IAEA TECDOC SERIES

IAEA-TECDOC-1785

Design Safety Considerations for Water Cooled Small Modular Reactors Incorporating Lessons Learned from the Fukushima Daiichi Accident

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DESIGN SAFETY CONSIDERATIONS FOR WATER COOLED SMALL MODULAR REACTORS INCORPORATING LESSONS LEARNED FROM THE FUKUSHIMA DAIICHI ACCIDENT

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DESIGN SAFETY CONSIDERATIONS FOR WATER COOLED SMALL MODULAR REACTORS INCORPORATING LESSONS LEARNED FROM THE FUKUSHIMA DAIICHI ACCIDENT

INTERNATIONAL ATOMIC ENERGY AGENCY
VIENNA, 2016

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Printed by the IAEA in Austria
March 2016

IAEA Library Cataloguing in Publication Data

Names: International Atomic Energy Agency.
Title: Design safety considerations for water cooled small modular reactors incorporating
lessons learned from the Fukushima Daiichi accident / International Atomic Energy
Agency.
Description: Vienna : International Atomic Energy Agency, 2016. | Series: IAEA TECDOC
series, ISSN 1011–4289 ; no. 1785 | Includes bibliographical references.
Identifiers: IAEAL 16-01027 | ISBN 978–92–0–100716–2 (paperback : alk. paper)
Subjects: LCSH: Nuclear reactors — Safety measures. | Nuclear power plants —
Accidents — Prevention. | Nuclear reactors — Design and construction. | Water cooled
reactors.

FOREWORD

The global future deployment of advanced nuclear reactors for electricity generation depends primarily on the ability of nuclear industries, utilities and regulatory authorities to further enhance their reliability and economic competitiveness while satisfying stringent safety requirements. The IAEA has a project to help coordinate Member State efforts in the development and deployment of small and medium sized or small modular reactor (SMR) technology. This project aims simultaneously to facilitate SMR technology developers and potential SMR users, particularly States embarking on a nuclear power programme, in identifying key enabling technologies and enhancing capacity building by resolving issues relevant to deployment, including nuclear reactor safety.

The objective of this publication is to explore common practices for Member States, which will be an essential resource for future development and deployment of SMR technology. The accident at the Fukushima Daiichi nuclear power plant was caused by an unprecedented combination of natural events: a strong earthquake, beyond the design basis, followed by a series of tsunamis of heights exceeding the design basis tsunami considered in the flood analysis for the site. Consequently, all the operating nuclear power plants and advanced reactors under development, including SMRs, have been incorporating lessons learned from the accident to assure and enhance the performance of the engineered safety features in coping with such external events.

In response to the Fukushima Daiichi accident, the IAEA established an Action Plan on Nuclear Safety. The preparation of this publication was carried out within the framework of the IAEA Action Plan on effectively utilizing research and development. The main objective of this publication is to present technology developers and users with common considerations, approaches and measures for enhancing the defence in depth and operability of water cooled SMR design concepts to cope with extreme natural hazards. Indicative requirements to prevent such an accident from recurring are also provided for States planning to adopt water cooled SMR designs and technologies.

The IAEA gratefully acknowledges the information on technology and safety aspects provided by SMR design organizations and information regarding technical requirements provided by several Member States. The IAEA officers responsible for this publication were M.H. Subki of the Division of Nuclear Power and M. Kim of the Division of Nuclear Installation Safety.

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1. INTRODUCTION

1.1. BACKGROUND

In the past few years, the Agency saw a substantial increase in the participation of Member States in its programme for the development of small and medium-sized or modular reactors (SMRs) technology. The current driving efforts in the development of such reactors include: fulfilling the need for flexible power generation for a wider range of users and applications; replacing the ageing fossil fuel-fired power plants; enhancing safety performance through inherent and passive safety features; offering better economic affordability; suitability for non-electric applications; options for remote areas; and synergetic energy systems that combine nuclear and renewable energy sources. For the context of this report, SMRs stand for small modular reactors and are defined in general as advanced nuclear reactors that produce equivalent electric power of up to 300 MW(e) and are designed to be built in factories and transportable to utilities for installation as demand arises. In this report, the focus is on water cooled SMR designs that are under development. Some of the designs are to be deployed as multi-module power plants. For water cooled SMRs, modularity is achieved by integrating major components of the reactor coolant system inside the reactor pressure vessel – in the same compartment with the reactor core and internals. Several countries are also pioneering in the development of transportable nuclear power plant (TNPP), including floating and marine-based SMRs.

