GUIDE NO. AERB/NPP&RR/SG/RW-8

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GOVERNMENT OF INDIA

AERB SAFETY GUIDE

DECOMMISSIONING OF NUCLEAR POWER PLANTS AND RESEARCH REACTORS

ATOMIC ENERGY REGULATORY BOARD

AERB SAFETY GUIDE NO. AERB/NPP&RR/SG/RW-8

DECOMMISSIONING
OF
NUCLEAR POWER PLANTS
AND
RESEARCH REACTORS

Atomic Energy Regulatory Board
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FOREWORD

Activities concerning establishment and utilisation of nuclear facilities and use of radioactive sources are to be carried out in India in accordance with the provisions of the Atomic Energy Act 1962. In pursuance of the objective of ensuring safety of members of the public and occupational workers, as well as protection of environment, the Atomic Energy Regulatory Board (AERB) has been entrusted with the responsibility of laying down safety standards and enforcing rules and regulations for such activities. The Board has, therefore, undertaken a programme of developing safety standards, safety codes and related guides and manuals for the purpose. While some of these documents cover aspects such as siting, design, construction, operation, quality assurance and decommissioning of nuclear and radiation facilities, other documents cover regulatory aspects of these facilities.

Safety codes and safety standards are formulated on the basis of nationally and internationally accepted safety criteria for design, construction and operation of specific equipment, structures, systems and components of nuclear and radiation facilities. Safety codes establish the safety objectives and set requirements that shall be fulfilled to provide adequate assurance for safety. Safety guides elaborate various requirements and furnish approaches for their implementation. Safety manuals deal with specific topics and contain detailed scientific and technical information on the subject. These documents are prepared by experts in the relevant fields and are extensively reviewed by advisory committees of the Board before they are published. The documents are revised when necessary, in the light of experience and feedback from users as well as new developments in the field.

One of the regulatory requirements is safe management of radioactive waste generated during the operation and decommissioning including site remediation of nuclear fuel cycle facilities. This safety guide addresses the safe decommissioning of nuclear power plants and research reactors. It brings out the need for considering decommissioning aspects right from the design stage and periodic updating at various stages as a part of plant life management. It provides guidance for selection of various decommissioning options, safe management of the activity so as to reduce exposure to occupational worker and public, to minimise the environmental impact and waste generation to simplify the dismantling procedures and management of resulting waste. In drafting this guide, extensive use has been made of the information contained in the relevant documents of the International Atomic Energy Agency issued under Radioactive Waste Safety Standards (RADWASS) Programme and other publications on radioactive waste management.

Consistent with the accepted practice, ‘shall’ and ‘should’ are used in the guide to distinguish between a firm requirement and a desirable option respectively. Annexures and references are included to provide information that might be helpful to the user. Approaches for implementation different to those set out in the guide may be acceptable, if they provide comparable assurance against undue risk to the health and safety of the occupational workers and the general public, and protection of the environment.

For aspects not covered in this safety guide, applicable national and international standards, codes and guides, acceptable to AERB should be followed. Non-radiological aspects such as industrial safety and environmental protection are not explicitly considered in this guide. Industrial safety is to be ensured through compliance with the applicable provisions of the Factories Act, 1948 and the Atomic Energy (Factories) Rules, 1996.

This guide has been prepared by specialists in the field drawn from the Atomic Energy Regulatory Board, Bhabha Atomic Research Centre, Nuclear Power Corporation of India Limited and other consultants. It has been reviewed by the relevant AERB Advisory Committee on codes and guides and the Advisory Committee on Nuclear Safety.

AERB wishes to thank all individuals and organisations who have prepared and reviewed the draft and helped in its finalisation. The list of persons, who have participated in this task, along with their affiliations, is included for information.

(S.K. Sharma)
Chairman, AERB

DEFINITIONS

Accident

An unplanned event resulting in (or having the potential to result in) personal injury or damage to equipment which may or may not cause release of unacceptable quantities of radioactive material or toxic/hazardous chemicals.

Ageing

General process in which characteristics of structures, systems or component gradually change with time or use [although the term 'ageing' is defined in a neutral sense - the changes involved in ageing may have no effect on protection or safety, or could even have a beneficial effect - it is commonly used with a connotation of changes that are (or could be) detrimental to protection or safety, i.e. as a synonym of 'ageing degradation'].

Clearance Levels

A set of values established by the regulatory body and expressed in terms of activity concentrations and/or total activity, at or below which sources of radiation may be released from regulatory control.

Conditioning of Waste

The processes that transform waste into a form suitable for transport and/or storage and/or disposal. These may include converting the waste to another form, enclosing the waste in containers and providing additional packaging.

Confinement

Barrier, which surrounds the main parts of a nuclear facility, carrying radioactive materials and designed to prevent or to mitigate uncontrolled release of radioactivity into the environment during commissioning, operational states, design basis accidents or in decommissioning phase.

Contamination

The presence of radioactive substances in or on a material/the human body or other places in excess of quantities specified by the competent authority.

Core Components

All items other than fuel, which reside in the core of a nuclear power plant and have a bearing on fuel integrity and/or utilisation (e.g. calandria, coolant channels, in-core detectors and reactivity devices).

Decommissioning

The process by which a nuclear or radiation facility is finally taken out of operation in a manner that provides adequate protection to the health and safety of the workers, the public and the evironment.

Decontamination

The removal or reduction of contamination by physical or chemical means.

Decontamination Factor

The ratio of initial level of contamination of radioactive material to residual level achieved through a decontamination process.

Design Life

The period of time for which the item will perform satisfactorily meeting the criteria set forth in the design specification.

Discharge Limits

The limits prescribed by the regulatory body for effluent discharges into atmosphere/ aquatic environment from nuclear/radiation facilities.

Disposal (Radioactive Waste)

The emplacement of waste in a repository without the intention of retrieval or the approved direct discharge of waste into the environment with subsequent dispersion.

Documentation

Recorded or pictorial information describing, defining, specifying, reporting or certifying activities, requirements, procedures or results.

Dose

A measure of the radiation received or absorbed by a target. The quantities termed absorbed dose, organ dose, equivalent dose, effective dose, committed equivalent dose, or committed effective dose are used, depending on the context. The modifying terms are used when they are necessary for defining the quantity of interest.

Emergency

A situation which endangers or is likely to endanger safety of the site personnel, the nuclear/radiation facility or the public and the environment.

Exempt Waste

Waste, which is cleared from regulatory control in accordance with clearance levels. The designation should be in terms of activity concentration and/or total activity and may include a specification of the type, chemical/physical form, mass or volume of waste.

Exemption

The deliberate omission of a practice, or specified sources within a practice, from regulatory control or from some aspects of regulatory control, by the regulatory body on the grounds that the exposures which the practice or sources cause or have the potential to cause are sufficiently low as to be of no regulatory concern.

Fuel Handling

All activities relating to receipt, inspection, storage and loading of unirradiated fuel into the core and unloading of irradiated fuel from the core, its transfer, inspection, storage and dispatch from the nuclear power plant.

Guaranteed Shutdown State (GSS)

A specified shutdown state of the reactor with sufficiently large reactivity shutdown margin, established by the addition of liquid poison into moderator to provide positive assurance that an inadvertent increase in reactivity by withdrawal of all other reactivity devices cannot lead to criticality.

Licence

A type of regulatory consent, granted by the regulatory body for all sources, practices and uses for nuclear facilities involving the nuclear fuel cycle and also certain categories of radiation facilities. It also means authority given by the regulatory body to a person to operate the above said facilities.

Long-lived Wastes

Radioactive wastes containing long-lived radionuclides having sufficient radiotoxicity and/or concentrations requiring long time isolation from the biosphere. The term long-lived radionuclides refers to half lives usually greater than 30 years.

Low and Intermediate Level Waste (LILW)

Radioactive wastes in which the concentration or quantity of radionuclides is above clearance levels established by the regulatory body, but with radionuclide content and thermal power below those of high level waste. Low and intermediate level waste is often separated into short lived and long lived wastes.

Low Level Waste (LLW)

Radioactive waste in which the concentration or quantity of radionuclides is above clearance levels established by the regulatory body but with the radionuclide content below those of intermediate and high level wastes. It does not require shielding during handling and transportation.

Nuclear Power Plant (NPP)

A nuclear reactor or a group of reactors together with all the associated structures, systems, equipment and components necessary for safe generation of electricity.

Plant Management

The members of Site Personnel who have been delegated responsibility and authority by the Responsible Organisation/Operating Organisation for directing work in the plant.

Prescribed Limits

Limits established or accepted by the regulatory body.

Primary Containment

The principal structure of a reactor unit that acts as a pressure retaining barrier, after the fuel cladding and reactor coolant pressure boundary, for controlling the release of radioactive material into the environment. It includes containment structure, its access openings, penetrations and other associated components used to effect isolation of the containment atmosphere.

Quality Assurance

Planned and systematic actions necessary to provide the confidence that an item or service will satisfy given requirements for quality.

Radioactive Waste

Material, whatever its physical form, left over from practices or interventions for which no further use is foreseen: (a) that contains or is contaminated with radioactive substances and has an activity or activity concentration higher than the level for clearance from regulatory requirements, and (b) exposure to which is not excluded from regulatory control.

Radioactive Waste Management Facility

Facility specifically designed to handle, treat, condition, temporarily store or permanently dispose of radioactive wastes.

Records

Documents, which furnish objective evidence of the quality of items and activities affecting quality; they include logging of events and other measurements.

Regulatory Inspection

An examination through review of documents, observation, measurement or test undertaken by or on behalf of the regulatory body during any stage of the regulatory consenting process, to ensure conformance of materials, components, systems and structures as well as operational and maintenance activities, processes, procedures, practices and personnel competence with predetermined requirements.

Repository

A facility where radioactive waste is emplaced for disposal. Future retrieval of waste from the repository is not intended.

Research Reactor

A critical/sub-critical assembly of nuclear fuel elements used for the purpose of research, teaching and production of radioisotopes.

Responsible Organisation

An organisation having overall responsibility for siting, design, construction, commissioning, operation and decommissioning of a facility.

Safety Guide

A document containing detailed guidelines and various procedures/ methodologies to implement the specific parts of a safety code, that are acceptable to the regulatory body, for regulatory review. This is issued under the authority of regulatory body and is of non-mandatory nature.

Secondary Waste

A form and quantity of waste that results as a by-product of the process when applying a waste treatment technology to the initial waste .

Short-lived Waste

Radioactive wastes in quantities and/or concentrations, which will decay to activity levels considered acceptably low from the radiological point of view within the time period during which administrative controls are expected to last. Radionuclides in short-lived wastes will generally have half-lives shorter than 30 years.

Site

The area containing the facility defined by a boundary and under effective control of the facility management.

Source

Anything that causes radiation exposure, either by emitting ionising radiation or releasing radioactive substances or materials.

Surveillance

All planned activities, viz. monitoring, verifying, checking including in-service inspection, functional testing, calibration and performance testing carried out to ensure compliance with specifications established in a facility.

Technical Specifications for Operation

A document approved by the regulatory body, covering the operational limits and conditions, surveillance and administrative control requirements for safe operation of the nuclear or radiation facilities. It is also called 'operational limits and conditions'.

Unrestricted Use

Any release or use of materials, equipment, buildings or site without any restriction imposed by the regulatory body.

Waste Management

All administrative and operational activities involved in the handling, pre-treatment, treatment, conditioning, transportation, storage and disposal of radioactive waste.

Waste Treatment

Operations intended to benefit safety and/or economy by changing the characteristics of the wastes by employing methods such as

(a) volume reduction;

(b) removal of radionuclides; and

(c) change of composition.

After treatment, the waste may or may not be immobilised to achieve an appropriate waste form.

Worker

Any person who works, whether full-time, part-time or temporarily, for an employer and who has recognised rights and duties in relation to occupational radiation protection. (A self-employed person is regarded as having the duties of both an employer and worker)

SPECIAL DEFINITIONS
(Specific for the Present Guide)

Site Remediation

Site remediation during and after decommissioning is the process of transforming the site undergoing decommissioning to an acceptable end state suitable for its intended future use by adopting appropriate remediation techniques leading to a reduction in radiation exposure and an improvement in the environment through the required management of contaminants. It does not imply restoration to a pristine environmental state.

CONTENTS

1. INTRODUCTION

1.1 General

1.1.1 Many nuclear power plants (NPP) and research reactors(RR) will be facing final shutdown in the coming years as they approach the end of plant life. In order to provide a consistent and harmonious approach to the decommissioning of nuclear reactors as well as to incorporate the lessons learnt from the decommissioning activities performed world wide, a need for preparing a decommissioning guide of NPP/RR was recognised.

1.1.2 The objective of decommissioning of a NPP/RR is to retire it safely, economically and with acceptably low risk to both site personnel and general public and to minimise the impact on the environment. It is a complex process involving activities such as radiological characterisation, decontamination, dismantling of equipment and structures, and management of resulting waste. When decommissioned, the site may be released for unrestricted use or for specified alternative use after the required site remediation and as approved by AERB. The time period to achieve this goal may range from a few years to several decades mainly to allow radioactive decay.

The decommissioning strategy can vary from case to case. It could be immediate dismantling, deferred dismantling or entombment. The various factors that should be considered for choosing most appropriate decommissioning strategy are brought out in this safety guide.

Experience of decommissioning of NPP/RR world wide has shown that adequate considerations should be given to decommissioning during the plant design and construction phase. The objectives of including design features to facilitate decommissioning are to reduce personnel exposure, minimise waste generation and other environmental impacts, simplify dismantling procedures and reduce costs. A variety of concepts that require consideration at the design and construction stages to facilitate various decommissioning activities, include decontamination and removal either as a single piece or segments of component/equipment.

1.2 Objective

The objective of this safety guide is to provide guidance to the plant management and regulatory body to ensure that the decommissioning process for NPP/RR is conducted in a safe and environmentally acceptable manner. To meet this objective, guidance in the following areas is provided in this safety guide:

- Suitable provisions in design and construction including plant layout to facilitate dismantling of piping, equipment and structures and the maintenance of the integrity of the required components and structures, till the facility is fully decommissioned

- Radiological characterisation in order to support the decommissioning planning effort, together with the methodology of performing such a characterisation of a shut down reactor
- Establishment of radiological practices and protection criteria
- Selection of the appropriate option, planning and organisation structure for decommissioning
- Proper documentation and reporting, right from design and construction stage till the end of decommissioning and reporting format to regulating body
- Management of decommissioning waste based on stipulated clearance levels.

1.3 Scope

1.3.1 The provisions of this safety guide apply to NPP/RR and their associated facilities. A separate safety guide titled 'Decommissioning of Nuclear Fuel Cycle Facilities other than Nuclear Power Plants' addresses decommissioning of other nuclear fuel cycle facilities. [Refer AERB/ NF/SG/RW-7]

1.3.2 This safety guide primarily addresses the issues related to decommissioning of nuclear reactors after the planned final shutdown. Many of these provisions are also applicable to decommissioning of a plant after a safety significant event that has resulted in serious damage or contamination of the plant. In such a case, this safety guide may be used as a basis for developing special decommissioning provisions with additional considerations as necessary.

1.3.3 Major activities to be carried out during decommissioning are given in Annexure I.

1.3.4 Non-radiological hazards, involving industrial safety aspects, would also arise during decommissioning activities. This safety guide does not address these hazards, but it is important that these be given due consideration during the planning process and in risk analysis. For these requirements, Atomic Energy (Factory) Rules 1996 are to be followed.

