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International Atomic Energy Agency

Guidelines for the Review of Research Reactor Safety

Reference Document for IAEA
Integrated Safety Assessment
of Research Reactors (INSARR)

Vienna, April 2024

IAEA Services Series 25 (Rev. 1)

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OF RESEARCH REACTOR SAFETY

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IAEA SERVICES SERIES No. 25 (Rev. 1)

GUIDELINES FOR THE REVIEW OF RESEARCH REACTOR SAFETY

REFERENCE DOCUMENT FOR IAEA INTEGRATED SAFETY
ASSESSMENT OF RESEARCH REACTORS (INSARR)

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2024

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FOREWORD

The Integrated Safety Assessment of Research Reactors (INSARR) safety review service is offered to IAEA Member States to support them in ensuring and enhancing the safety of their research reactors. The service consists of a comprehensive peer review and assessment of the safety of the involved research reactors that is conducted upon request against the IAEA safety standards and the provisions of the Code of Conduct on the Safety of Research Reactors. INSARR missions are carried out for the benefit of the operating organizations or the regulatory bodies of the requesting Member States and can cover new research reactors under design or operating research reactors, including those under a project and supply agreement with the IAEA.

INSARR missions and other limited scope safety review missions are carried out following the guidelines presented in this publication, which is a revision of IAEA Services Series No. 25, *Guidelines for the Review of Research Reactor Safety: Revised Edition*, published in 2013. The current revision takes into account the publication of relevant new and updated IAEA safety standards — including IAEA Safety Standards Series No. SSR-3, *Safety of Research Reactors* — and other publications providing guidance on all safety related aspects throughout the lifetime of a research reactor.

This publication describes the preparation, implementation, reporting and follow-up of safety review missions to research reactors as conducted by the IAEA under the INSARR safety review service. It is also intended to be of assistance to operating organizations and regulatory bodies in conducting safety assessments of research reactors to address individual safety issues such as ageing or major modifications, as well as other types of safety review such as internal reviews and audits by the reactor management, peer reviews and regulatory inspections.

The IAEA officers responsible for this publication were A.M. Shokr and D.F. Sears of the Division of Nuclear Installation Safety.

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CONTENTS

1.	INTRODUCTION	1
1.1.	BACKGROUND	1
1.2.	OBJECTIVE	2
1.3.	SCOPE.....	3
1.4.	STRUCTURE.....	3
2.	PREPARATION, CONDUCT, REPORTING AND FOLLOW UP OF AN INSARR... 3	
2.1.	PREPARATION OF AN INSARR	3
2.2.	PRE-INSARR MISSION	4
2.2.1.	Definition of objectives, scope, strategy and review methods	5
2.2.2.	Counterpart contact(s) for the review areas.....	6
2.2.3.	Selection of the size and membership of the review team.....	6
2.2.4.	Logistical matters.....	7
2.2.5.	Advance information and documentation package for the mission review team members	7
2.2.6.	Agenda and work plan	8
2.2.7.	Finalization of logistical matters.....	8
2.2.8.	Briefing of the review team	8
2.3.	CONDUCT OF THE MAIN INSARR MISSION	9
2.3.1.	INSARR activities	10
2.4.	REPORTING THE SAFETY REVIEW FINDINGS.....	12
2.4.1.	Technical notes	12
2.4.2.	Exit meeting.....	12
2.4.3.	Press release and press conference	13
2.4.4.	Mission report	13
2.5.	FOLLOW-UP INSARR MISSION	14
2.5.1.	Host organization responsibility	14
2.5.2.	Review team responsibility.....	15
3.	REVIEW AREAS AND GUIDELINES	15
3.1.	REVIEW AREAS.....	15
3.2.	DESIGN.....	16
3.2.1.	Objective.....	16
3.2.2.	Guidelines	16
3.3.	SAFETY ANALYSIS	22
3.3.1.	Objective.....	22
3.3.2.	Guidelines	23
3.4.	SAFETY ANALYSIS REPORT	24
3.4.1.	Objective.....	24
3.4.2.	Guidelines	25
3.5.	CONSTRUCTION	28
3.5.1.	Objective.....	28
3.5.2.	Guidelines	28
3.6.	COMMISSIONING.....	29
3.6.1.	Objective.....	29
3.6.2.	Guidelines	30

3.7.	SITING AND PROTECTION FROM EXTERNAL EVENTS.....	32
	3.7.1. Objective.....	32
	3.7.2. Guidelines.....	32
3.8.	OPERATIONAL LIMITS AND CONDITIONS.....	34
	3.8.1. Objective.....	34
	3.8.2. Guidelines.....	34
3.9.	SAFETY CULTURE.....	35
	3.9.1. Objective.....	35
	3.9.2. Guidelines.....	35
3.10.	REGULATORY SUPERVISION.....	37
	3.10.1. Objective.....	37
	3.10.2. Guidelines.....	37
3.11.	SAFETY COMMITTEE.....	39
	3.11.1. Objective.....	39
	3.11.2. Guidelines.....	39
3.12.	OPERATING ORGANIZATION AND REACTOR MANAGEMENT.....	41
	3.12.1. Objective.....	41
	3.12.2. Guidelines.....	41
3.13.	TRAINING AND QUALIFICATION.....	42
	3.13.1. Objective.....	42
	3.13.2. Guidelines.....	43
3.14.	CONDUCT OF OPERATIONS.....	44
	3.14.1. Objective.....	44
	3.14.2. Guidelines.....	44
3.15.	MAINTENANCE, PERIODIC TESTING AND INSPECTION.....	48
	3.15.1. Objective.....	48
	3.15.2. Guidelines.....	49
3.16.	MODIFICATIONS.....	51
	3.16.1. Objective.....	51
	3.16.2. Guidelines.....	52
3.17.	UTILIZATION AND EXPERIMENTS.....	53
	3.17.1. Objective.....	53
	3.17.2. Guidelines.....	53
3.18.	MANAGEMENT SYSTEM.....	55
	3.18.1. Objective.....	55
	3.18.2. Guidelines.....	55
3.19.	RADIATION PROTECTION.....	58
	3.19.1. Objective.....	58
	3.19.2. Guidelines.....	58
3.20.	RADIOACTIVE WASTE MANAGEMENT.....	60
	3.20.1. Objective.....	60
	3.20.2. Guidelines.....	60
3.21.	EMERGENCY PLANNING.....	62
	3.21.1. Objective.....	62
	3.21.2. Guidelines.....	63
3.22.	PLANNING FOR DECOMMISSIONING.....	65
	3.22.1. Objective.....	65
	3.22.2. Guidelines.....	65

APPENDIX I. TYPICAL MAIN MISSION REPORT FORMAT	67
APPENDIX II. CHECKLISTS FOR WALKDOWN	69
REFERENCES.....	73
CONTRIBUTORS TO DRAFTING AND REVIEW	77

1. INTRODUCTION

1.1. BACKGROUND

The IAEA performed its first research reactor evaluation on DIORIT (Switzerland) in October 1959 [1]. Six more evaluations were performed between 1960 and 1971. In 1972, the IAEA began to regularly review the safety of research reactors by means of safety review missions, in line with its statutory mandate and responsibilities. The IAEA conducted these safety review missions to Member States operating research reactors not subject to project and supply agreements, upon request from the relevant governments. According to the IAEA Statute, for missions to Member States with project and supply agreements, there is a requirement for the project to comply with relevant IAEA safety standards.

From 1972 to 1976, safety reviews to research reactors under project and supply agreements were referred to as “safety inspections”. During this period safety reviewers considered themselves IAEA safety inspectors and their main interest was to examine the legal framework and organization of the radiation protection and to examine the legal and operational radiological practices. Nuclear safety aspects such as safety analysis and operational procedures were gradually introduced into the scope of missions. From 1976 until 1987 the missions were no longer considered safety inspections but were referred to as “safety advisory missions” in various IAEA official documents. Mission objectives were mainly related to operational safety aspects and the scope included mainly nuclear safety related areas such as the safety analysis report (SAR), operational limits and conditions (OLCs), operating and maintenance procedures, reactor modifications and regulatory supervision. The reviews were conducted following a questionnaire based on the 1984 edition of IAEA Safety Series No. 35, Safe Operation of Research Reactors and Critical Assemblies¹.

In 1987, the IAEA announced the creation of a formal approach to providing a safety review service for research reactors². This approach was named the programme for Integrated Safety Assessment of Research Reactors (INSARR). INSARR missions are formally requested by a host organization in a Member State.

Since 1988, the objectives and scope of the safety reviews, formalized by the INSARR designation, have been expanded to cover design, commissioning, and siting and also to include emphasis on the exchange of information between reviewers and the host organizations. The missions are conducted following defined procedures (see Section 2). The format of the reports also became standardized, being changed only for differences in mission objectives and scope.

In 1997, the review procedures used for INSARR missions were formalized and documented in IAEA Services Series No. 1, Guidelines for the review of research reactor safety³. As the IAEA continued to issue safety standards within its safety series, establishing a consistent and comprehensive set of safety requirements and safety guides for research reactors, enhancements to the formal INSARR review procedure were made from about the mid-2000s and were

¹ INTERNATIONAL ATOMIC ENERGY AGENCY, Safe Operation of Research Reactors and Critical Assemblies, Safety Series No. 35, IAEA, Vienna (1984).

² In 1982, the IAEA had initiated a similar programme to INSARR, but to enhance the operational safety of nuclear power plants. This was the Operational Safety Review Team (OSART) mission programme.

³ INTERNATIONAL ATOMIC ENERGY AGENCY, Guidelines for the Review of Research Reactor Safety, IAEA Services Series No. 1, IAEA, Vienna (1997).

reflected in the format and content of the INSARR reports. Enhancements were mainly regarding mission preparation, conduct and follow-up. The current review procedure is presented in this publication. The IAEA research reactor safety standards collectively form the basis upon which current INSARR missions are performed; the key areas upon which the review guidelines are based are noted in Section 3.

Since 2000 a three-stage approach was applied for the INSARR, which comprises the following steps:

- (1) Pre-INSARR mission: to present the INSARR methodology, to discuss and define with the host organization the topics to be reviewed and the advance information and documentation package to be provided to the IAEA before the main mission, and to obtain preliminary information about the research reactor;
- (2) Main INSARR mission: to conduct the review and provide a report on the findings;
- (3) Follow-up INSARR mission: to determine the status of actions taken by the host organization in response to the main mission findings, to clarify any outstanding issues in response to mission findings and to obtain feedback on the effectiveness of INSARR.

The objective of INSARR missions is to conduct a comprehensive safety review of research reactor facilities and to assess conformance with the IAEA's safety standards. However, an important benefit from INSARR missions has also been the mutual transfer of knowledge and experience between mission experts and host organization personnel. Certain missions have identified areas where the host organization had developed particularly good approaches to certain safety topics, to the extent that the IAEA review team recognized these as good practices and recommended them for application at other facilities. Missions are not intended to be regulatory inspections but are based on a peer review approach.

This publication is a revision of IAEA Services Series No. 25, Guidelines for the Review of Research Reactor Safety: Revised Edition⁴. It presents the current INSARR organizational process and review guidelines, follows the structure of IAEA Safety Standards No. SSR-3, Safety of Research Reactors [2] and takes into account the provisions of the Code of Conduct on the Safety of Research Reactors [3]. The main review areas remain basically unchanged, but the references are revised to include the current IAEA safety standards.

1.2. OBJECTIVE

The objective of this publication is to provide guidance on the preparation, implementation, reporting and follow-up of INSARR missions at research reactors. The guidance on the preparation of the review mission is intended for both the host organization and the review team. The guidance on the conduct of the mission is mainly directed to the review team. In principle, the guidance in this report is to be used for safety reviews of research reactors conducted by the IAEA. However, the guidelines could also be used as one source of information for periodic safety reviews conducted by the operator of the research reactor or by the regulatory body.

⁴ INTERNATIONAL ATOMIC ENERGY AGENCY, Guidelines for the Review of Research Reactor Safety: Revised Edition, IAEA Services Series No. 25, IAEA, Vienna (2013).

1.3. SCOPE

The present publication is applicable for the INSARR missions requested by a host organization (government, regulatory body or operating organization) of a Member State. Also, with careful consideration, the guidance may be applied to the following types of safety reviews:

- (a) Peer reviews by research reactor operating personnel for other research reactors, or by other parts of the operating organization for their research reactors;
- (b) Self-review/audits by the research reactor management;
- (c) Self-assessment by the host organization;
- (d) Initial safety assessments of a research reactor project at the site evaluation or design stages;
- (e) Reassessments of a research reactor project in order to monitor adherence to good safety practices;
- (f) Long-term safety reviews when ageing of the research reactor facility may be of concern;
- (g) Regulatory inspections and audits.

In accordance with the IAEA publications on the safety of research reactors [2–13], the review guidelines in this publication are applicable to most types of research reactor. In the case of homogeneous reactors, their use may need to be adapted as appropriate for the given research reactor. For some specific aspects of research reactors with power above several tens of megawatts or non-water-cooled research reactors, some topics may need review of organizational and technical elements additional to those provided in the present publication. This publication does not cover the review of fast neutron research reactors or prototype power reactors.

1.4. STRUCTURE

The present publication consists of three sections and two Appendices. Section 2 presents guidance on the preparation, conduct and follow-up of an INSARR mission. Section 3 presents detailed guidelines on all review areas that may be covered in a comprehensive safety review. The guidance is provided in a structured way so that those developing a plan and programme for a specific review can cover selected areas to the necessary depth, in accordance with the scope and objectives of the review.

2. PREPARATION, CONDUCT, REPORTING AND FOLLOW UP OF AN INSARR

2.1. PREPARATION OF AN INSARR

Any research reactor safety review, whatever the defined scope and resources are, needs to be adequately prepared. The success of a specific assessment and the efficiency with which personnel, time and financial resources are used depend on the attention given to the preparation. Section 2 discusses full-scope INSARR missions. These missions represent the most comprehensive type of research reactor safety reviews/assessments performed. Other types of IAEA safety review missions include fact-finding missions, technical cooperation expert missions, and safety advisory/expert missions. These latter mission types utilize the elements of the INSARR guidelines but are generally more focused in scope and do not consist of the three-stage INSARR programme discussed below. Nevertheless, the INSARR

methodology is generally applicable and can be used as a reference for these other types of safety review missions.

A proposal for an INSARR may be initiated by the regulatory body or by the operating organization (e.g., the regulatory body of one Member State has a licensing requirement for an INSARR to be performed about every five years). In the case of safety review missions to research reactors under a project and supply agreement with the IAEA, the proposal may come either from the IAEA or from the Member State. The review team leader is an IAEA staff member with broad experience in all aspects of research reactors who has participated in other safety reviews. The preparation of the INSARR mission is done by the review team leader, with active participation of the counterpart organization.

Before embarking on the mission, funding arrangements should have been discussed and agreed upon between the requesting Member State and the IAEA. The funding arrangements can vary from the IAEA funding the host organization for most of the review, to the Member State providing the entire funding. In the latter case this should not affect the independence of the review. For some INSARR missions, financial obligations may be shared by the IAEA (bearing travel expenses and daily allowances for the review team members). The Member State counterpart is normally asked to provide local transportation for the review team members and in some cases may contribute towards local accommodation costs.

Since 2000, the INSARR programme has comprised three stages:

- (1) Pre-INSARR mission;
- (2) Main INSARR mission;
- (3) Follow-up INSARR mission.

2.2. PRE-INSARR MISSION

The implementation of an INSARR mission to review the safety of a particular research reactor starts with a pre-INSARR mission. This mission is typically of two to three days' duration, conducted by one to two IAEA staff members. The pre-INSARR meeting is normally held at the counterpart site to allow the participation of senior management and other stakeholder organizations. During the pre-INSARR mission all the details of the main mission are decided upon to ensure that the mission is carried out effectively, and to reach a common understanding on the conduct of the INSARR mission.

The IAEA team members discuss the main features of the INSARR mission, the counterpart's preparation for the review, including the availability of the necessary documentation and the review methods to be used. The pre-INSARR mission also includes a brief walkdown of the reactor facility with the IAEA review team members.

During the pre-INSARR mission the following items are agreed upon between the IAEA review team members and the counterparts:

- (a) Definition of objectives, scope (main review areas; see Section 3), strategy and review methods;
- (b) Type, format and contents of the advance information and documentation package;
- (c) Selection of the size and membership of the review team;
- (d) Logistical matters.

Subsequent to the pre-INSARR mission, but before the main mission, the review team leader and the counterpart communicate to arrange the following:

- (a) Finalization of the selection of review team members;
- (b) Advance information and documentation package for the mission review team members, including the counterpart contact(s) for the review areas;
- (c) Agenda and work plan;
- (d) Finalization of logistical matters;
- (e) Briefing of the review team.

Further information on the items in the two lists above is presented below.

2.2.1. Definition of objectives, scope, strategy and review methods

The objectives of an INSARR mission may be related to aspects of site selection and to the evaluation of design, construction, commissioning, operation, decommissioning or refurbishment of a research reactor. For any review, the objectives are to be clearly defined and agreed to by all parties (i.e., the IAEA and counterparts) concerned. In some cases, the Member State counterpart may set the general scope as well as the detailed mission objectives. Alternatively, a Member State may request the IAEA for an INSARR mission with a general objective, in relation to one or more of the aspects mentioned above, but request that the detailed review areas be developed by the review team leader, in agreement with the Member State counterpart. There is a wide variety of review objectives that may be chosen, but the overall objective of any review is to enhance the safety of the research reactor under review. The guidance provided in this publication has been developed to achieve this objective. It is not specifically directed towards enhancing the operability or the efficiency of a reactor, although this is frequently a by-product of the safety review mission. An INSARR mission itself is basically a peer review process and is not a regulatory inspection or audit. The review is based on the IAEA safety standards and takes into account international standards and best practices.

To ensure that the objectives of a specific mission are met, the precise scope of the review areas is also established. The mission scope defines the level of detail to which the objectives will be pursued. It is worthwhile to note that the scope of a specific mission is limited by the available time, personnel, and financial resources.

Various strategies are considered, to establish the most efficient and effective way of achieving the agreed objectives. In some cases, for example, where a comprehensive review of the safety of a facility is needed, a multidisciplinary review team examining the facility intensely over a relatively short period may be most efficient. If the aim of the mission is to focus on a chosen topic (e.g., an upgrade project for a primary coolant system) advice and direct interaction on a one-to-one basis over a prolonged period involving more than one mission may be appropriate.

The review methods are agreed upon. This involves a facility walkdown to cover all aspects of the mission scope, assessment of relevant documentation and discussions with research reactor staff. The review team should be able to observe or make their own radiation and contamination checks, with their own instrumentation, if the review team requests this. Special types of inspection (e.g., inspection of reactor vessel wall thickness, detailed corrosion inspection of equipment) or testing of equipment performance that necessitates special instrumentation or preparation is pre-defined in the mission scope.

