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GOVERNMENT OF INDIA

AERB SAFETY GUIDE

RADIATION PROTECTION ASPECTS IN DESIGN OF NUCLEAR POWER PLANTS

ATOMIC ENERGY REGULATORY BOARD

AERB SAFETY GUIDE NO. AERB/NPP/SG/RP-1

RADIATION PROTECTION ASPECTS IN DESIGN OF NUCLEAR POWER PLANTS

Atomic Energy Regulatory Board
Mumbai-400 094
India

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Orders for this Guide should be addressed to:

Chief Administrative Officer
Atomic Energy regulatory Board
Niyamak Bhavan
Anushaktinagar
Mumbai-400 094.
India

FOREWORD

The Atomic Energy Regulatory Board (AERB) was constituted in 1983, to carry out certain regulatory and safety functions envisaged under Section16, 17 and 23 of the Atomic Energy Act, 1962. AERB has powers to lay down safety standards and frame rules and regulations with regard to the regulatory and safety requirements envisaged under the Act. The Atomic Energy (Radiation Protection) Rules, 2004, provides for issue of requirements by the Competent Authority for radiation installations, sealed sources, radiation generating equipment and equipment containing radioactive sources, and transport of radioactive materials.

With a view to ensuring the protection of occupational workers, members of the public and the environment from harmful effects of ionizing radiations, AERB regulatory safety documents establish the requirements and guidance for all stages during the lifetime of nuclear and radiation facilities and transport of radioactive materials. These requirements and guidance are developed such that the radiation exposure of the public and the release of radioactive materials to the environment are controlled; the likelihood of events that might lead to a loss of control over a nuclear reactor core, nuclear chain reaction, radioactive source or any other source of radiation is limited, and the consequences of such events if they were to occur are mitigated.

The Regulatory documents apply to nuclear and radiation facilities and activities giving rise to radiation risks, the use of radiation and radioactive sources, the transport of radioactive materials and the management of radioactive waste.

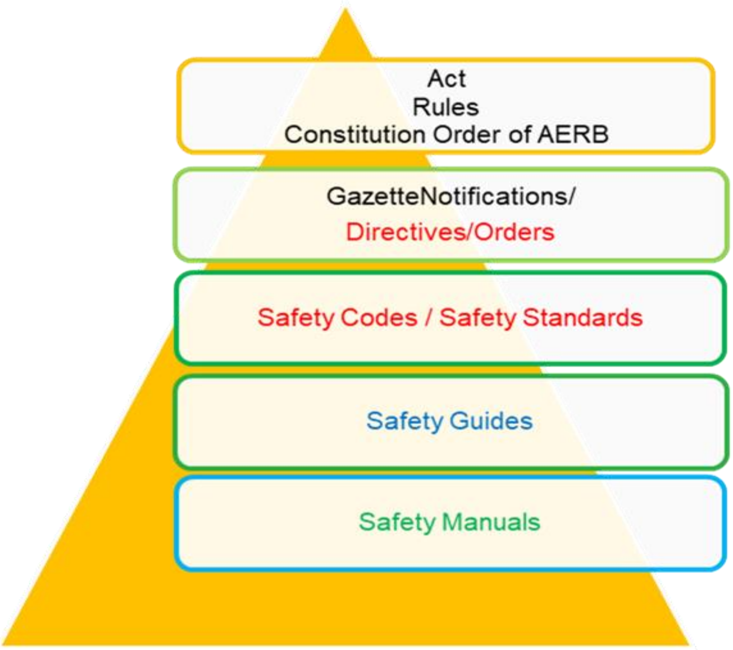

Fig. 1 Hierarchy of Regulatory Documents

Safety codes establish the objectives and set requirements that shall be fulfilled to provide adequate assurance for safety. Safety Standards provide models and methods, approaches to achieve those requirements specified in the safety codes. Safety guides elaborate various requirements specified in the safety codes and furnish approaches for their implementation. Safety manuals detail instructions/safety aspects relating to a particular application. The hierarchy of Regulatory Documents is depicted in Figure.1.

AERB issued a Safety Code titled 'Code of Practice on Design for Safety in Pressurised Heavy Water Reactor Based Nuclear Power Plants (AERB Code No. SC/D)' to spell out the minimum requirements for ensuring adequate safety in plant design. For elaborating the requirement of this safety code, AERB issued Safety Guide on Radiation Protection Aspects in Design for Pressurised Heavy Water Reactor Based Nuclear Power Plants (AERB/NPP-PHWR/SG/D-12) in 2005. This Safety Guide provided guidance to the plant designer on design aspects of radiation protection in Pressurised Heavy Water Reactor (PHWR) based Nuclear Power Plants (NPPs). Considering the review experience/ feedback on use of the Safety Guide, recent developments, different types of Nuclear Reactor designs and decommissioning aspects, the revision of the Safety Guide was taken up. Further, based on document development strategy and hierarchy in AERB, it was decided to place this Safety Guide under AERB Safety Code "Radiation Protection for Nuclear Fuel Cycle Facilities, AERB/NF/SC/RP, 2012" and retitled as "Radiation Protection Aspects in Design of Nuclear Power Plants AERB/NPP/SG/RP-1"

This Safety Guide elaborates the provisions for radiation protection to be incorporated in the design of Nuclear Power Plants. This guide will provide guidance to the Nuclear Power Plant designers to meet the relevant radiation protection related requirements stipulated in AERB Safety Code on Design of Pressurised Heavy Water Reactor Based Nuclear Power Plants AERB/NPP-PHWR/SC/D (Rev. 1), 2009, Design of Light Water reactor Based Nuclear Power Plants, AERB/NPP-LWR/SC/D, 2015 and Design of Sodium Cooled Fast Breeder Reactor based Nuclear Power Plants, AERB/NPP/SFR/SC/D (under preparation).

This Safety Guide considers the existing design of Nuclear Power Plants in India and gives generic guidance related to radiation protection aspects. This safety guide gives details of design basis for radiation protection, radiation protection considerations for NPP design, provisions for radiation protection in design of NPP and monitoring provisions for the same. In preparation of this safety guide, latest ICRP and IAEA standards have been used.

This Safety Guide supersedes the earlier version and is effective from the date of its issue and it applies to Nuclear Power Plants to be built after the issue of this Safety Guide.

The recommendations of international expert bodies, notably the International Commission on Radiological Protection (ICRP) and the International Atomic Energy Agency (IAEA) are taken into account while developing the AERB Regulatory safety documents.

The principal users of AERB regulatory safety documents are the applicants, licensees, and other associated persons in nuclear and radiation facilities including members of the public. The AERB regulatory safety documents are applicable, as relevant, throughout the entire lifetime of the nuclear and radiation facilities and associated activities. The AERB regulatory safety documents also form the basis for AERB's core activities of regulation such as safety review and assessment, regulatory inspections and enforcement.

Safety related terms used in this safety guide are to be understood as defined in the AERB Safety Glossary (AERB/GLO, Rev.1). The special terms which are specific to this safety guide are included under section on 'Special Terms and Interpretation'. In addition, the terms already defined in AERB Safety Glossary AERB/GLO, Rev.1, and being used in this safety guide with a specific context and requires interpretation or explanation are also included in this section.

Appendix is an integral part of the safety guide, whereas references and bibliography provide information that might be helpful to the user. For aspects not covered in this safety guide, applicable and acceptable National and International codes and standards shall be followed. Industrial safety shall be assured through good engineering practices and by complying with the relevant Industrial safety requirements under prevailing statutes.

This safety guide has been drafted by an in-house working Group. The draft was further reviewed by a Task Force with specialists drawn from technical support organisations and institutions, and other consultants. The comments obtained from all the major stakeholders have been suitably incorporated. The safety guide has been vetted by the AERB Standing Committee for Document Development (SCDD-1) and subsequently by the Advisory Committee on Nuclear and Radiation Safety (ACNRS). AERB wishes to thank all individuals and organizations who have contributed to the preparation, review and finalization of the safety guide.

D.K. Shukla.

Dinesh Kumar Shukla
Chairman, AERB

SPECIAL TERMS AND INTERPRETATION

Nil

Contents

1. INTRODUCTION

1.1 General

1.1.1 The mission of AERB is to ensure that the use of ionizing radiation and nuclear energy in India does not cause undue risk to the health of people and the environment. This requires that the radiation risks to occupational workers, members of the public and the environment arising from the nuclear power plants must be assessed, and controlled. AERB safety code(s) on design of Nuclear Power Plants (NPPs) lays down the minimum requirements for ensuring adequate safety in NPP design. Good design, prudent material selection, reliable equipment, good quality construction, QA in all the stages helps in ensuring a satisfactory Radiation Protection Program.

1.1.2 This safety guide deals with the provisions with respect to radiation protection to be considered in the design of nuclear power plants to protect the site personnel, the public and environment from undue exposure to ionising radiation during all plant states including operational states, accident conditions including design extension conditions and decommissioning. The guidance enumerated in this guide is based on latest ICRP recommendations, IAEA requirements and AERB codes relevant to radiation protection aspects.

1.2 Objective

The objective of this guide is to provide guidance for ensuring radiation protection in design of new NPPs, design modifications in existing NPPs and safety review and assessment of operating NPPs. This regulatory document will help in implementation of adequate design measures for radiation protection in accordance with the Safety Code on Design of Pressurized Heavy Water Reactor based Nuclear Power Plants (AERB/NPP-PHWR/SC/D (Rev 1), 2009), Safety Code on Design of Light Water Reactor based Nuclear Power Plants (AERB/NPP-LWR/SC/D, 2015), Safety Code on Design of Sodium Cooled Fast Reactor (AERB/NPP/FBR/SC/D, Draft) and Safety Code on Radiation Protection for Nuclear Fuel Cycle Facilities, (AERB/NF/SC/RP, 2012). This guide is meant for use by NPP designers, licensee/ plant operating personnel, regulatory body and technical support organizations.

1.3 Scope

1.3.1 This guide covers:

i) Principles and concepts such as dose limitation and optimization as a basis of radiation protection to be implemented in design of NPPs.

ii) General design basis aspects related to radiation protection during all operational states, accident conditions including design extension conditions and decommissioning.

iii) Radiation protection considerations in design of NPPs such as assessment of generation of radiation sources, layout of Structures, Systems and Components (SSCs), shielding, radiological zoning, ventilation and other relevant aspects during normal operations, Design Basis Accidents (DBAs), Design Extension Conditions (DECs) and Decommissioning.

iv) Provision for monitoring of radiation protection aspects in design for operational states for verification of design targets and for monitoring during accident conditions.

v) Organizational and human resources aspects related to radiation protection.

1.3.2 The design considerations for quantitative assessment of radiation 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, 1998). The procedural aspects of radiation protection are covered in safety manual on Radiation Protection in Nuclear Facilities (AERB/NF/SM/RP-1 (Draft)).

1.3.3 The guide covers only shielding and radiation protection related aspects of Engineered Safety Features (ESFs). The guidance on process details of ESFs can be found in relevant safety guides. This guide only addresses radiation protection in handling, treatment and storage of radioactive waste. The specific aspects of waste treatment and disposal are not addressed in this guide.

1.3.4 While this safety guide has been prepared specifically for radiation protection in design of Nuclear Power Plants, it may also be applicable, with suitable modifications, to other nuclear fuel cycle facilities.

2. RADIATION PROTECTION BASIS FOR NPP DESIGN

2.1 General

Protection of workers, public and the environment from harmful effects of ionising radiation is the fundamental safety objective from which the safety principles and requirements for minimising the radiological risks associated with nuclear power plants are derived. The fundamental safety objective applies to all stages in the lifetime of a nuclear power plant, including siting, design, construction, commissioning and operation, as well as decommissioning and also for DBA and DECs.

2.2 Safety Principles for Radiation Protection in NPP Design

2.2.1 The safety principles relevant to this Safety Guide are:

i) Optimization of protection:

Protection must be optimized to provide the highest level of safety that can reasonably be achieved.

ii) Limitation of risks to individuals:

Measures for controlling radiation risks must ensure that no individual bears an unacceptable risk of harm.

iii) Protection of present and future generations:

People and the environment, present and future, must be protected against radiation risks.

2.2.2 The measures to achieve the above objectives are based on the following fundamentals:

i) The practices causing radiation exposure to the individuals should be justified in terms of net positive benefit to the society.

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

iii) Radiation exposures of plant personnel and members of the public should not exceed the prescribed dose limits.

These should be adhered to throughput lifetime of NPP including decommissioning.

2.3 Dose Limits and Constraints during Operational States and Decommissioning

2.3.1 The design of Nuclear Power Plants should ensure that the dose limits and constraints prescribed by the regulatory body for occupational exposure of workers and members of the public as specified in AERB Safety Directive No 01/2011 are not exceeded. For members of the public, the dose should be restricted to dose constraint, a fraction of the public dose limit of 1 mSv/year.

