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Republic of the Philippines
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Commonwealth Avenue, Diliman, Quezon City

CPR PART 30

SAFETY REQUIREMENTS FOR RESEARCH REACTORS

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CPR PART 30

SAFETY REQUIREMENTS FOR RESEARCH REACTORS

I. GENERAL PROVISIONS

Section 1. *Purpose*

(a) This Part is promulgated pursuant to Republic Act No. 5207, otherwise known as the "Atomic Energy Regulatory and Liability Act of 1968", as amended, to establish the regulatory requirements for the protection of the health and safety of the workers and the general public with respect to the operation of research reactors, including critical and subcritical assemblies.

(b) This Part provides a basis for safety and for safety assessment for all stages in the lifetime of research reactors, including critical and subcritical assemblies.

(c) This Part is intended for use by organizations involved in the site evaluation, design, manufacture, construction, commissioning, operation, modification, maintenance, and decommissioning of research reactors, including critical and subcritical assemblies; in safety analysis, verification, and review, and in the provision of technical support.

Section 2. *Scope*

(a) The safety requirements established in this Part are applicable for research reactors, including critical and subcritical assemblies.

(b) All the requirements established in this Part are to be applied unless it can be justified that, for a specific research reactor, including critical and subcritical assembly, the application of certain requirements may be graded. Section 18 of this Part sets out factors to be considered in deciding whether the application of certain requirements established here may be graded.

(c) The requirements in this Part do not apply to any of the following:

(1) Matters relating to nuclear security other than the interfaces between nuclear safety and nuclear security, addressed in Section 99 or to the State system of accounting for, and control of nuclear material;

(2) Conventional industrial safety matters that under no circumstances could affect the safety of the research reactor;

(3) Non-radiological impacts arising from the operation of the research reactor facility.

Section 3. ***Definition of Terms***

As used in this Part:

(a) ***"Accident"*** means any unintended event, including operating errors, equipment failures and other mishaps, the consequences or potential consequences of which are not negligible from the point of view of protection or safety;

(b) ***"Accident conditions"*** means deviations from normal operation that are less frequent and more severe than anticipated operational occurrences, and which comprise design basis accidents and design extension conditions;

(c) ***"Activity"*** means the design, manufacture, construction, import, export distribution, sale, loan, commissioning, use, operation, maintenance, repair, transfer, decommissioning or possession of nuclear materials and radiation sources for industrial, energy production, education, research, agriculture and medical purposes; the transport of radioactive materials; the mining and processing of radioactive ores; the closing down of associated facilities; the clean-up of sites affected by the residues from the past activities; and radioactive waste management activities such as the discharge of effluents and such other activities as the PNRI shall from time to time determine;

(d) ***"Anticipated Operational Occurrence (AOO)"*** means an operational process deviating from normal operation which is expected to occur at least once during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety or lead to accident conditions;

(e) ***"Assessment"*** means the process, and the result, of analyzing systematically and evaluating the hazards associated with sources and practices, and associated protection and safety measures;

(f) ***"Authorized limit"*** means a limit on a measurable quantity, established or formally accepted by the Authority;

(g) ***"Commissioning"*** means the process during which systems and components of facilities and activities, having been constructed, are made operational and verified to be in accordance with the design and to have met the required performance criteria;

(h) ***"Common cause failure"*** means failure of two or more SSCs due to a single specific event or cause;

(i) ***"Critical assembly"*** means an assembly containing fissile material intended to sustain a controlled fission chain reaction at a low power level, used to investigate reactor core geometry and composition;

(j) ***"Design Basis Accident (DBA)"*** means a postulated accident leading to accident conditions for which a facility is designed in accordance with established design criteria and conservative methodology, and for which releases of radioactive material are kept within acceptable limits;

(k) ***"Decommissioning"*** means administrative and technical actions taken to allow the removal of some or all of the regulatory controls from a facility;

(l) ***"Defence in depth"*** means a hierarchical deployment of different levels of diverse equipment and procedures to prevent the escalation of anticipated operational occurrences and to maintain the effectiveness of physical barriers placed between a radiation source or radioactive material and workers, members of the public or the environment, in operational states and, for some barriers, in accident conditions;

(m) ***"Design Extension Conditions (DEC)"*** means postulated accident conditions that are not considered for design basis accidents, but that are considered in the design process of the facility in accordance with best estimate methodology, and for which releases of radioactive material are kept within acceptable limits;

(n) ***"Deterministic Safety Analysis"*** means an analysis using, for key parameters, single numerical values, taken to have a probability of 1, leading to a single value of the result. This safety analysis is performed under specific predetermined assumptions concerning the initial operational state and the initiating event, with specific sets of rules and acceptance criteria. Deterministic analyses can be conservative or best estimate;

(o) ***"Disposal"*** means emplacement of waste in an appropriate facility without the intention of retrieval;

(p) ***"Diversity"*** means the presence of two or more redundant systems or components to perform an identified function, where the different systems or components have different attributes so as to reduce the possibility of common cause failure including common mode failure;

(q) ***"Dose limit"*** means the value of the effective dose or the equivalent dose to individuals in planned exposure situations that is not to be exceeded;

(r) ***"Facility"*** means, for the purpose of this document, is the installation containing research reactor, including critical and subcritical assembly;

(s) ***"Fuel element"*** means a rod (or other form) of nuclear fuel, its cladding and any associated components necessary to form a structural entity;

(t) ***"Item important to safety"*** means an item that is part of a safety group and/or whose malfunction or failure could lead to radiation exposure of the site personnel or members of the public;

(u) ***"License"*** means a legal document issued by the PNRI granting authorization to perform specified activities related to a facility or activity;

(v) ***"Licensee"*** means a holder of a current and valid license issued by the PNRI pursuant to this Part. The licensee is the person or organization possessing all necessary licenses and having overall responsibility for the research reactor, including critical and subcritical assembly;

(w) ***"Maintenance"*** means the organized activity, both administrative and technical, of keeping SSCs in good operating condition, including both preventive and corrective (or repair) aspects;

(x) ***"Monitoring"*** means the measurement of dose or contamination for reasons related to the assessment or control of exposure to radiation or radioactive substances, and the interpretation of the results;

(y) ***"Normal operation"*** means operation within specified operational limits and conditions;

(z) ***"Operational Limits and Conditions (OLCs)"*** means a set of rules setting forth parameter limits, the functional capability and the performance levels of equipment and personnel approved by the PNRI for safe operation of an authorized facility;

(aa) ***"Operational states"*** means the states defined under normal operation and anticipated operational occurrences;

(bb) ***"Periodic Safety Review (PSR)"*** means a systematic reassessment of the safety of an existing facility or activity carried out at regular intervals to deal with the cumulative effects of ageing, modifications, operating experience, technical developments and siting aspects, and aimed at ensuring a high level of safety throughout the service life of the facility or activity;

(cc) ***"Postulated Initiating Events"*** means an event identified during design as capable of leading to anticipated operational occurrences or accident conditions;

(dd) ***"Probabilistic Safety Analysis"*** means a comprehensive and structured approach to identifying failure scenarios, constituting a conceptual and mathematical tool for deriving numerical estimates of risk;

(ee) ***"Protection system"*** means a system which monitors the operation of a reactor and which, on sensing an abnormal condition, automatically initiates actions to prevent an unsafe or potentially unsafe condition; the „system‟ in this case encompasses all electrical and mechanical devices and circuitry, from sensors to actuation device input terminals;

(ff) ***"Radiation protection"*** means the protection of people from the effects of exposure to ionizing radiation, and the means for achieving this;

(gg) ***"Radioactive material"*** means any material designated in national law or by the PNRI as being subject to regulatory control because of its radioactivity which includes sealed and unsealed sources and radioactive waste;

(hh) ***"Radioactive substance"*** means any material exhibiting radioactivity; emitting or relating to the emission of ionizing radiation or particles.

(ii) ***"Radioactive waste"*** means waste that contains, or is contaminated with, radionuclides at concentrations or activities greater than clearance levels as established by the Authority;

(jj) ***"Radioactive waste management"*** means all administrative and operational activities, involved in the handling, pretreatment, treatment, conditioning, transport, storage and disposal of radioactive waste;

(kk) ***"Reactivity"*** means a measure of the deviation from criticality of a nuclear chain reacting medium, such that positive values correspond to a supercritical state and negative values correspond to a subcritical state;

(ll) ***"Reactor management"*** means the licensee of the relevant research reactor, authorized by the Authority to operate the reactor;

(mm) ***"Reactor operator"*** means a person authorized to carry out operations in the control room and in the field;

(nn) ***"Redundancy"*** provision of alternative (identical or diverse) SSCs, so that anyone can perform the required function regardless of the state of operation or failure of any other;

(oo) ***"Research reactor"*** means a nuclear reactor used mainly for the generation and utilization of neutron flux and ionizing radiation for research and other purposes, including experimental facilities associated with the reactor and storage, handling and related to safe operation of the research reactor. Facilities commonly known as critical assemblies are included;

(pp) ***"Safe state"*** means state of the reactor facility, following an anticipated operational occurrence or accident conditions, in which the reactor is subcritical and the main safety functions can be ensured and maintained stable for a long time;

(qq) ***"Safety analysis"*** means an evaluation of the potential hazards associated with the operation of a facility or the conduct of an activity;

(rr) ***"Safety classification"*** means the assignment to a limited number of safety classes of systems and components and other items of equipment on the basis of their functions and their safety significance;

(ss) ***"Safety committee"*** means a group of experts from the licensee convened to advise on the safety of operation of an authorized facility;

(tt) ***"Safety culture"*** means the assembly of characteristics and attitudes in organizations and individuals, which establishes that, as an overriding priority, protection and safety issues receive the attention warranted by their significance;

(uu) ***"Safety feature"*** (for design extension conditions) means item that is designed to perform a safety function for or that has a safety function for design extension conditions;

(vv) ***"Safety limits"*** mean limits on operational parameters within which an authorized facility has been shown to be safe. Safety limits are operational limits and conditions beyond those for normal operation;

(ww) ***"Safety system"*** means a system important to safety, provided to ensure the safe shutdown of the reactor or the residual heat removal from the core, or to limit the consequences of anticipated operational occurrences and design basis accidents;

(xx) ***"Safety system settings"*** mean settings for levels at which safety systems are automatically actuated in the event of anticipated operational occurrences or design basis accidents, to prevent safety limits from being exceeded;

(yy) ***"Shift supervisor"*** means a person responsible and in charge of the operation shift;

(zz) ***"Single Failure"*** means a failure which results in the loss of capability of aa single system or component to perform its intended safety function(s), and any consequential failure(s) which result from it;

(aaa) ***"Spent fuel"*** means a nuclear fuel removed from a reactor following irradiation, which is no longer usable in its present form because of depletion of fissile material, buildup of poison or radiation damage;

(bbb) ***"Structures, Systems and Components (SSCs)"*** means a general term encompassing all of the elements of a facility or activity that contribute to protection and safety, except human factors.

(1) Structures are passive elements such as buildings, vessels and shielding.

(2) Systems are several components assembled in such a way as to perform a specific active function.

(3) Components are discrete elements of a system such as wires, transistors, integrated circuits, motors, relays, solenoids, pipes, fittings, pumps, tanks and valves.

(ccc) ***"Subcritical assembly"*** means a device which contains a mass of fissile materials that is insufficient to sustain a fission chain reaction and requiring a neutron source for its control and operation;

Section 4. ***Interpretation***.

Except as specifically authorized by the PNRI Director in writing, no interpretation of the meaning of the requirements in this Part shall be recognized to be binding upon PNRI.

Section 5. ***Communications.***

All correspondence, reports, applications, and other communications from the applicant or licensee to the PNRI concerning the regulations in this Part or individual license conditions shall be addressed to:

The Office of the Director
Philippine Nuclear Research Institute
Commonwealth Avenue,
Diliman, Quezon City

Section 6. ***Applicability of other Regulations and Requirements, and Resolution of Conflicts.***

(a) This Part shall be applied in conjunction with the radiation protection and safety requirements of CPR Part 3 – “Standards for Protection against Radiation”, the safe transport requirements of CPR Part 4 – “Regulations on the Safe Transport of Radioactive Material in the Philippines”, siting requirements of CPR Part 5 – “Requirements for Siting of Nuclear Installations”, general licensing process requiremenst for nuclear installations of CPR Part 7 – “Licensing of Nuclear Installations and the security requirements of CPR Part 26 – “Security of Radioactive Sources”, and CPR Part 27 – “Security Requirements in the Transport of Radioactive Materials” and other applicable regulations.

(b) This Part are in addition to, and not in place of, other applicable national and local laws and regulations.

(c) This Part does not relieve the applicant or licensee from complying with the applicable laws of the Republic of the Philippines and regulations of other responsible government agencies.

(d) If a conflict exists between requirements contained herein and other laws or regulations, the PNRI shall be notified of such conflict to initiate steps towards resolution.

(e) Nothing in this Part shall be construed as restricting any actions that may otherwise be necessary for protection and safety.

II. MANAGEMENT FOR SAFETY

Section 7. ***Responsibilities in the Management for Safety***

(a) The licensee shall have the prime responsibility for the safety of the research reactor over its lifetime from site evaluation, design, construction, commissioning, operation, including utilization and modification, and decommissioning, until its release from regulatory control.

(b) To ensure rigor and thoroughness at all levels of the staff in the achievement and maintenance of safety, the licensee shall:

(1) Establish and implement safety policies and shall ensure that safety matters are given the highest priority;

(2) Define clearly the responsibilities and accountabilities with corresponding lines of authority and communication;

(3) Ensure that it has sufficient staff with appropriate qualifications and training at all levels;

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

(5) Review, monitor and audit all safety related matters annually, and shall take appropriate corrective actions where necessary;

(6) Develop and sustain a strong safety culture, and shall prepare a statement of safety policy and safety objectives, which is disseminated to and understood by all staff.

(c) Whenever a change of stage in the lifetime of a research reactor is to be initiated by the licensee, it shall submit a detailed demonstration of safety, which shall include an adequate safety analysis, for review and assessment by the PNRI before the project is authorized to progress to the next stage.

(d) The licensee shall:

(1) submit to the PNRI any information requested in a timely manner;

(2) be responsible for making arrangements with vendors and suppliers to ensure the availability of any information that has been requested by the PNRI;

(3) be responsible for informing the PNRI of any additional new information on the research reactor and of any changes to information submitted previously; and

(4) provide complete and accurate information to the PNRI.

Section 8. ***Safety Policy***

(a) The licensee shall establish and implement safety policies that give safety the highest priority. The safety policy shall:

(1) promote a strong safety culture, including a questioning attitude and a commitment to excellent performance in all activities important to safety;

(2) stipulate clearly the leadership role of the highest level of management in safety matters. Senior management shall be responsible for communicating

and implementing the provisions of the safety policy throughout the organization; and

(3) include a commitment to achieving enhancements in operational safety. The strategy of the licensee for enhancing safety and for finding more effective ways of applying and, where feasible, improving existing standards shall be continuously monitored, periodically revised and supported by means of a clearly specified Program with clear objectives and targets.

Section 9. ***Safety Committee***

(a) A safety committee that is independent from the reactor manager shall be established to advise the operating organization on: (i) the safety assessment of design, commissioning and operational issues; and (ii) relevant aspects of the safety of the reactor and the safety of its utilization.

(b) Members of the safety committee shall be experts in different fields associated with the design and operation of research reactor.