To date, three reactors in the SMR category are under construction, i.e. in Argentina (CAREM25, an industrial prototype integral PWR), in the Russian Federation (KLT-40S, a barge mounted floating power unit) and in China (HTR-PM, an industrial demonstration plant of high temperature pebble bed gas cooled reactor). Dozens of advanced SMR designs are under development for near term deployment including in the United States of America (B&W mPower and NuScale's SMR design, both received government funding for design certification, as well as the Westinghouse SMR and the SMR-160). The System-integrated Modular Advanced ReacTor (SMART) from the Republic of Korea and CAREM25 from Argentina are the water cooled SMR designs that have obtained design approval from the respective governments. China has been developing the ACP100 design for potential near term deployment.

The accident at the Fukushima Daiichi nuclear power plant (NPP) on 11 March 2011 in Japan reveals the need for the nuclear community to prepare for unexpected circumstances that go beyond the design basis events. No matter how well plants are operated and maintained, there is always the potential for unexpected and high consequence situations. The Fukushima Daiichi accident imparted many valuable lessons on both technical and economic impacts in utilizing nuclear energy. The accident has disclosed various existing design weaknesses and vulnerabilities, especially when combination of unprecedented natural phenomena occurs. Actions taken by the operators and the emergency response team during the early phase of the accident showed that the weaknesses were not only in the hardware and the design of the reactor but also due to limitations of the human capability, accident management and emergency operating procedures, emergency infrastructures and regulatory framework.

Realizing that other unprecedented site/region-specific events could disrupt reactor operation in the same scale or more than what happened in the Fukushima Daiichi accident, not necessarily a large tsunami, nuclear community needs to take lessons from the accident and transform them into appropriate design enhancements, actions, and other

countermeasures in water cooled reactors, both those in operations and near term deployable designs.

The engineered safety features (ESFs) of the Fukushima Daiichi NPP were not damaged by the earthquake but the water and debris of the tsunamis crippled and disabled them. The disability of the ESF resulted in extensive damage to the nuclear power plant and released radioactivity to the environment. The accident prompted the nuclear industry to revisit the safety principles with further strengthening and additional assumptions that ESFs, otherwise considered robust and failsafe, are in fact vulnerable in some natural events or their never before assumed combination.

Based on the current trend, more attention should be given to water cooled SMR designs as this type of reactor is being considered as options to fulfill future energy demands due to their technological features to suit specific applications and deployments. The development of SMRs comes with several different concepts, coolants, neutron spectrum, deployment location and applications. Some are already in construction stage while others are in licensing process or early design phases. Each SMR employs particular design approach with its specific enabling technologies. Recently, the development trend has been towards modular integral PWR type – where all the major components or the reactor coolant systems, such as steam generators and pressurizer are contained inside the reactor pressure vessel. These reactors also adopt advanced features such as passive safety systems, multi module configuration, smaller emergency planning zone, underground and marine based deployment, etc. The designs and performance of these advanced features in anticipating and coping with the Fukushima Daiichi type accident should be reviewed and well understood. Therefore, a comprehensive review of the lessons learned from such an accident and identifying appropriate and practical countermeasures for SMR design will be timely and beneficial.

This TECDOC presents and discusses design safety considerations on appropriate and practical countermeasures to incorporate and address the lessons learned from the Fukushima Daiichi accident to enhance the design of engineered safety systems of water cooled SMRs currently under development.

This publication is a contribution from the IAEA Division of Nuclear Power in collaboration with the Division of Nuclear Installation Safety for the IAEA Action Plan on Nuclear Safety Item–12 on Utilization Effective Research and Development (R&D) [2]. This publication was derived from the result of extensive dialogues involving about thirty (30) experts from eleven (11) Member States and an international organization convened through three (3) consultancy meetings. This approach helped ensure a comprehensive representation of technical knowledge and experience. In these meetings, experts from Member States compiled and integrated the lessons learned that have been previously identified by the fact finding teams from several organizations/institutes including technology developers and IAEA in-house experts. They particularly discussed design safety considerations and options to enhance the performance of the ESF of water cooled SMRs incorporating the lessons learned from the Fukushima Daiichi accident. This publication is intended to be a preliminary compendium of general design safety considerations to enhance the performance of the ESFs of water cooled SMRs in coping with unprecedented external events.