2.4 Dose Criteria during Design Basis Accidents and Design Extension Conditions

2.4.1 For occupational workers, Guidance dose values prescribed by AERB (AERB/NF/SG/NRE-1) for initiating emergency response and mitigation actions to control the accident progression should be used. The specific design targets for any system, if required, should be arrived at considering the system operation and area occupancy.

2.4.2 For members of the public, design should demonstrate that the calculated doses at the exclusion zone boundary for design basis accidents and design extension accidents should not exceed the dose criteria as prescribed by AERB (AERB Safety Code Site Evaluation of Nuclear Facilities AERB/NF/SC/S Rev 1, 2014). 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.

3. RADIATION SOURCES CONSIDERATIONS FOR NPP DESIGN

3.1 General

3.1.1 Main objective of the radiation protection in design is to optimise collective dose to radiation workers and to control individual doses to radiation workers as well as members of the public within the limits. The primary consideration should be to optimize the protection against the sources of radiation in the plant. Radiation sources throughout the plant should be comprehensively identified and controlled so as to keep the radiation levels associated with them as low as reasonably achievable.

3.1.2 During operation, the major sources of radiation are activation products e.g. ^{16}N, ^{19}O, activated corrosion products fission products from tramp Uranium, gaseous systems exposed to reactor core although fission products may also be significant if there is significant amounts of fuel failure. These sources originate in the reactor core. The radioactive materials are then transported by the reactor coolant and by the moderator in liquid moderated reactors.

3.1.3 Material selection of reactor core and other components should be carried out keeping in view of design requirement and at the same time to minimise hazard due to activation/ corrosion products to meet the radiation protection requirements. Leak-tightness should be ensured as far as possible and leakage detection features should be provided.

3.1.4 The source term as estimated above should be used to assess radiation dose rates in various plant areas using validated models [1]. Public dose also needs to be evaluated for various exposure pathways as per approved methodology [2]. The dose rates within various plant areas and annual public dose to representative person thus estimated should be compared against the set design targets to evaluate the efficacy of the design.

This chapter contains details of radiation sources and doses to radiation workers and public during various plant states**.**

3.2 Operational States and Decommissioning

3.2.1 Radiation Sources

i) The magnitudes and locations of the sources of radiation in operational states and during decommissioning should be determined in the design phase. The significant sources of radiation during operational states and decommissioning are given in Appendix-I. These are:- the reactor core and vessel / Calandria; the reactor coolant and fluid moderator system; the steam and turbine system; the waste treatment systems; irradiated fuel; the storage of new fuel; decontamination facilities; miscellaneous sources. The major sources are the reactor core, irradiated fuel and spent resins, and the design should therefore be such as to ensure that personnel are not exposed to direct radiation from these sources.

ii) In the context of radiation sources, it is important to understand that a major source in a given operational state may become a minor one in a different operational state. Some isotopes that are of minor importance for dose rate considerations during operation become of major importance during decommissioning (e.g. ^{60}Co, ^{125}Sb, ^{65}Zn, concrete activation products). Also, even when dealing with reactors of the same type, changes in the design may have a strong influence on the relative importance of different sources.

iii) In a well-designed and operated reactor, the major radiation source will be the activation products in and near the core. The important radioisotopes will be those having a half-life of a few years or more. In many cases, ^{60}Co would be the pre-dominant radionuclide in initial years and later on ^{63}Ni arising from impurities becomes important. In such cases, the control of impurity levels during design will also be effective in controlling it during decommissioning.

iv) It should be recognised that, while consideration is given to decommissioning at the design stage itself, there will be significant and ongoing changes in radiation conditions during decommissioning. Measures should be taken in the design to reduce the significance of these changes, but this factor should also be recognised in the operational arrangements. Equally, access will be necessary during decommissioning to areas that are not normally accessed. Consideration should be given to this factor in the design of facilities and equipment.

v) The sources of radiation that contribute to doses received during decommissioning are the activation products in the components of the core and the surrounding materials, contamination in the primary and auxiliary circuits, and the accumulation of active material at the plant.

vi) In case of concrete, the magnitude of the source term can affect both the dose to radiation workers and the volume of radioactive waste that is generated. The source term in this case may be dominated by radionuclides that are not very important during operation, such as the rare earth isotopes, and control of such impurities may be an important aspect of the design process.

3.2.2 Assessment of Radiation Dose Rates

The design should adopt validated methods for the assessment of radiation dose rates on plant components/ equipment that are expected during operational states [1].

i) The design assessment of radiation sources starts with identification of radionuclide vectors responsible for radiation field or causing exposures. This may involve calculation of the core activity and the transport and redistribution of activated corrosion products and /or fission products carried in reactor coolant and deposited away from the point of origin.

ii) 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.

iii) The final step is to calculate the radiation dose rate by multiplying the radiation flux by the appropriate dose conversion factors.

3.2.3 Estimation of Dose to Public due to Effluent Releases

The following exposure pathways should be considered for evaluating exposure to representative person in public:

i) Exposure due to radioactive liquid effluents through the aquatic route.

ii) Exposure due to radioactive gaseous effluents (external plus internal dose from airborne and deposited activity) through the air route.

iii) 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 representative person should not be exceeded. The methodology for estimation of public dose should be as per AERB document [2].

3.2.4 Design Targets [1]

Design targets should be set at the start of the design process considering the annual collective doses, radiation levels at specified distance from equipment, area radiation levels and individual dose for members of the public.
In setting these design targets, account should be taken of experience at relevant plants and differences in the design, operational practices between the reference plants and the plant under design. These design targets should be included in safety analysis report of the NPP.

Typical design targets are provided in the Appendix II.

3.3 Accident Conditions

3.3.1 General

3.3.1.1 The principal design measures that are taken to protect the public against the possible radiological consequences of accidents are required to have the objectives of reducing the probability that accidents will occur (prevention of accidents) and reducing the source term and releases (mitigation of consequences) associated with accidents if they do occur. Accident prevention is

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[1] The design targets are not the dose limits and should serve as useful tools for optimisation process. These can be exceeded if justified. However, achieving design target itself does not satisfy optimisation principles. The doses should be reduced below the targets, if it is justified.

not explicitly addressed in this Safety Guide, but necessary references have been provided on the subject at appropriate places.

3.3.1.2 The design objectives for accident conditions are to limit to acceptable levels:- (1) the risks to the public from possible releases of radioactive material from the nuclear power plant; and (2) the risks to site personnel from these releases and from direct radiation exposure. These design objectives should be achieved by means of design and special features, such as safety systems and protection systems that are incorporated into the design of the plant. Achievement of the design objectives should be confirmed by means of a safety analysis [1]. Deterministic safety analysis and the associated dose assessments and probabilistic safety assessments for demonstrating compliance with the radiation dose limits should be based on conservative assumptions for the analysis of design basis accidents and realistic or best estimate assumptions for the analysis of severe accidents.

3.3.1.3 To achieve the design objectives mentioned, the necessary provisions and procedures (e.g. for access to the control room, maintenance of essential equipment or process sampling) should be such as to enable the plant operators to manage the situation adequately in an accident.

3.3.1.4 Practices that are similar to those used for operational states should also be used to ensure that proper plant design is achieved to provide adequate radiation protection for site personnel and the public under accident conditions.

3.3.1.5 The optimised design of plant systems and components for radiation protection under accident conditions should be achieved by means of consultation with experts in radiation protection, plant operations, plant design and accident analysis, and regulatory requirements. There should be continuous interactions among these groups throughout the design process to arrive at a design that provides radiation protection under accident conditions which is acceptable to the regulatory body. The design should also ensure that effective procedures for accident management can be implemented.

3.3.2 Radiation Sources

i) The magnitude, locations, possible transport mechanisms and transport routes of the sources of potential radiation exposure under accident conditions should also be determined in the design phase of the plant. Guidance on the same is provided in AERB documents [3].

ii) 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 during different design basis events (DBEs). The DBE scenarios analysed should be comprehensive and should include releases from:

a) reactor core during loss of coolant accidents,

b) reactivity accidents,

c) accidents due to loss/ breach of fuel containment barriers,

d) fuel handling accidents,

e) coolant or moderator circuits, due to major leakage of system heavy water in case of HWRs and

f) Core disruptive accident in case of fast reactors.

Details of such events are given in AERB Safety Guide Design Basis Events For Water Cooled Nuclear Power Plants AERB/NPP-WCR/SG/D-5 (Rev.1).

iii) Fission products are generally of major importance as compared to activation products and actinides in the determination of activity released from fuel. In addition, for heavy water-cooled reactors, 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 may result in exposure to the public, the following should be taken into account:

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

b) Transport of released radionuclides from fuel /core to environment taking in to account the thermal hydraulics of core, coolant and containment for the accident sequence under consideration. Validated computer codes should be used for estimating source terms for different DBAs and DECs. Alternatively, conservative methods of assessment of transport of fission products may be used wherever approved computer codes are not available [1].

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

a) Use of demisters before air filters,
b) Shielding of filters, and
c) Heat generation due to decay of accumulated activity.

vi) The main source of radiation in a nuclear power plant under accident conditions for which precautionary design measures are adopted consists of mainly radioactive volatile fission products (e.g. Iodine, Caesium, Tellurium etc.) and FPNGs. These are released either from the fuel elements or from the various systems and equipment in which they are normally retained. Examples of accidents in which there may be a release of fission products from the fuel elements are loss of coolant accidents and reactivity accidents in which the fuel cladding may fail due to over-pressurisation or overheating of the cladding material. Another example of an accident in which fission products may be released from the fuel rods is an accident in handling spent fuel, which may result in a mechanical failure of the fuel cladding from the impact of a fuel element that is dropped. The most volatile radionuclides usually dominate the accident source term during short and near term while for the long term, radionuclides like ^{90}Sr and ^{137}Cs may become important.

vii) Account needs to be taken of the possibility of radioactive material accumulating on and being released from air filters or components of the liquid waste treatment system after accidents. In comparison with the radiation emanating from fission products and actinides, activation products are usually of minor importance.

3.3.3 Considerations for Determining Potential Doses

3.3.3.1 In order to show compliance with the design targets, the potential consequences of the design basis events shall be determined. 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.

3.3.3.2 During accident conditions, the releases from the plant will be partly at the ground level and partly at an elevated level (through stack). The doses resulting from these releases should be estimated for the following:

i) On Site personnel
ii) Members of the public (living beyond exclusion zone boundary distance)

In addition, the dose to site personnel due to direct exposure from radioactivity present in the containment should be evaluated. 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 e.g. Supplementary control room etc. Such evaluations should include tritium exposures subsequent to large heavy water spillages for heavy water-cooled reactors. For assessing environmental radiation dose during the accident conditions, the following main radiation sources released from fuel should be considered:

i) Fission product noble gases(FPNG)
ii) Radio-iodines
iii) Radio Caesium
iv) Strontium, etc.

The details on source term evaluation and dose calculation are given in other AERB documents [3].

3.3.3.3 The adequacy of the provisions for the protection of the site personnel and public under postulated accident conditions should be judged by means of the comparison of calculated doses with the specified dose criteria that constitute the design targets for accidents. In general, the higher the probability of the accident condition, lower the specified design target should be.

3.3.4 Potential Doses to Workers

3.3.4.1 Control Room Operators and staff in On-site Emergency Support Center (OESC) would be handling accident situation following the Emergency Operating Procedures (EOPs)/Accident Management Guidelines (AMGs) as the case may be; so radiological conditions at these locations during accidents would be required to assess the personnel dose and estimate the required shielding to be provided for these structures (Main Control Room (MCR)/Supplementary Control Room (SCR)/OESC).

3.3.4.2 Further the hook-up points for fire/ emergency water supply in the field would also need to be accessed during DECs; so radiological conditions at these locations would be required to assess the personnel dose and required shielding, may be a temporary one. Acceptability of the above-estimated dose rates may then be decided based upon the applicable AERB Safety Guide “Management of Nuclear and Radiological Emergencies in Nuclear Facilities, AERB/NF/SG/NRE-1 (Draft)”.

3.3.4.3 Dose rate in MCR/SCR should be estimated for DBA and DECs. The shielding of OESC should be such that design dose target should be 20 mSv in 7 days due to direct radiation for the postulated accidents. Further details on aspects such as location of OESC, design of ventilation system etc have been provided in Section 4.9 of this guide.

3.3.4.4 For DEC-B, dose rate near OESC, Containment Filtered Venting System (CFVS), electrical & Fire/ emergency water hook-up points should be estimated. Acceptability of these may then be decided based upon the guidance available as brought out above and accordingly personnel dose can be limited through rotations/occupancy control.

