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PRIMARY HEAT TRANSPORT SYSTEM FOR PRESSURISED HEAVY WATER REACTORS

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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 to ensure 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, codes of practice and related guides and manuals for the purpose. These documents cover aspects such as siting, design, construction, operation, quality assurance, decommissioning and regulation of nuclear and radiation facilities.

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

The Code of Practice on 'Design for Safety in Pressurised Heavy Water Reactor Based Nuclear Power Plants' (AERB/SC/D, 1989) lays down the minimum requirements for ensuring adequate safety in plant design. This safety guide is one of a series of guides, which have been issued or are under preparation, to describe and elaborate the specific parts of the code. It is based on the current designs of 220 MWe and 500 MWe Pressurised Heavy Water Reactors. It prescribes guidelines for designing the primary heat transport systems of Pressurised Heavy Water Reactors (PHWRs).

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

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

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

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

Suhas P. Sukhatme
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DEFINITIONS

Accident conditions

Substantial deviations[1] 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.

Active Component

A component whose functioning depends on an external input, such as actuation, mechanical movement, or supply of power, and which therefore influences system process in an active manner. e.g., pumps, valves, fans, relays and transistors. It is emphasized that this definition is necessarily general in nature as in the corresponding definition of passive component. Certain components, such as rupture discs, check valves, injectors and some solid state electronic devices, have characteristics which require special consideration before designation as Active or Passive Component.

Anticipated Operational Occurrences[2]

An operational process deviating from normal operation which is expected to occur during the operating lifetime of a facility which in view of appropriate design provisions, does not cause any significant damage to Items Important to Safety nor lead to Accident Conditions.

Common Cause Failure

The failure of a number of devices or components to perform their functions, as a result of a single specific event or cause.

Decommissioning

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

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1. A substantial deviation may be a major fuel failure, a loss of coolant accident(LOCA) etc. Examples of engineered safety features are: an emergency core cooling system(ECCS) and containment.

2. Examples of anticipated operational occurrences are loss of normal electric power and faults such as turbine trip, malfunction of individual items, of a normally running plant, failure to function of individual items of control equipment, loss of power to main coolant pump.

Decontamination

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

Design

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

Design Basis Accidents (DBAs)

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

Diversity[3]

The presence of two or more different components or systems to perform an identified function, where the different components or systems have different attributes so as to reduce the possibility of common cause failure.

Independence

Independence of equipment, channel or a system is its ability to perform its function irrespective of the normal or abnormal functioning of any other equipment, channel or system. Independence is achieved by functional isolation and physical separation.

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 conditions, prevention of accidents or mitigation of accident consequences, resulting in protection of site personnel, the public and the environment from undue radiation hazards.

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3. Examples of such attributes are: different operating conditions of uses, different size of equipment, different manufacturers, different working principles and types of equipment that use different physical methods.

Operational States

The states defined under 'Normal Operation' and 'Anticipated Operational Occurrences'.

Passive Component

A component, which has no moving part, and, for example, 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 some solid-state electronic devices have characteristics which require special consideration before designation as an active or passive component.)

Physical Separation

A means of ensuring independence of an equipment through separation by geometry (distance, orientation etc.), appropriate barriers or combination of both.

Protection System

That part of Safety Critical System which encompasses all those electrical, mechanical devices and circuitry, from and including the sensors up to the input terminals of the safety actuation system and the safety support features, involved in generating the signals associated with the safety tasks.

Quality Assurance

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

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 any other.

Reliability

The probability that a structure, system, component or facility will perform its intended (specified) function satisfactorily for a specified period under specified conditions.

Residual Heat

The sum of the time dependent heat loads originating from radioactive decay and shutdown fission and heat stored in reactor-related structures and heat transport media in a nuclear reactor facility.

Safety

See 'Nuclear Safety'.

Safety Critical System (Safety System)

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

Safety Function

A specific purpose, that must be accomplished for safety.

Single Failure

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

Site

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

Ultimate Heat Sink

The atmosphere or a body of water or the ground water to which part or all of the residual heat is transferred during normal operation, anticipated operational occurrences or accident conditions.

CONTENTS

FOREWORD i

DEFINITIONS iii

1. INTRODUCTION 1
   1.1 Objective 1
   1.2 Scope 1

2. PRIMARY HEAT TRANSPORT AND ASSOCIATED SYSTEMS ... 2
   2.1 Primary Heat Transport System 2
   2.2 Associated Systems 2
   2.3 Primary Heat Transport and Associated System Components 2
   2.4 Support Systems 3

3. SAFETY PRINCIPLES 5

4. GENERAL REQUIREMENTS 7
   4.1 Material Selection 7
   4.2 Environmental Conditions and Qualifications 7
   4.3 Primary Coolant Activity 7
   4.4 Layout 8
   4.5 In-service Inspection 9
   4.6 Testing and Maintenance 9
   4.7 Seismic Considerations 9
   4.8 Static and Dynamic Loads 9
   4.9 Over-pressure Protection 10
   4.10 Pressure Boundary Leakages 10
   4.11 Instrumentation & Control and Electrical Requirements 11
   4.12 Decommissioning Aspects 11

5. DESIGN BASES FOR PHTAS 13
   5.1 Primary Heat Transport System 13
   5.2 Shutdown Cooling System 15
   5.3 Feed and Bleed System 16

	5.4	Pressure Relief System	17
	5.5	Emergency Core Cooling System	17
	5.6	Purification System	20
	5.7	Fuelling Machine Supply System	20
	5.8	Auxiliary Feed Water System for Steam Generators	20
6.	QUALITY ASSURANCE CONSIDERATIONS		21

FIGURE-1	:	PRIMARY HEAT TRANSPORT SYSTEM: TYPICAL PHWR SCHEMATIC	22
FIGURE-2	:	TYPICAL COOLANT CHANNEL (500 MWe PHWR)	23
APPENDIX-I	:	LIST OF DESIGN BASIS EVENTS FOR PHTAS	24
ANNEXURE-1	:	LEAK-BEFORE-BREAK (LBB) CONCEPT TO DESIGN THE PRIMARY HEAT TRANSPORT PIPING OF PHWR	32
ANNEXURE-2	:	CLASSIFICATION OF PHT SYSTEM COMPONENTS	35
ANNEXURE-3	:	MAJOR RADIOACTIVE ISOTOPES REMOVED BY PURIFICATION SYSTEM AND TYPICAL LIMITS FOR PRIMARY COOLANT ACTIVITY DURING NORMAL OPERATION	41
ANNEXURE-4	:	TYPICAL CHEMISTRY CONTROL PARAMETERS FOR PRIMARY COOLANT ACTIVITY DURING NORMAL OPERATION	42
ANNEXURE-5	:	TYPICAL VELOCITIES IN PHT SYSTEM COMPONENTS	43
ANNEXURE-6	:	LOADING COMBINATIONS FOR PHTAS COMPONENTS	44
ANNEXURE-7	:	MECHANISM FOR DEMONSTRATING COMPLIANCE WITH ECCS ACCEPTANCE CRITERIA	45

ANNEXURE-8 : TYPICAL PRIMARY HEAT TRANSPORT SYSTEM DESCRIPTION AND DESIGN EVOLUTION 46

REFERENCES 66

BIBLIOGRAPHY 67

LIST OF PARTICIPANTS 68

WORKING GROUP 68

ADVISORY COMMITTEE ON CODES, GUIDES AND ASSSOCIATED MANUALS FOR SAFETY IN DESIGN OF NUCLEAR POWER PLANTS (ACCGD) 69

ADVISORY COMMITTEE ON NUCLEAR SAFETY (ACNS) 70

PROVISIONAL LIST OF AERB SAFETY CODE, GUIDES AND MANUAL ON DESIGN OF PRESSURISED HEAVY WATER REACTORS 71

1. INTRODUCTION

1.1 Objective

The objective of this safety guide is to provide bases for the design of the primary heat transport and associated systems in Pressurised Heavy Water Reactors (PHWRs) so as to achieve nuclear safety, hereinafter referred to as 'safety' through appropriate design provisions and engineered safety features, as elaborated in section 0500 of AERB Code No. SC/D, Code of Practice on 'Design for Safety in Pressurised Heavy Water Based Nuclear Power Plants', 1989, hereinafter referred to as the 'code'.

1.2 Scope

This guide outlines the safety requirements for the design of Primary Heat Transport and Associated Systems (PHTAS). The extent of the PHTAS is described in section 2.

Apart from specifying design bases, it also includes general design guidance related to static and dynamic loads, material selection, layout considerations, coolant activity, decontamination and decommissioning. This guide includes a list of Design Basis Events (DBEs) relevant to PHTAS as Appendix-I (Ref. AERB safety guide on 'Design Basis Events for Pressurised Heavy Water Reactor', AERB/SG/D-5). Brief description of PHT system and design evolution is included as Annexure-8 to give background information on PHT system.

2. PRIMARY HEAT TRANSPORT AND ASSOCIATED SYSTEMS

The PHTAS comprises of the primary heat transport system as defined in section 2.1 and the associated systems as defined in section 2.2. Typical configuration of the PHTAS is illustrated in Fig.1.

2.1 Primary Heat Transport System

The primary heat transport system (PHTS) transports heat generated in the reactor core to the secondary system or to the ultimate heat sink. This includes the primary side of the steam generators, the primary coolant pumps, the coolant channels, end fittings, pressuriser, piping and piping components, etc. The pressure-retaining boundary of the PHTS extends up to and includes the first passive barrier or first active isolation device (when the circuit is traced from the core).

2.2 Associated Systems

The associated systems include the systems directly associated with PHTS and which perform the following functions:

- shutdown cooling
- feed and bleed (pressure control and storage)
- relief (over-pressure protection)
- emergency core cooling
- small leak handling
- purification
- fuelling Machine (FM) coolant supply and return
- auxiliary feed water for steam generator [1]
- leakage collection
- other associated systems such as delayed neutron monitoring, degassing, sampling,etc

2.3 Primary Heat Transport and Associated System Components

Following is the list of major PHTAS components:

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1 Included as a special case in view of the safety function performed and not covered in any other guide.

- coolant channel including end-fittings, seal plugs and fuelling machines when connected to a channel
- primary coolant pump and its gland supply (seal injection) system
- primary side of steam generator
- primary (D_2O) side of PHTAS heat exchangers
- bleed condenser
- pressuriser, if incorporated
- valves (including relief)
- reactor headers, feeders and piping
- primary system pressurising pumps
- components of emergency core cooling system like accumulators, pumps, valves, heat exchangers and piping components
- shutdown cooling system pumps, heat exchangers and piping components
- other associated systems such as delayed neutron monitoring, degassing, sampling,etc.

2.4 Support Systems

Support systems required to ensure intended functioning of PHTAS have been covered in separate safety guides. These include AERB safety guides on 'Safety Critical System' (AERB/SG/D-10), 'Emergency Electric Power Supply Systems for Pressurised Heavy Water Reactors' (AERB/SG/D-11), 'Ultimate Heat Sink and Associated Systems in Pressurised Heavy Water Reactors' (AERB/SG/D-15) and 'Safety Related Instrumentation and Control for Pressurised Heavy Water Reactor Based Nuclear Power Plants'. (AERB/SG/D-20).

Interface requirements, where needed, shall be specified as given below:

Requirements to ensure compatibility:

- flow rates
- loadings (including loading due to pressure and temperature, etc.)
- response times
- heat transfer capabilities

Requirements related to physical interface:

- interface device shall have same safety class as the higher safety class of the system to which it is connected

Examples for interface devices are:

- passive barriers (heat exchanger tubes)
- remotely operated valves
- manually operated valves
- supporting structures (anchors)

Requirements related to reliability:

- Diversity/redundant trains of the support systems, wherever needed, shall be specified, e.g., power supplies

3. SAFETY PRINCIPLES

3.1 The PHTAS have the primary safety objective of ensuring that the fuel in the core is cooled by circulation of an adequate quantity of coolant of appropriate quality under all operational states and during and following accident conditions so that specified limits on fuel defined in AERB safety guide on ‘Fuel Design for Pressurised Heavy Water Reactors’ (AERB/SG/D-6) are not exceeded. In addition, PHTAS pressure boundary shall function as a barrier to limit the escape of radioactive materials that may be present in the reactor coolant. PHTAS should also prevent deviation of PHTS parameters from set values during normal operating conditions. Design Basis Events related to PHTAS, referred in AERB safety guide on ‘Design Basis Events for Pressurised Heavy Water Reactors’ (AERB/SG/D-5) shall be analysed in detail and provisions made to counteract them as required.

3.2 Features such as redundancy, diversity, physical separation etc. should be taken into consideration to increase reliability, considering single failure criterion human factor engineering, operation and maintenance (O&M) aspects and aging. PHTAS components should be designed using good engineering practices and proven design principles and methodologies. Specific applications in PHTS are detailed in section 5.

3.3 The PHTAS design shall also take into account the requirements to perform appropriate safety functions listed in AERB safety guide on ‘Safety Classification and Seismic Categorisation for Structures, Systems and Components of Pressurised Heavy Water Reactors’ (AERB/SG/D-1). Structures, systems and components of PHTAS should be assigned appropriate safety classes depending upon the safety function they perform (refer to Annexure-2).

3.4 Systems or portions of systems of different safety classes shall be connected through appropriate interface devices. Each interface device shall have the same safety class as the higher safety class system to which it is connected.

Typical interface devices are:

(i) *Passive barriers,* such as heat exchanger tubes.

(ii) *Remotely operated valves:* Closure time of valves that are normally open and assumed to be safety class boundaries must be such that the safety function of the higher safety class components is maintained.

(iii) *Manually operated valves:* There shall be specific administrative procedures to ensure correct operation of manually operated valves, and it shall be demonstrated that there are means of detecting if the valve has been inadvertently left in the wrong state and that there is time to restore the situation before unacceptable consequences occur.