1.2. OBJECTIVES

The objectives of this publication are:

- To present technical lessons learned from sequence of events of the Fukushima Daiichi accident;
- To review the engineering designs and performance of the engineered safety features of water cooled small modular reactors in dealing with the design basis and severe accidents;
- To provide technical considerations for appropriate and practical countermeasures to address the lessons learned from the Fukushima Daiichi accident to improve the design of engineered safety systems of small modular reactors;
- To provide indicative requirements for embarking countries planning to deploy small modular reactors and advanced water cooled reactors to prevent Fukushima Daiichi type accident;
- To provide technology developers and users with considerations to enhance the performance of the engineered safety feature of water cooled small modular reactors

1.3. SCOPE

The publication consists of sections that will cover three areas concerning considerations to enhance the performance of ESFs in water cooled SMRs incorporating lessons learned from the Fukushima Daiichi accident. At first, a brief description of the Fukushima Daiichi accident with emphasis on the sequence of accident progression occurring in Units 1, 2 and 3 and some important facts leading to the lessons learned will be provided. Next, various ESFs employed in the existing design of water cooled SMRs and advanced water cooled reactors will be discussed. Here the current technologies used by existing reactor designs in dealing with the design basis accidents will be reviewed. The main feature of this publication is the discussion of lessons learned from the Fukushima Daiichi accident and the recommendation of practical countermeasures for water cooled SMR designs to cope with such an extreme external event. The recommended countermeasures are organized in tabular format, where a defence in depth level is used as a pointer to clarify the corresponding issues being addressed.

This report is organized as follows:

Section 1 includes background, objectives and scope of the publication.

Section 2 briefly describes the Fukushima Daiichi accident progression.

Section 3 discusses various designs of the engineered safety features of advanced reactors and SMRs which includes diverse trip system, residual heat removal system, safety injection system, containment system, and severe accident mitigation features.

Section 4 elaborates the recommended countermeasures to address the lessons learned from the Fukushima Daiichi accident in the design of water cooled SMRs.

Section 5 summarizes and highlights the recommended countermeasures.

ANNEXES I and II provide brief descriptions and parameters of the ESFs for each specific SMRs and advanced water cooled reactors under review.

1.4. APPROACH TO THE PREPARATION OF THIS PUBLICATION

The basis for the development of this publication is international experts' discussion result on the lessons learned from the Fukushima Daiichi accident which was produced in the following three (3) Consultancy Meetings:

- Consultancy Meeting on 'Incorporating Lessons Learned from the Fukushima Accident in SMR Technology Assessment for Design of Engineered Safety Systems' held at the IAEA Headquarters on 30 May – 01 June 2012.
- Consultancy Meeting on 'Preparation of Toolkit for SMR Technology Assessment on the Reliability of Engineered Safety Features' held at the IAEA Headquarters on 11 – 13 September 2012.
- Consultancy Meeting on 'Finalizing the TECDOC on Considerations to Enhance the Performance of Engineered Safety Features of Small Modular Reactors in Coping with Extreme External Events' held at the IAEA Headquarters on 2 – 5 March 2015.

In these meetings, experts from Member States compiled and integrated the lessons learned that have been previously identified, collected and published by the fact finding teams of several organizations/institutes including reactor designers. The experts discussed and produced integrated lessons learned and provided technical considerations and countermeasure options on how to enhance the performance of ESFs of water cooled SMRs [3]. The process diagram of the development of this publication is given in *FIG. 1*.