3.3.5 Dose to Public due to Releases

3.3.5.1 Time duration of the release (early release, delayed release) should be considered in Radiological Impact Assessment (RIA) of the various accident scenarios (DBA, ‘DEC-A’, ‘DEC-B’). The practice presently followed for RIA with respect to selection of atmospheric stability, is that early phase is termed as up to 2 hours and delayed phase is termed as beyond 2 hours. However, these number should be changed depending upon site–specific conditions (meteorological parameters etc), design features incorporated in the plant design, etc and actual persistence hours of wind duration and releases should be used in dose computation.

3.3.5.2 Phase-wise release (early release and delayed release for certain duration) should be calculated and then all these releases should be integrated to arrive at the total estimated release for a particular DBA/DEC. Estimate of concentration of radionuclide against time and the consequential dose should be plotted for at least one year. Further, for ‘DEC-B’, return time of relocated public is to be decided which would require dose estimation for a longer time. The exposure pathways and time period should be as per approved methodology [3].

4. PROVISIONS IN PLANT STRUCTURES, SYSTEMS & COMPONENTS FOR RADIATION PROTECTION

4.1 General

The design of plant structures should take into account the aspects related to ease of personnel access to various areas and required shielding structures such as compartmentalisation so as to limit the exposure to plant personnel. Various facilities should be designed and provided so as to facilitate ease of working access and support in field activities.

4.2 Plant Layout

4.2.1 The layout of the plant should be designed in such a way as to ensure personnel and equipment access requirements for operation, inspection, maintenance, repair, replacement and decommissioning of the equipment/ plant to limit the personnel exposure.

4.2.2 In layout design and arrangements, provision should also be made for the capability of performing the intended operations during and after design basis accidents and design extension conditions. Furthermore, the functions required for emergency management should be taken into account in the design.

4.2.3 Locations in the plant site where the dose rate remains low during an accident should be identified in the design phase. These locations should be used as assembly/shelter areas for gathering of radiation workers and passing on instructions to them during emergency.

4.2.4 In a multi-unit site, the layout design should consider presence of other facilities also. The requirements should include effluent releases and emergency management considerations among others.

4.3 Exclusion Zone and Operating Island

4.3.1 Exclusion Zone

Exclusion zone is physically isolated from outside areas and is under the control of the plant management. The exclusion zone should be such as to minimise the radiological impact of the plant on the public and should meet the specified radiation protection requirements as specified in AERB Safety Code “Site Evaluation of Nuclear Facilities, AERB/NF/SC/S (Rev.1), 2014” and AERB Safety Code “Management of Nuclear and Radiological Emergencies, AERB/SC/NRE, 2023”.[2].

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[2] Though Exclusion zone should consider the dose to the public as a radiation protection requirement, other considerations such as emergency response requirements, land usage (to accommodate all facilities and future expansion), security requirements and environmental factors (e.g. meteorological parameters) also should be considered.

4.4 Supervised Area and Controlled Area[3]

4.4.1 Supervised Areas

Within the operating island, the areas should be demarcated as Supervised Areas and Controlled Areas based on the potential of radiation exposure. The maximum permitted radiation dose rate in supervised areas should be limited to 1 μSv/h. There should not be any radiation sources and no surface contamination allowed in this area. Even though specific protective measures and safety provisions e.g. individual dose monitoring would not be required for these areas, radiological conditions should be monitored at specified frequencies.

4.4.2 Controlled Areas

The controlled areas should be defined by taking into account the magnitudes of the expected normal exposures, the likelihood and magnitude of potential exposures, the nature and extent of the required protection and safety procedures. The controlled areas should be delineated from supervised areas by engineering and administrative control measures. Controlled areas should have provisions for Individual dose monitoring, area radiation monitoring, use of personnel protective equipment (PPE) and personnel decontamination. The monitoring in controlled areas should be decided based on dynamic conditions of the systems. Entry to controlled area should be regulated through change room.

4.4.3 Zoning in Controlled Area

4.4.3.1 To restrict the radiation exposure, the controlled area of the NPP should be divided into various zones based on anticipated radiation levels (dose rates) or contamination levels (surface contamination levels/ airborne contamination levels), access requirements or any other specific requirements (e.g. need to separate safety trains). Each zone should be clearly demarcated and provided with appropriate ventilation design, access control, radiation monitoring programme and personal protective equipment including clothing.

4.4.3.2 Personnel movement in the plant should be from lower zones to higher zones for entry and vice versa for exit.

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

4.4.3.4 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.

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[3] These corresponds to Free Access Area and Controlled Access Area of VVER type reactors.

4.4.3.5 The guidelines and methodology for radiological zoning in controlled area is provided in Appendix III.

4.5 Access Control:

4.5.1 Based on the access requirements during reactor operation, location of the equipment and prevailing radiological conditions (e.g. dose rate, etc), controlled areas may be further classified as follows:

i) Accessible Areas[4]:
These are areas, which are accessible during all times of reactor operation, and the personnel exposure in these areas is not expected to be high.

ii) Shutdown Accessible Areas[5]:
These are the areas, which are accessible only during shutdown of the reactor or with very specific access provisions or limited time during operation owing to very high radiation levels. The provisions in designs for entry to these areas during reactor operation, if required, should be made only after approval from designated administrative authority.

iii) Inaccessible Areas:
These are the areas with very high dose rate or high persisting contamination at all times. The design should incorporate provisions to ensure that these areas are not accessible during lifetime of reactor and accessible only during decommissioning of the plant with special provisions. Such areas should be appropriately identified and sealed.

4.5.2 The shutdown accessible areas should be clearly marked and entry to these areas should be restricted through interlocks and administrative control measures during reactor operation. The design should incorporate appropriate provisions to ensure that whenever personnel are working in shutdown areas, reactor power cannot be raised.

For limiting entry in these areas during reactor operation, design should include features so that such instances are kept to minimum (e.g. taking out reading gauges to outside active area, CCTV monitoring of the critical locations, etc.).

4.5.3 Additionally there could be some areas in the plant, which need to be made inaccessible during certain operations (e.g. fuelling machine with spent fuel parked in fuelling machine service area, spent fuel in fuel transfer room etc in PHWRs). Such operations and respective areas should be identified and appropriate provisions should be incorporated in the design so as to ensure no personnel occupancy in such areas during identified operations.

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[4] These areas include general Reactor Building Areas in PHWR, Attended Areas, Periodically Attended Areas and Restricted Access Areas of Auxiliary building of VVER Reactors.

[5] This includes Shutdown Accessible areas of PHWRs and Restricted Access and Unattended Areas (in Reactor Building) of VVER Reactors.

4.5.4 The personnel access and occupancy in radiation areas and contamination areas should be controlled to minimise the radiation exposure. The following provisions should be provided for access and occupancy control:

i) Clear passageways for easy movement of personnel and equipment (with provisions for movement of portable shielding).

ii) Routes through radiation and contamination zones to be so provided as to ensure minimal transit times/ dose. Provisions of interlocks/ lockable doors to restrict access to high dose rate areas and provision of alarms to alert about non-functionality.

iii) Layout so designed as to ensure that personnel need not pass through zones of higher radiation for accessing lower radiation zones.

iv) Provisions of waiting areas in low radiation areas for use by radiation worker.

v) Provision of adequate space near equipment to carry out repair/ maintenance jobs and for cutting and segmentation operations for decommissioning purposes.

vi) Mounting of components that require frequent access at a height convenient for working and provisions of permanent ladders, access platforms and cranes in areas where it can be foreseen that these are required for maintenance or removal of plant equipment taking into consideration the provision of decommissioning also.

vii) Features to facilitate installation of temporary shielding and provisions for quick and easy removal of shielding and insulation in locations where it may be necessary to perform routine maintenance or inspection.

viii) Provision of special tools, equipment and remote handling devices for facilitating work to reduce exposure time.

ix) Provision of remote controlled equipment.

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

xi) Short emergency exit routes should be provided in the areas where the need may arise to operate equipment during an accident.

xii) Clear identification of rooms, placing of signboards and absence of obstacles to reduce period of exposure during accident conditions.

4.6 Buildings /Facilities

4.6.1 Reactor Building

Reactor building should house major active system components to ensure confinement of radioactivity and radiation shielding in operational states, and in accident conditions to the extent practicable. This should be leak tight and must be fitted with fast closing isolation dampers. Access should be through airlocks and must follow the required area/

zonal procedures. All the pipe penetrations, cables and duct penetrations have isolation bellows, glanding with leak testing provisions to ensure leak-tightness.

4.6.2 Auxiliary/ Service Building

The layout design of this building should include provisions to house auxiliary components and systems. The ambient radiation levels should be as per the zoning requirements. The facility for accident sampling may be housed in this building.

4.6.3 Control Building

The layout design of the control building should include provisions to house necessary control systems including main control room and located in supervised area. Additionally, the ventilation system should be such as to maintain habitability of the important locations during accident conditions. A Supplementary/ Emergency Control Room independent (physical separation and functional isolation) of Main Control Room (MCR) should be provided. Passage from MCR to SCR should be clearly marked and must remain accessible under postulated accident conditions.

4.6.4 New Fuel Storage

Layout and arrangements should ensure adequate sub-criticality margin during all operational states and design basis accidents. If the layout cannot fully ensure sub-criticality, additional means such as neutron absorbers should be used. Adequate shielding and confinement should be provided in new fuel storage and handling system to ensure protection of personnel against radiation. Appropriate ventilation should be provided to keep airborne contamination levels below the specified levels as per zoning criteria in case of any inadvertent airborne contamination due to fuel.

4.6.5 Spent Fuel Storage

Spent fuel should be stored in a designated area under water cover so as to ensure shielding, removal of decay heat and maintain the sub criticality. Water level should be maintained to a specified height and monitored continuously so as to control the radiation levels in the adjoining areas. Continuous temperature monitoring for the water should be provided. Provision for sampling for detecting any leakage from the storage bay should be provided. Low levels of radioactivity and proper chemistry control should be maintained in water though purification system using filters and ion exchange resins. The ventilation of the area should be connected to active ventilation circuit and must employ appropriate types of filters so as to prevent any activity going out during normal operation as well as during any unusual event. The facilities for retrieval of spent fuel and transportation to a dedicated storage facility for other purposes should be provided.

4.6.6 Gaseous Effluent Treatment Facility

4.6.6.1 Facility for treatment of gaseous effluents should be provided at the NPP. This should include provisions of pre-filters and filters (HEPA filter for particulates and activated charcoal filters for radioiodine). The system should also include vapour recovery dryers to contain any radionuclides associated in vapour form. The performance of filters/ adsorbers should be such as to achieve the effectiveness required by the safety analysis. Provision for hold-up should also be made for decay of short-lived radionuclides before discharge, if required.

4.6.6.2 Provisions for monitoring and isolation should be made along the discharge path. To avoid the possibility of out leakages, the ducts carrying gaseous effluents should be designed to be leak tight.

4.6.7 Stack

A stack of appropriate height should be provided for dispersion and dilution of effluents discharged from the Nuclear Power Plants. The stack design should be based on public protection as well as aerodynamic considerations with former being the dominant criteria. The stack should be of sufficient height in comparison to nearby structures so as to avoid wake effect from these structures. The stack should be located near the major facilities so as to minimise active duct lengths. The velocity of effluents released from stack should be such as to overcome the existing wind velocity (horizontal component of wind velocity so as to avoid downwash) above the stack. The dilution factor provided should be such that doses to the representative person will remain within the dose constraint specified by regulatory body for the corresponding discharges. The facilities for obtaining representative samples for analysis should be provided.

4.6.8 Waste Management Plant

4.6.8.1 Every NPP should be provided with a waste management plant for treatment of solid and liquid wastes. The design of the waste management plant should incorporate adequate radiation protection measures during all wastes handling/management operations. The radiation hazard of each area and equipment in waste management plant should be evaluated. Since various equipment in treatment systems may contain radioactive material in high concentrations, protection from these materials should be provided for radiation workers. The radiation hazard should be evaluated considering sources of high radiation levels such as Ion Exchange Resins etc. The changes in the activity concentration of waste that can occur as a result of treatment (particularly, the activity concentration increases in incinerated and compacted waste) should also be taken into account.

4.6.8.2 The waste management systems should be designed in such a way to reduce the leaks from the system particularly leakage of resin and concentrates from the tanks. Provisions for leak detection and leak containment should be made. The design should provide for remote control of major operations such as regeneration and change of resin among others. Equipment should be provided with suitable interlocks to prevent dangerous or incompatible operations.