(iv) *Active flow restriction devices:* An example of such an interface device is the excess flow check-valve in Fuel Handling system which restricts the flow from the PHTS to a value to ensure that the PHTS discharge is within the normal primary coolant make-up capability of pressurising pumps.

(v) *Passive flow restriction devices:* Examples of such interfaces are the restriction devices in Fuel Handling system, which restrict the flow from the PHTS (when fuelling machines are connected) to a value to ensure that the PHTS discharge is within the normal primary coolant make-up capability of pressurising pumps.

4. GENERAL REQUIREMENTS

The following sections identify the general requirements of the PHTAS. System-specific detailed design bases are given in section 5.

4.1 Material Selection

Materials for the PHTAS should be selected based on the details given in AERB safety guide on 'Materials Selection and Properties' (AERB/SG/D-16), which will cover requirements such as fatigue, creep, coolant channel material, stress corrosion cracking, embrittlement ,etc.

4.2 Environmental Conditions and Qualifications

Power operated valves and Instrumentation & Control (I&C) components of PHTAS and emergency core cooling system (ECCS) shall be designed to withstand the environmental conditions resulting from operational states and/or accident conditions as given below:

- temperature
- humidity
- pressure
- radiation fields

Safety systems components shall be qualified to demonstrate that they are capable of meeting the design basis performance taking the above into account.

In the qualification programme, consideration shall be given to the combined effect of various environmental factors and the integrated effect of normal ambient conditions over the required life of the equipment.

4.3 Primary Coolant Activity

Measures such as use of cobalt-free material shall be taken to minimise the primary coolant activity in order to keep the radiological consequences resulting from operational states and accident conditions, below the limit set by the Regulatory Body and As Low As Reasonably Achievable (ALARA). Delayed neutron monitoring system shall be provided to detect fuel failure. Provision shall be made for purification of the reactor coolant during normal operation to ensure specified chemistry and activity control. Crevices and the local configuration where radioactive sludge and debris can accumulate should be avoided. Provision for system decontamination should be made, to carry it out whenever necessary.

4.4 Layout

The layout and equipment accessibility of the PHTAS shall be such as to facilitate ease of inspection, maintenance and repair and to keep the radiological exposure of site personnel as low as reasonably achievable.

(a) This can be achieved by:

- remote controls
- proper shielding
- built-in provisions for equipment handling
- minimising piping that carries radioactive fluids in areas where personnel have frequent access or extended stay
- separation of radioactive components from non-radioactive components

(b) The layout of PHTAS shall take into account protection against internal missile and fire, by suitable means such as:

- physical separation of redundant safety channels
- protection against fire and internal missiles (refer to AERB safety guides on 'Protection Against Internally Generated Missiles and Associated Environmental Conditions', AERB/SG/D-3 and on 'Fire Protection in Pressurised Heavy Water Based Nuclear Power Plants', (AERB/SG/D-4).

The piping and equipment layout should aim for minimising the number of snubbers required to accommodate the dynamic loads, so as to improve reliability and reduce manrem consumption. The layout should aim at reducing the stresses and minimising cumulative fatigue usage factors in the system so as to reduce the number of locations where ISI may be required.

4.4.1 Protection against Pipe Failure

Design measures for the design, manufacture and installation of PHTAS shall ensure high integrity and minimise the likelihood of pipe failure and structural collapse or of loads due to falling components. Where pipe failures cannot be totally excluded, even after consideration of leak before break criteria, protection against pipe failure induced consequences shall be provided by PHTAS layout and use of protective barriers so that

(i) a break in pipe shall not cause rupture of another pipe that is necessary to mitigate consequences of the break, i.e., pipe break leading to loss of coolant accident (LOCA) shall not lead to incapacitation of ECCS;

(ii) a pipe break shall not impair containment integrity; and

(iii) a pipe break shall not incapacitate reactor shutdown capability

In order to implement the above, consideration should be given to provide pipe whip snubbers, mechanical stoppers etc.

4.5 In-service Inspection

The PHTAS coolant pressure boundary components shall be designed, manufactured and laid out in such a way that it is possible, throughout the service life of the plant to carry out, at appropriate intervals, adequate inspections and tests of the pressure boundary, as specified in AERB safety guides on 'In-service Inspection of Nuclear Power Plants' (AERB/SG/O-2) and 'Design for In-service Inspection' (AERB/SG/D-17).

4.6 Testing and Maintenance

PHTAS shall be designed to be tested, maintained and inspected for functional capability, including physical integrity during their intended life. Components should be designed to have a long life and easy maintainability so that radiation exposure involved in maintaining or replacing them is kept reasonably low. Design provision for important commissioning tests should be made.

4.7 Seismic Considerations

Components of PHTAS shall be assigned the appropriate seismic categories (refer to AERB safety guide on 'Safety Classification and Seismic Categorisation for Structures, Systems and Components of Pressurised Heavy Water Reactors', AERB/SG/D-1). PHTAS structures, systems and components shall be designed on the basis of seismic ground motion appropriate to the site (refer to AERB safety guide on 'Seismic Studies and Design Basis Ground Motion for Nuclear Power Plant Sites', AERB/SG/S-11).

The design shall ensure that failures of structures, systems and components, designed to seismic category II requirements, do not cause failure of seismic category I structures, systems and components.

4.8 Static and Dynamic Loads

PHTAS component analysis shall consider the design requirements for static, dynamic and thermal hydraulic loadings as per the acceptable national/ international codes and standards.

While considering ASME code section III requirement for class-I components following shall be considered:

(a) under level D condition, either safe shutdown earthquake (SSE) inertia or pipe rupture loads need to be considered one at a time. However, in case of equipment support/piping whose failure could lead to containment impairment or affect the reactor protective systems, both have to be considered at the same time.

(b) under service level C and service level D conditions, SSE seismic anchor movement (SAM) need not be considered in the qualification of the piping system, if OBE SAM is > ½ SSE SAM. However, SSE SAM loads need to be considered for the design of supports of piping and equipment.

For analysis, 10 equivalent maximum stress cycles per OBE shall be considered and 5 such OBEs shall be considered in the design.

Level C fatigue analysis is not called for as per the code.

Loading combinations for PHTAS components are given in Annexure-6.

4.9 Over-pressure Protection

All pressure retaining components of the PHTAS shall be protected against over-pressure (refer to ASME section III, NB-7000). The systems shall be protected against over-pressurisation by:

(i) adequate physical interfaces between the systems

(ii) provision of pressure relief devices

Where an adequate physical interface is not provided, the pressure relief capacity provided on the lower design pressure systems shall be related to the maximum pressure transient that can arise from the operation of the higher pressure system.

4.10 Pressure Boundary Leakages

Leak tightness of the system shall be such that total leakages from (PHTS) are within specified limits. These limits may be based on system make-up capability and activity release. The leakages are either open leaks or contained leaks.

An open leak is a leak whose location has not been precisely identified. By definition, these leaks are not directly collected, though the coolant leaked out is indirectly recovered through the dryer system.

All the contained leaks from the equipment, such as the valve stem, inter gasket leak-off, mechanical seal shall be appropriately guided to the heavy water leakage collection system.

Following considerations should be given for handling pressure boundary leakages:

(a) Detection: Operational features such as provision of floor beetles, pressure sensors, level sensors and activity monitoring including annulus gas monitoring should be provided for early detection of leakages.

(b) Mitigation: Mitigating systems like small leak handling, ECCS, shutdown systems and containment systems should be provided to limit the consequences of the entire spectrum of leaks.

4.11 Instrumentation & Control and Electrical Requirements

The instrumentation and control system shall provide sufficient information for reactor operators to determine the status of the PHTAS. The information provided to the operator shall be adequate to determine if:

- abnormal PHTAS conditions exist;
- PHTAS operating parameters are reaching the specified operational limits;
- a safety function related to the PHTAS needs to be performed;
- the PHTAS are ready (or not) to perform their safety functions, for example, if adequate coolant inventory exists;
- the PHTAS are carrying out safety functions; and
- safety functions have been accomplished.

The instrument lines are important interfaces between the PHTAS and I&C or protection system. Since these components contain primary or secondary coolant, they are part of the PHTAS. Their function is to transmit a parameter/ impulse from the PHTAS to a transducer. Instrumentation lines shall not unacceptably modify the characteristics of the parameter measured. They shall be designed for performance consistent with the safety requirements for the associated parameters under both steady and transient conditions (refer to AERB safety guides on ‘Safety Critical Systems’, AERB/SG/D-10 and ‘Safety Related Instrumentation and Control for Pressurised Heavy Water Reactor Based Nuclear Power Plants’, AERB/SG/D-20).

4.12 Decommissioning Aspects

During the design stage, priority shall be given to the safety design requirements for the planned operating life of the plant. However, the eventual need for decommissioning should be considered in terms of:

(a) the ability to maintain the plant in a safe state indefinitely, and

(b) dismantling feasibility.

Significant aspects to be taken into account are:

- layout for easy access;
- choice of materials to avoid dissemination of radioactive substances (long-lived nuclides) ;
- dismantling of equipment;
- transport of radioactive materials; and
- provision of connections for flushing and decontamination.

These aspects should be considered in relation to post- accident situation as well as the period after the end of planned life (refer to AERB safety manual on ‘Decommissioning of Nuclear Facilities’, AERB/SM/DECOM-1,1998).

5. DESIGN BASES FOR PHTAS

The schematic of PHTAS and coolant channel assembly are shown in Fig.1 and 2 respectively.

5.1 Primary Heat Transport System

In normal operation the PHTS transports the coolant and thereby the heat from the reactor core to the steam generators. The PHTS also forms part of the heat transport route for transferring heat from the reactor core to the ultimate heat sink during shutdown conditions.

The design bases for PHTS are as follows:

- high integrity of PHTS pressure boundary;
- adoption of leak before break (LBB) criteria which ensures slow propagation of any flaw to preclude catastrophic pipe failures for consideration of structural integrity of supports and jet impingement effects;
- adequate coolant circulation for ensuring reliable core cooling, (typical velocities in PHT system components are given in Annexure-5);
- PHT coast-down following class-IV power failure to be adequate for core heat removal;
- adequacy of natural convection to remove residual heat following class-IV power failure.
- component design and layout to facilitate ISI, maintenance and replacement;
- failure on secondary side should not lead to failure in primary side.

To implement effectively the leak before break concept (refer Annexure–1), apart from design considerations (refer to AERB safety guide on 'Materials Selection and Properties', AERB/SG/D-16), diverse means to monitor leakage should be provided (through suitably located detectors for moisture, annulus gas monitoring system, liquid level or flow rate in sump or tanks, detectors for containment atmosphere radioactivity, TV cameras at strategic points) to increase reliability of leak detection.

5.1.1 Design Requirements for Coolant Channel Assemblies and Sealing Plugs

Coolant tubes and end fittings are important components of the coolant channel assemblies which, along with end seal discs, form part of the primary coolant pressure boundary. They also house the fuel assemblies and serve to maintain thermal insulation between the coolant and the relatively cold moderator, to provide interfaces for the coupling to the primary heat transport system at the two ends of the coolant channel and to facilitate the fuelling operations. The coolant channel should have suitable features to facilitate its orientation within the given lattice positions, to accommodate thermal expansion and creep-growth related dimensional changes of the channel and to facilitate replacement of channels. Suitable annular space should be maintained around the coolant channel with the help of spacers at appropriate locations. Accordingly, these components shall meet the following design basis requirements[2] .

5.1.1.1 *Pressure Ttubes:* The material for coolant tubes shall be selected from considerations of neutron economy, lower creep rate in reactor environment, superior corrosion resistance and lower hydrogen pick-up characteristic. They shall be designed to withstand a postulated pressure rise in PHTS resulting from failure of pressure control followed by the opening of over- pressure protection devices. In the absence of availability of such a material listed in the ASME Code, the material chosen on the basis of past experience should meet, to the maximum extent possible, the intent of the code. Tubes made from cold worked Zircalloy-2 and Zirconium-2.5% Niobium alloys meeting such intent are acceptable (refer to AERB safety guide on 'Materials Selection and Properties', AERB/SG/D-16). Depending upon properties of materials, need for hot pressurisation shall be assessed and provision made, if required.

5.1.1.2 *End Fittings:* Design conditions and mechanical loads applicable to end fittings shall be identical to those applicable to the coolant tubes mentioned in 5.1.1.1 above. The material shall have low irradiation damage properties. Its thermal expansion co-efficient shall be compatible with that of the coolant tubes to enable a proper rolled joint. Total estimated degradation of material properties due to cumulative irradiation dose shall not render the end fitting unsuitable for the service.

5.1.1.3 *Seal Plugs:* The seal plugs (SP) close the ends of the coolant channel assemblies and prevent escape of the reactor coolant and provide a means for remote opening of the pressure boundary of the PHT system for refuelling the channel. The SP should be operable by rams of the Fuelling Machine. It should be provided with safety latch to prevent accidental unlatching from the end fitting. The design pressure and temperature conditions shall be identical to those specified for coolant channel assemblies in addition to ram forces imposed during F/M operations. The material shall have adequate

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2 Presently, pressure tube is designed on the basis of pressure temperature profile. Pressure tube thickness is optimised based on neutron economy and pressure consideration.

strength and resistance to stress corrosion cracking[3]. Sealing shall be achieved by a soft metallic gasket layer.

5.1.1.4 *Shield Plugs:* Shielding plugs are required to provide shielding to prevent radiation from axially streaming out of the coolant channel. They should locate the string of fuel bundles in a well-defined location within the coolant channel with a well-defined gap between the upstream shielding plug and the first fuel bundle. The shielding plugs are required to direct the coolant flow from radial to axial direction without causing undue turbulence in the first upstream fuel bundle. In addition, a fuel locator also may be provided in between shielding plug and first fuel bundle, if necessary. A safety latch mechanism to prevent accidental unlatching from the coolant channel is necessary. The mechanisms of the shielding plug shall be compatible with the operation of the fuelling machine. The material of construction should be corrosion resistant and compatible with the coolant.

