GUIDE NO. AERB/NPP-PHWR/SG/D-21

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GOVERNMENT OF INDIA

AERB SAFETY GUIDE

CONTAINMENT SYSTEM DESIGN FOR PRESSURISED HEAVY WATER REACTORS

ATOMIC ENERGY REGULATORY BOARD

AERB SAFETY GUIDE NO. AERB/NPP-PHWR/SG/D-21

CONTAINMENT SYSTEM DESIGN
FOR
PRESSURISED HEAVY WATER REACTORS

Atomic Energy Regulatory Board
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FOREWORD

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

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

The code of practice on ‘Design for Safety in Pressurised Heavy Water Based Nuclear Power Plants’ (AERB/SC/D) states the requirements to be met in the design of containment system. This guide is based on the current design of 220 MWe and 540 MWe and other new generation pressurised heavy water reactors (PHWRs). It specifically provides guidance on all aspects of safety in design of containment systems of PHWRs. In drafting this guide the relevant documents developed by International Atomic Energy Agency (IAEA) under the Nuclear Safety Standards (NUSS) programme, especially the safety guide on ‘Design of Reactor Containment Systems for Nuclear Power Plants’ (No. NS-G-1.10, 2004) and other international documents have been used extensively.

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

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

Specialists in the field drawn from the Atomic Energy Regulatory Board, the Bhabha Atomic Research Centre and the Nuclear Power Corporation of India Limited and other consultants have prepared this guide. It has been reviewed by experts and relevant AERB Advisory Committee on Codes and Guides and the Advisory Committee on Nuclear Safety.

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

(S.K. Sharma)
Chairman, AERB

DEFINITIONS

Acceptable Limits

Limits acceptable to the regulatory body for accident condition or potential expousre.

Accident

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

Accident Conditions

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

Anticipated Operational Occurrences

An operational process deviating from normal operation, which is expected to occur during the operating lifetime of a facility but which, in view of appropriate design provisions, does not cause any significant damage to items important to safety, nor lead to accident conditions.

Beyond Design Basis Accidents (BDBA)

Accidents of very low probability of occurrence, more severe than the design basis accidents, those may cause unacceptable radiological consequences; they include severe accidents also.

Confinement

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

Containment

(See ‘Primary Containment’/’Secondary Containment’/’Confinement’)

Containment Boundary

The outler limits of the containment system.

Containment Envelope

Structures and penetrations, which provide pressure retaining barrier to prevent or limit the escape of any radioactive material that could be released from the fuel during accident conditions.

Containment Isolation

The process of isolating or boxing up the containment so that there is no direct path from the system available for the radioactivity to reach the environment.

Containment Penetrations

Openings in the containment envelope for passage of personnel, materials, process piping and cables.

Containment Structure

The concrete portion and embedded parts of the primary and secondary containment systems.

Design Basis Accidents (DBAs)

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

Discharge Limits

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

Embedded Parts (EPs)

Any structural member, plate, angle, channel, pipe sleeve or other section anchored to a concrete structure through a direct bond or other anchors.

Emergency

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

Engineered Safety Features (ESFs)

The system or features specifically engineered, installed and commissioned in a nuclear power plant to mitigate the consquences of accident condition and help to restore normalcy, e.g. containment atmosphere clean-up system, containment depressurisation system, etc.

Environment

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

Explosion

An abrupt oxidation or decomposition reaction producing an increase in temperature, or in pressure, or in both simultaneously.

Fail Safe Design

A concept in which, if a system or a component fails, then the plant/component/system will pass into safe state without the requirement to initiate any operator action.

Flammable

Any medium which is capable of undergoing combustion in the gaseous phase, with emission of light during or after the application of igniting source.

Items Important to Safety (IIS)

The items which comprise:

- those structures, systems, equipment and components whose malfunction or failure could lead to undue radiological consequences at plant site or off-site;
- those structures, systems, equipment and components which prevent anticipated operational occurrences from leading to accident conditions;
- those features which are provided to mitigate the consequences of malfunction or failure of structures, systems, equipment or components.

Liner

Any metallic or non-metallic material applied to the surface of a base material for the purpose of protection against corrosion, abrasion or for leak tightness for the intended service conditions.

Loss of Coolant Accident (LOCA)

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

Main Steam Line Break (MSLB)

A break in steam pipeline which leads to discharge of high enthalpy steam.

Metal-Water Reaction

Reaction of water/steam with fuel cladding as a function of time and temperature during accident conditions.

Normal Operation

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

Nuclear Safety

The achievement of proper operating conditons, prevention of accidents or mitigation of accident consequences, resulting in protection of site personnel, the public and the environment from undue radiation hazards.

Operating Basis Earthquake (OBE)

An earthquake which, considering the regional and local geology and seismology and specfic characteristics of local sub-surface material, could reasonably be expected to affect the plant site during the operating life of the plant. The features of a nuclear power plant necessary for continued safe operation are designed to remain functional, during and after the virbatory ground motion caused by the earthquake.

Passive Component

A component which has no moving part and only experiences a change in process parameters such as, pressure, temperature, or fluid flow in performing its functions. In addition, certain components, which function with very high reliability, based on irreversible action or change, may be assigned to this category (examples of passive components are heat exchangers, pipes, vessels, electrical cables and structures. Certain components such as rupture discs, check valves, injectors and solid-state electronic devices have characteristics, whcih require special considerations before desingation as an active or passive component).

Postulated Initiating Events (PIEs)

Identified events during design that lead to anticipated operational occurrences or accident conditions, and their consequntial failure effects.

Prescribed Limits

Limits established or accepted by the regulatory body.

Primary Containment

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

Quality Assurance (QA)

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

Reactor Building

The concrete containment structure that contains and supports the reactor and other related systems such as the heat transport system, the moderator system, etc.

Redundancy

Provision of alternative structures, systems, components of identical attributes, so that any one can perform the required function, regardless of the state of operation or failure of the other.

Safe Shutdown Earthquake (SSE)

The earthquake which is based upon an evaluation of the maximum earthquake potential considering the regional and local geology, seismology and specific characteristics of the local sub-surface material. It is that earthquake which produces the maximum vibratory ground motion for which certain structures, systems and components are designed to remain functional. These structures, systems and components are those which are necessary to assure:

- the integrity of the reactor coolant pressure boundary; or
- the capability to shutdown the reactor and maintain it in a safe shutdown condition; or
- the capability to prevent the accident or to mitigate the consequences of accidents which could result in potential off-site exposures higher than the limits specified by the regulatory body; or
- the capacity to remove residual heat.

Safety System

System important to safety and provided to assure that under anticipated operational occurrences and accident conditions, the safe shutdown of the reactor followed by heat removal from the core and containment of any radioactivity, is satisfactorily achieved. (Examples of such systems are shutdown systems, emergency core cooling system and containment isolation system).

Secondary Containment

The structure surrounding the primary containment that acts as a further barrier to limit the release of radioactive materials and also protects the primary containment from external effects. It includes secondary containment structure and its access openings, penetrations and those systems or portions thereof, which are connected to the containment structure.

Severe Accident

Nuclear facility conditions beyond those of the design basis accidents causing significant core degradation.

Suppression Pool

A pool of water located at the lowermost elevation of the reactor building, into which steam resulting from LOCA/MSLB is directly led and condensed to reduce the pressure in the primary containment.

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1. INTRODUCTION

1.1 General

1.1.1 Containment system of a nuclear power plant (NPP) is a safety system provided to limit/mitigate the consequences of postulated accidents in order to protect the plant personnel, the public and the environment. The containment serves as the final physical barrier for the radioactivity contained in the reactor core during normal operation, anticipated operational occurrences, design basis accidents, potential severe accidents and, to the extent practicable, selected beyond design basis accidents (BDBA).

1.1.2 The containment systems include:

- primary containment (PC) envelope;
- containment isolation system;
- energy management systems;
- containment atmosphere clean-up systems;
- barriers separating volumes V_1[1] and V_2[2] ;
- the secondary containment (SC) envelope; and
- hydrogen management system.

1.1.3 Containment systems are provided to limit the release of radioactive fission products to the environment from the reactor core and reactor coolant system during and after accident conditions. For postulated accidents considered in the design basis, the estimated release shall be within acceptable limits. Design basis events that may dictate the design requirements of the containment, e.g. its pressure rating, leak tightness and performance requirements of the engineered safety features (ESFs), are large break loss of coolant accidents (LOCA), main steam line break (MSLB) and LOCA with impairment of emergency core cooling system (ECCS).

In addition, containment system also performs the following functions:

- shields from radiation during operational states and accident conditions;

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1 Volume V_1 (dry-well)
The areas of the reactor building (free space) containing all the high enthalpy systems of the reactor.

2 Volume V_2 (wet-well)
The areas of reactor building (free space) not falling in part of volume V_1 including the suppression pool.

- limits in conjunction with other systems, the release of airborne radioactivity to the environment within the prescribed limits during all operational states (i.e. during normal operation and shutdown); and
- serves as protective housing around reactor and associated systems.

1.2 Objective

One of the primary aims in siting, design, construction and operation of nuclear power plants is to ensure that during both normal operation, design basis accident and severe accident conditions, the consequences of radioactive releases to the plant personnel, the public and the ennvironment is within the acceptable limits. Containment system is provided in an NPP to achieve this goal. The objective of this guide is to bring out the design safety requirements of various parts of the containment system which are required to meet the provisions as stated in the AERB code of practice on 'Design for Safety in Pressurised Heavy Water Based Nuclear Power Plants' (AERB/SC/D).

1.3 Scope

This guide gives the functional requirements, design basis and design requirements of the containment system of pressurised heavy water reactors (PHWRs) to meet the above objective.

1.4 Structure

1.4.1 The sections and sub-systems of this guide cover general design considerations and design basis for the following:

- the primary and secondary containment envelopes;
- the containment isolation system; and
- the energy and radionuclide management systems.

1.4.2 Basis for arriving at containment design parameters is given in Appendix-A. Methodology for assessment of radiological release from containment during DBA is given in Appendix-B. A brief description of Indian PHWR containment systems is given in Annexure-I. The inter-relationship among various guides having a bearing on containment system design is given in Annexure-II. The categories of piping systems and their isolation provisions are given in Annexure III.

2. GENERAL DESIGN CONSIDERATIONS

2.1 General

The containment system shall be designed to cater to all DBAs. The design should be such that during any DBA, the release of radioactivity to the environment is within acceptable limits. In addition, consideration shall be given to the provision of features for the mitigation of the consequences of potential severe accidents.

2.2 Functional Requirements

The functional requirements of containment are as follows:

(i) The reactor core, as well as all piping of the main primary heat transport system should be totally enclosed within the containment structure; however, steam generators can be outside the primary containment, provided it is shown by analysis that there are no safety implications.

(ii) The containment structure should be able to withstand the maximum peak pressures arising from postulated loss of coolant accident (LOCA)/main steam line break (MSLB) events considered in the design, in conjunction with other loads. Considerations such as potential for generation and behaviour of flammable gases like hydrogen, assessment of ultimate load bearing capability of the primary containment structure, assessment of containment pressure build-up in the event of selected beyond design basis accidents should be given in the design for containment systems for postulated severe accidents.