2.2.2. Counterpart contact(s) for the review areas

The host organization provides to the review team leader a list of personnel to act as technical counterparts during the mission. Normally, there is one main counterpart who coordinates contact with other counterparts, depending upon the mission scope.

2.2.3. Selection of the size and membership of the review team

Proposals for the selection of the size and membership of the main INSARR mission review team are made during the pre-INSARR mission. The size of the review team depends on the type of research reactor (e.g., its power, design, and operational complexity, whether it is a unique or standard reactor type), on the experience of the reviewers and on the mission scope. Typically, the review team leader, deputy review team leader and another team member are from the IAEA staff, and there are between three and seven external experts. An administrative assistant from the IAEA may also be a team member, depending upon the scope and needs of the mission. Proposals/suggestions for the individual review team members may also be discussed at this time. The responsibility for determining the composition and size of the review team normally rests with the review team leader. The key characteristics needed for members of the review team are:

- (a) Expertise and experience in the topics to be reviewed;
- (b) Familiarity with the IAEA safety standards and other technical documents that form the basis of the review;
- (c) Familiarity with the type of research reactor to be reviewed;
- (d) Language capability.

In the selection process of the experts, special attention is paid to avoiding any potential conflict of interest with the Member States or with the host organization due to geopolitical or commercial considerations. Observers may be invited, with the mutual agreement of both parties.

The review team members are recruited taking into account their experience with similar facilities and with the topics that they will examine. Therefore, specific tasks within the review team are assigned in accordance with the specific competencies of the individual members. The ability of the review team members to prepare their contributions to the final mission report is also considered.

Other factors that may have to be taken into account in choosing the review team members depend upon the type of review, the scope and the objectives. If an INSARR mission is to undertake a comprehensive safety evaluation of a particular research reactor, consideration is given to ensuring that the review team provides a balanced international view of best practices. If the main aim of the review is to enhance safety awareness and influence attitudes within the host organization, greater consideration needs to be given to the ability of review team members to communicate effectively with a wide range of individuals within the host organization. Other considerations include the ability of review team members to support each other, such that a synergistic approach to the review is developed. As review team members learn from the experience obtained during a safety review, new review teams should include at least one member who has participated in a previous INSARR mission.

The final selection and confirmation of review team membership is made by the review team leader. The main mission is usually three to nine months after the pre-INSARR mission, so final

selection is made during this period, prior to the main INSARR mission. Proposed review team members are consulted by the review team leader before they are formally accepted by the Member State counterpart. The names and qualifications of the review team members are submitted to the counterpart for acceptance. Though the counterpart organization should not have any overriding influence on the choice of the review team members, there may be circumstances related to geopolitical considerations or potential conflict of interest that need to be carefully considered. Review team members with the same nationality as the counterpart Member State are normally excluded.

2.2.4. Logistical matters

The host organization for the INSARR mission confirms the date and duration of the mission in consultation with IAEA and usually asked to provide the necessary administrative resources (e.g., accommodation, transportation, office room(s) and office facilities, communication equipment, interpreter or translator (as necessary) and secretarial assistance). These resources need to be decided and agreed upon in advance, so that they are in place when the mission review team arrives to undertake the assessment.

Depending upon the duration and complexity of the mission, an administrative assistant may accompany the review team to facilitate preparation of reports, letters, and other documents. The organization hosting the INSARR mission also needs to make resources available to handle document translation and interpretation, if needed, and also to resolve logistic challenges that may arise before and during the mission. The involvement of media is not normally envisaged during the mission and review team members should not be involved with any public statements regarding the mission. Following the mission, there may be a press release issued, upon agreement with the Member State counterpart, and the IAEA Office of Public Information and Communication. The Member State may also issue its own press release following the mission.

2.2.5. Advance information and documentation package for the mission review team members

The advance information and documentation package (in English) needed for the main mission implementation is usually prepared by the host organization and sent, preferably in electronic format, to the review team leader. Its content is based on the objectives and scope of the mission.

The advance information and documentation package provides as much advance information as possible to facilitate adequate technical preparation for the mission review team members. The review team leader then coordinates the distribution of the advance information and documentation package prior to the main mission to the review team members. The review team leader may also send out the agenda and work plan, if finalized, at the same time, to the review team members. Since the team's preliminary review of the documentation provided can influence the working plan for the review, the advance information and documentation package should be received and distributed, if possible, about a month before the main mission to permit timely finalization of the agenda and working plan.

Regardless of the mission specific objectives, the advance information and documentation package from the host organization normally comprises, as a minimum:

- (a) General description of the main technical, nuclear, thermal-hydraulic, and operational characteristics of the research reactor,
- (b) Relevant safety analysis sections of the SAR,

- (c) Research reactor organizational chart, including functional responsibilities,
- (d) Mission-specific documentation.

During the main mission, the advance information and documentation package is supplemented with self-assessment results and other detailed information. Maintenance, test and operational records, operational procedures, operational flow sheets, drawings of buildings, systems and equipment, and electrical and instrumentation schematics are typical of this type of detailed information.

2.2.6. Agenda and work plan

The agenda and work plan describe the necessary tasks and the allocation of tasks to individual review team members. The agenda and work plan identify all tasks to be performed before, during and after the main mission. Each task is described in a manner such that each review team member who is assigned various tasks has a clear understanding of the specific objectives. The review team leader develops an initial plan and then members of review team, as well as the host organization, are consulted and involved in helping the review team leader finalize the agenda and work plan. Proper planning ensures that all tasks are executed according to procedures and to a time schedule that allows sufficient time for contingencies as a result of examination of particular topics, discussions with counterparts, review team meetings, preparation of an executive summary report and an exit meeting. The review team leader has the overall responsibility for fulfilling the objectives of the review and hence needs to ensure that the agenda and work plan are followed.

Regardless of the mission specific objectives, the agenda, provided by the review team leader, normally comprises, as a minimum:

- (a) Mission objectives, scope and expected output;
- (b) Work plan for the mission;
- (c) Timing of the assessment activities, including hold points, together with any interdependence;
- (d) References to relevant IAEA safety standards and other technical documents that will form the basis of the review;
- (e) Relevant information from previous missions, to be used as a reference.

The final agenda and work plan is sent by the review team leader to the host organization for approval, in advance of the main mission, so that the counterpart can commit to make the necessary staff and documentation and other logistical needs available at the appropriate time. The review team leader then ensures that the review team members are provided with the final agenda and work plan, prior to the main mission.

2.2.7. Finalization of logistical matters

Just prior to the main mission, the logistical matters (particularly for accommodation and provision of local transport for the team) are confirmed by the review team leader with the host organization.

2.2.8. Briefing of the review team

The review team leader is responsible for ensuring that the review team is adequately prepared. The scope of the preparation depends on the type of mission and the previous experience of review team members. In all cases, communication via email in the weeks prior to the main

mission ensures that all members of the review team (including any administrative support staff) are fully aware of the objectives and the scope of the mission and of the specific roles and responsibilities of each review team member. Any residual financial and other administrative arrangements and procedures are also finalized with review team members.

Prior to the main mission, the review team members need to have a clear understanding of the mission outputs and their format. There are three general outputs of the main mission:

- (a) Oral feedback provided by review team members via discussions with the counterparts throughout the mission;
- (b) An executive summary report, presented at the exit meeting, providing immediate advice to the counterpart organization, summarizing the main conclusions and recommendations of the review;
- (c) A final mission report (the main output) providing recommendations, suggestions, good practices, and comments for the host organization.

The review team leader is responsible for ensuring that review team members are informed of any changes to the final arrangements before they embark on travel.

2.3. CONDUCT OF THE MAIN INSARR MISSION

The main INSARR mission is conducted typically three to nine months after the pre-INSARR mission. The time gap between the missions may vary depending upon the proposed scope of the review and activities in the research reactor. The review team leader prepares the agenda and work plan, detailing the proposed work schedule, prior to the mission and forwards this to the review team members and the host organization.

Before starting the INSARR mission, the review team members normally arrive at the research reactor site at least one day before to receive a short briefing held by the review team leader. From the preliminary review of the advance information and documentation package, the review team members are expected to have some knowledge of the facility and the mission scope. Depending upon the extent and quality of the mission-specific documentation provided by the counterpart and the review preparation time, the review team is also expected to have some specific information on the condition of the research reactor.

The activities of the main mission start with an entry meeting which includes the introduction of review team members, observers (if any) and counterparts and a review of the agenda details. The INSARR review team acquires the mission information by review of documentation, interviews with research reactor staff, research reactor walkdown and direct observation of activities and status of systems and equipment. Throughout the mission, detailed discussions with the counterparts of the host organization are conducted to ensure an understanding of identified issues in order to formulate recommendations for improvement.

The INSARR review team meets each morning and evening to discuss the daily activities and to develop a consensus on emerging issues. The discussions of the review team help to ensure that all its members are well informed of the progress of the review and benefit from the observations of other experts. These meetings are also an opportunity for the review team leader to reinforce the review methodology. Those issues on which the review team has reached consensus are presented to the counterpart, preferably at the next morning meeting giving the research reactor representatives an opportunity to express their views regarding the issues.

As the assessment proceeds, each review team member drafts a summary on their review area and findings, which could be presented in the form of an Issue Page (see Appendix I), for the mission report and for possible inclusion in the executive summary report at the exit meeting. At the end of the assessment phase, a period of time is reserved for completing and presenting the executive summary report at the exit meeting and for rechecking any open topics, and for starting the preparation of the draft mission report. The exit meeting takes place in the morning of the last day of the mission. Upon completion of the exit meeting, a press conference may take place, if so requested and organized by the host organization. Final drafting and editing of the mission report is completed by the review team leader (with the assistance of the review team members when necessary) after the mission.

2.3.1. INSARR activities

The following activities are performed during a main INSARR mission:

- (a) Briefing of review team;
- (b) Entry meeting with counterpart and research reactor staff;
- (c) Examination and assessment of safety and technical documentation;
- (d) Research reactor walkdown;
- (e) Observation of operational activities;
- (f) Technical discussions with research reactor staff;
- (g) Exit meeting with the management of the host organization and the involved counterparts.

Information on these activities is provided below.

(a) Briefing of review team

As soon as possible upon arrival at the accommodation, the review team is briefed by the review team leader. This briefing basically reiterates the information forwarded after the pre-INSARR mission, with any recent updates and is mainly devoted to a detailed discussion of the programme, as documented in the agenda for the mission.

(b) Entry meeting with counterpart and the research reactor technical staff

Prior to starting the review, the counterpart, research reactor staff and others who may be involved in the mission activities, are adequately briefed. This briefing, which is normally the responsibility of the review team leader, takes the form of an entry meeting and addresses the following administrative items:

- (i) Introduction of the review team members to the counterpart;
- (ii) The needs of the review team, especially for technical documentation and staff communications;
- (iii) The method of communicating and reporting the results of the review;
- (iv) Discussion of the draft agenda.

At the entry meeting the draft agenda containing the detailed work schedule is discussed and finalized. If the preparation process has been timely and no unforeseen circumstances have occurred at the research reactor few changes are expected to the agenda. Minor changes during the week may be expected; these are incorporated into the final agenda, which is documented in the final mission report. The agenda and detailed work schedule are agreed upon during the entry meeting amongst the review team, counterpart, and research reactor staff. The agenda includes the following items:

- (i) Presentation by the counterpart on the general status of the research reactor, including planned activities and incidents;
- (ii) Daily meetings of the review team to discuss general progress, preliminary recommendations, suggestions and comments, and to revise the work schedule, if this becomes necessary;
- (iii) Special meetings to brief counterpart authorities on the preliminary review conclusions and recommendations;
- (iv) A final exit meeting where the review team presents to the management of the host organization and the involved counterparts a summary of the main conclusions and recommendations that are expected to be in the final report.

(c) Examination and assessment of safety and technical documentation

The examination and assessment of safety and operational documentation relevant to the objectives and scope of the mission is essential to the effectiveness of the safety review. Some review and assessment of the relevant document by the review team members should have preceded the main mission. This first review is usually limited to verifying or highlighting specific review items. However, in some missions, further documentation is often made available during the entry meeting or shortly after. Therefore, further review time during the main mission may be necessary to examine relevant information that was not previously provided. The review team leader should have reviewed previous IAEA mission reports, if any, and verified the implementation status of previous mission recommendations and suggestions.

(d) Research reactor walkdown

Direct observation of the reactor facilities via a walkdown to observe reactor structures, systems, and components (SSCs) is an important aspect of the review process. Observation of the reactor facilities is intended to allow the review team to obtain a general understanding of the reactor conditions and the visual status of safety related SSCs, as well as the general adherence to good housekeeping practices and industrial health and safety practices. As a part of the observations of the general condition of SSCs, the walkdown extends to support facilities and structures outside the reactor building such as emergency power supplies (usually diesels), back up cooling water supplies, and security access control features (which might influence emergency response capabilities). The walkdown outside the reactor building is also an opportunity to observe the potential for, and protection provided against, external hazards such as forest or bush fires, flooding, external traffic accidents, and large-scale storage of hazardous materials.

Appendix II provides an indicative checklist for a research reactor walkdown. The list is not intended to be used as a strict audit process, but rather as a guide to assist the reviewer during a walkdown.

(e) Observation of operational activities

Direct observation of operational and handling activities such as reactor startup, shutdown system operation, fuel handling, and tests such as responses of the reactor and equipment to anticipated operational occurrences is conducted.

The observation of operational activities may include checking the use of procedures and instructions, quality control practices, and operator responses. A better understanding of the operation and procedures is then obtained when direct interactions of the operating staff with equipment are observed. Because of the limited duration of a typical INSARR mission, the time

dedicated to observation of ongoing activities is rather short (typically one full working day or less). Observing activities may help the reviewers to obtain an understanding of the staff level of training, experience in the reactor systems and knowledge of the reactor procedures.

(f) Technical discussions with research reactor technical staff

Discussions with the management, operating staff and other reactor support staff provide important inputs for the INSARR mission. The discussions are used to:

- (i) Obtain additional information not covered by the available documentation;
- (ii) Obtain answers to questions and clarify issues that may have arisen from the documentation review, observations of operational activities and the research reactor walkdown;
- (iii) Identify needs for operating staff training.

Discussions with the research reactor staff are to be used to exchange information between reviewers and counterparts. In order to promote frank and open discussions, they are conducted in a cooperative manner and should not be perceived as an audit.

(g) Exit meeting with the management of the host organization and the involved counterparts.

The main findings and recommendations are provided to the host organization and involved counterparts at the exit meeting. This is discussed further in section 2.4.2.

2.4. REPORTING THE SAFETY REVIEW FINDINGS

The findings of the review team are communicated to those who have responsibilities for and influence on the safety of the research reactor. The extent and type of the final report depend upon the scope of the review and its objectives. However, all types of mission are expected to encompass the elements discussed below.

2.4.1. Technical notes

Team members should make notes on the main safety issues identified each day while conducting the INSARR mission. These notes form the basis for discussions during the daily meetings of the review team members, and for preparation of the mission report, which contains details on the facts, recommendations, suggestions, comments, and good practices observed by the review team members and recorded as issue pages (see Appendix I). The mission report contains sufficient information, with references, to ensure that individuals who were not involved in the safety review understand the issues covered.

2.4.2. Exit meeting

Before the review team leaves the research reactor, an exit meeting is always conducted where the review team leader orally conveys the main recommendations and areas of good practice identified to representatives of the operating organization and possibly to the regulatory body. The attendees at the exit meeting are determined by the counterpart organization. As a minimum, those with significant executive responsibilities for safety attend, including the reactor manager.

For the preparation of the exit meeting, a review team meeting is arranged where the results of the review team member's individual assessments are discussed and consolidated.

An executive summary report is provided to the operating organization during the exit meeting. The executive summary report is considered as a preliminary compilation of the review team's main conclusions, recommendations, and suggestions, including good practices that have been identified during the mission.

The exit meeting is conducted in an open and constructive manner. During the exit meeting, the commitment and follow-up actions necessary to enhance safety may be discussed, but it is up to the host organization to decide on its response to the formal mission report when it is issued. There may be minor technical items that still remain to be clarified by research reactor staff. Items such as these are normally expected to be communicated to the review team leader as soon as possible after the mission, for inclusion in the final mission report.

2.4.3. Press release and press conference

The IAEA issues a press release after the INSARR mission. The IAEA press release is prepared by the IAEA Office of Public Information and Communication in cooperation with the team leader of the mission. It is based on the executive summary that is provided to the IAEA Press Officer as early as possible, but at least one day before the exit meeting. The drafting of the press release is coordinated with the host organization so that it has an opportunity to comment on the draft, although the IAEA maintains the sole ownership of the press release. The press release is disseminated by the IAEA Press Officer and published on the IAEA website after the exit meeting.

Depending on the host organization's media communications strategy, a joint press conference may be organized after the exit meeting. Normally, the joint press conference involves a senior representative of the host organization, the media coordinator and the INSARR mission team leader. The decision whether to hold a joint press conference is made in advance of the main mission, usually during the preparatory meeting.

2.4.4. Mission report

Following the main mission, a final mission report is prepared on the basis of the executive summary, in order to provide a permanent record of the review. The final INSARR mission report is normally prepared under the responsibility of the review team leader, who coordinates the production of this report with the review team members. The counterpart is provided with a draft copy of the final mission report for comments on the findings and checking of the technical correctness of the information.

The format of the final mission report is provided in Appendix I. The main sections are self-explanatory. This report contains details on facts and findings of the review team members, along with recommendations, suggestions, comments and identified good practices, which are defined as follows:

- (a) Recommendations: A recommendation is a proposal intended to strengthen safety to be in line with the IAEA Fundamental Safety Principles or Safety Requirements. It is based on inadequate conformance with the IAEA safety standards and addresses the general concern rather than the symptoms of the identified concern. Recommendations are specific, realistic, and designed to result in tangible improvements. Recommendations are designated in the mission report with an 'R' followed by a number.

- (b) **Suggestions:** A suggestion is a proposal for changes to arrangements to better align with the IAEA Safety Requirements or Safety Guides. It offers advice on an opportunity for a safety improvement not directly related to inadequate conformance with the IAEA safety standards. It is primarily intended to make performance more effective, to indicate useful expansions to existing programmes and to point out possible superior alternatives to ongoing work. Suggestions are designated in the mission report with an ‘S’ followed by a number.
- (c) **Comments:** These are proposals for the implementation of the recommendations or suggestions, but do not constitute review team advice. Comments are designated in the mission report with a ‘C’ followed by a number.
- (d) **Good Practices:** These are proven performances, activities, or uses of equipment, which the review team considers to be markedly superior to that observed elsewhere. Good practices are designated in the mission report with ‘GP’ followed by a number.

The report conclusions show to what extent the objectives of the mission were achieved and, as such, provide a starting point for plans for future reviews. A general statement regarding the overall safety of the research reactor could also be included in the conclusions.

