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

DS511

DRAFT SPECIFIC SAFETY GUIDE

A revision of Safety Guide SSG-22

CONTENTS

1. INTRODUCTION

BACKGROUND

1.1. This Safety Guide provides recommendations on the use of a graded approach in the application of the safety requirements for research reactors, including critical and subcritical assemblies, established in IAEA Safety Standards Series No. SSR-3, Safety of Research Reactors [1].

1.2. For the purpose of this Safety Guide, a graded approach is the application of safety requirements commensurate with the risks associated with the research reactor. The use of a graded approach is intended to ensure that the necessary levels of analysis, documentation and actions are commensurate with, for example, the magnitudes of any radiation hazards, the nature and the particular characteristics of a facility, and the stage in the lifetime of a facility.

1.3. This Safety Guide was developed together with ten other Safety Guides on the safety of research reactors:

— IAEA Safety Standards Series No. DS509A, Commissioning of Research Reactors [2];
— IAEA Safety Standards Series No. DS509B, Maintenance, Periodic Testing and Inspection of Research Reactors [3];
— IAEA Safety Standards Series No. DS509C, Core Management and Fuel Handling for Research Reactors [4];
— IAEA Safety Standards Series No. DS509D, Operational Limits and Conditions and Operating Procedures for Research Reactors [5];
— IAEA Safety Standards Series No. DS509E, The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors [6];
— IAEA Safety Standards Series No. DS509F, Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors [7];
— IAEA Safety Standards Series No. DS509G, Ageing Management for Research Reactors [8];
— IAEA Safety Standards Series No. DS509H, Instrumentation and Control Systems and Software Important to Safety for Research Reactors [9];
— IAEA Safety Standards Series No. DS510A, Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report [10];
— IAEA Safety Standards Series No. DS510B, Safety in the Utilization and Modification of Research Reactors [11].

1.4. The terms used in this Safety Guide, including the definition of a graded approach, are to be understood as defined in the IAEA Safety Glossary [12].

1.5. This Safety Guide supersedes the IAEA Safety Standards Series No. SSG-22, Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors[1].

OBJECTIVE

1.6. The Safety Guide provides recommendations on the use of a graded approach in the application of the safety requirements for research reactors, which are established in SSR-3 [1]. This Safety Guide is intended for use by regulatory bodies, operating organizations and other organizations involved in the site evaluation, design, construction, commissioning, operation, and preparation for decommissioning of research reactors.

SCOPE

1.7. The application of a graded approach to all of the activities throughout the lifetime of a research reactor (site evaluation, design, construction, commissioning, operation and preparation for decommissioning) is addressed, including utilization and experiments which are specific features of research reactor operation. These activities are identified in SSR-3 [1]. A major aspect of this Safety Guide involves the use of a graded approach in the application of the safety requirements for design and operation of research reactors, so that the fundamental safety objective (see paras 2.2 and 2.3 of SSR-3 [1]) to protect people and the environment from harmful effects of ionizing radiation is achieved.

1.8. This Safety Guide is primarily intended for use for heterogeneous, thermal spectrum research reactors having a power rating of up to several tens of megawatts. Research reactors of higher power, specialized reactors (e.g. homogeneous reactors, fast spectrum reactors) and reactors having specialized facilities (e.g. hot or cold neutron sources, high pressure and high temperature loops) may need additional guidance.

1.9. Para 6.18 of SSR-3 [1] states that “The use of a graded approach in the application of the safety requirements shall not be considered as a means of waiving safety requirements and shall not compromise safety”. All requirements are applicable to all types of research reactor and cannot be waived. The recommendations provided in this Safety Guide are on whether and how a graded approach can be applied to these requirements in SSR-3 [1].

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Use of a Graded Approach in the Application of the Safety Requirements for Research Reactors, IAEA Safety Standards Series No. SSG-22, IAEA, Vienna (2012).

STRUCTURE

1.10. Section 2 provides a description of the basic elements of a graded approach and its application. The remaining sections provide recommendations on the application of a graded approach to requirements for regulatory supervision (Section 3); management and verification of safety (Section 4); site evaluation (Section 5); design (Section 6); operation (Section 7); and preparation for decommissioning (Section 8). Section 9 discusses Requirement 90 from SSR-3 [1] on the interfaces between safety and security. Sections 3– 9 have a similar structure to the corresponding sections of SSR-3 [1].

2. BASIC ELEMENTS OF A GRADED APPROACH FOR RESEARCH REACTORS

GENERAL CONSIDERATIONS OF A GRADED APPROACH

2.1. The use of a graded approach in the application of the safety requirements for research reactors in SSR-3 [1[is valid in all stages of the lifetime of a research reactor (see para. 1.7).

2.2. Research reactors are used for special and varied purposes, such as research, training, education, radioisotope production, neutron radiography and materials testing. These purposes call for different design features and different operational regimes. Design and operating characteristics of research reactors may vary significantly, in particular the use of experimental devices may introduce specific potential hazards. In addition, the need for flexibility in their use requires a different approach to achieving and managing safety.

2.3. Because of the wide range of designs, operating conditions, radioactive inventories and utilization activities, the safety requirements for research reactors are not applied to every research reactor in the same way. For example, the way in which requirements are demonstrated to be met for a multipurpose, high power research reactor might be very different from the way in which the requirements are demonstrated to be met for a research reactor with very low power and very low associated radiation hazard to facility staff, the public and the environment. SSR-3 [1], which applies to a wide range of research reactors, includes information on the application of the safety requirements in accordance with a graded approach (see paras 2.15–2.17 of SSR-3 [1]).

2.4. During the lifetime of a research reactor, the use of a graded approach in the application of the safety requirements should be such that safety functions and operational limits and conditions are preserved, and there are no undue radiation hazards to workers, the public or the environment.

2.5. The use of a graded approach should be based on safety analyses, regulatory requirements and expert judgement. Expert judgement implies that account is taken of the safety functions of structures, systems and components (SSCs) and the consequences of the failure to perform these functions and implies that the judgement is documented and subjected to appropriate review and approval using a process in the management system. Prescriptive regulatory approaches[2], resulting in very detailed regulatory requirements may restrict the use of a graded approach by the operating organisation on some of the topics in this Safety Guide. Other elements to be considered when applying a graded approach are the complexity and the maturity of the technology, operating experience associated with activities and the stage in the lifetime of the facility.

DESCRIPTION OF THE USE OF A GRADED APPROACH IN THE APPLICATION OF SAFETY REQUIREMENTS

2.6. The result of the use of a graded approach in the application of safety requirements should be a decision on the appropriate effort to be expended and appropriate manner of complying with a safety requirement, in accordance with the characteristics and the potential hazard of the research reactor

2.7. The overall method to determine the graded approach may be quantitative, qualitative or a combination of both. The graded approach presented in this Safety Guide has two steps. First is the qualitative categorization of the facility in accordance with its potential hazard (see para. 2.16 of SSR-3 [1]). Second is consideration of a specific safety requirement from SSR-3 [1], and the quantitative and/or qualitative analysis of any activities and/or SSCs associated with that requirement.

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

2.8. Qualitative categorization of the facility should be performed on the basis of the potential radiological hazard, using a multi-category system, as follows:

(a) Facilities with significant potential for an off-site radiological hazard: such facilities include research reactors with high operating power, a large radioactive inventory, or high-pressure experimental devices. These facilities are categorized as a high potential hazard.

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[2] Prescriptive and performance based regulatory approaches are described in para 2.80 of IAEA Safety Standards Series No. SSG-16 (Rev. 1), Establishing the Safety Infrastructure for a Nuclear Power Programme [13].

(b) Facilities with potential for an on-site radiological hazard only: such facilities include research reactors with operating power up to a few MW, a limited radioactive inventory, or no high-pressure experimental devices. These facilities are categorized as a medium potential hazard.

(c) Facilities with no potential radiological hazard beyond the research reactor hall and associated beam tubes or connected experimental facility areas: such facilities include facilities with low operating power, not requiring heat removal systems, or with a small radioactive inventory. These facilities are categorized as a low potential hazard.

Section 3 of DS509F [7] provides further guidance on evaluating the radiological hazard of research reactors.

2.9. Additional characteristics to be considered in deriving the category of the facility in accordance with its potential hazard are listed in para 2.17 of SSR-3 [1], which states:

> "The factors to be considered in deciding whether the application of certain requirements established here may be graded include:
>
> (a) The reactor power;
> (b) The potential source term;
> (c) The amount and enrichment of fissile and fissionable material;
> (d) Spent fuel elements, high pressure systems, heating systems and the storage of flammable materials, which may affect the safety of the reactor;
> (e) The type of fuel elements;
> (f) The type and the mass of moderator, reflector and coolant;
> (g) The amount of reactivity that can be introduced and its rate of introduction, reactivity control, and inherent and additional safety features (including those for preventing inadvertent criticality);
> (h) The design limitations of the containment structure or other means of confinement;
> (i) The utilization of the reactor (experimental devices, tests and reactor physics experiments);
> (j) The site evaluation, including external hazards associated with the site and the proximity to population groups;
> (k) The ease or difficulty in changing[3] the overall configuration."

On the basis of these characteristics, the application of expert judgement, and consideration of any other factors that might affect the potential radiological hazard from the facility, a high, medium or low potential hazard should be identified and used in the analysis in step 2.

[3] Modifications and experiments are an important aspect of research reactor design and operation. See paras 6.148-6.150 and 7.70 for specific recommendations

Step 2: Analysis and Application of a Graded Approach

2.10. Following the categorization of the facility in step 1, an analysis should be performed to determine the appropriate manner for meeting a specific safety requirement using a graded approach. A safety requirement may address a specific SSC, or an element of the management system. The safety significance of each SSC or management system element (including SSCs and management system elements related to experiments) can be determined through the step 2 analysis. Requirement 16 of SSR-3 [1] states that "All items important to safety for a research reactor facility shall be identified and shall be classified on the basis of their safety function and their safety significance".

2.11. The safety function and safety significance and potential risks of SSCs should be determined by conducting a safety assessment (see DS510A [10]). When identifying SSCs that are important to safety, classifying them by their importance to safety, and then considering a graded approach in their design, para 6.32 of SSR-3 [1] states that "The basis for the safety classification of the structures, systems and components shall be stated and the design requirements shall be applied in accordance with their safety classification." The application of design requirements commensurate with the safety classification of an SSC is the basis of a graded approach in the design process.

2.12. With regard to analysing the safety significance of elements of the management system, and then applying grading in meeting management system requirements, Requirement 7 from IAEA Safety Standards Series No. GSR Part 2, Leadership and Management for Safety [14] states:

> "The criteria used to grade the development and application of the management system shall be documented in the management system. The following shall be taken into account:
>
> (a) The safety significance and complexity of the organization, operation of the facility or conduct of the activity;
>
> (b) The hazards and the magnitude of the potential impacts (risks) associated with the safety, health, environmental, security, quality and economic elements of each facility or activity;
>
> (c) The possible consequences for safety if a failure or an unanticipated event occurs or if an activity is inadequately planned or improperly carried out."

Paras 2.37–2.40 of IAEA Safety Standards Series No. GS-G-3.1, Application of the Management System for Facilities and Activities 5] provide recommendations on how elements of the management system can be assessed, to support a graded approach in the application of management system requirements.

2.13. The analysis in step 2 to determine how requirements related to SSCs and/or management system elements are met should consider the overall categorization of the facility from step 1, the safety significance of the SSC and/or element of the management system which is affected, and therefore the appropriate level of effort needed in meeting the requirement, and the manner in which the requirement will be met. Expert judgement, from a single expert or a multidisciplinary group as appropriate, may be included in the analysis.

2.14. Specific recommendations on the use of a graded approach in the application of each safety requirement are provided in Sections 3–8, including on requirements to which a graded approach cannot be applied. Examples are given for the graded application of requirements for research reactors with a high, medium, or low potential hazard.

3. USE OF A GRADED APPROACH IN THE REGULATORY SUPERVISION OF RESEARCH REACTORS

3.1. The general requirements for the legal and regulatory infrastructure for facilities and activities are established in IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety [16], which including requirements on the use of a graded approach for the responsibilities and functions of the regulatory body. IAEA Safety Standards Series No. GSG-13, Functions and Processes of the Regulatory Body for Safety [17] provides recommendations on the core regulatory functions and processes, including the application of a graded approach (see paras 2.8–2.10 of GSG-13 [17]), to the following:

(a) Regulations and guides;
(b) Notification and Authorization;
(c) Review and assessment of facilities and activities;
(d) Inspection of facilities and activities;
(e) Enforcement;
(f) Emergency preparedness and response;
(g) Communication and consultation with interested parties.

THE USE OF A GRADED APPROACH IN THE LEGAL AND REGULATORY INFRASTRUCTURE

3.2. The requirements for the legal infrastructure established in GSR Part 1 (Rev. 1) [15] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

3.3. [1616]. are placed on the government (e.g. for the adoption of legislation that assigns the prime responsibility for safety to the operating organization and establishes a regulatory body) and on the regulatory body (e.g. for the establishment of regulations that results in a system of authorization for the regulatory control of nuclear activities and for the enforcement of the regulations). Regarding the application of these requirements, para 3.2 of SSR-3 [1] states that “The application of a graded approach that is commensurate with the potential hazards of the facility is essential and shall be used in the determination and application of adequate safety requirements.” Specific aspects of the legal and regulatory framework in a State may affect the extent to which a graded approach can be used.

3.4. In a State where the most hazardous nuclear facility is a single operating research reactor with a low potential hazard (see para. 0), the implementation of the national policy and strategy for safety may use a graded approach, with a less comprehensive set of policy mechanisms and internal resources than in a State with a large and diverse nuclear infrastructure. A graded approach to applying the requirements for a State’s legal and regulatory infrastructure[4] should include an analysis of the radiation risks associated with facilities and activities and also consider the following provisions that are necessary for the government to meet the fundamental safety objective:

(a) Human and financial resources;
(b) The type of authorization process;
(c) The provisions for regulatory review;
(d) Appropriate inspection and enforcement regulations;
(e) Communication and consultation with interested parties.

Further detail is provided in Requirements 1 and 2 of GSR Part 1 (Rev. 1) [16].

THE USE OF A GRADED APPROACH IN THE ORGANIZATION AND FUNCTIONS OF THE REGULATORY BODY

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

The regulatory body is required to be provided with sufficient authority, and a sufficient number of experienced staff and financial resources to discharge its assigned responsibilities (Requirement 3 of GSR Part 1 (Rev. 1) [15] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

[4] Some examples are shown in TECDOC-XXXX, “Application of graded approach in regulating nuclear power plants, research reactors and fuel cycle facilities”.

3.6. [16]). The responsibilities of the regulatory body should include establishing regulations, review and assessment of safety related information (e.g. from the safety analysis report), issuing authorizations, performing inspections, taking enforcement actions and providing information to other competent authorities and the public. External experts, technical support organizations or advisory committees may assist the regulatory body in these activities.

3.7. Examples of safety requirements for the regulatory body that can be met using a graded approach are requirements for: staffing; resources for in-house technical support; inspections; the content and detail of authorizations, regulations and guides; and the detail required from the licensee for submissions of documentation on the safety of the facility, including the safety analysis report. Areas where the regulatory body might use a graded approach are identified in IAEA Safety Standards Series GSG-12, Organization, Management and Staffing of the Regulatory Body for Safety [1818]. Regulatory requirements should be taken into account as they may limit the scope of a graded approach in the application of requirements for the regulatory body itself.

THE USE OF A GRADED APPROACH IN THE AUTHORIZATION PROCESS

3.8. The authorization process is often performed in steps for the various stages of the lifetime of a research reactor, as described in paras 3.4 and 3.5 of SSR-3 [1]. For a research reactor, these stages include:

(a) Site evaluation;
(b) Design;
(c) Construction;
(d) Commissioning;
(e) Operation, including utilization and modification;
(f) Decommissioning;
(g) Release from regulatory control.

3.9. At each of these stages, regulatory reviews and assessments are usually made and authorizations or approvals are issued. In some cases, some of these stages may be combined, depending on the nature of the facility and relevant laws and regulations.

3.10. The authorization process should be used by the regulatory body to exercise control during all stages of the lifetime of the research reactor. This control is accomplished by means of the following:

(a) Defining clear lines of authority for authorizations to proceed;
(b) Reviewing and assessing all safety relevant documents, particularly the safety analysis report;
(c) Issuing of licences;

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

(e) Reviewing, assessing, and approving operational limits and conditions;

(f) Authorizing construction;

(g) Authorizing commissioning;

(h) Authorizing operation;

(i) Authorization of operating personnel;

(j) Authorizing decommissioning.

3.11. The steps in the authorization process apply to all research reactors, including experiments and modifications (see DS510B [11]), at all stages of the reactor lifetime. However, at each step in the authorization process, a graded approach may be used in the application of the safety requirements by the regulatory body, depending on the potential hazard of the facility. For example, the level of detail required in the application for an authorization, the depth of review and human resource needed by the regulatory body when considering an application for authorization, and the duration of an authorization when it is issued, should be commensurate with the potential hazard from the facility being authorized.