1.3.5 This safety guide does not provide detailed information on handling and transportation of irradiated fuel and other radioactive materials. In case radioactive waste is to be transported through public domain before its final disposal then AERB safety code 'Transport of Radioactive material' (AERB Code No SC/TR-1) is to be followed.

2. DECOMMISSIONING DECISION AND OPTIONS

2.1 General

2.1.1 Decision to decommission a NPP/RR is influenced by several factors. Important factors, that should be taken into consideration for the decision are:

- Safety
- Technical/obsolescence aspects
- Major damage to the NPP/RR
- Economical aspects
- Requirement of the site.

2.1.2 Once the decision to decommission a NPP/RR is taken, decommissioning of the facility may be carried out in one continuous operation following the final shutdown, (i.e. immediate decommissioning) or it may be taken up in a series of discrete operations over an extended time period (i.e deferred decommissioning) or encasement of the highly active materials in a structurally sound material (entombment). A number of factors influence in deciding decommissioning options. Major factors, that should be taken into consideration for the selection of options of immediate decommissioning, deferred decommissioning or entombment are:

- Location
- Size and type
- Status of NPP/RR after normal operating phase
- Status of NPP/RR after major damage
- Site characteristics
- Availability of waste management facilities
- Availability of appropriate technologies for decommissioning.

2.2 Major Influencing Factors of Decommissioning Decision

2.2.1 Safety Considerations

Reliability of safety systems of NPP/RR should be maintained, to ensure their safety functions throughout the operation phase of the specified design life. If reliability of the safety systems is impaired due to ageing or it is not meeting the current safety standards, the same should be achieved through refurbishment and safety upgradation for further operation. If this option is not adoptable, the unit should be decommissioned.

2.2.2 Technical/Obsolescence Aspects

Closure of a RR on obsolescence grounds might arise when an alternate RR/facility is available, which offers better technology or has wider applications. In many cases, a RR/proto type NPP is commissioned for a specific programme, at the end of which the plant may be closed down for decommissioning.

2.2.3 Major Damage

In case of a severe accident, if a NPP/RR cannot be rehabilitated for further safe operation, it should be decommissioned. Depending on the nature of damage, special procedures may have to be adopted for carrying out decommissioning.

2.2.4 Economical Aspects

A decision to close the NPP/RR on economic grounds could be taken by the operating organisation if even after achieving all practicable cost saving operations related to NPP/RR, the operating costs are still significantly higher than the benefits accrued from the NPP/RR. In such a case planning for decommissioning should be done.

2.2.5 Requirement of the Site

In view of the considerable time, technical efforts and resources spent on selecting nuclear sites, they represent a valuable resource in terms of low seismic activity, availability of adequate cooling water, access to major transportation routes, desirable demographic patterns and above all public acceptance through environmental impact analysis. As a result, an operating organisation may consider the merits of replacement of RR/NPP at the same site. It may even be economically viable to dismantle the NPP/RR completely with a shorter cooling/decay period to make room for a replacement plant.

In multi-unit sites, personnel at the operating NPP/RR may be available, during safe enclosure period for continuing maintenance, surveillance and security for the initial radioactive decay, dismantling and/or decontamination stages, at relatively low incremental costs.

2.3 Selection of a Decommissioning Option

Three main decommissioning options are immediate dismantling, deferred dismantling and entombment. Selection of appropriate decommissioning option is influenced by the factors given below:

2.3.1 Location

Decommissioning option would also depend upon whether a site is comprising of a single unit or multi-units.

(a) In case of decommissioning of NPP/RR at a single-unit location, dedicated personnel should be provided for monitoring and surveillance during extended periods of decommissioning.

(b) In case of multi-unit location, the required technical support should be provided by the operating unit(s) at the location during extended periods of decommissioning.

2.3.2 Size and Type

Size and type of NPP/RR should be taken into consideration for estimation of the type and amount of radioactivity likely to be generated for decommissioning waste management. Decommissioning activities and safety considerations for NPP would vary depending on operating history and design such as Boiling Water Reactor (BWR), Pressurised Heavy Water Reactor (PHWR), Pressurised Water Reactor(PWR) and Fast Breeder Reactor (FBR) and, accordingly, decommissioning options would be different. Similarly for RRs such as heavy water tank type, light water pool type and other demonstration/test reactors; decommissioning options would be different. Additionally, in case of RR, decommissioning of various research and isotope handling facilities such as cold and hot neutron facilities, neutron collimators, pneumatic carrier facility for neutron activation analysis, in-pile engineering test loop and thermal column should be given due consideration in the selection of appropriate decommissioning options.

2.3.3 Status of NPP/RR after Normal Operating Phase

Status of NPP/RR with respect to mechanical stability of various structures, systems and components should be evaluated to decide prompt or deferred decommissioning option taking into account the following factors:

- Ageing as a result of irradiation damage along with thermal and chemical environment and the time-dependent degradation of material
- Change in external forces due to floods, high winds, rainfall and other natural events of large magnitude than considered in design, occurring during the extended period of radioactive decay
- Change in loads arising out of the decommissioning operation (e.g. cutting, demolition etc.) and their sequence
- Residual resistance to seismic activity or any other external events such as severe cyclones, floods etc
- Condition of ventilation system and integrity of confinement.

2.3.4 Status of a NPP/RR after Major Damage

In case of a NPP/RR, which is to be decommissioned following major damage or severe accident, the radiological and structural status of NPP/RR should be evaluated to decide the option of prompt or deferred decommissioning. Need for large scale decontamination, provision of temporary radioactive confinement, etc. should be taken into consideration before final decommissioning.

2.3.5 Site Characteristics

The following site characteristics, as updated at the time of decommissioning, should be taken into account:

- Local topography for continued availability of access to heavy transportation means.
- Geological conditions of site such as seismic wave propagation, soil characteristics, water table and its seasonal fluctuations etc.
- Local demography and ecology, projected population distribution, land use, infrastructure available in the area etc.
- Interface with other plants at the site.

2.3.6 Availability of Waste Management Facilities

Taking into account the estimated radioactive waste likely to be generated during decommissioning, requirement of on-site additional waste treatment and disposal facilities be evaluated and, if required, should be made available.

2.3.7 Availability of Technologies for Decommissioning.

Decommissioning option to be selected should be based on availability of the latest equipment, techniques and/or facilities resulting in efficient and/or better means of dismantling or one piece removal of major activated and/or contaminated equipment, thereby, saving cost and/or radiation doses. Improvement in remotisation of such equipment and techniques should also be considered to ensure that the chosen techniques of dismantling and decontamination aids in significantly improving the industrial and radiological safety of workers.

3. REGULATORY ASPECTS

3.1 General

This section highlights the planning issues of decommissioning and its preparation during various stages of NPP/RR starting from site selection and design stage. This will not only minimise safety issues during decommissioning process but also reduce transition time between final shutdown and placement of the facility in a safe and stable condition preparatory to safe enclosure and/or dismantling. It also gives regulatory requirements as applicable during the decommissioning period.

3.2 Regulatory Framework

3.2.1 The feasibility of safe decommissioning and subsequent site remediation of NPP/RR should be considered during site selection, design and construction stages of consenting process. Later, during operation phase, the conceptual plan of decommissioning (refer Annexure IIA) should be periodically updated and a detailed decommissioning plan should be submitted for approval by AERB before final shutdown. The necessary submissions in this regard for review by the regulatory body at the relevant stages are:

(a) During site selection, design and construction phases; provisions to facilitate decommissioning shall be identified in the safety analysis report of NPP/RR and the conceptual decommissioning plan shall be submitted to AERB to show that decommissioning of the facility can be carried out safely at a future date.

(b) During operational phase of NPP/RR, the above decommissioning plan should be suitably modified on the basis of operational experience and occurrences and shall be submitted to the regulatory body for review during periodic safety assessment.

(c) At the end of plant design life, five years before expiry of operating license, the operating organisation/responsible organisation shall carry out a complete review on the basis of which it may seek from AERB an extension of the operating authorisation beyond the original design life or submit an application for decommissioning containing the detailed plan supported by safety assessment relevant to decommissioning for review and approval. The decommissioning plan may require changes or further refinements as decommissioning proceeds, for which specific regulatory approval should be taken as the case may be. The detailed plan should cover, amongst others, the following:

    (i) Justification of the decommissioning option chosen viz. immediate or deferred or entombment decommissioning,

(ii) Identification of decommissioning organisation and management aspects,

(iii) Decommissioning activities,

(iv) Applicable quality assurance programme,

(v) Safety assessment for each phase of decommissioning and job hazard analysis (JHA) for each major activity,

(vi) Radiological protection programme and collective dose budgeting,

(vii) Emergency preparedness,

(viii) Radioactive waste management,

(ix) Site remediation,

(x) Proposed final radiation survey plan, and

(xi) Impact on environment.

A typical decommissioning plan is given in Annexure II B.

For facilities, which are already in operation, a conceptual decommissioning plan (refer Annexure-IIA) shall be prepared and submitted to AERB.

3.2.2 In case the responsible organisation decides to shutdown a NPP/RR without pre-planning, the plant should be brought to a guaranteed shutdown state (GSS) and the decommissioning plan should be prepared and submitted to AERB. Decommissioning shall not be started until the plan has been approved by AERB.

3.2.3 Decommissioning may be carried out in a sequence of operations separated by one or more periods of time (i.e phased decommissioning). Some of these periods (i.e decommissioning phases) may be dormant under safe enclosure. In such cases of multiple decommissioning phases, the operating organisation should submit the appropriate 'Technical Specifications' for each phase of decommissioning to the regulatory body for approval. In addition, procedural details on the following should also be submitted:

(a) The proposed surveillance and maintenance programme for the safety related structures, systems and components (SSC) during the entire decommissioning period.

(b) Existing or new systems or programmes necessary for maintaining the installation under proper control, such as engineered barriers, ventilation, drainage and environmental/safety monitoring. Ventilation requirements would be changing as per activity in progress. Arrangements shall be made to provide suction over the area which is source of increasing air activity and to lead the sucked air to ventilation duct so as to minimise spread of air activity.

(c) Systems to be installed or replaced to carry out deferred dismantling.

(d) The proposed frequency at which the above items would be reviewed.

(e) The number of staff needed during the period of deferment and their qualifications.

3.2.4 The regulatory control of decommissioning may be done by a single overall license, or by separate licenses (for each phase), whichever is considered to be the most appropriate in the circumstances.

3.2.5 The plant management should report to the regulatory body on a scheduled basis, as stipulated in the regulatory control mechanism e.g., licence, all safety related information e.g. monitoring systems data, radiological surveys. In case of any significant event, the plant management shall report, in a timely manner as stipulated, those data that are necessary to evaluate safety during such events.

3.2.6 Plant management should maintain the relevant documents and records of decommissioning for an agreed period to a specified quality and easily retrievable for guidance and verification.

3.2.7 The plant management should make the necessary documents and co-ordinate with AERB inspection team during their visit to the facility for inspections with respect to the following:

(i) Status of decommissioning activities

(ii) Health and safety aspects of decontamination/decommissioning activities

(iii) Availability of trained manpower

(iv) Availability of normal and emergency procedures

(v) Availability and health of monitoring instrumentation

(vi) Security aspects

(vii) Waste management aspects.

3.2.8 On completion of decommissioning, a final decommissioning report, including final confirmation survey, should be prepared and submitted to AERB. This report should include:

(a) deviations from planned activity along with necessary constraints encountered,

(b) audit report highlighting compliance with the regulatory stipulations,

(c) other statutory requirements if any, and compliance,

(d) quality and acceptance criteria and non-conformance if any,

(e) individual and collective dose to occupational workers,

(f) inventory of radioactive materials, including amounts and types of waste generated during decommissioning and their locations for storage and/or disposal,

(g) materials released for re-use and clearance levels used,

(h) details of site remediation work carried out after decommissioning,

(i) use of decommissioned site with restrictions if any,

(j) 'fit for purpose plan' to assure maintenance of safe conditions till the site is declared safe for unrestricted use,

(k) structures, areas or equipment designated for restricted use, and

(l) lessons learned.

3.3 Planning and Life Management

3.3.1 Requirement of future decommissioning of the plant after operational phase should be considered in the design and during operation phases. However, many of the NPP/RR have been operating for many years, and decommissioning may not have been adequately considered at the design stage. The planning of decommissioning for such installations should start as early as possible.

3.3.2 Decommissioning should be facilitated by planning and preparatory work undertaken during the entire lifetime of the nuclear installation. These actions are intended to minimise the eventual occupational and environmental impacts, which may occur during the active and passive processes during decommissioning and subsequent site remediation work.

3.4 Safety

3.4.1 Decommissioning of nuclear reactors invariably involves the generation of large quantities of radioactive and non radioactive materials. In the course of decommissioning, waste is generated in forms that are different from materials and wastes of the types routinely handled during the operational phase of a NPP/ RR. Subject to safety considerations, generation of radioactive waste should be kept down to the minimum practicable. Appropriate decontamination and dismantling techniques and the reuse or recycling of materials to reduce the waste inventory should be put in place. Plant management should ensure that the required means are available to manage the waste safely.

3.4.2 Decommissioning activities such as decontamination, cutting and handling of large equipment and the progressive dismantling or removal of some existing safety systems or shielding have the potential for creating new hazards. The safety impacts of the decommissioning activities should be assessed and managed so that these hazards are minimised. Safety assessment of the hazards involved during decommissioning (including accident analysis, where necessary) should be conducted to define protective measures, as part of defence in depth that takes into account the specifics of decommissioning. In some cases, such measures may be different from those in place during the operation of NPP/RR.

3.4.3 Decommissioning of NPP/RR involves the removal, at an early stage, of significant quantities of radioactive material, including fuel and operational waste. Even after this step, the total activity due to contamination and activation is significant and has to be taken into account in the safety assessment. Integrity of spent fuel, when still stored on-site in a fuel pool, should be considered and maintained. Fire safety measures for the complete site should be included in the decommissioning plan.

3.4.4 During decommissioning of NPP/RR, radioactive and non-radioactive streams may be discharged to the environment. These discharges should be controlled in compliance with the appropriate regulations of AERB as well as other applicable statutory authorities.

3.4.5 Plant management should establish and maintain emergency planning commensurate with the hazards associated with the pre-disposal management of radioactive waste and the decommissioning activities and should report safety significant events to AERB in a timely manner.

3.4.6 Plant management should identify an acceptable destination for the radioactive waste and shall ensure that radioactive waste is transported safely and in accordance with transport requirements specified in safety guide on 'Standards of Safety in Transport of Radioactive Material' (AERB/SG/TR-2). When it is proposed to store radioactive waste or to defer decommissioning for an extended period of time, consideration shall be given to the principle that radioactive waste shall be managed in such a way that ensures protection of human health and environment and does not cause undue burden on future generations.

4. DESIGN PROVISIONS FOR FACILITATING DECOMMISSIONING

4.1 General

4.1.1 Design provisions for facilitating decommissioning aim at reducing individual and collective dose during decommissioning, minimising waste generation, simplifying dismantling operation and reducing decommissioning costs. However, the considerations from the requirements of decommissioning should not be in conflict with the primary objectives namely safe and reliable operation of NPP/RR.

4.1.2 Design provisions for facilitating maintenance and in-service inspection are generally also beneficial for the decommissioning activities. Some additional features may be incorporated specifically to facilitate decommissioning. Any extra feature aimed purely at decommissioning should be evaluated against its maintainability over long period of time and it should be ensured that it does not have unacceptable conflicting effects. For example, any extra material handling equipment may reduce access spaces and may result in additional waste which may be radioactive.