The INSARR mission report is submitted by the IAEA through the official channels to the Member State concerned. This report is designated as a restricted⁵ distribution IAEA document, not to be released to the public. However, the operating organization of the research reactor may wish to use the document in an unrestricted manner. Some INSARR mission reports have been posted by the counterpart organization on the web, allowing public access. The operating organization may, in certain cases, also request the IAEA to remove the restricted designation, after the restricted version has been issued.

2.5. FOLLOW-UP INSARR MISSION

The purpose of the follow-up mission is to determine the status of implementation of the recommendations and suggestions presented in the final INSARR mission report and to form judgments on whether adequate actions have been taken by the research reactor operating organization. The follow-up INSARR mission is typically conducted one to two years after the main mission, depending on the number of recommendations and the urgency of their implementation. The responsibility for responding to the formal conclusions and recommendations of the review team rests with the operating organization. .

The follow-up mission is organized following the same procedures as for the main mission, but with a reduced scope to examine the areas object of recommendations and suggestions. The follow-up mission team comprises at least one member of Agency staff and one of the experts who participated in the main mission.

2.5.1. Host organization responsibility

The host organization has prime responsibility for considering and implementing the recommendations provided during the main mission, in advance of the request for a follow-up

⁵ In the case of other IAEA safety review services such as OSART and Integrated Regulatory Review Service (IRRS), the reports are automatically derestricted 90 days after their issuance by the IAEA unless the host Member States specifically request that the reports remain restricted.

mission. In many circumstances, the host organization prepares a formal response to the recommendations and suggestions from the main mission report.

Where the list of recommendations is relatively small, the operating organization usually keeps a list of action items updated to facilitate follow-up. If the list of recommendations is extensive, then the operating organization needs to develop a more detailed action plan to facilitate follow-up. Reports stating the progress in the implementation of the recommendations need to be prepared for the IAEA, prior to the follow-up mission.

2.5.2. Review team responsibility

Individual review team members have no formal responsibility to ensure that INSARR mission recommendations are followed up. However, the follow-up mission is used to determine the status of implementation of all recommendations and suggestions.

If a significant number of issues show insufficient progress, the review team leader may propose to the host organization that an invitation be issued for another follow-up mission. In addition, a support mission or workshop can be offered to the host organization.

The review team leader may disseminate non-restricted information to other Member States with similar facilities, so that such information leads to a broader enhancement of safety.

3. REVIEW AREAS AND GUIDELINES

3.1. REVIEW AREAS

This section provides a comprehensive list of review areas and associated detailed guidelines for an INSARR mission. Not all of these topics are covered in a typical INSARR mission. The coverage depends on the scope and objectives of the review, as determined by the counterpart in conjunction with the IAEA review team leader.

The comprehensive list of review topics, from which any individual INSARR scope can be formulated, is based upon the safety requirements established in SSR-3 [2] and the recommendations provided in IAEA Safety Standards Series No. SSG-20 (Rev. 1), Safety Assessment for Research Reactors and Preparation of the Safety Analysis Report [5]. The Code of Conduct on the Safety of Research Reactors [3], which contains provisions on best practices to achieve a high safety level, is an important basis for the definition of review areas.

The IAEA safety standards relevant to research reactors collectively form the basis upon which INSARR missions are performed and the reference for the review areas. Other IAEA publications, such as safety reports, provide detailed information and examples useful for the review.

The review areas that are covered by a full-scope INSARR mission include the following:

- (a) Design;
- (b) Safety analysis;
- (c) SAR;
- (d) Construction;
- (e) Commissioning;

- (f) Siting and protection against external events;
- (g) OLCs;
- (h) Safety culture;
- (i) Regulatory supervision;
- (j) Safety committees;
- (k) Operating organization and reactor management;
- (l) Training and qualifications;
- (m) Conduct of operations;
- (n) Maintenance and periodic testing;
- (o) Modifications;
- (p) Utilization and experiments;
- (q) Management system;
- (r) Radiation protection;
- (s) Radioactive waste management;
- (t) Emergency planning;
- (u) Planning for decommissioning.

Considerations of ageing management and the interface between safety and security are addressed in various review areas, as appropriate. The review of research reactor nuclear security aspects is covered by another IAEA review service. However, the interface between safety and security could be covered during the INSARR missions. Safety culture and emergency preparedness may also be addressed in depth by separate IAEA review services. A detailed review of ageing management and continued safe operation of research reactors may also be covered by a separate IAEA mission.

Review team members cover their assigned individual review areas to the extent necessary to be able to make well informed judgments. Not all the issues covered by the guidelines for a given review area need to be addressed during a INSARR mission. It is the responsibility of the reviewer to make an appropriate selection of subjects for questioning in accordance with the objectives, scope and duration of the review. This selection should be appropriate to identify strengths and weaknesses, to draw conclusions, to make recommendations on research reactor safety, and to address fully such issues in the mission report.

Guidelines on each of the above-mentioned review areas are presented in the following subsections.

3.2. DESIGN

3.2.1. Objective

The objective is to provide a basis for evaluation of the safety aspects of the design of research reactors. The reviewer verifies that the general requirements established in SSR-3 [2] have been met.

References

SSR-3 [2]: Requirement 5 and Section 6; SSGs [4–15].

3.2.2. Guidelines

- (1) Examine the overall design safety objective of the research reactor to verify the following:
 - (a) Radiation doses to individuals are in conformance with the principle of optimization of protection and safety;

- (b) Radiation doses to individuals do not exceed the limits prescribed by national authorities;
 - (c) The collective radiation dose commitment to individuals on and off the site is in conformance with the principle of optimization of protection and safety;
 - (d) The environmental impact is minimal and within the limits prescribed by national authorities;
 - (e) Generated radioactive waste, including radioactive effluents, is in conformance with the principle of optimization of protection and safety.
- (2) Verify that the reactor design is based on defence in depth concepts wherein multiple levels of protection are provided, ensuring that:
- (a) All reasonably practical design provisions have been made to prevent accidents;
 - (b) All reasonably practical design provisions have been made to mitigate the consequences of accidents for individuals on and off site and for the environment.
- (3) Verify that the reactor design takes into account the following:
- (a) All items important to safety are identified and safety classified;
 - (b) The design basis for all items important to safety is specified for all facility states, including acceptance criteria;
 - (c) The interaction between different systems ensure that the failure of one system does not affect the functioning and reliability of the other systems, and more stringent design criteria are followed where different systems are physically interconnected;
 - (d) The response of inherent and engineered features to deviations from normal operation minimize or exclude the actuation of safety systems;
- (4) Verify that the design of the reactor and experimental facilities include:
- (a) A set of design limits for all items important to safety for all operational states and accident conditions;
 - (b) Derivation of a set of design extension conditions and associated practicable safety features;
 - (c) Qualification of items important to safety for all service conditions that could be reasonably anticipated;
 - (d) Features necessary to facilitate commissioning including identification of temporary provisions for transition from commissioning to operation;
 - (e) Provisions for calibration, testing, maintenance, repair, replacement, inspection and monitoring of items important to safety to ensure their functional capabilities as specified in their design basis;
 - (f) Provisions for emergency preparedness and response including effective communications throughout the research reactor;
 - (g) Provisions for safe utilization and modification of the research reactor;
 - (h) Ageing management for items important to safety including design life determination, margins to deal with various degradation mechanisms and consequent ageing effects, and obsolescence;
 - (i) Provisions to ensure safety in long shutdown periods, such as preservation of SSCs, fuel safety, neutron poisoning of the reflector, and ageing management including maintenance, periodic testing and inspection;
 - (j) Provisions to eventually safely decommission the research reactor and experimental facilities.

- (5) Verify that the analysis of response of the reactor and associated systems considers all credible postulated initiating events including internal hazards and external events, both natural and human induced.
- (6) Verify that multiple means for ensuring each of the following main safety functions are provided:
 - (a) The capability to control reactivity, including to shut down the reactor and maintain it in a safe shutdown condition for all facility states (operational states and accident conditions);
 - (b) Adequate removal of heat from the core and from the fuel storage for all facility states;
 - (c) Confinement or containment of radioactive material to prevent or minimize its release to the environment.
- (7) Determine that conservative design margins were adopted for all operating parameters, including OLCs.
- (8) Verify that equipment and procedures are in place to:
 - (a) Prohibit deviations from normal operation;
 - (b) Prevent anticipated operational occurrences from progressing into accident conditions;
 - (c) Control and mitigate accidents;
 - (d) Manage the consequences of design extension conditions.
- (9) Verify that necessary facilities for emergency preparedness and response aimed at mitigating the effects of the release of radioactive effluents to the environment during on-site and off-site emergencies are in place.
- (10) Verify that applicable safety related systems fulfil the following criteria:
 - (a) Redundancy to improve reliability and to meet the single failure criterion;
 - (b) Diversity to enhance reliability and reduce the potential for common cause failures;
 - (c) Independence (or physical separation) to enhance reliability and minimize common cause failures;
 - (d) Fail-safe design to ensure that the failure of items important to safety does not prevent the performance of safety function;
 - (e) Testability to permit inspection, servicing and testing at prescribed intervals over the lifetime of the research reactor, and access to the SSCs for maintainability and replacement.
- (11) Verify that no single failure of a component can result in the loss of capability of a safety system to perform its safety function.
- (12) Verify that the reactor protection system is capable of automatically initiating protective action for all postulated initiating events to safely achieve and maintain a safe state.
- (13) Examine the design of the buildings and structures important to safety to verify that they are designed:
 - (a) For all operational states, for design basis accidents and, to the extent practicable, for design extension conditions;

- (b) To withstand loading arising from all postulated internal and external events;
- (c) In a manner to keep radiation levels and releases on and off the site during all operational states within prescribed limits;
- (d) For a degree of leak tightness consistent with reactor safety analysis;
- (e) With ventilation and air flow rates consistent with reactor safety analysis.

(14) Examine the design of the reactor core to verify that:

- (a) Individual fuel elements are consistent with the neutronic, thermal-hydraulic, mechanical, material, chemical, and irradiation constraints of the core as a whole;
- (b) Fuel damage is kept within acceptable limits during accident conditions;
- (c) The maximum permissible design limits specified for all operational states are not exceeded;
- (d) The reactor can be shut down for all operational states and accident conditions and be maintained in a subcritical state;
- (e) Criticality cannot be reached for any of the core configurations for subcritical assemblies.

(15) Examine the design of the reactivity control system to verify that:

- (a) Sufficient negative reactivity is provided to make the reactor subcritical, and to maintain it subcritical in all operational states, taking into account the core condition with the highest positive reactivity contribution;
- (b) The maximum positive reactivity addition rate is limited to the values justified in the reactor safety analysis.

(16) Examine the thermal-hydraulic design to verify that:

- (a) Reactor fuel parameters are maintained below the specified safety limits during all operational states;
- (b) Safety margins, including margins for error and engineering tolerances for the safety limits, are consistent with the reactor safety analysis.

(17) Examine the design of the reactor coolant system to verify that:

- (a) Adequate, reliable, long term core cooling is provided for all operational states;
- (b) Provisions for testing, surveillance and inspection are provided;
- (c) A reliable system is available for shutdown cooling, where required;
- (d) In reactor coolant systems with penetrations at or below core level, redundant, testable features have been considered to prevent core uncovering;
- (e) Where primary coolant drainage and subsequent core uncovering could occur, suitable arrangements such as syphon breaks are provided.

(18) Examine the reactor shutdown system design to verify that:

- (a) At least one fast acting, automatic shutdown system is incorporated;
- (b) Sufficient shutdown reactivity is provided to safely shut down the reactor and maintain it subcritical under all operational states and accident conditions;
- (c) The speed of action and shutdown margin provided are consistent with the reactor safety analysis assumptions;
- (d) A single failure will not prevent the system from completing its safety function.

(19) Examine the design of the reactor protection system to verify that:

- (a) It is automatic and independent of other systems;
- (b) Automatic protective actions, once initiated, proceed to completion and cannot be prevented or impaired by manual actions;
- (c) Redundancy and diversity are utilized to the extent that each postulated initiating event can be detected and protective action can be initiated in at least two different ways;
- (d) No single failure or credible common cause failures will prevent the safe shutdown of the reactor;
- (e) Fail-safe action is provided for ensuring the safe condition of the reactor in the event of a protection system failure;
- (f) All components can be functionally tested;
- (g) Protection system settings are established with an adequate margin between the initiation point and a safety limit to permit the protective action to correct the situation;
- (h) Protective interlocks and trips cannot be overridden;
- (i) Manual reactor trip initiation is provided;
- (j) High reliability can be demonstrated for the computer-based systems or diverse means are provided for fulfilling the protection function.

(20) Examine the design of the emergency core cooling system to verify that:

- (a) The capability exists to maintain core temperatures within the limits specified in the reactor safety analysis during all shutdown conditions;
- (b) Significant fuel failure is prevented for the range of loss of coolant accidents specified in the design and demonstrated in the reactor safety analysis;
- (c) A single failure in the system will not prevent the system from fulfilling its intended function;
- (d) Provisions for testing, surveillance and inspection are provided.

(21) Examine the design of the confinement or containment to verify that:

- (a) It is capable of withstanding loading from accident events including those arising from all postulated internal and external events;
- (b) Adequate margins are provided for the highest calculated pressure and temperature loads expected during design basis accident conditions;
- (c) Suitable means are provided to control the release of radioactive material during accident conditions;
- (d) The degree of leak tightness is commensurate with the findings of the reactor safety analysis;
- (e) Provisions are made for initial and periodic leak tests, routine testing, and ventilation filter efficiency testing and replacement.

(22) Examine the design of the instrumentation and control to verify that:

- (a) The arrangement of instrumentation and displays utilizes ergonomic principles and provides optimal conditions for assimilation of information;
- (b) Sufficient instrumentation is provided to monitor reactor systems and reactor core parameters during all operating, shutdown, refuelling and maintenance conditions, and for recording all safety related variables, including the position of the neutron source;

- (c) Adequate instrumentation is provided to monitor emergency conditions;
 - (d) Sufficient indicating and recording instrumentation is provided to monitor important reactor parameters following anticipated operational occurrences and accident conditions;
 - (e) Provision is made for startup neutron source and startup instrumentation, including for startup after long shutdown periods;
 - (f) Audible and visual alarms are installed to provide indication of unacceptable deviations in operating parameters and to indicate when trip points are reached;
 - (g) Provisions for periodic testing, inspection and maintenance are in place.
- (23) Examine the design of the electrical power supply system to verify that:
- (a) It conforms to the applicable requirements and is consistent with the reactor safety analysis;
 - (b) An emergency electrical power supply is provided when needed for systems that are essential to safe shutdown, cooling of the reactor and confinement of radioactive material;
 - (c) The maximum period for interruption of all electrical power specified in the reactor safety analysis is met;
 - (d) Provisions for periodic, functional testing of the emergency electrical power supply are in place when that system is required.
- (24) Examine the design of the research reactor supporting systems and auxiliary systems to verify that:
- (a) The systems act to prevent the uncontrolled release of radioactive material;
 - (b) The systems act to prevent the malfunction of items important to safety;
 - (c) Fuel handling and storage systems ensure provisions for sufficient fresh and irradiated fuel storage, subcriticality, adequate cooling, minimum corrosion, adequate containment, adequate radiation shielding, adequate ventilation, inspection, and tracking of individual fuel assemblies;
 - (d) Adequate ventilation, filtration and radiological monitoring are provided, particularly in areas that could experience airborne radioactivity;
 - (e) Provisions are made to prevent fires and explosions (e.g. use of fire-retardant material), to detect fires and to mitigate their consequences;
 - (f) Adequate lighting, including emergency lighting, is provided in all operational areas of the research reactor;
 - (g) Equipment is provided for safe lifting, lowering and movement of material in the research reactor;
 - (h) The quality of the compressed air supply to items important to safety is specified and monitored;
 - (i) Adequate communications and alarm systems are provided to ensure reactor safety and the safety of experimental facilities.
- (25) Examine the design of the main control room and verify that an independent and separate supplementary control room or remote shutdown panel are provided.
- (26) Examine the design of computer-based items important to safety to verify that the hardware and software are subject to adequate quality assurance, verification, validation and testing as per the management system.

(27) Examine the design of experimental devices to verify that:

- (a) In all operational states, the devices do not pose an unacceptable hazard to the reactor, other experiments, on-site personnel, the public or the environment;
- (b) Neither operation nor failure results in an unacceptable reactivity change in the reactor;
- (c) Provisions for monitoring parameters of experiments in the control room, safety limits and environmental conditions, as appropriate, are provided for experiments and experimental facilities;
- (d) Experimental devices are classified on the basis of their importance to safety and the interaction between experimental devices and the reactor are covered in the safety analysis.

(28) Examine the design of the radioactive waste systems to verify that:

- (a) Releases of radioactive material can be controlled, minimized, monitored, and maintained below limits set by the national authority;
- (b) Techniques such as shielding and decay are utilized to reduce doses to personnel and releases to the environment;
- (c) Adequate on-site storage and recovery facilities are provided;
- (d) Adequate means are provided to control, sample and monitor airborne and liquid effluent discharges to the environment.

(29) Based on consistent overall radiation protection concepts, examine the design of radiation protection systems to verify that:

- (a) Shielding, ventilation, filtration and decay systems are in place;
- (b) Adequate radiation area and airborne radioactivity monitoring systems are in place to monitor operational states and accident conditions;
- (c) Structural materials have been chosen to minimize doses to personnel and damage to equipment during operation, inspection, maintenance and repair of the reactor;
- (d) The effects of radionuclides such as nitrogen-16, tritium, and argon-41 produced by neutron activation in reactor process systems have been given due consideration in providing radiation protection;
- (e) Hot water layer is effective and operable (if applicable);
- (f) Access control is provided for areas that could have radiation levels in excess of those in normal operating areas.

3.3. SAFETY ANALYSIS

3.3.1. Objective

To provide a basis for evaluating the safety analysis in accordance with SSG-20 (Rev. 1)[5] for ensuring that the safety of the reactor has been analysed and evaluated to demonstrate that it is adequate.

References

SSR-3 [2]: Requirements 5, 10, 18, 19, 21 and 41, and Appendix I; SSG-20 (Rev. 1) [5]; Safety Reports Series No. 55 [16].

