IAEA-TECDOC-1624

Passive Safety Systems and Natural Circulation in Water Cooled Nuclear Power Plants

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FOREWORD

Nuclear power produces 15% of the world's electricity. Many countries are planning to either introduce nuclear energy or expand their nuclear generating capacity. Design organizations are incorporating both proven means and new approaches for reducing the capital costs of their advanced designs. In the future most new nuclear plants will be of evolutionary design, often pursuing economies of scale. In the longer term, innovative designs could help to promote a new era of nuclear power.

Since the mid-1980s it has been recognized that the application of passive safety systems (i.e. those whose operation takes advantage of natural forces such as convection and gravity), can contribute to simplification and potentially improve economics of new nuclear power plant designs. The IAEA Conference on The Safety of Nuclear Power: Strategy for the Future, which was convened in 1991, noted that for new plants 'the use of passive safety features is a desirable method of achieving simplification and increasing the reliability of the performance of essential safety functions, and should be used wherever appropriate". Some new designs also utilize natural circulation as a means to remove core power during normal operation. The use of passive systems can eliminate the costs associated with the installation, maintenance, and operation of active systems that require multiple pumps with independent and redundant electric power supplies. However, considering the weak driving forces of passive systems based on natural circulation, careful design and analysis methods must be employed to ensure that the systems perform their intended functions.

To support the development of advanced water cooled reactor designs with passive systems, investigations of natural circulation are conducted in several IAEA Member States with advanced reactor development programmes. To foster international collaboration on the enabling technology of passive systems that utilize natural circulation, the IAEA initiated a Coordinated Research Project (CRP) on Natural Circulation Phenomena, Modelling and Reliability of Passive Systems that Utilize Natural Circulation in 2004. As one output of this CRP, this publication describes passive safety systems in a wide range of advanced water-cooled nuclear power plant designs with the goal of gaining insights into the system design, operation, and reliability.

The IAEA officers responsible for this publication were J. Cleveland and J.H. Choi of the Division of Nuclear Power.

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1. INTRODUCTION

As part of the IAEA's overall effort to foster international collaborations that strive to improve the economics and safety of future water-cooled nuclear power plants, an IAEA Coordinated Research Project (CRP) was started in early 2004. This CRP, entitled Natural Circulation Phenomena, Modeling and Reliability of Passive Safety Systems that Utilize Natural Circulation, focuses on the use of passive safety systems to help meet the safety and economic goals of a new generation of nuclear power plants. This CRP has been organized within the framework of the IAEA Department of Nuclear Energy's Technical Working Groups for Advanced Technologies for Light Water Reactors and Heavy Water Reactors (the TWG-LWR and the TWG-HWR) and provides an international cooperation on research work underway at the national level in several IAEA Member States.

The use of passive safety systems was addressed in 1991 at the IAEA Conference on "The Safety of Nuclear Power: Strategy for the Future" [1]. Subsequently, experts in research institutes and nuclear plant design organizations from several IAEA Member States collaboratively presented their common views in a paper entitled 'Balancing passive and active systems for evolutionary water cooled reactors' [2]. The experts noted that a designer's first consideration is to satisfy the required safety function with sufficient reliability, and the designer must also consider other aspects such as the impact on plant operation, design simplicity and costs. The Safety Fundamentals of the IAEA Safety Standards [3] recommends "an appropriate combination of inherent and engineered safety features" for defence in depth. Design Requirements of the IAEA Safety Standards [4] mentions "following a postulated initiating event, the plant is rendered safe by passive safety features or by the action of safety systems that are continuously operating in the state necessary to control the postulated initiating event".

The use of passive safety systems such as accumulators, condensation and evaporative heat exchangers, and gravity driven safety injection systems eliminate the costs associated with the installation, maintenance and operation of active safety systems that require multiple pumps with independent and redundant electric power supplies. As a result, passive safety systems are being considered for numerous reactor concepts (including in Generation III and III+ concepts) and are expected to find applications in the Generation-IV reactor concepts, as identified by the Generation IV International Forum (GIF). Another motivation for the use of passive safety systems is the potential for enhanced safety through increased safety system reliability.

The CRP benefits from earlier IAEA activities that include developing databases on physical processes of significant importance to water cooled reactor operations and safety [5,6], technical information exchange meetings on recent technology advances [7-13], and Status Reports on advanced water cooled reactors [14,15]. In the area of thermal hydraulic phenomena in advanced water cooled reactors, recent IAEA activities have assimilated data internationally on heat transfer coefficients and pressure drop [5]; and have shared information on natural circulation data and analytical methods [5], and on experimental tests and qualification of analytical methods [8]. This CRP also benefits from a recent report issued by IAEA [16] on the status of innovative small and medium sized reactor designs.

In order to establish the progress of work in this CRP, an Integrated Research Plan with description of the tasks addressing the objectives of the CRP was defined. These tasks are:

- Establish the state-of-the-art on natural circulation
- Identify and describe reference systems
- Identify and characterize phenomena that influence natural circulation
- Examine application of data and codes to design and safety
- Examine the reliability of passive systems that utilize natural circulation.

The activity under the first task is aimed at summarizing the current understanding of natural circulation system phenomena and the methods used experimentally to investigate and model such phenomena. In November 2005, the IAEA issued a technical document [17], developed by the collaborative effort of the CRP participants and with major contributions from some selected experts in the CRP, aimed at documenting the present knowledge in six specific areas; advantages and challenges of natural circulation systems in advanced designs, local transport phenomena and models, integral system phenomena and models, natural circulation experiments, advanced computation methods, and reliability assessment methodology.

The activity for the third task is aimed at identifying and categorizing the natural circulation phenomena of importance to advanced reactors and passive safety system operations and reliability. This task is the major link between the second and the fourth tasks. The activities related to the second task and the fourth task including the fifth task are agreed to be published in two different IAEA-TECDOCs by the CRP participants. Since the third task is the backbone for both tasks, inclusion of this task in both IAEA-TECDOCs in an appropriate form is a logical consequence.

The aim of this publication is to describe passive safety systems in a wide range of advanced water-cooled nuclear power plant designs with the goal of gaining insights into the system design, operation, and reliability without endorsement of the performance. This publication has a unique feature which includes plant design descriptions with a strong emphasis on passive safety systems of the specific design. These descriptions of the passive safety systems together with the phenomena identification (including the definitions of the phenomenon to describe in some detail the titles of the phenomenon considered) are given in the Annexes and Appendix of this report, respectively. Based on the passive systems and phenomena, which are considered, a cross reference matrix has been established and also presented in this report. As basis for the phenomenon identification, earlier works performed within the OECD/NEA framework during 1983 to 1997 were considered. These are:

- Code validation matrix of thermal-hydraulic codes for LWR LOCA and transients [24],
- State of the art report (SOAR) on thermo-hydraulic of emergency core cooling in light water reactors [23],
- Separate effects test (SET) validation matrix for light water reactors [19],
- Integral facility tests validation matrix for light water reactors [20],
- Status report on relevant thermal-hydraulic aspects of advanced reactor designs [21].

Since the Generation III and III+ reactor designs contain technological features that are common to the current generation reactors, the phenomena identified during the work performed for first item to fourth item can be used as base knowledge. The fifth item provides the important and relevant thermal hydraulic phenomena for advanced reactor designs in addition to the relevant thermal hydraulic phenomena identified for the current generation of light water reactors (LWR). The list of relevant phenomena established in reference 21 has been taken as basis for the CRP work and has been modified according to the reactor types and passive safety systems considered in this report. It is to be noted that in identifying the relevant thermal hydraulic phenomena in the list which is provided in this report, expert judgement is the main contributor.

IAEA-TECDOC-626 provides definitions for safety related terms as applied to advanced reactors [16]. In that document, the concepts of passive and active safety systems are defined and discussed. The definition of a passive safety system is as follows: *Either a system which is composed entirely of passive components and structures or a system which uses active components in a very limited way to initiate subsequent passive operation*. Four categories were established to distinguish the different degrees of passivity.

Category A

This category is characterized by:

- no signal inputs of ‘intelligence’
- no external power sources or forces
- no moving mechanical parts, and
- no moving working fluid.

Examples of safety features included in this category are physical barriers against the release of fission products, such as nuclear fuel cladding and pressure boundary systems; hardened building structures for the protection of a plant against seismic and or other external events; core cooling systems relying only on heat radiation and/or conduction from nuclear fuel to outer structural parts, with the reactor in hot shutdown; and static components of safety related passive systems (e.g. tubes, pressurizers, accumulators, surge tanks), as well as structural parts (e.g. supports, shields).

Category B

This category is characterized by:

- no signal inputs of 'intelligence'
- no external power sources or forces
- no moving mechanical parts; but
- moving working fluids.

Examples of safety features included in this category are reactor shutdown/emergency cooling systems based on injection of borated water produced by the disturbance of a hydrostatic equilibrium between the pressure boundary and an external water pool; reactor emergency cooling systems based on air or water natural circulation in heat exchangers immersed in water pools (inside containment) to which the decay heat is directly transferred; containment cooling systems based on natural circulation of air flowing around the containment walls, with intake and exhaust through a stack or in tubes covering the inner walls of silos of underground reactors; and fluidic gates between process systems, such as 'surge lines' of pressurized water reactors (PWRs).

Category C

This category is characterized by:

- no signal inputs of 'intelligence'
- no external power sources or forces; but
- moving mechanical parts, whether or not moving working fluids are also present.

Examples of safety features included in this category are emergency injection systems consisting of accumulators or storage tanks and discharge lines equipped with check valves; overpressure protection and/or emergency cooling devices of pressure boundary systems based on fluid release through relief valves; filtered venting systems of containments activated by rupture disks; and mechanical actuators, such as check valves and spring-loaded relief valves, as well as some trip mechanisms (e.g. temperature, pressure and level actuators).

Category D

This category is characterized by:

- signal inputs of 'intelligence' to initiate the passive process
- energy to initiate the process must be from stored sources such as batteries or elevated fluids
- active components are limited to controls, instrumentation and valves to initiate the passive system
- Manual initiation is excluded.

Examples of safety features included in this category are emergency core cooling and injection systems based on gravity that are initiated by battery-powered electric or electro-pneumatic valves; emergency reactor shutdown systems based on gravity or static pressure driven control rods.

The reader of the present document should consider that:

(a) The information provided shall not be taken as an advertisement for any reactor type.
(b) The description of selected design does not imply a preference relative to other water cooled reactor systems that are not described.
(c) There is no implicit recommendation that passive systems should be preferred to active systems.
(d) Nomenclature in the Annexes may not be consistent with that in the main text. Harmonization was not attempted for the text provided in the Annexes for different reactor designs.

2. PASSIVE SAFETY SYSTEMS FOR CORE DECAY HEAT REMOVAL

This section describes the types of advanced reactor passive safety systems for removing the decay heat from the core after a reactor scram. The types of passive safety systems considered for this function are:

- Pre-pressurized core flooding tanks (accumulators)
- Elevated tank natural circulation loops (core make-up tanks)
- Gravity drain tanks
- Passively cooled steam generator natural circulation
- Passive residual heat removal heat exchangers
- Passively cooled core isolation condensers
- Sump natural circulation

A brief description of each is provided in the following sections. Combinations of these systems are incorporated into the designs described in Annexes I to XX.

2.1. Pre-pressurized core flooding tanks (accumulators)

Pre-pressurized core flooding tanks, or accumulators, are used in existing nuclear power plants and they constitute part of the emergency core cooling systems. They typically consist of large tanks having about 75% of the volume filled with cold borated water and the remaining volume filled with pressurized nitrogen or an inert gas. As shown in Figure 1, the contents of the tank are isolated from the reactor coolant system (RCS) by a series of check valves that are normally held shut by the pressure difference between the RCS and the fill gas in the tank. In the event of a loss of coolant accident (LOCA), the core pressure will drop below the fill gas pressure. This results in opening the check valves and discharging the borated water into the reactor vessel. This is a Category C passive safety system for conditions mentioned above.

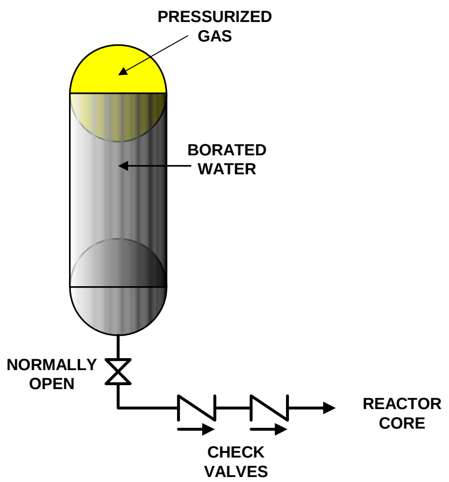

FIG. 1. Pre-pressurized core flooding tank (accumulator).

2.2. Elevated tank natural circulation loops (core make-up tanks)

Natural circulation loops represent an effective means of providing core cooling. Several advanced reactor designs implement elevated tanks connected to the reactor vessel or primary loop at the top and bottom of the tank as shown in Figure 2. The tanks are filled with borated water to provide coolant injection at system pressure. The tanks are normally isolated from the reactor vessel by an isolation valve located along the discharge line departing from the bottom of the tank itself. The fluid is always sensing full system pressure through the top connection line. In the event of an emergency, the bottom isolation valve is opened to complete the natural circulation loop and to permit cold borated water to flow to the core. In order to reduce the number of pipelines connected with the reactor pressure vessel, the delivery (or bottom) line of the core make-up tank (CMT) is in common with the emergency core coolant delivery line. In case of a number of accident scenarios, the CMT delivery can start before the accumulator delivery and end-up after the accumulator emptying. In those situations the CMT delivered flow-rate can be affected by the accumulator delivered flow-rate to a noticeable extent. Furthermore, specifically when the CMT delivery line is connected with the cold or hot leg (i.e. without the presence of the direct vessel injection), the direction of the fluid motion in the discharge line should be checked: in other terms there is the possibility that CMT liquid is used to cool the steam generator, or in any case, is diverted from its principal mission that is core cooling. This is a Category D passive safety system.

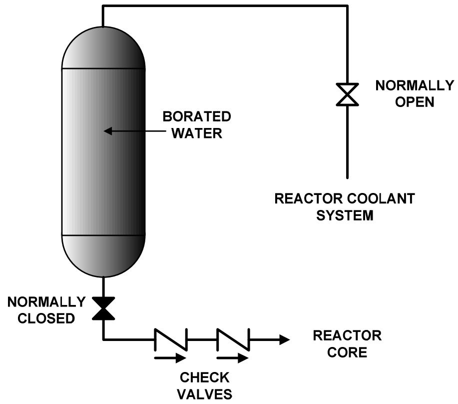

FIG. 2. Elevated tank natural circulation loops (core make-up tanks).

2.3. Elevated gravity drain tanks

Under low pressure conditions, elevated tanks filled with cold borated water can be used to flood the core by the force of gravity. In some designs, the volume of water in the tank is sufficiently large to flood the entire reactor cavity. As shown in Figure 3, operation of the system requires that the isolation valve be open and that the driving head of the fluid exceed the system pressure plus a small amount to overcome the cracking pressure of the check valves. The performance of the gravity drain tank may be limited under core uncovery conditions due to steam production in the core region. This is a Category D passive safety system.

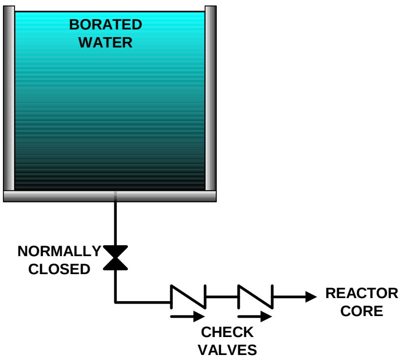

FIG. 3. Elevated gravity drain tank.

2.4. Passively cooled steam generator natural circulation

Some advanced PWR designs incorporate a system to remove decay heat passively through the steam generators. This is done by condensing steam from the steam generator inside a heat exchanger submerged in a tank of water or an air cooled system as shown in Figures 4 and 5, respectively. The system shown in Figure 4 has some similar characteristics to isolation condenser. These are Category D passive safety systems.

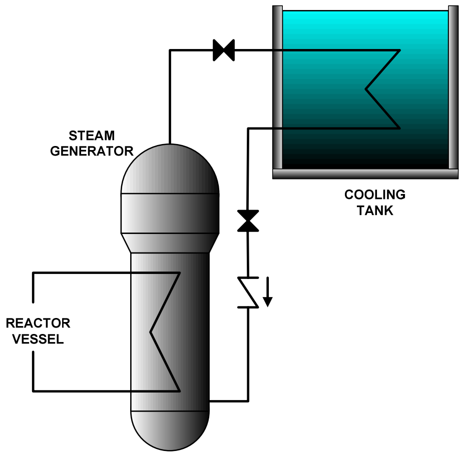

FIG. 4. Core decay heat removal using a passively cooled steam generator (water-cooled).

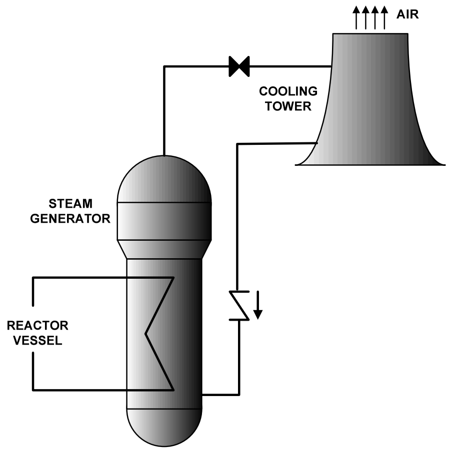

FIG. 5. Core decay heat removal using a passively cooled steam generator (air-cooled)[*].

2.5. Passive residual heat removal heat exchangers (single-phase liquid)

Passive residual heat removal (PRHR) heat exchangers are incorporated into several advanced PWR designs. Their primary function is to provide extended periods of core decay heat removal by transferring heat using a single-phase liquid natural circulation loop as shown in Figure 6. The PRHR heat exchanger loop is normally pressurized and ready for service. Single-phase liquid flow is actuated by opening the isolation valve at the bottom of the PRHR heat exchanger. The PRHR system design is optimized for single-phase (contrary to isolation condenser which is optimized for boiling and condensation) liquid heat transfer. It is particularly useful in mitigating the station blackout scenario. In general it eliminates the need for ‘bleed and feed’ operations for plant cool-down. This is a Category D passive safety system.

[*] This CRP does not deal with natural draft air flow cooling of tubes. Figure 5 is added for completeness.

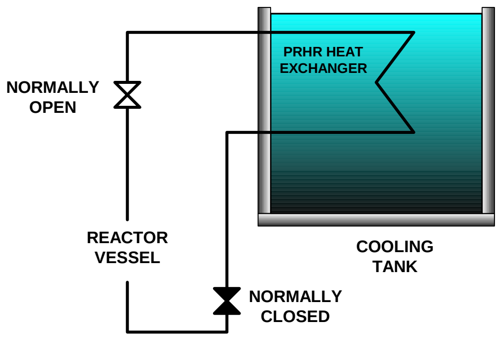

FIG. 6. Core decay heat removal using a water-cooled passive residual heat removal heat exchanger loop.

2.6. Passively cooled core isolation condensers (steam)

Passively cooled core isolation condensers are designed to provide cooling to a boiling water reactor (BWR) core subsequent to its isolation from the primary heat sink, the turbine/condenser set. As shown in Figure 7, during power operations, the reactor is normally isolated from the isolation condenser (IC) heat exchanger by closed valves. In the event that the core must be isolated from its primary heat sink, the valves located in the IC lines are opened and main steam is diverted to the IC heat exchanger where it is condensed in the vertical tube section. Heat is transferred to the atmosphere through the heat exchanger and the ICS/PCCS pool. The condensate returns to the core by gravity draining inside the tubes. This is a Category D passive safety system.

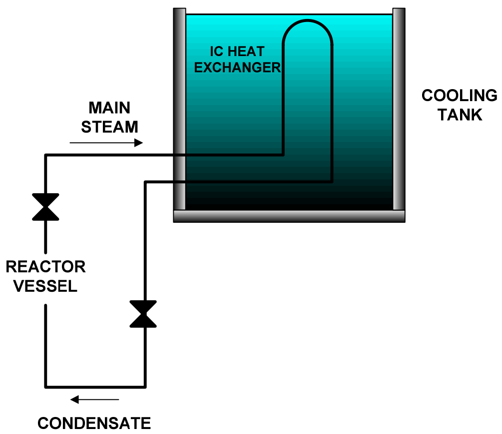

FIG. 7. Isolation condenser cooling system.

2.7. Sump natural circulation

Some designs utilize the reactor cavity and other lower containment compartments as a reservoir of coolant for core cooling in the event of a break in the primary system. As such, water lost from the reactor system is collected in the containment sump. Eventually the reactor is completely immersed in water and the isolation valves are opened. Decay heat removal occurs by boiling in the core. The steam generated in the core travels upward through an automatic depressurization system (ADS) valve that vents directly into containment. The density difference established in the situation depicted in Figure 8 between the core region and the pool produces a natural circulation flow that draws water up through the sump screen into the reactor vessel and is adequate in removing the decay heat. In some design cases, natural circulation inside the reactor vessel may be sufficient to remove decay heat without the need of ADS operation. This is a Category D passive safety system.

Annexes I through XX present descriptions of how different variations of these systems work in combination in various advanced water-cooled NPP designs to provide core cooling after a reactor scram.

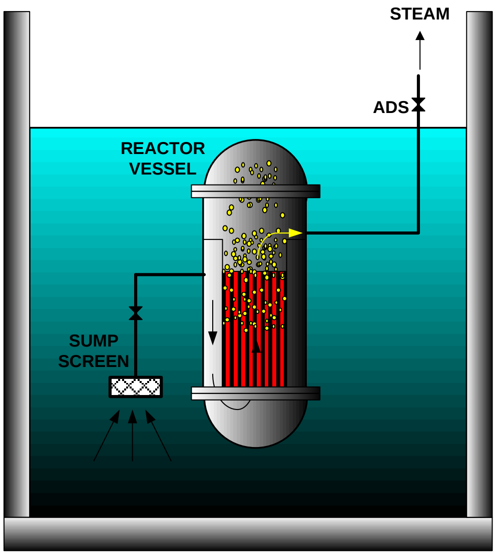

FIG. 8. Core cooling by sump natural circulation

3. PASSIVE SAFETY SYSTEMS FOR CONTAINMENT COOLING AND PRESSURE SUPPRESSION

This section describes the types of advanced reactor passive safety systems for removing the heat from the containment and reducing pressure inside containment subsequent to a loss of coolant accident. The types of passive safety systems being incorporated for this function are:

- Containment pressure suppression pools
- Containment passive heat removal/pressure suppression systems
- Passive containment spray

A brief description of each passive safety system for pressure suppression and containment cooling is provided in the following sections. Combinations of these systems are incorporated into the designs described in the Annexes I to XX.

3.1. Containment pressure suppression pools

Containment pressure suppression pools have been used in BWR designs for many years. Figure 9 presents a generic concept of a suppression pool. Following a LOCA, steam is generated into the drywell (the primary containment) following vaporization of liquid and/or steam expansion, both of these coming from the primary system typically due to a break. From drywell the steam-non condensable mixture is subsequently forced through large vent lines submerged in the water in the suppression pools. The steam condenses, thus mitigating a pressure increase in the containment. This is a Category B and C passive safety system.

3.2. Containment passive heat removal/pressure suppression systems

This type of passive safety system uses an elevated pool as a heat sink. Steam vented in the containment will condense on the containment condenser tube surfaces to provide pressure suppression and containment cooling. Three variations of the concept are presented in Figures 10 to12. In the first variation of the concept, Figure 10, an air heat exchanger (HEX) is connected with a pool located on the top of the containment. Single phase liquid is expected to flow inside the HEX driven by gravity gradient caused by the inclination of the same HEX. Experiments have been performed to prove the validity of the solution. In the second variation of the concept, Figure 11, a closed loop filled with single phase liquid connects an air HEX and a pool-type HEX. Natural circulation and heat removal capability are generated when the air HEX receives heat from the containment: this occurs through liquid heating and stratification that produces a difference between densities in the rising and descending leg of the pool-type HEX. In the third variation of the concept, Figure 12, two different zones of the containment, typically characterized by different pressures in case of accident (pressure is the same during normal operation), are connected with the rising and the descending side of a pool-type HEX. In this case, the steam-air mixture is the working fluid with condensate in the descending leg. Driving forces may be lower than in the previous cases and working condition may not be stable over a reasonably wide range of conditions. Positive driving forces may be low in all three cases and careful system engineering is needed. These passive safety systems are of Categories B and D.

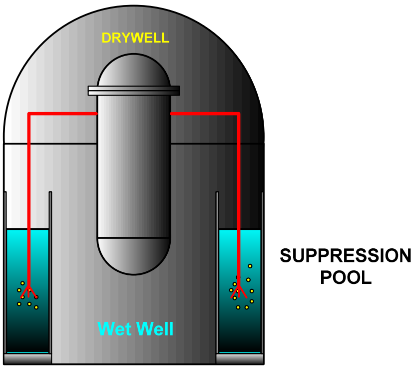

FIG. 9. Containment pressure reduction following a LOCA using steam condensation in suppression pools.

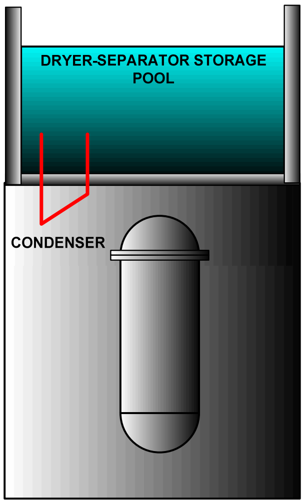

FIG. 10. Containment pressure reduction and heat removal following a LOCA using steam condensation on condenser tubes.

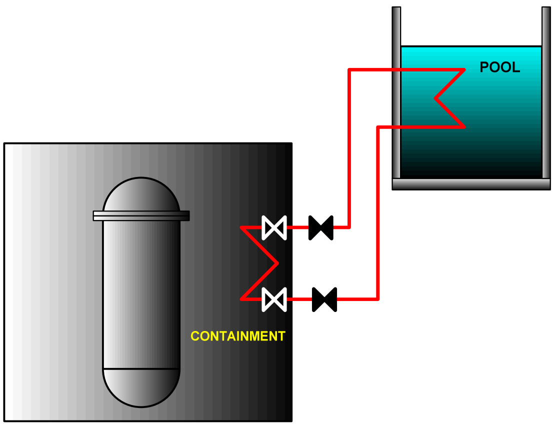

FIG. 11. Containment pressure reduction and heat removal following a LOCA using an external natural circulation loop.

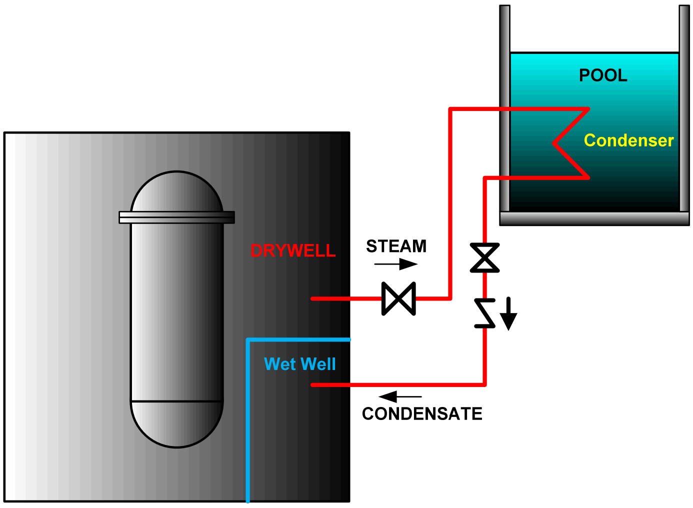

FIG. 12. Containment pressure reduction and heat removal following a LOCA using an external steam condenser heat exchanger.

3.3. Passive containment spray systems

Figure 13 shows a design that implements a natural draft air cooled containment. Subsequent to a LOCA, steam in contact with the inside surface of the steel containment is condensed. Heat is transferred through the containment wall to the external air. An elevated pool situated on top of the containment provides a gravity driven spray of cold water to provide cooling in a LOCA scenario. The air flow for the cooling annulus, that is generated by a chimney-like type effect, is a Category B passive safety system. The containment vessel sprays are a Category D passive safety system.

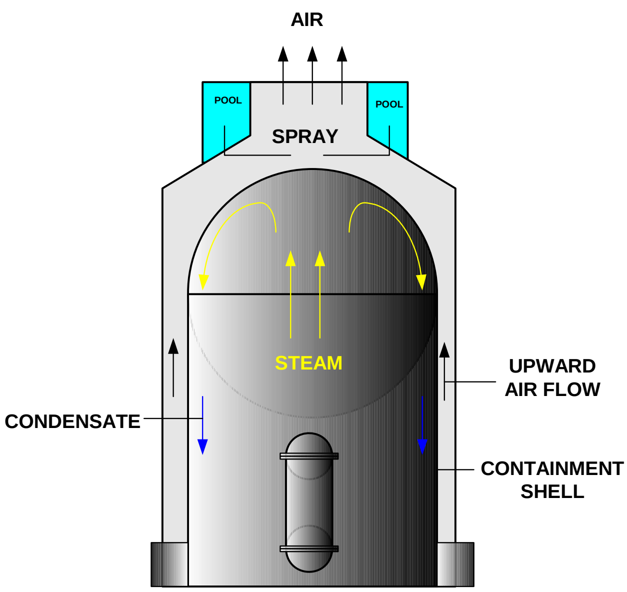

FIG. 13. Containment pressure reduction and heat removal following a LOCA using a passive containment spray and natural draft air.

4. PASSIVE SAFETY SYSTEM PHENOMENA

The geometrical and layout characterization of passive safety systems constitutes the key subject for Sections 2 and 3. The thermal-hydraulic characterization for the same systems requires the consideration of phenomena, i.e. passive safety system phenomena.

Therefore the aim of Section 4 is twofold:

- To classify thermal-hydraulic phenomena for passive systems;
- To establish a correlation between systems (as described in Sections 2 and 3) and phenomena (bullet above).

4.1. Thermal-hydraulic phenomena

Thermal-hydraulic phenomena and related parameter ranges that characterize the performance of passive systems do not differ, in general, from phenomena that characterize the performance of systems equipped with active components. This is specifically true for transient conditions occurring during safety relevant scenarios.

In other words, one can say that friction pressure drops or heat transfer coefficients are affected by local velocity and void fraction and not by the driving force that establishes those conditions, e.g. gravity head or centrifugal pump. The same can be repeated for more complex phenomena like two phase critical flow or counter-current flow limiting.

Thus, a large number of thermal-hydraulic phenomena that are expected to occur in passive systems during accident are classified in the OECD/NEA/CSNI documents ‘separate effect’ (SE) and ‘integral effect’ (IE) reported as references 19 and 20, hereafter. However, specific layout of passive systems and combination of parameter ranges brought the need of expanding the original list of phenomena in the same references 19 and 20. This was done in reference 21, where, mainly the passive systems proposed at the time of issuing of the report (1996) were considered.

The ‘expanded’ OECD/NEA list of phenomena for passive systems was up-graded in IAEA CRP on Natural Circulation Phenomena, Modeling and Reliability of Passive Safety Systems that Utilize Natural Circulation, considering the recently proposed passive systems by the industry. The description of the individual phenomena is given in the Appendix for the sake of completeness.

The identification and characterization of additional (i.e. with reference to the original SE and IE lists) phenomena for passive systems is presented in Table 1, which includes two main columns, other than the first column with numbering, which is consistent with the phenomena numbering in reference 19:

- Column 2: phenomena identification;
- Column 3: phenomena characterization based upon the individual phenomena description in the Appendix, considering the key layout of systems described in Sections 2 and 3.

The content of Table 1 is self-standing and directly understandable including the supporting description provided in the Appendix (as already mentioned). However, the following additional items should be noted:

- Specific geometry configurations or range of variations of affecting thermal-hydraulic parameters justify the presence of phenomena at rows 2, 5, 6 7 and 14 in both the present list and the list in reference 19. This is specifically true in the case of phenomenon 6 (natural circulation) that is expected to occur whenever a gravity environment exists.
- Natural circulation is also at the origin of the Core-make up Tank performance, phenomenon at row 15 in Table 1. However, the simultaneous presence of stratification in the tank, the possible condensation with level formation inside the tank, the specific loop connection and the values of boundary and initial conditions, suggest the consideration of a separate phenomenon.

TABLE 1. IDENTIFICATION AND CHARACTERIZATION OF PHENOMENA FOR PASSIVE SAFETY SYSTEMS

Phenomena identification		Characterizing thermal-hydraulic aspect
1	Behaviour in large pools of liquid	Thermal stratification
		Natural/forced convection and circulation
		Steam condensation (e.g. chugging, etc.)
		Heat and mass transfer at the upper interface (e.g. vaporization)
		Liquid draining from small openings (steam and gas transport)
2	Effects of non-condensable gases on condensation heat transfer	Effect on mixture to wall heat transfer coefficient
		Mixing with liquid phase
		Mixing with steam phase
		Stratification in large volumes at very low velocities
3	Condensation on containment structures	Coupling with conduction in larger structures
4	Behaviour of containment emergency systems (PCCS, external air cooling, etc.)	Interaction with primary cooling loops
5	Thermo-fluid dynamics and pressure drops in various geometrical configurations	3-D large flow paths e.g. around open doors and stair wells, connection of big pipes with pools, etc.
		Gas liquid phase separation at low Re and in laminar flow
		Local pressure drops
6	Natural circulation	Interaction among parallel circulation loops inside and outside the vessel
		Influence of non-condensable gases
		Stability
		Reflux condensation
7	Steam liquid interaction	Direct condensation
		Pressure waves due to condensation
8	Gravity driven cooling and accumulator behaviour	Core cooling and core flooding
9	Liquid temperature stratification	Lower plenum of vessel
		Down-comer of vessel
		Horizontal/vertical piping
13	Behaviour of emergency heat exchangers and isolation condensers	Low pressure phenomena
14	Stratification and mixing of boron	Interaction between chemical and thermo-hydraulic problems
		Time delay for the boron to become effective in the core
15	Core make-up tank behaviour	Thermal stratification
		Natural Circulation

- The phenomenon at row 3 is a containment related phenomenon: the phenomena discussed in the (OECD/CSNI) report at reference 22 should be connected for completeness with the present one.
- The phenomenon at row 9 is also relevant for characterizing the phenomenon at row 1: geometry peculiarities and boundary conditions suggest keeping two separate phenomena in the present list.
- The accumulator behaviour is at the origin of an individual phenomenon considered in reference 17. Furthermore, the accumulator performance is driven by gas pressure. However, other than the ‘passive nature’ for the component behaviour, similarity in geometrical configuration and in the ranges of variations of relevant parameters suggested to consider ‘accumulator behaviour’ together with ‘gravity flooding’.
- The list of relevant thermal-hydraulic aspects in the 3rd column of the table can be expanded consistently with descriptions in Sections 2 and 3, and, also, in the Appendix: an effort to make fully comprehensive or exhaustive, the contents of the information in this column has not been attempted.

4.2. Passive system design and thermal-hydraulic phenomena

The cross connection between passive systems described in Sections 2 and 3 and phenomena identified in Section 4.1 can be derived from Table 2, namely considering the first and the last column. This is consistent with the second bullet under the heading of the present section as elements of aims of Section 4.

In addition, an effort has been made to homogenize the nomenclature adopted by different designers: having defined the type of passive safety system (for core decay heat removal and for containment cooling and pressure suppression) and the related (key) phenomena, in column 3 of Table 2. The passive safety systems are listed and named according to the nomenclature provided by the designers (Annexes I to XX).

In the column 2 of Table 2 the passive safety systems are foreseen in ‘various advanced water cooled nuclear power plants’ (part I of annexes of the present document, i.e. Annexes I to XIII) are distinguished from ‘integral reactor systems’ (part II of annexes of the present document, i.e. Annexes XIV to XX).

As in the case of the previous table, the content of Table 2 is self-standing and directly understandable including the supporting description provided in Sections 2 and 3 and in Annexes I to XX (as already mentioned). The following additional items should be noted:

- An attempt has been made to list in the column 2 *all* passive systems installed in the reactors described in Annexes I to XX. However, the current stage of the design and the different level of detail in the descriptions prevent the possibility of an imperfection-free list. This is particularly true in relation to the accumultors (first row in the table).
- The variety of definitions adopted by designers for passive systems is wider than what is considered here.

5. SUMMARY OF REACTOR AND PASSIVE SAFETY SYSTEM CATEGORIES

The characterization of the nuclear reactor designs based on the passive (sub-) systems constitutes the objective of the present section. This can be achieved by combining the reactor descriptions given in Annexes I to XX and the passive systems identified in Sections 2 and 3. Furthermore, the ‘passive’ thermal-hydraulic phenomena characterized in Section 4 can be cross-correlated with the reactor configurations.

TABLE 2. CROSS LISTING OF THE PASSIVE SAFETY SYSTEM WITH PHENOMENA

Type of Passive Safety System	Passive Safety Systems of Advanced Designs	Related Phenomena
Pre-pressurized Core Flooding Tanks (Accumulators)[1] **- Section 2.1 -**	Accumulators (AP-1000) ECCS accumulator subsystem (WWER-640/V-407) First stage hydro-accumulators (WWER-1000/V-392) Advanced accumulators (APWR+) Standby liquid control system (ESBWR) Accumulator (AHWR) Emergency core coolant tanks (SMART)	8,2,5
Elevated Tank Natural Circulation Loops (Core Make-up Tanks) **- Section 2.2 -**	Core make-up tanks (AP-1000) Second stage hydro-accumulators (WWER-1000/V-392) Core make-up tanks (ACR-1000) Core make-up tanks (SCWR-CANDU) Emergency boration tanks (IRIS)	8,6,9,5,15
Elevated Gravity Drain Tanks **- Section 2.3 -**	Core flooding system (SWR 1000) IRWST injection (AP-1000) ECCS tank subsystem – Elevated hydro-accumulators open to the containment (WWER-640/V-407) Gravity-driven cooling system (SBWR and ESBWR) Suppression pool injection (SBWR and ESBWR) Gravity-driven core cooling system (LSBWR) Gravity-driven water pool (GDWP) injection (AHWR) Reserve water system (ACR-1000) Reserve water system (SCWR-CANDU) Containment suppression pool injection (IRIS)	8,5
Passively Cooled Steam Generator Natural Circulation (water cooled) **- Section 2.4 -**	SG passive heat removal system (WWER-640/V-407) Passive residual heat removal system (SMART) Emergency decay heat removal system (PSRD) Stand-alone direct heat removal system (IMR) Passive emergency heat removal system (IRIS)	13,1,6
Passively Cooled Steam Generator Natural Circulation (air cooled) **- Section 2.4 -**	Passive residual heat removal system via SG (WWER-1000/V-392) Passive core cooling system using SG - open loop (APWR+) Stand-alone direct heat removal system – late phase (IMR)	6,4
Passive Residual Heat Removal Heat Exchangers **- Section 2.5 -**	Passive residual heat removal system (AP-1000) Passive moderator cooling system – inside insulated PT without CT (SCWR-CANDU) Residual heat removal system on primary circuit (SCOR)	13,6,2,1

[1] Accumulators constitute the design practices of the operating water cooled reactors.

Passively Cooled Core Isolation Condensers **- Section 2.6 -**	Emergency condensers (SWR 1000) Isolation condenser system (SBWR and ESBWR) Passive reactor cooling system (ABWR-II) Isolation condenser (RMWR) Isolation condenser (AHWR) Residual heat removal system (CAREM)	13,6,1
Sump Natural Circulation **- Section 2.7 -**	Lower containment sump recirculation (AP-1000) Primary circuit un-tightening subsystem (WWER-640/V-407) ADS-steam vent valves and submerged blow-down nozzles (MASLWR)	6,1
Containment Pressure Suppression Pools **- Section 3.1 -**	ADS 1-3 steam vent into IRWST (AP-1000) Automatic depressurization through safety relief valves – vent into suppression pool (SBWR and ESBWR) Steam vent into suppression pool through SRV and DPV (LSBWR) Steam vent into suppression pool through safety valves (CAREM) Steam dump pool (SCOR) Containment pressure suppression system (SCOR) Steam vent into suppression pool through ADS (IRIS)	1,7,3
Containment Passive Heat Removal/Pressure Suppression Systems (Steam Condensation on Condenser Tubes) **- Section 3.2 -**	Containment cooling condensers (SWR 1000) Passive containment cooling system (AHWR)	4,1,2,3
Containment Passive Heat Removal/Pressure Suppression Systems (External Natural Circulation Loop) **- Section 3.2 -**	Containment passive heat removal system (WWER-640/V-407) Containment water cooling system (PSRD)	4,1,2,3
Containment Passive Heat Removal/Pressure Suppression Systems (External Steam Condenser Heat Exchanger) **- Section 3.2 -**	Passive containment cooling system (SBWR and ESBWR) Passive containment cooling system (ABWR-II) Passive containment cooling system (RMWR)	4,1,2,3
Passive Containment Spray Systems **- Section 3.3 -**	Passive containment cooling system (AP-1000) Passive containment cooling system (LSBWR) Containment cooling spray (ACR-1000) Containment cooling spray (SCWR-CANDU)	3,2,4

All of these is achieved by Tables 3 and 4 that make reference to two reactor categories, respectively:

(a) PWR, BWR and SCWR (Super Critical Water Cooled Reactor) systems, Annexes I to XIII;
(b) Integral Reactor Systems, Annexes XIV to XX.

The main information in Tables 3 and 4 connects the reactor type with the passive safety systems, e.g. column 1 and 4. Thermal-hydraulic phenomena are cross-connected with specific passive safety systems in columns 4 and 5. Finally columns 2 and 3 provide elements, as an example, namely the thermal power and the 'boiling' or 'pressurized' feature, that characterize the reactor system.

'Proven' technology reactors, i.e. with final design already scrutinized in a formal safety review process, or under construction, or with an already built and operated prototype, are listed in Table 3, with a few exceptions constituted by the RMWR, the LSBWR and the SCWR that are at different levels of early design stages.

TABLE 3. PWR, BWR AND SCWR SYSTEMS AND TYPES OF PASSIVE SAFETY SYSTEMS

Reactor System	**Reactor Type**	**Power (MW•th)**	**Passive Safety Systems**	**Related Phenomena**[2]
SWR 1000 AREVA, France	BWR	2778	Emergency Condenser System	13,6,1
			Core Flooding System	8,5
			Containment Cooling Condensers	4,1,2,3
Advanced Passive PWR AP 600 and AP 1000 Westinghouse Electric, USA	PWR	1940 3415	Passive Residual Heat Removal System	13,6,2,1
			Core Make-up Tanks	8,6,9,5,15
			Automatic Depressurization System 1-3 Steam Vent into IRWST	1,7,3
			Accumulator Tanks	8,2,5
			In-containment Refuelling Water Storage Tank Injection	8,5
			Lower Containment Sump Recirculation	6,1
			Passive Containment Cooling System	3,2,4
WWER-640/407 Atomenergoproject/Gidropress, Russian Federation	PWR	1800	ECCS Accumulator Subsystem	8,2,5
			ECCS Tank Subsystem	8,5
			Primary Circuit Un-tightening Subsystem	6,1
			Steam Generator Passive Heat Removal System	13,1,6
			Containment Passive Heat Removal System	4,1,2,3
WWER-1000/392 Atomenergoproject/Gidropress, Russian Federation	PWR	3000	First Stage Hydro-accumulators	8,2,5
			Second Stage Hydro-accumulators	8,6,9,5,15
			Passive Residual Heat Removal System via Steam Generator	6,4

[2] See Section 4 for characterization of the phenomena influencing natural circulation.

Advanced PWR (APWR+) Mitsubishi, Japan	PWR	5000	Passive Core Cooling System using Steam Generator	6,4
			Advanced Accumulators	8,2,5
Simplified Boiling Water Reactor (SBWR) General Electric, USA	BWR	2000	Gravity Driven Cooling System	8,5
			Suppression Pool Injection	8,5
			Isolation Condenser System	13,6,1
			Passive Containment Cooling System	4,1,2,3
			ADS-SRV Vent into Suppression Pool	1,7,3
Economic Simplified Boiling Water Reactor (ESBWR) General Electric, USA	BWR	4500	Gravity Driven Cooling System	8,5
			Suppression Pool Injection	8,5
			Isolation Condenser System	13,6,1
			Standby Liquid Control System	8,2,5
			Passive Containment Cooling System	4,1,2,3
			ADS-SRV Vent into Suppression Pool	1,7,3
Advanced BWR (ABWR-II) Tokyo Electric Power Company (TEPCO), General Electric, Hitachi and Toshiba, Japan	BWR	4960	Passive Reactor Cooling System	13,6,1
			Passive Containment Cooling System	4,1,2,3
Reduced-Moderation Water Reactor (RMWR) Japan Atomic Energy Agency (JAEA), Japan	BWR	3926	Isolation Condenser System	13,6,1
			Passive Containment Cooling System	4,1,2,3
Advanced Heavy Water Reactor (AHWR) Bhabha Atomic Research Centre, India	HWR	750	Gravity Driven Water Pool Injection	8,5
			Isolation Condenser System	13,6,1
			Accumulator	8,2,5
			Passive Containment Cooling System	4,1,2,3
Advanced CANDU Reactor (ACR 1000) Atomic Energy of Canada Ltd, Canada	HWR	3180	Core Make-up Tanks	8,6,9,5,15
			Reserve Water System (RWS)	8,5
			Containment Cooling Spray	3,2,4
Long operating cycle Simplified Boiling Water Reactor (LSBWR) Toshiba, Japan	BWR	900	Gravity Driven Core Cooling System	8,5
			Passive Containment Cooling System	3,2,4
			Steam Vent into Suppression Pool through SRV and DPV	1,7,3
SCWR-CANDU Atomic Energy of Canada Ltd, Canada	SCWR	2540	Core Make-up Tanks	8,6,9,5,15
			Reserve Water System	8,5
			Passive Moderator Cooling System	13,6,2,1
			Containment Cooling Spray	3,2,4

Integral type reactors are considered in Table 4. All of these are of PWR type, second column, and can be assumed to constitute a special class of Light Water Reactors (LWR). In integral PWR, the major components of the nuclear steam supply system (NSSS) such as the core, steam generators, main coolant pumps, and pressurizer are integrated into a reactor vessel without any pipe connections between those components. This makes integral PWR systems relatively compact.

As a difference from the reactors listed in Table 3, all the integral reactor systems in Table 4 are in a design stage and no-one of such design has undergone a comprehensive safety scrutiny process (i.e. the licensing). However, in some cases, e.g. CAREM and to a lower extent IRIS, the reactor systems are under design since couple of decades, thus testifying the technological difficulties encountered for the exploitation of the integral nuclear reactor configuration idea.

