ performed in many reactors in order to detect reduced flow through the core (~~either~~ due to either a by~~-~~pass or ~~to~~ a blockage);~~:~~ this measurement is~~n~~ generally ~~is~~ not ~~needed~~ necessary in a critical assembly or in a reactor operating in a natural convection cooling mode.

~~7.52.~~6.52. Another means of grading instrumentation and control ~~I&C~~ systems is by means of the choice of the level of redundancy. Triple and quadruple channel redundancy is often used for research reactors that need to operate continuously, in order to minimize spurious trips and to allow for testing and/or maintenance ~~on power~~ of instrumentation and control ~~I&C~~ equipment during operation at- power. For research reactors that operate for only a few hours per week~~,~~ or less frequently, such as critical assemblies ~~for example~~, a lower level ~~(~~i.e. two channel~~,~~ (one-out-of-two) redundancy ~~may~~ can be ~~selected~~applied, thus reducing design and operational complexity as well as costs.

~~7.53.~~6.53. The level of reliability as well as the accuracy required for measurement~~s~~ of ~~the~~ relevant parameters will depend on the importance to safety of the instrumentation and control equipment ~~and process requirements~~.

~~7.54.~~6.54. The degree of automation required for the control system, including the extent of manual control provided, can be graded.

~~7.55.~~6.55. The instrumentation and control ~~I&C~~ system ~~should~~ is required to monitor reactor parameters and to allow for appropriate response ~~for~~ to anticipated operational occurrences and ~~DBAs~~design basis accidents. If analysis shows that in some situations the main control room ~~can not~~ cannot be occupied, then a supplementary control room, separated and functionally independent[23] from the main control room, ~~should~~ is required to be provided in the design. The design and equipment of this ~~secondary~~ supplementary control room ~~is~~ can also ~~gradable~~ be graded in accordance~~ing to~~ with the reactor characteristics and foreseen accident conditions. If the need for ~~an emergency~~a supplementary control room is confirmed, ~~then~~ there should be an analysis of its operational requirements and, in particular, the parameters to be ~~supervised~~ monitored and controlled and the actions ~~required~~ necessary to maintain the reactor in a safe shutdown state. Typical features that ~~may~~ can be included in the supplementary control room, depending~~according to documented~~ on requirements, are: radiation monitors, fire detection systems and actuators of fire extinguishers, communication means, features for control of the ventilation system~~ control~~, scram and/or safe shutdown features, features for operation of experimental devices, and features for operation of cooling systems.

~~7.56.~~6.56. A complex and costly human machine interface ~~in a low power level research reactor facility may~~ might not be justified for a low power level research reactor facility.

Radiation protection systems

~~7.57.~~6.57. The requirements for ~~the~~ radiation protection systems are ~~specified~~ established in paras 6.145 to~~-~~ 6.148 of Ref. [1]. To achieve the basic requirement ~~of~~ established para. 2.2 of Ref. [1], as discussed in paras 6.26~~6.26~~ to 6.27~~6.27~~ of this ~~publication~~ Safety Guide, a wide range of radiation protection systems are required to be provided in the design "to ensure adequate monitoring for radiation protection purposes in operational states, ~~and accident conditions (~~d~~D~~esign b~~B~~asis ~~A~~accidents~~, DBAs,~~ and, as practicable, ~~B~~beyond d~~D~~esign b~~B~~asis a~~A~~ccidents~~, BDBAs~~". Paragraph~~.~~ 6.145 of Ref. [1] lists the radiation protection systems used in research reactor facilities and the purposes they serve. All these systems are likely to be required for research reactors. Grading ~~may~~ can be applied in determining the ~~level~~ degree to which these systems will provide~~of~~ adequate~~cy~~ monitoring for a specific facility.

~~7.58.~~6.58. ~~For e~~Examples of considerations in the grading of radiation monitoring are provided in the following:

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

(b) A research reactor with various experimental devices, such as~~:~~ beam tubes and neutron guides, neutron activation analysis~~ (NAA)~~, and radioisotope production

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

~~(RIP)~~ facilities, should ~~require~~ employ neutron and gamma monitors for the beam tubes and neutron guides and instruments, gamma monitors in the ~~NAA~~ neutron activation analysis facility and in the ~~RIP~~ radioisotope production handling systems ~~as well as~~and equipment for monitoring of contamination ~~monitoring~~.

(c) A low power reactor without beam tubes used only for teaching purposes would need only limited and basic monitoring equipment, such as gamma monitors at the open pool end or in the control console and contamination monitors.

(d) For high power level reactors, supplementary monitoring displays outside the control room should be ~~required~~ employed for displaying and recording radiation conditions at specific locations in the facility for ~~normal~~ operational states and accident conditions (~~large~~ wide range monitoring). Such additional radiation monitoring locations ~~may~~ might not be ~~required~~ necessary for very low power level facilities (less than~~<~~ 50 kW).

Fuel handling and storage system

~~7.59.~~6.59. The requirements for the fuel handling and storage system are ~~specified~~ established in paras 6.149 to ~~-~~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. ~~The scale at which the Rrequirements relatinged to the prevention of damage and to ensuring provision of security physical protection against theft and sabotage may be equally applicable to many research reactors, the only difference being that of scaleare applied can also be graded.~~

~~7.60.~~6.60. The application of the requirements to different reactors ~~may~~ can be graded in several aspects, in accordance~~ing to~~ with the design of the reactor and its utilization programme. 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~~ need special storage places for temporary storage of these assemblies;

(c) ~~Requirements~~ The need for decay heat removal may ~~vary~~differ for different reactors, leading to ~~requiring~~ different provisions in the design for decay heat removal.~~;~~

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

~~7.61.~~6.61. The requirements for ~~the~~ electrical power supply systems are ~~specified~~ established 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.

