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Replacement Research Reactor Project

SAR CHAPTER 2
SAFETY OBJECTIVES AND ENGINEERING DESIGN REQUIREMENTS

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2 SAFETY OBJECTIVES AND ENGINEERING DESIGN REQUIREMENTS

2.1 INTRODUCTION

Safety objectives and engineering design requirements of systems, structures, and components important to the safety of the Replacement Research Reactor Facility (Reactor Facility) are specified and described in this Chapter. Emphasis is placed on the principles used in the design of the reactor.

The objectives of this chapter are as follows:

1. To identify the overall safety objectives, the safety criteria and the general requirements relating to design of the Reactor Facility.
2. To identify the design requirements relating to safety for specific systems, structures and components important to safety.
3. To identify the approach to the classification of systems, structures and components for the purpose of design or analysis, including their safety categorisation, seismic classification and quality level determination.
4. To identify the design criteria for the resistance of systems, structures and components to external events.
5. To identify the codes and standards used in the design and analysis of systems, structures and components.
6. To identify and describe the design requirements for fire protection inside the Reactor Facility.
7. To identify and describe the design basis for the qualification of components to achieve intended safety functions and resist environmental conditions.

Other chapters in this SAR demonstrate how the as-built design of the Reactor Facility meets the safety objectives and engineering design requirements identified in this chapter.

End of Section

2.2 SAFETY OBJECTIVES AND CRITERIA

2.2.1 Safety Objectives

This section identifies the fundamental design objective and the supporting objectives and principles applicable to the Reactor Facility.

2.2.1.1 Fundamental Safety Objective

The fundamental safety objective for the Reactor Facility is to protect individuals, the general public and the environment from exposure to radiation due to operation of the facility. This will be achieved by establishing and maintaining in the Reactor Facility effective defences against radiological hazards.

2.2.1.2 Radiation Protection and Technical Safety Objectives

The fundamental safety objective is supported by radiation protection and technical safety objectives as follows:

1. That radiation exposure to site personnel and members of the public from planned releases of radioactive materials from the Reactor Facility during all operational states, will remain below limits prescribed by ARPANSA and will be kept As Low As Reasonably Achievable, economic and social factors being taken into account (ALARA).
2. That all practicable measures will be taken to prevent initiating events and minimise their potential to develop into accidents.
3. That all practicable measures will be taken to mitigate the consequences from accidents that might occur.
4. That, for design basis accidents, it will be shown with a high level of confidence that radiological consequences will be small and below prescribed limits.
5. That beyond design basis accidents which may result in significant off-site consequences will be extremely unlikely, and that any potential consequences will be minor.

2.2.1.3 Defence in Depth Principle

A defence in depth approach will be adopted to ensure that the Reactor Facility's design and operation incorporates multiple levels of protection against the release of radioactive materials. The five successive levels of defence in depth are:

a) Prevention of deviations from normal operation.

b) The detection and interception of such deviations and failures in order to prevent anticipated operational occurrences from escalating into accident conditions.

c) Control of the consequences of any resulting accident conditions in the unlikely event that escalation of certain anticipated operational occurrences is not arrested by a preceding level.

d) Control of severe conditions including prevention of accident progression and mitigation of the consequences of a severe accident.

e) Mitigation of the radiological consequences of significant releases of radioactive materials.

2.2.1.4 Safety Culture Objective

All organisations involved in the design, construction and operation of the Reactor Facility will actively support the development of a safety culture which ensures that an overriding priority is given to matters involving nuclear safety and radiation protection, and that these matters receive the attention warranted by their significance. The development of a safety culture will be supported at the organisational and individual level through establishment of an appropriate safety management system, specific training and the implementation of appropriate procedures.

2.2.2 Nuclear and Radiation Safety Criteria

The Reactor Facility will be designed, constructed and maintained to meet the nuclear and radiation safety criteria of ARPANSA and the safety criteria set out in the following sections.

2.2.2.1 Safety Criterion 1: Occupational Radiation Dose Limit

The effective dose to any occupationally exposed person from all activities associated with the operation, maintenance and utilisation of the Reactor Facility during normal operation and anticipated operational occurrences will not exceed 20 mSv annually, averaged over five consecutive years. The effective dose in any one year will not exceed 50 mSv.

The effective dose from normal operation of the Reactor Facility to any occupationally exposed person will be constrained to be less than 20 mSv per year.

2.2.2.2 Safety Criterion 2: Public Radiation Dose Constraint

The effective dose from normal operation of the Reactor Facility to any member of the public at the buffer zone boundary will be constrained to be less than 0.1 mSv per year. This will ensure that the effective dose to any member of the public from all activities associated with the operation, maintenance and utilisation of the Reactor Facility during normal operation and anticipated operational occurrences will not exceed 1 mSv annually

Discharges from the Reactor Facility will comply with levels given in the Discharge Authorisation set by the Regulator. Airborne discharges from the whole site will not lead to exposure of a member of the public greater than 0.01 mSv per year.

2.2.2.3 Safety Criterion 3: Occupational and Public Radiation Dose Optimisation

The magnitude of individual doses, the number of people who are exposed, and the likelihood of incurring exposures to radiation, will be ALARA, after taking into account economic and social factors.

2.2.2.4 Safety Criterion 4: Safety System Reliability Targets

The total frequency of exceeding any of the safety limits for the reactor core will be shown to be less than 10^{-4} per year.

The reliability of each safety system (i.e. Safety Category 1 systems; see section 2.5) will be shown to be better than 10^{-3} failures to perform its intended function per demand. Where available, manufacturer’s information for reliability will be provided. When necessary, a reliability analysis will be performed.

The safety analysis will show that the frequency of any single accident sequence that results in the safety limits for the reactor core being exceeded does not dominate the overall risk.

2.2.2.5 Safety Criterion 5: Maximum Public Dose for Design Basis Accidents

The maximum effective dose to the most exposed individual member of the public resulting from any design basis accident (assessed using conservative assumptions) will be less than the minimum intervention level recommended by ARPANSA for any emergency countermeasures.

2.2.2.6 Safety Criterion 6: Risk-Dose Criterion

ARPANSA's Regulatory Assessment Principles (RAPs) specifies that probabilistic methods using best-estimate methods and data may be used to confirm that the risks to the public comply with the safety limits in the following Table.

ARPANSA – SAFETY LIMITS AND OBJECTIVES from Table 2, ARPANSA RAPs		
Maximum effective dose to most exposed member of the public (milliSieverts)	Total frequency per reactor year	
	Safety Limit	Safety Objective
0.1 to 1	1	10^{-2}
1 to 10	10^{-1}	10^{-3}
10 to 100	10^{-2}	10^{-4}
100 to 1000	10^{-3}	10^{-5}
More than 1000	10^{-4}	10^{-6}

Note that the Safety Limit is the value that will not be exceeded. The Safety Objective is the value below which further reduction is not required. Between the Safety Limit and the Safety Objective, the risk must be shown to be as low as reasonably achievable, economic and social factors being taken into account.

End of Section

2.3 GENERAL DESIGN REQUIREMENTS

This section describes the general design requirements, including the requirements for normal operation, anticipated operational occurrences and design basis accidents, which ensure that the safety objectives specified in Section 2.2 are met. They are supplemented by additional specific design requirements identified in Section 2.4.

2.3.1 Quality Assurance

The design will be performed according to the design requirements of the Project Quality Assurance Program and the Design Plan.

The Project Quality Assurance Program will be in place throughout the project prior to commencement of normal operation. It will address the design process and meet the requirements of:

a) ARPANSA RG-5, Regulatory Assessment Criteria for the Design of New Controlled Facilities and Modifications to Existing Facilities.
b) International Atomic Energy Agency (IAEA) Safety Series No. 50-C/SG-Q, Quality Assurance for Safety in Nuclear Power Plants and other Nuclear Installations, 1996.
c) IAEA Safety Standard, Safety Requirements for Research Reactors, Draft DS272 April 2001.

The INVAP Project Quality Assurance Program will be developed as part of INVAP's overall Quality Assurance (QA) system, which is certified to ISO 9001:2000. INVAP's QA system will be subject to regular audits by ANSTO and by organisations independent of both INVAP and ANSTO.

The ANSTO Project Quality Management System will be developed as part of ANSTO's overall QA system. It will also be certified to OSO- 9001:2000 and subject to regular third party audits by independent organisations.

Quality Assurance relevant to normal operation is discussed in detail in Chapter 18.

2.3.2 Engineering Design Methodology

2.3.2.1 General

The design basis of the Reactor Facility will be developed from the demands on reactor design imposed by the conditions the reactor must be able to meet during its operational life. The design basis will take into account the challenges and conditions associated with all stages of the Reactor Facility's life, all operational states, site characteristics, design requirements and limits, and operating modes.

Conservative, high quality design methods, analyses and engineering practice will be used in the design of structures, systems and components important to safety. In particular, systems important to safety will be designed with conservative margins between applied loads and/or operation conditions and the design limits. These margins will take into account the most challenging parameter values for operational states and design basis accident conditions, the precision of the design methods applied, transient effects on the systems, measurement uncertainties and the ageing effects on materials properties where applicable.

2.3.2.2 Defence in Depth Applied to the Design

In accordance with the Defence in Depth Principle, a series of successive and diverse levels of protection against the unintentional release of radioactive material will be provided in the design of the Reactor Facility. Each level will provide adequate protection in the event that the previous one fails, including its failure as a consequence of the failure of another level.

The levels of defence in depth implemented in the design are:

Level	Objective	Essential means
Level 1	Prevention of abnormal operation and failures	Conservative, high quality, proven design and high quality in construction and operation.
Level 2	Control of abnormal operation and detection of failures	Process control and limiting systems and other surveillance features and procedures.
Level 3	Control of accidents within the design basis	Engineering safety features and accident procedures.
Level 4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures, accident management and on-site mitigation
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Off-site emergency response

The design of the Reactor Facility will be such that the first two levels of defence in depth carry the primary burden for nuclear safety, with the greatest emphasis being placed on the first level.

The design will also ensure that adequate defence in depth is maintained during all operational states of the facility.

2.3.2.3 Provision of Barriers

To complement the implementation of defence in depth principle, the design of the Reactor Facility will include a series of physical barriers against the release of radioactive material to the environment. The design of each successive barrier will address the failure of the previous barriers, including its failure as a consequence of the failure of other barriers. The following physical barriers will be provided:

a) fuel matrix and cladding;

b) pool coolant boundary and pool water; and

c) containment system.

As such, a large uncontrolled release of radioactivity from the Reactor Facility to the environment would require the failure of all three barriers.

2.3.2.4 Proven Engineering Practice

The technology used in the design, construction, commissioning and operation of the Reactor Facility will be based to the greatest extent practicable on conservative engineering practices that have been proven by operating experience and/or testing, and which are reflected in applicable codes and standards and other appropriately documented sources of information.

New technologies incorporated into the design of the Reactor Facility will be supported by appropriate research and development, and tested and validated prior to the Reactor Facility becoming operational.

2.3.2.5 Design Simplification

The design of the Reactor Facility will be kept as simple as practicable. Unnecessary design complexity will be avoided such that:

a) The operator has a clear and unambiguous understanding of plant behaviour and system responses. System behaviour and response indications are well defined and clearly understandable.

b) System interaction is reduced to a level that will still achieve the safety and utilisation requirements.

c) There are no complex service system dependencies.

d) Maintenance and inspection processes are well understood and easily carried out.

e) The Reactor Facility is simple to construct, commission, operate and decommission.

2.3.2.6 Fail-safe Features

The fail-safe principle will be incorporated into the design of systems and components important to nuclear safety, wherever practicable, taking into account plant availability requirements.

Fail-safe will be interpreted to mean that if a system or component fails, then the facility would automatically pass into a safe state without the need for the initiation of any protective actions.

2.3.3 Inherent Safety Features

Wherever practicable, inherent safety features will be incorporated into the design of systems important to safety, in particular systems which ensure the three basic safety functions of reactor shutdown, decay heat removal and containment of radioactive materials.

An inherent safety feature will be taken to mean a feature which ensures a change towards safe conditions in response to a postulated initiating event through reliance on intrinsic physical properties without requiring movement, displacement or actuation of any equipment.

2.3.4 Passive Safety Features

Wherever practicable, passive safety features will be incorporated into the design of systems important to safety, in particular systems which ensure the three basic safety functions of reactor shutdown, decay heat removal and containment of radioactive materials.

A passive safety feature will be taken to mean a feature which achieves its function without the need for an active power source (e.g. electricity, compressed air, etc)

2.3.5 Provision of Safety Systems

The design of the Reactor Facility provides safety systems to fulfil the following three basic safety functions:

Reactor shutdown and maintenance of sub-criticality.

Adequate heat removal.

Containment of radioactive materials.

Safety systems (i.e. Safety Category 1 systems; see section 2.5) will be provided to fulfil identified safety functions and will be established in a logical and systematic manner based on the safety analysis, such that sufficient protection is provided for every identified design basis fault sequence.

Safety systems will be subject to strict design requirements in order to ensure high reliability, and include provisions to facilitate regular inspection, testing and maintenance.

The principles of redundancy, diversity and independence, as identified below, will be applied to safety systems in order to ensure reliable performance and reduce the potential for common mode failures. The level of redundancy, diversity and independence applied to systems will be commensurate with the safety significance of the system.

The principles of redundancy, diversity and independence will also be applied in the design of safety-related process systems, to the extent appropriate to their safety categorisation (see section 2.5).

Compliance with these principles is discussed in the relevant “design evaluation” section associated with the individual system descriptions in Chapters 5 to 11 and in the safety analysis in Chapter 16.

2.3.5.1 Redundancy

Redundancy will be provided in the design of safety systems where necessary in order to:

a) Ensure the reliable performance of the safety function;

b) Meet the single failure criterion;

c) Ensure that reliability is maintained in systems where there is potential for undetected failures.

Where redundancy is provided, each set of redundant equipment will be capable of being individually tested.

2.3.5.2 Single Failure Criterion

Each safety system (or group of systems) provided to fulfil a safety function will be capable of fulfilling this safety function even in the event of a single random failure within the system (or group of systems).

In assessing compliance with the single failure criterion, design basis fault sequences will be considered in turn, with each component within the system (or group of systems) provided to fulfil that safety function assumed to fail in all credible modes. Consequential failures and the potential for common cause failures will also be considered, as well as the possible unavailability of other components and systems due to maintenance, repair or periodic testing.

Where necessary, individual non-compliance with the single failure criterion will be justified on a case-by-case basis based on one or more of the following:

a) Low frequency of the Postulated Initiating Event (PIE).

b) Small consequences of the fault sequence.

c) Unavailability of components or systems due to maintenance, repair or periodic testing only occurs for limited periods and/or during periods when the safety function is not likely to be required.

d) The single random failure is the failure of a passive component designed, manufactured, inspected and maintained in service to a high level of quality.

e) The component is a passive component of high quality, upon which reliance is not placed for an extended period.

Non-compliance with the single failure criterion will be avoided as far as reasonably practicable.

2.3.5.3 Diversity

Diversity will be applied, where appropriate, taking into account operational and maintenance requirements to redundant systems or components and elements within a system which perform the same safety function in order to enhance reliability and reduce the potential for dependent failures.

Diversity may be achieved by incorporating different attributes into the systems or components such as:

a) different principles of operation;

b) different operating conditions;

c) production by different manufacturers.

Diversity will also be provided between the various levels of defence in depth as far as reasonably practical.

2.3.5.4 Independence

The principle of independence will be applied as appropriate in the design of safety systems to enhance their reliability and reduce the potential for dependent failures.

The degree of independence will be effective under the following conditions:

Fire events;

Seismic events;

Flooding events;

Fault in a service system;

Collapse of supporting structures;

Internally generated missiles;

Physical/mechanical damage.

Independence will be provided as far as practicable between:

a) process control systems and safety systems;

b) redundant systems, subsystems or components;

c) different safety systems;

d) different levels of defence in depth.

Independence will be achieved by using both physical separation and functional isolation. Physical separation is achieved by spatial separation and/or by the use of barriers. Functional isolation includes the use of isolation devices and avoiding the sharing of process variable sensors across redundant subsystems.

2.3.6 Accident Prevention

The design of the Reactor Facility will incorporate appropriate design features which ensure that accidents are prevented. As such, the plant will be designed to be highly resistant to anticipated operational occurrences and design basis accidents by the application of sound design and engineering practices.

The basic philosophy applied in relation to accident prevention will be a simple design using proven technology, the use of conservative design margins, high quality materials and the defence in depth approach set out above.

2.3.7 Accident Management

The design of the Reactor Facility will include features and characteristics to limit the progression of accidents, including beyond design basis accidents, and to mitigate their consequences. These accident management measures will be aimed at maintaining heat removal from the core and maintaining the integrity of the Containment.