5.2 Shutdown Cooling System

The residual heat removal (RHR) during operational states is performed by a combination of other systems such as steam and feed water system. The RHR systems under these conditions are required for:

(a) removal of residual heat from the core and the PHTS during the initial phase of cool down and its transfer to ultimate heat sink (UHS) via the steam and feed water system (SFS) (refer to AERB safety guide on ‘Ultimate Heat Sink and Associated Systems in Pressurised Heavy Water Reactors’, AERB/SG/D-15).

(b) removal of residual heat from the core and the PHTS after the initial phase of cool down and its transfer to the UHS directly or via intermediate cooling circuit (e.g., Process Water Systems) using shutdown cooling system (refer to AERB safety guide on ‘Ultimate Heat Sink and Associated Systems in Pressurised Heavy Water Reactors’, AERB/SG/D-15).

The function of the shutdown cooling system is to remove residual heat during shutdown with the reactor coolant pressure boundary (RCPB) intact. Shutdown cooling system should be designed for PHTS full pressure and temperature.

Shutdown cooling system should be designed, as a minimum, to remove residual heat (after PHTS cool down through steam generator up to 150°C) during cool down from 150°C to 55°C and maintaining coolant at this temperature (55°C max.).

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3 Material ASTM-A 564 Gr X M13 (RH 13.8 MoSS) condition H 1050 for higher pressures and A 564 Gr 630 condition H 1075 for lower pressures are acceptable materials for sealing plugs.

The shutdown cooling system should be such as to cater to other functions such as draining/refilling of steam generators, header level control operation (for maintenance of PHTS isolation valves of primary coolant pumps and steam generators, if provided and other isolation valves, steam generators and primary coolant pumps).

The system shall be available in the event of loss of off-site power.

The systems providing RHR function shall be capable of removing the residual heat from the core and the PHTS at such a rate that the applicable specified fuel limits and the design limits of the PHT system pressure boundary are not exceeded. (refer to AERB safety guide on 'Fuel Design for Pressurised Heavy Water Reactors', AERB/SG/D-6)

They shall also be capable of removing residual heat at a faster rate to override emergencies.

For methodology of decay heat load calculations, (refer to AERB safety manual on 'Decay Heat Load Calculations', AERB/SM/D-1).

The design requirements for the systems performing the residual heat removal function include high functional reliability and compliance with single failure criterion to ensure proper RHR function during operational states and accident conditions.

5.3 Feed and Bleed System

Following functions are to be performed by this system:

- maintain sufficient primary coolant inventory for core cooling during and after all operational states.
- make up for handling small leaks (within pressurising pump capacity).
- pressure control in PHTS (with or without pressuriser).
- reactor coolant degassing and pH control to inhibit corrosion of PHTS component and to minimise deposition of crud on the fuel.
- auxiliary services to other PHTAS functions, e.g., cooling mechanical seals of reactor coolant pumps, coolant supply to fuelling machine, hydrotesting of PHTS, bleed condenser cooling, spray in pressuriser for cooling prior to its maintenance.
- boxing up PHTS pressure boundary in the event of failure of air supply to pneumatically actuated valves, excluding IRVs.
- providing purification flow required for clean-up (section 5.4).

Provision of redundancy in pumps, valves and diversity in supply of power from emergency power system is recommended in order to maintain high availability.

5.4 Pressure Relief System

Design shall provide over-pressure protection to PHTS. Relief system should be designed to ensure that the pressure in the weakest link does not exceed the permissible limit.

Following sequence of events which are simultaneous failures shall be taken as the basis for sizing pressure-relieving capacity for PHTS.

(i) turbine trip (sudden closure of turbine stop valves);

(ii) steam discharge valves (atmospheric and to condenser) fail to open (boiler pressure control failure);

(iii) signals giving power set back to reactor fail to regulate (reactor regulation system failure); and

(iv) reactor trip is available.

Instrumented Relief Valves (IRVs) shall meet the design intent of ASME code (NB 7000).

Over-pressure discharge from PHTS is collected in a bleed condenser which has adequate capacity to receive anticipated quantity of relief.

In case IRV opens (on genuine signal or spuriously) and remains stuck in open position, bleed condenser becomes part of PHTS and therefore, the over-pressure protection device on bleed condenser shall meet the requirements of ASME code NB-7000. On rising PHTS pressure the reactor trip precedes IRV opening. In case of spurious IRV opening, the reactor trips on low PHTS pressure or low level in pressuriser.

5.5 Emergency Core Cooling System

5.5.1 The safety function of the emergency core cooling system is to remove heat from the core after failure of primary coolant pressure boundary in order to limit fuel damage. The design shall provide adequate core cooling using intermediate cooling loops (refer to AERB safety guide on ‘Ultimate Heat Sink and Associated Systems in Pressurised Heavy Water Reactors’, AERB/SG/D-15). It shall meet requirements as specified in AERB safety guide on ‘Fuel Design for Pressurised Heavy Water Reactors’ (AERB/SG/D-6) and AERB safety guide on ‘Hydrogen Release and Mitigation Measures under Accident Conditions in Pressurised Heavy Water Reactors’ (AERB/SG/D-19) for the entire spectrum (sizes and locations) of failures considered. This system shall incorporate a high pressure injection and long-term recirculation system to prevent/limit the escape of fission products from the core as specified in AERB safety guide on ‘Loss of Coolant Accident Analysis for Pressurised Heavy Water Reactors’ (AERB/SD/D-18). ECCS shall be designed to cater to various sizes of pipe breaks in PHT system, the maximum being double-ended break of maximum diameter pipe.

5.5.2 ECCS Acceptance Criteria

(a) For all LOCA events, the release of radioactive material from the fuel in the reactor shall be limited such that the estimated radiological consequences at exclusion zone boundary are within acceptable limits for accidents as specified by AERB (refer to AERB safety guide on ‘Radiation Protection in Design’, AERB/SG/D-12).

(b) For LOCA events with break size upto equivalent of a double-ended feeder pipe break anywhere in the primary pressure boundary, there shall be adequate cooling to prevent fuel failures.[4] For single channel events, fuel damage may occur in the affected channel. The failure shall not propagate to other reactor channels.

(c) In all LOCA events, the configuration of fuel channels shall be maintained such that coolability of core by ECCS is ensured.

(d) Adequate long-term cooling capability of the fuel following the LOCA event shall be ensured. To this end, ECCS shall provide a reliable link between heat source and ultimate heat sink on a long-term basis.

(e) The metal water reaction of zircalloy in the core shall be limited such that the resulting hydrogen release anywhere in the containment should generally be maintained outside the bounds of deflagaration (flamability) limit in the ternary diagram.

The ECCS shall also meet the following design requirements to satisfy the ECCS acceptance criteria in 5.5.2:

(i) ECCS shall be available in the event of loss of off-site power.

(ii) Valves provided in inter-connected lines between two PHTS loops shall isolate the healthy loop from the breached loop to reduce inventory loss and core voiding, in case of two-loop PHT system, such as in 500 MWe PHWR.

(iii) The ECCS recirculation pump intakes shall be adequately protected by a suitable strainer. Pump intake design shall prevent any cavitation effects with conservative assumptions.

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4 Fuel failure here denotes clad rupture as a consequence of accident and not defects, viz., pin holes, etc. which may be encountered during normal operation. For guidance in predicting fuel failure (refer to AERB safety guide on ‘Fuel Design for Pressurised Heavy Water Reactors’, AERB/SG/D-6).

(iv) Reliability of ECCS shall be adequate to meet single failure criterion.

(v) ECCS shall be designed with sufficient redundancy to meet reliability targets. Redundant components shall be physically separated to prevent a common cause failure.

(vi) Mission time for ECCS recirculation mode should be on the basis of residual heat (Presently, based on decay heat level estimation, this period is two months).

(vii) Instrumentation shall be provided for monitoring and controlling the performance of ECCS function during and after the accident (AERB safety guide on ‘Safety Related Instrumentation and Control for Pressurised Heavy Water Reactor Based Nuclear Power Plants’, AERB/SG/D-20) provides guidance on design of Instrumentation & Control).

(viii) Periodic testing is required for all ECCS active components, which are required to operate or continue operating after the accident.

(ix) The design shall ensure that the maintenance and testing do not affect the system availability.

(x) Since the complete system operation cannot be tested on power, provision shall be made for conducting tests to ensure sub-system operation for availability checks.

(xi) Provision for fire-water injection to ECCS recirculation heat exchangers should be made (AERB safety guide on ‘Ultimate Heat Sink and Associated Systems in Pressurised Heavy Water Reactors’, AERB/SG/D-15).

(xii) Structural integrity of ECCS system shall be as per seismic class 1. (i.e. design of the pressure boundary piping and equipment and also the integrity of their supporting arrangement shall be carried out for both SSE and OBE). The active components in ECCS shall be designed for functionability during and after the occurrence of SSE event.

5.5.3 Testing and inspection of ECCS

Keeping in view the safety importance of the availability of ECCS and the fact that ECCS normally remains dormant, an elaborate functional test facility shall be provided, to check proper operation of all the instruments and equipment in the system without interfering the normal operation of the PHTS. The functional adequacy of ECCS shall be demonstrated by conducting integrated test during commissioning.

5.6 Purification System.

The primary function of the purification system is to remove dissolved chemical impurities, radioactive substances including fission products and suspended solids from the primary coolant so as to control the coolant chemistry and activity levels within specified limits.

The purification flow should be designed to provide for periodic clean-up of the PHTS piping and equipment during temperature or chemical excursions (crud burst). Clean-up half-life of primary coolant should be less than one hour, so that redeposition of corrosion and fission products on the system surface is avoided, thereby limiting the radioactivity levels in the plant.

Provision should be made for monitoring the water purity. Major isotopes expected are given in Annexure-3, Table-III-1. It should also be possible to keep the primary coolant activity during normal operation within the values given in Annexure-3, Table-III-2. Provision should be made to hook-up a separate system for decontamination, whenever necessary.

5.7 Fuelling Machine Supply System

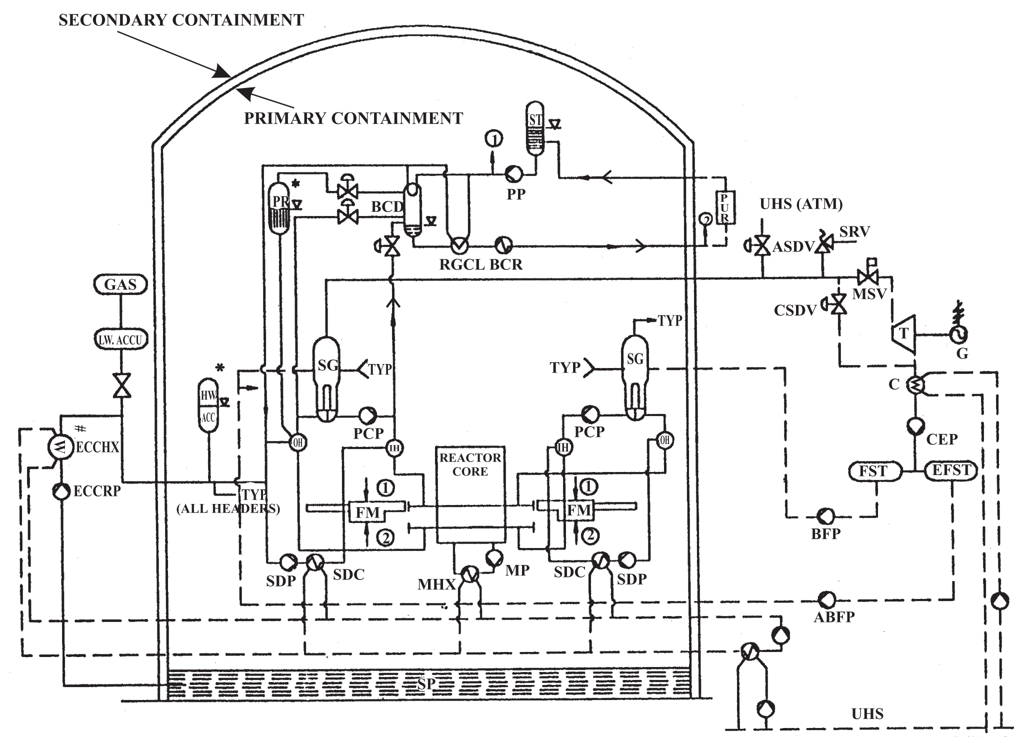
The design of fuelling machine supply system, is detailed in AERB safety guide on 'Design of Fuel Handling and Storage Systems for Pressurised Heavy Water Reactors' (AERB/SG/D-24).

5.8 Auxiliary Feed Water System for Steam Generators

The design of the Auxiliary (Emergency) Feed Water System is detailed in AERB safety guide on 'Ultimate Heat Sink and Associated Systems in Pressurised Heavy Water Reactors' (AERB/SG/D-15).

6. QUALITY ASSURANCE CONSIDERATIONS

6.1 A quality assurance programme shall be developed and approved prior to the commencement of the design activity for PHTAS. (refer to AERB safety guide on 'Quality Assurance in the Design of Nuclear Power Plants', AERB/SG/QA-1).