Safety analysis report

3.12. The requirements for the safety analysis report, which is used in the review and assessment of facilities and activities and in the authorization of research reactors, are established in Requirement 1 of SSR-3 [1]. The responsibilities of the regulatory body include the review and assessment of safety related information from the safety analysis report. A graded approach may be used in the application of these requirements. The level of detail in documentation related to the safety of the facility, including the safety analysis report, should be based on the potential hazard from the facility, and on the stage in the lifetime of the facility.

3.13. A graded approach should be used in the preparation a safety analysis report, for example, the level of detail necessary to demonstrate that acceptance criteria are met should be commensurate with the potential hazard of the research reactor. For research reactors with a higher potential hazard, typically more detailed analysis is necessary to demonstrate safety in all operating and accident conditions, with less use of large bounding analyses. For a facility with a low potential hazard, the safety analysis may include bounding analyses, due to large safety margins in the design, to demonstrate that the research reactor can be operated safely.

3.14. The use of probabilistic safety assessment to supplement deterministic safety analysis as appropriate, is another element of the safety analysis report that could vary in scope based on the potential hazard of the facility (see Requirement 41 in SSR-3 [1]). The Appendix in DS510A [10] provides recommendations on safety assessment and the safety analysis report for research reactors, including the application of a graded approach commensurate with the magnitude of the potential hazards.

THE USE OF A GRADED APPROACH IN INSPECTION AND ENFORCEMENT

Requirements for inspection and enforcement are established in paras 3.13–3.16 of SSR-3 [1]. For inspections, GSR Part 1 (Rev. 1) [15] INTERNATIONAL ATOMIC ENERGY AGENCY, Application of the Management System for Facilities and Activities, IAEA Safety Standards Series No. GS-G-3.1, IAEA, Vienna (2006).

3.15. [16] states:

> "The regulatory body shall develop and implement a programme of inspection of facilities and activities, to confirm compliance with regulatory requirements and with any conditions specified in the authorization. In this programme, it shall specify the types of regulatory inspection (including scheduled inspections and unannounced inspections) and shall stipulate the frequency of inspections and the areas and programmes to be inspected, in accordance with a graded approach."

In general, there should be fewer inspections and hold points for a research reactor with a low potential hazard, compared to those for a research reactor with a higher potential hazard.

3.16. Enforcement actions should be commensurate with the consequences of non-compliance. Regulatory bodies should allocate resources and apply enforcement actions or methods in a manner commensurate with the seriousness of the non-compliance, increasing them as necessary to bring about compliance with requirements.

3.17. Some of the factors that should be considered in determining the appropriate level of enforcement actions are as follows:

(a) The safety significance of the non-compliance or of the violation;
(b) Whether the non-compliance or violation is repeated;
(c) Whether there has been an intentional violation;
(d) Whether or not the authorized party identified and/or reported the non-compliance or the violation;
(e) Whether the non-compliance or violation impacted the ability of the regulatory body to perform its regulatory oversight function;
(f) The past safety performance of the authorized party and the performance trend;
(g) The need for consistency and openness in the treatment of authorized parties.

3.18. Enforcement actions in response to an intentional violation of a regulatory requirement should be commensurately serious.

4. USE OF A GRADED APPROACH IN THE MANAGEMENT AND VERIFICATION OF SAFETY OF RESEARCH REACTORS

Requirements for the management system for organizations operating nuclear installations, including research reactors, are established in GSR Part 2 [13] INTERNATIONAL ATOMIC ENERGY AGENCY, Establishing the Safety Infrastructure for a Nuclear Power Programme, IAEA Safety Standards Series No. SSG-16 (Rev. 1), IAEA, Vienna (2020).

4.1. [1414], including the requirement for the management system to be developed and applied using a graded approach. Additional requirements specific to research reactors are established in Requirements 2–6 of SSR-3 [1].

RESPONSIBILITIES IN THE MANAGEMENT FOR SAFETY

4.2. Requirements for responsibilities in the management for safety for research reactors are established in Requirement 2 of SSR-3 [1]. Paragraph 4.1 of SSR-3 [1] states:

> "In order to ensure rigour and thoroughness at all levels of the staff in the achievement and maintenance of safety, the operating organization:
>
> (a) Shall establish and implement safety policies and shall ensure that safety matters are given the highest priority;
>
> (b) Shall clearly define responsibilities and accountabilities with corresponding lines of authority and communication;
>
> (c) Shall ensure that it has sufficient staff with appropriate qualifications and training at all levels;
>
> (d) Shall develop and strictly adhere to sound procedures for all activities that may affect safety, ensuring that managers and supervisors promote and support good safety practices, while correcting poor safety practices;
>
> (e) Shall review, monitor and audit all safety related matters on a regular basis, and shall take appropriate corrective actions where necessary;
>
> (f) Shall develop and sustain a strong safety culture, and shall prepare a statement of safety policy and safety objectives, which is disseminated to and understood by all staff."

There are elements of this requirement which cannot be applied using a graded approach, for example, for the operating organization to have prime responsibility for the safety of the research reactor, and the requirement to develop and sustain a strong culture for safety.

4.3. The management of a research reactor should vary depending on the potential hazard of the facility, its complexity and size. For example, in a research reactor with a high potential hazard, the requirement for sufficient staff could result in a large operating organization, to enable continuous operation day and night, and provide maintenance and technical support. In a facility with a low potential hazard, such as some subcritical assemblies, the requirement for sufficient staff could result in a small operating organization, with the necessary training to operate, maintain, and ensure the safety of the research reactor. The organization structure for the operating organization, and the definition of minimum staff required in the facility during operation, should account for the operational response to anticipated operational occurrences and the emergency preparedness and response arrangements required for all accident conditions.

SAFETY POLICY

4.4. Requirement 3 of SSR-3 [1] states that "**The operating organization for a research reactor facility shall establish and implement safety policies that give safety the highest priority.**"

4.5. The requirement to establish and implement a safety policy cannot be applied using a graded approach. The safety policy is a central component of an integrated management system, to ensure that any activities across the operating organization place safety as the highest priority.

THE USE OF A GRADED APPROACH IN THE APPLICATION OF THE REQUIREMENTS FOR THE MANAGEMENT SYSTEM

4.6. Requirements for the management system for a research reactor facility are established in Requirement 4 of SSR-3 [1]. Paragraph 4.7 of SSR-3 [1] states that "The level of detail of the management system that is required for a particular research reactor or experiment shall be governed by the potential hazard of the reactor and the experiment".

4.7. In general, management system processes should be most stringent for items, services or processes where a failure or a non-conformance has the highest potential hazard. For other items, services or processes, the management system processes may be less stringent. The following are examples of elements of the management system where this requirement can be applied using a graded approach:

(a) Type and content of training;

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

(c) Need for and detail of inspection plans;

(d) Scope, depth and frequency of operational safety reviews and controls including internal and independent audits;

(e) Type and frequency of safety assessments;

(f) Records to be generated and retained;

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

(h) Maintenance, periodic testing and inspection activities;

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

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

(k) Need to analyse events and equipment failure data.

4.8. Procedures for a research reactor with a high potential hazard should be subject to a level of review and approval commensurate with their safety significance. A procedure for a simple maintenance task on a component in a non-active system with low safety significance could be written by an experienced member of the engineering personnel and reviewed by a maintenance supervisor. A procedure for use in the control room to start up the reactor should be subject to more rigour in the level of detail and extent of review. For a research reactor with a low potential hazard, the expertise necessary to write and review new procedures may not always exist within the operating organization and could involve experts from the reactor designer or another external organization with appropriate knowledge. The level of review for procedures should also be commensurate with their safety significance.

4.9. The approval of procedures is the responsibility of the reactor manager (see para 5.16 DS509D [5]). In every research reactor, regardless of potential hazard, every procedure in the management system should be periodically reviewed by the reactor manager or a designate, to enable improvements to be identified.

4.10. Paras 2.37–2.44 of GS-G-3.1 [115] also provide recommendations on a graded approach to the application of requirements for management system controls.

4.11. The requirement for the assessment and improvement of the integrated management system can be applied using a graded approach to identify and correct weaknesses commensurate with their safety significance, and with the potential hazard of the facility. For example, for a research reactor with a high potential hazard, the operating organization could be large, and the management system could include a large number of procedures to ensure operation, utilization and maintenance activities are conducted safely. An operating experience programme could be implemented by a small group of personnel within the operating organization to identify weaknesses and improvements in the management system on a weekly basis, for management to prioritize based on their safety significance. In parallel, the management system could be the subject of frequent external assessment, to identify where systemic improvements can be made. For a research reactor with a low potential hazard, the management system could consist of relatively few processes and procedures, the operating experience programme could be implemented by the operations personnel to identify improvements to the management system, and an audit of the management system could occur as part of the renewal of the authorization from the regulatory body.

THE USE OF A GRADED APPROACH IN THE APPLICATION OF THE REQUIREMENT FOR VERIFICATION OF SAFETY

Safety assessment

4.12. Requirements for safety assessment are established in Requirement 5 of SSR-3 [1]. This requirement can be applied using a graded approach, for example by considering the potential hazard of the research reactor when determining the frequency and scope of safety assessments throughout the lifetime of the facility such as self-assessments and peer reviews. For example, the frequency and scope of safety assessments, self-assessments and peer reviews, should be commensurate with the potential hazard of the facility, recent operating experience, the potential hazard of modifications (see para 7.70), or the results from previous periodic safety reviews.

4.13. The requirement to verify the adequacy of the design using safety assessment techniques can be applied using a graded approach based on the potential hazard of the facility and the number of SSCs important to safety, as discussed in para 3.13. Recommendations on the use of a graded approach in safety analysis of the design are provided in paras 6.85–6.91 of this Safety Guide.

Safety committee

4.14. Requirements for the safety committee are established in Requirement 6 of SSR-3 [1]. One element of this requirement that cannot be applied using a graded approach, is the establishment of a safety committee. The safety committee is required to be independent from the reactor manager, to 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 minimum list of items that the safety committee is required to review is provided in SSR-3 [1] (see also para 7.9 of this Safety Guide).

4.15. Aspects of this requirement which can be applied using a graded approach include, the number, size, and frequency of committee meetings; and the membership composition of the committee.

4.16. In a research reactor with a high potential hazard, the safety committee could have a busy schedule of work, requiring frequent meetings reviewing proposed experiments of safety significance, safety documentation, reports on doses to personnel and reports to the regulatory body. In such a research reactor, the safety committee may designate subcommittees with specific expertise to provide advice or recommendations on specific technical areas such as criticality safety or radiation protection, to reduce the workload on other safety committee members. The composition of the safety committee and its subcommittees typically includes a wide range of expertise on all technical areas of operation. The operating organization for such a facility typically can staff the safety committee from internal personnel. In a research reactor with a low potential hazard, the safety committee could be convened less frequently to review the status of safety and to provide advice to the reactor manager, with additional meetings arranged only as necessary. The operating organization for such a research reactor is typically smaller in size, and the safety committee could be staffed with a number of external personnel with experience from other facilities and in the appropriate technical areas.

5. THE USE OF A GRADED APPROACH IN SITE EVALUATION FOR RESEARCH REACTORS

5.1. The requirements for site evaluation for research reactors are established in IAEA Safety Standards Series No SSR-1, Site Evaluation for Nuclear Installations [15]. Recommendations for the application of those requirements for research reactors, using a graded approach, are provided in Section 6 of IAEA Safety Standards Series No. SSG-35, Site Survey and Site Selection for Nuclear Installations, [16].

5.2. Requirement 3 of SSR-1 [15] discusses a graded approach to the application of requirements for site selection specifically for facilities other than nuclear power plants. Paragraph 5.1 of SSR-3 [1] states that “The main safety objective in evaluating the site for a research reactor is the protection of the public and the environment against the radiological consequences of normal and accidental releases of radioactive material”. Accordingly, it is necessary to assess those characteristics of the site that may affect the safety of the research reactor, to determine whether there are deficiencies in the site and if they can be mitigated by appropriate design features, site protection measures and administrative procedures. For a graded approach to the application of site evaluation requirements, the scope and depth of site evaluation studies and evaluations should be commensurate with the potential radiation risk associated with the facility. The scope and detail of the site evaluation may also be reduced if the operating organization proposes to adopt conservative parameters for design purposes that reduce the potential for on-site and off-site consequences in the event of an accident, which may be a preferred approach for research reactors. For example, a conservative assumption for the design of a particular SSC that is readily accommodated in the overall design may permit simplification of the site evaluation.

5.3. Paragraphs 4.1–4.5 of SSR-1 [15] develop the basis for applying a graded approach to the various site related evaluations and decisions, commensurate with the radiological hazard of the research reactor. The main factors to be considered in site evaluation are the following:

(a) The amount, type and status of the radioactive inventory at the site (e.g. whether the radioactive material on the site is in solid, liquid and/or gaseous form, and whether the radioactive material is being processed in the nuclear installation or is being stored on the site);
(b) The intrinsic hazards associated with the physical and chemical processes that take place at the research reactor;
(c) The thermal power;
(d) The distribution and location of radioactive sources in the nuclear installation;
(e) The configuration and layout of installations designed for experiments, and how these might change in future;
(f) The need for active systems and/or operator actions for the prevention of accidents and for the mitigation of the consequences of accidents;
(g) The potential for on-site and off-site consequences in the event of an accident.

5.4. The requirements for site evaluation should applied use a graded approach, provided that there is an adequate level of conservatism in the design and siting criteria, to compensate for a simplified site hazard analysis and simplified analysis methods.

5.5. Section 10 of IAEA Safety Standards Series No. SSG-9 (Rev. 1), Seismic Hazards in Site Evaluation for Nuclear Installations [17] provides recommendations on a graded approach to the application of safety requirements for seismic hazard evaluation for nuclear installations other than nuclear power plants. The approach can be based upon the complexity of the installation and the potential radiological hazards, including hazards due to other materials. A seismic hazard assessment should initially apply a conservative screening process in which it is assumed that the entire radioactive inventory of the installation is released by an accident initiated by a seismic event. If such a release would not lead to unacceptable consequences for workers, the public or the environment, the installation may be screened out from further seismic hazard assessment. If the results of the conservative screening process show that the potential consequences of such a release could be significant, a seismic hazard evaluation should be performed.

5.6. Section 7 of IAEA Safety Standards Series No. SSG-21, Volcanic Hazards in Site Evaluation for Nuclear Installations [18] provides recommendations similar to those in SSG-9 (Rev. 1) [17] for a graded approach to the application of Requirement 17 from SSR-1 [15] with respect to volcanic hazards in site evaluation. A volcanic hazard assessment should initially apply a conservative screening process in which it is assumed that the entire radioactive inventory of the installation is released by an accident initiated by a volcanic event. If such a release would not lead to unacceptable consequences for workers, the public or the environment, the installation may be screened out from further volcanic hazard assessment. If the results of the conservative screening process show that the potential consequences of such a release could be significant, a more detailed volcanic hazard assessment should be performed, and a graded approach outlined in SSG-21 [18] should then be used to categorize the installation for the purposes of volcanic hazard assessment.

5.7. Recommendations on a graded approach to the application of Requirements 18, 19 and 20 of SSR-1 [15] on meteorological and hydrological hazards in site evaluation are provided in IAEA Safety Standards Series No. SSG-18, Meteorological and Hydrological Hazards in Site Evaluation for Nuclear Installations [19]. For the purpose of the evaluation of meteorological and hydrological hazards, including flooding, the installation should be screened on the basis of its complexity, the potential radiological hazards, and hazards due to other materials. If the results of a conservative screening process, similar to that described in SSG-9 (Rev. 1) [17] and SSG-21 [18], show that the consequences of a potential release could be significant, a detailed meteorological and hydrological hazard assessment for the installations should be carried out, in accordance with the graded approach outlined in Section 10 of SSG-18 [19].

5.8. Human induced events cannot be included in site evaluation using the same approach as other external events. Because human induced events are discrete and are not characterised by a range of frequency and severity, only one intensity level for each event is expected for consideration in the design basis. Recommendations on site survey and site selection, including the screening and analysis of human induced events, are provided in SSG-35 [20]. While the events themselves are discrete, the siting process for nuclear installations other than nuclear power plants can be applied using a graded approach, based on the potential hazard of the facility (see Section 6 of SSG-35 [20])

6. THE USE OF A GRADED APPROACH IN THE DESIGN OF RESEARCH REACTORS

6.1. Section 6 of SSR-3 [1] establishes requirements for design under three categories:

(a) Principal technical requirements: Paragraphs 6.2–6.18 of this Safety Guide provide recommendations on the use of a graded approach in the application of Requirements 7–15 of SSR-3 [1].

(b) General requirements for design: Paragraphs 6.19–6.91 of this Safety Guide provide recommendations on the use of a graded approach in the application of Requirements 16–41 of SSR-3 [1].

(c) Specific requirements for design: Paragraphs 6.92–6.150 of this Safety Guide provide recommendations on the use of a graded approach in the application of Requirements 42–66 of SSR-3 [1].