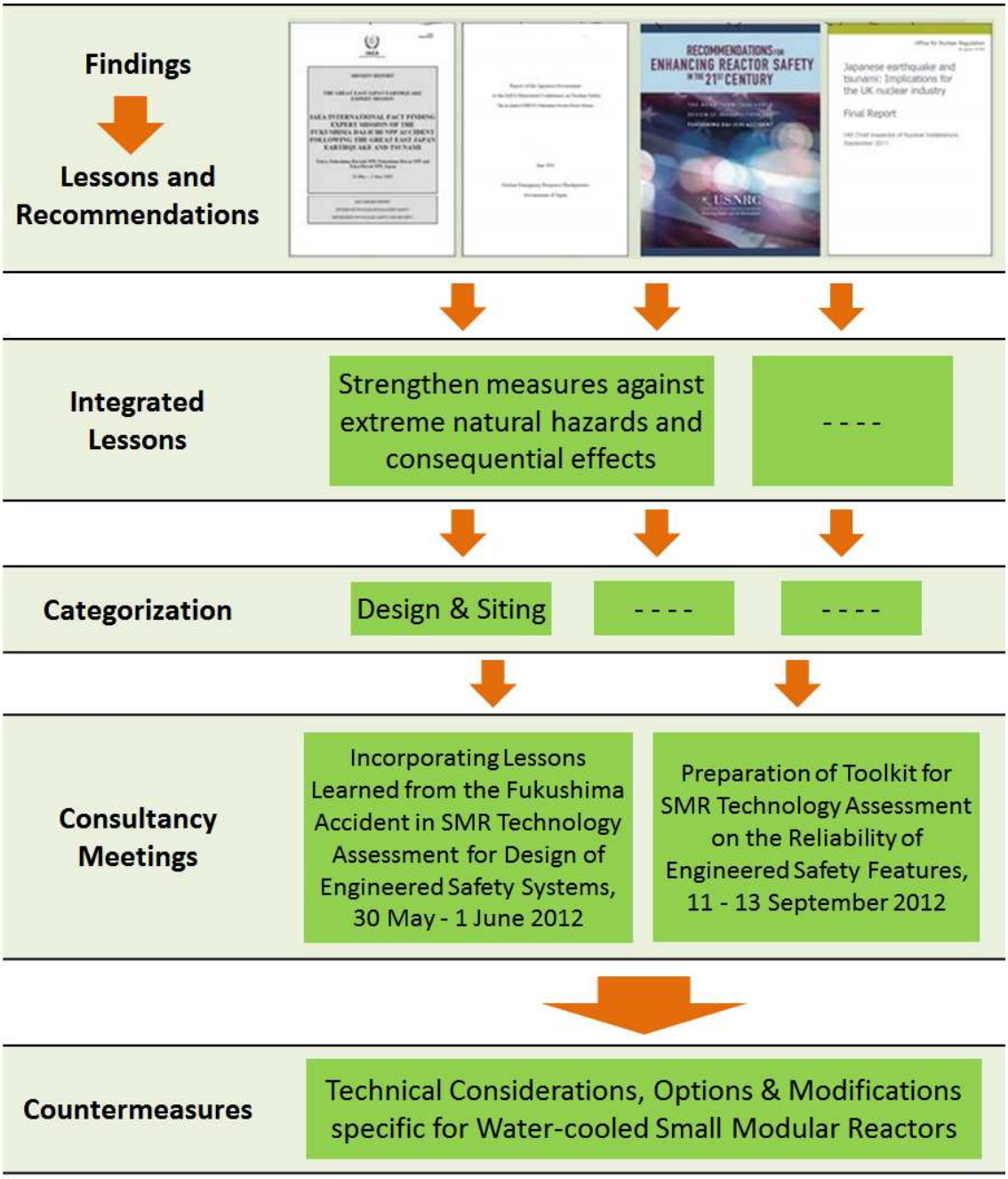

FIG. 1. Process diagram of document development.

2. OVERVIEW OF THE FUKUSHIMA DAIICHI ACCIDENT

The Fukushima Daiichi nuclear power plant (NPP) consists of six units of boiling water reactors (BWRs) which were commissioned between 1971 and 1979, with power ratings from 460 to 1100 MW(e). Unit 1 was a BWR/3 design with MARK I containment, Units 2–5 were BWR/4 designs with MARK I containment and Unit 6 was a BWR/5 design with MARK II containment, which was the first unit in Japan with a capacity of 1100 MW(e). A typical schematic diagram of BWR is shown in *FIG. 2* [1]. 28