(c) The safety committee shall be fully functioning before the design of the research reactor begins. The list of items that the safety committee is required to consider, provide advice on, or recommend approval of shall also be established. Such a list shall include, among other things, the following:

(1) The design of SSCs and in particular the design and qualification of nuclear fuel elements and reactivity control elements;

(2) Safety documents and their modifications;

(3) Proposed new tests, experiments, equipment, systems or procedures that have significance for safety;

(4) Proposed modifications to items important to safety and changes in experiments that have implications for safety;

(5) Violations of the OLCs, of the license and of procedures that are significant to safety;

(6) Events that are required to be reported or that have been reported to the PNRI;

(7) Periodic reviews of the operational performance and the safety performance of the research reactor facility;

(8) Reports on routine radioactive discharges to the environment;

(9) Reports on radiation doses to the personnel at the facility and to the public;

(10) Reports to be provided to the PNRI; and

(11) Reports on regulatory inspections.

Section 10. ***Management System***

(a) The licensee shall:

(1) Establish, implement, assess, use and continuously improve integrated management system requirements for the stages of site evaluation, design, construction, commissioning, operation, utilization, modification and decommissioning, in a safe manner and within the limits and conditions that are specified in the OLCs and established in the license.

(2) The licensee shall ensure that the management system for the research reactor is established and implemented by means of a graded approach based on the importance to safety of each item, service or process.

(b) The management system shall:

(1) Be developed and established at a time consistent with the schedule for accomplishing activities at all stages in the lifetime of the research reactor. In particular, activities for site investigation, which are usually initiated a long time before the establishment of a project, shall be covered by the management system;

(2) Include all the elements of management so that processes and activities important to safety are established and conducted in accordance with relevant requirements, including those relating to leadership, protection of health, human performance, emergency preparedness and response, protection of the environment, security and quality;

(3) Identify and include the following requirements:

(i) The national statutory and regulatory requirements;

(ii) The relevant IAEA safety standards; and

(iii) Any requirements formally agreed with interested parties.

(4) Be based on four functional categories:

(i) Management responsibility;

(ii) Management of resources;

(iii) Management of processes and activities;

(iv) Measurement, assessment and improvement of the management system

(c) The documentation of the management system shall be reviewed and made subject to approval at appropriate levels of management in the licensee and shall be submitted to the PNRI for review and assessment as requested.

III. ADMINISTRATIVE REQUIREMENTS

Section 11. ***Responsibilities of the Licensee***

The licensee shall:

(a) Have the prime responsibility for safety in the operation of the research rector, including critical and subcritical assembly. This prime responsibility shall cover all the activities relating to the operation directly and indirectly, including activities for experiments. The responsibility of the licensee for safety shall not be delegated.

(b) Establish an appropriate management system for the research reactor, including critical and subcritical assembly and shall provide for all necessary infrastructure for the conduct of reactor operations.

(c) Assign direct responsibility and authority for the safe operation of the reactor to the reactor manager. The organization for reactor operation shall include the reactor manager and the operating personnel.

(d) Ensure that adequate provision is made for all functions relating to the safe operation and utilization of the research reactor, including critical and subcritical assembly.

(e) Establish, in accordance with the management system, the functions and responsibilities for the key positions in the organization for reactor operation. Staff controlling changes in the operational status of the research reactor, including critical and subcritical assembly, shall be required to hold a valid license.

(f) Identify positions to be authorized and establish processes for staff to achieve this authorization including adequate training. In the assessment of an individual's competence and suitability as a basis for the issuance of a license by the PNRI, or other competent authority, documented criteria shall be used. At the minimum, the reactor manager, the shift supervisors, and the reactor operators shall hold a valid license issued by the PNRI, or other competent authority.

(g) Prepare and issue specifications and procedures in accordance with the classification of SSCs and the management system, in particular for the procurement, manufacturing, loading, utilization, unloading, storage, movement and testing of items important to safety, including fuel and core components and other fresh or irradiated fissile material.

(h) Prepare periodic summary reports on matters relating to safety as required by the PNRI and shall submit these reports to the safety committee and to the PNRI if so required.

(i) Be responsible to ensure the following:

   (1) Safety policies are issued and clearly understood by everyone.

   (2) The establishment of its advisory safety committee.

(3) The design enables the reactor to be operated safely, and the reactor is constructed in accordance with the approved design.

(4) An adequate safety analysis report is prepared and kept up to date.

(5) The commissioning process demonstrates that the design requirements have been met and that the reactor can be operated in accordance with the design assumptions.

(6) A system for reporting and reviewing abnormal events is established and operated.

(7) On-site emergency arrangements, including the emergency plan and procedures, are established and maintained.

(8) The research reactor is operated and maintained in accordance with the safety requirements by suitably qualified and experienced personnel recognized by the PNRI.

(9) Personnel with responsibilities relating to safe operation are adequately trained, and a training and retraining Program is established, implemented and kept up to date and periodically reviewed to verify its effectiveness.

(10) Adequate resources, facilities and services are made available during operation.

(11) Information on events with safety significance that are required to be reported to the PNRI, including any assessments of such events and the corrective actions intended, is submitted to the PNRI.

(12) Safety culture is fostered in the organization to ensure that the attitudes of personnel and the actions and interactions of all individuals and organizations are conducive to safe conduct of activities during operation of the facility.

(13) An integrated management system is established and implemented, in accordance with a graded approach.

(14) The reactor management is provided with sufficient authority and resources to enable it to fulfil its duties effectively.

(15) The research reactor is operated and maintained in accordance with the OLCs and operating procedures.

(16) The fissile material and radioactive material that are utilized or generated are controlled.

(17) Operating experience, including information on operating experience at similar research reactors, is carefully examined for any precursor signs of tendencies adverse to safety so that corrective actions can be taken before serious adverse conditions arise and recurrences can be prevented.

(18) An exclusion Program for foreign objects is implemented and monitored, in accordance with regulatory requirements.

Section 12. *Structure and Functions of the Licensee*

(a) The licensee organizational structure, functions, roles and responsibilities of its personnel shall be established, documented, and made available to staff and if required to PNRI.

(b) Functional responsibilities, lines of authority, and lines of internal and external communication for the safe operation of the research reactor in all operational states and in accident conditions shall be clearly specified in writing.

(c) The organizational structure and the arrangements for discharging responsibilities shall be documented in the SAR and made available to the staff and, if required, to the PNRI. The structure of the operating organization shall be specified so that all roles that are critical for safe operation are specified and described. Proposed organizational changes to the structure and associated arrangements, which might be of importance to safety, shall be analyzed in advance by the licensee and submitted to the PNRI for approval.

(d) The licensee shall be responsible for ensuring that the necessary knowledge, skills, attitudes, and safety expertise are sustained at the research reactor, and that long-term objectives for human resources are met and knowledge preservation policies are developed.

Section 13. *Operating Personnel*

(a) Reactor Manager

The reactor manager shall:

(1) Have overall responsibility for all aspects of operation, training, maintenance, periodic testing, inspection, utilization and modification of the reactor.

(2) Clearly document the duties, responsibilities, necessary experience and training requirements of operating personnel, and their lines of communication.

(3) Specify the minimum staffing requirements for the disciplines required to ensure safe operation for all operational states of the research reactor in accordance with the OLCs. The person with qualification and responsibility for the direct supervision of the operation of the reactor shall be clearly identified at all times. The availability of the staff that would be required to deal with accident conditions shall also be specified

(4) Be responsible for ensuring that the staff selected for reactor operation are given the training and retraining necessary for the safe and efficient operation of the reactor and that this training and retraining is appropriately

evaluated. Adequate training in the procedures to be followed in both operational states and accident conditions shall be conducted.

(5) Ensure that the operating personnel, including technical support personnel and experimenters have suitable training in radiation protection before the start of their duties. Periodic refresher training in operational radiation protection shall be carried out.

(6) Approve the detailed Program for the operation and experimental use of the research reactor.

(7) Be responsible for, and shall decide for, all the activities associated with core management and fuel handling and the handling of any other fissile material.

(8) Periodically review the operation of the research reactor, including experiments, and shall take appropriate corrective actions in respect of any problems identified.

(9) Seek the advice of the safety committee or shall call upon advisors to review important safety issues arising in the commissioning, operation, maintenance, periodic testing and inspection, and modification of the reactor and experiments.

(b) Operating Personnel

(1) The operating personnel shall operate the facility in accordance with the approved OLCs and operating procedures.

(2) Every licensed or authorized member of the operating personnel shall have the authority to shut down the reactor in the interest of safety.

(3) A maintenance group shall be established by the licensee to implement the programs for maintenance, periodic testing and inspection.

(c) Radiation Protection Personnel

A radiation protection group shall be established to prepare and implement a radiation protection Program and to advise the reactor management and the licensee on matters relating to radiation protection.

(d) Additional Support Personnel

(1) The licensee shall make provision as needed for additional technical personnel such as training officers, safety officers and reactor chemists.

(2) The licensee shall arrange for the provision of assistance by contractor personnel as required.

(e) Safety committee

(1) The safety committee shall advise the reactor manager on the safety aspects of the day to day operation and utilization of the reactor including safety of proposed experiments and modifications.

(2) The reactor manager shall have the authority to refuse or delay the performance of an experiment or a modification that he or she considers not safe and to request for additional review.

Section 14. ***Training, Retraining and Qualification of Personnel***

The licensee shall:

(a) Ensure that safety related functions are performed by suitably qualified, competent, and fit-for-duty personnel.

(b) Clearly define the requirements for qualification and competence to ensure that personnel performing safety related functions are capable of safely performing their duties.

(c) Select suitably qualified personnel and give the necessary training and instruction to enable them to perform their duties correctly for different operational states and in accident conditions, in accordance with the appropriate procedures.

(d) Establish and maintain suitable training and retraining programs for the operating personnel. The training Program shall include provision for periodic confirmation of the competence of personnel, which shall be documented, and for refresher training every three years. The refresher training shall also include retraining provision for personnel who have had extended absences from their authorized duties. The training shall emphasize the importance of safety in all aspects of reactor operation and shall promote safety culture.

(e) Establish and maintain procedures for the validation of the training to verify its effectiveness and the qualification of the staff.

IV. SAFETY VERIFICATION

Section 15. ***Safety Assessment***

(a) The adequacy of the design of the research reactor, including critical and subcritical assembly, shall be verified in accordance with the management system by means of comprehensive deterministic safety analysis and complementary probabilistic safety analysis as appropriate.

(b) The licensee shall conduct safety assessment commencing at an early stage in the design process. Deterministic safety analysis shall be the primary tool for safety assessment of the facility.

(c) The design shall be validated by independent verification by individuals or groups independent from those who originally performed the design work.

(d) Verification, validation, and approval of the reactor design shall be completed as soon as practicable in the design and construction processes, and in any case before commissioning of the facility is commenced.

(e) Safety assessment shall be part of the design process, with iterations made between the design activities and the confirmatory analytical activities and with increases in the scope and the level of detail of the safety assessment as the design progresses.

(f) The safety assessment shall be continued throughout all the stages of the research reactor's lifetime and shall be conducted in accordance with the potential magnitude and nature of the hazards associated with the facility or activity.

(g) The safety assessments and periodic safety reviews shall be documented to facilitate their evaluation.

(h) The licensee shall report to the PNRI the confirmed findings of the periodic safety review that have implications for safety.

(i) The licensee shall verify by analysis, surveillance, testing and inspection that the physical state of the facility, including experimental devices and facilities, is as described in the safety analysis report and other safety documents, and that the facility is commissioned and operated in accordance with safety requirements and the safety analysis and OLCs.

Section 16. ***Periodic Safety Review***

(a) The licensee shall perform systematic periodic safety reviews of the research reactor, including critical and subcritical assembly, throughout its operating lifetime, with account taken of operating experience, the cumulative effects of ageing, applicable safety standards and safety information from all relevant sources.

(b) Based on the results of the periodic safety review, the licensee shall take any necessary corrective actions and shall consider making justified modifications to enhance safety.

(c) The periodic safety review shall contain the information specified in Section 21 (d) of CPR Part 7.

(d) The safety assessments and periodic safety reviews shall be documented to facilitate their evaluation.

(e) The licensee shall report to the PNRI the confirmed findings of the periodic safety review that have implications for safety.

Section 17. *Safety Analysis of the Design*

(a) The licensee shall conduct a safety analysis of the design for a research reactor facility.

(b) These analyses shall be used:

(1) As the design basis for items important to safety;

(2) For the selection of the OLCs for the reactor;

(3) For the development of operating procedures, inspection and periodic testing programs, record keeping practices, maintenance schedules, proposals for modifications and emergency planning;

(4) In the case of DEC, to identify the additional accident scenarios to be addressed in the design and to plan practicable provisions for the prevention of such accidents or mitigation of their consequences if they do occur.

(c) The safety analysis shall provide assurance that defence in depth has been implemented and uncertainties have been given adequate consideration in the design.

(d) The scope of the safety analysis shall include:

(1) Characterization of the postulated initiating events that are appropriate;

(2) Analysis of event sequences and evaluation of the consequences of the postulated initiating events;

(3) Comparison of the results of the analysis with radiological acceptance criteria and design limits;

(4) 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;

(5) Specification of the design extension conditions and of how they are addressed;

(6) Determination of the OLCs for normal operation;

(7) The analysis of safety systems and the engineered safety features and the safety features for design extension conditions;

(8) The analysis of the means of confinement.

(e) For each postulated initiating event, qualitative and quantitative information about the following aspects shall be considered in the safety analysis:

(1) The input parameters, initial conditions, boundary conditions, assumptions, models, uncertainties and codes used;

(2) The sequence of events and the performance of reactor systems;

(3) The sensitivity to single failure modes and common cause failures;

(4) The sensitivity to human factors;

(5) Analysis of transients;

(6) The identification of damage states;

(7) The derivation of source terms;

(8) The evaluation of radiological consequences.

(f) For each accident sequence considered, the extent to which the safety systems and any operable process systems are required to function under accident conditions shall be indicated.

(g) Where applicable, the safety analysis shall include consideration of the experimental devices with regard to both their own safety aspects and their effects on the research reactor.

(h) The applicability of the methods of analysis, the analytical assumptions and the degree of conservatism used in the design of the research reactor shall be updated and verified for the as built facility.

Section 18. *Safety Analysis Report*

(a) The safety analysis report for a research reactor, including critical and subcritical assembly shall:

(1) Be prepared by the licensee;

(2) Provide a justification of the site and the design and shall provide a basis for the safe operation;

(3) Be reviewed and assessed by the PNRI before it is authorized to progress to the next stage;

(4) Be periodically updated over the research reactor’s operating lifetime to reflect modifications made to the facility and on the basis of experience and in accordance with regulatory requirements;

(5) Contain the information specified in Section 12 of CPR Part 7;

(6) Include information to demonstrate compliance with national legislation and requirements issued by the PNRI. The level of detail of the information to be presented in the safety analysis report shall be determined using a graded approach;

(7) Describe the expected experimental use and shall provide information demonstrating that provisions have been made to ensure that the experimental facilities and experiments are within the safety criteria established for the research reactor, the staff and the public;

(8) Cite references that may be necessary for its thorough review and assessment. This reference material shall be readily available to the PNRI and shall not be subject to any classification or limitation that would prevent its adequate review and assessment

(b) The safety analyses in the safety analysis report shall form the basis for the OLCs for the research reactor, including critical and subcritical assembly.

Section 19. ***Use of Graded Approach***

(a) The use of a graded approach shall not be considered as a means of waiving safety requirements and shall not compromise safety.