~~7.62.~~6.62. Consideration is required to be given to the need for an emergency electrical power system to back up the off-site power supply system. Grading ~~may~~ can be applied in the design of the power supply system and the emergency power supply system. Considerations relevant for grading include~~:~~ the type and number of safety functions~~,~~ and ~~ESFs~~engineering safety features~~,~~ for which emergency power ~~is~~ would be required. The reliability requirements ~~may~~ might be different for different reactors, for the various utilization programmes ~~for the same~~ of a particular 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.~~ In a g~~G~~raded~~ing~~ approach ~~would consider~~ the number, size~~,~~ and reliability of any necessary emergency power supply systems should be considered. ~~Examples would include the control system, the radiation protection system, the radiation monitoring system and the means for decay heat removal:~~.

a) A reactor may or may not need forced ~~circulation~~ convection cooling after shutdown. The emergency power supply ~~requirement~~ needs and the length of time ~~needed~~ after shutdown ~~to operate~~ that ~~theis~~ cooling ~~system~~ will be required will determine the specifications of the emergency power supply system. Depending on the reactor power, the power density and the~~,~~ duty cycle, this time could be hours, days~~,~~ or weeks, and this will provide input into determining the necessary ~~giving rise to~~ reliability of the emergency power supply system ~~considerations~~. Reliability requirements, in general, may call for a degree of redundancy and separation in the design, which will ~~that~~ depend~~s~~ on the ~~DBAs~~ design basis accidents ~~postulated~~ for the facility.

b) The reactor power will determine the extent of ~~process~~ cooling water requirements for power operation and decay heat removal.

Radioactive waste systems

~~7.63.~~6.63. The requirements for ~~the~~ radioactive waste systems are ~~specified~~ established in paras 6.162 to 6.166 of Ref. [1]. Radioactive ~~materials~~ waste (in solid, liquid and gaseous forms) ~~are~~ is generated from fuel, neutron and gamma irradiation of reactor core components and coolants, in-core experiments and irradiation facilities, and also ~~from~~ as operational waste[24].

~~7.64.~~6.64. A graded approach can be applied to ~~Tthe specific requirements for~~the handling, processing, storage, transport and disposal of radioactive waste and ~~the~~ for control and monitoring of solid, liquid and gaseous effluent discharges ~~are all gradable~~ and grading ~~should~~ can be related to the types and quantities of radioactive waste generated in the specific reactor facility. Reference~~.~~ [25~~44~~] provides information on grading of performance standards for package designs for the safe transport ~~regulations~~ of radioactive material and the appendix of Ref. [26~~45~~] ~~Appendix~~ provides detailed examples of grading for all aspects of transport of radioactive material. A detailed example of the ~~use~~ application of the graded approach ~~for~~ to the packaging of radioactive material is ~~given~~ provided in the Annex ~~1~~to this Safety Guide, taken from the appendix of Ref. [26~~45~~] ~~Appendix~~.

~~7.65.~~6.65. Grading ~~considerations should~~ can be ~~compatible with~~applied on the basis of safety analysis and in accordance with regulatory requirements, including the requirement for application of defence in depth in the design ~~requirements~~ for different types and quantities of radioactive waste, for example:

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

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

Buildings and structures

~~7.66.~~6.66. The requirements for ~~the~~ buildings and structures are ~~specified~~ established in paras 6.167 to 6.169 of Ref. [1]. The requirements relating~~ed~~ to the design of buildings and structures will depend on their intended safety function~~s~~ and their importance to safety.

~~7.67.~~6.67. The design basis for buildings and structures ~~may~~ can be graded by examining their safety function. For example, the reactor building ~~may~~can ~~be required to act as~~constitute

[24] ~~SOLIDS~~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~~Liquids: primary system ~~cooling~~coolant; 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~~ in active zones. ~~GASEOUS~~Gases: from the reactor tank or pool; from the cooling systems and from irradiation facilities; gases produced by active material created during reactor operation; fission product noble gases; tritium.

a confinement barrier and be designed accordingly. However, different reactor buildings may require different degrees of leak-tightness, which ~~should~~ is required to be determined in accordance with the ~~reactor's~~ safety analysis of the reactor.

~~7.68.~~6.68. Careful design of building and structures ~~should~~ will help in the application of grading ~~in~~ to other systems (or to avoiding costly refurbishment later). For example:

~~7.~~(a) Separation of areas according their potential hazard can ~~and the use of adequate structural building materials[??] can simplify the design of other SSCs (and consequently reduce the grade efforts necessary to construct themrequired of) other SSC and~~ minimize~~ing~~ the need -for radioactive waste handling,~~.~~ contribute to design for radiation protection, design for emergency preparedness and response~~,~~ and design for fire protection, and help to reduce ~~as well as~~ operational costs.

~~8.The architecture of the building should facilitate the provision of the control room and, where appropriatenecessary, anthe emergency supplementary control centreroom.~~

~~9.~~(b) Good site evaluation will help~~s~~ 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 (see Ref. [23], section 2.2.1).

Auxiliary systems

~~7.69.~~6.69. The requirements for ~~the~~ auxiliary systems are ~~specified~~ established 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.

~~7.70.~~6.70. Those auxiliary systems that ~~do not have an effect~~are not important to ~~on nuclear~~ safety ~~may~~can be designed to standards commensurate with good industrial practice.

7. OPERATION

GENERAL

~~8.2.~~7.1. Operation includes all activities performed to achieve the purpose for which the research reactor was designed and constructed or modified. Section 7 of Ref. [1], ~~Chapter 7~~ includes fifteen operational topics, and recommendations on application of a ~~the~~ graded~~ing aspects~~ approach to~~of~~ fourteen of these~~, (omitting physical protection as this is out of scope)~~

are ~~discussed~~ provided in this ~~chapter~~section (application of a graded approach to the requirements for physical protection is not in the scope of this Safety Guide).

APPLICATION OF GRADING TO ORGANIZATIONAL PROVISIONS

~~8.3.~~7.2. The organizational requirements for a research reactor are ~~presented~~ established in paras 7.1 to 7.26 of Ref. [1]. ~~Guidance~~ Recommendations on meeting these requirements ~~is presented~~are provided in Ref. [10~~5~~].

~~8.4.~~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~~ is required to be assigned to the reactor manager. This responsibility ~~should not~~cannot be graded. However, the manner in which the associated functions are performed ~~may~~ can be graded in accordance with their safety significance, maturity and complexity[25].

~~8.5.~~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 that have~~while maintaining~~ the same functionality~~ of those structures~~ can be established. 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, expertise in quality control, a large inventory of spare components, expertise in isotope production~~,~~ and maintenance personnel). A similar research reactor in a ~~MS~~ State with a large infrastructure and nuclear programme may not need such a large in-house capability, because support could be easily obtained from external organizations.