4.1.3 The design features to facilitate decommissioning can be classified into those aimed at reducing the radiation source, volume of radioactive waste, decommissioning time and enhancing ease of dismantling. Many features aimed at reducing the radiation exposure may also shorten decommissioning time and also result in substantial reduction of radioactive waste.

4.2 Reduction of Radiation Sources and Spread of Contamination

4.2.1 Components subjected to neutron radiation get activated during service. Long-lived isotopes formed in this manner are the major sources of radiation fields during decommissioning. Proper selection of materials will minimise the inventory of such long-lived isotopes resulting in reduction of radiation fields both during operation and decommissioning.

4.2.1 Design of reactor core, core supports and other reactor vault components should ensure that just the minimum components satisfying all functional requirements (This includes all safety, operability and redundancy requirements) are present in this region. Bare minimum items, without sacrificing operational safety, should be located near the reactor core so as to reduce burden of radioactive waste during decommissioning.

4.2.2 Material selection for any component is mainly governed by functional requirements such as strength, corrosion resistance, neutron economy, etc. Attention should be given to impurity contents, which form isotopes contributing significantly to radiation fields during decommissioning. Thus, to aid prompt decommissioning, Cobalt impurities should be minimised and for delayed decommissioning impurity contents of 94Nb and 104Ag should be minimised. Use of materials like stellites, which have high cobalt content, should be avoided wherever possible and alternative materials like colmonoy could be used.

4.2.3 Corrosion and erosion products of the components of various reactor systems get transported to core region where they get activated and finally deposit in low velocity zones (dead ends). These contribute significantly to radiation fields during normal maintenance as well as during decommissioning. Contribution of impurities in materials used in primary coolant system and other reactor systems in enhancing radiation field should also be studied and harmful impurities be controlled.

4.2.4 Water chemistry of primary coolant and moderator systems should be controlled so as to minimise generation of corrosion products.

4.2.5 Filtration and purification circuits should be provided in primary system so as to remove erosion and corrosion products. Condensate clean up system plays a major role in BWR water chemistry.

4.2.6 Inner surfaces of components of primary systems should be suitably treated, e.g. passivation of stainless steel surfaces or providing magenetite coating by hot conditioning for carbon steel feeder pipes, so as to make them corrosion resistant and amenable for decontamination.

4.2.7 Provisions should be made for carrying out periodic decontamination of external and internal surfaces so as to reduce the radiation sources due to deposits on these surfaces.

4.2.8 Surfaces of floors, sumps, etc., which are expected to come in contact with contamination due to spillage etc., should be easily decontaminable. This may be achieved by providing suitable paints, coatings or metallic liners.

4.2.9 Fuel assemblies should be designed and manufactured for high reliability. Provisions should exist in primary system for early detection and removal of failed fuel.

4.2.10 Provision of neutron shielding around the core reduces activation of the surrounding components and structures. Incorporation of some neutron aborbing material between core and vessel, between vessel and concrete can reduce activation of these components. However, such material must be capable of being inserted into the space and not cause difficulties during dismantling.

4.2.11 Effectiveness of biological shields may be enhanced by increasing hydrogen content, for example by using aggregates with higher water content or adding boron (mostly in non soluble form) in concrete mix. While doing so, the effects of such design changes on the basic requirements should be studied.

4.2.12 Use of steel bars, pre-stressing tendons or steel punching in biological shields, especially near regions facing neutron flux, should be minimised to the extent possible.

4.2.13 Piping layout should be carefully designed to avoid dead ends as they tend to accumulate crud and result in hot spots. Appropriate slope should be provided to ensure complete draining.

4.2.14 Design should avoid laying of sub-soil pipe lines carrying radioactive fluids. If such provisions become necessary, appropriate secondary containment in the form of pipe-in- pipe or leak proof concrete trenches should be provided with leakage detection facility.

4.3 Provisions for Ease of Decontamination and Reduction of Radioactive Waste

4.3.1 Accumulation of radioactive contaminants in different locations of the nuclear facility can give rise to special requirements during decommissioning. A reduction in such sources can be achieved by minimising the formation of activation products, by reducing the spread of fission products, and by good house-keeping during operation to avoid the accumulation of contaminants resulting from normal operation as well as from anticipated occurrences. The effectiveness of these provisions can be enhanced if accompanied by routine decontamination during operation.

Therefore, design features which could minimise contamination and/or facilitate decontamination, for example the availability of connections for a mobile decontamination unit or an effective drainage system, should be considered in the design phase.

4.3.2 Certain reactor systems such as moderator system, end shield cooling system are likely to develop high internal contamination/hot spots during operation but generally are not decontaminated during operation. Provisions should be made for decontaminating such systems to reduce activity levels prior to their decommissioning.

4.3.3 Surfaces, which are likely to come in contact with radioactive fluids, should be provided with decontaminable finish. Smooth, coated or lined concrete surfaces are preferable so that simple non-destructive method such as washing and stripping can be used instead of blasting, chipping or grinding. Strippable coatings aqueous/solvent based paints should be applied for such applications.

Other techniques including the pre-treatment of metallic surfaces to reduce contamination may be applied e.g. electro-polishing. This reduces surface roughness thereby reducing the propensity of oxide films forming on such surfaces to absorb radioactive contaminants.

4.3.4 For areas where spillage of radioactive fluids can occur, additional containment features, such as curbs, should be provided. Floors inside such curbs should be provided with suitable drainage to sumps. These floors should be provided with slope towards such drainage locations.

4.4 Facilitating Dismantlement

4.4.1 Design provisions for facilitating dismantlement are component specific and also depend upon the method proposed for handling or dismantling. One should also carry out cost-benefit analysis.

4.4.2 Components, which are expected to be replaced during service life, would inherently have design features facilitating their removal. The same approach may be considered for design of other components, which are to be dismantled during decommissioning.

4.4.3 Provision of accessibility for various decommissioning operations should be made in the layout design of plants. Removable panels and hatch blocks can be provided to have easy but controlled access.

4.4.4 Temporary routes used in construction stage should not be closed in irreversible fashion. These routes could be effectively used for dismantling equipment and removing cut segments.

4.4.5 For components proposed to be removed in 'intact mode' i.e. one piece removal, features such as lifting lugs, trunions, attachments for slings etc., may be provided.

4.4.6 Various dismantling aides can be pre-placed at the initial stage itself. These may include tracks used for guiding remotely controlled equipment, embedded parts in ceilings and walls for handling equipment and temporary supports, provisions for viewing, inspections, radiation monitoring etc.

4.4.7 Connectors, fasteners and hold down devices should be easily removable. Bolted connections should preferably be used instead of welded joints.

4.4.8 Thermal insulation should be easily detachable. Insulation material should be non-absorbing for spilled active liquids.

4.4.9 Design of shielding should be such that dismantling is made easy. Modular construction of interconnected blocks could be used wherever possible. Composite shield made of shielding material sandwiched between two steel plates is another alternative. Biological shields can be constructed in two layers so that active (inner) and non-active (outer) parts can be separated easily. However, the above mentioned changes may be adopted only if they do not adversely affect shielding and structural integrity requirements.

4.4.10 If any design feature, required from decommissioning considerations only, needs inspection or maintenance in radiation area during the operating phase of the plant, then the decision for implementation of such features should be arrived at after considering its pros and cons.

4.5 Design Features Influencing Decommissioning Options

Design features specific to decommissioning options are as follows:

4.5.2 For Immediate Decommissioning :

(a) For this option, the amount of radioactivity handled is large and suitable industrial infrastructure should be available for undertaking such a job.

(b) Infrastructure for large-scale disposal of radioactive waste should be available. Provisions for interim storage of waste should also be made at the site.

(c) Intact removal of core components would not be possible and sophisticated remotely operated cutting equipment would be necessary.

(d) Use of cobalt containing alloys should be avoided. Cobalt impurity content should also be kept as low as possible.

4.5.3 For Deferred Decommissioning :

(a) Under this option, after final shutdown, the plant is preserved in a safe enclosure till it is taken up for decommissioning. Existing plant structures such as containment or other building would be acting as safe enclosures.

Design life of structures, which would be functioning as safe enclosures should cover the proposed safe enclosure period with applicable loading. Means for controlling access to safe enclosures should be provided.

(b) Systems required during the period of safe enclosure are fire detection and suppression, ventilation, lighting, radiological and environmental monitoring, compressed air, drainage and waste treatment etc. Suitable power supplies for these systems would also be required.

(c) If safe enclosure period is more than 50 years, radio-activity due to cobalt would reduce considerably.

4.5.4 For Entombment :

(a) This option is selected when there is no further need of the site where plant is located.

(b) Under this option, highly active/contaminated materials are kept together in small area and encased in a structurally sound material.

(c) Surveillance is to be carried out for the entombment structure until radioactivity decays to a level permitting unrestricted release of site. However, based on radiological conditions, buildings around entombment structure may be used.

5. CHARACTERISATION

5.1 General

5.1.1 For proper planning and implementation of decommissioning of NPP/RR, knowledge of radiation field and contamination levels in the structures, systems and components (SSCs) is necessary. For this, quantitative and qualitative assessment of radionuclides contributing to residual activity in various SSCs should be carried out. The characterisation should provide a reliable database of information on the quantity and type of radionuclides, their distribution and their physical and chemical states.

5.1.2 The quantity and distribution of the radionuclides depends on the type and size of reactor, its operational history, the material of constructions used, corrosion rate and waste management practices adopted during the operational phase of the reactor.

5.2 Objectives

5.2.1 The prime objective of characterisation should be to arrive at appropriate options of decommissioning and their consequences, considering

(i) decontamination process, dismantling procedures (remote, immediate, deferred) and tools required,

(ii) radiological protection of workers, public and the environment,

(iii) waste classification, and

(iv) cost of decommissioning.

5.3 Data Generation and Preparation of Characterisation Report

5.3.1 For the evaluation of decommissioning strategy including preferred decommissioning procedure, manpower estimation and radiation protection criteria, adequate and appropriate data pertaining to waste management, cost evaluation and risk assessment should be generated by detailed calculation on induced radioactivity and by measurements.

5.3.2 A survey of radiological and non-radiological hazards is required for the safety assessment in decommissioning process and should be conducted to identify the inventory and location of radioactive and other hazardous materials. In planning and implementing surveys in a characterisation programme, survey of existing records and operating history should be made use of.

5.3.3 A characterisation report should be prepared which documents the information and data obtained during the characterisation process. The report should be retained as a part of the official record of the installation. The data collected should be updated as the decommissioning process progresses and the database generated should be amenable for revisions.

5.3.4 Process and methodology of characterisation is given in Annexure-III. In-situ measurement techniques for radiological characterisation of SSCs are given in Annexure-IV.

6. DECOMMISSIONING PROCESS

6.1 General

The process of decommissioning begins after the final shutdown of the reactor and ends with the release of site for reuse by the responsible organisation or for unrestricted use by the public. The removal of irradiated fuel from the reactor and other active process fluids such as coolant and moderator is one of the initial activities in decommissioning. Other major decommissioning activities are decontamination, disassembly/dismantlement (conversion of chemically active coolant into stable compound, in case of FBR) and waste management. The thought given to these major activities at design stage will identify the need for provisions of necessary facilities, which will be of help at the time of decommissioning. Methods followed for decontamination and disassembly, sources of wastes and techniques adopted for reduction/ conditioning that are appropriate for liquid and solid radioactive wastes resulting from decommissioning are discussed in succeeding paragraphs. The factors involved in selection of proper disposal facilities for various types of waste are also discussed.

6.2 Defuelling

6.2.1 The removal of irradiated fuel from the reactor at the end of its operational lifetime should preferably be as part of operations or as one of the initial activities in decommissioning. The procedures used for removal, storage and shipment of fuel would be expected to be same as those used during normal operation. 'Technical Specifications for Operations' would be applicable till complete removal of fuel from the core. While the fuel remains stored in operating island, it should be stored in such a way as to control any risk to the public and to the site personnel.

6.2.2 If a stock of fresh fuel is present at the time of final shutdown, necessary arrangements should be made for its safe and secure management.

6.2.3 In case on-site interim storage facilities for spent fuel are utilised, care should be taken with respect to possible interference with future decommissioning activities of the reactor.

6.2.4 Subsequent to removal of fuel (including blanket, in case of FBR) from the reactor, other active process fluids such as moderator and coolant systems may be removed after the structural cooling requirements are met. In case they are aiding shielding of other active components, the same should be retained till adequate radioactive decay is achieved.

6.2.5 After removal of fuel from the core the ‘Technical Specifications for Operations’ should be suitably modified as applicable for decommissioning and got approved from AERB.

6.3 Decontamination

6.3.1 The term decontamination covers the broad range of activities directed to the removal or reduction of radioactive contamination in or on materials, structures and equipment at a NPP/RR. Decommissioning of a reactor is normally preceded by partial or total decontamination. The process of decontamination associated with decommissioning may be conducted before, during or after dismantling.

6.3.2 The main objectives of decontamination include:

(a) Reduction of exposures during decommissioning activities. (This will make it feasible to use simpler techniques for dismantling rather than expensive use of high technology equipment like robot/remote manipulators),

(b) To minimise the potential of spread of contamination, and

(c) To reduce the contamination of components or structure to such low levels that they may be unconditionally released for recycling or reuse.

6.3.3 While selecting decontamination techniques, the following requirements must be considered :

(a) Efficiency :

Method should have high decontamination factor.

(b) Safety :

Application of method should not cause radiation hazard due to external/internal contamination of workers.

(c) Waste minimisation :

Method should not give rise to large quantities of secondary waste.

(d) Ease of application :

Method or technique used should not be labour intensive, difficult to handle or difficult to automate. It should not have non-radiological hazards (e.g. toxicity of solvents).

6.3.4 Over the years, several decontamination techniques have been developed for decontaminating nuclear installations for operation and maintenance purpose. In maintenance work, decontamination is done in a manner to avoid any damage to structural materials whereas in decommissioning, the main purpose of decontamination is to remove as much activity as possible and substantial reduction of thickness of materials may be acceptable. Hence in decontamination for decommissioning, aggressive decontamination methods are permitted provided that the secondary wastes thus generated are manageable.

Typical decontamination techniques that could be used during decommissioning are given in Annexure-V.

6.4 Containment Maintenance and Access Control

6.4.1 Containment systems should be retained as long as necessary as it was during operation. Its removal or modification should be with the approval of AERB. Containment systems may require modification during decommissioning as radioactive materials (viz. spent fuel, active fluids and other operational waste) are removed from the plant. In case certain containment barriers are removed or altered in the course of dismantling, acceptable confinement of residual radioactive material should be planned and implemented. Similarly, adequeate containment should be planned and demonstrated when cutting and dismantling operations are carried out which may give rise to airborne contamination.

6.4.2 In the case of deferred dismantling, structures and systems may have to perform for longer periods than their accepted design life. Hence the containment building should be kept in a state appropriate to the potential hazard and the atmosphere inside the building should be kept under control.

6.4.3 Access control should be enforced. Access into the facility should be subjected to monitoring and surveillance procedures.