(iii) The release of radioactivity from the containment should be controlled by a combination of the following:

(a) Containment isolation,

(b) Leaktightness of containment,

(c) Limiting the containment pressure by containment cool down/ depressurisation features/systems, and

(d) Reduction of radionuclide concentration by containment clean-up/confinement features/systems.

2.3 Design Features

2.3.1 The structural portion of the containment consists of two structures (i.e. primary and secondary conainment) and several interconnected compartments, housing various equipment in PC. The containment should withstand the pressure, thermal and mechanically induced loads resulting from various DBAs together with extreme environmental conditions (e.g. seismic, wind etc.).

2.3.2 The containment isolation features include the valves/dampers/systems and associated actuating devices and the related instrumentation and control systems to isolate the penetrations through the containment envelope.

2.3.3 Energy management features limit the internal pressure and temperature loadings on and within the containment envelope to design limits. Examples of energy management features are air coolers, pressure suppression system and primary containment controlled discharge (PCCD) etc..

2.3.4 The radionuclide management features limit the radiological consequences of the postulated accident conditions. These functions are achieved through the use of various filters. Examples of radionuclide management systems are: primary containment filtration and pump back (PCFPB) system, secondary containment filtration recirculation and purge (SCFRP) system. Plate-out on various surfaces and retention in the vapour suppression pool also help in reducing the concentration of the airborne radionuclides in the containment. These features work in conjunction with the energy management features and containment isolation features to meet the overall requirements of limiting the radioactivity release to the environment.

2.3.5 Combustible gas control features are designed to mitigate the consequences of release of hydrogen which may be generated by metal-water reaction in the core and radiolysis of water during and after accident conditions.

2.4 Quality Assurance Requirements

Quality assurance in design shall be followed as per AERB code of practice on ‘Quality Assurance for Safety in Nuclear Power Plants’ (AERB/SC/QA) and guide titled ‘Quality Assurance in the Design of Nuclear Power Plants’ (AERB/SG/QA-1).

2.5 Ageing Effects

The containment may be subjected to several ageing phenomena such as the corrosion of metallic components, loss of prestressing force (in prestressed containments), the reduction of resiliance in elastomeric seals, and the shrinkage and cracking of concrete. The detrimental effects of ageing cannot easily be identified during the plant lifetime. All ageing mechanisms are required to be identified and taken into account in design. Provision should be made for monitoring the ageing of the containment, for testing and inspection of components and where possible for periodically replacing items that are susceptible to degradation through ageing.

2.6 Testing and Inspection of Containment Structures

The design provisions for containment testing are covered in this section. Detailed requirements of containment proof and leakage rate testing are covered in AERB safety guide on ‘Proof and Leakage Rate Testing of Reactor Containments’ (AERB/NPP/SG/O-15). Provisions should be made during design to carry out the tests as brought out. In order to assure structural integrity and leak tightness of the containment structures and associated systems, tests are required to be conducted at specified conditions, prior to first criticality. In addition, in-service leakage rate tests are required to be conducted at specified intervals, during which structural monitoring will also be carried out using the long-term structural monitoring instruments provided for this purpose. Pre-operational and periodic testing of ESFs and control logics are also required.

The containment and associated systems should be designed to permit appropriate inspection and testing to ensure functionally correct and reliable actuation of the containment isolation valves and dampers and their leak tightness during the operational phase. Design provisions should be made for periodic inspection and testing of all containment systems.[3]

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3 AERB/NPP/SM/CSE-2 outlines the provisions for in-service inspection of civil engineering structures important to safety of nuclear power plants.

3. DESIGN BASES

3.1 General

3.1.1 The containment shall be designed to house the reactor, steam generators and auxiliary systems. The major function of containment system is to limit the radioactivity releases to the environment from the reactor core and from reactor coolant system during and after accident conditions. Its additional functions are to provide shielding during all postulated events, to minimise radioactive releases during operational states and to protect the reactor against external events.

3.1.2 The containment shall be designed to satisfy the requirements during all postulated events. In case of normal operating conditions, containment is designed for dead load (weights of components and structures), live load and the load during extreme environment conditions. It is also necessary that the accident conditions for which the containment is required to be designed should be selected in a conservative manner, which is at the same time physically credible. The accident conditions of relevance here are those related to the potential release of radioactivity outside the containment envelope from a source inside it. However, in some special situation, any other accident which does not involve radiological consequences but has the potential to affect containment structural integrity may also form the design basis.[4]

3.2 Considerations for Normal Operating Conditions

3.2.1 The containment shall be designed to withstand permanent gravitational/dead loads and temporary gravitational/live loads. It shall also be designed to withstand thermal loading due to temperature gradients across structural elements taking into account normal and extreme climatic conditions. Other environment conditions (i.e. wind and earthquake loads) also shall be considered (see subsection 3.3.2 for loads to be considered) in its design.

3.2.2 The containment structure is designed to protect the plant personnel from undue exposure to direct radiation from radioactivity contained in the containment or in the system within it. The adequacy of density of concrete used and thickness of the containment structure shall be checked against this requirement.

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4 For PHWRs of 220 MWe and 540 MWe design, main steam line break (MSLB) is the governing event for containment peak pressure.

3.2.3 The internal structures of reactor building are designed to provide atmospheric barriers between various areas which have different ventilation requirements and/or different activity levels (e.g. tritium). They are also designed to provide control of access to various rooms during reactor operation so as to prevent inadvertent entry of personnel into high radiation areas. To facilitate separation of areas of different activity levels, the openings between these are closed by suitable barriers. For openings which are required to be available for pressure equalisation during accident conditions, the barriers are designed to give way at specific pressure differential. The barriers could be in the form of louvres or plastic sheets etc.

3.2.4 The ventilation system is designed to provide clean, fresh and cool air in accessible areas. In addition, the exhaust flow is adjusted to maintain pressure gradient in such a way that no activity spread from relatively high active area to low active area could take place.

3.2.5 Reactor building coolers are provided at various areas within the containment building so that the heat load from hot parts of reactor systems could be effectively removed and specified air temperature could be maintained in various areas within the containment. These coolers may be designed to form part of energy management feature for accident conditions.

3.2.6 Provisions shall be made for access to reactor building for personnel and equipment movement. At least two independent paths shall be provided for personnel movement.

3.2.7 The opening of vent shafts through which volume V_1 and V_2 are interconnected should be covered with thin leak tight membrane such as plastic sheets to prevent supression pool water vapour getting into V_1 areas.

3.2.8 No permanent deformation of or damage to the containment structure shall occur. Structural integrity shall be ensured with large margins well within elastic limit.

3.3 Considerations for Accident Conditions

3.3.1 Internal Events

For containment design, the accident inside the containment is considered to be either LOCA or MSLB whichever may result in higher containment peak pressure. While secondary system pipe break accident does not involve release of any significant amount of fission products, LOCA on the other hand may involve fuel failure resulting in release of fission products to the containment. Therefore containment is designed to fulfill the following requirements:

(a) to maintain the structural integrity of the containment at maximum pressure encountered during DBA (higher of that resulting from LOCA or MSLB);

(b) to have low leakage rate through the containment envelope at the LOCA based pressure;

(c) to depressurise the reactor building (PC) following an accident (mainly by cooling down) to minimise fission product leakages to the environment;

(d) containment heat removal[5] and clean-up; and

(e) if the communication openings are covered to prevent activity spreading during normal operation, the same should be able to provide free path upon reaching a set pressure differential.

The determination of the maximum accident pressure and temperature over the entire break spectrum shall be based on the assumption that the accident occurs at nominal operating conditions. Following design parameters are required to be considered:

(a) Peak pressure of containment,

(b) Thermal loads arising from temperature gradients across structural elements,

(c) Peak differential pressure across the walls and floors of compartments within the primary containment,

(d) Peak negative pressure following DBA,

(e) Fluid jet impingement loads, pipe reaction loads, pipe impact loads and internally generated missiles,

(f) Overall integrated leakage rate (to be maintained within the specified limit), and

(g) Environmental conditions (resulting from accident and prevalent radiological conditions).

3.3.1.1 Basis for Containment Design Parameters

The details of bases for the calculation of containment design parameters during internal events (LOCA/MSLB) are given in Appendix-A.

3.3.1.2 Leakage Criteria

The maximum allowable leakage from containment shall not be more than the value used in safety analysis, which demonstrates that the acceptable dose

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5 Containment Heat Removal System
The system which cools the containment atmosphere and brings down the pressure and temperature inside the containment following any postulated initiating event including design basis accident.

values specified by regulatory body for accident conditions are not exceeded. Methodology to be adopted for radiological release calculations is given in Appendix-B. Acceptance criteria for measured leakage rate during containment integrated leakage rate test (ILRT) should ensure adequate margin over and above the allowable leakage rate to account for degradation between consecutive tests.

3.3.1.3 Containment structure shall be designed against the loading effects due to LOCA/MSLB as addressed in clause 3.3.1.1 following similar criteria specified in clause 3.2.8.

3.3.2 External Events

3.3.2.1 External events as applicable for the site shall be considered. The containment structure shall be designed for safe shutdown earthquake (SSE) for the site simultaneously with LOCA/MSLB loads. Local permanent deformations are acceptable for this condition. Structural integrity shall be ensured although with margins less than those for normal operation.

3.3.2.2 The containment shall be protected against or designed for the risks of external missiles, aircraft crash, floods, wind, external explosion and fire, depending on the nature and extent of the risks posed by surrounding site environment.

3.4 Considerations for Severe Accidents

3.4.1 Considerations shall be given in design of containment envelope and systems for potential severe accidents. Safety analysis should demonstrate the containment design response for potential severe accidents.

3.4.2 One of the features of relevance affecting the course of potential severe accident scenario is the presence of inventory of calandria vault water, and suppression pool water in current generation PHWRs, which may serve as corium cooling/retention mechanism for a significant period of time, in the event of loss of all active residual heat removal provisions (including PHT system cooling, ECCS and moderator cooling)[6] .

3.4.3 The following should be considered:

(i) Assessment of containment pressure build-up in the event of loss of all active residual heat removal systems, resulting in its slow re-pressurisation. This should consider release of non-condensibles from any predicted interaction between corium and concrete.

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6 For transferring the supression pool water to serve as corium cooling, appropriate design provisions should be made.

(ii) Free volume of the primary containment should be such that the global average concentration of hydrogen in containment building resulting from severe accident during the entire course of accident and during post-accident period will be less than that required for global detonation[7]. In this assessment, credit may be taken for passive mitigating measures such as catalytical recombiners. The evaluation should consider hydrogen contribution due to metal-water reaction as well as radiolysis of water, as applicable. The detailed assessment procedures and related guidelines are given in AERB safety manual on 'Hydrogen Release and Mitigation Measures under Accident Conditions in Pressurised Heavy Water Reactors', AERB/NPP-PHWR/SM/D-2.

(iii) During design the responsible organisation should pay particular attention for prevention of potential containment bypass in scenarios involving core degradation.

(iv) It should be demonstrated that for a reasonable period acceptable to AERB following the onset of core damage that containment leakage rate is maintained within acceptable limits. After this period, the containment must prevent uncontrolled releases of radioactivity.

