GUIDE NO. AERB/NPP-PHWR/SG/D-12

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GOVERNMENT OF INDIA

AERB SAFETY GUIDE

RADIATION PROTECTION ASPECTS IN DESIGN FOR PRESSURISED HEAVY WATER REACTOR BASED NUCLEAR POWER PLANTS

ATOMIC ENERGY REGULATORY BOARD

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RADIATION PROTECTION ASPECTS IN DESIGN FOR PRESSURISED HEAVY WATER REACTOR BASED NUCLEAR POWER PLANTS

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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 has been entrusted with the responsibility of laying down safety standards and framing rules and regulations for such activities. The Board has, therefore, undertaken a programme of developing safety standards, codes of practice 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 regulation aspects of these facilities.

Codes of practice and safety standards are formulated on the basis of 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 objectives and set minimum 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.

The Code of Practice on 'Design for Safety in Pressurised Heavy Water Reactor Based Nuclear Power Plants (AERB Code No. SC/D)' lays down the minimum requirements for ensuring adequate safety in plant design. This safety guide is one of a series of guides, which have been issued or are under preparation, to describe and elaborate the specific parts of the code.

The guide is based on the current designs of 220 MWe and 540 MWe Pressurised Heavy Water Reactors (PHWRs). It provides guidance to the plant designer on design aspects of radiation protection. In preparing the guide, International Atomic Energy Agency (IAEA) documents such as 'International Basic Safety Standards for Protection against Ionising Radiation and for the Safety of Radiation Sources' (IAEA Safety Series No. 115, 1996), 'Design Aspects of Radiation Protection for Nuclear Power Plants' (Safety Series No. 50-SG-D9, 1985) and other relevant documents (such as ICRP-60) have been used extensively.

Consistent with the accepted practice, 'shall', 'should' and 'may' are used in the guide to distinguish between a firm requirement, a recommendation and a desirable option, respectively. Appendices are an integral part of the document, whereas annexures, footnotes, bibliography and list of participants 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 guide, applicable and acceptable national and international standards, codes and guides should be followed. Non-radiological aspects of industrial safety and environmental protection are not explicitly considered. 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 safety guide has been prepared by specialists in the field drawn from the Atomic Energy Regulatory Board, Bhabha Atomic Research Centre, Indira Gandhi Centre for Atomic Research, 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.

Accident Conditions

Substantial deviations from operational states, which could lead to release of unacceptable quantities of radioactive materials. They are more severe than anticipated operational occurrences and include design basis accidents as well as beyond design basis accidents.

Activity

The quantity 'A' for an amount of radionuclide in a given energy state at a given time is defined as:

$$A = \frac{dN}{dt}$$

where 'dN' is the expectation value of the number of spontaneous nuclear transformations from the given energy state in a time interval 'dt'. The SI unit of activity is the reciprocal of second (s^{-1}), termed the Becquerel (Bq).

ALARA

An acronym for 'As Low As Reasonably Achievable'. A concept meaning that the design and use of sources, and the practices associated therewith, should be such as to ensure that exposures are kept as low as reasonably practicable, with economic and social factors taken into account.

Bio-assay

The determination of the kind, quantity, location, and/or retention of radionuclides in the body by in vitro analysis of material excreted or removed from the body.

Collective Dose

An expression for the total radiation dose incurred by a population and defined as the product of the number of individuals exposed to a source and their average radiation dose.

Committed Effective Dose, E (τ)

The time integral of the whole body effective dose rate following an intake of a radionuclide. The quantity 'E (τ)' is defined as

$$E(t) = \sum_{T} W_T \cdot H_T(t)$$

where '$H_T(t)$' is the committed equivalent dose to tissue 'T' over the integration time 't'. When 't' is not specified, it will be taken to be 50 years for adults and age 70 years for intake by children.

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.

Controlled Area

A delineated area to which access is controlled and in which specific protection measures and safety provisions are, or could be, required for

(a) controlling normal exposures or preventing the spread of contamination during normal working conditions; and

(b) preventing potential exposures or limiting their extent should they occur.

Countermeasures

An action aimed at alleviating or mitigating the consequences of accidental release of radioactive material into the environment.

Critical Group

A group of members of the public which is reasonably homogeneous with respect to its exposure for a given radiation source and given exposure pathway and is typical of individuals receiving the highest effective dose or equivalent dose (as applicable) by the given exposure pathway from the given source. When exposure occurs by more than one pathway, the term may also be used to mean the group which receives the highest total dose by all the pathways of exposure from a given source or practice.

Critical Pathway

The dominant environmental pathway through which members of the critical group are exposed to radiation.

Decontamination

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

Defence-in-Depth

Provision of multiple levels of protection for ensuring safety of workers, the public or the environment.

Derived Air Concentration (DAC)

That activity concentration of the radionuclide in air (Bq/m^3) which, if breathed by reference man for a working year of 2000 h under conditions of light physical activity (breathing rate of 1.2 m^3/h), would result in an inhalation of one ALI, or the concentration, which for 2000 h of air immersion, would lead to irradiation of any organ or tissue to the appropriate annual dose limit.

Design

The process and results of developing the concept, detailed plans, supporting calculations and specifications for a nuclear or radiation facility.

Design Basis Accidents (DBAs)

A set of postulated accidents which are analysed to arrive at conservative limits on pressure, temperature and other parameters which are then used to set specifications to be met by plant structures, systems and components, and fission product barriers.

Design Basis Events (DBEs)

The set of events, that serves as part of the basis for the establishment of design requirements for systems, structures and components within a facility. Design basis events (DBEs) include operational transients and certain accident conditions under postulated initiating events (PIEs) considered in the design of the facility (see also 'Design Basis Accidents').

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 not necessary for defining the quantity of interest.

Dose Constraint

A prospective and source-related restriction on the individual dose delivered by the source, which serves as a bound in the optimisation of protection and safety of the source. For occupational exposures, dose constraint is a source-related value of individual dose used to limit the range of options considered in the process of optimisation. For public exposure, the dose constraint is an upper bound on the annual dose that a member of the public should receive from the planned operation of any controlled source. The exposure to which the dose constraint applies is the annual dose to any critical group, summed over all exposure pathways, arising from the predicted operation of the controlled source. The dose constraint for each source is intended to ensure that the sum of doses to the critical group from all controlled sources remains within the dose limit. For medical exposure the dose constraint level should be interpreted as a guidance level, except when used in optimising the protection of persons, other than workers, who assist in the care, support or comfort of exposed patients.

Dose Limit

The value of the effective dose or the equivalent dose to individuals from controlled practices that shall not be exceeded.

Effective Dose

The quantity 'E' is defined as a summation of the tissue equivalent doses, each multiplied by the appropriate tissue weighting factor:

$$E = \sum_{T} w_T . H_T$$

where 'H_T' is the equivalent dose in tissue 'T' and 'w_T' is the tissue weighting factor for tissue 'T'.

Emergency Plan

A set of procedures to be implemented in the event of an accident.

Environment

Everything outside the premises of a facility, including the air, terrain, surface and underground water, flora and fauna.

Exclusion Zone

An area extending up to a specified distance around the plant, where no public habitation is permitted. This zone is physically isolated from outside areas by plant fencing and is under the control of the plant management.

Exposure

The act or condition of being subject to irradiation. Exposure can be either external (irradiation by sources outside the body) or internal (irradiation by sources inside the body). Exposure can be classified as either normal exposure or potential exposure; either occupational, medical or public exposure; and in intervention situations, either emergency exposure or chronic exposure. The term 'exposure' is also used in radiation dosimetry to express the amount of ions produced in air by ionising radiation.

Exposure Pathway

A route by which radiation or radionuclides can reach humans and cause exposure.

Limit

The value of a parameter or attribute (which is variable) used in certain specific activities or circumstances that must not be exceeded.

Loss of Coolant Accident (LOCA)

An accident resulting from the loss of coolant to the fuel in a reactor due to a break in pressure retaining boundary of the primary coolant system.

Member of the Public

Any individual in the population except for one who is subject to occupational or medical exposure. For the purpose of verifying compliance with the annual dose limit for public exposure, the member of the public is the representative individual in the relevant critical group.

Monitoring

The continuous or periodic measurement of parameters for reasons related to the determination, assessment in respect of structure, system or component in a facility or control of radiation.

Normal Operation

Operation of a plant or equipment within specified operational limits and conditions. In case of a nuclear power plant, this includes, start-up, power operation, shutting down, shutdown state, maintenance, testing and refuelling.

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.

Occupational Exposure

All exposures of personnel incurred in the course of their work.

Occupational Worker

Any person, working full time or part time in a nuclear or radiation facility, who may be employed directly by the 'consentee' or through a contractor.

Off-site

Area in public domain beyond the site boundary.

Operation

All activities following and prior to commissioning performed to achieve, in a safe manner, the purpose for which a nuclear/radiation facility is constructed, including maintenance.

Operational States

The states defined under 'normal operation' and 'anticipated operational occurrences'.

Practice

Any human activity that introduces additional sources of exposure or exposure pathways or extends exposure to additional people or modifies the network of exposure pathways from existing sources, so as to increase the exposure or the likelihood of exposure of people, or the number of people exposed.

Prescribed Limits

Limits established or accepted by the regulatory body.

Public Exposure

Exposure incurred by members of the public from radiation sources, excluding any occupational or medical exposure and the normal local natural background radiation, but, including exposure from authorised sources and practices and from intervention situations.

Quality Assurance (QA)

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

Safety Analysis

Evaluation of the potential hazards (risks) associated with the implementation of a proposed activity.

Site Personnel

All persons working on the site, either permanently or temporarily.

Source

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

Supervised Area

Any area not designated as a controlled area but for which occupational exposure conditions are kept under review even though specific protective measures and safety provisions are not normally needed.

CONTENTS

1. INTRODUCTION

1.1 General

1.1.1 This safety guide deals with the provisions to be made in the design of pressurised heavy water reactor (PHWR) based nuclear power plants to protect the site personnel and the public from undue exposure to ionising radiation during all the design basis events (operational transients and accident conditions).

1.1.2 It should be recognised that effective radiological protection is achieved by a combination of good design, quality assurance, proper selection of materials, high quality construction, safe operation and above all a safety culture that pervades all these aspects.

1.1.3 The radiation protection principles outlined in this guide are generally based on the ICRP Recommendations [1], IAEA Basic Safety Standards [2], and AERB safety manual on Radiation Protection for Nuclear Facilities [3].

1.2 Objective

The purpose of this safety guide is to provide guidelines for implementation of radiation protection in the design of nuclear power plants, consistent with the requirements of the Code of Practice on Design for Safety in Pressurised Heavy Water Reactor Based Nuclear Power Plants (AERB/SC/D) [4]. This guide is meant for use by nuclear power plant (NPP) designers and the plant operating personnel.

1.3 Scope

1.3.1 This guide covers:

i) Principles and concepts of dose limitation as a basis of radiological protection measures to be implemented in the design.

ii) Important sources of radiation and contamination during normal operation and design basis events (DBE).

iii) Design measures and provisions for radiological protection of plant personnel, members of the public and the environment.

iv) Radiation detection and monitoring systems and portable instruments required at the plant to verify whether the design meets the required level of radiation protection.