(b) The graded approach shall be used in determining that scope and level of details of the safety assessment carried out at each particular stage of the research reactor life consistent with the magnitude of the possible radiation risks arising from the facility.

(c) Grading of the application of requirements in this Part shall be justified by the applicant or licensee and supported by safety analysis or engineering judgement. All instances of grading shall be subject to review by the PNRI.

(d) The factors to be considered in deciding whether the application of certain requirements established here may be graded include:

(1) The reactor power;

(2) The potential source term;

(3) The amount and enrichment of fissile and fissionable material;

(4) Spent fuel elements, high pressure systems, heating systems and the storage of flammable materials, which may affect the safety of the reactor;

(5) The type of fuel elements;

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

(7) The amount of reactivity that can be introduced and its rate of introduction, reactivity control, and inherent and additional safety features (including those for preventing inadvertent criticality);

(8) The quality of the containment structure or other means of confinement;

(9) The utilization of the reactor (experimental devices, tests and reactor physics experiments);

(10) The site evaluation, including external hazards associated with the site and the proximity to population groups;

(11) The ease or difficulty in changing the overall configuration.

Section 20. ***Site Evaluation***

The licensee shall conduct a detailed site evaluation in accordance to CPR Part 5, “Requirements for Siting of Nuclear Installations”.

V. RESEARCH REACTOR DESIGN

PRINCIPAL TECHNICAL REQUIREMENTS

Section 21. ***Main Safety Functions***

(a) The design for a research reactor facility shall ensure the fulfilment of the following main safety functions for the research reactor for all states of the facility:

   (1) control of reactivity;

   (2) removal of heat from the reactor and from the fuel storage; and

   (3) confinement of the radioactive material, shielding against radiation and control of planned radioactive releases, as well as limitation of accidental radioactive releases.

(b) A systematic approach shall be taken to identifying those items important to safety that are necessary to fulfil the main safety functions and to defining the conditions and inherent features that contribute to, or affect the fulfilment of, the main safety functions for all states of the facility.

(c) Means of monitoring the status of the reactor facility shall be provided for ensuring that the main safety functions are fulfilled for all states of the facility.

Section 22. ***Application of the Concept of Defence in Depth***

(a) The defence in depth concept shall be applied to provide several levels of defence that are aimed at preventing consequences of accidents that could lead to harmful effects on people and the environment, and at ensuring that appropriate measures are taken for the protection of people and the environment and for the mitigation of consequences in the event that prevention fails.

(b) The design shall:

   (1) provide for successive verifiable physical barriers to the release of radioactive material from the reactor;

   (2) use conservative margins, and the manufacturing and construction shall be of high quality so as to provide assurance that failures and deviations from normal operation are minimized and that accidents are prevented as far as is practicable;

   (3) provide for the control of reactor behavior by means of inherent and engineered features, such that failures and deviations from normal

operation requiring actuation of safety systems are minimized or excluded to the extent possible;

(4) provide for automatic actuation of safety systems, such that failures and deviations from normal operation that exceed the capability of control systems can be controlled with a high level of confidence, and the need for operator actions in the early phase of such failures or deviations from normal operation is minimized;

(5) provide for SSCs and procedures to control the course of and, as far as practicable, to limit the consequences of failures and deviations from normal operation that exceed the capability of safety systems;

(6) provide effective means for ensuring that each of the main safety functions is performed, thereby ensuring the effectiveness of the barriers and mitigating the consequences of any failure or deviation from normal operation.

(c) To ensure that the concept of defence in depth is maintained, the design shall prevent, as far as is practicable:

(1) Challenges to the integrity of physical barriers;

(2) The failure of one or more barriers;

(3) The failure of a barrier as a consequence of the failure of another barrier;

(4) The possibility of harmful consequences of errors in operation and maintenance.

(d) The design shall ensure, as far as is practicable, that the first, or at most the second, level of defence in depth is capable of preventing an escalation to accident conditions for all failures or deviations from normal operation that are likely to occur over the operating lifetime of the research reactor.

(e) The levels of defence in depth shall be independent as far as practicable to avoid a failure of one level reducing the effectiveness of other levels. In particular, safety features for design extension conditions (especially features for mitigating the consequences of accidents involving the melting of fuel) shall as far as is practicable be independent of safety systems.

Section 23. *Proven Engineering Practices*

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

(b) Items important to safety shall preferably be of a design that has previously been proven in equivalent applications, and if not, shall be items of high quality and of a technology that has been qualified and tested.

(c) National and international codes and standards that are used as design rules for items important to safety shall be identified and evaluated to determine their applicability, adequacy and sufficiency, and shall be supplemented or modified as necessary to ensure that the quality of the design is commensurate with the associated safety function.

(d) Codes and standards applicable to SSCs shall be identified and their use shall be in accordance with the classification of the SSCs. In particular, if different codes and standards are used for different types of item, consistency between the codes and standards shall be demonstrated.

(e) In the case of SSCs for which there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar equipment having similar environmental and operational requirements or, in the absence of such codes and standards, the results of experience, tests, analysis or a combination of these shall be applied. The use of such a results-based approach shall be justified.

(f) Where an unproven design or feature is introduced or where there is a departure from an established engineering practice, a process shall be established under the management system to ensure that safety is demonstrated by means of appropriate supporting research programs, performance tests with specific acceptance criteria or the examination of operating experience from other relevant applications.

(g) Acceptance criteria shall be established for operational states and for accident conditions. In particular, the design basis accidents considered in the design of the research reactor and selected design extension conditions shall be identified for the purposes of establishing acceptance criteria, subject to review by the PNRI.

Section 24. *Provision for Construction*

(a) Items important to safety for a research reactor facility shall be designed so that they can be manufactured, constructed, assembled, installed and erected in accordance with established processes that ensure the achievement of the design specifications and the required level of safety.

(b) Due account shall be taken of relevant experience that has been gained in the construction of similar facilities and their associated SSCs. Where good practices from other relevant industries are adopted, such practices shall be shown to be appropriate to the specific nuclear application.

(c) The construction shall start only after the licensee has verified that the main safety issues in the design have been resolved and after the PNRI has granted a license.

Section 25. *Features to Facilitate Radioactive Waste Management and Decommissioning*

(a) Special consideration shall be given at the design stage of a research reactor facility to the incorporation of features to facilitate radioactive waste management and the future decommissioning of the facility. In particular, the design shall take due account of:

(1) The choice of materials so that amounts of radioactive waste will be minimized to the extent practicable and decontamination will be facilitated;

(2) The access capabilities and the means of handling that might be necessary;

(3) The facilities necessary for the processing and storage of radioactive waste generated in operation and provision for managing the radioactive waste that will be generated in the decommissioning of the research reactor facility.

(b) This requirement shall also be considered in the design of any modifications, new utilizations and experiments.

(c) The design shall be such as to ensure that the generation of radioactive waste and discharges are kept to the minimum practicable in terms of both activity and volume and that wastes and discharges are classified.

GENERAL REQUIREMENTS FOR DESIGN

Section 26. *Safety Classification of SSCs*

(a) All items important to safety for a research reactor facility shall be identified and shall be classified on the basis of their safety function and their safety significance.

(b) The licensee shall identify and classify the SSCs, including software, that are important to safety in accordance with their function and significance for safety. The basis for the safety classification of the SSCs shall be stated and the design requirements shall be applied in accordance with their safety classification.

(c) The classification of SSCs shall be based primarily on deterministic methods complemented, where appropriate, by probabilistic methods, with due account taken of factors such as:

(1) The safety functions to be performed by the item;

(2) The consequences of failure to perform a safety function;

(3) The frequency with which the item will be called upon to perform a safety function;

(4) The time following a postulated initiating event at which, or the period for which, the item will be called upon to perform a safety function.

(d) The failure of SSCs in one safety class shall not cause the failure of other SSCs in higher safety class.

(e) Equipment that performs multiple functions shall be classified in a safety class that is consistent with those functions having the highest safety significance.

Section 27. ***Design Basis for Items Important to Safety***

(a) The design basis for items important to safety for a research reactor facility shall specify the necessary capability, reliability and functionality for the relevant operational states, accident conditions and conditions arising from internal and external hazards, to meet the specific acceptance criteria over the lifetime of the research reactor.

(b) The design basis for each item important to safety shall be systematically justified and documented.

Section 28. ***Postulated Initiating Events***

(a) The design for the research reactor shall apply a systematic approach to identifying a comprehensive set of postulated initiating events such that all foreseeable events with the potential for serious consequences and all foreseeable events with a significant frequency of occurrence are anticipated and are considered in the design.

(b) Postulated initiating events shall be selected appropriately as listed in Appendix I for the purpose of analysis. It shall be shown that the set of postulated initiating events selected covers all credible accidents that may affect the safety of the research reactor.

(c) The postulated initiating events shall include all foreseeable failures of SSCs of the reactor facilities and experiments as well as operating errors and possible failures arising from internal and external hazards for all operational and shutdown states.

(d) The postulated initiating events shall be identified based on engineering judgement, operating experience feedback and deterministic assessment, complemented, where appropriate and available, by probabilistic methods.

(e) An analysis of the postulated initiating events shall be made to establish the preventive and protective measures that are necessary to ensure that the required safety functions will be performed.

(f) The expected behavior of the reactor in any postulated initiating event shall be such that the following conditions can be achieved, in order of priority:

(1) A postulated initiating event would produce no safety significant effects and would produce only a change towards a safer and more stable condition by means of inherent safety characteristics of the reactor.

(2) Following a postulated initiating event, the reactor would be rendered safe by means of passive safety features or by the action of systems that are operating continuously in the state necessary to control the postulated initiating event.

(3) Following a postulated initiating event, the reactor would be rendered safe by the actuation of active items important to safety that need to be brought into operation in response to the postulated initiating event.

(4) Following a postulated initiating event, the reactor would be rendered safe by following specified procedures.

(g) The postulated initiating events used for developing the performance requirements for the items important to safety shall be grouped into representative event sequences that identify bounding cases and that provide the basis for the design and the operational limits for the items important to safety.

(h) Where prompt and reliable action would be necessary in response to a postulated initiating event, provision shall be made in the design for automatic safety actions for the actuation of safety systems to prevent progression to more severe reactor conditions.

(i) The design shall include features for manual initiation of safety systems, or on other operator actions needed in response to PIEs, the account for:

(1) the time interval between detection of the postulated initiating event or accident and the required action;

(2) adequate procedures such as administrative, operational and emergency procedures specified to ensure the performance of such actions;

(3) the potential for an operator to worsen an event sequence through erroneous operation of equipment or incorrect diagnosis of the necessary recovery process.

(j) The operator actions necessary to diagnose the state of the reactor following a postulated initiating event and to put it into a stable long term shutdown condition in a timely manner shall be facilitated by the provision in the design of adequate instrumentation to monitor the status of the reactor, and adequate means for the manual operation of equipment.

Section 29. ***Internal and External Hazards***

(a) All foreseeable internal hazards and external hazards for a research reactor, including the potential for human induced events directly or indirectly to affect the safety of the research reactor, shall be identified and their effects, both individually and in credible combinations shall be considered in designing the layout of the facility and in the design of relevant items important to safety. The events to be considered shall include those that have been identified in the site evaluation.

(b) Items important to safety shall be designed and located with due consideration of other implications for safety, to withstand the effects of hazards or to be protected, in accordance with their importance to safety, against hazards and against common cause failure mechanisms generated by hazards.

(c) Credible combinations of individual events, including internal and external hazards, that could lead to anticipated operational occurrences or design basis accidents, shall be considered in the design. Deterministic safety assessments, complemented, as appropriate, by probabilistic safety assessments or engineering judgment shall be used for the selection of the event combinations. Where the results of the assessments indicate that combinations of postulated initiating events could lead to accident conditions, such combinations of events shall be considered to be design basis accidents or shall be included as part of design extension conditions, depending mainly on their likelihood of occurrence. Events that are consequences of other events shall be considered to be part of the original postulated initiating event.

(d) **Fire and Explosion**

(1) SSCs important to safety shall be designed and located, subject to compliance with other safety requirements, so as to minimize the effects of fires and explosions.

(2) Non-combustible or fire retardant and heat resistant materials shall be used wherever practicable throughout the research reactor facility. Flammable gases and liquids and combustible materials that could produce or contribute to explosive mixtures shall be kept to the minimum necessary amounts and shall be stored in adequate facilities to keep reactive substances segregated.

(3) Fires and explosions shall not prevent achievement of the main safety functions as well as monitoring the status of the facility. These shall be maintained by means of the appropriate incorporation of redundant SSCs, diverse systems, physical separation and design for fail-safe operation.

(e) **External Events**

(1) Natural external events and human induced external events arising from nearby industries and transport routes shall be addressed.

(2) In the short term, the safety of the facility shall not be dependent on the availability of off-site services such as the electricity supply and firefighting services. The design shall take due account of site specific conditions to determine the maximum delay time by which off-site services need to be available.

(3) A research reactor facility located in a seismically active region shall be equipped with a seismic detection system that actuates the automatic reactor shutdown systems if a specified threshold value is exceeded.

(4) Features shall be provided to minimize any interactions between buildings containing items important to safety (including power cabling and instrumentation and control cabling) and any other structure as a result of external events considered in the design.

(5) The design shall be such as to ensure that all items important to safety are capable of withstanding the effects of external events considered in the design, and if not, other features such as passive barriers shall be provided to protect the reactor facility and to ensure that the main safety functions will be achieved.

(6) The design shall provide for an adequate margin to protect items important to safety against levels of external hazards more severe than those selected for the design basis, derived from the site hazard evaluation.

Section 30. ***Design Basis Accidents***

(a) A set of accident conditions that are to be considered in the design for a research reactor shall be derived from postulated initiating events for the purpose of establishing the boundary conditions for the research reactor to withstand, without exceeding the limits established in CPR Part 3.

(b) The design shall be such that for design basis accident conditions, key reactor parameters do not exceed the specified design limits. Design basis accidents shall be used to define the design bases, including performance criteria, for safety systems and for other items important to safety that are necessary to control design basis accident conditions, with the objective of returning the reactor to a safe state and mitigating the consequences of any accident, and preventing the need for off-site emergency response actions. The design shall reduce demands on the operator as far as reasonably practicable during and following a design basis accident.

(c) The design basis accidents shall be analyzed in a conservative manner including the application of single failure criterion to safety systems, specifying design criteria and using conservative assumptions, models and input parameters in the analysis.

(d) The design of subcritical assemblies shall include technical provisions to prevent criticality.

Section 31. ***Design Limits***

A set of design limits consistent with the key physical parameters for each item important to safety shall be specified for each operational state of the reactor and its experimental devices and for accident conditions.

Section 32. ***Design Extension Condition***

(a) A set of design extension conditions for a research reactor shall be derived for the purpose of enhancing the safety of the research reactor by enhancing its capabilities to withstand, without unacceptable radiological consequences, accidents that are either more severe than design basis accidents or that involve additional failures.

(b) An analysis of design extension conditions shall be performed to determine whether the potential radiological consequences would exceed those deemed acceptable by the PNRI.

(c) The design extension conditions shall be used to define the design specifications for safety features and for the design of all other items important to safety that are necessary for preventing such conditions from arising, or, if they do arise, for controlling them and mitigating their consequences. For existing research reactors, complementary safety reassessment shall be performed to determine whether there is a need for mitigatory measures or modifications of the facility to be implemented.