TABLE 4. INTEGRAL REACTOR SYSTEMS AND TYPES OF PASSIVE SAFETY SYSTEMS

Integral Reactor System	**Reactor Type**	**Power (MW•th)**	**Passive Safety Systems**	**Related Phenomena**[3]
System-Integrated Modular Advanced ReacTor (SMART) Korea Atomic Energy Research Institute, Republic of Korea	PWR	330	Passive Residual Heat Removal System	13,1,6
			Emergency Core Coolant Tank	8,2,5
CAREM CNEA National Atomic Energy, Argentina	PWR	100	Residual Heat Removal System – Emergency Condenser	13,6,1
			Steam Vent into Suppression Pool through Safety Valves	1,7,3
Multi-Application Small Light Water Reactor (MASLWR) INL, OSU, Nexant, USA	PWR	150	ADS-Steam Vent Valves and Submerged Blow-down Nozzles	6,1
Passive Safe Small Reactor for Distributed Energy Supply System (PSRD) Japan Atomic Energy Agency (JAEA), Japan	PWR	100	Emergency Decay Heat Removal System	13,1,6
			Containment Water-Cooling System	4,1,2,3
Integrated Modular Water Reactor (IMR) Mitsubishi, Japan	PWR	1000	Stand-alone Direct Heat Removal System	13,1,6
			Stand-alone Direct Heat Removal System-Late Phase	6,4
Simple COmpact Reactor (SCOR) Commissariat à l'Energie Atomique, France	PWR	2000	Residual Heat Removal System on Primary Circuit RRP	13,6,2,1
			Steam Dump Pool	1,7,3
			Containment Pressure-Suppression System.	1,7,3
IRIS Westinghouse Electric, USA	PWR	1000	Passive Emergency Heat Removal System (EHRS)	13,1,6
			Emergency Boration Tanks (EBT)	8,6,9,5,15
			Containment Suppression Pool Injection	8,5
			Steam Vent into Suppression Pool through ADS	1,7,3

[3] See Section 4 for characterization of the phenomena influencing natural circulation

6. CONCLUSIONS

Passive systems are widely considered in 'innovative' or advanced nuclear reactor designs and are adopted for coping with critical safety functions. The spread and the variety of related configurations are outlined in the present document.

Twenty 'innovative' nuclear reactors are described, specially giving emphasis to the passive safety systems, in the annexes and distinguished in two groups; (see also Tables 3 and 4):

- Advanced water cooled nuclear power plants,
- Integral reactor systems.

The levels of development, or even the actual deployment of the concerned reactor designs (i.e. equipped with passive systems) for electricity production are very different, and the range of maturity of these extend from reactors already in operation to preliminary reactor designs which are not yet submitted for a formal safety review process.

A dozen different passive system types, having a few tens of reactor specific configurations, suitable to address safety functions in primary loop or in containment have been distinguished, as in Table 2. These include systems like the core make-up tanks, the containment spray cooling and the isolation condenser.

The thermal-hydraulic performance of the passive systems has been characterized by less than a dozen key phenomena at their time characterized through specific descriptions including a few tens of relevant thermal-hydraulic aspects, see Table 1 and the Appendix. Cross correlations between key thermal-hydraulic phenomena, reactor specific safety systems and 'innovative' nuclear plants have also been established (See Tables 2, 3, and 4).

There is the need to demonstrate the understanding of the key thermal-hydraulic phenomena that are selected for characterizing the performance of passive systems: this implies the identification of parameter ranges, the availability of proper experimental programs and the demonstration of suitable predictive capabilities for computational tools.

Comprehensive experimental and code development research activities have been conducted, also very intensely at an international level, in the past three to four decades in relation to the understanding of thermal-hydraulic phenomena and for establishing related code predictive capabilities for existing nuclear power reactors. In the same context, research activities also addressed some of the phenomena for passive systems. However, **a systematic effort for evaluating the level of understanding of thermal-hydraulic phenomena for passive systems and connected code capabilities appears to be limited and in general lacking**.

Appendix

DEFINITIONS OF PHENOMENA ASSOCIATED WITH NATURAL CIRCULATION[4]

Phenomena have been classified into two categories (a) phenomena occurring during interaction between primary system and containment; and (b) phenomena originated by the presence of new components and systems or special reactor configurations. This classification considers the information provided in the CSNI Report [25] which has been developed for the primary systems having in mind the safety assessment, and is intended to provide complementary aspects that are relevant to advanced water-cooled nuclear power plant designs, including containment designs. Therefore the descriptions given below are intended to supplement those in the CSNI Report.

Behaviour in large pools of liquid

Large pools of water (e.g. up to several thousand cubic meters) at near atmospheric pressure are incorporated into several advanced designs. These large pools provide a heat sink for heat removal from the reactor or the containment by natural circulation, as well as a source of water for core cooling. Examples include the pressure suppression pool (wet-well) of the ESBWR, the in-containment refuelling water storage tank of the AP-1000, the pool of the emergency condenser of the SWR-1000 and the gravity driven water pool of the AHWR.

Large pools may have a very wide spectrum of geometric configurations. Heat transfer in a limited zone in terms of volume (e.g. by condensing injected steam or by heat transfer from an isolation condenser) does not imply homogeneous or nearly homogeneous temperature in the pool. Three-dimensional convection flows develop affecting the heat transfer process, which results in a temperature stratification.

Steam generated by heat transfer or following injection may be released from the pool into the containment and influences the increase of the containment pressure. Compared to a homogeneous temperature distribution, the fluid at the top of the pool may reach the saturation temperature while the bulk fluid is sub-cooled. The evaporation from the top of the pool results in a pressure increase in the containment. Therefore the temperature stratification influences plant design. The three-dimensional nature of the temperature stratification requires appropriate modelling.

Effects of non-condensable gases on condensation heat transfer

Condensation occurs when the temperature of vapor is reduced below its saturation temperature. Presence of even a small amount of Non-condensable gas (e.g. air, N_2, H_2, He, etc.) in the condensing vapor leads to a significant reduction in heat transfer during condensation. The buildup of non-condensable gases near the condensate film inhibits the diffusion of vapor from the bulk mixture to the liquid film. The net effect is to reduce the effective driving force for heat and mass transfer. This phenomenon is the concern of industrial applications and nuclear reactor systems.

In nuclear plants, the condensation of steam in the presence of non-condensable gas becomes an important phenomenon during LOCA (loss of coolant accident) when steam released from the coolant system mixes with the containment air. Besides this, nitrogen gas in accumulators is a source of non-condensable gas, which can affect the condensation heat transfer inside the steam generator tubes of nuclear power plants, and may effect the core make-up tank performance. The effect of non-condensable gases on condensation heat transfer is also relevant to certain decay heat removal systems in advanced reactor designs, such as passive containment cooling systems.

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[4] The starting point for identification of phenomena associated with natural circulation was the OECD-CSNI report ‘Relevant Thermal Hydraulic Aspects of Advanced Reactor Design’ [26].

The effect of non-condensable gases on the condensation of steam has been extensively studied for both natural and forced convection flows. In each of them, geometries of interest (e.g. tubes, plates, annulus, etc.) and the flow orientation (horizontal, vertical) can be different for various applications. The condensation heat transfer is affected by parameters such as mass fraction of non-condensable gas, system pressure, gas/vapor mixture Reynolds number, orientations of surface, interfacial shear, Prandtl number of condensate, etc. Multi-component non-condensable gases can be present.

Condensation on containment structures

This phenomenon involves heat and mass transfer from the containment atmosphere towards the surrounding structures. This phenomenon would occur in existing reactors in case of a coolant release into the containment. It also occurs in advanced designs where containment surfaces are cooled externally, usually by natural mechanisms. Good examples are the designs of the AP series by Westinghouse, where the steel containment is cooled externally by water flowing on its exterior surface from a reservoir above the containment, and by ascending air driven by bouyancy.

Steam condensation is largely affected by conditions which can be split into two groups depending on the relevance of the physical dimensions of the system. The ‘scale-independent factors’ are variables like the fraction of non-condensables, the pressure, the gas composition and so on, the effect of which could be well investigated through separate effect tests. The ‘scale-dependent factors’ are those phenomena that require to be investigated in actual or scaled geometries (i.e. Integral Effect Tests) since physical dimensions largely influence their quantitative effect. Examples of this kind are the natural convection process at both sides of the metallic structures and the potential gas stratification.

Behaviour of containment emergency systems

Nuclear power reactor containments are equipped with safety systems which protect the containment integrity under various accident conditions. The focus of this phenomenon is the natural circulation cooling and heat transfer in various containment passive cooling systems under accident conditions to remove the energy out of the containment by natural circulation and condensation heat transfer. Typical systems are the tube condensers such as the passive containment cooling system (PCCS) and external air cooling system or external liquid film cooling and internal condensation of steam in the containment by natural circulation. The major purpose of these containment systems is to protect the containment under both design basis accidents and severe accidents involving serious core damages and to prevent the significant release of radioactive materials to the atmosphere. These systems are required to remove the load on the containment from the LOCAs and other accidents by removing the heat but containing the mass within the structure. Most of load comes from the released steam from the primary coolant system due to the LOCA or venting of the pressure relief valves. The major part of the non-condensable gases consists of the original containment atmosphere such as air or nitrogen, however with the core damage, hydrogen or fission gases can be also released into the containment atmosphere. The thermal-hydraulic phenomena of importance are tube surface condensation with non-condensable gases, natural circulation of steam and non-condensable gases, degradation of condensation by the accumulation of non-condensable gases and purging of non-condensable gases from condenser systems. The passive containment cooling system can be vertical or horizontal tube condensers in external water pool, exposed condenser tube system in the containment cooled by natural circulation water through the tubes from the external pool or by external air circulation and others.

Thermo-fluid dynamics and pressure drops in various geometrical configurations

Pressure drop is the difference in pressure between two points of interest in a fluid system. In general, pressure drop can be caused by resistance to flow, changes in elevation, density, flow area and flow direction. Pressure drops in natural circulation systems play a vital role in their steady state, transient and stability performance.

It is customary to express the total pressure drop in a flowing system as the sum of its individual components such as distributed pressure loss due to friction, local pressure losses due to sudden variations of shape, flow area, direction, etc. and pressure losses (the reversible ones) due to acceleration (induced by flow area variation or by density change in the fluid) and elevation (gravity effect). An important factor affecting the pressure loss is the geometry. In a nuclear reactor, we have to deal with several basic geometrical shapes (circular pipes, annuli, etc.) and a number of special devices like rod bundles, heat exchangers, valves, headers, plenums, pumps, large pools, etc. Other factors are concerned with the fluid status (single or two phase/one component, two-component or multi-component), the flow nature (laminar or turbulent), the flow pattern (bubbly, slug, annular, etc.), the flow direction (vertical upflow, downflow, inclined flow, horizontal flow, countercurrent flow, etc.), flow type (separated and mixed), flow paths (one-dimensional or multi-dimensional, open or closed paths, distributor or collector), and the operating conditions (steady state or transient).

An important focus of this phenomenon is the geometric conditions that hinder the establishment of fully developed flow specially when the fluid in question is a mixture of steam, air and water. This complex thermo-fluid dynamic phenomenon warrants special attention. However, it is worth mentioning here that though in many systems like the primary system of a nuclear power plant, flow is mostly not fully developed, pressure drop relationships used in these systems are invariably those obtained for developed flow. This practice is also experimentally proved to be more than adequate in most of the cases. However, in some specific cases like containment internal geometry, it is necessary to consider thermo fluid dynamics in the developing region.

A final, very important issue, is concerned with the driving force depending on whether the flow is sustained by a density difference in the fluid (natural circulation) or by a pump (forced convection), or whether there will be feedback between the pressure loss and the extracted power or not. Normally the pressure loss inside a device depends on the nature of flow through the device and not on the nature of driving head causing the flow. However, under some circumstances, because of local effects, the pressure loss may get influenced by the nature of driving force.

Natural circulation

The complex set of thermal-hydraulic phenomena that occur in a gravity environment when geometrically or materially distinct heat sinks and heat sources are connected by a fluid can be identified as Natural Circulation (NC). No external sources of mechanical energy for the fluid motion are involved when NC is established.

The above definition includes the situations of a heater immersed into a fluid, of an open flame in the air, of a chimney driven fire, of insurge of hot fluid into a pool of cold liquid, and of a heat source and sink (e.g. heater and cooler) consisting of separated mechanical components connected by piping and situated at different gravity elevations. Natural circulation also drives the occurrence of stratification in horizontal pipes.

Within the scope of this document, this phenomenon involves the following system configurations:

(a) Heat source and sink of primary loop constituted, respectively, by core and steam generator, or boiler, or primary side of heat exchanger, with core located at a lower elevation;

(b) Heat source and sink inside the pressure vessel, constituted, respectively, by core and (typically annular-like region of) vessel downcomer. ‘Steady-state’ NC between core and downcomer occurs owing to continuous cooling of the downcomer fluid by a heat exchanger (boiler or steam generator) or by continuous inlet of feed-water liquid at a temperature lower than core outlet temperature;

(c) Cooling of the containment atmosphere by a closed loop.

In the current generation of nuclear plants, the NC core power removal capability is exploited for accident situations to demonstrate the inherent safety features of the plant (with the noticeable exception of the Dodewaard commercial BWR unit, shutdown in 1997). The natural circulation is also occurring during various phases of the refuelling.

In future generation of nuclear plants, NC is planned to be used for ensuring the nominal operating conditions and for achieving safe cooling following accidents in a wider spectrum than foreseen for current generation reactors.

Steam-liquid interactions

Large pools may have a very wide spectrum of geometric configurations. Heat transfer in one very limited zone in terms of volume (e.g. by condensing injected steam or by heat transfer from a passive containment cooler) does not imply homogeneous or nearly homogeneous temperature in the pool. Many containment phenomena require steam-liquid interface. Steam discharge into a suppression pool of boiling water reactor is a good example of this case. After break-up of the originally created bubbles in the suppression pool, the subsequent formation of bubble plumes takes place. Consequently, complete condensation occurs and this induces mixing in the pool, the process is being determined by single and two-phase natural circulation. It is important to understand the break-up and plume-stirring process and mechanisms, because the system pressure ultimately controlled by the pressure in the vapour space above the water surface in the suppression chamber. This pressure is the sum of the partial pressures of steam and gas, the former controlled by the temperature at the pool surface. In turn, the pool surface temperature depends on the efficiency of steam condensation in the pool, and the degree of mixing in the pool.

The following is a listing of the steam-liquid interactions related phenomena:

- Direct contact condensation of steam in pool water
- Bubble formation and break-up and the subsequent formation of bubble plums
- Break-up and plume-stirring process and mechanisms inducing mixing in the pool

As example for steam-liquid interactions can be given passive containment cooling (PCC) venting into the suppression pool of ESBWR and also injection of steam-gas mixture through a downcomer vent line into the suppression pool.

Gravity driven cooling and accumulator behaviour

Gravity driven cooling provides emergency core cooling water by gravity draining, in events with loss of coolant. This system requires a large volume of water above the core, plus additional depressurization capacity, so that the primary coolant system can be depressurized to allow for gravity flow from the elevated suppression pool. Since there are no large reactor vessel pipes at or below the core elevation, this design ensures that the core will remain covered by water during all design basis accidents. In general, gravity driven cooling concept is mainly based on the depressurization of the reactor pressure vessel to sufficiently low pressures to enable reflood of the core by gravity feed from an elevated pool. When the gravity driven cooling operates, the gravity drain flow rate to the reactor pressure vessel depends on the piping geometry, the state of the fluid, and the pressure conditions in both the water pool and the reactor pressure vessel. Flow entering the reactor pressure vessel during the later stages of blowdown during a postulated loss-of coolant accident (LOCA) must be sufficient to keep the nuclear core flooded. The system which provides gravity driven cooling is a simple and economical safety system.

The following is a listing of the gravity driven cooling related phenomena:

- Depressurization of the reactor pressure vessel by discharging through depressurization valves into the drywell and increase of pressure in the upper part of containment;
- Evaporation in the reactor pressure vessel due to depressurization;

- Friction in the gravity driven cooling system and injection lines including the valves in these lines;
- Large amounts of cold water immediately floods the lower parts of the reactor pressure vessel, causing:
  - Collapse of voids
  - Condensation of steam
  - Suppression of boiling
  - Increase of water level inside the reactor pressure vessel;
- Condensing of steam out of the reactor pressure vessel and drywell gas space until air accumulates on the primary sides of the passive containment cooling system, resulting in termination of steam condensation.

As examples for gravity driven cooling system can be given gravity driven cooling system (GDCS) of ESBWR and passive core flooding system (HA-2 hydraulic accumulators of the second stage) of WWER-1000/392 and passive core flooding systems (ECCS tank)of WWER-640/407.

Liquid temperature stratification

Nuclear reactors that implement natural circulation passive safety systems may produce large temperature gradients in their working fluid as a result of local cooling caused by emergency core coolant (ECC) injection or local heating caused by steam condensation or heat exchanger heat transfer. Thermal stratification arises because the low flow condition typically encountered in a natural circulation system greatly reduces the amount of fluid mixing that can occur. Examples of thermal stratification during ECC injection include the formation of cold plumes in the downcomer, and liquid thermal stratification in the lower plenum, cold legs and loop seals.

ECC injection into horizontal piping partially filled with steam also results in liquid temperature stratification. The cooler liquid condenses the steam forming a saturated layer of liquid water on top of the sub-cooled liquid layer. This saturated layer is at a higher temperature than the sub-cooled layer, resulting in a stratified temperature condition. The formation of the saturated layer may mitigate occurrences of condensation-induced water hammer (CIWH) events.

Liquid temperature stratification can also arise in passive safety systems such as the natural circulation driven core make-up tanks (CMT) and the large liquid-filled tanks that serve as the heat sink for reactor core or containment passive cooling systems. Steam vented into the large safety tanks condenses in the cold liquid producing hot rising plumes that form thermal layers at the free surface of the tank. Thermal layers having different temperatures grow with time to create a large temperature gradient in the liquid.

Behaviour of emergency heat exchangers and isolation condensers

The removal of decay heat from a nuclear core can be accomplished by passive means using either an emergency heat exchanger or an isolation condenser (IC), depending on the system design. In some advanced pressurized water reactors (PWR), the emergency heat exchanger decay heat removal system consists of a closed loop that includes a shell and tube heat exchanger immersed in a large liquid pool that is elevated above the core. The relative elevation between the heat source and heat sink creates a buoyancy-driven natural circulation flow that eliminates the need for a pump. Decay heat is removed from the core by convective heat transfer from the fuel to the single-phase liquid in the reactor vessel. The heat stored in the liquid is carried by natural circulation to the emergency heat exchanger. Heat is transferred from the fluid through the emergency heat exchanger tubes into the pool by three mechanisms; single-phase convective heat transfer at the tube inside surface, heat conduction through the tube walls, and nucleate boiling at the tube outside surface. Some advanced PWRs use the steam generator as an intermediate emergency heat exchanger with a passively cooled, natural circulation feedwater loop.

Some advanced boiling water reactors (BWR) use isolation condensers as the means of removing core decay heat. The IC consists of a shell and tube heat exchanger immersed in a large liquid pool elevated above the core. In a BWR, core decay heat is removed by nucleate boiling. The steam generated by this process is condensed inside the IC tubes creating a low pressure region inside the tubes which draws in additional steam. Thus the driving mechanism for the flow is steam condensation. Heat is transferred through the IC tubes into the pool by three mechanisms; single-phase steam condensation (phase change) at the tube inside surface, heat conduction through the tube walls, and convective heat transfer at the tube outside surface. The condensate is returned as a single-phase liquid to the reactor vessel by gravity draining. Performance of the IC can be affected by the presence of non-condensable gases.

The following is a listing of the emergency heat exchanger related local phenomena:

- emergency heat exchanger loop flow resistance
- Buoyancy force
- Single-phase convective heat transfer
- Shell-side nucleate boiling heat transfer

The following is a listing of the IC related local phenomena:

- IC loop flow resistance
- Low pressure steam condensation
- Condensation heat transfer in the presence of non-condensable gases
- Shell-side convection heat transfer
- Condensate/steam countercurrent flow limitations

Stratification and mixing of boron

Boric acid is introduced into the reactor coolant to control long term reactivity. Forced coolant circulation during normal operation ensures that the boric acid is homogeneously distributed in the reactor coolant system (RCS) so that the boron concentration is practically uniform. Decrease of the boron concentration results in an increase of the reactivity. Causes for decreasing of boron concentration are injection of coolant with less boron content from interfacing systems (external dilution) or separation of the borated reactor coolant into highly concentrated and diluted fractions (inherent dilution). Examples of external dilution are the injection of coolant of reduced boron concentration by the makeup system, and injection of low-boron pump sealing water into the primary system. Inherent dilution can occur after reflux condenser heat transfer or back flow from the secondary system in case of primary-to-secondary leakage accidents.

Operation in the reflux condenser mode over a lengthy period of time could occur in the event of small-break loss-of-coolant accidents (SB-LOCA) concurrent with limited operability of the emergency core cooling (ECC) systems. In such an event the condensate descending down the cold-leg steam generator (SG) tubing into the SG outlet plenum and from there into the pump seal could form slugs of low-boron water. On restoration of natural circulation after refilling of the reactor coolant system such slugs would be transported towards the reactor core. However, on their way to the core, they would be mixed in the cold-leg piping, the reactor pressure vessel (RPV) downcomer and the lower plenum and thus increase in boron concentration.

Restarting of a reactor coolant pump (RCP) after a SB-LOCA or a SG tube rupture (SGTR) is very unlikely to occur as such events can be clearly identified on the basis of measured data and starting of a RCP is an action which would provide several individual actions and therefore some time. Assuming that an inadvertent demineralized water injection into one loop were to occur before starting of the RCP in this loop in spite of the monitoring and measurement of the boron concentration of the water injected into the RCS, a slug of demineralized water moves towards the core inlet after pump start.

Mixing of the diluted slug with the ambient coolant of higher boron content provides the only mitigation mechanism before the slug enters the core.

The main mixing mechanism in case of the low-boron water slug accelerated by the RCP start is turbulent mixing between the fluid flows having different velocities whereas in case of re-establishing the natural circulation after a reflux condensation phase the main mixing mechanism is buoyancy driven turbulent mixing. The density differences between the fluids are due to the temperature and the boron concentration differences.

Core make-up tank behaviour

Several advanced reactor designs implement core make-up tanks (CMTs) to provide natural circulation cooling to the core. CMTs are elevated tanks connected to the reactor vessel and primary loop at the top and bottom of the tank. Special lines connect the bottom of the tank with the vessel, and are termed direct vessel injection (DVI). In connection to this, an important interaction occurs between the CMT, the accumulator and the IRWST also considering the actuation signal for automatic depressurization. The tanks are filled with cold borated water and can provide coolant injection at system pressure. The tanks are normally isolated from the reactor vessel by an isolation valve located at the bottom of the vessel. The fluid is always sensing full system pressure through the top connection line. In the event of an emergency, the bottom isolation valve is opened to complete the natural circulation loop and permitting cold borated water to flow to the core. The relative elevation between the core and the CMT and the density difference between the hot primary system water and the cold CMT water creates a buoyancy-driven natural circulation flow that eliminates the need for a pump. Decay heat is removed from the core by convective heat transfer from the fuel to the single-phase liquid in the reactor vessel. CMT behaviour includes natural circulation, liquid thermal stratification in the tank, and liquid flashing during plant depressurization.

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Annexes

PART I: PASSIVE SAFETY SYSTEMS FOR VARIOUS ADVANCED WATER COOLED NUCLEAR POWER PLANTS

ANNEX I. ABWR-II
TEPCO, Ge, Hitachi Ltd, Toshiba Corporation

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Advanced BWR (ABWR--II) *Tokyo Electric Power Company (TEPCO), General Electric Company, Hitachi Ltd, and Toshiba Corporation*	BWR	4960	CORE/PRIMARY: • Passive Reactor Cooling System (Isolation Condenser) CONTAINMENT: • Passive Containment Cooling System

I-1. Introduction

The ABWR-II, an evolutionary reactor based on the advanced BWR (the ABWR), is now being jointly developed by six Japanese BWR utilities led by Tokyo Electric Power Company (TEPCO), General Electric Company, Hitachi Ltd, and Toshiba Corporation.

By adopting a large electric output (1700 MW(e)), a large fuel bundle, a modified ECCS, and passive heat removal systems, among other design features, a design concept capable of increasing both economic competitiveness and safety performance has been achieved.

The key objectives of ABWR-II are further improvement in economics against alternative forms of electric generation and enhancement of safety & reliability. ABWR-II plant system features are summarized in Figure I-1.

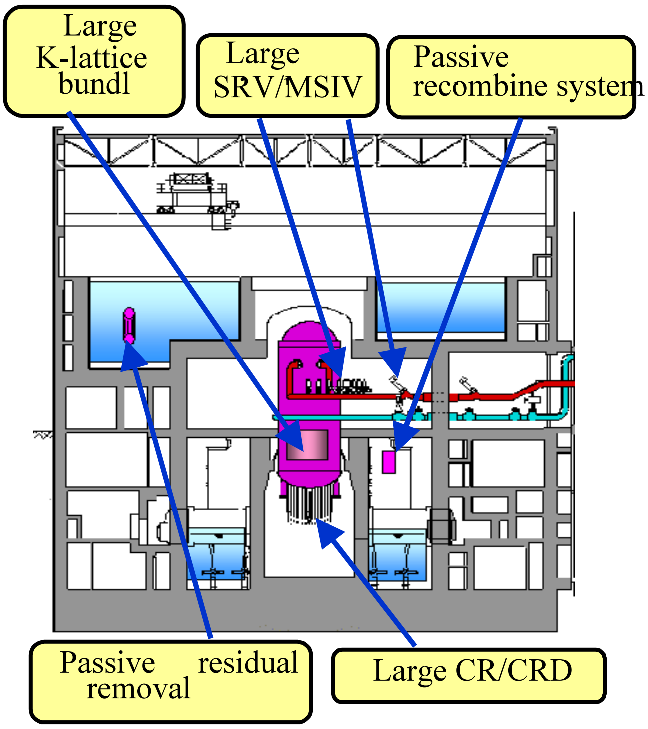

FIG.I-1. Features of ABWR-II plant system.

I-2. Description of passive reactor cooling system and passive containment cooling system

One of the new features of the ABWR-II safety design is the adoption of passive safety systems. The passive heat removal system (PHRS) consists of two dedicated systems, namely the passive reactor cooling system (PRCS: the same as Isolation condenser) and the passive containment cooling system (PCCS), that use a common heat sink pool above the containment allowing a one-day grace period (Figure I-2). These passive systems not only cover beyond DBA condition, but also provide in-depth heat removal backup for the RHR, and practically eliminate the necessity of containment venting before and after core damage as a means of overpressure protection.

With regard to ABWR-II development, the horizontal type PCCS has been designed with a focus on anti-seismic structure and decreasing PCCS pool depth instead of conventional vertical type PCCS. Figure I-3 shows PCCS functional schematic and an example of containment pressure transient following typical low pressure core melt scenario. Containment venting for overpressure protection under severe accident condition is practically excluded by adopting PCCS.

I-3. Conclusions

In the present, various testing programs are being performed or planned to consolidate their feasibilities and to find further room for improvements. ABWR-II results are planed to reflect the new next generation light water reactor development program sponsored by the government.

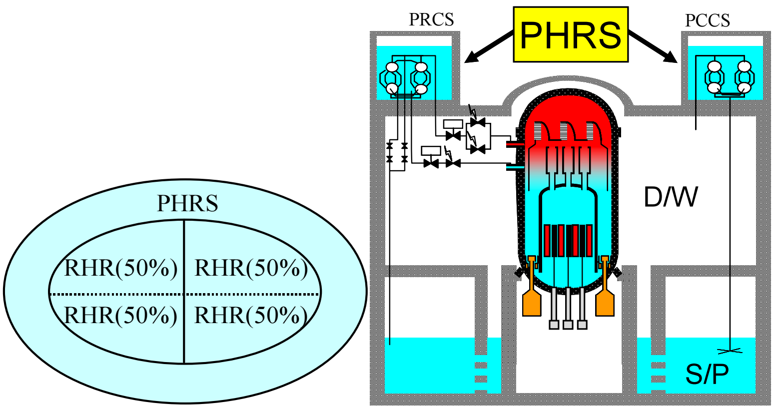

FIG. I-2. Passive heat removal system.

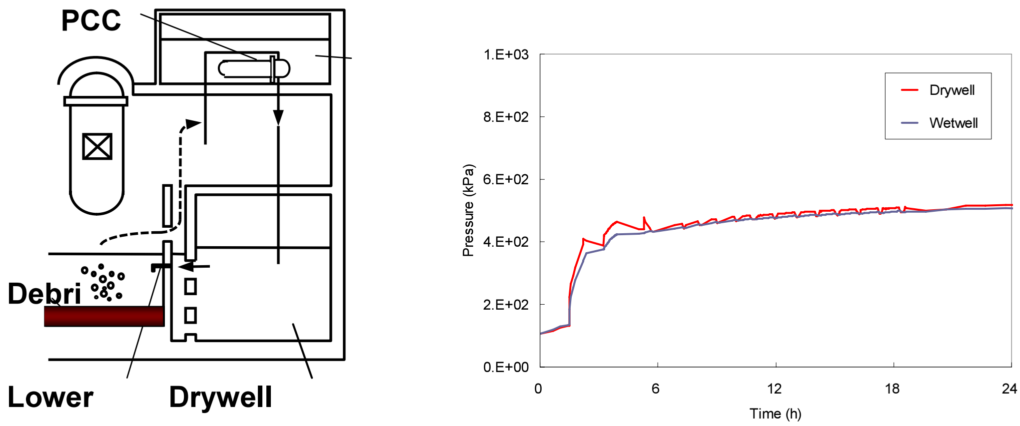

FIG. I-3. Example of containment pressure transient following typical low pressure core melt scenario.

ANNEX II. ACR-1000
Atomic Energy of Canada Ltd, Canada

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Advanced CANDU Reactor (ACR 1000) *Atomic Energy of Canada Ltd*	PHWR /PWR*	3180	The following systems have passive safety features CORE/PRIMARY: • Shutdown System 1 (Shutoff Rods) • Shutdown System 2 (Liquid Injection Shutdown System. • Core Make-up Tanks (CMT) • Reserve Water System (RWS) • Moderator & Reactor Vault CONTAINMENT • Containment Cooling Spray • Air Recirculation
*Note: In a CANDU reactor the primary heat transport system (PHTS) and the moderator system (MS) are separated. The PHTS is pressurized and contains light water in an ACR 1000 but the MS is a low pressure system and contains only heavy water.			

II-1. Introduction

The Advanced CANDU Reactor ™ (ACR™) is a Generation III+ pressure tube reactor designed by Atomic Energy of Canada (AECL). The ACR-1000 is an evolution of the proven CANDU reactor design. It is a light-water-cooled reactor that incorporates features of both pressurized heavy water reactors (PHWRs) and advanced light-water PWRs. It incorporates multiple and diverse passive systems, wherever necessary, for mitigation of any postulated accident scenarios, including severe accidents. The ACR-1000 uses passive design elements to complement active features, thus enhancing reliability and improving safety margins.

The ACR retains the core features of previous CANDU designs, such as horizontal fuel channels surrounded by a heavy water moderator. The major innovation in the ACR is the use of low enriched uranium fuel and light water as the coolant.

The overall layout of an ACR-1000 reactor and its primary components are shown in Fig. II-1 and Fig. II-2. The ACR-1000 reactor is designed by Atomic Energy of Canada Ltd to produce a nominal gross output of 1165 MW(e). The reactor employs active as well as passive safety features, the latter relying on gravity, compressed gas or natural circulation (thermosyphoning). As with all CANDU reactors, the high pressure, (11.1 MPa) heat transport system (HTS) and the low pressure and low temperature moderator are separate systems.

The low pressure, low-temperature moderator is contained in a tank called the Calandria. The primary coolant system consists of fuel channels, stainless steel feeders, four inlet headers, four outlet headers, four steam generators, four electrically driven heat transport pumps, and the interconnecting piping. There are 520 fuel channels and each fuel channel contains 12 nuclear fuel bundles (approximately 50 cm long × 10 cm in diameter). The primary coolant system is arranged in a two 'figure of eight' loop configuration. The pumps circulate the water in the two loops in opposite directions.

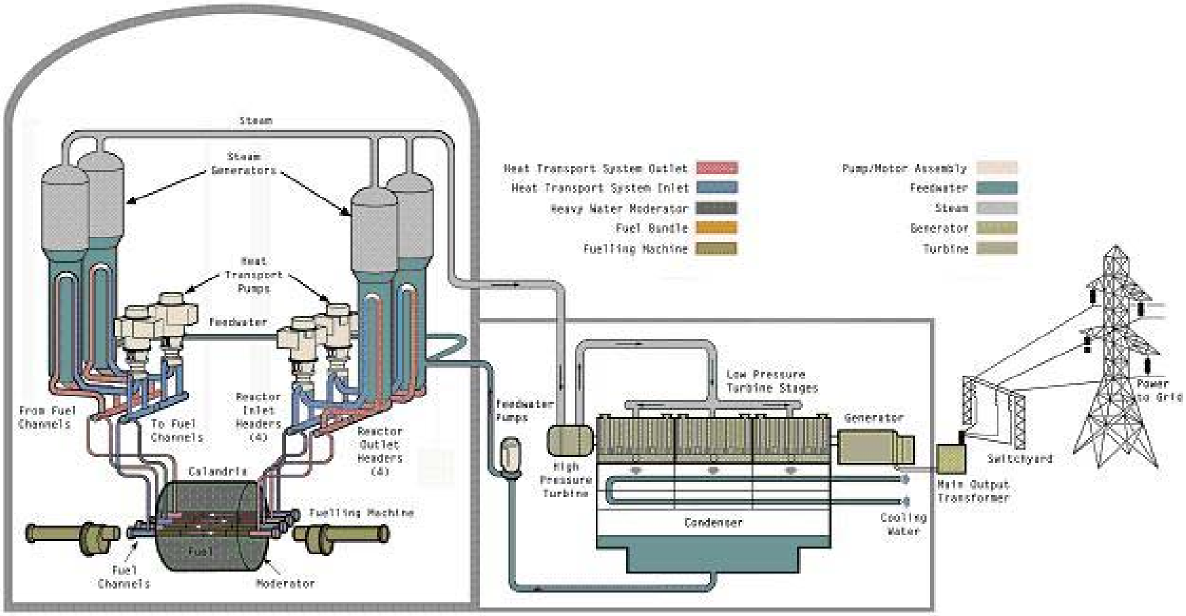

FIG.II-1. ACR-1000 plant layout.

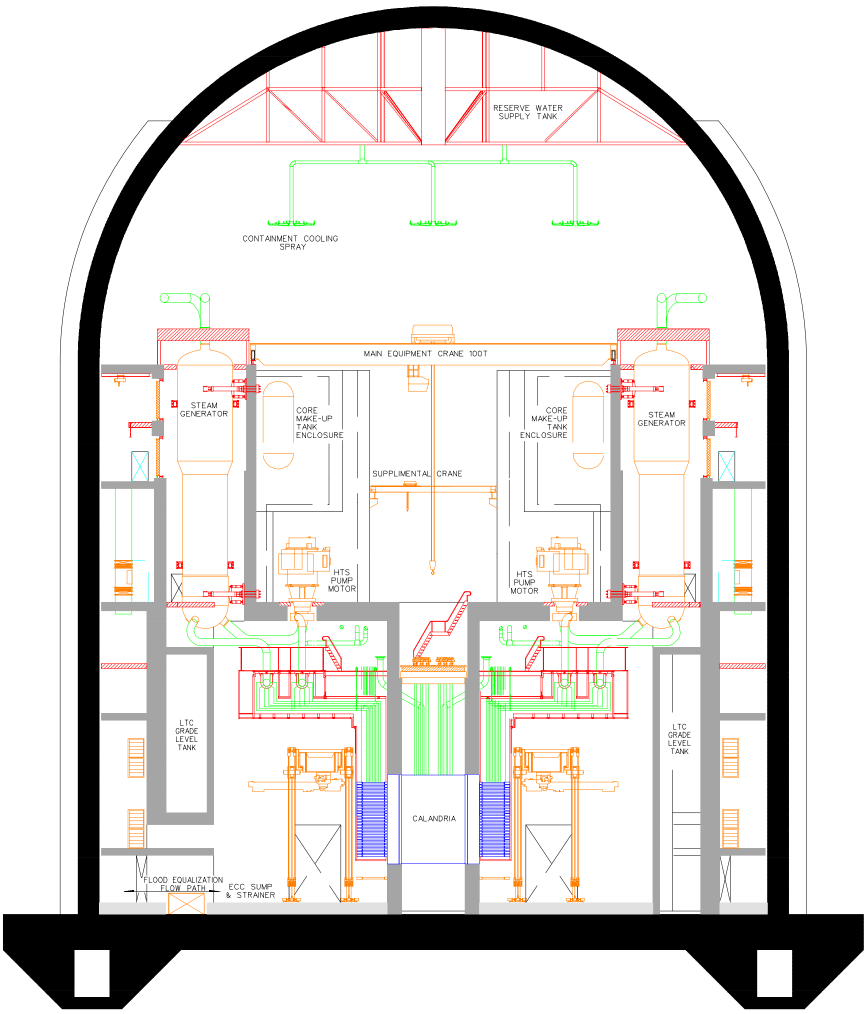

FIG. II-2. ACR-1000 reactor building layout.

In normal operation heat is generated in the reactor fuel bundles and exchanged with the fast-flowing coolant. The heated water is transported through the feeder pipes to the headers and into the U tubes inside the steam generators (SGs). The heat is then transferred to the water on the secondary side to produce steam, driving the turbine to generate electricity. The cooled water leaving the steam generator is pumped by four HTS pumps through two separate HTS loops connected to a common pressurizer, which maintains the HTS at a constant pressure.

II-2. ACR-1000 safety features and systems based on natural circulation

Some of the safety features included in the ACR-1000, which are discussed in this article, are:

- Two fast-acting, fully capable, diverse, and separate shutdown systems, which are physically and functionally independent of each other and also from the reactor regulating system;
- Emergency core cooling (ECC) system, comprised of:
    - Emergency coolant injection (ECI) system to refill the HTS and cool the fuel following a loss of coolant accident (LOCA);
    - Long term cooling (LTC) system, a four-quadrant system capable of operating in either 'shutdown cooling mode', taking suction from the reactor outlet headers and returning it to the inlet headers via heat exchangers for heat removal, or in post-LOCA 'recovery mode', drawing water from dedicated grade level tanks and the reactor building (RB) sumps to cool the fuel in the long term, and;
    - Core make-up tanks (CMTs) (see Fig. II-2) that provide make-up to limit the extent and duration of voiding in the intact HTS following a rapid cooldown event that depletes HTS inventory.
- Containment system (steel-lined containment structures with low leakage, containment isolation system, containment heat removal system, etc.);
- Containment cooling system, comprised of local air coolers (LACs) for active air circulation and heat removal, a containment cooling spray for post-accident pressure and temperature suppression, and provisions to interconnect the major volumes of the RB to establish an air flow path for natural circulation;
- Reserve water system (RWS, Fig. II-3), which provides an emergency source of water by gravity to steam generators, moderator system, and HTS, if required;
- Emergency feedwater system (EFW), a four-quadrant system that provides make-up feedwater to the steam generators from reserve feedwater tanks when the main feedwater system is unavailable;
- Primary coolant system (heat transport system), laid out with heat transport pumps and steam generators above the core to promote natural circulation of the primary coolant for accidents when the heat transport pumps are not operating, and;
- Moderator system, a low pressure and temperature heavy water system contained in a Calandria vessel, which moderates nuclear fission and acts as a heat sink following postulated accidents. Natural circulation of the moderator in the Calandria following an accident prevents localized heat-up of the moderator, and can prevent a severe accident even if all normal means of fuel cooling and decay heat removal are not available.

Natural circulation features in ACR are grouped into three generalized categories; cooling of the fuel in the primary heat transport system, cooling of the moderator in the Calandria, and circulation and cooling of the atmosphere inside containment.

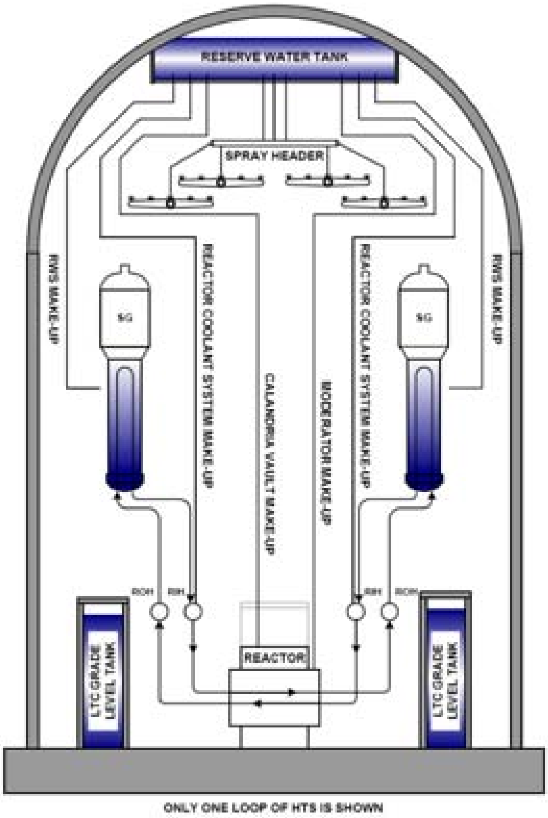

FIG. II-3. Reserve water system.

II-2.1. Natural circulation in heat transport system

The HTS is laid out with the heat transport pumps and steam generators above the core to promote natural circulation of the primary coolant for accidents when the heat transport pumps are not operating (see Fig. II-4). This natural circulation flow allows the plant to recover from any trip without relying on the heat transport pumps. Commissioning tests have been done (on current generation of CANDU plants) to validate the effectiveness of the design feature.

Core make-up tanks (CMTs, see Fig. II-2) are provided to passively limit the extent and duration of voiding in the HTS following events that cause a rapid depletion of the HTS inventory. The CMTs are located above the tops of the steam generator U tube bundles (i.e. at the highest point in the HTS) and are normally maintained at approximately the pressure and temperature of the reactor inlet headers. ‘Flashing’ of the CMT inventory when the HTS depressurizes to below the saturation pressure of the CMTs forces a rapid flow of coolant into the HTS, thus maintaining the HTS at a relatively high pressure and fully filled with water.

Keeping the HTS full and free of void ensures the thermosyphoning capability and allows operation of the LTC pumps (if available) in the shutdown-cooling mode (see Section 2) without the risk of void entrainment and consequential cavitation of the LTC pumps.

Thermosyphoning in the HTS is supported by provision of feedwater to the secondary side of the steam generators for heat removal. There are several feedwater options available in ACR-1000. The main feedwater system (using either the main or start-up feedwater pumps) and the four-quadrant EFW system are capable of supplying feedwater to the SGs at full pressure. If these active sources are not available, passive design feature of supply is provided by gravity from the RWS (see Fig. II-3) after auto-depressurization of the SGs.

With the HTS full and free of void, and with a continuous supply of feedwater to the secondary side, thermosyphoning can continue indefinitely to remove heat from the fuel.

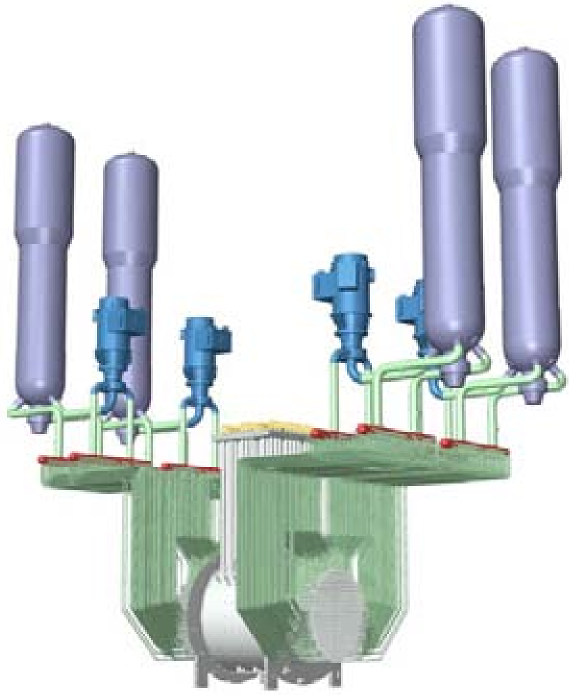

FIG. II-4. Heat transport system layout.

II-2.2. Natural circulation in Calandria vessel

In a CANDU reactor the fuel is contained within pressure tubes that run through a vessel called the Calandria. Low pressure and temperature heavy water contained in a Calandria vessel moderates nuclear fission. The pressure tubes are contained within another set of tubes called calandria tubes, separated from the pressure tubes by an annulus gap filled with carbon dioxide. This keeps the high pressure, high temperature coolant on the primary side separated and thermally isolated from the low pressure, low temperature moderator.

The moderator system consists of a closed heavy water recirculating loop that serves to cool and circulate the heavy water moderator through the calandria (see Fig. II-5). If the moderator pumps fail, natural circulation flows prevent formation of hot spots inside the Calandria vessel.

For accidents when all means of normal fuel cooling and long term decay heat removal are not available, the fuel and fuel channels will heat up until the pressure tubes contact the Calandria tubes, resulting in direct conduction of heat to the moderator. If the forced circulation moderator cooling system not available to remove heat, natural circulation inside the Calandria prevents further damage to the fuel channels and keeps the accident from progressing to a severe accident.

Without cooling, the heat transferred to the moderator ultimately exceeds the latent heat of vaporization, and steam is released to the RB atmosphere. Passive make-up water supply from the RWS (by gravity) will keep the Calandria full (see Fig. II-3). Adequate natural circulation flow will be maintained indefinitely for long term decay heat removal.

The LTC system design allows the system to recover and cool water from the RB sumps and pump it back to the RWS. Alternatively, the RWS inventory can be made up by supply from the firewater system. The replenished RWS inventory can then be directed to the moderator. This permits thermosyphoning in the moderator to be sustained, thereby preventing severe core damage, for an unlimited period of time.