THE USE OF A GRADED APPROACH IN PRINCIPAL TECHNICAL REQUIREMENTS

Main safety functions

6.2. Requirement 7 of SSR-3 [1] states:

> "**The design for a research reactor facility shall ensure the fulfilment of the following main safety functions for the research reactor for all states of the facility: (i) control of reactivity; (ii) removal of heat from the reactor and from the fuel storage; and (iii) confinement of the radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.**"

The use of a graded approach should result in design features which fully meet this requirement and are appropriate for the potential hazard from the research reactor. The control of planned radioactive discharges during normal operation is an element of this requirement that cannot be applied using a graded approach. The control of radioactive discharges is necessary to protect the public and the environment and ensure that facility operation meets applicable national environmental regulations.

6.3. A graded approach can be used in the application of some elements of the requirement for the main safety functions, as follows:

(a) Control of reactivity:

(i) The capability to shut down the reactor when necessary is a requirement for all research reactors, although the size of the subcriticality margin available and the speed of response of the shutdown system may vary according to the reactor design.

(b) Removal of heat from the reactor and from the fuel storage:

(i) For some research reactors (typically with a medium or high potential hazard and higher power) a forced convection cooling system to remove fission heat, could be necessary to meet the acceptance criteria for the design, in all operating conditions and accident conditions, whereas for research reactors with less demanding cooling needs, such as some critical and subcritical assemblies, fission heat could be generated at sufficiently low levels that it could be adequately removed without the need for an engineered system.

(ii) Similarly, for the removal of decay heat following shutdown, the design of the cooling system can use a graded approach based on factors such as the power of the reactor, the maximum level of fission products and the heat transfer characteristics of the fuel. For a research reactor with less demanding cooling needs, where no heat removal system is necessary during operation, no dedicated equipment is necessary for decay heat removal.

(iii) The scope and necessity of cooling systems, including emergency core cooling systems to replace the inventory of reactor coolant in the event of a loss of coolant accident, is verified through the safety analysis for the research reactor, which is required to demonstrate that for all operational states and accident conditions, the main safety function of heat removal is fulfilled.

(c) Confinement of radioactive material, shielding against radiation and control of planned radioactive releases:

(i) The design of SSCs to perform barrier or retention functions to confine radioactive material in operational states and accident conditions can use a graded approach. The approach can be based on the potential hazard of the facility, the inventory of fission products, the characteristics of the fuel, and the results of the safety analysis for the research reactor. (See also the description of the fourth level of defence in depth in para. 6.7).

(ii) The design of shielding for protection from radiation should be based on the magnitude of the radiation hazard which can be calculated for each location in the research reactor where actions by operating personnel are necessary in operational states and in accident conditions. The appropriate material and thickness of shielding that is commensurate with the hazard can then be included in the design.

(iii) The requirement for the control of planned radioactive discharges cannot be applied using a graded approach.

Radiation protection

6.4. Requirements for radiation protection in the design of research reactors are established in Requirement 8 of SSR-3 [1]. The requirement for the design to ensure that doses to reactor personnel and the public are kept as low as reasonably achievable should be applied using a graded approach considering the potential hazard of the research reactor, and its characteristics such as the inventory of fission products and the proximity to a population centre. Specific design provisions, or SSCs included in the design to protect reactor personnel and the public from radiation, e.g. an emergency filtration system, could be larger and more complex for a research reactor with a high potential hazard.

Design

6.5. Requirements for the design of a research reactor are established in Requirement 9 of SSR-3 [1]. The use of a graded approach in the application of this requirement should be based on the potential hazard of the facility and the factors in para. 2.9.

6.6. The requirement that adequate information on the design is available for operation, future modifications, and decommissioning can be applied using a graded approach based on the potential hazard of the research reactor, the number of SSCs important to safety and the number of SSCs in the facility with associated radiation hazards. The quantity of information that would be adequate to decommission a research reactor with a high potential hazard should be larger in scope than for research reactors with lower potential hazard, e.g. some low power reactors, critical and subcritical assemblies.

Application of the concept of defence in depth

6.7. Requirements for the application of the concept of defence in depth are established in Requirement 10 of SSR-3 [1]. Paragraph 2.12 of SSR-3 [1] describes the five levels of defence in depth for preventing or controlling deviations in normal operation, preventing accidents and mitigating radiological consequences of accidents.

6.8. Defence in depth is an important design principle that is required for all research reactors regardless of potential hazard; However, this requirement should be applied using a graded approach by recognizing that for low power research reactors, or critical and subcritical assemblies, accidents which need mitigation by the fourth or fifth level of defence in depth (see para. 2.12 of SSR-3 [1]) may not be physically possible.

6.9. For a facility with a low or medium potential hazard, the first four levels of defence in depth should be included in the design, however the design capability of the engineered safety features can use a graded approach, for example the decay heat load could be smaller, and typically a smaller fission product inventory needs to be confined or mitigated than for a research reactor with a high potential hazard.

Interfaces of safety with security and the State system of accounting for, and control of, nuclear material

6.10. Requirement 11 of SSR-3 [1] states:

> "**Safety measures, nuclear security measures and arrangements for the State system of accounting for, and control of, nuclear material for a research reactor shall be designed and implemented in an integrated manner so that they do not compromise one another.**"

This requirement is specifically for integration, and consequently it cannot be applied using a graded approach[5]. The design of the safety measures themselves are the subject of specific requirements of SSR-3 [1], and these requirements should be applied using a graded approach commensurate with the potential hazard of the facility.

Proven engineering practices

6.11. Requirement 13 of SSR-3 [1] states:

> "**Items important to safety for a research reactor shall be designed in accordance with the relevant national and international codes and standards.**"

This requirement, can be applied using a graded approach, following the detailed requirements in paras 6.19–6.24 of SSR-3 [1], for example, when no appropriate code or standard is available or when there is a departure from established engineering practice.

6.12. For SSCs for which there are no established codes or standards, SSR-3 [1] allows the use of related standards or the results of experience, tests or analysis, and requires that such an approach is justified. A graded approach can be used in the application of this requirement, based on the potential hazard of the facility, the safety classification of the SSC, and the availability of related codes and standards, such as those for nuclear power plants or from other industries. Expert judgement is necessary in using this approach and should be documented as part of the required written justification, and should be approved in accordance with a process in the management system.

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[5] Additional guidance on this topic is available in IAEA-TECDOC-1801, Management of the Interface between Nuclear Safety and Security for Research Reactors (2016).

6.13. If the design process does not follow established engineering practice, SSR-3 [1] requires that, “a process shall be established under the management system to ensure that safety is demonstrated”. A graded approach can be used in the application of this requirement based on the safety classification of the SSC, its reliability requirements and the consequences of failure established in the safety analysis. The effort required to develop the new process and its scope and level of detail should be commensurate with the hazard category of the research reactor and the safety classification of the SSC. In all cases, SSR-3 [1] requires that the SSC is monitored in service to verify that the research reactor facility operates as designed.

Provision for construction

6.14. Requirements for the provision for construction in the design of research reactors are established in Requirement 14 of SSR-3 [1].

6.15. The requirement for items important to safety to perform according to specification cannot be applied using a graded approach, and the ability of those SSCs to function as designed cannot be compromised by the manufacturing, construction and installation processes.

Features to facilitate radioactive waste management and decommissioning

6.16. Requirements for features to facilitate radioactive waste management and decommissioning are established in Requirement 15 of SSR-3 [1] and can be applied using a graded approach.

6.17. The choice of materials used in the design of a research reactor should use engineering judgement to address the utilization needs of the facility and the hazards in the decommissioning process that result from long-lived activation products. The effort and scope of design measures to minimize radioactive waste from decommissioning the research reactor should be commensurate with the potential hazard of the decommissioning process. For a research reactor with a high potential hazard, the elimination of materials that produce long-lived activation products may not be feasible, however minimizing them where possible will reduce the overall potential hazard for the decommissioning process. Planning for how those materials are managed during the operating lifetime and the decommissioning of the facility should include radiation protection considerations and could include specific technology or practices to prevent undue radiation exposure of personnel. For a research reactor with a low potential hazard, such as a subcritical assembly, the activation of core components could be insufficient to create a significant hazard from activation products. The level of detail of the characterization of the hazard to be included in the decommissioning plan, should be commensurate with the magnitude of the hazard, using a graded approach.

6.18. In addition to the original reactor design, this requirement applies to modifications made, and new experiments undertaken, during its operation. For example, this requirement could be applied using graded approach to the choice of material used in the design of new experimental equipment based on the potential hazard introduced for waste management and decommissioning.

THE USE OF A GRADED APPROACH IN GENERAL REQUIREMENTS FOR DESIGN

Safety classification of structures, systems and components

6.19. Requirements for the safety classification of structures, systems and components are established in Requirement 16 of SSR-3 [1].

6.20. All research reactors regardless of the potential hazard are required to classify the SSCs important to safety. The method for determining the safety significance of SSCs should be based on deterministic methods, complemented by probabilistic methods and engineering judgement. Research reactors with higher potential hazard and significant in-core experimental facilities, such as loops, typically require a greater number of SSCs that are in a higher safety class. The classification of SSCs important to safety is useful input when using a graded approach in the application of other requirements.

Design basis for items important to safety

6.21. Requirements for the design basis for items important to safety are established in Requirement 17 of SSR-3 [1]. The requirement to justify and document the design basis for each item important to safety can be applied using a graded approach based on the potential hazard of the facility and the level of detail for each SSC necessary to enable the operating organization to operate the research reactor safely.

6.22. Although it is not possible to apply the requirements in para 6.34 of SSR-3 [1] using a graded approach, the design basis for items important to safety in a research reactor or a critical or subcritical assembly with a low potential hazard, is typically less complex, and requires less analysis to demonstrate that its performance meets acceptance criteria, due to the low potential hazard of the facility. The classification of SSCs, based on their importance to safety, should be utilized to establish the design requirements for withstanding accident conditions without exceeding authorized limits.

Postulated initiating events

6.23. Requirements for identifying postulated initiating events are established in Requirement 18 of SSR-3 [1].

6.24. The requirement to identify the postulated initiating events cannot be applied using a graded approach. A comprehensive set of postulated initiating events is required for the safety analysis of a research reactor regardless of potential hazard, and can be identified using current safety standards and operational experience, including operational experience from similar facilities.

6.25. The analysis of the set of postulated initiating events should be commensurate with the hazard and complexity of the research reactor facility. A graded approach can be used in the safety analysis that follows from the identification of initiating events. The scope and level of detail of the safety analysis should be commensurate with the characteristics of the design and the potential hazard of the facility (see paras 6.85-6.91).

Internal hazards and external hazards

6.26. Requirements for identifying and evaluating internal hazards and external hazards are established in Requirement 19 of SSR-3 [1].

6.27. Identification of internal hazards (e.g. fire, explosion or flooding originating inside the facility) and external hazards (e.g. seismic activity, tornado or flooding external to the facility), that are applicable to the facility, should be based on the site characterization, and the design of the reactor. The application of this requirement cannot use a graded approach. A detailed list of postulated internal and external hazards is included in Appendix I of SSR-3 [1]. A graded approach can be used in applying the requirement to evaluate the effect of internal hazards and external hazards using safety analysis, based on the characteristics of the design and the potential hazard of the facility (see paras 6.85–6.91).

Design basis accidents

6.28. Requirements for identifying and considering design basis accidents are established in Requirement 20 of SSR-3 [1].

6.29. The requirement to identify a set of design basis accidents based on postulated initiating events (see para 6.23) cannot be applied using a graded approach. Because the postulated initiating events will correspond to the degree of complexity and the potential hazard of the facility, the resulting design basis accidents will also reflect the facility design. For example, a critical or subcritical assembly that does not require forced cooling flow may not have a design basis accident associated with loss of flow.

Design limits

6.30. Requirements for specifying the design limits are established in Requirement 21 of SSR-3 [1].

6.31. Design limit specifications are required to support design requirements for all relevant parameters for all operational states and design basis accidents. Design limits are limits on key physical parameters such as the maximum stress or temperature that items are exposed to, that ensure the integrity of barriers and the reliability of safety functions. Design limits should also be specified for experimental devices.

6.32. One aspect of this requirement that can be applied using a graded approach is the degree of conservatism included in design limits. The specification of design limits should include conservatism to ensure that the limits are effective, are not exceeded, and that the facility will withstand design basis accidents without acceptable limits for radiation protection being exceeded. The degree of conservatism can be adjusted according to the potential hazard of the facility and the approach taken for safety analysis. For example, a facility with a low potential hazard could apply conservative design limits and simplify the safety analysis, whereas a facility with a larger potential hazard could apply less conservatism, leading to greater effort in a more detailed safety analysis.

Design extension conditions

6.33. Requirements for the derivation and use of design extension conditions are established in Requirement 22 of SSR-3 [1]. The inclusion of design extension conditions in the safety analysis for a research reactor can use the overall graded approach for safety analysis as discussed in paras 6.85–6.91.

6.34. The requirement to derive a set of design extension conditions should be applied using a graded approach based on the potential hazard of the research reactor, engineering judgement and the results of the safety analysis of design basis accidents. The outcome from the analysis of these design extension conditions could result in additional design features in combination with an additional set of severe accident management procedures to the existing emergency plans and procedures. In a research reactor with a low potential hazard such as a subcritical assembly with few SSCs important to safety, accidental criticality could be the only event included in the analysis of design extension conditions.

Engineered safety features

6.35. Requirements for engineered safety features are established in Requirement 23 of SSR-3 [1].

6.36. For each design basis accident and selected design extension conditions, the safety analysis for the facility is required to demonstrate that operational parameters are maintained within the specified design limits by either passive or engineered safety features. As discussed in para 6.31, the requirements for design limits may be applied using a graded approach, which would have an effect on the design of engineered safety features. A research reactor with a high potential hazard including a large cooling system, could require specific engineered safety features to mitigate internal flooding caused by a leak of secondary coolant. Such a facility could also require an emergency core cooling system to collect and recirculate primary coolant inventory in response to a loss of coolant accident. The need for engineered safety features is identified by the safety analysis of the design. For a research reactor with a low potential hazard such as a critical assembly where the irradiated fuel can be safely stored in air, the safety analysis could demonstrate that no engineered safety feature is required to maintain fuel integrity in response to a loss of coolant accident.

Reliability of items important to safety

6.37. Requirements for the reliability of items important to safety are established in Requirement 24 of SSR-3 [1].

6.38. The reliability of items important to safety requires the application of the principles of redundancy, diversity, independence and fail-safe design including the application of relevant codes and standards, for example the level of redundancy or independence in a safety system.

6.39. The use of a graded approach in the application of this requirement should be based on the potential hazard of the facility, and the characteristics of the facility identified in the safety analysis. The analysis is required to demonstrate that the safety systems that prevent design limits from being exceeded (Requirement 20 from SSR-3 [1]) operate with sufficient reliability. Using a graded approach, the design of a safety system could use triplicate redundant channels to ensure a high reliability. If greater reliability is needed, the design could include a second system using diverse technology.

6.40. Where automatic or passive performance of a safety function is necessary or an inherent safety feature is used, a minimum level of reliability of the associated SSC should be established and maintained. Depending on the type of the research reactor, performance of one or more of the following safety functions may need to be automatic:

(a) Reactor shutdown;

(b) Initiation of emergency core cooling;

(c) Confinement of radioactive material.

6.41. To ensure the necessary reliability one of the following design principles may be applied:

(a) Single failure criterion;

(b) Design for common cause failures;

(c) Physical separation and independence;

(d) Fail-safe design;

(e) Qualification of items important to safety.

Recommendations on the application of these principles to research reactors in accordance with a graded approach are provided in paras 6.42–6.51.

Single failure criterion

6.42. Requirement 25 of SSR-3 [1] states that "**The single failure criterion shall be applied to each safety group incorporated in the design of the research reactor.**"

6.43. The requirement that no single failure prevents SSCs in a safety group from performing a main safety function, cannot be applied using a graded approach. The groups of equipment delivering any one of the main safety functions are required to be designed with redundancy, independence and diversity to ensure high reliability.

Common cause failures

6.44. Requirement 26 of SSR-3 [1] states:

> "**The design of equipment for a research reactor facility shall take due account of the potential for common cause failures of items important to safety, to determine how the concepts of diversity, redundancy, physical separation and functional independence have to be applied to achieve the necessary reliability.**"

Because the objective is to achieve a level of reliability necessary to ensure safe operation, this requirement can be applied using a graded approach for example, in the design of an emergency ventilation system. For a research reactor with a high potential hazard, where a design basis accident combined with the failure of emergency ventilation could result in off-site radiological consequences, to meet the acceptance criteria for the safety analysis, the design of the emergency ventilation system could exclude low-probability common cause failures through the use of diversity, redundancy and physical separation, whereas for a research reactor with a low potential hazard, the acceptance criteria may be met using a design with simple redundancy of SSCs.

Physical separation and independence of safety systems

6.45. Requirements for the physical separation and independence of safety systems are established in requirement 27 of SSR-3 [1].

6.46. Physical separation can be incorporated into a design to varying degrees, for example in a research reactor with a high potential hazard, system cable trains for two independent shutdown systems could be installed on separate floors of the facility to prevent a fault leading to a fire in one system affecting the second system. In a facility with a lower potential hazard, cable trains could be located in separate rooms or separated from each other within the same room and meet the required reliability in the safety analysis for the system.