3.4.4 For existing plants, the phenomena relating to possible severe accidents and their consequences should be carefully analyzed to identify design margins and measures for accident management that can be accounted for in safety analyses for severe accidents and mitigate the consequences of severe accidents. For these accident management measures, full use should be made of all available equipment, including alternative or diverse equipment, as well as of external equipment for the temporary replacement of design basis components. Furthermore, the introduction of complementary equipment should be considered in order to improve the capabilities of the containment systems for preventing or mitigating the consequences of severe accidents.

3.4.5 For new plants, potential severe accidents should be considered at the design stage of the containment systems. The consideration of severe accidents should be aimed at practically eliminating the following conditions:

- Severe accident conditions that could damage the containment in an early phase as a result of direct containment heating, steam explosion or hydrogen detonation;
- Severe accident conditions that could damage the containment in a late phase as a result of basemat melt-through or containment over-pressurisation;

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7 To prevent the risk of global detonation, a commonly acceptable limit on concentration of hydrogen in dry air is 10 % to 13 %. However, this may be higher in presence of steam.

- Severe accident conditions with an open containment notably in shutdown states;
- Severe accident conditions with containment bypass, such as conditions relating to the rupture of a steam generator tube or an interfacing system LOCA.

3.4.6 For severe accidents that cannot be practically eliminated, the containment systems should be capable of contributing to the reduction of the radioactive releases to such a level that the extent of any necessary off-site emergency measures needed is minimal.

3.4.7 Severe accident conditions may pose a threat to the survivability of equipment inside the containment owing to the high pressures, high temperatures, high levels of radiation (the effects of deposition of aerosols should be taken into account in estimating the values of temperatures and levels of radiation) and hazardous concentrations of combustible gases. Furthermore, the larger uncertainties in relation to the conditions in the containment following severe accidents should be taken into account by using appropriate margins in the survivability demonstration or in specifying protective measures (such as shielding). These factors should be taken into account in verifying the necessary survivability of equipment and instrumentation.

3.4.8 Leaktightness of containment structure shall be analysed and demonstrated for load combinations including dead loads, live loads, prestressing forces (if applicable) along with pressure and temperature arising out of severe accident conditions. A limited increase in leak rate is acceptable. The leak rate may exceed the design value but the leaktightness should be adequately assessed and should be considered in design.

3.4.9 Containment structure should demonstrate appropriate structural integrity in the event of potential severe accidents.

(i) When containment is subjected to long term pressurization, structural integrity should be ensured although with design margin smaller than that during normal operation/DBA conditions. However local permanent deformation may be permitted in these conditions as stated in (iii) below.

(ii) Such condition of structural integrity should also be satisfied for load combinations, including pressure, temperature and pipe reaction from severe accidents, which are similar to those considered in the design basis accident conditions.

(iii) Significant permanent deformation and some local damage are acceptable for load combinations that include local effects derived from potential severe accident conditions.

3.4.10 Structural integrity of the containment structure including major appurtenances (airlocks and penetrations etc., over which containment pressure bounday extends) against collapse should be demonstrated by calculating the ultimate load capacity considering both pressure and temperature. Ultimate load capacity of a containment structure should not be less than the maximum peak pressure calculated for potential severe accidents or two times the design pressure, whichever is higher.

4. PRIMARY CONTAINMENT SYSTEM

4.1 General

This section covers the requirements for the primary containment, its isolation system, energy management system and atmosphere control or clean-up systems. During normal operation, the PC is maintained under negative pressure as compared to atmosphere and volume V_1 is more negative than volume V_2.

4.2 Overall Requirements

4.2.1 Layout

The general layout of various compartments and equipment within the reactor building primary containment is dictated primarily by the functional requirements of the reactor systems (e.g. the need for locating steam generators at higher elevation than the reactor). However, to the extent that the layout affects the performance of the containment function, the following requirements should be considered:

(a) Piping and equipment containing high enthalpy fluids (viz. the PHT system and the secondary system piping and equipment), which are housed in PC, shall be located in volume V_1. Exception to this requirement is small diameter piping and tubings e.g. delayed neutron (DN) monitoring and other instrumentation impulse lines, failure of which in volume V_2 does not result in primary containment pressure exceeding its design value.

(b) Requirement of accessibility of equipment in PC during reactor operation should be considered. As a general rule, equipment requiring access during operation should not be located in shutdown accessible areas (which cover all of volume V_1 and some areas in volume V_2). Routing/location of high radioactivity equipment/piping in accessible areas should be avoided or they should be adequately shielded. Maintainability of the equipment located in the primary containment should be considered in the design.

(c) The layout should consider the possibility of internally generated missiles and jet impingement/pipe whip from postulated pipe breaks imposing loads on the containment structure. If these loads cannot be avoided by suitable layout, their effects on containment structure need to be evaluated.

(d) The layout of cables and pipes in the PC should follow the requirements of physical separation to ensure that common causes do not result in impairment of capability for reactor shutdown and decay heat removal.

(e) When non-metallic liner used for achieving the PC leaktightness requirement assists in decontamination, it should maintain its required function and should not get peeled/dislodged under both normal and accident conditions thereby clogging filters/strainers required for safety functions. Consideration should be given for thermal and radiation related degradation/ageing followed by LOCA environment. They should have good crack spanning ability, adhesion and low air permeability. After thermal ageing, there should be no loss of adhesion and no delamination or blistering.

4.2.2 The net free volume in PC should be considered while assessing the potential for global detonation of hydrogen generated in severe accident scenario (see subsection 3.4).

4.2.3 Performance

The containment shall meet the performance requirements for entire operating life of the plant. The containment isolation logics and devices should have the required speed of actuation and redundancy; the containment atmosphere energy removal (air coolers) and clean-up system capacity should be commensurate with that considered in safety analysis with adequate redundancy in equipment as required by single failure criterion. Various requirements are further elaborated in the following subsections.

The leak rates should be small enough to restrict radioactive releases to the environment to ensure that the relevant dose limits are not exceeded during normal operation or in accident conditions.[8]

For improved leak tightness of containment structure, measures such as minimising blank EPs, reduction of number of EPs by clubbing together the smaller EPs, EPs without corners and preferably round shaped, improvement in engineering of electrical EPs especially the mould and gland, long-term prestressing losses etc. should be considered in the design of containment structure.

In the design of new containment structures, engineered concrete mix with mineral admixures like high volume fly ash, silica fume and ground granulated blast furnace slag should be considered.

4.2.4 Reliability

The containment system shall be designed to have high functional reliability commensurate with the importance of the safety functions to be performed.

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8 Such limit may be taken as 1.5 % per day for unlined containment and 0.3 % per day for lined containment.

While designing for reliability of containment system, following requirements should be met:

(a) The single failure criterion shall be applied in accordance with AERB code of practice on ‘Design for Safety in Pressurised Heavy Water Based Nuclear Power Plants’, (AERB/SC/D).

(b) The reliability of safety support systems necessary for intended operation of the containment system shall be commensurate with the reliability requirement of the containment system.

(c) The components/equipment should be fail-safe to the extent practicable.

(d) Surveillance testing of logics and equipment which may be required for containment isolation should be able to be carried out without impairment of the minimum required containment functions, when primary heat transport system is hot.

(e) All necessary actions of containment equipment, which are initiated by automatic control logic, should also be capable of being initiated manually from the control room.

(f) Physical and functional separation of redundant instrument channels including sensors and power supplies associated with containment safety functions is required. Likewise support system supplies (instrument air, control and power supply) to redundant components shall be from redundant and physically separate trains of support system supplies.

4.2.5 Interface/Interaction with other Systems

4.2.5.1 Containment system shall be independent and decoupled from all process systems whose failure requires containment function, so that it acts as mitigating feature in case of failure of any process system such as PHT system.

4.2.5.2 Containment system shall be physically and operationally independent from other safety systems to the extent explained below:

No equipment that is part of the containment system shall be shared with other safety systems, if coincidental failure of these safety systems (as a result of failure of the shared equipment) has the effect of compounding the severity of the accident sequence. Alternately, the acceptability of sharing feature shall be adequately justified (e.g. high reliability, fail safe feature).

The above requirement does not apply to safety support systems as a shared feature.

4.2.6 Qualification

4.2.6.1 The term qualification means a formal test or appropriate analyses demonstrating the capability of structures, systems and components, to meet the conditions imposed on it during all design basis events and severe accidents as applicable. The containment structure and the various penetrations are subjected to a structural integrity and leakage testing. For any individual item of equipment, environmental or seismic qualification is often the overriding concern. The guidelines for qualifying the various equipment/ instruments are given in the following subsections.

4.2.6.2 To achieve environmental qualification, a programme shall be established to confirm that all containment system components are capable of continuously meeting, while being subjected to any environmental conditions defined in the design basis (pressure, temperature, humidity, radiation, etc.), the design basis performance requirements needed for their function. The various items of containment equipment shall be qualified for the environmental conditions that are likely to be encountered and for the period of time they are required to perform their function. Conservative estimates of the environmental conditions and their time periods should be made. The conditions shall include:

(i) Normal operating environment for the lifetime of the plant (or shorter durations, if equipment is proposed to be replaced at such durations),

(ii) The expected number of cycles of anticipated operational occurrences, and

(iii) The limiting effects resulting from the DBAs.

4.2.6.3 The methods of qualification are:

(i) Performance of a type test on equipment representative of that to be supplied,

(ii) Performance of an actual test on the supplied equipment,

(iii) Use of pertinent past experience in similar applications, and

(iv) Analysis based on reasonable engineering extrapolation of test data or operating experience under pertinent conditions.

The foregoing methods of qualification shall be used in combination as necessary.

4.2.7 Maintainability

In the design and layout of containment system, due consideration shall be given to maintenance. The containment system shall be maintainable without undue radiation exposure to plant personnel.

Adequate working space and shielding should be provided to aid in maintaining the system.

4.2.8 Safety Classification and Seismic Categorisation

Safety classification and seismic categorisation of structures, systems and components of the containment system shall be as per AERB safety guide on 'Safety Classification and Seismic Categorisation for Structures, Systems, and Components of Pressurised Heavy Water Reactors', AERB/NPP-PHWR/SG/D-1. Interfacing components related to containment systems should have the same safety class as the higher safety class system to which it is connected.

4.2.9 Instrumentation and Control

(a) The design of instrumentation for containment box up logics, ESF logics, etc. associated with the containment system function shall be in accordance with the AERB code of practice on 'Design for Safety in Pressurised Heavy Water Based Nuclear Power Plants', AERB/SC/D and AERB safety guide on 'Safety Systems for Pressurised Heavy Water Reactors', AERB/NPP-PHWR/SG/D-10 and 'Safety Related Instrumentation and Control for Pressurised Heavy Water Reactor Based Nuclear Power Plants', AERB/NPP-PHWR/SG/D-20.

(b) For normal operation:

Design provisions should be made to monitor:

- Pressure-normal range
- V_1-V_2 differential pressure
- Area temperatures
- Ventilation exhaust activity
- Tritium levels in atmosphere
- Area radiation levels
- Fire/smoke
- Suppression pool level and its pH

(c) For accident conditions:

Following parameters should be monitored at appropriate locations:

(i) Pressure

(ii) Temperature

(iii) Radiation levels

(iv) Concentration of Hydrogen inside the containment

Range of these instruments should be sufficient to take into account possibility of environmental conditions departing from design conditions and to the extent possible for severe accidents scenario.