1.3.2 The requirements and considerations for quantitative assessment of radioactive sources generated in the reactor and their transport as well as shielding aspects, have been briefly indicated. Radiation protection aspects during operation of NPPs are covered in the safety guide on Radiation Protection during Operation of NPPs (AERB/SG/O-5) [5].

1.3.3 While this safety guide has been prepared specifically for radiation protection in design of PHWR based NPPs, it may also be applicable, with suitable modifications, to other types of reactors.

2. DESIGN REQUIREMENTS

2.1 General

2.1.1 A general requirement is that radiological aspects of design should be addressed right from the initial design stages. Quality assurance in design should also address this aspect.

2.1.2 In accordance with the basic principles of radiation protection, provisions should be made to comply with the following design objectives.

(a) Radiation protection provisions shall be such as to keep exposures as low as reasonably achievable (ALARA), taking into account economic and social factors.

(b) Radiation exposures of plant personnel and members of the public shall not exceed the prescribed limits.

2.2 Dose Limits during Operational States

The individual dose limits and constraints for plant personnel and the public during operational states are given in AERB safety manual on Radiation Protection for Nuclear Facilities [3]. The most important limits to be used for design purposes are given in Annexure-I.

2.3 Exposure Criteria for Accident Analysis

Design should demonstrate that the calculated doses to the members of the public at the site boundary under design basis accidents (DBE Category-4 events) [6] should not exceed the reference doses prescribed by AERB [3].

For arriving at the acceptable risk, the philosophy adopted is that events having higher probability of occurrence should have low consequence while events associated with high consequence should have low probability of occurrence.

2.4 Operational States

2.4.1 Design Approach

The design targets should be set in terms of the following:

- Radiation level at a specified distance from equipment/ components and general radiation fields in different areas of the plant. Suggested design radiation levels are given in Annexure-II.

- Limits of air contamination levels in different zones of the plant. In full occupancy areas of the plant, ventilation requirement should be governed by AERB Safety Directive [7] (Annexure-III).

- Minimising collective dose for plant personnel.
- Design target for collective dose (approved by AERB)
- Dose limits for members of the public (vide Annexure-I)

2.4.2 Radiological Protection of Plant Personnel

The design considerations for radiological protection should address the following:

(1) The plant layout should be such that

- Areas are segregated according to their radiation levels and contamination potential,
- The dose received by plant personnel during operation/ maintenance is minimised, and
- Shielding and ventilation provisions are adequate.

(2) Minimisation of individual exposure and collective dose shall be ensured by:

(i) Reduction of dose rate in working areas by:

- source reduction,
- adequate shielding,
- remote handling techniques including robotics, and
- periodic decontamination of active systems.

(ii) Minimisation of occupancy in radiation areas by:

- conducting time and motion study for different operations,
- use of equipment with low failure rates,
- ensuring ease of maintenance or removal/replacement of equipment,
- use of CCTV to minimise personnel entries,
- provision of stand-by equipment,
- separation/segregation of radioactive/non-radioactive equipment, and
- ensuring ease of access and good lighting.

(iii) Minimising the number of workers for a particular job

(3) Fractions of target occupational collective dose should be budgeted to individual systems, to areas of the plant and to operational, maintenance and inspection work functions based on design and past experience. During plant design, estimates should be made of the following.

- contribution of each system to radiation exposure,
- additional contribution to radiation exposure by nearby piping/ equipment,
- contribution by airborne activity to radiation exposure, and
- dose received by different groups of workers, to identify the groups of workers who are likely to receive significant dose on routine basis. e.g. fuel handling group in PHWRs.

2.4.3 Limitation of Public Exposure

The sources contributing to generation of radioactive solid, liquid and gaseous wastes and their release to the environment shall be examined with respect to minimisation of waste at the source. The dose to public resulting from these releases shall be assessed and if necessary, appropriate design measures to reduce these releases should be introduced.

2.5 Accident Conditions

2.5.1 In order to achieve the design objectives and criteria for accident conditions outlined in Section 2.3 the following features should be incorporated.

(a) High level of safety and defence-in-depth provisions in system design

(b) Engineered Safety Features (ESFs) and containment for mitigation of consequences

(c) Proper siting of the plant and provision of an exclusion zone as stipulated by AERB [8]

Safety analysis shall demonstrate that these objectives are achieved by design.

2.5.2 To enable plant operators to adequately manage accident situations, necessary design provisions and procedures shall be made for the following. (Also see Section 6)

(a) Radiation levels in control room and supplementary control room and all other locations requiring personnel access should be kept low. Suggested levels are given in Annexure-IV.

(b) Adequate provisions should be incorporated to ensure personnel protection against airborne contamination during emergencies.

(c) Control room and supplementary control room and other areas requiring personnel access should also be designed to provide personnel protection against chemical hazards, (e.g., H_2S, chlorine) as applicable.

(d) Provisions for safe collection/analysis of samples from process systems and air/ water from within the containment.

2.6 Monitoring for Radiation Protection

2.6.1 For demonstrating compliance with prescribed limits and for providing information on changes in radiation levels, installed radiation monitoring systems should be provided in the design.

2.6.2 Provisions should be made for workplace and environmental monitoring. Facilities should be provided for individual external and internal monitoring of the occupational workers. Workplace monitoring should include monitoring for external radiation, airborne contamination and surface contamination. Monitoring for protection of public should include monitoring of effluents from the plant as well as the environmental monitoring during normal operation as well as during accident conditions.

3. RADIATION SOURCES DURING OPERATIONAL STATES

The significant sources of radiation during operation and shutdown conditions are given in Annexure-V, Table V-1. Radiological data pertaining to commonly encountered radionuclides are given in Annexure-V, Table V-2. The observed levels of activities as seen in some of the important systems in currently operating 220 MWe units are given in Annexure-V, Table V-3.

(i) Long-lived fission products and activation products contained in reactor systems give rise to a general increase of radiation levels in the entire circuit and in addition cause radioactive "hot spots" in certain parts of the circuits. Both these are of radiological safety concern particularly during periods of high personnel occupancy such as maintenance outages. While gamma radiation emitted from these sources gives rise to external hazard, beta emitting sources cause additional external exposure (skin exposure) and internal exposure hazards (due to airborne contamination) when active system/ equipment are opened up (for maintenance) or when there are leaks from the systems. Even in the initial stages of plant operation the PHT system contains some fission product contamination mostly due to tramp uranium arising from surface contamination of (typical value 1×10^{-2} mg/cm^2) fuel cladding surfaces and uranium contamination present as impurity in the zircalloy materials [9]. However, much higher amounts of uranium and fission products are also released if there are fuel defects/ failures and this constitutes the major source of contamination in the PHT circuit during later stages of operation. Optimum fuel irradiation and fuel handling programme should ensure that fuel failure rate and thus the fission product contamination of primary coolant are kept low. Suggested fuel performance targets are given in Annexure-VI.

(ii) Cobalt-60 is the most significant long-lived activation product, both in the PHT and moderator systems. At the design stage the choice of components/ materials used in all the reactor system circuits should ensure that cobalt impurity levels are as low as possible. Also to the maximum extent possible, the use of cobalt bearing materials (particularly stellite) should be avoided and these should be replaced by cobalt-free substitutes. A note on the control of activation product Co-60 and suggested limits of cobalt impurity levels in different materials are given in Annexure-IX.

(iii) The long-lived activation product C-14 produced in fuel, moderator and PHT systems remains confined mostly in the fuel and the spent ion-exchange resin of the purification systems. Thus for NPP operation it is considered to be of minor importance as contributor to occupational or environmental dose. However C-14 releases would be of significance during reprocessing of irradiated fuel [10, 11].

(iv) The other active systems as listed in Annexure-V, Table V-1 (other than Miscellaneous Sources) basically contain those radioactive sources, which originate in the reactor core, the PHT and/ or moderator systems. The levels of activities in these systems depend upon factors such as their concentrations in the original systems, rate of activity transport, extent of escape of active fluids from the systems, etc.

(v) The process water and boiler feed water/ steam systems are normally inactive. However, these systems can become contaminated in the event of ingress of active system fluids due to factors such as leaks of heat exchanger or steam generator tubes. The activities in both these secondary systems should be monitored and as soon as activity is detected, corrective action to isolate and stop the leak are to be initiated. Provisions to clean and flush the circuit and safe disposal of active water generated due to leak to the secondary circuit should be available. During the course of investigation/ corrective action subsequent to detection of a leak, some amount of activity would have to be permitted to be present in the circuit containing the offending heat exchanger or steam generator in order to permit identification of the leaky equipment. Guidelines on arriving at the target maximum activity concentration in the offending system are given in Annexure-VIII.

4. RADIATION SOURCES UNDER ACCIDENT CONDITIONS

(i) The safety analysis of the plant should determine the source term, i.e., the amount of radioactive material that is likely to be released to the environment under different design basis events (DBEs). The DBE scenarios analysed should be comprehensive [12, 6, 13] and should include releases from:

- reactor core during loss of coolant accidents,
- fuel handling accidents, and
- coolant or moderator circuits, due to major leakage of system heavy water.

(ii) The analysis should include calculation of:

- radioactivity inventory in the reactor core,
- tritium and fission product inventory in moderator and coolant,
- fraction of fission product inventory released to coolant and containment, and
- fraction of inventory released to environment.

(iii) Fission products are generally of major importance as compared with activation products and actinides in the determination of activity released from fuel. However, tritium exposure should also be taken into account in calculating on-site doses in case of large heavy water leakage accidents.

(iv) In determining the source term that my result in exposure to the public the following should be taken into account:

i) Fission products and activation products released from the fuel and coolant

ii) Transport of released radionuclides from fuel/ core to environment taking into account the thermal hydraulics of core, coolant and containment for the accident sequence under consideration. Approved computer codes specific to PHWRs should be used for estimating source terms for different DBEs. Alternatively conservative methods of assessment of transport of fission products may be used.

The assumptions as well as the retention factors used for calculating source term in PHWRs are given in Appendix-A.

(v) Account should also be taken of the possibility of radioactivity accumulating on air filters/adsorbers or components of the liquid waste treatment system after accidents. In addition, the following aspects should be considered during the design of the filters/ adsorbers.

- use of demisters before air filters,
- shielding of filters, and
- heat generation due to decay of accumulated activity.

5. PROTECTION OF PLANT PERSONNEL DURING OPERATIONAL STATES

5.1 Plant Layout

The design of plant layout should take into consideration personnel access required for operation, calibration, inspection, and maintenance including replacement of equipment in radioactive areas of the plant. The layout should facilitate these tasks besides limiting the exposure of plant personnel and spread of contamination. For this the following need to be provided:

- shielding,
- area segregation and zoning system,
- change room facilities,
- access control,
- appropriate ventilation arrangements,
- equipment handling facilities,
- remote handling facilities, and
- decontamination facilities.

The auxiliary facilities required for radiation protection at the plant are given in Section 13.

5.2 Supervised Areas and Controlled Areas

5.2.1 Design shall provide appropriate isolation, by fencing off of all the important facilities of the operating plant, including the active areas of the power station from the rest of site facilities. This fenced off area, called Operating Island, should have entry only for authorised personnel.

5.2.2 To permit effective control over personnel access to radiation areas, the station layout should establish supervised areas and controlled areas according to their radiation exposure potential [3, 5]. In PHWRs these areas comprise the following different zones (vide section 5.3)

Supervised areas	-	Zone 1 areas.
Controlled areas	-	Zone 2, 3 and 4 areas

The use of personal dosimeters should be mandatory in the controlled areas. There should be only single point entry/ exit to the Operating Island (except emergency exits).