Fail-safe design

6.47. Requirement 28 of SSR-3 [1] states “The concept of fail-safe design shall be incorporated, as appropriate, into the design of systems and components important to safety for a research reactor.”

6.48. The requirement for the use of fail-safe design features cannot be applied using a graded approach. However engineering judgement should be applied, considering the acceptance criteria used in the safety analysis of the design, to assess the appropriate extent of fail-safe design features in systems and components important to safety, to ensure that safety functions are sufficiently reliable in response to initiating events to prevent and mitigate design basis accidents and selected design extension conditions.

Qualification of items important to safety

6.49. Requirements for the qualification of items important to safety are established in Requirement 29 of SSR-3 [1].

6.50. Where the design of a research reactor includes provisions for safety functions to mitigate or prevent accident conditions, the SSCs performing those functions are required to be qualified for the appropriate environmental conditions. Maintenance and testing procedures for items important to safety should be developed recognizing the potential for a test to negatively affect the component being tested by imposing conditions of temperature, pressure or stress. The level of qualification of SSCs should be consistent with their safety classification (see para 6.20).

Design for commissioning

6.51. The requirements for the design to facilitate commissioning are established in Requirement 30 of SSR-3 [1].

6.52. The requirement to include features to facilitate the commissioning process cannot be applied using a graded approach. However, the requirement specifies the inclusion of such features “as necessary”. 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 commissioning tests and measurements, and to provide for such testing and measurement in the design. Additional recommendations on the use of a graded approach in the application of requirements for commissioning of research reactors, including experimental devices and modifications are provided in DS509A [2].

Calibration, testing, maintenance, repair, replacement, inspection and monitoring of items important to safety

6.53. Requirements for the calibration, testing, maintenance, repair, replacement, inspection and monitoring of items important to safety are established in Requirement 31 of SSR-3 [1]:

6.54. Where the performance of maintenance, periodic testing and inspection takes place in controlled areas, it is required that the activity does not result in undue exposure to radiation of the operating personnel (paras. 6.88 and 7.44 of SSR-3 [1]). This aspect of the requirement cannot be applied using a graded approach.

6.55. Aspects of this requirement which can be applied using a graded approach include:

(a) Provision for testing of SSCs during reactor operation;
(b) The storage and use of spare parts.

6.56. The design of a research reactor is required to accommodate the need for maintenance and testing of components during operation based on the reliability requirements of the SSC and its safety significance as well as the potential hazard of the facility, consistent with the manufacturer's recommendations and operating history. For example, for a research reactor with a high potential hazard, components in the reactor protection system could require testing more frequently than during shutdown periods. In such cases the design should incorporate specific features to enable testing of components or trains within a system without impairing the safety function. In a facility with a lower potential hazard, the reliable performance of SSCs in the reactor protection system could be adequately demonstrated with testing performed during periodic shutdowns.

6.57. The storage and use of spare parts for maintenance of items important to safety is an aspect of this requirement that can be applied using a graded approach, while meeting the requirements of applicable national codes and standards and regulatory conditions (e.g. admissible repair time) specified in the authorization and operational limits and conditions. For a research reactor with a high potential hazard, spare parts for some SSCs important to safety might need to meet the national standards for nuclear power plants, including requirements for procurement and storage.

6.58. There are two steps in determining the provisions for maintenance, periodic testing and inspection:

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

Design for emergency preparedness and response

6.59. Requirement 32 of SSR-3 [1] states:

> "**For emergency preparedness and response purposes, the design for a research reactor facility shall provide:**
>
> **(a) A sufficient number of escape routes, clearly and durably marked, with reliable emergency lighting, ventilation and other services essential to the safe use of these escape routes;**
>
> **(b) Effective means of communication throughout the facility for use following all postulated initiating events and in accident conditions.**"

6.60. The requirement for escape routes to meet national requirements for emergency preparedness cannot be applied using a graded approach. A graded approach can, however, be used for applying other aspects of this requirement including the following:

(a) The design of the escape routes and the location where personnel assemble;
(b) The design of the communication system used within the facility during an emergency.

6.61. The number, size and type of escape routes should be based on the layout and size of the facility, the number of personnel, and the potential hazards in various zones. For a research reactor with a high potential hazard and a large number of operating personnel, the design of escape routes could be relatively complex and the location where personnel assemble could need specific design features to protect personnel from hazards during an emergency. For a research reactor with a low potential hazard such as a critical or subcritical assembly with a small number of operating personnel, all the SSCs associated with the facility could be located in one or two rooms, and emergency routes could have simpler designs.

6.62. A communication system for use in a facility with a high potential hazard, with several floors and rooms to accommodate the facility systems, a large number of operating personnel, and elevated noise levels from equipment in some locations, could require a complex design including the ability to communicate via loudspeaker with specific rooms or zones within the building, and the ability for two-way communication between remote panels and the control rooms. The system design could also include diverse technology such as wired and wireless equipment, to increase its availability during an emergency. In a research reactor with a low potential hazard and a small number of operating personnel, where all the facility systems are contained in one or two rooms, a communication system could be a simple design to allow control room personnel to provide warnings and instructions in an emergency.

Design for decommissioning

6.63. Requirements for design to support decommissioning are established in Requirement 33 of SSR-3 [1]. A graded approach can be used in the selection of the design features to meet the requirements for the protection of workers, the public and the environment, for example, as follows:

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

6.64. The provisions in the design to enable the decommissioning process should be based on the potential hazard of the facility, the power level and duration of operation and associated level of activation of core components, the predicted number and characteristics of other SSCs with radiological hazards e.g. components in the primary coolant purification system, and the volume of material in the reactor building and reactor structure. In a facility with a low potential hazard, the used fuel and core components could require less additional shielding or specialized equipment for transport and storage than for a research reactor with a high potential hazard.

6.65. In addition to considerations for decommissioning in the original design of the reactor, this requirement also applies to design activities throughout the lifetime of the facility, including the design of modifications and new experimental devices and in preparation for decommissioning.

Design for radiation protection

6.66. Requirements for design for radiation protection are established in Requirement 34 of SSR-3 [1]. This requirement can be applied using a graded approach, for example: engineered features to maintain doses as low as reasonably achievable, or equipment to monitor and control access to the reactor and its experimental devices and facilities.

6.67. Paragraph 6.94 of SSR-3 [1] requires that adequate provision is made for shielding, ventilation, filtration and decay systems in the design of a research reactor. The design of ventilation systems can use a graded approach based on the potential radiological hazard, and the occupancy requirements for the room (in operational states and accident conditions). For a research reactor with a low or medium potential hazard, the number of locations within the facility requiring ventilation systems to mitigate radiological hazards is typically fewer than in a research reactor with a high potential hazard. Similarly, the design calculations and features necessary to ensure adequate shielding of SSCs with high radiation fields, should be fewer and less complex.

6.68. Design provisions to monitor and control access to SSCs with radiological hazards is an element of this requirement that can be applied using a graded approach. Based on the number of areas in the reactor building with a radiological hazard that requires access control, the frequency of entry, and the number of personnel, access control at a research reactor facility could be implemented using a range of design features from electronic locks with access cards, to a controlled set of keys administered by the control room, commensurate with the potential hazard of the facility and the complexity of the design.

6.69. Requirements for radiation protection (see para 6.4) and radioactive waste management (see para 6.16–6.18) at research reactors can also be applied using a graded approach and contribute to the objective of maintaining doses below prescribed dose limits and as low as reasonably achievable. Further recommendations on the use of a graded approach in the application of requirements for radiation protection and radioactive waste management are provided in DS509F [7].

Design for optimal operator performance

6.70. Requirements for the design for optimal operator performance are established in Requirement 35 of SSR-3 [1]. This requirement can be applied using a graded approach to aspects of human factors and ergonomics such as, the design of control room displays and audible signals for parameters important to safety, and the development of operating procedures as a tool to prevent human errors.

6.71. The design of the human–machine interface in the control room can use a graded approach, based on the potential hazard of the facility, the number of SSCs important to safety and the corresponding number of parameters important to safety that need to be monitored. The depth of analysis of the human–machine interface can also use a graded approach. In all cases, the analysis of the human–machine interface should consider all normal operating states, postulated initiating events, design basis accidents and selected design extension conditions, to ensure that combinations of alarms and indications in the control room are unambiguous.

6.72. For a research reactor with a medium potential hazard, the number of SSCs is typically smaller with fewer operation and maintenance procedures than in a facility with a high potential hazard. Development of procedures involves expertise in human factors to assess the human–machine interface and the possible interactions between SSCs, using a graded approach based on the size and complexity of the facility, the number of SSCs important to safety, and the potential consequences from an error made during operation or maintenance. Further recommendations on the development, use and improvement of operating procedures are provided in DS509D [5].

Provision for safe utilization and modification

6.73. Requirements for safe utilization and modification are established in Requirement 36 of SSR-3 [1]. The management system should include processes for new experiments and modifications to ensure a systematic approach to changes in the facility. The main elements of the requirement for the provision of utilization and modification in design are as follows:

(a) The reactor configuration is known at all times and it is appropriately assessed and authorized. This element of the requirement cannot be applied using a graded approach. Control of reactor configuration is an important objective of the design process in the management system (see para 5.86 of GS-G-3.5 [20]).
(b) New utilization and modification projects, including experiments that have a major significance for safety, are subject to safety analysis (see also para 6.85) and to procedures for design (see also para 6.5), construction, commissioning (see also para 6.51) and decommissioning (see also para 6.63) that are equivalent to those used for the research reactor itself. For less significant modifications and experiments, this element of the requirement can be applied using a graded approach, following the recommendations from the relevant section of this Safety Guide, based on the potential hazard of the research reactor and the potential hazard of the proposed modification (see para 6.74).
(c) Where experimental devices penetrate the reactor vessel or reactor core boundaries, they are designed to preserve the means of confinement and reactor shielding. This requirement cannot be applied using a graded approach.
(d) Protection systems for experiments are designed to protect the experiment and the reactor. This requirement cannot be applied using a graded approach. The system must protect both the experiment and the reactor.

6.74. DS510B [11] provides recommendations for designing and implementing new experiments or modifications at a research reactor. In particular, paragraphs 3.7–3.12 of DS510B [11], include the use of a categorization process to determine the safety significance of the experiment or modification, for the use of a graded approach in the application of this requirement. For a modification that is categorized as a 'major effect on safety', the operating organization is required to update the safety analysis for the research reactor and, as applicable, seek authorization from the regulatory body. The analysis of the modification should be reviewed by the safety committee and the regulatory body. For a modification categorized as a 'significant effect on safety', the existing safety analysis and authorization remain valid, but a change is required in the operating limits and conditions for the research reactor. In such cases, analysis is required to demonstrate that validity of the existing safety analysis report, and to justify the change in the operating limits and conditions. That analysis should be reviewed by the safety committee and approved by the reactor manager before the design process proceeds. New or modified operating limits and conditions are required to be reviewed and approved by the regulatory body prior to commencement of operation with the modification or new experiment (see para 7.33 of SSR-3 [1]). Modifications categorized as 'minor' or 'no effect on safety' have reduced recommendations for analysis and approval.

6.75. Requirement 90 in SSR-3 [1] includes the requirement for a change control process to evaluate new experiments or modifications and the effect the changes may have on safety or security. Technical guidelines on managing the interface between nuclear safety and security for research reactors are provided in Ref. **Error! Reference source not found.**].

6.76. The commissioning tests necessary to verify the acceptability of modifications is an aspect of this requirement that can be applied using a graded approach. For the most safety significant modifications a formal commissioning programme is required (see SSR-3 [1] para 6.110). Further recommendations on the use of a graded approach in the application of the requirement for a commissioning programme are given in paras 7.29–7.33. Recommendations on the commissioning programme for modifications in research reactors are provided in DS510B [11] and DS509A [2].

Design for ageing management

6.77. Requirements for design to support ageing management are established in Requirement 37 of SSR-3 [1] and can be applied using a graded approach, based on the potential hazard, the utilization and anticipated lifetime of the research reactor.

6.78. For a research reactor with a medium or low potential hazard, the ageing management programme, during the operation phase of the facility, should include a smaller number of items for monitoring, and fewer ageing management activities than the programme in a facility with a high potential hazard which typically has more SSCs important to safety. A design with less-accessible SSCs could be acceptable providing the programme is able to verify the condition of all items important to safety and ensure the required safety functions remain available. A graded approach can be used in the application of this requirement in such a facility, based on the safety classification of SSCs and expert judgement.

6.79. Accounting for ageing management in the design includes measures such as the use of materials resistant to degradation mechanisms, with sufficient design margins and provisions for testing, inspection and replacement. The extent to which this is applied to the design can use a graded approach, on the basis of the safety significance of the SSCs and their ease of replacement.

Provision for long shutdown periods

6.80. Requirements for the provision for long shutdown periods are established in Requirement 38 of SSR-3 [1]:

6.81. Research reactor designs normally include provisions necessary to ensure safety during shutdown of the facility and these provisions can typically be used during a long shutdown. A graded approach can be used in the application of this requirement. For all SSCs that are important to safety and which could suffer some degradation during the extended shutdown period, provision should be made for inspection, testing, maintaining, dismounting and disassembling during the shutdown period. It may be more convenient to remove equipment than to implement a preservation programme with the equipment in place; this decision is usually linked to the future of the research reactor. All modifications made to the facility due to extended shutdown are subject to Requirements 36 and 83 of SSR-3 [1], including review, assessment and approval by the regulatory body prior to implementation, when appropriate.

6.82. The design of a fuel storage location for a long shutdown period can use a graded approach, based on the number of irradiated fuel assemblies, the total fission product inventory, the decay heat generated and the specific criticality and corrosion characteristics of the fuel assemblies. For a research reactor with a high potential hazard, the design could include a separate storage pool for irradiated fuel assemblies, equipped with heat removal and purification systems. Operating limits and conditions could be implemented, after review, assessment and approval by the regulatory body, to prevent criticality safety events, and maintain the fuel assemblies in conditions where their integrity can be monitored and maintained. The design of the storage area, including cooling, purification and other support systems, should be based on safety analysis to ensure those systems are sufficiently reliable, using redundancy and the single failure criterion. For a research reactor with a low potential hazard such as a subcritical assembly with irradiated fuel containing a low fission product inventory that does not need shielding or water cooling, the irradiated fuel assemblies could be stored in a dry storage area of relatively simple design during a long shutdown period.

Prevention of unauthorized access to, or interference with, items important to safety

6.83. Requirement 39 of SSR-3 [1] states "**Unauthorized access to, or interference with, items important to safety at a research reactor facility, including computer hardware and software, shall be prevented.**"

This requirement cannot be applied using a graded approach because preventing unauthorized access to nuclear facilities is necessary regardless of the size or potential hazard of the research reactor. Access controls are needed for operating personnel, other personnel involved in the operation or use of the reactor (e.g. technical support personnel and experimenters), as well as the public, and emergency workers. A major objective of access control is to prevent the unauthorized removal of nuclear material.

Research reactors with a low potential hazard and a low inventory of fission products in irradiated fuel assemblies such as some critical and subcritical assemblies, should include specific design features for access control for those fuel assemblies.

Prevention of disruptive or adverse interactions between systems important to safety

6.84. Requirement 40 of SSR-3 [1] states:

> "**The potential for disruptive or adverse interactions between systems important to safety at a research reactor facility that might be required to operate simultaneously shall be evaluated, and any disruptive or adverse interactions shall be prevented.**"

This requirement cannot be applied using a graded approach because this evaluation is necessary for research reactors regardless of potential hazard. Design features to prevent disruptive or adverse interactions between systems are included in the safety analysis to demonstrate that systems important to safety perform reliably in response to all applicable initiating events. However, research reactors with lower potential hazard typically have fewer systems important to safety resulting in fewer adverse interactions between systems requiring evaluation.

Safety analysis of the design

6.85. Requirement 41 of SSR-3 [1] states:

> "**A safety analysis of the design for a research reactor facility shall be conducted in which methods of deterministic analysis and complementary probabilistic analysis as appropriate shall be applied to enable the challenges to safety in all facility states to be evaluated and assessed.**"

6.86. The associated requirements in paras 6.119–6.125 of SSR-3 [1] include several aspects that cannot be applied using a graded approach, for example, as follows:

(a) A safety analysis is required for every research reactor of any potential hazard level.

(b) Use of the results from the safety analysis to define operational limits and conditions, form the design basis for items important to safety, and demonstrate adequate defence in depth in the design, is also required.

(c) Comparison of the results from the safety analysis with radiological acceptance criteria is required for all research reactors regardless of potential hazard.

6.87. The safety analysis is also the basis for demonstrating the safety of the proposed design in support of an application for a licence, and should be used to confirm that any use of a graded approach in the application of safety requirements has been appropriate.

