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AERB DESIGN SAFETY GUIDE
(AERB/SG/ D-14)

CONTROL OF AIRBORNE RADIOACTIVE MATERIALS

IN

PRESSURISED HEAVY WATER REACTORS

GOVERNMENT OF INDIA
ATOMIC ENERGY REGULATORY BOARD
MUMBAI 400 094

FOREWORD

The assurance of safety of public and occupational workers and protection of the environment are important needs in the pursuance of activities for economic and social progress. These activities include the establishment and utilisation of nuclear facilities and use of radioactive sources and they have to be carried out in accordance with relevant provisions in the Atomic Energy Act 1962 (33 of 62).

Since the inception of nuclear power development in the country, maintaining high safety standards has been of prime importance. Recognising this aspect of nuclear power development, Government of India constituted Atomic Energy Regulatory Board (AERB) in November 1983 vide Standing Order No. 4772 notified in Gazette of India dated 31.12.1983. AERB has been entrusted with the responsibility of laying down safety standards and frame rules and regulations in respect of regulatory and safety functions envisaged under the Atomic Energy Act 1962. Under its programme of developing Safety Codes and Safety Guides, AERB has issued Codes of practice covering the following topics:

Safety in Nuclear Power Plant Siting

Safety in Nuclear Power Plant Design

Safety in Nuclear Power Plant Operation

Quality Assurance for safety in Nuclear Power Plants

The Safety Guides are issued to describe and make available methods of implementing specific parts of relevant Codes of Practice as acceptable to AERB. Methods and solution varying from those set out in the Guides may be acceptable if they provide at least comparable assurance that Nuclear Power Plants can be operated without undue risk to the health and safety of general public and plant personnel.

The Codes and Safety Guides are subject to revision as and when necessary in the light of experience as well as the current state of the art in science and technology. When an appendix is included in a document it is considered to be integral part of the document. Whereas annexures, foot notes, lists of participants and bibliography where included are only to provide information that might be helpful to the user.

In preparation of the Codes and Guides emphasis is on protection of site personnel and public from undue radiological hazard. However, for other 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 practice.

The plant designer, to ensure adequate safety, shall meet the general criteria specified in the Code of practice on Nuclear Power Plant Design. One of the

requirements in designing a nuclear power plant to meet these criteria is to assess the potential in the effect of initiating events including man-made events that may cause radiological consequences.

This Safety Guide on "Control of Airborne Radioactive Materials in PHWR" deals with safe operation of the NPP during normal operation and accident conditions. without exceeding acceptable design limits.

This Safety Guide has been prepared by the staffs of AERB, BARC, IGCAR and NPC professionals. This Safety Guide has been reviewed by experts and AERB Advisory Committees before issue. AERB wishes to thank all individuals and organisations who have contributed in the preparation, review and amendment of the Safety Guide. List of persons who have participated in the committee meetings and their organisations is included for information.

(P. Rama Rao)
Chairman, AERB

DEFINITIONS

Acceptable Limits
Limits acceptable to the regulatory body

Accident Conditions
Substantial deviation from operational states which are expected to be infrequent and which could lead to release of unacceptable quantities of radioactive materials if the relevant engineered safety features did not function as per design intent.

ALARA
An acronym for "As Low As Reasonably Achievable", a concept meaning that the design and use of radioactive sources (including nuclear facilities), and the practices associated with them, should be such as to ensure that exposures are kept as low as reasonably achievable, economic and social factors being taken into account .

Anticipated Operational Occurrences
All operational processes deviating from normal operation which may occur during the operating life of the plant and which in view of appropriate design provisions, neither cause any significant damage to Items Important to Safety nor lead to Accident Conditions.

Aerosols
A dispersion of very small particles and /or droplets in air.

Authorised Limits
Same as prescribed limits

Countermeasure
An action aimed at alleviating the consequences of an accident

Derived Limits
Values of measurable quantities, in a given , simplified exposure model corresponding to the dose limits recommended by the ICRP. Derived limits are the limits on concentration for the release rates and will provide quantitative limits between the measured concentration or release rates and the dose limits recommended.

Design Basis Accident
Accident conditions against which the NPP is designed according to established design criteria.

Dual Failure
A process system failure simultaneous with unavailability of a safety system.

Fail safe Design
Design such that logics and instrumentation would fail in the safe direction.

LOCA
Loss of coolant accident caused by pipe failures in the primary heat transport system.

Mission Time
Duration/period for which the clean up system operation must be ensured. Subsequent to this period the radionuclides would have decayed to such low levels /concentration which is considered safe from fire hazard point of view.

Normal Operation
Operation of a NPP within specified operational limits and conditions including starting up , shut down, power operation, maintenance, testing and refueling.

Operational States
The states defined under normal operation and anticipated operational occurrences together.

Prescribed Limits
Limits established or accepted by the regulatory body.

Redundancy
Provision of alternative (identical or diverse) elements or systems, so that any one can perform the required function, regardless of the state of operation or failure of any other.

Single Failure
A random failure, which results in the loss of capability of a component to perform its intended safety function. Consequential failures resulting from a single random occurrence are considered to be part of the Single Failure.

CONTENTS

1.0 INTRODUCTION

1.1 General

This safety guide forms part of AERB's programme in bringing out Codes, guides and associated manuals for safety in design of pressurised heavy water-based reactors (PHWRs). The intention of this safety guide is for the use of designers and operators dealing with the safe operation of nuclear power plants, environmental protection, radiological protection and waste management.

1.2 Objectives

The main objectives of this guide are to provide basic guidelines for:

- the design of systems and components to control airborne radioactive material within the working environment and its release to outside the environment in order to keep concentration of airborne radioactive materials within the limits stipulated by the Atomic Energy Regulatory Board (AERB) for normal and accident conditions.

- management of airborne radioactive materials.

1.3 Scope

The guide deals with a ventilation system with appropriate clean-up for the control of airborne radioactive material mainly for the reactor building.