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

i. The ~~N~~number and duties of operating personnel. For reactors with a low potential radiological hazard, an individual ~~may~~ could be assigned multiple duties. However, Ref. [1] requires that duties, responsibilities, experience

[25] A reactor manager of a large research reactor may have under her/his direct authority ~~the~~ a t~~T~~echnical s~~S~~upport ~~G~~group, a ~~S~~safety ~~A~~analysis group, a ~~T~~training ~~G~~group, and a ~~QA~~ quality assurance ~~G~~group for example. Smaller organizations may have similar groups not under the direct authority of the reactor manager. In either case, the reactor manager should always be kept informed and is required to be the person responsible for the implementation of all the relevant programmes and projects and for the safe operation of the reactor.

and lines of communication be documented; application of this requirement ~~is ~~cannot ~~gradable~~be graded;

ii. Membership of and ~~meeting ~~frequency of meetings of the~~a~~ safety committee(s) (see para. 4.9 ~~above~~of this Safety Guide);

iii. ~~Production ~~Preparation and periodic updating of the s~~S~~afety ~~A~~analysis r~~R~~eport (see discussion of the licensing process in para. 3.6 ~~to 3.136 ~~of Ref. [1]);

iv. Training, re~~-~~training and qualification programmes (see paras 7.5 to 7.7 of this Safety Guide);

v. Operating ~~P~~procedures (see paras 7.21 to 7.25 of this Safety Guide);

vi. Maintenance, periodic testing and inspection programmes (see paras 7.26 to 7.33 of this Safety Guide);

vii. Emergency planning and procedures (see para. 7.41 to 7.44 of this Safety Guide);

viii. The ~~R~~radiation protection programme (see paras. 7.51 to 7.56 of this Safety Guide); ~~and~~

ix. The ~~M~~management system (see para. 4.1 of this Safety Guide).

APPLICATION OF GRADING TO TRAINING, RETRAINING AND QUALIFICATION

~~8.6.~~7.5. Requirements for ~~T~~training, retraining and qualification ~~requirements ~~for research reactor staff and other personnel such as experimenters are ~~presented ~~established in paras 7.27 to 7.28 of Ref. [1]. ~~Guidance ~~Recommendations on meeting these requirements ~~is presented~~are provided in Ref. [10~~2~~].

~~8.7.~~7.6. Training, retraining and qualification requirements for research reactor staff and other personnel such as experimenters should be consistent with the complexity of the design, the hazard potential, the planned utilization of the facility, the available infrastructure and other functions that might be assigned to staff and other personnel. The required levels of educational~~ level~~, required ~~experience and ~~operational experience ~~requirements ~~(~~such as~~e.g. the minimum number of hours of operation~~al activity~~ per year) for the various ~~reactor ~~staffing positions and the contents and the duration of training ~~may ~~can be graded in accordance with the above criteria~~,~~ (see Ref. [1~~2~~0], para 1.10).

~~8.8.~~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~~personnel. The requirement that there be adequate training and that it be implemented ~~is not gradable~~cannot be graded. The nature and details of the training ~~is gradable~~can be graded (see Ref. [1~~2~~0], para. 5.13). Reauthorization after absences ~~may~~ can be approached in a graded manner, with retraining, requalification and examinations commensurate with the duration of the absence, the complexity of the facility, and the changes to the facility and its operation during the absence of the individual.

APPLICATION OF GRADING TO OPERATIONAL LIMITS AND CONDITIONS

~~8.9.~~7.8. The requirements for research reactor ~~OLC~~operational limits and conditions[26] are ~~presented~~ established in paras 7.29 to 7.41 of Ref [1]. ~~Guidance~~ Recommendations for the preparation and implementation of ~~OLC~~operational limits and conditions ~~is presented~~are provided in Ref. [27~~9~~].

General

~~8.10.~~7.9. Since the ~~OLC~~operational limits and conditions are based on the reactor design and on the information from the ~~SAR~~safety analysis report concerning conduct of operations, grading will have ~~already taken place~~been employed, as discussed in ~~other s~~Sections 3 and 6 of this ~~publication~~Safety Guide.

Safety ~~L~~limits

~~8.11.~~7.10. The ~~need~~ requirement ~~for~~ to ~~establishing~~set safety limits and corresponding operational limits to protect the integrity of physical barriers cannot be graded. However, the depth of analysis used to establish the limits ~~may~~ can be graded~~able~~.

Safety ~~S~~system ~~s~~Settings

~~8.12.~~7.11. For each safety limit, ~~there should be~~ at least one safety system ~~instrument~~ is required to be put in place~~used~~ to monitor parameters and to provide a signal to ~~cause~~ accomplish an action (e.g.~~,~~ to shut down the reactor) to ~~preclude approaching~~prevent the parameter from ~~exceeding~~approaching the safety limit. The ~~set point~~ safety system setting

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

should be ~~established~~ set ~~to provide~~at an acceptable safety margin ~~between the point of the action and~~from the safety limit. For safety actions of particular importance, such as neutronic trips (scrams), redundant systems should be employed. The analysis performed to establish a suitable safety margin ~~may~~ can be graded, along with the level of redundancy.

~~8.13.~~7.12. Another ~~grading~~ possibility for grading that is related to the redundancy and diversity of instruments lies in the selection~~g~~ of the types and varieties of safety system settings relating~~ed~~ to the safety limits and ~~to~~ the ~~OLC~~operational limits and conditions. For example, in a low power reactor, ~~the safety system setting parameter related to the fuel temperature could be~~ the coolant~~ing~~ outlet temperature could be selected as the parameter relating to the fuel temperature for which a safety system setting is defined, while in a higher power reactor, to prevent ~~from reaching~~ the safety limits from being approached, a complex system of variables should have defined safety system settings, such as the coolant outlet temperature, the inlet temperature, the coolant flow rate, the differential pressure across the core and~~,~~ the primary pump discharge pressure, ~~and~~ as well as parameters from experimental facilities.

Limiting ~~C~~conditions for ~~S~~safe ~~O~~operation

~~8.14.~~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 the safety system settings during start-up, operation, shutting down and shutdown. Appendix I of Ref. [27~~9~~] provides a list of operational parameters and equipment to be considered in establishing limiting conditions for safe operation. Appendix I of Ref. [27] recommends the selection~~g~~ of only the appropriate items, in accordance with the type of reactor and conditions of operation. Grading ~~should~~ can also be applied in the type of analysis performed in establishing a limiting condition for safe operation, on the basis~~ed~~ of~~n~~ the selection from the list in appendix I of Ref. [27] in accordance with the type of reactor and conditions of operation.