3.3.2. Guidelines

- (1) Determine whether the safety analysis provides:
 - (a) Analyses of the response of the reactor to a range of postulated initiating events that could lead either to anticipated operational occurrences or to accident conditions;
 - (b) Due consideration of defence in depth and uncertainties;
 - (c) Consideration of experimental devices with respect to both their own safety aspects and their effect on the reactor safety.
- (2) Verify that the safety analysis has been used as a basis for:
 - (a) The design of items important to safety;
 - (b) Establishment of the OLCs of the reactor;
 - (c) Development of operating procedures;
 - (d) Periodic testing and inspection programmes;
 - (e) Record keeping;
 - (f) The maintenance programme;
 - (g) Ageing management;
 - (h) Modification proposals;
 - (i) Training of operating personnel;
 - (j) Emergency preparedness and response.
- (3) Verify that the postulated initiating events and the bounding design basis accident for the reactor have been properly identified.
- (4) Verify that the results of the safety analysis are reflected in the SAR.
- (5) Determine that, as a minimum, the set of postulated initiating events for the safety analysis includes:
 - (a) Loss of electrical power supplies;
 - (b) Insertion of excess reactivity;
 - (c) Loss of flow;
 - (d) Loss of coolant;
 - (e) Erroneous handling or failure of equipment or components;
 - (f) Special internal events such as fires, explosions, flooding and security related events;
 - (g) External events such as earthquakes, weather emergencies, floods, fires and aircraft crashes;
 - (h) Human error.
- (6) Verify that the set of postulated initiating events covers all credible accidents that influence the safety of the reactor.
- (7) Verify that the deterministic safety analysis and complimentary probabilistic safety assessment (as appropriate, if conducted) have been validated by independent verification by individuals or groups independent from those who performed the work, and that the results have been used to assess the adequacy of the design.
- (8) Determine for each postulated initiating event that the following information has been considered in the evaluation:

- (a) Input parameters, initial conditions, boundary conditions, assumptions, models and codes used and their verification and validation;
 - (b) Sequence of events and performance of reactor systems;
 - (c) Sensitivity to single failure modes and common cause failures;
 - (d) Sensitivity to human factors;
 - (e) Potential for fission product releases and radiation exposures;
 - (f) The extent to which the safety systems and any non-failed process systems need to function under accident conditions.
- (9) Determine that design limits for all relevant parameters have been specified for each operational state of the reactor and for accident conditions.
- (10) Verify that the safety analysis includes:
- (a) Characterization of the appropriate postulated initiating events;
 - (b) Analysis of event sequences and evaluation of the consequences of the postulated initiating events;
 - (c) Comparison of the results of the analysis with technical and radiological acceptance criteria and design limits;
 - (d) Analysis of safety systems and the engineered safety features;
 - (e) Demonstration that the management of anticipated operational occurrences and design basis accidents is possible by means of an automatic response of safety systems in combination with prescribed operator actions;
 - (f) Analysis of the design extension conditions and specifications of and practicable provision for safety features for preventing such conditions from arising or mitigating their consequences if they do arise.
- (11) Verify if the periodic safety review is performed and the safety analysis is updated and covers as a minimum the following:
- (a) Validity of the SAR and other documents in view of current regulatory requirements and the status of the research reactor;
 - (b) Changes in site characteristics;
 - (c) Changes in the utilization programme;
 - (d) Cumulative effects of ageing and modifications;
 - (e) Changes to procedures;
 - (f) Use of feedback from operating experience;
 - (g) Developments in the research reactor technology;
 - (h) Compliance of SSCs and software with the design requirements.

3.4. SAFETY ANALYSIS REPORT

3.4.1. Objective

To provide a basis for determining the adequacy of the SAR to meet its purpose. General requirements are provided in SSR-3 [2] and detailed guidance on the SAR content are presented in SSG-20 (Rev. 1) [5]. The review team assesses the consistency of the methods used in the SAR with the methods and practices described in SSG-20 (Rev. 1) [5].

References

SSR-3 [2]: Requirement 1; SSG-20 (Rev. 1) [5].

3.4.2. Guidelines

- (1) Determine whether the content of the SAR covers the following:
 - (a) Introduction and general description of the reactor;
 - (b) Safety objectives and engineering design requirements;
 - (c) Site characteristics;
 - (d) Buildings and structures;
 - (e) The reactor and its design;
 - (f) Cooling systems and connected systems;
 - (g) Engineered safety features;
 - (h) Instrumentation and control systems;
 - (i) Electric power systems;
 - (j) Auxiliary systems;
 - (k) Reactor utilization;
 - (l) Operational radiation safety;
 - (m) Conduct of operations;
 - (n) Environmental assessment;
 - (o) Commissioning;
 - (p) Safety analysis;
 - (q) OLCs;
 - (r) Integrated management system;
 - (s) Decommissioning;
 - (t) Emergency preparedness and response.
- (2) Determine whether the SAR reflects the current status and configuration of the facility.
- (3) Determine whether the SAR provides:
 - (a) A basis for the OLCs;
 - (b) A basis for establishing the management system, and for preparing operating procedures and emergency plans;
 - (c) Sufficient information to allow the regulatory body to make an independent assessment of the safety of the reactor as a basis for the licensing and approval of reactor operation;
 - (d) Sufficient information for operators and experimenters to understand the facility.
- (4) Determine whether the SAR gives:
 - (a) A detailed description of the reactor site;
 - (b) Safety principles and general design criteria to protect the reactor, operating personnel, the public, and the environment;
 - (c) An analysis of the potential risks to the reactor, provisions to mitigate against them, and the impact of any reactor incident;
 - (d) An analysis of accident scenarios and incorporated safety features to avoid them or mitigate their consequences, including those for design extension conditions.
- (5) Determine whether a new or amended SAR was needed and has been completed for the following situations:

- (a) New construction;
- (b) Relicensing;
- (c) Refurbishment;
- (d) Significant modifications;
- (e) Power level increase;
- (f) Core fuel conversion;
- (g) Changes in OLCs;
- (h) Equipment upgrade, replacement or repair;
- (i) Changes in site characteristics;
- (j) Experiments with significant relevance to reactor safety;
- (k) The advent of modern safety standards, new regulatory requirements, or international codes of practice.

(6) Ensure that the SAR presents:

- (a) A general description of the facility including, as appropriate, a historical review of upgrades and modifications, comparison with similar facilities, safety features, the experimental programme, management organization, and facility drawings;
- (b) Safety principles and general design criteria including, as appropriate, overall safety objectives, safety principles important to design, design criteria applied to safety related systems, classification of SSCs, external events, codes and standards, design methods, qualification of structures, components and equipment, and design for internal hazards (e.g. fire, flooding);
- (c) Site characteristics including, as appropriate, a general site description, natural external events, geology, seismology, meteorology, hydrology, oceanography, nearby industrial, transportation, and military facilities, population distribution, natural environment, land, and water usage, baseline radiological levels, atmospheric dispersion of radioactive material, dispersion of radioactive material through surface water and groundwater, and mitigation measures for postulated accidents;
- (d) Building and structures including, as appropriate, reactor building design features, drawings, tests and inspections, and auxiliary structure descriptions that are important to safety;
- (e) Reactor information including, as appropriate, a summary description of the reactor, fuel element design, fuel properties, lifetime operating characteristics, reactivity control system, core nuclear design, thermal hydraulic design, and materials utilized in the core structure, fuel and absorbers;
- (f) Reactor coolant systems and connected systems including, as appropriate, a summary description supported by drawings and elevation diagrams of the primary coolant system, the secondary coolant system, the moderator system, the emergency core cooling system, the decay heat removal system, the primary purification system, and the primary make-up system;
- (g) Engineered safety features including, as appropriate, types, locations and brief descriptions of the engineered safety features with information on such aspects as reliability, redundancy, diversity, and the ability of materials to withstand accident conditions and, to the extent practicable, design extension conditions;
- (h) Instrumentation and control including, as appropriate, the power regulating system, the reactor protection system, alarm systems, interlocks, other instrumentation systems required for safety, and the control room layout and ergonomic assessment;
- (i) Electric power systems including, as appropriate, normal AC power supply, uninterruptable DC/AC power supply, cables and routing;

- (j) Means of containment or confinement, including methods and characteristics of normal and emergency ventilation;
- (k) Auxiliary systems including, as appropriate, fuel storage and handling, water systems, process auxiliaries such as compressed air, sample handling, air conditioning, heating, cooling, ventilation, and fire protection;
- (l) Reactor utilization including, as appropriate, experimental facilities, irradiation facilities, design safety principles, safety analyses, method of review and approval for new experimental devices, and materials forbidden in experiments;
- (m) Operational radiological safety including, as appropriate, radiation protection policy of the organization, overall radiation protection programme, quantitative account of sources of radiation at the facility, facility design for radiological safety, handling and movement of radioactive material, dose assessment for normal operation, procedures and training, equipment and instrumentation, environmental monitoring, access control and zoning, shielding, ventilation for radiological control, area and effluent radiation monitoring, solid, liquid and gaseous waste, and anticipated direct radiation exposures within the facility;
- (n) Conduct of operations including, as appropriate, organizational structure, staff selection, training and qualification, review and audit functions, operating procedures, maintenance, testing and inspection programmes including ageing management, records and reports, and fire protection procedures;
- (o) Environmental assessment to the extent required by the national regulatory body and that may include analysis of gaseous and aqueous release for all operational states and accident conditions, effects of radionuclide uptake in plant and animal life, ultimate heat sink effects, transport and disposal of spent fuel and radioactive waste, and both the positive and negative socioeconomic impacts of the facility on the neighbouring population;
- (p) Commissioning programme including, as appropriate, summary, details of the commissioning organization, management system, stages for commissioning, OLCs, equipment installation and test procedures, test schedules and sequence of commissioning systems and equipment, and summary of commissioning results;
- (q) Safety analysis summary and details as appropriate (also see 3.3);
- (r) OLCs as determined in the safety analysis including:
 - (i) Safety limits on important process variables which, if exceeded, could result in undue exposure to personnel or release of undue amounts of radioactivity to the environment;
 - (ii) Safety system settings for those variables and parameters that, if not controlled, could result in safety limits being exceeded;
 - (iii) Establishment of limiting conditions for safe operation to provide acceptable margins between normal operating values and safety system settings;
 - (iv) Surveillance requirements that prescribe the frequency and scope of tests to demonstrate that performance levels are being met;
 - (v) Administrative requirements including organizational structure and responsibility, staffing requirements, review and audit of facility operations, required procedures, operational event review requirements, and reports and record requirements;
- (s) A management system including, as appropriate, provision for control of all activities associated with the facility such as design, procurement, construction, manufacturing and operation, services and procedures to which quality assurance applies, quality assurance implementation, including review and audit, means to demonstrate compliance with acceptance criteria, regulatory requirements, design bases, codes and

- standards, validation and verification of control measures, test programmes, resolution techniques for non-conformance events and evaluation of their effects on safety, management of the interface between safety and security, and management system procedures including distribution and control of the safety analysis report;
- (t) Planning for decommissioning, including, as appropriate, feasibility of decommissioning without undue risk to personnel, the public and the environment, and evidence that considerations for decommissioning have been included in the design, construction and operational lifetime of the reactor;
 - (u) Emergency planning and preparedness for radiological accidents including, as appropriate, approval by the appropriate authorities, agreement with relevant off-site emergency services including public authorities and regulatory bodies for provision of assistance and emergency support, on-site emergency response actions for design basis accidents and design extension conditions, development of credible accident scenarios with emergency actions for each, availability of resources and communications to manage each accident scenario, provision for drills and tests at prescribed time intervals, and provision for review and updating of the emergency plan.

3.5. CONSTRUCTION

3.5.1. Objective

The objective is to provide a basis for evaluating the construction of the reactor and associated requirements established in SSR-3 [2] and their adequacy. The review team should determine whether the as-built facility meets the design intention regarding major safety systems. Detailed guidance is provided in IAEA Safety Standards Series No. SSG-38, Construction for Nuclear Installations [17].

References

SSR-3 [2]: Requirement 14; SSG-38 [17]

3.5.2. Guidelines

- (1) Verify that the management system has established processes to ensure:
 - (a) Project control and responsibilities including quality assurance provisions;
 - (b) Control of material of construction including traceability;
 - (c) Proper control of the purchasing, delivery, receipt, handling, storage and installation of SSCs important to safety;
 - (d) Qualification of the contractors;
 - (e) Control of intentional and unintentional deviations during construction;
 - (f) Observance of construction hold points;
 - (g) Control of construction records and as built drawings;
 - (h) Compliance with relevant regulations and safety requirements.
 - (i) Transfer of responsibilities to commissioning group.
- (2) Verify that the operating organization is responsible for safety during construction of the research reactor.
- (3) Verify that the construction of the buildings and SSCs important to safety:

- (a) Meets the design assumptions of the reactor safety analysis;
 - (b) Conforms to construction and installation drawings and design specifications including materials of construction;
 - (c) Meets relevant codes and standards for construction, and good practices of local, national and international organizations;
 - (d) Has been reviewed and authorized by the appropriate regulatory body, as necessary;
 - (e) Includes protection to prevent internal and external damage during the construction period.
- (4) Determine whether buildings and structures:
- (a) Form the required barrier against uncontrolled release of radioactivity to the environment;
 - (b) Act as barriers to keep radiation levels and radioactive releases on and off the site in conformance with the principle of optimization of protection and safety, and within prescribed limits during all operational states;
 - (c) Provide protection against external events to enclosed safety systems;
 - (d) Provide a degree of leak tightness and ventilation air flow consistent with the reactor safety analysis;
 - (e) Permit testing of penetrations, doors, and airlocks.
- (5) Determine whether:
- (a) Deviations (non-conformities) between the design and as built systems have been documented and adequately resolved;
 - (b) Facility documentation, including drawings, has been updated to reflect the as built condition;
 - (c) All temporary provisions made for the construction which are not required further such as scaffoldings, temporary connections for utilities (e.g., power, water, air) have been removed.
- (6) Determine whether:
- (a) Access to SSCs is available, including for maintenance, inspection, repair or replacement as necessary;
 - (b) The facility layout permits unhindered escape routes for personnel;
 - (c) The safe movement of equipment and material, including transportation vehicles, is possible.

3.6. COMMISSIONING

3.6.1. Objective

To provide a basis for evaluation of the commissioning programme and procedures for research reactors (see IAEA Safety Standards Series No. SSG-80, Commissioning of Research Reactors [7]) and to ensure the requirements established in SSR-3 [2] are met. The review team determines the adequacy of the organization for commissioning, scheduled tests, and related procedures. If the commissioning stage has already been completed, the review team reviews the accuracy of the results as compared to the design specifications.

References

SSR-3 [2]: Requirements 30 and 73; SSG-80 [7].

3.6.2. Guidelines

- (1) Verify that an adequate commissioning programme has been prepared for the purpose of demonstrating that all design objectives have been achieved, and that the programme has been reviewed by the safety committee and approved by the regulatory body.
- (2) Verify that the necessary features and additional equipment needed only for commissioning (e.g., special neutron detectors, counting equipment, filters, filling, draining provisions, instrumentation, provisions for operation with transition cores) have been incorporated into the design of the reactor and related facilities to facilitate the reactor commissioning process.
- (3) Verify that experimental devices have been given adequate consideration in the commissioning programme, and that new experimental devices are subject to appropriate additional commissioning procedures.
- (4) Determine whether development of the commissioning programme has been a cooperative effort involving the operating organization, designers, manufacturers, and constructors.
- (5) Verify the adequacy of the organization that has been set up for commissioning, regarding the various groups involved, and their staffing, responsibilities, and training.
- (6) Verify that a suitable process exists for transfer of as built SSCs, including relevant documents, records, drawings, and manuals from the construction group to the commissioning group and from the commissioning group to the operating group.
- (7) Verify the adequacy of the management system, including the quality assurance programme for commissioning, regarding its provisions associated with the management, performance, and evaluation of commissioning activities.
- (8) Examine the commissioning stages and verify the adequacy of the tests and prerequisites included in each of the stages.
- (9) Verify that appropriate procedures have been prepared for each of the tests envisaged in the commissioning programme and that these procedures have been prepared and reviewed in accordance with the quality assurance programme for commissioning.
- (10) Verify that the commissioning programme includes adequate provisions for dealing with verifications, reviews, audits, deviations, non-conformances and keeping of records and updating of the SAR.
- (11) Review the commissioning programme and determine its adequacy with respect to the following objectives:
 - (a) Determination by measurement under realistic conditions of all safety relevant reactor characteristics;
 - (b) Verification, on the basis of measured data, of the relevant safety requirements;

- (c) Provision of additional information and data from commissioning to complete safety documentation, technical documentation and operating procedures;
 - (d) Provision of familiarization and training opportunities for operating and maintenance personnel.
- (12) Verify that commissioning tests are arranged in stages, functional groups and in a logical sequence that includes:
- (a) Stage A: Pre-operational tests prior to fuel loading;
 - (b) Stage B: Fuel loading tests, initial criticality tests and low power tests;
 - (c) Stage C: Power ascension test and power tests.
- (13) Verify that:
- (a) An emergency plan with implementing procedures is in place and tested before fuel loading;
 - (b) Personnel involved in commissioning are trained to cope with emergencies.
- (14) Verify that the procedures covering commissioning tests include:
- (a) Purpose of the test and results expected;
 - (b) Acceptance criteria;
 - (c) Safety provisions required to be in force for the test;
 - (d) Prerequisites for each stage and precautions;
 - (e) Test instructions;
 - (f) Provisions for data collection, data analysis, evaluation of the results, identification of deficiencies, and corrective actions.
- (15) Verify the accuracy of the results from the commissioning tests and the involvement of the operating personnel in the performance of the tests.
- (16) Ensure that all commissioning test results, whether produced by the operating organization or by suppliers, are available to the operating organization and are maintained for the lifetime of the facility.
- (17) Ensure that the analysis of postulated accidents and the capability of the safety systems to limit their consequences for the as built reactor, are fully documented before loading fuel.
- (18) Verify that all facility components and systems have been constructed in accordance with their design intent and that they meet the safety criteria.
- (19) Verify that a comprehensive commissioning report has been prepared that presents and assesses the results of commissioning, in particular, the action taken for unsatisfactory test results, if any.
- (20) Verify that the results of commissioning have been incorporated into the SAR and that the approved OLCs include the commissioning results.

3.7. SITING AND PROTECTION FROM EXTERNAL EVENTS

3.7.1. Objective

To provide a basis for evaluating the safety aspects of the reactor siting and design as established in SSR-3 [2] and IAEA Safety Standards No. SSR-1, Site Evaluation for Nuclear Installations [18]. Further guidance is available in Refs [14, 15, 19–24].

References

SSR-1 [18]: Requirement 1; SSR-3 [2]: section 5; Refs [4, 14, 15, 19–24].

3.7.2. Guidelines

The review team verifies that the site chosen in the case of a new research reactor, together with the design of the facility, take into account the principle of optimization of protection and safety with regard to potential public doses from normal operation and from accidents. For an existing facility, the review team examines documentation and records and visits the site to establish that changes in the characteristics of the site, such as surrounding population and other external changes of the facility, such as an increase in power level, do not significantly affect the safety of the site. If the review team is examining the design, then the design basis parameters for external hazards and protection of safety related structures and components from these are included in the review.