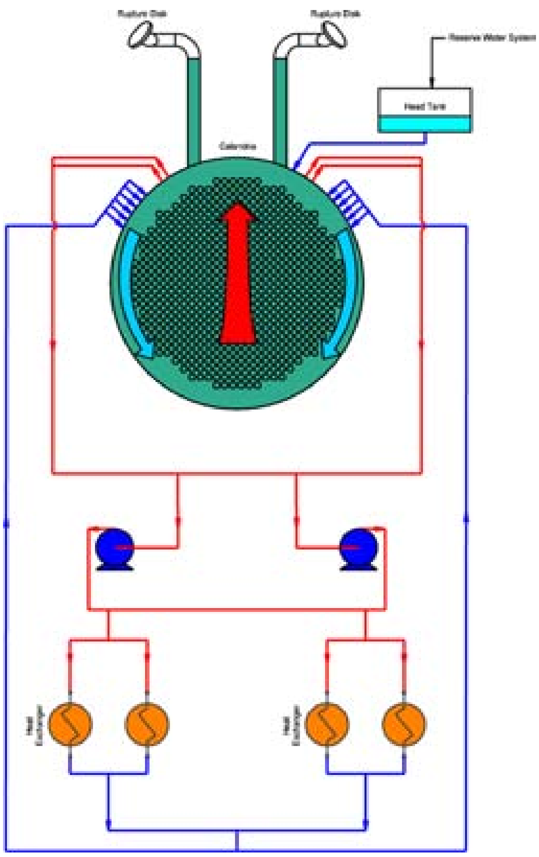

FIG.II-5. Moderator system & Calandria.

II-2.3. Natural circulation in containment

The containment cooling system (see Fig. II-6) includes design features for circulation and cooling of the RB atmosphere both during normal operation and following an accident.

During normal operation local air coolers (LACs) operate to remove heat from the RB atmosphere. Two fuelling machine vaults and steam generator enclosures are atmospherically isolated from the rest of the RB.

Following an accident large airflow paths are established by , interconnecting the fuelling machine vaults and steam generator enclosures with the rest of the RB, permitting natural circulation flows to mix the RB atmosphere, without the need for fans or other active components (i.e. LACs), .This feature allows hydrogen dispersion/dilution throughout the larger RB volume and prevents formation of regions of locally high temperature. Hydrogen control design features are provided to restrict the concentration of hydrogen to below the limit for deflagration or detonation.

If needed, post-accident pressure and temperature suppression is performed by a containment cooling spray system supplied from the RWS.

A combination of passive and active features are provided in the ACR-1000 for atmospheric hydrogen control; passive auto catalytic recombiners, and active igniters (or ‘glow plugs’) that limit the concentration of hydrogen in the RB atmosphere to below the threshold limit at which deflagration or detonation could occur.

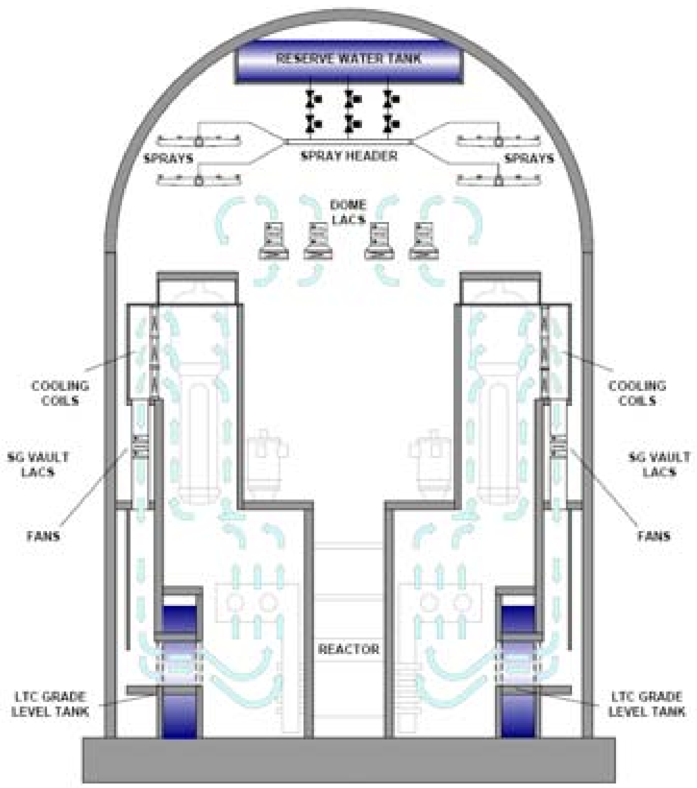

FIG. II-6. Containment cooling system.

II-3. Integrated system response to a large break LOCA

The most effective means of describing natural circulation in the HTS, Calandria and containment is to integrate their operation in response to a large break loss of coolant accident (LOCA). The first phase of a LOCA is the subcooled blow down phase. During this phase, high pressure subcooled liquid is vented from the break under choked flow conditions. The primary HTS pressure and liquid inventory will decrease. When a LOCA is detected, the following automatic actions are initiated:

- Shut down the reactor,
- Initiation of the ECC system,
- Open the RWS valves to HTS,
- Crash cooldown (rapid depressurization of the steam generators),
- Isolate the broken loop of the HTS from the intact loop, and
- Trip the HTS pumps.

II-3.1. Decay heat removal – broken loop

Following reactor shutdown, the HTS loops are isolated and crash cooldown (rapid depressurization of the secondary side of the steam generators) is initiated. This depressurizes the HTS and facilitates refill of the broken loop by either the ECC system, or by gravity from the RWS (as a back-up to ECC).

Heat is removed from the broken loop by the LTC system in recovery mode (see Section II-2). If the LTC system fails (low probability event), the fuel and fuel channels of the broken loop will heat up until the pressure tubes sag into contact the Calandria tubes, resulting in direct conduction of heat to the moderator (refer to Section II-2.1). The moderator cooling system, assisted by natural circulation in the Calandria, serves as a back-up heat sink, preventing further core damage.

If the moderator fails as a heat sink (e.g. if moderator make-up from the RWS fails), the fuel channels will fail. The shield vault contains a large volume of water that can cool and contain the damaged fuel channels inside Calandria and halt the progress of the accident. Inventory is allocated in the RWS to make-up inventory boiled off from the shield vault during this time. As long as make-up can be provided to the shield vault to cool the fuel, Calandria vessel integrity will be maintained and corium-concrete interaction will be prevented.

II-3.2. Decay heat removal – intact loop

After the heat transport system is crash cooled and the loops of the HTS are isolated automatically by the LOCA signal (see discussion for broken loop above), the CMTs replenish the inventory of the intact loop and limit the extent and duration of voiding that occurs.

When the HTS is full and free of void, natural circulation can be relied upon to remove heat from the fuel (see Section II-2.1). This is supported by provision of feedwater to the secondary side of the steam generators as a heat sink. Feedwater is supplied from either the main or emergency feedwater systems. Feedwater may also be supplied from the RWS, though this make-up must be initiated manually.

II-3.3. Containment heat removal and atmospheric mixing

Following a LOCA, large flow paths are established interconnecting the fuelling machine vaults and SG enclosures with the rest of the reactor building. These flow paths allow natural circulation flows to prevent formation of regions of locally high temperature and/or hydrogen concentration to protect the integrity of containment.

For LOCA events with the LTC system available, the LTC system alone is capable of removing decay heat and maintaining indefinitely the integrity of the reactor building.

If the LTC system fails, steam will be discharged to the reactor building both from the break and eventually, from the moderator system as well (see the discussion for the broken HTS loop above). In this case, either the LACs or the containment cooling spray will suppress the pressure and temperature inside containment. If the LACs are unavailable to provide forced circulation, the natural circulation airflows will prevent regions of locally high temperature and/or hydrogen concentration inside containment.

II-4. Summary and conclusions

The ACR-1000 incorporates multiple features utilizing natural circulation to prevent or mitigate accidents. The intrinsic passivity and inherent reliability of these features greatly strengthens the safety case for the ACR-1000.

- Thermosyphoning of the HTS primary coolant can remove decay heat from the fuel upon shutdown for normal maintenance outages and/or when forced circulation is not available.
- Natural circulation of the moderator in the calandria prevents formation of ‘hot spots’ when forced moderator circulation is unavailable. Natural circulation in the calandria can also limit damage to the fuel channels, thereby preventing severe accidents.
- Natural circulation airflows in containment prevent formation of regions of locally high temperature and dilution of hydrogen concentration following an accident.

The inclusion of features using passive natural circulation, in addition to the ‘traditional’ active safety systems used for active mitigation, is intended to improve the overall safety of the ACR design, by virtue of the simplicity and reliability inherent to passive designs.

Enhanced reliability through use of passive natural circulation features helps prevent events from progressing to the level of ‘severe’, or ‘beyond design basis’ accidents, and reduces severe core damage frequency for the ACR.

By utilizing natural circulation to remove heat from the HTS and the moderator, and to cool and circulate the containment atmosphere, the ACR-1000 design objectives are to improve overall reliability for the key safety functions, and to have a greatly enhanced safety case for postulated severe accidents.

ANNEX III. AHWR
Bhabha Atomic Research Centre, India

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Advanced Heavy Water Reactor (AHWR) *Bhabha Atomic Research Centre*	HWR	920	CORE/PRIMARY • Gravity Driven Water Pool • Isolation Condenser for decay heat removal • Accumulator for ECC injection • Passive heat removal from core under normal operating conditions CONTAINMENT • Passive Containment Cooling System • Passive Containment Isolation System

III-1. Introduction

The Indian nuclear power program consists of three stages. The first stage envisaged setting up of pressurized heavy water reactors (PHWRs) and the necessary fuel cycle facilities. Comprehensive capability in the design, construction, and operation of PHWR has been achieved. The second stage envisages development of fast breeder reactors using Plutonium and Depleted Uranium obtained from the first stage. Fast breeder test reactor (FBTR) has been operated successfully for 21 years and the construction of 500 Mw(e) prototype fast breeder reactor (PFBR) has started. The third stage aims at the development of reactors based on Uranium-233 obtained from irradiated thorium. The Kamini reactor uses Uranium-233 as a fuel and has operated since 1995. To transition to thorium based systems, an advanced heavy water reactor (AHWR) is being developed at Bhabha Atomic Research Centre.

The Bhabha Atomic Research Centre (BARC), located in Mumbai, is the premier multidisciplinary nuclear research centre of India having excellent infrastructure for advanced research and development. It has expertise covering the entire spectrum of nuclear science, engineering, and related areas. Detailed design, research, and development work on AHWR is being conducted in BARC.

The AHWR is a 300 MW(e) boiling light water cooled, heavy water moderated, vertical pressure tube type reactor designed to produce most of its power from thorium with an associated 500 $m^{3/}$day capacity desalination plant. The core consists of $(Th\text{-}U^{233})\ O_2$ and $(Th\text{-}Pu)O_2$ fuel. A simplified sketch of the reactor is depicted in Fig. III-1. Some important features of the reactor are given below.

- Thorium based fuel with a negative void coefficient of reactivity,
- Advanced coolant channel design with easily replaceable pressure tubes,
- Passive systems for core heat removal (under both normal operating and shutdown condition), containment cooling and containment isolation,
- Direct injection of ECCS water into fuel bundle,
- Accumulator for high pressure ECC,
- Gravity driven water pool (GDWP) at high elevation,
- No emergency planning in public domain,
- Design life of 100 years,
- Associated desalination plant.

A number of passive systems that utilize natural circulation have been incorporated in AHWR. Some of them are briefly described below.

III-2. Passive core cooling system

In AHWR, natural circulation is used to remove heat from the reactor core under normal as well as shutdown conditions. Fig. III-2 shows the main heat transport (MHT) system and the passive decay heat removal system of AHWR. The two-phase steam water mixture generated in the core flows through the tail pipes to the steam drum, where steam gets separated from water. The separated water mixes with the subcooled feed water and flows down the downcomers to the reactor inlet header. From the header it flows back to the core through inlet feeders.

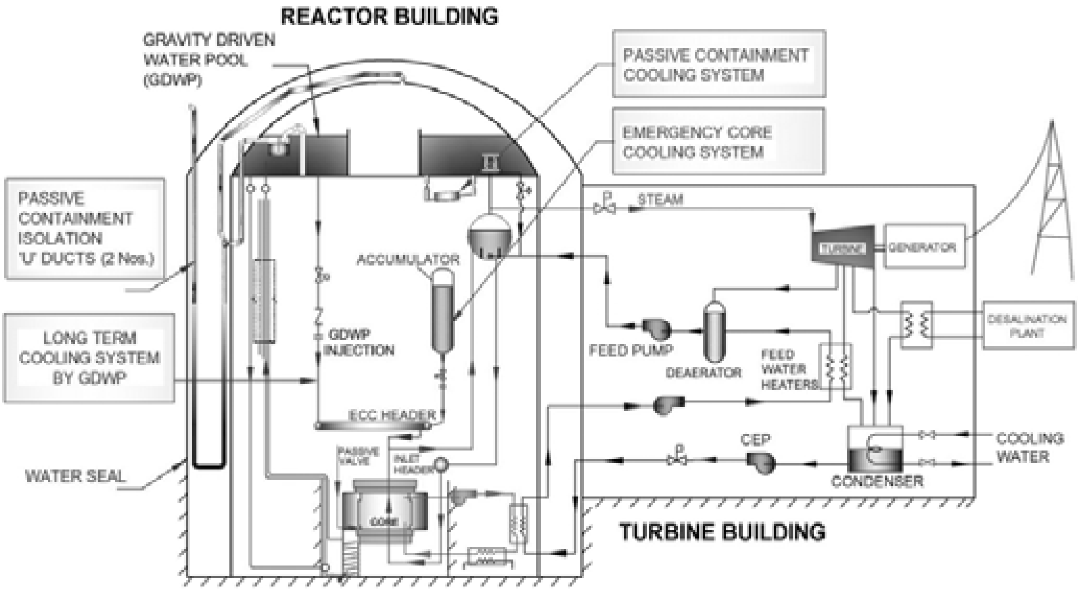

FIG. III-1. Simplified flow sheet of AHWR.

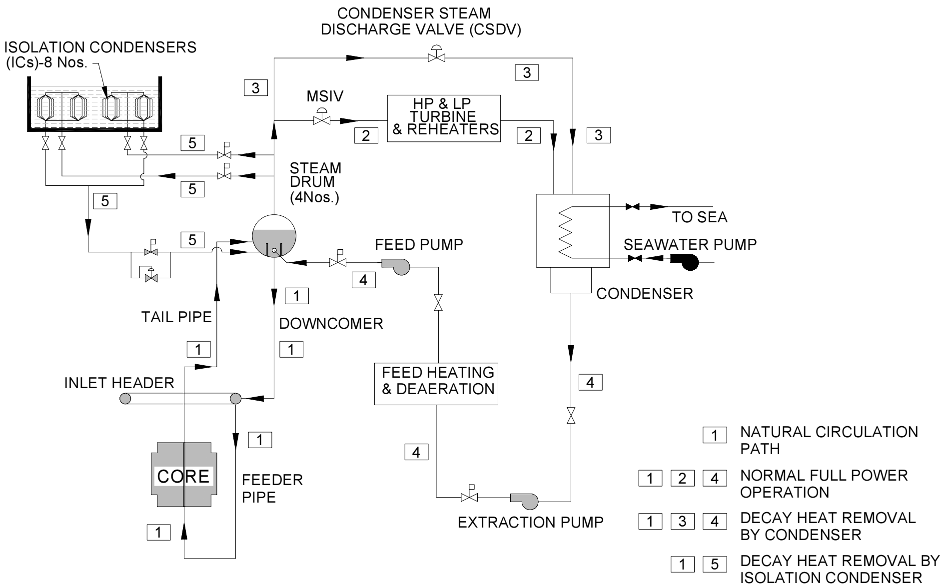

FIG. III-2. MHT and decay heat removal system.

Larger density differences between hot and cold legs are possible to be achieved in two-phase flow systems compared to single-phase natural circulation flow systems. The absence of pumps not only reduces operating cost, but also eliminates all postulated transients and accidents involving failure of pumps and pump power supply.

Steady state flow prevails in a natural circulation loop when the driving buoyancy force is balanced by the retarding frictional forces. However, the driving force in a natural circulation system is much lower compared to a forced circulation system. With a low driving force, measures are needed to reduce the frictional losses. The methods adopted to reduce frictional losses include, elimination of mechanical separators in the steam drum and the use of large diameter piping. The larger pipes increase the amount of coolant needed in the primary system.

Elimination of mechanical separators makes the system dependent on natural gravity separation at the surface in the steam drum, which may increase carryover and carryunder. Carryover is the fraction of the liquid entrained by the steam, whereas carryunder is the fraction of vapour that is carried by the liquid flowing into the downcomer. Excessive carryover can damage the turbine blades due to erosion, whereas carryunder can significantly reduce the driving buoyancy force and hence the natural circulation flow rate. The steam drum size is chosen to keep carryunder and carryover within acceptable limits.

A rational start-up procedure of the AHWR has been worked out for low pressure and temperature conditions. For this, after the MHT is filled with water to a desired level in the steam drum, the MHT system is pressurized to an initial desired pressure by using steam generated from an external boiler. Subsequently, the control rods are partially withdrawn and coolant heating up continues at about 2% full power. Core boiling will start only after the steam drum pressure reaches 70 bar and the coolant temperature attains 285^{o}C. The reactor power is increased gradually with controlled subcooling at the inlet of the reactor core until full power is reached.

Emergency core cooling system (ECCS) is designed to remove the core heat by passive means in case of a postulated loss of coolant accident (LOCA). In the event of rupture in the primary coolant pressure boundary, the cooling is initially achieved by a large flow of cold water from high pressure accumulators. Later, cooling of the core is achieved for three days by low pressure injection of cold water from gravity driven water pool (GDWP) located near the top of the reactor building. Fig. III-1 shows the emergency core cooling system.

III-3. Core decay heat removal system

During normal reactor shut down core decay heat is removed by passive means utilizing Isolation condensers (ICs) immersed in a gravity driven water pool (GDWP) located above the steam drum. Core decay heat, in the form of steam enters the IC tube bundles. The steam condenses inside the tubes and heat is transferred to the surrounding water pool. The condensate returns by gravity to the steam drum. The water inventory in the GDWP is adequate to cool the core for more than 3 days without any operator intervention and without boiling of GDWP water. Fig. III-2 depicts the core decay heat removal system comprising isolation condensers. A separate GDWP cooling system is provided to cool the GDWP inventory in case the temperature of GDWP inventory rises above a set value.

III-4. GDWP as ultimate heat sink

Isolation condensers (ICs) for removal of decay heat are immersed in the gravity driven water pool (GDWP). The pool is having a capacity of 6000 m^3 of water and is divided into 8 symmetry sectors, each containing one IC. In normal operation the pool water circulates through heat exchangers to maintain the pool temperature. Figure III-1 shows the recirculation and cooling system of GDWP water. The decay heat generated in the reactor during shut down is stored in the form of sensible heat of water. However, stratification may influence heat transfer to pool to a great extent and heat storage capacity of the pool in the form of sensible heat is significantly reduced. The GDWP acts as heat sink for passive containment cooling system also (Fig. III-3).

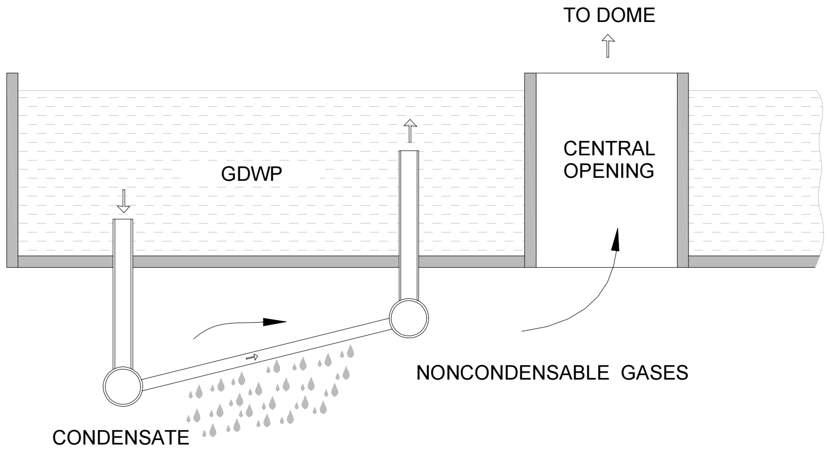

FIG.III-3. Schematic of passive external condenser.

III-5. Passive containment cooling system

Containment is a key component of the mitigation part of the defence in depth philosophy, since it is the last barrier designed to prevent large radioactive release to the environment.

In advanced heavy water reactor (AHWR), a passive containment cooling system (PCCS) is envisaged which can remove long term heat from containment following loss of coolant accident (LOCA). Immediately following LOCA, steam released is condensed in water pool by vapor suppression system. For subsequent long term cooling, PCCS is provided. PCCS, by definition is able to carry out its function with no reliance on external source of energy.

As mentioned earlier, gravity driven water pool acts as the heat sink for a number of passive heat removal systems including the PCCS. The passive external condensers (PECs) of the PCCS are connected to the pool as shown in Fig. III-3. The containment steam condenses on the outer surface of the tubes of PEC. The water inside the tubes takes up heat from air/vapor mixture and gets heated up. Due to the heating up of water, the natural circulation of water from the pool to PEC and from PEC to pool is established.

One important aspect of PCCS functioning is the potential degradation of heat transfer on PEC outer surface due to the presence of noncondensable gases in the containment. The presence of noncondensable (NC) gases in vapor can greatly inhibit the condensation process. Extensive R&D work is in progress to address this issue. Another aspect of PCCS functioning is the blockage of passive external condenser by noncondensable gas due to the stratification of noncondensable gas/vapor in the containment. In case of AHWR, the noncondensable gas is likely to escape through the central opening provided in the GDWP to the dome region. Experiments are planned to confirm this.

III-6. Passive containment isolation system

The reactor has double containment system viz., primary and secondary containments. Between the two containments, a negative pressure with reference to atmosphere is maintained to ensure that there is no release of radioactivity to atmosphere under accidental conditions. The primary containment envelops the high enthalpy and low enthalpy zones designated as volume V1 and V2 respectively. The volume V2 is normally ventilated to atmosphere through a ventilation duct.

A scheme for containment isolation for AHWR under accidental condition without any active actuation is conceived. The scheme consists of an isolation water tank comprising of two compartments, one in communication with volume V1 through a vent shaft while the other is in communication with volume V2 via the normal ventilation duct as shown in Fig. III-4. A vertical baffle plate, running from the top of the tank, separates the two compartments. The baffle plate however, does not run through the full height of the tank. The bottom portion of the tank allows the two compartments to be in communication. It may be noted that the volume V2 is normally ventilated to atmosphere through a ‘U’ duct, which has a branched connection to isolation water tank outlet. In the event of volume V1 reaching certain pressure, the water level in other compartment of tank rises to spill the water in to the ‘U’ duct. Thus, isolation of volumes V1 and V2 from atmosphere is ensured by securing a water seal at the base of U duct. It is required that the seal be formed in a minimum possible time, typically of the order of few seconds, to ensure fast isolation.

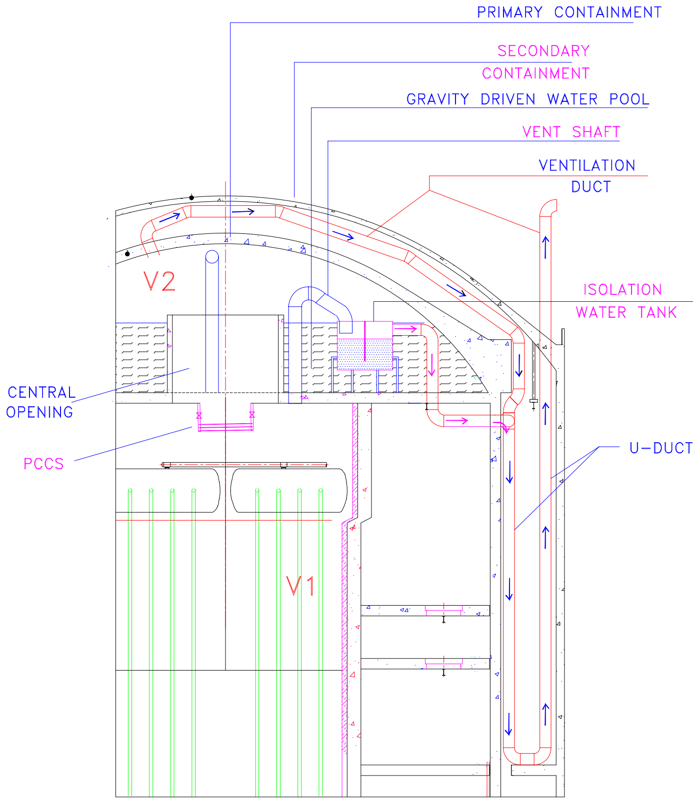

FIG.III-4. Passive containment isolation system.

III-7. Ongoing R&D activities

Several inherent and passive systems have been adopted in Indian innovative reactor, AHWR. Analyses have been performed to prove design concepts of these systems. Experiments and further analyses of these systems are being carried out rigorously. Several major areas of R & D have been identified for detailed study and the required development activities are in progress.

Extensive work has been carried out in the area of natural circulation. A transparent rectangular loop has been installed to study natural circulation. The stability of natural circulation with different heater and cooler orientations has been studied in the loop. Start-up procedure and instability studies are being carried out at high pressure natural circulation loop (HPNCL). Flow pattern transition instability studies using neutron radiography have been conducted at APSARA reactor in flow pattern transition instability loop (FPTIL). Integral behaviour experiments are being conducted in an integral test loop (ITL). The ITL (Fig. III-5) was commissioned in 2005 in BARC. The performance of isolation condensers for reactor decay heat removal is being evaluated in the ITL. A parallel channel experimental facility is set up in BARC to investigate parallel channel instability. Void coefficient of reactivity has been simulated in the loop. The pre-test single channel stability and parallel channel stability analyses by RELAP 5 and other in house codes have been carried out for the ITL and parallel channel loop. Two-phase low flow pressure drop experiments are being conducted in 3 MW boiling water loop (BWL) across the various components of coolant channel. Earlier single phase and two phase (air-water) pressure drop experiments were performed on simulated full scale fuel bundle of AHWR in flow test facility at low pressure.

Thermal stratification inside a water pool is being investigated. One dimensional theoretical model and computer code for solving two dimensional Navier-Stokes equations have been developed to study the stratification phenomena in the water pool. Other generalized computer codes available are also being used for this purpose.

An experimental set up to study phenomena associated with passive containment isolation is being set up.

The effect of non-condensable gas on steam condensation inside a vertical tube has been investigated experimentally. A comparison of local heat transfer coefficient determined by theoretical model with experimental data has been carried out. An experimental facility to investigate the effect of non condensable gas on condensation of steam on the outer surfaces of tubes of passive external condensers has been commissioned and experiments are in progress.

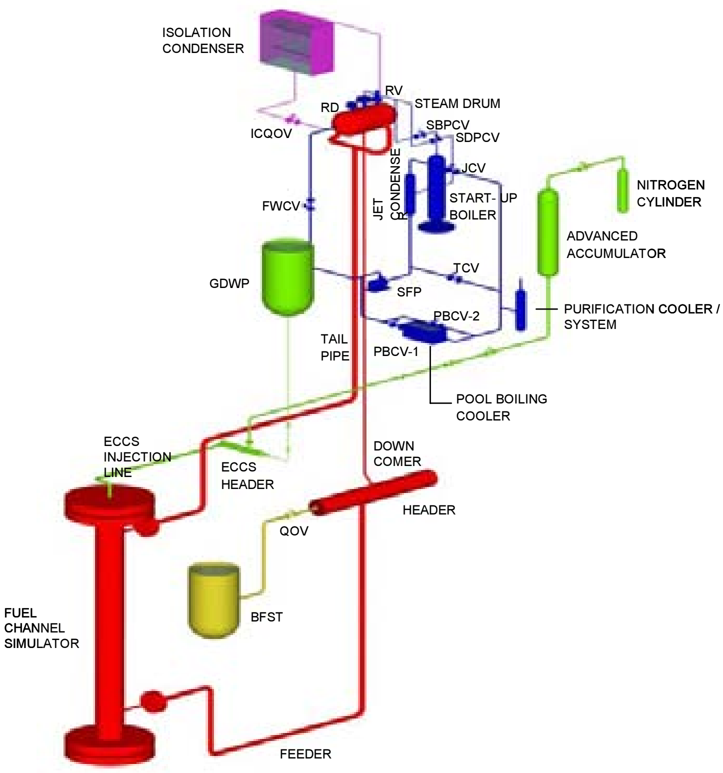

FIG. III-5. 3-D view of ITL.

ANNEX IV. APWR+
Mitsubishi, Japan

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Advanced PWR (APWR+) *Mitsubishi, Japan*	PWR	5000	CORE/PRIMARY: • Passive core cooling system using steam generator • Advanced Accumulators • Advanced Boric Acid Injection Tank

IV-1. Introduction

The APWR+ is a four loop type PWR with 1750 MW(e) output, which is being developed as the successor of APWR and conventional PWRs, aiming at more enhancements in economy, safety, reliability, reduction of the operators' workload, and harmony with the environment. The development is being carried out by Japanese PWR utilities (Kansai Electric Power, Hokkaido Electric Power, Shikoku Electric Power, Kyushu Electric Power, and The Japan Atomic Power Co.) and Mitsubishi Heavy Industries.

APWR+ employs the following concepts for its safety system including passive features:

a) Passive safety equipment and system
   - Core cooling using steam generator and natural circulation of the primary system during accident
   - 'Advanced Accumulators' and 'Advanced Boric Acid Injection Tank'.

b) Four train configuration

c) Confinement of energy release to the containment
   - The core and the loop piping are submerged in case of a LOCA

d) Enhancement of the design diversity
   - Emergency power supply by Diesel and gas turbine
   - Heat removal paths from the containment

Figure VI-1 shows the features of APWR+ safety related systems.

IV-2. Description of passive core cooling system using steam generator

Passive core cooling system using a Steam Generator (SG) is implemented in case of a loss of reactor coolant accident (LOCA) to cool-down and depressurize the primary system using SGs. Figure IV-2 shows a scenario for small break LOCA.

To show the validity of this system, it is important to verify natural circulation flow characteristics of the primary side and heat transfer capability through SGs because supplied water from accumulators includes non-condensable gas (nitrogen). From this point of view, a simulation test had been performed in addition to the simple flow test in a single SG tube and numerical analyses using CANAC3-3D code. This test was performed by Kansai Electric Power, Kyushu Electric Power, The Japan Atomic Power, and Mitsubishi Heavy Industries.

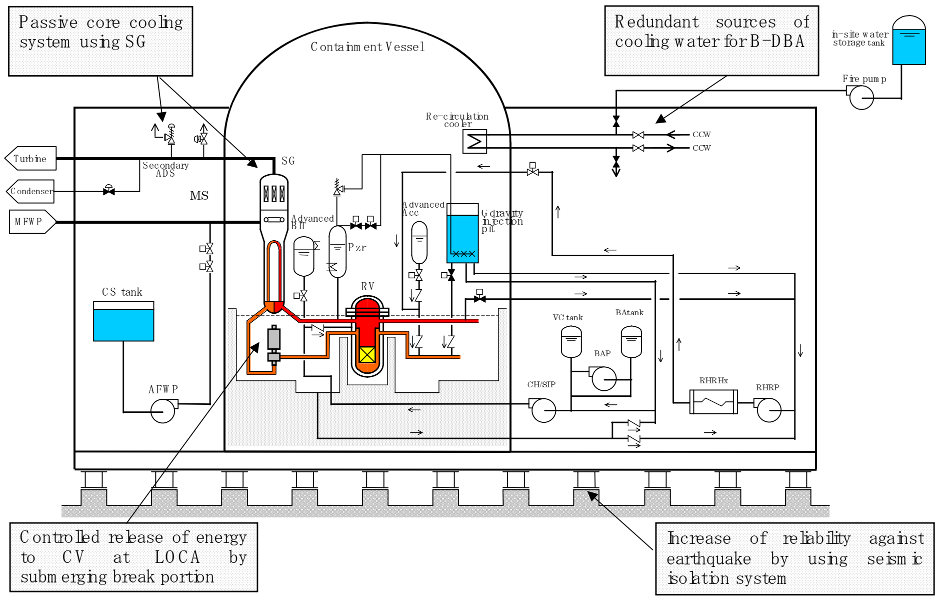

FIG.IV-1. Features of APWR+ safety related systems.

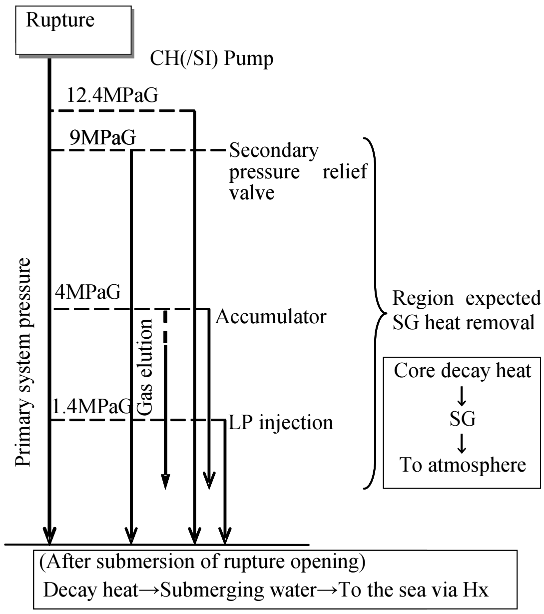

FIG.IV- 2. Small LOCA scenario of APWR+.

IV-3. Conclusions

In the safety concept of APWR+ against small LOCA event, Steam Generators (SGs) are used for decay heat removal. Therefore, it is important to maintain natural circulation in the primary system during small LOCA events to transfer the decay heat from the core to the SG. Non-condensable gas, which dissolves in water from the accumulators and injected to primary loop on LOCA event, may accumulate in the top of U-bend tube and cause siphon brake for natural circulation at the U-bend region of SG.

Small LOCA test was performed using the 'EOS' simulation loop of a PWR to verify natural circulation and siphon break in SGs, and we concluded that natural circulation with non-condensable gas was maintained during LOCA event and heat removal by the SGs was continuously effective.

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ANNEX V. AP600 & AP1000
Westinghouse Electric, USA

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Advanced Passive PWR (AP600) and (AP100) *Westinghouse Electric, USA*	PWR	1940 3415	CORE/PRIMARY: • A Passive Residual Heat Removal System • Two Core Make-up Tanks • Four Stage Automatic Depressurization System • Two Accumulator Tanks • In-containment Refuelling Water Storage Tank • Lower Containment Sump Recirculation CONTAINMENT • Passive Containment Cooling System

V-1. Introduction

The AP600 and AP1000 are pressurized light water reactors designed by the Westinghouse Electric Corporation to produce 600 MW and 1100 MW of electric power, respectively. Both designs employ passive safety systems that rely on gravity, compressed gas, natural circulation, and evaporation to provide for long term cooling in the event of an accident.

Figure V-1 shows the overall layout of the plant. Figure V-2 is a schematic that illustrates the primary system components. The primary loop consists of the reactor vessel, which contains the nuclear fuel assemblies; two hot legs, which connect the reactor vessel to the steam generators; two steam generators; a pressurizer; four canned motor pumps; and four cold legs.

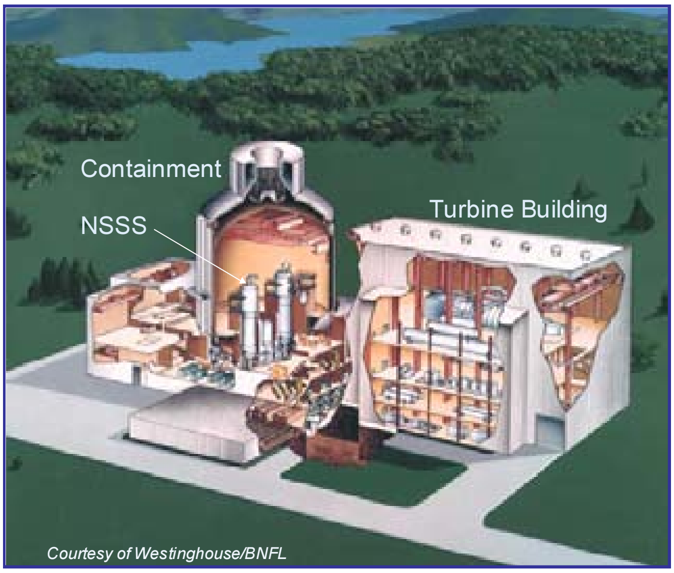

FIG. V-1. General layout of the AP600 and AP1000 plants.

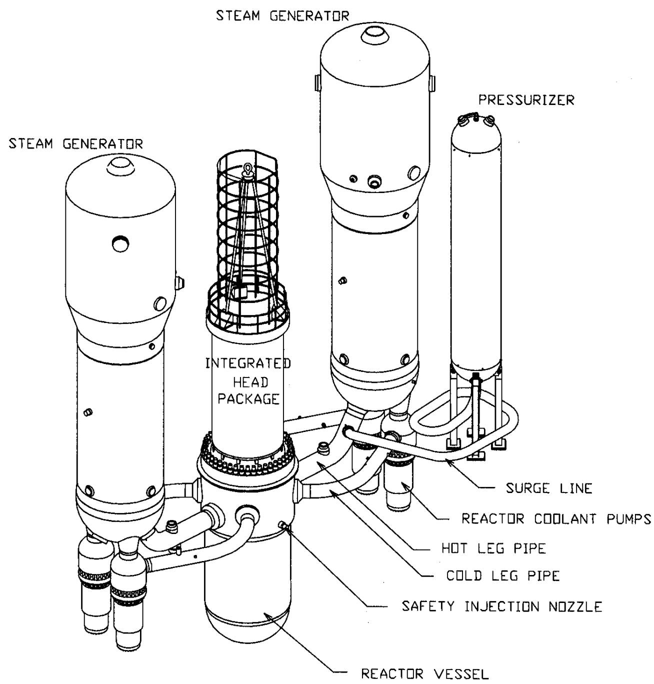

FIG. V-2. Schematic of primary loop of the AP600 and AP1000 plants.

During normal power operation, heat is generated in the reactor fuel. This heat is transported by conduction through the fuel and its cladding and transferred by convection to the water. Since the entire system operates at 15.5 MPa (2250 psia), bulk boiling of the water does not occur. The heated water is transported through the hot legs to the U tubes inside the steam generators. The energy of the primary coolant inside the tubes is transferred to the water on the secondary side by forced convection inside the tubes, conduction through the tube walls and boiling on the outside surface of the U tubes. The cooled water leaving the steam generator is pumped by four canned motor pumps, through four cold legs, back into the reactor vessel where the heating cycle is repeated. Primary system pressure is maintained constant by the pressurizer.

V-2. AP600/AP1000 passive safety systems

With respect to thermal hydraulic phenomena, normal full-power operation is typical of most pressurized water reactor (PWR) systems. Under shutdown cooling conditions, a key feature of the AP600 and AP1000 designs is that it uses core decay heat to drive the core cooling process by natural circulation. In fact, the AP600/AP1000 designs use core decay heat to drive the following six natural circulation processes:

- Primary system natural circulation (2 × 4 loop)
- PRHR loop circulation (1 loop)
- CMT loop circulation (2 loops)
- Lower containment sump recirculation (2 loops)
- Containment internal circulation (steam)
- Containment external circulation (air)

Figure V-3 presents a schematic that describes the connections of the primary system passive safety systems.

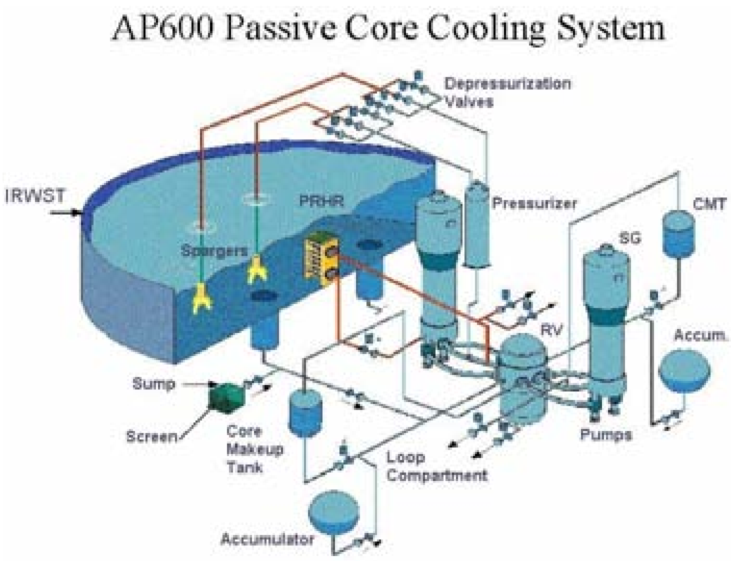

FIG. V-3. passive safety systems used in the AP600/AP1000 designs.

The AP600/AP1000 passive safety systems consist of:

- A passive residual heat removal (PRHR) system
- Two core make-up tanks (CMTs)
- A four stage automatic depressurization system (ADS)
- Two accumulator tanks (ACC)
- An in-containment refueling water storage tank (IRWST)
- A lower containment sump (CS)
- Passive containment cooling system (PCS)

V-2.1. Passive residual heat removal (PRHR) system

The passive residual heat removal (PRHR) consists of a C-Tube type heat exchanger that resides in the water-filled In-containment refueling water storage tank (IRWST) as shown in the schematic given in Figure V-4. The PRHR provides primary coolant heat removal via a natural circulation loop. Hot water rises through the PRHR inlet line attached to one of the hot legs. The hot water enters the tubesheet in the top header of the PRHR heat exchanger at full system pressure and temperature. The IRWST is filled with cold borated water and is open to the containment Heat removal from the PRHR heat exchanger occurs by boiling on the outside surface of the tubes. The cold primary coolant returns to the primary loop via the PRHR outline line that is connected to the steam generator lower head.

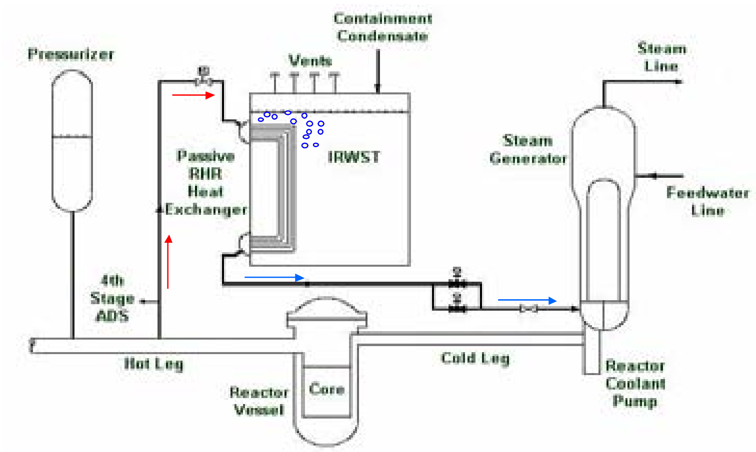

FIG.V- 4. Passive residual heat removal (PRHR) system.

V-2.2. Core make-up tank (CMT)

The core make-up tanks effectively replace the high pressure safety injection systems in conventional PWRs. Each CMT consists of a large volume stainless steel tank with an inlet line that connects one of the cold legs to the top of the CMT and an outlet line that connects the bottom of the CMT to the direct vessel injection (DVI) line. The DVI line is connected to the reactor vessel downcomer. Each CMT is filled with cold borated water. The CMT inlet valve is normally open and hence the CMT is normally at primary system pressure. The CMT outlet valve is normally closed, preventing natural circulation during normal operation. When the outlet valve is open, a natural circulation path is established. Cold borated water flows to the reactor vessel and hot primary fluid flows upward into the top of the CMT.

V-2.3. Automatic depressurization system (ADS)

The automatic depressurization system consists of four stages of valves that provide for the controlled reduction of primary system pressure. The first three stages consist of two trains of valves connected to the top of the pressurizer. The first stage opens on CMT liquid level. ADS stages two and three open shortly thereafter on timers. The ADS 1-3 valves discharge primary system steam into a sparger line that vents into the IRWST. The steam is condensed by direct contact with the highly subcooled water in the IRWST. The fourth stage of the ADS consists of two large valves attached to ADS lines on each hot leg. The ADS-4 valves open on low CMT liquid level and effectively bring primary side pressure down to containment conditions. The ADS-4 valves vent directly into the containment building.

V-2.4. Accumulators (ACC)

The accumulators are similar to those found in conventional PWRs. They are large spherical tanks approximately three-quarters filled with cold borated water and pre-pressurized with nitrogen. The accumulator outlet line is connected to the DVI line. A pair of check valves prevents injection flow during normal operating conditions. When system pressure drops below the accumulator pressure (plus the check valve cracking pressure), the check valves open allowing coolant injection to the reactor downcomer via the DVI line.

V-2.5. In-containment refueling water storage tank (IRWST)

The In-containment refueling water storage tank is a very large concrete pool filled with cold borated water. It serves as the heat sink for the PRHR heat exchanger and a source of water for IRWST injection. The IRWST has two injection lines connected to the reactor vessel DVI lines. These flow paths are normally isolated by two check valves in series. When the primary pressure drops below the head pressure of the water in the IRWST, the flow path is established through the DVI into the reactor vessel downcomer. The IRWST water is sufficient to flood the lower containment compartments to a level above the reactor vessel head and below the outlet of the ADS-4 lines.

V-2.6. Containment sump recirculation

Figure V-5 illustrates the automatic depressurization system, the passive safety injection, and sump recirculation flow paths and components.

After the lower containment sump and the IRWST liquid levels are equalized, the sump valves are opened to establish a natural circulation path. Primary coolant is boiled in the reactor core by decay heat. This low-density mixture flows upward through the core and steam and liquid is vented out of the ADS-4 lines into containment. Cooler water from the containment sump is drawn in through the sump screens into the sump lines that connect to the DVI lines.

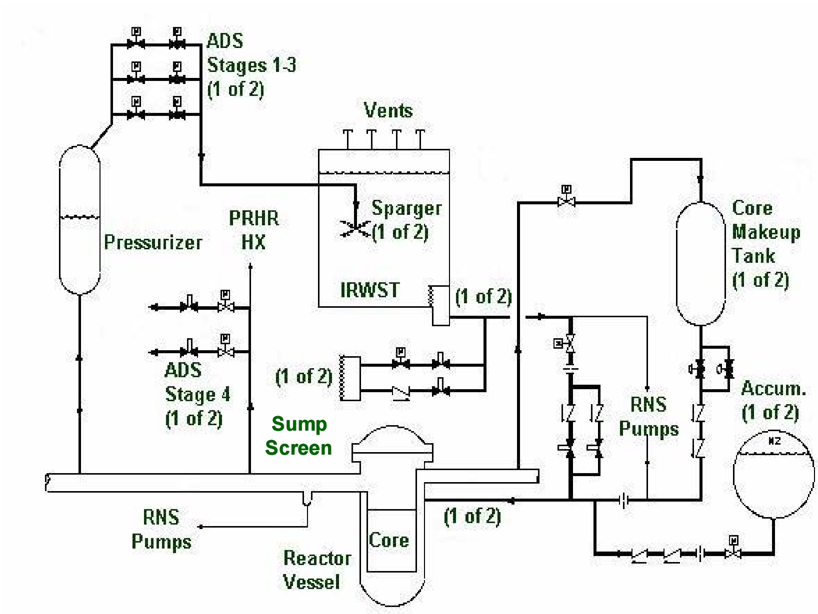

FIG. V-5. Passive safety injection and sump recirculation.