On March 11, 2011, at 14:46 (Japan Standard Time) a great earthquake of 9.0 Richter scale shook the northeast coast of Japan and about one hour later two tsunami waves smashed the Fukushima Daiichi NPP which was operated by Tokyo Electric Power Company (TEPCO). About 10 minutes after the first wave, the second and largest wave, with a run up height of 14–15 m, overwhelmed the seawalls and inundated the site. It engulfed all structures and equipment located at the seafront, as well as the main buildings (including the reactor, turbine and service buildings) at higher elevations. The wave flooded and damaged the unhoused seawater pumps and motors at the seawater intake locations on the shoreline. This meant that essential plant systems and components, including the water cooled Emergency Diesel Generators (EDGs) could not be cooled to ensure their continuous operation. Water entered and flooded buildings, including all the reactor and turbine buildings, the common spent fuel storage building and diesel generator building. The water damaged the buildings and the electrical and mechanical equipment inside at ground level and on the lower floors. The damaged equipment included the EDGs or their associated power connections, which resulted in the loss of emergency AC power. Only one of the air cooled EDGs – that of Unit 6 – was unaffected by the flooding. It remained in operation, continuing to supply emergency AC power to the Unit 6 safety systems and allowing cooling of the reactor [1].

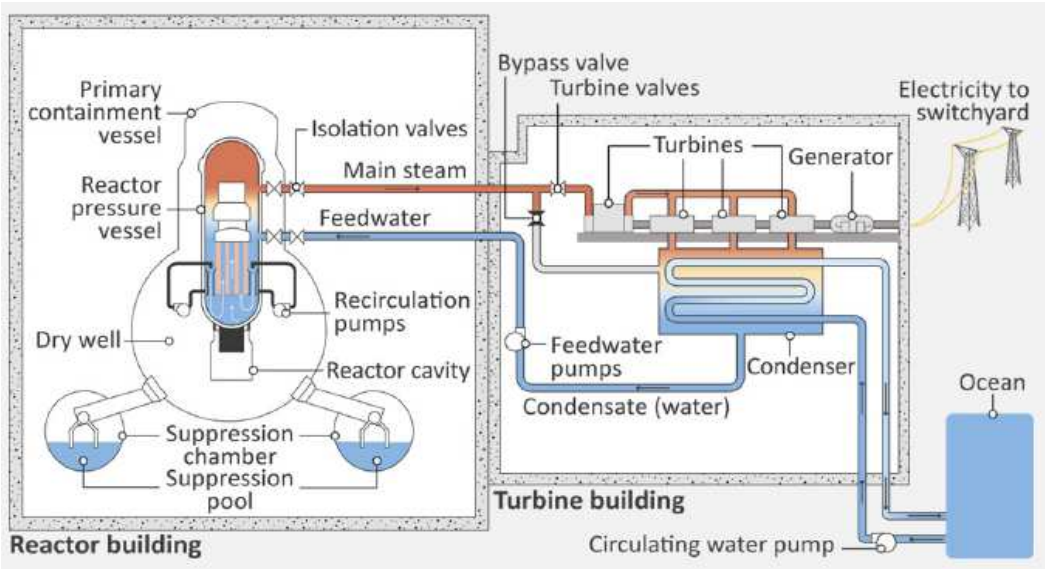

FIG. 2. Typical BWR with MARK I containment [1].

The earthquake and the tsunami impacted on multiple units at the Fukushima Daiichi NPP. The tsunami also caused widespread destruction of buildings, doors, tanks, water intakes, roads and other site infrastructures which lead to the loss of emergency core cooling capability and eventually loss of the ultimate heat sink from the sea. Almost all power

sources and supporting systems and equipments expected to be activated in case of accidents became inoperable. As a result of these events, Units 1–5 lost all AC power, a situation referred to as a station blackout. The entire site was in a blackout situation after the tsunami. The units at the Fukushima Daiichi NPP, similar to other plants of the same age, were designed to withstand a station blackout for eight hours, based on the capacity of the DC batteries in the reactor units. The units responded to the initiating event – the earthquake and the concurrent loss of off-site power – as intended by the designers and as stipulated in the operating procedures, except for some operator actions that were restricted or delayed by the aftershocks [1].