6.5 Dismantling

6.5.1 There are many dismantling techniques such as thermal, machanical, water jet cutting etc., available for reactor decommissioning. Each technique has certain advantages and disadvantages. Hence, after proper evaluations, appropriate technique should be selected. Selection of methods and techniques to be used in safe dismantling should take into account aspects such as:

(a) The types and characteristics of materials, equipment and systems to be dismantled,

(b) The availability of proven equipment and suitable adoption of the same for a particular task,

(c) The associated radiation hazard,

(d) The environmental conditions of the workplace,

(e) The deleterious effects on nearby structures it may cause,

(f) The radioactive waste and air particulate produced during dismantling,

(g) The non-radioactive waste produced, and

(h) Availability of on-site waste disposal facility or other sites needing waste transportation for long distances in public domain and the packaging required.

6.5.2 Each dismantling task should be analysed to determine the most effective and safe method for its performance. Some considerations are:

(a) equipment to be simple to operate, decontaminable and maintainable,

(b) effective methods for controlling easily airborne radionuclides (provision of local air clean-up system),

(c) effective control of discharges to the environment,

(d) effect of each task on adjacent systems and structures and on other works in progress, and

(e) waste containers, handling systems and routes for transportation to storage/disposal sites.

6.6 Waste Management

6.6.1 Decommissioning of NPP/RR invariably involves generation of large amount of radioactive waste including some of the waste that is different in characteristics from the normal operational wastes. Decommissioning wastes can be broadly of two types, namely primary waste and secondary waste. Primary waste consist of highly radioactive components from the reactor structure region and moderately active reactor systems. Secondary waste are generated during decommissioning work (e.g. active liquids during decontamination, spent resins, filters etc.). A waste management plan, which is an important part of overall decommissioning plan, should consider the different categories of waste produced and aim at reducing its quantum and safe management. It should also indicate whether existing waste management systems are capable of coping with anticipated decommissioning waste, if not, based on the specific requirement, availability of new facilities should be established before start of decommissioning. The waste disposal facility could be located on-site or could be off-site. In case of off-site disposal facility the need for transporting the radioactive waste should be adequately factored into the planning process. The following constitute essential elements of such a plan:

(a) The origin, amount, category and nature of the different types of wastes likely to be generated during decommissioning and their potential impact on the workers, public and environment

(b) Criteria for segregation of waste materials and eventual clearance for withdrawal of regulatory control with a view to unrestricted release without any radiological concern

(c) Waste minimisation practices and strategies for possible reuse and recycle of materials, equipments and premises

(d) Procedures for monitoring radioactivity particularly for the materials to be reused and the relevant analytical aspects

(e) Details regarding proposed treatment, conditioning, packaging, transportation, storage and disposal modes

(f) Industrial safety and chemical hazards associated with activities of decontamination, dismantling and waste management.

6.6.2 Significant reductions in volumes of radioactive waste should be achieved through decontamination programmes, appropriate dismantling and volume reduction/conditioning techniques, contamination control, and suitable radiological and administrative control measures. Reuse and recycle strategies coupled with release of very low active material cleared from regulatory control can substantially reduce the amount of solid waste material.

6.6.3 The radiation exposure to workers and the public may vary according to waste minimisation strategy. A proper optimisation should be arrived at in regard to minimisation of waste and radiation exposures.

6.6.4 Scrap material can be released without any restriction if the total activity or concentration of the radionuclide present does not exceed the values specified by AERB.

6.6.5 For guidance on predisposal management of low and intermediate level radioactive waste refer AERB safety guide 'Predisposal Management of Low and Intermediate Level Waste' (AERB/NRF/SG/RW-2) and for predisposal management of high level waste (HLW) refer AERB safety guide 'Predisposal Management of High Level Radioactive Waste' (AERB/NRF/SG/RW-3).

6.7 Emergency Preparedness

During the period of decommissioning, various contingency measures such as emergency operating procedures (EOPs) pertaining to the activity to be executed should be available before start of the work. List of EOPs for decommissioning should be available before commencement of decommissioning. Operating personnel should be trained in executing these procedures. In case, fuel is stored in the operating island then measures should be incorporated into emergency planning in order to deal with accidents or incidents involving fuel.

In case of any occurrence of significant event the same should be evaluated through root-cause analysis and effective corrective measures should be implemented to prevent re-occurrence.

7. RADIOLOGICAL SAFETY

7.1 General

7.1.1 The area of radiation protection in decommissioning operations needs special attention to cater to a sizable number of occupational workers involved in decontamination and dismantlement of highly radioactive components and handling of large quantities of radioactive waste. Therefore, to achieve an effective radiological protection programme, good practices have to be clearly documented and practised.

7.2 Radiological Safety Considerations

7.2.1 The following health and safety aspects should be considered during decommissioning process :

(a) Information regarding the decommissioning preparation, including works planning and collective and individual dose estimates

(b) The operational radiation protection procedures to enforce dose limits, dose constraint and dose budgeting

(c) Precautions to be observed and monitoring to be carried out for preventing spread of contamination

(d) Methodology for operational monitoring of external and internal exposures and management of radiological zoning and radiation protection training to occupational workers including contract labourers/temporary workers

(e) Radiation protection measures considered, for implementing ALARA taking into account social and economic conditions

(f) Radiological safety taking into account the status of original containment barriers

(g) Emergency planning and preparedness to cope up with radiological safety significant events

(h) Safety of occupational workers, members of public and environment due to routine or accidental release of radioactivity

(i) Hazards to workers and the public during shipment of radioactive materials to disposal site.

7.3 Radiation Protection Procedures

7.3.1 For an overall optimisation programme of radiation protection measures, the features that will result in reduction of source inventory such as de-fuelling, draining of coolant and moderator system after the mandatory cooling period and early treatment of operational waste, after the reactor is shut down, should be adopted. Also, minimising the number of decommissioning personnel to ensure collective dose reduction should be adhered to. Apart from the routine personnel exposure control and contamination-monitoring programme, the radiological protection programme also should cater to special situations, which may arise during the course of decommissioning, and to safety significant events.

7.3.2 All decommissioning work should be planned and carried out by following special work procedures. The radiation protection personnel should establish a clear liaison with the plant management and regulators.

7.3.3 With respect to the need for the radiation protection procedure, the following factors should be considered :

(a) Use of shielding and protective equipment such as lead aprons, polythene suits and respirators etc. to limit the internal and external exposures and to minimise doses

(b) Use of alarming dosimeters especially while working in areas having streaming fields or hot spots

(c) Having the appropriate number of experienced, trained and qualified radiation protection personnel in ensuring the safe conduct of decommissioning tasks

(d) Monitoring for radioactivity on-site and off-site for demonstrating the effectiveness of decommissioning control procedures and integrity of the containment

(e) Radiation/contamination monitoring of working environment/ components and materials during handling, packaging, transportation and storage

(f) Good house- keeping practices to reduce doses and to prevent spread of contamination

(g) Zoning of the reactor installation in accordance with the guidelines prescribed by AERB and appropriate rezoning procedures as decommissioning progresses so as to ensure spread of contamination does not take place

(h) Record keeping and documentation of all radiation protection measures.

7.4 Radioactivity Measurements and Control

7.4.1 The factors to be addressed in the radioactivity measurements during decommissioning should include:

(a) The presence and nature of all types of radioactive contamination and, in particular, alpha contamination

(b) The significantly higher radiation levels in some facilities, necessitating the consideration of remote measurement

(c) The increased hazards associated with the possible build up of daughter radio nuclides (more significant for actinides)

(d) The complexity of strategies for waste management owing to the diversity of waste streams and adherence of the clearance levels.

7.4.2 Minimisation of radiation exposure to personnel requires a system of radiological control procedures. This system should include establishment of controlled access working zones, issuance of special work permits (SWPs), use of protective clothing and respiratory protection, control of storage and prompt removal of all dismantled components and parts, use of remote tools and shielding techniques.

7.4.3 Monitoring of decommissioning site should include dose rate measurements, radioactivity in air, water and soil in and around the site and assessment of exposure to personnel during and after the work. Techniques to monitor personnel may include hand and foot monitors, alarm dosimeters, TLDs, whole body counting, bioassay etc.

7.4.4 Persons involved in the radiation protection programme should be trained, qualified and should have first hand knowledge in use of appropriate monitoring instrument/equipment for carrying out radiation surveys, assessment of external and internal exposure, surface and air contamination and similar radiological surveillance programme as deemed necessary during the course of decommissioning.

7.4.5 A pre-decommissioning on-site survey, to collect data towards the detailed decommissioning plan and for dose budgeting, should include maximum and average radiation field and contamination levels on SSCs, and the radiological characterisation data. It should also ensure that critical release points are adequately monitored. The subsequent measurements should be carried out as appropriate for detailed planning of the activities during decommissioning. This ensures adequacy of controls and helps in mitigating radiological hazards.

7.4.6 The on-site monitoring programme should ensure availability of :

(a) appropriate monitoring equipment for dose rate and contamination surveys for workplaces, components and materials during decontamination, dismantling and handling,

(b) appropriate monitoring equipment and schedules for packaging and handling of radioactive waste within site, as well as for transportation of the waste off-site (The handling, packaging, storage and transport of radioactive materials should conform to the regulatory requirements on transport of radioactive substances), and

(c) appropriate monitoring equipment, like bulk waste monitors, for timely screening of large quantities of low level radioactive material for clearance purposes.

7.4.7 An environmental surveillance programme should be in place, to ensure that discharge of radionuclides via airborne and liquid pathways are within the stipulated values. Records of radioactive effluent discharges during various stages of decommissioning should be maintained.

7.4.8 Materials and equipment released from the site should be monitored and the values should conform to the clearance values specified by AERB. Release of materials to the environment via air route and liquid stream shall be in accordance with the technical specifications for decommissioning approved by AERB for decommissioning phase.

7.5 Final Radiological Assessment

7.5.1 At the completion of decontamination and dismantling activities and subsequent site remediation, a radiological survey of the facility should be performed to demonstrate that the residual activity is acceptably low and within the criteria set by AERB and that the decommissioning and remediation objectives have been fulfilled. The design and implementation of the survey should be discussed with the regulatory body during the planning period for the survey. This survey may be carried out in phases, as decommissioning work is getting completed, to enable parts of the facility or site to be released from the regulatory control.

7.5.2 The criteria established by AERB in terms of measurable quantities should be compared with field measurements to demonstrate the compliance.

7.5.3 The survey data should be documented in a final survey report and submitted to AERB. The report should include:

(a) The criteria used,

(b) The methods and procedures used to verify compliance with the criteria, and

(c) The measurement data, including appropriate statistical analysis.

7.5.4 The results of the survey shall be included in the final decommissioning report.

7.5.5 Typical contents of the final radiological survey report for NPP/RR are provided in Annexure-VI.

7.6 Clearance

7.6.1 During decommissioning, appreciable quantities of valuable materials may become available for recycling or reuse. Useful materials arising out of decommissioning can be released from the regulatory regime for subsequent disposal, reuse or recycling if the radioactivity content is below clearance levels specified by AERB.

7.6.2 The guiding radiological criteria for clearance are expressed in terms of dose and for practical application this should be converted into quantities derived in terms of mass activity concentration (Bq/g) and/or surface contamination (Bq/cm^2). Radionuclide specific clearance levels should be used for all practical purposes as per AERB stipulation.

7.6.3 From the regulatory view point, it is necessary to be able to verify that the cleared material conforms to the applicable clearance levels. This should be done by direct measurement on the material in question, by laboratory measurements on representative samples, by the use of properly derived scaling factors or by other means that are accepted by AERB.

8. QUALITY ASSURANCE

8.1 General

The plant management shall establish an appropriate quality assurance programme for decommissioning of a NPP/RR covering all activities, which would have an influence on decommissioning in a safe manner. This should be put in place before commencement of decommissioning. The programme shall provide a systematic approach to all activities affecting safety and quality, including, where appropriate, written verification that each task has been performed in accordance with prescribed limits, regulations and approved procedures. The quality assurance programme shall be in conformance with the 'Quality Assurance in Nuclear Power Plants' [AERB/NPP/SC/QA (Rev. 1)-2009].

8.2 Objective

The objective of the quality assurance programme is to ensure that all the mandatory requirements are met with regard to safety of the public, workers and protection of the environment.

8.3 Manual

8.3.1 A quality assurance manual for decommissioning shall be prepared by the responsible organisation/operating organisation for each NPP/ RR. It shall cover all aspects of decommissioning programme taking into consideration the requirements intended in the technical specification for decommissioning of the nuclear reactor. Where applicable, it shall relate to the requirements of other statutory codes and standards. The manual shall be approved by the competent authority and shall be available in time to enable the intended functions to be performed in an orderly manner by qualified persons.

8.3.2 The quality assurance manual should be prepared with the aim of ensuring the following:

(a) The requirement of quality assurance for various decommissioning activities are recognised and verified for conformance.

(b) Procedures for decommissioning activities are prepared, reviewed and approved, including revisions resulting from major changes.

(c) All information relevant to decommissioning shall be adequately documented, stored and made retrievable for future reference upto 30 years after the site is released for 'fit-for purpose' use.

8.4 Quality and Safety Management

To ensure that, the decommissioning activities are carried out in a safe manner, a QA group consisting of competent persons should be established to carry out verification and internal audit functions as per approved plan.

They should inform the decommissioning management any noncompliance to technical specifications and also ensure that event report/significant event report is prepared for all safety related incidents.

9. ORGANISATION AND MANAGEMENT

9.1 General

9.1.1 In order to facilitate safe and orderly execution of the decommissioning programme the responsible organisation should establish a clear-cut hierarchical organisation. A typical organisation chart for executing the activities is shown in Annexure-VII.

9.1.2 The responsible organisation should formulate decommissioning policies and strategies for handling important activities like contracts and materials, health and safety aspects, quality assurance, waste management, documentation and records, staffing and training, financial aspects and administrative support.

9.1.3 The plant management should function broadly under the policies and strategies set out by the responsible organisation and should generally consist of a site decommissioning in-charge (Decommissioning Superintendent) with appropriate support services belonging to the civil, mechanical, electrical, instrumentation and waste management groups reporting to him.

9.1.4 The last activity after decommissioning and site remediation would be detailed radiation survey and monitoring of the site for any residual radioactivity. The group to carry out the activity should consist of staff specialising in monitoring at micro level with sensitive instrumentation. All measurements made shall be methodically recorded for verification by AERB.

9.1.5 After the release of the site either for unrestricted public use or for other monitored uses by the responsible organisation, the decommissioning organisation at the particular site may be wound up.

9.2 Staffing and Training

The staff for decommissioning may be divided into two groups :

(a) The responsible organisation at headquarters.

(b) The decommissioning organisation at the respective sites.

9.2.1 Headquarter Group

9.2.1.1 The headquarters group may normally consist of qualified and experienced senior staff who would formulate procedures to dismantle active components in the mechanical, electrical, instrumentation and civil structural areas. These core group should have the necessary inputs and experience to train the site groups to undertake the decommissioning activities. The head quarters should establish training facility to train personnel in the decommissioning/ dismantling areas. The mock up of the requisite tools and tackles should also be available at the training facility for imparting hands-on training to personnel. Training should also cover radiation and industrial safety measures.

9.2.1.2 The headquarters group should have the following responsibilities:

(a) Preparation of the decommissioning plan and its submission to AERB for securing its approval

(b) Liaison with AERB, environmental monitoring agencies, if any, and statutory bodies at national/state level regarding release of chemical effluents, suspended particulate matter etc. to the environment

(c) Overall supervision of execution of the approved decommissioning plan and preparation and submission of the final decommissioning report

(d) Management and co-ordination for the removal of recoverable radioactive materials like fuel, process fluids etc. applicable from the facility and its safe despatch out of the site

(e) Supervision of site activities for compliance on the following:

(i) Exposure control of decommissioning personnel including overexposure investigations, if any

(ii) Safe disposal of radioactive waste

(iii) Release of radioactive materials below prescribed levels for recycling

(iv) Release of materials below clearance level for unrestricted use.