6.88. The use of enveloping events in the safety analysis to include a range of input parameters, initial conditions, boundary conditions, and assumptions, is an aspect of this requirement that can be applied using a graded approach. For a facility with a high potential hazard, the use of enveloping events, combining several such conditions, may not be possible if those enveloping events are too severe to meet the acceptance criteria. The safety analysis for such facilities typically makes limited use of enveloping events, and as a result includes a larger number of individual events for analysis. For a research reactor with a lower potential hazard, the conditions from separate events may be combined in enveloping events which, although more severe than any specific design basis accident, can be demonstrated to meet the acceptance criteria. The use of enveloping events for the safety analysis of these facilities simplifies the analysis process and requires less resources from the operating organization.

6.89. The scope and depth of the safety analysis should be based on the potential hazard of the facility, as discussed in para 1.3 and annex I of Ref. [21]. The appendix of DS510A [10] provides recommendations on the content of the safety analysis report for research reactors and indicates where elements may not be applicable for subcritical assemblies. Paras 3.1–3.7 of IAEA Safety Standards Series No. GSR Part 4 (Rev. 1), Safety Assessment for Facilities and Activities [22] contain requirements on a graded approach to safety assessment. Paragraph 3.3 of GSR Part 4 [27] states:

> "The main factor to be taken into consideration in the application of a graded approach is that the safety assessment shall be consistent with the magnitude of the possible radiation risks arising from the facility or activity."

6.90. The safety analysis required for a small facility with a relatively small number of SSCs and applicable postulated initiating events would be simpler than that for a large and complex facility. Other examples of a graded approach include the following:

(a) Analysis may demonstrate that for some identified postulated initiating events the potential for a release of radioactive material from the core is physically impossible (or can be considered with a high level of confidence to be extremely unlikely), which would remove the need for extensive engineered safety features and analysis of their failure.

(b) The presence of passive or inherent safety features and/or the absence of in-core experiments may also result in a reduction of the scope and depth of the safety analysis.

(c) The use of conservative methods and criteria is a means of simplifying the safety analysis. Facilities with low potential hazard may use conservative criteria in safety analysis, with low impact on the facility design and operation or cost.

6.91. A graded approach is also required to be used in updating the safety assessment (see para. 5.10 of GSR Part 4 (Rev. 1) [22]). The frequency at which the safety assessment is updated and the level of detail of the safety assessment should be based on the following:

(a) The number and extent of modifications to the research reactor systems and the safety significance of these modifications;

(b) Changes to procedures;

(c) Results of compliance monitoring of operational limits and conditions;

(d) Evidence of component ageing;

(e) Results from research or internal and external operating experience;

(f) Changes in site conditions;

(g) Changes to input data used in safety analysis;

(h) New regulatory requirements.

THE USE OF A GRADED APPROACH IN SPECIFIC REQUIREMENTS FOR DESIGN

Buildings and structures

6.92. Requirements for buildings and structures are established in Requirement 42 of SSR-3 [1].

6.93. A graded approach can be used for the design of shielding throughout the facility, based on the number of rooms in the building where SSCs could be a source of radiation under operational states or accident conditions, and the characteristics of the radiation risk. The buildings and structures are required to be designed to maintain radiation levels as low as reasonably achievable. For a research reactor with a high potential hazard, a larger number of rooms where equipment associated with reactor operation, isotope production, experimental devices or radioactive waste storage could require shielding as part of the building design. In a facility with a lower potential hazard, with a small number of rooms where a radiation risk is present, the design of structures to provide adequate shielding could be less complex.

6.94. Specific features in the design of buildings and structures will contribute to the application of other requirements using a graded approach, for example, as follows:

(a) Separation of areas according to their potential radiological hazard can minimize the need for radioactive waste handling, contribute to design for radiation protection, design for emergency preparedness and response, and design for fire protection.

(b) Up-to-date site evaluation can help to reduce excessive conservatism in engineering requirements for buildings and structures to ensure protection against external events (see section 2.2.1 of Ref. [21]).

Means of confinement

6.95. Requirements for the means of confinement are established in Requirement 43 of SSR-3 [1]. The results of safety analysis, considering factors such as the fission product inventory in the core, and the proximity to population centres, can provide the basis for a graded approach in the application of this requirement.

6.96. For research reactors with a high potential hazard, in some cases safety analysis might demonstrate the need for a confinement system which includes a pressure-retaining containment structure (see footnote 25 in SSR-3 [1]) to meet the acceptance criteria.. The necessary reliability of the safety functions performed by containment SSCs is determined by the acceptance criteria for off-site consequences under design basis accidents and selected design extension conditions. For a facility with a medium or low potential hazard, the reactor building could be designed without a pressure-retaining function, but with a ventilation system with features to control or mitigate radioactive releases, and meet the acceptance criteria. In all cases, the results of safety analysis should be used to determine how a graded approach is used in the design of the means of confinement, for example whether iodine traps are necessary in the event of a release of fission products from the reactor.

Reactor core and fuel design

6.97. Requirements for reactor core and fuel design are established in Requirement 44 of SSR-3 [1].

6.98. Paragraph 6.143 of SSR-3 [1] states that "The reactor core shall be designed so that the reactor can be shut down, cooled and maintained subcritical with an adequate margin for all operational states and accident conditions." This requirement cannot be applied using a graded approach. A graded approach can, however, be used in the application of other elements of this requirement, for example, provisions in the design for monitoring the physical conditions and integrity of the fuel, and analysis and experiments necessary to demonstrate the acceptability of fuel.

6.99. For a research reactor with a high potential hazard, monitoring of parameters such as temperature and flow, or monitoring radiation to detect fission products, in each fuel channel, could be design features that ensure an automatic response from the reactor protection system, or an action by operating personnel to an alarm. Such design features could be necessary to protect the facility in response to specific initiating events, demonstrated in the safety analysis; however, the implementation of such a monitoring system could add additional SSCs to the research reactor design. In a facility with a lower potential hazard, bulk monitoring of coolant parameters such as pressure, temperature and radiation could be sufficient for the safety analysis to demonstrate an adequate automatic response from safety systems and operator action in response to alarms, following postulated initiating events.

6.100. The requirement to consider neutronic, thermohydraulic, mechanical, material, chemical and irradiation related factors in the design and qualification of fuel elements, can be applied using a graded approach based primarily on the potential hazard of the research reactor, and on existing analysis and qualification documents including experience from other facilities. The extent of analyses and experiments necessary to demonstrate the acceptability of a reactor design with previously qualified fuel, could be substantially smaller, particularly in a research reactor with a medium or low potential hazard, than that necessary for reactor designs that make use of new types of fuel assemblies (where a fuel-qualification process should be conducted).

Provision of reactivity control

6.101. Requirements for the provision of reactivity control are established in Requirement 45 of SSR-3 [1]. Adequate reactivity control is required for all research reactor designs and the application of this requirements cannot use a graded approach. Further recommendations on requirements for the main safety functions are provided in para 6.3.

Reactor shutdown systems

6.102. Requirement 46 of SSR-3 [1] states:

> "**Means shall be provided for a research reactor to ensure that there is a capability to shut down the reactor in operational states and in accident conditions, and that the shutdown condition can be maintained for a long period of time, with margins, even for the most reactive conditions of the reactor core.**"

6.103. Paragraph 6.152 of SSR-3 [1] states that "No single failure in the shutdown system shall be capable of preventing the system from fulfilling its safety function when required." This requirement cannot be applied using a graded approach.

6.104. Paragraph 6.155 of SSR-3 [1] states that "It shall be demonstrated in the design that the reactor shutdown system will function properly under all operational states of the reactor and will maintain its reactor shutdown capability under accident conditions, including failures of the control system itself." This requirement cannot be applied using a graded approach.

6.105. A graded approach can be used when determining how many redundant shutdown channels are necessary, how redundant channels will be credited in the safety analysis (see section 3 of Ref. [21]), and the extent of instrumentation required for monitoring the state of the shutdown system, based on the potential hazard of the facility.

Design of reactor coolant systems and related systems

6.106.Requirement 47 of SSR-3 [1] states that "**The coolant systems for a research reactor shall be designed and constructed to provide adequate cooling to the reactor core.**"

6.107.. The coolant system is required to be designed to provide adequate cooling to the reactor with an acceptable and demonstrated margin. Adequate cooling is required not only during normal operation at the authorized power levels, but also after shutdown, under a range of anticipated operational occurrences and accident conditions that involve loss of flow or loss of coolant transients. A graded approach can be used in the design of the cooling system. The coolant system can range from the provision of forced cooling with emergency electrical power being available to power some or all of the main coolant pumps, to no emergency power for any of the coolant pumps, to a system where natural convection cooling is used for both heat removal under full power operation as well as decay heat removal. Cooling by natural convection might be adequate for some small research reactors.

6.108.In a research reactor with a high potential hazard and a high-power, the design of the SSCs to control the coolant temperature and pressure could be complex. In a research reactor with a medium potential hazard, SSCs to monitor water temperature, and pool volume could be of a simpler design while still meeting the requirements established in paras 6.73–6.81 of SSR-3 [1]. For a research reactor with a low potential hazard that does not have a heat removal system, such as some critical and subcritical assemblies, the safety analysis could confirm that there is no requirement to monitor certain parameters of the coolant such as pressure.

6.109.The requirement to monitor and control the properties of the reactor coolant (e.g. the pH and conductivity) is also applicable to all water-cooled research reactors of any power level including subcritical assemblies, to ensure that water conditions do not degrade reactor SSCs important to safety, especially boundaries that prevent the release of fission products, such as the fuel cladding:see para 6.162 of SSR-3 [1].

Emergency cooling of the reactor core

6.110.Requirement 48 of SSR-3 [1] states that "**An emergency core cooling system shall be provided for a research reactor, as required, to prevent damage to the fuel in the event of a loss of coolant accident.**" A graded approach can be used in the application of this requirement, based on the characteristics of the reactor and the fuel.

6.111. The need for an emergency cooling system should be defined in the design stage, and emergency operating procedures should be established, as necessary, taking into consideration the timescale needed for safe removal of the decay heat. In a research reactor with a high potential hazard, the design and safety analysis could demonstrate that loss of coolant accidents require an emergency core cooling system to recover water lost from the primary cooling system, collect it in a sump and recirculate it back to cool the core. For a research reactor with a medium potential hazard, a simple system to replace the coolant inventory in the pool could be sufficient to prevent significant fuel failure from loss of coolant accidents (see para 6.164 of SSR-3 [1]). For a facility with a low potential hazard, such as some subcritical assemblies, where the irradiated fuel is normally stored in dry conditions, safety analysis could demonstrate that no emergency core cooling system is necessary to mitigate the consequences of a loss of coolant accident.

6.112. For a research reactor where an emergency core cooling system is required, the system is required to perform its intended function in the event of any single failure (see para 6.165 of SSR-3 [1]).

THE USE OF A GRADED APPROACH IN INSTRUMENTATION AND CONTROL SYSTEMS

Provision of instrumentation and control systems

6.113. Requirements for the provision of instrumentation and control systems are established in Requirement 49 of SSR-3 [1].

6.114. The requirement for instrumentation and control systems can be applied using a graded approach based on the potential hazard of the facility, for example, in the provision of audible and visual alarms.

6.115. In a research reactor with a high potential hazard, there could be a large number of process variables and system parameters that necessitate audible or visual alarms or both, to provide early indication of changes in the operating conditions of the facility. Alarms may be necessary at locations other than the control room to ensure personnel are aware of the status of the facility and take appropriate action. In a research reactor with a low potential hazard such as some critical and subcritical assemblies, there could be a small number of process parameters that necessitate audible or visual alarms located in the control room. In all cases the number of alarms and their location is assessed in safety analysis and emergency preparedness and response planning for the research reactor.

6.116. A graded approach should be taken in determining the types of measurement, locations of measurement and number of measurements to be taken of reactor parameters, such as temperature, pressure, flow, pool or tank water level, gamma radiation, neutron flux and water chemistry parameters. Operational limits and conditions should provide the basis for a graded approach in the application of this requirement. For example, the pressure drop across the core is measured in many reactors in order to detect reduced flow through the core. This measurement is typically not necessary in a critical assembly or a subcritical assembly.

6.117. A graded approach in the design of instrumentation and control systems can be based on the type of reactor, the potential hazard, and the role of the SSC stated in the safety analysis. Examples of features that can be included in a design using a graded approach include the following:

(a) Redundancy and diversity (see also para 6.37);
(b) Accuracy and precision;
(c) Response time;
(d) Level of quality assurance, as determined by the safety classification;
(e) Level of automation.

6.118. An example of a graded approach in the application of safety requirements for instrumentation and control systems is the choice of the level of redundancy. Triple channel redundancy is often used for research reactors that need to operate continuously, in order to minimize spurious scrams and to allow for testing and/or maintenance of instrumentation and control equipment during operation at power. For research reactors that operate for only a few hours per week or less frequently, such as some critical assemblies, a lower level, i.e. two channel (one-out-of-two), redundancy can be applied, thus reducing the complexity of the design and of operation, as well as costs.

Reactor protection system

6.119. Requirement 50 of SSR-3 [1] states that "**A protection system shall be provided for a research reactor to initiate automatic actions to actuate the safety systems necessary for achieving and maintaining a safe state.**"

6.120. The reactor protection system is required to automatically initiate the necessary protective actions for the full range of postulated initiating events to achieve a safe state. A reactor protection system is required for all research reactor designs regardless of potential hazard. However, this requirement can be applied using a graded approach, based on the potential hazard of the facility and the number of initiating events identified in the safety analysis. For example, in a research reactor with a high potential hazard, typically there are a large number of SSCs important to safety and most of the postulated initiating events from Appendix I of SSR-3 [1] are included in the design and the safety analysis. The reactor protection system in such a facility typically monitors a large number of process parameters to ensure that automatic action can be initiated in response to any postulated initiating event. In a research reactor with a lower potential hazard, natural convection cooling and no high-pressure experimental devices, fewer postulated initiating events should be applicable for the reactor protection system design and safety analysis, for example primary pump failure, or loop rupture for a fuel testing experimental device. In such a facility the reactor protection system could be designed with less sensors for process parameters, with corresponding reduced complexity throughout the system. Other aspects of the facility design and location will affect the design of the reactor protection system include the following:

(a) At sites that could be affected by significant seismic events, a seismic sensor may be necessary to shut down the reactor, while at other sites with minimal seismic activity, such protection would not be necessary.

(b) Initiation of emergency core cooling may be necessary for certain reactors, while in others it would not be necessary (see para 6.3 (b) (iii)).

Reliability and testability of instrumentation and control systems

6.121. Requirements for the reliability and testability of instrumentation and control systems are established in Requirement 51 of SSR-3 [1].

6.122. In systems with high safety significance such as a safety system in a research reactor with a high potential hazard, a design that includes a self-checking function within each channel of the instrumentation would allow an alarm to indicate a loss of function as soon as it occurred and minimise the time for which the fault was present. In systems important to safety where safety analysis has demonstrated that a loss of redundancy could exist for 24 hours and the system meets acceptable reliability targets, a daily function test should be performed to confirm the availability of each channel of instrumentation, and the design should support that level of testing. In a system with lower safety significance, the instrumentation and control equipment could be tested weekly or monthly and perform sufficiently reliably.

Use of computer-based equipment in systems important to safety

6.123. Requirement 52 of SSR-3 [1] states:

> "**If a system important to safety at a research reactor is dependent upon computer-based equipment, appropriate standards and practices for the development and testing of computer hardware and software shall be established and implemented throughout the lifetime of the system, and in particular throughout the software development cycle. The entire development shall be subject to an integrated management system.**"

6.124. A graded approach cannot be used in the application of these requirements, including the verification and validation of computer-based equipment in systems important to safety.

Control room

6.125. Requirements for the control room are established in requirement 53 of SSR-3 [1]. Based on the potential hazard of the research reactor and the accident conditions identified in the safety analysis report, the requirements for control room design can be applied using a graded approach. In a research reactor with a high potential hazard, accident conditions identified in the safety analysis could involve combinations of severe conditions of radiation, heat and humidity. In a research reactor with a low potential hazard, such as some critical and subcritical assemblies, the safety analysis may not identify any conditions during design basis accidents that would necessitate additional protective measures in the control room. Under all conditions identified by safety analysis, the control room design is required to enable the research reactor to be maintained in a safe state, or returned to a safe state. In all cases, the control room design should consider the potential hazard of the facility, its environment its seismic resistance, ventilation systems, and fire protection.

Supplementary control room

6.126. Requirements for the supplementary control room are established in Requirement 54 of SSR-3 [1]. The supplementary control room is required to support the fulfilment of the main safety functions, and the display of important parameters and radiological conditions in the facility. A graded approach, based on the research reactor characteristics, potential hazard, and accident conditions identified in the safety analysis report can be used in the design of the supplementary control room, or a remote shutdown panel. The use of a graded approach could affect, in particular, the number of parameters to be monitored and controlled, and the actions necessary to maintain the reactor in a safe shutdown state, as well as, for example, information from radiation monitors, fire detection systems, and fire suppression systems in the research reactor, and emergency communication equipment.

6.127. Requirement 54 includes consideration for research reactors with a low potential hazard. Para 6.188 of SSR-3 [1] states that "A supplementary control room might not be necessary for critical assemblies and subcritical assemblies. In this case, the decision shall be justified on the basis of a comprehensive analysis". The safety analysis report for such a research reactor is required to demonstrate that the facility meets all acceptance criteria without a supplementary control room being included in the design.