(d) For subcritical assemblies, the likelihood of criticality shall be sufficiently remote to be considered a design extension condition. To ensure subcriticality, the design shall include safety provisions such as the use of only natural uranium or limited amounts of fissile materials, or a fixed fuel/moderator ratio. If no such provisions can be provided, measures for mitigating the consequences shall be determined and implemented based on safety analysis.

(e) The analysis undertaken shall include identification of the safety features that are designed for use in, or that are capable of preventing or mitigating, events considered in the design extension conditions. These features shall be:

   (1) independent, to the extent practicable, of those used in more frequent accidents;

   (2) capable of performing, to the extent practicable, in the environmental conditions pertaining to design extension conditions, as appropriate;

   (3) reliable commensurate with the function that they are required to fulfil.

(f) The design shall be such that the possibility of conditions arising that could lead to an early radioactive release or a large radioactive release is practically eliminated. The design shall be such that for design extension conditions, protective measures that are limited in terms of times and areas of application shall be sufficient for protection of the public, and sufficient time shall be available to take such measures.

(g) Where the results of engineering judgement and deterministic safety assessments, complemented, as appropriate, by probabilistic safety assessments, indicate that combinations of postulated initiating events could lead to accident conditions, such combinations of events shall be considered to be design basis accidents or shall be included as part of design extension conditions, depending mainly on their likelihood of occurrence. Certain events might be consequences of other events, such as a flood following an earthquake. Such consequential effects shall be part of the original postulated initiating event.

Section 33. ***Engineered Safety Features***

(a) Engineered safety features shall be provided for a research reactor to prevent anticipated operational occurrences and design basis accidents and to mitigate their consequences if they occur.

(b) The necessity and capabilities for engineered safety features shall be determined from the safety analysis. The accidents where these systems are required to be able to cope shall be specified and analyses shall be provided to demonstrate that the systems fulfil the requirements. Those systems and subsystems that are essential for the proper operation of the engineered safety features shall be provided.

(c) The various modes of operation of an engineered safety feature shall be determined in detail, including the extent to which the engineered safety feature is automated and the conditions for which its manual overriding is warranted. The following shall be considered in the design of engineered safety features:

  (1) Component reliability (including reliability of supporting and auxiliary systems necessary for operating the engineered safety features, independence, redundancy, fail-safe characteristics, diversity and physical separation of redundant systems, preference of passive systems over active systems, and functional separation of redundant safety systems;

  (2) The use of materials to withstand the postulated accident conditions;

  (3) Provisions for maintenance, periodic testing and inspection (including under simulated design basis accident conditions where possible) to verify that the engineered safety features continue to function or are in a state of readiness to perform their functions reliably and effectively upon demand.

Section 34. ***Reliability of Items Important to Safety***

(a) The reliability of items important to safety for a research reactor facility shall be commensurate with their safety significance.

(b) The design of items important to safety shall be such as to ensure that the equipment can be qualified, procured, installed, commissioned, operated and maintained to be capable of withstanding, with sufficient reliability and effectiveness, all conditions specified in the design basis for the items.

(c) In the selection of equipment, consideration shall be given to both spurious operation and unsafe failure modes. Preference shall be given in the selection process to equipment that exhibits a predictable and revealed mode of failure and for which the design facilitates repair or replacement.

Section 35. ***Single Failure Criterion***

(a) The single failure criterion shall be applied to each safety group incorporated in the design of the research reactor.

(b) Spurious action shall be one mode of failure when applying the single failure criterion to a safety group or safety system.

(c) The design shall take due account of the failure of a passive component, unless it has been justified in the single failure analysis with a high level of confidence that a failure of that component is very unlikely and that its function would remain unaffected by the postulated initiating event.

(d) Multiple sets of equipment that cannot be tested individually shall not be considered redundant.

(e) The degree of redundancy adopted shall reflect the potential for undetected failures that could degrade reliability. Possible failures shall be considered undetectable if there is no test or method of inspection by which they could be found. For undetected failures, either the failure shall be considered to occur at any time or other methods shall be applied, such as the surveillance of reference items, validated methods of calculation and the use of conservative safety margins.

Section 36. ***Common Cause Failures***

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

(b) The principle of diversity shall be adopted wherever practicable, after consideration of its possible disadvantages from complications in operating, maintaining and testing the diverse equipment.

Section 37. ***Physical Separation and Independence of Safety Systems***

Interference between safety systems or between redundant elements of a system for a research reactor facility shall be prevented by means such as physical separation, electrical isolation, functional independence and independence of communication or data transfer, as appropriate.

Section 38. ***Fail-safe Design***

(a) The concept of fail-safe design shall be incorporated, as appropriate, into the design of systems and components important to safety for a research reactor.

(b) Systems and components important to safety shall be designed for fail-safe behavior, as appropriate, so that their failure or the failure of a support feature does not prevent the performance of the intended safety function.

Section 39. ***Qualification of Items Important to Safety***

(a) A qualification Program shall be implemented for a research reactor facility to verify that items important to safety are capable of performing their intended functions when necessary, and in the prevailing environmental conditions, throughout their design life, with due account taken of reactor conditions during maintenance and testing.

(b) Any environmental and service conditions that could reasonably be anticipated and that could arise in specific operational states shall be included in the qualification Program.

(c) The environmental conditions considered in the qualification Program for items important to safety at a research reactor shall include the variations in ambient environmental conditions that are anticipated in the anticipated operational occurrences and the design basis accidents for the facility.

(d) The qualification Program for items important to safety shall include the consideration of ageing effects caused by environmental factors over the expected service life of the items important to safety. When the items important to safety are subject to natural external events and are required to perform a safety function during or following such an event, the qualification Program shall replicate as far as is practicable the conditions imposed on the items important to safety by the natural event, either by test or by analysis or by a combination of both.

Section 40. ***Design for Commissioning***

The design for a research reactor facility shall include features as necessary to facilitate the commissioning process for the reactor facility, including experimental facilities.

Section 41. ***Calibration, Testing, and Maintenance***

(a) Items important to safety for a research reactor facility shall be designed to be calibrated, tested, maintained, repaired or replaced, inspected and monitored as required to ensure their capability of performing their functions and to maintain their integrity in all conditions specified in their design basis.

(b) The layout of the reactor shall be such that activities for calibration, testing, maintenance, repair or replacement, inspection and monitoring are facilitated and can be performed without undue exposure to radiation of the operating personnel.

(c) Provision shall be made in the design of the reactor to facilitate maintainability and the replacement of items important to safety as well as to facilitate routine in-service inspection.

Section 42. ***Design for Emergency Preparedness and Response***

For emergency preparedness and response purposes, the design for a research reactor facility shall provide:

(a) A sufficient number of safe escape routes, clearly and durably marked, with reliable emergency lighting, ventilation and other building services essential to the safe use of these escape routes;

(b) Suitable alarm systems and reliable and diverse means of communication so that all persons present at the reactor facility and on the site can be given warnings and instructions in an emergency. The availability of reliable and diverse means of communication necessary for safety within the reactor facility shall be ensured at all times, with due account taken of postulated initiating events that may compromise their availability.

Section 43. ***Design for Decommissioning***

In the design of the research reactor and its experimental facilities and in any modifications of them, consideration shall be given to facilitation of decommissioning. In accomplishing this, the following shall be considered:

(a) The selection of materials so as to minimize activation of the materials with regard to decommissioning and radioactive waste management and to provide for easy decontamination;

(b) Optimizing of the facility's layout and access routes to facilitate the removal of large components and the detachment and handling of activated components;

(c) The predisposal management of radioactive waste arising from operation and decommissioning of the reactor.

Section 44. ***Design for Radiation Protection***

(a) The design of a research reactor facility shall be such as to ensure that radiation doses to workers and other personnel at the research reactor facility and to members of the public do not exceed the established dose limits, as provided in CPR Part 3 and that they are kept as low as reasonably achievable (ALARA) for operational states for the entire lifetime of the research reactor facility, and that they remain below acceptable limits and ALARA in, and following, accident conditions.

(b) The design shall ensure that facility states that could lead to high radiation doses or large radioactive releases are practically eliminated and that there are no, or only minor, potential radiological consequences for facility states with a significant likelihood of occurrence.

(c) The relevant dose constraints shall be taken into consideration in the design of the research reactor.

(d) For all operational states and accident conditions, adequate provision shall be made in the design, on the basis of the radiation protection Program, for shielding, ventilation, filtration and decay systems for radioactive material and for monitoring instrumentation for radiation and airborne radioactive material inside and outside the controlled area.

(e) The dose values used for design purposes shall be set with a sufficient margin to ensure that the limits established in CPR Part 3will not be exceeded. Protection and safety shall be optimized by means of suitable provision in the design and layout of the reactor and its experimental devices and facilities to limit exposure and contamination from all sources. The shielding, ventilation, filtration and decay systems of the reactor and its associated facilities shall be designed to allow for uncertainties in operating practices and in all operational states and design basis accidents.

(f) Structural materials, in particular those used near the core, shall be selected to limit the doses to personnel during operation, maintenance, testing and inspection, and decommissioning, consistent with the requirements of optimization in CPR Part 3, as well as to fulfil their other functions. The effects of radionuclides produced by neutron activation in reactor process systems shall be given due consideration in the provision of radiation protection for people on and off the site.

(g) The design shall include any necessary provisions to segregate materials in accordance with their radiological, physical and chemical characteristics and to facilitate their handling. Surfaces shall be appropriately designed to facilitate their decontamination.

(h) The design shall include any necessary provisions for establishing zones within the facility in accordance with CPR Part 3.

(i) The design shall include the shielding required for the reactor and for experimental devices and associated facilities.

(j) Provision shall be made for controlling the release and preventing the dispersion of radioactive substances and contamination at the facility. Ventilation systems with appropriate filtration shall be provided for use in operational states and accident conditions.

(k) Provision shall be made in the design for safe handling of the radioactive waste generated at the research reactor facility. Provision shall be made for appropriate decontamination facilities for both personnel and equipment and for handling the radioactive waste arising from decontamination activities.

Section 45. ***Design for Optimal Operator Performance***

(a) Systematic consideration of human factors, including the human–machine interface, shall be applied in the design process for a research reactor facility, including its experimental facilities.

(b) Consideration shall be given in design to ensuring that, if reliance on administrative controls and procedures is necessary, such controls are feasible and the associated procedures are applicable.

(c) Consideration shall be given to human factors and the application of ergonomic principles in the design of the control room and reactor systems.

(d) The human–machine interface shall be designed to provide the operators with comprehensive but easily manageable information, in accordance with the necessary decision times and action times, the physical environmental conditions expected and the possible psychological pressure on the operator. The information necessary for the operator to make a decision to act shall be simply and unambiguously presented and shall enable:

(1) Assessment of the general state of the facility in any condition;

(2) Operation of the facility within the specified limits on parameters associated with facility systems and equipment;

(3) Confirmation that safety actions for the actuation of safety systems are automatically initiated when needed and that the relevant systems perform as intended;

(4) Determination of both the need for and the time for manual initiation of the specified safety actions.

(e) The design shall support operating personnel in the performance of their tasks and shall limit the effects of operating errors on safety. Due consideration shall be given

(f) in the design process to the layout of the facility and equipment, and to procedures, including procedures for maintenance and inspection, for facilitating intervention of the operating personnel on the reactor SSCs in all states of the research reactor.

Section 46. ***Provision for Safe Utilization and Modification***

(a) The design for research reactor facility shall include provision for the safe utilization and modification of the research reactor.

(b) Precautions shall be taken in the design regarding the utilization, experimental equipment and modification of the research reactor to ensure that the configuration of the reactor is known at all times. In particular, consideration shall be given to experimental equipment since:

(1) It can cause hazards directly if it fails.

(2) It can cause hazards indirectly by affecting the safe operation of the reactor.

(3) It can increase the hazard due to an initiating event by its consequent failure and the effects of this on the event sequence.

(c) Every proposed modification to the reactor or to an experiment that may have a major significance for safety shall be designed in accordance with the same principles as apply for the reactor itself. In particular, all experimental devices shall be fully compatible in terms of the materials used, the structural integrity and the provision for radiation protection. The radioactive inventory and the generation and release of energy shall be considered in the design of all experimental devices.

(d) Modifications of research reactors and experimental devices shall be designed such that the means of confinement and shielding of the reactor is preserved. Protection systems for experimental devices shall be designed to protect both the device and the reactor.

(e) The requirements relating to the anticipated utilization of the reactor, including the requirements for power stability, shall be taken into account in the design. The design shall be such that the response of the reactor and its associated systems to a wide range of events, including anticipated operational occurrences, will allow its safe operation.

Section 47. *Design for Ageing Management*

(a) The design life of items important to safety at a research reactor facility shall be determined. Appropriate margins shall be provided in the design to take due account of relevant mechanisms of ageing, and of the potential for age related degradation, to ensure the capability of items important to safety to perform their necessary safety functions in operational states and accident conditions in case of demand throughout their design life.

(b) The design for a research reactor shall take due account of physical ageing, the effects of wear and tear and obsolescence in all operational states for which a component is credited, including testing, maintenance, and operational states during and following a postulated initiating event.

(c) The design shall include provisions for the necessary monitoring, testing, sampling and inspection for the detection, assessment, prevention and mitigation of ageing effects.

Section 48. *Provision for Long Shutdown Periods*

In the design of the research reactor facility, consideration shall be given to ensure the safety of the facility in long shutdown periods.

Section 49. *Prevention of Unauthorized Access to, or Interference with, Items Important to Safety*

(a) Unauthorized access to, or interference with, items important to safety at a research reactor facility, including computer hardware and software, shall be prevented.

(b) Provision shall be made in the design for the control of access to the reactor facility and/or to equipment by operating personnel and reactor users, including emergency workers and vehicles, with particular consideration given to the prevention of any unauthorized entry of persons and goods to the site or to buildings on the site, for the main purposes of preventing theft or the unauthorized removal of nuclear material and preventing sabotage.

Section 50. *Prevention of Disruptive or Adverse Interactions Between Systems Important to Safety*

(a) The potential for disruptive or adverse interactions between systems important to safety at a research reactor facility that might be required to operate simultaneously shall be considered in the design.

(b) In the analysis of the potential for disruptive or adverse interactions of systems important to safety, due account shall be taken of physical interconnections and of the possible effects of one system's operation, spurious operation or malfunction on local environmental conditions of other systems, to ensure that changes in environmental conditions do not affect the reliability of systems or components in functioning as intended.

(c) If two systems important to safety and containing fluid are interconnected and are operating at different pressures, either the systems shall both be designed to withstand the higher pressure, or provision shall be made to prevent the design pressure of the system operating at the lower pressure from being exceeded.

Section 51. *Buildings and Structures*

(a) The buildings and structures important to safety for a research reactor facility shall be designed to keep radiation levels and radioactive releases on and off the site as low as reasonably achievable and below authorized limits for all operational states, for design basis accidents and, as far as practicable, for design extension conditions.

(b) The buildings and structures important to safety shall be designed for all operational states, for design basis accidents and, as far as practicable, for design extension conditions.

(c) The required leak tightness of the reactor building or of other buildings and structures containing radioactive material and the requirements for the ventilation system shall be determined in accordance with the safety analysis of the reactor and its utilization.

Section 52. *Means of Confinement*

(a) Means of confinement shall be provided for a research reactor to ensure, or to contribute to, the fulfilment of the following safety functions: (i) confinement of radioactive substances in operational states and in accident conditions; (ii) protection of the reactor against natural external events and human induced events; and (iii) radiation shielding in operational states and in accident conditions.