When the earthquake occurred, Units 1, 2 and 3 were in operation at their rated power and Units 4, 5 and 6 were in refueling outage. At Unit 4, all the fuels were stored in the spent fuel pool (SFP) for the core shroud replacement work. The outage for Units 5 and 6 was nearly complete and the fuels were already loaded into the reactor pressure vessels (RPV). A few seconds after the earthquake, all the three operating units shutdown automatically by the insertion of control blades. Turbine generators were also tripped and main steam isolation valves (MSIVs) were closed. The earthquake had no significant impact on plant's structures, but it damaged the electrical grid infrastructures and interrupted electric supply lines to the site, and the tsunami caused substantial destruction of the operational and safety infrastructure on the site. The combined effect led to the loss of off-site and on-site electrical power for all the six units which resulted in the loss of offsite power event, and consequently the loss of the cooling function at the three operating reactor units as well as at the spent fuel pools.

Units 1–3 were automatically isolated from their turbine systems due to the power interruption, resulting in increases in the temperature and pressure of the reactors due to the decay heat. The cooling of these reactors following the isolation was accomplished by means of the following design and operational provisions:

- In Unit 1, as the reactor pressure increased, both loops of the IC system started automatically and continued to cool the reactor. The operation of both ICs loops lowered the reactor pressure and temperature so rapidly that the operators manually stopped them, in accordance with procedures, in order to prevent thermal stress on the RPV. Afterwards, only one of the loops was used by the operators to control the cooling rate in a range prescribed by the procedures.
- In Units 2 and 3, the increase in reactor pressure automatically activated safety relief valves, which were designed to protect the reactor from over pressurization by releasing steam from the reactor vessel to the suppression pool section of the primary containment vessel. This resulted in a decrease in the reactor water levels. The operators manually activated the reactor core isolation cooling (RCIC) system in accordance with procedures [1].

Following the loss of offsite power, EDGs provided essential power to all emergency systems as designed for about 50 minutes until the big tsunami hit the Fukushima Daiichi complex, flooding electrical switchboards, battery room and crippling EDGs. Consequently, DC power was gradually lost in Units 1, 2 and 4 during the first 10–15 minutes of the flooding, making it difficult to cope with the station blackout. The elevations and locations of structures and components at the Fukushima Daiichi NPP with reference to the Onahama Port is shown in *FIG. 3* [1].