Accident management features that are required to operate under design basis or beyond design basis accident conditions will be designed to be capable of fulfilling their intended safety function taking into account the possible environmental conditions existing at the time of the accident.

Accident management features incorporated into the design will include the following:

a) A Post-Accident Monitoring (PAM) system capable of allowing operators to monitor the plant status and take appropriate mitigation actions during accident conditions.

b) An Emergency Control Centre (ECC) separate from the Main Control Room (MCR) and designed to be habitable during design basis accidents should the MCR become unavailable.

c) An Emergency Make-up Water System (EMWS) to maintain heat removal from the reactor core in the unlikely event of pool drainage.

d) A Reactor Containment that minimises any release of radioactive material to the environment following a reactor accident.

e) A Containment Energy Removal System (CERS) which removes heat from an isolated Containment.

f) A Containment Pressure Relief and Filtered Vent System (CPRFVS), and a Containment Vacuum Relief System (CVRS) to minimise the differential pressure between Containment and the outside environment in the very unlikely event of failure of the CERS.

g) An effective communications systems and site emergency warning system.

h) Escape routes and emergency lighting.

i) A fire protection system.

2.3.8 Assessment of Human Factors

The design of the Reactor Facility will systematically consider human factors, the human-machine interface and ergonomic principles to reduce the potential for human error, facilitate correct actions by operators, minimise operator stress and ensure an appropriate and clear distinction of functions between operating personnel and the automatic systems provided.

The plant and control room layout and the mode of presentation of information during operational states, including maintenance and inspection, will facilitate the ergonomic disposition of data and controls for actions important to safety, including accident management.

Safety systems will be designed to be automatically initiated and require no operator action within thirty minutes of the occurrence of the initiating event, while permitting operator initiation or action. Once initiated, the actuation of the safety systems will not be able to be interrupted or interfered with by the operator.
The design of the Reactor Facility will minimise the demands on the operator during and following an accident condition. This will include the provision of diagnostic aids to resolve questions important to safety and to monitor the status of the facility, self-checking diagnostic techniques and alarm prioritisation.

Engineered features, which are distinct from administrative controls, will be implemented in the design whenever possible, to prevent the occurrence of initiating events.

Manuals and procedures will be produced with consideration of potential error likely situations. Qualitative and quantitative Human Factor Analysis techniques will be implemented to evaluate manuals and procedures during the design stage. Particular attention will be paid to tasks and operations performed at the reactor pool top.

An independent review of the human-machine interface and human factors will be performed by ANSTO consistent with a formal review plan that identifies the appropriate codes and standards to be used (eg NUREG and IEEE standards).

2.3.9 Radiation Protection

The Reactor Facility will be designed so that throughout the life of the facility, radiation doses to the public and staff arising from normal operation and anticipated operational occurrences do not exceed the limits and constraints specified in Section 2.2.2.

The design of the Reactor Facility will meet the nuclear and radiation safety standards defined in Section 2.7 including occupational and industrial safety and licensing requirements. The design will also meet appropriate Commonwealth and New South Wales State Occupational Health and Safety Acts and Regulations.

The design of the Reactor Facility will optimise the level of radiation protection provided such that individual and collective radiation doses are kept ALARA and within the specified dose constraints, and that radioactive discharges from the facility are ALARA and within limits.

The design of the Reactor Facility will incorporate adequate provisions for shielding, ventilation, filtration, radioactive material decay systems and radiation monitoring instrumentation for all operational states and accident conditions.

The design of the Reactor Facility will use structural materials, particularly near the reactor core, which ensure low radiation dose rates to personnel during operation, inspection, maintenance and decommissioning, while fulfilling all other necessary functions.

The design of the Reactor Facility and its layout will incorporate provisions for the segregation of radioactive materials through access control and zoning.

End of Section

2.4 SPECIFIC DESIGN REQUIREMENTS

The following are specific, nuclear safety-related, design requirements for safety relevant systems, equipment and activities.

2.4.1 Reactivity Control Criteria

This section specifies the reactivity control criteria applicable to the design of the facility. The implementation of these criteria is discussed in Chapter 5 and 11 while their acceptability is demonstrated in the various design evaluation sections in Chapter 5 and the safety analysis in Chapter 16.

2.4.1.1 Shutdown System Requirements

The reactor will be provided with two independent and diverse shutdown systems. These will consist of the First Shutdown System (FSS) which shuts down the reactor by insertion of control rods into the core and the Second Shutdown System (SSS) which shuts down the reactor by the partial dumping of heavy water from the reflector vessel.

Each shutdown system will insert sufficient negative reactivity to bring the reactor subcritical and maintain it subcritical for all core configurations under all operational states and design basis accident conditions. This will be demonstrated assuming the facility is in its most reactive state and, in the case of the FSS, the most reactive control rod is stuck fully out of the core.

The FSS and SSS are Engineered Safety Features (ESFs) and will meet the requirements of Section 2.4.10.

The effectiveness and response times of the FSS and the SSS will be such that the reactor is shutdown before any safety limit is reached for any design basis fault sequence.

2.4.1.2 Reactivity Control System Requirements

The reactivity process control system utilises the same control rods as the FSS. It controls core reactivity and maintains stable flux shapes and levels within operating limits during normal operation. As such, the design will ensure that the shutdown function of the FSS has priority over the process control function so that the process control function cannot prevent the insertion of the control rods into the core upon demand.

The total power feedback coefficient of the reactor in any operating condition (even without process control systems), and all temperature and void coefficients of reactivity associated with the fuel, coolant and moderator will be shown to be either negative for all operational states and accident conditions, or it will be shown that the coefficient has an insignificant effect.

2.4.1.3 Reactivity Limits

The maximum reactivity worth of each reactor irradiation facility, rig or experiment (fixed or not) as well as the total reactivity worth of all irradiation facilities, rigs and experiments will be specified and limited in order to ensure that their operation or failure does not result in an unacceptable change in reactivity.

The maximum rate of addition of positive reactivity allowed by the insertion, withdrawal or operation of an irradiation facility, rig or experiment will be specified and limited to a value which provides a conservative margin to safety limits, as demonstrated by the safety analysis.

The maximum rate of addition of positive reactivity allowed by the reactivity control system will be specified and limited to a value which provides a conservative margin to safety limits.

These specifications and limitations will be contained in the Operational Limits and Conditions (OLCs).

2.4.1.4 Shutdown Margins

The design of the Reactor Facility will specify limits on minimum shutdown capability for the core which ensure a margin to safety limits for design basis accidents. Limits will be specified for:

a) Shutdown capability of the FSS, assuming the most reactive control rod is fully out of the core.
b) Shutdown capability of the SSS, assuming a single failure within the system.

2.4.2 Thermal-hydraulics Design Criteria

This section specifies the thermal-hydraulics design criteria applicable to the design of the facility. Their implementation is discussed in Chapter 5 and their acceptability is demonstrated in the various design evaluation sections in Chapter 5 and the safety analysis in Chapter 16.

The design of the Reactor Facility will ensure that thermally or hydraulically induced fuel damage during any operational state or design basis accident can not occur.

Thermal-hydraulic design limits will be established with a margin to safety limits to take into account uncertainties and engineering tolerances.

To ensure the integrity of the fuel cladding, maximum fuel cladding and fuel meat temperature limits will be established with margins to safety limits. The safety limit for fuel meat temperature will correspond to the phenomenon of blistering, while the safety limit for fuel cladding temperature during normal operation will take into account the need to minimise corrosion.

For all steady state, normal operating conditions, the thermal-hydraulic design of the Reactor Facility will ensure that the phenomena of Onset of Nucleate Boiling (ONB) is avoided.

To avoid fuel plate vibrations, which may result in local overheating and possibly blockage of the coolant channels, the maximum coolant velocity will be limited to a value which ensures a conservative margin to fluid-structure instability effects.

The thermal hydraulic design of the Reactor Facility will ensure that the total power peaking factor is limited to a value which provides a conservative margin to safety limits, as demonstrated by the safety analysis.

2.4.3 Reactor Core Integrity Requirements

This section specifies the reactor core integrity requirements applicable to the design of the facility. Their implementation is discussed in Chapter 5 and their acceptability is demonstrated in the various design evaluation sections in Chapter 5 and the safety analysis in Chapter 16.

The reactor core will be designed and constructed such that the design limits are not exceeded for any operational state or design basis accident. In particular, it will be demonstrated that:

a) Core components are designed and constructed so as to withstand the static and dynamic loading expected during operational states and design basis accidents, taking into account thermal, hydraulic, radiation, seismic and environmental effects.

b) Core distortion or movement during all operational states or design basis accidents does not impair the effectiveness of the reactivity control system or shutdown systems, or prevent cooling of the fuel.

c) Fuel parameters are maintained within limits during all operational states and design basis accidents so that significant damage to the core does not occur.

The design of the reactor core will ensure that the reactor can be shutdown and held subcritical for all operational states and design basis accident conditions.

Design margins will be used in determining design limits for the reactor core to take into account uncertainties, engineering tolerances and the anticipated properties of materials at the end of their useful life. Materials used in the core and reflector will be selected to withstand the conditions of operational states and design basis accidents over the life of the facility. Consideration will also be given to the effects of corrosion, radiation-induced swelling and radiation-induced mechanical property changes.

The design of the reactor core will consider foreseeable reactor core configurations from the initial core through to the equilibrium core.

The design of the Reactor Facility will reduce the potential for the inadvertent dropping of heavy objects onto the reactor core. In particular, the design of the reactor hall and the crane will ensure that movement of heavy items directly above the core is avoided. The design will also incorporate appropriate means of protection for vulnerable components to ensure that any dropped loads do not result in damage to the reactor core or the reactor pool pressure boundary.

2.4.4 Protection Against Flow Instabilities and Power Oscillations

This section specifies the requirements applicable to the design of the facility with respect to protection against flow instabilities and power oscillations. Their implementation is discussed in Chapter 5 and 8 and their acceptability is demonstrated in the various design evaluation sections in Chapter 5 and the safety analysis in Chapter 16.

The Reactor Facility will be designed so that flow instabilities and power oscillations are avoided taking into account interactions between the nuclear system and other facility systems.

Protection against flow instabilities is ensured by the following:

a) Boiling will be avoided during all steady operational states. The primary cooling system design will ensure that the steady state fuel cladding temperature is below the temperature for ONB with an appropriate safety margin (see Section 2.4.2).

b) A margin will be established between normal operation and flow instability phenomenon that can occur in arrangements of parallel channels subject to power inputs (see Section 2.4.2).

c) Appropriate venting provisions will be provided within the primary cooling system to ensure proper venting of the system during reactor start-up.

Protection against power oscillations will be ensured by the following:

a) Power oscillations induced by the formation of vapour (bubbles) in the core cannot occur as boiling is avoided during steady state conditions in operational states.

b) The automatic power-regulating system will be designed with a set of dynamic parameters that assure a smooth and stable behaviour of the reactor power.

c) The rate of insertion of reactivity will be limited.

2.4.5 Fuel Design Limits

This section specifies the fuel design limits applicable to the design of the facility. Their implementation is discussed in Chapter 5 and their acceptability is demonstrated in the various design evaluation sections in Chapter 5 and the safety analysis in Chapter 16.

2.4.5.1 General Fuel Design Requirements

Fuel assemblies (FAs) will be designed such that under all operational states and design basis accident conditions:

a) The fuel plate cladding maintains its integrity.

b) Variations in geometry and dimensions are within allowable tolerances defined by design.

FAs will be of proven design and fully qualified through analyses supported by data from experiments and irradiation experience.

Design limits will be specified for all parameters relevant to FA performance as identified in the following sub-sections. Design margins will be used in determining design limits for FAs to take into account uncertainties and engineering design tolerances.

2.4.5.2 Thermal Effects

The following thermal effects will be taken into account in the design of FAs:

a) Reduction in coolant channel thickness due to thermal expansion of fuel plates.

b) Formation of oxide layer on the cladding surface.

c) Thermal stress on fuel plates because of non-uniform temperature distribution and external restraints to thermal expansion.

d) Risk of plate elastic buckling due to lateral compressive loads.

e) Cooling flow instabilities because of excessive fuel plate surface temperature.

The following design requirements ensure geometrical and dimensional stability of fuel plates with respect to thermal effects :

a) A fuel cladding surface temperature limit for normal operation will be specified taking into account the need to minimise corrosion effects.

b) A fuel meat temperature limit will be conservatively set to avoid fuel plate blistering.

c) A maximum allowable compressive thermal stress limit will be specified with a conservative margin to safety limits for fuel plate buckling.

d) A limit for the total calculated thermal stresses at operating temperatures will be specified below the beginning of plastic strains, in order to prevent permanent distortion or deflection of the fuel plates.

2.4.5.3 Hydraulic and Mechanical Effects

The following hydraulic and mechanical effects will be taken into account in the design of fuel assemblies:

a) Hydraulic instability due to excessive coolant velocity.

b) Mechanical stress on FA bottom nozzle due to interaction with the core grid plate and fuel assembly clamp.

c) Mechanical stress on FA structure during refuelling operations.

The following design requirements ensure geometrical and dimensional stability of fuel plates with respect to hydraulic and mechanical effects:

a) The maximum coolant velocity in the cooling channels will be limited with a conservative margin to the critical velocity value. The critical velocity is the threshold above which large flow-induced deflection of the plates can occur which could lead to obstruction of the cooling channel.

b) The maximum hold-down force on the FA will be limited to prevent excessive stresses in the contact surfaces and bottom nozzle.

c) To avoid stresses during refuelling operations, the pull out force on the FA will be limited.

2.4.5.4 Radiation Effects

The following radiation effects will be taken into account in the design of fuel assemblies:

a) Swelling and blistering of fuel plates due to fission product build-up.

b) Changes in physical and mechanical properties of fuel materials due to irradiation (thermal conductivity, yield and ultimate strengths and ductility).

The irradiation swelling and blistering will be accounted for in the calculation of fuel plate thickness changes during irradiation. The risk of a fuel assembly failure due to swelling, blistering or corrosion will be kept low by ensuring that:

a) The maximum fuel cladding and meat temperatures are below safety limits (see Section 5.1).

b) The residence time in the core is relatively short.

c) The composition of the fuel meat provides adequate proportions of fuel, aluminium and voids that reduce overall swelling.

2.4.5.5 Chemical Effects

The following chemical effects will be taken into account in the design of fuel assemblies:

a) Chemical interaction with coolant.

b) Uniform corrosion of exposed surfaces, especially on the hottest fuel plates.

c) Localised corrosion phenomena, such as pitting and galvanic corrosion.

The maximum reduction in cooling channel width due to thermal expansion, oxide formation and swelling will be limited to ensure that adequate cooling of the fuel is maintained.

Uniform corrosion is not significant due to the moderate operational temperatures and low core residence time. The water chemistry will be rigorously controlled to further lower the impact of uniform corrosion on FA performance as well as avoiding pitting. The cleanliness of objects to be introduced into the reactor pool will also be controlled to avoid chemical contamination.

Galvanic corrosion will be prevented by an adequate selection of materials and a proper design of contact zones where different materials are present (e.g., avoidance of stagnant water conditions which can lead to local variations in water chemistry).

2.4.6 Design Criteria for the Reactor and Service Pools

This section specifies the design criteria for the reactor and service pools. Their implementation is discussed in Chapters 5 and 6 and their acceptability is demonstrated in the various design evaluation sections in Chapters 5 and 6 and the safety analysis in Chapter 16.

The reactor and service pools structure, including their penetrations, will provide a high standard of leak tightness and have features to prevent and monitor water leakage.

ASME Boiler and Pressure Vessel Code has been used as a guide in design, construction and test for Atmospheric Storage Tanks, with supplementary inspection requirements for the Reactor and Service Pool liners and the Transfer Canal

The reactor and service pool liners will be designed and constructed to withstand the static and dynamic loads anticipated during operational states and design basis accidents, including thermal and hydraulic effects.

Sufficient design margin will be applied in the design of the reactor and service pool liners to allow for:

a) Deterioration over the life of the facility due to erosion, corrosion, creep, fatigue, radiolysis and the effects of the radiation and chemical environment.

b) Uncertainties in determining the initial state of components and the rate of deterioration.

c) Anticipated properties of materials at the end of their useful life.

The design of the reactor and service pool liners will ensure that:

a) flaws are very unlikely to initiate.

b) flaws, if initiated, propagate in a regime of high resistance to propagation.

c) situations which could lead to brittle behaviour are avoided.

The reactor and service pool liners will be designed to facilitate inspections and tests in order to ascertain the effects of irradiation and to detect the occurrence of leaks, stress corrosion, cracking, brittle fractures and ageing of structural materials.