4.6.8.3 The waste generated during decommissioning is almost similar in characteristics to that of waste generated during operation of the NPP with the difference that large quantity of waste containing only small concentrations of radionuclides needs to be treated. Design of waste management plant should be able to cater to these needs.

4.6.8.4 During accident conditions, radioactive waste of high concentration and high dose rates needs to be managed. Thus, the design of waste management plant should include considerations for handling of such wastes. For the NPPs, where the waste management plant already exists, the nature and amount of waste that can be generated during accident conditions should be assessed and additional provisions such as separate streams for waste treatment, provisions for temporary shielding etc should be provided.

4.6.9 Temporary Storage of Radioactive Waste

4.6.9.1 The design should provide facilities for temporary storage of radioactive waste taking into account the form, radionuclide content and types of waste. The design should ensure that radioactive waste can be received, handled, stored and retrieved without any undue impact on radiation workers. The design should provide for

i) Adequate space and confinement of the waste
ii) Provisions for shielding and contamination control
iii) Provision for ventilation of the area
iv) Provision for continuous radiation monitoring
v) Necessary arrangement for remote handling for various operations particularly handling waste packages with high dose rates.

4.6.9.2 In addition to radiological hazards, non-radiological hazards (for example, fire, flood or explosion), which may contribute to radiologically significant consequences, should also be considered in the design of storage facilities. The waste storage should address the waste generated during off-normal conditions and consideration for the same should be included in design.

4.7 Auxiliary Facilities:

4.7.1 Change Room/ Areas

Change rooms/ areas should be provided at designated places in controlled areas at the entrance of controlled area and locations where potential of contamination exists to prevent the spread of contamination. The change room/ area should have facilities corresponding to the requirements for entering to higher contaminated areas including decontamination facilities for personnel, monitoring instruments and storage areas for protective clothing. The change rooms/ areas should be designed for meeting the requirements of higher manpower anticipated during shutdown and maintenance jobs. The ventilation design should ensure prevention of spread of contamination. Appropriate barriers should be provided to separate the clean area from the potentially contaminated area. Sufficient space should be provided so that cross contamination between personnel is avoided.

4.7.2 Decontamination (Equipment, Tools and Personnel)

4.7.2.1 The provisions for decontamination should be considered in the design of the NPP for reduction in radiation exposure. All the components of the systems, which come in direct contact with primary coolant or moderator as well as floors and drains of rooms in which radioactive systems are located, should be considered as potential items requiring decontamination. The floors and walls of the rooms having potential for radioactive liquid leaks should be designed waterproof to the height up to which water level may rise in case of a design basis accident. 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). Floor drains with filtration provision should also be provided in all such rooms. Leak detection instruments (e.g. beetles etc) should be provided for timely detection of spills.

4.7.2.2 Provisions for decontamination of active components and equipment at their locations should be provided to the extent possible. The components that need decontamination and their transport arrangements shall be designed in such a way that the detachment and transfer of components for decontamination does not result in significant radiation doses to workers. The reactor building (for PHWRs) should be provided with vacuum mopping system to rapidly collect spilled water from floors subsequent to a major leak. Facilities/ equipment to safely collect spills should also be made available at all vantage locations in the building.

4.7.2.3 Decontamination Centre should be provided to remove surface contamination from radioactive material transport containers, tools, equipment, system parts, etc. Separate designated space should be provided for the decontamination of highly activated and contaminated components. The provisions to handle the components and objects to be decontaminated remotely or using a shield should be provided. Drains from the decontamination centre should connect to the active drainage system. Facilities for decontamination of personnel and protective equipment should be provided. The design should consider provisions for decontamination and other operations (cutting & transport) of large equipment during decommissioning.

4.7.3 Radioactive Source Storage

Designated facility for storage of radioactive sources should be provided at the station. The shielding of the facility should be such that there will not be any increase in the ambient background radiation levels outside the facility during use of maximum capacity sources. The source specific shielding should be designed in such a way to limit the radiation levels as per the designated zone radiation levels. The storage area for sources should be properly barricaded with administrative control. The entry should be administratively controlled and allowed only to authorized personnel. All other safety and regulatory requirements specified by regulatory body should be adhered to.

4.8 Health Physics Facilities

The facilities that are needed for effective radiological safety control during the operation and maintenance of the nuclear power plant and for responding to emergencies should be included in the design. These are required for limiting the spread of contamination within the controlled area and preventing the spread of contamination outside the controlled area, for carrying out adequate monitoring of the workplace and individual monitoring, for providing the workers with the required protective equipment, and for managing other health physics operations. These facilities should include the following:

i) Health Physics operations office
ii) Health Physics laboratory (for housing tritium analysers, low level beta counters, gamma spectrometers, etc.)
iii) Personnel Dosimetry laboratory
iv) Facility for calibration of radiation instruments
v) Adequate space for housing personnel dosimeter racks, personnel contamination monitors (Friskers, Hand and Foot monitors, Portal monitors, etc.)

4.9 Emergency Facilities

4.9.1 Appropriate and adequate facilities to deal with stipulated emergencies should be provided at the NPP. This should include the emergency control room, emergency control centres for decision makers to deal with emergencies. Specified and accessible locations should be provided for storage of equipment needed for emergency response. Adequate number of emergency shelter areas and emergency assembly areas should be provided in the design. These should be provided with provisions of ventilation cut-off and other necessary facilities as specified by regulatory body. Facilities for first-aid decontamination of personnel, equipment and areas should be provided.

4.9.2 An Onsite Emergency Support Centre (OESC) with adequate shielding and support provisions to house the emergency response personnel for prolonged period of time should be provided. OESC should be located preferably at the highest elevation, away from water intake and outfall points, least dominant wind direction and in shadow of plant buildings with access from two different routes. The space should be sufficient to accommodate the staff (around 100) required for emergency response functions for 7 days without any external support and have provisions for rest so that that can be rotated in the said period.

4.9.3 OESC should have its own survival ventilation system (provision of switching off in case of passing of radioactive plume) with air suction from two different directions, in addition to normal air conditioning system. The filters in the survival ventilation system of OESC should be designed to operate for at least one month without need of replacement. OESC should have adequate numbers of portable/ self-contained breathing apparatus sets.

4.9.4 The communication facilities, tools and equipment for OESC should be so as to enable it to function as an emergency control centre. Provisions for uninterrupted power supply for 7 days should be provided. Facilities for radiation and contamination monitoring at the entrance, decontamination centre, toilet/shower rooms and change rooms should be provided.

4.10 Systems and Components

4.10.1 Systems and Components should be designed in such a way that doses to operating personnel will be maintained below the prescribed limits and will be kept as low as reasonably achievable (ALARA). Even though specific requirements with respect to radiation protection are covered in this section, the generic requirements for all the SSCs that need to be ensured in their design are given below:

i) Design should provide adequate considerations (say in terms of space, amenability, spares, etc.) for maintenance/repair/overhauling, periodic surveillance on the system components.
ii) Auxiliary systems that will be needed during decommissioning should be designed such that during decommissioning they are available and will only require isolation from structures, systems and components that will have to be taken out of service and dismantled early in the decommissioning project.
iii) Mounting of components that require frequent access at a height convenient for working and provisions should be made 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) Design consideration should ensure that external non-nuclear systems and services do not become contaminated. These then can be subjected to conventional industrial demolition.
v) Systems should have provisions for installation of "test coupons" to facilitate radiological characterisation.
vi) Provision of clear passageways of adequate dimensions for easy movement of personnel and equipment should be provided. The routes should be as short as possible so as to minimise radiation exposure and possibility of spread of contamination.
vii) All major components should be made removable using permanently installed lifting equipment.
viii) Materials used in the manufacture of SSCs should be such as to minimise activation of the material as far as is reasonably practicable.
ix) Plant equipment subject to frequent maintenance or manual operation should be located in areas of low dose rate to reduce the exposure of radiation workers.

4.10.2 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 active components (like pumps, and valves) placed in high radiation zones should be adequately shielded to ensure low radiation levels during their in-situ maintenance.

iii) As far as possible, process parameters indicators (level / pressure / temperature/dew point, etc.), drive units, control equipment, auxiliary equipment/units and other non-radioactive equipment should be installed in low radiation areas.

iv) Adequate number of mask-air supply points should be available at all active areas (in PHWR and SFR).

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

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

vii) Provision should be made available for quick and easy removal of shielding and insulation in locations where it may be necessary to perform routine maintenance or inspection. This would facilitate ease in decommissioning also.

4.11 Shielding

4.11.1 General

The provision of shielding can be an effective form of engineered control. At the design stage, adequate shielding should be provided to give an acceptable level of protection to the workers during normal as well as abnormal situations. The adequacy of the shielding in abnormal conditions, including accident situations leading to maximum foreseeable (worst-case) radiological consequences, should be evaluated and, where necessary, additional shielding or other engineered controls (e.g. interlocks) should be considered. Particular attention should be given on the transfer and storage of spent fuel and active components removed from the reactor core. Adequacy of shielding should be tested during fabrication and installation. These shields should be designed for easy to assemble and disassemble in the work locations.

4.11.2 Permanent Shielding

To reduce radiation levels at all accessible locations of the plant to acceptable levels, plant systems/equipment and Embedded Penetrations (EPs) (wall /floor openings) should be shielded against gamma and / or neutron radiation as applicable. The shield design should take into consideration the following factors:

i) Source strength of both normal and transient sources and build-up of long-lived activity during the lifetime of the plant.

ii) Source strength for accident conditions

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

iv) Selection of shielding materials should be made on the basis of shielding properties, structural properties, space availability and weight considerations.

v) The shielding efficiency of materials should not be affected by environmental conditions (e.g. temperature, pressure, humidity & radiation) or process conditions

(.e.g. water-filled end-shields in PHWRs should be provided with venting provision to ensure their complete filling irrespective of process temperature.).

vi) Adequate precautions should be taken during design stage that the neutron and gamma shielding provided should itself not become a source of radiation due to activation or production of secondary radiation resulting from interactions with the primary radiation.

vii) If any augmentation is required to be done in the permanent shielding already provided in the plant design, then it should qualify, inter-alia, the requisite seismic considerations, so that their failure under seismic event does not jeopardize any safety related SSCs.

4.11.3 Temporary Shielding

i) Provision should be made in design for installation of temporary shielding (such as lead or concrete bricks, lead mats) for reducing radiation levels during maintenance, in-service inspection (ISI) activities etc.
ii) The concerned area/location should have adequate space and its floor should have the required load-carrying strength.
iii) Also, certain areas in the plant should be provided with shielding enclosures for temporary storage of active equipment/ components for maintenance.

4.11.4 Shield Penetrations

4.11.4.1 Penetrations are required in bulk shields for various purposes such as running pipelines, cable-trays, or for entry of personnel/equipment. They introduce additional path through which streaming of neutrons and gamma rays can occur unless carefully shielded.

4.11.4.2 The following requirements are to be fulfilled in such cases:

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

a) Minimizing the cross-sectional area and number of all straight-through paths containing material of very low density (e.g. gases including air, water) and, wherever unavoidable, such straight penetration EPs should be oriented towards ceiling of the room (i.e. much above the radiation workers passage).

b) Providing shield plugs.

c) Placing shields of larger diameter than those of the penetrations, to cover the ends of the penetrations.

d) Providing zigzag or curved pathways in order to ensure that adequate shielding is available along any line-of-sight path.

e) Filling the gap between the penetrations and the structural wall/floor with grout or other compensatory shielding material like lead balls, borated rubber etc.

ii) Effect of any plant modifications done later on from process considerations should be critically reviewed with respect to its impact on loss of shielding and resulting in higher radiation levels in the adjoining areas.

4.12 Remote Handling

4.12.1 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/ maintenance of equipment. Some of the jobs may be semi-remote as these involve personnel entry in active areas for installation/ maintenance 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 exchangers and steam generator tubes, inspection of reactor core components (e.g. reactor pressure vessel in LWRs, coolant channels in PHWRs, main vessel triple weld joints in sodium cooled reactors etc) collection of active samples, activities involving exposure to high dose rates (e.g. handling of spent Self Powered Neutron Detectors (SPNDs), En-masse Coolant Channel Replacement (EMCCR) operations, etc.).

4.12.2 Remote handling operations should be aided by CCTV equipped with radiation resistant camera and adequate area-lighting, so that operator in inactive area can smoothly operate the remote handling tool to effect the task in active areas.

4.13 Decontamination

4.13.1 Provisions for the decontamination of equipment, piping, surfaces, etc., that are likely to have radioactive contamination, should be considered in design. All the components of the systems, which come in direct contact with primary coolant or moderator, should be considered as potential items requiring decontamination. Decontamination provisions for floors, walls, drains, etc., in the rooms, where active systems are located, which may get contaminated should be considered.