2.0 DESIGN CRITERIA

2.1 General

Design criteria is based on radiological protection which is directed to avoid unnecessary radiation exposure and to keep the unavoidable radiation exposures as low as reasonably achievable. In order to achieve this objective, AERB prescribes primary dose limits to control the radiation exposures of plant personnel and the members of the public. These limits are based on the AERB/SG/D-12- “Radiological Protection” for the purpose of design.

2.2 Design Criteria for Air cleaning and Ventilation Systems.

2.2.1 Primary Dose Limit

The primary dose limit for the members of the public is apportioned for the different modes of releases such as air, water and land from a nuclear installation keeping some reserve for the future expansion. The apportioned limits are translated into derived working concentrations or release limits for various nuclides. Compliance with these secondary limits is the means by which the requirements of primary dose limits are met.

2.2.2 Sources of Airborne Radionuclides

The steps needed to restrict the exposure of individuals can be taken by applying actions in the network (see Annexure-I para A..2) at any point linking the sources (source here means airborne radionuclide) to individuals. These actions may be applied to the source, to the environment or to the individual. However, it is most effective to apply control measures to the source and that will be least disruptive also.

2.2.3 Systems for Control of Exposure

Adequate systems shall be provided to maintain control over the exposure of the occupational workers due to airborne radioactive material and to keep the quantity and concentration of the particulate and gaseous radioactive effluent discharge within the limits prescribed by AERB for members of the public.

2.2.4 Use of Models

In order to demonstrate compliance with the dose limits stipulated by AERB tp the members of the public, assessment has to be made with the use of environmental, metabolic and dosimetric models. [Annexure-I describes the methodology for arriving at the effective dose to a member of the public from gaseous radioactive effluent releases].

2.2.5 Counter-measures during Accidents

An evaluation of the potential exposure is needed to assess the requirements of any counter measures to be adopted for reducing the actual exposures under accident conditions. The techniques dealing with potential exposures include safety analyses to identify possible causes of accident and methods available to reduce their likelihood and severity.

There should be plans for dealing with accidents, should they occur. For accident conditions, the prescribed dose limits for normal operations are not applicable and are replaced by specially developed acceptable limits based on concentrations of radioactivity in different environmental matrices and specified levels of external exposure. Reference comparison with these specially developed acceptable limits gives guidance for initiating action or counter measures.[Ref. AERB safety Guide No.SG/HS-1 " Intervention Levels and Derived Intervention Levels for the Off-Site Radiation Emergency" (1992)]

2.3 Design Criteria for Ventilation Systems of Radioactive Areas

Ventilation system is generally provided to ensure personal comfort and proper functioning of plant structures, systems and components by providing air flows at appropriate temparature and humidity conditions. Design criteria for ventilation systems from radiological considerations are as described below :

- Primary Objective : The primary objective of the air cleaning and ventilation system is to control the buildup of airborne radioactive contamination in the working areas in order to keep the radiation exposure and the intake of radionuclides as low as reasonably achievable for occupational workers and to ensure that prescribed limits are not exceeded.

- Access Requirement: The system should be adequately designed to reduce concentration of radioactive substances to levels compatible with the access requirement of an area. [Ref. AERB/SG/D-21- "Containment System" and AERB/SG/D-12 on "Radiological Protection"]

- Airflow Pattern: As far as possible it is to be ensured that air flow takes place from low active areas to high active areas. This would help achieve the ultimate objective of preventing unacceptable dispersion of airborne radioactivity within the plant.

- Negative Pressure: It is required to maintain reactor containment building under negative pressure to ensure that there is no uncontrolled discharge of contaminated air.

- Control on Releases: The system should serve to keep the release of airborne radioactive substances to environment within prescribed limits during normal operation and acceptable levels under accident conditions.

- Fire Protection: The system should incorporate features (i.e. fire dampers) to minimise spread of fire. [Ref. AERB/SG/D-4 “ Fire Protection “]

- Qualification: The components of the system should be qualified to withstand and function as per design intents under environmental and seismic levels that can be expected to occur at the site.[Ref. AERB/SG/D-1- “ Safety Classification and Seismic Categorisation of Structures Systems and Components” and AERB/SG/D-3- “Environmental and Missile Effects”]

- Reliability: The air-cleaning system should be sufficiently reliable and so designed that the required retention factors for radioactive materials are achieved under the normal and as also accidental conditions. Systems important to safety are to be so designed such that their performance can be periodically tested and evaluated during normal operation of the plant. [AERB/SG/D-2 –‘‘Single Failure Criteria”]

3.0 SOURCES OF AIRBORNE RADIOACTIVE MATERIALS

3.1 General

In this section, airborne contaminants in the form of radioactive gases, vapours and aerosols released into the containment atmosphere during normal operation and accident conditions are discussed.

3.2 Normal Conditions

Airborne contaminants in the form of radioactive gases, vapours and aerosols may get released to the environment atmosphere through escapes from primary coolant and moderator system in the form of chronic leakages. It is recognised that it is difficult to quantify the source term for designing air cleaning system.

Some of the gaseous and volatile contaminants are tritium, fission products-noble gases, elemental iodine, or organic iodides and submicro aerosols comprising of a variety of fission products (e.g. Sr-90, Cs-137, I-131) and activated corrosion products(e.g. Co-60, Fe-55)

Tritium contributes significantly to airborne radioactivity during normal operation. Iodine as well as fission product noble gases may be present in negligible amounts due to timely detection and removal of failed fuel. Tritium, a beta emitter is mainly produced in the primary coolant and moderator by neutron capture reaction with deuterium. It appears in air as HTO/ DTO alongwith heavy water which escapes from valve glands, pump seals and non-welded joints as chronic leakages.

Fission products originate in the fuel and find their way into the primary coolant through failed fuel and to the containment atmosphere through leakages.

Ar-41, which is an activation product of Ar-40 present in air used for calandria vault cooling and in annulus gas monitoring system is not expected to be present consequent to the adoption of a water filled design for calandria vault and CO_2 filled annuli between pressure tube and calandria tube.