V-2.7. Containment and passive containment cooling system (PCCS)

Figure V-6 presents a schematic of the AP600/AP1000 containment. It consists of a large steel vessel that houses the nuclear steam supply system (NSSS) and all of the passive safety injection systems. The steel containment vessel resides inside of a concrete structure with ducts that allows cool outside air to come in contact with the outside surface of the containment vessel. When steam is vented into containment via a primary system break or ADS-4 valve actuation, it rises to the containment dome where it is condensed into liquid. The energy of the steam is transferred to the air on the outside of containment via conduction through the containment wall and natural convection to the air. As the air is heated, it rises through the ducts creating a natural circulation flow path that draws cool air in from the inlet duct and vents hot air out the top of the concrete structure. The condensate inside containment is directed back into the IRWST and the containment sump where it becomes a source of cool water in the sump recirculation process. In a LOCA, cold water is sprayed by gravity draining onto the containment vessel head to enhance containment cooling. A large tank of water, located at the top of the containment structure, serves as the source of water for this operation.

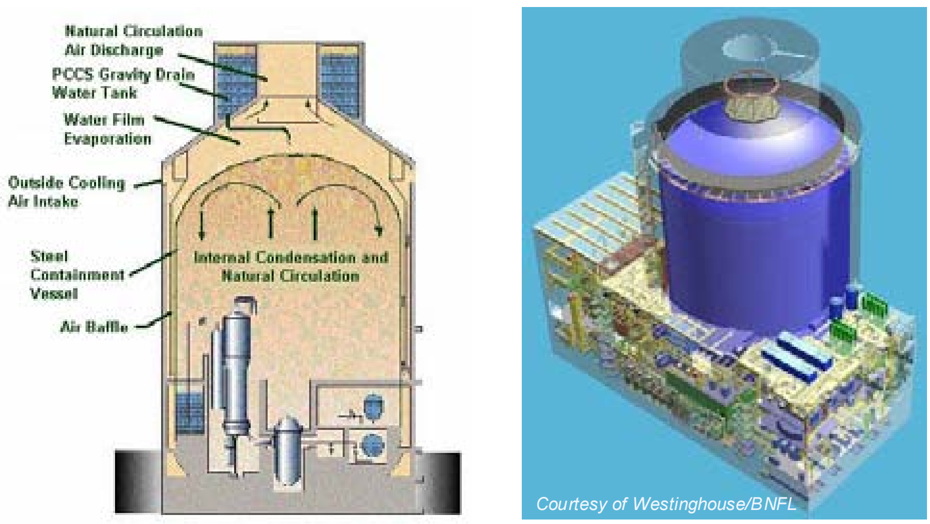

FIG.V- 6. Containment and passive containment cooling system (PCCS).

V-3. Integrated passive safety system response during a SBLOCA

The most effective means of describing the function of each of these passive safety systems is to relate their operation in response to a small break loss of coolant accident (SBLOCA). The first phase of a SBLOCA is the subcooled blowdown phase. During this phase, high pressure subcooled liquid is venting from the break under choked flow conditions. The primary system pressure and primary liquid inventory will be decreasing. When low pressure or low liquid level is sensed in the pressurizer, a safety signal is issued resulting in the following automatic actions:

- Scram reactor
- Open the PRHR inlet and outlet valves
- Open the CMT outlet valves
- Isolate Steam Generators (feedwater and main steam)
- Trip reactor coolant pumps (coastdown).

Natural circulation is established in the PRHR loop and the CMT loops. Boiling occurs on the PRHR tubes and hot water begins to fill the top of the CMTs. If the plant continues to depressurize, eventually the primary system reaches the saturation pressure corresponding to the hot leg temperature. Depending on the break size, the system pressure will reach a plateau during which the loop will experience a period of two-phase natural circulation.

If primary coolant inventory continues to decrease, eventually the CMTs will begin to drain. At a predetermined CMT level, the ADS-1 valves will open followed by the ADS 2-3 valves. System pressure will drop very quickly as a result of the ADS 1-3 venting steam into the IRWST. The primary system pressure soon drops below the accumulator tank pressure; and significant quantities of cold borated water are injected from the accumulator into the reactor vessel.

If the CMT liquid level continues to decrease, the ADS-4 actuation setpoint will be reached. The ADS-4 valves open, dropping the primary system pressure below the head pressure of the IRWST liquid. The IRWST drains by gravity into the reactor vessel, out the break and ADS 4 valves into the containment sump. Eventually the IRWST and containment sump liquid levels equalize and the sump valves are opened, establishing long term sump recirculation cooling.

Steam vented through the ADS-4 valves condense on the inside surfaces of the containment vessel. The containment vessel is externally cooled by air and water as needed. The condensate inside the containment is returned to the containment sump and IRWST, where it is available for sump recirculation.

V-4. Description of main control room habitability system (VES)

The main control room (MCR) is also augmented by passive systems. The main control room habitability system (VES) will pressurize, cool, and provide fresh air to the MCR in the event of an accident. This system is initiated by a high radiation signal in the MCR. It isolates the MCR ventilation system and pressurizes the room to slightly higher than atmospheric pressure, thereby limiting infiltration by airborne contaminants.

V-5. Conclusions

The AP600 was the first passively safe nuclear plant to be certified in the United States. The certification was based on comprehensive integral system and separate effects testing conducted by Westinghouse and the U.S. Department of Energy at the SPES test facility in Italy and at the APEX test facility at Oregon State University. The U.S. Nuclear Regulatory Commission (NRC) conducted confirmatory tests at the ROSA-AP600 test facility in Japan and the APEX Facility at Oregon State University. The AP600 received Final Design Approval from the NRC in September, 1998 and Design Certification in December, 1999. The AP1000 received NRC Final Design Certification in January 2006.

ANNEX VI. ESBWR
General Electric, USA

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Economic Simplified Boiling Water Reactor (ESBWR) *General Electric, USA*	BWR	4500	CORE/PRIMARY: • Gravity-Driven Cooling System • Automatic Depressurization System • Isolation Condenser System • Standby Liquid Control System CONTAINMENT: • Passive Containment Cooling System • Suppression Pool

VI-1. Introduction

General Electric (GE) has developed a new passive safe boiling water reactor called the economic simplified boiling water reactor (ESBWR), which is based on the previous simplified boiling water reactor (SBWR) design with some modifications of safety systems and the containment size relative to the reactor power [1, 2]. Major differences between the current boiling water reactors (BWR) and the ESBWR are in the simplification of the coolant circulation system and the implementation of a passive emergency cooling system. The ESBWR reactor core has a rated thermal output of 4500 MW•th.

The ESBWR relies on natural circulation and proven passive systems to improve safety, economics, and performance. In ESBWR concepts, the safety is accomplished by eliminating the recirculation pump, thus relying on natural circulation cooling. The coolant is circulated by natural circulation as a result of the density difference between the high void, two-phase fluid in the chimney and the exterior single-phase liquid in the downcomer. The tall chimney not only enhances the natural circulation flow, but also ensures the ample time for core uncovery before the emergency core cooling system (ECCS) comes in play. The emergency core cooling and containment cooling systems do not have an active pump injecting flows and the cooling flows are driven by pressure differences. Large volumes of suppression pool (SP) functions not only as a primary heat sink during the initial blow down, but also as coolant inventory to prevent the core uncovery through the gravity equalization lines.

By relying on natural circulation at operating pressures (7.2 MPa) and increased chimney height, the ESBWR has enhanced natural circulation flow inside the vessel. The schematic of natural circulation inside the reactor pressure vessel (RPV) is shown in Figure VI-1. The driving head of core flow is proportional to the core and chimney height and void fraction inside the downcomer shroud. In ESBWR, the differential water level is increased by approximately 8.2 m compared to the conventional BWRs. The greatly increased driving head enhances the natural circulation flow in the ESBWR compared to the conventional BWRs. Aforementioned ESBWR design features results in an average core flow per bundle over three times greater than that of a conventional BWR under natural circulation at similar bundle power. The use of natural circulation eliminates pumps, motors, controls, piping and many other components that could possibly fail.

VI-2. ESBWR passive safety systems

The basic nature of the passive safety systems and accident management strategies for the SBWR and ESBWR are similar. The major engineered safety systems and safety grade systems in the ESBWR are:

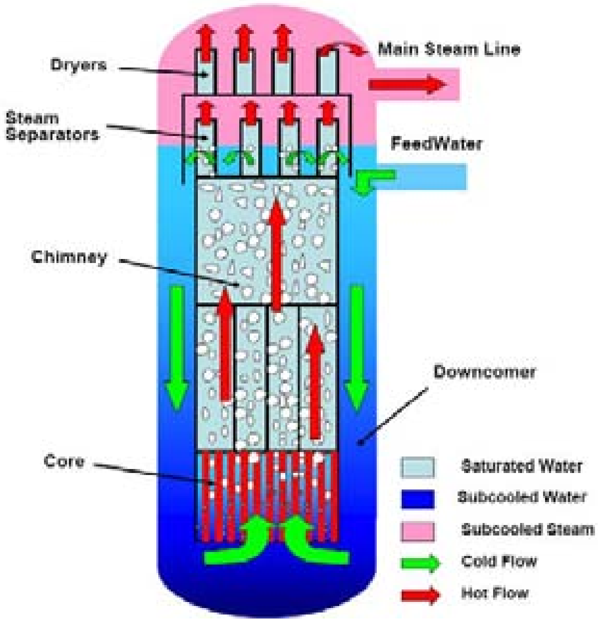

FIG. VI-1. Schematic of the natural circulation flow inside the RPV.

- Gravity driven cooling system (GDCS),
- Automatic depressurization system (ADS), which consists of the depressurization valve (DPV) and safety relief valve (SRV),
- Isolation condenser system (ICS),
- Standby liquid control system (SLCS),
- Passive containment cooling system (PCCS), and
- Suppression pool (SP).

Figure VI-2 presents a schematic of the ESBWR including passive safety system. The GDCS and PCCS are unique to the ESBWR and SBWR. They represent the passive ECCS and containment cooling systems of currently operating BWRs. The ADS is actuated at a prescribed RPV condition and depressurizes the reactor pressure vessel so that the GDCS can be actuated to supply water into the RPV. The goals of the safety systems are to adequately cool the core by maintaining a water level above the active core and to provide a sufficient heat sink via the PCCS and SP in the wetwell (WW) to keep the containment pressure and temperature below the design criteria. The ICS is an engineered safety system, which participates in the most accident transients. It is functionally similar to that in operating BWR and acts as a passive decay heat removal system.

VI-2.1. Gravity-driven cooling system (GDCS)

The GDCS provides emergency core cooling after events that threaten the reactor coolant inventory. Following confirmed RPV water Level 1 signal, the ADS depressurizes the RPV to allow the GDCS injection. Once the reactor is depressurized, the GDCS is capable of injecting large volumes of water into the depressurized RPV to keep the core covered for at least 72 hours following loss of coolant accident (LOCA).

The GDCS requires no external AC electrical power source or operator intervention. The cooling water flows from the GDCS pool to the RPV through simple and passive hydrostatic head. A schematic of the GDCS injection is shown in Figure VI-3. The actual GDCS flow delivered to the RPV is a function of the differential pressure between the reactor and the GDCS injection nozzles, as well as the loss of head due to inventory drained from the GDCS pool. As shown in Figure VI-3, the GDCS can be considered as two separate systems: a short term safety system and long term safety system.

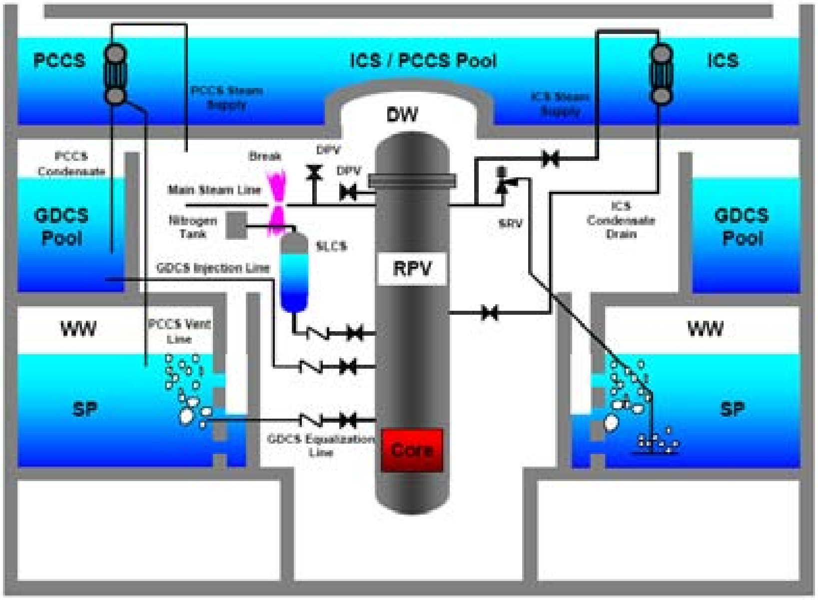

FIG. VI-2. Schematic of connections and safety system in the ESBWR [1].

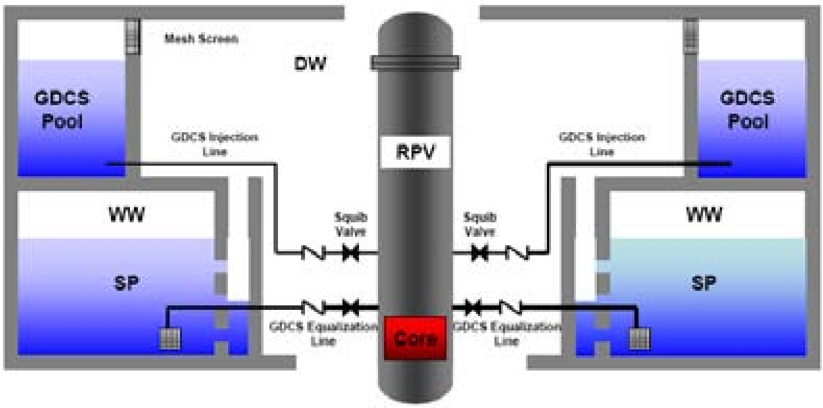

FIG. VI-3. Schematic of the GDCS injection.

The short term safety system is designed to provide short term water makeup to the reactor vessel for maintaining water level higher than the top of active fuel (TAF). There are four identical GDCS drain lines. Each GDCS drain line consists of a pipe exiting from the GDCS pool and squib valves. For short term cooling requirement, each line takes suction from three independent GDCS pools positioned in the upper elevations of the containment. In the ESBWR, there are two GDCS tanks with 602 m^3 and one GDCS tank with 795 m^3. Flow through each drain line is controlled by squib valves, which remain open after initial actuation. Each short term GDCS drain line connects to two GDCS injection nozzles on the RPV.

In case of a pipe break of GDCS lines at lower RPV elevations, large amounts of GDCS water can be lost and water inventory can be reduced close to the TAF. To compensate aforementioned loss of coolant in the GDCS lines, the long term cooling water is provided by a second GDCS subsystem fed by the SP. This is called GDCS equalization system. This GDCS equalization lines are opened after a prescribed time delay so that the short term GDCS pools have time to drain to the RPV and the signal of the initial RPV water inventory reduction as a result of the blowdown does not make the equalizing lines open. For long term event, the GDCS equalization lines are open when the RPV coolant level decreases to 1 m above the TAF. The squib valves are actuated in each of four GDCS equalizing lines, which connect to one RPV injection nozzle per line. The open equalizing lines leading from the SP to the RPV make long term coolant makeup possible. As shown in Fig. VI-3, GDCS equalization line nozzles are placed at a lower elevation on the RPV than those of the short term GDCS, so the GDCS equalization lines make it possible to prevent the core uncovery even though the short term GDCS injection fails. This long term GDCS equalization system also functions through purely passive hydrostatic head differences.

The GDCS pools are placed above the RPV with their air space connected to the DW. This connection effectively increases the DW air space and provides a larger volume for the released gases produced during a severe accident. After the GDCS pools are drained, the total volume of the GDCS pools are added to the volume of the DW air space.

VI-2.2. Automatic depressurization system (ADS)

The ADS is a part of the ECCS. The ADS depressurizes the RPV in the event of LOCA to allow the GDCS water injection into the vessel, preventing the core uncovery. Once the ADS actuates, it continuously operates to keep the reactor depressurized for GDCS injection after an accident initiation. The ADS in the ESBWR is composed of ten safety relief valves (SRVs) and eight depressurization valves (DPVs). The SRVs are mounted on the top of the main steam lines in the DW and discharge the steam through lines routed to quenchers in the SP. Four DPVs are horizontally mounted on horizontal stub tubes connected to the RPV at about the elevation of the main steam lines. The other four DPVs are horizontally mounted on horizontal lines branching from each main steam lines. Main function of the DPVs is to discharge the steam directly from the RPV to the DW in order to depressurize the RPV during the initial phase of LOCA.

The ADS automatically actuates on a reactor low level (Level 1) signal that persists for at least 10 seconds. A two-out-of-four Level 1 logic is used to activate the SRVs and DPVs. The 10-second persistence requirement for the Level 1 signal ensures that momentary system perturbations do not actuate the ADS when it is not required. The two-out-of-four logic ensures that a single failure does not cause spurious system actuation.

VI-2.3. Isolation condenser system (ICS)

During a LOCA, the reactor shuts down and the RPV is isolated by closing the main steam line isolation valves. The ICS removes decay heat after any reactor isolation. In other words, the ICS passively removes sensible and core decay heat from the reactor when the normal heat removal system is unavailable. Decay heat removal limits further increases in steam pressure and keeps the RPV pressure below the safety set point. The arrangement of the IC heat exchanger is shown in Fig. VI-4. The ICS consists of four independent loops, each containing two heat exchanger modules that condense steam inside the tube and transfers heat by heating/evaporating water in the IC pool, which is vented to the atmosphere. This transferring mechanism from IC tubes to the surrounding IC pool water is accomplished by natural convection, and no forced circulation equipment is required.

The ICS is initiated automatically by any of the following signals: high reactor pressure, main steam line isolation valve (MSIV) closure, or an RPV water Level 2 signal. To operate the ICS, the IC condensate return valve is opened whereupon the standing condensate drains into the reactor and the steam water interface in the IC tube bundle moves downward below the lower headers.

VI-2.4. Passive containment cooling system (PCCS)

The PCCS is a passive system which removes the decay heat released to the containment and maintains the containment within its pressure limits for design basis accidents such as a LOCA. The schematic of the PCCS is shown in Fig. VI-5. The PCC heat exchangers receive a steam-gas mixture from the DW, condense the steam and return the condensate to the RPV via the GDCS pools. The noncondensable gas is vented to the WW gas space through a vent line submerged in the SP. The venting of the noncondensable gas is driven by the differential pressure between the DW and WW. The PCCS condenser, which is open to the containment, receives a steam-gas mixture supply directly from the DW. Therefore, the PCCS operation requires no sensing, control, logic or power actuated devices for operation.

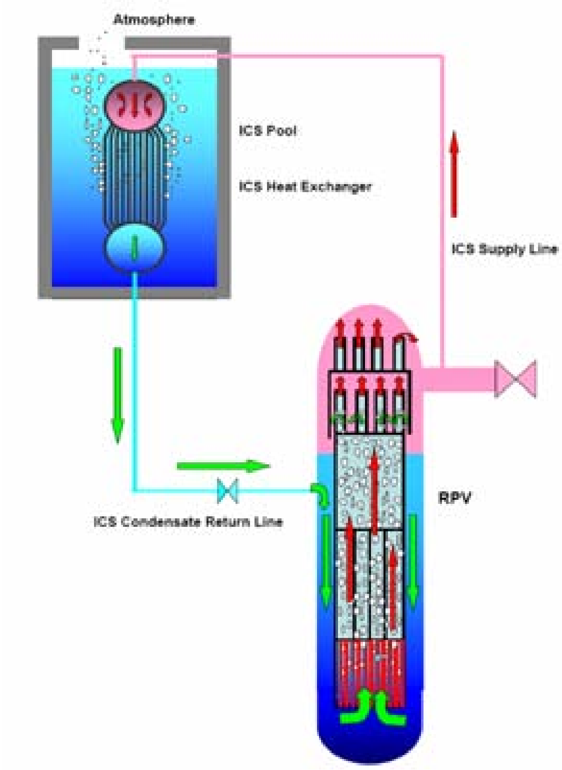

FIG. VI-4. Isolation condenser arrangement.

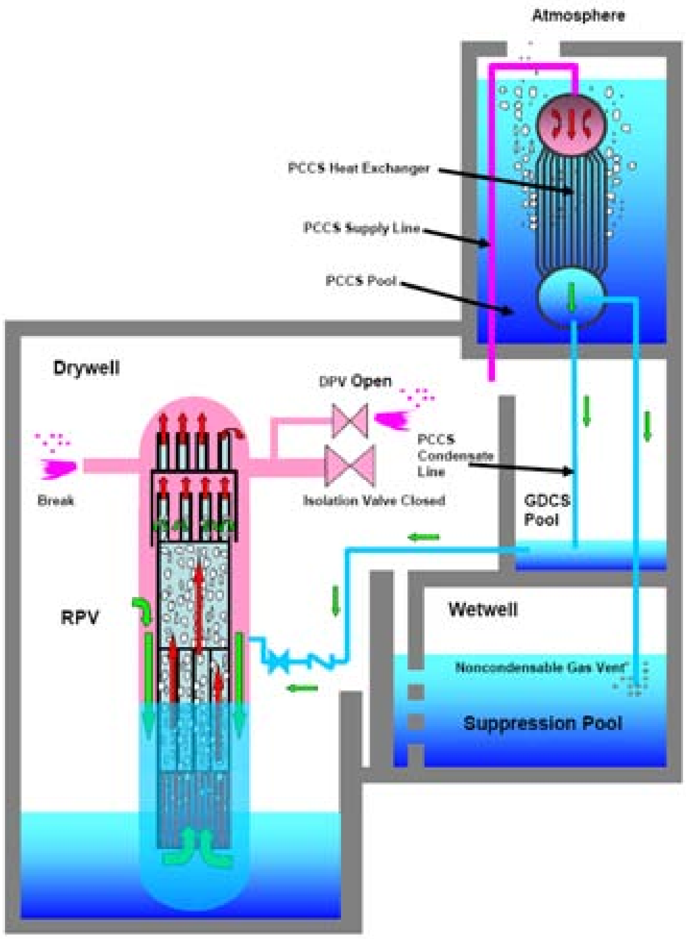

FIG. VI-5. Passive containment cooling condenser arrangement.

The PCCS consists of six PCCS condensers. Each PCCS condenser is made of two identical modules and each entire PCCS condenser two-module assembly is designed for 11 MWt capacity. The condenser condenses steam on the tube side and transfers heat to the water in the IC/PCC pool. The evaporated steam in the IC/PCC pool is vented to the atmosphere. PCCS condensers are located in the large open IC/PCC pool, which are designed to allow full use of the collective water inventory, independent of the operational status of any given PCCS loop.

VI-2.5. Suppression pool (SP) in the wetwell (WW)

The WW is a large chamber with connection to the DW. During the initial blow down, the WW is directly communicated with the DW through the horizontal vents. For long term phase of the LOCA, the WW is communicated with the DW through the PCCS supply lines, condensers and the PCCS vents. When the WW pressure increases above the DW pressure, the vacuum breaker check valves are open. This action ensures that the WW is depressurized by discharging noncondensable gas from the WW to DW and the PCCS is functional. Approximately one-half of the WW volume is filled with a large volume of water that is called SP. The gas space in the WW acts as a receiver for noncondensable gases during a severe accident.

The SP plays an important role in passive safety performance because it provides: 1) A large heat sink, 2) Quenching of steam, which flows through the horizontal vents during LOCA, and 3) Scrubbing of fission products, which flow through the horizontal vents and the PCCS vents.

The WW is directly connected to the DW through twelve vertical/horizontal vent modules. Each module consists of a vertical flow steel pipe, with three horizontal vent pipes extending into the SP water. Each vent module is built into the vent wall, which separates the DW from the WW. The WW boundary is the annular region between the vent wall and the cylindrical containment wall and is bounded above by the DW diaphragm floor.

In the event of a pipe break within the DW, the increased pressure inside the DW forces a mixture of noncondensable gases, steam and water through either the PCCS or the vertical/horizontal vent pipes and into the SP where the steam is rapidly condensed. The noncondensable gases, which are transported with the steam and water, are contained in the free gas space volume of the WW. Performance of the pressure suppression in condensing steam has been demonstrated by a large number of tests.

There is sufficient water volume in the SP to provide adequate submergence over the top of the upper row of horizontal vents, as well as the PCCS return vent. When water level in the RPV reaches at one meter above the TAF, water is transferred from the pool to the RPV through the GDCS equalization lines. Water inventory of the SP and the GDCS pool is sufficient to flood the RPV to at least one meter above the TAF.

VI-2.6. Standby liquid control system (SLCS)

Although the SLCS performs no design basis safety-related functions, it is classified as safety-related system. The SLC system is designed to provide makeup water to the RPV to mitigate the consequences of a LOCA. The ECCS and the SLC are designed to flood the core during a LOCA to provide sufficient core cooling. By providing core cooling following a LOCA, the ECCS and SLCS, in addition to the containment, limit the release of radioactive materials to the environment. The SLCS provides the reactor with an additional liquid inventory in the event of DPV actuation. This function is accomplished by firing squib-type injection valves to initiate the SLCS. The SLCS is operational at the high pressure. The SLCS is pressurized by utilizing a nitrogen charging subsystem including a liquid nitrogen tank, vaporizer, and high pressure pump for initial accumulator charging.

The SLCS is manually initiated for its shutdown function. In addition, the SLCS is automatically initiated for events beyond the safety design basis, such as a LOCA event. The SLCS contains two identical and separate trains. Each train provides 50% injection capacity. In addition to providing water inventory to the RPV, the SLCS removes the remaining reactivity by injecting boron solution into the RPV after the reactor shutdown.

VI-3. Integrated passive safety system performance during the LOCA

The most effective means of describing the function of each of these passive safety systems is to relate their operation in response to a LOCA. As shown in Fig. VI-6, the GE divides the LOCA transient into three regions: 1) blowdown period, 2) GDCS period, and 3) long term PCCS period.

Throughout overall LOCA transient and ECCS operation, the long term core decay heat is removed in three steps. First, the GDCS injects water into the RPV, removing core energy by boiling and venting steam into the DW through DPVs, which remain open once activated. Heat is removed from the core by natural circulation flow within the RPV. Second, the PCCS transfers energy from the DW to the PCCS/ICS pools by condensing steam from the DW in the PCCS condensers. Third, the PCCS/ICS pools transfer their energy to the atmosphere outside the containment by vaporizing pool water and venting it. The PCCS also feeds the condensate to the GDCS pools and vents noncondensable gases to the SP which enhances condensation in the PCCS pools.

There are two natural circulation mechanisms during a LOCA event. One natural circulation mechanism is natural circulation inside the RPV. Once the core is shut down, the natural circulation flow is established inside the RPV to cool down the core. The other natural circulation mechanism is natural circulation through the RPV, DW, PCCS and GDCS. The steam ejected to the DW, is condensed in the PCCS and returned to the RPV via the GDCS.

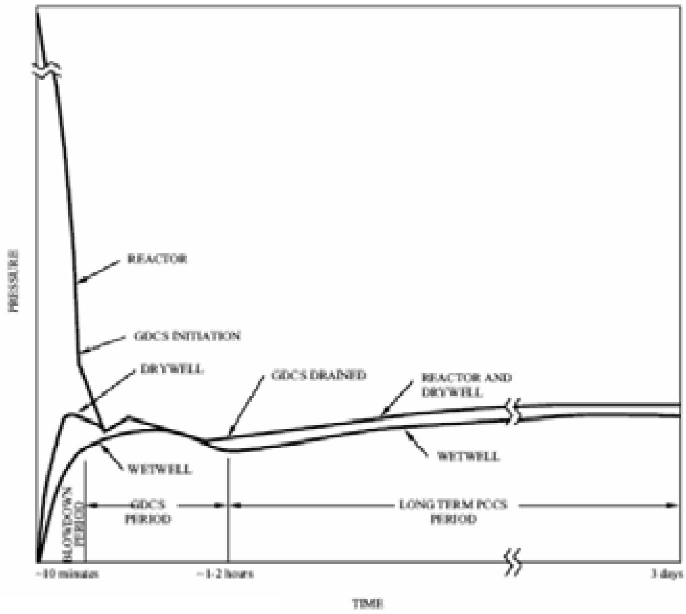

FIG. VI-6. Integral passive system responses during LOCA [3].

VI-3.1. Blowdown period

During a LOCA or transient, the control rod drive system (CRDS) shuts down the reactor. After the reactor shutdown, the main concern to safety is how to remove the decay heat. To depressurize the RPV, the ADS is actuated. As shown in Fig. VI-6, the RPV pressure decreases rapidly as soon as the ADS is actuated. The steam vented through the SRVs is sent to the SP where it is condensed, but the steam vented through DPVs goes to the DW. Large amount of steam and noncondensable gases are vented from the DW to the SP through primary horizontal and vertical vents. During this period, the SP is the primary heat sink to prevent over-pressurization of the containment and the DW pressure is adjusted by venting steam through the horizontal vents into the SP.

Since the ADS is actuated at the Level 1 signal, it is necessary to remove the decay heat in order to prevent the over-pressurization of the RPV. The ICS is designed to remove the decay heat although the RPV is at very high pressure. Similarly, the SLCS is designed to makeup the water into the RPV at high pressure before the GDCS injects water.

VI-3.2. GDCS period

The reactor is operated at high pressure (7.2 MPa) which needs to be depressurized to a value closer to the containment pressure in order to gain a driving head between the GDCS tanks and the RPV. The depressurization of the RPV is achieved through the ADS during blowdown period. Once the RPV is depressurized, the GDCS injects water into the RPV without relying on any active systems by using the gravitational head between the GDCS tanks and the RPV. The GDCS functions to remove the core decay heat to avoid the core uncovering and melting. Although the PCCS is actuated during the GDCS injection period, the PCCS is not functional since there are no driving forces between the DW and WW. In other words, the GDCS water cools down the RPV and the containment pressure keeps reducing during this period.

VI-3.3. Long term PCCS period

After the GDCS injection period is terminated, the core starts to boil again due to the long term decay heat. The long term DW pressure increase is mitigated through PCCS condensers. The PCCS removes the core decay heat energy, released to the containment from the RPV, to outside of the containment. Steam condensate from the PCCS is returned to the RPV via GDCS tanks and noncondensable gases are vented to the SP.

The water inventory of the GDCS pools is slowly replenished with condensate draining from the PCCS. For all design basis events, the closed loop of PCCS condensation and GDCS drainage to the RPV results in long term coverage of the core. Beyond design basis events, the GDCS equalization flow may be necessary where multiple failures are assumed. When the vessel water level reaches 1 m above TAF and at least 30 minutes has passed since the Level 1 signal is confirmed, the GDCS equalization line will open to inject water from the SP to the RPV.

VI-4. Conclusion

The ESBWR program is based on the earlier SBWR program, which was sponsored by the US Department of Energy (DOE). The ESBWR program was started in 1993 to improve the economics of the SBWR. There are some significant differences between the SBWR and ESBWR designs. The major differences are: 1) increased core thermal power, 2) much higher core power density, 3) considerably reduced DW and WW volumes relative to the reactor power, 4) increased GDCS tank volume, and 5) increased component numbers of the PCCS and ICS in the ESBWR compared with those in the SBWR. In addition to these, the number of the main steam lines was increased to four in the ESBWR. A multi-year, four-phase program was defined to complete the technology, develop a detailed design, and secure certification with regulatory bodies. Evaluation of the overall design showed that the plant was considerably simplified and that the overall material quantities were significantly lower than those for the SBWR design and other GE designs.

The ESBWR is equipped with a passive safety system that is basically similar to that of the SBWR. This reactor design features the simplification of the coolant circulation system and implementation of passive safety system. There are several engineered safety systems and safety-grade system in the ESBWR which are directly related to the relevant issues and objectives of the present program. The performance of these safety systems under a LOCA and other important transients is a major concern. Since the ECCS is driven by the gravitational head, interactions between the ADS, GDCS, PCCS and other auxiliary systems are important. The safety systems and various natural circulation phenomena encountered after initial blowdown in the ESBWR are somewhat different from the system and phenomena studied by the nuclear community in the existing commercial nuclear reactors.

Comprehensive integral system and separate effects testing have been conducted to verify the functionality of passive safety system [1]. GE performed tests to assess the GDCS performance in a low pressure full-height GIST facility. Results of this study demonstrated the feasibility of the GDCS concepts. GE also performed tests to assess the PCCS performance in a low pressure, full-height Toshiba GIRAFFE facility in Japan. A PANDA facility in Switzerland, with a low pressure and full-height, was built for testing the PCCS performance and containment phenomena in the SBWR. Later, PANDA was modified to partially simulate the ESBWR configuration.

Purdue University designed and constructed an integral test facility, called PUMA (Purdue University Multi-dimensional integral test Assembly), sponsored by the U.S. NRC. Originally, the PUMA facility was designed to address the functionality of SBWR safety system in 1994. The facility was modified to simulate the safety system in the ESBWR in 2006. The facility contains all of the important safety systems of the ESBWR that are pertinent to the postulated LOCA transient.

The design and technology program for the ESBWR involves several utilities, design organizations and research groups. In mid 2002, the technology base of the ESBWR was submitted to the U.S. NRC for review with the objective of obtaining closure of all technology issues. This was a first and necessary step toward obtaining NRC design certification.

REFERENCES TO ANNEX VI

[1] ISHII, M., et al., Second Scaling and Scientific Design Study for GE ESBWR Relative to PUMA Facility with Volume Ratio of 1/475, Purdue University Report PU-NE-04-04 (2004).

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ANNEX VII. LSBWR
Toshiba Corporation, Japan

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Long operating cycle Simplified Boiling Water Reactor (LSBWR) *Toshiba Corporation, Japan*	BWR	900	CORE: • Gravity Driven Core Cooling System CONTAINMENT: • Passive Containment Cooling System • Suppression Pool

VII-1. Introduction

The long operating cycle simplified boiling water reactor (LSBWR) is a modular boiling water reactor (BWR) plant that is designed by Toshiba Corporation. The reactor concept described in this section has a small power output, a capability of long operating cycle, and a simplified BWR configuration with comprehensive safety features. To be economically competitive, simplification of systems and structures, modularization for short construction period, and improvement in availability are included into the LSBWR design. For comprehensive safety features, the aim is to need no evacuation by utilizing highly reliable equipment and systems such as large RPV inventory, bottom located core layout, in-vessel retention (IVR) capability and passive emergency core cooling system (ECCS) and primary containment vessel (PCV) cooling. Figure VII-1 shows conceptual drawing of the LSBWR.

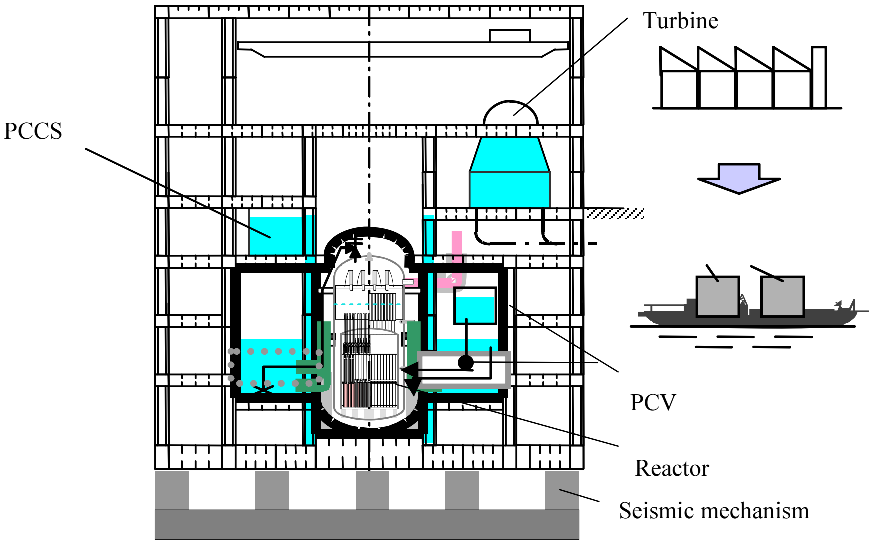

FIG. VII-1. Conceptual drawing of the LSBWR.

VII-2. Description of passive core cooling system

Natural circulation core cooling is applied for eliminating recirculation pumps. This results in high reliability in operation. For attaining natural circulation core cooling, the fuel length is shortened to 2.2m from the conventional 3.7m to decrease the pressure drop. Figure VII-2 shows the core and fuel bundle for the LSBWR. Figure VII-3 shows the LSBWR reactor internals and configuration.

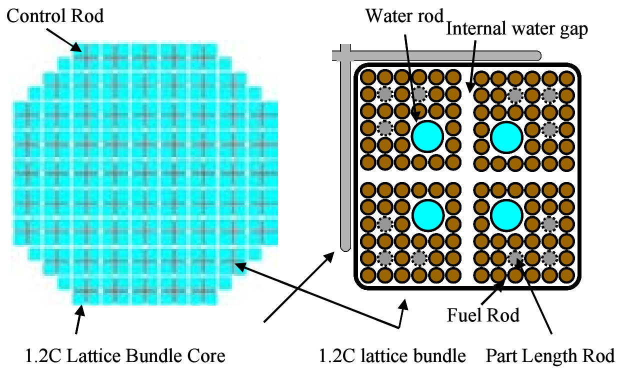

FIG. VII-2. Core and fuel bundle for the LSBWR.

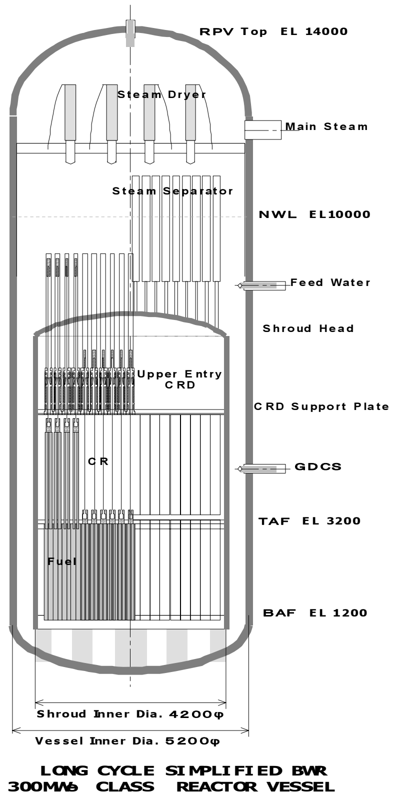

FIG.VII- 3. LSBWR reactor concept.

VII-3. Description of passive containment cooling system (PCCS)

The cylindrical type drywell with small diameter can be designed by routing the safety relief valve piping through the spacing between the RPV and the drywell wall and the main vent pipe from the RPV top to the suppression pool. The drywell air space is minimized and contains only SRV and depressurization valve (DPV) components, the gravity driven core cooling system (GDCS) and drywell loading piping. Since MS and FW piping is routed through suppression pool air space, which are protected by the guard pipe, GDCS piping is contained in the access tunnel placed in the lower part of suppression pool, and isolation valves are installed outside PCV etc.

Since the reactor core is placed at the bottom of the RPV, the emergency coolant injection system consisted of DPV and GDCS can achieve high reliability of the water coverage of the reactor core following an accident.

The containment wall with ship hull structure is filled with cooling water that is boiled off to the atmosphere to cool the PCV passively during an accident. This containment wall cooling system is also used for the drywell cooling during the normal operation and therefore the drywell arrangement is simplified without drywell cooling components used in the current BWR containment.

When cooling water in the PCCS pool above PCV is exhausted, external pool or seawater is supplied by gravitation so that the highly reliable and long term PCV cooling is achieved.

The double cylindrical raised suppression pool with the ship hull structure is installed around the cylindrical drywell and above the core elevation. This makes the structure stronger and simpler, and the suppression pool water can be easily used for GDCS and drywell lower part flooding. LSBWR safety system concept including PCCS is shown in Figure VII-4.

The performance of the safety system has been analyzed for a feedwater line break accident. The analysis has been performed using TRAC code incorporated with the heat transfer models for the natural convection cooling and the steam condensation cooling with a noncondensable, which have been used to estimate the heat transfer coefficients in the containment space and the containment wall coolant channels. The analysis results for the containment pressure and the heat removal rate by the passive containment cooling system are shown in Figure VII-5 and VII-6. After taking its peak value during the blowdown phase, the containment pressure keeps decreasing while the GDCS coolant flow is sufficient to suppress the steam production in the reactor core. The containment pressure begins to increase around 3 hours since the GDCS flow decreases and the steam is produced by the decay heat. The pressure increase is, however, suppressed by the containment wall cooling and is maintained well below the design pressure for 24 hours. The heat removal rate of the containment wall cooling becomes almost comparable with the decay heat after 12 hours. The condensate produced by the containment wall cooling flows from the drywell to the RPV through the GDCS injection line and the reactor core is kept covered.

VII-4. Conclusions

The LSBWR design is still at the conceptual design stage and licensing reviews have not yet been started. Recently, Compact Containment BWR development is going on based on various experience obtained at LSBWR development.

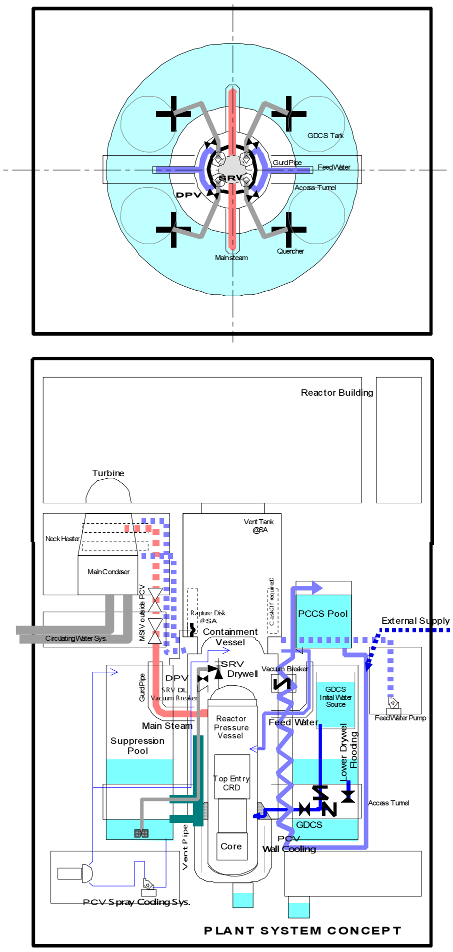

FIG. VII-4. LSBWR safety system concept.

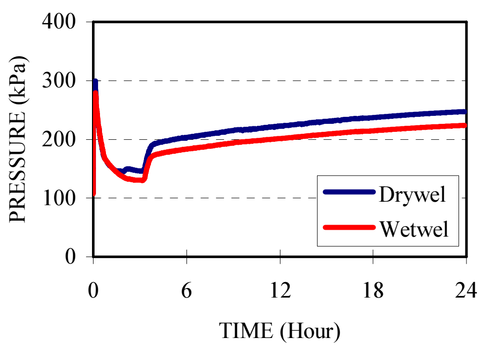

FIG. VII-5. PCV pressure response.

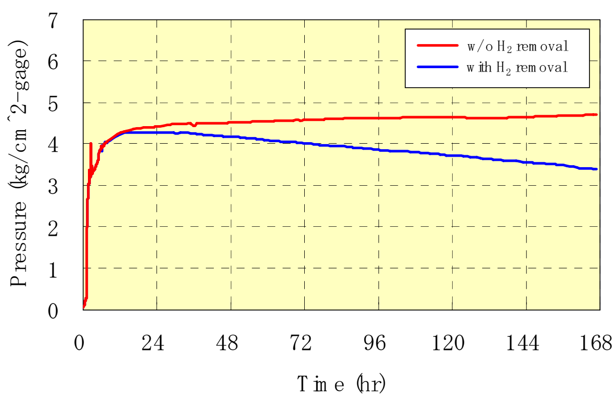

FIG. VII-6. Long term containment pressure transient.

ANNEX VIII. RMWR
Japan Atomic Energy Agency (JAEA), Japan

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Reduced-Moderation Water Reactor (RMWR) *Japan Atomic Energy Agency (JAEA)*	BWR	3926	CORE/PRIMARY: • Isolation Condenser CONTAINMENT: • Passive Containment Cooling System

VIII-1. Introduction

The Reduced-Moderation Water Reactor (RMWR) is a light-water cooled high-conversion reactor that is being developed by the Japan Atomic Energy Research Institute (JAERI) with collaboration from the Japanese industries. The design study conducted so far has indicated that the RMWR can realize the favourable characteristics of high conversion ratio of more than one, high burn-up, long operation cycle, and multiple recycling of plutonium. The design is characterized by the use of the 'double flat core' which consists of two flat core parts and three blanket parts in the vertical direction. This geometry is adopted to increase the neutron leakage from the core to make the void reactivity coefficient negative. The fuel assembly consists of the MOX fuel rods tightly arranged in the triangular lattice with the gap width of typically 1.3 mm to increase the fuel-to-coolant volume ratio.

Although several types of the RMWR systems have been investigated, the current study focuses mainly on the BWR type due to advantages regarding the core performances and the system simplification. Among the BWR type RMWRs, two system designs have been developed for different core powers. The larger one is the 1300 MW(e) RMWR based on the ABWR design (IAEA-TECDOC-1391). Table VIII-1 summarizes the major characteristics of this system, which has 900 fuel assemblies, each of which consists of 217 fuel rods with the outer diameter of 13.7 mm arranged in the triangular lattice in gap width of 1.3 mm (see Figures VIII-1 and VIII-2). The double flat core consists of the lower and upper core parts with the height of 205 and 195 mm, and the lower, medium, and upper blanket parts with the height of 190, 295, and 220 mm as shown in Figure VIII-3. The fissile plutonium-enrichment of the MOX fuel is 18% for the reload fuel at the equilibrium core. The blanket material is depleted UO2.

TABLE VIII-1. MAJOR CHARACTERISTICS FOR THE 1300 MWE RMWR

Item	Unit	Design value
Electric power output	MW(e)	1300
Core outer diameter	M	7.6
Core average burnup	GWd/t	45
Core effective height	M	1.105
Enrichment of reload fuel at equilibrium core	Wt%	18
Fuel cycle length	Month	24

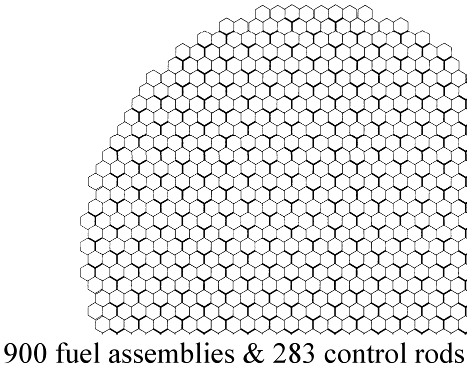

FIG. VIII-1. Core configuration for the 1300 MW(e) RMWR.