- (1) Review the siting and protection of the facility from external events with experienced members of the operating organization, and examine relevant documentation such as the SAR and siting justification to assess the degree to which:
 - (a) The site is commensurate with the potential hazard from the facility and site evaluation has taken into account all relevant natural and human induced external events, combinations of external events and consequential internal events;
 - (b) The site chosen ensures that the number of people likely to be affected by accidental releases from the facility is minimized, taking due account of other aspects of the population distribution such as the nature and distribution of the population around the site (e.g. the site chosen for a facility with a significant off-site hazard potential should avoid proximity to institutions with relatively large numbers of immobile people, such as hospitals or senior citizen homes, and with large concentrations of the population in the prevailing wind direction);
 - (c) Controls exist and have been applied to ensure that changes in the characteristics of the site have not adversely affected the safety of the facility by either increasing the probability of an accident occurring or increasing the consequences of an accident. In cases where such changes in the site characteristics have occurred, the reviewer assesses the extent to which these have been offset by additional engineered and/or administrative protection and mitigation features.
- (2) When reviewing the siting of the facility in greater detail, the reviewer examines the SAR, emergency plan, siting justification documentation, and periodic reviews of this documentation, and discusses site issues with relevant personnel, to establish that:
 - (a) The on-site and off-site emergency plan is compatible with the safety analysis for the facility and the characteristics of the site are taken into account;
 - (b) The capability of the ultimate heat sink is adequate for all operating conditions.

- (3) Determine whether the site evaluation and SAR provide a comprehensive review which takes into account all identified characteristics of the site and their impact from the safety point of view and includes:
- (a) General site description;
 - (b) Natural external events;
 - (c) Population in the vicinity of the reactor;
 - (d) Geology, seismology, meteorology, hydrology, oceanography, etc.;
 - (e) Nearby industrial and military facilities, including their potential effects on the reactor site;
 - (f) Potentially hazardous plant or activities in the vicinity of the site which could have an impact on the safety of the facility under review;
 - (g) Routes, types and frequency of aircrafts, and other types of transport such as trains, trucks and ships carrying potentially hazardous materials;
 - (h) Natural environment, land and water usage;
 - (i) Baseline radiological levels;
 - (j) Buildings or natural features that could affect the dispersion of radioactive releases from the site or which may affect the civil engineering aspects of the site in a way that may, during their construction or afterwards, create a hazard to the site;
 - (k) Dispersion of radioactive material (atmospheric and through aquifers, ground water, and surface water);
 - (l) Aspects of the topography and road structure of the area around the site which could affect the movement of people in an emergency or the access to the site of the external emergency team(s) in case of severe external event such as earthquakes. This will establish the changes in the potential impact of the facility to the population and environment and the design basis for external events (natural and man-induced) affecting the facility;
 - (m) Mitigation measures required for postulated accidents;
 - (n) Meteorological conditions (the predicted conditions may be different from those taken into account).
- (4) Check that a programme of review of the site characteristics (e.g., periodic safety review) is in place to monitor any change, to verify the original design assumptions and database, and to update the relevant documents such as the SAR.
- (5) Check that interfaces between safety and security aspects are adequately considered in the site selection and site evaluation process.
- (6) Determine whether a site reassessment has been done, taking into account the changes in the site characteristics and lessons learned from the Fukushima Daiichi accident. If yes, identify the findings and if any measures were implemented to overcome the deficiencies.

3.8. OPERATIONAL LIMITS AND CONDITIONS

3.8.1. Objective

To provide a basis for evaluating the adequacy of the research reactor OLCs.

References

SSR-3 [2]: Requirement 71; IAEA Safety Standards Series No. SSG-83, Operational Limits and Conditions and Operating Procedures for Research Reactors [6].

3.8.2. Guidelines

- (1) Determine that a detailed set of OLCs has been developed for the reactor, either as part of the safety analysis or incorporated into a separate document, which contains:
 - (a) Safety limits on important process variables which, if exceeded, could result in undue exposure of personnel, or the release of undue amounts of radioactivity to the environment;
 - (b) Safety system settings for those variables and parameters that, if not controlled, could result in safety limits being exceeded;
 - (c) Limiting conditions for safe operation to provide acceptable margins between normal operating values and safety system settings, including equipment requirements (both the characteristics and number of items of such equipment in operational conditions) to ensure safe operation;
 - (d) Surveillance requirements that prescribe the frequency and scope of tests of safety systems to ensure compliance with the approved OLCs;
 - (e) Administrative requirements such as:
 - (i) Organizational structure and responsibilities;
 - (ii) Staffing requirements;
 - (iii) Facility review and audit requirements;
 - (iv) Procedure requirements to ensure limits are not exceeded;
 - (v) Operational event review requirements;
 - (vi) Reports and records requirements.
- (2) Verify that the OLCs are derived from the reactor safety analysis and constitute a comprehensive envelope for safe operation to protect staff, the public, and the environment.
- (3) Verify that the applicable OLCs have been reviewed and approved by the regulatory body and that up-to-date copies are available to the operating personnel at the point of use.
- (4) Verify that the approved OLCs are presented with clear statements of their objectives, applicability, specification, and bases (justification).
- (5) Verify that the operating organization regularly reviews the approved OLCs in order to make revisions arising out of modifications, operational experience and/or changes in regulations.
- (6) Verify that the OLCs include actions to be taken if a safety limit, safety system setting or limiting condition for safe operation is not satisfied.

- (7) Determine whether any safety limits, safety system settings or limiting conditions for safe operation have been violated during reactor operation and, if so, which actions have been taken.
- (8) Determine the controls established by the operating organization to ensure compliance with the approved OLCs and to facilitate verification that the reactor has been operated in compliance with them.

3.9. SAFETY CULTURE

3.9.1. Objective

The basis for evaluating safety culture within the organization is considered as part of safety verification. The review establishes the extent to which an adequate set of safety culture attributes, and attitudes in individuals and organizations exist, as described in Refs [25–27], which ensures that safety issues, as an overriding priority, receive the attention that their significance warrants.

References

IAEA Safety Standards Series No. GSR Part 2 Leadership and Management for Safety [25]: Requirements 12 and 14; SSR-3 [2]: Requirement 3; IAEA Safety Standards Series No. GS-G-3.1, Application of the Management System for Facilities and Activities [27]

3.9.2. Guidelines

- (1) Examine the organizational structure and the organization policy to determine whether:
 - (a) A safety culture development programme is established;
 - (b) Safety is considered and given priority in all decision making;
 - (c) All individuals in the organization contribute to fostering and sustaining a strong safety culture.
- (2) Examine whether senior managers and all other managers advocate and support the following:
 - (a) A common understanding of safety and of safety culture, including awareness of radiation risks and hazards relating to work and to the working environment; an understanding of the significance of radiation risks and hazards for safety; and a collective commitment to safety by teams and individuals;
 - (b) Acceptance by individuals of personal accountability for their attitudes and conduct with regard to safety;
 - (c) An organizational culture that supports and encourages trust, collaboration, consultation and communication;
 - (d) The reporting of problems relating to technical, human and organizational factors and reporting of any deficiencies in SSCs to avoid degradation of safety, including the timely acknowledgement and reporting of actions taken;
 - (e) Measures at all levels in the organization to encourage a questioning and learning attitude and to discourage complacency with regard to safety;
 - (f) The means by which the organization seeks to enhance safety and to foster and sustain a strong safety culture, and using a systemic approach (i.e., an approach relating to the system as a whole in which the interactions between technical, human and organizational factors are duly considered);
 - (g) Understanding of safety culture.

- (3) Verify that the senior management:
 - (a) Ensures that self-assessment of leadership for safety and of safety culture takes place at all organizational levels and for all functions in the organization, making use of recognized experts in such assessment;
 - (b) Ensures that an independent assessment of leadership for safety and of safety culture is conducted for enhancement of the organizational culture for safety (i.e., the organizational culture as it relates to safety and as it fosters a strong safety culture in the organization).
- (4) Verify that the results of self-assessments and independent assessments of leadership for safety and of safety culture are communicated at all levels in the organization. The results of such assessments are acted upon to foster and sustain a strong safety culture, to improve leadership for safety and to foster a learning attitude within the organization.
- (5) Verify that adequate management structures are available to ensure the leadership for safety is clear and to provide clear accountability for safety with clearly delegated functions and responsibilities (see para. 5.2(b) of GSR Part 2 [25]).
- (6) Verify that adequate and appropriate resources are devoted to safety.
- (7) By discussion with management and individuals and by observing areas such as operational practices and housekeeping, determine whether:
 - (a) The relevant staff clearly understand the safety aspects of licence conditions and OLCs, the safety consequences of malfunction of items, and the safety significance of their actions;
 - (b) The system of rewards and sanctions, management attitudes and communication with individuals promotes good attitudes and motivation among staff members;
 - (c) Individuals are encouraged to have a questioning attitude towards matters affecting safety and a rigorous and prudent approach to their tasks, and are not afraid to communicate their concerns on safety matters and suggestions for improvement to line managers and others;
 - (d) A system of obtaining, addressing, and communicating feedback exists, along with procedures for the continuous improvement of the system.
- (8) Determine whether a safety culture encouragement and improvement programme is defined and implemented by ensuring that:
 - (a) Working conditions and environment encourage safety culture;
 - (b) The subjects of safety and safety culture are part of regular meetings;
 - (c) Safety culture awareness and safety performance by staff is evaluated and reported periodically;
 - (d) Safety culture improvement actions are identified;
 - (e) There is a formal process for recording safety issues and documenting the corrective actions.
- (9) Examine safety practices at the corporate level by posing the following questions:
 - (a) Does the corporate board have expertise in safety?
 - (b) Do formal meetings at this level include agenda items on safety?
 - (c) Do operating staff attend formal meetings to discuss the safety performance of the facility?

- (d) Is there an active nuclear safety review committee which reports its findings at corporate level?
- (e) Is there a senior member of the board with safety as a prime responsibility? How is the member supported and assisted in their duties? What is this member's standing compared with that of the heads of other functions?
- (f) Are the resource requirements for the safety function reviewed periodically at corporate level, and which results have been achieved?

(10) Safety responsibility should be questioned as follows:

- (a) Has the assignment of safety responsibilities been clearly enunciated?
- (b) Has the responsibility of the reactor manager been clearly stated and accepted?
- (c) Are the documents that identify safety responsibilities kept up to date and reviewed periodically, and with what result?

(11) Examine whether the safety attitude of the managers is factored in the performance appraisal or promotions.

3.10. REGULATORY SUPERVISION

3.10.1. Objective

To provide the basis for evaluating the regulatory activities as described in IAEA Safety Standards Series No. GSR Part 1 (Rev. 1), Governmental, Legal and Regulatory Framework for Safety [28] and IAEA Safety Standards Series No. GSG-13 Functions and Processes of the Regulatory Body for Safety [29].

References

GSR Part 1 (Rev 1) [28]: Requirements 3 and 4; SSR-3 [2]: Section 3; GSG 13 [29].

3.10.2. Guidelines

The reviewer determines that the reactor is subject to independent assessment and inspection and that the facility operates in compliance with the licence requirements. In particular, the review team verifies that the facility is regulated by an appropriate regulatory or institutional body and that an adequate legal framework exists. The requirements on regulatory supervision of nuclear facilities in general are established in GSR Part 1 (Rev. 1) [28] and guidance on how to meet them is provided in GSG-13 [29]. The IAEA Integrated Regulatory Review Service (IRRS) is also available to provide peer review advice to regulatory bodies.

- (1) By examining documentation and discussing with the staff of the regulatory body and the operating organization, the reviewer identifies whether:
 - (a) A regulatory body, or an equivalent institutional body, exists to regulate reactor safety related activities;
 - (b) The regulatory body is effectively independent of the operating organization or, in the case of a country with a relatively small nuclear programme, is at least independent of the management of the reactor facility;
 - (c) The regulatory body is responsible for governmental surveillance and control with regard to nuclear safety in all activities of nuclear research reactors;

- (d) The relationship between the operating organization and the regulatory body is based on mutual understanding, respect and confidence;
 - (e) A licensing process has been established, with a clear process for the submission of safety related information including their schedules and contents;
 - (f) The regulatory staff are adequately qualified and knowledgeable and have sufficient resources to fulfil their responsibilities; and
 - (g) Despite the responsibilities of the regulatory body for surveillance and control with regard to all problems relevant to nuclear safety in the siting, design, construction, commissioning, operation and decommissioning of research reactors, the operating organization management accepts that the responsibility for safety of the facility, and for demonstrating an adequate level of safety, remains with them, and not with the regulatory body.
- (2) For a more in-depth review, the reviewer examines:
- (a) The institutional bases for the regulatory body (e.g. laws, regulations);
 - (b) The lines of communication between the operating organization, the regulatory body, and the government;
 - (c) The licensing process, as an ongoing activity;
 - (d) Documents submitted to the regulatory body, their disposition and follow-up actions;
 - (e) The qualifications of the regulatory body staff and the availability of outside consultants where and when necessary;
 - (f) The provision of sufficient resources, commensurate with the size of the nuclear programme.
- (3) The reviewer examines the details of the licensing programme to ensure that the regulatory body:
- (a) Has established or adopted safety policies, principles, associated criteria, regulations and guidance upon which the regulatory activities are based;
 - (b) Has provided the operating organization with information on its regulatory approach, organization, procedures and decisions;
 - (c) Has free unimpeded access to all relevant documentation including SARs, management arrangements, facility operating records, quality assurance records, and safety committee minutes and documents;
 - (d) Has free unimpeded access to all accessible areas of the facility subject to the access control rules and procedures.
- (4) The reviewer also examines the following:
- (a) The existence of a SAR to facilitate the safety assessment of the reactor facility;
 - (b) The existence of independent means of review and approval within the operating organization that can provide judgment on the adequacy of the safety of the facility and may endorse proposals for action by the reactor manager;
 - (c) Safety related documentation submitted to the regulatory body by the operating organization;
 - (d) The programme of review and assessment as applied to the various stages of the licensing process;
 - (e) The adequacy of licensing review with regard to the reactor design, construction and operation;
 - (f) That only suitably qualified and experienced persons hold posts that can affect safety;

- (g) That regulatory inspections are carried out to ensure conformance with the OLCs and applicable regulations, codes and standards;
 - (h) Information concerning safety related incidents including its treatment;
 - (i) Planned modifications that may have a significant effect on safety;
 - (j) Experiments that may have a major effect on safety and are beyond the scope of the existing OLCs and the SAR;
 - (k) Regulatory inspection (planned and unannounced) frequency, scope and details, including sample reports;
 - (l) Enforcement of regulations, including conditions attached to the licence and any necessary corrective actions.
- (5) Determine whether a clear and established process for communicating routine and non-routine radioactive discharges, operational occurrences, violations of OLCs and other regulatory requirements to the regulatory body, and follow-up on necessary corrective actions has been established and followed.

3.11. SAFETY COMMITTEE

3.11.1. Objective

The objective is to determine whether an appropriate safety committee exists and meets regularly to supervise safe reactor operation (see SSR-3 [2]). The establishment of an appropriate safety committee or advisory group to provide independent advice on safety matters to the management of the operating organization is a proven and recommended means to ensure safety.

References

SSR-3 [2]: Requirement 6.

3.11.2. Guidelines

- (1) By examining documentation and holding discussions with the staff of the operating organization, the reviewer identifies whether:
 - (a) A safety committee or an equivalent advisory group exists to review safety aspects of the operation of the research reactor and its associated facilities, including the safety of its utilization;
 - (b) The committee includes members who are independent of the operating organization;
 - (c) The safety committee is composed of members who are adequately qualified to perform their task (various members may be experts in special fields but the committee as a whole provides advice covering sufficient safety areas);
 - (d) The committee advises on all aspects affecting the safety of the facility during design, construction, commissioning, operation and decommissioning;
 - (e) Terms of reference for the safety committee have been agreed on by the committee and the operating organization and approved/agreed by the regulatory body;
 - (f) The advice of the committee is transparent to the management of the facility and the regulatory body.

- (2) For a more in-depth review, the reviewer examines:
- (a) The records of the meetings of the safety committee and of the operating organization to examine how complete these records are, what matters are discussed, whether all points of view are recorded, the frequency of the meetings, to what extent the advice of the committee is followed, and what follow-up action is initiated;
 - (b) The terms of reference for the safety committee to check whether it needs to consider and provide advice on:
 - (i) Safety policy matters;
 - (ii) Management arrangements devised to ensure safety, and any proposed changes to these arrangements;
 - (iii) Safety aspects of facility design, construction, commissioning, operation, maintenance, testing, experiments, utilization, modification, and decommissioning.
 - (c) The qualifications and experience of members of the safety committee;
 - (d) The procedure for the operating organization to reject the advice of the safety committee;
 - (e) The way in which urgent safety proposals are handled;
 - (f) Reporting of incidents and faults affecting reactor safety to the safety committee.
- (3) The reviewer examines whether there is a list of documents that the safety committee needs to review, and verify that the list includes the following, as stated in para 4.27 of SSR-3 [2]:
- (a) The design of structures, systems and components and in particular the design and qualification of nuclear fuel elements and reactivity control elements;
 - (b) Safety documents and their modifications;
 - (c) Proposed new tests, experiments, equipment, systems or procedures that have significance for safety;
 - (d) Proposed modifications to items important to safety and changes in experiments that have implications for safety;
 - (e) Violations of the OLCs, of the licence and of procedures that are significant to safety;
 - (f) Events that are required to be reported or that have been reported to the regulatory body;
 - (g) Periodic reviews of the operational performance and the safety performance of the research reactor facility;
 - (h) Reports on routine radioactive discharges to the environment;
 - (i) Reports on radiation doses to the personnel at the facility and to the public;
 - (j) Reports to be provided to the regulatory body;
 - (k) Reports on regulatory inspections.
- (4) The reviewer examines the records to verify that the safety committee has considered or reviewed management arrangements related to:
- (a) Appointment of persons to posts which can have an impact on safety;
 - (b) The working methods of the safety committee;
 - (c) Control of construction and installation of a new reactor facility;
 - (d) Control of modifications to the design of a reactor facility under construction;
 - (e) Commissioning and decommissioning;
 - (f) Control of modifications to an existing reactor facility;
 - (g) Control of experiments and research proposals that may affect the safety of the facility;

- (h) Examination, inspection, maintenance and testing of items which affect safety;
- (i) Ageing management activities;
- (j) Control of fissile material and radioactive material;
- (k) Record keeping of matters important to safety;
- (l) Training of persons who have an impact on safety;
- (m) Emergencies;
- (n) Preparation, approval, updating and review of safety documentation;
- (o) Peer review of the safety of the facility;
- (p) Quality assurance aspects of items and systems important to safety;
- (q) Radiation protection;
- (r) OLCs;
- (s) Operating instructions;
- (t) Control and discharge of radioactive waste;
- (u) Regulatory supervision;
- (v) Adequacy of resources (e.g., personnel, funds) to ensure safe operation.