Reactor and service pool penetrations will be designed such that a leak will not cause uncovering of the core or loss of core cooling. As such, process piping penetrations through the reactor or service pool liners that connect with the pool water will be located well above the reactor core and incorporate appropriate siphon breakers and isolation devices. Penetrations through the reactor pool liner at or below the level of the reactor core through which reactor pool water could potentially leak will be provided with redundant seals.

The design of the Control Rod Drive (CRD) room below the reactor pool will take into account the potential for leakage of pool water through penetrations at the bottom of the pool.

The reactor and service pool coolant boundary is an ESF and as such, will meet the requirements for ESFs specified in Section 2.4.10.

2.4.7 Design Criteria for Structures Within the Reactor and Service Pools

This section specifies the design criteria for structures within the reactor and service pools that perform a safety function or whose failure could impact on reactor safety. Their implementation is discussed in Chapter 5 and their acceptability is demonstrated in the various design evaluation sections in Chapter 5 and the safety analysis in Chapter 16.

The structures within the reactor and service pools will be designed to withstand the operating conditions in temperature, water quality and radiation during the whole life cycle of the facility.

In the design of structures inside the reactor and service pools conservative margins will be applied between stresses and the design strength which allow for ageing, irradiation effects and engineering tolerances.

Corrosion of structures inside the reactor and service pools will be minimised by the selection of appropriate materials and by strict control of the water chemistry.

The materials for structures within the reactor and service pools will be selected so as to minimise doses to personnel and damage to equipment during operational states, including inspection, maintenance and repair activities, and ultimately, decommissioning. They will also be selected to withstand design basis accident conditions, such as increased heat loads and irradiation during transients, without compromising their performance after recovery of the facility.

The structures within the reactor and service pools will be designed to withstand an SL-2 seismic event, remaining in elastic behaviour. Adequate margin will be provided to remain in the elastic region for earthquakes of lower return frequency than the SL-2 earthquake.

The design will facilitate the periodic inspection of structures and components inside the reactor and service pools.

2.4.8 Cooling Systems Criteria

This section specifies the criteria and requirements applicable to the design of the reactor, reflector vessel and pool cooling systems. These consist of:

Primary Cooling System (PCS)

Reactor and Service Pools Cooling System (RSPCS)

Reflector Cooling and Purification System (RC&PS)

Secondary Cooling System (SCS)

Emergency Make-up Water System (EMWS)

Associated systems include the Reactor Coolant Purification System, the Secondary Coolant Treatment System and the Reactor Pool Hot Water Layer System. These systems are discussed in Chapter 6 and their acceptability is demonstrated in the various design evaluation sections in Chapter 6 and the safety analysis in Chapter 16.

2.4.8.1 General Requirements

Cooling systems will provide adequate cooling to the reactor core, the reflector vessel and the reactor and service pools (including the irradiation facilities) with a margin for all operational states and design basis accidents.

The design of all cooling systems will include an integrated approach to keeping corrosion within acceptable limits for the design life of the facility. This will be achieved by providing systems for coolant quality monitoring, purification and chemical treatment, to ensure the maintenance of appropriate water quality. These systems will also include provisions for monitoring and removal of radioactive substances from the coolant, including corrosion and fission products and tritium, where appropriate.

High standards of design and fabrication will be applied to all cooling systems. In particular, Safety Category 1 vessels will be designed to ASME Boiler and Pressure Vessel Code.

Equipment and components in cooling systems will be designed and constructed to withstand the static and dynamic loads anticipated during operational states and design basis accidents, including thermal and hydraulic effects.

The design margins applied in the design of systems containing reactor coolant will be sufficient to allow for:

a) Deterioration over the life of the facility due to erosion, corrosion, creep, fatigue, radiolysis and the effects of the radiation and chemical environment.

b) Uncertainties in determining the initial state of components and the rate of deterioration.

c) Anticipated properties of materials at the end of their useful life.

The design of the primary coolant boundary will ensure that:

a) Flaws are very unlikely to initiate.

b) Flaws, if initiated, propagate in a regime of high resistance to propagation.

c) Situations which could lead to brittle behaviour are avoided.

Systems containing reactor coolant will be designed to facilitate inspections and tests of the coolant boundary in order to ascertain the effects of irradiation, and to detect the occurrence of leaks, stress corrosion, cracking, brittle fractures and ageing of structural materials, as appropriate.

The design of cooling systems will include the provision of inventory control systems with adequate capacity, taking into account volumetric changes and normal losses.

2.4.8.2 Primary Cooling System

The Primary Cooling System (PCS) will provide sufficient cooling of the reactor core to ensure that the fuel cladding remains an intact barrier against the release of fission products from the fuel.

The PCS will remove fission heat from the core and maintain adequate subcooling at the inlet to the core during the Power State by forced cooling.

The design of the PCS will ensure that there is adequate coolant flow through the reactor core during pump run down in the event of loss of one or more primary coolant pumps, and during the transition between forced cooling and natural circulation cooling for the reactor core.

The PCS will include a reliable means of decay heat removal that is capable of adequately cooling the reactor core during routine reactor shutdowns and following design basis fault sequences. These means will be designed with inherent and passive safety features by utilising natural circulation within the reactor pool to provide long-term heat transfer from the reactor core to the pool water which acts as the ultimate heat sink.

Redundancy will be incorporated in the design of the core residual heat removal system by the provision of dual cooling paths and flap valves which are fully testable to verify that they will function when required.

The means of decay heat removal from the reactor core by natural circulation is an ESF and as such, will meet the requirements for ESFs specified in Section 2.4.10.

2.4.8.3 Reactor and Service Pools Cooling System

The Reactor and Service Pool Cooling System (RSPCS) will provide sufficient forced cooling for all operational states and design basis accidents.

The RSPCS will include a reliable means of residual heat removal that is capable of adequate cooling during routine reactor shutdowns, and following all design basis fault sequences. These means will be designed with inherent and passive safety features by utilising natural circulation within the reactor pool to provide long-term heat transfer to the pool water which acts as the ultimate heat sink.

Redundancy will be incorporated in the design of the means of residual heat removal from the irradiation rigs by the provision of dual cooling paths and flap valves which are fully testable to verify that they will function when required.

The means of residual heat removal from the irradiation rigs by natural circulation is an ESF and as such, will meet the requirements for ESFs specified in Section 2.4.10.

2.4.8.4 Reflector Cooling and Purification System

The Reflector Cooling and Purification System (RC&PS) will be designed to remove the heat generated in the heavy water within the reflector vessel under normal operating conditions. The RC&PS will also incorporate a means to remove heat from specific components within the reflector vessel as appropriate following actuation of the SSS when the heavy water is partially drained.

The RC&PS will minimise the possibility of heavy water leakage into the SCS and the environment by the provision of an intermediate system between the primary system (containing heavy water) and the SCS.

2.4.8.5 Secondary Cooling System

The Secondary Cooling System (SCS) will remove heat from the PCS, the RSPCS and the RC&PS and discharge it to the atmosphere via the SCS using cooling towers under all normal operating conditions, for all design basis meteorological conditions. The SCS cooling towers will be capable of operation without damage in case of bushfires in close proximity to the Site.

The SCS will be designed such that the potential for the release of radioactivity from the PCS, the RSPCS or the RC&PS to the environment is low. In addition, the SCS will include provisions to monitor for radioactivity within the secondary coolant.

2.4.8.6 Emergency Make-up Water System

An Emergency Make-up Water System (EMWS) will be provided to maintain the core covered with water in the event of a Beyond Design Basis Loss of Coolant Accident (LOCA) in order to prevent damage to the fuel. The system will be defined as a Beyond Design Basis Accident Mitigation system (defence in depth level 4).

Operation of the EMWS will be initiated automatically when required regardless of the availability of electrical power supplies.

2.4.9 Instrumentation and Control Systems Criteria

This section specifies the criteria and requirements applicable to the design of the Instrumentation and Control (I&C) systems. These consist of:

- First Reactor Protection System (FRPS)
- Second Reactor Protection System (SRPS)
- Shutdown Systems' Instrumentation
- Reactor Nucleonic Instrumentation
- Post Accident Monitoring (PAM) System
- Reactor Control and Monitoring System (RCMS)
- Radiation Monitoring System (RMS)
- Main Control Room (MCR)
- Emergency Control Centre (ECC)
- Irradiation Facilities, Beam Facilities, and Cold Neutron Source (CNS) Instrumentation
- Seismic Monitoring Instrumentation

These systems are discussed in Chapter 8 (with some additional information in Chapter 12 relating to radiation protection and waste monitoring) and their acceptability is demonstrated in the associated design compliance sections and the safety analysis in Chapter 16.

2.4.9.1 General Instrumentation and Control System Requirements

Process instrumentation and control systems will provide sufficient instrumentation for monitoring the operation of the reactor, and its irradiation and beam facilities, during all operational states and for recording all variables important for safety.

As far as practicable, process variables that are used in systems important to safety will be measured directly (in contrast to being calculated using assumed relationships from other measurements) such that they detect the actual conditions that require safety action.

When a system variable deviates from operational limits during normal operation or an anticipated operational occurrence, an action of the RCMS is initiated to bring the reactor to a safe state without the need for the reactor protection systems to invoke safety systems. The corrective actions of the RCMS affect a wide range of facility systems (e.g., reactor power control by insertion or removal of the central control plate, PCS and RSPCS pump trip due to low pool water level signal).

Monitoring and control systems will be fully automatic, although manual actions by the operator are possible. However, interlocks will inhibit manual actions that may adversely impact on reactor safety.

I&C systems for monitoring and controlling the reactor during operational states and design basis accidents will be centralised in a Main Control Room (MCR). The design will ensure that the MCR is habitable during all operational states.

Audible and visual alarms will be provided to warn the operators of abnormal situations within the safety system settings. Once safety system settings are reached, the Reactor Protection Systems (RPS) will override the action of the process control systems.

An Emergency Control Centre (ECC), separate from the MCR, will be provided for monitoring and controlling the reactor during design basis accidents which could result in the MCR becoming uninhabitable or unavailable. The ECC will only be used when the reactor is shutdown and will not be capable of operating the reactor at power.

2.4.9.2 Reactor Protection Systems

2.4.9.2.1 General Requirements

The reactor protection systems are the First Reactor Protection System (FRPS) and the Second Reactor Protection System (SRPS). This section identifies common requirements applicable to both systems whilst requirements specific to each system are identified in the subsequent sections.

The FRPS and the SRPS will be diverse and independent systems. No single component failure in either shutdown system will be able to cause failure of the other system to perform its function. The FRPS and SRPS will also be fail-safe.

The RPS will employ diversity to enable all postulated initiating events to be detected in a minimum of two different ways (where physically possible).

The FRPS and SRPS will ensure that the reactor is brought to and maintained in a safe shutdown condition.

The FRPS and SRPS will comply with Institute of Electrical and Electronics Engineers (IEEE) standards for Class 1E equipment.

The FRPS and SRPS are ESFs and will meet the requirements for ESFs specified in Section 2.4.10.

The FRPS and the SRPS will both be triple redundant systems. Redundant trains will be independent to reduce the potential for common cause failure (including failures due to the effects of design basis accidents) and to ensure that no single component failure in any train can result in the failure of another train to perform its function.

Redundancy will be applied in the coincidence voting logic of both the FRPS and SRPS such that the outputs of redundant trains are combined using 2 out of 3 voting logic. This voting logic will also ensure that maintenance can be carried out on defective channels within trains without the need for the reactor to be shutdown. In addition, all components in the FRPS and SRPS will be capable of being functionally tested and adequate provision is made for maintenance and inspection of equipment.

Electric power to the FRPS and the SRPS system will be supplied from uninterruptible power supplies with each of the three redundant trains in each system supplied from an independent Uninterruptible Power Supply (UPS) system.

The FRPS and SRPS will each be designed to protect against design basis accident conditions by automatically initiating the operation of appropriate systems to ensure that safety limits are not exceeded during any design basis fault sequence. The operation of the FRPS and SRPS will be fully automatic and, following initiation, requisite actions will proceed to completion with no manual intervention required. However, provision will be made for manual initiation of a reactor trip via either the FRPS or the SRPS, including the capability to initiate reactor shutdown from a remote location.

Both the FRPS and SRPS will minimise the likelihood that operator actions could defeat the effectiveness of either system during operational states and design basis accidents. However, neither system will negate correct operator actions during design basis fault sequences.

The FRPS and SRPS will not be automatically reset. Deliberate operator action will be required to reset either system.

The FRPS and SRPS will be independent of other systems. In particular:

a) Approved isolation devices will be provided where the FRPS or SRPS interfaces with Reactor Control and Monitoring System (RCMS).

b) The FRPS and SRPS will be independent of other protection systems (e.g. the CNS protection system).

c) The redundancy and independence of the FRPS and the SRPS will not be jeopardised where signals from utilisation devices are fed into the FRPS or SRPS.

Instrumentation and tripping systems will enable trip levels to be established. FRPS and SRPS set points will be established with sufficient margin between the set point and the safety limit such that actions initiated by either system are able to control the process before the safety limit is reached. Margins between set points and safety limits will allow for instrumentation inaccuracies, calibration uncertainties, instrument drift and instrumentation and shutdown system response times.

The FRPS and SRPS will alert the operator when trip settings are reached and allow for the cause of a safety system trip to be determined. The capability to identify the parameter that initiated the trip and the sequence of subsequent trips will be included in the RCMS.

The design of the FRPS and SRPS will ensure that safety interlocks and trips cannot be inadvertently bypassed and that safety system settings cannot be inadvertently changed.

The FRPS will incorporate dynamic signal and watchdog features to supervise its functioning. It will also incorporate automatic verification routines to check whether the logic voting units are working according to their design specifications.

2.4.9.2.2 First Reactor Protection System

The FRPS will provide signals for the First Shutdown System (FSS) and other safety systems. It will also initiate Containment isolation, perform specific control actions within the CERS and trip the reactor on request from the Cold Neutron Source Protection System (CNSPS).

The computer based digital FRPS system will be subject to hardware and software verification and validation. In particular, it will comply with the requirements defined in IEEE standards for the application of computer systems in reactor safety systems and associated software quality assurance and management.

2.4.9.2.3 Second Reactor Protection System

The SRPS will provide signals for the Second Shutdown System (SSS) only.

2.4.9.3 Reactor Shutdown Systems' Instrumentation

FSS and SSS instrumentation will comply with IEEE standards for Class 1E equipment. In addition, the instrumentation of each shutdown system will be to the same standards as the FRPS and SRPS such that:

a) Each reactor shutdown system comprises redundant trains that are independent and isolated from each other, to reduce the potential for common cause failure.

b) No single component failure can result in the failure of the reactor shutdown system to perform its function.

c) Failure of either reactor shutdown system cannot inhibit the safe shutdown of the reactor by the non-failed system.

d) The shutdown systems are designed to be fail-safe.

The effectiveness and response times of the FSS and the SSS will be such that the reactor is shutdown before safety limits are reached for design basis accidents.

2.4.9.4 Reactor Nucleonic Instrumentation

Reactor nucleonic instrumentation will be provided to enable monitoring of the neutron flux during operational states and design basis accidents and to provide the required inputs to the FRPS and SRPS. Independent and diverse instrumentation will be provided for the FRPS and SRPS in accordance with the general requirements identified above.

2.4.9.5 Post Accident Monitoring System

The Post-Accident Monitoring (PAM) system is an ESF and as such, will meet the requirements for ESFs specified in Section 2.4.10.

The PAM system complies with IEEE standards for Class 1E equipment.

The PAM system will incorporate two redundant trains that are independent. This is required to reduce the potential for common cause failure (including failures arising from design basis accidents) and to ensure that no single component failure in any train can cause failure of the other channel to perform its function.

Electric power to the PAM system will be supplied from uninterruptible power supplies with each PAM train supplied from an independent UPS system.

PAM instrumentation will be designed such that design basis accidents do not result in loss of function.

PAM instrumentation will:

a) Monitor the course of design basis accidents.

b) Monitor the state and the effectiveness of all safety systems.

c) Monitor neutron flux during design basis accident conditions.

d) Facilitate operator actions that may be necessary to place the facility in a stable long term shutdown condition.

e) Determine the type, location and quantities of radioactive material that might be released.

Operator displays for the PAM will be centralised in the MCR and duplicated in the ECC.

The PAM system will provide sufficient instrumentation for monitoring the reactor and for recording variables important for safety during anticipated operational occurrences and design basis accidents.

PAM instrumentation will be designed to facilitate regular testing, inspection and maintenance.

2.4.9.6 Reactor Control and Monitoring System

There will be a separate reactor control, instrumentation, monitoring display, alarm and warning system (the RCMS) and support facilities to serve the normal operation of the reactor plant. The RCMS will be provided in addition to, and independent of, the reactor protection systems.

The input and output signals associated with the operation and control of the plant will be kept as separate as practicable.