5.3 Zoning

5.3.1 General

In order to minimise contamination and also to control its spread, the entire plant area should be divided into distinct zones of different contamination potential. Each zone should be clearly demarcated and provided with interzonal check points equipped with contamination monitors (vide Section 11.3.7). Personnel movement in the plant should be from zones of lower to higher contamination potential for entry and vice versa for exit. At the final exit point (zone 2 to zone 1 interface) automatic portal surveillance monitors should also be installed.

5.3.2 The guidelines for classification of different zones are given in Annexure-IX.

5.3.3 Space/ facilities for setting up rubber stations should be provided at all locations where there is contamination potential. Additionally, highly contaminated areas should be provided with rubber change stations (double rubber stations).

5.3.4 Provisions should be made for storing protective clothing/ equipment and materials required for setting up rubber stations. Facilities for personnel decontamination should be available at important locations (such as just outside reactor building) to control spread of contamination.

5.4 Change Room

A change room should be provided at the entrance of zone-2 (controlled area). It should have the following facilities:

- separate lockers for plant and personal clothing,
- personnel decontamination facilities (showers and wash basins),
- clean plant clothing storage crib and issue counter,
- containers for contaminated clothing, and
- contamination monitoring facilities.

The layout and ventilation of the main change room should be such as to prevent the spread of contamination to areas outside the controlled areas. Within this room, a barrier should clearly separate the clean areas from a potentially contaminated area. The change room should have adequate capacity to cater for normal and peak loads of manpower.

5.5 Access and Occupancy Control

5.5.1 Areas within the station shall be divided into:

i) Shutdown Areas: Areas, which are accessible only during reactor shut down (or very low power), as at high power these areas may have very high radiation fields. Examples of such areas are PHT circulating pump and steam generator room, moderator room, etc.

ii) Accessible Areas: Areas, which are accessible at all times. These areas should have only low levels of radiation such that the exposure of personnel required to work here is not expected to be high.

The Accessible and shutdown Areas should be clearly marked. The shutdown areas should have doors and interlock systems to ensure that inadvertent personnel entry into these areas is prevented by design. Wherever not feasible, entry to locations of high radiation exposure potential (such as Fuel Transfer Room) should be restricted by strict administrative control measures. The design should incorporate appropriate provisions to ensure that personnel located inside shutdown areas exit from the area at all power levels.

5.5.2 The personnel occupancy time required in radiation and contamination areas should be consistent with the ALARA principle. This should be achieved in the plant layout by:

i) Provision of clear passageways of adequate dimensions for easy movement of personnel and equipment. The routes should be as short as possible so as to minimise radiation exposure and possibility of spread of contamination.

ii) Provision of adequate space in the vicinity of plant equipment for ease of working; for example, to carry out repairs or inspection, including removal of a plant equipment.

iii) Mounting of components that require frequent access at a height convenient for working and provision of permanent ladders, access platforms and crane rails (or cranes) in areas where it can be foreseen that these are required for maintenance or removal of plant equipment.

iv) Provision of facilities for quick and easy removal of shielding and insulation in locations where it may be necessary to perform routine maintenance or inspection.

v) Provision of special tools, equipment, robotics, remote handling devices, etc., in order to reduce exposure time.

vi) Provision of facilities for communication with the control room and between personnel working in radiation or contaminated areas.

5.6 Control of Activity in Coolant and Moderator Systems

5.6.1 The circuits associated with these systems contain most of the sources of radioactivity and are therefore the main contributors to personnel exposure.

5.6.2 The inventory of activated corrosion and erosion products in the systems should be minimised by:

i) reducing the corrosion and erosion rates by proper selection of materials and control of system chemistry, and

ii) minimising the use of cobalt bearing materials (e.g stellite for valve seats, bearing, etc.) and specifying materials, components and piping with low cobalt impurity content (vide Annexure-VII).

5.6.3 The activity build-up in these systems should be minimised by the following:

i) Fuel failures should be controlled by proper design and in-core management of fuel. On-line failed fuel detection system should be installed to enable early detection of failed fuel. (Limits for fuel failures and PHT system fission product activity are given in Annexure-VI).

ii) Clean-up systems (e.g., filters and ion exchange resin) should be provided. Their capacities should be adequate to cope with activity spikes during start-up, cool down and depressurisation phases.

iii) The possibility of build-up/trapping of activity in the system circuits should be reduced by the following:

- Sharp bends and dead ends that can act as traps should be avoided as far as possible.
- Number of welded joints should be kept to a minimum and the welds should have minimum roughness.
- Drains should be minimised and these should be properly positioned to avoid residual stagnant pockets of liquid when the circuits are drained.
- System tanks should have provisions for flushing and draining (to reduce crud/sludge build-up).
- Facilities for carrying out system decontamination (whenever required) should be provided. Design should ensure feasibility to carry out decontamination campaigns in these systems whenever required.
- Vent lines should, as far as possible, run vertically from the pump bowl.

5.6.4 To control air contamination (particularly tritium) the circuits should be designed for maximum leak tightness by:

- minimising the number of valves and components in the circuit, and
- use of leak tight valves (such as bellow-sealed valves).

5.7 Spent Fuel Storage Bay

Fuel storage bay water should be maintained at a low activity level by means of a cleanup system employing particulate filters and ion exchange resins. Experience has shown that a turnover rate for water of the order of once a day will keep the water clear and the activity at an acceptable level. (Normally the activity in the spent fuel bays should be maintained below 20 GBq/m^3 (500 mCi/l) for tritium and 2 GBq/m^3 (50 mCi/l) for gross beta activities).

5.8 System Design

In order to keep radiation exposures ALARA the following design features should be incorporated:

i) As far as possible standby equipment should be provided for active systems.

ii) Workspace around pumps and valves in high radiation zones should be shielded from radiation emanating from active equipment to ensure low radiation fields.

iii) As far as possible, indicators, auxiliary units, drive units, control equipment and other non-radioactive equipment, which do not have to be mounted close to active components, should be installed only in low radiation areas.

iv) For sampling active liquids from systems, remote techniques should be employed. All sample points should be provided with drip trays draining to active waste systems. Sample stations should have ventilated hoods.

v) Pipe runs of active systems should be short. It should also be ensured that active pipelines are not routed through high occupancy areas.

5.9 Component Design

5.9.1 While the choice and design of components is mainly dependent upon system requirements, they should have high reliability, requiring minimum maintenance.

5.9.2 Means for quick installation/ removal of components and whole units in high radiation zones should be provided to reduce exposures of workers. (examples: quick disconnect couplings, reduction in the number of hold-down nuts or bolts for installation of jig saw panels on the E-face to the minimum necessary.)

5.9.3 Components, which come in contact with active liquids/ gases, should be designed for ease of decontamination. For this, the following should be incorporated:

- provide smooth surfaces,

- avoid angles and pockets (to minimise hot spots), and
- provide means for isolation, flushing and draining of circuit.

5.10 Remote Handling

To the maximum extent practicable, the initial design should incorporate use of remote techniques for working in high radiation areas. These techniques should include arrangements for remote inspection and removal/installation of equipment. Some of the jobs may be semi-remote as these involve personnel entry in active areas for installation of equipment or rigs followed by remote testing operation. Such jobs should be optimised to achieve net reduction in total exposure. Examples of jobs where remote techniques should be used are ultrasonic inspection of welds, inspection of heat exchanger and steam generator tubes, coolant channel creep measurements, garter spring location, collection of system samples for chemical and radioactivity analysis, etc.

5.11 Decontamination

5.11.1 Requirements for the decontamination of equipment, piping, surfaces, etc., that are likely to have radioactive contamination should be considered at the design stage itself. All the components of the systems, which come in direct contact with primary coolant or moderator, should be considered as potential items requiring decontamination. Floors, walls, drains, etc., in the rooms, where active systems are located, which also get contaminated due to leakages or spills of active liquids, will require decontamination.

5.11.2 Depending upon the system to be decontaminated the following types of decontamination techniques may be adopted:

- decontamination of active components and equipment at their locations,
- transportation of contaminated equipment to a central decontamination centre, and
- on-line decontamination of circuit piping/equipment using suitable chemical decontamination techniques.

Design should provide adequate space and facilities for carrying out the decontamination operations.

5.11.3 The walls and floors of the rooms/areas containing active equipment/systems should have surfaces, which can be easily decontaminated (e.g., specially coated floors with curved edges between the floor and wall sloped towards local drains). In order to restrict spread of contamination, suitable bunds and leakage detection instruments (beetles) should be provided for localisation and timely detection of spills. Floors, those along high traffic routes, should have hardwearing surface paint.

5.11.4 The reactor building should be provided with vacuum mopping system to rapidly collect D_2O from floors subsequent to a major leak. Facilities/ equipment to safely collect D_2O spills (such as vacuumax) should also be made available at all vantage locations in the building.

5.11.5 Decontamination facilities to remove surface contamination from radioactive material transport containers, tools, equipment, system parts, etc., should be provided.

5.11.6 Drains of the decontamination facilities should be connected to active drainage system. The design should provide for periodic cleaning/ decontamination of any hot spots on the active drains.

5.11.7 Facilities for decontamination of personnel and protective equipment should be provided.

5.12 Shielding

5.12.1 Shield Design

To reduce radiation levels at all accessible locations of the plant to acceptable levels plant systems should be shielded against both gamma and neutron radiation. The shield design should take the following factors into consideration:

i) Criteria for radiation levels in full occupancy areas given in Annexure-III and Ref. [7].

ii) Source strength of both normal and transient sources and build up of long-lived activities during the lifetime of the station.

iii) Loss of shielding due to penetrations (such as pipes, cables, etc.)

iv) Choice of shielding materials should be made on the basis of shielding properties, mechanical properties and space and weight limitations. The shielding efficiency of materials should not be affected by environmental or process conditions.

v) Water-filled shields such as end-shield, should be provided with venting provision to ensure complete filling.

5.12.2 Temporary Shielding

Provision should be made in design for installation of temporary shielding (such as lead or concrete bricks, lead mats) for reducing radiation fields from active equipment, such as pipes, valves, actuators, heat exchangers, etc., during maintenance. The concerned area/location should have adequate space and the floor, the required load-carrying strength. Also, certain areas in the plant, such as service area, should be provided with shielding enclosures for temporary storage (for maintenance) of active equipment/ components.

5.12.3 Shield Penetrations

5.12.3.1 Penetrations are required in bulk shields for various purposes such as running pipelines or for entry of personnel/equipment. They introduce additional pathways along which streaming of neutrons and gamma rays can occur. Some design approaches for shield penetrations are given in Ref. 18.

5.12.3.2 In general the basic means, which should be used for minimising radiation streaming due to penetrations, are:

- Minimising the area and number of all straight-through paths containing material of very low density (e.g. gases, including air)
- Providing shield plugs
- Placing shields of larger diameter than those of the penetrations, to cover the ends of the penetrations
- Providing zigzag or curved pathways in order to ensure that adequate shielding is available along any line-of-sight path
- Filling the gap with grout or other compensatory shielding material.

5.13 Ventilation

5.13.1 To maintain appropriate ambient conditions (both temperature and air contamination control) in working areas, ventilation system shall be provided in all active areas of the plant.