(f) Provision of resources such as financing, trained manpower etc.

(g) Research and development activities relating to development and testing of various facilities/mechanisms for decontamination and dismantling.

9.2.2 The Site Group

9.2.2.1 The site group should be headed by a decommissioning superintendent. He should be supported by various specialised groups. A typical organisation chart is shown in Annexure-VII. The organisation structure would depend on decommissioning activity at each stage.

The group should have competent staff to cover the following areas adequately:

(a) Safety requirements of the license

(b) Physical protection

(c) Familiarity with reactor systems

(d) Radiation protection

(e) Engineering support

(f) Decontamination

(g) Dismantling and demolition

(h) Robotics and remote handling

(i) Fuel handling

(j) Waste management

(k) Quality control (QC) and QA

(l) Project management.

9.2.2.2 The Decommissioning Superintendent should co-ordinate the activities of all the groups in order to achieve the objectives of the decommissioning programme at the site and give a feedback to head quarters on the status and further requirements and time schedules. He should also co-ordinate the decommissioning activities with AERB, in order to ensure that all health and environmental safety aspects are fully met.

9.2.2.3 The site group should be assigned with the following responsibilities:

(a) Implementation of the decommissioning activities at site as per approved decommissioning plan

(b) Preparation/submission of periodic progress reports on execution of decommissioning activities including major deviations and constraints encountered

(c) The final radiation survey after completing decommissioning and site remediation

(d) Assistance to headquarters group in preparing the final decommissioning report including the manner in which the decommissioning plan objectives were achieved together with supporting data and lessons learned and

(f) Documentation and record-keeping.

9.3 Emergency Preparedness

The head quarters group, which formulates all procedures with regard to decommissioning should also evolve procedures to handle emergency situations. These situations should include radiation emergencies (like uncontrolled radioactive releases/exposures during the various stages of decommissioning), industrial and site specific hazards and emergencies like fire, flooding etc. Site staff should be trained to handle the probable emergencies and appropriate EOPs should be made available.

9.4 Physical Protection

Appropriate physical protection and surveillance of the reactor facility and site should be maintained during decommissioning. This should be given higher consideration in case one adopts a strategy of deferred decommissioning for an extended period.

9.5 Administrative Control

9.5.1 The main control should be exercised from headquarters which should ensure training, availability of personnel and other facilities to carryout, monitor and handle all aspects of the decommissioning programme.

9.5.2 The site administrative control should rest with the site group overall in-charge who should also look after all site related activities in addition to health and well being of all site personnel.

9.5.3 The organisation chart of both the headquarters and site groups should clearly specify all the duties and responsibilities of all the personnel indicated in the organisation chart.

10. DOCUMENTATION AND RECORD KEEPING

10.1 General

10.1.1 Careful and detailed advanced planning is required for safe and successful decommissioning of NPP/RR. Right from design stage till the final shutdown of a nuclear facility, i.e. through construction, commissioning and operational phases, large information/data normally would be generated and documented. Some of this information/data are essential for proper planning and executing the decommissioning programme effectively.

10.1.2 Preparation of initial/preliminary decommissioning plan should be initiated at a suitable stage during operation phase. This plan should be periodically updated till preparation of final decommissioning plan. The on-going process of updating decommissioning plan during operation phase would help in identifying further information/data required to be documented for future decommissioning.

10.2 Basis for Documentation

10.2.1 Since all the records/data generated over the operation phase of the facility are not relevant to decommissioning aspects, it would be necessary to select and collect the information/data needed for decommissioning and preserve them in an easily retrievable form in a cost effective manner. This task should be an on-going process. This task of generating and maintaining a set of decommissioning related records could be part of normal record management system. Basis for records that are to be generated during operation phase of the installation is established and accordingly temporary and permanent records are identified and maintained. Similarly, basis for generating and maintaining a set of decommissioning related records should be established.

10.2.2 Since the decommissioning strategy for any operating nuclear installation may not be decided during early phase of reactor operation, generating and maintaining of decommissioning related records should meet the needs of different possible decommissioning strategies till the final choice is made.

10.2.3 Details of documents that would be useful during decommissioning and, therefore, should be properly stored are given in Annexure-VIII.

10.3 Documentation during Decommissioning

The progress of decommissioning should be documented by the decommissioning organisation. All radioactive materials that were present at the start of decommissioning should be properly accounted for until they reach their final destination. Periodic reports on decommissioning activities and waste disposal should be sent to AERB. The report should also mention up-to-date status of the reactor plant and the site. At the completion of decommissioning and site remediation, a final report should be prepared containing all required information, such as final radiological survey report, summary of events that have occurred, occupational doses and lessons learned etc. and should be submitted to AERB.

10.4 Organisation for Documentation and Record Keeping

Organisation of a nuclear installation should include record management unit or documentation cell depending on size of the installation. Record manager should have added experience on plant O and M and computerised record systems. Since record could be in the form of hard copy or soft copy (computer files) or in both forms, periodic training for access, proper understanding, modifications, security and safety of records is required based on the latest developments in computer systems. On final shutdown of the nuclear facility, record management unit could become part of decommissioning organisation.

ANNEXURE-I

TYPICAL MAJOR DECOMMISSIONING ACTIVITIES
(Ref. subsection 1.3.3)

I.1 Major Activities during Preparatory Steps

I.1.1 Major activities include the following:

(a) Transfer of fuel (including blanket, in case of FBR) from the plant boundary as practicable.

(b) Decontamination of active systems

(c) Draining of moderator, coolant and other auxiliary system liquids after the required structural cooling requirements are met.

(d) Dismantling of equipment and components such as secondary and tertiary cooling systems which are not required for further decommissioning work.

(e) Stabilisation and sealing of systems containing radioactive material.

Stabilisation in a nuclear facility means putting the facility in a stable condition by ensuring sub-criticality of the fissile materials present, adequate cooling of radioactive materials, and adequate control on spread of radioactivity and on spread of contamination.

I.1.2 If plant components, systems and structures, which are highly radioactive and not easily accessible, are left intact; the plant should remain under surveillance of trained and qualified staff till further decommissioning is taken up

I.1.3 The contamination barrier should be kept as it was during the operation phase. All openings of the process systems shall be permanently blocked and sealed with valves, plugs, blanks etc. as necessary. Checks should be carried out to ensure that there are no visible leaks.

I.1.4 Containment building should be kept in a state appropriate to the potential hazard of the atmosphere inside the building.

I.1.5 Access control should be enforced. Access into the facility should be subject to monitoring and surveillance procedures.

I.1.6 All utilities such as ventilation system, electric power supply, compressed air system, waste handling and treatment systems, fire fighting systems, service water system, mechanical handling equipment including monitoring systems for radiation and fire hazards should be operable for further safe decommissioning.

I.1.7 All relevant as-built construction drawings including alterations should be available.

I.2 Major Activities during Decommissioning Phase for Restricted Site Release

I.2.1 Transfer of fuel (including blanket, in case of FBR) from the plant boundary, if it had not been done earlier.

Extensive decontamination and/or dismantling of radioactive parts of the plants may be carried out. The parts so treated may be stored for further radiological decay or disposed off as radioactive waste. Parts whose radioactivity level conforms to the clearance level (CL) may be released as appropriate for recycle and reuse. Decontaminated areas themselves may be considered for restricted or unrestricted use as authorised by AERB.

Sealing of remaining radioactive parts of the plant may be carried out by physical means for further reduction in radiation dose rates due to radiological decay under surveillance.

I.2.2 Reduction of the contamination barrier to the required size after decommissioning and/or dismantling parts of the plant.

I.2.3 Modifications to containment building as also to the nuclear ventilation system as per the radiological safety requirement.

I.2.4 Continuance of surveillance as appropriate for radiological safety, structural integrity and operability of remaining utilities.

I.3 Major activities during Decommissioning Phase for Unrestricted Site Use

I.3.1 Removal of all radioactive parts of the plant

(a) Either for release to the public after achieving required decontamination to the clearance level (CL) or

(b) To solid waste management facility.

I.3.2 The final radiation survey shall be carried out to ensure that the residual radioactivity level on the site conforms to clearance level (CL). Following the survey, the site is released for 'fit for purpose' use and no further surveillance, inspection or tests are required. However, record on all decommissioning activities and final radiation survey should be kept for verification and audit for a period stipulated by AERB.

ANNEXURE-IIA

TYPICAL CONTENTS OF CONCEPTUAL DECOMMISSIONING PLAN (Ref. subsection 3.2.1)

Information to be provided by responsible/operating organisation for new projects/ operating units is as follow:

IIA.1 Facility Details

IIA.1.1 Name and complete location details of the facility

IIA.1.2 Brief description of system structures and components

IIA.2 Design Provision for Decommissioning

IIA.2.1 General

IIA.2.2 Design provision for reduction of radiation sources

IIA.2.3 Provision for ease of decontamination and reduction of radioactive waste

IIA.2.4 Design provision for ease in dismantling

IIA.2.5 Any other special design features

IIA.3 Decommissioning Strategies

IIA.3.1 Decommissioning objectives

IIA.3.2 Options for decommissioning

IIA.3.3 Selection of appropriate option for decommissioning

IIA.4 Radioactive Waste Management

IIA.4.1 Estimated inventory of material and their radioactive content

IIA.4.2 Types and volumes of radioactive wastes anticipated and the method for their disposal

IIA.5 Decommissioning Aspects

IIA.5.1 Planning

IIA.5.2 Safety and environment

IIA.5.3 Waste disposal facility

IIA.5.4 Provision for funding

IIA.5.5 Record and documentation

IIA.6 Decommissioning Process and Prerequisites

IIA.7 Organisation and Management for Decommissioning

ANNEXURE-II B

TYPICAL CONTENTS OF DECOMMISSIONING PLAN
(Ref. subsection 3.2.1)

Information to be provided by responsible/operating organisation, while applying for decommissioning to AERB, is as follows:

IIB.1 Facility Details - At The Time Of Final Shutdown

IIB.1.1 Name and complete location details of the facility

IIB.1.2 Brief description of systems structures and components

IIB.1.3 Date and validity period of present regulatory consent

IIB.1.4 Proposed modification of regulatory consent for decommissioning

IIB.1.5 Estimated inventory of material and their radioactive content, dose rate etc. furnishing the calculational methods and measurement used

IIB.1.6 History of significant events related to spread of activity

IIB.2 Decommissioning Strategy

IIB.2.1 Decommissioning objectives

IIB.2.2 Decommissioning strategy considered with time periods to meet the objectives and their justification

IIB.2.3 Types and volume of radioactive wastes anticipated and the method for disposal in the options considered

IIB.2.4 Dose estimates for the options considered

IIB.2.5 Safety analysis of the decommissioning strategy considered

IIB.3 Decommissioning Organisation and Project Management

IIB.3.1 Organisation structure

IIB.3.2 Function and responsibilities, qualification, training and experience

IIB.3.3 Assistance from outside organisations including contract works

This should include details such as:

Identification of activities to be executed
Commitment to comply with relevant regulatory provisions
Training of contractor's personnel and their records

IIB.3.4 Site security with reference to access control and material movement

IIB.3.5 Project review and monitoring methods

IIB.3.6 Reporting and record keeping

IIB.4 Decommissioning Activities

IIB.4.1 Estimated levels of radioactivity at the time of final shut down and their expected levels at different time periods during decommissioning

IIB.4.2 Requirement of the utilities/service systems and equipment and their surveillance at different time periods of decommissioning

IIB.4.3 Techniques and procedures to be used for decommissioning, dismantling and waste management based on experience/feedback from construction, commissioning and operation phases.

IIB.4.4 Requirement/ development of robotics/remote handling tools/ systems for various decommissioning jobs

IIB.4.5 Revised technical specifications as applicable to decommissioning phase

IIB.4.6 Detailed decommissioning activity sequence diagram with time schedule

IIB.5 Quality Assurance Programme Applicable To Decommissioning

IIB.5.1 Preparation of quality assurance manual applicable to decommissioning of the facility, ensuring that all activities pertaining to decommissioning are carried out in a safe manner

IIB.6 Safety Analysis

IIB.6.1 Availability of safety systems and equipment such as containment, ventilation system, radiation monitoring instruments etc.

IIB.6.2 Safety assessment of probable events during decommissioning

IIB.6.3 Preparedness for management of plant and site emergency

IIB.7 Radiological Protection and Industrial Safety Programme

IIB.7.1 Radiological history of the facility

IIB.7.2 Identification of systems, equipment, areas of the site that might have been inaccessible during the plant's lifetime and may be excessively contaminated

IIB.7.3 Radiation survey and mapping of contaminated systems, structures and components

IIB.7.4 Management's approach to ALARA

IIB.7.5 Implementation of occupational radiation protection programme having the required radiation protection procedures, ensuring the availability of monitoring and survey instruments, establishing effluent analysis and personnel dosimetry methods and an effective contamination control programme.

IIB.7.6 Identification of potential sources of radiation or contamination exposure to workers or to members of the public that are likely to be generated during decommissioning activities and their control.

IIB.7.7 Radiation protection policies for safety of contractor's personnel while working in restricted areas and the method of implementing these policies.

IIB.7.8 Management approach to industrial and fire safety.

IIB.8 Radioactive Waste Management

IIB.8.1 Systems to be used for handling, storing and disposal of radioactive wastes.

IIB.8.2 Estimation of waste generation during decommissioning stages with details such as volume, radionuclide concentration, waste forms, classifications etc.

IIB.8.3 Precautions to be taken to avoid contamination/mix-up of large quantity of inactive material with active material, while taking up decontamination/ dismantling.

IIB.8.4 Details regarding interim/temporary storage of wastes including quantitites, expected period of storage, location of storage, radiation levels at accessible places and methods of control and accounting.

IIB.9 Planned Final Radiation Survey

IIB.9.1 Description of survey procedure

IIB.9.2 Proposed method of survey to ensure all structures, systems, equipment and plant site are included and sufficient data is collected for meaningful survey

IIB.9.3 Description and data on background radiation

IIB.9.4 Type, specification and operating conditions of the instrument to be used

IIB.9.5 Expected radiation and contamination levels towards release of facility for unrestricted use and justification, if regulatory criteria are not likely to be met with.

ANNEXURE-III

PROCESS AND METHODOLOGY OF CHARACTERISATION
(Ref. subsection 5.3.4)

III.1 Process of Characterisation

III.1.1 The process of characterisation essentially comprises of assessment of neutron activated materials and contaminated materials in the reactor.

III 1.2 A comprehensive characterisation programme should comprise review of operation of facility, theoretical calculations for assessment of induced activity of SSCs, adequate number of radiation and contamination surveys, maximum and average dose rates and contamination levels of inner and outer surfaces of structures or components throughout the reactor installation, in-situ measurements, sampling and analysis.

III.1.3 The reactor core and the adjacent thermal shields are most activated part of the reactor structure. The other important components of the reactor structure, exposed to relatively lower neutron fluxes are the liners, biological shield, reflectors and the materials located in and near the reactor core. The significant radionculides to be considered for the activation reaction are tabulated in Table-III.1. Induced activity for long years of effective full power years (EFPY) operation will be almost close to equilibrium activities for significant gamma emitters such as ^{60}Co.