Requirements for ~~maintenance~~inspection, periodic testing and ~~inspection~~maintenance

~~8.15.~~7.14. In order to ensure that safety limits and limiting conditions for safe operation are met, the relevant SSCs ~~should~~ are required to be maintained, monitored, inspected, checked, calibrated and tested in accordance with an approved surveillance programme (see paras 7.56 to 7.64 of Ref. [1]). Surveillance requirements ~~specify~~ can the frequency ~~,~~and scope of inspections and acceptance criteria for each SSC. Grading should be used in establishing these requirements on the basis~~ed~~ o~~n~~f the importance to safety and necessary reliability of ~~the~~ each SSC~~s~~. Additional information is provided in paras 7.26 to 7.33~~7.33~~.

Administrative r~~R~~equirements

~~8.16.~~7.15. Administrative requirements include those for the organizational structure and responsibilities, minimum staffing, training and retraining, ~~safety~~ review and ~~verification~~audit~~,~~ procedures, records and reports, and event investigation and follow up (see para. 7.38 of Ref. [1]). The grading ~~which~~ that may be possible in relation to some of these activities is discussed in paras 7.3~~7.3~~ and 7.4~~7.4~~ of this Safety Guide.

~~8.17.~~7.16. The requirement for action after a violation ~~is not gradable~~cannot be graded. The nature of the action ~~is~~ can be graded~~able~~ depending on the severity of the violation, i.e. whether ~~a safety limit or~~ an ~~LCO~~ operational limit and condition has been exceeded.

APPLICATION OF GRADING TO COMMISSIONING

~~8.18.~~7.17. The safety requirements for commissioning ~~a~~ of research reactors are ~~presented~~ established in paras ~~4.5, and~~ 7.42 to 7.50 of Ref. [1]. ~~Guidance~~ Recommendations for commissioning of research reactors ~~commissioning is~~are ~~presented~~ provided in Ref. [28].

~~8.19.~~7.18. The commissioning process itself cannot be graded, in the sense that all SSCs, activities and experiments ~~should~~ are required to be commissioned. However, grading ~~may~~ can be applied to the commissioning programme in the areas of:

(a) O~~o~~rganizational structure;

(b) P~~p~~reparation of procedures;

(c) N~~n~~umber of hold points and tests;

(d) D~~d~~ocumentation;

(e) R~~r~~eporting.

~~8.20.~~7.19. While grading ~~may~~ can be applied ~~in~~ to the number of hold points imposed, ~~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 (see Ref. [28], ~~Appendix~~ para. A.2 of the appendix). The extent and type of tests to be performed ~~being~~ should be determined on the basis of their importance to safety of each item and the overall hazard potential of the reactor.

~~8.21.~~7.20. The principles applied in commissioning for the initial approach to criticality, reactivity device calibrations, neutron flux measurements, determination of core excess reactivity and shut-down margins, for power raising tests and testing of the containment~~/confinement~~ system ~~testing~~ or other means of confinement ~~should be~~are similar for all research reactors.

~~8.22.~~7.21. The requirements for ~~research reactor~~ the operating procedures for research reactors ~~(OPs)~~ are ~~presented~~ established in paras 7.51 to 7.55 of Ref. [1]. ~~Guidance~~ Recommendations for the preparation of operating procedures ~~OPs is presented~~are provided in Ref. [27~~9~~]. Appendix II of Ref. [27~~9~~] presents an ~~extensive~~ indicative list of operating procedures ~~OPs~~ for ~~a~~ research reactors.

~~8.23.~~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 s~~S~~afety a~~A~~nalysis r~~R~~eport and the ~~OLC~~operational limits and conditions. In addition, grading will have been employed in preparation and implementation of the management system ~~programme which~~ that governs the format, development, initial and periodic review, control and~~,~~ training on the use of and implementation of operating procedures.

~~8.24.~~7.23. The list of operating procedures presented in Appendix II of Ref. [~~9~~27] should be graded for applicability to a specific research reactor. Consequently, the number of operating procedures developed will depend~~s~~ upon the research reactor and will be ~~smaller~~ less for simpler reactors with low potential hazard.

~~8.25.~~7.24. Grading ~~should~~ can be applied to ~~the~~ staff training in the use of the operating procedures. However, all personnel using the operating procedures ~~should~~ are required to be thoroughly familiar with them and proficient in their use.

~~8.26.~~7.25. While all operating procedures ~~should [~~are required to be prepared, reviewed and submitted for approval~~ed based~~ on the basis of criteria established by the operating organization and regulatory requirements, operating procedures ~~may~~ can be graded ~~based~~ on the basis of their importance to safety. Several examples are:

(a) The procedure for regeneration of an ion-exchange system for producing the de-mineralized water inventory in a storage tank will be of low safety significance and will involve mature and non-complex technology. The ~~safety~~ implications for safety of an error in the regeneration process are low. Consequently, the operating procedure governing this application ~~itself may~~ can be simplified.

(b) By contrast, an operating procedure ~~that is~~ developed for an application in which an error ~~has the potential implications for safety significance and or [?]~~ could caus~~eing~~ a violation of the ~~OLC~~operational limits and conditions ~~would~~ should be more detailed. An example ~~would be~~is the procedure for regeneration of an ion-exchange system for the primary cooling water purification system. While it ~~may~~ involves the same basic technology as the example in item (a) above, the safety implications of an error in this application could be much more significant (e.g.~~,~~ ~~an error which allowed~~if resin were allowed to enter the primary cooling water and hence enter into the reactor core). Design features and/or procedural arrangements should therefore take into account the greater hazard ~~from mis-~~ associated with operation of this system~~.[“~~ and the development, review and approval of operating procedures governing such safety significant activities should follow a stringent process.

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

APPLICATION OF GRADING TO INSPECTION, PERIODIC TESTING AND MAINTENANCE

~~8.27.~~7.26. The requirements for ~~research reactor~~ maintenance, periodic testing and inspection of research reactors are ~~presented~~ established in paras 7.56 to 7.64 of Ref. [1]. ~~Guidance~~ Recommendations for maintenance, periodic testing and inspection ~~is~~ are ~~presented~~ provided in Ref. [29~~10~~].

~~8.28.~~7.27. Grading can be applied to the frequency of maintenance, periodic testing and inspection of individual SSCs and is required to be adjusted on the basis~~ed~~ of~~n~~ experience and on the importance to safety of the SSC concerned.

~~8.29.~~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~~SSC concerned, to the complexity of the maintenance, testing or inspection activity ~~operation~~ and to the experience of the ~~maintenance~~ staff and their familiarity with the systems ~~to be maintained~~. Grading of procedures ~~was~~ is discussed in paras 7.21 to 7.25 of this Safety Guide.