3.3 Accident Conditions

Plant safety analysis for postulated PIE's (as defined in AERB Safety Guide SG/D5) should be carried out and the features which have the potential of significant release of radioactivity to the environment should be examined in detail. The sources and magnitude of radioactivity under various accident conditions are determined in this process.

The main source of radioactivity under accident conditions is radioactive fission products against which precautionary protection design measures should be taken. These are released from various systems in which they are normally contained or from fuel bundles as a consequence of a LOCA/ fuel handling accident.

The highest number of cladding failures for the limiting design basis accidents should

be assessed. [AERB/SG/D-12 “Radiological Protection”]. The fraction of each of the fission products released from failed fuel should be calculated. The subsequent release of fission products from the coolant to the containment and the behaviour within the containment (e.g. removal by various transport mechanisms such as plate out, vapour suppression pool water trapping and by containment atmosphere clean-up system) should be evaluated.

As an alternative to a detailed analysis of limiting DBA and in the absence of quantification of fission products release, it is an acceptable practice to specify the fraction of core inventory of fission products that may reach the containment atmosphere consequent to the accident. This fraction is specified differently for different categories of chemical elements but will be usually independent of the design measures against such type of accidents. Thus this fraction is set as an assumed upper limit irrespective of the performance of the emergency core cooling system providing necessary conservatism.

The physical and chemical forms of different fission products released are of interest for the design of ventilation related engineered safety features (i.e. primary containment clean-up system primary containment controlled discharge system, secondary containment clean-up system etc.) and are listed in Table-1.

TABLE-3.3
Physical and Chemical Forms of Different Fission Products Release from Reactor Core during and after Accident Conditions

Sr. No.	**Nuclide**	**Chemical Form**	**Physical Form**
1.	Noble Gases	Kr and Xe	Gases
2.	Cesium	CsI, CsOH, Cs_2O, CsO	Vapour & Aerosols
3.	Iodine	CsI, I, I_2, HI, HIO	Vapour & Aerosols
4.	Tellurium	Te_2, Te, TeO, TeO_2, Te_2O_2, H_2Te	Aerosols
5.	Tritium	HTO, HT, T_2	Vapour/ Gas
6 7.	Strontium Others	Sr-90, Yt-90 In multitude of chemical species	Aerosols Aerosols

4. CONTROL OF AIRBORNE RADIOACTIVE MATERIALS

4.1 General

In this chapter, the significance of ventilation system during normal operation and accident conditions is described.

4.2 Normal Conditions

4.2.1 Use of Particulate Filters

By means of confinement features such as barriers and appropriate pressure differentials, the spread of contamination can be limited. Effective segregation of heavy water and light water areas is an important factor that contributes towards minimising spread of radioactivity. Release to environment is controlled by provision of exhaust air cleaning system consisting of coarse Pre filters and high efficiency particulate (HEPA) filters.

In designing a ventilation system for control of airborne activity, the following should be taken into account :

- thermal and mechanical mixing mechanism in any large volumes
- the limited effectiveness of dilution in reducing airborne contamination.
- the need to exhaust air from potentially contaminated areas
- the difficulty in detecting leakage from non welded joints and the resulting possibility of chronic leakages

For the type of contamination present in containment atmosphere, the basic air cleaning processes such as high efficiency particulate filtration of aerosols absorption of iodine (e.g. by use of charcoal filter, Silver–zeolite etc.) and dehumidification / drying for tritium are recommended to be used.

4.2.2 Flow Pattern

In general an once through ventilation system of pull through type is provided. Interzonal air-flows are to be directed from regions of lower airborne contamination to regions of higher contamination. In some areas, e.g. those housing high enthalpy heavy water system it may be desirable to have closed loop recirculation system including filters, dryers, and coolers as appropriate, with only a small controlled flow discharge. Sizing of flow capacity of those systems shall give due consideration to the source term, the airborne radioactivity levels to be maintained in the controlled areas and also to the releases to environment. In order to limit airborne tritium activity level in these areas, escape of heavy water needs to be minimised and dryers should be adequately sized.

4.2.3 Spread of Contamination

A separate ventilation system shall be provided for primary and secondary containment. A slightly negative pressure should be maintained in the containment and other areas wherein airborne radioactive material may be present to prevent ground level releases and to permit monitoring discharge of contaminated air. One of the ways of achieving this is by the provision of exhaust fans with fresh air supply achieved by induction effect. Care should be taken in design to reduce the probability of spread of contamination in case of power failure.

4.2.4 Local Exhaust

Provision of local exhaust / portable ventilation system may be made in areas where airborne contamination may arise during maintenance. Provision shall be made for sufficient space and appropriate electric points for operating these systems. Such systems should discharge into contaminated ventilation exhaust or if equipped with filters, it may be acceptable to discharge the exhaust into the atmosphere of the building directly.

4.2.5 Use of Filters in Air Inlet

To reduce carry-over of dust from the external environment into the nuclear power plant and a possible increase in contamination transport, filters should be used in air inlet ducts.

4.3 Accident Conditions

Safety analysis of present design of PHWR has indicated that because of large inventory as heat sink in these reactors, accidents Beyond Design Basis Accidents (BDBA) are not considered. Though the probability of occurrence of a DBA in a nuclear power plant is very low, the mitigating features should be designed to take care of a DBA and the main focus should be on severe design basis accident.

4.3.1 Engineered Safety Features (ESF)

In order to reduce uncontrolled release to the environment and to mitigate the radiological consequences under design basis accidents, ESF i.e. vapour suppression pool, containment coolers and air cleaning systems are provided.

[Ref. AERB/SG/D-1 “Safety and Seismic Classifications of System, Structures and Components”
AERB/SG/D-2 “ Single Failure Criteria”
AERB/SG/D-3 “ Environmental and Missile Effects”]

5. PROTECTION OF PLANT PERSONNEL, PUBLIC AND ENVIRONMENT

5.1 General

It is essential to protect plant personnel, public and environment from the hazards involved in the operation of the NPP both during the operational phase and under accident conditions. The design of airborne radioactivity control mechanisms shall be so engineered to meet the objectives of reducing the consequences during normal operation and accident conditions.(See Annexure-III for details)

5.2 Protection under Normal Conditions

During operational states the total quantity of radioactivity released to the environment through air route should be reduced to prescribed levels. This should be achieved by the ventilation control measures that are to be implemented to protect plant personnel and public from avoidable radioactivity exposure, such as, appropriate air flow direction, provision of filtered exhaust ventilation system connected to stack, dryers for areas with potential D_2O escape and charcoal filters in the exhaust from areas like fuel transfer room with potential for iodine activity escape.