(d) For earthquake:

Instrumentation for earthquake shall be provided as per AERB safety guide on 'Seismic Studies and Design Basis Ground Motion for Nuclear Power Plant Sites', AERB/SG/S-11. For a seismically homogeneous site, where identical units are constructed, seismic instrumentation on a single unit is sufficient on condition that alarms are provided in control rooms of all the units.

(e) Design shall provide capability to monitor the status of the following related to containment and its ESFs:

(i) Containment isolation devices- open/closed status of each damper or valve

(ii) ESFs related valves (open/close status) and other equipment (on-off status)

(iii) On/off status of each fan motor

(iv) Bed temperature for the charcoal filters

(v) Differential pressure across HEPA filters.

Instrumentation shall be provided to perform the operability check for each of the containment related ESF devices like circulation fans and isolation damper/valve.

(f) For long term structural monitoring:

Instrumentation shall be provided to monitor, during the operational life of the containment, the state of pre-stress, i.e. the residual compression in general section of the primary containment and corrosion status of prestressing system to enable assessment of the capability to withstand the over pressure under design basis accident.

4.2.10 Shielding

The concrete thickness of the containment structures should ensure that during normal operation, the radiation dose outside the containment is within specified limit (typically less than 1 μSv/h (0.1 mrem/h).

For postulated accident conditions, an assessment should be made regarding expected radiation fields outside the containment in areas requiring personnel occupancy during these conditions, especially all the control rooms. This assessment should be done for the limiting design basis accident conditions involving large release of fission products from reactor core to the containment atmosphere. The dose limits during DBAs for control room operating personnel are specified in safety guide on 'Radiation Protection Aspects in Design for Pressurised Heavy Water Reactor Based Nuclear Power Plants', AERB/NPP-PHWR/SG/D-12.

4.2.11 Compressed Air in-leakage

During all design basis events requiring box-up of the containment, the in-leakage of compressed/instrument air into the containment would result in its gradual re-pressurisation, thus adversely affecting radioactivity releases from the containment. Design provisions should therefore be made to enable manual isolation of compressed air within 1 to 2 hours after the accident and minimising instrument air supply to the containment during such accident conditions. To enable minimising instrument air leaks, it may be necessary to have separate header for air supply to some valves/equipment which require air supply on extended basis for safety functions to be performed under the relevant accident conditions.

The design should ensure that, should an initiating event results in a guillotine break of compressed air lines, the pressurisation of the containment due to the air ingress is limited to and will not exceed the maximum containment pressure assessed in the safety analysis, and shown to be acceptable.

4.2.12 Power Supply

The containment system shall be able to perform its safety function without credit for availability of Class IV electric power supplies. For further guidance refer safety guide on 'Emergency Electric Power Supply Systems for Pressurised Heavy Water Reactor', AERB/SG/D-11 and 'Safety Systems for Pressurised Heavy Water Reactors', AERB/NPP-PHWR/SG/D-10.

4.2.13 Recovery after Accidents

To the extent reasonably practicable, design provisions should be made for assisting in recovery after accidents (e.g. easily decontaminable internal surfaces, means for disposal of liquids and gases inside containment, which become contaminated in an accident).

4.2.14 Decommissioning

Attention shall also be paid to the features, which would assist final decommissioning of the plant (e.g. selection of construction materials to reduce activation products during operation). For further guidance refer AERB safety manual on 'Decommissioning of Nuclear Facilities', AERB/SM/DECOM.-1.

4.3 Containment Isolation System

To ensure that the containment isolation requirements are not defeated, piping systems or other openings that penetrate the containment envelope shall have appropriate provisions for containment isolation in response to any containment isolation signals. These features make up what is known as the containment isolation system.

4.3.1 Design Criteria

The design basis for the containment isolation system is derived primarily from the requirement of early closure of all the openings in the containment envelop which could constitute path for radioactive release to the environment and maintaining leak tightness thereafter. To achieve the purpose of limiting release of radioactivity to the outside of the containment, the isolation devices should be fail safe and shall close at a speed which takes proper account of the potential release hazard. However, it may be necessary to limit the closing speed of valves or dampers, particularly for larger penetrations, to ensure proper functioning and tight sealing. Sealing material should be able to withstand the high radiation level, pressure, temperature and humidity condition prevailing following the postulated accident condition.

The containment isolation system includes dampers, airlocks, isolation valves and other devices which are required to seal/isolate the penetrations through the containment envelope. It also includes instrumentation and controls which actuate their closure in response to the containment isolation signals and maintain in closed condition.

Single failure criterion should be satisfied to ensure reliable isolation. Credit can be taken for passive system boundary if it is capable of withstanding the containment design pressure, temperature and seismic loads. The isolation system should be of same safety class as that of the containment structure.

The isolation provisions covered in subsection 4.3 are for primary containment alone. Isolation provisions required for secondary containment are covered in section 5.0.

4.3.2 Piping and Ducts

The basic principle in the containment isolation philosophy is that, two isolation barriers shall be provided for each penetration. For piping and duct systems these barriers can be a combination of the system boundary, isolation devices, etc., and each shall be capable of retaining the applicable pressure and preventing the release of radionuclides. The categories of piping systems and their isolation provisions are given in Annexure III.

(a) Each line that is not part of a closed loop that penetrates the containment and which:

   (i) directly communicates with the reactor coolant during normal operation or accident conditions, or

(ii) directly communicates with the containment atmosphere during normal operation or accident conditions;

shall be provided with two isolation valves in series. Each valve shall be either normally closed or automatic. Preferably one valve should be inside and other outside the containment boundary. Each valve shall be reliably and independently actuated. For the above case, portion of the piping or duct passing through the containment wall and upto the second isolation valve should be designed for containment design pressure, temperature and seismic loads.

(b) Lines penetrating the containment and forming part of closed loop either inside or outside the containment shall have at least one isolation valve outside the containment at each penetration. However, the pressure boundary of the closed loop system should be able to withstand pressure, temperature and environmental conditions of the containment following the design basis accident alongwith seismic loads. Also, leak tightness integrity of the pressure boundary should be at least of the same quality as that of containment. Where the failure of a closed loop is required to be assumed as a design event, only a single isolation valve may be provided if it can be demonstrated that the release of radionuclides is within acceptable limits assuming a single failure. This valve shall be either automatic, normally closed or remotely operated as required to accommodate the postulated accident sequences. Where this cannot be demonstrated, the same requirements as referred at (a) above shall be applied to each line of the closed loop.

(c) Lines penetrating the containment and forming part of closed loops both inside and outside the containment envelope shall have at least one isolation valve, remotely operated, normally closed or automatic, outside the containment envelope at each penetration.

The design shall recognise the conflict arising between the requirements for containment isolation provisions and the requirements for necessary safety systems that penetrate the containment envelope. In such cases, consideration of the isolation provisions shall be balanced by the required availability of the necessary safety systems and the need to avoid the escalation of the accident condition. For those pipelines penetrating the containment and which cannot be isolated during containment box up signal in view of their safety functions, it should be ensured during design that such of these system boundaries are qualified for containment pressure boundary.

4.3.3 Valves, Dampers and Access Doors

4.3.3.1 For the selection of valve types and their location with respect to the containment envelope, the following guidelines are to be applied to each of the three categories (see subsection 4.3.2) of isolation, as indicated in Annexure-III:

- Check valves that depend only on system pressure for closure shall not be used as automatic containment isolation valves,
- Isolation valves shall be located as close to the structural boundary of the containment as is practical,
- Process valves may be used for containment isolation if they meet all requirements for the containment isolation system, and
- As an exemption to the requirements in subsection 4.3.2, small dead ended instrument lines (e.g. 25 mm inside diameter or smaller) require only one manually operated valve outside the containment. Those closed both inside and outside the containment and designed to withstand the DBAs for the containment system (e.g. sealed fluid filled tubing with protective shielding) may not require isolation valves.

4.3.3.2 Airlocks [main airlock (MAL), emergency airlock (EAL), fuelling machine airlock (FMAL) etc.] used for access of personnel or equipment, shall be provided with at least a double-door arrangement, and there shall be an interlocking arrangement so that one door, the inner or the outer is always closed.

4.3.4 Isolation Signals

Containment shall be isolated on the basis of the following signals:

- High radiation in the containment atmosphere
- High pressure in V_1 volume

In any accident condition, certain lines which penetrate the containment envelope are required for the operation of performance of safety functions (e.g. ECCS lines etc.). Such lines should either not have automatic containment isolation feature, or else, there should be provisions to override the containment isolation signal and it shall be ensured by other means that releases of radioactivity through the containment envelope do not exceed acceptable limits.

4.3.5 An integrated ‘containment isolated’ indication in control room should be provided indicating completion of closure of all the penetrations, valves, dampers and doors after containment box up, on containment isolation signal.

4.3.6 Status of containment isolation valves, dampers and doors shall be monitorable from control room.

4.4 Primary Containment Pressure and Energy Management System[9]

Energy management is a generic term used to describe all those features that affect in some way the energy balance within the containment envelope. No specific requirements for utilisation of individual energy management systems can be given, because it is the total effect of the combination of these systems that is important in reducing the pressure and temperature within the containment envelope. To the extent practicable, the energy management systems should be passive.

The design parameters of energy management systems form the inputs for containment analysis for pressure and temperature transient and consequent activity releases. The requirements applicable to this analysis are brought out in Appendix-A.

Typically different energy management systems may be the following:

(i) Containment structure

(ii) Vapour suppression system

(iii) Reactor building air coolers

(iv) Primary containment controlled discharge system

The concept and overall requirements of each of these systems are described in the following subsections:

4.4.1 Containment Structure

The net free volume within the containment is the primary physical parameter determining peak pressure after postulated pipe rupture events.

In addition to providing a pressure resistant and leak tight envelope, the containment structure also acts as a passive heat sink. The heat transfer to structures is an important parameter in determining the pressure and temperature transients. The primary heat transfer mechanism is the condensation of steam on exposed surfaces. The thermal conductivity of the structure plays an important part in determining the rate of heat transfer to the structure. The presence of coatings shall be considered in determining the heat transfer rate to the structures.

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9 Containment Pressure and Energy Management System
The system provides the initial pressure suppression, subsequent reduction of temperature and pressure and long-term heat removal, in the containment following an accident. This system includes reactor building atmosphere cooling system and suppression pool system.

The design should provide for ample flow routes between separate compartments inside the containment. The cross-section of openings between compartments should be of such dimensions as to ensure that the pressure differentials occurring during pressure equalization in AOO and DBA do not result in damage to the pressure bearing structure or to other systems of importance in limiting the effects of the accidents.

4.4.2 Vapour Suppression System

In the event of postulated pipe rupture accident, the steam and air mixture flows from volume V_1 to volume V_2 via the vapour suppression pool. During the flow passage, the steam gets condensed in cold pool water and non-condensables (e.g. air), leave the pool water surface after getting cooled. The effectiveness of the vapour suppression system in absorbing the fraction of the total energy released and limiting the peak pressure depends on the primary containment volume, volume V_1 to V_2 ratio and the vent area. Considerations should be given to these aspects taking into account other constraints like equipment layout. For details refer safety guide on 'Vapour Suppression System (Pool Type) for Pressurised Heavy Water Reactor' (AERB/SG/D-22).