3.12. OPERATING ORGANIZATION AND REACTOR MANAGEMENT

3.12.1. Objective

To provide a basis for evaluating the adequacy of the operating organization and reactor management.

References

SSR-3 [2]: Requirements 67–69; IAEA Safety Standards Series No. SSG-84, The Operating Organization and the Recruitment, Training and Qualification of Personnel for Research Reactors [8]

3.12.2. Guidelines

The review team examines whether the following have been established and implemented:

- (a) Structure and responsibilities of the operating organization;
 - (b) Roles and responsibilities of:
 - (i) Operating personnel;
 - (ii) Radiation protection personnel;
 - (iii) Additional support personnel.
- (1) The reviewer examines the organizational chart, management section of the safety analysis report, quality assurance programme, policy documents, and job descriptions for safety related posts, and discuss with relevant personnel to ensure that:
- (a) A clearly defined organizational structure is available to and understood by relevant personnel, and the organizational structure is accepted/approved by the regulatory body;
 - (b) Functions and responsibilities are clearly defined and understood by relevant personnel;
 - (c) The operating organization has overall responsibility for safety;
 - (d) The reactor manager has the responsibility for the safe operation of the reactor, and this has been clearly defined in a written delegation of responsibility by the operating organization to the reactor manager;

- (e) The reactor manager has the necessary authority and resources to fulfil the responsibilities of the post;
 - (f) Other members of the operating organization performing functions of prime importance to safety, such as radiation protection officers and reactor physicists, have the necessary authority and resources to fulfil their duties.
- (2) In undertaking a deeper review of the organization, the reviewer examines the structure and responsibilities of all the support functions as well as those directly involved in the operation of the reactor. These include staff involved in maintenance, training, chemical analysis, quality assurance, radiation protection, engineering support, and peer review groups (see para. 7.2 of SSR-3 [2]).
 - (3) The reviewer examines the degree of independence of those performing radiation protection and quality assurance functions from the reactor management.
 - (4) The reviewer verifies that all the posts that can affect safety have been identified, that qualification, training, and experience needed for the posts are defined, that any national licensing or certification needed is defined, and that appropriate written description of responsibility and associated authority are provided to the individuals holding the posts.
 - (5) Conduct interviews with a selection of staff, including the reactor manager and at least one operator to assess the degree to which the responsibilities of their posts are defined and understood. During the interviews, assess whether the operating staff have adequate standards of, and a proper attitude toward, safety.
 - (6) Verify that:
 - (a) Clear lines of authority and communication are established for the reactor manager, safety committee, radiation protection groups, maintenance groups, management system personnel and experimenters;
 - (b) Periodic summary reports and abnormal event reports are prepared and submitted to the safety committee and regulatory body as appropriate;
 - (c) An integrated management system is established;
 - (d) Safety relevant programmes such as operating experience feedback and foreign material exclusion are established;
 - (e) Long term human resources and knowledge preservation policies are developed;
 - (f) Staff requirements for normal operation and accident conditions are specified and available;
 - (g) Training and retraining programme for all operating personnel and reactor users (experimenters) are established;
 - (h) Groups such as maintenance, radiation protection, and additional support personnel are identified and available;
 - (i) Management of the interface between safety and security is addressed.

3.13. TRAINING AND QUALIFICATION

3.13.1. Objective

To provide a basis for evaluating training programmes and personnel qualification at research reactors. The review team should verify that programmes for personnel training and retraining are in place and are consistent with the guidelines described in SSG-84 [8].

References

SSR-3 [2]: Requirement 70; SSG-84 [8]

3.13.2. Guidelines

- (1) Determine whether a formally reviewed and approved training and qualification programme is established and implemented.
- (2) Examine the training and qualification programme to determine whether it contains:
 - (a) Training organization and administration;
 - (b) Provisions for the training of all operating personnel such as operations, maintenance and radiation protection personnel;
 - (c) Requirements for initial operator training and certification;
 - (d) Requirements for requalification training;
 - (e) Adequate provisions for training reactor users (experimenters).
- (3) Determine whether the organization and administration of the training and qualification programme includes:
 - (a) The structure of the training organization;
 - (b) Requirements for qualifications of instructors;
 - (c) Provision of resources for training support;
 - (d) A description of training materials and the content of those materials;
 - (e) Procedures for selection of personnel;
 - (f) The educational qualifications, experience, and competence of all operating personnel.
- (4) Determine whether initial operator training and certification includes:
 - (a) An organized curriculum with written and practical examinations;
 - (b) Specific and adequate instruction in reactor theory, radiation protection, OLCs, facility systems, operating procedures for operational states and accident conditions, and emergency preparedness and response;
 - (c) Written examinations and practical checkouts;
 - (d) Certification for all phases of qualification with minimum levels for satisfactory completion;
 - (e) Maintenance of training records for at least three years following termination of employment.
- (5) Determine whether operator requalification training:
 - (a) Is conducted on a regular basis such as a one or two-year cycle, and in cases when an operator has not performed their duty for a certain period, such as six months;
 - (b) Has an organized curriculum with written and practical examinations;
 - (c) Provides certification for completion of requalification with minimum levels for satisfactory completion;
 - (d) Provides for maintenance of requalification records for at least three years following termination of employment.

- (6) Determine whether training is provided in areas not specifically required for operator certification, such as calibration and maintenance, life saving techniques and firefighting.
- (7) Examine the training programme considering the following questions:
 - (a) Does all training and retraining important to safety culminate in formal assessment and approval for duties? What is the success/failure record? What is the proportion of operating staff time devoted to training and how does this compare with the practices of other research reactors for a similar size and type of facility?
 - (b) Which resources are allocated to training? How does this compare with the allocations in similar facilities?
 - (c) Is the quality of the training programme assessed at the corporate and reactor management levels?
 - (d) Is there a periodic review of the applicability, correctness, and results of training courses? Does this review take into account operating experience feedback?
 - (e) How frequently are production requirements permitted to interfere with scheduled training?
 - (f) Do staff members understand the significance of the operational limits of the research reactor in their areas of responsibility?
 - (g) Are the staff educated in the safety consequences of the malfunction of facility items?
 - (h) Are the staff trained in the special importance of operating procedures? Are they regularly reminded? Are they trained in the safety basis of the procedures?
 - (i) Can training staff cite examples of operating errors that have resulted in modifications to a training programme?
 - (j) For maintenance personnel, do training sessions make use of mock-ups before a complex maintenance activity is performed?
 - (k) Does the training programme include operating personnel response under accident conditions?
 - (l) Do training programmes address safety culture?

3.14. CONDUCT OF OPERATIONS

3.14.1. Objective

To provide a basis for evaluation of the conduct of operations at a research reactor. The review team verifies that operations are conducted in accordance with written procedures, that housekeeping is acceptable, and that records and reports are well maintained (see Refs [2, 6, 9, 30–32]).

References

SSR-3 [2]: Requirements 11, 39, 62, 72, 74–76, 78, 79, 82, 87 and 88; SSG-83 [6], IAEA Safety Standards Series No. SSG-82, Core Management and Fuel Handling for Research Reactors [9], IAEA-TECDOC-1387 [30], IAEA-TECDOC-1801 [31].

3.14.2. Guidelines

- (1) Verify that adequate up-to-date written operating procedures are available to operators in the reactor control room and effectively used.
- (2) Determine whether personnel involved in the operation and use of the reactor are adequately trained in the procedures and their use.

- (3) Verify that core management and fuel handling are carried out in accordance with the requirements of nuclear and radiation safety.
- (4) Verify that up-to-date information concerning the design, construction, commissioning, and operation of the reactor facility, including site and environmental data, design specifications, details of material and equipment, as built drawings, operating and maintenance manuals, and quality assurance documents are available and retrievable.
- (5) Ensure that written operating procedures include adequate, technically accurate, and complete written instructions for the following activities:
 - (a) Startup, operation and shutdown of the reactor and, where appropriate, operation of experimental devices;
 - (b) Loading, unloading and movement within the reactor of fuel elements and assemblies, reflector assemblies, experimental devices, and other core components;
 - (c) Routine maintenance of major components or systems that could affect reactor safety;
 - (d) Surveillance required by the OLCs including periodic inspections and tests of SSCs that are essential for the safe operation of the reactor;
 - (e) Implementation of a radiation protection and control programme consistent with applicable regulations;
 - (f) Authorization of operation and maintenance activities and conduct of experiments and irradiations that could affect reactor safety or insert reactivity in the core;
 - (g) Response of the operator to anticipated operational occurrences, design basis accidents and, to the extent feasible, design extension conditions;
 - (h) Emergency actions;
 - (i) Handling of radioactive waste and monitoring and control of radioactive releases;
 - (j) Utilization;
 - (k) Modifications;
 - (l) Surveillance, as required, of the reactor and its auxiliary systems during reactor shutdown periods.
- (6) Determine that operating procedures are periodically reviewed and updated.
- (7) Verify that changes to operating procedures are made in accordance with predetermined internal procedures and that changes are made known to operating personnel.
- (8) Verify that procedures for the operation and use of the reactor are consistent with the OLCs.
- (9) For activities related to operations, tests, maintenance or experiments not covered by existing procedures, determine whether appropriate procedures were prepared, reviewed, and approved prior to the start of the activities, and whether training of relevant staff was conducted.
- (10) Verify that adequate facilities are provided for handling, storage and disposal of spent fuel.
- (11) Verify that new fuel is handled and stored in a manner that will prevent criticality and that will minimize the possibility of theft.
- (12) Verify that a validated calculation basis exists for the loading of fuel, reflectors, safety activation devices and experimental devices in the core and that all core configurations are in accordance with design intent and assumptions specified in the OLCs.

- (13) Verify that the core management includes:
- (a) Assessment of the safety implications of core components or materials proposed for irradiation;
 - (b) Investigations of fuel or experiment failures;
 - (c) Assessment of effects of irradiation on core components;
 - (d) Failed fuel detection arrangements;
 - (e) Procedures for handling failed fuel and other core components.
- (14) Check that the packaging and transportation of fresh and irradiated fuel elements have been completed in accordance with national and international regulations and, as appropriate, in accordance with IAEA Safety Standards Series No. SSR-6 (Rev. 1), Regulations for the Safe Transport of Radioactive Material [32].
- (15) Check that habitability and good conditions are maintained in the control room. Also check that there is a supplementary control room and/or panel. Check that the functioning of the supplementary control room and/or panel are tested to ensure their readiness.
- (16) For facilities with multiple control centres (e.g., main control room, supplementary control room, control rooms for experiments, radiation protection and security), check that a clear line of communication and hierarchy of command is established to avoid conflicting inputs.
- (17) Determine whether the following records and reports related to the operation of the reactor are available and retrievable either as self-standing documents or as parts of procedures:
- (a) Routine operating data including logbooks, reading sheets, checklists and automatically recorded data;
 - (b) Core management, fuel behaviour and fuel handling activities;
 - (c) Performance evaluation of safety systems;
 - (d) Current operational status and components out of service;
 - (e) Written instructions for temporary procedures or procedures that vary from existing, approved procedures;
 - (f) Maintenance, periodic testing and inspection;
 - (g) Safety categorization of experiments and modifications;
 - (h) Location and transfer of radioactive sources and fissile materials;
 - (i) Staff qualification and training;
 - (j) In-service failures, safety related occurrences and incidents;
 - (k) Radiation exposure and medical records;
 - (l) Radioactive waste storage and shipment;
 - (m) Radioactive effluent releases;
 - (n) Environmental monitoring results;
 - (o) Management system audits and reviews;
 - (p) Relevant commissioning records, including startup test reports;
 - (q) Records relevant to decommissioning;
 - (r) Communications with regulatory bodies.
- (18) Verify that entries in logbooks, checklists and other operating records are timely and complete, adequately reflect facility activities and status, and are signed and dated.
- (19) Ensure that adequate storage of records and reports is provided, including backups as appropriate.

- (20) Verify that only the most current version of each document is used by operating personnel.
- (21) Check that adequate retention periods for records are specified.
- (22) Verify that for facility and equipment operations:
- (a) Lines of supervision are clearly defined;
 - (b) Shift turnovers are formal and complete;
 - (c) Responsibilities and authority are clearly defined by position descriptions and procedures.
- (23) Verify that the following elements are taken into account or fulfilled during the facility and equipment operations:
- (a) Operations are in compliance with established OLCs;
 - (b) Alarm set points are correct and adhered to;
 - (c) Operator vigilance and supervision are ensured;
 - (d) There is a programme to track and repair out-of-service equipment;
 - (e) Changes of system and component status are recorded in the documentation;
 - (f) Off-normal conditions are apparent to operators;
 - (g) Lighting is adequate;
 - (h) Instrumentation is unobstructed, clearly readable, and understandable to operators;
 - (i) Defective and out-of-service instrumentation is clearly identified and made apparent to operators;
 - (j) Communications equipment is properly operating;
 - (k) Radiation survey instruments are available at appropriate locations and in proper working order;
 - (l) SSCs important to safety are labelled and clearly identifiable;
 - (m) Housekeeping and cleanliness are satisfactory and temporary storage is controlled and limited;
 - (n) Degraded or damaged equipment is identified and attended to in a timely manner.
- (24) Determine that equipment is operating properly and that:
- (a) There is protection from adverse environmental conditions;
 - (b) Instruments are calibrated and the calibration period is valid;
 - (c) Drive belts show no excess wear;
 - (d) Fasteners, cover guards and supports are properly installed;
 - (e) Electrical cables are properly laid out and supported;
 - (f) Insulation is installed where appropriate.
- (25) Verify that a proper water chemistry control and monitoring programme is established and implemented in accordance with the OLCs and approved procedures.
- (26) Verify that, to manage an extended shutdown state of the reactor, appropriate arrangements have been considered for the following:
- (a) Unloading and safe storage of the fuel;
 - (b) Revision of OLCs to cater for the extended shutdown state;
 - (c) Removal of components for protective storage;
 - (d) Measures to prevent ageing related degradation such as corrosion;
 - (e) Security of the nuclear material;

- (f) Availability of adequate procedures and resources to ensure safety;
- (g) Retention of adequate staff.

(27) Verify that:

- (a) An operating experience feedback programme is established which includes internal as well as external operating experience and provides feedback to the national and international systems;
- (b) Events are investigated to determine root causes and implement corrective actions;
- (c) Feedback is used to disseminate information and to improve facility programmes such as procedures and training.

(28) Verify that adequate fire safety arrangements are available that include:

- (a) Fire safety provisions based on a fire hazard analysis of the facility;
- (b) Control of combustible materials and ignition sources;
- (c) Arrangements for fire detection, location and alarms (in the control room and in an alternate location, such as the security office);
- (d) Availability and adequacy of different types of fire extinguisher, and other arrangements such as fire water;
- (e) Availability of personal protective equipment for firefighting;
- (f) Automatic or manual actuation of fire extinguishers;
- (g) Training for the facility staff (including security staff) in fire safety aspects, including the use of personal protective equipment and fire extinguishers, and training for the firefighting staff in radiation protection aspects;
- (h) Communication arrangement with the local fire safety authorities (fire brigade);
- (i) Maintenance, inspection and testing programme to ensure the operability and availability of fire safety equipment, including the monitoring system;
- (j) Conduct of periodic fire drills.

(29) Verify that:

- (a) The interface between safety and security is well managed for all processes during the operational stage of the research reactor;
- (b) Unauthorized access to nuclear material and items important to safety, including computer hardware and software, is prevented;
- (c) Access control ensures that only authorized persons enter the designated areas and at the same time, any safety issue such as evacuation is possible without undue delay.

3.15. MAINTENANCE, PERIODIC TESTING AND INSPECTION

3.15.1. Objective

To provide a basis for evaluating the preparation and implementation of maintenance, periodic testing, and inspection programmes.

The review team should verify that such programmes exist and ensure that they satisfy the requirements. The reviewer should also conduct a facility walkdown and observe, if possible, some of these activities to assess the quality and performance aspects.

References

SSR-3 [2]: Requirements 77 and 86; IAEA Safety Standards Series No. SSG-81, Maintenance, Periodic Testing and Inspections of Research Reactors [10]; IAEA Safety Standards Series No. SSG-10 (Rev. 1), Ageing Management for Research Reactors [12].

3.15.2. Guidelines

- (1) Determine that an overall programme for maintenance, periodic testing and inspection of systems and equipment exists, with the following components:
 - (a) Systems and equipment covered;
 - (b) Mode of maintenance and inspection;
 - (c) Frequency;
 - (d) Responsible person by name or title;
 - (e) Authorization mechanism;
 - (f) Procedures for testing and resumption of normal operation.
- (2) Determine that maintenance, periodic testing and inspection activities are conducted to ensure:
 - (a) Compliance with OLCs;
 - (b) Adequacy of the safety status of the reactor.
- (3) Verify that maintenance work on installed equipment, removal of equipment from operation for maintenance purposes, and reinstallation of equipment after maintenance is:
 - (a) The responsibility of the reactor manager; and
 - (b) Conducted with due regard to maintaining the level of safety of the reactor as specified in the OLCs.
- (4) Verify that there are written procedures for the maintenance, periodic testing and inspection of reactor equipment, in particular of all items important to safety, and that the procedures are based on the reactor safety analysis and manufacturers' recommendations.
- (5) Verify that:
 - (a) A clearly defined structure for authorization of performance of all maintenance and periodic testing exists;
 - (b) Removal, replacement, repair and service of items important to safety are performed only by specifically authorized personnel;
 - (c) A system of work permits is used including appropriate check-off procedures, before and after the conduct of work in accordance with a quality assurance programme;
 - (d) The results of maintenance and periodic testing are assessed by properly qualified personnel in order to verify compliance with the OLCs.
- (6) Determine that the frequency of maintenance and periodic testing of individual SSCs ensures adequate reliability, taking into account:
 - (a) Their relative importance to safety;
 - (b) The likelihood of their failure to function as intended;
 - (c) Requirements established in the reactor safety analysis and any subsequent revisions.