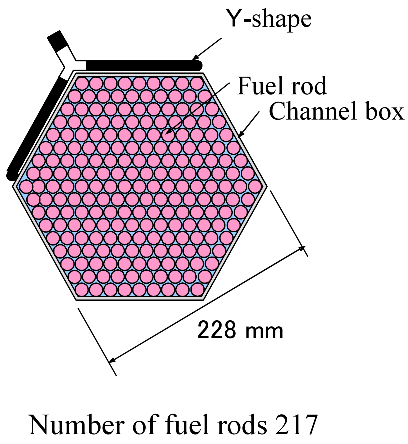

FIG. VIII-2. Fuel assembly.

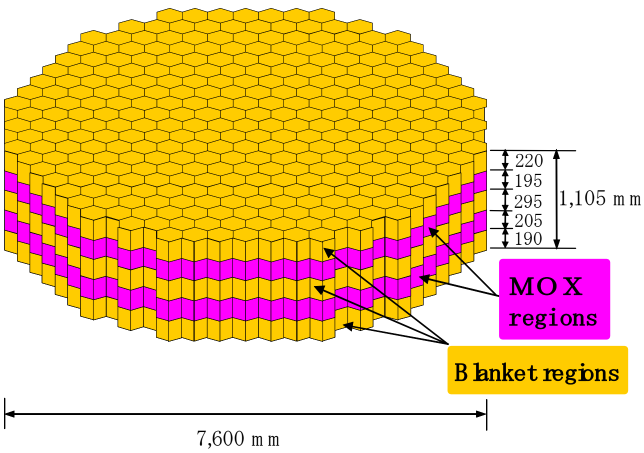

FIG. VIII-3. Schematic of axial core configuration for the 1300 MW(e) RMWR.

The smaller one is the 330 MW(e) RMWR system designed to seek benefits of the low initial capital cost and the flexibility in the plant installation corresponding to the power demand (IWAMURA, T. et al., 2002). To overcome the economic disadvantage for small reactors, the system was simplified by adopting natural circulation core cooling and passive safety features. Since the RMWR utilizes the flat and short length core and operates under lower core flow conditions, the steam velocity in the chimney region is smaller than for the conventional BWR. This characteristic allows further simplification: the steam/water separator and the steam dryer may be eliminated since the gravitational steam/water separation is expected to be possible at the free surface. The major characteristics of the reactor are listed in Table VIII-2. A breeding ratio of 1.01, negative void coefficient and natural circulation cooling of the core were realized under the discharged burn-up of 60GWd/t. The core consists of 282 of hexagonal fuel bundles: each has 217 fuel rods arranged in triangular lattice of 1.3 mm in gap width as listed in Table VIII-2. The core height and outer diameter are 1300 and 4140 mm, respectively, as shown in Figure VIII-4. Figure VIII-5 summarizes the plant system concept. The passive safety features include the accumulator injection system, the isolation condenser, and the passive containment cooling system.

VIII-2. Description of natural circulation core cooling for the 330 MW(e) RMWR

The natural circulation loop consists of the core, the divided chimney, the downcomer, and the lower plenum, which is similar to the other natural circulation cooling BWR concepts such as the SBWR. There are, however, several differences in the characteristics comparing to the SBWR, which includes:

1) One-order smaller absolute value for the void reactivity coefficient, which makes the neutronics-thermal-hydraulic coupling to be almost negligible,
2) Smaller frictional pressure drop across the core because of the shorter core length and the lower core flow rate despite the smaller core flow area
3) Hexagonal cross-sectional shape of the divided chimney
4) Elimination of the conventional separator.

The first characteristic makes the evaluation of the instability problem much simpler. Thus, the characteristic can be regarded as a benefit in view of the thermal-hydraulic analysis. The second one is also considered as a benefit for the realization of the natural circulation system. The third one causes the prediction of the void fraction in the chimney to be difficult because the previous studies are almost nonexistent for this geometry. However, the thermal-hydraulic relationships among the pressure, vapor and liquid flow rates, and void fractions are supposed to be basically the same as those for the circular geometry. So this characteristic is not an essential problem but just requires confirmation tests using the actual geometry of the system to get the relationships for the design finalization. Since the free-surface separation was utilized for the old type natural circulation BWR, the fourth characteristic does not create a new problem. Careful consideation, however, is necessary because the conditions are not completely the same between the old BWRs and the RMWR. The mockup test will be required to confirm especially the phase separation characteristics on the free surface.

VIII-3. Description of passive safety systems

The isolation condenser will be utilized for the 330 MW(e) RMWR, which is basically the same as the conventional one and is, therefore, not described in here. The passive containment cooling system (PCCS) with horizontal heat exchangers will be utilized for the large-size RMWR, details of which are described in the section for the ABWR-II in this report.

TABLE VIII-2. MAJOR CHARACTERISTICS FOR THE 330 MWE RMWR

Item	Unit	Design value
Electric power output	MW(e)	330
Core circumscribed radius	M	2.07
Core average burn-up	GWd/t	60
Core effective height	M	1.3
Core exit quality	%	52
Core void fraction	%	69
Core pressure drop	MPa	0.04
Enrichment of reload fuel at equilibrium core	%	18
Conversion ratio	-	1.01
Max. power density	kW/m	42
MCPR	-	1.3
Void reactivity coefficient	10^{-}	-0.5
Fuel cycle length	month	24

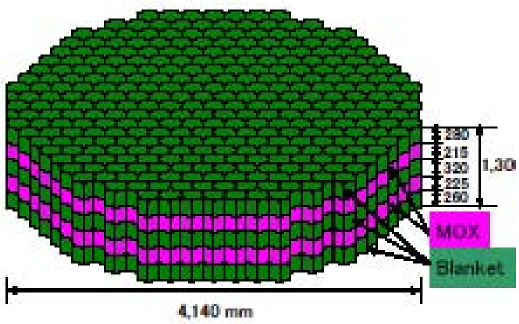

FIG. VIII-4. Schematic of axial core configuration for the 330 MW(e) RMWR.

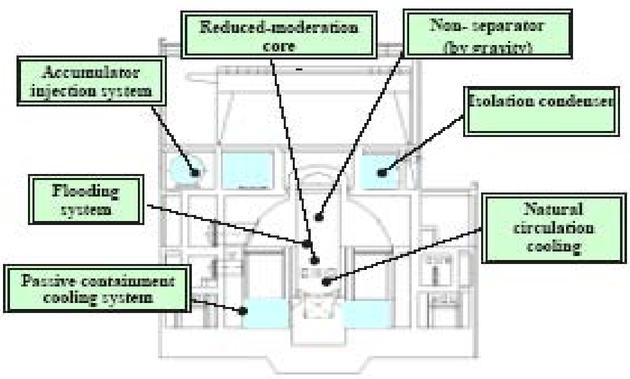

FIG. VIII-5. Plant system concept for the 330 MW(e) RMWR.

VIII-4. Conclusions

Research and development (R&D) activities for the RMWR have been conducted by the JAERI in collaboration with the Japanese industries, which have clarified the favourable characteristics of the reactor including high conversion ratio of more than one, high burn-up, long operation cycle, and multiple recycling of plutonium. So far, the R&D has been conducted under several domestic frameworks, including a) the research corporation program between the Japan Atomic Power Company (JAPC) and JAERI where many RMWR systems were studied, b) the innovative and viable nuclear energy technology (IVNET) development project where the 330 MW(e) RMWR was developed with JAPC, Hitachi Ltd. and Tokyo Institute of Technology (IWAMURA, T., 2002), c) the program sponsored by the ministry of education, culture, sports, science and technology (MEXT) on the innovative nuclear reactor technologies where the bundle core heat transfer tests are conducted together with broad ranges of R&D for fuel and neutronics (OHNUKI, A., 2004, KURETA, M., 2004, YOSHIDA, H., 2004), and d) the feasibility study on the commercialized fast reactor cycle systems conducted by the Japan nuclear cycle development institute. In addition to the above domestic frameworks, the JAERI entered into the agreement with the USDOE on the tight lattice core design in December, 2004.

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ANNEX IX. SBWR
General Electric, USA

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Simplified Boiling Water Reactor (SBWR) *General Electric, USA*	BWR	2000	CORE/PRIMARY: • Gravity-Driven Cooling System • Automatic Depressurization System • Isolation Condenser System CONTAINMENT: • Passive Containment Cooling System • Suppression Pool

IX-1. Introduction

The General Electric (GE) Nuclear Energy has developed a boiling water reactor called the simplified boiling water reactor (SBWR). Major differences between the current boiling water reactors (BWR) and the SBWR are in the simplification of the coolant circulation system and the implementation of a passive emergency cooling system. There are no recirculation pumps to drive the coolant in the vessel of the SBWR. The emergency core cooling and containment cooling systems do not have active pump injected flows.

IX-2. Gravity-driven cooling system (GDCS)

The emergency core cooling systems (ECCS) of the SBWR are the GDCS and automatic depressurization system (ADS). As part of the ECCS, the GDCS plays a major role in the SBWR safety mechanisms. The GDCS can be considered as two separate systems: a short term safety system and long term safety system.

The short term safety system is designed to provide short term water makeup to the reactor vessel for maintaining fuel cladding temperatures below safety limits. Three separate water pools, located within the upper drywell (DW) at an elevation above the active core region, provide gravity-driven water makeup to the reactor vessel. The ADS provides depressurization with the Safety Relief Valves (SRVs) and Depressurization Valves (DPVs). Once the reactor is depressurized, the GDCS provides gravity-driven flow from three separate water pools located in the DW at an elevation above the active core region. The GDCS actuation signal is related to the confirmed Level 1 signal. Level 1 is 3.93m higher than the level of the top of the active fuel (TAF). Three squib valves are activated, 150 seconds after confirmed Level 1 signal, one in each of the injection lines connecting the GDCS pools to reactor pressure vessel (RPV).

The long term GDCS safety system is designed to provide long term vessel cooling by keeping the core region covered with water, again through gravity-driven flow. This is accomplished through three GDCS equalization lines connecting the WW to the RPV. Each line is independent and designed to open when the water level in the RPV reaches 1 m above the TAF and at least 30 minutes has passed after Level 1 confirmation. After sufficient time delay has passed since Level 1 confirmation, and when the RPV coolant level decreases to 1 m above the TAF, squib valves are opened in all three equalization lines. The time delay and the above criterion ensure that the GDCS pools have had time to drain into the RPV and enough time has passed since Level 1 is confirmed. In Table IX-1, the GDCS parameters are given. As shown in Table IX-1, the minimum equalization line driving head of 1 m is determined by the elevation difference between the top of the first WW horizontal vent and the centerline of the equalization line RPV nozzle. The 193.7 mm (8 inch Sch. 80) lines from each GDCS pool branch out into two 146.3 mm (6 inch Sch. 80) lines just before they enter RPV.

TABLE IX-1. GDCS AND EQUALIZATION LINE PARAMETERS [1]

GDCS Pool Numbers	3
Each GDCS Pool Minimum Drainable Inventory	329 m^3
Minimum Surface Elevation of GDCS Pool above the RPV Nozzle	13.3 m
SP Inventory 1 Meter above TAF	1475 m^3
Minimum Equalization Line Head	1 m
GDCS Line Size from GDCS Pool (three total)	193.7 mm, Sch. 80
GDCS Line Size (six total)	146.3 mm, Sch. 80
GDCS-Injection Line Nozzle Size at RPV (six total)	76.2 mm
Equalization Line Nozzle size at RPV (three total)	50.8 mm

IX-3. Automatic depressurization system (ADS)

The function of the automatic depressurization system (ADS) is to systematically depressurize the RPV in the event of LOCA or transient, to allow the GDCS water injection to the vessel, preventing the core uncovery and maintaining the peak clad temperature below design limits. The ADS also keeps the reactor depressurized for continued operation of the GDCS after an accident initiation. The ADS consists of the eight SRVs, six DPVs and their associated instrumentation and controls. There are four SRVs and one squib-type DPV on each main steam line (MSL). Four DPVs are flange-mounted on horizontal stub lines connected directly to the RPV at about the elevation of the MSLs. Three of the four stub tubes have an IC steam supply line connected. The SRVs discharge into the wetwell (WW) through spargers. The DPVs discharge into the upper DW.

The SRVs and DPVs are actuated in several groups at staggered times as the reactor undergoes a controlled depressurization. This minimizes reactor water mixture level swell during the depressurization phase. The ADS is activated when the low water level (Level 1) signal persists for at least 10 seconds. First, four SRVs (two from each MSL) open and discharge steam to the SP. The remaining four SRVs open after an additional 10-second time delay. At 55 seconds after ADS actuation, the first group of two DPVs (on MSLs) starts to discharge to the DW. Likewise, the second group of two DPVs opens after 100 seconds and the third group of two DPVs opens after 145 seconds of ADS actuation. In Table IX-2, SRV and DPV parameters are given.

TABLE IX-2. SRV AND DPV PARAMETERS [1]

SRV Inlet Line Diameter	193.7 mm
SRV Outlet Line Diameter	242.9 mm
SRV Minimum Flow Area	67 cm^2
DPV (MSL) Inlet Line Diameter	257.2 mm
DPV (RPV) Inlet Line Diameter	366.7 mm
DPV Minimum Flow Area	248.5 cm^2

IX-4. Isolation condenser system (ICS)

The ICS removes core decay heat from the reactor by natural circulation. It can function with minimum loss or no loss of coolant inventory from the reactor when the normal heat removal system is unavailable. For example, it can be activated for the following events: (1) sudden reactor isolation from power operating conditions, (2) reactor hot standby mode, and (3) safe shutdown condition.

The ICS consists of three independent high pressure loops, each of which contains a steam isolation condenser (IC). The steam is condensed in the vertical tube section and transfers the heat to a large ICS/PCCS pool by evaporating water to the atmosphere. Each IC is designed for 30 MW•th capacity and consists of two identical condenser modules.

Each IC is located in a sub-compartment of the ICS/PCCS pool, and all pool subcompartments are interconnected with each other by low-elevation pipe runs and openings that penetrate through the lower portions of pool sub-compartment walls. This arrangement ensures that the entire ICS/PCCS pool water inventory is made available to each IC heat exchanger. Steam condenses inside vertical tubes and is collected in the lower header. One pipe from each of the two lower headers carries the condensate to a common drain line leading to the RPV. Also, a small purge line connected to the steam inlet piping and leading to a point in the MSLs is normally open. When a reactor is supplying steam to the turbine, a pressure differential between the intake to this purge line and the MSL exists. Thus, this drives a continuous small live steam flow through the IC heat exchanger steam supply line and out to the MSL, sweeping out any non-condensable gases that otherwise may accumulate in this steam supply line. Non-condensable gases are purged through vent lines into the WW. The purging operation in the ICS vent lines is performed manually by an operator; hence, it is not passive as in the PCCS vent lines.

During LOCA transients, the ICS is activated when the reactor water level falls below Level 2 which is 7.93 m higher than the TAF level. In Table IX-3, the ICS/PCCS pool and ICS parameters are given. Note that the ICS supply inlet shares a stub line with one of the DPVs.

TABLE IX-3. IC POOL AND ICS PARAMETERS [1]

IC Pool: Depth Air Space Volume above top of tubes	 4.4 m 1165.83 m^3 1250 m^3
IC Inlet Line Size (from DPV Stub/Tube)	242.9 mm
IC Condensate Return Line Size	146.3 mm
IC Vent Line Size	18.9 mm
Number of Units	3
Modules per Unit	2
Condenser Tube: Length OD ID Number of Tubes per module	 1.8 m 50.8 mm 46.6 mm 120

IX-5. Passive containment cooling system (PCCS)

The PCCS is an engineered safety feature and therefore it is a safety-related system. The PCCS removes the core decay heat energy, rejected to the containment after a loss of coolant accident (LOCA), to outside of the containment. It provides containment cooling for a minimum of 72 hours after a LOCA. The PCCS consists of three PCCS condensers. The condenser is sized to maintain the containment within the design pressure limits of 483 kPa (70 psia) for design basis accidents (DBAs). The PCCS is designed as a passive system without power actuated valves or other components that must be activated during the accident.

Each PCCS condenser assembly is composed of two identical condenser modules. One PCCS condenser assembly is designed for 10 MW•th capacity under conditions of saturated steam in tubes at 412 kPa (50 psia) and 134 °C and pool water at atmospheric pressure and 101 °C. The noncondensable gas purging system is driven by the pressure difference between the DW and WW.

The PCCS related parameters are given in Table IX-4. Each PCCS condenser has two identical modules with 248 tubes per model. The tubes are 1800 mm in height, with 50.8 and 47.5 mm O.D. and I.D., respectively. Each PCCS condenser is connected to a 254.5 mm (10 inch Sch. 40) line that vents the noncondensables to the WW, and a 154.1 mm (6 inch Sch. 40) line that returns the condensed water to the GDCS pool. The inlet supply to the PCCS condenser is 254.5 mm (10 inch Sch. 40) line. This inlet is always open to the DW to allow free flow of steam/gas from the DW to the PCCS condenser tubes. The vertical condenser tubes of the PCCS modules are connected between two drums acting as the inlet and outlet plenum. The driving head of the PCCS is provided by the pressure difference between the DW and WW. There are no valves, pumps or fans in the PCCS, which makes it a passive system by design.

TABLE IX-4. PCCS PARAMETERS [1]

Number of Units	3
Modules per Unit	2
Tubes per Module	248
Total Heat Transfer Area Inside/Outside	400/430 m^2
Total Flow Area	2.6 m^2
Condenser Tube: Length OD ID Material	 1.8 m 50.8 mm 47.5 mm Stainless Steel
Headers: Length OD	 2.4 m 750 mm

IX-6. Suppression pool (SP) in the wetwell (WW)

The SBWR has a low-leakage containment which is divided into the DW and WW. The WW is comprised of a gas volume and suppression pool water volume. The containment is a cylindrical, steel-lined, reinforced concrete structure integrated with the reactor building. The WW design conditions are 483 kPa (70 psia) and 121 ^{0}C. The WW is higher in elevation than the top of the core. This provides a gravitational driving head for injecting WW water into the vessel when the vessel is depressurized and the equalization lines (total of three) between the WW and the vessel are opened.

The gas space in the WW serves as the LOCA blowdown gas reservoir for the upper and lower DW nitrogen and other noncondensable gases, which pass through the eight DW-to-WW vertical vent pipes. Each vent pipe has three horizontal vents located below the WW pool surface. There are a total of 24 horizontal vents between the DW and WW. To prevent the over-pressurization of the WW relative to the DW, there is a vacuum breaker system between the WW gas space and the DW. The vacuum breakers consist of check valves which open when the WW pressure exceeds the DW pressure at a preset pressure difference. In Table IX-5 relevant WW parameters are given.

TABLE IX-5. WW PARAMETERS [1]

WW Gas Volume	3819 m^3
WW Water Volume	3255 m^3
WW Vertical Vent Area	9 m^2
WW Surface Area	588 m^2
Vertical Vent Pipe Inside Diameter	1.2 m
Vertical Vent Pipe Height	12.7 m
Horizontal Vent Diameter	0.7 m

IX-7. Conclusions

There are several engineered safety systems and safety-grade systems in the SBWR which are directly related to the relevant issues and objectives of the present program: 1) the ADS, 2) GDCS, 3) PCCS, and 4) ICS in addition to the standard reactor scram systems and containment which can be divided into the DW and WW. The GDCS and PCCS are new designs unique to the SBWR and do not exist in operating BWRs. The ICS is functionally similar to those in some operating BWRs. Both GDCS and PCCS are designed for low pressure operation (less than 1.03 MPa), but the ICS is capable of high pressure operation as well (up to 7.58 MPa).

The performance of these safety systems under a LOCA and other important transients is a major concern. Since the emergency core cooling systems are driven by the gravitational head, interactions between the ADS, GDCS, PCCS and other auxiliary systems are important. The emergency core cooling systems depend not only on the gravitational head but also on the relative static pressure differences among the RPV, DW, and WW. The safety systems and various natural circulation phenomena encountered after the initial vessel depressurization in the SBWR are somewhat different from the system and phenomena studied by the nuclear community in the existing commercial nuclear reactors.

GE has performed tests to assess the GDCS performance in a low pressure full-height GIST facility with a volume scale of 1/508 [2]. Results of this study have demonstrated the feasibility of the GDCS concepts. The GIST facility was scaled from an old SBWR design, in which the GDCS pools were combined with the WW. The PCCS was absent in the GIST facility, hence parallel operation of the GDCS and the PCCS was not observed in the GIST experiments. GE has also performed tests to assess the PCCS performance in a low pressure, full-height Toshiba GIRAFFE facility in Japan with a volume scale of 1/400 [3]. The GIRAFFE tests provided data to help model the prototypic SBWR PCCS units, and demonstrated the feasibility of the noncondensable venting concept. However, the GRAFFE facility was scaled from an older SBWR design, and it did not investigate GDCS injection in the vessel.

A PANDA facility in Switzerland is a low pressure, full-height facility with a volume scale of 1/25[4]. The main focus of the PANDA facility is on PCCS performance and containment phenomena in a relatively large-scale facility so that three-dimensional effects can he assessed. Like GIRAFFE, however, the PANDA facility is not designed for assessing GDCS injection into the vessel. Although GE has performed experimental and analytical studies for the PCCS and GDCS systems and associated phenomena, the U.S. Nuclear Regulatory Commission (NRC) has identified a need to develop additional independent confirmatory data from a well-scaled integral test facility built to reproduce major thermal-hydraulic phenomena at relatively low pressure (<1.03MPa).

Purdue University has designed and constructed an integral test facility, called PUMA (Purdue University Multi-Dimensional Integral Test Assembly), sponsored by the U.S. NRC. The PUMA facility has a scale of 1/4 in height, 1/400 in volume, and a time scale of 1/2 [5]. The power is scaled by 1/200 of the prototype and pressure is scaled 1:l. The facility contains all of the important safety and non-safety systems of the SBWR that are pertinent to the postulated LOCA transient. The PUMA has adequate instrumentation to provide phenomena comprehension, quantification, evaluation, and can be used for model development and assessment of the RELAP5 code. Three kinds of LOCAs, i.e. MSL break, bottom drain line break, and GDCS drain line break, were conducted and compared with the RELAP5 code with good agreement.

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ANNEX X. SCWR-CANDU
Atomic Energy of Canada Ltd, Canada

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
SCWR-CANDU AECL, Canada	SCWR	2540	CORE/PRIMARY: Two independent shutdown systems (spring-assisted, gravity-driven shutoff rods and pressure driven poison injection) • Core make-up tanks • Reserve water system • Passive Moderator Cooling System CONTAINMENT • Passive Containment Cooling System

X-1. Introduction

The CANDU®-SCWR is a pressure tube SCWR reactor concept under development by AECL as part of the Canadian Generation IV program. The main mission of this reactor is expected to be electricity production but other non-electricity applications such as hydrogen production and process heat are also being investigated. Figure X-1 shows a possible layout of the plant. Preliminary design parameters are shown in Table X-1 [2].

The coolant enters the reactor core into the individual pressure tubes at 25MPa and a subcritical temperature of 350°C. The temperature of the coolant rises above the critical point along the fuel channels and exits at about 625°C. The coolant enters the supercritical turbine, which is compact and can be placed inside the containment. Depending on the mission of the reactor, the high- energy stream from the supercritical turbine outlet can be used to generate electricity using conventional turbines or can be used for non-electricity applications.

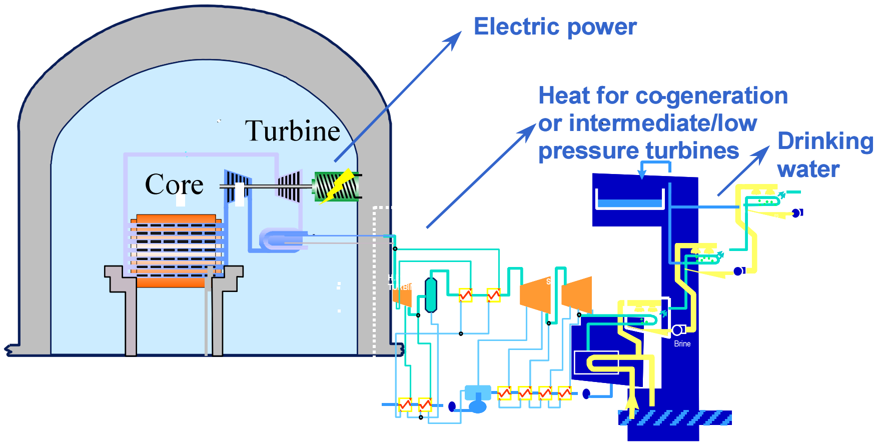

FIG. X-1. CANDU SCWR schematic.

® CANDU is a registered trademark of Atomic Energy of Canada Ltd (AECL).

TABLE X-1. CANDU-SCWR PRELIMINARY SPECIFICATIONS

Spectrum	Thermal
Thermal Power, MW	2540
Electric Power, MW	1220
Thermal Efficiency,%	48
Pressure, MPa	25
Inlet Temperature, ^oC	350
Outlet Temperature, ^oC	625
Flowrate, kg/s	1320
Calandria Diameter, m	4
Fuel	UO_2/Th
Enrichment,%	4
Number of Fuel Channels	300
Number of Fuel Elements	43 or 61
Cladding Material	Ni alloy
Cladding Temperature, ^oC	< 850
Moderator	Heavy water
Coolant	Light water

The CANDU-SCWCR is similar to a typical CANDU reactor but with the following major differences:

- The lattice pitch is tighter to reduce heavy water cost,
- The coolant is light water at supercritical conditions,
- Uses a modified fuel channel design with internal insulation to accommodate the higher coolant pressure and temperature.

X-2. CANDU-SCWR passive safety systems

The safety systems that will be employed in the CANDU-SCWR will be based on those employed in the ACR[TM] design [3]. However, an additional passive safety system that utilizes the separation between moderator and coolant in the CANDU design will also be used with enhancements that are expected to significantly reduce the probability of core damage.

X-2.1. ACR Passive safety systems

These systems are described elsewhere in this document for the ACR-1000 design. Differences between the CANDU-SCWR and the ACR-1000 will be taken into consideration when selecting which of these systems will be selected in the reference CANDU-SCWR design. For example, the reserve water system (RWS) will not be used to provide cooling via the steam generators since the CANDU-SCWR uses a direct cycle. Implementation of the following ACR passive safety systems in the CANDU-SCWR is currently under evaluation:

[TM] ACR™ (Advanced CANDU Reactor™) is a trademark of Atomic Energy of Canada Ltd (AECL).

a) Two independent shutdown systems: these consist of shutoff rods and liquid injection shutdown systems. Both systems are located in the low pressure moderator system and are driven either by gravity (spring-assisted shutoff rods) or pressurized gas (poison liquid injection).

b) Emergency coolant injection (ECI) system: the ECI system consists of accumulator tanks filled with make-up water and pressurized by compressed gas to provide emergency cooling when the core pressure falls below the pressure of the accumulators.

c) Reserve water system (RWS): this system provides gravity-driven cooling water from the reserve water tank (RWT) to several systems such as the containment spray cooling system and the moderator cooling system.

d) Containment cooling system: this system consists of containment cooling spray and is actuated automatically for any event resulting in pressures or temperatures that challenge the integrity of the containment. Once actuated, operation of the sprays relies only on gravity to deliver water from the RWT to the spray headers, which are located at a high elevation in the reactor building. Spray nozzles connected to the spray headers diffuse the cooling water into fine droplets, which fall through the containment atmosphere [3].

X-2.2. Passive moderator cooling system

The separation between coolant and moderator in CANDU reactors provides a backup safety system through the moderator cooling system. The moderator cooling system is used to remove heat deposited in the moderator during normal operation due to gamma and neutron heating. This heat, which is approximately 5% of reactor thermal power, is of the same order of magnitude as decay heat shortly following reactor shutdown; therefore, this system can be used to remove decay heat if the emergency core cooling (ECC) system fails.

The moderator cooling system used in existing CANDU reactors uses a pumped loop (see Figure X-2), and its role as a backup safety system can be significantly enhanced if a natural circulation loop is used. A passive moderator cooling system (PMCS) concept for CANDU reactors has been under investigation for the past several years and is described in [4]. In the PMCS concept, the moderator operates close to saturation so that two-phase flow is generated by flashing in a riser that is connected to a heat exchanger (see Figure X-3).

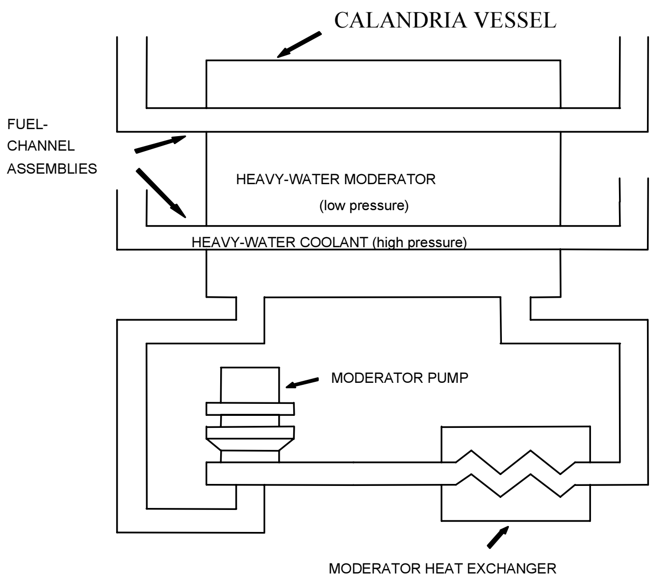

FIG. X-2. Current CANDU moderator cooling system.

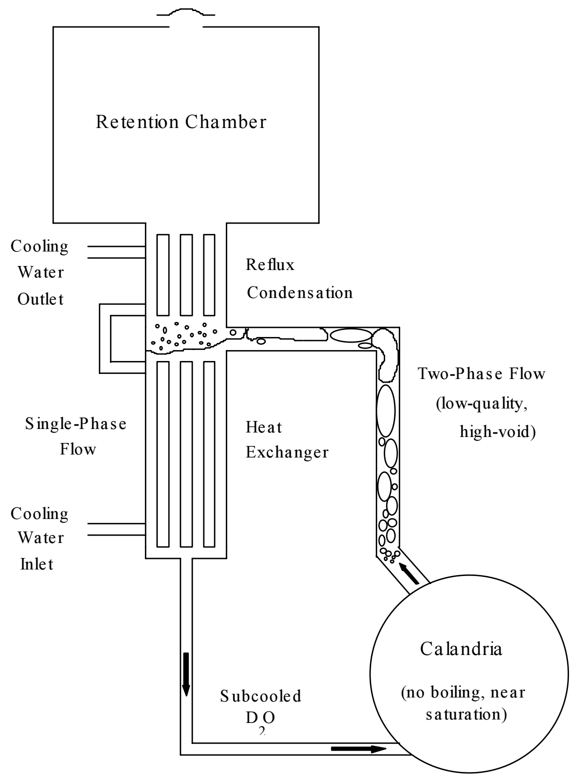

FIG. X-3. Passive moderator cooling system.

The passive loop design in Figure X-3 ensures the following: 1) single-phase exists in the calandria (for effective neutron moderation), and 2) two-phase exists in the riser so that sufficiently high flowrates are achieved for decay heat removal following certain postulated accident scenarios. The stability of the PMCS loop was verified experimentally and compared to code predictions using a scaled loop [4]. For the CANDU-SCWR, the objective is to operate the passive moderator loop during normal operation to ensure that this loop is available following accidents without operator intervention. This is made possible by the design of the CANDU-SCWR fuel channels, which is different from the design used in existing CANDU Reactors. Existing CANDU reactors use a fuel channel design that consists of a pressure tube that is insulated from the cool heavy water moderator by a gas gap and a calandria tube (Figure X-4). In this configuration, the pressure tube operates close to the coolant temperature while the calandria tube operates close to the moderator temperature.

During normal operation, heat is deposited directly in the moderator by direct gamma and neutron heating (~5% of the total thermal power). Heat transfer from the coolant to the moderator is negligible because of the presence of the gas gap between the pressure tube and the calandria tube. The moderator heat load is of the same order of magnitude as the decay heat shortly after reactor shutdown. This makes the moderator an attractive heat sink for decay heat removal following certain postulated accident scenarios provided that heat can be efficiently transferred from the fuel channel to the moderator.

Existing CANDU reactors rely on pumps to remove heat deposited in the moderator during normal operation, and to remove decay heat during certain accident scenarios. Furthermore, existing CANDU reactors require the moderator to operate with a certain degree of subcooling to avoid film boiling on the calandria tube if the pressure tube balloons into contact with the calandria tube following a loss of coolant accident (LOCA) combined with loss of Class IV power [5]. This restriction on the moderator temperature has to be removed in order to implement the passive moderator cooling loop during normal operation. This is possible by the use of an alternate fuel channel design for the CANDU-SCWR, which is required to protect the pressure tube from the high temperature coolant. This alternate fuel channel design is shown in Figure X-5 and utilizes an internally insulated pressure tube.

FIG. X-4. Current CANDU fuel channel.

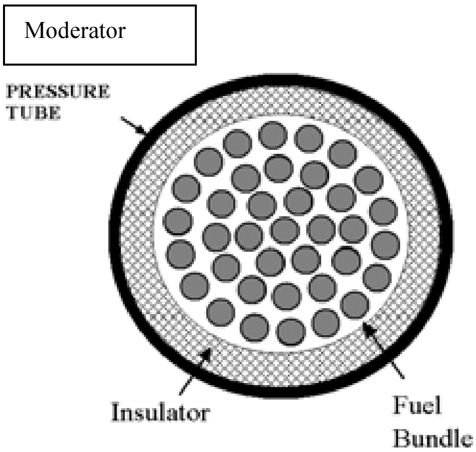

FIG. X-5. CANDU SCWCR fuel channel.

The main differences between the fuel channel design shown in Figure X-4 and that shown in Figure X-5 are: 1) the insulator in Fig. X-5 is on the inside of the pressure tube to keep the pressure tube cooler (close to the moderator temperature), and 2) the calandria tube is eliminated. Elimination of the calandria tube removes the subcooling restriction on the moderator because this restriction is mandated by the design shown in Figure X-4 where the pressure tube could balloon into contact with the calandria tube following certain accident scenarios [5]. Therefore, the moderator could operate close to saturation and the PMCS concept can be used during normal reactor operation. This results in two important advantages: 1) the PMCS is demonstrated to be functional at all times, and 2) there is a potential for cost reduction because of the elimination of the moderator pumps.

X-3. Response to LBLOCA followed by loss of ECC system

A brief description is given here to a postulated large break loss of coolant accident (LBLOCA) followed by loss of ECC. During normal operation, the PMCS rejects heat deposited in the moderator to an ultimate heat sink that could be provided by elevated heat exchangers connected to the RWS. Following LOCA and reactor shutdown, the ECC system, which consists of the ECI and potentially a pumped system for long term cooling will remove decay heat. In the event that the ECC system fails, the fuel gets hot and heat is radiated from the fuel elements to the insulator and is then transferred to the moderator by conduction and convection. Preliminary simulations showed that the insulator thickness could be optimized to provide minimal heat loss to the moderator during normal operation (compared to the 5% loss due to gamma and neutron heating) but sufficient heat conduction during accident conditions.

X-4. Conclusions

The CANDU-SCWR is expected to use existing CANDU passive safety features. In addition, the CANDU SCWCR employs an advanced fuel channel design that makes it possible to use a flashing-driven passive moderator cooling system to enhance the role of the moderator as a backup safety system. A flashing-driven loop can be designed to reject the moderator heat load under normal operating conditions. Since the moderator heat is comparable to the decay heat load immediately following reactor shutdown, this loop can also be used to reject decay heat under accident conditions through radiation to the insulator and conduction through the moderator-cooled pressure tube. This passive loop can be used during normal operation, which guarantees the functionality of the passive loop at all times, and can potentially eliminate scenarios that could lead to severe core damage.

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ANNEX XI. SWR 1000
Areva, France

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
SWR 1000 *Areva, France*	BWR	2778	CORE/PRIMARY SYSTEM: • Emergency Condenser System • Core Flooding System • Passive Pressure Pulse Transmitters CONTAINMENT • Containment Cooling Condensers • Drywell Flooding System

XI-1. Introduction

In 1992, German utilities awarded FRAMATOME (former Siemens) a contract to develop a new BWR nuclear power plant using passive safety systems, and together with the utilities FRAMATOME started development work on a new BWR with a net capacity of 750 MW(e). During the conceptual phase, lasting from February 1992 until September 1993, priority was given to developing passive safety systems to replace or supplement active systems. At the end of the conceptual phase, it was decided that the new requirements for this advanced BWR, especially economic aspects, justified a concept with a reactor thermal output of 2778 MW and a net electrical output of 977 MW. Since 2000 the net electrical output was further increased to 1254 MW.

The four-year basic design phase for the resulting 'SWR 1000' plant started in mid-1995. In parallel, an experimental testing program was conducted at FRAMATOME's own testing facilities and at other German and European research centers to provide verification of the mode of operation and effectiveness of the SWR 1000's passive safety systems.

In the following sections, after a description of the general design principles and the safety systems, the passive safety components that use natural circulation are described in detail. In particular, descriptions are provided for the emergency condenser, the containment condenser and the pressure vessel cooling after a hypothetical severe accident.

XI-2. Design principles

The main goal of this advanced BWR is to replace the active safety systems used in current designs with passive safety systems enabling:

- Reliable control of the various design basis accidents;
- Low probability of beyond-design-basis accidents (core damage frequency);
- Limitation of the consequences of a core melt accident to the plant itself;
- High plant availability;
- Economic competitiveness.

Various features have been changed compared to existing BWR designs, including:

- Larger water inventory in the reactor pressure vessel (RPV) above the core permits passive core cooling;
- Larger water storage capacities inside and outside the reactor containment providing long grace periods and avoiding the need for prompt operator intervention, especially during and after accidents;

- For transients as well as for accident control, emergency condensers and containment cooling condensers to passively remove decay heat from the core and containment, respectively;
- Activation of key safety functions such as reactor scram, containment isolation and automatic depressurization is backed up by passive systems (passive pressure pulse transmitters);
- Passive cooling of the reactor pressure vessel exterior in the event of a core melt accident ensures in-vessel melt retention;
- Despite the introduction of passive safety systems for accident control the operating experience gained from current BWR plants constitutes the basis for the new concept;
- Simplification of reactor auxiliary systems and systems used for normal power operation.

XI-3. General description of the system

Three main steam lines connecting the reactor pressure vessel to the high pressure turbine section serve to transport the steam generated in the reactor to the turbine. Each main steam line inside the containment is allocated a specific number of safety-relief valves for overpressure protection of the reactor pressure vessel. For automatic depressurization, the safety-relief valves are opened either by solenoid pilot valves or by the passive pressure pulse transmitters and diaphragm pilot valves.

The core of the SWR 1000 was reduced in active height and the fuel assembly was enlarged. A consequence of reducing the active core height is that the core can be positioned lower down inside the reactor pressure vessel. This provides a larger water inventory inside the reactor pressure vessel above the core, a feature which facilitates accident control. The fuel assemblies were enlarged resulting in fewer fuel assemblies in the core. This reduces handling times during refueling. Reducing the number of fuel assemblies also reduces the number of control rods, and hence the number of control rod drives as well. The average power density is approximately 51 kW/L.

The reactor pressure vessel has numerous nozzles for connecting the piping of the main steam, feedwater, emergency condenser, shutdown cooling and vessel head spray systems, as well as for accommodating the internal reactor water recirculation pumps, the control rod drives, the core flux monitoring assemblies and the reactor water level, pressure and temperature measuring instrumentation. The four emergency condensers as well as the four standpipes (which connect the passive pressure pulse transmitters and the condensation pots of the reactor pressure vessel level measuring equipment to the reactor vessel) are regarded as being external extensions to the vessel since they are connected to it via non-isolatable lines.

The main auxiliary systems of the SWR 1000 are:

- the residual heat removal (RHR) system;
- the reactor water cleanup system and fuel pool cleanup system;
- the fuel pool cooling system.

Apart from these systems many other auxiliary systems such as waste processing systems, a chilled water system, and HVAC systems exist for normal operation of the plant. The SWR 1000 design concept includes two active systems for low pressure core injection/flooding and residual heat removal. As in earlier plant designs these systems perform the following tasks:

- Cooling of the reactor core during and after normal plant shutdown;
- Water transfer operations before and after refueling;
- Operational heat removal from the core flooding pools and the pressure suppression pool;
- Heat removal from the containment in the event of loss of the main heat sink by cooling the pressure suppression pool and core flooding pool water;
- Low pressure coolant injection into the reactor pressure vessel and simultaneous heat removal in the event of loss-of-coolant accidents.

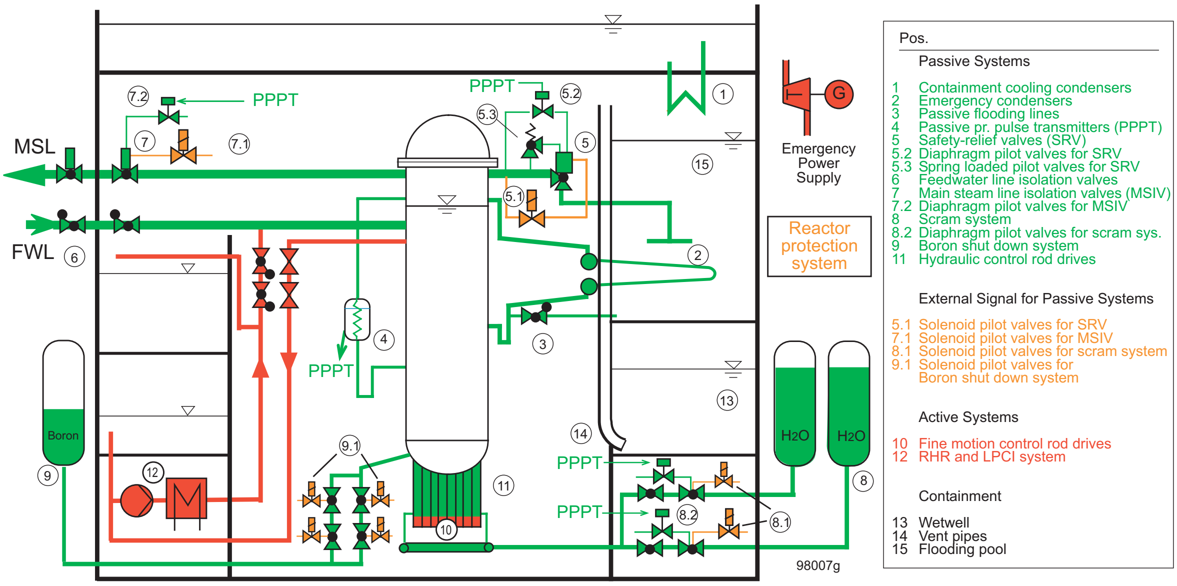


FIG. XI-1. SWR 1000 – Active and passive safety systems.

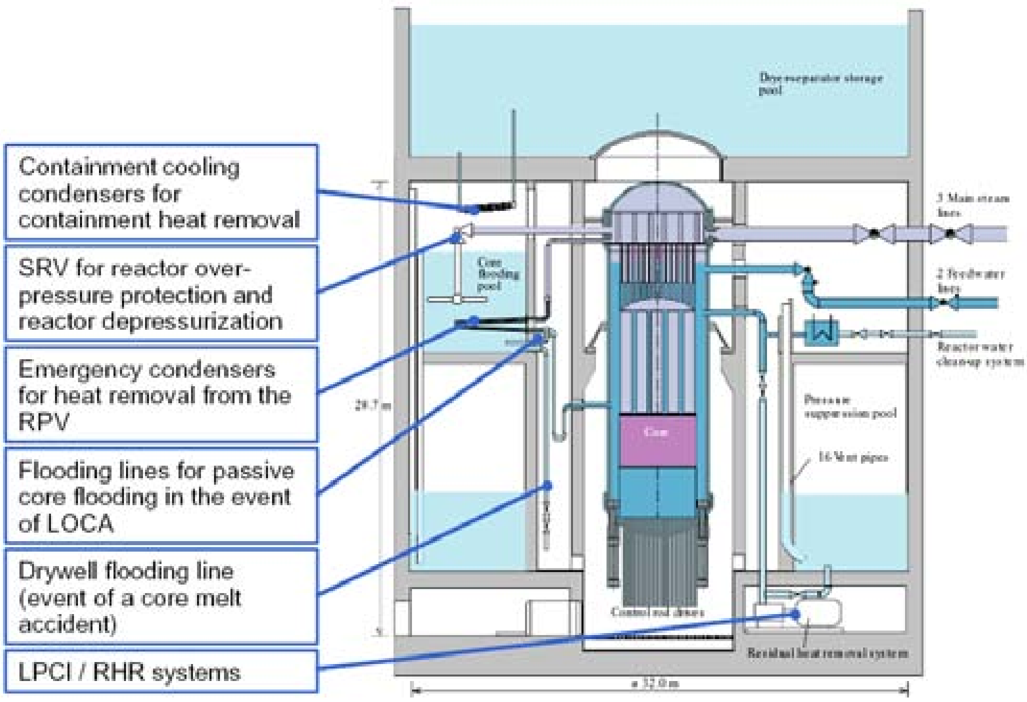


FIG. XI-2. SWR-1000 safety concept.

XI-4. Passive safety systems and features

The fundamentally new concept for accident control incorporated into the SWR 1000 includes equipment which, in the event of failure of the active safety equipment, will bring the plant to a safe condition without the need for any instrumentation and control signals or external power. This passive safety equipment (Figures XI-1 and XI-2) includes the following:

Emergency condensers

The function of the emergency condenser system is to remove, in the event of an accident, the decay heat still being generated in the core as well as any sensible heat stored in the reactor pressure vessel to the core flooding pools, without any coolant inventory being lost from the reactor pressure vessel. The system thus replaces the high pressure coolant injection systems used in existing BWR plants. The emergency condenser system also provides a means for reactor pressure relief that is diverse with respect to the safety-relief valves.

Containment cooling condensers

The function of the four containment cooling condensers is to remove — by entirely passive means — decay heat from the containment following accidents that result in the release of steam into the drywell, and in this way to limit the increase of containment pressure. They provide redundancy and diversity with respect to the residual heat removal system.

Core flooding system

The core flooding system is a passive low pressure flooding system for controlling the effects of loss-of-coolant accidents. It is installed at an elevation which ensures that, following automatic depressurization of the reactor, it can passively flood the reactor core by means of gravity flow. The system provides redundancy and diversity with respect to the core flooding function of the residual heat removal system.