Uninterruptible Power Supply will be provided for the RCMS.

The reactor control and monitoring system will include sufficient redundancy to ensure provision of a highly reliable system and that no single point of failure exists.

The design of the RCMS will allow for online testing of the system.

The RCMS and the control and monitoring system for the irradiation facilities and the CNS are safety-related process systems and as such, will meet the requirements for Safety Category 2 systems (see section 2.5).

The computer based digital RCMS will be subject to hardware and software verification and validation in accordance with appropriate standards at each phase of its development.

2.4.9.7 Radiation Monitoring System

This section identifies the requirements applicable to the Radiation Monitoring System (RMS). This system covers all aspects of radiation monitoring and the implementation of these requirements is discussed in Chapters 8 and 12.

2.4.9.7.1 Liquid Waste Monitors

Liquid effluent monitors will be provided to generate alarm signals to warn of radioactive liquid discharges from the facility approaching or exceeding the allowed limits.

2.4.9.7.2 Air Waste Monitoring

Air effluent monitoring will be provided to allow continuous on-line sampling of all the radioactive stack emissions during operational states.

The air effluent monitoring will provide real-time data regarding the quantity and isotopic composition of radioactivity released in the stack emissions. The air effluent monitoring system will monitor for radioactive particulates, iodine, noble gases, and tritium.

2.4.9.7.3 Failed Fuel Monitoring

The PCS will be provided with on-line monitoring to provide an early warning of fuel cladding failure. On-line monitoring will also be provided within the RSPCS to provide an early warning of irradiation target failure.

2.4.9.7.4 Liquid System Monitoring

On-line sampling will be provided for all systems that contain potentially radioactive cooling water. The principal systems that contain radioactive water are the PCS, the RSPCS, the RC&PS and the liquid waste system.

2.4.9.7.5 Area Radiation Monitoring

Fixed-area monitors will be provided to cover the area around the reactor and areas where radiation may be present.

2.4.9.7.6 Neutron Dose-Rate Monitoring Systems

A fixed system of radiation monitors to measure neutron dose-rates will be provided at appropriate locations.

2.4.9.7.7 Tritium Monitoring System

A tritium monitoring system capable of measuring tritium concentrations in both the reactor building and in stack discharges will be provided.

2.4.9.7.8 Contamination Monitoring

Contamination monitors will be provided at exits from areas where the potential for personal contamination exists.

Alarms from contamination monitors will be provided in the MCR and in the health physics office via video display units (VDUs).

2.4.9.8 Main Control Room Instrumentation and Equipment

The MCR will be provided with instrumentation and controls to enable operating personnel to operate the reactor and efficiently manage normal operating situations and anticipated operational conditions.

2.4.9.9 Emergency Control Centre Instrumentation and Equipment

An ECC will be provided which is protected from the effects of fault sequences that could potentially result in the MCR becoming uninhabitable or unavailable.

The ECC will be provided with sufficient hard-wired instrumentation and controls to enable operating personnel to ensure the shutdown the reactor, maintain it in a safe shutdown state, and monitor the status of the Reactor Facility in the event of the MCR becoming unavailable. The ECC will only be used when the reactor is shutdown and will not be capable of operating the reactor at power.

The ECC will contain sufficient duplication and separation (according to IEEE standards) of instrumentation to monitor Safety Category 1 and 2 system parameters.

The ECC will be suitable for occupation, and its access routes fully accessible, throughout all postulated design basis accidents in which the MCR becomes uninhabitable.

2.4.9.10 Facility Emergency Warning System

A facility emergency warning system will provide a reliable means of communication with all persons present in the facility.

The facility warning system will interface with the site warning system.

2.4.9.11 Irradiation, Beam Facilities, and Cold Neutron Source Instrumentation

2.4.9.11.1 Irradiation Facilities Instrumentation and Control

All instrumentation and computerised controls associated with all irradiation facilities will be separate from the RCMS. Appropriate alarms will provide input into the RCMS and MCR displays. The system will be designed to the same level of performance as the RCMS.

Relevant signals to verify normal functioning of the RSPCS (which cools the irradiation facilities) or deviation from normal operation will be sent to the FRPS and initiate reactor trip if appropriate. These signals will follow the same standard as for the FRPS signals, with triple redundancy, qualification to IEEE Class 1E standards and sensors independent of the RCMS instrumentation.

2.4.9.11.2 Beam Port Instrumentation

Indication of primary and secondary beam shutter position will be shown on local displays near each port.

Primary beam shutter positions will input into the RCMS for indication only.

Leak detection will be provided for possible leakage through the beam ports.

2.4.9.11.3 Cold Neutron Source Instrumentation

Instrumentation and computerised controls associated with the CNS will be separate from the RCMS. Selected CNS alarms will provide input into the RCMS and MCR displays.

The CNS control system will be designed to the same level of performance as the RCMS.

CNS instrumentation will monitor the CNS and will cause the FRPS to shutdown the reactor in the event of a failure in the CNS with the potential to damage the CNS in-pile assembly.

2.4.9.12 Seismic Monitoring Instrumentation

Motion Accelerometers will be provided to measure seismic activity. These instruments input into the FRPS and the SRPS and as such, will be qualified to IEEE Class 1E standards and be provided with power from the uninterruptible power supplies.

The reactor will be automatically shutdown by both the FRPS and SRPS, with two sets of three independent seismic switches, in case of a seismic event reaching the safety system settings.

2.4.10 Engineered Safety Features Design Criteria

This section specifies the criteria and requirements applicable to the design of the Engineered Safety Features (ESFs). These consist of:

- Reactor Protection Systems
- Post Accident Monitoring System
- First Shutdown System
- Second Shutdown System
- Core cooling by natural circulation

Rigs cooling by natural circulation

Reactor pool coolant boundary

Containment

Emergency Control Centre Ventilation and Pressurisation System

Standby Power System

These features are discussed in Chapter 7 and in the appropriate functional chapter (e.g. shutdown systems are discussed in Chapter 5, cooling systems are discussed in Chapter 6 etc). Their acceptability is demonstrated in the associated design evaluation and compliance sections and in the safety analysis in Chapter 16.

2.4.10.1 Safety Functions

Engineered Safety Features (ESFs) will be provided to perform the following basic safety functions:

a) To shutdown the reactor and maintain it in a safe shutdown state for design basis accidents.

b) To provide for adequate heat removal from the core and rigs after shutdown, including design basis accidents.

c) To contain radioactive materials in order to ensure releases to the environment are small.

d) To fulfil other functions (e.g., habitability after accident) – refer to Section 2.5.1.1.

All ESFs will be classified as Safety Category 1 systems (see section 2.5).

2.4.10.2 Reliability Requirements

High reliability of the ESFs will be achieved by the application of one of more of the following means, as discussed in other sections of this chapter:

a) Use of proven engineering practice and conservative design margins.

b) Use of inherent safety features, where practicable.

c) Use of passive safety features, where practicable.

d) Use of fail safe design features, where practicable.

e) Use of redundancy, diversity and independence.

f) Compliance with the single failure criterion, where practicable.

g) Qualification of equipment and components.

h) Regular inspection, testing and maintenance.

The reliability of each safety system (i.e. Safety Category 1 systems; see section 2.5) will be shown to be better than 10^{-3} failures to perform its intended function per demand in accordance with section 2.2.2.4.

2.4.10.3 Testing, Inspection and Maintenance Requirements

The ESFs and their support systems will be designed to permit testing, inspection and maintenance during commissioning and the operational life of the Reactor Facility as part of assuring their ability to fulfil their safety function.

The design of ESFs will facilitate the testing of the operation of each individual redundant train/component of the ESF, as well as the ESF in its entirety.

The design of ESFs will ensure that personnel doses associated with testing, inspection and maintenance of ESFs and their support systems are ALARA.

ESFs will be designed such that sufficient redundancy is maintained where a redundant system or component is removed from service for inspection or testing purposes.

The testing, inspection or maintenance of a redundant train of an ESF will not impede the functionality of the other train(s).

2.4.10.4 Disabling/Bypassing of ESFs

The design will reduce the need for disabling or bypassing of all, or part of, an ESF. In each case where provision is made for disabling or bypassing part or all of an ESF:

a) Annunciation will be provided of the disabling/bypass of an ESF.

b) Redundant safety systems can only be disabled/bypassed one at a time (where practicable).

2.4.10.5 Independence of Process and Safety Systems

Process control systems will be kept physically and functionally independent of safety systems as far as practicable, such that:

a) a system is not relied upon to perform both control and safety functions; and

b) failure of a process system cannot cause failure of a safety system.

2.4.10.6 Identification of Safety Systems

Safety systems and safety related systems will be distinguished by clear markings so that they stand out in contrast to systems that are not important to safety.

Safety systems will additionally be distinguished from safety-related systems by clear markings to emphasise their different safety categorisation (see section 2.5).

2.4.11 Containment

This section specifies the specific requirements and criteria applicable to the Containment, consisting of the following:

Reactor containment boundary

Containment isolation valves (CIVs)

Containment Energy Removal System (CERS)

Containment Pressure Relief and Filtered Vent System (CPRFVS)

Containment Vacuum Relief System (CVRS)

The RCS is discussed in Chapter 7 (with additional information provided in Chapters 4, 6, 8, 9 and 10) and its acceptability demonstrated in the associated design evaluation sections and in the safety analysis in Chapter 16.

2.4.11.1 General Requirements

The Containment, in conjunction with other ESFs, will limit the release of radioactive material to the environment to less than prescribed limits.

The Containment will remain intact following accidents that place a demand on the Containment, taking into account design basis conditions and environmental conditions, including transient conditions and those arising from internal and external events. Containment is a design function that is provided for radioactive releases that exceed the design basis.

The design of the Containment will employ design margins. The design margins will allow for uncertainties in defining accident phenomena and Containment response, and potential inaccuracies in calculational models.

The performance of the Containment under accident conditions will be determined in the safety analysis, taking into account the source term, environmental conditions, pressure and temperature.

The Containment is an ESF and as such, will meet the requirements for ESFs in Section 2.4.10.

2.4.11.2 Reactor Containment Boundary

The reactor containment boundary will completely enclose the reactor systems containing radioactive materials.

During normal operation, the pressure within the reactor containment boundary will be established at a level such as to prevent uncontrolled outward release from the building. In establishing this pressure, variations in atmospheric conditions will be taken into account.

The number of penetrations through the reactor containment boundary will be kept to a practical minimum.

The reactor containment boundary will be designed to facilitate performance testing.

Reactor containment boundary penetrations will be designed to facilitate testing of individual penetrations.

Personnel and equipment access to the Containment will be via airlocks equipped with doors that are interlocked to ensure that at least one door is closed whenever Containment integrity is required. Emergency access/escape doors will not be interlocked as this would interfere with their function of providing emergency egress. Containment hatches will not be interlocked as they require the use of cranes to move them.

2.4.11.3 Containment Isolation Valves

Containment isolation valves (CIVs) will automatically close, isolating the necessary Containment penetrations, on receipt of commands from the FRPS for relevant design basis accidents.

Containment isolation will be initiated by the FRPS for the penetrations that are in contact with the Containment atmosphere. Containment is an ESF and, as such, will meet the requirements for ESFs in Section 2.4.10. The I&C aspects will also comply with IEEE standards for Class 1E equipment.

The closure time of Containment penetrations will be taken into account in the design of the isolation actions and equipment.

Penetrations of the Containment involving piping or ducting that may potentially be open to the internal atmosphere will be fitted with at least two means of isolation, one on either side of the Containment boundary. Instances where this is not possible will be adequately justified.

Penetrations of the Containment involving pipe-work closed to the internal atmosphere will be fitted with a single means of isolation outside of the Containment boundary.

Each means of isolation will be located as close to the reactor containment boundary as practicable and will be designed to be fail safe.

The CIVs of the ventilation system will be able to be remotely actuated from the MCR and the ECC.

Indication of the status of CIV closures will provided in the MCR and in the ECC.

2.4.11.4 Containment Energy Removal System

A Containment Energy Removal System (CERS) will be provided to remove heat from an isolated Containment. .

The CERS is an ESF and as such, will meet the requirements for ESFs in Section 2.4.10. For other than the chiller units, the electrical aspects will use IEEE 308 and associated standards, where relevant as the basis for system design; equipment procurement, installation and system operation and I&C aspects will comply with IEEE standards for Class 1E equipment. Because there will be an installed spare chiller unit provided, the chiller units will be to high quality commercial standards.

Valves and/or switches in the CERS will be designed to fail safe.

2.4.11.5 Containment Pressure Relief and Filtered Vent System

A Containment Pressure Relief and Filtered Vent System will be provided for accident management purposes. It will provide a means of pressure relief via a filtered vent path to the environment in the event of over pressurisation of the containment. It will also provide a means of venting the containment to the environment via a filtered path under manual control for beyond design basis accidents. It will not be required to maintain the Containment pressure within the design limit following any design basis accident.

The discharge vent to the environment will be provided with filtration with appropriate retention factors to limit the release of radioactive substances to the environment.

2.4.11.6 Containment Vacuum Relief System

A Containment Vacuum Relief System will be provided to minimise the differential pressure between an isolated Containment and the environment. It will not be required following any design basis accident.

2.4.12 Active Area Heating, Ventilation and Air Conditioning Systems

This section specifies the requirements and criteria applicable to the Heating, Ventilation and Air Conditioning (HVAC) systems serving active areas. These systems are discussed in Chapter 10 (with addition information provided in Chapter 12 in relation to contamination control) and their acceptability demonstrated in the associated design evaluation section and the safety analysis in Chapter 16.

Appropriate HVAC systems will be provided to control the movement of airborne radioactivity in active areas during normal operation and following relevant design basis accidents (i.e. design basis accidents that result in a release of radioactivity in active areas). Non-containment area HVAC systems will incorporate appropriate features to prevent and mitigate (where prevention fails) the release of radioactivity to the environment following accident conditions arising in those areas.

The HVAC systems will be designed so as not to compromise the independence of redundant safety components and systems.

Provision will be made to monitor and filter the HVAC exhaust to the atmosphere.

HVAC systems serving areas containing safety systems will be designed to control environmental conditions in accordance with the requirements of the safety systems in those areas. They will incorporate appropriate provisions to prevent the ingress of gases and airborne particles that might impair the operation of the safety systems.

2.4.13 Emergency Control Centre Ventilation and Pressurisation System

An ECC Ventilation and Pressurisation System will be provided to ensure the continued habitability of the ECC during accidents that result in the unavailability or other loss of the MCR.

The ECC Ventilation and Pressurisation System is an ESF and will meet the requirements for ESFs in Section 2.4.10. The electrical aspects will use IEEE 308, and associated standards, where relevant as the basis for system design, equipment procurement, installation and system operation and I&C aspects of the system will comply with IEEE standards for Class 1E equipment.

The ECC Ventilation and Pressurisation System is discussed in Chapter 10 (with additional information in Chapter 8 relating to the function of the ECC itself) and its acceptability is demonstrated in the associated design evaluation.

2.4.14 Sharing of Structures and Systems Important to Safety Between the Reactor Facility and Other Facilities at LHSTC

This section specifies the requirements and criteria applicable to the sharing of structures and systems important to safety between the Reactor Facility and other facilities at the Lucas Heights Science and Technology Centre (LHSTC). These are mainly discussed in Chapters 4 and 10 with additional information provided in other chapters as appropriate (e.g. in Chapter 9 with respect to the electrical power supplies interfaces). Their acceptability is demonstrated in the associated design evaluation and compliance sections of these chapters.

During normal operation, the Reactor Facility will be supplied with some services from other facilities within the LHSTC. However, in each case, an alternative source of supply dedicated to the Reactor Facility will be provided to ensure continuity of supply in the event that the connection with the rest of the LHSTC is lost or degraded as follows:

a) Fire: the facility will have its own water reservoir, pumps and distribution network (see Chapter 10).

b) Electric power supply: the facility will have its own electrical power supply with redundant diesel generators (see Chapter 9).

c) Cooling capability: the facility cooling towers will have a pool capacity large enough to ensure decay heat removal in case of loss of make-up water supply from LHSTC (see chapter 6).

d) Liquid waste: the facility will have its own tanks for collecting liquid wastes from its laboratories and operational areas (see Chapter 12).

e) Compressed air: the facility will have its own compressed air system including compressors and distribution network (see Chapter 10).

f) Telecommunications: the facility will have its own telecommunications arrangements, including emergency warning system (see Chapter 8).

g) Fuel for diesel generators: the facility is provided with its own fuel storage tanks for the diesel generators (see Chapter 9).

2.4.15 Consideration of Human Factors and Ergonomic Design Principles

This section specifies the requirements and criteria applicable to the design with respect to human factors and ergonomic design principles. These complement the requirements identified in section 2.3.8 and are principally discussed in Chapter 8.