4.4.3 Reactor Building Air Coolers

Reactor Building Coolers are designed to remove the thermal energy from the containment atmosphere during accident. In general, the coolers have a number of air cooling units and are located at different places in the containment. Each air cooling unit consists of cooling coils, preferably direct driven fan-motor unit, filter sections, coolant, condensate drain and cabinet[10]. For heat removal during accident conditions, credit should be taken for those coolers which are specifically designed for the duty, with their power supply drawn from Class-III power and cooling water from safety related process water system.

During pipe rupture events, the air coolers operate largely in the condensing heat transfer mode. It is therefore, important that appropriate heat transfer correlations be used in the design assessment. Coolers should be designed taking into account the increased air density as a result of high building pressure during the accident. Location of these coolers should be such that they have high effectiveness (e.g. near ceiling of a room, in the area of high temperature etc.).

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10 Air cooling units, provided in accessible area, i.e. volume V_2 of primary containment and in secondary containment, are designed to remove heat from the respective areas during 'normal operation/shutdown' of the reactor. Heat from volume V_2 and secondary containment is removed by chilled water/process water coolers. These coolers are operated on Class-IV power supply. No credit is taken for these coolers for accident analysis.

The building air coolers together with other heat sinks should have sufficient capacity to bring down containment pressure and temperature in reasonable period of time, such that ground level releases, considering operation of radionuclide management systems, are within acceptable limits. The coolers shall be designed for continuous, long-term and automatic operation.

The equipment shall be designed for both safe shutdown and operating basis earthquake conditions.

The construction material for all the components of the system should be selected properly taking into account the postulated radiaton levels, and mechanical considerations. The selected material for electrical insulation should have adequate radiation resistance and should be suitable for post accident environment.

Adequate redudancy shall be provided in the coolers system. The system shall be designed against failures due to common causes. The system should also be protected from the internal events. The power supply to the mutually redundant equipment shall be from different routes so that even during failure of one route, availability of minimum capacity is ensured.

Operating status of each pump room and fuelling machine vault coolers should be available in control/control equipment room. Standby coolers should start automatically on demand i.e. tripping of operating cooler or on signal indicating an accident. Facility for actuation of each cooler should be available in control/ control equipment room.

Provision shall be made for periodic functional test of the system. Equipment should be tested as per applicable standards.

4.4.4 Primary Containment Controlled Discharge (PCCD) System

While the energy management features/systems described above cool down and depressurise the containment atmosphere to a low pressure, further depressurisation (say below 0.05 kg/sq.cm.g) may be difficult to be achieved by cooling alone. Also in-leakage of instrument air may lead to a gradual re-pressurisation over a period of time. To cater to the requirement of releasing the pressure, a design provision should be made for resorting to the option of controlled discharge to stack via filters if the circumstances demand. The operation of the system would reduce ground level release while adding to the release via stack. In the current Indian PHWRs using double containments, the PCCD option is envisaged to be used (if at all required) only after a delay of atleast 48 hours.

The range of pressure in the containment after which the PCCD system can be allowed to be operated should be specified and the system process design should permit operation over this pressure range without the differential pressure rating of any components including filters being exceeded. Typical current designs allow this sytem to be operable for containment pressures up to 0.4 kg/sq.cm.g. The intake for PCCD should be taken from volume V_2, where activity concentration is expected to be relatively low and the discharge should be routed via stack.

During post LOCA blow down phase, rate of pressure reduction in volume V_1 might be faster than that in V_2. Suitable means should be provided to equalise the pressure between volume V_1 and V_2.

4.5 Primary Containment Atmosphere Control System (PCAS)[11]

Primary containment atmosphere control system (PCAS), should include systems for ventilation, radionuclide management (containment clean-up system), and those for combustible gas control.

Radionuclide management should include all the systems used to control the movement of radionuclides released within the containment envelope so that release to atmosphere is minimised.

To estimate the total release of radionuclides within the containment envelope, conservative analysis shall be made and these shall be used as the basis for the design of any radionuclide management system. Some radionuclide management systems also perform the function of energy management systems. Combustible gas control system[12] should include systems for monitoring hydrogen in containment and its mitigation during DBAs.

4.5.1 Containment Cleanup System (CCS)/Radionuclide Management System

Design of this system should take into account the following guidelines:

(i) The clean-up systems should be designed to operate efficiently and reliably under the environmental conditions resulting from DBA.

(ii) The enivronmental conditions prior to the DBA may affect the performance of the filters and adsorbers in the clean-up system. It should be ensured by design and operating practices that industrial

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11 Primary Containment Atmosphere Control System (PCAS)
The system comprises of venitillation system, radionuclide management system (i.e. containment cleanup system) and combustible gas control system.

12 Combustible Gas Control System
The system provided to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere during accident conditions to prevent explosion or deflagration which could jeopardise the containment integrity. This system is also sometimes termed as ‘Containment Atmosphere Control System’.

contaminants, pollutants, temperature and relative humidity levels do not significantly reduce the capability of CCS to perform the intended function. Possibility of build-up of moisture in the adsorbers due to condensation of leaked out steam or any other reason should be prevented by design.

(iii) In general the CCS consists of most or all of the following components:

- Demisters
- Prefilters
- High efficiency particulate air (HEPA) filters
- Adsorbers
- Fans
- Associated ducting and dampers
- Instrumentation

Demister is provided to remove entrained water droplets from the inlet stream thereby protecting filters and adsorbers from water plugging/accumulation. Heaters alone or in conjunction with cooling coils should be used when humidity is to be controlled before filtration. Pre-filters are provided to remove the large size particles and prevent excessive loading of HEPA filters. To some extent demisters may also perform this function. The HEPA filters are provided to remove the fine particulate matter from the air stream. The adsorbers remove gaseous iodine (elemental and organic iodides) from the air stream. HEPA filters provided in some systems at the downstream of the adsorbers would also collect fine particles and offer redundant protection against particulate release in case of failure of the upstream HEPA filters.

(iv) Adequate redundancy shall be provided in CCS and/or their components to meet single failure criterion as required for active components. However, exceptions should be properly justified. The redundant systems should be physically separated, so that both the redundant sub-systems/components do not get damaged due to any reason. CCS should also be protected from the possible missiles generated from high pressure systems, rotating machinery failure or natural phenomenon to the extent possible.

(v) The design of each adsorber section should be based on the estimated concentration and relative abundances of iodine species (elemental, particulate and organic) in the containment following DBA. Credit for reduction in concentration of iodines due to water trapping and plate-out on the surfaces may be taken based on experimental studies and/ or best available knowledge of radionuclide deposition on surfaces.

(vi) If impregnated activated charcoal is used as the adsorbent, the adsorber section, in typical designs, is designed for a residence time of atleast 0.25s per 50 mm of bed thickness. The adsorber section should be designed for a maximum loading of 2.5 mg of total iodine (radioactive+stable) per gram of activated charcoal.

(vii) The design of the adsorber section should consider possible iodine desorption and auto-ignition of charcoal that may result from the temperature rise due to heat from the fission products. Reliable provision shall be made to limit the charcoal temperature below the desorption temperature for iodines.

(viii) If adsorbent other than impregnated activated charcoal (say metal zeolites) is proposed to be used in CCS or if the quality of carbon used is not as per specified requirements, then the performance of the proposed adsorbent shall be demonstrated to be satisfactory.

(ix) The CCS should be suitably instrumented for monitoring the physical status and performance of the system. Provision should be made for measuring differential pressure across filters, temperature of charcoal in the adsorber section, etc.

(x) The HEPA filters should be designed, constructed and tested as per ANSI-N-509 or equivalent. Each HEPA filter should be shop tested using diOctyl phthalate (DOP) and penetration should not be more than 0.03 % at rated flow for particle size down to 0.3 microns when tested as per MIL Standard 282 or BIS-3928 or equivalent. In-situ testing for these HEPA filter should however be conducted for particule sizes of 0.5 microns.

(xi) Provison should be made to facilitate periodic testing of the systems. It should be ensured by design and operation that there is no undue radiological impact due to use of radio-iodine during periodic testing of adsorber sections for filter efficiency.

(xii) All structures, systems and components of CCS should be designated to safety class and seismic category and should be qualified for LOCA environment.

(xiii) Power supply to the blowers of CCS should be fed from Class III buses.

Any portion of the ducting or piping of these systems passing through floors/ walls separating volumes V_1 and V_2 and are potentially communicating with these volumes, should have two isolation valves located preferentially in volume V_2, and the portion between the valves (including the valves) should be able to withstand the following:

(a) containment design pressure externally, and

(b) the portion upto the 2nd valve in V_2 should be able to withstand the peak differential presure between V_1 and V_2 internally.

Design considerations and requirements specific to individual atmospheric control systems are described below.

4.5.1.1 Primary Containment Filtration and Pump Back System (PCFPB)

This system is used to remove iodine activity on a gradual basis from the RB atmosphere following an accident. Intake to the system should be from a region of higher activity concentration during accident (volume V_1) and exhaust to volume V_2. The design should provide adequate iodine filter capacity commensurate with fan flow rate and iodine concentration within containment. The system should be maintained in such a way that contamination by pollutants is avoided. Filter capacity should be adequate so as not to exceed the limits of mechanical loading (filter choking consideration) or thermal loading (over heating consideration). Adequate cooling means should be provided to avoid over heating of filters after they are loaded. The iodine filter should be preceded by moisture separator and particulate filters to remove moisture and to reduce the load on iodine filter. The efficiency of the sorption material (iodine filters) in removing iodine in humid condition shall be demonstrated in laboratory test. Provisions shall be provided to test the filter system insitu (leak tightness of the filter assembly).

4.5.1.2 Primary Containment Controlled Discharge System (PCCD)

The primary function of PCCD system is to depressurise the containment under post accident conditions. The HEPA/charcoal filter in the system, removes the fission products from the gaseous discharges through the stack.

4.5.1.3 Containment Structure

The containment structure and its internals provide a large surface area for radionuclide deposition. The plate-out and desorption factor allocated to the containment structure shall be conservatively based on the best available knowledge of radionuclide deposition on surfaces.

4.5.1.4 Suppression Pool

The radionuclide management function of the suppression pool is to dissolve or entrap radioactive substances which come out of the fuel and otherwise would have been airborne. This is mainly achieved when pool water is used as a recirculation water for the long term core cooling. This serves to limit the radiological consequences resulting from containment atmosphere leakage during an accident. Chemical additives (such as sodium hydroxide) can be mixed in water to enhance the trapping of radionuclides. Removal of radioiodine is of particular importance because of its higher specific dose consequence. Appropriate pH value (alkaline) of pool water should be maintained by proper additive to enhance the iodine trapping in such a way that a large fraction of iodine remains in aqueous form.

4.5.2 Ventilation System

Ventilation System performs atmospheric control function in PC during normal operation. Design requirement of this system are given in subsection 3.2.4.

4.5.3 Combustible Gas Control

After a loss of coolant accident, a mixture of hydrogen and air could be formed in the containment atmosphere. The design provisions for management of hydrogen should be addressed based on assessment of the hydrogen generation and its dispersion in the containment. The assessment of hydrogen generation should take into account its production due to metal-water reaction as well as radiolytic decomposition of water and other combustible releases possible during accident conditions.