- (7) Determine that measuring and test equipment is:
 - (a) Calibrated against national or international standards at recommended intervals;
 - (b) Subject to controlled use;
 - (c) Tagged and removed from service when out of tolerance.
- (8) Determine that a means of immediate rectification exists when calibrations, checks and inspections reveal a non-conformance with safety system settings or limiting conditions for safe operation of the reactor.
- (9) Verify that resumption of normal operation is permitted only by the person responsible for the coordination of maintenance work.
- (10) Ensure that records of maintenance and periodic testing conform to the requirements of the quality assurance programme. In particular, verify that these records:
 - (a) Are technically accurate, adequate and complete;
 - (b) Are current, dated, periodically reviewed, signed and available to operating staff.
- (11) Determine that procedures exist for all maintenance, periodic testing and inspection activities required by the OLCs and that these procedures:
 - (a) Are technically accurate, adequate and complete;
 - (b) Are current, dated, periodically reviewed, signed and available to operating staff;
 - (c) Include acceptance criteria;
 - (d) Provide for corrective action following an out-of-specification calibration;
 - (e) Provide criteria for successful calibration.
- (12) Determine that a master maintenance schedule exists for maintenance, periodic testing, and inspection, and that there are means for ensuring the completion of these activities, record retention, and compliance with the frequency requirements.
- (13) Verify that the maintenance, periodic testing, and inspection programme is periodically reviewed and adjusted, taking into account the experience within the facility and from other similar facilities.
- (14) Verify that all maintenance, periodic testing, and inspection activities are performed with proper safety equipment and radiological protection, with procedures available and in use, and are properly documented.
- (15) Determine that a routine preventive maintenance programme exists and that it includes an appropriate lubrication programme, rotation in operation of redundant equipment, verification of spare equipment operation, drive belt replacement, painting, filter replacement, and draining of systems subject to freezing.
- (16) Determine that the preventive maintenance programme in place pays special attention to systems, equipment, and components to ensure that ageing effects do not cause failure.
- (17) Determine that an ageing management programme exists and covers all stages of the facility life. Such a programme should include both physical ageing and obsolescence of the SSCs. In many research reactors, an ageing management programme could be associated with the maintenance, periodic testing, and inspection programme.

- (18) Verify that the ageing management programme includes the following elements:
- (a) List of SSCs important to safety, including experimental devices that are included in ageing management programme;
 - (b) Applicable degradation mechanisms for each SSC;
 - (c) Activities to minimize the ageing effects;
 - (d) Activities to detect and monitor the ageing effects and analyse the trends;
 - (e) Means to mitigate ageing effects;
 - (f) Criteria to accept the continued operation of an SSC;
 - (g) Corrective actions needed/taken to keep the SSC in service;
 - (h) Continuous improvement;
 - (i) Records and data of the ageing management programme.
- (19) Verify that the ageing management programme has appropriate interfaces with other technical areas including:
- (a) Maintenance, inspection and periodic testing;
 - (b) Periodic safety review;
 - (c) Equipment qualification;
 - (d) Water chemistry programme;
 - (e) Design basis of the facility;
 - (f) Facility configuration management;
 - (g) Post service surveillance.
- (20) Determine that a periodic safety review or equivalent programme exists in the facility (see Requirement 5 of SSR-3 [2]). Also check if such a programme is in accordance with regulatory requirements.
- (21) Verify that the periodic safety review programme is agreed with the regulatory body and includes the review of safety factors relating to:
- (a) The facility;
 - (b) Safety analysis;
 - (c) Operational experience;
 - (d) Organizational effectiveness;
 - (e) Environment.
- (22) For a more in-depth review, examine the last periodic safety review report (if available) and determine what corrective actions and safety improvements were recommended and their implementation status.

3.16. MODIFICATIONS

3.16.1. Objective

To provide a basis for evaluating research reactor modifications. The review team verifies that modifications are assessed to determine their safety impact and taken into account in the safety documents of the facility.

References

SSR-3 [2]: Requirements 36 and 83; IAEA Safety Standards Series No. SSG-24 (Rev. 1), Safety in the Utilization and Modification of Research Reactors [11].

3.16.2. Guidelines

- (1) Verify that there is a procedure for classifying modifications into those that have safety significance and those that do not.
- (2) Verify that modifications with safety significance are, in accordance with the procedures and as appropriate, subject to safety analyses, and to design, construction, and commissioning procedures equivalent to those envisaged for the original design, with particular attention being given to site environment changes such as population movement and general site usage.
- (3) Verify that the following types of modifications having safety significance are submitted for review and approval by the regulatory body:
 - (a) Changes in approved safety limits and safety system settings;
 - (b) Changes in approved limiting conditions for safe operation, surveillance and administrative requirements in OLCs;
 - (c) Modifications that could have a significant impact on safety;
 - (d) Modifications that create new safety hazards or hazards not previously addressed in the reactor safety analysis;
 - (e) Modifications that reduce an existing margin of safety.
- (4) Determine that all reactor and facility modifications have been subjected to an internal review by the facility operations staff and by the safety committee, in accordance with the procedures.
- (5) Determine that a procedure for controlling modifications exists, that it has been implemented, and that, as appropriate, it includes:
 - (a) A description of the proposed modification;
 - (b) Justification for the modification such as ageing, backfitting or upgrading;
 - (c) Internal organization, arrangements associated with the modification and specific responsibilities;
 - (d) Design requirements and criteria;
 - (e) A safety assessment that supports the modification;
 - (f) Specifications of the manufacturing processes;
 - (g) Installation procedures;
 - (h) Commissioning process;
 - (i) Testing and inspection of the completed modification;
 - (j) Review of operational and emergency procedures;
 - (k) Involvement of external agencies including suppliers and contractors;
 - (l) Documentation updating;
 - (m) Special requirements for training and operator certification;
 - (n) Decommissioning provisions;
 - (o) Quality assurance requirements;
 - (p) Routes of review and approval based on the safety significance of the modification.

- (6) Verify that modifications have to be and have been optimized with respect to reducing the radiation exposure of personnel.
- (7) Verify that clear delineation of responsibilities for design, installation, commissioning and acceptance of modifications has been established and observed.
- (8) Determine, for each facility or system, that due consideration is and has been given to the following items in assessing and implementing the modification:
 - (a) Core reactivity effects;
 - (b) Radiation protection including shielding, waste, the potential for increased personnel exposures, and the potential for uncontrolled release of radioactive material within the site and to the environment;
 - (c) Safety devices including interactions with the reactor protection system and any deleterious effects;
 - (d) Heat generation;
 - (e) Cooling;
 - (f) Internal pressure and potential for explosion;
 - (g) Flux perturbations;
 - (h) Protection against external hazards.
- (9) Verify that temporary modifications are limited in number and time and that a procedure exists to control them.
- (10) Verify that:
 - (a) A change control process is established that ensures any changes to the design or layout, including experiments, are reviewed considering the interface between safety and security aspects;
 - (b) Effective control is established to prevent the introduction of any weakness, especially during construction or major modifications, that may jeopardize safety or security at a later stage.

3.17. UTILIZATION AND EXPERIMENTS

3.17.1. Objective

To provide a basis for evaluation of research reactor utilization and experiments. The reviewer verifies that procedures for experiments exist and are used and, in the case of experiments with a significant impact on safety, verifies that these experiments have followed a formal licensing process, including commissioning.

References

SSR-3 [2]: Requirements 36 and 83; SSG-24 (Rev. 1) [11].

3.17.2. Guidelines

- (1) Verify that each new experiment is reviewed through an established internal procedure for its safety significance.

- (2) Verify that each experiment judged to be of safety significance is submitted to an internal safety committee for review, and to the regulatory body, if appropriate.
- (3) Verify that modifications to experimental devices are subject to the same procedures used for the original device.
- (4) Verify that a safety analysis has been prepared for every new reactor experiment which may significantly affect reactor safety.
- (5) Verify that all experimental devices loaded into or directly connected to the reactor are designed to the same standards as the reactor itself and are fully compatible in terms of material used, structural integrity, and radiological safety.
- (6) Verify that experimental devices and equipment are designed:
 - (a) So that in all operational states, they will not cause unacceptable operational and radiological consequences to the reactor, other experiments, site personnel, the public or the environment;
 - (b) So that neither operation nor failure results in an unacceptable reactivity change to the reactor;
 - (c) With appropriate monitoring of experimental parameters in the reactor control room;
 - (d) With appropriate OLCs;
 - (e) So that the confinement or containment and shielding of the reactor are preserved if the devices or equipment penetrate the reactor boundaries;
 - (f) With protection systems that protect both the device and the reactor from any hazard arising from the experimental device.
- (7) Verify that procedures are in place to ensure that when the reactor itself is utilized to produce experimental results, all aspects of the design assumptions as reflected in the OLCs are met.
- (8) Verify that the use and handling of experimental devices is controlled by written procedures that state the responsibilities for those involved with experiments.
- (9) Determine that a procedure for reviewing proposed experiments and reactor utilization exists and that it includes:
 - (a) The provisions for integrating the experimental device with the reactor system;
 - (b) The selection and justification of the criteria used in the design of the experimental device with consideration given to specific items such as reactivity effects, temperature, pressure build-up, heat generation, and explosive materials;
 - (c) A safety assessment of the device itself and of its effects on the reactor and personnel;
 - (d) Review by the safety committee;
 - (e) Requirements for the preparation and validation of special operating and maintenance documentation;
 - (f) Requirements for any special personnel training in operating procedures, radiological rules and instructions associated with performance and handling of the experiment, and emergency arrangements;
 - (g) Commissioning and functional testing requirements;
 - (h) Decommissioning considerations and procedures;

- (i) Procedures to ensure adequate communication and intervention between operators and experimentalists;
- (j) Disposal of radioactive waste generated by the experimental programme;
- (k) Application of the quality assurance programme.

(10) Review the overall utilization and experiment programme to verify that:

- (a) Operating personnel are responsible for coordination and safety of all reactor experiments;
- (b) Operating personnel have available all information necessary for safe operation of experiments;
- (c) Close cooperation and communication exists between operating personnel and experimenters;
- (d) Procedures are established in advance for modifications of experiments;
- (e) The reactor manager or a designated member of the operating personnel has the authority to direct any operation of experimental equipment;
- (f) Approved methods and procedures are used for handling experiments;
- (g) Compliance with OLCs on experiments, including for production of radioisotopes, is observed;
- (h) Appropriate encapsulation and radiation protection controls are observed.

(11) Verify that the materials and the physical state(s) prohibited for irradiation are identified.

(12) Verify that the experimental devices and equipment not in use are removed or properly isolated and included in the ageing management, maintenance, inspection and testing programmes, and decommissioning plan as appropriate.

3.18. MANAGEMENT SYSTEM

3.18.1. Objective

The objective is to provide the basis for evaluating the adequacy of the management system and procedures at research reactors. The review team verifies that the responsibilities of the operating organization are defined and implemented.

References

SSR-3 [2]; Requirements 2, 4, 80 and 90; GSR Part 2 [25]; GS-G-3.1 [27]; IAEA-TECDOC-1801 [31].

3.18.2. Guidelines

- (1) Review the documentation of the facility and interview the staff of the operating organization to verify the following:
 - (a) A management system that covers safety, quality, health, security, environment, and economic considerations in an integrated manner is established and effectively implemented;
 - (b) The management system covers all life cycle stages of the facility and all processes in the facility and takes into account all statutory, regulatory and other requirements;
 - (c) The provisions of the management system are based on the four functional categories, namely:

- (i) Management responsibility;
 - (ii) Management of resources;
 - (iii) Management of processes and activities;
 - (iv) Measurement, assessment and improvement of the management system;
- (d) The management system has been considered by the safety committee and, where required, by the regulatory body;
- (e) There is evidence that audits and reviews are conducted to verify the effectiveness of the management system;
- (f) An individual has been identified who is responsible for the implementation of the programme.
- (2) Examine if the management system is well documented and covers as a minimum:
- (a) The policy statements of the organization;
 - (b) A description of the management system;
 - (c) A description of the structure of the organization;
 - (d) A description of the functional responsibilities, accountabilities, levels of authority and interactions of those managing, performing and assessing work;
 - (e) A description of the processes and supporting information that explain how work is to be prepared, reviewed, carried out, recorded, reported, assessed, and improved.
- (3) Verify if all the processes including the procedures, instructions, checklists and forms are clearly identified, implemented, periodically assessed and improved. Further verify that the management system includes the following processes:
- (a) Operation of the reactor;
 - (b) Maintenance and modification of items important to safety;
 - (c) Reactor safety;
 - (d) Radiation protection;
 - (e) Occupational health;
 - (f) Environmental protection;
 - (g) Emergency preparedness and response;
 - (h) Management of the interface between nuclear safety and security;
 - (i) Selection and training of personnel;
 - (j) Self-assessment and independent assessment;
 - (k) Management system review and improvement;
 - (l) Non-conformances and corrective and preventive actions;
 - (m) Control of documents and records of irradiations, waste and control of purchasing;
 - (n) Communication;
 - (o) Management of organizational change.
- (4) As part of a more in-depth review of the management system aspects:
- (a) Verify that safety policy statements are made available to all staff and that responsibilities and attitudes to safety are well known and understood, thus ensuring that safety is a clearly recognized value;
 - (b) Examine in detail the management system to assess whether it is in accordance with SSR-3 [2], GSR Part 2 [25] and GS-G-3.1 [27];
 - (c) Examine the management system to check that items important to safety, as identified by the safety analysis, are adequately covered by quality assurance procedures;

- (d) Verify that an up to date set of formally reviewed, approved and implemented procedures and instructions exists;
 - (e) Examine the records of the facility to check on the frequency, depth and quality of management system audits and reviews;
 - (f) Discuss with a variety of individuals from the operating organization the degree to which management system concepts are understood and followed by those with an influence on safety;
 - (g) Discuss the management system and its application with the persons responsible for it, to assess their understanding of their responsibilities and whether they have access to senior management in the organization independent of the reactor management;
 - (h) Ensure that the management system covers all aspects of safety including reactor operation, experiments, and emergency arrangements;
 - (i) Verify that the management system includes monitoring and evaluation of safety performance indicators;
 - (j) Ensure that violations, deficiencies and non-conformities necessitating corrective actions are properly identified, recorded and rectified, including those relating to the management system itself;
 - (k) Ensure that adequate management system records are kept and archived;
 - (l) Determine whether matters related to safety are controlled by well documented management procedures and arrangements, and the degree to which these are followed, reviewed and reinforced;
 - (m) Examine the extent to which managers ensure that staff are fully competent for their duties;
 - (n) Examine the degree to which management institutes a programme of maintenance practices by audit, review, and comparison.
- (5) Verify that activities related to the following topics are subject to particular controls established in written procedures:
- (a) Reactivity and criticality management, including core configuration changes, manipulation of equipment or material in the vicinity of the reactor core, and fuel storage;
 - (b) Core thermal safety, including changes in the core loading and geometry;
 - (c) Safety of experimental devices, including their design, construction, installation, commissioning, operation and decommissioning;
 - (d) Reactor modifications including their assessment and implementation, and resumption of reactor operation;
 - (e) Component and material manipulations, including fuel and objects which may cause interference (mechanical, thermal, electrical or nuclear) with the reactor;
 - (f) Human surveillance as applied to experimenters, visitors and trainees;
 - (g) Maintenance, periodic testing and inspections, including authorization, performance and verification of tests, repairs and changes;
 - (h) Commissioning testing and evaluation of results;
 - (i) Procurement of items important to safety;
 - (j) Recruitment, training and qualification of operating personnel;
 - (k) Preparation of safety related documents such as operating procedures and records.
- (6) Verify that adequate arrangements are available for non-radiation-related safety, in particular:

- (a) Inclusion of occupational safety aspects in work permits and work procedures;
 - (b) Procedures for working at heights, confined spaces, and in areas such as the reactor pool or spent fuel pool;
 - (c) Use of personal protective equipment such as helmets, safety shoes, gloves, eye shields, safety belts and coveralls for protection against chemicals;
 - (d) Procedures to work on electrical equipment, including lock out, isolation, earthing and electrical discharge;
 - (e) Railings and guards to prevent the inadvertent fall of personnel;
 - (f) Protection against internal hazards such as missile generation.
- (7) Verify that the management system ensures that:
- (a) Safety and security measures are implemented in such a manner that they do not compromise one another;
 - (b) Effective coordination and communication are established between the persons responsible for safety and security.

3.19. RADIATION PROTECTION

3.19.1. Objective

To provide a basis for evaluating the radiation protection programme, procedures and practices. The review team should verify that a radiation protection programme is established and implemented according to SSR-3 [2] and IAEA Safety Standards Series Nos GSR Part 3, Radiation Protection and Safety of Radiation Sources: International Basic Safety Standards [33], SSG-85, Radiation Protection and Radioactive Waste Management in the Design and Operation of Research Reactors [34] and GSG-7, Occupational Radiation Protection [35].

References

SSR-3 [2]: Requirements 57 and 84; GSR Part 3 [33]; SSG-85 [34]; GSG-7 [35].

3.19.2. Guidelines

- (1) By examining the radiation protection programme, determine whether the operating organization:
- (a) Has established a clear policy on radiation safety;
 - (b) Demonstrates support for radiation safety;
 - (c) Has provided adequate resources to the radiation protection organization.
- (2) Verify that a radiation protection programme has been established consistent with regulatory requirements and that it provides:
- (a) Control of radiation doses to individuals present on the site, including exposure limits and actions to be taken if limits are exceeded;
 - (b) Control of the amounts of radioactive substances released to the environment from operation of the reactor, including release limits and actions to be taken if limits are exceeded;
 - (c) Reference levels for radiation dose rates that take into account the particular characteristics of the reactor and experimental facilities;
 - (d) Monitoring and records of internal and external exposure of personnel, including lifetime doses and action reference levels;

- (e) Dose constraints;
 - (f) Reports to the regulatory body.
- (3) Determine that the radiation protection programme, in addition to administrative procedures to enable its implementation, includes:
- (a) Sufficient and appropriate instrumentation and equipment for personnel monitoring and protection;
 - (b) Workplace radiological monitoring and surveys (external radiation level and contamination monitoring), including alarms;
 - (c) Monitoring of airborne activity, including stack release;
 - (d) Monitoring of radioactivity, including specific radionuclides in process fluid systems (on-line and off-line);
 - (e) Monitoring of effluents prior to or during their discharge;
 - (f) Radiation monitoring at gates and other entrance/exits to monitor material being moved;
 - (g) Environmental radiological surveillance;
 - (h) Decontamination of personnel, equipment and structures;
 - (i) Compliance with applicable regulations for disposal of radioactive material;
 - (j) Compliance with applicable regulations for shipment of radioactive material;
 - (k) Maintenance of records and reporting of activity releases including dose estimates up to the site boundary;
 - (l) Record keeping of inventories of radiation sources;
 - (m) Adequate training in radiation protection practices;
 - (n) Periodic reviews, audits and updates to ensure that the programme's objectives are being satisfied;
 - (o) Facilities, equipment and instrumentation for contamination monitoring and for decontamination of personnel and equipment;
 - (p) Worker responsibility for radiation protection and safe work practices;
 - (q) Review by the radiation protection staff, operations staff, and new employees of lessons learned from past occurrences;
 - (r) Calibration of survey and monitoring equipment on a regular basis;
 - (s) Review and analysis of the hazards associated with experimental programmes and individual experiments;
 - (t) Clear, well written procedures for radiation protection evaluations with a mechanism for review, approval, and feedback.
- (4) Verify that the radiation protection staff operate with sufficient independence from the reactor management. Evaluate whether cooperation exists between the radiation protection staff and the operating personnel and experimenters in preparing operating and maintenance procedures and in providing direct assistance when radiation hazards are anticipated.
- (5) Verify that the operating organization has adequate and qualified radiation protection staff and support personnel with:
- (a) Clearly defined authority and functional responsibilities;
 - (b) Reporting lines independent of reactor management;
 - (c) A cooperative working relationship with reactor management;
 - (d) Access to levels of management that have authority to establish and enforce operational procedures.