5.3 Protection Under Accident Conditions

5.3.1 Design of the System and its Assessment

The design of airborne radioactivity control equipment should have the objective of reducing the consequences of accident conditions. The design should be assessed by means of safety analysis. In cases where calculated values exceed the acceptable levels, for accidents additional protection features are to be incorporated in the design to meet those levels.

5.3.2 Design Measures

Design measures that are required to be taken to achieve reduction in radiation exposure due to air-borne radioactivity releases include :

- Increasing the leak tightness of the system components and containment
- Provision of clean up system to reduce the airborne radioactivity present in the containment
- Filtering the controlled discharge air from containment in order to reduce the release of airborne radioactivity
- Providing means for minimising compressed air ingress into the reactor

building so that the requirement for purging the containment is minimal

- Delayed operation of controlled discharge system to allow natural flow of

  radionuclides within the building as well as their removal by primary containment clean up system(Primary containment filtration and pump back system) where provided.

- fast acting containment isolation valves for early isolation of containment,

- Providing means for reducing explosion hazard potential of H_2 that could be released in an accident

- maintaining negative pressure in secondary containment to avoid ground level release of radioactivity.

ANNEXURE-I

MATHEMATICAL MODEL FOR ASSESSMENT OF DOSE

A.1 Computation for normal/routine releases

A.1.1 Concentration of radionuclides in atmosphere:

The Gaussian plume model with sector averaging can be used with observed meteorological data to compute the concentration of radionuclides in atmosphere due to normal releases from the Nuclear Power Plants (NPPs). The annual average concentration of a radionuclide at a distance due to contributions in the wind direction and adjacent sector directions of continuous and constant release rate (Bq/s) can be calculated from the following equation:

$$\chi = \sum_{A=S}^{F} \frac{Qf_{s1}W_{s1}\exp(-H^2/2\sigma^2_{zs})}{\pi\sigma_{ys}\sigma_{zs}U_{s1}} + \frac{Qf_{s2}W_{s2}\exp(-H^2/2\sigma^2_{zs})}{\pi\sigma_{ys}\sigma_{zs}U_{s1}} + \frac{Qf_{s3}W_{s3}\exp(-H^2/2\sigma^2_{zs})}{\pi\sigma_{ys}\sigma_{zs}U_{s3}} \qquad (1)$$

where,

χ = Annual average concentration of radionuclide at ground level, (Bq/m^3)

Q = Release rate of radionuclide, (Bq/s)

σ_{ys}, σ_{zs} = Horizontal and vertical dispersion coefficients at atmospheric stability,s,

U_{s1},U_{s2},U_{s3} = Annual average wind speed in the direction of concern and in adjacent sector at atmospheric stability, s

H = Effective discharge height, (m)

f_{s1}, f_{s2}, f_{s3} = Annual average frequency of wind blown towards direction of concern and adjacent sectors for atmospheric stability, s.

W_{s1}, W_{s2}, W_{s3} = Averaging coefficients in the direction of concern and adjacent sectors for atmospheric stability,s.

The averaging coefficients can be calculated using the following equations:

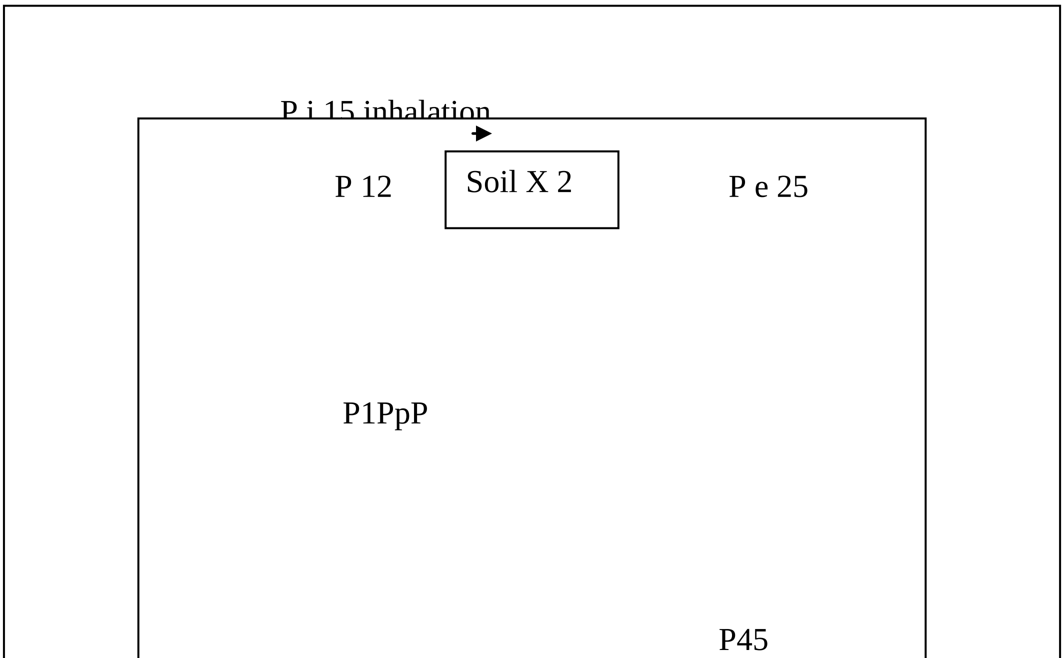
$$Ws_1 = \int_0^{\pi x/16} \frac{\exp(-y^2 / 2\sigma^2{}_y)dy}{\pi x / 16}$$

$$Ws_2 = Ws_3 = \frac{\left[\int_0^{3\pi/16} \exp(-y^2 / 2\sigma^2{}_y)dy - \int_0^{\pi x/16} \exp(-y^2 / 2\sigma^2{}_y)dy \right]}{2\pi x / 16}$$