III.2 Radionuclide Inventory Of Activated Components

III.2.1 The inventory of the radionuclides would vary with the type of reactor, the material composition and neutron fluence seen. The inventory calculation requires input of sufficient information on the following variables:

- Geometry, volume and mass of activated materials
- Chemical composition of the materials (an analytical investigation should be made to provide information on the concentration of all elements which get activated to long lived products in reactor structural materials)
- Neutron fluence (time, space and energy distribution of the neutron population inducing activation.)
- Operational history (it is necessary to know the flux levels throughout operation life and shutdown periods, due to the variation of plant operation due to power changes, geometry modifications etc.)

III.2.2 Number of computer codes are available for material activation calculations. Typical codes used to calculate spatial and energy distribution of the neutrons are ANISN (one dimension) and DORT(two dimension). For very complex geometries, codes based on the Monte Carlo method like MORSE, KENO, MCNP and TRIPOLI are suitable. The most commonly used code for neutron activation calculations is ORIGEN2. The main input of the code is the material of irradiation and the irradiation history. Many versions of ORIGEN code such as ORIGEN-S and ORIGEN2 have been developed.

III.3 Radionuclide Inventory of Contaminated Materials

III.3.1 Contamination of the system components arise from the activation and redeposition of the corrosion and erosion products carried by the coolant and from the dispersion of the fission products through breach of cladding. In addition, contamination results from leakages in the primary circuit, processing and storage of effluents and wastes, maintenance and repair activities, fuel discharge operations and incidents during working. Airborne contamination also would result in deposition of radioactive substances on walls, ceilings and ventilation ducts. The contamination occurs on reactor core hardware, primary circuit piping, auxiliary circuits and associated equipment, near fuel charging equipment, storage pools and processing and storage facilities for radioactive wastes and effluents. The probability of contamination in secondary circuits as well as outside systems should not be ruled out while accounting for the contaminated material inventory. Contamination generally accumulates on the facility and equipment surfaces and generally does not penetrate very deep. In biological shields, like concrete, the penetration can be deep.

Various long-lived fission products and actinides to be considered from contamination point of view are tabulated in Table-III.2. They are usually low in reactors that have operated with good fuel performance and are likely to be present in spent fuel storage bays.

III.3.2 Certain computer codes are also available for calculation of surface contamination. These are less reliable than those used for calculation of neutron-induced activity. Codes and models are reactor specific and hence codes developed and used should be validated against experiments or theoretical benchmarks.

III.4 Methods and Techniques of Characterisation

III.4.1 The characterisation programme should be optimised to get the most appropriate and essential information for decommissioning programme. The induced activity and the radionuclide concentrations present in neutron irradiated components and the associated γ-dose rate should be estimated by neutron activation calculations as mentioned and subsequent sampling and in-situ measurements. Remote dose rate measurements and sampling techniques should be employed to assess these parameters to reasonable accuracy. Internal and external contamination of the plant systems and surfaces can be determined by direct in-situ measurements and in the case of removable surface contamination assessment, smear samples should be analysed. For the qualitative and quantitative estimation of residual activity contributing to the dose rates, high-resolution gamma spectrometric techniques are to be employed. For hard-to-measure radionuclides, radioactivity estimates should be done by correlation techniques. Radiochemical separation and analysis may have to be carried out for certain pure beta emitting radionuclides like ^{90}Sr and ^{63}Ni.

III.4.2 Within the wide spectrum of radio nuclides representing radioactive inventory of a shut down nuclear reactor, there are several radionuclides, which have very low energy gamma signatures, which is difficult to measure in the presence of background spectrum. Correlation techniques or scaling factors should be employed to assess these radionuclides. Suitable correlation factors should be developed for these hard-to-detect radionuclides. It is possible to use direct measurements, appropriately decay corrected, of strong gamma emitters in the mixture to infer the inventories of the hard-to-detect radionuclides. For a statistically significant correlation, a large number of measurements on hard-to-detect radionuclides have to be made.

III.5 Assessment of Hard-to-detect Radionuclides.

^{60}Co, which is always present in a shut down nuclear reactor, can be paired for neutron activated corrosion/erosion products like ^{55}Fe, ^{59}Ni, ^{63}Ni or ^{94}Nb to arrive at the correlation factors. Among the fission products, ^{137}Cs which is easily measurable, can be used to arrive at the correlation factors in determining the hard-to-detect fission products like ^{129}I or ^{99}Tc. Due to strong chemical similarities between ^{144}Ce and many actinides, ^{144}Ce is a good candidate for arriving at the scaling factors for hard-to-detect actinides. Such methods should be well tested and the uncertainties should be known.

TABLE III.1

TYPICAL RADIONUCLIDES OF CONCERN IN NEUTRON ACTIVATED MATERIALS

Isotope	Half-life (year)	Means of Production	Emission	Energy (MeV)
	Base Material: Carbon and Stainless Steel			
^{14}C	5730.0	^{14}N (n, p)	β^-	0.156
^{49}V	0.906	^{52}Cr (p, α)	γ, β^-	0.6 [a]
^{54}Mn	0.856	^{56}Fe (d, α)	γ	0.835
^{55}Fe	2.6	^{54}Fe (n, γ)	γ	0.23 [a]
^{59}Ni	8 x 10^4	^{58}Ni (n, γ)	ε	1.06 [a]
^{63}Ni	100.0	^{62}Ni (n, γ)	β^-	0.066
^{65}Zn	0.667	^{64}Zn (n, γ)	γ, ε, β^+	1.115, 1.352, 0.325
^{55}Co	0.194	^{55}Mn (α, n)	β^+, γ	0.474, 0.810
^{60}Co	5.263	^{59}Co (n, γ)	β^-, γ, γ	0.314, 1.17,1.33
^{93}Mo	3.5 x 10^3	^{92}Mo (n, γ)	ε	Nb X-rays
^{94}Nb	2 x 10^4	^{93}Nb (n, γ)	β^-, γ, γ	0.49, 0.702, 0.871
^{95}Nb	0.096	^{95}Zr decay	β^-, γ	0.16, 0.765
^{95}Zr	0.175	^{94}Zr (n, γ)	β^-, γ, γ	0.396, 0.724, 0.756
	Base Material: Concrete			
^{14}C	5730.0	^{14}N (n, p)	β^-	0.156
^{35}S	0.238	^{34}S (n, γ)	β^-	0.167
^{36}Cl	3.01 x 10^5	^{35}Cl (n, γ)	β^-, ε	0.714, 1.18 [a]
^{37}Ar	0.0953	^{36}Ar (n, γ)	ε	0.81[a]
^{39}Ar	269	^{38}Ar (n, γ)	β^-	0.565
^{40}K	1.28 x 10^9		β^-, γ	1.314, 1.46
^{41}Ca	8 x 104	^{40}Ca (n, γ)	γ	K- X-rays
^{45}Ca	0.446	^{44}Ca (n, γ)	β^-	0.257
^{46}Sc	0.229	^{45}Sc (n, γ)	β^-, β^-, γ, γ	1.48,0.357, 0.889, 1.12
^{54}Mn	0.856	^{56}Fe (d, α)	γ, ε	0.835, 0.829, 1.379
^{55}Fe	2.6	^{54}Fe (n, γ)	γ	0.23
^{59}Fe	0.122	^{58}Fe (n, γ)	β^- γ, γ	1057, 1.1, 1.29
^{55}Co	0.194	^{55}Mn (α, n)	β^+, γ	0.474, 0.81
^{60}Co	5.263	^{59}Co (n, γ)	β^-, γ, γ	0.314, 1.17, 1.33

TABLE III.1 (CONT.)

TYPICAL RADIONUCLIDES OF CONCERN IN NEUTRON ACTIVATED MATERIALS (CONT.)

Isotope	Half-life (year)	Means of Production	Emission	Energy (MeV)
	Base Material: Concrete			
^{59}Ni	8×10^4	^{58}Ni (n, γ)	ε	1.06 [a]
^{63}Ni	100	^{62}Ni (n, γ)	β	0.067
^{65}Zn	0.667	^{64}Zn (n, γ)	γ, ε, $β^-$	1.115, 1.352, 0.325
^{94}Nb	2×10^4	^{93}Nb (n, γ)	$β^-$, γ, γ	0.49, 0.702, 0.871
^{95}Nb	0.096	^{95}Zr decay	$β^-$, γ	0.16, 0.765
^{93}Mo	3.5×10^3	^{92}Mo (n, γ)	γ	Nb x rays
	Base Material: Aluminium			
^{46}Sc	0.229	^{45}Sc (n, γ)	$β^-$, $β^-$, γ, γ	1.48, 0.357, 0.889, 1.12
^{54}Mn	0.856	^{56}Fe (d, α)	γ	0.835
^{55}Fe	2.6	^{54}Fe (n, γ)	γ	0.23 [a]
^{53}Fe	0.122	^{52}Fe (n, γ)	$β^-$, γ, γ	1.57, 1.1, 1.29
^{60}Co	5.263	^{59}Co (n, γ)	$β^-$, γ, γ	0.314, 1.17, 1.33
^{65}Zn	0.667	^{64}Zn (n, γ)	γ, ε, $β^-$	1.115, 1.352, 0.325
^{110}Ag	0.69	^{109}Ag (n, γ)	$β^-$, γ[b]	0.087, 0.6577

a - continuous spectrum of x-ray energies below this number due to Bremsstrahlung.

b - energy of most probable β and most probable γ.

ε - electron capture.

TABLE III.2

TYPICAL FISSION PRODUCTS AND ACTINIDES OF CONCERN IN DECOMMISSIONING

ISOTOPE	HALF-LIFE	PRINCIPLE EMISSION	ENERGY (MeV)
^{90}Sr	29.1 years	β^-	0.546
^{99}Tc	2.1*10^5 years	β^-	0.294
^{106}Ru	374 days	β^-, γ, γ	0.039, 0.512, 0.622
^{129}I	1.6 x 10^7years	β^-	0.154
^{137}Cs	30 years	β^-, β^-, γ	0.662, 0.514, 1.2716
^{144}Ce	285 days	β^-, β^-, γ	0.318, 0.185, 0.133
^{238}Pu	87.7 years	α	5.49
^{239}Pu	24110 years	α	5.157
^{241}Pu	14.35 years	β^-	0.021
^{241}Am	432 years	α, γ	5.486, 0.060
^{242}Cm	162.8 days	α, α	6.112, 6.069
^{232}U	72 years	α	5.32
^{233}U	1.6 x 105 years	α	4.82
^{234}U	2.5 x 10^5 years	α	4.775
^{235}U	7 x 10^8 years	α	4.398
^{236}U	2.3 x 10^7 years	α	4.494
^{238}U	4.5 x 10^9 years	α	4.198

ANNEXURE-IV

IN-SITU MEASUREMENTS FOR RADIOLOGICAL CHARACTERISATION (Ref. subsection 5.3.4)

IV.1 General

Three kinds of in-situ measurements may be used in relation to characterisation namely dose rate measurements, radioactive contamination measurements and measurement of relative individual radionuclide activities by spectrometry. In each case, particular attention must be paid to ensure that the methods of measurements take into account the geometry, the surface conditions and the extent of radioactive contaminants. As far as possible, the counting geometry and the nature of samples should be consistent in repetitive measurements. Suitable efficiency correction factors should be included depending on the sample geometry to arrive at the specific activity concentrations. The in-situ measurements take into account the nature and type of radiation emitted, the physical and geometrical conditions and the accuracy requirements.

IV.2 Measurement of Dose Rates

IV.2.1 Radiation dose rates emanating from reactor structural components such as reactor pressure vessel internals and the external walls or any test sections situated in the vessel can be calculated by using transport theory codes such as ANISN, DOT/DORT, TORT or a point kernel code from the knowledge of the calculated activations.

IV.2.2 The gas filled detectors (ionisation chambers, proportional and Geiger-Mueller counters) are employed in assessment of external radiation field. The dose rate measurements in inaccessible areas may require suitable on line measurement systems with matching hardware and software options. Thermoluminiscent dosimeters can also supplement these readings. Use of chemical dosimeters and ESR techniques are also required for qualitative estimation of high dose rate areas. Mock-up measurements and use of phantoms loaded with personnel monitoring devices are required for general estimates of the dose rates, especially in inaccessible areas of reactors. Dose budgeting for the entire decommissioning programme should be derived from such measurements.

IV.3 Contamination Measurements

IV.3.1 Contamination is detected either by the direct method using a Geiger or scintillation detector held close to the surface or by indirect method in which a specified area of 100-300 cm^2 of the component surface is wiped with an absorbent paper which is then counted in a shielded detector. For sampling purpose, in specific areas, where minor surface contamination is found, thin layers of ~1-2 mm of the representative section can be removed until the surface is found free from any transferable contamination. For items such as pipes and equipment, the inner surface should be sampled by scraping. As an alternative, metallic equipment can be melted and sampled homogeneously to arrive at the mass specific activity. Repetitive sampling from various locations should be collected to get a gross representation of the area/equipment sampled. Whenever surface contamination is found to have penetrated deep, more complex methods can be employed like mechanical or chemical techniques for contamination removal and measurements.

IV.3.2 In the direct method, total or fixed external contamination is measured by using fixed detectors commonly coupled with an instrument that integrates the counts over a select time or by scanning a surface systematically with a portable detector. A clear distinction of these readings with the background radiation field should be made during such measurements.

IV.3.3 Buried pipelines, ventilation ducts or hardware embedded in concrete, are often to be excavated for the estimation of contamination. Representative sampling is carried out from the exposed surfaces and gross estimate is derived in such cases.

IV.3.4 In order to restrict the contaminated areas to a minimum, as also to prevent the spread of contamination during decommissioning stages, various areas in the reactors can be divided into radiological zones. Re-designation of zones should be decided depending upon the contamination potential during various stages of decommissioning.

IV.4 Spectrometry Measurements

IV.4.1 To accomplish characterisation of neutron induced activation, samples of system components such as the pressure vessel components and spent fuel assembly hardware may be required. These measurements should empirically determine the concentration of all significant intermediate and long-lived radionuclides. One important use of biological shield sampling is to determine the depth to which it is contaminated above clearance levels. This measurement is of utmost importance in management of decommissioning waste because of the volume and weight of the biological shields.

IV.4.2 Spectrometric techniques can be employed for alpha, beta or gamma emitting radionuclides. In-situ gamma spectrometric measurements can be carried out to assess the internal contamination of hardware like pumps, valves, pipelines and other components. By using appropriate algorithms it is possible to transform the measured gamma spectrum to a radionuclide specific surface contamination and thus to validate the computer codes which simulate the build up and inventory of activated corrosion products.

IV.4.3 In assessing the radionuclides, the selection criteria can be suitably formulated to discard short-lived radionuclides since they have little bearing on the potential detriment to humans during decommissioning operations. In doing so, caution should be exercised to account for the short-lived radionuclides formed from equilibrium activities of long-lived radionuclides.

IV.4.4 High purity Ge-detectors coupled with the state-of-the-art multi-channel analysers can accurately determine the gamma emitting radionuclides. Use of alpha spectroscopy equipment and liquid scintilators can be employed for spectra analysis. High volume scintillators like NaI (Tl) crystals are ideally used for simple spectra when the efficiency rather than resolution is important. Pre-calibrated bulk monitors can be developed for gross characterisation of waste materials in assorted wastes.