4.13.2 Depending upon the system to be decontaminated, provisions should be made for the following types of decontamination techniques:

i) On-line decontamination of circuit tanks/piping/equipment using suitable chemical decontamination flushing and/or ultrasonic decontamination, as feasible.
ii) Decontamination of active components and equipment at their locations.
iii) Off-line decontamination of contaminated equipment by transporting it to a centralised decontamination centre.

4.14 Control of Activity in Coolant and Moderator Systems

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

4.14.2 Inventory of activated corrosion and erosion products in the systems should be minimised by:

i) Equipment design features provided to limit erosion (e.g., protective layers, smooth surfaces or design flow rates and baffles to help reduce erosion)
ii) Chemistry controls to minimize corrosion

4.14.3 Proper selection of materials e.g. use of low cobalt impurity materials. The activity build-up in these systems should be minimised by the following:

i) On-line detection/ sampling system should be part of design of NPP to enable early detection of failed fuel.

ii) Clean-up systems (e.g., filters, ion exchange resin) should be provided. Their capacities should be adequate to cope with activity spikes during start-up, cool down and depressurisation phases. Features should be provided to control release of activated corrosion products due to change in system conditions during shutdown/ maintenance.

iii) Clean up systems (e.g., filters, ion exchange resin, dryers, re-combiner) for cover gas should be provided.

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

a) As far as possible sharp bends and dead ends that can act as traps should be avoided.

b) Number of welded joints should be kept to a minimum and the welds should have minimum roughness.

c) Drains should be minimised in number and properly positioned to avoid residual stagnant pockets of liquid when the circuits are drained.

d) System tanks should have provisions for flushing and draining (to reduce crud/sludge build-up).

e) 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.

f) Vent lines should, as far as possible, run vertically from the pump bowl/ equipment.

4.14.4 To control air borne contamination, these circuits should be designed for maximum leak tightness by:

i) minimising the number of valves and components in the circuit, and
ii) use of leak tight valves (such as bellow-sealed valves, inter-gasket/ inter-gland/inter seal leak off collection provisions).

4.15 Ventilation System

4.15.1 General

To maintain appropriate ambient conditions (both temperature and air contamination control) in working areas, ventilation system should be provided in all active areas of the plant. For radiation protection purposes, the primary objective of providing a ventilation system should be to control the contamination of the working environment by airborne radionuclides and to reduce the need to wear respiratory protection. Careful attention should be given to the design of the ventilation and containment systems network, including the calculation and verification of rates and velocities of airflow, to ensure that they are adequate for controlling airborne contamination. The system should meet the following requirements:

i) The ventilation system design should ensure that air contamination levels in plant areas are maintained ALARA. In full occupancy areas, the air contamination levels should be kept below 0.1 DAC. The required number of air changes in partial occupancy areas should be arrived at by taking into account the air contamination potential specified for different areas.

ii) The airflow should be directed from regions of lower to higher contamination potential. Design should prevent spread of air contamination to lower contamination areas in case of failures like power failure, instrument air supply failure, choking of filters, cooling failure among others.

iii) To reduce air borne radioactive contamination in active areas, a closed loop air circulation system with dryers, filters and coolers along with a small purge flow should be employed.

iv) Ventilation exhausts of areas such as Fuel Transfer Room and Spent Fuel Storage area, where there is possibility of air contamination due to particulate & iodine activities, should have provisions for air filtration using suitable particulate and iodine filters.

v) Provisions should be made in design for Portable ventilation system in areas where airborne contamination may arise during maintenance. The exhaust should be discharged into the active ventilation exhaust system.

vi) The ventilation system should have sub-systems such that few or no modifications will be required during decommissioning.

vii) The vents of tanks containing radioactive fluids should be connected to the active exhaust ventilation system.

viii) Design should also envisage specific conditions during operation or shutdown like purging a system (e.g. Cover gas system/ Annulus Gas Monitoring System), in which case the design should have provision for the radioactive exhaust to be directed to stack through the active duct.

ix) The system should limit spread of contamination and environmental releases by provision of filters (of specified filtration efficiency) and maintaining pressure differential across the locations.

x) Elements (e.g. filters, fans) of cleanup equipment should have adequate redundancy in compliance with the safety classification of the relevant system, to ensure their reliability during maintenance and replacement of filter media.

4.15.2 Reactor Containment Isolation System and Survival Ventilation System during Accident Conditions

4.15.2.1 All the systems required to be operated under accident conditions should be powered by DEC power source to ensure their functionality during their mission time even in case of Station Black Out (SBO).

4.15.2.2 Containment Isolation System

i) The system configuration in terms of number of isolation dampers, location and mode of their operation (auto/manual, etc.) should be as per the AERB Safety Guide Containment System Design for Pressurised Heavy Water Reactors AERB/NPP-PHWR/SG/D-21, 2007.

ii) The system actuation should be reliable based on triplicated logic, and derived from diverse signals such as process signal (e.g. Containment pressure high), radiation based signal (e.g. Very High activity in ventilation duct) etc,.

iii) Redundant, diverse and fast acting system should be in place to minimise the radioactivity releases and dose to the public during accident conditions.

iv) In case of conflicting requirements wherein the isolation valves/dampers are to be closed manually by the operator after due diligence (e.g. Passive Decay Heat Removal System (PDHRS) vent line isolation valves normally kept open to facilitate PDHRS functionality in SBO may have to be manually closed under LOCA), the design should provide for remote manual actuation of the such isolators from MCR so that operator does not have to go into the field whereby he may get exposed to contaminated atmosphere. Electrical power supply to such valves should be ensured to be from reliable and uninterrupted power so that even in case of LOCA initiated Severe Accidents, its remote operability can be ensured.

v) Compressed air receivers should be provided for isolation system as per specified regulatory requirements.

4.15.2.3 Survival Ventilation System for Main Control Room

The main control room (MCR) including computer room should be treated as a survival area during emergency condition as outside air may be contaminated due to accidental release of radioactivity, and hence be provided with "Survival ventilation system".

i) The system should comprise, as a first line of defence, dedicated battery of air-bottles/canisters stored at secured location for the purpose.

ii) Provision should be made so that the normal fresh air supply to the A/C system for the above areas can be cut-off in case of radioactivity release or toxic gas release in the atmosphere and the survival ventilation system be valved in.

iii) Outside air, if at all required to be used for make-up, should be only after adequate filtration through pre-filters, combined absolute and charcoal filters. Intake point for outside air into the survival ventilation system to be resorted to during such an exigency should be located away from the contaminated exhaust of the nearby buildings. Also, consideration should be given in design such that its functionality should not be affected by any common cause which has rendered the normal ventilation unusable (e.g. failure of electric power supply to system components, Instrumentation and relevant computer based systems, structural failure during seismic event, inundation in extreme flood, etc).

4.16 Containment Atmosphere Clean up System for Accident Conditions

4.16.1 There should be systems provided in design to cleanup Containment atmosphere (Primary Containment Filtration & Pump Back System (PCFPB) and/or Vapour Suppression Pool (VSP), or Containment Spray System (CSS), following an accident, so that ground level release of radionuclides is minimised.

4.16.2 These systems should be designed in such a way that they have extensive coverage within various compartments of Containment with respect to removal of airborne radionuclides with a focus on the most active compartment. The decontamination factors should be as per AERB Safety Document [3].

4.16.3 Specific requirements for these systems are given below:

i) Primary Containment Filtration and Pump Back System

a) Adequate consideration should be given to the filter-bank temperature rise and control.

b) Design should give consideration for periodic surveillance on PCFPB filter-banks for their efficiency and replacement.

ii) Vapour Suppression Pool

Chemical dosing of VSP water should be ensured so that it can effectively trap the radionuclides in the steam-air mixture passing through it, and also prevents its re-suspension later on. Appropriate level should be maintained to ensure the effectiveness of VSP system. The associated systems e.g. pumps, strainers, pipes etc should be periodically checked and inspected.

iii) Containment Spray System

a) The CSS spray water should be chemically dosed to have its chemistry commensurate with the intent of maximum possible radioactivity removal.

b) Design should provide inspection platform for periodic surveillance of CSS header and checking of thoroughness of individual spray nozzles.

c) Provision may be made so that CSS header with its nozzles can be used under Design Extension Conditions for spraying water in Containment by injecting fire water through external hook-up.

4.17 Waste Management System

This section deals with management of effluents, viz.: liquid & gaseous waste. Details on solid waste management plant design are given in AERB Safety Guides [4], [5], [6].

4.17.1 Liquid Waste Management Systems for Operational States and Decommissioning

i) The design of liquid waste management systems should take account of all the applicable requirements for protection of site personnel such as:

   a) Implementation of radiological zoning
   b) Provision of shielding
   c) Use of remote techniques.

ii) The design should also consider the provisions for storage and subsequent management of off-normal waste (e.g. liquid waste generated during SG tube failure, moderator heat exchanger (HX) failure etc.).

iii) In order to manage the interim storage and treatment of liquid wastes optimally, the design should incorporate a liquid waste segregation system, that would segregate liquid waste from different process at source and accordingly sent for treatment and conditioning before discharging to environment.

iv) Considering the direct bearing of liquid effluents in the aquatic route of the environment, efforts should be made in the design to minimise liquid effluents, (e.g by converting liquid waste to gaseous effluents using evaporator system).

v) Liquid effluents discharge point to the receiving water body should be chosen to ensure adequate dilution and prevent back-flow of discharged effluents to the plant water intake. In case of limitations for dilution in receiving water body, a suitable engineering provisions should be made available to achieve the adequate dilution.

vi) The monitoring point should be established where a representative sample can be obtained. The monitoring system should be designed for detecting the type of activity released, e.g. gross Beta, tritium etc.

vii) On high activity detected in the effluent discharge stream, the pump-out should be automatically stopped through an appropriate interlock logic.

viii) In case of multi-unit site, single point discharge (main outfall) is preferable for exercising effective administrative control on the environmental release. However, this warrants a system in place to properly assign the discharges to the respective source stations.

ix) The plant design should be such that adequate barriers are provided against inadvertent release of radioactivity through the liquid route (e.g. direct cooling of active HX by atmospheric water body should not be resorted to.).

x) Provisions should be made available to augment the existing capacity during decommissioning.

4.17.2 Gaseous Effluent Discharge during Operational State

i) These effluents should be discharged into the atmosphere after filtration for removal of particulate radioactivity.

ii) To avoid the possibility of out-leakages, the ducts carrying gaseous waste should be leak-tight.

iii) The ducting should not run through high occupancy area so as to keep the personnel dose minimum.

iv) Design should ensure that the ventilation ducting does not pose direct streaming path for radiation from high active area to low active area.

4.17.3 Gaseous Effluents Discharge during Accident Conditions

The systems employed to discharge contaminated atmosphere of Containment to outside environment under accident conditions should satisfy the following requirements:

i) A filtration/scrubbing mechanism for active ingredients should be provided in the exhaust. Provision for containment and decay of short-lived radionuclides (e.g. short-lived noble gases) is a desirable feature.

ii) Monitoring of discharges and availability of their reading in MCR as well as in on site Emergency Support Centre (OESC) should be ensured. For DECs, additional monitoring provisions may also be made available in OESC.

iii) The system should have provision of remote operation (manually) so that emergency crew does not have to go to the field wherein the location/ area following accident will be contaminated.

iv) The releases from these systems should be at an elevation (through stack) thereby taking credit of atmospheric dilution.

v) The source term should be estimated as per approved methodology [3] factoring in (also the Containment leak rates) the decontamination factor as specified therein for respective type of radionuclides (particulate, elemental, organic, etc.).

5. PROVISIONS FOR MONITORING OF RADIATION PROTECTION

5.1 General

5.1.1 Radiation monitoring is required to ensure protection of site personnel and public. This should be done using a combination of fixed devices and portable devices for workplace monitoring as well as personnel monitoring devices for individual monitoring.

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

5.1.3 The monitoring should include the parameters and quantities, type, the methodology (including the location and frequency of data collection), the required resolution and precision of any measurements.

5.2 Operational States and Decommissioning

5.2.1 General

i) For an effective implementation of the radiation protection of site personnel and the public in design, a well-planned radiation monitoring programme for both plant operation and decommissioning should be established comprising of the following major aspects:

   a) Personnel monitoring (both internal and external).
   b) Area monitoring.
   c) Process monitoring.
   d) Effluent monitoring.
   e) Environmental monitoring (including background radioactivity measurements).

ii) Equipment for implementing the above programme should be provided in design. A typical list of different monitoring systems required is given in Appendix -IV.

iii) The rationale for the selection of ranges, alarm set-points and locations for the monitoring systems should be documented at the design stage. Adequate reliability/ redundancy of the equipment should be ensured. Some important features of the installed monitoring systems are given in Appendix V.

iv) The guidance provided for radiation monitoring during plant operation is generally applicable for decommissioning also. However, since decommissioning is multi-phase activity, some of the initial monitoring equipment may need to be removed or some new monitoring provisions may be required for decommissioning purposes. These should be appropriately considered in the design of radiation monitoring system.