(b) Means of confinement shall be designed to ensure that a release of radioactive material following an accident involving disruption or damage of the nuclear fuel, core components or experimental devices does not exceed acceptable limits.

(c) For the proper functioning of the means of confinement, the pressure within a barrier shall be set at such a level as to prevent the uncontrolled release of radioactive material to the environment through the barrier. In setting this pressure, variations in atmospheric shall be taken into account.

(d) In the design of the means of confinement, the effects of extreme conditions and environmental conditions due to accidents, including conditions arising from the external and internal events listed in Appendix I, as relevant, shall be taken into account.

(e) Barriers shall be designed to withstand, with suitable margins, the highest calculated pressure and temperature loads expected in design basis accident conditions.

(f) The release rate under accident conditions and associated consequences shall be determined, with account taken of the source term and other parameters such as extent of filtration, the point of release, environmental conditions, and the pressure and temperature under design basis accident conditions.

(g) In the event of an accident, the leakage from the barrier shall be controlled by means of appropriate engineering features to prevent the release of radioactive material to the environment in excess of acceptable limits.

(h) Provisions to enable initial and periodic performance tests to check air leakage rates and to enable monitoring of the operational performance of the ventilation system shall be included in the design.

(i) Where confinement is dependent on the efficiency of filters, the design shall include provisions as appropriate for in situ periodic testing of the efficiency of the filters.

(j) For structures and components performing the function of confinement, coverings and coatings shall be such as to ensure their safety functions and to minimize interference with other safety functions in the event of their deterioration.

(k) For research reactors that have greater potential hazards associated with them, a containment structure shall ensure that, in design basis accident conditions, any release of radioactive material would be kept below authorized limits and that, in design extension conditions, any release of radioactive material would be kept below acceptable limits.

REACTOR CORE AND ASSOCIATED FEATURES

Section 53. *Reactor Core and Fuel Design*

(a) Research reactor core components and fuel elements and assemblies for a research reactor shall be designed to maintain their structural integrity and to withstand satisfactorily the conditions in the reactor core in all operational states and in design basis accident conditions.

(b) Appropriate neutronic, thermohydraulic, mechanical, material, chemical and irradiation related considerations associated with the reactor as a whole shall be taken into account in the design and qualification of fuel elements and assemblies, the reflectors and other core components.

(c) Analyses shall be performed to show that the intended irradiation conditions and limits such as fission density, total fissions at the end of lifetime and neutron fluence are acceptable and will not lead to undue deformation or swelling of the fuel elements. The anticipated upper limit of possible deformation shall be evaluated. These analyses shall be supported by data from experiments and from experience with irradiation. Consideration shall be given in the design of the fuel elements to the requirements relating to the long-term management of irradiated elements, which may include reprocessing or conditioning for disposal.

(d) All foreseeable reactor core configurations, including the initial core configuration through to the equilibrium core configuration, as appropriate, shall be considered in the core design. The effect of the inserted experimental devices or materials under irradiation shall also be considered. For subcritical assemblies, this includes assurance that all of these configurations are subcritical with justified margins.

(e) The reactor core shall be designed to maintain the relevant parameters below predetermined limits in all operational states. Provisions shall be considered in the design for monitoring the physical conditions and integrity of the fuel. The design shall ensure that inadvertent movement of fuel elements or core components is not possible.

(f) The reactor core, including fuel elements, reactivity control mechanisms and experimental devices shall be designed and constructed so that the maximum permissible design limits that are determined for all operational states are not exceeded. A suitable margin, including margins for uncertainties and engineering tolerances, shall be incorporated in setting these limits.

(g) The reactor core shall be designed so that the reactor can be shut down, cooled and maintained subcritical with an adequate margin for all operational states and accident conditions. The end state of the reactor core shall be assessed for selected design extension conditions.

(h) Wherever possible, the design of the reactor core shall make use of inherent safety characteristics to minimize the consequences of accident conditions due to transients and instabilities.

(i) The design and construction of the core of a subcritical assembly shall ensure that criticality cannot be reached for any core configuration, temperatures, moderation and reflection circumstances.

Section 54. ***Provision of Reactivity Control***

(a) The design of a research reactor shall provide adequate means to control the reactivity.

(b) It shall be demonstrated in the design that the reactivity control system will function properly under all operational states of the reactor and will also maintain its reactor shutdown capability under all design basis accidents, including failures of the control system itself.

(c) Sufficient negative reactivity shall be available in the reactivity control devices so that the reactor can be brought to a subcritical condition and maintained subcritical in all operational states and in accident conditions, with account taken of the experimental arrangements with the highest positive reactivity contribution. In the design of reactivity control devices, account shall be taken of wear and tear and the effects of irradiation, such as burnup, poison buildup and decay, changes in physical properties and the production of gas. This requirement might not apply to some subcritical assemblies; however, subcriticality shall be justified for any configuration.

(d) The maximum rate of addition of positive reactivity allowed by the reactivity control system or an experiment shall be specified and shall be limited to values justified in the safety analysis report and documented in the OLCs.

(e) If a subcritical assembly will remain subcritical in any condition even in the most reactive case, reactivity control devices might not be required.

Section 55. ***Reactor Shutdown Systems***

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

(b) At least one automatic shutdown system shall be incorporated into the design. The provision of a second independent shutdown system may be necessary, depending on the characteristics of the reactor, and this shall be given due consideration in the design.

(c) The effectiveness, speed of action and shutdown margin of the reactor shutdown system shall be such that the conditions and the design limits for the fuel specified in the safety analysis report are met.

(d) No single failure in the shutdown system shall be capable of preventing the system from fulfilling its safety function when required.

(e) A capability to initiate manual reactor emergency shutdown shall be provided. This manual reactor trip signal shall also be provided as an input to the reactor protection system. The manual reactor trip shall be able to shut down the reactor directly.

(f) Instrumentation shall be provided and tests shall be performed to ensure that the means of shutdown are in the state stipulated for the given condition of the reactor.

(g) It shall be demonstrated in the design that the reactor shutdown system will function properly under all operational states of the reactor and will maintain its reactor shutdown capability under accident conditions, including failures of the control system itself.

Section 56. ***Design of Reactor Coolant Systems and Related Systems***

(a) The coolant systems for a research reactor shall be designed and constructed to provide adequate cooling to the reactor core.

(b) Systems containing reactor coolant shall be designed to allow pre-service and in-service tests and inspections to detect the possible occurrence of leaks, cracks and brittle fractures. Consideration shall be given in the design to ensuring material characteristics that ensure the slow propagation of failures.

(c) In the design of water cooled reactors, particular attention shall be paid to preventing the uncovering of the core.

(d) Where the primary cooling system is not designed for cooling the core after shutdown, a reliable separate system shall be provided for the removal of residual heat.

(e) For reactor systems that use flappers or equivalent systems for the transition from forced to natural circulation cooling, or for operation with natural circulation cooling, and for which this mode is part of the safety system or is considered an engineered safety feature, the single failure criterion shall be applied. Instrumentation to verify their functioning and to provide signals to the reactor protection system shall be provided.

(f) If two coolant systems that are operating at different pressures are interconnected, the requirement of Section 48 (c) applies.

(g) Provision shall be made in the design for controlling the volume, temperature and pressure of the reactor coolant in any operational state of the facility, with due account taken of volumetric changes and leakage.

(h) Provisions shall be made in the design to monitor and control the properties of the reactor coolant and/or the moderator, and to remove radioactive substances, including activated corrosion products and fission products, from the coolant. Despite the fact that subcritical assemblies might not require cooling systems for heat removal, such provisions shall be applied to the fluids contained within such assemblies, to preserve fuel elements and SSCs and to avoid radioactive releases.

(i) Design features (such as leak detection systems, appropriate interconnections and capabilities for isolation) and suitable redundancy and diversity shall be provided to fulfil the requirements of Section 29 to Section 30 with adequate reliability for each postulated initiating event. Such measures also apply to subcritical assemblies.

Section 57. ***Emergency Cooling of the Reactor Core***

(a) An emergency core cooling system shall be provided for a research reactor, as required, to prevent damage to the fuel in the event of a loss of coolant accident.

(b) The emergency core cooling system shall be capable of preventing significant failure of fuel for the range of accidents specified in the design basis. Under design basis accidents, damage to the fuel and releases of radioactive material shall be kept within authorized limits. Special procedures for cooling the core shall be considered in the case of selected design extension conditions.

(c) For design basis accidents, the emergency core cooling shall be designed to perform its intended function in the event of any single failure in the system.

(d) The emergency core cooling system shall be designed to permit the periodic inspection of components and shall be designed for appropriate periodic functional testing for the verification of performance.

INSTRUMENTATION AND CONTROL SYSTEMS

Section 58. ***Provision of Instrumentation and Control Systems***

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

(b) The reactor shall be provided with sufficient instrumentation and recording means to monitor important reactor parameters and the status of essential equipment of the reactor including the position of neutron source and associated experimental devices in all facility states. The expected response of such instrumentation and control systems in an emergency shall be assessed and taken into account in the emergency arrangements.

(c) The reactor shall be provided with appropriate controls, both manual and automatic as appropriate, to maintain parameters within specified operating ranges.

(d) In the design of the instrumentation and control systems, provision shall be made as appropriate for startup neutron sources and dedicated startup instrumentation for conditions in which they are needed. This requirement shall be fulfilled for commissioning and startup after long shutdown periods.

(e) Audio and visual alarm systems, as appropriate, shall be provided for the early indication of changes in the operating conditions of the reactor that could affect its safety.

(f) Interconnections between reactor instrumentation and control systems and systems for controlling experimental devices shall in general be prohibited. Exceptions shall be permitted only if interconnections for controlling specific parameters of experimental devices are mandatory for the safe operation of the reactor.

Section 59. ***Reactor Protection System***

(a) A protection system shall be provided for a research reactor to initiate automatic actions to actuate the safety systems necessary for achieving and maintaining a safe state.

(b) The reactor protection system shall be independent of other systems and shall be capable of overriding unsafe actions of the control system.

(c) The reactor protection system shall be capable of automatically initiating the required safety actions, for the full range of postulated initiating events, to actuate the safety systems necessary for achieving a safe state.

(d) The reactor protection system shall be designed in such a way that, once the sequence of protective actions has been initiated automatically by the reactor protection system, it will proceed to completion and no manual actions will be necessary within a short period of time following activation of the reactor protection system. Such automatic actions by the reactor protection system shall not be self-resetting, and deliberate operator action shall be required to return to normal operation.

(e) The possibility of bypassing interlocks and trips of the reactor protection system that might result in the bypassing of a safety function shall be carefully evaluated and justified. Appropriate means of preventing interlocks and trips that are important to safety from being inadvertently bypassed shall be incorporated into the reactor protection system.

(f) The design of the reactor protection system shall be such that no single failure could result in the loss of automatic protective actions.

(g) The reactor protection function shall be designed to bring the reactor to a safe condition and to maintain it in a safe condition even if the reactor protection systems are subjected to a credible common cause failure.

(h) The reactor protection system shall be designed to permit periodic testing of its functionality.

(i) It shall be ensured in the design that the set points can be established with a margin between the initiation point and the safety limits such that the action initiated by the reactor protection system will be able to control the process before the safety limit is reached. Some of the factors to be considered in establishing this margin are:

   (1) The accuracy of the instrumentation;

   (2) Uncertainties in calibration;

   (3) Instrument drift;

(4) Instrument and system response times.

(j) Where a computer-based system is intended to be used in a reactor protection system, the following requirements apply:

(1) Hardware and software of high quality and proven design shall be used.

(2) The whole development process, including control, testing and commissioning of the design, shall be systematically documented and reviewable.

(3) In order to confirm the reliability of the computer-based systems, a systematic, fully documented and reviewed assessment shall be undertaken by expert personnel who are independent of the designers and the suppliers.

(4) Protection shall be provided against accidental disruption of, or deliberate interference with, system operations.

(k) Where the necessary high reliability of a computer-based system that is intended for use in a reactor protection system cannot be demonstrated with a high level of confidence, diverse means of ensuring fulfilment of the protection functions shall be provided.

Section 60. ***Reliability and Testability of Instrumentation and Control Systems***

(a) Instrumentation and control systems for items important to safety at a research reactor shall be designed for high functional reliability and periodic testability commensurate with the safety functions to be performed.

(b) The required level of reliability shall be achieved by means of a comprehensive strategy that uses various complementary means including an effective regime of analysis and testing at each phase of development of the system and a validation strategy to confirm that the design requirements for the system have been fulfilled. The conditions in which equipment is to be used and stored and the effects of possible environmental factors shall be taken into account in the reliability analysis.

(c) Design techniques such as testability, including a self-checking capability where necessary, fail-safe characteristics, functional diversity and diversity in component design and in concepts of operation shall be used to the extent practicable to prevent loss of a safety function.

Section 61. ***Use of Computer Based Equipment in Systems Important to Safety***

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

(b) For computer-based equipment in safety systems and systems important to safety:

    (1) A high quality of, and best practices for, hardware and software shall be used, in accordance with the importance of the system to safety.

    (2) The entire development process, including the control, testing and commissioning of design changes, shall take into account all phases of the life cycle of the computer-based system, shall be systematically documented and shall be reviewable.

    (3) An assessment of the equipment shall be undertaken by experts who are independent of the design team and the supplier team to provide assurance of its high reliability.

    (4) When the necessary high reliability of the equipment cannot be demonstrated with a high level of confidence, diverse means of ensuring fulfilment of the safety functions shall be provided.

    (5) Common cause failures deriving from software shall be taken into consideration.

    (6) Protection shall be provided against accidental disruption of, or deliberate interference with, system operation.

    (7) Appropriate verification and validation and testing of the software systems shall be performed.

Section 62. ***Control Room***

(a) A control room shall be provided at a research reactor facility from which the facility can be safely operated in all operational states, either automatically or manually, and from which measures can be taken to maintain the research reactor in a safe state or to bring it back into a safe state after anticipated operational occurrences and accident conditions.

(b) Appropriate measures shall be taken and adequate information shall be provided for the protection of occupants of the control room, for an extended period of time, against hazards such as high radiation levels resulting from accident conditions, releases of radioactive material, fire, or explosive or toxic gases.

(c) Special attention shall be paid to identifying those events, both internal and external to the control room, that could challenge its continued operation, and the design shall provide for practicable measures to minimize the consequences of such events. The design shall provide for escape routes for the occupants in case of events necessitating the evacuation of the control room.

(d) The design of the control room shall provide an adequate margin against natural hazards more severe than those selected for the design basis.

Section 63. *Supplementary Control Room*

(a) Provision of a supplementary control room for a research reactor facility, separate and functionally independent from the main control room, shall be considered in the design.

(b) The means provided in the supplementary control room shall be sufficient for fulfilment of the main safety functions in the event of an emergency. Information on important parameters and the radiological conditions in the facility and its surroundings shall be made available in the supplementary control room. Systems designed for this purpose shall be considered items important to safety. A supplementary control room might not be necessary for critical assemblies and subcritical assemblies. In this case, the decision shall be justified on the basis of a comprehensive analysis.

Section 64. *Emergency Response Facilities on the Site*

(a) A research reactor facility shall include the necessary emergency response facilities on the site. Their design shall be such that personnel will be able to perform expected tasks for managing an emergency under conditions generated by accidents as well as initiating events.