~~8.30.~~7.29. The period ~~that~~ for which a SSC ~~may~~ is permitted to be out of service while reactor operation continues is usually stated in the ~~OLC~~operational limits and conditions for the research reactor and ~~may~~ can be graded. ~~As a result~~For example, ~~any~~ no outage time whatsoever ~~may not~~might be acceptable for automatic shutdown systems, while outage times of up to several days ~~may~~ might be acceptable for other systems (e.g.~~,~~ for purification system monitoring the primary coolant pH). The allowed~~able~~ outage time will depend on the extent to which safety is impacted, or the ease in applying compensatory measures.

~~8.31.~~7.30. In a similar way, the frequency for periodic testing ~~may~~ can be graded. A balance ~~is necessary~~should be sought between the improvement in ~~unrevealed fault~~ detection of faults ~~due~~ owing to more frequent testing and the risk that testing ~~may~~ could be performed incorrectly and leave the SSC in a degraded state. The testing frequency could also be increased to the point where testing~~ failures~~ causes more frequent failures of SSCs, and so it should be recognized there is always an optimum test frequency. This consideration also applies also for periodic maintenance.

~~8.32.~~7.31. At times it may become necessary to perform maintenance, periodic testing ~~and~~ or inspection in ~~radiation~~ controlled areas or on components ~~which~~ that are radioactive. ~~While~~ Although the procedure for the inspection, periodic testing ~~and~~ or maintenance activity may have been graded, controls should be put in place to ensure that the radiation exposure~~s~~ of workers ~~are~~ is within the prescribed limits. The radiation protection ~~control~~ measures ~~may~~ can be graded ~~based~~ on the basis of the potential for occupational exposure.

~~8.33.~~7.32. When maintenance, periodic testing ~~and~~ or inspection of an SSC is uncomplicated and operating experience indicates a high reliability of the SSC, a review and re-grading of the maintenance, periodic testing or inspection activity leading to a change in the procedure ~~may~~ might be justified. However, a change in the procedure should be subjected to the established preparation, review and approval process.

~~8.34.~~7.33. In weighing the importance to safety, maturity and complexity of ~~some~~ maintenance, periodic testing and inspection activities for grading purposes, it ~~may~~ might be concluded that ~~the required~~ some activities are highly specialized and involv~~eing~~ complex and sophisticated techniques. Such activities are often performed by contracted, external experts. ~~This~~ Such outsourcing should be carefully considered by the o~~O~~perating o~~O~~rganization 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 the performance of maintenance, periodic testing and inspection is discussed in Ref. [29~~10~~].

APPLICATION OF GRADING TO CORE MANAGEMENT AND FUEL HANDLING

~~8.35.~~7.34. The requirements for core management and fuel handling are ~~presented~~ established in paras 7.65 to 7.70 of Ref. [1]. ~~Guidance~~ Recommendations for core management and fuel handling ~~is presented~~are provided in Ref. [9~~11~~].

~~8.36.~~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~~ might not apply to these low power level reactors. For these reasons, the requirements for core management and fuel handling should be graded for applicability to ~~a~~ the particular research reactor (see Ref. [9~~11~~], paras 1.11 and 2.2).

~~8.37.~~7.36. Reference~~.~~ [3~~2~~0] 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, G~~grading ~~in~~of the analysis and verification associated with the proposed core management and fuel handling activities may be possible, on the basis of their safety significance. See also ~~Reactor Utilization and Modification in~~ paras 7.47 to 7.50 of this Safety Guide.

APPLICATION OF GRADING TO FIRE SAFETY

~~8.38.~~7.37. The requirements for fire safety are ~~presented~~ established in paras 6.22 to 6.25 and 7.71 of Ref. [1]. ~~Guidance~~ Recommendations for fire safety ~~is presented~~are provided in Refs~~.~~ [3~~2~~1], ~~and Ref.~~ [3~~4~~2].

~~8.39.~~7.38. ~~Since fire protection is important to safety, all the requirements for fire safety of Ref. [1] are safety significanthave to be complied with .~~ ~~However, in the SAR T~~the potential fire hazards should be discussed in the safety analysis report and an indication ~~given~~ should be provided of their relative importance (i.e. in terms of likelihood and consequences) in the facility. This information can serve as a basis for grading the implementation of the fire prevention and protection measures.

~~8.40.~~7.39. Grading of the measures for ~~operational~~ fire protection ~~may~~ might be facilitated by provisions incorporated into the design ~~corresponding to~~in accordance with the fire hazard analysis, as required to be performed ~~in~~ [1~~4~~2], and which should ~~and to~~ be periodically reviewed and updated ~~in~~ [3~~2~~1], as well as by siting considerations.

~~8.41.~~7.40. Since techniques for 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~~ The analysis should employ techniques that have been proven adequate in similar facilities elsewhere.

~~8.42.~~7.41. The requirements for emergency planning are ~~presented~~ established in paras 6.20 and 7.72 to 7.78 of Ref. [1]. ~~Guidance~~ Further requirements for emergency planning and response ~~is presented~~are established in Ref. [33~~22~~]. Detailed approaches for a wide range of emergencies, suitable for the application of grading, are discussed in Ref. [34~~8~~].

~~8.43.~~7.42. The emergency plan and its implementing procedures are required to be based on the ~~DBA~~ accidents analy~~z~~sed in the ~~SAR~~safety analysis report (design basis accidents) as well as those additionally postulated for the purposes of emergency planning ~~(BDBA)~~(beyond design basis accidents). These analyses will allow the development of a source term ~~to be~~for use~~d~~ ~~for~~ in emergency planning. For some research reactors, it may be possible to demonstrate~~d~~ 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~~each research reactor is required for preparing an appropriate emergency plan.

~~8.44.~~7.43. In ~~conformance~~ accordance with the concept of a graded approach, Ref. [33~~18~~], paras 3.6 to 3.7 establishes ~~utilize~~ a ~~nuclear and radiation emergency~~ categorization scheme for nuclear and radiation related threats is required to be used that~~which~~ provides a basis for developing optimized arrangements for preparedness and response. This scheme requires that an ~~emergency~~ urgent protective action planning zone be ~~considered~~specified. The threat categories are:

Category I: Facilities for which on-site events are postulated that could give rise to severe deterministic health effects off the site, or for which such events have occurred in similar facilities.

Category II: Facilities, such as some types of research 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~~are in threat ~~C~~category II or III. This grading may lead to an ~~emergency~~ urgent protective action planning zone as small as the reactor building itself or large enough to extend off the ~~-~~site.