ANNEXURE-V

TYPICAL DECONTAMINATION TECHNIQUES

Stage of Decontaminiation	System	Method	Goal
Before dismantling	Pipelines	Chemical + Mechanical	To reduce occupational exposure
Before dismantling	Pool, Tank	Hydro-jet + Blast + Strippable coating method	To reduce occupational exposure
After dismantling	Pipe, Components	Electropolishing + Chemical immersion + Blast method + Ultrasonic Method + Gel Method	To reduce contaminated metal and to reduce radwaste
After dismantling	Concrete surface	Mechanical methods such as scabbler + shaver + drill & spawling + steel grit blast, Thermal stress method such as microwave irradiation and flame scarfing.	Unconditional release of building and reduction of radioactive concrete waste

ANNEXURE-VI

TYPICAL CONTENTS OF FINAL RADIOLOGICAL SURVEY REPORT
(Ref. subsection 7.5.5)

1. **Facility Name**
2. **Facility Description**
3. **Type and Location of Facility**
4. **Background**

   Reason for decommissioning

   Management approach.
5. **Operating History**

   Licensing and operations

   Operating history and utilisation

   Waste disposal practices.
6. **Decommissioning Activities**

   Objectives

   Results of previous surveys including characterisation data

   Dose budget

   Decontamination and dismantling procedures.
7. **Activities for Waste Management**

   Generated quantities of waste (volume, activity)

   Treatment and conditioning

   Disposal, including on-site and/or off-site shipment.
8. **Final Survey**

   Sampling parameters

   Background/baseline levels identified

   Major contaminants identified

   Clearance level criteria.

Equipment and procedures selected

Instruments and equipment

Techniques for instrument use

Procedures followed

Findings of the survey

Summary of findings

Techniques for reducing/evaluating data

Statistical evaluation

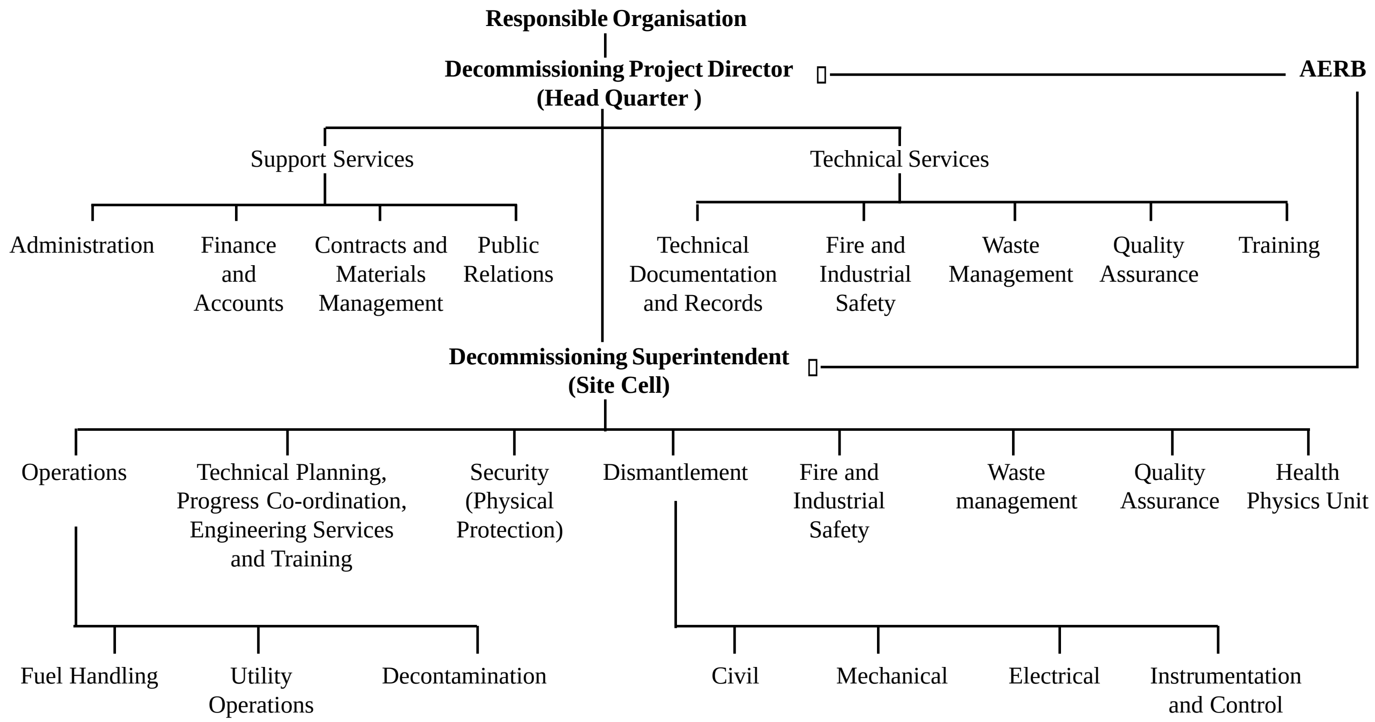
Comparison with guideline values

Assessment of acceptability.

9. Summary/Attachments/Detailed Survey Data with Drawings

ANNEXURE-VII

TYPICAL ORGANISATION STRUCTURE FOR DECOMMISSIONING



ANNEXURE-VIII

DOCUMENTS FOR DECOMMISSIONING
(Ref. subsection 10.2.3)

VIII.1 As-built construction drawings

As-built drawings are the very basic requirement of decommissioning planning. Literature shows number of examples wherein decommissioning of nuclear installation is affected and methods and sequence of dismantling had to be altered due to difference in details as existing at location and as shown in available construction drawing. Based on experience during operation phase and difficulties faced during refurbishment jobs, as-built drawing should be confirmed to be as per site details.

VIII.2 Photographs and video recordings

Photographs and video recordings, taken during construction and installation could be additional tools to decommissioning manager and contractors for selection and working out details on decommissioning activities and their sequence.

VII.3 Constructional sequences/methods

Documentation on construction sequences/methods used during construction phase of nuclear installation can help in deciding appropriate methods and sequence of dismantling. Identification of construction joints and details about metal lining of structures can aid in planning the correct sequence of dismantling.

Details on material handling methods and equipment used during installation could help in organising preparations for various dismantling activities. Documentation on experience gained during major maintenance activities on large equipment during refurbishment could be of great help, as it would give additional input on radiological aspects of handling/dismantling of large equipment.

VIII.4 Scale model of nuclear installation

Scale model of nuclear installation, if made during construction stage, could be an additional aid to decommissioning manager and contractors for working out details on decommissioning activities and for training the decommissioning staff.

VIII.5 Surveillance/Conditioning monitoring of systems, structures and components

Operating plants are required to monitor and carry out surveillance on SSC to assess their ageing effect and residual life. Based on such studies, for plant life extension, refurbishing of various SSC is carried out. Data/document generated during ageing studies and subsequent refurbishment activities, would become an additional input to decommissioning planning.

For example, ageing studies and subsequent refurbishment, if done, could indicate whether material-handling systems in the installation would be able to carry out the required works during decommissioning in safe manner. Containment system of the nuclear facility is required, for confinement of the radioactivity arising during decommissioning activities in the installation. Surveillance during the service period and ageing studies could aid in predicting the condition of the containment which would be in place during decommissioning period which could be few years in case of prompt decommissioning or tens of years in case of deferred decommissioning. Based on the acceptable condition of containment, decision on decommissioning, prompt or deferred, could be taken.

VIII.6 Relevant details of design basis/calculation/material test certificates:

Design basis, calculation, data on material properties based on material test certificates of various SSC particularly thermal/biological shields of reactor structure, would be an aid in assessment of activated radiouclides that would be generated due to irradiation during the life of reactor installation.

VIII.7 Significant event/event reports

Documentation on significant events in the facility till the final shut down can provide useful input for assessing detailed and correct radiological status of SSC, residual fissile material in SSC, alterations/modifications in SSC and changes in operational practices.

Events during operation phase might have changed radiological condition of part of the installation. Details on the same are required for planning decommissioning activities, such as:

- radioactive spillage due to breakage of piping, tanks carrying radioactive fluid could set localised permanent contamination spots,
- excessive fuel clad failures could highly contaminate piping and equipment of primary coolant circuit requiring full-scale decontamination operation,
- inadequate shielding in certain area may generate higher activated products than anticipated in design,
- failures in metal lining of structures (such as sumps, spent fuel storage ponds) holding radioactive material may have contaminated substantial portion of base structure.

VIII.8 Radiation and contamination level mapping as per the periodic radiation survey

During the service period of installation, as per surveillance requirement, radiation and contamination level mapping in different areas/locations in installation is done. This documentation marks the areas, which need special attention for radiological aspects that should be taken into consideration for decommissioning planning.

VIII.9 Accounting of spent/fresh fuel till dispatch of all fuel to off-site location

For any nuclear installation, documentation on accounting for (fresh and spent) fuel inventories and their movement is done in detail and is available for the entire operation phase. This documentation could aid in ensuring the removal of all remaining spent and fresh fuel inventories during initial phase of decommissioning.

VIII.10 Material handling systems

Documents/test reports are required to be maintained on healthiness of material handling systems/equipment (such as cranes, crawlers and other lifting tackles) to comply with relevant provisions of regulatory practices. This documentation is necessary to check for preparedness for decommissioning activities.

VIII.11 Decontamination and dismantling experience during operation/ refurbishment phase

Over the period of operating phase, radioactivity build-up in some systems (e.g. primary coolant) necessitates their full-scale decontamination for reducing collective dose consumption for maintenance outages. Documentation generated on experiences gained through such decontamination and dismantling works and various processes/facilities/mechanisms developed during operation/refurbishment phase, could be employed for decontamination and dismantling during decommissioning.

VIII.12 Environmental monitoring

Records generated on bore-wells around the facility should comprise radioactivity levels of various radionuclides in bore-wells water sediment samples and water levels in bore-wells. These data generated till final shut down would be a reference and will indicate migration of radioactivity, if any, from the facility to the surrounding environment during long duration of decommissioning phase and subsequent site remediation.

Change in water table, if any, due to any site-specific events such as earthquake, sea shoreline change, flooding, etc., would be useful for safety analysis for assessment of confinement of radioactivity.

VIII.13 Additional reports to be maintained:

- Satisfactory functioning of radiological monitoring instruments, their calibration and frequency of testing;
- An up-to-date record on personnel exposure (individual and collective);
- record of radioactive waste under temporary storage, radioactive effluents released and waste finally disposed.
- Results of environmental monitoring and frequency of monitoring.
- Demonstration of compliance with national, state and local government legislation on environmental control; and
- Fire detection and fire fighting provisions in the facility and their testing.
- System/equipment performance including surveillance testing including ISI on all utilities e.g. ventilation system including containment, compressed air, waste handling systems.

ANNEXURE-IX

SPECIAL FEATURES OF DECOMMISSIONING DIFFERENT TYPES OF NUCLEAR POWER PLANT

IX.1 Pressurised Heavy Water Reactor

IX.1.1 Pressurised heavy water reactors (PHWR) are horizontal pressure tube type reactor, with heavy water as coolant and moderator in separate systems. It uses natural uranium fuel in oxide form and short length bundles made of multiple fuel pins. Important features of PHWRs are pressure tube design, low pressure moderator, on power refueling, diverse shut down systems and availability of large low temperature heat sink around the reactor core.

IX.1.2 The reactor consists of a cylindrical vessel called calandria, welded at both ends to the end shields. The fuel channels pass through the two end shields and calandria and contain fuel bundles. The fuel assemblies consist of natural uranium dioxide fuel pellets in zirconium clad, similar to LWR fuel. However, short, cylindrical bundles of pins allow a unique on-line fueling scheme whereby a machine attaches to each end of a single coolant tube and with fuel bundles inserted from one side and discharged on the other side simultaneously.

IX.1.3 The calandria is filled with low temperature, low pressure heavy water in the space between the calandria tubes and acts as a moderator. Reactivity control devices enter the calandria shell from top as well as from sides (in 540 MWe) and from top and bottom (in 220 MWe design). Stand pipes and thimbles welded on to calandria nozzle provide extension of core in the accessible region outside the calandria vault for mounting of reactivity control devices. The calandria is housed in steel lined concrete calandria vault filled with light water which provides shielding and cooling of the vault structures. This also brings down considerably Argon-41 activity in the ventilation air which is one major contributor of dose to public during normal operation. The steam-cycle is in the secondary loop, like the PWR, with the primary pressurised heavy-water loop transferring heat energy to a loop of light water for steam production.

IX.1.4 Indian PHWRs have several design features, which would contribute to ease of decommissioning. Some of these features are unique to the inherent design of the PHWR. Other features have been incorporated primarily to aid operation and maintenance or in-service replacement of components, and are expected to play a positive role at the time of decommissioning. Some of these features are detailed below:

(i) Water filled calandria vault :

The space between the reactor and the concrete shield around it is filled with neutron absorbing de-mineralised water, thereby substantially reducing the activation of the concrete of the calandria vault.

(ii) Steel ball filled end shield :

The end shields consist of steel shells filled with balls of steel. This design would facilitate segmenting of the end shield plates after removing the steel balls.

(iii) Removal of coolant channels (in core components) :

The PHWR reactor core has number of coolant channels housing reactor fuel. Fuel can be replaced remotely during power operation by fuelling machines, a design provision that could be used for the final removal of fuel at the decommissioning stage. Also, the design of the PHWR considers the requirements of en-masse replacement of coolant channels during the life of the plant. The design features, provisions and tools provided for this replacement can be directly used at the time of decommissioning.

(iv) Removal of steam generators :

The steam generators are installed by lowering them into the reactor building vertically through hatch openings in the reactor-building dome. These openings are adequately sealed for containment during normal operation and can be opened for replacement of steam generators during the life of plant and also for decommissioning.

(v) Reactivity control assemblies :

The adjustor rods for reactivity control can be loaded with control assemblies containing cobalt for the purpose of producing the radioactive cobalt isotope. Design provisions and tools are provided for the periodic replacement of these assemblies. This feature would again be used during decommissioning for the removal of the reactivity control assemblies.

(vi) Design provision for minimisation of personnel exposure :

Extensive design improvements have been carried out in Indian PHWRs for reducing personnel exposure during operation and maintenance (selection of materials to minimise activation, etc.). In particular, the use of cobalt-bearing alloys is severely restricted in vulnerable areas. Decontaminability of surfaces prone to being contaminated from radioactive spills is also considered in the design. These features would limit personnel exposure as well as radioactive wastes during decommissioning.

IX.1.5 After the final shutdown of the reactor, the foremost aspect before removal of the core components is to ensure that their decay heat has been reduced to an acceptable level. The start of decommissioning activity involves defuelling the channels and transfer of fuel bundles to spent fuel bay, draining of heavy water from moderator and primary heat transport system (PHT), decontaminating PHT and part of fuel transfer systems including fuelling machines, and removal of the systems, which are not required during the decommissioning.

IX.1.6 Amongst the radionuclides, ^{55}Fe, ^{60}Co, ^{63}Ni continue to dominate for a few years after shutdown, following 30-40 years of operation. From radiological point of view ^{60}Co is the most dominant among these nuclides during the first three decades because of its hard gamma radiation. The other nuclides considered are primarily soft beta and X-ray emitters.

IX.1.7 In PHWRs, the main radioactivity is contained only in fuel channels and provision has been made to handle the fuel channels during decommissioning. Hence handling of active components is simpler compared to that in BWRs or PWRs. But in PHWRs, the number of components to be handled are more compared to BWRs or PWRs. Most of the components in the PHWRs can be dismantled without much difficulty with a few exceptions. Considering the problem associated with handling large size radioactive components, the following major components have to be cut to manageable size so that shielding and handling operations enables safe handling: (i) fuel channels (ii) calandria tube (iii) end shield (iv) calandria (v) dump tank (in some designs) (vi) calandria vault structure (vii) thermal shield (viii) biological shield. The size of the individual segments of these components would be suitably determined based on their level of radioactivity.