5. SECONDARY CONTAINMENT SYSTEM

5.1 General

5.1.1 Secondary containment (SC) is designed to intercept and hold up the radionuclides leaking from PC during a DBA, BDBA and potential severe accident conditions. It also protects the PC from external effects such as missiles, solar radiation, and wind. Also, the combined thickness of PC and SC concrete wall provides shielding for areas outside the containment. A filtered purge system, which discharges to stack, provided in SC quickly reduces its pressure to a negative gauge pressure following DBA. If the negative gauge pressure is not maintained on account of high leakage from PC or low purge rate, ground release of unfiltered radionuclides would take place. However, because of dilution of the concentration of radionuclides in the SC space, the consequence of this release would be much less as compared to the direct release from PC.

5.1.2 The SC system includes the SC structure and the safety related systems provided to control the ventilation and clean up of potentially contaminated SC volume following a DBA.

5.2 Design Criteria

The following criteria applies to the design of secondary containment:

(i) SC system should be able to intercept leaks for various leak paths from PC. Activity analysis for radiological release should take into account the fraction by which PC could be bypassed.

(ii) The functional capability of SC system should be such as to quickly achieve and/maintain a negative pressure (typically atleast 12 mm water column for 220 MWe PHWRs) during design basis accident.

(iii) Thickness of these structures shall be governed by design basis load including design pressure and by shielding requirement. Seismic load and design pressure should be combined together. For further details, refer AERB safety standard 'Design of Concrete Structures Important to Safety of Nuclear Facilities', AERB/SS/CSE-1.

Structural design should consider potential natural events (e.g. wind, solar radiation, earthquake, etc.) as well as man-induced events including externally generated missiles, etc. if applicable.

The structures should be designed for both serviceability as well as strength requirements under all design conditions.

(iv) SC should have required leaktightness at the specified pressure to ensure that radiological releases under design basis accident is within specified limit.

(v) Provision should exist for periodic inspection and functional testing of the SC structure.

(vi) Safety classification and seismic categorisation of the containment system shall be as per applicable AERB safety guide.

(vii) Requirement of single volume of annular space between PC and SC to maximise the mixing and dilution of any radioactive material released from the primary containment in the event of an accident.

5.3 Design Features

To meet the above design criteria, SC should have following design features (or suitable alternatives):

(i) The containment boxup signals specified in subsection 4.3.4, shall be instrumented to effect closure of valves/dampers, etc. required for SC isolation.

(ii) Provision should be made to restrict the PC leakage bypassing the SC consistent with 5.2 (i)

(iii) Appropriate sealing arrangement should be made at various leak paths including penetrations and construction joints to ensure leak tightness. Construction practice for concreting should consider leak tightness requirement.[13]

(iv) Provision should be made for SC recirculation, filtration and purge system with suitable fans and filters to maintain negative pressure of recommended range (12 mm to 24 mm of WC) within SC during accident condition.

(v) Pipe penetrations/opening into PC and passing through SC shall have at least one isolation valve/dampers after SC wall.

(vi) Appropriate paint liner may be used for achieving the SC leaktightness requirement. The paint used in this application should be qualified for thermal ageing consistent with its intended life and the expected normal environment (temperature, humidity, etc). After thermal ageing, there should be no loss of adhesion and no delamination or blistering. If significant radiation is expected during normal or accident condition, this should also be included in the qualification programme.

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13 A typical value of leakage rate equal to 400 S cubic meter per hour is specified at over pressure of 200 mm of water column (WC) for 220 MWe PHWRs.

5.4 Secondary Containment Atmosphere Control System

5.4.1 General

This system is also called as secondary containment filtration, recirculation and purge (SCFRP) system and is required to be operated under accident conditions when containment is isolated. This system provides filtration and mixing by recirculation within SC space, and also maintains negative pressure within the SC space. To achieve these functions, a fixed amount of air from SC space is passed through iodine and particulate filters by fan, after which the majority of the flow recirculates back to the SC space, while the remaining small fraction is purged through stack via additional filters to maintain a negative pressure in SC space.[14]

5.4.2 Design Requirement

This system should meet the following requirements.

(i) The system should have enough purge capacity to be able to quickly reduce the secondary containment pressure to a sub-atmospheric value and maintain it over an acceptable pressure range.

(ii) The valves in the purge exhaust line should be 'fail to open' type on failure of power/air supplies.

(iii) HEPA filter(s) in-situ efficiency shall be better than 99.97% for 0.5 micron DOP particles.

(iv) Charcoal filter in-situ efficiency shall be better than 99% using iodine labelled as radio-iodine.

(v) This system shall be started on auto in the event of a containment box-up signal.

(vi) There shall be standby provision for fans.

(vii) The design should have adequate capacity for iodine filter commensurate with fan flow rate/iodine concentration within the containment.

(viii) System shall be SSE qualified with provision of class-III power supply.

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14 The pressure of SC is maintained between certain values (typically 10 to 40 mm WC for current designs) by auto opening and closing of purge dampers.

5.5 Instrumentation Requirements

The design of instrumentation for secondary containment isolation and engineered safety systems shall be in accordance with the AERB safety code, 'Code of Practice on Design for Safety in Pressurised Heavy Water Based Nuclear Power Plants', AERB/SC/D and AERB safety guide on 'Safety Systems for Pressurised Heavy Water Reactors', AERB/NPP-PHWR/SG/D-10 and AERB safety guide on 'Safety Related Instrumentation and Control for Pressurised Heavy Water Reactor Based Nuclear Power Plants', AERB/NPP-PHWR/SG/D-20 as applicable.

(a) For Normal Operation

Instrumentation shall be provided to monitor pressure of secondary containment during normal operation.

(b) For Accident Conditions

Design provision shall be made for automatic stoppage of ventilation system and SC isolation and starting of SCFRP to establish sub-atmospheric pressure in SC by purging the air through filters under accident conditions.

(c) Design shall provide capability to monitor the status of following containment related features in the control room:

(i) Secondary containment ventilation isolation dampers (open/close status).

(ii) ESF related valves/dampers (open/close status) and other equipment (on-off status).

5.6 Protection against High Enthalpy Pipe Rupture within Secondary Containment

The high enthalpy lines within the SC, whose rupture could pressurise the SC shall be provided with guard pipe in such a way that in the event of rupture of the steam pipe segment within SC, the release of high enthalpy steam could be directed to outside atmosphere. Alternatively, it shall be ensured by design that this length of pipe does not rupture. Otherwise the SC should be provided with blowout panel to restrict the pressure rise and should be designed for suitable differential pressure at which it should open up. This requirement is important in case the steam generator is kept out of PC.

5.7 Design Evaluation

Following data shall be considered in the analysis of SC system:

(i) Leakage from PC to SC.

(ii) Purge flow rate and recirculation flow rate.

(iii) Initial conditions assumed for the SC structure and atmosphere.

(iv) The ingress of compressed air in SC during accident conditions.

5.8 Interception of in-line Leaks from Primary Containment

5.8.1 In containment system designs (viz. in 220 MWe PHWRs with provision of 3 doors in airlocks and three containment isolation dampers in series), following provisions should be made:

(i) The interspace between second and third door (from outside) should be communicated to the annular space of SC through a vent line, and

(ii) The size of vent line shall be kept small enough so that even in case of 3rd barrier failing to close during accident, the pressure of the SC interspace is maintained at negative value.

5.8.2 For all the NPPs having SC enveloping PC and if credit is taken in estimatation of radiation dose to members of public under accident condition due to interception of leaks by SC; interception provided by the secondary containment should be tested periodically and results ensured to be in line with safety analysis report (SAR). Design provisions should be made for monitoring interception.

APPENDIX-A

BASIS FOR ARRIVING AT CONTAINMENT DESIGN PARAMETERS

A.1 General

A.1.1 The PC pressure and temperature analysis determines the system design parameters. A spectrum of high energy line breaks in the primary heat transport (PHT) system or secondary systems shall be considered in performing the analysis to determine the PC peak pressure and associated temperature. This spectrum shall consider the effects of break area, location and inital reactor power level to indentify the break yielding the maximum pressure and associated temperature. This spectrum shall include instantaneous double-ended breaks of the largest pipe in the system. Each analysis shall be carried out for a sufficient duration to ensure that the maximum peak pressure and temperature have been ascertained.

A.1.2 All considerations regarding modelling, assumptions and input, as well as relating to the spectrum of cases and events shall be taken in a manner which will render conservative design parameters for PC. Above analysis shall incorporate the effects of most severe single failure [e.g. failure of one of the atmospheric steam disharge valves (ASDVs), etc.].

A.1.3 Assumptions which are not confirmed by experiments shall be made in conservative way.

A.2 Primary Containment Analysis Model

A.2.1 An analytical model of the PC shall be developed for the purpose of transient analysis of the primary containment atmosphere following a postulated pipe break. This model shall be based on the equations for mass and energy conservation for the PC system. In developing the model, typically two distinct regions, the vapour and liquid regions should be identified. The vapour region (steam, non-condensable components, and water droplets) may be considered to be homogeneously mixed and in thermal equilibrium with each other.

A.2.2 The thermodynamic state conditions for the steam component shall be described using real gas equation or industrially accepted steam table. The thermodynamic state conditions for the non-condensable components shall be described using the ideal gas equations of state. The liquid region consists of water on the floor, either condensed steam or suppression pool water. Industrially accepted steam tables shall be used to define the thermodynamic state conditions for the water region.

A.3 Specific Modelling Aspects

Mass and energy discharge rates from the postulated breaks are evaluated by system thermal-hydraulic blowdown computer codes and form input for the containment analysis code.

A.3.1 Initial Conditions

A.3.1.1 Initial conditions shall be chosen from range of normal operating conditions, to yield a conservatively high PC pressure and temperature. Typically this requires an upper bound estimate of initial primary containment pressure, a lower bound estimate of initial PC relative humidity and net free volume. The assessment of net free volume should take into account the volume occupied by major structures and components and fluids (e.g. suppression pool water volume etc.).

A.3.2 Maximising reactor coolant system water inventory and stored thermal energy is conservative for reactor coolant system mass and energy release calculations. Stored energy in all reactor coolant system pressure boundary and internal metals thermally in contact with the reactor coolant system water shall be included. Core stored energy, fission energy and fission product decay energy shall be included. All uncertainties shall be biased in the direction which leads to the maximum PC pressure.

A.3.3 Vent Models

A.3.3.1 In case of vapour suppression pool type of containment, the volume V_1 to volume V_2 vents shall be included in the analysis of the primary containment system to determine the pressure and temperature transients. The two phases in the vent model are vent clearing phase and steam-air flow phase.

A.3.3.2 In the vent clearing phase, the boundary conditions are the pressures at the liquid interface in the vents and in the wet well. Pressure losses in the vents due to flow in the vent system shall be accounted for. Inertia of water slug shall be considered in the modelling of vent clearing. For suppression pool with horizontal vent, variation in area of opening available for steam-air flow shall be modelled.

A.3.3.3 The steam-air flow phase follows the vent clearing phase immediately. Steam condensation efficiency and non-condensable cooling efficiency shall be assumed conservatively. One dimensional incompressible and homogeneous flow model may be used for vent system. Details of the methodology to model vent clearing and steam-air flow are given in AERB safety guide on ‘Vapour Suppression System (Pool Type) for Pressurised Heavy Water Reactor’ (AERB/SG/D-22).