Drywell flooding system

A postulated severe accident involving core melt is controlled such that the molten core is retained inside the reactor pressure vessel. For this purpose the section of the drywell surrounding the reactor pressure vessel is flooded with water in order to cool the exterior of the reactor pressure vessel and thus remove heat from the reactor.

Passive pressure pulse transmitters

The passive pressure pulse transmitter is a completely passive switching device which is used to directly initiate the following safety functions (as a minimum), without the need for instrumentation and control equipment: reactor scram, containment isolation at the main steam line penetrations, and automatic depressurization of the reactor pressure vessel. The passive pressure pulse transmitter comes into action as a result of a drop or increase in reactor water level as well as an increase in reactor pressure. For activating the various safety functions, passive pressure pulse transmitters of redundant design are installed at two elevations. The upper passive pressure pulse transmitters, situated at an elevation beneath that of the normal water level of the reactor pressure vessel, are responsible for initiating reactor scram. The lower passive pressure pulse transmitters, arranged at a lower elevation, activate automatic depressurization of the reactor as well as closure of the main steam containment isolation valves. Further passive pressure pulse transmitters installed at appropriate locations respond to a rise in reactor water level above the main steam nozzles and likewise activate containment isolation at the main steam line penetrations.

XI-4.1. Description of the emergency condenser

The SWR-1000 is equipped with 4 emergency condensers having a maximum power of about 60 MW at a pressure of 7 MPa. The emergency condensers are horizontal tubes arranged between two collectors (see Fig. XI-3). Each bundle consists of 104 tubes arranged in four vertical levels. The tubes have a length of 10 m, an inner diameter of 38.7 mm, a wall thickness of 2.9 mm and are made of austenitic steel (design state 1998). The upper collector is connected via the feed line and the lower connector via the back line connected to the reactor pressure vessel. Both lines can not be blocked. The back line is equipped with an anti circulation loop to avoid the establishment of a stratified counter current flow during the normal operation.

Under normal operation conditions the water level in the core is above the level in the emergency condenser bundles. The water inside the tubes is entirely single phase and no heat transfer to the flooding pool takes place (Fig. XI-3 left side).

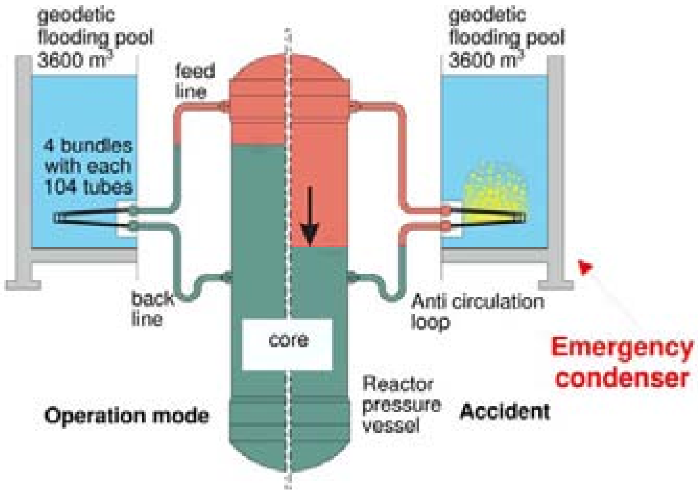


FIG.XI-3. Operation scheme of the emergency condenser.

During an accident the water level in the core sinks below the bundle level of the condenser. The condenser tubes are filled with steam and the emergency condenser acts as a heat sink (Fig. XI-3 right side). In addition, owing to the pools' elevation, the water in the core flooding pools is used for passive flooding of the reactor core following reactor pressure vessel depressurization in the event of a loss of coolant accident. For this function, spring-assisted check valves open the core flooding lines automatically. Passive core flooding serves as a diverse means of providing reactor pressure vessel coolant makeup which supplements the active core cooling systems.

XI-4.2. Description of the containment cooling condenser

The flooding pools are equipped with active coolers. If the active cooling fails, the 3600 m^3 pool will not start to evaporate for twelve hours. During a failure the pressure in the containment is increased and the heat is transferred by four containment cooling condensers into the upper shielding pool (see Fig. XI-1, XI-2 and XI-4). Each condenser is able to transfer up to 5.5 MW at a pressure of 0.25 MPa. Each condenser consists of 124 finned tubes arranged in two layers (design state 1998). The tubes have a length of 4 m, an inner diameter of 32 mm and a tube wall thickness of 3 mm. The tubes are arranged with a twelve degree incline. In addition, a second version of the cooling condenser was designed to remove non condensable gases.

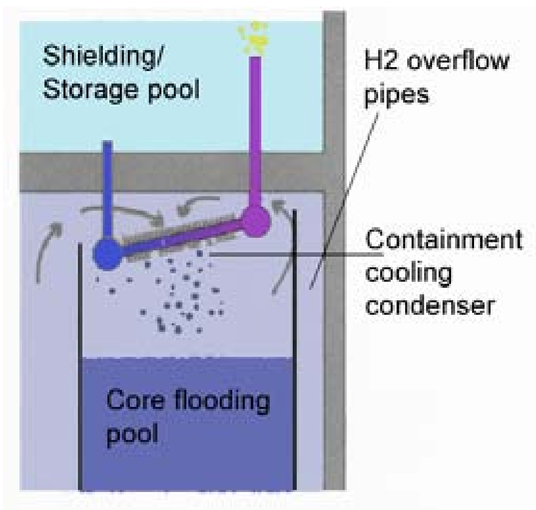


FIG. XI-4. Working principle of the containment cooling condenser.

XI-4.3. Description of the cooling of the core melt within the reactor pressure vessel in the event of a severe accident

In the event of a core melt accident, pools of water located outside of the reactor pressure vessel will flood the lower part of the reactor cavity outside the pressure vessel. The pools are located above the pressure suppression chamber and are approximately two-thirds full of water. They are physically separated from each other by four equipment compartments containing mechanical components, piping and ventilation equipment. Each pool houses an emergency condenser, a containment cooling condenser (above the water surface), a core flooding line connection, and the SRV discharge pipes with steam quenchers (see Fig. XI-3 to XI-5). In addition, a drywell flooding line leads to the bottom of the drywell for cooling the exterior of the reactor pressure vessel.

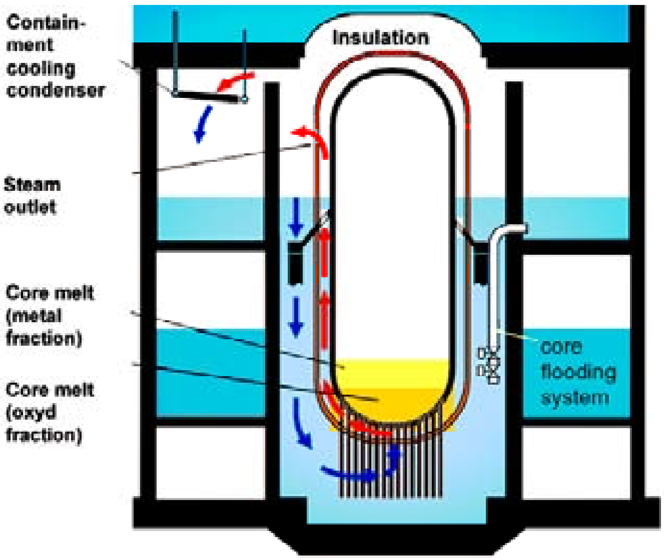


FIG.XI-5. Severe accident melt-in pressure-vessel cooling.

XI-5. Conclusions

In the past, different investigations concerning the emergency cooling were done. Besides the R&D work of FRAMATOME an experimental testing program was conducted at other German and European research centers to provide verification of the mode of operation and effectiveness of the SWR 1000's passive safety systems. During 1996-1998 the European Union supported the BWR R&D Cluster of innovative passive safety systems, where the FZ Jülich, Germany, the GRS Cologne, Germany, the NRG Petten, The Netherlands, NRG Arnheim, The Netherlands, PSI, Switzerland, SIET Piacenca, Italy, STORK NUCON, The Netherlands and VTT Espoo, Finland were involved.

Two large test facilities (NOKO, FZ Jülich for separate effect tests and PANDA, PSI for integral tests) delivered experimental data. In addition operational data from the Doodewaard NPP were used. Post-test calculations were performed using system codes: ATHLET, RELAP and TRAC; lumped parameter codes such as COCOSYS, RALOC and GOTHIC. The CFD-codes applied were CFX and PHOENIX.

During the single effect tests, the heat transfer capability of the emergency condensers and the containment cooling condensers were investigated thoroughly. Special attention was directed to the simulation of condensation in horizontal tubes and to 3D stratification phenomena in the surrounding pools.

ANNEX XII. WWER-640/407
Atomenergoproject/Gidropress, Russian Federation

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
WWER-640/407 *Atomenergoproject/Gidropress, Russian Federation*	PWR	1800	CORE/PRIMARY: • ECCS Accumulator Subsystem • ECCS Tank Subsystem • Primary Circuit Un-tightening Subsystem • Steam Generator Passive Heat Removal System

XII-1. Introduction

The design of WWER-640/407 (V-407) was developed by the FSUE SPAEP (St. Petersburg, Russian Federation), FSUE EDO 'Gidropress' (Podolsk, Russian Federation) and the Russian National Research Centre 'Kurchatov Institute' (Moscow).

The design basis of the V-407 is that the prescribed exposure dose limits and the standards for the release of radioactive substances into the environment should not be exceeded under normal operation, anticipated operational occurrences, and in design and beyond-design-basis accidents, for the life (50-60 years) of the plant.

For this purpose, the new WWER-concept V-407 makes wide use of passive systems and components. A number of relatively innovative passive safety systems are implemented in these designs to ensure the fundamental safety functions: reactivity control, fuel cooling, and the confinement of radioactivity.

One important problem related to the implementation of passive systems is the lack of sufficient operating experience using the passive systems/components under actual plant conditions. Besides, the existing computer codes for transient and accident analysis are not sufficiently validated for the conditions and phenomena which are relevant to the passive system functioning (low pressure, low driving pressure and temperature heads, increased effect of non-condensable, boron transport at low velocities, etc.).

The new design features are envisaged to be verified experimentally at a large-scale test facility (1:27 volume and power scale). The design is developed in accordance with the latest Russian safety regulations for nuclear power plants, which meet modern world requirements.

The passive systems of WWER-V-407 using natural circulation consist of:

- Containment passive heat removal system (C-PHRS)
- Emergency core cooling system with three subsystems
- Steam generator passive heat removal system (SG-PHRS).

The overall configuration of the WWER-640/407 reactor systems is shown in Figure XII-1.

XII-2. Description of containment passive heat removal system

Containment passive heat removal system (C-PHRS) removes heat from the containment in case of a LOCA and is designed to fulfil the following functions: (1) emergency isolation of service lines penetrating the containment and not pertaining to systems intended to cope with the accident; (2)

condensation of the steam from the containment atmosphere; (3) retention of radioactive products released into the containment; (4) control of the iodine levels released into the containment atmosphere. The steam from the containment atmosphere condenses on the internal steel wall of the double-containment being cooled from the outside surface by the water stored in the tank.

The system operates due to natural circulation of the containment atmosphere and water storage tank. The design basis of this system is to condense the amount of steam equivalent to decay heat release within 24 hours after reactor trip without water storage tank replenishment. This system does not require electric power supply to operate. The system consists of four independent trains with the redundancy of 4 × 33%. A diagram of the passive heat removal system is shown in Figure XII-2.

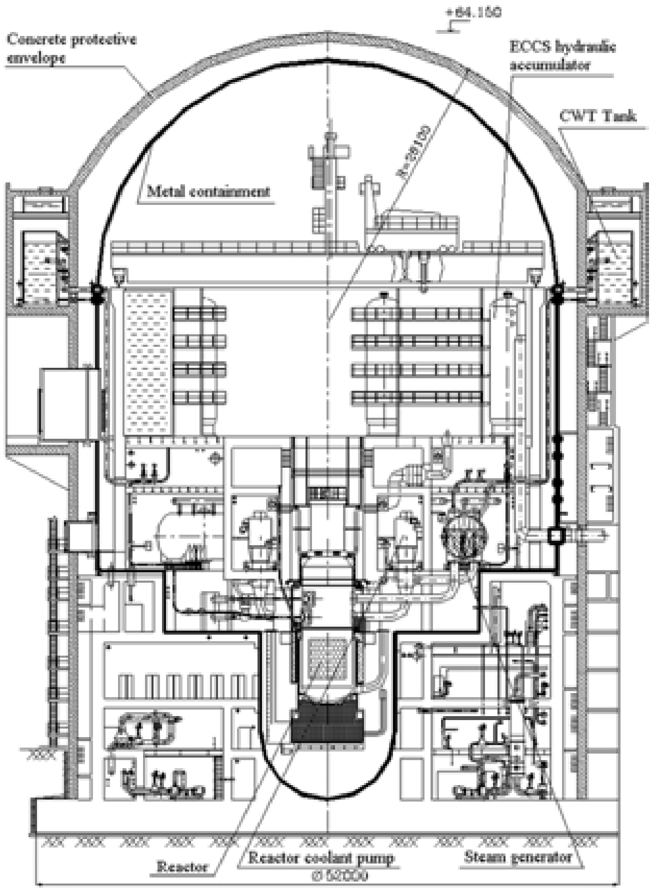


FIG. XII-1. Reactor building.

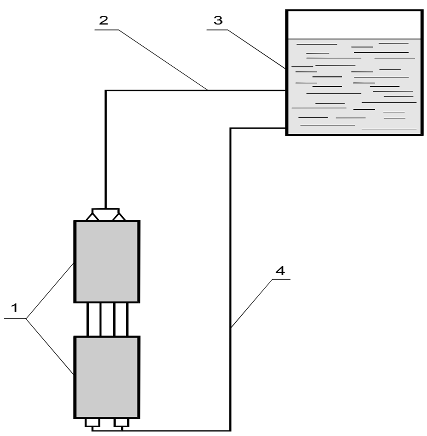


FIG. XII-2. Passive heat removal system for LOCAs (1 - cooler of primary containment; 2 - discharge pipeline; 3 - emergency heat removal tank; 4 - make-up pipeline).

XII-3. Description of the emergency core cooling system

The emergency core cooling system of V-407 reactor comprises three automatically initiated subsystems: (1) hydroaccumulators with nitrogen under pressure (which are traditional ECCS accumulators used for operating WWER-1000 reactors), (2) elevated hydroaccumulators open to the containment, and (3) equipment for forced emergency depressurization of the primary circuit. All these subsystems are based on the principle of passive operation providing for long term residual heat removal in case of a loss-of-coolant accident concurrent with plant blackout (i.e. AC power supply is not required for ECCS operation). Schematic diagram of emergency core cooling system is also shown in Figure XII-3.

XII-3.1. ECCS accumulator subsystem

ECCS accumulator system is intended to supply cold water at the core inlet and retain the primary coolant inventory when the primary pressure reduces below 4 MPa(when leaks occur). The system consists of four independent trains with redundancy of 4 × 50%. Each train includes an accumulator tank with boron solution under pressure; the pressure is created by a nitrogen cushion. The accumulators are connected to the reactor downcomer. The check valves are installed on the connecting pipelines to isolate the accumulators from the reactor when the primary pressure exceeds 4.0 MPa. The system operates in accordance with the passive principle, and no signal is required to activate the system.

XII-3.2. ECCS tank subsystem

ECCS tanks (tanks of atmospheric type) are intended to flood the reactor core allowing long term residual heat removal when operating at a pressure below 0.29 MPa. The system consists of four independent trains with redundancy of 4 × 50%. Each train includes the tank filled with boron solution, valves, and pipelines. The ECCS tank is connected to the reactor downcomer by the pipeline with two check valves. The system operates under the hydrostatic force effect.

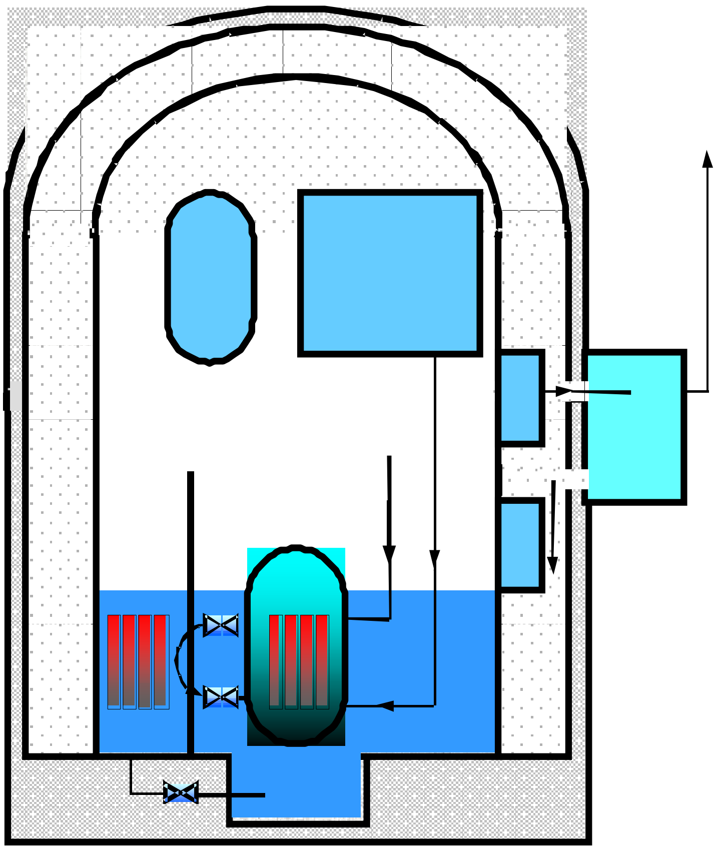


FIG. XII-3. Emergency core cooling system for LOCAs.

XII-3.3. Primary circuit untightening subsystem

The primary circuit untightening subsystem is designed to ensure reliable water flow from atmospheric ECCS tanks and natural circulation of coolant along the circuit ‘reactor-emergency pool’. This system operates during LOCAs when the primary circuit pressure decreases to 0.6 MPa. The system consists of two independent trains with redundancy of 2 × 100%. Each train comprises two pipelines and valves connecting the primary hot and cold legs to the emergency pool located in the lower part of the containment. This provides long term residual heat removal after the ECCS accumulator tanks and atmospheric tanks are emptied. The system train includes untightening valve units, pipeline and valves for connecting the refueling pool with the emergency pool, repair valves, and a screen filter.

XII-4. Description of steam generator passive heat removal system (SG-PHRS)

The steam generator passive heat removal system (SG-PHRS) for the V-407 reactor does not require a power supply and is designed to remove the decay heat in case of a non-LOCA. It also supports the emergency core cooling function in case of a LOCA. The reactor coolant system and passive heat removal equipment arrangement provides heat removal from the core following reactor shutdown via steam generator heat transfer to the tanks of chemically demineralized water (CWT) outside the containment and further to the atmosphere by natural circulation as shown in Figure XII-4. Reactor power that can be removed from the core by coolant natural circulation is about 10% of the nominal value, which ensures a reliable residual heat removal. In case of a non-LOCA the decay heat is removed by coolant natural circulation to steam generator boiler water. The steam generated comes into the passive heat removal system where steam is condensed on the internal surface of the tubes that are cooled on the outside surface by the water stored in the demineralized water tank. These tanks are stored outside the containment. The water inventory in this tank is sufficient for the long term heat removal (at least 24 hours) and can be replenished if necessary from an external source.

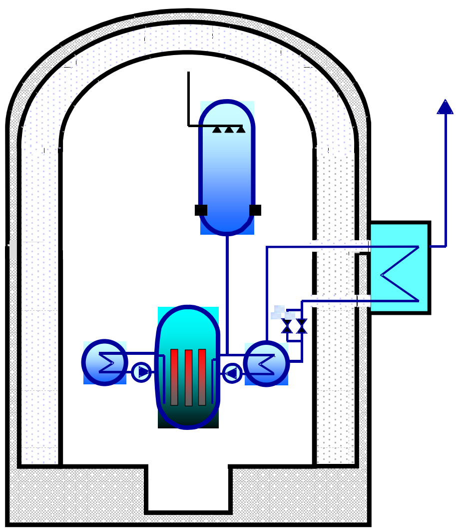


FIG. XII-4. Passive heat removal for non-LOCAs.

XII-5. Integrated passive safety system response during a LBLOCA accompanied by blackout

The initial state of the NPP is the operation at rated power. As a result of break of main coolant pipeline, the discharge of significant mass of coolant of the primary circuit takes place. When the pressure of the primary circuit becomes less than 13.7 MPa, the scram is initiated. Stop valves are closed after trip of the reactor. In this condition, the main coolant pump is switched off and coast down when the difference between the temperature of coolant in a hot leg and saturation temperature under actual pressure becomes bellow 10 °C. The steam generator PHRS starts up by the fact of decrease of the difference between primary and secondary side pressure. The boric acid solution is supplied to the reactor from traditional hydroaccumulators with initial pressure of 4.0 MPa by corresponding primary pressure decrease, without any signal actuation. The containment PHRS will provide condensation of the steam in the containment.

Thus, in the first stage of the accident, primary pressure is decreased due to loss of coolant and operation of the passive heat removal system. Further cooling down and pressure decrease are realized via steam generator PHRS and containment PHRS. When the pressure difference between primary circuit and containment is decreased to 0.6MPa, passive valves of the emergency depressurization system (primary circuit untightening subsystem) open to connect reactor inlet and outlet with the fuel pond space. When the reactor and containment pressure difference is decreased below 0.3 MPa, the ECCS tanks (elevated hydroaccumulators open to the containment) begin to flood the reactor. This sequence results in creating of so called emergency pool where the reactor coolant system is submerged to and in connection of this emergency pool with the spent fuel pond. The natural circulation along the flow path shown in Figure XII-3 (reactor inlet plenum - core - reactor outlet plenum - 'hot' depressurization pipe - fuel pond - 'cold' depressurization pipe - reactor inlet plenum) provides the long term heat removal from the core in case of the LOCA combined with complete loss of power supply. The water in the emergency pool and spent fuel pool reaches the saturation point in about 10 hours. The steam generated will condense on the internal surface of the steel inner containment wall, and condensate flows back into the emergency pool. This configuration ensures also the heat removal from reactor vessel bottom to keep the corium inside the reactor in case of postulated core melt event.

XII-6. Conclusions

Several passive safety systems based on natural circulation phenomena are used in WWER-640/407 reactors to fulfill the fundamental safety functions of reactivity control, fuel cooling, and radioactivity confinement. Implementation of these systems made it possible to significantly increase power plant safety in terms of the expected severe core damage and excessive radioactivity release frequencies.

REFERENCES TO ANNEX XII

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Natural circulation data and methods for advanced water cooled nuclear power plant designs, IAEA-TECDOC-1281 (IAEA, Vienna, 2000).
[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of advanced light water reactor designs 2004, IAEA-TECDOC-1391 (IAEA, Vienna, 2004).

ANNEX XIII. WWER-1000/392
Atomenergoproject/Gidropress, Russian Federation

Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
WWER-1000/392 *Atomenergoproject/Gidropress, Russian Federation*	PWR	3000	CORE/PRIMARY: • Passive quick boron supply system • Passive subsystems for reactor flooding (first and second stage hydro-accumulators) • Passive residual heat removal system via steam generator CONTAINMENT: • Maintain low inter-containment gap (annulus) atmosphere pressure • Passive core catcher

XIII-1. Introduction

The design of WWER-1000/392 (V-392) was developed by FSUE 'Atomenergoproject' (Moscow, Russian Federation), FSUE EDO 'Gidropress' (Podolsk, Russian Federation) and the Russian National Research Centre 'Kurchatov Institute' (Moscow).

The primary purpose of the V-392 is to ensure the safety of the personnel, the public, and the environment against radiation effects exceeding the specified (prescribed) radiation doses. This principle also addresses the standards for releases of radioactive substances and their content in the environment under normal operation conditions, anticipated operational occurrences, design, and beyond-design-basis accidents during the plants life. The objective of the reactor plant design and nuclear plant process systems is to achieve estimated probability of severe core damage not above 1.0E-5 per reactor-year and the probability of accidental radioactive releases not above 1.0E-7 per reactor-year. These values are specified in Russian safety standards.

The design of NPP with WWER-1000/392 improves technical and economic parameters. Wide application of passive safety means, using natural physical processes, along with the traditional active systems is a specific feature of this design. Each plant designer must solve problems caused by implementing passive safety systems. The passive systems have their own advantages and drawbacks in comparison with the active systems both in the area of plant safety and economics. Therefore, a reasonable balance of active systems and new passive means is adopted in V-392 design to improve safety and public acceptability of nuclear energy.

The passive systems of WWER-1000/392 [1,2] are:

- Passive quick boron supply system,
- Passive subsystem for reactor flooding HA-1 (hydroaccumulators of first stage),
- Passive subsystem for reactor flooding HA-2 (hydroaccumulators of second stage),
- Passive system to maintain low inter-containment gap (annulus) atmosphere pressure,
- Passive residual heat removal system via steam generator (PHRS),
- Passive core catcher.

The reactor main coolant pump flywheel inertia provides the initial boron injection to the primary system. Nitrogen gas at high pressure injects borated water from the hyrdoaccumulators in two stages to the core. Gravity provides the driving force in some of the passive safety systems. The last three systems operate in a natural circulation loop providing decay heat removal from the core.

The overall configuration of the safety systems based on passive principles mentoned above is shown in Figure XIII-1.

The overall functions of the safety systems based on passive principles mentioned above in comparison with active safety systems are shown in Figure XIII-2.

XIII-2. Description of quick boron supply system

The quick boron supply system (QBSS), being developed as an additional reactor trip system, comprises a system of 4 special loops bypassing the main coolant pumps. Each loop consists of a hydroaccumulator containing concentrated boron acid solution and pipelines with quick-acting valves that do not require electric power for their opening. These valves are opened when the reactor fails to scram. When this occurs concentrated boron solution is pressed out of the hydro-accumulators into the primary loops and further into the reactor. In case of a station blackout the boron solution delivery occurs in the period of reactor coolant pump (RCP) coast-down.

The RCPs have a large flywheel inertia which provides the possibility of ejecting all boron concentrate from the QBSS hydro-accumulators. The amount and concentration of the boron solution are chosen to provide a certain equivalency from the viewpoint of reactivity inserted by this system and by the solid absorber scram. In fact, this system, being a part of the primary coolant circulation system increases the ‘inherent’ safety of the reactor design.

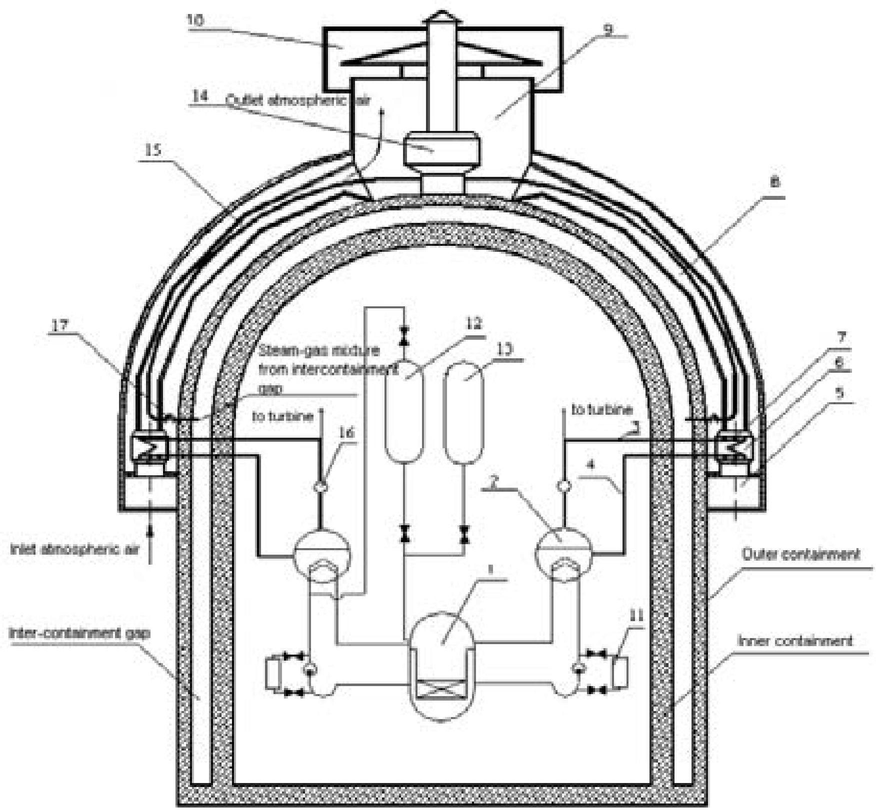


FIG. XIII-1. Containment and the systems of WWER-1000/392 that used passive principles. (1 – reactor; 2 – steam generator; 3 – steam path; 4 – condensate path; 5 – inlet circular header; 6 – PHRS heat exchanger; 7 – PHRS slide valve; 8 – PHRS draught tube; 9 – PHRS exit header ;10 – deflector; 11 – quick boron supply system; 12 – HA-2; 13 – HA-1; 14 – filtering unit; 15 – tube of passive filtering system; 16 – steam header; 17 – valve).

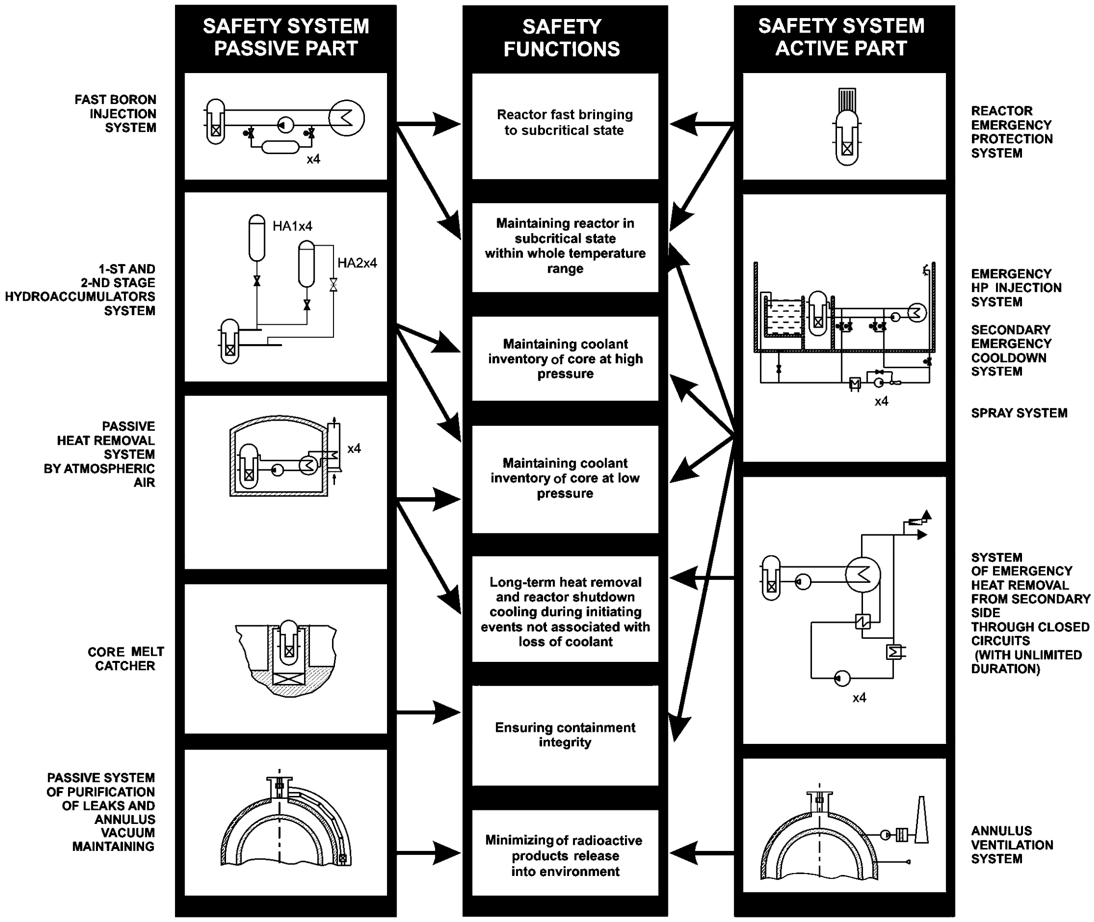


FIG. XIII-2. Safety function.

XIII-3. Description of HA-1

The emergency core cooling subsystem has been designed to provide long term residual heat removal in case of a primary leak accident concurrent with a station blackout. The hydroaccumulators with nitrogen under pressure will provide the coolant injection during the first stage of such an accident. The active subsystem is connected before the hydroaccumulators are emptied.

XIII-4. Description of HA-2

The emergency core cooling subsystem has been designed to possibly provide a long term residual heat removal in case of primary LOCA concurrent with a station blackout.

The passive core flooding subsystem includes four groups of hydroaccumulators under atmospheric pressure (2nd stage HA) which are coupled with the pipelines connecting the ECCS hydroaccumulators and the reactor. The hydroaccumulators of the passive core flooding system are connected to the primary system at 1.5 MPa. The hydrostatic pressure of the water column will flood the core removing the residual heat at the last stage of a LOCA for at least 24 hours.

XIII-5. Description of passive system to maintain low inter-containment gap (annulus) atmosphere pressure

To substantially limit the release of fission products beyond the containment; a permanent under pressure is maintained in the inter-containment gap of the V-392 design. This safety function, one of the most important, is fulfilled by two systems: (1) an exhaust ventilation system equipped with a filtering plant with suction from the inter-containment gap and outlet into the stack; (2) a passive system of suction from the inter-containment gap. The first system is intended to control removal of a steam-gas mixture from the inter-containment gap under accidents with the total loss of power. The system is capable to remove at least 240 kg per hour that is equivalent to the inner containment leaks of 1.5% containment volume per 24 hours. The second system consists of the lines connecting the inter-containment gap with PHRS exhaust ducts, which are always in the hot state. This enables permanent removal and purification of inner containment leaks regardless of power supply and operator actions. According to estimations, the under pressure is maintained at any point of the inter-containment gap with inner containment leaks up to 2.8% of containment volume per day (the design basis for the containment is 0.3%). The technical solution described above in combination with the systems for the containment pressure decrease (traditional spray system and new passive heat removal system) allows us to give up the filtered venting system designed for V-392. Even though this system satisfies the current requirements to filtered venting; this should not increase the risk of containment failure. Filtered venting is not required within the short term of a core melt accident.

XIII-6. Description of passive residual heat removal system via steam generator

A passive residual heat removal system (see Figure XIII-3) is included in the design to remove heat from the reactor plant. The PHRS is designed in case of a station blackout including the loss of emergency power supply. The removal of residual heat should be provided without damage to the reactor core and the primary system boundary for an unlimited time. The PHRS consists of four independent trains, each of them being connected to the respective loop of the reactor plant via the secondary side of the steam generator. Each train has pipelines for steam supply and removal of condensate, valves, and an air-cooled heat exchanger outside the containment. The steam that is generated in the steam generators due to the heat released in the reactor; condenses and rejects its heat to the ambient air. The condensate is returned back to the steam generator. This occurs by natural circulation.

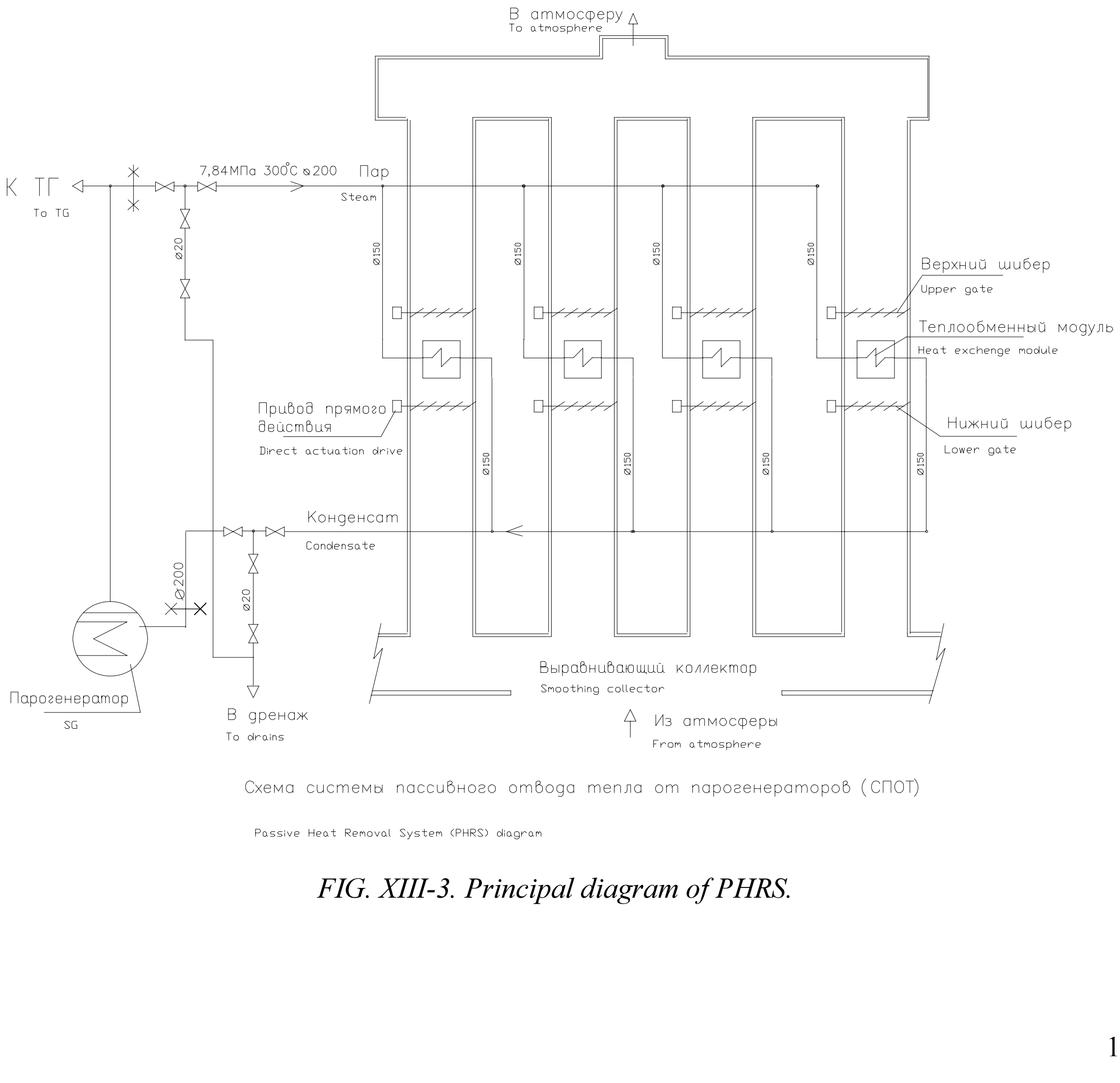


FIG. XIII-3. Principal diagram of PHRS.

XIII-7. Description of passive core catcher

The passive core catcher increases the safety barrier and confinement by preventing radioactive material releases from the primary circuit and the reactor vessel. The passive core catcher provides receiving and subsequent cooling of liquid and hard corium fractions released from the damaged reactor vessel. It is installed in a concrete pit below the reactor vessel. This device comprises four variable parts located (top-to-bottom) in the direction of corium movement from the reactor vessel to the concrete pit basement. The variable parts are the lower plate, vent header, barrel with the filler, and heat exchanger.

The lower plate is designed as a guiding structure, similar to a funnel, in which the corium from damaged reactor vessel flows into the passive core catcher.

The vent header is installed under the lower plate and is designed as a thermal shield protecting the thermal insulating structures. It also protects the reactor vessel from corium outflow installed on the concrete cantilever and in the foundation of lower plate at the stage of corium outflow from the reactor vessel.

This allows to increase operating period of the specially installed thermal shields and to lower the intensity of their damage in the course of radiation heat exchange with corium and aerosol atmosphere in the concrete pit sub-reactor room.

The vent header increases the operating time and reduces the radiation intensity to the thermal shields.

The barrel with filler functions as a corium diluent and thermal absorber for the surrounding peripheral structures (the core catcher). The optimum content of ferric, aluminium oxides, and structural steel in sacrifice material allows to lower volumetric power density in the corium melt, release of gas and radionuclide masses into the confinement, melt temperature.

The evaporation of water causes a period of corium direct cooling through a heat exchanger. The heat exchanger is gravity fed from the inspection well and begins after the corium chemical reaction with the filler and melting of barrel steel structures. Heat exchanger passive make-up and corium direct cooling by the water supplied through concrete cantilever pipelines allows us to avoid human intervention for 24 hours (the moment of reactor vessel damage with corium).

XIII-8. Integrated passive safety system response during a station blackout

The initial state of NPP is the operation at rated power. As a result of an initial event, loss of all sources of alternating current electrical power, all MCPs (main coolant pumps) are tripped, stop valves of a turbine generator are closed, the primary circuit makeup-blowdown system is disconnected, the power supply to pressurizer is disconnected, BRU-Ks (steam dump valve to turbine) are disconnected, and the main and auxiliary feedwater systems of secondary circuit are stopped. Besides as a result of diesel generators failure to start all active safety systems do not function.

After scram (because of three or more MCPs switching-off) the reactor power reduces down to a residual heat level. After the ending of MCPs coastdown the natural circulation of the primary coolant is established. Firstly the heat removal from the primary circuit is carried out due to BRU-A (steam dump valve to the atmosphere) operation, and then via steam generator pulse safety device. As a result of SPOT operation the part of heat from primary circuit is removed in an environment, and other heat is removed through BRU-A (loss of the boiler water from steam generator proceeds). After appropriate decreasing of residual heat, dumping devices of the second circuit are closed, and loss of boiler water from steam generator stops. The heat removal from the primary circuit is carried out due to steam generator PHRS operation under the closed circuit, steam is condensed in heat exchanger modules, and the condensate returns back in steam generator. The reliable cooling of the core is provided. Thus, results of calculation show that steam generator PHRS operation prevents any core damage in considered BDBA.

XIII-9. Conclusions

A number of relatively innovative passive safety means are used in the new Russian plant designs with V-392 reactors to fulfil the fundamental safety functions; such as reactivity control, fuel cooling, and radioactivity confinement. For example, the estimated core melt frequency for V-392 is three orders of magnitude less than the V-320 reactor. The engineering solutions using natural circulation in V-392 design increases the safety level compared to operating WWER-1000 plants.

REFERENCES TO ANNEX XIII

[1] INTERNATIONAL ATOMIC ENERGY AGENCY, Natural circulation data and methods for advanced water cooled nuclear power plant designs, IAEA-TECDOC-1281, Vienna (2000).

[2] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of advanced light water reactor designs 2004, IAEA-TECDOC-1391, Vienna (2004).

Annexes

PART II: PASSIVE SAFETY DESIGN FEATURES FOR INTEGRAL REACTOR SYSTEMS

ANNEX XIV. CAREM
National Atomic Energy Commission (CNEA), INVAP, Argentina

Integral Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
CAREM *CNEA (National Atomic Energy Commission), INVAP, Argentina*	PWR	100	CORE • Residual Heat Removal System – Emergency Condenser

XIV-1. Introduction

CAREM is an Argentine project to achieve the development, design, and construction of an innovative simple and small Nuclear Power Plant (NPP), which is jointly developed by CNEA (National Atomic Energy Commission) and INVAP. This nuclear plant has an indirect cycle reactor with some distinctive and characteristic features that greatly simplify the design, and contributes to a higher safety level. Some of the high level design characteristics of the plant are: integrated primary cooling system, self-pressurised primary system and safety systems relying on passive features.

The CAREM concept was first presented in March 1984 in Lima, Peru, during the IAEA conference on small and medium size reactors. CAREM was, chronologically, one of the first of the present new generation of reactor designs. The first step of this project is the construction of the prototype of about 27 MW(e) (CAREM-25). CAREM has been recognized as an International Near Term Deployment (INTD) reactor by the Generation IV International Forum (GIF).

XIV-2. Description of passive core cooling system

The CAREM nuclear power plant design is based on a light water integrated reactor. The whole high-energy primary system, core, steam generators, primary coolant and steam dome, is contained inside a single pressure vessel (Fig. XIV-1).

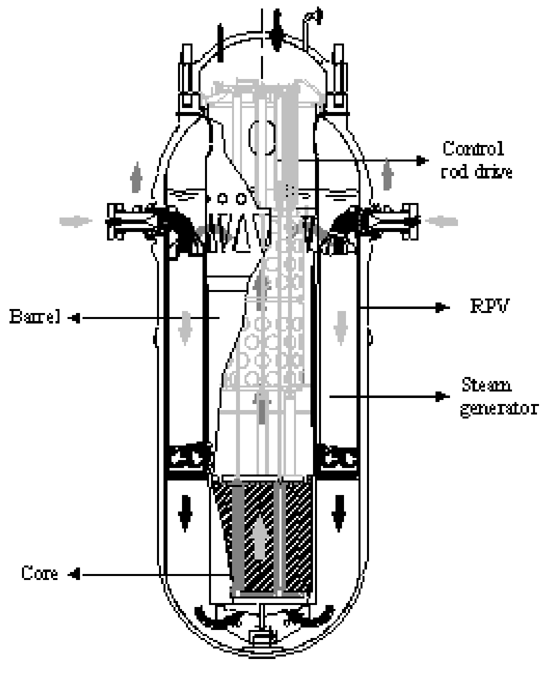
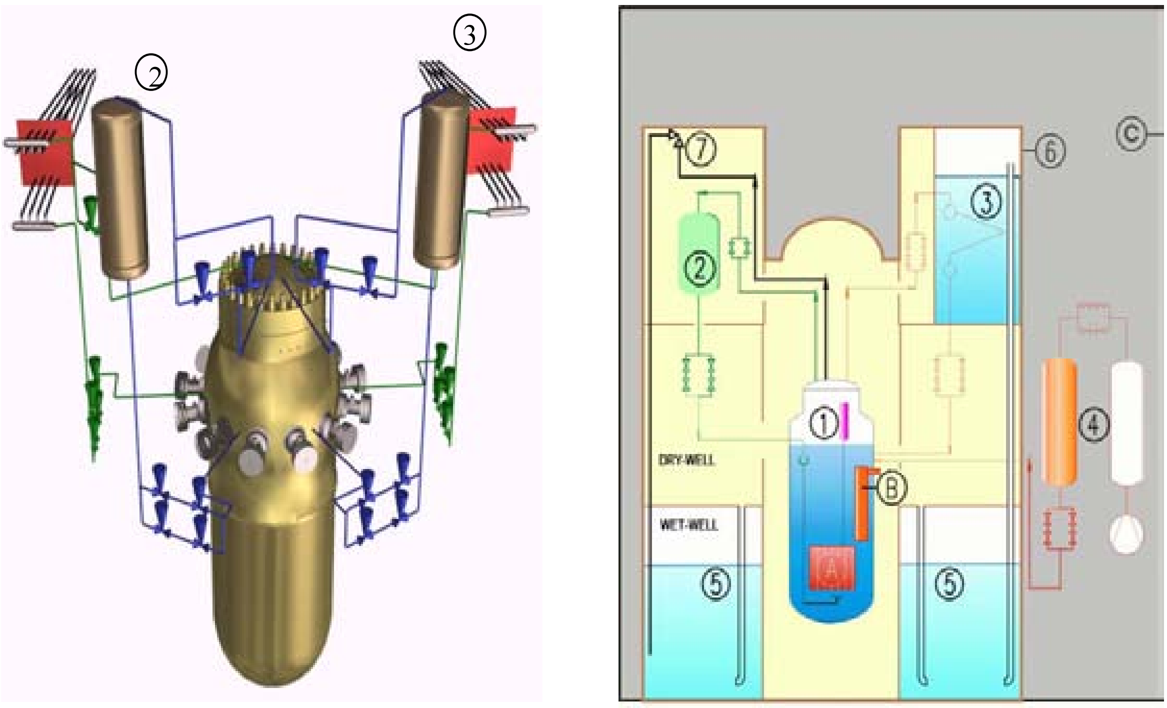


FIG. XIV-1. Reactor pressure vessel.