Emergency response facilities on the site

6.128. Requirements for the emergency response facilities on the site are established in Requirement 55 of SSR-3 [1]. The requirements for the scope and functions of emergency response facilities can be applied using a graded approach, based on the nature and severity of the accident conditions identified in the safety analysis report, along with other emergency scenarios included in the scope of the design for the emergency response facilities. Aspects of this requirement that could be applied using a graded approach include the structure and number of the emergency response facilities and the provision of information and equipment for communication.

6.129.For a research reactor with a high potential hazard, the conditions near the on-site emergency response facilities could be hazardous during an emergency, including high radiation levels. To respond adequately to emergencies, separate emergency response facilities could be designed, to protect personnel involved in emergency response and to support the provision of technical support, operational support, and on-site emergency management, in an integrated manner [23]. For a research reactor with a low potential hazard, such as some subcritical assemblies, where the safety analysis does not identify a significant hazard outside the reactor building as a result of any design basis accident, the emergency response facility could be of a simpler design, with no additional protective measures.

Electrical power supply systems

6.130.Requirements for electrical power supply systems are established in Requirement 56 of SSR-3 [1]. These requirements can be applied using a graded approach based on factors including the potential hazard of the research reactor, the type and number of safety functions and engineered safety features for which normal or emergency power is needed, and the accident conditions identified for the facility which the electrical power supply needs to withstand. The reliability requirements might be different for different reactors, for the various utilization programmes of a particular reactor and for the needs of experimental devices. In applying a graded approach, the number, size and reliability of any necessary emergency power supply systems should be considered.

6.131.For a research reactor with a high potential hazard, where forced cooling is needed to remove decay heat, the level of redundancy and the number of separate channels in the emergency power supply system should be based on the results of safety analysis, including the frequency of abnormal occurrences and accident conditions for which emergency power is needed. The duration for which the emergency power supply needs to deliver power should be based on the characteristics of the fuel and the nature of the accident conditions. In a research reactor with a low potential hazard, such as some critical assemblies and subcritical assemblies with very low inventories of fission products and no significant decay heat, emergency power for cooling is not necessary.

6.132.Requirement 32 from SSR-3 [1] includes the requirement for a “means of communication throughout the facility for use following all postulated initiating events and in accident conditions”. This requirement can be applied using a graded approach (see para 6.62). On the same basis, a graded approach can be used in the application of the requirement for emergency power for the communications system. A means of communication is required following the loss of normal electrical power. As described in para 6.62, for research reactors of different potential hazards, the design of the emergency communications system should vary, based on the size and complexity of the facility and the number of locations within it where audible communication are necessary for emergency response. The emergency power supply to that system is required to be of commensurate design and reliability.

6.133. Requirement 49 of SSR-3 [1] states:

> "**Instrumentation shall be provided for a research reactor facility for monitoring the values of all the main variables that can affect the performance of the main safety functions and the main process variables that are necessary for its safe and reliable operation, for determining the status of the facility under accident conditions and for making decisions for accident management.**"

Because this requirement applies during accident conditions, the monitoring function is required during and following the postulated initiating event of loss of normal electrical power.

6.134. Requirement 49 of SSR-3 [1] can be applied using a graded approach for research reactors of different potential hazards. On the same basis, a graded approach can be used in the application of the requirement for emergency electrical power for these monitoring functions.

6.135. Most research reactors, irrespective of potential hazard, need, as a minimum, an emergency power supply for lighting (see Requirement 62 of SSR-3 [1]), instruments for monitoring the status of the facility (see Requirement 49 of SSR-3 [1]), emergency communication equipment (see Requirement 32 of SSR-3 [1]), and fire protection systems (see Requirement 61 of SSR-3 [1]), after a failure of normal electrical power.

Radiation protection systems

6.136. Requirements for radiation protection systems are established in Requirement 57 of SSR-3 [1]. The requirements for radiation protection systems can be applied using a graded approach, to ensure that the design of radiation protection systems provides adequate monitoring for the facility and is commensurate with the nature and extent of the radiological hazards. Paragraph 6.193 of SSR-3 [1] lists the radiation protection systems used in research reactor facilities and the purposes they serve. Each of these systems should be considered in the design for a research reactor, regardless of potential hazard.

6.137. Examples of considerations in the use of a graded approach to radiation monitoring include the following:

(a) The number and extent of deployment of fixed radiation monitoring instruments should be commensurate with the potential hazard of the research reactor, and the number of rooms or areas where a potential radiological hazard could arise during operational states or accident conditions.

(b) A research reactor with more SSCs where a radiological hazard from neutrons may be present, such as beam tubes and neutron guides, should deploy sufficient neutron and gamma monitors near those SSCs, as well as equipment for monitoring of contamination.

(c) A research reactor with a low potential hazard, used for teaching purposes could need only limited monitoring equipment, such as gamma monitors at the open pool end or in the control console and contamination monitors.

(d) For research reactors with a high potential hazard and a larger number of personnel, supplementary monitoring displays elsewhere in the facility, outside the control room, should be used for displaying the radiological conditions at specific locations in the facility for operational states and accident conditions (wide range monitoring).

Handling and storage systems for fuel and core components

6.138. Requirements for the handling and storage systems for fuel and core components are established in Requirement 58 of SSR-3 [1]. The aim of this requirement is to ensure safety in the handling and storage of fresh and irradiated fuel, core components and experimental devices. The main concerns are the prevention of inadvertent criticality and fuel damage from mechanical impacts, corrosion or other chemical damage. Two elements of this requirement cannot be applied using a graded approach: the requirement to prevent criticality by an adequate margin (para 6.198 (a) of SSR-3 [1]) or to enable individual fuel elements and assemblies to be identified and tracked (para 6.198 (i) of SSR-3 [1]). The application of other elements of the requirements can use a graded approach, based on the potential hazard of the facility, the design of the reactor and its utilization programme. For example, the design of the storage location for irradiated fuel could be a separate fuel storage pool with systems for cooling and purification, or an area within the reactor pool designated for fuel storage, or for a research reactor with a low potential hazard such as some critical and subcritical assemblies, the irradiated fuel assemblies could be safely stored in a dry storage area in the reactor hall.

6.139. A graded approach for the design of storage systems should be based on the storage requirements of all types of irradiated fuel assembly used in the research reactor, and for experimental fuel as well as experimental devices or equipment and materials used in isotope production. Other considerations include the means of decay heat removal and protection from mechanical impacts or corrosion.

Radioactive waste systems

6.140. Requirements for the design of radioactive waste systems are established in Requirement 59 of SSR-3 [1]. A graded approach can be used in the application of the requirements for the handling, processing, storage, transport and disposal of radioactive waste, and for control and monitoring of solid, liquid and gaseous effluent discharges, based on the characterisation, types and quantities of radioactive waste generated in the research reactor facility.

6.141.IAEA Safety Standards Series No. SSR-6 (Rev. 1), Regulations for the Safe Transport of Radioactive Material, 2018 Edition [24] includes a graded approach to performance standards for package designs for the safe transport of radioactive material, including radioactive waste, and the appendix of IAEA Safety Standards Series No. TS-G-1.4, The Management System for the Safe Transport of Radioactive Material [25] provides detailed examples of a graded approach for all aspects of transport of radioactive material. A graded approach for the design of shielding in radioactive waste systems should be based on the characteristics and radiological hazard of the waste produced at the facility.

THE USE OF A GRADED APPROACH IN SUPPORTING SYSTEMS AND AUXILIARY SYSTEMS

Performance of supporting systems and auxiliary systems

6.142.Requirements for performance of supporting systems and auxiliary systems are established in Requirement 60 of SSR-3 [1]. A research reactor with a lower potential hazard typically has fewer and simpler SSCs important to safety, including supporting and auxiliary systems. The design of supporting systems and auxiliary systems is required to be commensurate with those systems which they support and as a result this requirement cannot be applied using a graded approach as each system must comply with the design and the performance characteristics stated in the safety analysis.

Fire protection systems

6.143.Requirements for fire protection systems are established in Requirement 61 of SSR-3 [1]. Requirements for fire protection systems can be applied using a graded approach based on the results of safety analysis, fire hazard analysis or expert judgement, while remaining in compliance with regulatory requirements. For example, fire protection systems are required to provide alarms and information on the location of fires. In a research reactor with a high potential hazard, the facility typically comprises a large number of rooms on different floors of the reactor building, whereas a research reactor with a low potential hazard could be located in a single reactor hall. Using a graded approach, based on the results of a fire hazard analysis and the layout of the facility, the information displayed by the fire protection system could vary in scope and complexity.

Lighting systems

6.144.Requirements for lighting systems are established in Requirement 62 of SSR-3 [1]. The requirement can be applied using a graded approach on the basis of safety analysis and expert judgement.

Safety analysis should identify where actions by operating personnel are necessary in response to accident conditions, and which areas of the reactor building could be accessed during an emergency response. The outcome of that analysis should be used as the basis for the design of lighting systems. For a research reactor with a high potential hazard, lighting and emergency lighting systems could be extensive and include emergency electrical power. For some research reactors with a low potential hazard, the facility is located in a single reactor hall where the provision of adequate lighting is straightforward.

Lifting equipment

6.145. Requirements for lifting equipment are established in Requirement 63 of SSR-3 [1]. The requirements in para 6.210 (a)–(e) in SSR-3 [1] cannot be applied using a graded approach. The design of lifting equipment in a research reactor facility is required to prevent the lifting of excessive loads, prevent the dropping of loads with radiological consequences, permit the safe movement of lifting equipment, and permit periodic inspection. Lifting equipment used in areas where equipment important to safety is located, is required to be seismically qualified.

Air conditioning systems and ventilation systems

6.146. Requirements for air conditioning systems and ventilation systems are established in Requirement 64 of SSR-3 [1]. For a research reactor with a high potential hazard, the design may include normal and emergency ventilation systems based on the results of safety analysis and the characteristics and locations of potential airborne radiological hazards. If the research reactor has a potential tritium hazard, the ventilation system may include additional features to detect and mitigate that hazard. For a research reactor with a low potential hazard, based on the results of safety analysis, airborne radiation monitoring could be performed by periodic checks on an air filter, with no other special ventilation equipment needed in the design.

Compressed air systems

6.147. Requirements for compressed air systems are established in Requirement 65 of SSR-3 [1]. For a compressed air system serving an item important to safety at a research reactor, the design is required to specify three parameters: quality, flow rate and cleanness; this requirement cannot be applied using a graded approach.

Experimental devices

6.148. Requirement 66 of SSR-3 [1] states:

> "**Experimental devices for a research reactor shall be designed so that they will not adversely affect the safety of the reactor in any operational states or accident conditions. In**

particular, experimental devices shall be designed so that neither the operation nor the failure of an experimental device will result in an unacceptable change in reactivity for the reactor, affect operation of the reactor protection system, reduce the cooling capacity, compromise confinement or lead to unacceptable radiological consequences."

The requirement states that the operation or failure of an experimental device shall not result in specific consequences. That aspect of the requirement cannot be applied using a graded approach, those consequences must be prevented.

6.149.Some aspects of the requirement for the design of experimental devices can be applied using a graded approach, based on the potential hazard of both the facility and the experimental device. A graded approach can be applied to the design of alarm and trip signals of experiments interconnecting with the reactor protection system, and/or the control signals of the experiment connected to the reactor instrumentation and control system. For a research reactor with a high potential hazard and involving an experimental device that affects the reactivity of the core, such as a fuel testing facility, the experimental device could include specific instrumentation for the reactor protection system to initiate a scram. In the same reactor, a simple experimental device for performing irradiations that requires no cooling, may not include any instrumentation in its design. A graded approach can also be applied to the monitoring of the experimental devices from the control room(s).

6.150.The design, analysis and the authorization process (see also para 7.70), for experimental devices should be commensurate with the potential hazard of both the facility and the experimental device, the operating organization's familiarity with the experiment, and any existing, relevant safety analyses. For the installation of a new experimental device where the potential hazard is high, and a failure of the experimental device represents a new initiating event outside the scope of the safety analysis report, a revision of the safety analysis report is required, and any necessary revision of the operational limits and conditions must be submitted to the regulatory body for review, assessment and approval prior to commencement of operation with the new experimental device. For an experimental device with a low potential hazard, equivalent to experiments that have previously been installed in the facility, such as an irradiation experiment that does not need active cooling, analysis and authorization could be simplified by confirming that the irradiation conditions are bounded by those in the existing safety analysis. Recommendations for a categorization process for experimental devices are provided in section 3 of DS509A [11].

7. THE USE OF A GRADED APPROACH IN THE OPERATION OF RESEARCH REACTORS

THE USE OF A GRADED APPROACH IN ORGANIZATIONAL PROVISIONS

Responsibilities of the operating organization

7.1. The requirements for the responsibilities of the operating organization are established in Requirement 67 of SSR-3 [1]. Recommendations on meeting these requirements are provided in DS509E [6].

7.2. The general responsibilities and functions of the operating organization as well as responsibilities, functions, and line of communications of the key positions within the reactor operation organization, apply equally to all research reactors regardless their potential hazard. The application of the requirement to staff positions that require licensing or certification in accordance with the legal framework of the State is not subject to the use of a graded approach. The direct responsibility and the necessary authority for the safe operation of the reactor are required to be assigned to the reactor manager. Responsibility for the safety of the research reactor cannot be delegated.

7.3. Para 7.2 of SSR-3 [1] states:

> "The operating organization shall ensure that adequate provision is made for all functions relating to safe operation and utilization of the research reactor facility, such as maintenance, periodic testing and inspection, radiation protection, quality assurance and relevant support services."

The manner in which these functions can be performed could be subjected to the use of a graded approach in accordance with their safety significance, maturity, and complexity. For example, in a facility with a low potential hazard and subject to work arrangements that ensure effective quality checks, the maintenance, periodic testing and inspection activities could be performed by the reactor operators.

7.4. The implementation of a management system is an aspect of this requirement that can be applied using a graded approach (see also paras 4.6–4.11). In comparison with a research reactor with a low potential hazard, a research reactor of a high potential hazard will involve a large amount of management system documentation to include roles and responsibilities, procedures for operation and maintenance of reactor SSCs, and programmes for radiation protection, ageing management, environmental monitoring, and utilization.

Structure and functions of the operating organization

7.5. Requirements for the structure and functions of the operating organization are established in Requirement 68 of SSR-3 [1]. The requirement that the organizational structure is documented, including the roles that are critical for safe operation, cannot be applied using a graded approach. Changes to the documented organizational structure are required to be analysed before implementation (see para 7.11 of SSR-3 [1]).

7.6. The use of a graded approach in the application of this requirement should be based on the potential hazard of the research reactor and the State infrastructure. For similar reactors belonging to different operating organizations, different operational structures that have the same functionalities can be established. For example, a research reactor in a State with a limited nuclear programme may need a large and complete in-house capability (such as a technical support group, expertise in quality control, a large inventory of spare components, expertise in isotope production and maintenance personnel). A similar research reactor in a State with a large infrastructure and nuclear programme may not need such a large in-house capability because support could be easily obtained from external organizations.

7.7. The use of a graded approach in the application of requirements on organizational functions can be applied in the following areas:

(a) The number and duties of operating personnel i.e. for a research reactor of a low potential hazard, which is typically less complex with fewer SSCs compared to a facility with medium or high potential hazard, an individual could be assigned multiple duties. In this case, arrangements should be established to ensure functional independence (e.g. in the radiation protection function) and effective quality checks.
(b) Membership of and frequency of meetings of the safety committee(s) (see para 4.14 of this Safety Guide).
(c) Preparation and periodic updating of the safety analysis report (see DS510A [10]).
(d) Training, retraining and qualification programmes (see paras 7.10–7.15).
(e) Operating procedures (see paras 7.34–7.38).
(f) Maintenance, periodic testing and inspection programmes (see paras 7.42–7.51).
(g) Emergency planning and procedures (see paras 7.63–7.67).
(h) The radiation protection programme (see paras 7.76–7.83).
(i) The management system (see Section 4).

Operating personnel

7.8. Requirements for operating personnel are established in Requirement 69 of SSR-3 [1]. Irrespective of the potential hazard of the research reactor, the key positions within the operating organization include the reactor manager, operating personnel, including maintenance staff, radiation protection personnel, additional support staff such as training officers and safety officers, and reactor safety committee members. However, the number of personnel in some of these positions should be subjected to the use of a graded approach. For example, a larger number of operating personnel are typically needed for a research reactor with a high potential hazard, depending on its operating schedule (e.g. operation in shifts), and other factors such as the level of automation, and the number of maintenance activities. See Para 7.7 for the use of a graded approach in the application of this requirement on the number of operating personnel.

7.9. A reactor safety committee is required for all research reactors, as established in paras 7.26–7.27 of SSR-3 [1]. A graded approach should be used in the application of this requirement with respect to the size of the safety committee and the frequency of meetings, based on the potential hazard and the utilization schedule of the facility, or the number and complexity of planned modifications with safety significance.