A.3.4 Flow through Inter-compartment Openings

Orifice or momentum flow model may be used to calculate flow through inter-compartment openings. Mixture of air-steam may be assumed to be homogeneously mixed. Average density model is used to solve the momentum equation.

A.3.5 V_1 to V_2 Bypass Leakage

A bypass leakage analysis shall be performed to determine the maximum allowable leakage area for direct transfer of steam from V_1 to V_2 air space without passing through the vents and into the pool. The results of this analysis serve to specify maximum allowable leakage area to prevent over pressurisation. For the purpose of design calculation a drywell (volume-V_1) bypass area of around 0.1 m^2 may be used.

A.3.6 Structural Heat Transfer

A.3.6.1 In taking credit for the energy removal capabilities of the PC's structural heat sinks, a lower bound estimate of surface area of structural heat sinks shall be used in the analysis. All modes of heat transfer shall be considered and those modes that are significant shall be modelled. Acceptable film type condensation heat transfer coefficient correlations may be used in dry well.

The following are acceptable correlations for calculating heat transfer coefficient for modelling of heat transfer due to condensation of steam in volume V_1.

For LOCA:

(i) Diffusion based heat transfer coefficient,

(ii) Tagami correlation of condensing heat transfer coefficient, and

(iii) Other correlations of condensing heat transfer coefficient in the presence of non-condensables.

For MSLB:

(i) Diffusion based heat transfer coefficient,

(ii) Uchida correlation of condensing heat transfer coefficient, and

(iii) Other correlations of condensing heat transfer coefficient in the presence of non-condensables.

For volume V_2 appropriate natural convection correlation can be used.

A.3.6.2 The thermal properties used to describe the heat sink materials shall be chosen to represent the heat absorption properties of the material in its anticipated constructed state and provide a conservatively low estimate of the storage and transmission capabilities of the heat sink.

A.3.6.3 Where two distinct materials interface, consideration shall be given to thermal contact resistance and its effect on the heat transfer capabilities of that heat sink. Within a heat sink, the temperature profile shall be determined by an appropriate solution of the transient heat conduction equation.

A.3.6.4 Vapour region steam which has condensed on the surface of structural heat sinks shall be assumed to go directly to the liquid region at a specific enthalpy corresponding to condensate temperature. However, latent heat of condensation shall be transferred directly to the structure.

A.3.7 Containment Energy Removal Systems

A.3.7.1 Credit may be taken for removal of energy from PC system by means of containment energy removal systems, e.g. PC air coolers, vapours suppression pool. Consideration shall be given to the mechanism and efficiency of energy removal for each component and the effects incorporated in the modelling of this energy removal. Coolant temperature shall be assumed to be at their highest credible temperature throughout the accident. In addition, modelling shall incorporate the effects of fouling, condensate build up or any other conditions which may degrade the operating capabilities of the RB coolers during the accident.

A.4 Analysis for Differential Pressure on Internal Structures

A.4.1 General

The design of compartments and internals inside the containment vessel shall take into consideration so that the differential pressure developing in the course of the pressure equalisation process during accidents shall not damage the containment wall and its safety related internals. The stability of portions and/or internals shall be assured considering the differential pressure that may occur. The determination of differential pressure shall proceed as follows from the nominal operating conditions.

A.4.2 Nodalisation

When using multi-compartment code, a sufficiently fine compartmentalisation shall be chosen.

A.4.3 Blow down Discharge

The possible maximum release rate at the beginning of the blow down process shall be used for the release of the energy and mass contents.

A.4.4 Break Configuration

The most unfavourable break configuration shall be used for each compartment.

A.4.5 Heat Absorption by Structure

Heat transfer to the structures may be considered in a conservative manner.

A.5 Assessment of Temperature Profile Across Walls

A.5.1 Inside structures of containment are exposed to high temperature steam environment during accident conditions. It is also possible that both the surfaces of the structure may be subjected to the same environment or different environments. Due to condensation of steam on the structure, the surface temperature of structure would increase which in turn would result in heat transfer within structure. Heat transfer inside the material of structure would depend on surface temperature and the properties of material i.e. thermal conductivity, specific heat and density. Due to the various modes of heat transfer a non-linear temperature gradient would be established during transient.

A.5.2 Concrete structure of containment shall be designed to withstand the thermal stresses expected to be developed due to anticipated temperature gradient across structure during accident conditions.

A.5.3 Following consideration shall apply for the design of structure for temperature gradient across its thickness:

(a) Paint shall be modelled appropriately if the surface of concrete is painted. While using Tagami correlation for condensation heat transfer coefficient for paint, value of constant used in correlation shall be same as that of steel.

(b) Nodalisation of structure shall be fine near the inner surface to model non- linear behaviour of temperature gradient across concrete thickness.

A.6 Design for Vacuum

A.6.1 The containment is required to be designed for negative pressure expected to occur during accident conditions due to late actuation of isolation dampers. In the event of LOCA/MSLB, a significant amount of air-steam mixture would be expelled from containment atmosphere following delay in actuation of isolation damper, which would, in turn result in negative pressure in containment due to subsequent condensation of steam present in containment atmosphere.

A.6.2 For the purpose of peak negative pressure of containment, the delay in initiation of isolation dampers of containment shall be assumed conservatively.

A.6.3 Considerations should also be given to the maximum negative pressure developed during normal operation assuming inlet dampers closed with ventilation exhaust fans continuing to operate.

A.7 Single Failure Criterion

Primary containment response analysis shall incorporate the effects of the most severe single failure of active components concurrent with the pipe break. The single failure may be postulated to occur in the emergency electric power system (diesel generator), containment energy removal system (fan, pump or valve failure) or the core cooling systems (pump or valve failure). The failure chosen shall result in the highest calculated primary containment atmosphere pressure and temperature for that postulated break and shall be consistent with that chosen for the generation of mass and energy release data. In addition, the loss of class-IV electric power to the plant shall be postulated if such an occurrence yields more severe consequences.

APPENDIX-B

METHODOLOGY FOR ASSESSMENT OF RADIOLOGICAL RELEASE FROM CONTAINMENT DURING DBA

B.1 General

Activity leakage from PC activity inventory is modelled as in-leakage into the SC which form the SC activity inventory and direct leakage to environment. Activity gets released from the secondary containment by direct leakage constituting ground release, and through stack via filter system. Release through stack comes from PC controlled discharge also.

B.2 Fission Products Release Analysis from Primary and Secondary Containment

B.2.1 Calculation Methodology

B.2.1.1 This analysis is based on fission product inventory balance in PC and SC respectively. All the engineered safety systems of PC and SC, which can influence the inventory of fission products, should be appropriately modelled in the analysis.

B.2.1.2 While major fraction of activity leaked from PC will go into SC and form the SC inventory, some part may get released directly to atmosphere which should be taken into account as ground level leakage. From SC, the activity gets released by direct leakage constituting ground release (during periods when SC pressure may be above atmosphere), and through stack via SC filtration, purge and recirculation system. Release through stack will also take place from PC, when PC controlled discharge system is operated. PC filtration and pump back system should be modelled to account for the concentration of fission products. Plate out on the structural/containment surfaces inside PC or SC should also be modelled for the purpose of inventory balance. Realistic value of compressed air ingress to PC and SC should be modelled for the purpose of pressure calculation.

B.2.1.3 The calculation of activity leakage from PC should consider:

(i) Time dependent activity inventory in PC atmosphere, following the accident,

(ii) Time dependent PC pressure following the accident,

(iii) PC leakage rate as a function of pressure, and

(iv) Any direct leakage from PC bypassing SC.

B.2.1.4 Similarly, calculation of activity release from SC should consider:

(i) Time dependent activity inventory in SC, which in turn depends on activity leakage from PC and SC walls and release from SC filtration, purge and recirculation system,

(ii) Time dependent SC pressure, which in turn depends on PC leakage and SC purge, and

(iii) SC leakage rate as a function of pressure.

B.3 Assumptions

B.3.1 Release from Fuel and Containment

(i) For the purpose of conservative analysis of current PHWR containment, 100% of core iodine and noble gases are assumed to be released from fuel. However, if justified by an appropriate analysis, a lower value may be used.

(ii) For iodine release in aqueous phase appropriate partition coefficients between air and water phases should be considered for assessing the airborne concentration of iodine. For accident scenarios such as LOCA with ECCS failure, 50% of water trapping in PHT circuit may be considered to calculate net iodine released to containment atmosphere. However, a different value may be used with appropriate justification.

(iii) Although release may take place slowly with time, but for the purpose of conservatism, all the fission products are considered to be released instantaneously to containment atmosphere. A more realistic time dependent release can be accepted with proper justification.

(iv) Even though for DBAs large scale fuel melting and associated aerosol releases are not anticipated, appropriate particulate filters with adequate efficiency should be considered in the assessment.

B.3.2 Factors for Reduction of Iodine Inventory during Transport in the Containment

(i) Plate-out on containment surfaces

Appropriate plate-out half-life for deposition of iodine on PC and SC surfaces are considered taking into account the environmental conditions i.e. moisture content. Currently a plate-out half-life of 1.5 h and 2 h are considered for PC and SC. For conservatism, it is assumed that once the airborne concentration has reached 10% of the original value, further plate-out should not be effective. This assumption is based on the consideration for organic iodine which does not participate in plate-out on containment surfaces.

(ii) Iodine plate-out through leak paths

From all the leak paths of PC, a plate-out factor of 10 is considered. Assumption of this value is conservative with respect to experimental results.

B.3.3 Reduction of Iodine through ESF

Appropriate consideration should be given for removal of iodine through the ESF namely PCCD, PCFPB and SCFRP, taking into account the filter efficiency and delay in the operation of these systems.

B.3.4 Containment Leakage Rate

Analyses should be performed based on the leakage rates as stipulated in technical specification for PC and SC respectively.

ANNEXURE-I

BRIEF DESCRIPTION OF INDIAN PHWR CONTAINMENT SYSTEMS

I.1 General

The containment systems have undergone progressive improvements. The design of the standardised 220 MWe and 540 MWe reactors has seen several notable improvements in the containment and related ESFs for minimising the release of radioactivity to the environment under accident conditions.

I.2 Containment Systems

I.2.1 Double Containment

The single most important factor for reducing radioactivity releases to environment following postulated accidents is the use of double containment envelope instead of a single envelope.

The primary containment of current Indian PHWR consists of a pre-stressed concrete perimeter wall topped by a pre-stressed concrete dome. The containment building is made of concrete with epoxy/vinyl coating for leak tightness and has a passive suppression system. The concept of double containment is extended to cover the entire reactor building (except the base slab). The outer secondary containment envelope consists of a reinforced concrete cylindrical wall topped by a reinforced concrete dome. The annular gap between them is maintained at a negative pressure by purging the interspace. The purge exhaust is discharged via filters through elevated ventilation stack. Thus ground level release of radioactivity during accident conditions are reduced very significantly, the effectiveness being determined mainly by the extent to which the PC leaks are intercepted by SC. However, a small fraction of leakage from primary containment, through some inline leak paths, may bypass the secondary containment. For purpose of analysis, upto 10% of the specified leakage rate from primary containment has been considered to bypass the secondary containment.