For low power modules (below 150 MW(e)), the flow rate in the reactor primary systems is achieved by natural circulation. Figure XIV-1 shows a diagram of the natural circulation of the coolant in the primary system. Water enters the core from the lower plenum. After it's heated the coolant exits the core and flows up through the riser to the upper dome. In the upper part, water leaves the riser through lateral windows to the external region. Then it flows down through modular steam generators, decreasing its enthalpy. Finally, the coolant exits the steam generators and flows down through the down-comer to the lower plenum, closing the circuit. The driving forces obtained by the differences in the density along the circuit are balanced by the friction and form losses, producing the adequate flow rate in the core in order to have the sufficient thermal margin to critical phenomena. Reactor coolant natural circulation is produced by the location of the steam generators above the core.

XIV-3. Description of the residual heat removal system

CAREM safety systems are based on passive features that don't require active actions to mitigate accidents for a long period. They are duplicated to fulfil the redundancy criteria. One of them that relies on natural circulation is the residual heat removal system (RHRS*)*, Figure XIV-2. It has been designed to mitigate a Loss of Heat Sink, reducing the pressure of the primary system to values lower than the values at hot shutdown, by removing the decay heat. The RHRS is a simple and reliable system that operates condensing steam from the reactor dome in the emergency condensers. This establishes a stratified two-phase natural circulation loop with the primary system.



1: First Shutdown System
2: Second Shutdown System
3: Residual Heat Removal System
4: Emergency Injection System
5: Pressure suppression pool7: Safety valves
6: Containment
A: Core B: Steam Generators C: Reactor Building

FIG. XIV-2. Safety system.

The emergency condensers are heat exchangers consisting of an arrangement of parallel horizontal U tubes between two common headers. The top header is connected to the reactor vessel steam dome, while the lower header is connected to the reactor vessel at a position below the reactor water level. The condensers are located in a pool filled with cold water inside the containment building. The inlet valves in the steam line are always open, while the outlet valves are normally closed, therefore the tube bundles are filled with condensate. When the system is triggered, the outlet valves open automatically. The water drains from the tubes and steam from the primary system enters the tube bundles and condenses on the cold surface of the tubes. The condensate is returned to the reactor vessel forming a natural circulation circuit. In this way, heat is removed from the reactor coolant. During the condensation process the heat is transferred to the pool water by boiling process. This evaporated water is then condensed in the suppression pool of the containment. The pool of the RHRS has a volume sufficient to provide an autonomy greater than the grace period for the prototype (48 h).

The RHRS main characteristics are listed in Table XIV-1.

TABLE XIV-1. RESIDUAL HEAT REMOVAL SYSTEM-EMERGENCY CONDENSER FOR CAREM PROTOTYPE

Operation Mode	Steam Condensation
Maximum power of one module (at reactor nominal operational conditions)	2 MW
Tube length	13.3 m
Tube external diameter	60.3 mm
Tube inner diameter	42.8 mm
Redundancy	Condenser 2 × 100%
Valves:	4 × 100%
Autonomy	> 48 hours

In case of a very small LOCA (lower than 3/4') the RHRS is also demanded by the reactor protection system to depressurize the primary system to allow the emergency injection system to act.

XIV-4. Conclusions

In order to assess the CAREM primary circuit numerical modelling a High Pressure Natural Circulation Rig (CAPCN) was built in the decade of the 90s. The CAPCN resembles CAREM in the primary loop and steam generators, while the secondary loop is designed just to produce adequate boundary conditions for the heat exchanger. The CAPCN rig reproduces most of the dynamics phenomena of the RCCS, except for 3-D effects. Several tests were performed covering thermal hydraulics, reactor control and operating techniques around the nominal operational point.

Indicative values of the main variables corresponding to the CAPCN operation at full power are shown in Table XIV-2.

TABLE XIV-2. CAPCN NOMINAL CONDITIONS

Variable	
Thermal power	238 kW
Primary circuit mass flow rate	1.49 kg/s
Secondary circuit mass flow rate	0.105 kg/s
Steam dome pressure	110 bar
Cold leg temperature	288 °C
Steam generators feed water temperature	209 °C

Most of the tests performed consisted of an initial self-steady state in which a pulse-wise perturbation induced a transient. In this case the perturbation is a thermal unbalance as severe as possible for operational transients. The following test groups were performed:

- Preliminary tests to characterize components and equipment;
- Thermal balance test, instrumentation calibration and evaluation of their accuracy;
- Dynamic test around the operational nominal point without control: thermo-hydraulic response evaluation;
- Dynamic test with the control loops (power pressure);
- Dynamic test at low pressure and temperature.

ANNEX XV. IMR
Mitsubishi, Japan

Integral Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Integrated Modular Water Reactor (IMR) *Mitsubishi, Japan*	PWR	1000	• Stand-alone Direct Heat Removal System • Stand-alone Direct Heat Removal System – Late Phase

XV-1. Introduction

The Integrated Modular Water Reactor (IMR) is one of the integrated primary system reactors (IPSRs) with the reference output of 1,000 MWt (350 MW(e)). The design targets of IMR are to achieve the electricity generating cost comparable to that of a large-scale nuclear reactor and higher-level safety by removing the sources that cause fuel failures by design. To achieve these targets, IMR employs integrated design with in-vessel CRDM (control rod drive mechanisms), hybrid heat transport system (HHTS) which employs two-phase natural circulation for the primary heat transportation, and stand-alone direct heat removal system (SDHS) for accident heat removal from the primary system.

IMR started its conceptual design study in 1999 at Mitsubishi Heavy Industries (MHI) reflecting changes of business environment such as less growth of economy and electricity demand, and deregulation of electricity market in Japan. An industry-university group led by MHI (Kyoto University, Central Research Institute of Electric Power Industries (CRIEPI), the Japan Atomic Power Company (JAPC), and MHI) has been developing related key technologies funded by Japan Ministry of Economy, Trade and Industry from 2001 to 2004. In this project, the concepts and the feasibility of HHTS and SDHS have been built and tested through three series of experiments. They are (1) air-water scale test to confirm void distribution and void behavior in the reactor, (2) high temperature natural circulation test to study two-phase natural circulation in the reactor with the actual temperature, pressure, and axial dimension of IMR, (3) SDHS test to study passive heat transport with the actual temperature, pressure, and axial dimension of SDHS. These test facilities were built and set at MHI Takasago R&D centre. In-vessel CRDM technology is based on marine reactor (MRX) development by Japan Atomic Energy Agency (JAEA) and MHI.

Here, the concepts and the feasibility test results of HHTS and SDHS are summarized.

XV-2. Description of hybrid heat transport system (HHTS)

Figure XV-1 shows a cross sectional view of IMR reactor. As shown in the figure, IMR employs several key concepts for the reactor design. The first one is the integrated primary system concept. Fuel assemblies, control rods, steam generators (SGs), and control rod drive mechanisms (CRDMs) are all installed inside the reactor vessel (RV), and there is no main coolant piping and no large penetration of RV.

The second one is the hybrid heat transport system (HHTS), which is a kind of two-phase natural circulation system operating under the self-pressurized saturation condition of the primary coolant. The coolant starts boiling at the upper part of the core, two-phase coolant flows up in the riser, and is condensed and sub-cooled by SGs. In order to control the amount of boiling and the system pressure, IMR has two kinds of SGs, i.e. SG in liquid region (SGL) and SG in vapor plenum (SGV). Table XV-1 shows the major specifications and operating condition of the IMR primary system.

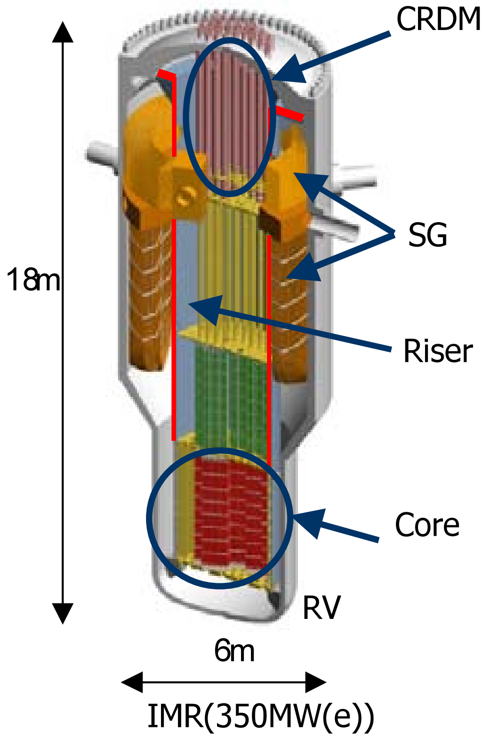


FIG. XV-1. Configuration of the IMR primary systems.

TABLE XV-1. MAJOR SPECIFICATIONS OF IMR

Item	**Specification**
Thermal/electric output	1000 MWt/350 MW(e)
Reactor vessel ID/H	6m (max.)/18m
Primary coolant	Light water
Primary pressure	15.5 MPa
Max. coolant temperature	345deg-C (core outlet)
Primary coolant flow rate	3000 kg/s
Core outlet void fraction	20%
Type of fuel	UO_2, UO_2+Gd_2O_3
Fuel enrichment	<5wt%
Fuel assembly type/number	Square 21 × 21/97
Core height	3.7m
Power density	40kW/L
Cycle length	>24EFPM
Control rod absorber/number	Enriched B_4C/92
Type of in-vessel CRDM	Electric motor driven

This concept allows eliminating a pressurizer and reactor coolant pumps, which simplify the reactor design and is beneficial to reduce maintenance work. This concept also increases the driving force of the coolant flow. The average void fraction in the riser (20%) is optimized to minimize the required RV height while keeping the appropriate thermal margin of the core. A small amount of boiling is allowed in the core but the core characteristics are still very similar to that of conventional PWRs, because the void fraction is relatively low and most of the core operates under sub-cooled condition. From the safety point of view, this concept realizes a reactor design free from large-scale break of the primary boundary (i.e. LOCA), control rod ejection accident, and loss of flow accident. As the results, no safety injection system and containment spray system is required for IMR. Additionally, such downsizing and safety feature allows applying very small dry containment, which greatly reduces construction cost.

The feasibility of HHTS is one of the most important subjects for IMR because of less knowledge of two-phase flow behavior under such high temperature (345deg-C) and pressure (15.5MPa). In order to verify the feasibility, two series of experiments have been performed. The first one is an air-water scale experiment and the second one is a natural circulation experiment under the actual temperature, pressure, and axial dimension.

XV-3. Description of stand-alone direct heat removal system (SDHS)

IMR realizes ‘no cause of fuel failure’ concept by reactor design and the core is always submerged in primary coolant without safety injection. Therefore, only a heat removal function is required as the safety system. Figure XV-2 shows the configuration of the stand-alone direct heat removal system (SDHS). SDHS is a closed natural circulation system that removes residual heat directly from inside the RV to the atmosphere via SGs and passive SG coolers (PSGCs). The reactor is cooled down and depressurized without opening the primary system pressure boundary. Residual heat in the early stage of an accident is removed by latent heat of cooling water by submerging PSGCs. After the cooling water was dried out, residual heat will be removed by air-cooling, since the dried water pool and ducts form a wind tunnel. Therefore, the heat transfer mode in PSGCs automatically changes from water-cooling to air-cooling following the pool water evaporation and SDHS works a long time without any operator action and external support.

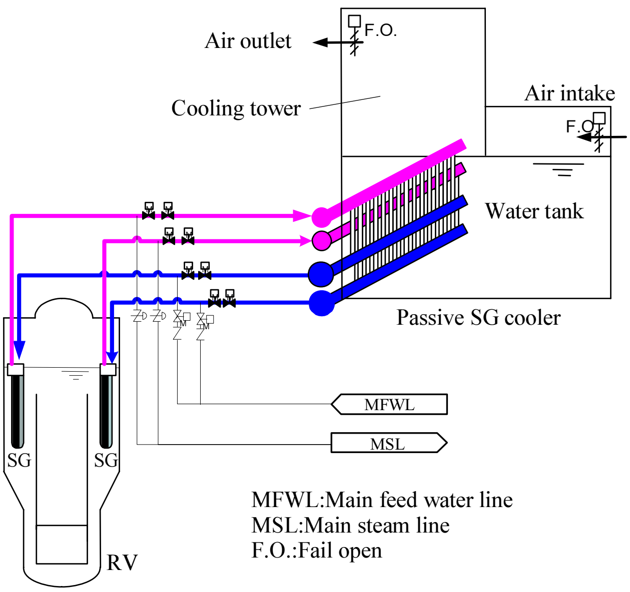


FIG. XV-2. Stand-alone direct heat removal system.

The reactor design and SDHS concept greatly simplify the safety system of IMR compared with conventional PWRs. There is no safety injection system and containment spray system, and SDHS makes support systems to change into non-safety systems, which are component cooling water system, essential service water system, and emergency AC power system. To show the feasibility of SDHS, the safety system of IMR, natural circulation and heat removal capability of the system including effect of non-condensable gas has been experimentally examined. SDHS is designed to accumulate non-condensable gas into the lower header of PSGC and the feed water tank of the SGs.

XV-4. Conclusions

The Integrated Modular Water Reactor (IMR) employs two natural circulation systems, the hybrid heat transport system (HHTS) for the primary system and the stand-alone direct heat removal system (SDHS) for the safety heat removal. The design concepts of IMR and these systems have been built and the test results showed the feasibility of the system. In the next design phase of IMR, 3D effects of two-phase natural circulation flow behaviour will be tested as a part of basic design study.

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ANNEX XVI. IRIS
Westinghouse Electric Corporation, USA

Integral Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
IRIS *Westinghouse Electric, USA*	PWR	1000	CORE/PRIMARY: • Passive Emergency Heat Removal System (EHRS) • Emergency Boration Tanks (EBT) • Automatic Depressurization System (ADS) • Containment Suppression Pool Injection CONTAINMENT: • Containment Pressure Suppression System (PSS) • Containment Cavity

XVI-1. Introduction

IRIS is a pressurized water reactor that utilizes an integral reactor coolant system layout. The IRIS reactor vessel houses not only the nuclear fuel and control rods, but also all the major reactor coolant system components including pumps, steam generators, pressurizer, control rod drive mechanisms and neutron reflector. The IRIS integral vessel is larger than a traditional PWR pressure vessel, but the size of the IRIS containment is a fraction of the size of corresponding loop reactors, resulting in a significant reduction in the overall size of the reactor plant. IRIS has been primarily focused on achieving design with innovative safety characteristics. The first line of defence in IRIS is to eliminate event initiators that could potentially lead to core damage. In IRIS, this concept is implemented through the 'safety-by-design' ™ IRIS philosophy, which can be simply described as 'design the plant in such a way as to eliminate accidents from occurring, rather than coping with their consequences.'

XVI-2. The integral reactor coolant system

The IRIS reactor vessel (RV) [1] houses not only the nuclear fuel and control rods, but also all the major reactor coolant system (RCS) components (see Fig. XVI-1): eight small, spool type, reactor coolant pumps (RCPs); eight modular, helical coil, once through steam generators (SGs); a pressurizer located in the RV upper head; the control rod drive mechanisms (CRDMs); and, a steel reflector which surrounds the core and improves neutron economy, as well as it provides additional internal shielding. This integral RV arrangement eliminates the individual component pressure vessels and large connecting loop piping between them, resulting in a more compact configuration and in the elimination of the large loss-of-coolant accident as a design basis event.

As the IRIS integral vessel contains all the RCS components, it is larger than the RV of a traditional loop-type PWR. It has an inner diameter of 6.21 m and an overall height of 22.2 m including the closure head. Water flows upwards through the core and then through the riser region (defined by the extended core barrel). At the top of the riser, the coolant is directed into the upper part of the annular plenum between the extended core barrel and the RV inside wall, where the suction of the reactor coolant pumps is located. Eight coolant pumps are employed, and the flow from each pump is directed downward through its associated helical coil steam generator module. The primary flow path continues down through the annular downcomer region outside the core to the lower plenum and then back to the core completing the circuit. Additional details of the IRIS integral vessel can be found in IAEA-TECDOC-1451. [2]

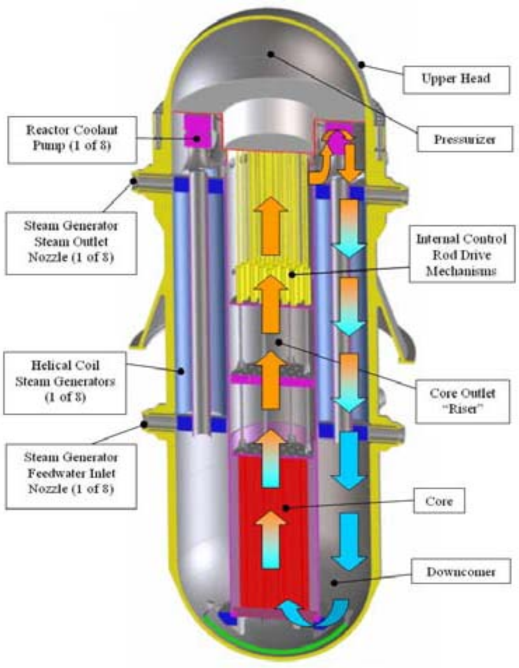

FIG. XVI-1. IRIS integral layout.

XVI-3. IRIS safety approach

The overall approach to safety in IRIS may be represented by the following three-tier approach:

1. The first tier is the safety-by-design™, which aims at eliminating by design the possibility for an accident to occur, rather than dealing with its consequences. By eliminating some accidents, the corresponding safety systems (passive or active) become unnecessary as well.
2. The second tier is provided by simplified passive safety systems, which protect against the still remaining accidents and mitigate their consequences.
3. The third tier is provided by active systems, which are not required to perform safety functions (i.e. are not safety grade) and are not considered in deterministic safety analyses, but do contribute to reducing the core damage frequency (CDF).

XVI-3.1. First tier

The first tier is embodied in the IRIS 'safety-by-design'™. Nuclear power plants consider a range of hypothetical accident scenarios. The IRIS 'safety-by-design'™ philosophy is a systematic approach that aims—by design—at eliminating altogether the possibility for an accident to occur, i.e. to eliminate accident initiators, rather than having to design and implement systems to deal with the consequences of the accident. It should be noted that the integral configuration is inherently more amenable to this approach than a loop-type configuration, thus enabling safety improvements not possible in a loop reactor. To give only the most obvious example, loss of coolant accidents caused by a large break of external primary piping (LBLOCA) are eliminated by design since no large external piping exists in IRIS. Additionally, in cases where it is not possible or practical to completely eliminate potential initiators of an accident, safety-by-design™ aims at reducing the severity of the accident's consequences and the probability of its occurrence. As a result of this systematic approach, the eight Class IV design basis events (potentially leading to most severe accidents) that are usually considered in LWRs, are reduced to only one in IRIS, with the remaining seven either completely eliminated by design, or their consequences (as well as probability) reduced to a degree that they are no longer considered Class IV events.

XVI-3.2. Second tier

The second tier consists of the passive safety systems needed to cope with the still-remaining potential accidents. Notably, the elimination of the possibility for some accidents to occur enables simplifications of IRIS design and passive safety systems, resulting simultaneously in enhanced safety, reliability, as well as economics. In other words, the increased safety and improved economics support each other in the IRIS design.

XVI-3.3. Third tier

The third tier has been addressed within the PRA/PSA (Probabilistic Risk Assessment/Probabilistic Safety Assessment) framework. In fact, PRA was initiated early in the IRIS design, and was used iteratively to guide and improve the design safety and reliability (thus adding 'reliability by design'). The PRA has suggested modifications to the reactor system designs, resulting in reduction of the predicted CDF. After these modifications, the preliminary PRA level 1 analysis estimated the CDF due to internal events (including anticipated transients without scram, ATWS) to be about 2×10^{-8}, more than one order of magnitude lower than in advanced LWRs. A subsequent evaluation of the LERF (large early release frequency) also produced a very low value, of the order of 6×10^{-10}, which is more than one order of magnitude lower than in advanced loop LWRs, and several orders of magnitude lower than in present LWRs.

XVI-4. IRIS safety features

To complement its safety-by-design™, IRIS features limited and simplified passive systems as shown in Figure XVI-2. They include:

XVI-4.1. Passive emergency heat removal system (EHRS)

IRIS implements a passive emergency heat removal system made of four independent subsystems, each of which has a horizontal, U tube heat exchanger connected to a separate SG feed/steam line. These heat exchangers are immersed in the refuelling water storage tank (RWST) located outside the containment structure. The RWST water provides the heat sink to the environment for the EHRS heat exchangers. The EHRS is sized so that a single subsystem can provide core decay heat removal in the case of a loss of secondary system heat removal capability. The EHRS operates in natural circulation, removing heat from the primary system through the steam generators heat transfer surface, condensing the steam produced in the EHRS heat exchanger, transferring the heat to the RWST water, and returning the condensate back to the SG. The EHRS provides both the main post-LOCA depressurization (depressurization without loss of mass) of the primary system and the core cooling functions. It performs these functions by condensing the steam produced by the core directly inside the reactor vessel. This minimizes the break flow and actually reverses it for a portion of the LOCA response, while transferring the decay heat to the environment.

XVI-4.2. Emergency boration tanks (EBT)

IRIS has two full-system pressure emergency boration tanks (EBTs) to provide a diverse means of reactor shutdown by delivering borated water to the RV through the direct vessel injection (DVI) lines. By their operation these tanks also provide a limited gravity feed makeup water to the primary system.

XVI-4.3. Automatic depressurization system (ADS)

A small automatic depressurization system (ADS) from the pressurizer steam space, which assists the EHRS in depressurizing the reactor vessel when/if the reactor vessel coolant inventory drops below a specific level. This ADS has one stage and consist of two parallel 4 in. lines, each with two normally closed valves. The single ADS line downstream of the closed valves discharges into the pressure suppression system pool tanks through a sparger. This ADS function ensures that the reactor vessel and containment pressures are equalized in a timely manner, limiting the loss of coolant and thus preventing core uncover following postulated LOCAs even at low RV elevations.

TABLE XVI-1. IMPLICATIONS OF SAFETY-BY-DESIGN APPROACH

IRIS Design Characteristic	Safety Implication	Accidents Affected	Condition IV Design Basis Events	Effect on Condition IV Event by IRIS Safety-by-Design
Integral layout	No large primary piping	• Large break Loss of Coolant Accidents (LOCAs)	Large break LOCA	Eliminated
Large, tall vessel	Increased water inventory Increased natural circulation Accommodates internal Control Rod Drive Mechanisms (CRDMs)	• Other LOCAs • Decrease in heat removal various events • Control rod ejection, head penetrations failure	Spectrum of control rod ejection accidents	Eliminated
Heat removal from inside the vessel	Depressurizes primary system by condensation and not by loss of mass Effective heat removal by Steam Generators (SG)/Emergency High Removal System (EHRS)	• LOCAs • LOCAs • All events for which effective cooldown is required • Anticipated Transients Without Screen (ATWS)		
Reduced size, higher design pressure containment	Reduced driving force through primary opening	• LOCAs		
Multiple, integral, shaftless coolant pumps	Decreased importance of single pump failure No shaft	• Locked rotor, shaft seizure/break • Loss of Flow Accidents (LOFAs)	Reactor coolant pump shaft break Reactor coolant pump seizure	Eliminated Downgraded
High design pressure steam generator system	No SG safety valves Primary system cannot over pressure secondary system Feed/Steam System Piping designed for full Reactor Coolant System (RCS) pressure reduces piping failure probability	• Steam generator tube rupture • Steam line break • Feed line break	Steam generator tube rupture Steam system piping failure	Downgraded Downgraded
Once through steam generators	Limited water inventory	• Feed line break • Steam line break	Feedwater system pipe break	Downgraded
Integral pressurizer	Large pressurizer volume/reactor power	• Overheating events, including feed line break • ATWS		
			Fuel handling accidents	Unaffected

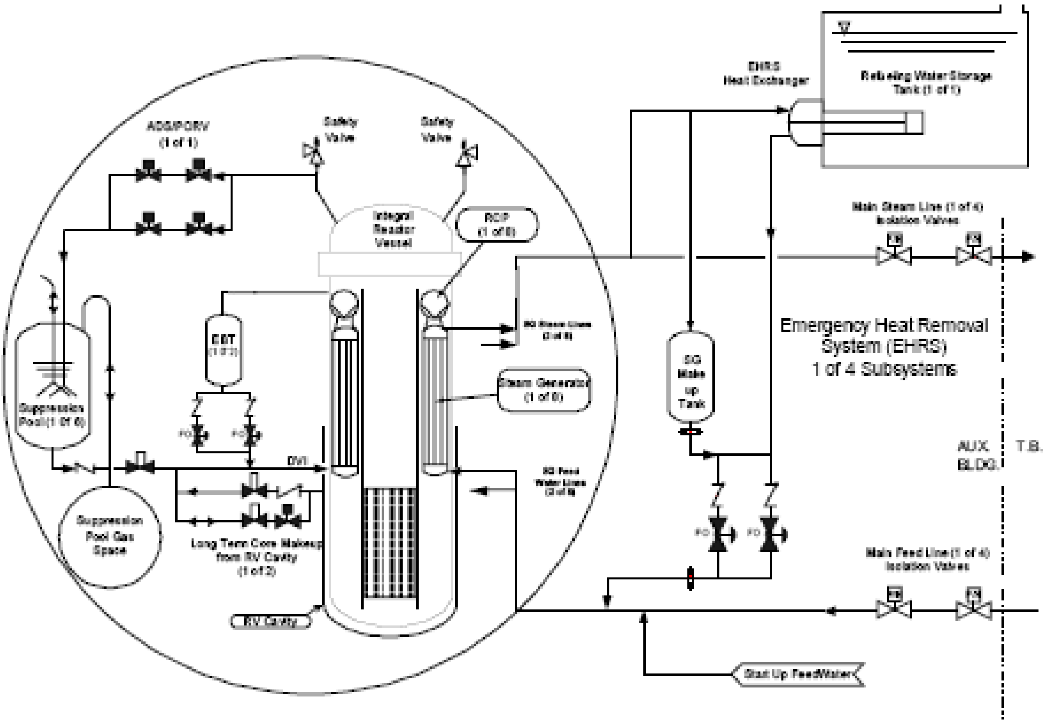

FIG. XVI-2. IRIS passive safety system schematic.

XVI-4.4. Containment pressure suppression system (PSS)

The containment pressure suppression system (PSS), shown in Figure XVI-3, consists of six water tanks and a common tank for non-condensable gas storage. Each suppression water tank is connected to the containment atmosphere through a vent pipe connected to a submerged sparger so that steam released in the containment following a loss of coolant or steam/feed line break accident is condensed. The suppression system limits the peak containment pressure, following the most limiting blowdown event, to less than 1.0 MPa (130 psig), which is much lower than the containment design pressure. The suppression system water tanks also provide an elevated source of water that is available for gravity injection into the reactor vessel through the DVI lines in the event of a LOCA.

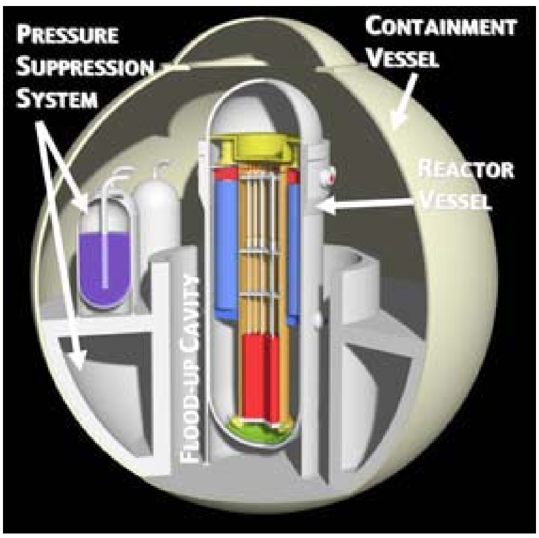

FIG. XVI-3. IRIS spherical steel containment arrangement.

XVI-4.5. Containment cavity

The IRIS design includes a specially constructed lower containment volume that collects the liquid break flow, as well as any condensate from the containment, in a cavity where the reactor vessel is located. Following a LOCA, the cavity floods above the core level, creating a gravity head of water sufficient to provide coolant makeup to the reactor vessel through the DVI lines. This cavity also ensures that the lower outside portion of the RV surface is or can be wetted following postulated core damage events.

As in the AP600/AP1000, the IRIS safety system design uses gravitational forces instead of active components such as pumps, fan coolers or sprays and their supporting systems. The safety strategy of IRIS provides a diverse means of core shutdown by makeup of borated water from the EBT in addition to the control rods; also, the EHRS provides a means of core cooling and heat removal to the environment in the event that normally available active systems are not available. In the event of a significant loss of primary-side water inventory, the primary line of defence for IRIS is represented by the large coolant inventory in the reactor vessel and the fact that EHRS operation limits the loss of mass, thus maintaining a sufficient inventory in the primary system and guaranteeing that the core will remain covered for all postulated events. The EBT is capable of providing some primary system injection at high pressure, but this is not necessary, since the IRIS strategy relies on ‘maintaining’ coolant inventory, rather than ‘injecting’ makeup water. This strategy is sufficient to ensure that the core remains covered with water for an extended period of time (days and possibly weeks).

Thus, IRIS does not require and does not have the high capacity, safety grade, and high pressure safety injection system characteristic of loop reactors. Of course, when the reactor vessel is depressurised to near containment pressure, gravity flow from the suppression system and from the flooded reactor cavity will maintain the RV coolant inventory for an unlimited period of time. However, this function would not be strictly necessary for any reasonable recovery period since the core decay heat is removed directly by condensing steam inside the pressure vessel, thus preventing any primary water from leaving the pressure vessel.

The IRIS design also includes a second means of core cooling via containment cooling, since the vessel and containment become thermodynamically coupled once a break occurs. Should cooling via the EHRS be defeated, direct cooling of the containment outer surface is provided and containment pressurization is limited to less than its design pressure. This cooling plus multiple means of providing gravity driven makeup to the core provide a means of preventing core damage and ensuring containment integrity and heat removal to the environment that is diverse from the EHRS operation.

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ANNEX XVII. MASLWR
Idaho National Laboratory, Oregon State University, Nexant, USA

Integral Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Multi-Application Small Light Water Reactor (MASLWR) *INL, OSU, Nexant, USA*	PWR	150	CORE/PRIMARY: • Steam Vent Valves • Automatic Depressurization System • Sump Recirculation Valves CONTAINMENT: • High pressure containment vessel • Internal pressure suppression pool • Passive Containment Cooling Pool

XVII-1. Introduction

The multi-application small light water reactor (MASLWR) is a 150 MW(t) modular nuclear reactor that uses natural circulation for primary loop cooling. The MASLWR system was designed by the Nuclear Energy Research Initiative (NERI), a program of the United States' Department of Energy (DOE). Other collaborators included Idaho National Engineering and Environmental Laboratory (INEEL), Oregon State University (OSU), and NEXANT-Bechtel, all in the United States of America.

The design philosophy was to use existing pressurized water reactor (PWR) knowledge to develop a safe and economical small power source using only natural circulation in the primary system. Through economic considerations, the project moved towards design of factory-fabricated modules, designed to operate in a power farm of 30 units producing 1050 MW(e), although a fraction of that, down to a single module, could also be used in a given location. A single MASLWR module produces 35 MW(e), using 8% enriched fuel and can last for 5 full-power years without replacement.

Figure XVII-1 shows a single power generation module for the MASLWR design. MASLWR implements an integrated reactor vessel with an internal helical coil steam generator. The reactor vessel is enclosed in a high pressure steel containment vessel that is partially filled with water to serve as a suppression pool. The containment vessel in turn resides in a large exterior cooling pool that acts as the ultimate heat sink.

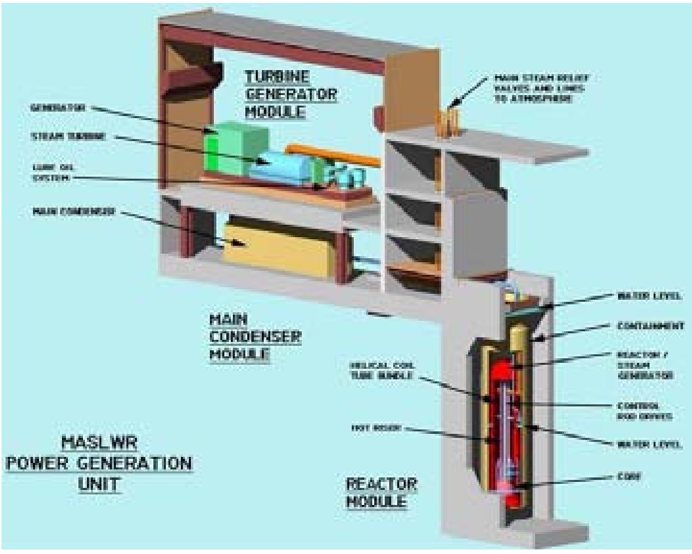

FIG. XVII-1. Schematic of the MASLWR exterior cooling pool and turbine-generator set.

XVII-2. MASLWR primary loop design

Because MASLWR uses natural circulation for primary loop flow, reactor coolant pumps are not needed. In this regard, its primary flow loop is quite simple as shown in Figure XVII-2. The long vertical tube in the centre of the reactor vessel is called the riser and functions like a chimney to enhance the driving head of the natural circulation flow. Starting from the bottom of the riser, fluid enters the core, which is located in a shroud connected to the riser entrance. While the fluid travels through the core, it is heated and rises by buoyancy through the riser. Hot fluid in the surrounding annulus, outside the riser is cooled by convective heat transfer to a helical coil steam generator. The fluid inside the tubes is at a lower pressure, hence boiling occurs inside the tubes to generate superheated steam. The steam produced within the tube side of this coil travels on to the turbine generator set where it is used to produce electrical power. The cooled primary fluid in the annulus is negatively buoyant and descends to the bottom of the vessel and the inlet of the core thereby completing its loop.

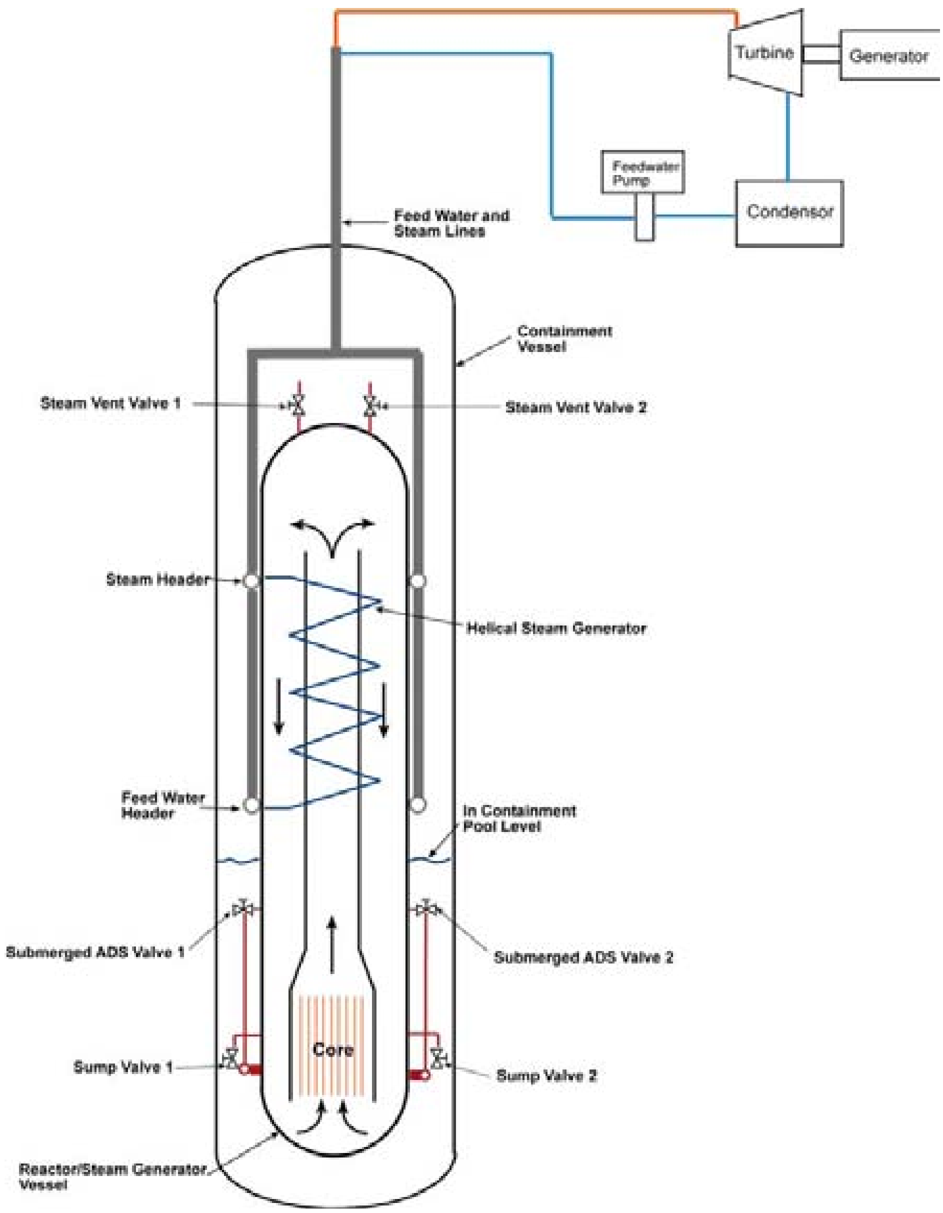

FIG. XVII-2. Schematic of the MASLWR reactor cooling system and containment.

Table XVII-1 lists the steady-state operating conditions for MASLWR. The design provides a 53°K temperature rise from the core inlet to the core outlet. In addition, it is designed to provide superheated steam at the helical coil outlet to eliminate the need for separators and driers. The secondary side pressure was selected so that off-the-shelf low pressure steam turbines could be implemented.

TABLE XVII-1. MASLWR STEADY-STATE OPERATING CONDITIONS

Primary Side	
Reactor Power (MW_t)	150.00
Primary Pressure (MPa)	7.60
Primary Mass Flow Rate (kg/s)	597.00
Reactor Inlet Temperature (K)	491.80
Reactor Outlet Temperature (K)	544.30
Saturation Temperature (K)	565.00
Reactor Outlet Void Fraction	0.00%
Secondary Side	
Steam Pressure (MPa)	1.50
Steam Outlet Quality	1.00
Steam Temperature (K)	481.40
Saturation Temperature (K)	471.60
Feedwater Temperature (K)	310.00
Feedwater Flowrate (kg/s)	56.10

XVII-3. MASLWR passive safety system SBLOCA operations

This section briefly describes the evolution of a small break loss of coolant accident in MASLWR. Because MASLWR is an integrated reactor system, there are very few plausible primary break scenarios. In the event of a small break, the MASLWR passive safety systems would respond to the accident. The passive safety systems consist of the following components:

- Two, independent, small diameter, steam vent valves (SVV)
- Two, independent, small diameter, automatic depressurization system (ADS) valves
- Two, independent, small diameter, sump recirculation valves (SV)
- A high pressure containment vessel with an internal pressure suppression pool, and
- An external cooling pool that serves as the ultimate heat sink for the high pressure containment and reactor decay heat.

Let us postulate the inadvertent opening of an ADS valve. Figure XVII-3 provides a schematic of the postulated pressure trend for illustration purposes. As shown in the figure, the transient begins with a relatively short blowdown period that consists of a subcooled blowdown into the suppression pool within the stainless steel containment. The suppression pool consists of the annular space bounded by the exterior surface of the reactor vessel and inner surface of the containment walls. It is partially filled with water. This water region is integral to the long term removal of decay heat following system depressurization (blowdown). The rapid rise in containment pressure results in a Safety Injection signal, which automatically opens the steam vent valves, the ADS valves, and the sump recirculation valves. The ADS blowdown period serves to further reduce the reactor vessel pressure well below the saturation pressure corresponding to the hot leg temperature. A major advantage to the small volume, high pressure containment is that the blowdown quickly results in equalizing the containment and reactor vessel pressures, effectively terminating the blowdown. As the pressures become equalized, a natural circulation flow path is established in which the sump fluid enters through the sump recirculation valves, descends through the downcomer region in the lower portion of the reactor vessel, rises through the core and riser, and finally exits through the upper vent valve into the containment as a saturated vapor. From the vent valve, the fluid returns to the sump via condensation on the containment walls and/or water surface, thus completing its circuit.

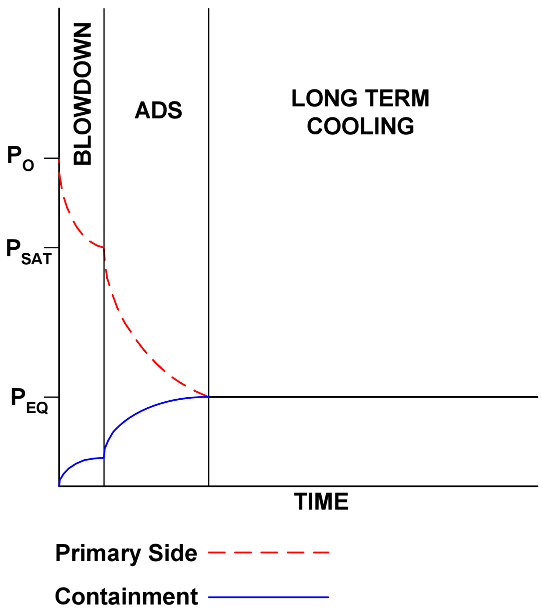

FIG. XVII-3. Illustration of transient phases for a MASLWR SBLOCA.

Finally, to ensure the long term removal of heat from the containment and thus to moderate the containment pressure, the containment itself is submerged in an outer pool of water which is open to the atmosphere. Within this pool, thermal energy is transferred from the outer containment wall to the atmosphere via natural convection and circulation of the water. The pool is formed in the space between the outer containment wall and the inner wall of the concrete structure in which the containment is placed.

In conclusion, a SBLOCA in MASLWR can be divided into three phase, or modes of operation, a blowdown phase, an ADS operation phase and a long term cooling phase. A more detailed description of the MASLWR design is provided in IAEA-TECDOC-1536.

ANNEX XVIII. PSRD
Japan Atomic Energy Agency (JAEA), Japan

Integral Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Passive Safe Small Reactor for Distributed Energy Supply System (PSRD) *Japan Atomic Energy Agency (JAEA)*	PWR	100	CORE/PRIMARY: • Emergency Decay Heat Removal System • Hydraulic Force Valve CONTAINMENT: • Containment Water-Cooling System

XVIII-1. Introduction

The PSRD (passive safe small reactor for distributed energy supply system)-100 is an integral type PWR with the thermal power of 100 MWt developed at the Japan Atomic Energy Research Institute based on the technologies developed for the marine reactor MRX (KUSUNOKI, T. et al., 2000; ISHIDA, T., et al., 2003-1; IAEA-TECDOC-1391). The reactor is designed for the electricity generating for a small power-grid, heat supply, and/or sea water desalination. The major characteristics are listed in Table XVIII-1.

Siting in the demand area is one of desired characteristics for this kind of reactors, which requires the extremely-higher level of the safety compared to that for the current generating LWRs without compromising economic competitiveness. In order to realize this, several futures are incorporated in the design, which includes:

- Elimination of the pipes connecting to the primary pressure boundary by adopting in-vessel steam generators (SG) and disconnecting volume control and purification systems during the power-operation period to limit the primary pipes only to those for the safety valve lines, which significantly reduces the possibility of a loss-of-coolant accident (LOCA);
- Adoption of a small and high pressure water-filled containment that can mitigate effects of a LOCA by terminating the primary coolant discharge before the core is exposed to the steam environment;
- Adoption of an in-vessel control rod drive mechanism (INV-CRDM) to eliminate the possibility of a reactivity insertion accident caused by a control-rod withdrawal;
- Natural circulation core cooling to eliminate the possibility of a transient caused by a circulation pump failure and simplify the system;
- Full utilization of passive cooling systems to enhance the reliability of the decay heat removal during accidental conditions and simplify the system; and
- Adoption of the core design that enables continuous full power operation for five years without refuelling.

Figure XVIII-1 shows the cross-section of the reactor pressure vessel (RPV) and the containment vessel (CV). Inside the RPV, the core is located in the lower part, the SGs in the middle part, and the INV-CRDMs in the upper part. The core is radially surrounded by the radiation shield located outside the core barrel. The RPV is not fully filled with water: the nitrogen gas occupies the top part of the RPV to absorb the primary liquid volume change. The SG is the once-through, helical coil tube type, where the secondary side coolant flows up inside the tubes. The volume control and purification systems are not connected to the primary loop during the power operation to limit the pipes composing the primary pressure boundary to those for the safety valve lines. The CV is filled mostly with water so that the RPV, and the heat exchangers for the emergency decay heat removal system (EDRS) and the containment water-cooling system (CWCS) are submerged. The water inside the CV has the role of radiation shielding. The RPV is covered with the water-tight shell (WTS) to house stainless-steel felt between the RPV and WTS for the thermal insulation. The heat loss from the RPV with this insulation is estimated less than 1% of the rated power.

XVIII-2. Description of natural circulation core cooling system

The core is cooled mostly by the single-phase free convection due to natural circulation. The natural circulation loop consists of the core, the chimney, the downcomer and the lower plenum. In order to establish natural circulation at different RPV liquid levels during normal operations and accidents, the core barrel has large communication-holes above the SG top elevation and small holes at several elevations below that. The bypass flow through the small holes is approximately 5% of the core flow during the full-power operational condition. Since the RPV coolant is self-pressurized, that is, the pressure is determined by the temperature at the liquid level in the RPV, the core outlet is saturated. The fuel assembly is based on the 17×17 fuel assembly design with Zircaloy-4 cladding UO_2 pellets that is used for the current PWRs except for the fuel pin pitch. The pitch of the PSRD is 13.9mm, which is 1.3mm wider than that of the current PWRs to enhance burn-up by increasing the moderation effect. This geometry decreases the flow resistance along the core.