~~8.45.~~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~~ can be ~~possible~~applied, inter~~-~~ alia, in the following areas:

(a) ~~t~~The organization needed to carry out the emergency ~~plan~~response;

(b) T~~t~~he ~~emergency~~ urgent protective action planning zone;

(c) T~~t~~he identification and ~~categorization~~ classification of the hazard~~emergenciesy~~;

(d) N~~n~~otification requirements for informing the authorities;

(e) T~~t~~he amount, nature and storage location of ~~the~~ equipment needed to survey and monitor people and the environment ~~during the~~in the event of an emergency;

(f) T~~t~~he number, identity, training of and agreements with off-site agencies (e.g. police, fire services, medical treatment and medical transport) that ~~are involved~~will help in an emergency. Although the emergency ~~may~~ might not have an off~~-~~site impact, it is generally prudent to establish contact with off~~-~~site ~~authorities~~agencies ~~(e.g. police, fire services, medical transport, medical treatment)~~ to ensure their ~~concurrence~~ agreement ~~upon~~if a request for assistance is issued;

(g) T~~t~~he time~~-~~scales envisaged for ~~going through~~ the various phases of the response to an~~the~~ emergency;

(h) T~~t~~he types and the extent of ~~the~~ exercises and drills;

(i) T~~t~~he nature and amount of other resources needed for preparedness for and response to an ~~to handle the~~ emergency ~~situation~~; ~~and~~

(j) T~~t~~he facility's proximity to populated areas, which can significantly increase or decrease the ~~grading in~~ scope and the content of the emergency planning.

APPLICATION OF GRADING TO RECORDS AND REPORTS

~~8.46.~~7.45. The requirements for records and reports are ~~presented~~ established in paras 7.81 to 7.84 of Ref. [1]. ~~Guidance~~ Requirements for the ~~maintenance~~ control of records ~~and preparation of reports is presented~~are established in ~~Ref. [4]~~ paras 5.21 and 5.22 of Ref. [5] and recommendations are provided ~~Ref. [5],~~ paras~~.~~ 5.35 to 5.49 and a~~A~~nnexes I, II and III of Ref. [6].

~~8.47.~~7.46. Consistent with the purpose for which reports are prepared and records are kept, Ref. [6~~5~~], para. 2.44 lists specific examples of where a graded approach can be applied to controls for a~~the~~ records management ~~system~~process ~~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

~~8.48.~~7.47. The requirements for reactor utilization and modification are ~~presented~~ established in paras 7.85 to ~~-~~7.92 of Ref. [1]. ~~Guidance~~ Recommendations for reactor utilization and modification ~~is presented~~ are provided in Ref. [3~~2~~0].

~~8.49.~~7.48. The operating organization ~~should~~ is required to establish~~develop~~ criteria for ~~classifying~~ categorizing a proposed experiment or modification in accordance with its importance to safety. The resulting ~~classification~~ categorization should then be used to determine the types and extent of the analysis and approvals to be applied to the proposal.

~~8.50.So As far as possible, requirements for future utilization or modification of the reactor requirements should have been anticipated during reactor design, analyszed in the sSafety aAnalysis rReport, confirmed during the commissioning of the reactor and incorporated into the OLCoperational limits and conditions. Implementation at some later dateof future modifications may can be graded, relying in accordance withon the work already performed.~~

~~8.51.~~7.49. In ~~other~~ cases where an experiment~~al~~ or modification ~~requirement may might~~was not ~~have been~~ anticipated in the design, ~~requiring a determination of~~ its safety significance should be determined. Paragraph 1.11 of Ref. [9~~11~~], ~~para. 1.11~~ and annex I of Ref. [3~~2~~0]~~, 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~~Alternatively, a two category system can be used. The first category is the category for which the ~~a~~ modification or experiment is submitted to the regulatory body for review and approval. ~~The first category~~It includes modifications or experiments ~~which~~that:

(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

~~8.52.~~7.50. The requirements for radiation protection are ~~presented~~ established in paras~~.~~ 7.93 to 7.107 of Ref. [1] and in ~~Basic Safety Standards,~~ Ref. [35~~27~~]. ~~Guidance~~ Recommendations for radiation protection in the design and operation of research reactors ~~is~~ are ~~presented~~ provided in Ref. [36~~19~~].

~~8.53.~~7.51. While the content of the radiation protection programme ~~at a research reactor~~ will depend on ~~its~~ the design, power level and utilization of the particular research reactor, many aspects of the programme will be similar ~~at~~ for all research reactors.

~~8.54.~~7.52. The application of grading to the radiation protection programme should be consistent with the reactor's design and with its utilization (see Ref. [36~~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.

~~8.55.~~7.53. ~~It should be noted that a~~ A critical assembly may present a higher ~~hazard~~ risk of external radiation exposure ~~for~~ of operating personnel than a higher power research reactor, but the latter may ~~have~~ present a higher potential ~~hazard~~ risk ~~for~~ of ~~personnel~~ contamination of personnel causing internal radiation exposure. ~~Also~~ In addition, because critical assemblies are sometimes located within conventional industrial standard buildings, ~~critical assembly~~ reactivity accidents involving a critical assembly could result in a higher ~~potential hazard, for~~risk of contamination outside the building, compared to ~~larger source term~~ higher power reactors with a larger source term that have a containment structure.

~~8.56.~~7.54. Working areas within a research reactor should be classified (graded) into supervised areas and controlled areas, according to the magnitudes of the expected normal exposures, the likelihood and magnitude of potential exposures, and the nature and extent of the required radiological protection ~~procedures~~measures. Controlled areas themselves should be subject~~ed~~ to classification (grading) according to the protective measures employed or the expected radiological level~~,~~ (see Ref. [36~~19~~], paras~~.~~ 5.44 to 5.46 and 5.48).

~~6.1.~~• For a high power research reactor, it may be necessary to further grade the controlled areas into different levels, for example, ~~controlled area~~ levels I, II and III. ~~Residence at controlled area level II may require S~~specific procedures might be prescribed for work in level II controlled areas (in addition to those procedures prescribed ~~required~~ for ~~area~~ level I areas), ~~that~~ which may require, in so~~a~~me cases, ~~require~~ the use of protective garment, equipment~~,~~ or tools. ~~Controlled area L~~level III controlled areas will normally be closed by a physical barrier (e.g. an airlock door) that is opened only by authorized workers. Furthermore, opening of ~~this~~ a door to a level III controlled area during reactor operation ~~may~~ could be set to result in ~~an~~ automatic shutdown of the reactor~~ shutdown action~~.