The primary containment is further sub-divided into two accident based volumes called V_1 and V_2. These two volumes are interconnected through vapour suppression by means of vent systems consisting of vent shafts, distribution headers and in some cases by down-comer pipes. The vapour suppression pool, which passively absorbs energy during LOCA, also helps in scrubbing of a part of fission products released if any, in the event of accident leading to core damage. The availability of suppression pool water provides a large heat sink to reactor building for long-term decay heat removal.

I.2.2 Containment Isolation System

The concept of double containment is also extended to penetrations and piping open to the containment atmosphere such as airlocks, ventilation ducts and other air handling systems.

Automatic isolation of the containment is initiated in the event of pressure rise or activity buildup in the containment or emergency core cooling system actuation.

Instrumentation for containment isolation signals is supplied with Class II electric supply. However, the containment isolation dampers are pneumatically operated and of fail-safe design (close on air failure or control system power failure). The instrumentation for isolation signal is triplicated and operates on fail safe two-out-of-three logic. Airlock seals are supplied with Class-III powered compressed air supply backed up with local air cylinders.

I.2.3 Containment Engineered Safety Systems for Removal of Fission Products

I.2.3.1 Primary Containment Controlled Discharge (PCCD)

In order to minimise the integrated out-leakage of radioactivity from within the containment to the atmosphere, ESFs such as reactor building coolers, primary containment controlled discharge (PCCD) systems are provided to depressurise the containment. Some of the RB coolers are normally operating during reactor operation and the remaining ones come on sensing increase in RB pressure. For depressurisation of the primary containment below 3.43 kPa (g) (0.035 kg/cm^2 (g)), which may be difficult to be achieved by cooling alone, there is a provision of resorting to controlled discharge to stack via filters. Because of the double containment barrier, it is possible to allow the primary containment to have a small overpressure over an extended period of time without significantly adding to ground level release. Thus, the option of using controlled discharge could be used, if required, after a delay of 48 hours or more by which time the primary containment clean-up system would have reduced the iodine activity in the primary containment atmosphere. So, discharge is significantly reduced. To facilitate delay in operation of controlled discharge via stack, the PCCD system has been designed to operate upto a containment overpressure of 0.4 kg/cm^2 (g) for standared PHWR 220 MWe and 0.48 kg/cm^2 (g) for PHWR 540 MWe. This design feature also allows for delayed PCCD operation in case of persistent compressed air in-leakage into the containment.

In addition, the provision of instrument air supply to selected valves in RB during containment box-up permits main instrument air to RB to be cut off, thereby reducing instrument air in-leakage to RB, consequently, further reducing the need for operating the PCCD.

Initiation of controlled gas discharge is a manual action. It is proposed that this discharge will be initiated after a nominal delay of at least 48 hours following the accident if required. This delay will be acceptable because of the presence of secondary containment. Advantage of this delay is that the release of activity to the stack will get reduced.

The decision on starting the PCCD is expected to be taken by the team handling the site emergency based on the following guidelines:

(i) No PCCD during first 48 hours after initiation of accident.

(ii) Beyond the above period, the decision for opening the PCCD line to be established based on:

   (a) Primary containment pressure and its trend. If it is around 4.9 kPa (g)(0.05 kg/cm^2 (g)) or higher and has increasing trend then operation of PCCD may be justified.

   (b) Meteorological condition: Stable condition (Pasquill category E, F) are preferable. PCCD should not be started if weather conditions are characterised by Pasquill category A, B, etc. or during temperature inversion condition. Under no condition, RB pressure should be allowed to increase beyond the maximum allowed pressure during the re-pressurisation phase if any due to in-leakage of compressed air.

   (c) Activity levels in primary containment as determined from samples.

PCCD is the controlled release system and would be operated at the time of favourable weather conditions and hence corresponding dose conversion factors for long-term may be used.

The use of PCCD following a postulated LOCA could lead to accumulation of fission products in the charcoal filter in the duct, with the potential for unacceptable heat-up of the filter. To limit the overheating of filters, provision exists for passing of air drawn from stack plenum through the filters when the temperature of filters exceeds 80^{0}C.

I.2.3.2 Primary Containment Clean-up System (PCCS)

The environmental releases of fission products can also be minimised by reducing the concentration of iodine fission products by means of primary containment clean-up system. This system would be operated normally four hours into the accident in order to take advantage of plate-out and also decay of short lived radioisotopes of iodine so that the thermal loading of the charcoal filter remains limited to reasonable value during its operation. This is to avoid excessive heating of charcoal filter. As operation of this system will aid mixing of volume V_1 and volume V_2 atmosphere, any hydrogen present in volume V_1 will also get diluted.

In this system air flow is re-circulated within the primary containment through charcoal filters. The system is designed to perform containment atmosphere cleanup operation on a long term basis after an accident and is intended to start manually with a nominal delay of seven hours following the accident. This system would give substantially lower iodine removal half-life.

Accumulation of fission products in the filter during operation of the system in the accident conditions will add heat source. As long as flow is maintained through the filter, the heat is effectively removed.

The temperature of the charcoal filter bed is continuously monitored by RTD type temperature sensor. The temperature exceeding the set value of 80^{O}C is annunciated in the control room computer. For all dampers fail-safe mode is open position. In case of failure of both the fans in this circuit, flow will be maintained by natural circulation of air. The maximum expected temperature in the charcoal filter bed, for the worst scenario is analysed to be within the ignition temperature of charcoal.

I.2.3.3 Secondary Containment Filtration, Re-circulation and Purge System (SCFRP)

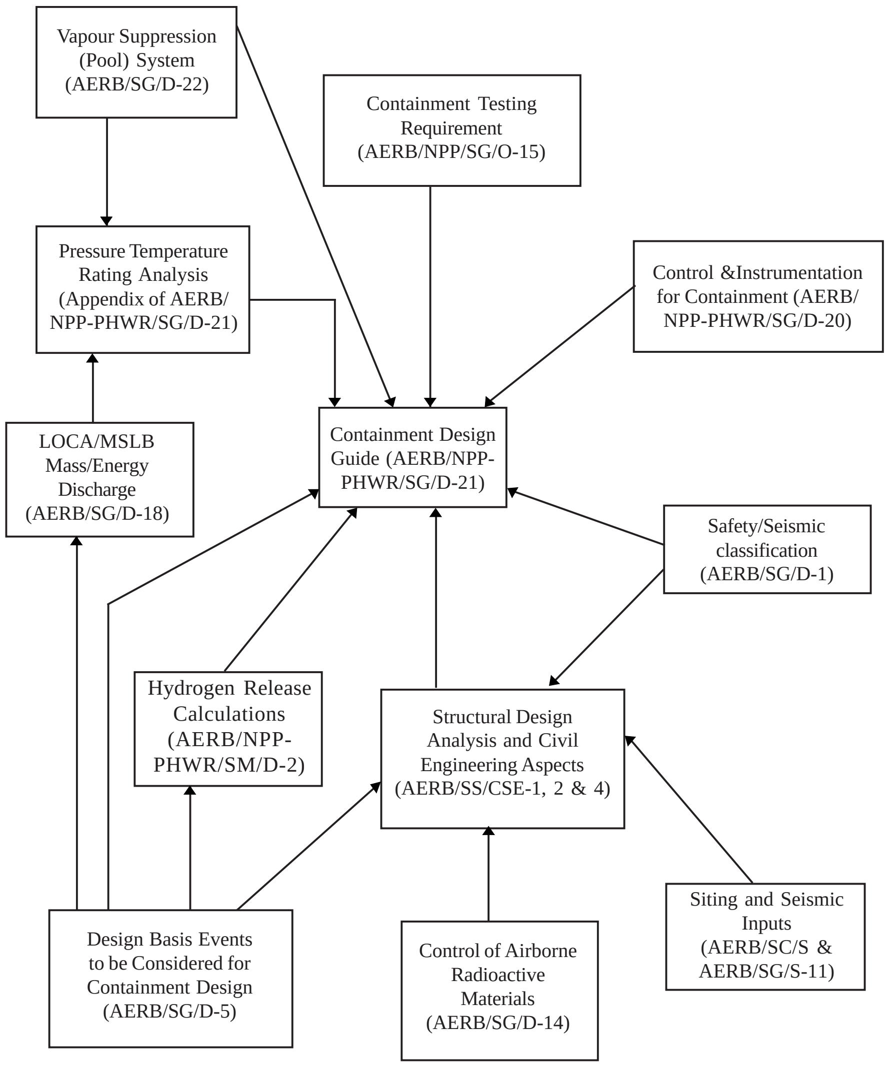
This system provides multipass filtration and mixing by re-circulation within the secondary containment space, and also maintains negative pressure within the secondary containment space. To achieve these functions, about 1700 m^3/h (1000 $ft^3/minute$) of air from the secondary containment space is passed through iodine filters by a fan, after which a large portion of the flow re-circulates back to the secondary containment space, while the remaining small fraction is purged to stack via additional filters to maintain a negative pressure (between 12 to 24 mm WC) in the secondary containment space. Iodine filters and fans have 100% standby.

The normal ventilation in secondary containment maintains a slightly negative pressure in this space. Following an accident, as the secondary containment re-circulation and purge takes over, the negative pressure in the secondary containment will be maintained and, therefore, the net ground level release of activity will be limited to the factor of PC leakage bypassing the secondary containment (i.e. not intercepted by secondary containment). This system starts on auto on RB high pressure signal.

The aforesaid features, together with the provision of double containment concept, with arrangement for filtration, re-circulation and purge in the secondary containment space, are intended to limit radioactivity release to environment to levels well below the reference dose values.

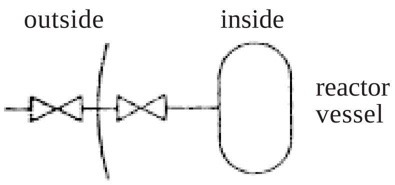
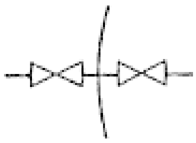
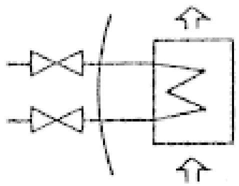
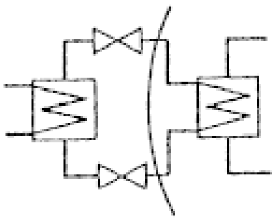
ANNEXURE-II

INTER-RELATIONSHIP BETWEEN VARIOUS AERB REGULATORY DOCUMENTS RELATED TO CONTAINMENT SYSTEMS DESIGN



ANNEXURE-III

ILLUSTRATION OF CATEGORIES OF ISOLATION FEATURES

Reference to subsection 4.3.2	Schematic configuration	Example
(a)(i)		- PHT purification system
(a)(ii)		- Ventilation duct
(b)		- Ventilation cooling inside containment
(c)		- Intermediate cooling (Process water)

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		January 31, 1997	February 14, 1997
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		April 17, 1997	April 25, 1997
		July 15, 1997	September 12, 1997
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SG/D-26	Standard Format & Content of Safety Analysis Report for Nuclear Power Plants
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AERB/NPP-PHWR/ TD/D-1	Decay Heat Load Calculations in Pressurised Heavy Water Reactor Based Nuclear Power Plants

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