(b) Information about important reactor parameters and radiological conditions at the reactor facility and the site, and information from monitoring systems and laboratory facilities that is to be used to determine the need to initiate emergency measures, as well as information to be used for continuing assessment, shall be provided to the relevant emergency response facilities. Each emergency response facility shall be provided with means of communication with the control room, the supplementary control room and other important locations at the facility, and with on-site and off-site emergency response organizations.

POWER SUPPLIES

Section 65. *Electrical Power Supply Systems*

(a) The design for a research reactor facility shall include reliable normal electrical power supply systems and shall consider reliable emergency electrical power supply systems.

(b) Reliable electrical power supplies for essential safety functions shall be available in normal operational states and in accident conditions.

(c) The design shall consider the provision of uninterruptible power supplies for those safety systems that require a continuous energy supply, such as the reactor protection system and the radiation monitoring system.

(d) In the design basis for the emergency power supply, due account shall be taken of the postulated initiating events and the associated safety functions to be performed to determine the requirements for capability, availability, duration of the required power supply, capacity and continuity.

Section 66. *Radiation Protection Systems*

(a) Equipment shall be provided at a research reactor facility to ensure that there is adequate radiation monitoring in operational states and accident conditions.

(b) The design of radiation protection systems shall include:

(1) Stationary dose rate meters for monitoring the local radiation dose rate at places routinely accessible by operating personnel and at other places where the changes in radiation levels in operational states could be such that access is allowed only for certain specified periods of time.

(2) Stationary dose rate meters to indicate the general radiation levels at suitable locations of the facility in anticipated operation occurrences and accident conditions. The stationary dose rate meters shall provide sufficient information in the control room or in the appropriate control position that operating personnel can initiate protective actions and corrective actions if necessary.

(3) Monitors for measuring the activity of radioactive substances in the atmosphere in those areas routinely occupied by personnel, including experimental areas, and where the levels of airborne activity may be expected to be such as to require protective measures.

(4) Stationary equipment and laboratories for determining, in a timely manner, the concentrations of selected radionuclides in fluid process systems, and in gas and liquid samples taken from the research reactor facility or the environment, in operational states and accident conditions.

(5) Stationary equipment for monitoring and controlling effluents prior to or during their discharge to the environment.

(6) Devices for measuring radioactive surface contamination.

(7) Installations and equipment for measuring doses to and contamination of personnel.

(8) Radiation monitoring at gates and other entrances of the facility to detect radioactive material being moved without permission or unintentional contamination.

(c) In addition to monitoring within the facility, arrangements shall also be made to assess exposures and other radiological impacts in the vicinity of the facility, where necessary.

Section 67. *Handling and Storage Systems for Fuel and Core Components*

(a) The design for a research reactor facility shall include provisions for the safe handling and storage of fresh and irradiated fuel and core components.

(b) The design shall include provisions for safely storing a sufficient number of spent fuel elements and irradiated core components. These provisions shall be in accordance with the programs for core management and for removing or replacing fuel elements and core components.

(c) The design shall include provisions to unload all fuel from the core safely at any time.

(d) The implications of the storage of irradiated fuel and core components over an extended time period shall be considered in the design, where applicable.

(e) The handling and storage systems shall be designed:

(1) To prevent criticality by an adequate margin, by physical means such as the use of an appropriate geometry and fixed absorbers;

(2) To permit periodic inspection and testing;

(3) To minimize the probability of loss of, or damage to, the fuel;

(4) To prevent the inadvertent dropping of heavy objects on the fuel;

(5) To permit the appropriate storage of suspect or damaged fuel elements;

(6) To provide for radiation protection;

(7) To provide a means for controlling the chemistry and activity of the storage medium;

(8) To prevent unacceptable levels of stress in the fuel elements;

(9) To identify and track individual fuel elements and assemblies.

(f) Handling and storage systems for irradiated fuel shall be designed to permit adequate heat removal and shielding in operational states and accident conditions.

(g) Critical assemblies and subcritical assemblies are unlikely to include spent fuel or significantly irradiated fuel and therefore the requirements relating to handling and storage of spent fuel or significantly irradiated fuel might not apply.

Section 68. ***Radioactive Waste Systems***

(a) The design of a research reactor facility and its associated experimental facilities shall include provisions to enhance safety in waste management and to minimize the generation of radioactive waste. Systems shall be provided for treating solid, liquid and gaseous radioactive waste to keep the amounts and concentrations of radioactive releases as low as reasonably achievable and below authorized limits on discharges.

(b) Appropriate means, such as shielding and decay systems, to reduce the exposure of personnel and radioactive releases to the environment shall be considered in the design and provided as necessary.

(c) Means shall be provided in the design for the handling, processing, storage, removal from the site and disposal of radioactive waste. Where liquid radioactive waste is to be handled, provision shall be made for the detection of leakage and the recovery of waste, if appropriate. Where gaseous radioactive material is to be handled, provision shall be made for the detection of leakage and the prevention and control of releases to below authorized limits for a radioactive release.

(d) Systems shall be provided for the handling of solid or concentrated radioactive waste and for its storage at the site for a reasonable period of time.

SUPPORTING SYSTEMS AND AUXILIARY SYSTEMS

Section 69. *Performance of Supporting Systems and Auxiliary Systems*

(a) The design of supporting systems and auxiliary systems for a research reactor shall be such as to ensure that the performance of these systems is consistent with the safety significance of the system or component that they serve at the research reactor.

(b) The failure of any auxiliary system, irrespective of its importance to safety, shall not be able to jeopardize the safety of the reactor. Adequate measures shall be taken to prevent the release of radioactive material to the environment in the event of the failure of an auxiliary system containing radioactive material.

Section 70. *Fire Protection Systems*

(a) Fire protection systems for a research reactor facility, including fire detection systems and fire extinguishing systems, fire containment barriers and smoke control systems, shall be provided throughout the research reactor facility, with due account taken of the results of the fire hazard analysis.

(b) The fire protection systems installed at the research reactor shall be capable of dealing safely with postulated fire events. The design of the fire protection system shall consider the potential for accidental criticality in a critical assembly or subcritical assembly. Fire hazards due to experiments shall be considered.

(c) Fire extinguishing systems shall be capable of automatic actuation where appropriate. Fire extinguishing systems shall be designed and located to ensure that their rupture or spurious or inadvertent operation would not impair the capability of items important to safety.

(d) Fire detection systems shall be designed to provide alarms and prompt information on the location and spread of fires that start in the reactor facility at any time.

(e) Fire detection systems and fire extinguishing systems that are necessary to protect against a possible fire following a postulated initiating event shall be appropriately qualified to resist the effects of the postulated initiating event.

(f) Non-combustible or fire retardant and heat resistant materials shall be used wherever practicable throughout the facility, in particular in locations such as the means of confinement and the control rooms.

Section 71. ***Lighting Systems***

Adequate lighting shall be provided in all operational areas of a research reactor facility for operational states and in accident conditions.

Section 72. ***Lifting Equipment***

(a) Equipment shall be provided for lifting and lowering items important to safety at a research reactor facility, and for lifting and lowering other items in the proximity of items important to safety.

(b) The lifting equipment shall be designed so that:

(1) Measures are taken to prevent the lifting of excessive loads, including those for experimental programs;

(2) Conservative design measures are applied to prevent any unintentional dropping of loads that could affect items important to safety or could cause a radiological hazard;

(3) The facility layout permits safe movement of the lifting equipment and of items being transported, in accordance with analyzed safe load pathways;

(4) Such equipment for use in areas where items important to safety are located is seismically qualified;

(5) Such equipment can be inspected on a periodic basis.

Section 73. ***Air Conditioning Systems and Ventilation Systems***

(a) Systems for air conditioning, air heating, air cooling and ventilation for a research reactor facility shall be provided as appropriate in areas at the facility to maintain the required environmental conditions.

(b) Systems shall be provided for the ventilation of buildings at the reactor facility with appropriate capability for the conditioning and cleaning of air:

(1) To prevent unacceptable dispersion of airborne radioactive substances within the facility;

(2) To reduce the concentration of airborne radioactive substances to levels compatible with the need for access by personnel to the area;

(3) To keep the levels of airborne radioactive substances in the reactor facility below authorized limits and as low as reasonably achievable;

(4) To ventilate rooms containing inert gases or noxious gases without impairing the capability to control radioactive effluents;

(5) To maintain the required efficiency of the filtration system and to control releases of gaseous radioactive material to the environment and maintain them below the authorized limits on discharges and to keep them as low as reasonably achievable.

Section 74. ***Compressed Air Systems***

The design basis for any compressed air system that serves an item important to safety at a research reactor facility shall specify the quality, flow rate and cleanness of the air to be provided.

Section 75. ***Experimental Devices***

(a) Experimental devices for a research reactor shall be designed so that they will not adversely affect the safety of the reactor in any operational states or accident conditions. In particular, experimental devices shall be designed so that neither the operation nor the failure of an experimental device will result in an unacceptable change in reactivity for the reactor, affect operation of the reactor protection system, reduce the cooling capacity, compromise confinement or lead to unacceptable radiological consequences.

(b) A design basis shall be established for each experimental device associated directly or indirectly with the reactor. Experimental devices shall be classified on the basis of their importance to safety. The radioactive inventory of the experimental device as well as the potential for the generation or release of energy shall be taken into consideration. A safety analysis shall also be performed, including an analysis of the damage that would be caused to the experimental devices by the postulated initiating events of the reactor. The safety analysis shall also cover the interaction between the experimental devices and the reactor.

(c) Where necessary for the safety of the reactor and the safety of the experiment, the design shall provide appropriate monitoring of the parameters for experiments in the reactor control room.

(d) The design of experiments and experimental devices shall facilitate their dismantling operations, interim storage and final disposition.

VI. OPERATION OF RESEARCH REACTOR FACILITIES

Section 76. ***Operational Limits and Conditions (OLCs)***

(a) General.

(1) The licensee shall ensure that the research reactor, including critical and subcritical assembly, is operated in accordance with the OLCs.

(2) The facility shall be operated within the OLCs to prevent situations arising that could lead to anticipated operational occurrences or accident conditions, and to mitigate the consequences of such events if they do occur.

(3) The OLCs shall be developed to ensure that the reactor is being operated in accordance with the design assumptions and intent, as well as in accordance with its license conditions.

(4) The set of OLCs important to reactor safety shall include safety limits, safety system settings, limiting conditions for safe operation, requirements for surveillance, testing and maintenance, and administrative requirements. The OLCs shall be submitted to the PNRI for approval before the commencement of operation. All OLCs shall be substantiated by a written statement or by analysis of the reason for their adoption.

(5) The OLCs shall be adequately defined, clearly established and appropriately substantiated by clearly stating for each OLC its objective, its applicability, its specification and its basis. The selection of, and the values for, the OLCs shall be based on the safety analysis, on the reactor design or on aspects relating to the conduct of operations. The OLCs shall be demonstrably consistent with the updated safety analysis report, shall reflect the present status of the reactor and shall correspond to the license conditions imposed by the PNRI.

(b) Safety Limits.

Safety limits shall be set to protect the integrity of the physical barriers that protect against the uncontrolled release of radioactive material or exposure over limits established in CPR Part 3.

(c) Safety System Settings.

Safety system settings shall be defined so that the safety limits are not exceeded.

(d) Limiting Conditions for Safe Operation.

(1) Limiting conditions for safe operation shall be established to ensure that there are acceptable margins between normal operating values and the safety system settings. Limiting conditions for safe operation shall include limits on operating parameters, requirements relating to the minimum availability of operable equipment and minimum staffing levels, and prescribed actions to be taken by operating personnel to preserve the settings of the safety system.

(2) Maximum authorized unavailability limits for operation of the research reactor shall be established for items important to safety to ensure the reliable performance of safety functions. The unavailability limits shall be documented in the operational limits and conditions.

(e) Requirements for Maintenance, Periodic Testing and Inspection.

(1) Requirements shall be established for the frequency and scope of inspections, periodic testing and maintenance, operability checks and calibrations of all items important to safety to ensure compliance with the safety analysis report.

(2) The requirements for maintenance, surveillance, periodic testing and inspection shall include a specification that clearly defines the objectives and the applicability, prescribes the frequency for the performance of activities and establishes criteria for acceptable deviations. In order to provide operational flexibility, the specification shall prescribe the frequency of activities in terms of average intervals with a maximum interval that is not to be exceeded. Deferrals that exceed the maximum interval shall be justified and made subject to approval, and safety measures shall be put in place where necessary.

(f) Administrative Requirements.

The OLCs shall include administrative requirements or controls concerning organizational structure and the responsibilities for key positions for the safe operation of the reactor, staffing, the training and retraining of facility personnel, review and audit procedures, modifications, experiments, records and reports, and required actions following a violation of the OLCs.

(g) Violations of OLCs.

(1) In the event that the operation of the reactor deviates from one or more OLCs, corrective actions shall be taken.

(2) Actions shall be prescribed to be taken by the operating staff within an allowed time if a limiting condition for safe operation is violated. The reactor management shall conduct an investigation of the cause and the consequences and shall take appropriate actions to prevent a recurrence. The PNRI shall be notified in due time.

(3) If a safety limit is exceeded, the reactor shall be shut down and maintained in a safe state and inspections on challenged items important to safety shall be performed. Under such circumstances, the PNRI shall be promptly notified, an investigation of the cause shall be carried out by the licensee and a report shall be submitted to the PNRI for assessment before the reactor is returned to operation.

Section 77. ***Performance of Safety Related Activities***

(a) The licensee shall ensure that safety related activities are adequately analyzed and controlled to ensure that the risks associated with harmful effects of ionizing radiation are kept as low as reasonably achievable.

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

(c) All activities important to safety shall be carried out in accordance with approved written procedures to ensure that the research reactor is operated within the established OLCs. Acceptable margins shall be ensured between normal operating values and the established safety system settings to avoid undesirably frequent actuation of safety systems.

(d) No experiments shall be conducted without adequate review and justification. If there is a need to conduct a non-routine operation or test that is not covered by existing operating procedures, a specific safety review shall be performed and a special procedure shall be developed and made subject to approval in accordance with national or other relevant regulations.

Section 78. *Commissioning Program*

(a) The licensee shall prepare an adequate commissioning Program for the testing of reactor components and systems after their construction or modification to demonstrate that they are in accordance with the design objective and meet the performance criteria. The commissioning Program shall:

(1) establish the organization and responsibilities for commissioning, the commissioning stages, the suitable testing of SSCs on the basis of their importance to safety, the test schedule, the commissioning procedures and reports, the methods of review and verification, the treatment of deficiencies and deviations, and the requirements for documentation;

(2) include provisions and procedures for audits, reviews, and verifications intended to ensure that it has been conducted as planned and that its objective have been fully achieved; and

(3) be submitted to PNRI for review and approval before being implemented.

(b) The licensee shall carry out a comparison during construction and commissioning between the as built reactor facility and its design parameters. A comprehensive process shall be established under the management system of the licensee to address non-conformances in design, manufacturing, construction and operation. Resolutions to correct differences from the initial design and non-conformances shall be documented and reviewed before starting the commissioning.

(c) Experimental devices and their potential impact on reactor operations shall be given adequate consideration by the licensee during the commissioning of the reactor. Experimental devices shall be subject to an adequate commissioning Program prior to being placed in service.

Section 79. *Organization and Responsibilities for Commissioning*

(a) The licensee, designers and manufacturers shall be involved in the preparation and execution of the commissioning Program. The commissioning process shall involve cooperation between the licensee and the supplier to ensure an effective means of familiarizing the licensee with the characteristics of the particular reactor.