A.1.2 Schematic Pathway for Terrestrial Environment

The simplified representation of the pathways to man from atmospheric releases is shown in figure below:

Fig.A.1.2



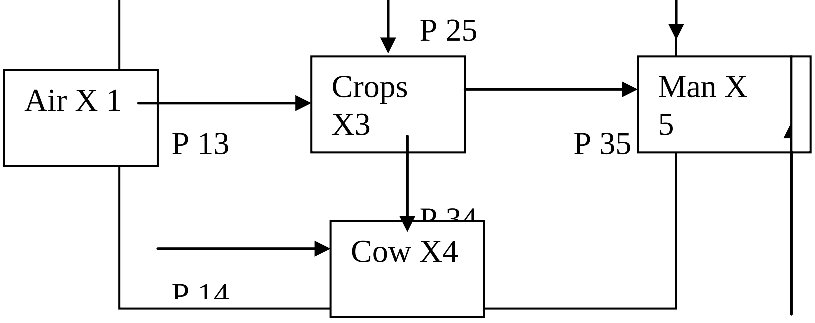


TABLE-A.1.2

TRANSFER FACTORS FOR ATMOSPHERIC RELEASES SHOWN IN FIG-1

TRANSFER FACTOR	**IDENTIFICATION**	**UNITS**
$P_{i,15}$	Dose factor for inhalation	Sv/Y Per Bq/m^3
P_{12}	Deposition on to soil	m^3/kg
$P_{e,25}$	External dose from soil	Sv/Y Per Bq/kg
P_{13}	Deposition on to Vegetation	m^3/kg
P_{23}	Bv in Vegetation	Kg(s)/kg(v)
P35	Dose from intake of Vegetation	Sv/Y per Bq/kg
X_1	Atmospheric Concentration	Bq/m^3
X_2	Soil concentration	Bq/kg
X_3	Vegetation Concentration	Bq/kg
X_4 X_5	Concentration of Animal Produce Dose to man	Bq/kg Sv/year
P_{14}	Inhalation of plume by Animals	m^3/kg
P_{45}	Dose from intake of milk or meat	Sv/Y Per Bq/kg

$P_{e,15}$	Dose factor for cloud immersion	Sv/Y Per Bq/m^3
P_{34}	Vegetation intake to Animal produce	Kg(s)/kg(m)

Parameters: s- soil; v-vegetation ; f- animal feed; m- meat or milk

A.1.3 Concentration of Radionuclides in Terrestrial Components

Concentration of Radionuclides in Vegetation

Cereals, leafy vegetables, milk and meat are the main agricultural products considered for the oral intake pathways. The concentration of radionuclides in vegetation due to

$$Cv = \chi V_g \left[\frac{R[1 - \exp(-\lambda_e t_e)]}{Y\lambda_e} + \frac{B_v[1 - \exp(\lambda_t t_b)]}{P\lambda_i} \right]$$

interception of atmospheric radionuclide flux and root uptake can be calculated from the following equation:
Where

C_V = concentration of radionuclide in vegetation (Bq/kg)

χ = mean annual concentration of radionuclide in air(Bq/m^3)

V_g = Deposition velocity (m/s)

R = Fraction of interception of the material on vegetation

λ_e = effective removal constant for reduction of activity from vegetation (d^{-1})= (λ+λw)
λw = rate constant for reduction of activity deposited on the surface of vegetation due to processes other than radioactivity decay(d^{-1})

t_e = time of exposure during growing season (d)

Y = agricultural density (kg/m^2)

B_v = transfer factor from soil to vegetation (kg soil/kg veg)

λ = radioactive decay constant (d^{-1})

λi = environmental dacay constant for removal of activity from soil (d-1) = ($\lambda + \lambda s$)

λs = rate constant for reduction in concentration of radionuclide in root Zone of soil other than radioactive decay λ

t_b = time of long term accumulation of radionuclide in soil (d)

P = effective soil density (kg/m^2)

For tritium concentration in vegetation, a simplified version of equation can be employed as given below:

$$C_T = \frac{\chi * 0.75}{AH} \longrightarrow (5)$$

where,

C_T = concentration of tritium in vegetation (Bq/kg)
AH = Absolute humidity (kg/m^3)
0.75 = fraction of water in the vegetation

Concentration of Radionuclide in Milk

The concentration of radionuclide in milk can be obtained in terms of pasture intake as :

$$C_m = C_v * Q_f * F_m * \exp.(-\lambda\ t_f) \quad \rightarrow \quad (6)$$

where,

C_m = concentration of radionuclide in milk (Bq/L)

C_v = concentration of radionuclide in feed (Bq/kg)

Q_f = feed intake rate (kg/d)

F_m = transfer factor for fraction of animal's daily intake of radionuclide that appears in each litre of milk(d/L) at equilibrium (ratio of concentration of a radionuclide in milk to the product of same radionuclide concentration in feed and daily intake of feed by the animal)

t_f = Average time of the transport of activity from the feed into the milk and to the receptor (assumed to be 4 days for fresh milk in the absence of site specific information) (d)

Concentration of Radionuclide in Meat

The concentration of radionuclide in meat due to intake of animal feed can be calculated from the relation:

$$C_f = C_v * Q_f * F_f * \exp(- \lambda t_s) \qquad \rightarrow \qquad (7)$$

where

C_f = concentration of radionuclide in meat (Bq/kg)

F_f = Transfer factor for fraction of animal's daily intake of radionuclide that appears in each kg. of meat (d/kg) at equilibrium (ratio of the concentration of a radionuclide in meat to the product of same radionuclide concentration in the feed and daily intake of feed)

t_s = Average time of the transport of activity from the feed to slaughter to consumption, assumed to be 20 days in the absence of site specific information.