Training, retraining and qualification of personnel

7.10. Requirements for training, retraining and qualification of personnel are established in Requirement 70 of SSR-3 [1]. Recommendations on meeting these requirements are provided in DS509E [6].

7.11. A training programme, for the training, retraining, and qualification of research reactor operating personnel and other staff is required regardless of the potential hazard of the facility. The need for a systematic approach to training, including assessment of training needs, and the design, development, implementation, and evaluation of both initial and continuing training is applicable to all research reactors. However, a graded approach should be used in the application of the requirement for training, retraining, and qualification, in that these activities should be consistent with the complexity of the research reactor design, the potential hazard, the planned utilization of the facility, and the functions that might be assigned to the personnel being trained.

7.12. The elements that could be subjected to the use of a graded approach include education level and operational experience of trainees, content and duration of initial and continuing training, training material, the nature of assessment of completed training, and qualification, which can depend on the complexity of the reactor design, potential hazard, planned utilization, and available infrastructure.

7.13. The required levels of education (e.g. post-graduate university degree, university degree, or technician qualification) and operational experience (e.g. the minimum number of hours of operation per year) for the various staffing positions could be subjected to use of a graded approach in accordance the above criteria. The contents and the duration of initial training and continuing training can be graded in accordance with the same criteria.

7.14. The training programme should cover theoretical and facility-specific knowledge, as recommended in DS509E [6]. The contents and duration of the theoretical training should be the same for all research reactors, but training on facility-specific knowledge should be more extensive for facilities with high potential hazards and those of more complex designs. The topics included in a continuing training course and the appropriate duration are dependent on the potential hazard, the complexity of the research reactor and the utilization programmes, including recent changes made to the SSCs, operating procedures, and operational limits and conditions. While the duration of continuing training could be a few days per year for research reactors of low or medium potential hazards, this duration could be of up to few weeks per year for complex facilities of high potential hazards. See also Section 4 and Annex II of DS509E [6] for guidance on the contents and duration of initial and continuing training of operating personnel for research reactors.

7.15. A graded approach to the application of the requirement for reauthorization after absences (see para. 5.13 of DS509E [6]), should ensure that retraining, requalification and examinations are commensurate with the duration of the absence, the complexity and potential hazard of the facility, and the changes to the facility and its operation during the absence of the individual. For example, in a research reactor with a high potential hazard, retraining for a reactor operator after an extended absence, could be significant, whereas for a research reactor with a lower potential hazard, retraining after a similar absence could be accomplished in less time.

OPERATIONAL LIMITS AND CONDITIONS

7.16. Requirements for operational limits and conditions are established in Requirement 71 of SSR-3 [1]. Recommendations for the preparation and implementation of operational limits and conditions are provided in DS509D [5].

7.17. Since the operational limits and conditions are based on the reactor design and on the information from the safety analysis report concerning conduct of operations, a graded approach will have been used in the application of those requirements for design and safety analysis, as discussed in Sections 3 and 6 of this Safety Guide.

Safety limits

7.18. The safety limits are established in the design stage as a result of safety analysis. A graded approach cannot be used in the application of the requirements on establishing safety limits to protect the integrity of the physical barriers against release of radioactive material. For example, the value of the safety limit on the maximum cladding temperature should be based on the physical properties of the cladding material and its environment, regardless of the potential hazard of the facility. However, the depth of analysis that is used to establish the safety limit should vary depending on the potential hazard of the facility.

Safety system settings

7.19. Paragraph 7.36 of SSR-3 [1] states that "Safety system settings shall be defined so that the safety limits are not exceeded."

7.20. For each safety limit, at least one safety system is required to be put in place to monitor parameters and to provide a signal to accomplish an action (e.g. to shut down the reactor) to prevent the parameter from approaching the safety limit. The safety system setting should be at an acceptable safety margin from the safety limit. For protective safety actions of particular importance, such as neutronic trips (scrams), redundant systems should be employed. The depth of the analysis, including the use of methods to evaluate uncertainty, performed to establish a suitable safety margin can use a graded approach. The value of an acceptable (minimum) safety margin could be subjected to use of a graded approach depending on the potential hazard of the facility.

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

Limiting conditions for safe operation

7.22. Limiting conditions for safe operation are operational constraints and administrative limitations on parameters and equipment that are established to provide acceptable margins between normal operating values and the safety system settings during start-up, operation, shutting down and shutdown. The selection of the facility-specific number and scope of limiting conditions for safe operation is an aspect of this requirement that can be applied using a graded approach. For example, a research reactor with a high potential hazard typically has more SSCs important to safety and a greater number of parameters which require limiting conditions for safe operation to be specified, than a research reactor with a medium or low potential hazard and few SSCs important to safety. Appendix I of DS509D [5] provides a list of operational parameters and equipment to be considered in establishing limiting conditions for safe operation. A graded approach could be used to determine the type and depth of analysis performed in establishing a limiting condition for safe operation in accordance with the type of reactor and conditions of operation.

Requirements for maintenance, periodic testing, and inspection

7.23. In order to ensure that safety limits and limiting conditions for safe operation are met, the relevant SSCs are required to be reliable and available. To ensure adequate reliability, SSCs important to safety are required to be maintained, monitored, inspected, checked, calibrated and tested in accordance with approved maintenance, periodic testing, and inspection programmes: see Requirement 77 of SSR-3 [1]. Surveillance requirements in the operational limits and conditions specify the frequency and scope of inspections and acceptance criteria for each SSC. A graded approach should be used in the application of these requirements on the basis of the importance to safety and the required reliability of each SSC. Additional information is provided in paras 7.42–7.51 of this Safety Guide.

Administrative requirements

7.24. Administrative requirements include those for the organizational structure and responsibilities, minimum staffing, training and retraining, review and audit procedures, records and reports, and event investigation and follow-up (see para. 7.40 of SSR-3 [1]). In a research reactor with a high potential hazard and continuous operation during day and night, the operational limits and conditions could specify several administrative requirements for shift turnover, minimum staffing levels, requirements for technical specialists such as chemistry or radiation protection personnel, operating logs, and reporting of events which may not be needed, or needed in the same level of detail, for research reactors of medium or low potential hazard or those that have a limited operation schedule.

Violations of operational limits and conditions

7.25. The requirement for action after a violation of operational limits and conditions, cannot be applied using a graded approach. The nature of the action will be determined by the regulatory framework of the State and will typically depend on the severity of the violation.

PERFORMANCE OF SAFETY RELATED ACTIVITIES

7.26. Requirements for the performance of safety related activities are established in Requirement 72 of SSR-3 [1].

7.27. Paragraph 7.44 of SSR-3 [1] states:

> "All routine and non-routine operational activities shall be assessed for potential risks associated with harmful effects of ionizing radiation. The level of assessment and control shall depend on the safety significance of the task."

7.28. For a research reactor with a high potential hazard, the operating organization could include a group to plan, assess, and control operation and maintenance tasks. For research reactors with lower potential hazard, the smaller number of operation and maintenance tasks could be planned, assessed, and controlled by the same personnel who perform the operation and maintenance of the facility. Expertise in radiation protection is necessary to assess all tasks involving exposure to radiation.

THE USE OF A GRADED APPROACH IN COMMISSIONING

7.29. Requirements for the commissioning programme are established in Requirement 73 of SSR-3 [1]. Recommendations on meeting Requirement 73 are provided in DS509A [2].

7.30. A commissioning process is required for all SSCs, activities and experiments regardless of the potential hazard of the reactor facility. However, a graded approach can be used in the application of the requirement for a commissioning programme in the following areas:

(a) Organization for commissioning;
(b) Commissioning tests and stages;
(c) Commissioning procedures and reports.

7.31. An organization structure for commissioning, including for utilization and modifications important to safety, is required regardless of the potential hazard of the facility. However, the number of personnel within this structure, including the number of the operating group and the necessary expertise, can vary depending on the potential hazards of the facility and its design. For example, research reactors of low potential hazard and subcritical facilities typically have fewer personnel in the operating group and less or no expertise on power rise tests and operation at high power levels.

7.32. Stage C of commissioning (power ascension tests and power tests up to rated full power as defined in para 3.17 and paras 5.30–5.37 of DS509A [2]) is not necessary for subcritical assemblies, and the scope, extent, and duration of Stage C is much less for low power research reactors (typically of low potential hazard) compared to those of higher power levels. The scope, number and types of commissioning tests, procedures and reports, as well as the number of hold points in the commissioning process are much less for research reactors of low potential hazard and less complex design compared to those for facilities with medium or high potential hazards. The number of hold points in the commissioning process should be determined in part by the potential hazard of the facility and the safety significance of the subsequent step in the commissioning procedure. Regardless of the potential hazard of the facility, there should always be a hold point for tests prior to fuel loading (pre-operational tests). A graded approach to testing should be adopted (see Para. A.2 of the Appendix of DS509A [2]). The extent and type of tests to be performed should be determined on the basis of the importance to safety of each item and the potential hazard of the reactor. Further recommendations on use of a graded approach in application of safety requirements on commissioning are provided in DS509A [2].

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

THE USE OF A GRADED APPROACH IN OPERATION

Operating procedures

7.34. Requirements for operating procedures are established in Requirement 74 of SSR-3 [1]. Recommendations for the preparation of operating procedures are provided in DS509D [5]. Appendix II of DS509D [5] presents an indicative list of operating procedures for research reactors.

7.35. Prior to operation, a graded approach should have been used in the application of the requirements for research reactor design, construction, and safety analysis, including the development of operational limits and conditions, based on the potential hazard, the design, and the complexity of the facility. In addition, a graded approach should have been used in the application of the requirements for the establishment and implementation of the management system that governs the format, development, review, control, and training on the use of operating procedures. The use of a graded approach in the application of safety requirements for operating procedures should be consistent with the use of that approach in these programmes and activities.

7.36. The list of operating procedures presented in Appendix II of DS509D [5] should be assessed for applicability to a specific research reactor. The number of operating procedures developed should be dependent upon the characteristics of the research reactor and should be less for reactors with fewer SSCs important to safety and a low potential hazard. For example, in a facility with a low potential hazard such as some critical and subcritical assemblies, procedures related to surveillance of systems such as cooling and ventilation systems may not be necessary, and fewer procedures may be needed related to fuel handling.

7.37. All personnel using operating procedures are required to be thoroughly familiar with them and proficient in their use. However, a graded approach should be used in application of the requirement for personnel to be adequately trained in the use of operating procedures. For example, in a facility with a high potential hazard, with complex SSCs important to safety, training on a specific procedure may require extensive prerequisite training on related SSCs. Training for the use of a simple procedure for the maintenance of a component in a research reactor with a low potential hazard could take less time.

7.38. While all operating procedures are required to be prepared, reviewed and submitted for approval on the basis of criteria established by the operating organization and regulatory requirements, the detail of operating procedures can differ on the basis of their importance to safety. For example:

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

Main control room, supplementary control room and control equipment

7.39. Requirements for the main control room, the supplementary control room and control equipment are established in Requirement 75 of SSR-3 [1]:

7.40. Para 7.63 of SSR-3 [1] states:

> "The habitability and good condition of control rooms shall be maintained. Where the design of the research reactor foresees additional or local control rooms that are dedicated to the control of experiments that could affect the reactor conditions, clear communication lines shall be developed for ensuring an adequate transfer of information to the operators in the main control room."

In a research reactor with a high potential hazard, the supplementary control room could include more monitoring and control equipment than a shutdown panel for a research reactor of low or medium potential hazard. The number of shutdown panels in locations other than the reactor control rooms should be commensurate with the potential hazards of the facility. The frequency and scope of tests performed by the operating organization to confirm that the supplementary control room and the shutdown panels are in a proper state of operational readiness should be commensurate with the nature of the equipment and the potential hazard of the facility.

Material conditions and housekeeping

7.41. Requirements for material conditions and housekeeping are established in Requirement 76 of SSR-3 [1]. High standards of material conditions and housekeeping, including cleanliness, accessibility, adequate lighting, appropriate storage conditions, and identification and labelling of safety equipment are required regardless of the potential hazard of the facility. A research reactor of a low potential hazard and fewer SSCs important to safety, should necessitate less effort for a high standard of housekeeping and cleanliness compared to those facilities of medium and high potential hazards with a larger number of SSCs.

Maintenance, periodic testing and inspection

7.42. Requirements for maintenance, periodic testing and inspection are established in Requirement 77 of SSR-3 [1]. Recommendations for maintenance, periodic testing and inspection for research reactors are provided in DS509B [3].

7.43. A maintenance, periodic testing and inspection programme is required for all research reactors regardless of their potential hazards. The scope, extent of the programme, and the resources required for planning, implementation and assessing this programme should be commensurate with the potential hazards of the facility and could vary significantly depending on the design, size and complexity of the reactor. For a facility with a low potential hazard and fewer SSCs important to safety, these activities can be performed by the operating personnel, but a dedicated maintenance group is typically needed for a large research reactor facility with more SSCs and a high potential hazard. The number of maintenance staff should also be commensurate with the potential hazards of the facility.

7.44. Three aspects of Requirement 77 should be applied using a graded approach: the development of procedures, the frequency of maintenance, periodic testing and inspection, and the work permit system used to implement these procedures. The graded approach should be based on the potential hazard of the facility, the safety significance of the SSCs involved, the complexity of the maintenance, periodic testing and inspection activity, and the potential radiation risk of relevant tasks.

7.45. In developing the procedures for maintenance, periodic testing and inspection, consideration should be given to the importance to safety of the SSC concerned, and to the complexity of the maintenance, testing or inspection activity, and to the experience of the staff and their familiarity with the SSCs. A graded approach to the application of requirements for procedures is discussed in paras 7.34–7.38 of this Safety Guide.

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

7.47. The frequency of maintenance, periodic testing and inspection of individual SSCs is also required to be established and adjusted on the basis of experience and the importance to safety of the SSC concerned (see para 7.72 of SSR-3 [1]). For example, some instrumentation in the scram safety actuation system could require daily testing to demonstrate the functional operability and availability whereas a sump pump could be tested at a lower frequency based on the results of the safety analysis.

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

7.49. The period for which an SSC is permitted to be out of service while reactor operation continues is usually stated in the operational limits and conditions for the facility and can be based on the availability requirement for the SSC from the safety analysis. For example, outage times of any duration might not be acceptable for automatic shutdown systems, while outage times of up to several days might be acceptable for other systems, with appropriate compensatory measures (e.g. for a purification system monitoring the primary coolant pH, the system could be unavailable for several days, but pH measurements could be taken manually each shift). The allowed outage time should depend on the extent to which safety is impacted, or the ease of applying compensatory measures.

7.50. The use of a work permit system is required in all research reactors of all levels of potential hazard :see para 7.69 of SSR-3 [1]. This element of the requirement can be applied using a graded approach. All work permits for activities with potential radiation risk should be reviewed by radiation protection personnel to ensure that doses from the activity are within prescribed limits and are as low as reasonably achievable. Further recommendations on radiation protection in research reactor operation are provided in DS509F [7].

7.51. Some maintenance, periodic testing and inspection activities are highly specialized and involve complex and sophisticated techniques. Such activities are often performed by contracted experts external to research reactor operating organizations. Such outsourcing should be carefully considered by the operating organization to ensure that external support is secured and that resources will be available throughout the operating lifetime of the facility. Recommendations on the use of external contractors for the performance of maintenance, periodic testing and inspection are provided in DS509B [3].

Core management and fuel handling

7.52. Requirements for core management and fuel handling are established in Requirement 78 of SSR-3 [1]. Recommendations for core management and fuel handling are provided in DS509C [4].

7.53. The requirement to establish procedures for core management and fuel handling to ensure compliance with the operational limits and conditions and consistency with the utilization programme, is applicable to all research reactors regardless of their potential hazards. In addition, the requirements on monitoring the integrity of reactor core and fuel and on confinement of failed fuel, as established by Para 7.82 of SSR-3 [1], apply equally to research reactors regardless of the potential hazard from the facility.

7.54. Research reactors with a low potential hazard, requiring infrequent changes to core configuration, may need a less comprehensive core management and fuel handling programme. These reactors 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 DS509C [4] should be considered, some might not apply to these research reactors with a low potential hazard. Some research reactors, including some critical and subcritical assemblies, may undergo frequent changes to core configuration and fuel handling operations. As a result, these facilities require a more comprehensive core management and fuel handling programme.

7.55. Changes to research reactor core management and fuel handling procedures are modifications of major safety significance. DS510B [11] provides recommendations on a method for determining the safety significance of modifications to a research reactor and this method is applicable to core management and fuel handling. A graded approach to the application of requirements for analysis and verification associated with the proposed changes of core management and fuel handling activities may be possible on the basis of their safety significance (see also paras 7.70–7.75 of this Safety Guide).

7.56. A graded approach can also be used in determining the appropriate level of detail of the documentation and records on the status of fuel and core components. In comparison with small research reactors of low potential hazard, research reactors of a high potential hazard may need a more comprehensive system to document the status, and the evolution of this status with time, of each fuel assembly and core component including experimental devices. In some research reactors of a high potential hazard with more complex systems and utilization programme, a dedicated group for core management and fuel handling may be necessary.