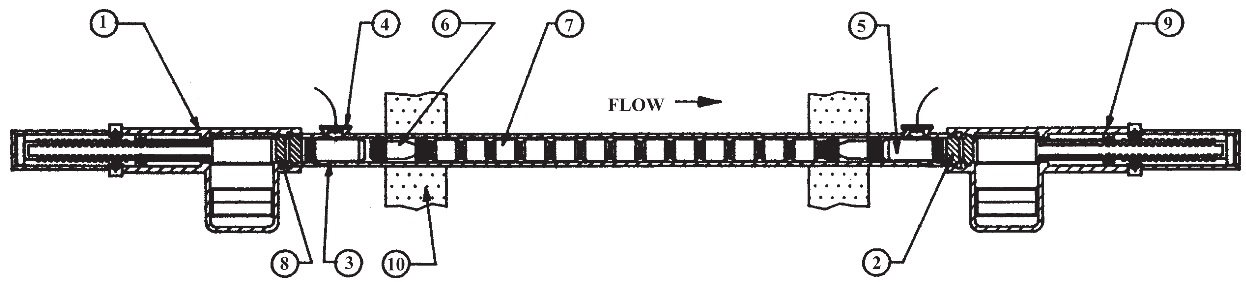


ABFP	AUXILIARY BILER FEED PUMP
ATM	ATMOSPHERE
ASDV	ATM STEAM DISCHARGE VALUE
BCD	BLEED CONDENSER
BCR	BLEED COOLER
BFP	BOILER FEED PUMP
C	CONDENSER
CEP	CONDENSATE EXTRACTION PUMP
CSDV	CONDENSER STEAM DUMP VALVE
ECCHX	EMERGENCY CORE COOLING HX
ECCRP	EMERGENCY CORE COOLING RECIRCULATION PUMP
EFST	EMERGENCY FEED WATER STORAGE TANK
FST	FEED WATER STORAGE TANK
FM	FUELLING MACHINE
G	GENERATOR
HW ACC	HEAVY WATER ACCUMULATOR
IH	INLET HEADER
LW ACC	LIGHT WATER ACCUMULATOR
MP	MODERATOR PUMP
MHX	MODERATOR HEAT EXCHANGER
MSIV	MAIN STEAM ISOLATION VALVE
OH	OUTLET HEADER
PCP	PRIMARY COOLANT PUMP
PP	PRESSURISING PUMP
PR	PRESSURISER
PUR	PURIFICATION
RGCL	REGENERATIVE COOLER
SDC	SHUTDOWN COOLER
SDP	SHUTDOWN PUMP
SG	STEAM GENERATOR
SP	SUPPRESSION POOL
SRV	STEAM RELIEF VALVE
ST	STORAGE TANK
T	TURBINE
UHS	ULTIMATE HEAT SINK

* EITHER PR OR HW ACCUMULATOR

THIS PORTION OF ECCS MAY BE INSIDE OR OUTSIDE CONTAINMENT

FIG.1 PRIMARY HEAT TRANSPORT SYSTEM : TYPICAL PHWR SCHEMATIC



1. UPSTREAM FM HEAD
2. SEALING PLUG
3. END FITTING
4. FEEDER PIPE COUPLING
5. SHIELDING PLUG
6. FUEL LOCATOR
7. FUEL BUNDLE
8. SNOUT PLUG
9. DOWNSTREAM FM HEAD
10. END SHIELD

FIG.2 TYPICAL COOLANT CHANNEL (500 MWe PHWR)

 23

APPENDIX I

LIST OF DESIGN BASIS EVENTS FOR PHTAS

Following tables give the design basis events which form the basis of design analysis of PHTAS.

(Figures indicated against each item correspond to event number for the item as appearing in the AERB safety guide on 'Design Basis Events for Pressurised Heavy Water Reactors' AERB/DG/D-5).

TABLE-1 : CATEGORY-1 EVENTS

NORMAL OPERATION AND OPERATIONAL TRANSIENTS

C1-1 Reactor startup from cold to 100% full power (FP)

C1-2 Reactor power operation

C1-3 Reactor shutdown from 100% FP to cold and maintaining at shutdown state

C1-4 Reactor trip and its re-startup before poison out

C1-5 Reactor start-up just after poison out

C1-6 Reactor trip and cool-down

C1-8 Operational hydro-test. (viz., after repairs affecting the system pressure boundary)

C1-9 Reactor operation with specific items of equipment out of service or under test as may be permitted by the technical specifications for the plant. (viz., operation with unavailability of two main primary coolant pumps under 1-1 mode)

C1-11 Plant disturbances due to electric power supply fluctuation (changes in frequency and voltage)

C1-12 Power changes between from 0% to 100% FP (including sudden changes)

Note : Ck-m type numbering convention has been followed for numbering of events. First number 'Ck' refers to category, second number 'm' refers to serial number of the event. This note is applicable to Table-1 only.

TABLE-2: CATEGORY-2 EVENTS

EVENTS OF MODERATE FREQUENCY

C2-2 Decrease in PHT System Inventory

C2-2.1 Rupture at any location of any small pipe (e.g., instrument line) connected to PHT system

C2-2.2 Rupture of tube(s) of heavy water heat exchangers other than steam generator (like gland cooler, shutdown cooler and bleed cooler)

C2-2.3 Failure of PHT pressure control system with PHT system (cold/hot) [for example, feed valves are stuck closed and bleed valves are stuck open simultaneously as a result of spurious signals from pressure controller]

C2-2.4 Rupture at any location of PHT system up to size of double ended largest feeder pipe

C2-2.5 Unlatching of fuelling machine head from coolant channel without re-sealing

C2-2.6 Failure at any location of any coolant channel assembly (including failure at any location of coolant channel followed by failure of its calandria tube)

C2-2.7 Failure resulting in opening of instrumented relief valves of the PHT system and failure of the relief valve on the bleed condenser to re-close

C2-2.8 Failure of mechanical seals of a single PHT pump

C2-2.9 Rupture of a single steam generator tube

C2-3 Increase in PHT System Inventory

C2-3.1 Failure of PHT pressure control system with PHT system (cold/hot). Consider, also all such incidents when fuelling machine is coupled to PHT system (cold/hot) [for example: feed valves stuck open, bleed valves stuck closed, bleed isolation valves closed by mistake of the operator during maintenance]

C2-3.2 Inadvertent operation of ECCS during cold shutdown condition leading to pressure tube brittle failure

C2-4 Increase in Heat Removal by the Secondary System

C2-4.1 Feed water system malfunctions that result in decrease in feed water temperature

C2-4.2 Feed water system malfunctions that result in increase in feed water flow

C2-4.3 Failures that result in increase in steam flow (For example: boiler pressure controller malfunction, inadvertent opening of main steam line relief or safety valve, steam discharge/dump valve)

C2-5 Decrease in Heat Removal by Secondary System

C2-5.1 Loss of external electrical load

C2-5.2 Turbine trips

C2-5.3 Loss of condenser vacuum

C2-5.4 Loss of normal feed water flow (multiple trains)

C2-6 Decrease in PHT System Flow Rate

C2-6.1 Single and multiple primary coolant pumps trip

C2-6.2 Credible flow blockage in any reactor coolant channel assembly

C2-6.3 Shutdown cooling system pump failure

C2-8 Malfunctioning of Support/Auxiliary Systems

C2-8.1 Process water system circulation failure

C2-8.2 Class-IV electrical power supply failure

C2-8.7 Instrument air failure

C2-9 Others

C2-9.3 Operating Basis Earthquake (OBE): [Ref. AERB/SC/S. 1990: Code of Practice on Safety in Nuclear Power Plant Siting]

Note : Ck-m.n type numbering convention has been followed for numbering of events. First number 'Ck' refers to category, second number 'm' refers to functional classification and third number 'n' is serial number for a particular functional group. Serial numbers are given sequentially in all categories of events. This note is applicable to Tables-2, 3 and 4.

TABLE -3 : CATEGORY-3 EVENTS

EVENTS OF LOW FREQUENCY

C3-2 Decrease in PHT System Inventory

C3-2.9 Rupture at any location of PHT system piping of a size bigger than double-ended largest feeder pipe and including up to double-ended guillotine break of biggest piping in the system

C3-2.10 Failure of a coolant channel leading to ejection of fuel bundles from coolant channel and consequential LOCA

C3-4 Increase in Heat Removal by the Secondary System

C3-4.4 Steam system pipe or header break inside and outside containment

C3-5 Decrease in Heat Removal by Secondary System

C3-5.5 Feed water pipe break

C3-6 Decrease in PHT System Flow Rate

C3-6.4 Primary heat transport main coolant pump shaft seizure or pump shaft break

C3-8 Malfunction of Support/Auxiliary System

C3-8.10 Loss of on-site electrical power supply (class-III, II or I power; one at a time)

C3-8.11 Rupture at any location of any pipe in process water system/process water cooling system

C3-9 Others

C3-9.5 Safe shutdown earthquake (SSE) [Ref. AERB/SC/S, 1990: Code of Practice on Safety in Nuclear Power Plant Siting]

C3-9.7 Design basis flood [Ref. AERB/SC/S, 1990: Code of Practice on Safety in Nuclear Power Plant Siting]

C3-9.8 Design basis cyclone [Ref. AERB/SC/S, 1990: Code of Practice on Safety in Nuclear Power Plant Siting]

C3-9.9 Loss of normal and auxiliary feed water flow

C3-9.10 Dam failure leading to loss of ultimate heat sink

Note : Ck-m.n type numbering convention has been followed for numbering of events. First number 'Ck' refers to category, second number 'm' refers to functional classification and third number 'n' is serial number for a particular functional group. Serial numbers are given sequentially in all categories of events. This note is applicable to Tables-2, 3 and 4.

TABLE-4 : CATEGORY-4 EVENTS

MULTIPLE FAILURES AND RARE EVENTS

C4-2.0 Decrease in PHT System Inventory

C4-2.11 Small or large LOCA coupled with any one of the following:

1. failure of ECCS (in injection or recirculation mode)
2. failure to close the isolation devices on the interconnects between the PHT loops
3. failure of steam generator auto-crash cooling
4. containment impairment characterised by any one of the following:
   - (a) degraded operation of reactor building air coolers
   - (b) failure of one set of containment isolation dampers
   - (c) failure of containment isolation logic
   - (d) one door of main airlock stuck open and seals on second door deflated
   - (e) excessive communication between volumes V1 and V2 of containment (bypassing suppression pool)
   - (f) degraded operation of primary containment clean-up system
   - (g) excessive leakage from primary containment
   - (h) failure of secondary containment clean-up and purge system

C4-2.12 Failure of tube(s) in PHT System heavy water heat exchangers other than steam generator coupled with any one of the following:

1. failure of emergency core cooling system (in injection/recirculation mode)
2. failure to close the isolation devices on the interconnection between PHT loops
3. failure of steam generator auto-crash cooling actuation
4. failure to close the isolation devices on the pipes carrying process water to and from the heat exchangers

C4-9 Others

C4-9.11 Station blackout (Simultaneous failure of Class-III and Class-IV electrical power supply) for specified duration

C4-9.12 Safe Shutdown earthquake (SSE) simultaneous with loss of coolant accidents (LOCA):

This is to be considered only for the purpose of design of those equipment/ systems/structures whose failure could impair integrity of containment.

Note : Ck-m.n type numbering convention has been followed for numbering of events. First number 'Ck' refers to category, second number 'm' refers to functional classification and third number 'n' is serial number for a particular functional group. Serial numbers are given sequentially in all categories of events. This note is applicable to Tables-2, 3 and 4.

TABLE-5

BEYOND DESIGN BASIS EVENTS*

BDBE-1	Loss of coolant accident (LOCA) plus failure of both the reactor shutdown systems
BDBE-2	Loss of coolant accident plus failure of emergency core cooling system followed by loss of moderator heat sink

* An agreed methodology for analysis of these events should be arrived at by the utility and the regulatory body.

ANNEXURE I [1, 2, 3, 4, 5]

LEAK-BEFORE-BREAK (LBB) CONCEPT TO DESIGN THE PRIMARY HEAT TRANSPORT PIPING OF PHWR

I.1 BACKGROUND

Historically, the hypothetical double-ended guillotine break (DEGB) has been taken as the most severe reactor loss of coolant accident in nuclear power plant (NPP) design. The original purpose and intent of the postulated DEGB was to provide a limiting basis for emergency core cooling and containment systems. However, the postulation was extended to the design of high energy primary heat transport (PHT) system piping, resulting in the requirement of massive pipe whip restraints and jet impingement shields, because no alternative acceptable design measure was available. This introduced a number of problems in the design of PHT piping, namely, restricted access for in-service inspection, complication in design, extra cost, increased heat loss to surrounding environment and unanticipated thermal expansion stresses. The DEGB postulation was further extended to the design for environmental qualification of safety-related equipment. For many years, the commercial nuclear industry has recognised that a DEGB is unlikely even under severe accident conditions, and that a LOCA based on DEGB is too restrictive a design requirement. Consequently, the leak-before-break (LBB) concept has emerged to design the PHT piping of pressurised heavy water reactors (PHWR).

I.2 LEAK-BEFORE-BREAK METHODOLOGY

The LBB behaviour for a particular piping system is demonstrated through the following steps:

Step 1: Screening

- Demonstrate that the candidate piping is not susceptible to failure from any degradation mechanism in service such as, water hammer, creep, erosion, corrosion and excessive fatigue. The operating history and measures to prevent or mitigate this degradation must be reviewed.

Step 2: Design with Factor of Safety

- Design the piping system with a well-defined factor of safety (FOS) as given in, for example, section III of ASME ‘Boiler and Pressure Vessel’ code. This design is done without postulation of any cracks in the pipes. However, the FOS will take care to some extent the enhanced stress due to any acceptable flaw.