- For a low power research reactor~~s~~, ~~controlled area~~ level~~s~~ II~~I~~ controlled areas and~~or~~ level III controlled areas~~,~~ may not be needed.

~~8.57.~~7.55. Reference~~.~~ [36~~19~~] provides general recommendations concerning the nature and scope of an operational radiation protection programme. The application of these general recommendations ~~may~~ can be graded ~~based on the basis of the above assessmentaspects described in paras 7.53 to 7.56 [???]~~ to determine the nature and scope of the elements of the specific operational radiation protection programme.

APPLICATION OF GRADING TO SAFETY ASSESSMENTS

~~8.58.~~7.56. The requirements for safety assessment~~s~~ are ~~presented~~ established in para. 7.108 of Ref. [1]. ~~Guidance~~ Recommendations for performing safety assessments ~~is presented~~are provided in Ref. [11~~23~~].

~~8.59.~~7.57. ~~Chapter~~Sections 4 and 7 of Ref. [1] ~~discuss~~ establish the requirements for management and verification of safety and ~~discuss~~ for safety assessment throughout all the stages ~~in~~ of the lifetime of the reactor. Recommendations on ~~G~~grading in the management and verification of safety ~~has~~ have been ~~discussed~~ provided in ~~Chapter~~Section 4 of this ~~publication~~Safety Guide.

~~8.60.~~7.58. Reference~~.~~ [7~~28~~], paras 3.1 to 3.7 ~~specifies~~ establishes 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~~ is required to be consistent with the magnitude of the possible radiation risk arising from the facility or activity.

~~8.61.~~7.59. The application of a graded approach ~~should~~ will vary according to the stage of the safety assessment as the ~~facility potential~~ possible radiation risks arising from the facility are clarified. At the design concept stage, for example, the safety case will focus on a statement of intent and principles. As the maturity of the facility progresses into the operational stage, much more detail and analysis should be ~~required~~provided in the safety assessment. The safety assessment for the decommissioning stage should ~~require~~ contain significantly less detail and analysis than that for the operational stage. The scope, and level ~~and~~ of detail of the safety assessment and the resources ~~required~~ deployed to produce it should be adjusted accordingly.

~~8.62.~~7.60. The main factors influencing the radiation risk and thus the level of detail ~~used for~~of a safety assessment at the operational stage ~~would be~~are: the~~;~~ predicted or historical ~~operational~~ radioactive releases and radiation ~~doses to~~exposure of ~~on-site staff~~workers and public; the consequences of anticipated operational occurrences and accidents with respect to ~~facility~~ damage to SSCs and radiation ~~doses to~~exposure of ~~staff~~ workers and public;~~,~~ and the potential consequences (in terms of radiation ~~dose~~ exposure and ~~SSC~~ damage to SSCs)~~,~~ from low probability events with potentially high consequences.

~~8.63.~~7.61. The graded approach should also be applied to the requirements for updating the safety assessment~~s,~~ (see Ref. [7~~28~~], para. 5.10). The frequency at which the safety assessment is updated and the ~~depth~~ level of detail of the safety assessment~~s~~ should be graded ~~depending upon~~ on the basis of the number and extent of modifications ~~for~~ to reactor systems as well as experimental facilities, changes to procedures, results of compliance monitoring of ~~OLC~~operational limits and conditions, modifications of safety significance, evidence of component ageing, ~~developments~~ feedback from operating experience and ~~historical unplanned incident experience~~from events, changes in site conditions and new regulatory 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 review of the safety assessment for the overall facility every 5 years would be appropriate. However, it is suggested that the~~A~~ maximum time between periodic reviews of the safety assessment~~s is though suggested to~~ be no more than 10 years, ~~regardless~~ irrespective of ~~reactor~~ the type or ~~usage~~utilization of the research reactor. With regard to reactors with more than 20 years of operation, more emphasis should be placed on safety assessment~~s~~ of component ageing~~ would be expected~~, particularly with regard to control systems and safety ~~-~~related passive components in locations in which it may be difficult ~~-~~to~~-~~ inspect and carry out repairs ~~locations~~ (e.g. inaccessible coolant piping and reactor tanks or ~~/~~vessels).

APPLICATION OF GRADING TO AGEING RELATED ASPECTS

7.62~~3~~. The requirements for ageing related aspects are ~~presented~~ established in para. 7.109 of Ref. [1]. ~~Guidance~~ Recommendations on ageing ~~related aspects of~~management for research reactors ~~is presented~~are provided in Ref. [38~~24~~].

~~6.29.~~7.63. While selection of materials and the effects of the operating environment on their properties ~~have to [is required to?/should?]~~ be taken into account~~ed for~~ in the design of all research reactors, the ~~use of a~~ grad~~inged approach~~ can be ~~made in~~applied to develop~~menting~~ of ~~the in-service inspection and~~ the ageing management programme~~s~~, including in-service inspection, ~~during~~ throughout the operating lifetime of the facility.

~~6.30.~~7.64. Grading ~~may~~ can be applied in determining the appropriate frequency of inspections, in selecting detection methods, as well as in establishing measures for ~~ageing~~ prevention and mitigation ~~measures~~of ageing effects, which ~~may~~ could be based on the estimated ~~lifetime~~ service lives of the SSCs, their complexity and their ease of replacement. In most research reactors, it is feasible to inspect most SSC~~'~~s periodically and to replace ~~the~~ components~~,~~ if ~~needed~~necessary. Particularly important material ageing concerns are corrosion in reactor tanks and vessels, where leak detection can be difficult and repair or replacement might not be practical. Similarly, the management of corrosion of inaccessible primary coolant piping and associated components is of key importance for reactor longevity. An important knowledge management area, which supplements the appropriate selection of materials and the management of ageing related effects, is the need for human resources management to address the ageing of research reactor personnel. Other key knowledge management areas are configuration management, document control and programmes for feedback of operating experience.

~~6.31.~~7.65. Grading may also be applicable to the resources ~~needed~~ necessary to implement the ageing management programme. While a dedicated organizational unit may be needed to implement such a programme for higher power research reactors, the ageing management activities for research reactors having a low power ~~may~~ might be performed by the ~~facility~~ maintenance personnel of the facility.

APPLICATION OF GRADING TO EXTENDED SHUTDOWN

~~6.32.~~7.66. The requirements for the safety of a research reactor in extended shutdown are ~~presented~~ established in paras 6.71 and 7.111 ~~to~~ and 7.112 of Ref. [1]. Further information on extended shutdown is provided in paras~~.~~ 6.34 to 6.35 of this ~~publication~~ Safety Guide and in Ref. [39~~25~~].