- (6) Verify that for all operational states and accident conditions adequate provisions have been made for:
 - (a) Radiation shielding, ventilation, filtration and decay systems;
 - (b) Area radiation and airborne radioactivity monitoring instrumentation.
- (7) Determine that:
 - (a) The radiological zoning is adequately defined and implemented in the different locations in the research reactor;
 - (b) Access control is provided to areas that could have radiation levels in excess of those in normal operating areas;
 - (c) Reactor operation and facility modifications are planned, reviewed, supervised, and implemented from the perspective of avoiding unnecessary exposure to radiation and keeping unavoidable exposure in conformance with the principle of optimization of protection and safety.
- (8) Determine that:
 - (a) The operating organization conducts reviews or audits to ensure the correct implementation of the radiation protection programme, to ensure objectives are being met and that the programme is reviewed periodically and updated if necessary;
 - (b) Trends in personnel dose (individual and collective) and environmental discharges are monitored.
- (9) Verify that the interface between safety and security is taken into account in the implementation of the radiation protection programme (e.g. for access control).

3.20. RADIOACTIVE WASTE MANAGEMENT

3.20.1. Objective

To provide a basis for evaluating the monitoring and control programme of airborne and liquid effluents and solid waste and their release to the environment at research reactors. The reviewer verifies that provisions have been made to ensure that effluent releases are appropriately controlled and within regulatory limits, and that solid waste is appropriately managed, including its transport.

References

SSR-3 [2]: Requirements 59 and 85; SSR-6 (Rev.1) [32]; SSG-85 [34]; IAEA Safety Standards Series Nos GSR Part 5, Predisposal Management of Radioactive Waste [36] and GSG-9, Regulatory Control of Radioactive Discharges to the Environment [37].

3.20.2. Guidelines

- (1) Verify that effluent releases are within national limits or regulatory limits.
- (2) Review airborne and liquid effluent releases to determine whether:
 - (a) Procedures for control of effluent releases including radioactive gases and liquids are in place, and operations, radiation protection and management responsibilities are specified;

- (b) Installed sampling and monitoring equipment is appropriate for the effluents being monitored;
 - (c) Environmental monitoring is adequate and includes monitoring radioactivity in air, water (groundwater) and vegetation, and dose measurement, as appropriate.
 - (d) The radiological exposure of the general public from the release of radioactive effluents is in conformance with the principle of optimization of protection and safety;
 - (e) Periodic reviews of the release control systems (e.g., filter efficiency, sampling, monitoring equipment) are performed to ensure that they fulfil their intended purpose.
- (3) Verify that records of effluent releases and environmental monitoring are being maintained.
 - (4) Review the installed effluent monitoring systems, including number of instruments, type, location, interlocks, sensitivity, calibration, and maintenance. Check records of calibration and maintenance.
 - (5) Review the analytical procedures used to sample and evaluate effluent releases for accuracy and adequacy.
 - (6) Review the procedures for calibration of effluent monitoring instrumentation for adequacy and correctness. Check if up to date records of calibration are available.
 - (7) Verify that ventilation is adequate in all reactor facility areas occupied by operating personnel to ensure proper personnel protection from radioactive airborne effluents.
 - (8) Discuss with the personnel responsible for airborne and liquid effluent monitoring and surface contamination monitoring to ensure that they:
 - (a) Understand the proper use of sampling, filtering and monitoring equipment;
 - (b) Have an understanding of measurements taken by instruments for radiation counting of gaseous and liquid samples and surface contamination.
 - (9) Verify that consideration was given to minimizing effluent releases in the design of experimental facilities and devices.
 - (10) Verify that atmospheric dispersion data used for radioactive airborne effluent releases have not changed or been affected by new buildings or other construction in the near vicinity of the reactor facility.
 - (11) Verify the validity of the models used to evaluate the impact of surface water and groundwater contamination due to releases of radioactive liquid effluents.
 - (12) Determine whether adequate facilities exist commensurate with operation of the reactor and its experimental programmes for handling, segregation, treatment, conditioning, transportation, storage and disposal of solid and liquid radioactive waste.
 - (13) Determine whether consideration is given to minimizing gaseous, solid and liquid radioactive waste generation during the design, construction and operation of the reactor and experimental facilities.

- (14) Determine whether solid and liquid radioactive waste is handled and stored to maintain exposure to operating staff and health physics personnel according to the principle of optimization of protection and safety.
- (15) Determine whether periodic reviews of sampling and monitoring systems of solid and liquid radioactive wastes are conducted to ensure that they fulfil their intended purpose.
- (16) Determine whether:
- (a) Adequate interim storage arrangements are available for solid and liquid waste;
 - (b) Adequate written procedures exist for the handling, segregation, treatment, conditioning, transportation, storage and disposal of solid and liquid radioactive waste;
 - (c) Production of solid and liquid radioactive waste is reported periodically to the national regulatory body in accordance with its requirements;
 - (d) Classification and characterization, treatment, conditioning, transportation, storage and disposal of solid and liquid radioactive waste is carried out in accordance with the requirements of relevant local and national authorities;
 - (e) Appropriate records are maintained of the quantities, types and characteristics of stored solid and liquid radioactive waste and waste that is removed from the reactor site;
 - (f) Goals have been set by the operating organization to minimize generation of solid waste.
- (17) Review the solid and liquid waste monitoring systems, including number of instruments, type, location, interlocks, sensitivity, calibration and maintenance. Check records of calibration and maintenance.
- (18) Interview the personnel responsible for solid and liquid waste monitoring to ensure that they:
- (a) Understand the proper use of sampling and monitoring equipment;
 - (b) Have an understanding of measurements taken by instruments for radiation counting of solid waste samples.

3.21. EMERGENCY PLANNING

3.21.1. Objective

To provide a basis for evaluating emergency planning and emergency preparedness. The review team verifies that an emergency preparedness and response programme exists and that it is implemented through written procedures.

References

SSR-3 [2]: Requirements 32, 55, and 81; IAEA Safety Standards Series Nos GSR Part 7, Preparedness and Response for a Nuclear or Radiological Emergency [38]; GSG-2, Criteria for Use in Preparedness and Response to a Nuclear or Radiological Emergency [39]; EPR-Research Reactor [40].

3.21.2. Guidelines

- (1) Determine whether the reactor facility, within its management system, has an organizational framework for efficient on-site emergency arrangements and for effective coordination with off-site authorities, and whether:
 - (a) Roles and responsibilities relating to emergency preparedness and response are clearly specified and assigned to various positions as part of the routine organizational structure, and are documented in emergency plans and procedures;
 - (b) Processes are in place to assess hazards, to define goals in emergency preparedness and response, to develop adequate arrangements for achieving these goals, and to boost leadership and individual commitment to emergency preparedness and response;
 - (c) Resources (human, technical and financial) are allocated for emergency preparedness and response;
 - (d) Coordination mechanisms and effective relationships with off-site response organizations are established, as appropriate;
 - (e) A programme is in place to ensure the availability and reliability of all supplies, equipment, facilities, plans, procedures and other arrangements for emergency preparedness and response whenever they are needed.
- (2) Verify that the emergency plans and procedures are based on the accidents analysed in the SAR, including accidents postulated for the purposes of emergency preparedness and response, and that they include:
 - (a) Results of analysis of design basis accidents and design extension conditions;
 - (b) Results of any hazard assessment (including events of very low probability not considered in the design, and nuclear security events) to inform the need for emergency arrangements and any lessons from operating experience and from past emergencies, including conventional emergencies.
- (3) Verify that the emergency plans and procedures:
 - (a) Are periodically reviewed and updated (including consideration of feedback from drills and exercises);
 - (b) Take into account coordination with off-site response organizations;
 - (c) Are approved, including relevant updates, by the regulatory body, as appropriate.
- (4) Verify that the emergency plans and procedures contain the following information:
 - (a) Definitions, including emergency planning zone, site boundaries and emergency classifications based on severity levels of emergencies;
 - (d) A description of the reactor including authorized power level, fuel type and fission product inventory;
 - (e) A description of the location of the reactor facility including, as appropriate, surrounding population density, nearby industrial activity and access routes;
 - (f) Identification of the owner/operator;
 - (g) A statement of the objective of the emergency plan;
 - (h) Reactor utilization and operating mode;
 - (i) Emergency organization and responsibilities, including the names and duties of those individuals authorized to act as site emergency director;

- (j) Off-site organizations to be notified, including the names of specific officials;
 - (k) Arrangements and authority to activate the emergency organization;
 - (l) Response actions for each emergency classification by the operating organization and off-site organization;
 - (m) Description of emergency facilities and equipment, including locations;
 - (n) Checklist of assessment actions and recommended intervention levels;
 - (o) Conditions and indications for termination of the emergency;
 - (p) Arrangements and authority to terminate the emergency provision of information on the emergency to relevant off-site response organizations;
 - (q) Communication arrangements with all on-site and off-site personnel and authorities.
- (5) Verify that procedures and arrangements for implementing the essential functions for an effective on-site emergency response are in place and that they include:
- (a) Appropriate management of operations on the site, including for the transition from normal operations to operations under emergency conditions;
 - (b) Prompt identification of emergency conditions and declaration of emergency class, activation of on-site emergency response and notification of off-site notification point(s), including provision of sufficient information for effective off-site emergency response;
 - (c) Implementation of mitigatory actions on the site including provisions to obtain support from off-site emergency services;
 - (d) Assessment of hazards and possible development of hazardous conditions initially and throughout the emergency to inform decisions of necessary emergency response actions on the site;
 - (e) Implementation of necessary urgent protective actions to protect all persons present on the site in an emergency;
 - (f) Availability of suitable, reliable and diverse means of communication for use in taking protective actions on the site, and for communication with relevant off-site officials, taking into account that some means of communication might not be available during an emergency;
 - (g) Clearly marked escape routes with emergency lighting, assembly points, contact details of the responsible persons, and instructions that are displayed prominently;
 - (h) Effective communication with the public, ensuring consistency with the messages of relevant off-site response organizations;
 - (i) Protection of emergency workers responding on the site and assessment of hazardous conditions in which emergency workers might have to perform response functions;
 - (j) Documentation, protection, and preservation, to the extent practicable, of data and information important for an analysis of the emergency and for emergency response.
- (6) Verify that:
- (a) A training programme exists that ensures the personnel staffing the emergency response organization have the necessary knowledge, skills and abilities to perform their assigned functions;
 - (b) A drill and exercise programme exists that allows for emergency arrangements covering the range of functions to be fulfilled in an emergency response (including off-site organizations, as appropriate) to be tested at suitable intervals, and for necessary improvements to be identified;

- (c) The emergency training and drills conducted are commensurate with the potential magnitude of credible emergencies at the research reactor;
 - (d) Evaluation processes are applied to the training and drill and exercise programme against pre-established objectives.
- (7) Determine if all other essential infrastructure elements are in place to provide the capability for fulfilling the functions that are essential for effective emergency response.
- (8) Determine whether off-site organizations are familiar with the research reactor, adequately trained and prepared to respond to reactor emergencies.

3.22. PLANNING FOR DECOMMISSIONING

3.22.1. Objective

To provide a basis for evaluating safety aspects of research reactor planning for decommissioning.

References

SSR-3 [2]: Requirements 33 and 89; IAEA Safety Standards Nos GSR Part 6, Decommissioning of Facilities [41], SSG-47 Decommissioning of Nuclear Power Plants, Research Reactors and Other Nuclear Fuel Cycle Facilities [42] and WS-G-5.2, Safety Assessment for the Decommissioning of Facilities Using Radioactive Material [43].

3.22.2. Guidelines

- (1) Determine whether provisions to facilitate decommissioning activities have been taken into account during the design and construction of the reactor, experimental devices and modifications, and whether further requirements for decommissioning, including the preparation of a detailed decommissioning plan, have been implemented by the operating organization.
- (2) Determine whether, during the operational life of the reactor, the operating organization and reactor management have:
- (a) Maintained up-to-date reactor documentation;
 - (b) Recorded experience gained from handling contaminated and irradiated SSCs during maintenance to facilitate planning for decommissioning.
- (3) In the case that a detailed decommissioning plan has been prepared, determine whether it includes:
- (a) A detailed set of decommissioning activities and tasks, and their duration;
 - (b) A decommissioning organization with assigned responsibilities;
 - (c) A training programme;
 - (d) Details of contractor assistance;
 - (e) Facility radiological status including baseline information;
 - (f) As built drawings and related information;
 - (g) Radiation protection;
 - (h) Radioactive waste management;
 - (i) Accident analysis;
 - (j) Final radiation survey plan;

- (k) Cost estimate of the decommissioning method selected and funding provisions;
 - (l) Technical and environmental specifications in effect during decommissioning;
 - (m) Quality assurance provisions in place during decommissioning.
- (4) Determine whether the decommissioning plan includes an evaluation of appropriate decommissioning methods which might include:
- (a) Protective storage in an intact condition after removal of all fuel assemblies and readily removable radioactive components and waste;
 - (b) Provisions for dismantling, handling, storage and disposal of decommissioned radioactive equipment;
 - (c) Removal of all radioactive material and thorough decontamination of the remaining structures to permit unrestricted use.
- (5) Determine whether the decommissioning plan has been reviewed and approved by the regulatory body.
- (6) Determine whether reactor management is aware of the true technical and financial issues associated with the decommissioning process and whether the operating organization is capable of addressing them.

APPENDIX I.

TYPICAL MAIN MISSION REPORT FORMAT

1. BACKGROUND

- 1.1 History of facility
- 1.2 Summary description of facility and utilization programme
- 1.3 Summary of pre-INSARR mission
- 1.4 Objectives and scope of mission
- 1.5 Basis for review and documents provided by the counterpart

2. CONDUCT OF THE MISSION

- 2.1 Method of conducting the review
- 2.2 Review criteria
- 2.3 Results of the facility walkdown

3. CONCLUSIONS AND MAIN RECOMMENDATIONS

APPENDIX 1: ISSUE PAGES

ANNEX 1: AGENDA

ANNEX 2: LIST OF PERSONS MET DURING THE MISSION

ISSUE PAGE FORMAT

REVIEW AREA
ISSUE I1:
OBSERVATIONS:
BASIS AND REFERENCES:
POSSIBLE SAFETY CONSEQUENCES:
COUNTERPART'S VIEW AND MEASURES ON THE FINDINGS:
RECOMMENDATIONS: R1:
COMMENTS: C1:
SUGGESTIONS: S1:

APPENDIX II.

CHECKLISTS FOR WALKDOWN

A. HOUSEKEEPING AND GENERAL OBSERVATIONS		Comment
A.1	Note the physical condition of the major building structures of the facility.	
A.2	Note housekeeping and cleanliness throughout the facility	
A.3	Is portable equipment (ladders, scaffolding, heavy maintenance equipment, lifting and rigging equipment and fire protection equipment) stored in designated areas when not in use?	
A.4	Are working areas tidy with equipment and materials neatly laid out?	
A.5	Are equipment and systems free of significant amounts of dust and debris?	
A.6	Are equipment identification tags or labels provided for all equipment, and are they readable and affixed in a secure and durable manner?	
A.7	Are garbage / trash containers readily available and tidy?	
A.8	Is temporary storage controlled and limited?	
A.9	Are parts and materials in inactive work areas stored after work has been clearly completed?	
A.10	Are there any incompatible chemicals, flammable or toxic volatile materials stored in undesignated places in the facility?	
A.11	Are chemical storage and bottled gas storage correctly labelled for condition and content?	
A.12	Are radioactive material storage areas correctly identified and uncluttered, and are radiation fields identified?	
A.13	Are radiation and contamination monitors provided and functional at the entry/exit points of controlled areas?	
A.14	Are step off pads provided at the radiation zone change points?	
A.15	Are there pools of water or oil evident in any areas?	
A.16	Is there a monitoring programme for housekeeping, cleanliness and a fire protection equipment checks?	
A.17	Are the equipment and pipelines regularly painted (where applicable)?	
A.18	Are the equipment and piping properly supported?	
A.19	Are there any signs of corrosion of equipment and pipelines?	
A.20	If pipes and supports are of different materials, is a protective barrier installed between them?	

A. HOUSEKEEPING AND GENERAL OBSERVATIONS		Comment
A.21	Is the environment in different areas of the facility well maintained (e.g. temperature, humidity, ventilation)?	
A.22	Are cables routed in a tidy way and are sharp bends avoided?	
A.23	Are there any cracks/damage on the insulation of cables?	
A.24	Are junction boxes closed?	
A.25	Are civil structures properly protected by painting or coating?	
A.26	Are there any signs of spalling, cracks, or wetness in the plaster?	
A.27	Are there any signs of swelling, bowing or rusting rebar in concrete civil structures?	
A.28	Is there evidence of routine facility walkdown by the reactor manager?	

B. MAINTENANCE OF SSCs		Comment
B.1	Do managers and supervisors encourage reporting of minor deficiencies with equipment?	
B.2	Is there a formal work planning and work protection programme with suitable levels of procedures?	
B.3	Is the maintenance section appropriately sized and resourced?	
B.4	Are maintenance procedures satisfactory?	
B.5	Is there an adequate equipment maintenance, testing and inspection programme, and well archived historical record keeping?	
B.6	Is there an adequate spare parts system?	
B.7	Is there a satisfactory maintenance log and tracking mechanism?	
B.8	Is there a balance between preventive maintenance and corrective maintenance?	
B.9	Are the spare parts and consumables stored properly?	
B.10	Does the maintenance programme cover the equipment in long term storage?	

C. INDUSTRIAL HEALTH AND SAFETY		Comment
C.1	Is the protective equipment (respiratory and for maintenance activities) and clothing storage area well stocked and tidy?	
C.2	Are there eye wash facilities and eye protection equipment?	
C.3	Is there safe storage, use and disposal of hazardous chemicals?	
C.4	Are there confined space entry procedures?	
C.5	Are there procedures and training for the use of scaffolding, hoisting and crane equipment?	
C.6	Are the guards for rotating equipment (e.g. coupling and belt guards) in place and properly supported?	
C.7	Is there an industrial training programme (e.g. first aid, firefighting)?	
C.8	Are movable equipment and vehicles stored safely?	
C.9	Are there fire and smoke detectors, an alarm system and personnel response, particularly in the off shifts?	
C.10	Are fire detectors and fire extinguishers located near the fire loads?	
C.11	Do fire extinguishers have refill date and expiry labels?	
C.12	Are cage guards provided on tall ladders?	
C.13	Are safety rails provided around the pool and other high altitude areas?	
C.14	Are there water leakage detectors, an alarm system and personnel response, particularly in the off shifts?	

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