(b) The licensee shall maintain close liaison with the PNRI throughout the commissioning process. In particular, the results and analyses of tests directly affecting safety shall be submitted to PNRI for approval.

Section 80. *Commissioning Tests and Stages*

(a) The licensee shall arrange a commissioning tests in functional groups and in a logical sequence. Commissioning test shall include:

(1) Stage A: Test prior to fuel loading or pre-operational tests;

(2) Stage B: Fuel loading tests, initial criticality tests, low power tests; and

(3) Stage C: Power ascension and power tests.

(b) No test sequence shall proceed unless the required previous steps have been successfully completed.

Section 81. *Commissioning Procedures and reports*

(a) The licensee shall prepare, review, and made subject to approval commissioning procedures for each commissioning test prior to the commencement of the tests. Commissioning activities shall be performed in accordance with approved written procedures. The commissioning procedures shall:

(1) include hold points for the notification and involvement of the safety committee, external agencies, manufacturers and the PNRI; and

(2) be prepared in accordance with the management system covering the scope, sequence and expected results of these tests.

(b) The licensee shall keep the commissioning reports for the entire lifetime of the facility including the decommissioning stage. The reports shall cover the following:

(1) The purpose of the tests and the expected results;

(2) The safety provisions required to be in force during the tests;

(3) Precautions and prerequisites;

(4) The test procedures;

(5) The test reports, including a summary of the data collected and their analysis, an evaluation of the results, the identification of deficiencies, if any, and any necessary corrective actions.

(c) The licensee shall make available and maintain the results of all commissioning tests for the lifetime of the facility.

Section 82. ***Operating Procedures***

(a) The licensee shall develop operating procedures for the research reactor and its associated facilities that apply comprehensively for normal operation, anticipated operational occurrences and accident conditions, in accordance with the policy of the licensee and the requirements of the PNRI.

(b) The licensee shall develop operating procedures for normal operation to ensure that the reactor is operated within the OLCs.

(c) The licensee shall develop operating procedures for all safety related operations that may be conducted over the entire lifetime of the facility, including for:

(1) Commissioning;

(2) Operation in normal operational states;

(3) The maintenance of major components or systems that could affect reactor safety;

(4) Periodic inspections, calibrations and tests of items important to safety;

(5) Radiation protection activities;

(6) The review and approval process for operation and maintenance and the conduct of irradiation and experiments that could affect reactor safety or the reactivity of the core;

(7) The reactor operator's response to anticipated operational occurrences and design basis accidents, and, to the extent feasible, to design extension conditions;

(8) Emergencies;

(9) Handling of radioactive waste and monitoring and control of radioactive releases;

(10) Utilization;

(11) Modifications;

(12) The management system.

(d) The operating procedures shall be:

(1) developed by the reactor operating personnel, in cooperation whenever possible with the designer and manufacturer and with other staff of the licensee, including radiation protection staff;

(2) reviewed and updated by the licensee in accordance with the management system; and

(3) made available as relevant for the particular mode of operation of the reactor.

(e) The licensee shall ensure that all personnel involved in the operation and use of the reactor are adequately trained in the use of these procedures.

(f) When activities that are not covered by existing procedures are planned, an appropriate procedure shall be prepared by the licensee in accordance with the management system before the activity is started. Additional training of relevant staff in these procedures shall be provided.

Section 83. ***Main Control Room, Supplementary Control Room and Control Equipment***

The licensee shall:

(1) ensure that the operation control rooms and control equipment are maintained in a suitable condition;

(2) develop clear communication lines for ensuring an adequate transfer of information to the operators in the main control room where the design of the research reactor foresees additional or local control rooms that are dedicated to the control of experiments that could affect the reactor conditions;

(3) keep operable and free from obstructions, as well as from non-essential material that would prevent their operation, the supplementary control room or a shutdown panel and all other safety related local control rooms or operational panels outside the control room;

(4) periodically confirm that the supplementary control room or shutdown panel and all other safety related operational panels are in the proper state of operational readiness, including proper documentation, communications and alarm systems as well as sufficient power supply; and

(5) establish a hierarchy of precedence between the supplementary and the main control rooms to prevent conflicting inputs being given from different control rooms or panels.

Section 84. ***Material Conditions and Housekeeping***

(a) The licensee for shall develop and implement programs to maintain a high standard of material conditions, housekeeping and cleanliness in all working areas.

(b) The licensee shall establish administrative controls to ensure that operational premises and equipment are maintained, well lit and accessible, and that temporary storage is controlled and limited. Equipment that is degraded shall be identified, reported and corrected in a timely manner.

(c) The licensee shall ensure that the identification and labelling of safety equipment and safety related equipment, rooms, piping and instruments are accurate, legible and well maintained, and that they do not introduce any degradation.

Section 85. *Maintenance, Periodic Testing and Inspection*

(a) The licensee shall establish and implement programs for maintenance, periodic testing and inspection.

(b) Maintenance, periodic testing and inspection shall be:

   (1) conducted following approved written procedures to ensure that SSCs are able to function in accordance with the design intent, in compliance with the OLCs; and

   (2) reviewed periodically to incorporate lessons learned from experience.

(c) In accordance with the requirements of the management system, a system of work permits shall be used for maintenance, periodic testing and inspection, including appropriate procedures and checklists before and after the conduct of the work.

(d) Non-routine inspections or corrective maintenance of systems or items important to safety shall be performed to a specially prepared plan and procedures. In-service inspections conducted for safety purposes and on a programmatic basis shall be performed in a similar manner.

(e) The decision to carry out maintenance work on installed equipment, to remove equipment from operation for maintenance purposes or to reinstall equipment after maintenance shall be:

   (1) the responsibility of the reactor manager;

   (2) in accordance with the objective of maintaining the level of safety of the reactor as specified in the OLCs.

(f) The frequency of maintenance, periodic testing and inspection of individual SSCs shall be adjusted on the basis of experience and shall be such as to ensure adequate reliability.

(g) Equipment and items used for maintenance, periodic testing and inspection shall be identified and controlled to ensure their proper use.

(h) Maintenance shall not be performed in such a way as to result, either deliberately or unintentionally, in changes to the design of the system under maintenance. If a maintenance activity requires a design change, procedures for the implementation of a modification shall be followed.

(i) The licensee shall designate properly qualified personnel, who shall verify that the activities have been accomplished as specified in the appropriate procedures, demonstrate compliance with the OLCs and assess the results of maintenance, periodic testing and inspection.

(j) The licensee shall inform PNRI of any non-conformance that is significant to safety. The licensee shall assess the impact of the non-conformance on the maintenance Program.

Section 86. ***Core Management and Fuel Handling***

(a) The licensee shall:

(1) establish core management and fuel handling procedures for a research reactor, including critical and subcritical assembly, to ensure compliance with OLCs and consistency with the utilization Program; and

(2) document in the OLCs applicable safety requirements for core management and fuel handling.

(b) Core components and fuel loaded into the core shall comply with the quality requirements established in the management system.

(c) To ensure safe operational cores, in addition to the demonstration of conformance with the safety analysis report and OLCs, the licensee shall:

(1) determine, using validated methods and codes, the locations for fuel and reflectors, the appropriate positions of experimental devices and moderators in the core and the effectiveness of the safety devices as well as the relevant thermohydraulic and neutronic parameters.

(2) analyze the possible interactions, both chemical and physical, between core components and with experimental devices.

(3) keep and update information on the parameters for the fuel and core configurations.

(4) detect, identify and unload failed fuel.

(5) load and unload fuel in accordance with the procedures for core management and fuel handling.

(d) In addition to the above activities, other activities shall be undertaken in the core management Program to ensure the safe use of the fuel in the core or to facilitate the basic activities for core management, such as:

(1) The assessment of the safety implications of any core component or material proposed for irradiation;

(2) The assessment of the effects of irradiation on core components and core support structure materials.

(e) For fuel handling activities, the licensee shall:

(1) prepare procedures for the handling of fuel assemblies and core components to ensure their quality and safety and to avoid damage or degradation;

(2) establish OLCs and prepare procedures for dealing with failures of fuel elements, control rods, reflectors or moderators, experimental devices or any other core components so as to minimize the amounts of radioactive material released;

(3) continuously monitor the integrity of the reactor core and the fuel by a system for the detection of failures of the cladding integrity;

(4) investigate causes of fuel failures;

(5) store failed fuel in a manner that prevents the release of radioactive material while still maintaining the requisite degree of residual heat removal and shielding and subcriticality conditions; and

(6) carry out the packaging and transport of fresh and irradiated fuel assemblies in accordance with national and international requirements and, as appropriate, in accordance with CPR Part 4.

(f) The licensee shall maintain a comprehensive records system in compliance with the management system to cover core management and the handling and storage of fuel, and core components.

Section 87. *Fire Safety*

(a) The licensee shall make arrangements for ensuring fire safety. These arrangements shall cover the following:

(1) adequate management for fire safety;

(2) preventing fires from starting;

(3) detecting and extinguishing quickly any fires that do start;

(4) preventing the spread of those fires that have not been extinguished;

(5) providing protection from fire for SSCs that are necessary to shut down the reactor safely;

(6) applying of the principle of defence in depth;

(7) controlling of combustible materials and ignition sources;

(8) maintaining, testing and inspecting of fire protection measures;

(9) establishing of a manual firefighting capability at the reactor facility;

(10) assigning responsibilities and training and exercising of personnel; and

(11) assessing of the impact of modifications on fire safety measures.

(b) The licensee shall give attention to cases for which there is a risk of release of radioactive material in a fire and establish appropriate measures for the radiation protection of firefighting personnel and the management of releases of radioactive material to the environment.

(c) The licensee shall develop a comprehensive fire hazard analysis for the research reactor, including critical and subcritical assembly, and shall be periodically reviewed and, if necessary, updated.

Section 88. ***Non-Radiation-related Safety***

The licensee shall establish and implement a Program to ensure that safety related risks associated with non-radiation-related hazards to personnel involved in activities at the reactor facility are kept as low as reasonably achievable.

Section 89. ***Emergency Preparedness***

The licensee shall:

(a) Prepare emergency arrangements for preparedness for, and response to, a nuclear or radiological emergency. The emergency arrangements shall be commensurate with the hazards assessed and the potential consequences of an emergency should it occur. Emergency arrangements shall cover the following:

(1) capability of maintaining protection and safety in the event of an emergency;

(2) mitigating the consequences of accidents if they do occur;

(3) protection of site personnel, emergency workers and the public;

(4) protection of the environment;

(5) communicating with the public;

(6) prompt recognition, classification, declaration and notification of an emergency;

(7) timely initiation of coordinated and pre-planned response;

(8) assessment of the progress of the emergency, monitoring of radioactive releases, its consequences and any actions that need to be taken on the site;

(9) treatment and first aid of a limited number of contaminated and/or overexposed person; and

(10) necessary provision of information to the off-site authorities.

(b) Establish appropriate emergency arrangements from the time that nuclear fuel is first brought to the site, and all emergency arrangements shall be completed before the commencement of fuel loading.

(c) Develop emergency plans and procedures for on-site preparedness and response to an emergency in relation to the research reactor, including critical and subcritical assembly. Emergency plans and procedures shall be:

(1) based on the accidents analyzed in the safety analysis report as well as those additionally postulated for the purposes of emergency preparedness and response on the basis of the hazard assessment;

(2) based on lessons learned from relevant experience, incidents, and accidents;

(3) periodically reviewed and shall be revised as necessary to ensure that feedback from experience and other changes; and

(4) subject to approval by the PNRI.

(d) Control on-site emergencies according to the approved arrangements and shall coordinate emergency activities with off-site response organizations with responsibilities in emergency preparedness and response in accordance with the RADPLAN.

(e) Have sufficient number of qualified and trained personnel to respond to emergencies. All personnel involved shall be retrained periodically in accordance with their assigned duties and shall be fit for their intended duty.

(f) Conduct exercises to test emergency arrangements at suitable intervals and shall involve, to the extent practicable, all persons with duties in responding to the emergency.

(g) Keep facilities, instruments, tools, equipment, documentation and communication systems available and in good operating condition to be used in an emergency, including those necessary for communication with off-site authorities.

(h) Ensure that the relevant information on the research reactor safety parameters and facility conditions is available in the emergency center and that communication is effective between the control rooms and the emergency center in the event of an accident. These capabilities shall be tested periodically.

Section 90. ***Records and Reports***

The licensee shall:

(a) Establish and maintain a system for the control of records and reports.

(b) Retain full details of the design requirements and of information relating to the site and its final design, construction and modification, such as the ‘baseline’ radiological characterization, as built drawings relating to the facility’s layout, piping and cable penetrations, as information necessary for decommissioning;

(c) Retain all essential information concerning the design, construction, commissioning, current configuration and operation of the reactor;

(d) Maintain information up to date throughout the operational stage of the reactor and shall be kept available during decommissioning;

(e) Document and keep updated information on experience with the handling of contaminated or activated SSCs in the maintenance or modification of the reactor to facilitate the planning of decommissioning;

(f) Develop administrative procedures consistent with the management system for the generation, collection, retention and archiving of records and reports. Information entries in logbooks, checklists and other appropriate records shall be properly dated and signed; and

(g) Prepare and retain records of non-compliance and the measures taken to return the research reactor, including critical and subcritical assembly, to compliance, and shall be made available to the PNRI.

(h) The licensee shall specify the records to be retained and their retention periods, in accordance with the period specified by the appropriate regulations, license condition or OLCs. If a retention period is not otherwise specified, these records must be retained until the PNRI terminates the license or, in the case of a provisional permit, until the permit expires.

(i) The licensee shall ensure that the arrangements made for storing and maintaining records and reports are in accordance with the management system.

Section 91. ***Utilization and Modification of a Research Reactor***

(a) The licensee shall establish and implement a Program to manage utilization and modifications of the reactor.

(b) The licensee shall have the overall responsibility for all safety aspects of the preparation and performance of a modification or experiment.

(c) The licensee shall be responsible for ensuring the following:

(1) Safety analyses of the proposed utilization or modification are conducted to ascertain whether all applicable safety requirements and provisions have been satisfied.

(2) Limiting conditions for safe operation are prepared for utilization equipment and activities and are incorporated into the OLCs of the research reactor, including critical and subcritical assembly.

(3) The relevant safety documentation for the experiment or modification is prepared and submitted to PNRI for approval when appropriate.

(4) The disposition path of any materials irradiated in the experiment is defined and made subject to approval.

(5) All personnel who will be involved in making a proposed modification or in conducting the proposed utilization are suitably trained, qualified and experienced.

(6) All documents affected by the experiment or modification that relate to the safety characteristics of the reactor, the OLCs, and the relevant procedures for operation, maintenance and emergencies, are updated as necessary, prior to the new utilization or to the commissioning of the modification.

(7) Safety precautions and controls are applied with regard to all personnel involved in the performance of the experiment or modification.

(d) Proposals for the utilization and modification of the research reactor, including critical and subcritical assembly, shall be categorized and relevant criteria for this categorization shall be established.

(e) Utilization and modification projects including temporary modifications, having major safety significance shall be subject to safety analyses and to procedures for design, construction and commissioning.

(f) The licensee shall establish a procedure for the review and approval of proposals for experiments and modifications and for the control of their performance.

(g) The use and handling of experimental devices shall be controlled by means of written procedures. The possible effects on the reactor, particularly changes in reactivity or radiation levels, shall be taken into account in these procedures.

(h) Temporary modifications shall be limited in time and number to minimize their cumulative safety significance. Temporary modifications shall be clearly identified at their location and at any relevant control position. Personnel shall be informed of temporary modifications and of their consequences for the operation and safety of the facility.