Step 3: Fatigue Crack Growth Analysis

- Postulate a circumferential semi-elliptical ‘part-through’ crack on the basis of coincidence of the highest fluctuating stress and material properties for base metals, weldments and ‘safe ends’. The aspect ratio of the crack should be six and remain constant throughout the fatigue crack growth analysis. The reference defect size shall be two times the maximum crack size which can escape non-destructive examination. However, if this data is not available during the design stage, one can consider the reference defect size as 25% of the wall thickness.
- Perform the fatigue crack growth analysis. In the analysis, the loading cycles and the loading time history from service level A and B should be taken into account. The rain-flow method is recommended for counting the loading cycles.
- The final flaw depth at the end of reactor life period should be less than maximum allowable crack depth.
- The maximum allowable crack depth after the growth is the smaller of:
    - 60% of the wall thickness;
    - the depth at which the plastic zone size is equal to the remaining ligament.
- The final length of the crack must be less than ½ of the plastic collapse crack length under the superimposed normal operation and accident loading.
- If this limits cannot be met, the LBB concept is not applicable.

Step 4: Unstable Crack Extension Analysis

- Postulate a through wall crack at the critical locations determined on the basis of coincidence of highest stresses and the poorest material properties for base metals, weldments and ‘safe ends’.
- The size of the flaw should be large enough so that any leakage from this postulated flaw size is sure to be detected. When the pipe is subjected to the normal operating loads, it should be demonstrated that there is a factor of at least 10 between the leakage from the so-called leakage size crack (LSC) and the plant’s installed leak detection

capability. It is recommended to use at least three independent leak detection systems in the plant. A leakage analysis which has been bench marked against experimental or plant data is required. The margin on leakage is required to account for uncertainties in the crack opening area, crack surface roughness, two-phase flow and leak-detection capability.

- It should be demonstrated that the postulated flaw is stable under normal and Safe Shutdown Earthquake (SSE) loads, and that there is at least a factor of 1.4 between the load that will cause flaw instability and the normal operating plus SSE loads. This factor of safety can be reduced to 1.0 if the loads are summed up absolutely.

- The flaw size margin should be determined by comparing the leakage size crack (LSC) with the critical crack size. Under the normal plus SSE loads, it should be demonstrated that there is a factor of at least two between the critical crack size and the LSC.

- Alternatively, R6 method or limit load analysis can also be used, if applicable.

Step 5: High Level Cyclic (SSE) Crack Growth Analysis

- Cyclic J-integral range ΔJ, responsible for fatigue crack growth rate in large-scale yielding shall be less than fracture resistance i.e. $J < J_R$, where J is 4 J_{max}. Number of stress cycles considered for crack growth analysis shall be 100. It includes a safety margin of two on the stress cycles conventionally used in the design.

ANNEXURE II

CLASSIFICATION OF PHT SYSTEM COMPONENTS

Mechanical Components

Sl. No.	Components	Safety function	Safety class	Remarks
1	PHT main circuit including reactor coolant system piping (headers, feeders, main circuit piping and valves including associated system piping upto and including first isolation valve penetrating the containment, primary side of the steam generator), pressure and relief system upto bleed condenser level control valves and relief valves.	k	1	
2	Primary coolant pumps pressure - retaining components	k	1	
3	Instrument supply line from PHT main circuit upto the first isolation valve	k	2	
4	PHT purification system	o	3	
5	PHT sampling system	o	3	
6	PHT inventory control beyond first isolation valve	e	3	
7	D_2O (PHT) leakage collection system	n	3	
8	PHT service circuits (includes system required for PHT header level control and boiler filling and draining)	e,n	3	
9	Primary coolant pump gland supply system	i	3	
10	Fuelling machine D_2O system (high pressure) up to first isolation valve	k	1	

CLASSIFICATION OF PHT SYSTEM COMPONENTS (contd.)

Mechanical Components (contd.)

Sl. No.	Components	Safety function	Safety class	Remarks
11	Fuelling machine supply and return circuit	e,q	3	
12	Shutdown cooling system	g	2	
13	PHT deuteration and de-deuteration system	n	4	
14	Emergency core cooling system	f	2	

For these components for safety class, ASME Boiler and Pressure Vessel Code, section III is used for mechanical design of the components as per the following sections mentioned against safety class:

Class 1 : NB

Class 2 : NC

Class 3 : ND

Supports : NF and

Class 4 : Section VIII Division-1 or equivalent.

CLASSIFICATION OF PHT SYSTEM COMPONENTS (contd.)

Electrical Components

Sl. No.	Components	Safety function	Safety class	Remarks
1	PHT pump motors	-	EB	
2	PHT system isolation valve actuator	k	EA	
3	Shutdown cooling system pump motors	g	EA	
4	Shutdown cooling system valve actuators	g, i	EA	
5	Pressurising pump motors	e	EA	
6	Isolating valve actuator with motors	e	EA	
7	Pressuriser heater excluding pressure boundary part	-	EB	
8.	ECCS pump motors	f	EA	
9.	ECCS valve actuators	f	EA	

CLASSIFICATION OF PHT SYSTEM COMPONENTS (contd.)

I&C Components

Sl. No.	Components	Safety function	Safety class	Remarks
1	High pressure trip and IRV actuation	k	IA	
2	Low pressure trip	j	IA	
3	PHT pressure control	v	IB	
4	Bleed condenser and bleed cooler instrumentation	k,v	IB	
5	Reactor trip logic	j	IA	Excluding circuitry associated with class IV switchgear which is NNS.
6	I&C for bearing temperature vibration, oil supply flow, etc., for PHT pumps	-	NNS	
7	Gland seal circuit, gland return valve closure	j, k	IB	
8	Rest of I&C on gland seal circuit	-	NNS	
9	PHT purification system			
9.1	For isolating valves for closure on RB penetration	l	IA	
9.2	Rest of I&C	-	NNS	
10	PHT storage circuit			
10.1	Reactor trip on low level, if provided	e	IA	
10.2	Storage tank level measurement	e	IB	
10.3	Small leak handling system	e	IB	
10.4	Cover gas pressure control	-	NNS	

CLASSIFICATION OF PHT SYSTEM COMPONENTS (contd.)

I&C Components (contd.)

Sl. No.	Components	Safety function	Safety class	Remarks
11	Pressuriser circuit			
11.1	Level protection circuit and isolation valve closure	e,k	IA	
11.2	Rest of I&C like heater control, level control	v,e	IB	
12	Fuelling machine supply and return circuit			
12.1	For keeping PHT solid and supply to FM as PHT system pressure boundary and for fuel cooling	j,k	IB	
12.2	Rest of I&C	-	NNS	
13	Shutdown cooling system.			
13.1	For maintaining flows on both D_2O and H_2O sides of heat exchanger and for core cooling during shutdown	g,h	IA	Instrumentation of individual loop is presently designed as per IB. The realiability requirement of 1A is met with two independent process loops having independent instrumentation.
13.2	Rest of I&C	-	NNS	
14	Leakage collection and service circuit			

CLASSIFICATION OF PHT SYSTEM COMPONENTS (contd.)

I&C Components (contd.)

Sl. No.	Components	Safety function	Safety class	Remarks
14.1	Level of leakage collection tank and C&I of associated pumping back circuit	e	IC	
14.2	FM vault D_2O leakage collection system	j,e,u	IB	
15	Deutration and de-deutration system	-	NNS	
16	D_2O addition and transfer system			
16.1	For remote operated isolating valves on RB penetration, if provided	l	IA	
16.2	Rest of I&C	-	NNS	
17	Coolant flow and temperature monitoring			
17.1	Channel flow monitoring associated with low flow trip	j	IA	
17.2	Flow monitoring in remaining loop	-	NNS	IB, if used in reactor regulating system
17.3	Channel temperature monitoring	j	IB	
17.4	Selected channel delta T monitoring	-	NNS	IB ,if used in reactor regulating system

ANNEXURE III

TABLE-III.1 : MAJOR RADIOACTIVE ISOTOPES REMOVED BY PURIFICATION SYSTEM

Sl.No.	Radioactive Isotope	Half-life	Gamma Energy(MeV)
1.	^{131}I	8.04 d	0.36
2.	^{60}Co	5.27 y	1.17,1.33
3.	^{58}Co	71d	0.81
4.	^{65}Zn	245d	1.11
5.	^{54}Mn	291d	0.84
6.	^{59}Fe	45.1d	1.1,1.29
7.	^{137}Cs	30y	0.662
8.	^{95}Zr	65d	0.72,0.76

Note: Sl.No.1 to 8 are the radioactive isotopes removed during normal course and Sl.No.2 and 6 are removed during PHT decontamination.

TABLE-III.2 : TYPICAL LIMITS FOR PRIMARY COOLANT ACTIVITY DURING NORMAL OPERATION[5]

Sl.No.	Parameter/Consituent	Range/Limit	Remarks
1.	^{131}I activity[6]	* 3.7 MBq/l (100 micro curies/l) Special limit 7.4 MBq/l (200 micro curies/l)	I-131 activity in the coolant indicates the presence of defects or failure in the fuel cladding.
2.	Gross beta-gamma (Decontamination factor)	Activity in main system divided by activity in ion exchange outlet = 10	This ratio gives a measure of the efficiency of removal of radio-activity by ion exchange purification

* If the value exceeds the limit, it shall be brought back below the limit in 72h, failing which the reactor shall be shutdown.

5 The above values of typical limits are for preparation of technical specifications.

6 ^{131}I activity under normal operating conditions is below 10 MBq/1 without fuel failure.

ANNEXURE IV

TYPICAL CHEMISTRY CONTROL PARAMETERS FOR PRIMARY COOLANT ACTIVITY DURING NORMAL OPERATION

Sl. No.	Parameter/ Constituent	Range/Limit	Remarks
1.	Isotopic purity (% w/w D_2O)	As per reactor physics considerations	Included in chemistry specifications as analytical service is entirely provided by chemical unit.
2.	pH at 25°C	10 – 10.4	This is optimum pH range for minimising corrosion of carbon steel and deposition on fuel bundles.
3.	Chloride (mg/litre)	0.3 (max)	To minimise possible localised corrosion especially of stainless steel components.
4.	Dissolved oxygen (micro gm/litre at 25°C)	10 (max)	High dissolved oxygen tends to enhance the concentration and transport of corrosion products.
5.	Specific conductivity at 25°C (micro siemens/cm)	30 (max)	Related to the LiOH concentration in the coolant for maintaining pH within specified limits and the presence of other ionic impurities.
6.	Lithium (mg/litre)	0.8 – 1.5	For maintenance of pH.
7.	Crud (mg/litre)	0.1 (max) for steady operation. 0.1 (max) before applying nuclear heat (>= 2% FP) during start up	Circulating crud gets activated in the core and deposits on the fuel bundles and PHT system surfaces. This leads to high radiation fields on the outer core surfaces of PHT system.
8.	Dissolved deuterium (ml at STP/ litre D_2O)	2 – 10	Maintained in the range by injecting H_2 to the system to suppress radiolysis of heavy water. Upper limit is set to minimise possible H_2 pickup by zircalloy.

ANNEXURE V

TYPICAL VELOCITIES IN PHT SYSTEM COMPONENTS

Sl. No.	Component	Velocity (m/s)
1.	Feeders	17
2.	Coolant channel	10
3.	PHT main piping	12.5
4.	Steam generator tubes	3.5
5.	Heat exchanger tubes	3

ANNEXURE VI

LOADING COMBINATIONS FOR PHTAS COMPONENTS

TABLE-VI.1 : LOADING COMBINATION FOR EQUIPMENT AND PIPING PRODUCTS

Sl No.	Loading combi-nation	Internal pressure	Dead Weight	Thermal	Seismic				Pipe rupture load	Fatigue analysis
					OBE		SSE			
					I	SAM	I	SAM		
1.	Design condition	X	X		X					
2.	Level 'A'	X		X						X
3.	Level 'B'	X	X[#]	X	X	X				X
4.	Level 'C'	X	X				X			
5.	Level 'D'	X	X				X[@]		X[@]	
6	Test	X	X							

TABLE-VI.2 : LOADING COMBINATION FOR EQUIPMENT NOZZLES AND SUPPORTS

Sl No.	Loading combi-nation	Internal pressure	Dead Weight	Thermal	Seismic				Pipe rupture load	Fatigue analysis
					OBE		SSE			
					I	SAM	I	SAM		
1.	Design condition	X	X		X					
2.	Level 'A'	X		X[▯]						X
3.	Level 'B'	X	X	X[▯]	X	X				X
4.	Level 'C'	X	X	X[▯]			X	X		
5.	Level 'D'	X	X	X[▯]			X[@]	X	X[@]	
6	Test	X	X							

SAM : Seismic Anchor Movement

I : Inertial

@ : Either SSE inertial or pipe rupture load one at a time to be considered, excepting equipment support whose failure could lead to containment impairment (e.g., steam generator supports)

: Dead weight to be considered only for primary stress check as per equation 9 (NB 3654.2(a)) for design condition with additional pressure

▯ : Thermal loads imposed by piping systems on the supports and equipment nozzles are treated as primary loads on the supports and nozzles. However, thermal stresses within the supports or nozzles due to their own expansion/contraction are secondary in nature

NOTE : As per Appendix N-1214, cyclic load basis for fatigue analysis of earthquake loading of equipment and components is 10 equivalent maximum stress cycles per earthquake. Hitherto, practice has been to consider 5 OBE events to take care of lower intensity seismic events

ANNEXURE VII

MECHANISM FOR DEMONSTRATING COMPLIANCE WITH ECCS ACCEPTANCE CRITERIA

VII-1 LOCA analysis is carried out for a range of break locations and sizes upto and including double ended break size of reactor inlet header and outlet header as per AERB safety guide on ‘Loss of Coolant Accident Analysis for Pressurised Heavy Water Reactors’ (AERB/SG/D-18). Based on this analysis, the limiting break size, i.e. critical break (which maximizes fuel failures in terms of radiological releases) is identified. Fuel failures in this case are calculated as per guidelines of AERB safety guide on ‘Fuel Design for Pressurised Heavy Water Reactors’ (AERB/SG/D-6). The fission product release (source term) from the failed fuel elements to the PHT system are estimated based on fuel temperature vs. elapsed time during the accident transient as per procedure given in Appendix of AERB safety guide on ‘Fuel Design for Pressurised Heavy Water Reactors’ (AERB/SG/D-6). The release of activity from containment is calculated as per AERB safety guide on ‘Containment Systems Design’ (AERB/SG/D-21). From this, the calculated radiological consequences are estimated and maintained within the dose limits given in AERB safety guide on ‘Radiation Protection in Design’ (AERB/SG/D-12).