5.13.2 The ventilation system design should ensure that air contamination levels in plant areas are maintained ALARA. In full occupancy areas the levels shall be kept below 1/10 DAC [7]. The required number of air changes in partial occupancy areas should be arrived at by taking into account the air contamination potential and the radiation levels specified in the different areas (vide Section 2.4.1 and Annexure-II).

5.13.3 The airflow should be directed from regions of lower to higher contamination potential. Care should be taken in design to reduce possibility of spread of air contamination in case of power failure.

5.13.4 To reduce tritium-in-air contamination and to recover D_2O from room atmosphere a closed loop ventilation system with dryers and coolers along with a small purge flow should be employed.

5.13.5 Ventilation exhausts of areas such as Fuel Transfer Room and Spent Fuel Storage Bay, where there is possibility of air contamination due to particulate and iodine activities, should be filtered using suitable particulate and iodine filters.

5.13.6 Temporary local exhaust should be made available in areas where airborne contamination may arise during maintenance. Such exhaust should be discharged into the active ventilation exhaust system.

5.13.7 The vents of tanks containing radioactive fluids (e.g., D_2O storage tanks or leak collection tanks) should be led to the active exhaust ventilation system.

5.13.8 Adequate number of mask-air supply points should be available at all active areas such as FT room, FM vault, Pump Room, SFSB.

5.14 Solid and Liquid Waste Management

5.14.1 The design of the solid waste management plant shall incorporate adequate radiation protection and ALARA measures during all waste handling/ management operations. Details on solid waste management plant design are given in AERB safety guide on Solid and Liquid Waste Management (AERB/ SG/D-13) [19].

5.14.2 For the assessment of radiation hazard in the waste management plant the sources which generate high radiation levels (such as ion-exchange resins, discarded radioactive components, used filters, etc) should be considered. Account should also be taken of possible increase in specific activity of waste residues as a result of treatment (such as incineration and compaction).

5.14.3 The radiological protection provisions in the waste management system design should also take into account the periodic refurbishing operations such as en-masse coolant tube replacement, etc.

5.14.4 Since on-load failed fuel removal capability enables PHT system activity to be kept low, activity concentrations in liquid waste arising in PHWRs are normally not high. The most significant activity is tritium. The design requirements for liquid waste management are given in Ref. 19.

5.14.5 The design should also take into account off-normal conditions that give rise to high volumes and activities in liquid waste management plant. (vide Section 7.2.2)

5.14.6 The main sources of high levels of activity which require particular attention are:

- Spent resin fixation/ transfer,
- Concreting for fixation of sludges/concentrates/incineration ash,
- Filters and ion exchange columns used in the treatment, and
- Sludges and cruds deposited at bottoms of waste hold up and disposal tanks.

5.14.7 The design of liquid waste management systems shall take account of all the applicable requirements of this guide for protection of site personnel such as, designation of zones, requirements for system and component design, provision of shielding and use of remote techniques. The waste management systems are provided primarily to minimise radioactive releases to the public domain. Further details of requirements for their design are given in Section 7.2.

5.14.8 Liquid effluent discharge point to the receiving water body should be chosen to ensure adequate dilution and prevent back-flow of discharged effluent to the plant water intake. The monitoring point should be established where representative sample is obtained.

5.15 Gaseous Waste Management System

The sources of activity in the gaseous wastes are given in Table V-1 & V-3 of Annexure-V. Since only moderate amounts of activities are present, these wastes are directly discharged to atmosphere after filtration through HEPA filters. To avoid the possibility of outleakages, the ducts carrying gaseous waste should be designed to be leaktight.

6. PROTECTION OF PLANT PERSONNEL UNDER ACCIDENT CONDITIONS

(i) An assessment shall be made of the strength and locations of the radiation sources that might exist during and after all categories of design basis events (particularly under Category-4) defined in AERB safety guide on Design Basis Events (AERB/SG/D-5) [6]. These data should be considered in the design of the nuclear power plant. Suitable provisions should be built into the design to ensure that personnel can have access to and occupy essential locations such as the main control room in order to operate and maintain essential equipment. This will call for the provision of the following [9, 12]:

- Adequate shielding of the containment building (vide Section 2.5.2)
- Room sealing and air cleaning provisions for control room (survival ventilation system)

In general, provisions should be made for automatic or remote controlled equipment.

(ii) There should be provision for remote collection, transfer and analysis of gaseous and liquid samples from the containment during accident conditions without incurring excessive individual exposures.

(iii) There should be provisions for alerting and assembling all site personnel including those not involved in accident control or fire fighting. Communications are required between the main control room, supplementary control room and emergency assembly areas and emergency control centre. The assembly areas should be chosen such that the radiation background is expected to be low.

(iv) Provisions should be made for easy identification of rooms with clearly marked signs and free passageways to enable quick movement of site personnel.

(v) All other facilities required as per the On-site Emergency Plan for the plant should be provided [20].

7. PROTECTION OF THE PUBLIC DURING OPERATIONAL STATES

7.1 Exposure Pathways

The following exposure pathways should be considered for evaluating public exposure:

- Exposure due to liquid effluents (mainly internal exposure) through the aquatic route
- Exposure due to radioactive gaseous effluents (external plus internal dose) through the air route
- Exposure resulting from solid waste disposal through the terrestrial route

The ALARA principle should be applied while releasing radioactive effluents from the plant and the authorised limits of dose to the critical group (residing beyond the exclusion distance) should not be exceeded (see Section 2.2).

7.2 Radioactive Waste Management Systems

Flow rates and radioactivity concentrations of the liquid and gaseous effluents shall be monitored and controlled to remain within authorised limits and further to meet the requirements of ALARA.

7.2.1 Gaseous Wastes

i) For removal of radioactive materials in gaseous effluents appropriate systems e.g. filters (both pre-filter and HEPA filters for particulates), dryers (for tritiated water vapour) and charcoal adsorbers (for radioiodines) should be provided as necessary. The performance of filters/ adsorbers should be such as to achieve the effectiveness required by the safety analysis as a minimum. Design should provide for a stack of adequate height for release of airborne effluents to achieve adequate dilution both for normal and accidental/ abnormal releases.

ii) Continuous micrometeorological measurements (wind speed, direction, vertical temperature profile, etc.) shall be carried out at the site meteorological laboratory. These data should also be available at plant control room and these should be used for assessment of radiation exposures in the public domain resulting from the release of radioactive materials through air route. For details see AERB safety guide on Release of Airborne Radioactive Materials in PHWRs (AERB/ SG/D-14) [21].

7.2.2 Liquid Wastes

Major sources of contaminated water requiring treatment include decontamination facilities for plant equipment/tools and for the spent fuel shipping cask, laundries and change room showers, waste water resulting from dedeuteration of spent ion-exchange resins, primary circuit leakage, and effluents from chemistry laboratories. Waste water may also occasionally originate from spent fuel storage bays, suppression pool, failed heat exchangers and steam generators tubes etc. The capacity of the liquid management plant should take all these sources into account. Suitable waste treatment methods (such as chemical co-precipitation, ion exchange, reverse osmosis and dilution) should be used to treat the liquid waste before disposal. For details refer AERB safety guide on Liquid and Solid Radioactive Waste Management in PHWR based NPPs (AERB/SG/D-13) [19].

7.2.3 Solid Wastes

The packaging, activity categorisation, choice of facilities used for disposal (such as earth-trenches, RCC vaults, tile holes, high integrity containers) and the location/characterisation of the solid waste disposal site used by the concerned NPP should be such that the migration of radioactivity under all conceivable conditions is kept to a minimum. Analysis and monitoring should demonstrate that the dose to the public resulting from this (terrestrial) route of exposure is kept below the limits stipulated by AERB. Likewise, installation of waste incinerator may be considered for volume reduction of radioactive combustible waste. The incinerator, if used, should have appropriate air cleaning systems to ensure very low levels of airborne radioactivity discharges having a negligible impact in the public domain. For details see AERB safety guide on Liquid and Solid Radioactive Waste Management (AERB/SG/D-13) [19].

8. PROTECTION OF THE PUBLIC UNDER ACCIDENT CONDITIONS

(i) The design measures taken to protect the public against accidents should have the twin objectives of reducing the probability of their occurrence as well as the consequences. (Also see Section 2.3)

(ii) The plant shall be provided with an exclusion zone of radius as stipulated by AERB, where no public habitation shall be permitted [8].

(iii) The design should be assessed by means of a safety analysis as given in the safety analysis report of the plant. Safety provisions in the design should be such that under design basis event (DBE) conditions, the exposure of a member of the public at the site boundary does not exceed the reference doses specified by AERB. (vide Section 2.3)

(iv) Design measures, which should be used to achieve reduction in radiation exposure in public domain during accident conditions, include the following.

- Containment system for control of activities released in the reactor building (AERB/SG/D-21) [22]
- Closed loop system for process water and steam and feed water circuits
- Provision of adequate storage capacity for liquid waste generated during accident conditions
- All the facilities required as per the off-site emergency plan (AERB/SG/O-6 [23] and Ref. 24)

9. GUIDELINES FOR DETERMINING RADIATION DOSE RATES DURING OPERATIONAL STATES

(i) The design should adopt validated methods for the assessment of radiation dose rates on plant equipment/systems that are expected under normal operation.

(ii) The first step in any calculation of dose rate is to evaluate the source strength and its distribution. This may involve calculation of the core activity and the transport and redistribution of activated corrosion products or fission products carried in reactor coolant and deposited away from the point of origin. The second step is to calculate the fluence rate (flux) at point of interest as a result of radiation transport from the source to the point of interest. The final step is to calculate the radiation dose rate by multiplying the radiation flux by the appropriate conversion factors.

(iii) The methodology adopted for calculation of radiation sources in the reactor and associated radiation transport through shielding are given in Annexure-X.

10. CONSIDERATIONS FOR DETERMINING POTENTIAL DOSES UNDER ACCIDENT CONDITIONS

(i) In order to show compliance with the design target (see Subsection 2.3) the potential consequences of the design basis accidents shall be determined.

(ii) Generally only the atmospheric releases are evaluated for accident conditions since a release of large quantities of radioactivity to water bodies is unlikely as most of the active liquid waste generated can be contained in the reactor building. Further, liquid wastes can be diverted to suitable storage facilities and can be released in a controlled manner after treatment or dilution. This should be evaluated for each plant.

(iii) During accident conditions, the releases from the plant will be partly at the ground level and partly at an elevated level (through stack). The methodology of calculation of source term and releases has been given in AERB safety guide on Containment System Design (AERB/SG/D-21) [22] (Also vide Section 4(iv)). The doses resulting from these releases should be estimated for the following:

- plant and site personnel
- members of the public (living beyond exclusion distance)

In addition, the dose to site personnel due to direct radiation from radioactivity present in the containment should be evaluated.

(iv) The design shall demonstrate that the radiation conditions within the plant will permit safe occupancy of plant personnel in the control room and other vital locations (also vide sections 2.5.2 & 6(i). Such evaluations should include tritium exposures subsequent to large heavy water spillages.

(v) For assessing environmental radiation doses during accident conditions the following main radioactive sources released from fuel should be considered.

- Fission product noble gases (FPNG)
- Radioiodines
- Radio Caesium, Strontium, etc.