(i) Any modifications made to experimental devices shall be subject to the same procedures for design, operation and approval as were followed for the original experimental device.

Section 92. ***Radiation Protection Program***

(a) The licensee shall establish and implement a radiation protection Program. The radiation protection Program shall:

(1) ensure that for all operational states and accident conditions, doses due to exposure to ionizing radiation at the research reactor facility or doses due to any planned releases of radioactive material from the facility are kept below authorized limits and are as low as reasonably achievable;

(2) have sufficient independence and resources to be able to advise on and enforce radiation protection regulations, standards and procedures, and safe working practices;

(3) be established by the licensee in accordance with CPR Part 3 and shall be subject to the approval of the PNRI;

(4) include a policy statement from the licensee that includes the fundamental safety objective of protecting people and the environment and statement of the licensee’s commitment to the principle of optimisation of protection;

(5) include in particular measures for the following:

(i) Ensuring that there is cooperation between the radiation protection staff and other operating staff and experimental staff in establishing operating procedures and maintenance procedures when radiation hazards are anticipated, and ensuring that direct assistance is provided when required;

(ii) Providing workplace monitoring and environmental monitoring;

(iii) Providing for the decontamination of personnel, equipment and structures;

(iv) Verifying compliance with applicable regulations for the transport of radioactive material;

(v) Detecting and recording any releases of radioactive material;

(vi) Recording the inventory of radiation sources;

(vii) Providing adequate training in practices for radiation protection;

(viii) Providing for the review and update of the Program in the light of experience;

(ix) Providing the review and analysis of materials, equipment and conditions for experiments.

(b) The licensee shall verify, by means of surveillance, inspections and audits, that the radiation protection Program is being correctly implemented and that its objectives are being met. The radiation protection Program shall be reviewed on a regular basis and shall be updated if necessary.

(c) The licensee shall establish dose constraints to ensure that radiation doses are kept as low as reasonably achievable.

(d) If the applicable dose limits for occupational or public exposure or the authorized limits for radioactive releases are exceeded, the PNRI shall be informed.

(c) All personnel who may be occupationally exposed to radiation at significant levels shall have their doses measured, assessed and recorded the and made available to the health surveillance Program, the reactor manager and to PNRI in accordance with CPR Part 3.

Section 93. *Management of Radioactive Waste*

(a) The licensee shall establish and implement a Program for the management of radioactive waste.

(b) The Program for the management of radioactive waste shall be in accordance with CPR Part 3. Processing and storage of radioactive waste shall be controlled in a manner consistent with the requirements for the predisposal management of radioactive waste.

(c) The reactor and its experimental devices shall be operated to minimize the generation of radioactive waste of all kinds, to ensure that releases of radioactive material to the environment are kept as low as reasonably achievable in accordance with CPR Part 3 and to facilitate the handling and disposal of waste.

(d) Releases of liquid and/or gaseous radioactive effluents to the environment shall be monitored and the results shall be recorded in order to verify compliance with the authorized limits in accordance with CPR Part 3. They shall be reported periodically to the PNRI.

(e) Written procedures shall be followed for the handling, processing, transport and storage of radioactive waste.

(f) An appropriate record shall be kept of the quantities, types and characteristics of the radioactive waste processed and stored on the reactor site or removed from the reactor site for the purpose of processing, storage or disposal.

Section 94. ***Ageing Management***

(a) The licensee shall ensure that an effective ageing management Program is implemented to manage the ageing of items important to safety so that the required safety functions of SSCs are fulfilled over the entire operating lifetime of the research reactor, including critical and subcritical assembly.

(b) The ageing management Program shall:

(1) determine the consequences of ageing and the activities necessary to maintain the operability and reliability of SSCs;

(2) be coordinated with, and be consistent with, other relevant programs, including the programs for in-service inspections, periodic safety review and maintenance;

(3) adopt a systematic approach to provide for the development, implementation and continuous improvement;

(4) include inspection and periodic testing of materials, the results of which are to be used in reviewing the adequacy of the design at appropriate interval; and

(5) include the management of obsolete SSCs and the management of spare parts.

Section 95. ***Extended Shutdown***

(a) If an extended shutdown is planned or occurs, the licensee shall establish and implement arrangements to ensure the safe management, planning, effective performance and control of work activities during the extended shutdown.

(b) The licensee shall take appropriate measures during an extended shutdown to ensure that materials and components do not seriously degrade. The following measures shall be considered:

(1) Unloading the fuel elements from the reactor core to appropriate and safe storage conditions;

(2) Changing the OLCs in accordance with the requirements for the shutdown reactor;

(3) Removing components for protective storage;

(4) Taking measures to prevent accelerated corrosion and ageing;

(5) Retaining adequate staff in the facility for the purposes of performing the necessary maintenance, periodic testing and inspection.

(c) The licensee shall be responsible for establishing programs and issuing procedures for managing extended shutdown and for the provision of adequate resources for ensuring the safety of activities during extended shutdown. Priority shall be given to safety related considerations in the process of planning and performing activities in the extended shutdown state. Special attention shall be given to maintaining the reactor configuration up to date in accordance with the OLCs.

(d) The licensee shall take the necessary decisions as soon as possible to reduce the period of extended shutdown to a minimum. During a period of extended shutdown, the licensee shall continue to comply with license conditions, and requirements for emergency planning and for the qualification of the operating staff unless PNRI approves the license conditions modification based on justification by the licensee. Security shall be provided for as long as nuclear fuel or other radioactive material is present at the facility.

Section 96. ***Feedback of Operating Experience***

(a) The licensee shall establish a Program to learn from events at the reactor facility and events in other research reactors and from the nuclear industry.

(b) The licensee shall report, collect, screen, analyze, trend, document and communicate operating experience at the reactor facility in a systematic way. These activities shall be performed in accordance with the management system.

(c) Events with significant implications for safety shall be investigated to identify their direct, root and/or contributory causes, including causes relating to equipment design, operation and maintenance, or to human and organizational factors. The results of such analyses shall be included, as appropriate, in relevant training programs and shall be used in reviewing procedures and instructions.

(d) Information on operating experience shall be examined by competent persons for any precursors to, or trends in, adverse conditions for safety so that any necessary corrective actions can be taken before serious conditions arise.

VII. PREPARATION FOR DECOMMISSIONING OF A RESEARCH REACTOR

The licensee shall be responsible for the preservation of knowledge of the reactor facility and for the retention of key personnel to facilitate decommissioning.

The implications for safety of the activities in the transition period, if any, between permanent shutdown of operation and approval of the final decommissioning plan shall be assessed and shall be managed by the licensee so as to avoid undue hazards and to ensure safety.

Section 97. ***Decommissioning Plan***

(a) The licensee shall prepare a decommissioning plan and shall maintain it throughout the lifetime of the research reactor, unless otherwise approved by the PNRI, to demonstrate that decommissioning can be accomplished safely and in such a way as to meet the specified end state. The decommissioning plan shall:

(1) be prepared at the design stage and shall be updated in accordance with changes in regulatory requirements, modifications to the SSCs, advances in technology, changes in the need for decommissioning activities and changes in national policies for decommissioning and/or the management of radioactive waste;

(2) be submitted for PNRI approval before decommissioning activities are commenced; and

(3) include an evaluation of one or more approaches to decommissioning that are appropriate for the reactor concerned and are in compliance with the requirements of the PNRI

(b) In developing the decommissioning plan, all aspects of the design shall be reviewed. The plan shall include all steps to completion of decommissioning to the point that safety can be ensured with minimum or no surveillance.

(c) The licensee shall establish in advance procedures for the handling, dismantling and disposal of experimental devices and other irradiated equipment that require storage and eventual disposal if the equipment concerned has already been constructed and such procedures are not in place.

Section 98. ***Decommissioning of a Research Reactor***

The licensee shall prepare and maintain relevant documents and records before, during and after decommissioning, until such time that the PNRI authorizes records disposal.

VIII. INTERFACES BETWEEN SAFETY AND SECURITY FOR A RESEARCH REACTOR

Section 99. *Requirements for the Interfaces between Nuclear Safety and Nuclear Security*

(a) The licensee shall:

(1) design, implement and maintain technical and administrative measures relating to the interfaces between safety and security;

(2) maintain coordination with relevant organizations that are involved in safety and security;

(3) ensure availability of adequate trained personnel with knowledge and skills relating to the interfaces between safety and security, as part of the management system;

(4) establish and implement adequate measures at all stages in the lifetime of the research reactor, including critical and subcritical assembly, to ensure effective communication and coordination among individuals with different objectives and backgrounds to ensure that safety measures and security measures do not compromise one another;

(5) select a research reactor site based on both safety and security related criteria;

(6) establish a change control process to ensure that any proposed changes of design, including new experimental facilities, of the layout of the research reactor facility or of procedures are evaluated to verify that they do not jeopardize safety or security;

(7) implement measures to prevent the inadvertent or intentional introduction of weaknesses, devices or any threat that could lead to a security breach or radioactive releases during operation and utilization of the reactor; and

(8) implement adequate measures during the operation stage to ensure effective management of the interfaces between safety and security. The management of the interfaces shall ensure appropriate balance between safety and security. In case of long shutdown periods and extended shutdown, safety and security of fuel shall be established and maintained.

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

IX. EFFECTIVITY

Section 100. ***Effectivity Date.***

This Part shall take effect fifteen (15) days following the publication in the Official Gazette or in a newspaper of general circulation.

APPROVED:

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Director

Date: 23 June 2020

APPENDIX I. SELECTED POSTULATED INITIATING EVENTS FOR RESEARCH REACTORS

I.1. The following are examples of selected postulated initiating events for research reactors. Certain research reactors may have additional postulated initiating events depending on specific characteristics of the design:

(a) Loss of electrical power supplies: Loss of normal electrical power.

(b) Insertion of excess reactivity: Criticality during fuel handling and loading (due to an error in fuel insertion);
- Startup accident;
- Control rod failure or control rod follower failure;
- Control drive failure or control drive system failure;
- Failure of other reactivity control devices (such as a moderator or reflector);
- Unbalanced rod positions;
- Failure or collapse of structural components;
- Insertion of cold or hot water;
- Changes in the moderator (e.g. voids, leakage of D2O into H2O systems or leakage of H2O into D2O systems);
- Effects of experiments and experimental devices (e.g. flooding or voiding, temperature effects, insertion of fissile material or removal of absorber material);
- Insufficient shutdown reactivity;
- Inadvertent ejection of control rods;
- Maintenance errors with reactivity devices;
- Spurious control system signals;
- Removal of poisons from the coolant or moderator.

(c) Loss of flow: Primary pump failure;
- Reduction in flow of primary coolant (e.g. due to valve failure or a blockage in piping or a heat exchanger);
- Effect of the failure or mishandling of an experiment;
- Rupture of the primary coolant boundary leading to a loss of flow;
- Fuel channel blockage or flow reduction (e.g. due to foreign material);
- Improper power distribution due to, for example, unbalanced rod positions in core experiments or in fuel loading (power–flow mismatch);
- Reduction in coolant flow due to bypassing of the core;
- Deviation of system pressure from the specified limits;
- Loss of heat sink (e.g. due to the failure of a valve or pump or a system rupture).

(d) Loss of coolant: Rupture of the primary coolant boundary;
- Damaged pool;
- Pump-down of the pool;
- Failure of beam tubes or other penetrations.

(e) Erroneous handling or failure of equipment or components: Failure of the cladding of a fuel element;

- Mechanical damage to core or fuel (e.g. mishandling of fuel or dropping of a transfer flask onto fuel);
- Failure of the emergency core cooling system;
- Malfunction of the reactor power control;
- Criticality in fuel in storage;
- Failure of the means of confinement, including the ventilation system;
- Loss of coolant to fuel in transfer or storage;
- Loss or reduction of proper shielding;
- Failure of experimental apparatus or material (e.g. loop rupture);
- Exceeding of fuel ratings.

(f) Special internal events: Internal fires or explosions, including internally generated missiles;
  - Internal flooding;
  - Loss of support systems;
  - Security related incidents;
  - Malfunctions in reactor experiments;
  - Improper access by persons to restricted areas;
  - Fluid jets or pipe whip;
  - Exothermic chemical reactions;
  - Drop of heavy loads.

(g) External events: Earthquakes (including seismically induced faulting and landslides);
  - Flooding (including failure of an upstream or downstream dam and blockage of a river and damage due to a tsunami or high waves);
  - Tornadoes and tornado missiles;
  - Sandstorms;
  - Hurricanes, storms and lightning;
  - Tropical cyclones;
  - Explosions;
  - Aircraft crashes;
  - Fires;
  - Toxic spills;
  - Accidents on transport routes (including collisions into the research reactor building);
  - Effects from adjacent facilities (e.g. nuclear facilities, chemical facilities and waste management facilities);
  - Biological hazards such as microbial corrosion, structural damage or damage to equipment by rodents or insects;
  - Extreme meteorological phenomena;
  - Electromagnetic interference (e.g. from solar events);
  - Lightning strikes;
  - Power or voltage surges on the external supply line.

(h) Human errors.

APPENDIX II. OPERATIONAL ASPECTS OF RESEARCH REACTORS WARRANTING PARTICULAR CONSIDERATION

II.1. This appendix highlights operational aspects of research reactors that warrant particular consideration.

REACTIVITY AND CRITICALITY MANAGEMENT

II.2. The core configuration of a research reactor is frequently changed and these changes involve the manipulation of components, such as fuel assemblies, control rods and experimental devices, many of which represent considerable reactivity value. Care shall be taken to ensure that the relevant subcriticality limits and reactivity limits for fuel storage and core loading are not exceeded at any time.

CORE THERMAL SAFETY

II.3. The frequent changes in core loading affect the nuclear and thermal characteristics of the core. Measures shall be established to ensure, for each change, that these characteristics are correctly determined and that they are checked against the relevant conditions for nuclear and thermal safety before the reactor is put into operation.

SAFETY OF EXPERIMENTAL DEVICES

II.4. Experimental devices used in research reactors may, by virtue of their technical, nuclear or operational characteristics, significantly affect the safety of the reactor. Measures shall be taken to ensure that the technical, nuclear and operational characteristics of experimental devices are adequately assessed for their safety implications and that this assessment is suitably documented.

MODIFICATION OF RESEARCH REACTORS

II.5. Research reactors and their associated experimental devices are often modified in order to adapt their operational and experimental capabilities to changing requirements for their utilization. Special attention shall be given to the need to verify that every modification has been properly assessed, documented and reported in terms of its potential effects on safety, and that the research reactor is not restarted without formal approval after the completion of modifications with major implications for safety.

MANIPULATIONS OF COMPONENTS AND MATERIAL

II.6. In pool type research reactors in particular, components, experimental devices and material are frequently manipulated in the vicinity of the reactor core. Care shall be taken to ensure that the operating personnel carrying out these manipulations adhere strictly to the procedures and restrictions established to prevent any nuclear or mechanical interference with the reactor, to minimize the probability of a blockage in the fuel cooling system by uncontrolled foreign objects, and to prevent radioactive releases and undue radiation exposures.

SAFETY MEASURES FOR VISITORS

II.7. Guest scientists, trainees, students and other persons who visit the research reactor may have access to controlled areas and may be actively involved in the operation or utilization of the reactor. Measures such as procedures, restrictions and controls shall be established to ensure that visitors have safe working conditions that their activities will not affect the safety of the reactor and that safety instructions are strictly observed.