VII-2 Analysis of LOCA events with break sizes upto double-ended feeder break, including single channel events are carried out. The fuel sheath temperatures in the core are estimated and fuel failures are checked as per fuel failure criteria (refer to section 7 of AERB safety guide on ‘Fuel Design for Pressurised Heavy Water Reactors’, AERB/SG/D-6).

VII-3 For the LOCA events, the circumferential and axial temperature profiles of pressure tube are estimated. Based on this, the structural integrity of the channel, taking into account the high temperature creep of pressure tube, is checked.

VII-4 ECCS system has been designed to provide reliable, adequate long-term cooling capability, by sizing of ECCS pipes and the capacity of pumps and heat exchangers. Provision of redundancy in pumps and valves and provision of power supply from class III ensure the reliability of the system.

VII-5 Metal-water reaction of the zircalloy components in the core and consequent hydrogen release to containment are estimated as per AERB safety guide on ‘Hydrogen Release and Mitigation Measures under Accident Conditions in Pressurised Heavy Water Reactors’ (AERB/SG/D-19).

ANNEXURE VIII

TYPICAL PRIMARY HEAT TRANSPORT SYSTEM DESCRIPTION AND DESIGN EVOLUTION

VIII-1. INTRODUCTION

The main circuit of primary heat transport system provides the means for transferring heat produced in the fuel (located inside the pressure tubes of the reactor) to the steam generators in which the steam to run the turbine is generated from ordinary water.

The heat transport medium is pressurised heavy water and is circulated through the main circuit by primary coolant pumps.

The primary heat transport (PHT) system is one of the vital systems in a nuclear power plant. The primary coolant in the system is in intimate contact with radioactive fuel bundles and has to remove the heat generated in such a way that under all conceivable plant operating conditions and credible accident conditions, fuel failure due to over heating is prevented. These functional and safety objectives are realised by incorporating the following salient features:

VIII-1.1 Salient Features :

The principal features which the system incorporates are listed below:

(a) continuous circulation of coolant through the reactor at all times by various modes as listed below:

(i)	Normal operation	: by primary coolant pump.
(ii)	Sudden loss of power to pumps	: by inertia of pump fly-wheel to avoid a sudden drop in the coolant pump.
(iii)	Thermo syphoning	: by placing main equipment above the elevation of reactor core.
(iv)	Shutdown cooling	: by shutdown cooling pumps and heat exchangers which are independent of steam generators.
(v)	Loss of coolant accident	: by receiving emergency injection of coolant from pre-charged accumulators while depressurisation of primary heat transport takes place. After initial supply from accumulators is exhausted, long-term cooling is established by ECC recirculation pumps.

(b) pattern of coolant flow rates through the coolant tubes is compatible with the pattern of heat production across the reactor that yields a common increase in coolant temperature in all channels

(c) controlled pressure at the reactor outlet headers to avoid boiling

(d) over-pressure relief to protect the PHT main system from pressure exceeding the design values

(e) addition of coolant to and removal from the system to control the coolant inventory in the main circuit

(f) control of dissolved gases in the coolant

(g) purification and pH control of the coolant

(h) provision for supply of high-pressure heavy water to the fuelling machines

(i) accessibility of all components during shutdown and accessibility of some during operation

(j) heavy water leakage collection from potential leak points in the system

(k) study of corrosion coupons in autoclaves

VIII-1.2 Design Basis

(a) To achieve the design objectives mentioned in para VIII-1.1, the first defence is to keep the primary system pressure boundary intact. Primary system pressure boundary comprises pressure retaining components that contain the coolant and their connecting pipes upto the first passive barrier or the first active isolation device.

This includes any component internals necessary to ensure proper flow of reactor coolant through the core. Components such as valve actuators, pump-motors and supporting features for components are also treated as the pressure boundary components from structural integrity considerations.

It is imperative that these systems and components be designed to the appropriate safety class. Based on this, following part of primary pressure boundary are classified under safety class-1 and are designed, fabricated, inspected and tested as per the requirements of ASME section-III, sub-section NB for class-1 components.

(i) main circuit (primary pressure boundary)

(ii) shutdown cooling system

(iii) relief system

(iv) bleed cooling system

(v) service system up to second isolation valve

(vi) all instrument lines/take off connections, including DN monitoring system, up to the first isolation valve

(b) The second line of defence for achieving the design objective is to prevent or minimize the consequences from postulated initiating events, interrupting normal operation of the PHT system. This will include the systems involved in inventory control and in transporting the core heat (such as ECC recirculation system) to the ultimate heat sink.

The importance of these systems makes it imperative to have at least the second highest ranking in safety class and hence the following systems are classified under safety class-2 and are designed, fabricated, inspected and tested as per the requirements of ASME section-III, sub-section NC for class-2 components.

(i) feed, storage and gland cooling system

(ii) emergency core cooling system

(iii) feed to fuel handling system

(iv) small leak handing system

(v) secondary system (inside RB) and steam generator emergency injection system

(vi) purification system up to isolation inside containment

(c) Systems which are not directly involved for core heat removal are classified under safety class-3 and designed, fabricated, inspected and tested as per ASME section-III, sub-section-ND for class-3 components.

These systems are:

(i) purification system (outside RB)

(ii) leakage collection system

(iii) service system (after second isolation valve)

(iv) D_2O sampling system (after first isolation valve)

VIII-2 DESCRIPTION OF MAIN CIRCUIT

The heavy water flows through the feeders into coolant tubes, through the end fittings and feeders to the reactor outlet headers. The reactor utilizes restriction orifices in selected inlet feeders to achieve the flow required by the reactor channel ratings, commensurate with equal outlet temperature from all channels. The reactor outlet headers distribute the flow through the steam generator and is collected at the primary coolant pumps (PCPs). Each PCP is associated with a respective steam generator through an individual suction line. The PCPs discharge the flow into the reactor inlet header.

From the reactor inlet header the heavy water flows through the feeders and end fittings into the reactor coolant tubes.

The primary heat transport circuit is shown schematically in Fig.1 and is typical for 220 and 500 MWe PHWRs.

The basic components of the main circuit are:

(i) coolant channels

(ii) end fittings and sealing plugs

(iii) feeders (inlet and outlet)

(iv) headers (inlet and oulet)

(v) steam generator isolation valves (inlet and outlet valves in older designs only)

(vi) primary coolant pumps

(vii) steam generators

(viii) pump discharge valve (in old designs only)

VIII.3 SYSTEM CHEMISTRY

Corrosion products and fission products are removed from the system by the purification circuit, which also helps to achieve a specific conductivity value between 20 and 30 micro siemens/centimetre (µs/cm). In addition, it reduces radiolytic decomposition of heavy water by controlling ionic impurities. Particulate matters are removed by a number of filters and strainers at suitable places.

Strict control of the chemical conditions of water in the main circuit is required to minimize corrosion. Apart from the strength reduction of pipe walls caused by corrosion, the radioactivity of the circuit and its components is directly affected. Corrosion products circulating through the reactor become radioactive and then these particles plate out on the inside of pipes; as a result components activity problem becomes severe.

VIII-3.1 Purification System

The principal requirement of the purification system is to maintain a satisfactory chemistry of heavy water in the heat transport system with a pH value of 10.2 to 10.5 and specific conductivity between 20 and 30 ms/cm.

The principal functions of the system are as follows:

(a) to minimize problems in the operation and maintenance of the plant resulting from the burden of activated plant and core material corrosion products, fission products and impurities inherent in the coolant

(b) to reduce radiolytic decomposition of heavy water by removing ionic impurities

(c) to provide a potential decontamination facility for heat transport system equipment and piping with a total flow-handling capacity of approximately 1800 lpm

The purification system employs disposable type filters and ion exchange columns. (in latest versions *in-situ* slurrying type non-disposable ion exchange columns are considered)

VIII-3.2 General Purification Circuit

The purification system comprises of a bank of filters, a bank of ion-exchange columns, strainers, crud filters and the associated instrumentation, piping and valves.

During normal operation the flow to the purification system consists of the bleed flow in the range of 450-600 lpm from the heat transport system and the heat transport leakage collection return. The system has been designed to provide sufficient ion exchange and filter capacity for a maximum flow of 1800 lpm.

The ion-exchanger effluent is returned through one of the two strainers to the storage tank. The strainers prevent the resin beads getting into the system in case of IX column screen failure. A flowmeter, down-stream of strainers, measures the return flow from the purification circuit.

VIII-3.3 Sampling

For monitoring system condition, sample points are provided on the following locations in primary heat transport (PHT) system.

(a) sampling from bleed line

(b) sampling for gases from bleed condenser

(c) sampling from PHT cover gas

(d) sampling from leakage-collection pump discharge line

(e) sampling in purification plant for crud

(f) delayed neutron activity monitoring for fission products. For this purpose, separate sample tubes are taken from individual outlet feeders.

(g) all D_2O/H_2O heat exchangers are provided with sampling provision on H_2O side for monitoring leakage of D_2O.

VIII-4 FEED-AND-BLEED SYSTEM

The main function of the feed-and-bleed system is to maintain the pressure in the main circuit of the primary heat transport system within very narrow limits, under most operating conditions and also during heating and cooling of the system. In doing so, the feed-and-bleed system compensates for changes in volume caused by losses from, or additions of heavy water to, the main circuit and by temperature variations which occur during the normal startup and shutdown operations. Heavy water is pumped to the heat transport system by the bleed circuit. The capacity of the bleed circuit is supplemented by the relief system which removes swells in the heat transport system that are above the capacity of the bleed system.

Subsidiary functions of the feed-and-bleed system are to supply high pressure, cool heavy water to the primary pump glands, hot and heavy water to fuelling machine circuits and heavy water for continuous purification of primary heat transport coolant. Feed system also supplies heavy water to bleed condenser as reflux and spray flows for controlling temperature and pressure in bleed condenser.

VIII-4.1 Pressurising and Pressure Control System

The pressure is established and maintained in the primary heat transport circuit by a feed-and -bleed system without the help of a vapour or gas volume in the system. The pressure, which is controlled, is the mean of the pressures in the two reactor outlet headers.

The feed flow for the system is provided by one of the two multi-stage vertical centrifugal pumps. These pumps, known as pressurising pumps, are installed in parallel, with one pump normally operating and the other available as standby. The motors are powered by class III power. The pressurising pumps supply the flow to the glands of the primary circulating pumps and also the feed directly to the main circuit when required.

The pumps draw water from the portion of the low pressure heavy water storage circuit to which D_2O storage tank is connected. Purification return flow is directed into the storage tank. The storage tank remains in flow path and at the same time can supply make-up water to the main system, whenever needed.

The feed-and-bleed system as described above is adequate for warming up or cooling down the plant at 3^{o}C/min. and to cope with reactor power changes at 0.6 per cent per second and 1 per cent per second during reactor start up and set-back respectively. A further transient which can be accepted by the reactor in the operating condition is the starting of an extra primary coolant pump or the stopping of a primary coolant pump. Taking for reference the flow condition with four pumps in one bank (RAPS/MAPS) reduces the overall flow in the reactor by about 8 per cent. This reduced cooling causes a relatively sudden but small swelling in the primary system, and causes the pressure rise. This is within the allowed range. In the event of loss of power to all primary coolant pumps, the flywheels serve to keep the flow going for some time, but during the first second the flow falls off faster than does the heat output from the fuel. This gives rise to a momentary small pressure surge. This pressure surge is not expected to rise to the trip setting but anyway reactor would be tripped on loss of pumps.

All primary circuit feed-and-bleed valves are provided in duplicate with an independent control system. The arrangement is such that if any one of the four valves or its control system should fail and cause that valve to open or close completely, the reactor would still be able to carry on operating properly and could be shut down without reactor trip, provided the other three valves operate correctly.

With the advent of 500 MWe PHWRs, the PHT main circuit inventory has also gone up and necessitated incorporation of a pressuriser which significantly reduces onerous demand on the feed-and-bleed system. In these reactors, the feed-and-bleed system acts mainly as an inventory control system and backup for pressure control when pressuriser is not available.

VIII-4.2 Bleed Cooling

The bleed from the primary system enters the bleed valves. The outflow, after passing through the bleed valves, is flashed to a substantially lower pressure in the bleed condenser which is equipped to condense the flashed vapour.

The feed pumps provide the cooling water via the reflux cooler to the bleed condenser which condenses the flashed bleed vapour. They also provide the water to the sprays in the bleed condenser, which is used to supplement the condensing capacity of bleed condenser. Degassing of the system can be carried out by opening the gas control valve on bleed condenser, whenever degassing is required.

The outflow from the bleed condenser is cooled in a heat exchanger, known as ‘bleed cooler’ and returned via the purification system to the storage tank. In newer designs a regenerative cooler is incorporated prior to bleed cooler to use and feed back the heat in bleed flow to the main circuit.

Control valves in this line regulate the level of water in the bleed condenser.

Flow through the bleed cooler is measured downstream of the bleed condenser level-control valves. The bypass valve around the purification system is controlled to prevent the pressure rising above a preset value of approximately 10.50 kg/cm^2 (g). Use of the purification bypass should be avoided if possible in order to minimize the spread of radioactivity into accessible areas.