(vi) The following main exposure pathways should be considered for determining dose in public domain.

- External dose from the plume and submersion dose from airborne effluents
- Inhalation dose due to intake of contaminated air

- Ingestion dose due to consumption of contaminated food stuffs
- External dose from ground contamination.

(vii) The methodology of calculation of exposures to members of the public due to atmospheric releases in public domain is given in Ref. 21 and 25.

11. MONITORING OF RADIATION DURING OPERATIONAL STATES

11.1 General

11.1.1 For an effective implementation of the design provisions for radiological protection of site personnel and the public, the following monitoring programme shall be implemented. (Also see section 2.6)

i) personnel monitoring

ii) area monitoring

iii) process monitoring

iv) effluent monitoring

v) environmental monitoring

In NPPs most of the radionuclides encountered are beta or gamma emitters. Hence the radiation monitors are predominantly for measurement of beta and gamma radiation.

11.1.2 Equipment for implementing this programme shall be provided. A typical list of different monitoring systems is given in Annexure XI, Table XI-1. The rationale for the selection of ranges, alarm set-points and locations for the monitoring systems should be documented. Adequate reliability/ redundancy of the equipment should be ensured. Some important features of the monitoring systems are given in Annexure XI, Table XI.2.

11.2 Personnel Monitoring

11.2.1 Personal monitoring is required to measure the dose (both external and internal) received by individual workers. For details refer AERB safety guide "Radiation Protection during Operation of NPPs" (AERB/SG/O-5) [5].

11.2.2 The personnel monitoring arrangements should include

- External dose measuring devices (TLDs, DRDs, extremity dosimeters, neutron dosimeters, etc.)
- Internal dose measurement techniques (bioassay, WB counting)
- Personal contamination monitors (friskers, hand/ foot monitors) and portal surveillance monitors (see also section 11.3)

The personal monitors measure the dose received by the person due to beta, gamma and neutron radiations.

11.2.3 Additional information on the monitoring of individuals is given in AERB/SG/O-5 [5].

11.2.4 A computerised personnel dose management system should be provided at the plant.

11.3 Area Monitoring in the Plant

11.3.1 Area monitoring is required to provide information on radiation fields, air activity and contamination levels in the different areas of the plant.

11.3.2 Area monitoring system shall include the following:

- Monitors for measurement of external dose rates
- Air contamination monitors
- Surface contamination monitors

11.3.3 The area radiation monitors should be installed at appropriate locations at the plant both in controlled and supervised areas. The suggested ranges are given in Annexure-XI, Table XI-3.

11.3.4 For monitoring special maintenance operations of short duration and for monitoring in areas where potentially high fields may occur, portable or mobile radiation monitors shall be provided with built-in alarms for dose rates exceeding the pre-set values.

11.3.5 Installed monitoring systems shall be provided for detecting air contamination due to tritium, iodines and particulate radioactivities. The suggested locations for these systems are given below:

Tritium:	Ventilation exhaust ducts in reactor building areas
Iodine and Air Particulates:	Accessible areas of reactor building, fuel transfer room and spent fuel storage bay

In addition to installed monitors, portable or mobile air activity monitors or samplers should be used for detection of air activity at work locations.

11.3.6 Surface contamination monitors shall be installed at a few locations to enable checking of contamination status on the plant floors and equipment. Such monitoring is required to be done both during entries/ and work in areas containing radioactive system/ equipment.

11.3.7 Personal contamination monitors (both friskers and hand/foot contamination monitors) shall be installed at inter-zonal transition points for use by personnel before exiting from higher activity zones. Contamination monitors should also be installed at the entrance of rubber stations whenever they are set up. Portal monitors should be installed at the final exit point of the plant.

11.3.8 Gate radiation monitors should be installed at the plant exits/gates for detecting any inadvertent carry-over of activity/active material from the plant premises. All the installed monitors should have audio-visual alarms.

11.4 Process Monitoring

Activity monitoring in both gaseous and liquid systems is required for detecting system malfunctions and equipment/component failures, which may also result in higher radiation levels or contamination at the plant or excessive release of radioactivity to the environment. Suitable process gamma radiation monitors shall be installed at the following locations:

i) Ventilation exhaust ducts of reactor building, service building, decontamination centre, etc.

ii) Process water discharge lines of the PHT and moderator system heat exchangers (to detect failure of HX tubes)

iii) Steam lines (to detect failure of steam generator tubes).

Besides gamma activity these ducts/lines should also be monitored for tritium activity. The heavy water leak detection sensitivity will be low for this purpose during initial periods. However as tritium in system heavy water builds up with irradiation, the sensitivity of leak detection improves while the reactor operation proceeds. It may be noted that during reactor shutdown minute failures of heat exchanger or steam generator tubes can be detected only by tritium measurements.

11.5 Waste Monitoring

11.5.1 Monitors shall be installed for the continuous monitoring of the following airborne effluents released through stack:

i) Tritium

ii) Fission product noble gases (FPNG)

iii) Argon-41

iv) Iodine-131

v) Particulate radioactive materials

11.5.2 Liquid activity monitor shall be installed at the liquid effluent discharge line to plant out-fall. This monitor should have the sensitivity to detect activity concentration at the limiting level stipulated by AERB. The plant out-fall line shall also be provided with a continuous out-fall sampling system.

11.5.3 Suitable monitoring methodology should be provided for assessing activity content in solid waste before disposal.

11.6 Environmental Monitoring

11.6.1 Environmental monitoring is provided to check if there is any increase in radiation background due to the operation of the plant and also to provide an indication of any abnormal releases from the plant. Environmental surveillance data also give confirmatory evidence that the plant releases are within the prescribed limits. This monitoring is done in the following two areas.

i) On-site monitoring-Monitoring within the site boundary (up to exclusion distance)

ii) Off-site monitoring-Monitoring beyond the plant exclusion zone:

11.6.2 Environmental monitoring within the plant boundary should be done using the following equipment.

i) Continuous environmental monitors installed at certain locations around the plant. The range of these monitors should be adequate to cover normal operation and accident conditions. The reading of these monitors shall be available both locally and in the control room so that any accidental releases can be quickly detected.

ii) Thermoluminescent dosimeters (TLDs) installed around the plant.

iii) Borewell water sampling system around the radioactive waste storage area.

iv) Monitoring of storm water drains.

The number of these monitors should be adequate to cover all the directions. TLDs, being inexpensive, can be used in a large number of locations.

11.6.3 The off site (beyond exclusion zone) environmental surveillance programme shall be carried out by setting up an environmental survey and micrometeorological laboratory (ESML) near the plant at least 3 years prior to plant operation [5]. The laboratory should have equipment and facilities for detection/analysis of very low levels of radioactivity in the environmental samples [26]. This laboratory should also provide base line data for both environmental radioactivity and site micrometeorological parameters. The laboratory should be equipped to provide environmental radioactivity data during accidents and for subsequent accident management operations. Details of the requirements of ESML are given in safety guide on Regulations and Criteria for Health and Safety of Nuclear Power Plants Personnel, the Public and Environment (AERB/SG/G-8) [27].

12. MONITORING OF RADIATION UNDER ACCIDENT CONDITIONS

12.1 General

12.1.1 The accident detection and monitoring equipment provided shall be suitable for enabling the operator to assess accident conditions and take necessary corrective actions. The expected radiological conditions both within the plant and environment during the postulated accidents should be considered for the choice of sensitivity and range of the instruments. The instruments should be capable of satisfactorily performing under the worst environmental conditions anticipated during the accident [28].

12.1.2 The procedures used for radiological monitoring and selection of monitoring locations should be as per the emergency preparedness plan for the plant.

12.1.3 The emergency actions should be taken based upon the actual levels of external radiation and contamination levels in the air, surfaces and food chain (vide AERB/SG/HS-1) [29].

12.1.4 Suitable communication systems should be provided to enable information/ instructions to be transmitted between different locations, emergency survey teams and other emergency agencies.

12.2 Monitoring within the Plant

Instruments/equipment shall be available for the following measurements:

i) Radiation fields on reactor building exhaust ducts (for "boxing up" of reactor building in case of high activity).

ii) Radiation fields in the reactor building. Accident monitors should have measurement range upto about 10^4 Gy/h.

iii) Sample collection systems for determination of air contamination levels in the reactor building (for inert gases, particulates and tritium activities).

iv) Sample collection systems for determination of activity in the coolant (PHT) and suppression pool.

v) Measurement of different radionuclides released through the primary containment depressurisation system to stack.

The sample collection systems should be provided with shielding and shielded containers to safely handle and transport the active samples to the laboratory for analysis. Likewise, the laboratory should have provisions for handling and analysing these high active samples.

12.3 Monitoring outside the Plant and Environment

Besides the normal monitoring provided by environmental survey laboratory and micrometeorological laboratory, equipment and facilities shall be available for monitoring and initiating countermeasures and remedial actions during accidents. The radiation measurements are obtained from:

i) Environmental radiation monitors and TLDs located around the plant and population centres nearby (vide Section 11.6.2)

ii) Environmental survey vehicle equipped with equipment/ instrument for monitoring, sampling and analysing the environmental samples.

13. AUXILIARY FACILITIES

The plant design should include auxiliary facilities that are necessary for effective radiological control in the operation and maintenance of the nuclear power plant and for responding to emergencies [30]. In particular, facilities are required:

i) To limit the spread of contamination within the controlled area and to prevent the spread of contamination outside the controlled area.

ii) To carry out adequate monitoring of the workplace and individual monitoring

iii) To manage other health physics operations.

Annexure-XII gives typical lists of the facilities/ equipment that are required to be provided for establishing a radiation protection programme in a nuclear power plant.

APPENDIX-A

ASSUMPTIONS FOR SOURCE TERM CALCULATIONS

An example of the assumptions used for calculation of release of radioactive materials from fuel to containment and from the containment to environment during maximum DBE conditions is given here. The following considerations are applied. (Ref: safety guide on Containment System Design, AERB/SG/D-21).

A.1 Release into Containment

- 100% of core inventory of iodines and noble gases are assumed to be released from fuel. However, if justified by appropriate analysis, a lower value may be used.
- For accident scenarios involving loss of coolant accident (LOCA) with failure of emergency core cooling system (ECCS), a water trapping factor of 2 in PHT circuit is applied.
- For LOCA with availability of ECCS, the value of water/ air partition factor for iodines is assumed to be 2×10^5.
- All fission products are considered to be released instantaneously to the containment atmosphere. A more realistic time-dependent release can be accepted with proper justification.
- Appropriate particulate filters with adequate efficiency should be incorporated.

A.2 Factors for Reduction of Iodine Inventory During Transport in the Containment

- The following plate-out half-lives for deposition of iodines may be used.

Primary containment	-	1.5 h
Secondary containment	-	2.0 h

- Once the air concentration reaches 10% of original value, further plate out is considered to be no longer effective (due to organic iodines).
- A plate out factor of 10 is considered for all leak paths in the containment.

A.3 Reduction of Iodine through Engineered Safety Features (ESF)

The removal of iodine through ESF i.e., primary containment controlled discharge (PCCD), primary containment filtration and pump back (PCFPB) and secondary containment recirculation and purge (SCRP) should take into account filter efficiency and delay in operation of these systems. The following values of filtration efficiencies should be used.