5.2.2 Personnel Monitoring

5.2.2.1 The personnel monitoring arrangements should include

i) External dose (Whole body, extremity, skin and eye lens) by measuring devices (Thermoluminescence dosimeters (TLDs), Direct Reading Dosimeters (DRDs)/ Electronic Personnel Dosimeters (EPDs), eye lens dosimeters, neutron dosimeters, etc.)

ii) Internal dose measurement techniques (bioassay, Whole Body counting (WBC), etc.)

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

5.2.2.3 NPP should be registered on NDRS (National Dose Registry System) e.g. NODRS (National Occupational Dose Registry System developed by BARC).

5.2.2.4 To meet the necessary accuracy and precision, individual dosimetry should be performed by an accredited dosimetry service.

5.2.2.5 Further details on personnel monitoring are given in AERB safety guide "Radiation Protection during Operation of NPPs" (AERB/SG/O-5) and AERB Safety Guide on Monitoring and Assessment of Occupational Exposure due to Intake of Radionuclides (AERB/NRF/SG/RP-1).

5.2.3 Area Monitoring

5.2.3.1 Area monitoring includes measurement of radiation dose rates, airborne contamination levels and surface contamination levels in different areas of the plant. Passive detectors (e.g. TLDs) might be deployed for backup and retrospective evaluation of radiation environment.

5.2.3.2 Area monitoring system provided in design should include the following:

i) Monitors for measurement of external dose rates
ii) Air contamination monitors
iii) Surface contamination monitors

5.2.3.3 The continuously operating area radiation monitors with a local alarm should be installed at appropriate locations in the controlled areas at the plant. The suggested ranges for area monitors are given in Appendix VI. As a general consideration, radiation monitors should be installed at

i) Various areas of reactor building and reactor auxiliary building including ventilation duct.
ii) Spent fuel storage facility.
iii) System/ areas associated with fuel handling including transport route for fuel.
iv) Radioactive waste management facilities.
v) Decontamination facilities.
vi) Active equipment maintenance workshops.
vii) On various lines carrying highly active reactor fluids as deemed appropriate.
viii) Emergency response facilities.

5.2.3.4 Provisions for monitoring should be made for other radioactive areas e.g. fresh fuel storage, active waste storage etc. Provisions should be made deployment of portable radiation monitors with built-in alarms for dose rates for monitoring of special maintenance operations of short duration and for monitoring in areas where potentially high radiation levels may occur.

5.2.3.5 Permanently installed monitors for detection of air borne contamination should be provided at appropriate locations in NPPs. The activity concentration in air should be determined, at least for those accessible rooms of the controlled area where airborne radioactive substances may be present in amounts that could influence the radiation doses to workers. In selection of monitors, physical and chemical form of the radionuclides should be taken into account. Provisions should be made for obtaining the representative sample. As a general requirement, airborne contamination level monitor should be installed at

i) Areas of reactor building and reactor auxiliary building.

ii) Ventilation duct from Reactor Buildings.

iii) Fuel transfer room and spent fuel storage facility.

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

5.2.3.6 Surface contamination monitors should be installed at a few locations to enable checking of contamination status on the equipment and nearby floors.

5.2.3.7 Personal contamination monitors (both friskers and hand/foot contamination monitors) should be installed at inter-zonal transition points for use by personnel before exiting from higher contamination 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 from the controlled areas of the plant.

5.2.3.8 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.

5.2.3.9 Radiation monitoring instruments should be permanently installed at workplaces where the possibility of a sudden unexpected increase in exposure necessitates continuous monitoring. The display should be routed to a control room, where appropriate, for initiating prompt action.

5.2.4 Process/ System Monitoring

5.2.4.1 The purpose of process/ system activity monitoring is to detect fuel failures and equipment/ component malfunction or failure.

5.2.4.2 System for detection of fuel failure should be provided. This should be done by measuring specific fission products activity in the coolant systems or measurement in off-gas systems. The system should work (online or offline) to locate and identify the specific fuel bundle/assembly.

5.2.4.3 Radiation measuring equipment for beta gamma activity should be installed for monitoring activity concentration in:

i) Primary circuit coolant water.

ii) Main steam lines.

iii) Ventilation exhaust ducts.

iv) Process Water outlet lines.

For HWRs, these lines should be monitored for tritium activity also.

5.2.5 Effluent Monitoring

5.2.5.1 The gaseous effluents discharged through the stack should be monitored and recorded. This should include monitoring of Noble Gases (Fission Products and Activation Products), Aerosols (Particulates), tritium (for HWRs), Iodine (^{131}I), ^{14}C etc. The monitoring should be done for both gross activity and composition of effluent discharges.

5.2.5.2 The monitoring should be continuous and must have stand-by equipment with provisions of changeover. For radionuclides where online monitoring is not feasible due to technical issues, continuous sampling with laboratory measurements (typically once a day) should be ensured.

5.2.5.3 Additionally, the system/ equipment/ building contributing major part of effluents may separately be monitored for better control and accounting.

5.2.5.4 Activity monitor should be installed at the liquid effluent discharge line to plant Main outfall. This monitor should have the sensitivity to detect activity concentration at the limiting level stipulated by AERB. The plant Main Outfall line should also be provided with a continuous outfall sampling system at appropriate location ensuring adequate dilution.

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

5.2.6 Environmental Monitoring

5.2.6.1 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 should be done in the following two areas;

i) On-site monitoring-within the site boundary (up to exclusion zone boundary).

ii) Off-site monitoring beyond the exclusion zone boundary.

5.2.6.2 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 and placement of these monitors should be adequate to cover normal operation and provide indication for any accident releases as well as to determine the direction of dispersion during the accident situation. These monitoring instruments should also be installed in the direction of water ways, if reasonably possible and required.

ii) The readings of these monitors should be available both locally and in the main control room. These should be provided with battery backup in case of loss of power till the restoration of availability of another mode of measurement (preferably 48 hours). The provisions for recording and storage of the data should be provided for time as deemed appropriate. The guidance on number and locations of these monitors has been provided in Appendix VII.

iii) Thermo luminescent dosimeters (TLDs) installed around the plant.

iv) Borewell water sampling system around radioactive waste disposal area.

v) Monitoring of storm water drains.

5.2.6.3 The off-site (beyond exclusion zone boundary) environmental surveillance programme should be carried out by setting up an environmental survey and micrometeorological laboratory. The laboratory should have equipment and facilities for detection/analysis of very low levels of radioactivity in the environmental samples. This laboratory should also provide baseline 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. The details of exposure pathways and dose assessment is covered in safety guide “Methodologies for Environmental Radiation Dose Assessment (AERB/NF/SG/S-5)”.

5.2.7 Monitoring during Decommissioning Stage

5.2.7.1 In designing the plant, the requirements for radiation protection during decommissioning of the plant should be taken into account. The dose limits and radiation protection principles are the same for decommissioning as those for the operations of the reactor. The radiation sources and amounts of activity during the decommissioning of a nuclear power plant should be assessed in the design phase of the plant.

5.2.7.2 In general, the radiological conditions will not change abruptly during decommissioning as expected during operation. The major changes are expected only during dismantling of the structures. Thus, provisions should be made available in design for assessment of radiological conditions during such activities. These should include provisions of power supply and installation of monitoring systems at appropriate locations.

5.2.7.3 Additionally, the waste generated during decommissioning will be of different type and mainly consists of low active large volume waste with some components having very high dose rates and contamination levels. The facility for measurements of such wastes should be assessed and provided in the design.

5.2.7.4 Further, during decommissioning, need will arise to access locations hitherto inaccessible, during reactor operations and radiological conditions will be unknown. The design should provide for monitoring and handling of such conditions.

5.2.7.5 There will be large number of personnel working in radioactive areas and in varying radiological conditions during decommissioning. The design should consider personnel monitoring of these workers and facilitate suitable dosimetry services.

5.3 Accident Conditions

5.3.1 General

The accident detection and monitoring equipment should be provided to enable the operator to assess accident conditions and take necessary corrective actions in specified time period. The expected radiological conditions both within the plant and the environment during the postulated accidents should be considered for selection of sensitivity and range of the instruments. The instruments should be capable of performing satisfactorily under the worst environmental conditions anticipated during the accident. The power supply to continuous monitoring system should meet single failure criterion.

5.3.2 Plant Monitoring

5.3.2.1 The type of equipment/system should be available for monitoring during DBAs and as far as practicable in DECs;

i) High range radiation monitors in specified locations in reactor building.

ii) Provisions to obtain representative sample from reactor atmosphere (inert gases, particulates, iodine, tritium etc) and RB sump/ vapour suppression pool water.

iii) Portable monitoring instruments with ranges to serve accident conditions.

iv) System for sampling and measurement of activity in gaseous effluents.

v) Monitoring provisions in Spent Fuel Storage Area.

5.3.2.2 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 adequate shielding to reduce background due to contaminated environment in case of accidents and also instrumentation and ventilation provisions for handling and analysing these high active samples.

5.3.2.3 The continuous radiation monitoring system should be supplied with emergency power supply system. The data from these should be available in MCR and other designated location for emergency response personnel.

5.3.3 Environmental Monitoring

5.3.3.1 Besides the normal monitoring provided by Environmental survey laboratory (ESL) and Micrometeorological laboratory (MML), equipment and facilities should be available for monitoring during accidental conditions for initiating protective actions and remedial actions as per regulatory guidance. The radiation measurements are obtained during accident conditions from:

i) Environmental radiation monitors and TLDs located around the plant.

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

5.3.3.2 Decision Support System

An automatic external radiation-measuring network coupled with a modelling software capable of determining the source term on basis of real-time environmental radiation levels should be provided. The system should be capable of providing support to decision makers in taking protective actions in combination with prevailing metrological and demographic conditions. The system output should be visible in MCR and OESC.

6. ORGANIZATIONAL ASPECTS, HUMAN RESOURCES AND QUALITY ASSURANCE

6.1 General

Adequate provisions, for radiation protection, made during design stage are very helpful during operation stage of NPP. Responsible organization should establish system/ mechanism to implement and oversee the radiation protection aspects in design. As the radiation protection aspects cut across other aspects of design also, interfaces should be established to ensure proper coordination.

6.2 Organizational Aspects

6.2.1 The responsible organization should have strong commitment for to ensure radiation safety in design and should make systematic efforts to induce safety awareness and propagate safety culture amongst staff. The responsible organization should review the provisions as highlighted in this guide to determine the adequacy for successful implementation of the Radiation Protection Programme.

6.2.2 Since radiation protection has many interfaces with other sub-areas within the construction of a nuclear power plant, those interfaces should be acknowledged in order to reach good results in the design and radiological planning.

6.3 Human Resources

6.3.1 The responsible organization should ensure availability of sufficient number of technically qualified and adequately trained staff. The design team should be fully aware of the radiological protection measures that should be incorporated into the design. Further, interfaces with several technical areas should be taken in account when training personnel in radiation protection.

6.3.2 Design organizations should invite experts from relevant operating organizations to participate in activities in relation to the design of new plants and design modifications to an existing plant, to assist in ensuring that the requirements for radiation protection and waste management are met. Moreover, the applicable operating experience should be transferred to the design organization. In this way, the interrelation between design aspects and operational procedures can be properly taken into account.

6.3.3 In order to implement this structured approach, the design organization should have an optimization culture, in which the importance of radiation protection is recognised at each stage of the design. An optimisation culture is established by ensuring that all participants in a project are aware of the general requirements for ensuring radiation protection and of the direct and indirect impact of their individual activities or functions on the provision of radiation protection for site personnel and members of the public.

6.3.4 The safety analysis reports of the nuclear power plant and the documentation of the systems, structures and components should contain a description of how the requirements addressed in this Guide will be fulfilled or have been fulfilled in design of the nuclear power plant. The reports should also contain information on the most important radiological safety aspects by means of which the ALARA principle has been complied within the design of the nuclear power plant.

6.3.5 The dose assessment and the factors leading to doses should be documented in the safety analysis reports. The expected dose rates in the working areas, factors that may cause changes in the dose rates and possible corrective measures should be documented. Additionally, assessment for dose rates, radioactivity in reactor systems, working time, number of workers required should be analysed and documented for major jobs that are expected to be carried out during the initial years of operation for comparison purposes.