Structures, systems and components will be ergonomically designed for all human-machine interfaces including operation of controls, monitoring of instrumentation, lifting and handling facilities, steps, handrails etc.

The man-machine interface will be centralised in the MCR and designed such that:

a) Displays provide optimal conditions for assimilation of information.

b) Operators are provided with comprehensive, but easily managed information.

c) The manner in which information is provided is compatible with decision times and action times.

d) Information displayed provides an overall picture of the status of the facility, allowing the operators to readily assess the facility status and determine appropriate operator initiated safety actions.

The selection and arrangement of instrumentation and the means of display will be systematically considered in the design to take into account ergonomic principles, to allow operators to readily assimilate information and take appropriate safety related actions, thus reducing the possibility of human error.

The design will consider human factors such as human anthropometric, perceptual, cognitive, physiological and motor response and limitations.

Anthropometric considerations will be taken into account to set the basis of physical requirements for the dimensions and form of control panels, boards and equipment.

The design will ensures that operations involving reactivity change are carried out under control and observation while the control and instrumentation systems are operational.

System design and interlocks will prohibit unauthorised and unassisted, either in error or deliberately, withdrawal of control rods.

Audio and visual aspects of the plant will be considered such that the audio level, sound frequency, intensity levels, maximum ambient noise and illumination levels permit tasks to be carried out in an efficient and exact manner and compatible with Australian regulations and standards. The environmental conditions for each operation will be taken into consideration in the elaboration of operational procedures.

The design of the reactor pool will allow for clear visibility of as many of the core components as practicable.

Valves will be designed such that:

a) They are located in accessible positions and, where possible, do not require operators to climb ladders or other equipment to operate them.

b) They are clearly marked to identify open and close operations.

c) Provision for locking is available where the valve may present a danger to personnel, plant or the environment if operated incorrectly.

d) All safety related locked valves give alarm in the control room when unlocked.

Gauges and measuring equipment will be designed such that:

a) Equipment is mounted in a manner that permits easy and accurate reading.

b) Gauges have sufficient capacity to operate over the maximum range of the parameter being monitored, including accident conditions without compromising the integrity of the system involved.

c) Gauges have scales and reading accuracy commensurate with the level of accuracy applicable to the system or condition being monitored.

d) Instruments are mounted for ease of reading and for easy access for calibration, adjustment, removal and repair.

2.4.16 Design Analysis Techniques

This section specifies the requirements and criteria applicable to the design analysis techniques used in the design and analysis of the facility. These techniques are principally discussed in Chapters 5, 6, 7, 8 and 16 with their acceptability also demonstrated in these chapters.

Design analysis will be carried out using validated techniques, models, prototypes and computer codes.

Validation will be performed against one or more of the following:

a) theoretical formulations and/or solutions;

b) values reported in the literature;

c) experimental measurements on prototypes or scaled models;

d) numerical benchmarks;

e) other design techniques or models.

2.4.17 Design Criteria for Reactor Utilisation

This section specifies the requirements and criteria applicable to reactor utilisation including the neutron beams and the irradiation facilities. These requirements are discussed in Chapter 11 and their acceptability demonstrated in the associated design evaluation sections and the safety analysis in Chapter 16.

2.4.17.1 General Requirements

The design of the Reactor Facility will be such that reactor utilisation does not adversely affect the safety of the reactor during any operational states.

All utilisation devices loaded into, or directly connected to, the reactor are designed to equivalent standards as the reactor itself and are fully compatible in terms of material used, structural integrity and radiation protection.

2.4.17.2 Beams

2.4.17.2.1 Cold Neutron Source System

The moderator vessel of the CNS will be designed, manufactured and tested in accordance with ASME to provide adequate strength and rigidity to ensure safe operation during operational states and design basis accidents.

The CNS cooling system will be simple and robust in order to promote reliable and safe operations during its lifetime. It will be capable of removing all heat from the CNS system, including moderator vessel and associated components, for full power reactor operation with a safety margin.

All instrumentation and control systems associated with the CNS will be supplied by the Uninterruptible Power Supply.

The system will be designed to ensure safe operation at all times and so that postulated failures of the cold source cannot affect reactor safety.

In case of mechanical failure or accident conditions, the working fluids will be retained inside the system to limit the spread of contamination and release of the fluids to the reactor building and any other confined space. Non-hazardous working fluids will be exhausted into a suitable vent stack.

The CNS system will be controlled automatically and monitored from the MCR.

A system will be provided to warn of abnormal operating conditions and malfunctions of the CNS or its supporting services and, in extreme cases, to request a reactor shutdown via the FRPS.

2.4.17.3 Irradiation Facilities

2.4.17.3.1 General Requirements

Irradiation rigs within the reflector vessel will be firmly secured during operation. Connections between the reflector vessel structure and rigs will be designed for ease of use.

Sufficient cooling capacity will be provided to each irradiation facility to dissipate the maximum heat being generated. Facilities will be designed to avoid boiling on the surface of the irradiated element for steady state normal operating conditions.

The design of irradiation facilities will incorporate sufficient instrumentation to ensure the safe operation of the irradiation rigs.

Limitations will be specified on the quantities of irradiated materials handled by each system.

All the elements in contact with the coolant will be resistant to corrosion.

Adequate means and procedures will be provided to control the loading of irradiation rigs and targets into the reactor and their paths once they have been irradiated.

Removal of fixed irradiation targets will only be possible with the reactor in the shutdown state.

2.4.17.3.2 Interfaces between Irradiation, Transfer and Processing Facilities

The target handling facilities will have sufficient shielding for the safe handling of irradiation targets and rigs containing irradiated targets at the maximum credible radiation levels subject to procedures such as a minimum cooling period following removal from the reactor.

The reactor hot cells will have suitable provisions to:

a) Ensure the radiation dose to operators will be within limits.

b) Provide means to ensure the safe manipulation of the materials inside the cell, including lighting, manipulators, seeing devices.

c) Monitor radiation fields in the area.

2.4.18 Protection against Dependent Failure

Protection against dependent failure (common cause failure) will be provided through the following design conditions:

a) Use of diversity, where practicable.

b) Provision of independence between redundant systems and components and between levels of defence in depth. Independence may include both functional isolation and physical separation.

c) Provision of on line testing and diagnostic techniques, and in particular, the implementation of watch-dog and dynamic-signal state and periodic auto testing techniques in the FRPS and SRPS.

2.4.19 Capability for Surveillance and Maintenance of Safety Related Equipment

This section specifies the general requirements and criteria applicable to the surveillance and maintenance of safety related equipment and systems. These requirements are discussed in the various chapters and sections where safety related equipment is described.

The Reactor Facility will be designed to minimise the maintenance required on structures, systems and components.

The design of the Reactor Facility will incorporate provision for appropriate accessibility, adequate shielding, remote handling and decontamination, as required, to facilitate maintenance and repair.

All systems important to safety will incorporate features to enable surveillance and maintenance to ensure they fulfil their safety functions with high reliability.

The design of the Reactor Facility will allow for the inspection of passive components whose ability to function is not normally verified by routine operations, and the surveillance of materials whose properties may change in service due to ageing.

The design of structures, systems and components to facilitate maintenance and surveillance will take into account:

a) The ease of performing inspections, tests and repairs.

b) The degree to which inspections and tests represent real conditions.

c) The need to maintain the safety function during inspection, testing or repair.

Special equipment, namely valves, interlocks and monitoring systems, will be provided to assist the performance of maintenance operations, and to ensure that the systems are properly configured after a maintenance operation has come to an end.

2.4.20 Electrical Power Supply Design Criteria

This section specifies the specific requirements and criteria applicable to the provision of electrical supplies to the facility. These systems consist of:

Normal Power System (NPS)

Standby Power System (SPS)

These systems each receive electric power from one of three sources, normal power from off-site sources, standby power from on-site diesels and uninterruptible power from UPS units and batteries. The NPS and SPS are discussed in Chapter 9 and demonstrated to be acceptable in the associated design evaluation sections in Chapter 9 and the safety analysis in Chapter 16.

2.4.20.1 General Requirements

Sufficient normal, standby and uninterruptible sources of electrical power will be provided to bring about and maintain a safe shutdown state for operational states and design basis accidents.

Protective relaying will be provided throughout the electric supply system to detect and isolate faulted equipment from the system with a minimum of disturbance in case of equipment failure.

Monitoring of electrical power supply equipment (generators, transformers, and circuits) will be provided in the MCR. In addition, all equipment required to control and operate under abnormal or incident conditions will be capable of being remotely operated from the MCR. Display and limited actions will be available in the ECC.

2.4.20.2 Standby Power System

The SPS will be capable of supplying and have sufficient capacity to supply electrical power to all systems requiring essential electrical power concurrently, including starting load requirements, assuming the coincidental loss of off-site power. The SPS will include uninterruptible power supplies for loads which cannot tolerate a break in supply.

The SPS will also be designed such that the effects of design basis accidents do not result in loss of function.

The SPS is classified as an ESF and as such, will meet the requirements for ESFs identified in section 2.4.10. The Standby Power System will use IEEE 308, and associated standards, where relevant, as the basis for system design, equipment procurement, installation and system operation with the exception of the diesel generators which will consist of components to high quality commercial standards.

The SPS will be designed to facilitate testing of the functional capability of the system.

The principles of redundancy, diversity and independence will be applied in the design of the SPS in accordance with the IEEE standards. The degree of redundancy and independence used is consistent with that used in the relevant ESFs being served.

The reliability of electrical power supply to safety systems will be evaluated and demonstrated to be acceptable.

The standby power diesel generators in the SPS will be started and loaded automatically.

2.4.21 Auxiliary System Criteria

This section specifies the requirements and criteria applicable to auxiliary systems. These are discussed in Chapter 10 and their acceptability is demonstrated in the associated design evaluation sections and the safety analysis in Chapter 16.

2.4.21.1 Fuel Handling and Storage Systems

The design of the Reactor Facility will include provisions for the handling and storage of fresh and irradiated fuel.

Ample fuel storage capacity will be provided to allow the core to be unloaded at all times.

Fuel handling and storage facilities will be designed to:

a) Prevent inadvertent criticality of fresh or irradiated fuel.

b) Maintain adequate shielding and cooling for irradiated fuel during operational states and design basis accidents.

c) Permit periodic inspection and testing of irradiated fuel.

d) Minimise the probability of loss or damage to fresh or irradiated fuel.

e) Permit the storage of suspect or damaged irradiated fuel.

f) Provide physical protection against theft and sabotage.

2.4.21.2 Other Auxiliary Systems

The other auxiliary systems: Fire Protection System, Communications System, Conventional Areas HVAC System, Compressed Air System, Demineralised Water Supply System, Nitrogen, Helium, Oxygen and Argon Supply Systems, Service Water Systems, Breathing Air System, will be designed to function as required during operational states and relevant design basis accidents.

The design of the Reactor Facility will ensure that a failure in an auxiliary system that is not required to shutdown the reactor or maintain it in a safe shutdown condition will not prevent those systems that are required so to do from performing their safety functions.

2.4.22 Radiation Protection in the Design

This section specifies the requirements and criteria applicable to the radiation protection measures incorporated into the design of the facility. These complement the requirements identified in section 2.3.9 and are discussed in Chapter 12 where their acceptability is also demonstrated.

Radiation shielding will be provided so that during normal operation all doses to operational staff will be ALARA.

Contact dose rates on the outside of external shielding will not be greater than $10\mu Svh^{-1}$. Areas of continuous occupancy will have dose rates not greater than $0.5\mu Svh^{-1}$.

The dose rate at the operation bridge above the reactor pool will not be greater than $8\mu Svh^{-1}$.

The shielding of the reactor will be developed from established designs to ensure that dose limits to operators are not exceeded.

The design of the Reactor Facility will take full account of the tritium build-up in the heavy water system.

The design of the irradiation facilities will include measures to minimise the argon build-up in irradiation tubes and pneumatic facilities.

Radiation shielding will be provided and access control patterns established to allow properly trained operating staff to control radiation doses within the limits of applicable regulations during operational states.

The PCS and the RSPCS will be provided with delay tanks to allow for the decay of nitrogen-16 contained in the primary coolant.

2.4.22.1 Design Criteria to Reduce Exposures

Special emphasis will be placed on the plant layout and personnel circulation paths in order to limit exposure of the plant staff and propagation of contamination.

This may be achieved by the following means:

a) separation of radiation zones;
b) adequate dispositions to permit ventilation;
c) special installations for equipment manipulation;
d) special installations for dressing rooms;
e) control of access;
f) techniques of remote control;
g) installations for decontamination and shielding;
h) making use of other different aspects of systems and components design.

The length of routes to be followed by the staff through sub-zones exposed to radiation and contamination will be minimised in order to reduce the transit time.

In order to minimise radiation doses to the personnel, the layout of the controlled zone will be such that personnel do not have to traverse zones of more intense radiation to reach areas of less intense radiation.

2.4.22.2 Prevention of Inadvertent Criticality

Nucleonic instrumentation will be provided to allow measurement of core neutron population variations from the source range in order to approach criticality in a deliberate fashion.

Storage and handling equipment for new and irradiated fuel will be designed to ensure that there can be no critical assembly of the fuel.

2.4.22.3 Waste Management

Waste management facilities will include appropriate facilities for the safe handling and temporary storage of radioactive and non-radioactive waste. These will include shielded facilities for delay and decay.

Waste management arrangements will comply with:

a) relevant waste management standards produced by the IAEA.
b) the requirements of state and federal authorities.
c) the waste operations acceptance criteria for liquid wastes used on the site.

End of Section

2.5 CLASSIFICATION OF SYSTEMS, STRUCTURES AND COMPONENTS

This section specifies the requirements and criteria applicable to the safety, seismic and quality classification of systems, structures and components during the design and construction phase.

All systems, structures and components will be classified based on their importance to safety, their seismic analysis requirements, their importance to plant availability and applicable quality assurance requirements.

2.5.1 Safety Classification Methodology

Systems (including their subsystems), structures and components will be classified using three Safety Categories:

Safety Category 1: Any system, structure or component that forms a primary means of ensuring nuclear safety.

Safety Category 2: Any system, structure or component that makes an important additional contribution to nuclear safety.

Safety Category 3: Any system, structure or component that is not allocated to Safety Category 1 or 2.

The methodology applied in the safety classification of systems, subsystems, structures and components of the Reactor Facility will be as follows:

a) The safety functions applicable to the Reactor Facility are identified with each safety function being assigned a nominal safety category.

b) The safety functions performed by each system, sub-system, structure or component are identified.

c) Each system, sub-system, structure or component is allocated a safety category consistent with the categorisation of the safety function(s) it performs.

The safety functions and their associated safety category are identified in Table 2.5/1.

2.5.1.1 Design Requirements for Safety Category 1 Systems, Structures and Components

Safety Category 1 systems, structures and components will be identified as such in accordance with the safety classification system.

The following items will be identified for each Safety Category 1 system, structure or component:

a) The safety functions that the system, structure or component is required to fulfil.

b) Performance requirements during operational states and following design basis fault sequences.

Safety Category 1 systems, structures and components will be capable of performing their safety function under normal operating conditions and anticipated operational occurrences.

Safety Category 1 systems, structures and components will be designed in accordance with the applicable codes and standards.

Design for Safety Category 1 systems, structures and components will comply with the principle of conservative design and construction standards.

Safety Category 1 systems, structures and components will be qualified so that they can fulfil their function following design basis accidents. This will be demonstrated by tests in compliance with applicable standards or by other means as identified during the detail engineering phase.

Materials of Safety Category 1 systems, structures and components will meet the applicable specifications and have a proven service record.

The ability of each Safety Category 1 system, structure or component to meet its identified performance requirement(s) and thus fulfil its required safety functions, will be demonstrated.

A Dependent Failure Assessment will be performed for Safety Category 1 components, structures and systems.

Failure Modes and Effects Analyses (FMEAs) will be performed as necessary for Safety Category 1 systems, structures and components. These analyses will incorporate single failure assessments to demonstrate compliance with the single failure criterion of section 2.3.

2.5.1.2 Design Requirements for Safety Category 2 Systems, Structures and Components

Safety Category 2 systems, structures and components will be identified in accordance with the safety classification system.

The following will be identified for Safety Category 2 systems, structures and components:

a) The safety functions that the system, structure or component is required to fulfil either individually or in conjunction with other systems, structures or components.

b) Performance requirements during operational states and following relevant design basis fault sequences.

Safety Category 2 systems, structures and components will be capable of performing their function under normal operating conditions and anticipated operational occurrences.

Safety Category 2 systems, structures and components will be designed in accordance with the applicable codes and standards.

Design for Safety Category 2 systems, structures and components will comply with appropriate national or international industrial codes and standards with particular consideration being given to demonstrating the ability to perform the required safety function.