HEPA filter (for 0.3 μ particulates)	-	99.9%
Charcoal filter (for iodines)	-	90.0%

A.4 Containment Leak Rate

Analyses should be performed based on the leakage rates as stipulated in technical specifications for primary and secondary containment respectively.

ANNEXURE-I

DOSE LIMITS FOR DESIGN PURPOSE

For design purposes the following effective dose limits shall be used:

Occupational workers	:	20 mSv/year
Members of the public	:	1 mSv/year

(for all radiation sources, present and future, at site)

For members of the public the dose should be restricted to the apportioned dose for the plant, which is a fraction of the 1 mSv per year resulting from all exposure pathways. The critical group and critical pathways should be identified by carrying out preoperational environmental surveys around the plant site. The dose will include both external and internal. For calculating the annual internal dose the committed effective dose resulting from intake of radioactive materials through inhalation and ingestion in that year should be taken into account. In order to comply with the apportioned dose limits the release of radioactive effluents should be kept below the authorised release limits.

ANNEXURE-II

DESIGN RADIATION LEVELS IN PLANT AREAS

S. No.	Area	Maximum Radiation Level (μSv /h)
(i)	Normal full time occupancy areas * (Supervised Areas)	1
(ii)	Reactor Building ** (Controlled Areas)	
	(a) Areas accessible during power operation (Accessible Areas)	
	Occupancy 8 h/d	5
	” 4 h/d	10
	” 2 h/d	20
	” 1 h/d	40
	(b) Area inaccessible during reactor operation (Shutdown Areas)	
	- General field during reactor shutdown	40
	- Areas with limited occupancy (70 h/y)	150
(iii)	Radioactive hot spots should be shielded to meet the above radiation levels.	
(iv)	There should not be any neutron field in the accessible areas.	

Notes:

* Ref. AERB Safety Directive 2/91 [7]

** The suggested radiation levels in other areas of reactor have been arrived at on the following basis:

- A dose of 10 μSv/y is allocated for external exposures. The full occupancy time is taken to be 2000 hours per year.
- Plant operating factor is taken as 75% and occupancy in shutdown areas is 50% of shut down time.

The shielding design at the Darlington GS, Canada, has been based on the following objectives:

Non-radiation area fields:	< 0.25 μSv/h
Average Accessible Area fields	< 6.0 μSv/h
Average Shutdown Area fields	< 40 μSv/h
Maximum Shutdown Area fields of frequent attendance (80 h/a)	< 170 μSv/h

ANNEXURE-III

(Extract of AERB Safety Directive 2/91)

AERB SAFETY DIRECTIVE 2/91

July 22, 1991

All future plants/ facilities including those under design shall be based on ICRP-60 which recommends that the dose constraint for optimisation should not exceed 20 mSv in a year for occupational exposures and 1 mSv in a year for the public.

a) The shieldings to be provided shall be such that the dose rates in full occupancy areas do not exceed 1 mSv per hour (0.1 mrem/h).

b) The ventilation designs shall be such that the air concentration of activities in full occupancy areas do not normally exceed 1/10 of the new derived air concentrations (DAC). The new DAC values can be obtained by dividing the annual limit on intake (ALI) values given in ICRP-61 by 2.4×10^{3}. These ALI values for commonly encountered radionuclides are given on the reverse.

c) All effluent discharges from a plant/ facility / practice shall be so controlled that the exposure of the critical group does not exceed the public dose limit of 1 mSv in a year (excluding natural background and medical exposures) from all practices at the site.

Sd/-
(S.D.Soman)
Chairman, AERB

Distribution:

MD, NPC
ED (O), NPC
Director, H&S, NPC
CS, TAPS
CS, RAPS
CS, MAPS
CS, NAPS
CMD, IRE
ED (P), NPC
Shri Ch. Surendar, Director (Engg)
NPC 500 MWe

Director, BARC
Director, RG, BARC
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Director, VECC, Calcutta
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ED, OPSD, AERB
Director, H&S Group, BARC
Head, HPD, BARC
Head, DRP, BARC
Head, RSSD, BARC
Secretary, AERB

ANNEXURE-IV

SUGGESTED MAXIMUM RADIATION DOSE RATES IN CONTROL ROOM AND OUTSIDE REACTOR BUILDING DURING ACCIDENT CONDITIONS

The suggested maximum radiation dose rates in control room and outside reactor building walls during emergency conditions are 0.5 mSv/h and 5 mSv/h respectively (Ref.9).

ANNEXURE-V

SOURCES OF RADIATION

TABLE V-1: SOURCES OF RADIATION DURING NORMAL OPERATION

(Source: These data have been summarised from Safety Analysis Reports of PHWRs, Annual Health Physics Reports of PHWR type NPPs and IAEA Draft Safety Guide Radiation Protection Aspects of Design for Nuclear Power Plants, DS 313, June 2003).

S.No.	System	Main Sources of Radiation Hazard
1.	**Reactor Core**	
	During operation:	(i) Fast and thermal neutrons and gamma photons from fission process (ii) Capture gammas (iii) Photoneutrons due to (γ, n) reaction in D_2O (iv) Fission products (v) Activation products
	During shutdown:	(i) Fission products in core and (ii) Activation products generated in core, fuel clad, pressure and calandria tubes, control rods, etc.
2.	**PHT**	
	During operation:	(i) Coolant activation products such as N-16, O-19, N-17, H-3 (ii) Fission products in coolant (iii) Photoneutrons (iv) Tritium and other beta emitters from system D_2O leaks
	During shutdown:	(i) Long-lived fission products (I-131, Cs-134, Cs-137, etc.) (ii) Activation and activated corrosion products (Mn-54, Co-58, Co-60, Fe-59, Zr-95, Nb-95, etc.) (iii) Tritium from system leaks and other β emitters
3.	**Moderator**	
	During operation:	(i) Activation products such as N-16, O-19, N-17, H-3 (ii) H-3 from system D_2O leaks

ANNEXURE-V (Contd.)

TABLE V-1: SOURCES OF RADIATION DURING NORMAL OPERATION (Contd.)

	During shutdown:	(i) Long-lived activation products (such as Zn-65, Co-60, Mn-54, Zr/Nb-95) (ii) Tritium from system leaks and other b emitters
4.	**Calandria Vault and End Shield Cooling System**	
	During operation:	N-16, O-19, N-17
	During shutdown	Small amounts of corrosion products such as Co-60, Zn-65, Fe-59
5.	**Waste Management System**	
	(a) Gaseous Wastes:	
	During operation:	(i) H-3, C-14 (ii) Fission product noble gases (mainly Xe-133) (iii) Ar-41 (mainly from calandria vaults in RAPS/MAPS and small amounts from annulus gas monitoring systems of other PHWRs) (iv) Small quantities of radioiodines and particulates
	During shutdown:	(i) H-3 and C-14 (ii) FPNG (mainly Xe-133 for a few days)
	(b) Liquid Wastes:	Fission and activation products (mainly H-3, I-131, Cs-134, Cs-137, Co-60, etc.)
	(c) Solid wastes:	Fission and activation products contained in different categories of waste packages
6.	**D_2O Cleanup Systems**	
		Long-lived fission and activation products corresponding to activities present in system concerned (PHT or moderator)
7.	**Spent Fuel (S.F) and S.F Handling Systems**	
	(a) Spent Fuel	(i) Fission and activation products contained in irradiated fuel (ii) Activation products in fuel bundle components
	(b) S.F Bay	Fission products released to pool water from failed irradiated fuel bundles (mainly I-131, Cs-134, Cs-137, etc.)
	(c) S.F Bay Cooling	Fission and activation products removed and Cleanup Circuit from pool water (filters/IX columns, HX, piping)

ANNEXURE-V (Contd.)

TABLE V-1: SOURCES OF RADIATION DURING NORMAL OPERATION (Contd.)

8.	**Decontamination Facilities**	Activity contained in equipment/ materials to be decontaminated and waste solutions (fission and activation products corresponding to equipment being contaminated)
9.	**Fresh Fuel**	Low level radiation from fresh natural UO_2 fuel, depleted uranium and thorium
10.	**Miscellaneous Sources**	Radiation sources (a,b,g and n) required for calibration of instruments and for reactor start up (neutron sources)
11.	**Annulus Gas System**	Ar-41 due to activation of air impurity in CO_2 (annulus gas)

ANNEXURE-V (Contd.)

TABLE V-2 RADIOLOGICAL PROPERTIES OF RADIONUCLIDES LIKELY TO BE ENCOUNTERED IN NPPs

(Source: Health Physics and Radiological Health Handbook, Shlein Bernard-Onley, 1984)

S. No.	Radionuclide	Half-life	Energy in MeV & (Yield %)	
			Beta	Gamma
1	H-3	12.4 y	0.0186 (100)	-
2	C-14	5728.8 y	0.157 (100)	-
3	N-16	7.13 s	1.55 (1), 3.31 (5), 4.288 (68)	2.74 (1), 6.13 (69), 7.13 (5)
4	O-19	26.9 s	-	0.109 (3), 0.197 (96), 1.36 (50), 1.44 (3), 1.55 (1)
5	Na-24	15 h	1.39 (100)	1.37 (100), 2.75 (100)
6	Ar-41	109.5 m	1.198 (99)	1.29 (99)
7	Sc-46	83.76 d	0.357 (100)	0.89 (100), 1.12 (100)
8	Cr-51	27.7 d	-	0.32 (10)
9	Mn-54	312.5 d	-	0.84 (100)
10	Fe-59	44.5 d	0.13 (1), 0.27 (45), 0.47 (53)	0.143 (1), 0.19 (3), 1.1 (57), 1.29 (43)
11	Co-58	70.86 d	-	0.51 (29), 0.81 (99), 0.86 (1)
12	Co-60	5.27 y	0.32 (100)	1.17 (100), 1.33 (100)
13	Cu-64	12.7 h	0.58 (37)	0.51 (36)
14	Ni-65	2.52 h	0.66 (28),1.02 (10) 2.14 (61)	0.37 (5), 1.12 (15), 1.48 (24)
15	Zn-65	243.91 d	-	0.51 (3), 1.12 (51)
16	Sr-90	29.12 y	0.55 (100)	-
17	Y-90	2.67 d	2.28 (100)	-
18	Kr-85	10.72 y	0.69 (99.5)	0.51 (0.4)
19	Rb-88	17.8 m	0.8 (2), 2.1(1), 2.58 (13), 3.48 (4), 5.32 (78)	0.9 (14), 1.38 (1), 1.84 (21), 2.68 (2)

ANNEXURE-V (Contd.)

TABLE V-2 RADIOLOGICAL PROPERTIES OF RADIONUCLIDES LIKELY TO BE ENCOUNTERED IN NPPs (Contd.)