6.4 Quality Assurance

A comprehensive quality assurance program should be established to ensure the effectiveness of assessment and execution of engineering activities performed during different stages to evaluate radiation protection aspects of the nuclear power plant. All the parameters and assumptions used in dose computation including shielding calculations should be thoroughly reviewed and properly documented. Validated models should be used for analysis purpose.

APPENDIX- I

RADIATION SOURCES IN OPERATIONAL STATES AND DECOMMISSIONING

I.1 General

The identification of all sources of radiation is an important task as they can affect the radiation levels throughout the plant areas. All practical means need to be employed for reducing the radiation sources in the nuclear power plant. The subsequent paragraphs give guidance for consideration of radiation sources and approaches to reduce them to the extent practicable.

I.2 Reactor Core and Vessel

I.2.1 During power operation, intense neutrons and gamma rays are emitted from the core as a result of fission process and the decay of fission products. Neutrons and gamma rays are also emitted as a result of neutron capture in the core and the surrounding material, which acts as supplementary source during shutdown periods and will be a major source of radiation during the decommissioning of the plant. The most significant are usually the isotopes of the noble gases, iodine and caesium, but others such as strontium and the isotopes of plutonium may also be important.

I.3. Reactor Coolant and Moderator System

I.3.1 Fission products that are released from fuel pins with defective cladding are a source of radiation in the reactor coolant. These need to be minimized by reducing fuel cladding defects and removal of failed fuel from the core as soon as possible. In addition, for PWRs, a spiking phenomenon is observed for fission products during the shutdown phase. If the defects are of larger size, water can enter the fuel pin and wash out the fission products. Fission products also enter the coolant from residual surface contamination of the cladding by uranium (it will be in ppm). The tramp uranium in the system should be limited to reduce the activity.

I.3.2 If fuel cladding defects is such that a non-negligible mass of fuel (in grams) is released to the coolant, then the alpha activity of the water and of the deposits need to be accounted for internal exposure when circuits and components are opened for maintenance and repair or during decommissioning.

I.3.3 If the coolant contains oxygen (in water cooled reactors), a major source of radiation during power operation will be ^{16}N, which is formed by the interaction of fast neutrons with ^{16}O. Since the half-life of ^{16}N is short (7.1 s), the significance of this isotope will be reduced where the transport time between the core and a component in the coolant system is long compared with the half-life, else this has to be appropriately considered in shielding. Other activation products of the coolant such as, ^{19}O and ^{18}F (Water cooled reactors) may also contribute to the radiation levels.

I.3.4 In HWRs, tritium is an important source of internal radiation exposure. Tritium as DTO is an important source in liquid and gaseous effluents. Photo neutrons emitted from interaction of gamma rays with deuterium also forms another source of exposure.

I.3.5 For SFRs with sodium coolant, the dominant sources are ^{22}Na and ^{24}Na. Shielding is required to yield acceptable dose rates on the operating floor. Tritium that is generated in the fuel by ternary fission is released to the primary coolant through the stainless steel cladding of the fuel (the principal mechanism is diffusion). Fission products such as iodine and caesium are released to the coolant if cladding defects occur. The Na coolant may be covered by an inert gas such as argon. The activation of the cover gas gives rise to active ^{39}Ar and ^{41}Ar.

I.3.6 Corrosion products present in coolant get activated due to temporary deposition in core and passage of coolant through core. Major contributors are ^{60}Co, ^{58}Co, ^{110m}Ag, ^{124}Sb, ^{54}Mn, ^{59}Fe and ^{51}Cr especially during repair and maintenance activities. These need to be minimized by a) reducing erosion and corrosion rate by proper material selection and chemistry control, b) provision of corrosion/erosion product removal systems and c) surface treatment to reduce adhesiveness among others.

I.3.7 The presence of materials with a high cobalt content (which produces the activation product ^{60}Co), needs to be reduced by applying the optimization principle. Other materials such as silver (for example in control rods and seals) and antimony (used in seals and pump bearings) that produce activation products contributing significantly to occupational exposure need also be reduced or eliminated by applying optimisation processes.

I.3.8 In the cleanup systems of water cooled and moderated reactors, there will be an accumulation of radioactive material in filters and ion exchange resins. This will consist of fission products such as iodine and caesium that have escaped to the coolant through fuel cladding defects, and of radioactive corrosion products that are transported by the coolant or moderator. All components in which an accumulation of radioactive products occurs, will generate very high activities that require shielding. Radioactive noble gases may be formed in these filters by the decay of iodine isotopes. In HWRs, photo neutrons are produced in the heavy water by the photons from ^{16}N. This source is significant in determining the shielding requirements of the coolant circuit external to the core.

I.3.9 Carbon-14 is produced in LWRs and HWRs by (n, α) reactions with the ^{17}O present in the oxide fuel and moderator, by (n, p) reactions with the ^{14}N present in impurities in the fuel and by ternary fission. Because of the large moderator mass, ^{14}C is produced mainly from ^{17}O reactions in the moderator in HWRs. The sources of radiation will be C-14 contamination levels on internal of moderator heat exchangers and accumulation in moderator ion exchange resin. This may also be the main contributor to the global long term collective dose commitment.

I.4 Steam and Turbine System

I.4.1 In direct cycle water reactors like BWRs, ^{16}N, which is carried over to the steam phase, will be the major source of radiation during power operation. The sky shine effect needs to be checked carefully for buildings with potentially light structures, such as the roof of the turbine building. Downstream from the condenser, ^{19}O also needs to be considered as a major source of radiation. In the event of fuel pin failures, an additional source of radiation will be volatile fission products, mainly the noble gases, and volatile fission products such as iodine and caesium. During power operation, this source will be of minor importance compared with ^{16}N, but after reactor shutdown, these isotopes and their progeny (e.g. ^{140}Ba) will be the major radiation source in this system. Another source may be non-volatile corrosion products that are carried over in steam.

I.4.2 In PWRs and PHWRs, the steam and turbine system are separated from the radioactive systems by a material barrier (the heat exchanger tubes). Thus, in these reactors radioactive material can only reach the steam and turbine system if leaks occur between the primary and secondary circuits. If the leak rates are monitored (e.g. by measurement of the activity of the water of ^{16}N in the secondary circuit) and kept to such a level that the activity in the secondary system is low, protective measures against direct and scattered radiation from this system are not necessary. However, provision needs to be made for cleaning the fluid circuits and for waste disposal from the secondary side in case primary to secondary leaks do occur. The leakage of primary coolant to the secondary circuit can also be detected by monitoring tritium in the feedwater.

I.4.3 In direct cycle plants, an additional source of secondary system contamination that needs to be considered is leakage from equipment for concentrating radioactive waste that involves steam heating. One such source of contamination is through tube leaks that allow contaminated waste to enter the condensed heating steam. Contaminated condensed water from such steam may then be introduced into the secondary system.

I.4.4 In SFRs, the secondary sodium coolant may become activated to ^{22}Na and ^{24}Na. This can give rise to dose rates in parts of the buildings outside the containment if the delay for the sodium transport from the steam generator to these areas is not long compared with the half-lives of ^{22}Na and ^{24}Na.

I.5. Waste Treatment Systems

I.5.1 The composition of liquid wastes (i.e. activity concentration and solid and chemical content) varies according to their origin. This mainly contains fission and activation products. The segregation of liquid wastes could be made in accordance with the following categories:

- High purity liquids (e.g. leakage wastes from the primary circuit of PWRs during power operation);
- High chemical content liquids (e.g. decontamination, fluids/ effluents);
- High solid content liquids (e.g. liquid wastes from floor drains);
- Liquid wastes containing detergents (e.g. liquid wastes from laundry drains and personnel showers);
- Liquid wastes containing oil (e.g. liquid wastes from floor drains from the area of the lubricating oil tank for the circulator);
- Tritium containing liquid wastes (for PHWRs).

I.5.2 Since the liquid waste treatment system processes active liquids, radioactive substances will accumulate in parts of the system such as filters, ion exchangers and evaporators.

In most cases, the accumulated radionuclide content consists of activated material such as ^{60}Co, ^{58}Co, ^{51}Cr, ^{54}Mn and ^{59}Fe (depending upon the composition and corrosion rates of the material used in the primary circuit). Fission products such as isotopes of iodine, caesium and strontium may be important if failure of fuel cladding occurs.

I.5.3 The major sources in gaseous waste consists of

- Fission gases (including gases formed from decay of fission products) released from the fuel through fuel cladding defects
- Small quantities of radioiodine's and particulates
- Activation gases in coolant (e.g.^{16}N, ^{19}O, ^{3}H etc)
- Activation gases for various associated gaseous systems (e.g. cover gas system, annulus gas system etc).

I.5.4 These gases need to be removed from the source of generation, treated and discharged to the atmosphere. Components such as holdup tanks, holdup pipes, charcoal delay beds or cryogenic devices are to be provided in the gas treatment systems to delay the release to the environment sufficient to allow for a large fraction of the radioactive material to decay. Vents for these gases need to be so located that the radioactive substances they contain are kept away from the occupancy areas.

I.5.5 The following constitute the major solid radioactive wastes:

- Components and structures that become activated or contaminated and have to be removed (e.g. control rods, neutron source assemblies, defective pumps, flux measuring assemblies, structures or parts thereof);
- Ion exchange resins, filter material, filter coating material, catalysts, desiccants and similar;
- Concentrates from evaporators, precipitates;
- Contaminated tools;
- Contaminated clothing, towels, plastic sheet, wet and dry mops.

I.5.6 In most cases, long lived activation products such as ^{60}Co and when fuel cladding defects have occurred, long lived fission products (particularly ^{134}Cs and ^{137}Cs) are the major radiation sources.

I.6 Spent Fuel

I.6.1 The major sources associated with spent fuel includes fission and activation products (transuranics) and activation products in associated components. For on-load refuelling systems, delayed neutrons that are emitted from the fuel while it is in the refuelling system also have to be taken into account.

I.6.2 For wet fuel storage and handling systems, water clean-up systems with particulate filtration and ion exchange need to be provided. The radioactive content of the water is removed by the filters and ion exchange resins, which themselves become sources of radiation. Contamination of the handling, clean-up and heat removal systems also gives rise to additional sources.

I.7 New Fuel

I.7.1 The activity of new fuel (natural UO_2) is generally low and is of not much significance as source of radiation.

I.7.2 However, in the case of mixed oxide fuel, the new fuel may be radioactive as a result of the recycled plutonium or recycled uranium. In this case, the new fuel will be a significant source of both neutrons and gamma rays~~.~~ The magnitude of the neutron source term will depend upon the time that has elapsed since the plutonium was created, since actinides that emit neutrons will be produced as the plutonium decays.

I.7.3 In the case of ^{232}Th–^{233}U fuel, the new fuel may be highly radioactive owing to the presence of ^{232}U progeny (such as ^{208}Tl and ^{212}Bi). It needs to be shielded and contained at all times until it is inserted into the reactor.

I.8 Decontamination Facilities

I.8.1 The radioactive material in the waste solutions consists mainly of the corrosion products containing radionuclides such as ^{60}Co, ^{58}Co, ^{51}Cr, ^{59}Fe, ^{54}Mn. This material arises from the decontamination of components, of contaminated areas, of reusable protective clothing and possibly also of personnel in the facilities that are provided to remove radioactive contamination from the surfaces. Whereas the activity concentrations in the waste arising from the decontamination of personnel and of clothing are low, concentrations may be medium or high in solutions arising from the decontamination of components before major repair work.

I.9 Miscellaneous Sources

There are also other sources of radiation at nuclear power plants, such as neutron startup sources, corrosion samples, in-core and ex-core detectors, calibration sources for instruments and sources that are used for radiographic inspection.

APPENDIX-II

DESIGN TARGETS

Collective Dose of Radiation Workers

The design dose targets should be set taking into consideration following factors

- Collective dose for the best performing similar type of NPP.
- Identifying major activities with high collective dose (typically 50-70% of collective dose budget) and ensuring dose optimization for these activities
- Estimating annual collective dose for a period of 10 year.

It is always prudent to set design targets for the collective dose to the groups of workers that are likely to receive the highest doses.

Dose to Members of the Public

The design target should take into consideration operating experience and use of best available technology for optimization.

Radiation Dose Rates in Various Areas

Design target for radiation dose rate should be as per the zoning criteria specified in Appendix III.