Safety Category 2 Systems, structures and components will be qualified such that they can fulfil their function following relevant design basis accidents. This will be demonstrated by testing, by comparison with the same (or similar) components that have survived similar conditions, or by other processes as determined during the detail engineering phase.

The ability of each Safety Category 2 system, structure or component to perform the required safety function will be demonstrated by suitable analysis techniques.

2.5.1.3 Design Requirements for Safety Category 3 structures, Systems and Components

Safety Category 3 systems, structures and components will be identified in accordance with the safety classification system.

Normal industrial standards will be applied in the design of Safety Category 3 systems, structures and components the minimum standard being the appropriate Australian standard for the equipment or system.

2.5.2 Seismic Classification Methodology

Systems, structures and major components will be allocated to an appropriate seismic class that takes into consideration the specific site characteristics to ensure achievement of the safe shutdown state following an earthquake. The Seismic Classes are:

a) Seismic Class 1: Items within this class are designed to withstand the consequences of ground motion associated with earthquake level SL-2 (Seismic Level 2, also denoted as Safe Shutdown Earthquake).

b) Seismic Class 2: Items within this class are designed to withstand the consequences of ground motion associated with earthquake level SL-1 (Seismic Level 1, also denoted as Operational Basis Earthquake).

c) Seismic Class 3: Items within this class are designed to withstand the consequences of ground motion associated with normal building and industrial codes.

d) Not Applicable: This class involves those items for which no seismic analysis applies, namely:

   (i) Systems, structures or components for which seismic analysis is inappropriate (e.g. software, fluids).

   (ii) Items fixed to buildings (the support structure of these items will be analysed, but the items will not be, e.g. cables).

   (iii) Items not fixed to buildings or structures (e.g. portable equipment, casks).

Seismic Class 1 will include:

a) Items whose failure could directly or indirectly cause accident conditions.

b) Items required for shutting down the reactor, monitoring critical parameters, maintaining the reactor in a shutdown condition and removing residual heat.

c) Items required to prevent radioactive releases.

The items in this class have to maintain their structural and be able to carry out their safety function during and after the SL-2 earthquake, even though they may remain non-operational.

Seismic Class 2 will include:

a) Items that are not in Seismic Class 1 but which are required to prevent the escape of radioactivity in excess of normal operational limits.

b) Items that are not in Seismic Class 1 but which are required to mitigate accident conditions which last for sufficiently long periods that there is a reasonable likelihood that an earthquake may occur during this period.

c) Items without nuclear connotation but for which it is reasonably a protection against an earthquake greater than the one associated with normal building and industrial codes, taking into account the economic value of the item.

The items in this class will remain operational during and after the SL-1 earthquake. The item will only need inspection or minor repair prior to recommencing operation after such an event.

Seismic Class 3 will include all systems, structures and components not classified as Seismic Class 1 and 2. They will be designed to industrial and building standards and codes. No requirement is placed on their structural and functional state following SL-1 and SL-2 seismic events.

According to IAEA Safety Series 50-SG-D15, when, as the result of an earthquake, the collapse, falling, dislodgement or any spatial response of an item is expected on the basis of analysis, test or experience to occur and could jeopardise the functioning of items in a higher class:

a) Such items will receive the same categorisation as the endangered items.

b) Under the reference ground motion, the absence of collapse or loss of function of the lower class items will be demonstrated.

c) The endangered items will be suitably protected so their functionality is not jeopardised.

2.5.3 Quality Classification Methodology

The quality classification methodology is a systematic means of identifying the appropriate quality assurance rating to systems and sub-systems. The systems and subsystems in the facility will be quality-rated using an assessment process, which considers the safety and availability classifications of each item, its complexity and previous design experience.

To perform the quality classification, the ratings of the different factors will be added to obtain a total quality rating for each system and subsystem.

Factors Determining Total Rating of Each Component/System/Subsystem

In the process of designing the facility a number of factors have been used in determining the rating of the components, systems and sub-systems. Included is a safety factor, availability requirements and design complexity.

End of Section

Table 2.5/1 Definition of Safety Functions

Safety Function	Safety Function Description	Safety Category
A	To form the primary barrier against the release of fission products from fuel.	1
B	To store and manipulate new and irradiated fuel.	2
C	To start protective actions in order to shutdown the reactor, to cool and contain radioactive materials, and to mitigate accident consequences.	1
D	To provide Post Accident Monitoring of the reactor in order to: a) Provide information to operators to indicate whether plant safety functions are being accomplished. b) Indicate the successful operation of individual safety systems c) Alert operators to take safety actions for initiating a system function that is not automatic. d) Indicate to operators when barriers to fission product release have the potential for being breached. e) Determine the magnitude of release of radioactive materials.	1
E	To shutdown the reactor whenever necessary in order to: a) avoid anticipated operational occurrences that could lead to accidental conditions, thus preventing fuel design and other limits from being exceeded b) allay the eventual consequences of accident conditions.	1
F	To control the plant keeping the reactor parameters within operational limits without reaching safety limits. This includes data gathering, processing and displaying of information on plant status to the operator to give early warning on plant deviation from normal operation to start control/corrective actions.	2
G	To control the reactor core reactivity during operational states, including the prevention of unacceptable reactivity transients.	2
H	To prevent the occurrence of inadvertent criticality in fuel handling or storage.	2
I	To keep available sufficient quantities of reactor coolant, and/or to remove heat from the core in order to cool it during and after any design basis accident.	1
J	To remove decay heat from the core during the various operational situations and design basis accidents where the reactor coolant boundary remains intact.	1
K	To keep available sufficient quantities of reactor coolant in order to cool the core and/or reflector during operational states.	2
L	To transfer heat derived from other safety systems to the atmosphere.	2
M	To limit radioactive material releases from the reactor containment during and after accidents	1
N	To limit discharge or release of radioactive waste or radioactive effluents in suspension in the water or air to levels below pre-set values during any operational situation.	2

Safety Function	Safety Function Description	Safety Category
O	To prevent the failure of a component or structure which, should it occur, would: a) endanger a safety function, b) alter the core geometry/configuration. or to limit the consequences of such a failure.	2
P	To ensure all necessary provisions (such as electrical supply, electric grounding, compressed air, water, hydraulic pressure, etc.) to safety systems in order to ensure their capability to perform their safety functions when required.	2
Q	To protect operators during the handling and manipulation of radioisotopes.	2
R	To keep environmental conditions inside the Reactor Facility under control so that safety systems will function appropriately, ensuring adequate protection against the adverse effects of factors such as: fire, explosion, flooding, lightning, temperature, humidity, aircraft impact, breach of plant security.	2
S	To provide adequate protection from radiation exposure to operating staff and research personnel, by means such as: a) Shielding. b) Measurements and warnings. c) Maintenance of sufficient pressure differential between different parts of the containment. d) Transport delay to allow decay of radioactive products. e) Purification and filtering of the containment.	2
T	To provide a system of barriers that ensure adequate protection of the public during the transport of spent fuel assemblies outside the reactor site.	1
U	To provide a secure and safe place from where an emergency can be managed in case that the Main Control Room has been evacuated.	1
V1	To provide the primary means of protection to the core or safety systems against the hazards imposed by reactor uses with a large effect on nuclear safety.	1
V2	To control the conditions of reactor use or radioisotope targets with a major effect on nuclear safety by such means as: a) keeping the system parameters within limits b) limiting the pressure and temperature build-up.	2
W	To remove decay heat from the reactor pool during operational situations.	2
X	To provide a means for mitigating the consequences of beyond design basis accidents.	2

End of Section

2.6 DESIGN CRITERIA TO WITHSTAND EXTERNAL EVENTS

The facility is designed to withstand the external events appropriate to the LHSTC site. Design criteria in relation to seismic, aircraft impact, wind/tornado and flood events are described in the following sections and their bases are provided in Chapter 3.

2.6.1 Design Criteria for the Resistance against Seismic Hazard

The methodology for seismic definitions and qualification will be adopted in accordance with the provisions of the IAEA and the provisions of the U.S. Nuclear Regulatory Commission, by means of the following main documents:

IAEA Safety Series 50-SG-S1 (Rev 1) - Earthquake and Associated Topics in Relation to Nuclear Power Plant Siting.

IAEA Safety Series 50-SG-D15 - Seismic Design and Qualification for Nuclear Power Plants.

US NRC Regulatory Guide 1.61 - Damping values for seismic design of nuclear power plants.

US NRC Regulatory Guide 1.92 - Combining modal responses and spatial components in seismic response analysis.

US NRC Regulatory Guide 1.122 - Development of floor design response spectra for seismic design of floor-supported equipment or components.

The scope of these regulations is for nuclear power plants and exceeds the requirements for research reactors. Nevertheless, they will be adopted due to the lack of appropriate regulations applicable to research reactors.

IAEA-TECDOC-348 (Earthquake Resistant Design of Nuclear Facilities with Limited Radioactive Inventory) is a Technical Document that presents simplified methods and procedures related to seismic safety design for facilities with limited radioactive inventory. It uses a simplified and conservative approach with the emphasis on appropriate construction and detailing principles rather than sophisticated dynamic analysis. TECDOC 348 (1985 issue) specifies that “for specific structural arrangements such that the simplified approach may not be valid and in cases where a large safety margin is required for nuclear safety reasons a more complete and refined approach, as those discussed in IAEA safety Guides 50-SG-S1, 50-SG-S2 and 50-SG-S8 may be preferred”. Therefore, the more rigorous IAEA-SG guides will be used for seismic definitions and qualification while TECDOC 348 is used for the provisions related to anchorage, piping supports, etc.

In accordance with the IAEA Safety Series, two levels of earthquake will be adopted for the structural design of the facility:

a) a lower level SL-1 , or Operating Base Earthquake (OBE); and

b) a higher level SL-2, or Safe Shutdown Earthquake (SSE).

A third level of earthquake, the SL-0 earthquake, will also adopted for the design of civil structures not important to safety.

The ground motions for the SL-1 and SL-2 levels of earthquake will be defined by means of Design Response Spectra (DRS). For each particular spectrum the following characteristics are defined:

a) Peak Ground Acceleration (PGA).

b) Control Points defining the amplifications of the PGA with period.

c) Adopted damping factor.

To justify the adopted values for design, and to facilitate the comparisons with international requirements, the spectra for both events will be defined first as a Reference or Basic Response Spectrum (BRS). These correspond to the typical 5 % of critical damping, whose spectral ordinates are taken as reference values and scaled appropriately according to the adopted damping coefficient to get the final DRS for each event.

As the building will be designed to have essentially an elastic response for both levels of earthquake, the response of the building, and therefore the ductility factor, will be set to unity for the purpose of design. Therefore, the input will be defined in both cases as Elastic Design Response Spectra (EDRS) and no allowance is permitted to reduce the spectral accelerations. Notwithstanding and recognising the level of uncertainties involved in the definition of the seismic input, the occurrence of the SL-2 event is expected to result in the overall structure undergoing some degree of inelastic response beyond the elastic strength defined for that earthquake. In consequence, the detailing of the reinforced concrete critical sections will be such that brittle modes of failure are either avoided or delayed, to achieve a certain degree of local energy dissipation

2.6.1.1 Design Basis Ground Motions

The design basis ground motions for the three seismic levels adopted for the whole facility are:

a) SL-2: Peak Horizontal Ground Acceleration: 0.37g

Peak Vertical Ground Acceleration: 0.25g

The acceleration response spectrum shape is taken as an envelope between the IGNS spectrum scaled to 0.37g and the US NRC Regulatory Guide 1.60 scaled to the 0.30g Peak Ground Acceleration. This envelope will maximise the acceleration over the whole frequency range.

b) SL-1: Peak Horizontal Ground Acceleration: 0.09g

Peak Vertical Ground Acceleration: 0.06g

The acceleration response spectrum shape is taken from US NRC Regulatory Guide 1.60 scaled to the Peak Ground Acceleration

c) SL-0: This level corresponds to the earthquake loads for civil structures. It is specified in accordance to AS1170.4, Minimum Design Loads on Structures (SAA Loading Code) Part 4: Earthquake Loads.

2.6.1.2 Basic Response Spectra

2.6.1.2.1 Safe Shutdown Earthquake, SL-2

SL-2 BRS has been taken as an envelope between the local response spectrum at the Reactor Facility site developed by IGNS (Institute of Geological & Nuclear Sciences, July, 2001) and the US NRC Regulatory Guide 1.60. With this method, the maximum of each spectrum has been considered in order to maximise the design acceleration at all periods.

Experience with previously built Research Reactors and the review of relevant documentation on seismic input demonstrates that US NRC 1.60 is a worldwide standard that bounds the design requirements for research reactors.

This BRS complies with the recommendation given by the IAEA Safety Series 50-SG-S2, yields the following horizontal acceleration spectral ordinates for the control points:

Period, T (sec)	Acceleration (g)	Dynamic Amplification Factor, A
0.00	0.37	PGA
0.03	0.37	1.00
0.11	0.78	2.11
0.20	0.86	2.32
0.40	0.94	2.54
1.00	0.46	1.24
3.85	0.14	0.38

2.6.1.2.2 Operating Basis Earthquake, SL-1

The US NRC Regulatory Guide 1.60 is adopted as a basis for the BRS spectrum. In this case the reference ordinates are scaled to give a 0.09g PGA, which results in the following horizontal accelerations for the control points:

Period, T (sec)	Acceleration (g)	Dynamic Amplification Factor, A
0.00	0.09	PGA
0.03	0.09	1.00
0.11	0.23	2.55
0.20	0.26	2.89
0.40	0.28	3.11
1.00	0.14	1.56
3.85	0.04	0.44

2.6.1.3 Design Response Spectra

The Response Spectra to be used in the design both for buildings and components will be derived from the Basic Response Spectra (BRS). The applicable Design Response Spectra (DRS) will be taken as a function of the seismic level and damping for all items, and the level location in the building for components.

For equipment and components analysis the applicable Design Floor Response Spectra are derived from the Basic Response Spectra.

No reduction factors will be used in order to reduce the Design Response Spectra either for buildings and components. This means ductility equal to 1 (see Section 2.6.1.5).

2.6.1.3.1 Damping Values

Damping is a measure of the energy dissipation in a vibratory body due to hysteresis, friction, impact, joints, etc. Damping values for the analysis of structures, components and equipment are based on the provisions of Regulatory Guide 1.61.

2.6.1.4 Soil-Structure Interaction

The effect of soil-structure interactions will be included in the seismic analysis of the Reactor Building structure.

2.6.1.5 Ductility

2.6.1.5.1 Seismic Categories 1 and 2

All systems, structures, and components will be designed to remain in the elastic range for the seismic level corresponding to them. This means that the ductility factor, as it is usually defined in order to reduce the DRS, is taken as equal to 1 (one). Therefore, no reduction factor will be required in the design of both structures and components.

Concrete civil structures will be designed to avoid brittle behaviour. This will be been carried out by limiting steel percentage, ensuring reinforcement continuity and adopting recommended joint details from seismic codes.

2.6.1.5.2 Seismic Category 3

For structures, systems and components classified as Seismic Category 3, beyond elastic behaviour is allowed, especially for civil structures. Thus, the DRS can be reduced in the application of this code.

2.6.1.6 Seismic Qualification

Seismic qualification refers to the demonstration of structural integrity and/or the ability of the system to perform its required function during and/or after the applicable design earthquake.

Seismic qualification will be achieved by means of:

- Analysis
- Testing
- Earthquake experience
- Indirect methods

The analysis, testing and earthquake experience methods are also named as direct methods. The approach adopted will depend on the type, size, shape, and complexity of the item.

An analytical approach will be adopted when the dynamic behaviour of a system can be adequately modelled.

Testing will be used when the behaviour of a system cannot be reliably predicted by analysis. It is also possible to use combinations of these methods. A combination of methods will be adopted where no individual method can provide adequate proof of seismic qualification due to the nature of the item.

2.6.1.7 Seismic Design Criteria

2.6.1.7.1 Civil Structures

In order to assure an appropriately safe seismic design, the following points will be evaluated in the building analysis:

a) The adequacy of the supporting strata.

b) The suitability of types of foundation supports or of different types of foundations under interconnected structures.

c) A balanced and symmetrical arrangement of structural frames and shear walls to achieve optimum stiffness, load and weight distribution, with minimum torsional effects.

d) The need to prevent collision between adjacent buildings as a consequence of their dynamic deformations.

e) The adequacy of the connections of annexes and appendages to the main structure.

f) The need to ensure sufficient resistance of essential structural elements, especially to lateral shear forces.

g) The need to ensure sufficient ductility and avoid brittle failure by shear or compression, for example by ensuring that there is an adequate amount of reinforcement steel, in particular enough hoop ties for columns.

h) The arrangement and distribution of reinforcing bars.

i) The need for the design of joints between structural elements and the anchorage of items cast into concrete to ensure ductile failure modes.

j) An evaluation of the bending moments arising from the vertical forces and the lateral deformation due to the P-Δ effect of the earthquake on the structures when large deformations are permitted.