20	Nb-95	34.95 d	0.16 (100)	0.77 (100)
21	Zr-95	63.98 d	0.37 (55), 0.4 (44), 0.89 (1)	0.72 (44), 0.76 (55)
22	Ru-103	39.5 d	0.70 (3%), 0.21 max	0.497 (88%), 0.610 (6%)
23	Ru-106	368.2 d	0.04 (100)	-
24	Rh-106	29.9 s	1.98 (2), 2.4 (10), 3.02 (8), 3.54 (79)	0.51 (21), 0.62 (10), 1.05 (2)
25	Sn-113	115.1 d	-	0.26 (2), 0.39 (65)
26	Sb-125	2.78 y	0.09 (1), 0.13 (2), 0.24 (2), 0.3 (40), 0.45 (7), 0.62 (4), 1.25 (6)	0.18 (7), 0.43 (29), 0.46 (10), 0.6 (22), 0.64 (11), 0.67 (2)
27	Te-131	25 m	1.1 (10), 1.15 (3), 1.37 (1), 1.65 (22)	0.45 (69), 0.45 (18), 0.49 (5), 0.6 (4), 1.15 (8)
28	I-131	8.01 d	0.25 (2), 0.33 (7), 0.61 (89)	0.08 (3), 0.28 (6), 0.365 (82), 0.64 (7), 0.72 (2)
29	Xe-133	5.29 d	0.35 (99)	0.08 (37)
30	Cs-134	2.07 y	0.09 (27), 0.47 (3), 0.66 (70)	0.56 (23), 0.6 (16), 0.61 (98), 0.8 (94)
31	Cs-137	30.14 y	0.51 (95), 1.17 (5)	0.032 (4), 0.662 (86)
32	Ba-140	12.8 d	0.45 (26), 0.57 (10) 0.99 (37), 1.01 (22)	0.16 (7), 0.3 (5), 0.42 (3), 0.44 (2), 0.54 (26)
33	La-140	1.68 d	1.24 (17), 1.28 (2), 1.3 (6), 1.33 (45), 1.4 (5), 1.68 (21), 2.16 (5)	0.33 (21), 0.43 (3), 0.49 (46), 0.82 (24), 0.92 (3), 0.93 (7), 1.6 (95)
34	Ce-141	325 d	0.581 max	0.145 (48%)
35	Ce-144	284.5 d	0.18 (20), 0.24 (5), 0.32 (77)	0.04 (8), 0.08 (2), 0.13 (11)

ANNEXURE-V (Contd.)

TABLE V-3: TYPICAL LEVELS OF ACTIVITY IN SOME IMPORTANT SYSTEMS OF OPERATING 220 Mwe UNITS

(after ~10 years of operation)

(Source: Data derived from Health Physics Quarterly Reports of PHWRs)

System	Activity
PHT	
H-3*	6.6-125 TBq/m^3
I-131	6.6-125 TBq/m^3
Gross-beta (other than H-3)	0.1-3 GBq/m^3
Moderator	
H-3*	112-703 TBq/m^3
Gross-beta (other than H-3)	0-0.6 GBq/m^3
Spent Fuel Pool	
H-3	0.5-10 GBq/m^3
Gross beta (other than H-3)	0.01-0.8 GBq/m^3

* Tritium concentrations in PHT and moderator circuits are theoretically expected to reach saturation values of 93-175 TBq/m^3 in PHT and 2.1-5 sGBq/m^3 in moderator respectively. However, these values are generally lower due to periodic make up of system D_2O.

ANNEXURE-VI

DESIGN FUEL FAILURE TARGETS

Based upon the operating experience, the following target design values have been proposed:

- Fuel failure rate: below 0.1%
- Maximum Iodine-131 concentration in PHT system (during steady state operation): 0.37 GBq/m^3

Note:

i) Fuel failure rate is defined as the percentage of suspected failed fuel bundles to the number of fuel bundles discharged during the year.

ii) Cumulative fuel failure rate is defined as the percentage of suspected failed fuel bundles discharged to the sum of the total number of fuel bundles discharged from the core so far and bundles in the core.

ANNEXURE-VII

CONTROL OF ACTIVATION PRODUCT COBALT-60

Co-60 ($T_{1/2}$ 5.3 years) is one of the most important activation products, which results in substantial dose to plant personnel, particularly during maintenance periods. It is formed by activation of Cobalt-59, which occurs as impurity in the reactor materials. Cobalt is also used in hard facing alloys (wear resistant). The use of such materials should be avoided and suitable substitute (alloys without cobalt) should be used. Some of the cobalt-free substitute materials are Pantanex 25, Actinit DUR 300, Nitronic 60, Tribaloy T 700, LC-1C, SS17-4 PH and Colmonoy 4. Several other proprietary materials have been found satisfactory substitutes [VII-1]. Also the cobalt impurity levels in the structural materials and in those used in heat exchangers, steam generators, etc., stainless steels, carbon steels, cupronickel, Inconel, Zircaloy, etc., should be kept as low as possible. The materials used in PHWRs in the important systems are given in Table Annexure VII-1. Based upon a review the suggested limits of cobalt impurity levels in the different materials are given in Table VII-2.

References:

VII-1 INTERNATIONAL ATOMIC ENERGY AGENCY “Design of Nuclear Power Plants to Facilitate Decommissioning” Technical Reports Series No. 382, STI/DOC/010/382, IAEA, 1997

VII-2 BHABHA ATOMIC RESEARCH CENTRE, “Dose Reduction in PHWR Plants through Selection of Materials at the Design Stage” Note prepared by V.K. Sharma, Health Physics Division, BARC (Jan 1999)

ANNEXURE-VII (Contd.)

TABLE VII-1: MATERIALS USED IN REACTOR SYSTEMS

System	Component	Materials Used
1. PHT	Incore:	
	Coolant Tube:	Zircaloy-2/Zr-2.5 Nb
	Fuel Clad:	Zircaloy-4/ Zircaloy-2
	Garter spring:	Zr-2.5 Nb-0.5 Cu alloy
	Steam generators:	Incoloy 800 with Inconel 600 clad for tube sheet
	End Fittings:	AISI-403 Martensitic SS
	Feeders and Headers:	Carbon steel
	Pump impeller and casing:	Cast SS
	Channel seal bellows:	Inconel 600
2. Moderator	Calandria vessel and internals:	SS Type 304 L
	Calandria tubes:	Zircaloy-2/ Zircaloy-4
	Heat exchangers:	Cupro-nickel/ SS316L
	Moderator pump:	SS
	Other equipment:	Austinitic SS
	Reactivity mechanism	Zircaloy-2, SS,
	Components (ball screws and bearings):	Waukesha-88, Stellite, 440 C/ 440B
3. Annular Gas Cooling		SS 304 L
4. Calandria vault Cooling System	Liner:	Carbon steel (zinc metallised), SS 304L
	HX tubing:	Cupro-nickel
	Piping:	SS in vault and CS elsewhere
5. End Shield Cooling System	Body:	SS 304 L
	Piping and steel balls:	CS
	HX tubing:	Cupro-nickel/ S.S

ANNEXURE-VII (Contd.)

TABLE VII-2: SPECIFICATION OF LIMITS OF CONCENTRATION FOR COBALT IMPURITY IN REACTOR MATERIALS

Material	**Content (ppm)**
Stainless steels - High flux areas [a]	 520
Zirconium alloys [b]	20
Steam generator tube materials [c]	150
Cupro-nickels [d]	500
Carbon steels [e]	260
Monel [f]	300

Notes:

(a) This includes calandria, end fittings, landed inserts, liner tube, end shield.
(b) This includes fuel clad, pressure tubes, calandria tubes, guide tubes.
(c) This applies to Incoloy 800 tubing and Inconol 600 components in the SG.
(d) This applies to heat exchanger tubes and components (From NAPS onwards cupronickel is not being used for nuclear HXs).
(e) This applies to all carbon steels used in PHT, end shield, vault cooling systems.
(f) This applies to steam generator tube materials for RAPS-1 & 2 and MAPS-1 & 2 only.

ANNEXURE-VIII

PROCESS WATER/ FEED WATER CONTAMINATION LIMITS

In order to assure that the stipulated daily discharge limits are not exceeded, the maximum activity concentration in the process water (PW) and feed water (FW) should be arrived at by taking into consideration the tritium releases during normal operation and the derived release levels (D_l).

The target maximum activity concentration, C_{max} (TBq/m^3), in the offending system should be arrived at by using the formula

$$C_{max} = \frac{D_l - 0.185}{L}$$

where, D_l (Bq/d) is the derived daily release limit for aquatic route and L (m^3/d) is the leak rate (make-up rate) from the system. (In Indian PHWRs the normal average value of aquatic route tritium activity discharges from PHWRs is about 0.185 TBq/d (5 Ci/d). The above formula is arrived at by assuming that all the water leaking out of the systems gets released to the liquid route.

In the event of such exigencies, it should also be ensured that the active process water spillages do not unduly contaminate the non-active areas of the plant (tritium-in-air contamination of these areas should be maintained below 0.01 DAC)

ANNEXURE-IX

GUIDELINES FOR CLASSIFICATION OF ZONES IN PHWR

The various areas in the nuclear power plant should be classified into four zones as per the following guidelines.

IX.1 Zone-1

(a) This zone contains no radioactive equipment and is kept free of contamination at all times.

(b) Typically it comprises control room, pump house, DG room, turbine building and switch yard.

IX.2 Zone-2

(a) This zone contains no radioactive source and should not normally become contaminated. However some contamination may, though inadvertently, spread into this area with the movement of personnel and equipment from Zone-3. Contamination in Zone-2 should be cleaned up as soon as it is detected.

(b) Typically this zone includes inactive workshops (mechanical, electrical, control), laboratories, change room, wash room, etc.

IX.3 Zone-3

(a) This zone contains contained sources of radiation.

(b) It includes the service areas for active equipment and materials that are potential sources of contamination when they are opened up. Equipment layout and work procedures should be planned to keep contamination localised and loose contamination should be cleaned up whenever it occurs.

(c) Shops and laboratory (Bio-assay Lab and Chemical Lab) areas handling contaminated equipment and sources fall in this zone. Reactor auxiliary building, some parts of decontamination centre, laundry, waste management facility, etc., are also part of this zone.

IX.4 Zone-4

(a) This zone normally contains open sources of radiation and contamination.

(b) Contamination in this zone is kept localised and under control by routine clean up operations to the maximum extent feasible; but some parts may remain contaminated.

(c) Whole of reactor building, some parts of decontamination centre, waste management plant and spent fuel storage bay fall in Zone-4.

ANNEXURE-X

RADIATION SOURCES AND RADIATION TRANSPORT THROUGH SHIELDING

X.1 Source Calculations

X.1.1 For determining the source strength in the core, which depends upon fission rate, neutron emission rate and spatial and energy distribution of neutron flux within the core, suitable validated code should be used [X-1]. The code should take into account the spatial distribution of materials in the core and the changes in fuel composition, the production of actinides and fission products with burnup. The neutron emission rates and flux distribution determined in core calculations should be used as input data for calculation of neutron flux energy and spatial distribution through the coolant, moderator and structural and shielding materials surrounding the core. These neutron flux distributions should be used for determining the production of gamma ray sources in the core and surrounding materials. The sources due to both fission and activation processes should be determined for this purpose. In the case of activation sources the decay of radionuclides and irradiation time in neutron flux should be taken into account for determining gamma ray source strength. Mostly it is the gamma ray source strength, which determines the dose to personnel.

X.1.2 As a result of mass transport of activation, activated corrosion and fission products generated in the core, transfer of the sources from their point of origin to other parts of the circuits of the concerned active system (PHT or moderator) takes place. These system activities and corresponding dose rates should be estimated in the following manner:

X.1.2.1 Activation Products: The activation product activity carried by the system fluid (PHT or moderator) is mainly due to neutron capture by the fluid (D_2O) nuclei. The neutron capture reaction with D_2O nuclei leads to the formation of activation products such as N-16, O-19, etc. The core exit activities in the fluid circulating through the core will be proportional to the average thermal neutron flux and the duration for which the nuclei are exposed to neutron flux.