~~6.33.~~7.67. The operating staff of a research reactor in extended shutdown ~~may~~ could be smaller in number than that for an operating reactor. However, a large reduction in the overall numbers of ~~reactor facility staff level~~personnel ~~may~~ might be inappropriate. Concerns such as ~~the~~ a possible loss of the operating experience and knowledge of the facility ~~which~~ that will be necessary for ~~the~~ restart ~~of the facility~~ may ~~mitigate against~~preclude a large reduction in ~~staff~~personnel.

~~6.34.~~7.68. A graded approach should be applied to the scope and details of the activities, the measures to be implemented, the level of reviews, the frequency and extent of maintenance, testing and inspection activities during ~~the~~ an extended shutdown, and the extent of relief from requirements that apply during the normal operating regime.

8. DECOMMISSIONING

8.1. The requirements for decommissioning are ~~presented~~ established in paras 8.1 to 8.8 of Ref. [1]. ~~Further guidance~~ Recommendations are provided ~~can be found~~ in Ref. [40~~26~~]~~, under revision as DS 402~~. Reference~~.~~ [16~~39~~] is intended to assist the regulatory~~s~~ body, the operating~~ors~~ organization and ~~supporting~~ technical ~~specialists~~ support organizations in the application of a graded approach to the development and review of safety assessments for decommissioning activities. Section 3 of Ref. [~~40~~41] ~~Chapter 3 discusses~~ establishes the requirements for application of a graded approach ~~requirements applicable~~ to the development of the decommissioning plan and ~~similarly~~ Ref. [42~~1~~]~~,~~ provides recommendations on application of the graded approach ~~to obtain~~in the release of sites from regulatory control.

APPLICATION OF GRADING TO DECOMMISSIOING

~~(a)~~8.2. Decommissioning requirements are applicable to every research reactor. The range of decommissioning activities for which a safety assessment is required is broad, and the scope, extent and level of detail of the safety assessment should be commensurate with the types of 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~~meeting the requirements (e.g.~~,~~ ~~in~~ for the preparation and review of ~~the~~ decommissioning plans and procedures) ~~may~~ can be graded. ~~The grading of the plan and procedures~~A graded approach ~~may~~can ~~include~~ be applied to the scope of the required analyses and investigations, the number and variety of procedures to be prepared, the scope and depth of the reviews, the controls imposed, the number and types of approvals needed, the extent of measures for radiation protection ~~measures~~ and the scope of surveillance activities during decommissioning (see Ref. [41~~0~~], ~~Chapter~~Section 3).

~~(b)~~8.3. Decommissioning of a research reactor facility can be graded ~~based~~ on the basis of the activities~~y~~ and types 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 k~~Knowledge of the facility, which might be lost ~~due to retirement and loss of experienced personnel~~ when the reactor is permanently shutdown owing to the retirement or departure of experienced personnel, ~~are also very important to~~should be managed, in order to facilitate efficient and safe decommissioning operations.

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

~~–~~(a) Critical assemblies ~~may~~ might not represent a substantial concern from ~~the~~a ~~radiological~~ radiation protection or radioactive waste management point of view. A critical assembly should, however ~~although it would~~ be ~~necessary to~~ monitored for activation products before commencing disassembly, ~~and~~ although the dismantling activities ~~would~~ can generally be conducted without the need for special tools or highly qualified personnel. In many cases, the building and other installations housing a critical assembly ~~will~~may later~~may~~ be used for ~~different~~ another purpose~~s~~.

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

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

–(d) Release of the site of a research reactor from regulatory control ~~may~~ might not be appropriate, as in some cases~~ as~~ the ~~country~~ State may ~~need~~ intend to continue using the existing infrastructure ~~of a research reactor~~ for other ~~uses~~purposes, such as~~:~~ for storage of radioactive sources, for storage of radioactive waste~~ installation~~, or as a gamma irradiation facility.

8.4 The operating organization should ~~define~~ determine the ~~boundary conditions~~scope of the decommissioning plan by considering the most significant criteria, such as the resources available for decommissioning, the time ~~implementation~~ period to decommissioning and ~~final decommissioning~~the required end state~~ required,~~ (such as ~~immediate dismantling, deferred dismantling, entombment~~full or partial decontamination and/or dismantling or ~~site~~ release of the site from regulatory control). ~~With these conditions defined, the degree of~~ The scope of the decommissioning plan ~~should~~can then be graded~~ing possible~~, e.g.~~,~~ on the basis~~ed~~ o~~n~~f the ~~present~~current ~~state~~ status of the installation and its possible future use~~s of the decommissioned installation or site, can be established~~.

–8.5 Regulatory review of the safety assessment for decommissioning ~~plan~~ should ~~adopt~~ follow a graded approach and account should be taken of the following~~,~~ (see Ref. [16~~39~~], paras 3.1 to 3.5 and paras 5.6 to 5.8):

(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 in p~~t~~roducing 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).

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ANNEX ~~I:~~

EXAMPLE OF STEPS IN THE GRADED APPROACH FOR PACKAGING OF RADIOACTIVE MATERIAL~~S~~[27]

~~(Reproduced from Ref [45] Appendix)~~

A~~.6~~-1. 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. [A-2~~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 A-1~~2~~);

(3) Specification of the management controls required and assignment of a quality category (grade) to each (Table ~~3~~A-2).

A-2~~.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-3~~.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, etc.) may also be included. Typically, traceability of material, control of chemical and physical properties of the material, and isolation of the material from non-conforming material are used to ensure proper fabrication. When applicable, subtier vendors should control the quality of the component materials used.

A~~.9~~-4. 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.

[27] *This Annex reproduces paras A.6 to A.10 and tables 2 and 3 from the appendix to Ref. [A-1].*

A~~.10~~-5. 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~~A-1. 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~~ leaktightness 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, as well as ~~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 the safety or quality of the product or service need not be included in this graded system. An ~~E~~example of such non-graded items ~~may include some~~is software that facilitates routine operation, handling and/or use of the package or packaging.

TABLE ~~3~~A-2. 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	**X**		

Graded management controls	Quality categories		
	Grade 1	**Grade 2**	**Grade 3**
or international standard.			
The manufacturing planning specifies complete traceability of raw materials and the use of certified welders and processes.	X		
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 v~~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

REFERENCES TO THE ANNEX

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