Seismic Categories 1 and 2 buildings will be analysed by means of mathematical models with the response spectrum method.

2.6.1.7.2 Equipment and Components

2.6.1.7.2.1 Mechanical Components

The codes specified in the following sections are general for structural design and not only for seismic analysis. The earthquake loads will be combined with all the other loads.

2.6.1.7.2.1.1 Pressurised Components

Pressurised components and parts of such components will be designed following the stress limits criteria specified in:

ASME Boiler and Pressure Vessel Code.

AS 1210 Pressure Vessels.

2.6.1.7.2.1.2 Internal Pools Structures and Not Pressurised Components

In the absence of specific codes or standards for the design of these components, the stress limits criteria will be adopted in accordance with ASME Boiler and Pressure Vessel Code.

Specific stress limits will be indicated in the respective analysis of each component presented in the corresponding PSAR Chapters.

Earthquake loads will be taken according to the seismic category and the corresponding Floor Response Spectrum.

2.6.1.7.2.1.3 Structural Steel

The design criteria for components constructed with structural steel and not designed to ASME, will be selected to comply with the requirements stated in Section 2.6.1.5.

2.6.1.7.2.1.4 Electrical and Electronic Components

The seismic qualification of Safety Category 1 electrical and electronic components will be made in accordance with the IEEE standards:

The scope of these IEEE standards is intended for Nuclear Power Plants and exceeds the requirements for a Research Reactor. Therefore, while the relevant and general procedures will be adopted in the design, an indication will be given where specific requirements are not applicable.

According to these standards, the seismic qualification will be performed by analysis or testing. The qualification tests or analysis for equipment purchased as “Qualified 1E” will be provided by the supplier of the equipment.

The qualification of an electronic or electrical component will include all the other sources of possible loads (namely vibration, temperature, radiation, ageing, etc) in combination with the seismic loads.

When the component is mounted on a flexible support (such as racks and consoles) the flexibility of this support will be considered in order to amplify the seismic action on the component. If the support is firmly fixed to the building structure, avoiding relative displacements, then it can be assumed that the equipment will be subjected to the motion represented by the floor response spectrum at the corresponding level.

2.6.1.7.2.1.5 Reactor Hall Crane

The Reactor Hall Building Crane is classified as Seismic Category 2. Therefore, the crane will be designed to resist the SL-1 earthquake without losing its operational condition after the event. Nevertheless, because it is a heavy load structure over the Reactor Hall and the pool, the crane will also be designed to resist the SL-2 earthquake against collapse. The operational condition during and after the SL-2 event is not required, but the requirement does exist that neither the structure nor its components shall fall into the Reactor Hall.

The bridge will be designed to remain on the runway with brakes applied, and the trolley should remain on the crane girders with brakes applied, during and after an SL-2 seismic event. Non elastic behaviour is allowed in both the structure and mechanism of the crane.

2.6.1.7.2.1.6 Anchors

Anchor is defined as the element that joints a support, structure, component or equipment to the building concrete structure.

In order to provide enough seismic resistance in anchors, practical provisions specified in TECDOC 348 will be adopted.

Anchors will be divided in two main groups: embedded and drill-in.

For embedded anchors, the minimum factor of safety against failure in any mode including pull out is 2.5. For drill-in anchors, the minimum factor of safety against failure in any mode including pull out is 4.0.[1] For expansion anchors, a minimum 6-diameter embedded length shall be provided. Embedded anchors will be preferred to fix heavy loads while drill-in anchors will be used to fix medium and light loads.

Redundancy will be used in the design of anchors for Seismic Category 1 components. The general concept of redundancy is that the failure of an anchor must not produce the lack of structural capacity of the fixation or to produce the modification of the constraint characteristics with the consequent change of vibrational behaviour. In order to comply with these assumptions all the anchors (designed with n bolts) will be capable of carrying the full design load with the failure of one bolt (designed with n-1 bolts). Moreover at least two bolts will be used per location.

Transverse shear forces will be assumed to be applied directly to the bolts, unless shear keys are provided.

2.6.1.7.2.2 Distribution Systems

Distribution systems refer to the networks of piping, ducts and raceways that distribute fluids, ventilation and cabling respectively around the facility.

2.6.1.7.2.2.1 Piping

Piping will designed in accordance with ASME and AS standards:

A minimum requirement for piping span for dead weight will be established, according to IAEA TECDOC 348. This requirement imposes the maximum distance between supports.

In order to support the seismic load, the natural frequency of the piping and the corresponding acceleration from the Response Floor Spectrum will be analysed. The stresses developed by this load will be checked with the allowable limits given in ASME.

Lateral bracing to the concrete building structure is provided to piping in order to withstand transversal seismic loads.

2.6.1.7.2.2.2 Ducts

Connections between duct sections will be provided by connectors that can develop axial and bending load strength capacity.

[1] IAEA Tecdoc 348 Earthquake resistant design of nuclear facilities with limited radioactive inventory (1985 issue)

2.6.1.7.2.2.3 Raceways

Effective lateral bracings will be provided for raceways, joining them directly to the building structure by means of horizontal and diagonal bars. The seismic bracing will support the forces both in transversal and longitudinal directions.

2.6.2 Design Criteria for the Resistance against Aircraft Impact

The facility will be capable of safe shutdown in the case of impact of a light aircraft.

The design will follow IAEA Safety Series Nº 50-SG-D5 (Rev 1), 1996.

2.6.3 Design Criteria for the Resistance against Wind and Tornado Loading

The design will take into account the effect of high winds on the building and structures of the facility, including the diesel generator enclosures. The following parameters will be taken into account:

a) Design wind velocity, its recurrence interval, vertical velocity profiles and applicable gust factors, applied forces.

b) Tornado loading, translational velocity, tangential velocity, pressure differential and its associated time interval, spectrum and general characteristics of tornado-generated missiles.

2.6.4 Design Criteria for the Resistance against Flood

The design of the Reactor Facility will incorporate provisions for the effect of flooding near the facility. The design will incorporate flood-relief features, such as (but not limited to):

a) flood protection measures for SL-1 structures;

b) permanent de-watering systems;

c) storm water system.

End of Section

Table 2.6/1 Design Ground Response Spectra

Period, T (sec)	Acceleration (g)
0.00	0.37
0.03	0.37
0.11	0.68
0.20	0.75
0.40	0.82
1.00	0.40
3.85	0.12

End of Tables

2.7 CODES AND STANDARDS

The codes and standards applied to the design of the Reactor Facility will follow Australian law and regulations, and international best practice. All Australian Standards (AS) and regulations pertinent to the design and construction of a nuclear facility will be followed. IAEA guides for the design and safety analysis of research reactors will be adopted. Applicable guidelines have been extracted from IAEA Safety Guides for nuclear power plants and appropriate international standards relating to the design of nuclear power plants are applied. The reactor will also be designed to meet the Regulations issued by the Autoridad Regulatoria Nuclear (ARN, Argentine Nuclear Regulatory Agency), within the guidelines of Australian regulations.

Codes and standards will be applied to structures, systems and components in a manner commensurate with the safety classification (refer Section 2.5.1).

Where different codes and standards are used for different aspects of the same item, the consistency between them will be demonstrated.

For systems, structures and components for which there are no appropriate established codes or standards, an approach derived from existing codes or standards for similar equipment will be applied. In the absence of such codes or standards, the results of experience, tests, analysis or a combination thereof will be applied and justified.

The design will follow the applicable and relevant sections of the following Codes and Standards:

Reference	
1.	Act - Australian Nuclear Science and Technology Organisation Act, 1987.
2.	Act - Australian Radiation Protection and Nuclear Safety Act, 1998 and Regulations 1999.
3.	Act - Construction Safety Act (1912-1978) and Regulations.
4.	Act - Environment Protection (Impact of Proposals) Act, 1974.
5.	Act - Environment Protection (Nuclear Codes) Act, 1978.
6.	Act - Environmental Offences and Penalties Act, 1989.
7.	Act - Environmental Planning and Assessment Act, 1979.
8.	Act - Factories, Shops & Industries Act, 1962.
9.	Act - National Environment Protection Council (New South Wales) Act 1995.
10.	Act - National Parks and Wildlife Act, 1974.
11.	Act - Noise Control Act, 1975.
12.	Act - NSW Clean Air Act 1961 and Regulations (Control of Burning) 1995, (Plant and Equipment) 1997.
13.	Act - NSW Clean Waters Act 1970 and Regulations 1972.
14.	Act - NSW Contaminated Land Management Act 1997 and Regulations 1998.
15.	Act - NSW Dangerous Goods Act 1975.
16.	Act - NSW Environmentally Hazardous Chemicals Act 1985 and Regulations 1994.
17.	Act - NSW Occupational Health & Safety Act 1983.

Reference	
18.	Act - NSW Pollution Control Act 1970 and Regulations 1985.
19.	Act - NSW State Emergency and Rescue Management Act 1989.
20.	Act - NSW Waste Minimisation and Management Act 1995 and Regulations 1996.
21.	Act - NSW Waste Recycling and Processing Service Act 1970.
22.	Act - Nuclear Non-Proliferation (Safeguards) Act 1987.
23.	Act - Occupational Health & Safety (Commonwealth Employment) Act 1991.
24.	Act - Pollution Control Amendment (Load Based Licensing) Act, 1997.
25.	Act - Protection of the Environment Operations Act, 1997.
26.	Act - Waste Minimisation and Management Act, 1995.
27.	Act - Workers Compensation Act.
28.	AECP 1054, Atomic Energy Code of Practice – Ventilation of radioactive areas.
29.	AECP 1058, Atomic Energy Code of Practice – Drainage of Radioactive Areas.
30.	ANSI N42.17A-1989 - Performance Specifications for Health Physics Instrumentation-Portable Instrumentation for use in Normal Conditions.
31.	ANSI/IEEE 352-1987, Standard Guide for General Principles of Reliability Analysis of Nuclear Power Station Safety Systems (relevant sections to be adapted for Research Reactors).
32.	ANSI/ISA - RP67.04-part II-1994, Methodology - Set Point for Nuclear Safety Related Instrumentation (relevant sections to be adapted for Research Reactors).
33.	ANSTO Engineering Procedure EP9.10, Piped Gas Systems.
34.	ANSTO Nuclear Technology Procedure NP 9.1 Criticality Safety Assessment.
35.	ANSTO Quality Policy, APOL 1.1, 26 August 1998.
36.	ANSTO Radioactive Waste Management Policy: Preliminary Environmental Review, D M Levins et al, ANSTO/E728, May 1996.
37.	ANSTO Safety Directive 2.1 Emergency Planning and Response arrangements for Lucas Heights Science and Technology Centre, May 1999.
38.	ANSTO Safety Directive 3.1 Radiation Protection Principles, May 2000.
39.	ANSTO Safety Directive 3.2 ANSTO Policy on As Low As Reasonably Achievable, May 2000.
40.	ANSTO Safety Directive 3.3 Annual Limits on Intake and Derived Air Concentrations of Some Common Radionuclides, May 2000.
41.	ANSTO Safety Directive 3.4 Radiological Classification of Areas and Access Control to Radiological Areas, May 2000.
42.	ANSTO Safety Directive 3.5 Personal Dosimetry, May 2000.
43.	ANSTO Safety Directive 3.6 Safe Movement and Transport of Radioactive Materials, May 2000.
44.	ANSTO Safety Directive 3.7 Criticality Control, May 2000.
45.	ANSTO Safety Directive 3.8 Nuclear Safety Principles, May 2000.
46.	ANSTO Safety Directive 4.2 Control of Chemical Hazards at the Lucas Heights Science and Technology Centre, May 2000.

Reference	
47.	ANSTO, General Specification for Installation of Access Control at LHSTC.
48.	ANSTO, Radioactive Waste Management Policy (approved July 1995).
49.	ANSTO, Safety Division document HE.1 Rev 1, 27/11/96, ANSTO Emergency Plan (DISPLAN).
50.	ANSTO, Safety Division document HE.2 Rev 1, 27/11/96, The Lucas Heights Science and Technology Centre Emergency Plan.
51.	ANSTO/Sydney Water Agreement Consent to Discharge Trade Wastewater, Category 3, Agreement No: 4423, 18th January 1999.
52.	ARPANSA, 1999a, Australian Radiation Protection and Nuclear Safety Agency, draft Safety Assessment Principals for Controlled Facilities (April 1999).
53.	ARPANSA, 1999b, Australian Radiation Protection and Nuclear Safety Agency, draft document RG-4, Criteria for the Siting of Controlled Facilities (April 1999).
54.	ARPANSA, 1999c, Australian Radiation Protection and Nuclear Safety Agency, draft discussion document RG-8, Criteria for the licensing of new controlled facilities (August 1999).
55.	ARPANSA, 1999d, Australian Radiation Protection and Nuclear Safety Agency, draft discussion document RG-12, Criteria for the safety analysis of controlled facilities (August 1999).
56.	ARPANSA/Nuclear Safety Bureau, Regulatory Guideline RG-5, Criteria for the Design of Nuclear Installations (Draft, 23 December 1998).
57.	AS 2344, Limits of Electromagnetic Interference from overhead A.C. power lines and high voltage installations in the frequency range 0.15 to 1000MHz (also see Draft DR 95281).
58.	AS 3735, Concrete structures for retaining liquids: Supp. 1: Commentary.
59.	ASME/ANSI B16.5 – Pipe Flanges and Flanged Fittings
60.	ASME/ANSI B31.1 – Process Piping Code
61.	ASME/ANSI B31.3 – Power Piping Code
62.	ASME Section III, Rules for Construction of Nuclear Power Components
63.	ASME Section VIII, Rules for the Construction of Pressure Vessels.
64.	Australian Attorney General’s Department document, Security Construction and Equipment Committee Criteria for Endorsed Equipment (available for reference at ANSTO).
65.	Australian Bridge Design Code, Austroads.
66.	Australian Draft Standard DR 92194, Wastewater Diversion Systems.
67.	Australian Draft Standard DR 98321 CP, Electromagnetic compatibility (EMC) – Environment - Classification of electromagnetic environments.
68.	Australian Government document, Guidelines for DSD, Secure Communication Equipment Installations (available for reference at ANSTO).
69.	Australian Government Publishing Service document, Protective Security Manual (current edition - available for reference at ANSTO)).
70.	Australian Standard AS 1012, Methods of testing concrete.
71.	Australian Standard AS 1018, Recommendations for partial discharge testing.

Reference	
72.	Australian Standard AS 1023, Low voltage switchgear and control-gear - protection of electric motors.
73.	Australian Standard AS 1024, Direct recording electrical measuring instruments and their accessories.
74.	Australian Standard AS 1025, High-voltage AC switchgear and control-gear – Switches and switch-disconnectors. Part 1 - Rated voltages above 1 kV and less than 52 kV.
75.	Australian Standard AS 1026, Impregnated paper insulated cables for electricity supply at working voltages up to and including 33 kV.
76.	Australian Standard AS 1029, Low voltage contactors.
77.	Australian Standard AS 1034, High-voltage current limiting fuses.
78.	Australian Standard AS 1042, Direct-acting indicating electrical measuring instruments and their accessories.
79.	Australian Standard AS 1078, Guide to loading of oil immersed transformers.
80.	Australian Standard AS 1102.101 to 1102.111, Graphical symbols for electro-technology.
81.	Australian Standard AS 1103, Diagrams, charts and tables for electro-technology.
82.	Australian Standard AS 1163, Structural steel hollow sections.
83.	Australian Standard AS 1170.4, Minimum design loads on structures. Part 4: Earthquake loads.
84.	Australian Standard AS 1178, Concentric wire neutral cables – XLPE insulated for electricity supply for working voltages 0.6/1kV.
85.	Australian Standard AS 1202, AC motor starters (up to and including 1000 V).
86.	Australian Standard AS 1210, Pressure vessels.
87.	Australian Standard AS 1221, Fire hose reels.
88.	Australian Standard AS 1243, Voltage transformers for measurement and protection.
89.	Australian Standard AS 1265, Bushings for alternating voltages above 1000 V.
90.	Australian Standard AS 1276, Methods for determination of sound transmission class and noise isolation class of building partitions.
91.	Australian Standard AS 1284, Electricity meters.
92.	Australian Standard AS 1302, Steel reinforcing bars for concrete.
93.	Australian Standard AS 1303, Steel reinforcing wire for concrete.
94.	Australian Standard AS 1304, Welded wire reinforcing fabric for concrete.
95.	Australian Standard AS 1306, High voltage AC switchgear and control-gear – Disconnectors (isolators) and earthing switches.
96.	Australian Standard AS 1310, Steel wire for tendons in pre-stressed concrete.
97.	Australian Standard AS 1311, Steel tendons for pre-stressed concrete - 7-wire stress-relieved steel strand for tendons in pre-stressed concrete.
98.	Australian Standard AS 1313, Steel tendons for pre-stressed concrete - Cold-worked high-tensile alloy steel bars for pre-stressed concrete.
99.	Australian Standard AS 1314, Anchorings and couplings.