A.1.4 Dose to Man

Applying concentration factor model, the dose to man, X_5 (compartment model 5) through various atmospheric pathways can be represented as :

$$X_5 = \{(Pi_{15} + Pe_{15}) + (P_{12}\, Pe_{25}) + (P_{13} + P_{12}\, P_{23})\, P_{35} + (P_{14} + P_{13}\, P_{34} + P_{12}\, P_{23}\, P_{34})\, P_{45}\} X_1 \qquad \rightarrow (8)$$

The identification of transfer parameter is shown in Figure. The transfer parameter, $P_{i,j}$ is defined as the ratio at steady state of the radionuclide concentration Xj and X_i in the compartments of j and i respectively. The identification of the compartment and transfer parameters are depicted in Table A.1.2 along with their units.

A.2 Computation for Release under Accident Conditions

A.2.1 Concentration of Radionuclides in the Atmosphere

The computation of radiological impact due to release of effluents to atmosphere under worst meteorological condition is important when releases are considered during an accident scenario. The approach can be broadly considered in two steps:

- evaluate ground level concentration of effluent in air and concurrent deposition flux (when applicable) to the ground under the postulated worst meteorological condition to obtain conservative estimates of the impact.

- use appropriate transfer factors to compute external and internal (by inhalation and ingestion routes) doses.

Computation of Ground Level Concentration (GLC) and Deposition Flux:

In computing the GLC the worst meteorological condition is considered based on atmospheric stability parameters. In terms of Pasquill Stability classification scheme this is usually taken as Pasquill-F weather category for ground level effluent releases. For elevated (stack or buoyant plume) releases, the worst weather category to be considered will depend on downwind distance and hence to be selected based on factors like effective release height, exclusion boundary etc. The approach in such cases would be to obtain concentration as a function of downwind distance for each of the stability class. An envelope curve joining the maximum value for each of the curves can be used to obtain the maximum concentration at any downwind distance.

The basic model used in estimates of atmospheric dispersion is the double Gaussian straightline plume model. For dose estimates, it is desirable to use the Time Integrated Concentration (TIC) since the dose is proportional to it. In most of the cases of accident scenarios, meteorological conditions can be assumed to remain effectively unchanged over the duration of release and ground level TIC (Bq.s/m3) can be obtained for elevated releases from the equation [1]:

$$\chi = \frac{Q}{\pi u \sigma_{ys} \sigma_{zs}} \exp(-H^2 / 2\sigma^2_{zs}) \longrightarrow \qquad (9)$$

where

Q = source strength, total quantity released (Bq)

u = mean wind speed (m/s) at height of release

In further discussions, the dependence of σy, σz on stability class s will not be explicitly stated.

For ground level release the equation reduces to

$$X = \frac{Q}{\pi u \sigma_y \sigma_z} \quad \rightarrow \qquad (10)$$

However, generally ground level releases cannot be considered as point source releases and the effect of finite volume of the building from which the release takes place needs to be taken into account. This is accounted for by modifying equation (10) as:

$$X = \frac{Q}{(\pi\sigma_y\sigma_z + CA)u} \qquad \rightarrow \qquad (11)$$

where,

A = area of cross section (m^2) of building normal to wind flow
C = building shape factor taken to be 0.5

Selection of parameters for the evaluation of stability class of the atmosphere in a particular situation is important. Pasquill defined wind speed and insolation and/or cloud cover as stability parameters which are widely used. Table 1 gives conservative set of wind speed for use in equations (9-11) corresponding to each stability class. For effluents with significant deposition velocity (eg.iodines) , the deposition flux and depletion of plume due to deposition is to be considered when arriving at GLC. This is done using the standard source deposition model given by the equation

$$D = V_g \frac{Q_d}{\pi\sigma_y\sigma_z u}\exp\left(\frac{-H^2}{2\sigma_z^2}\right) \qquad \rightarrow \qquad (12)$$

$$\chi_d = \frac{Q_d}{\Pi\sigma_y\sigma_z u} \exp\left(-H^2/2\sigma_z^2\right) \qquad (13)$$

where,

D = deposition flux to ground at plume centreline (Bq/m^2)
V_g = deposition velocity (m/ s)
Q_d = deposition corrected source strength (Bq)
χ_d = GLC after plume depletion ($Bq.s.m^{-3}$)

Qd = QF_d and

The deposition corrected GLC for such effluents is given by

$$F_d = \left\{\exp \int_0^z \frac{dx}{\sigma_z} \exp\left(\frac{-h^2}{2\sigma_z^2}\right)\right\}^{-\sqrt{\frac{2}{\pi}}\frac{Vg}{u}} \qquad \rightarrow \qquad (14)$$

Equations (13) and (12) should be used for evaluating ground level concentration of a plume which undergoes depletion by dry deposition and consequent dry deposition flux. Plume depletion by dry deposition is important for effluents with significant deposition velocity (>0.1 cm/s). Radioactive decay of the effluent during its travel from source to receptor should be taken into account in GLC computation. In addition plume depletion could occur due to washout (when effluent species are dissolved in falling rain drops) or rainout (when the effluent is scavenged within a precipitating cloud and precipitated) during its dispersion in the atmosphere. The method of computing the plume depletion factor and the resultant ground deposition by such wet deposition processes is outlined in IAEA/SGS3 [1] and can be used in GLC computations.

A.2.2. Concentration of Radionuclides in Terrestrial Components

The methodology outlined in section A.1.3 can be used to compute the concentration of radionuclides in various terrestrial components during accident scenario. The appropriate transfer factor values for the accident situation should be used where applicable [2].

A.2.3.Computation of Doses

Doses due to effluent release during accident situation can be from

- external dose from plume and deposited activity
- inhalation dose from activity inhaled during passage of plume
- Dose due to ingestion of deposited activity

In the following, the methodology of obtaining the doses or required dose factors are outlined.