Fire safety

7.57. The requirements for fire safety are established in Requirement 79 of SSR-3 [1]. Recommendations for fire safety are provided in IAEA Safety Standards Series No. DS503, Protection against Internal and External Hazards in the Operation of Nuclear Power Plants [26] and IAEA Safety Standards Series No. DS494, Protection Against Internal Hazards in the Design of Nuclear Power Plants [27]. Compliance with national requirements for fire safety cannot be subject to a graded approach.

7.58. The potential fire hazards should be discussed in the safety analysis report and an indication should be provided of their relative importance (i.e. in terms of likelihood and consequences) in the facility. This information can serve as a basis for the use of a graded approach in the implementation of the fire protection measures. For example, a fire affecting the instrumentation in the control room of a research reactor with a high potential hazard could be identified in the safety analysis as an event with a potential high consequence, and mitigated by the automatic action of an inert gas extinguishing system, combined with manual firefighting action from trained personnel. Fires in administrative areas of a reactor building, with a low safety consequence identified in the safety analysis could be mitigated by the deployment of hand-held fire extinguishers and the actions of fire-fighting personnel.

7.59. Use of a graded approach to implement the measures for fire protection might be facilitated by provisions incorporated into the design in accordance with the fire hazard analysis, which is required for all research reactors regardless of potential hazard (see para 7.87 on SSR-3 [1]), and which should be periodically reviewed and updated (see DS503 [26]). Use of a graded approach to fire protection might also be facilitated by siting considerations.

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

7.61. The use of a graded approach to the application of the requirement for fire protection measures incorporated into the design in accordance with the fire hazard analysis (see para 7.87 of SSR-3 [1]) is discussed in para 6.143.

Non-radiation-related safety

7.62. Requirements for a non-radiation-related safety programme are established in Requirement 80 of SSR-3 [1]. Each non-radiation hazard should be adequately addressed based on the nature of the hazard itself. The scope and level of detail of the programme should be developed using a graded approach based on the size and complexity of the research reactor facility and the specific hazards arising from its SSCs and operation.

Emergency preparedness

7.63. The requirements for emergency preparedness are established in Requirement 81 of SSR-3 [1]. Further requirements for emergency planning and response are established in IAEA Safety Standards Series No. GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [23].

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

7.65. As a basis for using a graded approach, Requirement 4 of GSR Part 7 [23], establishes a categorization scheme for nuclear and radiation related threats which is required to be used, and which provides a basis for developing optimized arrangements for preparedness and response. Most research reactor facilities are in emergency preparedness category II or III (see para 4.19 of GSR Part 7 [23]), dependent on whether the facility can generate events that require an off-site response as well as an on-site response.

7.66. The magnitude of the potential source term, the facility's proximity to population groups, and the engineered safety features are the most important factors to be considered in applying the requirements for the emergency plan using a graded approach, for example in the following areas:

(a) The organization needed to carry out the emergency response;
(b) The size of the urgent protective action planning zone;
(c) The identification and classification of the hazard;
(d) Notification of and communication with local, regional and national authorities, as appropriate;
(e) The amount, nature and storage location of equipment needed to survey and monitor people and the environment in the event of an emergency;
(f) The number and identity of external organizations (e.g. police, fire services, medical treatment and medical transport) that will help in an emergency, and the training of these organizations and

the nature of agreements with the operating organization. Although the emergency might not have an off-site impact, it is generally prudent to establish contact with appropriate local, regional or national authorities to ensure their agreement if a request for assistance is issued;

(g) The timescales envisaged for the various phases of the response to an emergency;

(h) The types, frequency and extent of training, exercises and drills of on-site and off-site emergency response;

(i) The nature and amount of other resources needed for preparedness for and response to an emergency.

7.67. For a research reactor of high potential hazard, there could be a need for a large amount of portable radiation protection equipment and emergency response equipment to be available at on-site locations. This equipment could be used in emergency preparedness drills and in training of on-site personnel and personnel from off-site organizations. For a research reactor with a lower potential hazard and no potential for off-site radiological consequences, far fewer portable radiation protection instruments and emergency equipment could be necessary for emergency response. In all cases, stored equipment for use in emergency response is required to be maintained in good operational condition, and should be included in the maintenance and periodic testing and inspection programme for the research reactor.

Records and reports

7.68. Requirements for records and reports are established in Requirement 82 of SSR-3 [1]. Requirements for the control of records and documentation are also established in Requirements 8 and 10 of GSR Part 2 14], and recommendations are provided in paras 5.35–5.49 of GS-G-3.1 [115]. The requirement for design information to be kept up to date for the duration of the operational stage of the research reactor, and the requirement for information in logbooks and other records to be properly dated and signed cannot be applied using a graded approach.

7.69. Consistent with the purpose for which reports are prepared and records are kept, para. 2.44 of GS-G-3.1 [1] lists specific examples of where a graded approach could be applied to controls for the records management process, as follows:

(a) Preparation of documents and records;

(b) Need for and extent of validation;

(c) Degree of review and the individuals involved;

(d) Level of approval to which documents are subjected;

(e) Need for distribution lists;

(f) Types of document that can be supplemented by temporary documents;

(g) Need to archive superseded documents;

(h) Need to categorize, register, index, retrieve and store document records;
(i) Retention time of records;
(j) Responsibilities for the disposal of records;
(k) Types of storage medium, in accordance with the specified length of time of storage.

Utilization and modification of a research reactor

7.70. Requirements for the utilization and modification of research reactors are established in Requirement 83 of Ref [1]. Recommendations for the utilization and modification of research reactors are provided in DS510B [11].

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

7.72. In cases where an experiment or modification was not anticipated and analysed in the design, its safety significance should be determined. DS510B and its Annex I [11] provide guidance for and an example of categorization for the treatment of modifications according to their potential hazard. It uses a safety screening checklist which divides modifications into four categories, as follows:

(a) A major effect on safety;
(b) A significant effect on safety;
(c) A minor effect on safety;
(d) No effect on safety.

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

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

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

7.75. The level of detail and depth of analysis that are necessary for design, safety analysis, quality assurance, installation procedures, commissioning plan, training for personnel who will implement the modification as well as those who will use the SSC after modification, can be implemented using a graded approach. Similarly, the scope and level of detail of the review performed by the regulatory body can use a graded approach based on effect on safety of the modification.

Radiation protection programme and environmental monitoring programme

7.76. The requirements for a radiation protection programme at a research reactor are established in Requirement 84 of SSR-3 [1]. Radiation protection requirements are also established in IAEA Safety Standards Series No. GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [28]. Recommendations for radiation protection in the design and operation of research reactors are provided in DS509F [7].

7.77. While the content of the radiation protection programme depends on the design, power level, radiological hazards, and utilization of the particular research reactor, many aspects of the programme should be similar for all research reactors. Para 7.110 of SSR-3 [1] lists measures which are required in radiation protection programmes for research reactors of all levels of potential hazard and hence cannot be applied using a graded approach.

7.78. The application of the requirements for the radiation protection programme should be consistent with the reactor's design and its utilization (see paras 1.5 and 1.9 of DS509F [7]).

7.79. The scope of the environmental monitoring programme is dependent on the potential radiological hazard of the reactor. For example, a facility located close to a densely populated area should result in a more extensive environmental monitoring programme.

7.80. Working areas within a research reactor should be categorized into supervised areas and controlled areas, according to the magnitudes of the expected exposures, the likelihood and magnitude of potential exposures, and the nature and extent of the required radiological protection measures (see paras 5.44–5.46 and 5.48 of DS509F [7]).

7.81. For a research reactor facility with a high potential hazard, it may be necessary to further categorize the controlled areas into different levels, for example, levels I, II and III. Specific procedures may be prescribed for work in level II controlled areas (in addition to those procedures prescribed for level I areas), which may involve, in some cases, the use of protective garments, equipment or tools. Level III controlled areas should normally be closed by a physical barrier (e.g. an airlock door) that is opened only by authorized workers. Furthermore, opening a door to a level III controlled area during reactor operation could be designed to result in automatic shutdown of the reactor. For a research reactor with less diverse radiological hazards, and a small number of areas where radiation hazards are present, the controlled area could be categorized into a small number of levels where additional radiation protection measures are needed. For a research reactor with a low potential hazard, with no locations where high dose rates are present, level II and level III controlled areas may not be needed.

7.82. A critical assembly could present a higher risk of external radiation exposure of operating personnel than a higher power research reactor but the latter could present a higher risk of contamination of personnel causing internal radiation exposure. In addition, because critical assemblies are sometimes located within conventional industrial buildings, reactivity accidents involving a critical assembly could result in a higher risk of contamination outside the building, compared with higher power reactors with a larger source term that have a containment structure. These elements should be considered in use of a graded approach in the application of the requirements for a radiation protection programme for these types of facilities.

7.83. Allocating sufficient resources for the radiation protection programme to advise on and enforce radiation protection regulations, standards and procedures (see para 7.108 of SSR-3 [1]), is an aspect of this requirement that can be applied using a graded approach. For example, at a research reactor with a high potential hazard and many SSCs with potential radiation hazards, the radiation protection group in the operating organization could include a large number of personnel, working in shifts, trained to use a number of instruments for detecting and characterising sources of radiation, and involved in the planning and execution of activities in the facility. In a research reactor, with a low potential hazard such as some critical and subcritical assemblies, radiation protection tasks could be performed by one or two personnel who are also trained in other operational activities.

Management of radioactive waste

7.84. Requirements for the management of radioactive waste in research reactors are established in Requirement 85 of SSR-3 [1]. Recommendations for management of radioactive waste are provided in DS509F [7], including the characterization, classification, processing (i.e. pre-treatment, treatment and conditioning), transport, storage and disposal of radioactive waste.

7.85. The scope of the radioactive waste management programme should be consistent with the size and complexity of reactor operations. This requirement can be applied using a graded approach based on the quantity and characteristics of radioactive wastes generated, the quantity and characteristics of liquid and/or gaseous effluents generated, and the corresponding national regulatory limits. For a research reactor with a high potential hazard, there may be a diverse range of radioactive waste generated including waste oil from maintenance activities, liquid and gaseous effluents from reactor operation, solid and liquid waste from isotope production, contaminated disposable materials from radiation protection and decontamination activities. In contrast, the quantity of waste generated, and the types of radiation risk from the waste for a research reactor with a low potential hazard are typically less. The provisions in the waste management programme should be commensurate with the waste generated. For any design of research reactor, levels of waste generated should be minimized, to ensure that releases of radioactive material to the environment are kept as low as reasonably achievable (see para 7.116 of SSR-3 [1]).

Ageing management

7.86. Requirements for ageing management of research reactors are established in Requirement 86 of SSR-3 [1]. Recommendations on ageing management for research reactors are provided in DS509G [8].

7.87. Aspects of this requirement that can be applied using a graded approach include the following:

(a) The frequency of inspections for the detection of ageing effects;
(b) The resources necessary to implement an ageing management programme;
(c) The implementation of corrective actions resulting from a periodic safety review.

7.88. The appropriate frequency of inspections, and the measures for mitigation of ageing effects, should be based on the importance to safety, estimated service life, complexity and ease of replacement of individual SSCs. In most research reactors, it is feasible to inspect most SSCs periodically and to replace components if necessary. For a research reactor with a high potential hazard, inspections should be prioritized where degradation mechanisms have been identified. For a research reactor with a low potential hazard, the SSCs that perform the main safety functions should be prioritised for ageing management inspections.

7.89. Allocating the resources necessary to implement the requirements for an ageing management programme can also use a graded approach. For a research reactor with a high potential hazard, a dedicated organizational unit may be needed to implement such a programme, to plan and perform ageing management activities, with an appropriate interface with the maintenance programme (see DS509G [8]). For a research reactor with a low potential hazard, the ageing management programme activities might be planned, supervised and performed by the maintenance personnel in the operating organization.

7.90. The requirement to implement corrective actions resulting from a periodic safety review can be applied using a graded approach. The global assessment of the findings from this review should apply risk-based significance levels to all proposed corrective actions. The operating organization may decide not to implement a corrective action for an issue of low safety significance where there is sufficient justification. This approach to the corrective actions from a periodic safety review is applicable to all research reactors regardless of potential hazard.

Extended shutdown

7.91. Requirements on extended shutdown of a research reactor are established in Requirement 87 of SSR-3 [1]. Additional information on extended shutdown is provided in paras 6.80–6.81 of this Safety Guide. Requirement 2 of SSR-3 [1] states that "The operating organization for a research reactor facility shall have the prime responsibility for the safety of the research reactor over its lifetime". That responsibility remains during the period of extended shutdown, when the decision has not been made to decommission or restart the research reactor.

7.92. A graded approach should be used for the scope and details of the activities, the measures to be implemented, the level of reviews, the frequency and extent of maintenance, and the testing and inspection activities during an extended shutdown, and the extent of relief from requirements that apply during the normal operating regime, including operational license conditions such as an operational limit or condition. Such a relief should be subjected to safety analysis and regulatory review and assessment.

Feedback of operating experience

7.93. Requirements for feedback of operating experience of a research reactor are established in Requirement 88 of SSR-3 [1]. The requirement for the operating organization to report, collect, screen, analyse, trend, document and communicate (including communication to support organizations such as manufacturers) operating experience at the reactor facility in a systematic way (see para 7.126 of SSR-3) applies regardless of the potential hazard of the research reactor.

7.94. The resources necessary to implement the operating experience programme should be commensurate with the potential hazard of the research reactor, the number and complexity of SSCs important to safety and the size of the operating organization. The number of items of operating experience identified by operating personnel and reviewed by reactor management should be commensurate with the size of the operating organization, the scope of the utilization programme, and the number of SSCs important to safety in the facility.

8. USE OF A GRADED APPROACH IN THE PREPARATION FOR DECOMMISSIONING OF RESEARCH REACTORS

8.1. Requirement 89 of SSR-3 [1] states:

"**The operating organization for a research reactor facility shall prepare a decommissioning plan and shall maintain it throughout the lifetime of the research reactor, unless otherwise**

approved by the regulatory body, to demonstrate that decommissioning can be accomplished safely and in such a way as to meet the specified end state."

8.2. The scope, extent, and level of detail of the safety assessment for decommissioning and the decommissioning plan should be commensurate with the hazards associated with the decommissioning of the research reactor. The effort associated with meeting the requirements for the preparation and review of decommissioning plans and procedures should also be based on the potential hazards associated with the decommissioning of the facility. Depending on these hazards, and on its design, complexity, and history of its operation and utilization, a graded approach can be used to determine the most appropriate level and depth of analyses, the type and number of decommissioning procedures to be prepared as well as the scope and depth of safety reviews and assessments. A graded approach should also be used in determining the appropriate extent and type and level of details of surveillance and radiation protection measures, including monitoring, during transition from operation to decommissioning.

8.3. Preparation for decommissioning should include consideration of knowledge of the facility which might be lost when the reactor is permanently shut down because of possible retirement or departure of experienced personnel. The requirement for the operating organization to retain personnel and preserve knowledge of the research reactor (see para 8.7 of SSR-3 [1]) should be applied using a graded approach, based on the potential hazards of the facility as well as based on the knowledge of the facility and its safety significance to decommissioning. For research reactors with a smaller operating organization, preserving the knowledge of a small number of key personnel may be essential for preparation for decommissioning.

8.4. The scope and level of details of the decommissioning plan should use a graded approach based on the potential hazard of the shut down facility (e.g. with nuclear fuel removed), resources available for decommissioning, time period to decommissioning and the required end state of the facility (e.g. full or partial decontamination and/or dismantling or release of the site from regulatory control).

9. USE OF A GRADED APPROACH TO THE INTERFACES BETWEEN SAFETY AND SECURITY FOR RESEARCH REACTORS

9.1. Requirement 90 of SSR-3 [1] states:

"**The interfaces between safety and security for a research reactor facility shall be addressed in an integrated manner throughout the lifetime of the reactor. Safety measures and security measures shall be established and implemented in such a manner that they do not compromise one another.**"

9.2. The requirement that safety and security issues are addressed in an integrated manner, cannot be applied using a graded approach. Safety and security are two distinct areas essential for reactor operation. A graded approach should be used in the application of safety requirements and also in the application of security recommendations. A graded approach can be also used in the activities that are required for effective management of the interface between safety and security. This includes the following:

(a) The number and extent of coordinated safety and security regulatory inspections and emergency drills;
(b) The extent and level of detail of review of the access control procedures by safety specialists;
(c) The extent and level of detail of review of the operating and maintenance procedures by security specialists;
(d) The extent of reviews by security specialists of modifications important to safety;
(e) The extent of reviews of modifications of nuclear security systems by safety specialists while ensuring appropriate information security;
(f) The contents of training of safety aspects for security specialists and vice versa.

9.3. Recommendations related to the interfaces between safety and security are included in the Safety Guides referenced in para 1.3, in particular DS509E [6] and DS510B [11].[6]

[6] Additional guidance on the use of a graded approach and the safety and security interface is available in IAEA-TECDOC-1801, Management of the Interface between Nuclear Safety and Security for Research Reactors (2016).

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