The dose rate on the system equipment should be calculated considering the decay of the core exit activities for the time period required for the transport of nuclei from the core to the location/ equipment of interest.

X.1.2.2 Corrosion Product Activity: This activity should be calculated on the basis of the specified concentration of crud in the system fluid. For this past experience on crud concentration should also be taken into account. The radiation levels on the system/ pipes/ equipment should then be estimated on the basis of the estimated crud activity.

X.1.2.3 Fission Product Activity: Fission product activity is expected only in the coolant. If no failed fuel is present in the system the source of this fission product activity is the tramp uranium present as impurity in:

(i) the coolant

(ii) the zircaloy of the coolant tubes and the fuel-clad material, and

(iii) on the fuel-clad surface.

The main source (~ 90%) of fission product activity in the coolant (PHT) is defective fuel (with defect sizes varying from pin-hole size perforations to major defects).

The relative contributions due to the above-mentioned sources of uranium contamination and fission product releases in the coolant should be assessed. For the assessment of fission product activity released to the coolant, the in-core neutron exposure of the uranium impurity in the coolant should be considered. For this purpose the average thermal neutron flux in the coolant for the period required to achieve equilibrium (500-600 days) should be considered. The fission product activity calculation can be done using a validated computer code, such as ORIGEN-2 [X-1]. In the presently operating Indian PHWRs small amounts of fission product activity has also been observed in the moderator system heavy water and cover gas. This is assumed to arise due to uranium contamination of structural materials that come in contact with system heavy water.

Typical equipment/systems, which show significant radiation fields are given in Table-X-1.

X.1.3 In addition to the above mentioned sources the radiation and contamination arising from the spent fuel storage system, which consists of fuel storage bays, bay cleanup circuits and dry storage systems should also be assessed.

X.1.4 Contamination resulting from the following important sources should be assessed:

- Tritium-in-air contamination resulting from leakages/spillages of moderator or PHT system heavy water
- Air and surface contamination due to other activation, activated corrosion and fission products released from the leakages/spillages of PHT system D_2O

X.1.5 The discharges of radioactivity in the liquid and gaseous effluents from the plant may result in certain concentrations of radioactivity in the environment. The distributions of nuclides in the atmosphere and water bodies, the resultant surface sources from deposition and concentration of radionuclides in foodstuff should be assessed to determine the dose incurred by the public [X-2]. The methodology of calculating environmental doses due to airborne releases is outlined in AERB safety guides SG/D-14 and SG/S-5 [X-3 & X-2] and BARC Report No. 1412 [X-4].

ANNEXURE-X (Cont.)

TABLE-X-1: EQUIPMENT/SYSTEMS WITH HIGH RADIATION LEVELS

System	Equipment
PHT	-PHT lines in both Accessible and Shutdown Areas -D_2O storage tank (bottom) -Hot spots in circulating pump-casings, valves
Moderator	-ALPAS tank -Heat exchanger tubes -System strainers -Hot spots on adjuster rod drive mechanisms
Fuel Handling	-Circuit strainers -Drain tanks -Hot spots on fuelling machine heads -Fuel transfer circuit equipment (hot spots)
End Shield Cooling	-System lines (in Accessible Areas) during operation
D_2O Cleanup	-IX columns and evaporators
Annulus Gas	-System lines (during operation due to activation of air impurity in annulus gas)

X.2 Radiation Transport Through Shielding

X.2.1 A detailed description of the methods of calculating the fluence from the radiation source and the data used is outside the scope of this Guide. Ref. X-5 contains extensive bibliographies on this subject. The shielding design should take into account neutron and gamma radiation source terms. Suitable validated codes developed for this purpose should be used. Examples of such codes are SHELTX, SHELRAD, GS-EXT [Ref. X-6 & X-7].

X.2.2 The shielding design should take into account both the gamma and neutron radiation source terms. The required shielding should be governed by the AERB stipulations/guidelines on acceptable dose rates. (Ref. X-8 and Annexure-II). The estimated design based upon these calculations should also be compared with observed data from similar operating plants.

X.2.3 The calculation of radiation dose rates at locations where there are penetrations or streaming paths is more complex [X-5]. However, the following precautions should be taken to avoid radiation streaming due to penetrations.

- Use of compensatory shielding around penetrations
- Avoid installing penetrations with straight line paths to a source

References to Annexure-X

X-1 ALLEN G. CROFF, ORIGEN 2, 'A Versatile Computer Code for Calculating the Nuclide Compositions and Characteristics of Nuclear Materials'; Nuclear Technology, Vol. 62, (1983).

X-2 ATOMIC ENERGY REGULATORY BOARD, 'Methodologies for Environmental Radiation Dose Assessment'; AERB Safety Guide No. AERB/SG/S-5, Mumbai, India (2005).

X-3 ATOMIC ENERGY REGULATORY BOARD, 'Release of Airborne Radioactive Materials in Pressurised Heavy Water Reactors'; AERB Safety Guide No. AERB/SG/D-14, Mumbai, India (2001).

X-4 HUKKOO. R.K., BAPAT. V.N. AND SHIRVEIKAR. V.V., 'Manual of Dose Estimation from Atmospheric Releases'; BARC Report No. 1412, Mumbai, India, (1988).

X-5 USNRC, 'Source Term Related Technical and Licensing Issues Pertaining to Evolutionary and Passive Light Water Reactor Designs'; SECY-94-302, USNRC (1994)

X-6 NUCLEAR POWER CORPORATION OF INDIA LIMITED, 'Shielding'; NPCIL Design Manual No. 01150, Kaiga Atomic Power Project, Mumbai, India (1994).

X-7 QAD-CGGP 'A Combinatorial Geometry Version of QAD-P5A, A Point Kernel Code System for neutron and Gamma Ray Shielding Calculations Using the GP Build-up Factor'; RSIC, Computer Code Collection CCC-493, Radiation Shielding Information Centre, ORNL, USA, (1989).

X-8 ATOMIC ENERGY REGULATORY BOARD, 'Design Basis Events for Pressurised Heavy Water Reactors'; AERB Safety Guide No. AERB/SG/D-5, Mumbai, India (1999).

ANNEXURE-XI

FEATURES OF RADIATION MONITORING SYSTEMS AT NPPs

TABLE XI-1: MONITORING SYSTEMS

S. No.	Type of Monitoring System
1.	Area radiation monitors (low range and high range) and accident monitors
2.	Airborne contamination monitors including gaseous, particulate, iodine and tritium-in-air monitors.
3.	Surface contamination monitors
4.	Personal contamination monitors for interzonal and final exit points
5.	Portal monitors
6.	Process radiation monitors
7.	Effluent radioactivity monitors for liquid and gaseous effluents (Liquid effluent monitors should include tritium and gross beta activity monitoring systems and gaseous effluent activity monitors should include monitoring systems for gross activity, tritium, iodines, particulates and inert gases)
8.	Personal monitors such as thermoluminiscent dosimeters (TLDs), direct reading dosimeters (DRDs), neutron dosimeters, alarming dosimeters
9.	Portable or semi-portable radiation monitors, contamination monitors (including instruments with telescopic probes) for on-the-spot radiation monitoring of systems/ equipment
10.	Laboratory instruments such as low background beta counting systems, multichannel gamma spectrometers, tritium analysers

ANNEXURE-XI (Contd.)

XI.2: IMPORTANT FEATURES OF INSTALLED RADIATION MONITORING SYSTEMS

S. No.	Features
(i)	Range of measurement should cover reactor shutdown, normal operation and operational transients. Multiple-range instruments should preferably have automatic range switching facility.
(ii)	The instruments should have required measurement sensitivity for the type of radiation monitored and its energy spectrum. They should be capable of discriminating between the radiation monitored and interfering activities.
(iii)	The instruments should be perform correctly in the environmental conditions in the field
(iv)	They should have the following features: - adjustable alarm set points - provision for both local and remote (in control room) audio and visual alarm - no signal alarm feature, to annunciate failure of detection system or its associated signal processing unit - continued operation during external power supply failure (availability of back up power supply) - availability of testing and calibration provisions
(v)	The data obtained should be processed by a computerised system with trend monitoring, recording and display features in control room and shift health physics room.

ANNEXURE-XI (Contd.)

TABLE XI-3: SUGGESTED RANGES OF AREA RADIATION MONITORS

	Location	Suggested Range
(i)	Normally inactive and high occupancy areas (e.g., Control Room, Service Area)	0-100 mGy/h
(i)	Active/potentially active and continuously accessible areas (e.g. reactor building, waste management plant, cleanup room, decontamination centre)	0-10 mGy/h
(iii)	Shutdown Areas *(Reactor Building)	(1) 0-1 Gy/h (2) 0-10 mGy/h

* Shutdown Area monitors should have two ranges (with automatic switch over facility)

- Range 1 for use during reactor operation
- Range 2 for use during reactor shutdown
- The recommended set points for high and very high alarms are twice and ten times the existing background radiation levels.

ANNEXURE-XII

AUXILIARY FACILITIES

XII.1 The plant design should incorporate the following equipment and facilities.

- Protective clothing, plastic suits, rubbers, gloves, etc.
- Respiratory protection equipment
- Air samplers and other equipment to measure airborne contamination
- Portable radiation monitors and personnel and surface contamination monitors
- Portable shielding, radiation sign boards, cordons, etc.
- First aid equipment

XII.2 The following facilities should be provided for the management of health physics operations:

- Health physics operations office
- Radioactive source room and calibration facilities for radiological instruments
- Personnel decontamination facility
- Equipment decontamination facility
- Laundry facilities for contaminated clothing
- First aid room
- Health Physics and Bioassay laboratory (for housing tritium analysers, low level beta counters, gamma spectrometers, etc.)
- Contaminated equipment workshops
- Transit Waste storage facility
- Dosimetry laboratory
- Whole-body counter (to be located in environmental survey laboratory)
- Adequate space for housing personnel dosimeter racks, personnel contamination monitors (friskers, foot monitors, portal monitors, etc.) as appropriate, at interzonal checkpoint and final exit point.
- Protective equipment testing and servicing room

XII.3 For handling accident situations the following additional facilities should be provided.

- On-site emergency control centre
- Off-site emergency control centre
- Assembly areas and shelters for plant/site personnel and members of the public.

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PROVISIONAL LIST OF SAFETY CODES, GUIDES AND MANUALS ON DESIGN OF PRESSURISED HEAVY WATER REACTORS

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AERB/NPP-PHWR/SG/D-1	Safety Classification and Seismic Categorisation for Structures, Systems and Components of Pressurised Heavy Water Reactors
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AERB/SG/D-11	Emergency Electric Power Supply Systems for Pressurised Heavy Water Reactors
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AERB/SG/D-13	Liquid and Solid Radwaste Management in Pressurised Heavy Water Reactor Based Nuclear Power Plants
AERB/SG/D-14	Control of Air-borne Radioactive Materials in Pressurised Heavy Water Reactors

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AERB/NPP-PHWR/SG/D-20	Safety Related Instrumentation and Control for Pressurised Heavy Water Reactor Based Nuclear Power Plants
AERB/SG/D-21	Containment System Design
AERB/SG/D-22	Vapour Suppression System for Pressurised Heavy Water Reactors
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