External Dose

External gamma dose can be obtained from dose conversion factors [4] for routine releases which essentially give the submersion doses. However, under accident conditions this may not be appropriate especially for elevated releases and short down wind distances from the source for which plume doses may be important. These can be obtained from Tables given in the manual of dose computation from atmospheric releases [3]. These tables give single plume and sector averaged plume doses at various distances by computing gamma radiation fluxes at the receptor location from elemental volumes in which the plume is divided. Tables for computing submersion doses are also given in the manual. The doses are given for different stability classes release heights and gamma energies. External dose due to ground deposited activity is important for long lived isotopes with significant deposition velocity. The methodology outlined in Section A.1.3 (the route P_{12} to Pe_{25}) can be used for this purpose

Inhalation Dose

The inhalation dose is given by

$$D_{inh} = \chi \, Q B \, \varepsilon_{inh} \quad \rightarrow \qquad (15)$$

where

B = breathing rate (m^3/s)
χ = GLC (exposure) (Bq.sec/m^3 per Bq release)
Q = Source strength (Bq)

ε_{inh} = Effective dose coefficient per unit intake (Sv/Bq) for whole body for critical organ considered

The dose conversion factors for various critical organs corresponding to different isotopes are given in (4). For iodines detailed analysis of metabolic data applicable to Indian conditions have been made and the dose conversion parameters for inhalation dose are shown in Table 2. It is recommended that where available metabolic parameters pertinent to Indian conditions be employed and in their absence default values [2,4] may be used.

Ingestion Dose

Stability Class	A	B	C	D	E	F
Wind Speed	1	1	2	3	2	2

ngestion dose
s for Indian
Iodines from

TABLE-A.2.3.(a)
Wind speed (m/s) for Different Stability Categories

TABLE-A.-2.3.(b)
PARAMETER VALUES USED IN COMPUTATION OF INGESTION AND INHALATION DOSES FOR THE CHILD*

PARAMETER	**VALUES USED IN PRESENT COMPUTATION**
Dosimetric parameters	
Thyroid mass(g)	1.8

Inhalation uptake Fraction fa	0.2
Ingestion uptake Fraction, fb	0.3
Effective elimination Constant(1/d)	0.115
Effective energy MeV/dis	0.2
Committed dose per unit Intake by ingestion (Sv/Bq)	3.99E-6 (for child thyroid)
Committed dose per unit Intake of inhalation (Sv/Bq)	2.66E-6
Transfer parameters	
Forage intake (kg/d)	8.75
Density of forage (kg/m^2)	2.0
Effective decay constant (1/d)	0.1546
Transfer factor in milk Fm (d/L)	6.7E-3
Milk intake, (L/d)	0.2
Breathing rate (m^3/d)	3.8
Ding Sv/Bq/m^2	6.64E-7
Dinh Sv/Bq.s/m^3	2.34E-10

- *BARC /1996/K004 – Stack height for Fuel Reprocessing Plant at Kalpakkam V.Sitaraman et al, 1996

ANNEXURE-II

REACTOR BUILDING VENTILATION SYSTEM AND ENGINEERED SAFETY FEATURES IN KAIGA-1,2 & RAPP-3,4.

Introduction :

Reactor building (RB) in all Indian PHWR from Narora Atomic Power Station (NAPS) onwards is a full double containment design with a Primary Containment (PC) and a Secondary Containment(SC). P C is again divided into two volume V1 and V2. The volume V1 houses all the high enthalpy D2O areas like the pump-room and F/M vaults. The remaining areas of PC constitute volume V2. V1 and V2 are separated by a suppression pool.

Ventilation Philosophy for Buildings Having Potential of Release of Radioactivity

The ventilation philosophy of buildings other than R.B. is broadly outlined below :-

1. Identification of contaminated/active zones, maintain inter zone flow from areas of lower contamination to areas of higher contamination.

2. Maintain negative pressure inside the building to prevent out leakages.

3. Maintain airborne contamination/activity level below the stipulated limit.

4. Use of pre and HEPA filters in the exhaust of the building before releasing through stack.

5. Use of charcoal filter locally wherever there is possibility of release of iodine activity.

6. Other requirements are similar to that of conventional ventilation. However, a higher degree of quality in manufacture and reliability of operation is ensured.

Reactor Building Ventilation System

In normal and shut down condition of the reactor two separate once through ventilation systems for volume V2 of the P C and S.C. are operational. Volume V1 is provided with a closed loop vapour recovery system with a small purge through dryer to maintain a negative pressure in V1 volume with respect to V2. (R.B. Volume V1 dryer system schematic is shown in sketch--2). Also these ventilation systems establish a negative pressure gradient between the outside atmosphere and P.C. through S.C. and eliminate any possibility of outward ground level leakage of activities during normal reactor operation. P.C. Ventilation System exhausts from volume V2, 8500 SM^3/hr of air during normal reactor operation and 10500 SM^3/hr of air during purge condition when occupancy is higher than normal in the RB as during shut down condition of the reactor. Based on RAPS, MAPS, NAPS and KAPS experience, the above flow rates have been considered adequate from the point of view air borne activity level, minimising heavy water loss, fresh air needs of the operational personnel and system economy. The heat load is taken care of by chilled water cooled air handling units located in various areas of volume V2. Charcoal filter has been provided in the Fuel Transfer room exhaust duct to adsorb iodine activity in case of any iodine activity release during handling of damaged fuel bundles. The R.B. exhaust air is filtered for particulate activity by the Pre and HEPA filters before releasing through stack. (R.B. primary Containment Ventilation System schematic is shown in sketch-1).

Engineered Safety Features

Suppression Pool and Building Coolers

In case of a LOCA almost all the energy released is absorbed by suppression pool and the building coolers provided in volume V1 (i.e. F/M vault and pump room coolers). The major communication under this condition between volumes V1 and V2 is through suppression pool. The primary containment pressure comes down to about 0.05 kg/cm^2g in about 3 hours with the help of building coolers. Cooling capacity under normal and accident conditions is 1.812×10^6 Kcal/hr & 6×10^6 Kcal/hr respectively.

Air Clean-up Systems

Engineered Safety Features like S.C. filtration, recirculation and purge system (S.C.clean up and purge system in 500 MWe reactor), P.C. Filtration and Pump Back System (PCFPB) and Primary Containment Controlled Discharge System (PCCD) have been provided to mitigate the consequences of postulated accidents and limit the radioactivity release to the environment.