Reference	
100.	Australian Standard AS 1318, Industrial safety.
101.	Australian Standard AS 1319, Safety signs for the occupational environment.
102.	Australian Standard AS 1324.2, Air filters for use in general ventilation and air-conditioning – Application, performance and construction.
103.	Australian Standard AS 1345, Identification of the contents of pipes, conduits and ducts.
104.	Australian Standard AS 1348, Road and traffic engineering glossary of terms.
105.	Australian Standard AS 1359, Rotating electrical machines – General requirements: Part 0: Introduction and list of parts. Part 2: Dimensional symbols. Part 3: Direction of rotation and marking of terminals. Part 10: Designations and dimensions. Part 20: Classification of types of enclosure. Part 30: Preferred outputs and frame sizes. Part 31: Three-phase induction motors – Operation on unbalanced voltages. Part 41: General characteristics. Part 46: Brushes, brush-holders, commutators, and slip rings - Glossary of terms. Part 47: Brushes and brush-holders – Dimensions. Part 51: Noise level limits. Part 101: Rating and performance. Part 102.1: Methods for determining losses and efficiency – General. Part 106: Methods of cooling (IC code). Part 107: Classification of types of construction and mounting arrangements (IM code). Part 109: Noise limits. Part 114: Vibration measurements and limits.
106.	Australian Standard AS 1360, Rotating electrical machines of particular types or for particular applications.
107.	Australian Standard AS 1379 Supp. 1, Specification and supply of concrete - Commentary (Supplement to AS1379).
108.	Australian Standard AS 1379, Specification and supply of concrete.
109.	Australian Standard AS 1384, Transducers for electrical measurement.
110.	Australian Standard AS 1386, Non-laminar flow clean - rooms - class 3500.
111.	Australian Standard AS 1397, Steel sheet and strip - Hot-dipped zinc-coated or aluminium/zinc coated.
112.	Australian Standard AS 1411, Methods for sampling and testing aggregates.
113.	Australian Standard AS 1418, Cranes (including hoists and winches). This Standard has 16 Parts. Part 1 is General requirements.
114.	Australian Standard AS 1429, Electric cables – Polymeric insulated for working voltages 1.9/3.3kV to 19/33kV.
115.	Australian Standard AS 1431, Low voltage switchgear and control-gear – Control circuit devices and switching elements.
116.	Australian Standard AS 1470, Health & safety at work – Principles and practices.
117.	Australian Standard AS 1478, Chemical admixtures for concrete.
118.	Australian Standard AS 1479, Code of Practice for the use of chemical admixtures in concrete.

Reference	
119.	Australian Standard AS 1530.4, Methods for fire tests on building materials, components and structures - Fire-resistance tests of elements of building construction.
120.	Australian Standard AS 1603, Automatic fire detection and alarm systems.
121.	Australian Standard AS 1627, Metal finishing – Preparation and pre-treatment of surfaces: Part 1: Cleansing using liquid solvents and alkaline solutions. Part 4: Abrasive blast cleaning. Part 9: Pictorial surface preparation for painting steel surfaces.
122.	Australian Standard AS 1650, Hot-dipped galvanised coatings on ferrous articles.
123.	Australian Standard AS 1657, Fixed platforms, walkways, stairways and ladders - Design, construction and installation.
124.	Australian Standard AS 1660, Test methods for electrical cables, cords and conductors.
125.	Australian Standard AS 1660.5, Test methods for electric cables, cords and conductors - Fire tests.
126.	Australian Standard AS 1668, The use of mechanical ventilation and air conditioning in buildings: Part 1: Fire and smoke control in multi-compartment buildings. Part 2: Mechanical ventilation for acceptable indoor-air quality. Part 2 (Suppl.): Mechanical ventilation for acceptable indoor-air quality – Commentary (1991).
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End of Section

2.8 TECHNICAL DESIGN METHODS

2.8.1 Neutronic, Thermal-hydraulic and Shielding Design

The neutronic, thermohydraulic and shielding design of the Reactor Facility will be performed with computer codes and programs developed by either INVAP or other internationally recognised organisations. The codes are described in Chapter 5, together with software validation and verification programs.

The design of shielding against ionising radiation is generally performed in two stages:

a) Determination of radiation source parameters, i.e., intensity, activity, isotopic composition, etc.

b) Transport calculations through the proposed shielding and evaluation of the desired response (dose, energy deposition, etc.) at points of interest.

The software used in shielding calculations will include ORIGEN-2, MERCURE (HXS_LAC, HXS_BUF) and MCNP

Chapter 5 presents a description of the methodology and the software used in shielding calculations.

2.8.2 Analysis of Transients

Analysis of transients after PIEs will be performed with PARET-PC (Reactivity Insertion Transients) and RETRAN-02 (Loss of Flow, Loss of Heat Sink and Loss of External Electric Power Transients) and RELAP V (Loss of Coolant). Chapter 16 presents a description of these codes, as well as the validation and verification process.

2.8.3 Mechanical Stress Analysis

Verification of stresses and displacements of the reactor mechanical components will be performed with COSMOS/M software (described in Chapter 4)

This code solves for nodal displacement and element stress on finite elements models. It has an extensive element library, load options and fast solution scheme for handling large problems. The analytical capabilities of this module include mechanical loads, temperature loads, stress stiffening, and buckling analysis. Linear dynamic analysis can also be performed including natural frequencies and mode shapes.

Linear and non-linear analysis can be handled, including: large deflection, large strains, non-conservative loading, plasticity, creep, gap, friction, non-linear buckling, direct time integration analysis.

Structural and seismic analysis of the building and structures will be performed with the ETABS and SAP2000 software (considered in Chapter 4). This analysis will include mechanical loads, thermal loads and natural frequencies and mode shapes.

2.8.4 Design and Analysis of Process Systems

The process systems will be designed with a set of computational tools that allow the calculation of the characteristics of the piping and components used to fulfil the cooling requirements of the core. The software used in the design and analysis of the process systems will include CHEMCAD 5.0.02, Engineers Aid SINET-XLT 5.3 and CRANE Companion 2.50 ABZ. These codes are considered in Chapter 6.

2.8.5 Experimental Testing

Experimental testing of a system or component will be carried out in cases where there is no previous experience and there are no experimental data available from other sources to validate the analytical or numerical analysis.

The correct functioning of a design will be tested with a mock up or prototype. New construction techniques or materials will be verified through mechanical testing.

Each system requiring a prototype or experimental testing will be identified and the methodology described in the corresponding chapter.

End of Section

2.9 DESIGN FOR INTERNAL FIRE PROTECTION

2.9.1 Preventive Protection

2.9.1.1 General

Preventive protection will be aimed at controlling and reducing as much as possible the most frequent causes of fire, *viz*, human errors and equipment or installation failures.

Building layout and zoning, location of fire barriers and the selection of reactor materials will minimise the internal fire hazard.

Systems, structures, and components important to safety will be designed and located to minimise the frequency and consequences of fires and explosions caused by internal and external events.

Structures, systems and components important to safety will also be designed such that the capability for shutdown, residual heat removal and containment of radioactive material is maintained in the event of internally or externally initiated fires and explosions.

In order to minimise human errors, tasks and operations involving equipment and reactor systems will be controlled.

2.9.1.2 Fire Prevention

A fire prevention program will be implemented by trained personnel permanently at hand at the Reactor Facility. Detection and extinguishing equipment will be provided.

An emergency program will be prepared which will include, as a minimum, the following aspects:

a) inspection programs;

b) organisation of fire combat roles;

c) activities and behaviour of individuals when faced with an emergency;

d) sanitary assistance;

e) decontamination procedures.

2.9.1.3 Handling of Combustible Products

Non-flammable products and solvents will be used for cleaning tasks, wherever practicable.

Possession, use, and storage of flammable products will be subject to review. The amount of combustible materials stored will be restricted to no more than the required quantities for plant operation and/or maintenance.

The design of the facility will ensure that flammable compressed gases are only installed outside the reactor building. An exception is the deuterium used in the Cold Neutron Source. Prevention measures will be in place in the CNS design.

2.9.1.4 Electrical Equipment

In view of the central importance of electrical and control equipment to safety system function, adequate preventive measures will be considered during the design, selection and equipment distribution stages.

Fire safety will be a key consideration in the selection of electrical equipment and the electrical protection of cables and electrical/electronic equipment.

Appropriate grounding and lightning protection will be installed in electrical systems.

Cables will be anti-flame, with self-extinguishing sheaths, where appropriate, consistent with applicable codes and standards, and the fire hazard analysis.

The use of electrical equipment and accessories in the Reactor Facility will be controlled.

2.9.1.5 Furnishings

Non-combustible or self-extinguishing furnishings will be used wherever possible. Preference will be given to metallic shelves, cupboards, file cabinets and desks.

2.9.2 Passive Protection

Passive, or structural, protection will be incorporated into the design of the facility to reduce potential fire sources as well as their spreading or propagation. This protection will include the provision of isolation and separation, the selection of appropriate structural and non-structural materials, the consideration of building layout and zoning, the use of fire barriers and consideration of the layout and protection of safety systems. These aspects are discussed in Chapter 10.

Special attention will be paid to emergency exits, and to specifying appropriate escape routes.

Non-combustible, fire retardant and heat resistant materials will be used wherever practicable throughout the facility, particularly in locations such as the MCR and in the Reactor Building.

2.9.3 Detection and Suppression Systems

Fire detection and suppression systems of appropriate capacity and capability will be provided. These are discussed in Chapter 10.

Fire suppression systems will be designed and located to ensure that their spurious or inadvertent operation does not impair the capability of structures, systems and components important to safety to fulfil their safety functions.

2.9.4 Handling Emergencies

The locations within the facility where special arrangements are necessary in relation to the handling of emergencies will be identified. Chapter 20 considers the handling of emergencies.

End of Section

2.10 QUALIFICATION OF COMPONENTS

This section specifies the general requirements and criteria applicable to the qualification of systems, structures and components important to safety. Such qualification is necessary to give confidence that the operation of systems, structures and components claimed as a means of protection or mitigation will occur as envisaged.

Systems, structures and components important to safety will be qualified to ensure that safety functions are able to be achieved for both the operating and environmental conditions under which the equipment or component is required to function.

The design method applied will depend on the safety category of the system (refer Sections 2.5.1.2, 2.5.1.3, 2.5.1.4), its complexity, and whether the proposed design is novel or similar designs are available.

Type tests will be carried out for equipment and components as required by its safety and quality classification.

Safety Category 1 electronic equipment and components will be classified as IEEE Class 1E, and will be qualified to IEEE Class 1E standards.

Safety Category 1 electrical equipment and components within the SPS will use IEEE and associated standards, where relevant as the basis for system design, equipment procurement, installation and system operation with the exception of the diesel generators. The diesel generators will be to high quality commercial standards.

Safety Category 2 electrical and electronic equipment will be qualified in accordance with a specific plan. A qualification procedure will be followed for all Safety Category 2 equipment and components in order to demonstrate the ability of the system, component or structure to perform its required function. This demonstration is achieved by means of verification and validation tests, operational experience and analysis of the systems, components or structures. In addition, seismic verification will be undertaken in accordance with seismic classification.

The performance of hardware and software used in computer-based systems important to safety will be subject to appropriate standards and practices for development, verification and validation.

2.10.1 Prototype Testing

Safety significant components of novel design and safety significant components which are based on a proven design but with significant modifications, have been qualified by building and testing a prototype. The detailed engineering of the component has taken into account the results of the prototype tests.

Designing, manufacturing and testing prototypes are considered part of the detailed design and engineering process and have been undertaken in accordance with the QA system for design.

2.10.1.1 Control Rod Drives

The power regulating control rod drive design will be based on the experience developed by INVAP in designing and constructing CRDs. The design of the CRD for the Reactor Facility is based on the CRD of the ETRR-2 in Egypt, with minor adaptations.

A prototype has been constructed to confirm the design and demonstrate its reliability and availability. The prototype has been cycled to a number of cycles that demonstrate the reliability of the system during its lifetime. Information has been obtained to determine the frequency of maintenance tasks and the extent of surveillance tests.

The CRDs are discussed in Chapter 5.

2.10.1.2 Neutron Beam Shutters

Neutron beam shutters are substantial mechanical components that require precise positioning. The proposed design differs from previous shutters manufactured by INVAP.

Due to its effect on the positioning and alignment of the neutron guides, a prototype shutter has been built and tested.

The prototype has served to adjust the manufacturing processes, optimise the design, and verify the maintainability and reliability figures of the proposed system.

The neutron beam shutters are discussed in Chapter 11.

2.10.1.3 Cold Neutron Source

The metallic vessel containing the cold neutron source is a component customised to the reactor. Prototypes of sections of the source inside the reflector vessel have been manufactured and tested for compliance with specified requirements.

As the final source test can only be done after the Reactor Facility is in operation, the goal of the prototypes (one or more) will be to verify design assumptions and models, calculation models and manufacturing processes.

The CNS is discussed in Chapter 11.

2.10.1.4 Second Shutdown System

The Second Shutdown System shuts down the reactor by dumping the heavy water in the reflector vessel into a storage tank.

A prototype has been built to test the actuation time of the Second Shutdown System and to provide a negative reactivity insertion curve for the safety analysis.

The Second Shutdown System is described in Chapter 5.

2.10.1.5 Fuel Assembly and Fuel Clamp

A prototype of the FA has been built and tested in a hydraulic loop to verify its mechanical behaviour under full flow rate. Several FAs will be irradiated to typical burn up ratios and mechanically tested to verify the effect of irradiation on the mechanical properties. The prototype will serve to adjust the manufacturing process and verify the performance of the FA.

A prototype of the fuel clamp has been built and tested in a hydraulic to loop to verify its behaviour when submitted to the velocities encountered in the Reactor Facility.

FAs and fuel clamps are described in Chapter 5.

2.10.1.6 Pneumatic Conveyor System

The Pneumatic Conveyor System allows the insertion of irradiation targets by moving them through pneumatic transport tubes.

A prototype of the Pneumatic Conveyor System has been built to fine tune the can injection and extraction time, as well as confirm the operational parameters of the system and its reliability.

The Pneumatic Conveyor System is described in Chapter 11.

2.10.1.7 Flap Valves

The flap valves are passive components within the piping inside the reactor pool, that open in the absence of forced circulation. The opening of the flap valves enables a natural circulation flow path to be formed that extracts the decay heat from the core and irradiation rigs respectively.

Prototypes of the flap valves have been built to test their performance and adjust the mechanical parameters to ensure the flap valves fulfil their function.

Flap valves are described in Chapter 6.

End of Section

2.11 CONCLUSIONS

This Chapter has presented the safety objectives and engineering design requirements to which the Reactor Facility is designed. Specifically, the following issues have been covered for the Reactor Facility:

1. The overall safety objectives relating to the whole Reactor Facility.
2. The general design requirements relating to the safety of the whole Reactor Facility, with respect to the following:
   a) Quality assurance
   b) Ensuring a high standard of engineering design including the use of conservative design margins, provision of barriers etc;
   c) The use of inherent, passive and fail-safe design features;
   d) The use of redundancy, diversity and independence in the design of ESFs;
   e) The defence in depth philosophy as applied to the design;
   f) Accident prevention and accident management features, including accident mitigation;
   g) The use of proven engineering practice and the use of accepted codes and standards;
   h) Consideration of human factors and dependent failures; and
   i) The radiation protection principles.
3. Specific design requirements relating to the safety of the Reactor Facility with respect to the following:
   a) Monitoring and control of reactor variables
   b) Reactor core integrity
   c) Protection against flow instabilities and power oscillations
   d) Sharing of common systems, structures or components between facilities
   e) Human factors and ergonomic principles
   f) Reactivity control
   g) Core cooling
   h) Fuel and materials design
   i) Reactor utilisation
   j) Safety system design criteria
   k) Reliability requirements
   l) Equipment qualification design basis
   m) Protection against dependent failures
   n) Surveillance and maintenance of safety equipment
   o) Radiation protection.
4. A listing of the safety functions.

5. A listing of the codes and standards to be applied.
6. The technical design methods to be employed.
7. Design requirements relating to internal fire protection of the Reactor Facility.
8. Requirements relating to the qualification of components important to safety to be used in the Reactor Facility.

End of Section