TABLE XVIII-1. MAJOR CHARACTERISTICS OF THE PSRD-100

Reactor power (MWt) Power output (MW(e))	100 27 to 31
Reactor coolant Operation press. (MPa) Inlet/Outlet temp. Flow rate (kg/s)	 10 270.4/311 450
Reactor core Diameter/height (m) U^{235} enrichment (Wt%) Fuel inventory (t)	1.62/1.50 ~4.9 7.1
Fuel Outer dia./pitch (mm) Burnable poison No. of fuel assembles	9.5/13.9 9%Gd_2O_3 37
Control rod and CRDM Absorber No. of control rods	B_4C 24 × 37
Steam generator Type Temp./press.(ºC/MPa)	Once-through 289/4.0
Reactor vessel Inner dia./height(m)	4/10
Containment Type Design press. (MPa) Inner dia./height(m)	Water-filled 2 7.3/14.6

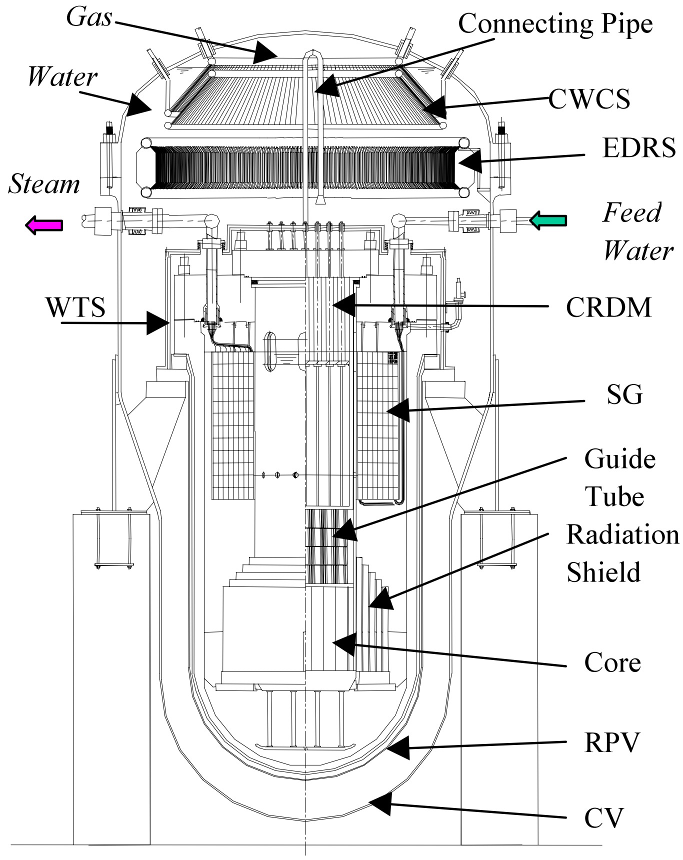

FIG. XVIII-1. Concept of PSRD

So far, the PSRD natural circulation core cooling has been evaluated for steady and transient states by using the thermal-hydraulic analysis codes. During such analyses, the load following capability was analyzed using the RETRAN-02 code by decreasing the feedwater flow rate from 100% to 50% in 200 sec, keeping it at 50% for 200 sec, and then increasing from 50% to 100% in 200 sec to represent a typical day-load change as shown in Fig. XVIII-2(a) (ISHIDA, T., 2003-2). The automatic reactor control system was not used to clarify the inherent self-controllability of the system. The calculation results showed that the reactor power responded well with the delay time of 50 to 100 sec and the overshoot of up to 10% as shown in Fig. XVIII-2 (b). The response of the natural circulation flow rate was also stable without showing high-frequency oscillations as shown in Fig. XVIII-2 (c). The results indicated the inherently stable nature of the PSRD natural circulation cooling system. The results also indicated that the responses will be more stable for the actual load follow condition with the operation of the reactor automatic control system without excessively depending on the use of the control rod operation.

XVIII-3. Description of passive cooling systems

The passive cooling systems of the PSRD shown in Fig. XVIII-3 consist of four natural circulation loops: the first one is to transfer decay heat from the core to the SG; the second one from the SG to the EDRS heat exchanger (HEX); the third one from the EDRS-HEX and the CWCS-HEX inside the CV; and the fourth one from the HEX inside the CV to that outside the CV in the CWCS where the heat is transferred to the atmosphere. The second system called EDRS starts the operation passively after the feedwater pump termination due to the safety signal generation upon an abnormal event occurrence. This passive initiation relies on the hydraulic force valve developed at the JAERI. The hydraulic force valve has the pressure-imposing line connecting to the feedwater pump to control the valve condition. The valve remains closed only when the pressure in the pressure-imposing line is higher than that inside the valve, i.e. the SG secondary pressure for this application. The feedwater pump termination, therefore, creates the pressure condition that opens the valve. After the hydraulic valves are passively opened, the steam will flow up toward the EDRS-HEX from the main steam line, while the liquid initially existed in the EDRS-HEX will flow down to the feedwater line.

Since the water is filled in the containment, the liquid-phase natural circulation between the EDRS-HEX and CWCS-HEX starts immediately after the temperature difference is generated between the two HEXs. The CWCS is a heat pipe system utilizing a refrigerant. The operation of the CWCS does not require the valve operation: the valves in the CWCS, if any, are opened for the normal condition. In summary, the decay heat removal system operation of the PSRD relies only on the operation of the hydraulic valve having inherently fail-safe nature, which means the passive cooling system of the PSRD is very reliable.

The capability of these natural circulation systems has been analyzed by using the RELAP5 code, which confirmed the feasibility of the basic concepts of this system (ISHIDA et al., 2003-2). Although the natural circulation behaviour in the containment is difficult to be analyzed by the RELAP5, much more accurate analysis can be performed by using the computational fluid dynamics (CFD) code because the analysis target is the single-phase liquid natural circulation.

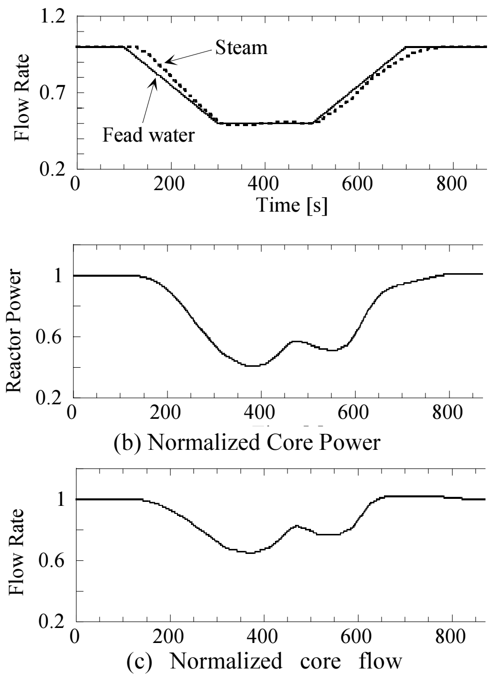

FIG. XVIII-2. *Reactor responses to load change calculated using RETRAN-02: (a) imposed feedwater change, (b) core power response, (c) core flow response.*

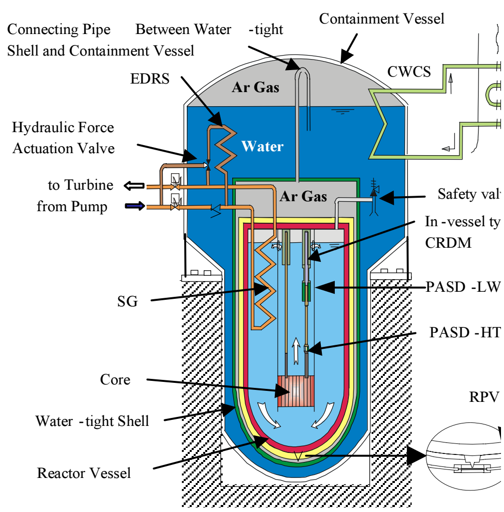

FIG. XVIII-3. The passive cooling systems of the PSRD: EDRS for emergency decay heat removal system, CWCS for containment water cooling system.

XVIII-4. Conclusions

The basic design for the PSRD has been completed. The feasibility of the concepts of the PSRD has been confirmed through the analyses with thermal-hydraulic codes developed for the current-generation LWRs. The basic nature of the load follow characteristics has been characterized with the RETRAN code. The analysis results indicated that the reactor can respond smoothly to a typical load change due to a rather large negative moderator density reactivity coefficient. The functions of the passive safety systems to transfer the decay heat to the atmosphere and maintain the RPV coolant inventory have been confirmed through the LOCA analyses with the RELAP5 code. The current research focuses on the optimization of the system and measures to site in the demand area such as the placement in a deep pit filled with sea-water in a harbor.

REFERENCES TO ANNEX XVIII

[1] ISHIDA, T., et al., Development of In-vessel Type Control Rod Drive Mechanism for Marine Reactor, J. Nucl. Sci. and Tech., vol.38, No.7 (2001) pp.557-570.

[2] ISHIDA, T., et al., Concept Of Passive Safe Small Reactor For Distributed Energy Supply System, Proc. of 11th International Conference on Nuclear Engineering, (2003) ICONE11-36470.

[3] ISHIDA, T., SAWADA, K., YNOMOTO T., Performance of Safety System of Passive Safety Small Reactor for Distributed Energy Supply System, Proc. of GENES4/ANP2003 (2003).

[4] KUSUNOKI, T., et al., Design Of Advanced Integral-Type Marine Reactor MRX, Nuclear Engineering and Design, Vol. 201 (2000) pp.155-175.

[5] INTERNATIONAL ATOMIC ENERGY AGENCY, Status of advanced light water reactor designs 2004, IAEA-TECDOC-1391 (2004) pp.755-769.

ANNEX XIX. SCOR
Commissariat à l'Energie Atomique, France

Integral Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
Simple COmpact Reactor (SCOR) *Commissariat à l'Energie Atomique, France*	PWR	2000	CORE/PRIMARY: • Residual Heat Removal System on Primary Circuit (RRP). CONTAINMENT: • Dedicated steam dump pool to prevent radioactivity release into the atmosphere in case of steam generator tube rupture. • Containment pressure-suppression system. • In-vessel core retention with corium cooling by pit flooding. • Inert atmosphere in the reactor vessel compartment to prevent hydrogen combustion.

XIX-1. Introduction

SCOR (Simple COmpact Reactor) is a 2000 MW•th, integrated pressurized light water reactor (PWR), whose design was developed at the Nuclear Reactor Division of Commissariat à l'Energie Atomique at Cadarache in France.

The plant design, begun in 2000, was mainly defined for electricity production but could be adapted for desalination. The net power is 630 MW(e) (31.5% net efficiency). The SCOR concept respects the European Utility Requirements. The SCOR design is based on a compact reactor vessel that contains all the reactor coolant system components, including the pressurizer, the reactor coolant pumps, the control rod drive mechanism and the dedicated heat exchangers of the passive decay heat removal systems. The single steam generator is located above the reactor vessel. When applicable, the SCOR development adopted the generic notion of « Safety by design »[5], sort of 'design oriented safety approach' which give the priority to a whole plant inherent behaviour able to meet an increased and well mastered level of safety vis à vis of the different safety functions.

XIX-2. Passive features of the SCOR reactor

The integral SCOR design incorporates PWR technology proven by more than 40 years of experience, and adopts passive safety systems studied at CEA in the 1990s. The safety approach of SCOR strengthens the prevention of abnormal operation and acceptability of failures for a large set of conditions, thereby reducing the expected frequencies of accident initiators and consequences. The design approach reduces the number and complexity of the safety systems and simplifies the required operator actions compared to those of a standard loop type PWRs (Generation II).

Nuclear reactor characteristics described here are quite similar to well-known PWRs (core geometry and materials, reactor control type, etc.). The SCOR design is characterized by the following innovative solutions:

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[5] See, for example, the detailed description of the effect of 'safety by design' given in IAEA-TECDOC-1391 Status of Advanced LWR Designs: 2004 in the description of the IRIS design (p. 591-592).

- Suppression of the large diameter connections on the reactor pressure vessel;
- Passive and integrated emergency core cooling system, based only on natural circulation and using external air as ultimate heat sink;
- Reactor operating with a soluble boron-free core; the control rod mechanisms are integrated in the vessel;
- Choice of a small power density, enabling large operating margin on core power parameters (i.e. to DNBR);
- Easy testability and maintenance of all safety systems.

Consequences of a significant number of accidents are either outright eliminated or reduced by conception, i.e. without any need of active or passive systems. The major safety systems are passive; they require no operator action or off-site assistance for a long time after an accident. Moreover, core and containment cooling is provided for lasting a long time without alternative current power. This approach is generic for different reactor designs.

XIX-3. Description of the nuclear systems

XIX-3.1. Primary circuit and its main characteristics

The SCOR is an integrated PWR with a compact primary circuit. The reactor pressure vessel houses the main primary system components including the core, pressurizer, steam generator, reactor coolant pumps, control rod drive mechanism (CRDM) and the heat exchangers of the decay heat removal system. This design configuration eliminates large penetrations in the reactor vessel, thus excluding the possibility of large break loss of coolant accidents.

The single steam generator acts as the reactor vessel head as shown in Figure XIX-1. The flow path of the reactor coolant is illustrated in this figure. From the lower plenum, water flows upward through the core and the riser and through the centre of the pressurizer.

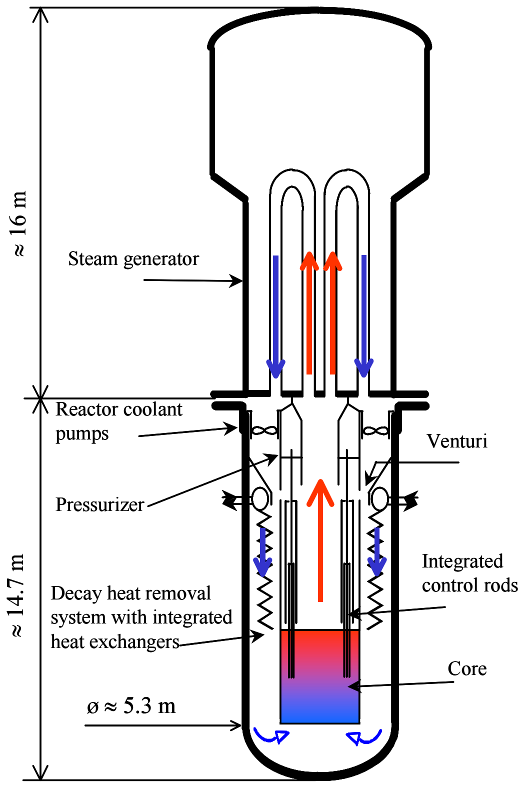

FIG. XIX-1. Diagram of SCOR design.

In the upper portion of the vessel, the fluid flows upward and then downward inside the U tubes of the steam generator. Then the fluid is collected in an annular plenum and passes to the inlet of the reactor coolant pumps. From the pump outlet, the coolant flows through a venturi and then across the tubes of the decay heat exchangers to the lower plenum.

XIX-3.2. Core and fuel features

The core is similar to the core of French 900 MW(e) PWRs with 28% lower power density than the standard French PWR. The core consists of 157 assemblies of 264 fuel rods in a 17 × 17 square array, with an active fuel height of 3.667 m. The central position in each fuel assembly may be used for in-core instrumentation, and 24 positions have guide tubes for control rods. The core thermal power is 2 000 MW.

It is a soluble boron free core. The reactivity is controlled with integrated CRDM (control rod drive mechanism).

XIX-3.3. The heat exchanger-pump modules

The large annular space between the core barrel shell and the reactor vessel contains the heat exchanger-pump modules. Each of the sixteen modules of Figure XIX-2 comprises a primary pump and a heat exchanger used to remove residual heat.

As shown in the figure, the submerged pump, is supplied with water exiting the steam generator. The coil-type motor is submerged downstream of the impeller. The pump motor has a stationary cylindrical stator surrounding the rotor that is connected to the pump's impellers. Electrical power is supplied to each of the 16 pumps by cables through small penetrations in the reactor vessel. The primary water that flows around the outer ring cools the coils; therefore associated piping penetrations through the reactor vessel for cooling water are eliminated. The spool pump needs an additional external motor to provide high inertia during coastdown in order to mitigate the consequences of Loss-Of-Flow Accidents (LOFAs). This additional inertia, provided by an external motor with an adequate flywheel, is linked to the spool type pump by an electrical connection. At the outlet of the pump, water is accelerated by a venturi, passes into a diffuser and then through the decay heat exchanger tube bundles.

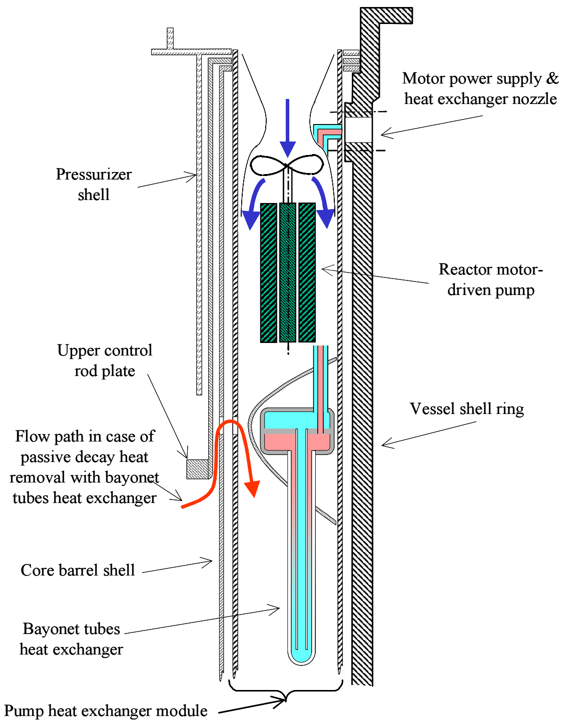

FIG. XIX-2. Decay heat exchanger-pump module.

The decay heat exchanger of the RRP system (see XIX-4.1.2) consists of bayonet tubes whose outside surface is wetted by the primary fluid. The secondary water flows first in the internal tube and then upwards through the annular space bound by the two tubes. The water box is located in a dead zone, behind the venturi. This type of heat exchanger is of interest, as it does not require a water box at the exit of the heat exchanger. This reduces the primary pressure drop and allows the free expansion of the tubes. Thermal loadings are reduced leading to an increased mechanical resistance and an enhanced reliability.

A flow bypass is installed where the venturi is located, between the core exit and the cold leg. It allows a natural convection of the primary fluid during pump shutdown. During normal operation, high flow velocity at the venturi throat leads to a decrease of the local pressure. The cross section area of the venturi throat is designed to balance the pressure between the hot leg (core exit) and the cold leg (heat exchanger-pump module), in order to prevent bypass flow in normal condition. This primary flow layout with a venturi and an integrated heat removal system is issued from the CEA patent N° 92 05220

The decay heat exchanger-pump module can be easily extracted from the reactor vessel once the steam generator has been removed. The pump power supply and the heat exchanger secondary feed-lines are set in the vessel via a removable opening in the upper part of the reactor vessel.

XIX-3.4. Pressurizer

The SCOR pressurizer is integrated into the upper part of the riser, just below the steam generator. The pressurizer region is designed as an annular shape in the form of an inverted U, see Figure XIX-1. The coolant flows through the central part of the pressurizer. The bottom portion of the inverted U contains the opening to allow water insurge and outsurge to/from the pressurizer region

The electric heaters are located in a small volume tank outside the reactor vessel and act as a steam source. The cold water supply is tapped off just downstream the pumps and the two-phase mixture is reinjected at the top of the pressurizer.

Owing to the low pressure and low temperature operating point leading to smaller variations of water density versus temperature, the volume of the pressurizer is smaller than those used in plants with a classical, separate, pressurizer vessel. The SCOR pressurizer has a total volume of ~21 m^3. This volume is large enough to manage a blackout without steam release through the safety valve of the pressurizer.

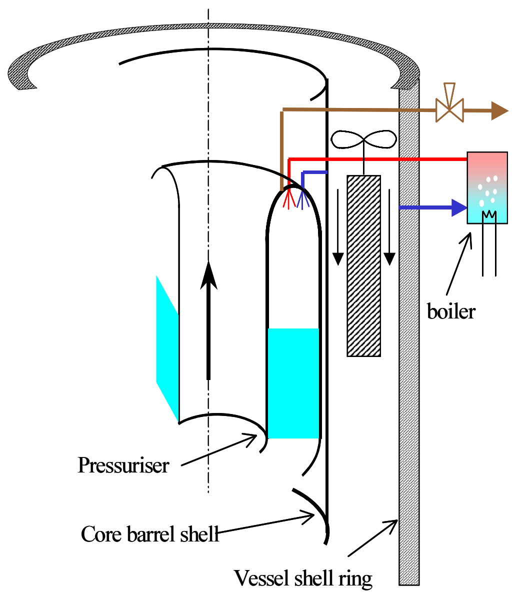

FIG. XIX-3. Pressurizer diagram.

XIX-3.5. Steam generator

There is only one U tube boiler steam generator. Like propulsion reactors, the SG is placed above the core. In contrast to standard SGs, the present generator has an axial symmetry. The hot leg is located in the centre and the cold leg is located in periphery.

XIX-3.6. Operating point

According to the results from a low pressure PWR studies, SCOR operates at a pressure of 88 bars. The secondary pressure at the turbine inlet is 30 bars, which is lower than that of standard PWRs. The net thermodynamic efficiency is 31,5%, slightly smaller than that of standard PWRs.

XIX-4. Description of the safety systems

XIX-4.1. Decay heat removal systems

Since the reactor has only one steam generator, the decay heat removal systems are diversified on both the primary and secondary circuit.

XIX-4.1.1. System on the secondary circuit

In case of an accident, the heat removal device should not release steam in the atmosphere during a SGTR (steam generator tube rupture). In case of over pressure transient, the released steam is condensed in a dedicated pool. The steam generator is not considered as the main system for decay heat removal. It acts as a thermal buffer until the safety systems on the primary side are fully operational.

XIX-4.1.2. System on the primary circuit

The primary system is cooled by means of the heat exchangers located in the downcomer (Figure XIX-4). Each exchanger has a dedicated heat sink. So there are sixteen independent loops, called RRP system (Residual heat Removal on Primary circuit). There are two types of heat sinks:

- Four RRP are cooled by immersed heat exchanger in a pool (RRPp);
- Other twelve are cooled by heat exchanger in air-cooling tower (RRPa).

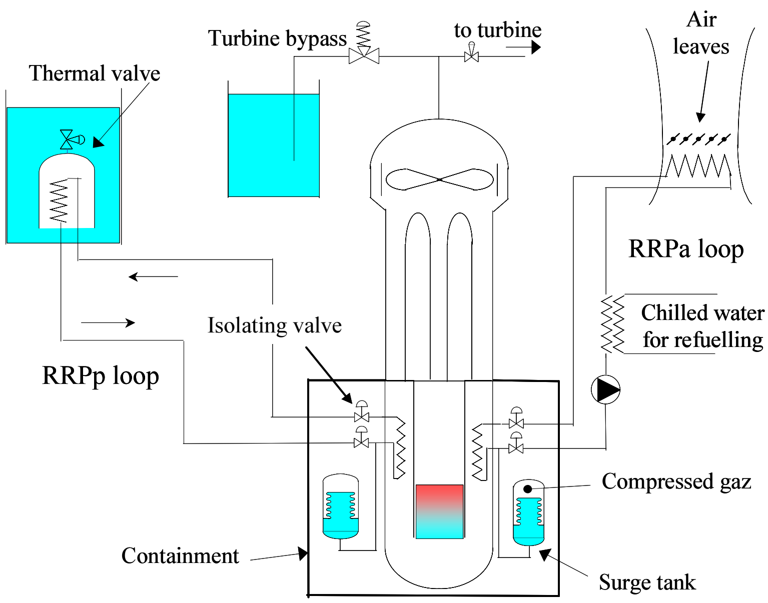

FIG. XIX-4. Cooling systems on the primary circuit.

All RRP are able to operate in natural convection in the loop and in the heat sink.

The design of the circuits is very simplified. The RRP loops are designed to resist the primary pressure. Isolating valves are placed on the circuit to avoid the risk of primary water passage outside the containment in the event of heat exchanger tube rupture. A surge tank for the water dilations from the cold shutdown to the full power operating condition carries out the pressure control of RRP circuit.

There is no control valve on the secondary loop, those are placed on the level of the heat sink: thermal valves or air leaves so that the temperature of the RRP loop remains high when the reactor is in power. Therefore, the RRP is ready to passively operate, just by the opening of the air leaves on RRP air coolers or by the opening of the thermal valves on RRP pools.

Forced convection is only required when the chilled cooling is needed for core refuelling. The twelve RRPa are able to cool the primary system down to cold shutdown. They replace the normal reactor heat removal system.

Safety features of the RRP systems

The maximum removed power by each RRP loop is about 5 to 7 MW according to the operating conditions. This small removed power, whatever the reactor power, makes it possible to test the heat removal system, with the reactor in operation, without significantly disturbing operating conditions. This procedure of test constitutes a significant element with respect to the reliability of these systems.

The RRPp are safety grade chilled water loop and pumps excepted.

XIX-4.2. Normal residual heat removal system

In hot conditions, the residual heat is removed through the steam generator. The steam is discharged to the atmosphere, and the Steam Generator is fed by the start-up shutdown system (SSS). The system is not safety grade. Then, at low temperature, the RRP with the air-cooling tower (RRPa) removes the decay heat.

When the vessel is opened, especially during the refuelling operating, decay heat is removed by the twelve RRPa cooled by chilled water in order to obtain a very low primary temperature compatible with the maintenance action conditions. The primary circuit operates in natural convection and the RRPa loops operate in active condition.

Chilled water is only used during the refuelling operation. In case of chilled water circuit or RRPa pump failure, the heat sink is backed up by the air-cooling tower of the RRPa. The RRP system replaces the normal residual heat removal system with an external loop like in standard PWRs.

XIX-4.3. Safety injection

Since large LOCAs are eliminated by design and since the primary system thermal inertia is larger than that of loop-type PWRs, the safety injection system requires devices with a small flow rate. Given the intrinsic low pressure option for the reactor, there is only one type of safety injection with a pressure of about 20 bars. The needed pump power for the safety injection is very small, about 35 kWe.

XIX-4.4. Description of the passive management of accidents in the design basis conditions

Among the design basis conditions, the main accidental transients (blackout, steam line rupture, steam generator tube rupture, and LOCA) were studied with the CATHARE code. All calculations were performed with 4 out of 16 available RRP loops. These four RRPs are automatically actuated with the shutdown signal. In case of failure of the actuation, a control system will actuate other RRP loops.

Blackout

For this transient, the power is firstly removed by the SG and later by the RRP system. The RRP system reaches its full operation at about one thousand seconds. After one hour and half, the removed power by RRP is enough to cool the reactor.

Cold shock

In case of steam line rupture, the magnitude of cold shock is 22°C at core inlet. The number of the control rods is sufficient to stop the reactivity increase until the cold shut down. After the trip, the residual power is firstly removed by the released steam of the SG, and secondly it is stored in the large thermal inertia of the primary circuit. The RRP loops reach their full power in one thousand seconds. One hour after the beginning of the transient, the RRP remove all residual power. In this transient there's no released water at the safety valve of the pressurizer.

Loss of coolant accident

The largest LOCA considered (2 × 50 mm) is the break of the line between the vessel and the boiler of the pressurizer, Figure XIX-3. At the beginning, the power is removed by the break and by the steam generator. As for blackout, the RRPa reach their full power in one thousand seconds. After a stabilisation at the pressure of the secondary safety valve, the primary pressure reaches the threshold pressure of the safety injection system at 4 000 s.

Steam generator tube rupture

During the first thousands seconds, the residual power is removed by the SG and by the RRP system. To prevent steam release to the atmosphere, the steam is condensed in a dedicated pool. With 4 RRP loops and 5 tubes ruptured, the mass of released steam is about 40 to 50 tons according to this heat sink of the RRP; with 8 RRPa, the released steam is 20 tons. After six thousand seconds, the RRP system is enough to cool the reactor and the released steam by SG is stopped.

Conclusion of the design basis accident management

The intrinsic behaviour of SCOR is considerably improved comparing a standard PWR. This is due to the large thermal inertia, the elimination of the large LOCAs and the suppression of soluble boron. Without any action (i.e. no RRP and no safety injection), the delay before core dewatering is one and half-hour after the beginning of the most penalizing transient.

Safety calculations showed that all the transients were correctly managed in a passive way (in the vessel, in the RRP loop, and in the heat sink) with only 4 out of 16 RRP, whatever the heat sink: pool or air-cooling tower. This represents a redundancy of 16 times 25%. The RRP operation is compatible with an active or passive way whatever the primary pressure or the temperature. The in-vessel heat exchangers of the RRP loop being located very close to the core, and thanks to the flow bypass of the venturi, the RRPs are operational in two-phase flow mode in case of small primary water inventory. Long term cooling may be ensured in total passive condition thanks to the RRP with air-cooling tower. Only a safety injection at 20 bars with a small flow rate is needed one hour after the beginning of the most penalizing LOCA, that is a double break of the pressurizer line (2 × 50 mm). In the event of SGTR, the steam released from the safety valves of the secondary circuit is condensed in a dedicated pool. There's no released steam in the atmosphere.

The comparison of the typical design basis events between standard PWRs and SCOR is summarized in Table XIX-1.

TABLE XIX-1. STANDARD PWRS AND SCOR RESPONSE TO ACCIDENTAL CONDITIONS

Initiating event	***Transient progress in standard PWRs***	***Transient progress in SCOR***
NPP blackout	• Natural convection in the primary circuit; • Need for an external electric source (diesel) for systems in support (seal pump, safety injection, etc.); • Heat sink covers few hours.	• Natural convection in the primary circuit, • Very few systems in support (reduced power of the diesels or battery), • Infinite autonomy of RRP systems with air heat sink.
Steam line rupture	• Risk of recriticality; • High pressure safety injection (HPSI) with borated water required.	• No risk of recriticality; • Not need for safety injection.
LOCA	• Possible fast core uncovering depending on break size; • Need for three types of safety injection systems: HPSI, accumulators, low pressure safety injection (LPSI); • request for quick safety injection (according to break size); • Long term cooling by LPSI (active system).	• No fast core dewatering (at least 1.5 hours after the transient start with no RRP operation); • Only one type of safety injection - LPSI with small flow rate is needed; • No request for immediate LPSI, • Long term cooling by the RRP systems in passive mode.
SG tube rupture	• Risk of primary water release through the broken SG; • Request for safety injection disturbing the transient management; • Delicate management of the decreasing pressure to prevent secondary water without boron from flowing into the primary circuit through the steam generator broken tubes.	• No steam release to the atmosphere (steam is condensed in a pool); • Cooling by the RRP systems; no need for safety injection; • Primary coolant has no soluble boron; hence, no risk of dilution by secondary coolant.

RRP: Residual heat removal on primary circuit.
HPSI/LPSI : High/low pressure safety injection system.

XIX-4.5. Beyond design basis accidents

XIX-4.5.1. Severe accidents

Compared to the standard PWR, safety is improved by the elimination of initiating events based on a specific design, as mentioned previously (optimum between safety, economy and human interface): large breaks on the primary circuit, reactivity insertion accident by rod ejection. However, the hypothetical case of a core meltdown is manageable in the following manner:

- Core meltdown: corium cooling should be ensured by reactor vessel pit flooding, because the core power density is small, and the large grace delay before an hypothetical core meltdown reduces the decay heat when the corium enters the lower plenum.

- Hydrogen risk: the Reactor Vessel Compartment atmosphere is inerted (cf. Figure XIX-5) to prevent hydrogen combustion like BWRs.

XIX-4.5.2. Management of the design extension condition

For SCOR, the transients in the design extension conditions are practically eliminated.

- H1 (total loss of the heat sink): SCOR concept is based on several independent decay heat removal loops (RRP) ready to operate in passive mode, having their heat sink either on pools with a limited availability of several hours, or on air cooling tower whose availability is almost infinite.
- H2 (total loss of feed water of the Steam Generator): the decay heat is removed by systems located on the primary circuit with a large redundancy (16 × 25%). There is no need of safety auxiliary feed water system.
- H3 (total loss of the power supplies): natural convection is possible on all the decay heat removal systems with the integrated heat exchangers, from the primary circuit to the heat sink.
- H4 (loss of the containment spray or loss of the low pressure safety injection): SCOR has no containment spray, because the containment is a pressure suppression containment type. The low pressure safety injection has a less significant role than in standard PWRs, because of the large primary inertia, the suppression of the large LOCAs and the strong effectiveness of decay heat removal systems.
- ATWT (Anticipated Transient Without Trip): SCOR has two independent shutdown systems. These transients will be treated individually as for the standard PWRs. The management of this complementary condition is eased owing to the always negative and higher moderator temperature coefficient than that of standard PWRs.
- Multi steam generator tubes rupture and non-isolable containment: the discharge of the Steam Generator is carried out in a dedicated pool.
- Failure of High Pressure Safety Injection: no HPSI is foreseen in SCOR.

XIX-5. Containment

The compactness of the primary circuit of SCOR leads to design of a pressure suppression containment similar to BWRs. The containment building in Figure XIX-5 consists of two physically separate compartments. The lower compartment is the reactor containment. The upper compartment is the mainly building again external hazard to protect the secondary side. The two compartments of the containment building are:

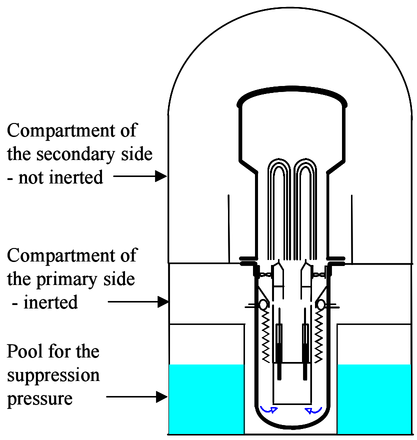

FIG. XIX-5. Containment building.

- The **compartment of the primary side** (or primary containment) is located under the reactor vessel-SG mating surface. It contains the vessel and all its primary pipe connections. Its volume is small. A pressure suppression device, as in BWR, controls the pressure. This compartment is inerted to manage the hydrogen risk.
- The **compartment of the secondary side** (or secondary containment) houses the steam generator. It is not inerted because it has no contact with the primary circuit when the vessel is closed.

XIX-6. Conclusions

SCOR is a simple compact PWR operating under low pressure. All the components are located inside the vessel, including the decay heat removal systems. The reactor has only one SG acting as the vessel head. The soluble boron free core operates with a low power density.

The configuration offers significant safety improvements over traditional loop-type PWRs (generation II) in achieving safety by design. The SCOR design eliminates the large LOCA. The decay heat removal system located very close to the core is very efficient. Calculations made with the CATHARE code have shown that, for the most penalizing accidents of the Design Basis Conditions, the core remains safely cooled with only four out of sixteen passive decay heat removal systems. For the most penalizing LOCA, a Low Pressure Safety Injection with small flow is required for a short time, one hour after the beginning of the transient.

The compactness of the reactor leads to the use of a pressure suppression containment type. It is inerted to prevent the hydrogen risk and, in case of a hypothetical core melting, the corium is cooled by pit flooding.

Technical data of SCOR

General plant data

Power plant output, gross		MW(e)
Power plant output, net	630	MW(e)
Reactor thermal output	2000	MW(t)
Power plant efficiency, net	31.5	%

Nuclear steam supply system

Number of coolant loops Compact RCS		
Primary circuit volume, including pressurizer	278	m3
Steam flow rate at nominal conditions	987	kg/s
Feedwater flow rate at nominal conditions	987	kg/s
Steam temperature/pressure	237/3.2	°C/MPa
Feedwater temperature/pressure	183/	°C/Mpa

Reactor coolant system

Primary coolant flow rate	10465	kg/s
Reactor operating pressure	8.8	MPa
Coolant inlet temperature, at core inlet	246.4	°C
Coolant outlet temperature, at riser outlet	285.4	°C
Mean temperature rise across core	39.5	°C

Reactor core

Active core height	3.66	m
Equivalent core diameter	3.04	m
Average linear heat rate	12.9	kW/m
Average core power density (volumetric)	75.3	kW/L
Rod arrays square,	17 × 17	
Number of fuel assemblies	157	
Number of fuel rods/assembly	264	
Number of control rod guide tubes	25	
Cladding tube wall thickness	0.57	mm
Outer diameter of fuel rods	9.5	mm
Active length of fuel rods	3660	mm

ANNEX XX. SMART
Korea Atomic Energy Research Institute, Republic of Korea

Integral Reactor System	Reactor Type	Power (MW•th)	Passive Safety Systems
System-Integrated Modular Advanced ReacTor (SMART) *Korea Atomic Energy Research Institute, Republic of Korea*	PWR	330	CORE/PRIMARY • Passive Residual Heat Removal System • Emergency Cool-Down Tank • Emergency Core Coolant Tank • Emergency Boron Injection Tank

XX-1. Introduction

The SMART (System-Integrated Modular Advanced ReacTor) is an advanced pressurized light water reactor that is being continuously studied at KAERI (Korea Atomic Energy Research Institute) with a rated thermal power of 330 MW. The reactor is proposed to be utilized as an energy source for sea water desalination as well as for small scale power generation. Advanced technologies such as inherent and passive safety features are incorporated in establishing the design concepts to achieve inherent safety, enhanced operational flexibility, and good economy. The SMART is designed to supply 40,000 tons of fresh water per day and 90MW of electricity to an area with an approximate population of 100,000 or an industrialized complex. In order to demonstrate the relevant technologies incorporated in the SMART design, the SMART-P (i.e. a Pilot plant of the SMART) project is currently underway at KAERI.

The prominent design feature of SMART is the adoption of integral arrangement. The major components of the NSSS such as the core, steam generators, main coolant pumps, and pressurizer are integrated into a reactor vessel without any pipe connections between those components. The schematic diagram of the SMART NSSS is shown in Figure XX-1.

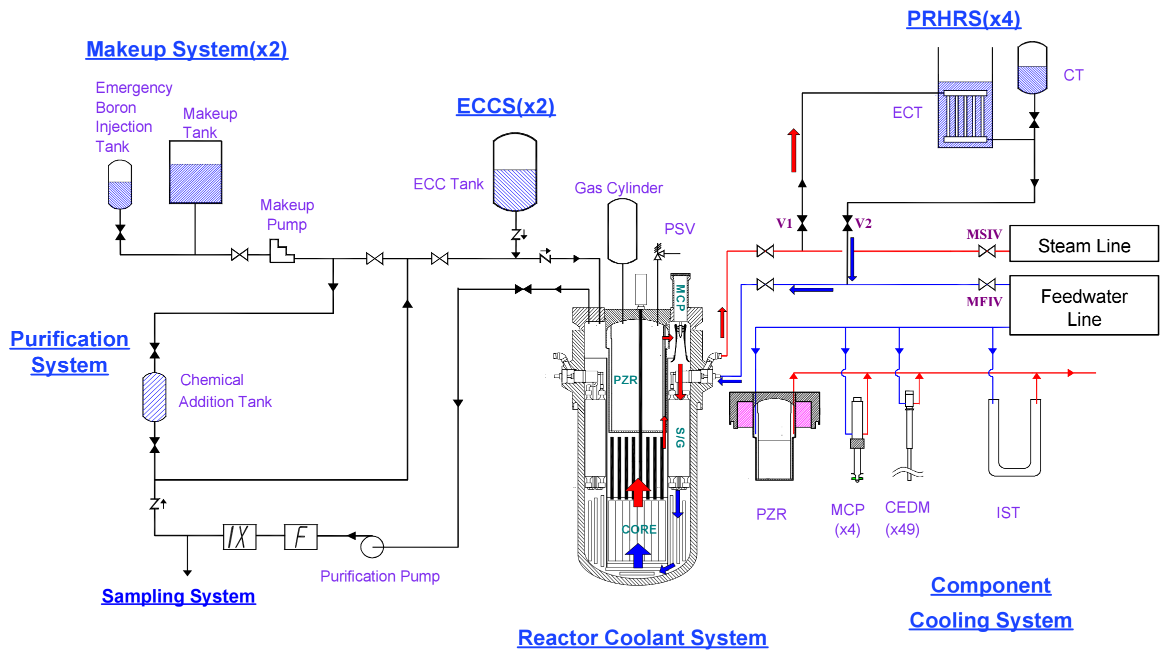

FIG. XX-1. Schematic diagram of the SMART NSSS.

The SMART core is currently being designed with the fuel design based on existing Korea Optimized Fuel Assembly (KOFA) which is in 17 × 17 rectangular rod arrays. The SMART core design is characterized by an ultra long operation cycle with a single or modified single batch reload scheme, low core power density, soluble boron-free operation, enhanced safety with a large negative Moderator Temperature Coefficient (MTC) at any time during the fuel cycle, a large thermal margin, inherently free from xenon oscillation instability, and minimum rod motion for the load follow with coolant temperature control. Due to soluble boron-free operation, an important design requirement for the SMART CEDM is a fine maneuvering capability to control the excess core reactivity. A linear step motor type CEDM is employed for easy maintenance. The minimum step length is 4mm that is short enough for the fine reactivity control. Forty-nine CEDMs are installed in the fifty-seven fuel assemblies of the SMART core.

Twelve identical SG cassettes are located in the annulus formed by the RPV and the core support barrel. Each SG cassette is a once-through type with helically coiled tubes wound around the inner shell. The primary coolant flows downward in the shell side of the SG tubes, while the secondary feedwater flows upward in the tube side. Therefore, the tubes are under compressive loads from the greater primary pressure, reducing the stress corrosion cracking and thus reducing the probability of tube rupture. The 40°C superheated steam at the exit of the helically coiled tubes eliminates the necessity of a steam separator during normal operations. The twelve SGs are divided into four sections. Each section consists of the neighboring three steam generator cassettes which are connected together with the steam and feedwater pipes. If there is a leakage in one or more of the tubes, the relevant section is isolated and SMART can be operated with reduced power until the scheduled shutdown.

The SMART adopted an in-vessel self-controlled pressurizer (PZR) located in the upper space of the RPV. The volume of the PZR is filled with water, steam, and nitrogen gas. The self-pressurizing design eliminates the active mechanisms such as spray and heater. The system pressure is determined by a sum of the steam and nitrogen partial pressures. In order to minimize the contribution of the steam partial pressure, a PZR cooler is installed for maintaining the low PZR temperature, and wet thermal insulator is installed to reduce the heat transfer from the primary coolant. The coolant temperature of the core outlet is controlled during a power maneuvering so as to minimize the system pressure variation by counterbalancing the increase of the coolant volume of the hot part with the decrease of the coolant volume of the cold part.

The SMART MCP is a canned motor type pump that eliminates the problems connected with conventional seals and associated systems. Four MCPs are installed vertically on the RPV annular cover. MCP is an integral unit consisting of a canned asynchronous 3-phase motor and an axial flow single-stage pump. The motor and pump are connected through a common shaft rotating on three radial and one axial thrust bearings. The impeller draws the coolant from above and discharges downward directly to the SG. This design minimizes the pressure loss of the flow.

There are many inherent safety features in the SMART design. Those include a large negative moderator temperature coefficient due to the boron-free operation, a low core power density, and the reduced xenon oscillations. Furthermore, enhanced safety of the SMART is accomplished with highly reliable engineered safety systems. The engineered safety systems consist of a reactor shutdown system, passive residual heat removal system, emergency core cooling system, safety vessel, reactor overpressure protection system and containment overpressure protection system. As the result of the probabilistic safety assessments for 10 internal events, the core meltdown frequency is predicted as 8.56×10^{-7}.

XX-2. Description of passive residual heat removal system

The passive residual heat removal system (PRHRS) is designed to remove the core decay heat during the accident conditions when the active systems are not available. In the case of a normal shutdown of the SMART, the residual heat is removed through the steam generators by a turbine bypass system. During accident conditions, the coolant temperature of the primary system goes down to a certain lower level due to the heat transfer through steam generators that is attained by the natural circulation flow paths established in the primary and the secondary systems of the SMART. The PRHRS consists of four independent trains with 50% of the heat removal capacity for each train. Two trains are sufficient to remove the decay heat generated in the primary system after the reactor trip. Each train is composed of an emergency cool down tank (ECT), a condensation heat exchanger, a compensating tank (CT), and several valves, pipes, and instrumentations as shown in the Figure XX-1. The condensation heat exchanger consists of inlet and outlet headers connected with several straight tubes for the heat exchange with the inner diameter of 13 mm. The compensating tank is filled with the water and pressurized nitrogen gas, which can be used to make up the losses of initial inventory in the PRHRS. The system is designed to prevent core damage for 72 hours after the postulated design basis accidents without any corrective actions by operators.

Three natural circulation circuits are involved in the operation of the PRHRS. In case of design basis events, the main steam isolation valve (MSIV) and the main feedwater isolation valve (MFIV) are closed automatically according to the reactor trip signal. After the automatic opening of the cut-off valves (V1 and V2), a natural circulation path is established between the heat exchanger in ECT and the steam generator due to the density difference of the two elevations. The ECT is located high enough relative to the steam generator in order to retain the heat removal capability during the events by supplying sufficient driving forces to the natural circulation flow. In the primary system, after the RCP trip, a natural circulation path is established between the reactor core and the steam generators. The decay heat generated in the reactor core is transported to the steam generators by the natural circulation flow. The third natural circulation path is established around the heat exchanger inside the ECT. The heat carried by the natural circulation flow in the primary and secondary systems is transferred to the ultimate heat sink through the natural convection at the vicinity of the heat exchanger.

XX-3. Conclusions

The PRHRS provides an ultimate heat sink when the off-site power is not available during the design basis events. The reliability of the PRHRS is being examined at KAERI through a high temperature and high pressure thermal-hydraulic test facility, named VISTA (experimental Verification by Integral Simulation of Transients and Accidents). The VISTA is an integral test facility simulating the primary and secondary systems as well as the major safety-related systems of the SMART-P. The scale ratios of the VISTA relative to the PRHRS of the SMART-P are 1/1 by the height and 1/96 by the volume. The primary system of the VISTA consists of the reactor vessel with electrical heaters, the main coolant pump, the pressurizer, and the helical coil steam generator. They are connected with pipes for easy installation of the instrumentation and simple maintenance. The secondary system is designed to remove the primary heat source by employing a single train of the PRHRS. Preliminary investigations have been conducted on the natural circulation performance of the PRHRS and the primary system as well as the heat transfer characteristics of the heat exchanger in the ECT, by employing the VISTA facility.

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