Primary Containment Controlled Discharge (PCCD)

The suppression pool and building coolers reduce the P.C. pressure to about 0.05 g/cm^2g in about 3 hours. Primary Containment Control discharge at the rate of 1700 m^3/hr is put into operation at the discretion of the station authority preferably after 48 hours (24 hours in case of NAPS & KAPS) from the occurrence of accident to bring down the P.C.pressure to atmospheric level. This delay of 48 hours enables the PCFPB system to clean the containment atmosphere to a large extent and allows sufficient time for short lived activity to die down considerably and thus reduces the stack release during operation of PCCD. Provision has been made to operate PCCD even at P.C. pressure of 0.4 kg/cm^2g. This provision helps in delaying PCCD operation even beyond 48 hours to enable the station authority to look for a favourable weather condition. (Primary Containment Control Discharge System schematic is shown in sketch-3).

S.C. Recirculation & Purge System (S.C.R. & P.)

Subsequent to a postulated accident R.B. gets boxed up by automatic closure of all building isolation dampers and immediately the secondary containment recirculation and purge system comes into operation. This system continuously re-circulates S.C. Containment air through suitable HEPA and charcoal filters, purges part of the air through the stack. The amount of purge is decided based on the building leak rate and instrument air ingress into the secondary containment. This purge maintains a negative pressure (-12 to -24 mm WG) in the S.C. with respect to atmosphere and any leakage of activity from P.C. to S.C. is intercepted by the latter and ground level leakage is reduced to a negligible level. This system enables to plan and delay the operation of PCCD based on meteorological conditions for a considerable length of time. (S.C. Recirculation & Purge System schematic is shown in sketch-4).

P.C. Filteration & Pump Back System (PCFPB)

After about 4 hours from the occurrence of DBA, the primary containment filteration and pump back system is put into operation. This delay helps to decay the short lived iodine isotopes considerably, allows aerosols to settle down, and reduce the load on charcoal filters. This system sucks air from V1 volume, passes through combined

HEPA-charcoal filters to clean the containment atmosphere by adsorbing radioactive iodine and recirculates the air to V2. The system is designed to handle the entire core iodine inventory with a removal half life of 2.7 hrs (3.5 hrs in case of NAPS & KAPS). (P.C. Filtration & Pump Back System schematic is shown in sketch-5).

ANNEXURE-III

This describes the various aspects of gaseous waste treatment systems.

1. Pressure relief device discharge

Discharge from pressure relief devices provided on tanks shall be piped into a building exhaust ventilation system or to another system capable of containing the discharge and with provision for monitoring of releases.

1.1 Purge system

Tank design shall provide for purging or inserting of the tank gas space with nitrogen to remove residual gaseous activity or potentially combustible gases. Purge and vent connections should be piped into a closed building exhaust ventilation system or to another vent with provisions for monitoring releases, with consideration given to the properties of the resultant gas mixture, e.g. combustible gas with oxygen or air.

1.2 Drainage

The drain shall have provisions to verify that the system is drained without permitting escape of contained gases unless the drain line is designed to accept the gases.

2. Charcoal absorption tanks

Tanks filled with charcoal can provide more delay of gaseous wastes than pressurised tanks, due to absorption on the charcoal.

2.1 Absorbent bed gas velocity

The charcoal absorbent tanks shall be designed so that if gas flows vertically upward the bed is not fluidised when minimum flow conditions occur. The vessel should be sized such that the superficial velocity through the charcoal for the design flow rate and for the absorbent volume will prevent loss of efficiency due to axial diffusion.

2.2 Process flow bypass

Bypass of charcoal absorbent tanks may have to be provided for the event of humidity in off-gas systems.

3. Recombiners

Catalytic recombiners should be provided as a means to reduce a combustible gaseous mixture by recombining the hydrogen and oxygen available.

3.1 Catalyst replacement

The recombiner shall be designed to allow removal and replacement of the catalyst.

3.1.1 Catalyst supports

Catalyst support elements design shall take into consideration the additional weight of wet catalyst and the pressure drop and the temperature associated with high startup flow transients and the prevention of catalyst loss from the vessel.

3.1.2 Moisture removal

The recombiner sub system shall have a preheated with the following features:

1) Adequately sized heated inlet drain

2) A sump or similar provisions to remove all water from the process flow during and transients as well as during normal system operation

3) adequately sized preheater steam side drain.

3.2 Compressors

Compressors shall be equipped with reliable, high quality seals to minimise leakage and to process gas without contamination by oil. Diaphragm type compressors, if used, should be provided with leak detection capability.

3.2.1 Gas filters

High efficiency particulate air (HEPA) filters meeting regulatory requirements shall be used for gaseous waste filtration. Prefilters and charcoal filters may also be used before filters.

3.2.1.1 Filter element removal (in radiation area)

The filter vessel should be furnished with guides to locate the the filter element in the vessel.

3.2.1.2 Filter testing

The filter system should be designed with injection and sampling points to allow for periodic testing to verify filter element and seal integrity as well as unit efficiency.

3.2.1.3 Vessel piping

The filter vessel piping shall be arranged for easy disassembly and filter removal.

3.2.2 Gas demisters and dryers

Various types of dryers and a demister may be used to remove moisture from the gas stream.

3.2.3 Special requirements

3.2.3.1 Design for explosion conditions

The gaseous waste handling and treatment system shall be deigned to withstand and minimise effects of a hydrogen explosion . Appropriate controls and detection devices shall be incorporated .

3.2.3.2 Leakage prevention

Joints should be seal welded to avoid leakage of radioactive material into environment. Valves with bellows sealed stems, diaphragm valves or valves with similar leaktightness characteristics should be used to minimise leakage.

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WORKING GROUP-14 (RECONSTITUTED in 1995)

Dates of meetings:
May 17/ 1996, June 6/ 1996, Nov 14/1996, Dec17/1996, May 5/1997
June 28/1999, July 5/1999

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Dates of Review : Sept. 7, 1995
Oct. 12 & 13, 1995
Sept 14, 1998
July 15,1999

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