The bleed condenser pressure can be regulated either by spray flow, reflux condenser flow or by a combination of the two flows. Normally the pressure is controlled by reflux flow only and the sprays are used to supplement the capacity of the reflux condenser.

VIII-5 SHUTDOWN COOLING AND MAINTENANCE

Removal of heat from the reactor is best performed by the steam generators, as long as feed water is supplied and PHT is in subcooled state. Steam generators are used for all-system cool-down operations upto 150° C. Below 150°C and for holding the heat transport system temperature low enough to perform maintenance, shutdown cooling system is required. It is proposed to provide two independent shutdown circuits for each loop between the two reactor inlet and outlet headers. The circuit remains isolated from main system by isolation valves during normal reactor operation. The system has been designed to endure Operating Basis Earthquake and also Safe Shutdown Earthquake (SSE). The active components like valves, their actuators and pumps shall remain operable even after SSE.

The shutdown cooling heat exchangers are of tube-in-shell design with U-tube bundle designed to withstand rapid temperature transients as indictated by the service conditions.

The shutdown cooling system is capable of cooling the main circuit from hot shutdown (zero power hot) condition also for some cycles in earlier designs.

The shutdown cooling system pressure boundary upto and including isolation valves on pressure boundary are classified as class I components of ASME Boiler and Pressure Vessel Code Section III for nuclear components.

The shutdown cooling circuit is normally isolated from the main circuit. When the system cool down is required, initial cooling is provided by releasing steam through the steam-discharge valves on the steam mains. A small flow through the shutdown circuit is simultaneously provided for warming up the system. When the system temperature reduces to 150°C, the shutdown isolating valves are opened and the power to shutdown cooling pumps is switched on. This provides for a normal cooling rate of 3° C/min.

When maintenance work on the secondary side (steam side) of the steam generators is required, the PHT system is cooled through shutdown cooling circuit. The secondary side is drained through steam generator blow-off connections. Provision of nitrogen blanketing to minimize oxidation is under consideration.

VIII-6 EMERGENCY CORE COOLING SYSTEM

Emergency core cooling system (ECCS) is one of the engineered safety systems provided to mitigate the consequences of loss of coolant accident (LOCA) in the event of a break in primary circuit pressure boundary. The ECCS is designed to provide enough coolant to the primary system and to transport heat from the core to the ultimate heat sink in such a way as to ensure, with sufficient reliability, adequate core (reactor) cooling during all phases of the accident.

The ECCS is so designed to be available even in the event of loss of off-site power by providing emergency power supply from diesel generator sets. The system design accommodates single failure criterion.

This ECCS in conjunction with recovery-and-inventory control system (storage + feed and bleed) is capable of handling the entire spectrum of break size in primary pressure boundary as follows:

VIII-6.1 Small Leak (small size of break)

Small leaks resulting due to a small rupture in primary pressure boundary is defined to be one which can be handled by recovery-and-inventory control system without invoking light water injection (and break of rupture disc). The system coolant is not significantly downgraded. The spilled heavy water inventory from the break point is collected in the recovery tank located below the calandria vault and is pumped back to the PHT D_2O storage tank where it is injected into the primary circuit by primary feed pumps. The maximum leakage rate (and the failure size) thus dictated by the capacity of primary feed pumps and recovery pumps is about 1000 lpm. In order to provide for the make-up required because of transport time between break point and recovery tanks, additional reserve heavy water inventory can be transferred to PHT system from D_2O reserve storage tank in the reactor auxiliary building.

VIII-6.2 Medium LOCA

This class of LOCA results from medium-size rupture, which is in between small break and large break. The break size is greater than that corresponding to small leak and therefore cannot be handled by normal recovery-and-inventory control system. It is smaller than that corresponding to large LOCA to the extent that the outflow from the break is inadequate to cool the core on long-term basis and this LOCA has to be handled by a combination of ECCS recirculation pumps acting as inventory make-up pumps and shutdown cooling pumps for coolant circulation through the core. Loss of inventory from PHT D_2O storage tank and pressurizer would be the first signal for operator intervention if the break size is on the lower side.

VIII-6.3 Large LOCA

This class of LOCA results from large-size rupture in primary pressure boundary and all the mitigating actions are automatic. The largest possible rupture in 500 MWe PHWR is due to double-ended break of 610 mm OD pipe connecting steam generator to the primary coolant pump. In order to reduce time of injection and increase reliability of detection, it is proposed to inject emergency coolant into all the eight headers. Isolated and unbreached loop, however, would not accept injection because of prevailing higher pressure in the loop and operator action would be required to isolate/close injection valves on the headers, close to the break point to avoid core bypassing from this path.

VIII-7 PRIMARY SYSTEM RELIEF

The purpose of the primary relief system is to prevent the system pressure from exceeding that for which it is designed.

Once the reactor has been operated at significant power, the fuel will continuously supply heat. If normal heat removal fails and normal pressure control fails or their capacities are exceeded, the increase in coolant volume caused by the reactor heat would be passed out of the primary system by the relief valves.

A system consisting of relief valves and extra volume in bleed condenser is provided to relieve the over-pressure. However, because the heat capacity and bleed condenser volume are limited, the heat and volume could exceed the capacity of the condenser and therefore, the relief system by itself cannot offer full protection but can do so only in association with an operating reactor shutdown system.

In the event of a certain compound failure in the pressure control system or during certain transients which could result from fault conditions in other systems, the pressure in the main control circuit rises above its design pressure. Instrumented relief valves are connected to the circuit to prevent the circuit pressure under such conditions from exceeding 110% of the (coolant tube) design pressure, to maintain the pressure within the limits permitted by the ASME code with full flow in the relief valves.

VIII-8 PRIMARY SEAL LEAKAGE

The purpose of the heat transport D_2O collection system is to :

- collect heavy water leakage from valve packings, pump seals and the intergasket cavities on valves and equipment flanges of primary and associated system equipment

- collect gases vented manually during the filling of equipment at startup and after maintenance work
- collect drainage from equipment when required for maintenance work
- transfer the collected heavy water to the purification system or to the D_2O transfer system which through the clean up system finally leads to the upgrading system
- transfer the vented gases to the fuelling machine vault air-dryers for recovery of the heavy water vapour
- add D_2O to PHT system from addition station in service building

A recirculation circuit is provided on the collection tank pump out-circuit to enable the pumps to be operated when a sample of the water in the tank is required. The sample is obtained at the heat transport system glove-box sampling station.

As heavy water draining to the D_2O collection system can be hot at times, a cooing coil is provided inside the tank. This cooling arrangement is necessary to prevent vapour locking and damage of the canned motor pumps. To recover D_2O vapours, the tank is vented to the fuelling machine vault-dryers through vent condensers.

VIII-8.1 Pump Seal Leakage

Dry service air is supplied to the vapour seals on the glands of the primary coolant pumps, shutdown cooling pumps and pressurising pumps to prevent the following:

- release of tritiated D_2O vapour to the environment from water leaking past shaft seals
- entrance of H_2O-laden air from the surrounding atmosphere, which could downgrade D_2O in the collection system
- leakage of D_2O along pump shafts on failure of mechanical seals

VIII-8.2 Leakage from Valves

D_2O leakage is collected from one or both of the following two areas on all heat transport system valves excepting the bellows seal valves

(a) The lantern ring cavity between the valve stem packing

(b) The intergasket cavity on double gasket, body and bonnet joints.

The widest possible use has been made of bellow seal valves in an endeavour to prevent leakage problems rather than provide for them.

As a result of this, the number of gate valves (with the exception of boiler and standby system isolated valves) has been kept to something less than 10 per cent as this type of valve is not readily available with bellows stem seal because of the comparatively long stem travel. In addition, gate valves generally contain flanged body joints - an area which constitutes another potential leak source.

In these high temperature valves requiring collection from double-packed glands and double -gasket body joints, the leakage from both sources is taken to the collection system through line. Provision is made at the valve to enable disconnection of each line to determine the major leak source. This arrangement, although detracting from the facility of the collection system to indicate leak source quickly, is considered to be justified by reduction in the tubing installation. Further, operating experience indicates that in approximately 95 per cent of cases valve glands rather than body joints are responsible for leaks. Operators can therefore assume, with a high degree of confidence, that leakage from a valve emanates from the gland.

VIII-9 DESIGN EVOLUTION AND STANDARDISATION

The system and component design for RAPS-1 was executed in the early sixties and was based on the design of Douglas Point Generating Station done by AECL. The component and system then was governed by ASME Boiler and Pressure Vessel Code section-VIII (and to limited extent as per section-III) and piping as per ANSI B 31.1. Major components, like primary coolant pumps were designed from functional and fabrication point of view. However, with the development of section-III Code and better methods of detailed stress analysis, like Finite Element, detailed calculations and development of non destructive examination techniques called for quick adoption of these developments in nuclear technology in general and primary pressure boundary in particular since reliability and radiation safety are of paramount importance. Table VIII.1 gives progressive changes that have taken place in design parameters of PHT main circuit from RAPS-1 through KAPS and the 500 MWe PHWR. Tables VIII.2, VIII.3 and VIII.4 give important data on three major components of the system, viz. steam generators, primary coolant pumps and reactor headers. Fig. VIII-1 provides the schematic arrangement of primary heat transport pressure control and associated systems. Fig. VIII.2 and VIII.3 give the schematic arrangement of the main primary heat transport system in 220 MWe and 500 MWe. Continuously growing awareness regarding radiation safety has been recognised the world over and IAEA prepared a design guide on ‘Reactor Coolant and Associated Systems in Nuclear Power Plants’ viz. 50-SG-D13 which identifies the areas of concern in PHT system and provides the guidance on design from radiological safety point of view.

A study of the Tables VIII.2,VIII.3 and VIII.4 conveys that the design evolution process has continued even up to KAPS. All efforts have been put in the KAPS equipment design to incorporate present state of engineering capabilities to meet stringent radiological safety standards. Use of impact tested material, restriction on residual cobalt content, maximum use of forgings, elimination of welded joints and amenability for periodic inspection during service have been the guiding principles in the process of design evolution.

No major changes in these areas are expected in the future and the KAPS design is being adopted for all 220 MWe future stations as standard design. This would enable manufacturers to take up manufacturing of substantial 'lot size' with shorter manufacturing time.

The major components of PHT system that have been the cause of concern from design and manufacturing point of view for 220 MWe units were identified very early in 500 MWe programme and advance action is being taken to avoid recurrence of similar experiences. The design of first 500 MWe unit shall be repeated at least in ten units to follow and this standardization would also help in reducing the total design and manufacturing time.

TABLE-VIII.1 : DESIGN DATA

Sl.No.	Description	RAPS	MAPS	NAPS	KAPS	500 MWe
1	Station output MWe	208	235	235	235	525
2	Heat transfer across steam generator MWe	659	758	796	796	1736
3	No. of steam generators	8	8	4	4	4
4	No. of coolant channels	306	306	306	306	392
5	No. of primary coolant pumps	8	8	4	4	4
6	Coolant flow 10^6 kg/h.	10.23	10.5	13.55	13.55	32.00
7	No. of reactor headers	4	4	4	4	8
8	Design pressure kg/cm^2(g)	112	112	112	112	126
9	Design temperature °C	299	299	299	299	310

TABLE-VIII.2 : STEAM GENERATOR FOR PHT SYSTEM

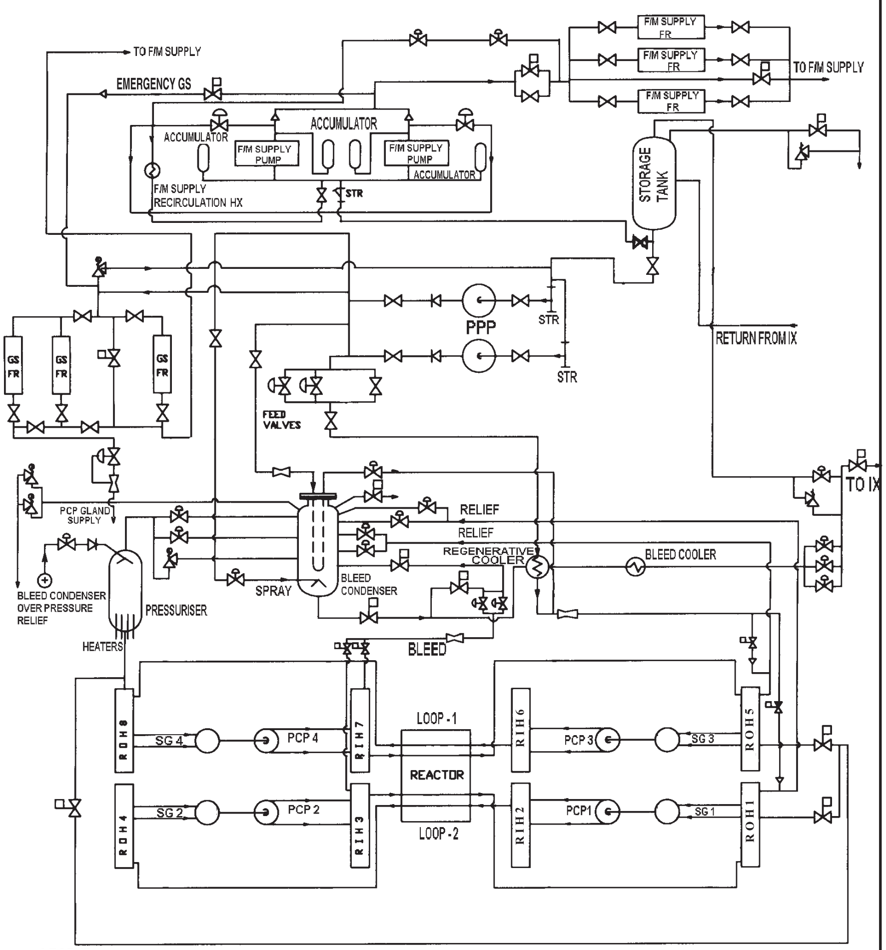
	Description	RAPS	MAPS	NAPS	KAPS	500 MWe
a)	Type	10 hair pins with common steam drum with preheater	11 hair pins with common steam drum with preheater	Mushroom type single tube sheet and integral steam drum with preheater	As in NAPP	As in NAPP without preheater
b)	Tube material	Monel 400	Monel 400	Incoloy 800	Incoloy 800	Incoloy 800
c)	Number of tubes/SG	195 x 10	195 x 11	1834	1834	2489
d)	Tube sheet diameters (m)	0.254 & 0.33	0.254 & 0.33	1.88	1.88	2.5
e)	Design Code ASME:					
	Primary	Section III Division 1	Section III	Section III	Section III Class 1	Section III Class 1
	Secondary	Section VIII Division 1	Section VIII	Section III Class 2	Section III Class 2	Section III Class 1