IX.1.8 Feasibility for reusing not so easily available materials like Zirconium alloy, Monel etc. from the decommissioned reactors is also being studied. These materials can be reused if their radioactivity comes down to an acceptable level by chemical treatment or by keeping them under mothballing for decay.

IX.1.9 Cost of decommissioning is accounted for, in the cost of electricity.

IX.2 Fast Breeder Reactors

IX.2.1 Presently fast reactor uses liquid sodium as coolant compared to heavy water/ light water in thermal reactors. It has three loops such as primary, secondary and tertiary compared to two/one loop in thermal reactors. The major concern in the decommissioning process of FBRs is disposal of large quantity of used (radioactive/non-radioactive) sodium (~ 1150 Te for a typical 500 MWe pool type plant or ~ 140 Te in the case of presently operating fast breeder test reactor at IGCAR) either by converting it into chemically stable component or by reusing in reactors or in industries like pharmaceutical industries.

IX.2.2 The primary sodium circuit components of pool type FBRs are housed in the main vessel, which is surrounded by a concentric safety vessel. The main vessel is supported at the top from roof slab. The roof slab together with the small rotatable plug (SRP) and large rotatable plug (LRP) form the top shield. The major primary circuit components, viz. primary sodium pumps (PSP) and the intermediate heat exchangers (IHX) are supported on the roof slab, and the bulk of the sodium of the primary circuit is contained within the main vessel. The control plug, housing the control and safety rod drive mechanisms (CSRDM) and diverse safety rod drive mechanisms (DSRDM), is supported on the SRP. The transfer arm (TA), for in-vessel handling of core sub-assemblies (CSA), is supported on the SRP, and the four safety-grade decay heat exchangers (DHX) are supported on the roof slab. The core sub-assemblies are supported on the grid plate, which transfers the load to the main vessel through the core support structure. A core catcher is incorporated as a defense in depth approach to take care of containment and cooling requirements of molten fuel during the unlikely scenario of core disruptive accident. The inner vessel, which separates the hot and cold pools of the primary sodium, is supported on the grid plate. Brief description of the steps involving fast reactor decommissioning is described below.

IX.2.3 After the final shutdown of FBRs, the foremost aspect before removal of the CSA is to ensure that their decay heat has been reduced to an acceptable level. The start of decommissioning activity (after completion of removal of CSA) involves maintaining the primary sodium temperature in main vessel, using immersion heaters or surface heaters. Subsequently, the fuel subassemblies, absorber rods, blanket, inner B_4C, and first two rows of shielding sub-assemblies are handled using the transfer arm and the inclined fuel transfer machine (IFTM) and removed from the reactor core using the normal fuel-handling route. All these CSA would be cleaned of sodium, decontaminated and transferred to the designated external storage locations, from where the fuel and blanket subassemblies would be sent for reprocessing while the other subassemblies would be sent for disposal to waste management facility

IX.2.4 From the data on most of the operating sodium cooled reactors, the radioactive nuclides of concern are mainly ^{22}Na, ^{137}Cs, ^{134}Cs, ^{54}Mn, ^{65}Zn, ^{60}Co, ^{3}H (tritium) and Pu. Of these, all nuclides except ^{60}Co would be found in liquid sodium while ^{60}Co and part of ^{54}Mn would be found on the components. Cooling period of the order of 5-6 years would be allowed before draining, handling and disposal of primary sodium. Hence, at the start of decommissioning activity, the out-of-core primary sodium purification facility would be installed primarily to remove ^{134}Cs and ^{137}Cs by Cesium traps. The getter material in these adsorption-based cesium traps is reticulated vitreous carbon (RVC), which forms an intercalated compound with cesium thereby retaining the radionuclides in the trap location. In parallel, the tertiary loop would be dismantled.

IX.2.5 The water vapour-nitrogen process would be used for cleaning of large components, such as the pumps, IHX, DHX etc., while the steam-nitrogen or steam carbon-dioxide process would be employed for components with non-drainable sodium. In these processes, the sodium adhering to the component surface is made to react with water vapour/steam to form sodium hydroxide. These processes supplemented by washing with demineralised water would be followed to dissolve out the sodium sticking to the surface of components before their decontamination and disposal.

IX.2.6 The main sources of contamination of the primary components at the end of reactor life are : (i) activated corrosion products from the core that are released into the sodium during normal operation of the reactor; and (ii) fission product and fuel material that are released in the event of clad rupture. A mineral acid based process, employing a suitable mixture of H_2SO_4 and H_3PO_4, would be used for decontamination of the various components of FBR. In this process, a thin layer of the component surface is chemically milled out thereby dissolving out the radionuclides diffused deep into the matrix at relatively high temperatures experienced during reactor operation.

IX.2.7 The sodium disposal would be carried out by converting it to sodium hydroxide employing the NaOH process by injecting sodium into a large pool of water or dilute sodium hydroxide that is kept vigorously stirred. In fact, sodium disposal by the NaOH process is considered the only industrially viable process for destroying large quantities of sodium with a capacity to destroy about 3Te/day. This process would involve converting the NaOH formed during the sodium disposal into solid waste. The acidic liquid waste generated during decontamination of components would be neutralised using the NaOH produced and would be disposed off as high active waste.

IX.2.8 No special cleaning is needed for the secondary sodium as, in principle, this sodium can be used as such after taking care that the tritium and ^{22}Na levels are within the allowable limits. This sodium can be used in the newly constructed FBRs, and in the chemical, pharmaceutical, metallurgical and engineering industries. However, limitations on radioactive and non-radioactive impurities are imposed on sodium.

IX.2.9 Complete draining of sodium from the main vessel is a necessary prerequisite, from safety considerations, before opening the top shield to enable easy access to inside of the main vessel and facilitate faster removal of the internals. Suitable shielding would be erected over the opening in conformity with the radiation levels at the openings. As the reactor assembly components have been suitably designed, the individual components can be removed with minimum preparatory effort. The removal of the RA components would more or less follow the 'first in last out' sequence (i.e. the last component that would be removed would be the one that was first erected.)

IX.2.10 However, considering the problems associated with handling large sized radioactive components, the following major components may have to be cut to manageable sizes so that the associated shielding and handling operations enable safe handling, and removed in the order : (1) inner vessel, (2) grid plate, (3) core support structure, (4) thermal baffle, (5) large rotating plug, (6) main vessel, (7) roof slab, and (8) safety vessel. The sizes of the individual items of these components would be suitably determined based on their level of radioactivity.

IX.3 Boiling Water Reactors

IX.3.1 In case of a Boiling water Reactor (BWR) direct cycle is used wherein steam is produced through boiling of coolant cum moderator water in the reactor core region and then collected through separators and transferred to steam turbine. Thus, radioactive contamination is transported from reactor to other systems such as turbine generator, condenser, condensate extraction pump, valves and pipes. This increases the number of radio active materials to be handled when decommissioning of BWR plant is taken up.

IX.3.2 If prompt decommissioning option is not selected then these active systems may be kept isolated so that radio activity present in the system is allowed to decay. Subsequently some parts can be decontaminated for recycle and reuse. Those components which are not amenable for decontamination, it will have to be disposed off as solid waste suitably. The fuel from the reactor as well as from the spent fuel bay may be removed from the site and kept at the designated location for its extended storage or subsequent reprocessing as the case may be.

IX.3.3 The reactor vessel, drywell, containment, spent fuel bay and other areas of the reactor building will have fixed as well as loose contamination to a varied extend. Based upon the requirement this will have to be decontaminated before any dismantlement work is taken up to reduce manrem expenditure for decommissioning. The reactor vessel, secondary steam generators and other bigger equipments may be transported as a single piece and disposed off as solid waste to reduce quantum of the work as well as manrem expenditure. Similarly all the equipments which are bulky like pumps, valves, heat exchangers may be transported as single piece for solid waste disposal. Some very heavy parts like condenser may need separate handling procedures for its disposal as these parts are to be broken down first and then taken up for disposal. The turbine building will also have contamination on the floor, walls and other structures inside to varied extent. This will also require decontamination of all these parts and components before dismantlement for waste minimisation.

IX.4 Pressurised Water Reactors

IX.4.1 The VVER-1000 model 412 (under construction in Kudankulam) belongs to the family of pressurized water reactors (PWRs) which are the predominant type in operation, world over. In this reactor, light water is coolant and moderator and enriched uranium (about 4.4% U235)is used as fuel. Primary coolant system consists of four circulating loops, each containing a horizontal steam generator, a main circulating pump. The loops are connected with the reactor pressure vessel assembly by interconnecting piping. Pressurising system and passive parts of ECCS (Accumulators) are also connected to the system.

IX.4.2 The reactor vessel is designed to contain the vessel internals and fuel assemblies of the core. The reactor along with control rod drive units has overall height of 19 meters and diameter of 4.5 meters. It is a vertical cylindrical container made of 190 mm thick high purity low alloy steel ring forgings welded together and cladded inside with austenitic stainless steel. It has a detachable top cover with sealing elements.

IX.4.3 The reactor vessel houses core barrel, which in-turn houses all the core components including the fuel assembly. All core components are made of austenitic stainless steel. The reactor with the top cover is kept in a concrete pit inside the containment. On the top of the reactor vessel, the head assembly, containing the control rod drive mechanisms, is mounted. The vessel flange has 54 stud holes and two grooves to receive metallic "O" rings for sealing the main flange joint.

IX.4.4 Core internals are used essentially for reactor coolant channeling, shielding the vessel against irradiation from the core, support and hold down the fuel assemblies, securing the irradiation specimen and mechanical guiding of control assemblies and in-core instrumentation. There are 163 nos. of fuel assemblies arranged in a hexagonal lattice pattern within the reactor core with the help of stainless steel supporting structure. There are 121 control rods for reactor protection made of B_4C and Dysprosium Titanate. There are also burnable absorber rods (BARs) made of CrB_2 in Aluminium matrix. The cladding material is Zr+1%Nb.

IX.4.5 A core catcher is incorporated as a defense in depth approach to protect the containment integrity and retention of the molten corium within the hermitically sealed area during hypothetical accidents.

IX.4.6 This reactor has several design features, which could contribute to the ease of decommissioning. Some of these features are inherent in design. Other features have been incorporated primarily to aid operation and maintenance or in-service inspection/replacement of components, but would pay a positive role at the time of de-commissioning. Some of these features are given below:

(a) The reactor pressure vessel (RPV) is placed inside a concrete reactor pit, which acts as a biological shield and thereby substantially reduces the activation of the other concrete structures.

(b) On-site independent fuel storage facility normally available.

(c) Primary coolant loop components and other reactor internals are mostly replaceable.

(d) Low Co and low Ni steels are used as ingredient materials for the components.

(e) Amenable plant lay-out for easy dismantling during the decommissioning process.

(f) Low contamination due to good water chemistry control, selection of materials and high-capacity primary coolant purification system.

(g) In-situ decontamination is possible at the very beginning of the decommissioning activity by utilizing the existing plant systems, experienced plant personnel and available infrastructure to the maximum extent. Full system decontamination prepares the entire primary circuit including the RPV and most of the plant primary and auxiliary systems for dismantling in one single step.

IX.4.7 The following approach could be followed for decommissioning:

(a) After the final shutdown of the reactor, the foremost aspect before removal of the core components is to ensure that their decay heat is reduced to an acceptable level.

(b) De-fuel the reactor and transfer the entire fuel bundle into spent fuel storage facility and/or away from reactor spent fuel facility.

(c) Remove all the contaminated equipment, piping and system from various building. Maintain the facility in this condition for 10 years, by that time radioactivity level are likely to be reduced by factor of 5.

(d) Mothballing initially for 30 years. This approach would reduce the total volume of radioactive waste. Subsequent step will be finalized after assessing the situation.

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LIST OF PARTICIPANTS

EXPERT COMMITTEE FOR SAFETY GUIDE ON DECOMMISSIONING OF NUCLEAR POWER PLANTS AND RESEARCH REACTORS (ECSGRW-8)

Dates of meeting	:	November 7, 2002 January 3, 2003 January 29, 2003 April 21, 2003 August 7, 2003 March 25, 2004 April 19, 2004 August 19, 2004

Members and Invitees of ECSG RW-8:

Shri S. Sankar (Chairman)	:	BARC (Former)
Shri K.B. Dixit	:	NPCIL
Shri B.L. Sharma	:	TAPS, NPCIL
Shri K.J. Vakharwala	:	BARC
Shri R. Venkatraman	:	AERB
Dr. R.K. Gopalkrishnan	:	BARC
Shri R.K. Chugha (Member Secretary)	:	AERB

Invitees:

Shri K.P. Dwivedi	:	NPCIL (on 25th March, 19th April, 2004)
Shri D.K. Jain	:	NPCIL (on 25th March & 19th August, 2004)
Dr J.L. Goswamy	:	BARC (on 7th August, 2003 & 19th April 2004)
Shri S. Datta	:	BARC (on 7th August 2003 & 19th April 2004)

ADVISORY COMMITTEE FOR SAFETY DOCUMENTS ON RADIOACTIVE WASTE MANAGEMENT (ACSDRW)

Dates of meeting	:	October 21, 2002 December 17, 2004

Members of ACSDRW:

Shri K. Balu (Chairman)	:	BARC (Former)
Shri P.D. Sharma	:	NPCIL (Former)
Shri S. Sankar	:	BARC (Former)
Shri N.K. Bansal	:	BARC (Former)
Shri R.M. Sharma	:	BARC (Former)
Shri A.R. Sundararajan	:	AERB (Former)
Shri S.K. Agarwal	:	AERB (Former)
Shri R.C. Rastogi	:	BARC & IAEA (Former)
Shri P.Hajra	:	AERB (Former)
Shri P.M. Gandhi	:	BARC
Shri S.S. Ali	:	BARC (Former)
Shri K.K. Chandraker	:	AERB (Former)
Dr. P.Vijayan (Secretary)	:	AERB

ADVISORY COMMITTEE ON NUCLEAR SAFETY (ACNS)

Date of meeting	:	March 31, 2009

Members of ACNS:

Shri G.R. Srinivasan (Chairman)	:	AERB (Former)
Shri S.C. Hiremath	:	HWB (Former)
Shri S.S. Bajaj	:	NPCIL (Former)
Shri D.S.C. Purushottam	:	BARC (Former)
Shri A.K. Anand	:	BARC (Former)
Shri H.S. Kushwaha	:	BARC
Shri R.K. Sinha	:	BARC
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Dr. S.K. Gupta	:	AERB
Shri K. Srivasista (Member-Secretary)	:	AERB

PROVISIONAL LIST OF SAFETY CODES AND GUIDES ON RADIOACTIVE WASTE MANAGEMENT

Safety Series No.	Titles
AERB/NRF/SC/RW	Management of Radioactive Waste
AERB/NRF/SG/RW-1	Classification of Radioactive Waste.
AERB/NRF/SG/RW-2	Pre-disposal Management of Low and Intermediate Level Radioactive Waste.
AERB/NRF/SG/RW-3	Pre-disposal Management of High Level Radioactive Waste.
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AERB/NPP/SG/O-11	Management of Radioactive Waste Arising from Operation of Pressurised Heavy Water Reactor Based Nuclear Power Plants.
AERB/SG/D-13	Liquid and Solid Radwaste Management in Pressurized Heavy Water Reactor Based Nuclear Power Plants.

NOTES

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