Fuel Failure Rate:- Below 0.1% (percentage of suspected failed fuel bundles/ assemblies to total number of fuel bundles/ assemblies discharged in a specific time period)

APPENDIX-III

METHODOLOGY FOR RADIOLOZICAL ZONING

III.1 Zoning In Controlled Area

Zoning is done for effective radiological monitoring, radiation protection surveillance, individual dose control and contamination control.
The various methodologies for zoning classification as practiced are as follows;

1. **PHWR Type Reactors**– In this case, the zoning is based on minimizing spread of contamination. The various zones as separated by inter-zonal monitors (Four zones) can be used for zoning purpose. If required, zones can be merged provided the requirements of zoning as specified are met. In such cases, even three zones may also be sufficient for contamination control purpose. Typical zoning scheme can be depicted as follows;

Zones	**Surface Contamination**	**Airborne Contamination**	**Radiation Levels**	**Additional Considerations**
Zone I/ Supervised Areas	No surface contamination	No airborne contamination	Radiation level below 1 μSv/h	1. No restriction on personnel occupancy. 2. No need of individual monitoring, however periodic area monitoring may be undertaken.
Provisions for change room facilities, personnel decontamination and contamination monitoring should be provided at zone change.				
Controlled Area				
Zone II	No surface contamination (Maximum 0.37 Bq/cm^2 in case of infrequent occurrence)	Below 0.1 DAC	Maximum area design dose rate below 1 μSv/h	1. No radiation source is allowed in this zone. 2. Periodic area monitoring is mandatory. 3. Continuous occupancy 4. Individual monitoring needs to be undertaken.
Provisions for contamination monitoring and storage facilities for active clothes should be provided at the zone change.				

Zone III	Surface contamination if any for active work areas should be below 3.7 Bq/cm^2.	Below 0.1 DAC	a) Area design dose rate in full occupancy areas should be below 1 µSv/h. b) Area design dose rate in other areas should be below 5 µSv/h.	1. Occupancy should be controlled to optimize the exposure. 2. This area may include active maintenance workshops, places for decontamination, general areas of reactor auxiliary building etc. There may be certain areas within zone III that have potential for increase in airborne contamination and surface contamination during some jobs for a specified period of time. This area may include Ion Exchange column areas of some low active equipment. Such areas may be provided with additional appropriate design features to facilitate exposure control e.g. air tap points, vacuum mopping systems, etc.
Provisions for contamination monitoring, decontamination facilities and clothing facilities for active clothes should be provided at the zone change.				
Zone IV	Surface contamination will be of the order of 3.7 Bq/cm^2	Below 1 DAC in accessible areas.	Accessible area design dose rate should be below 40 µSv/hour. Further guidance on radiation	1. Individual monitoring is required for these areas. Additional, dosimeters may be required for

			levels in various areas on basis of occupancy is provided in Annexure I.	assessment of dynamic conditions of the areas. 2. This area may include passageways and other general areas in reactor building. Some equipment in zone IV may have high dose rates causing increase in area background levels. Such equipment should be provided with appropriate provisions for shielding. In zone IV, there may be areas having equipment with high radioactivity potential especially during reactor operation. Access to these areas should be administratively controlled. Specific shielding and ventilation provisions should be provided for entry to these areas.

Example: This zoning scheme corresponds to typical zoning scheme practiced in Indian PHWR type reactors. Other reactors such as SFRs are also designed based on this philosophy.

2. **PWR Type Reactors-** In this case, the access control is done based on radiation levels and it is possible that adjacent rooms may fall in different zones. The dose control is mainly through access control by administrative provisions (e.g. interlocks and lock & key arrangements). The

major factor to be considered is the occupancy of the areas and the frequency of access required to these areas. Typical scheme can be depicted as follows;

Area	**Radiation Levels**	**Remarks**
Free Access Area	Below 1 μSv/h	1. Unrestricted occupancy areas 2. Periodic monitoring of radiological conditions
Controlled Access Area		
Attended Areas	Below 10 μSv/h	1. Continuous occupancy areas 2. Low potential for airborne contamination
Periodically Attended Areas	Below 20 μSv/h	1. Periodic occupancy access areas 2. Occupancy control through RWP.
Restricted Access Area	More than >20 μSv/h	1. Entry after authorization from relevant authority 2. May be further sub-divided based on radiation levels
Unattended Areas	-	Entry only during Shutdown

The rubber areas and rubber change areas should be in controlled access areas to check the spread of contamination from locations of contaminated/potentially contaminated areas to cleaner areas. This control should be exercised independent of area classification as depicted above.

III.2 Design Radiation Levels in Plant Areas

The following guidance design radiation levels for various areas in reactor building on basis of occupancy could be used for shield design and placing of various equipment,

S. No	**Area and Occupancy**	**Radiation Level (μSv/h)**
1	**Accessible Areas**	
	8 hour/day	5
	4 hour/day	10
	2 hour/day	20
	1 hour/day	40
2	**Areas accessible during Shutdown**	
	General radiation level during shutdown	40
	Areas with limited occupancy (70 h/year)	150
3	There should not be any neutron radiation level in accessible areas.	

APPENDIX- IV

TYPICAL LIST OF TYPES OF MONITORING EQUIPMENT

S. No	Type of Monitoring System
1	Area radiation monitors (low range and high range), environmental monitors and accident monitors
2	Airborne contamination monitors including gaseous, particulate, iodine and tritium-in-air monitors.
3	Surface contamination monitors (Beta gamma)
4	Personal contamination monitors for inter zonal and final exit points (Beta gamma)
5	Portal monitors (Beta gamma)
6	Process radiation monitors
7	Effluents 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 beta activity, tritium (for HWRs), iodine, and inert gases)
8	Personal monitors such as Thermoluminescent Dosimeters (TLDs), Direct Reading Dosimeters (DRDs)/ Electronic Personnel Dosimeter (EPD) with alarm facility, neutron 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

APPENDIX-V

IMPORTANT FEATURES OF INSTALLED RADIATION MONITORING SYSTEMS

S. No.	Features
1	Range of measurement should cover reactor shutdown, normal operation and operational transients. Multiple-range instruments should have automatic range switching facility.
2	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.
3	The instruments should perform correctly in the environmental conditions in the field
4	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 backup power supply) - availability of testing and calibration provisions - Local monitoring units of Shutdown accessible areas monitors and accident monitors to be preferably located in accessible areas
5	The data obtained should be processed by a computerised system with trend monitoring, recording and display features in control room and Shift Health Physicist room.

APPENDIX-VI

TYPICAL RANGES OF AREA RADIATION MONITORS

Location	**Typical Range**
Normally inactive and high occupancy areas (e.g., Control Room, Service Area)	0-1000 µSv/h
Active/potentially active and continuously accessible areas (e.g. reactor building, waste management plant, cleanup room, decontamination centre, spent fuel storage area)	0-10 mSv/h
Shutdown Areas *(Reactor Building)	0-1 Sv/h
Accident Radiation Monitors (General reactor building areas, fuel integrity assessment, containment monitoring etc)	Upto 10^4 Sv/h

* Shutdown Area monitors should be able to cater to requirement for both reactor operation and Shutdown.

- *The recommended set points for high and very high alarms are twice and ten times the existing background radiation levels*

APPENDIX-VII

GUIDANCE ON NUMBER AND LOCATIONS OF ENVIRONMENTAL RADIATION MONITORS AROUND NUCLEAR POWER PLANT (NPP)

The continuously operating Environmental Radiation Monitors (ERMs) should be installed around the Nuclear Power Plant within Exclusion Zone (EZ) boundary to detect any increase in radiation background due to operation of the plant and to provide an indication of any abnormal release from the plant. The ERMs detect increase in radiation levels mainly due to fall out of plume or shine from a passing plume. The range of these monitors should be such as to cover the entire range of operation and to indicate the possibility of any abnormal operating conditions and accidental release.

The number and location of ERMs around the plant (for one NPP only) should be decided considering the following aspects;

1. **Area and direction**: Entire radial direction over land surface around the plant and in the water bodies (if this is reasonably possible) should be covered.
2. **Sectoral Coverage:** They should cover all sectors including of predominant wind direction, maximum population sector etc.
3. **Distance from the stack:** They should be located at such a distance from the stack so as to detect the increase in radiation level from passing plume for all stability classes. In general, a distance between 300-500 m may be considered appropriate for this purpose.
4. **Angular Direction and Width:** The angle should not be more than 60° with stack as the centre. The spacing should be so as to ensure detection of plume releases in all kinds of dispersion conditions. In general, 6 ERMs around the plant may be deployed.
5. **Shadow effect due to Infrastructure around Plant:** These should not be located very near to any building to avoid shadowing effects.
6. **Feasibility of location:** The location should ensure seamless flow of signal in all conditions and feasibility for periodic maintenance in all seasons.
7. **Multiunit site**: In case of multi-unit sites, consideration of releases from other units and their impact on the ERM should be taken care so as to avoid any confusion.

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LIST OF PARTICIPANTS OF WORKING GROUP

REVISION OF AERB SAFETY GUIDE ON RADIATION PROTECTION ASPECTS IN DESIGN FOR PRESSURISED HEAVY WATER REACTOR BASED NUCLEAR POWER PLANTS (AERB/SG/D-12)

Dates of meeting:	March 07, 2019 May 29, 2019 June 11, 2019

Members of the Working Group:

Shri Shyam Vyas (Convener)	:	AERB
Shri Soumen Sinha	:	AERB
Shri Virendra Dhotre	:	AERB
Shri N. Khandelwal	:	AERB
Smt Bharati Ingavale	:	AERB
Shri Vipin Chander	:	AERB
Shri Jaydeb Mandal	:	AERB
Shri Basuki Baral	:	AERB
Shri Nishikant Tyagi (Member Secretary)	:	AERB

LIST OF PARTICIPANTS OF TASK FORCE

REVIEW OF R0 DRAFT OF SAFETY GUIDE ON "RADIATION PROTECTION ASPECTS IN DESIGN OF NUCLEAR POWER PLANTS" (AERB/NPP/SG/D-12) AND PREPARATION OF DRAFT R1 OF SAFETY GUIDE

Dates of meeting:	September 01, 2021
	September 09, 2021
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	October 07, 2021
	October 21, 2021
	November 17, 2021
	December 01, 2021
	January 10, 2023
	January 17, 2023
	April 03, 2024

Members and Invitees of Task Force:

Shri Aloke Varshney, Former AD(HSE), NPCIL	:	Convener
Shri S.K. Pawar, DRP&E, AERB	:	Co-Convener
Shri K. Venkata Ramana, ACE (HP), HSE, NPCIL	:	Member
Shri Pradeep Bhargava, HS&EG, BARC	:	Member
Shri Rajnish Kumar, NPSD, AERB	:	Member
Shri Kuldeep Sangwan, OPSD, AERB	:	Member
Shri Jaydeb Mandal, DRP&E, AERB	:	Member
Shri Nishikant Tyagi, DRP&E, AERB	:	Member Secretary
Shri Bijendra Kumar, R&DD, AERB	:	Permanent Invitee

STANDING COMMITTEE FOR DOCUMENT DEVELOPMENT (SCDD-1)

Dates of meeting: March 17, 2023, August 04, 2023, February 02, 2024 and February 15-16, 2024

Members and Invitees of SCDD-1:

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Shri S.B. Chafle (Vice Chairman)	:	Former (AERB)
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Shri S Harikumar	:	AERB
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Shri SK Sinha	:	BARC
Shri Utkarsh S. C	:	AERB
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Shri Ranjay Sharan	:	NPCIL
Shri Rupak Agrawal	:	NPCIL
Shri Malay Pal (Member-Secretary)	:	AERB

ADVISORY COMMITTEE ON NUCLEAR AND RADIATION SAFETY (ACNRS)

Dates of meeting: December 14, 2019 and July 30, 2024

Members and Invitees of ACNRS:

Shri S.S. Bajaj (Chairman)	:	AERB (Former)
Shri D.K. Shukla	:	AERB (Former)
Shri S. B. Chafle	:	AERB
Dr. N. Ramamoorthy	:	BARC (Former)
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Shri K.V. Suresh Kumar	:	BHAVINI
Director (T), NPCIL	:	NPCIL
Director (T-LWR), NPCIL	:	NPCIL
Prof. C.V.R. Murthy	:	IIT, Madras
Shri H.S. Kushwaha	:	BARC (Former)
Shri K.K. Vaze	:	BARC (Former)
Shri S.K. Ghosh	:	BARC (Former)
Shri A.R. Sundararajan	:	AERB (Former)
Dr. A.N. Nandakumar	:	AERB (Former)
Dr. (Smt.) Sadhana Mohan	:	BARC (Former)
Dr. L.R. Bishnoi	:	AERB (Former)
Shri S. T. Swamy (Member-Secretary) (2019)	:	AERB (Former)
Dr. R. B. Solanki (Member-Secretary)	:	AERB

TECHNICAL EDITING
Shri S. A. Hussain, Former, Head, RASD, AERB

COPY EDITING
Shri K. Srivasista, Former, Head, R&DD, AERB

LIAISON OFFICER
Shri Bijendra Kumar, RDS, IT&RDD, AERB

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