TABLE-VIII.3 : PRIMARY COOLANT PUMPS FOR PHT

	Description	RAPS	MAPS	NAPS	KAPS	500 MWe
a)	Pump type	Vertical single suction, single discharge	Same as in RAPS	Same as in RAPS	Same as in RAPS	Vertical single suction, double discharge
	Rated Head (m)	146.3	149	178	178	214
	Rated capacity (m^3/h)	1547	1550/1700	3550	3550	8400
b)	Gland sealing	1 pressure reducing throttle; 1 mechanical seal; 1 vapour seal; labyrinth type external seal injection	2 mechanical seals; 1 vapour seal; carbon ring type external seal injection	3 mechanical seals; 1 vapour seal; external seal injection	Same as in NAPP	3 mechanical seals; 1 vapour seal; 1 back-up seal; external seal injection
c)	RPM	3000	3000	1500	1500	1500
d)	Motor rating (KW)	875	1050	2800	2800	6600
e)	Voltage (KV)	3.3	6.6	6.6	6.6	6.6

TABLE-VIII.4 : REACTOR HEADERS FOR PHT

	Description	RAPS	MAPS	NAPS	KAPS	500 MWe
a)	Material	CS-A 105 Grade II	CS-A 105 Grade II	CS-A 105 low cobalt	CS-SA 350 LF 2 low cobalt and impurities	Same as NAPP
b)	Size	400 mm NB	Same as RAPP	Same as RAPP higher thickness	Same as NAPP; higher thickness	500 mm ROH 450 mm RIH
c)	Diameter of big opening	300 mm NB	300 mm NB	400 mm NB	400 mm NB	500 mm NB and 45 mm NB
d)	Feeder connections 40-65 mm NB pipes	153	153	153	153	98 x 4 = 392
e)	Type	Pull out for big openings; sit-on fitting for feeders	Reducing tees for big opening; sit-on fitting for feeders	Equal tees for opening; sit-on fittings for feeders	One piece forging with contour m/c for opening	

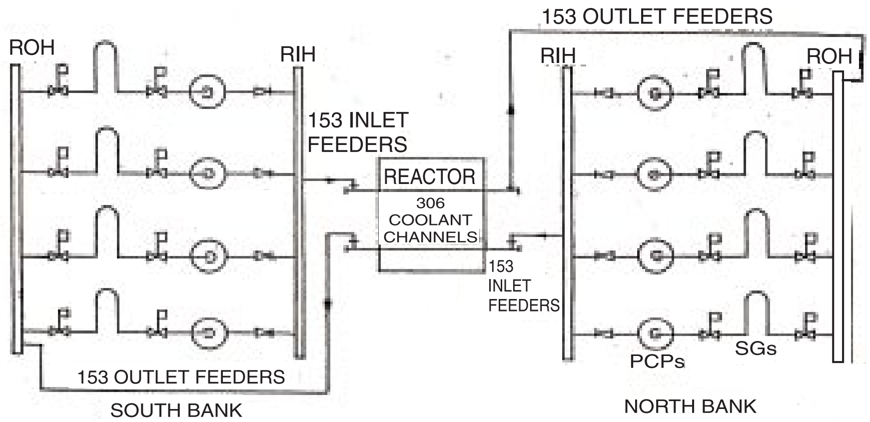


LEGEND :

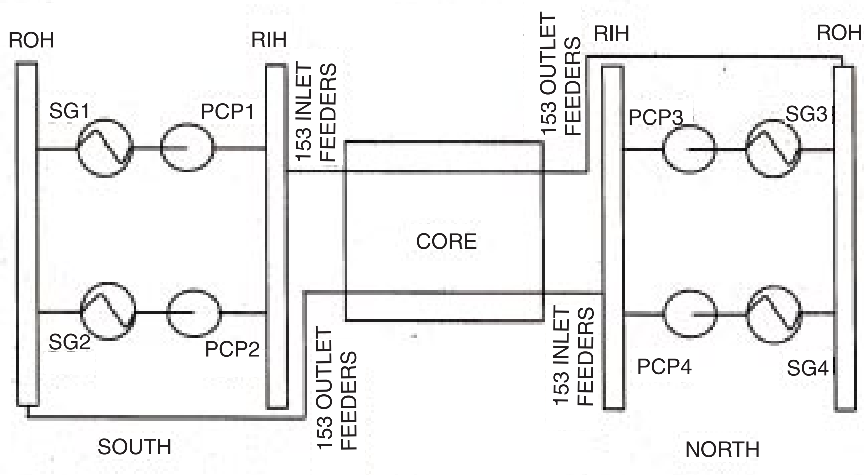
RIH : REACTOR INLET HEADER
GS : GLAND SUPPLY
STR : STRAINER
ROH : REACTOR OUTLET HEADER
FM : FUELING MACHINE
GR : GLAND RETURN
PCP : PRIMARY COOLANT PUMP
FR : FILTER
SG : STEAM GENERATOR
PPP : PRIMARY PRESSURISING PUMP

FIGURE-VIII.1 - PRIMARY HEAT TRANSPORT PRESSURE CONTROL AND ASSOCIATED SYSTEM

63



RAPS MAIN CIRCUIT



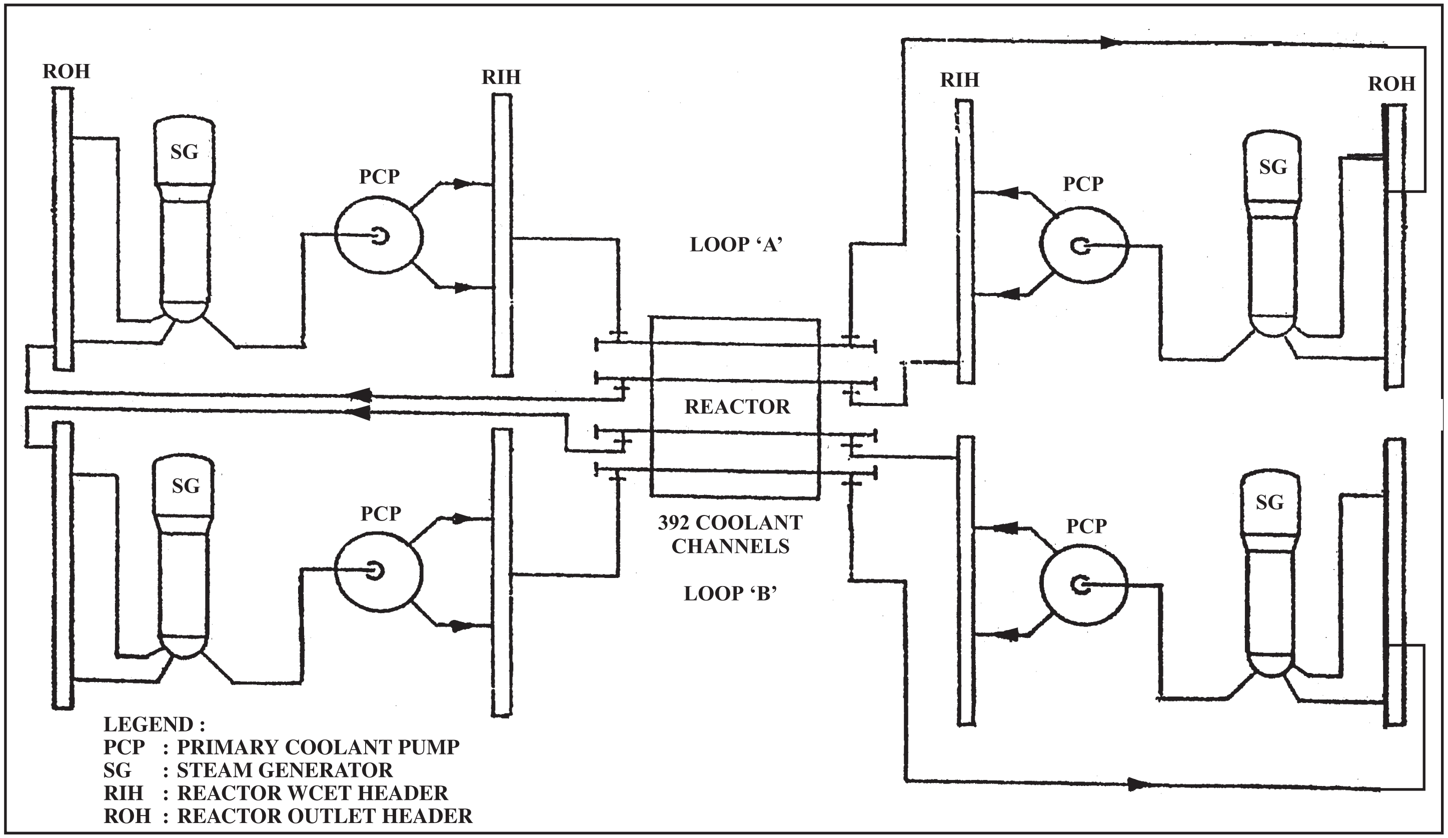
KAIGA MAIN CIRCUIT

LEGEND :

ISOLATING VALVE
CHECK VALVE

PCPs PRIMARY COOLANT PUMPS
SGs STEAM GENERATORS
RIH REACTOR INLET HEADER
ROH REACTOR OUTLET HEADER

FIGURE-VIII.2 : PHT MAIN CIRCUIT (220 MWe)



VIII. 3 500 MWE PHWR PHT MAIN CIRCUIT

 65

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LIST OF PARTICIPANTS

WORKING GROUP

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Shri S.G. Ghadge	:	NPCIL
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ADVISORY COMMITTEE FOR CODE, GUIDES AND ASSOCIATED MANUALS FOR SAFETY IN DESIGN OF NUCLEAR POWER PLANTS (ACCGD)

Dates of meeting	:	January 22&23, 1996 December 3, 1997 January 6, 1998 August 28, 1998 February 3, 1999 November 7, 2000

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Shri S.A. Khan (Member-Secretary)	:	AERB

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Date of meeting	:	August 28, 1999

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Shri K. Srivasista (Member-Secretary)	:	AERB

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

Safety Series No.	Provisional Title
AERB/SC/D	Code of Practice on Design for Safety in Pressurised Heavy Water Based Nuclear Power Plants
AERB/NPP-PHWR/ SG/D-1	Safety Classification and Seismic Categorisation for Structures, Systems and Components of Pressurised Heavy Water Reactors
AERB/SG/D-2	Structural Design of Irradiated Components
AERB/SG/D-3	Protection Against Internally Generated Missiles and Associated Environmental Conditions
AERB/SG/D-4	Fire Protection in Pressurised Heavy Water Reactor Based Nuclear Power Plants
AERB/SG/D-5	Design Basis Events for Pressurised Heavy Water Reactors
AERB/NPP-PHWR/ SG/D-6	Fuel Design for Pressurised Heavy Water Reactors
AERB/SG/D-7	Core Reactivity Control in Pressurised Heavy Water Reactors
AERB/NPP-PHWR/ SG/D-8	Primary Heat Transport System for Pressurised Heavy Water Reactors
AERB/SG/D-9	Process Design
AERB/SG/D-10	Safety Critical Systems
AERB/SG/D-11	Emergency Electric Power Supply Systems for Pressurised Heavy Water Reactors
AERB/SG/D-12	Radiation Protection in Design
AERB/SG/D-13	Liquid and Solid Radwaste Management in Pressurised Heavy Water Reactor Based Nuclear Power Plants
AERB/SG/D-14	Control of Air-borne Radioactive Materials in Pressurised Heavy Water Reactors

PROVISIONAL LIST OF SAFETY CODES, GUIDES AND MANUAL ON DESIGN OF PRESSURISED HEAVY WATER REACTORS (contd.)

Safety Series No.	Provisional Title
AERB/SG/D-15	Ultimate Heat Sink and Associated Systems in Pressurised Heavy Water Reactors
AERB/SG/D-16	Materials Selection and Properties
AERB/SG/D-17	Design for In-Service Inspection
AERB/SG/D-18	Loss of Coolant Accident Analysis for Pressurised Heavy Water Reactors
AERB/NPP-PHWR SG/D-19	Hydrogen Release and Mitigation Measures under Accident Conditions in Pressurised Heavy Water Reactors
AERB/NPP-PHWR SG/D-20	Safety Related Instrumentation and Control for Pressurised Heavy Water Reactor Based Nuclear Power Plants
AERB/SG/D-21	Containment System Design
AERB/SG/D-22	Vapour Suppression System for Pressurised Heavy Water Reactors
AERB/SG/D-23	Seismic Qualification
AERB/SG/D-24	Design of Fuel Handling and Storage Systems for Pressurised Heavy Water Reactors
AERB/SG/D-25	Computer Based Safety Systems
AERB/SG/D-26	Deterministic Safety Analysis of Nuclear Power Plants
AERB/SM/D-1	Decay Heat Load Calculations