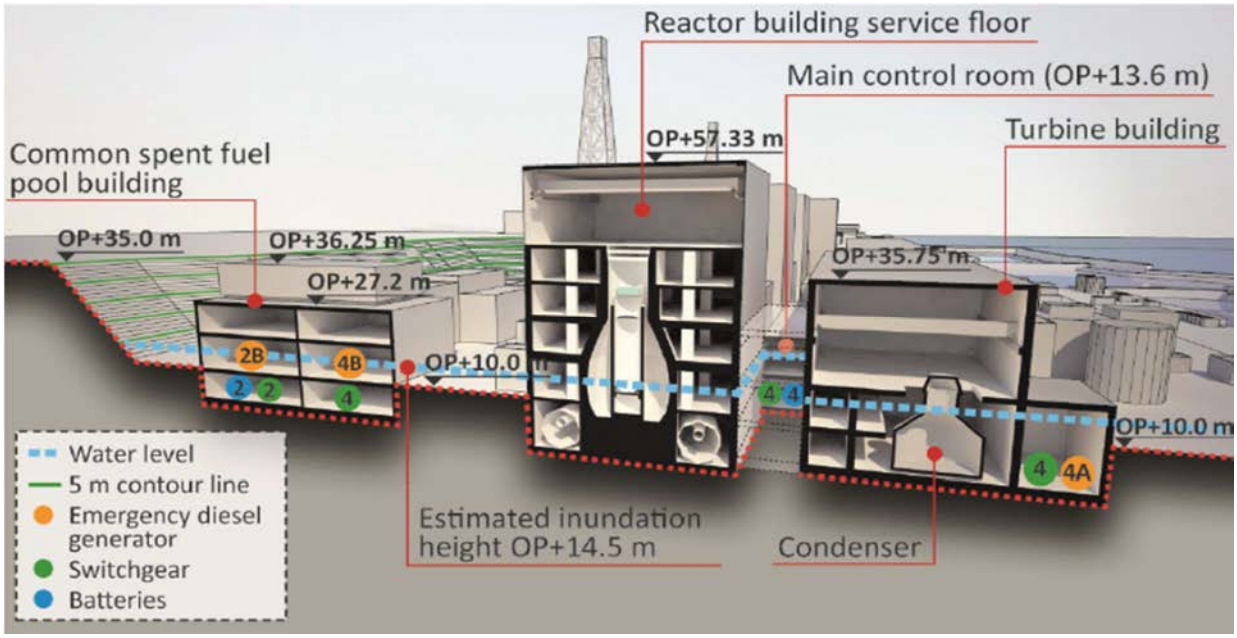

FIG. 3. The elevations and locations of structures and components at the Fukushima Daiichi NPP with reference to the Onahama Port [1].

When the reactors became isolated due to the closure of MSIVs, core cooling for Unit 1 was provided by the activation of IC which was under operator's control before the tsunami stroke. The schematic diagram of IC system is given in *FIG. 4* [1]. However, loss of DC power because of flooding in the battery room led to loss of indications for the reactor water level, and made water level in the core unknown. The operators were not certain whether or not the IC system properly functioned, so later on the operator decided to terminate the IC system operation. As a result, there was no cooling mechanism to remove decay heat from the reactor. In short, the fundamental safety function of core cooling at Unit 1 was lost when the IC was stopped by the operators just before the tsunami, and the Unit 1 core heated up from that time. The pressure in the containment drywell rose rapidly as suppression pool became hotter. The high temperature and high pressure level in the containment for an extended time damaged the electrical penetrations and door/hatch seal which later led the leakage of hydrogen into the reactor building.

The volume of Mark I containment vessel, unlike Mark II or PWR, is small for its reactor output and therefore the density of hydrogen may reach high enough under poor venting condition to detonate in a short time once hydrogen begins to discharge during severe accident condition. Nearly 24 hours after the station blackout, seawater injection and AC power supply were connected to Unit 1. However, within minutes of connection, an explosion in the Unit 1 reactor building damaged both of these arrangements before they could be put in use.

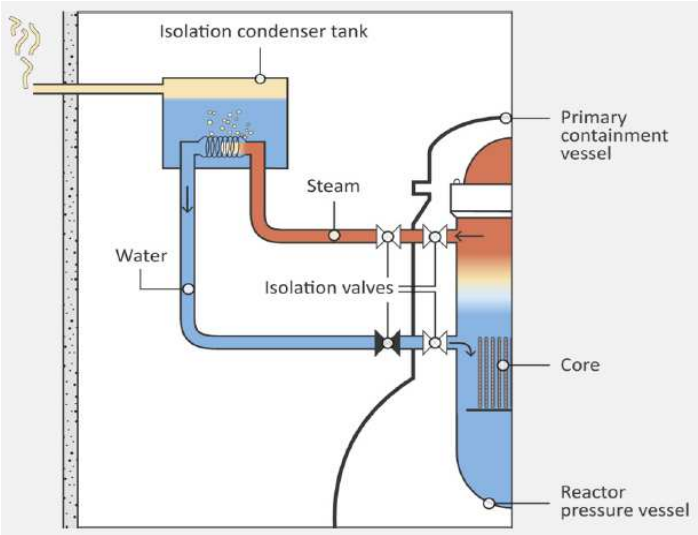

FIG. 4. Isolation Condenser system [1].

Unit 2 had a different design for removing residual heat from the reactor core. The RCIC system use steam from the RPV to drive a turbine which pumped water into the reactor vessel. The schematic diagram of a RCIC system is provided in *FIG. 5* [1]. The core cooling after reactor trip and MSIV closure was provided by the RCIC system. The operation of RCIC was manually started by operator a few minutes after the earthquake and the system would automatically trip if a high reactor water level was achieved. When tsunami reached the nuclear complex the DC distribution system was submerged in water and lost its function. As a result all indications of important parameters gradually disappeared and the high pressure coolant injection (HPCI) system which requires DC power to operate also became unavailable. Consequently, operators were not sure if the RCIC was operating because the indicator light had gone out. To confirm, a small team was dispatched to inspect the system locally and it concluded that the RCIC was operating and the reactor water level can be maintained.

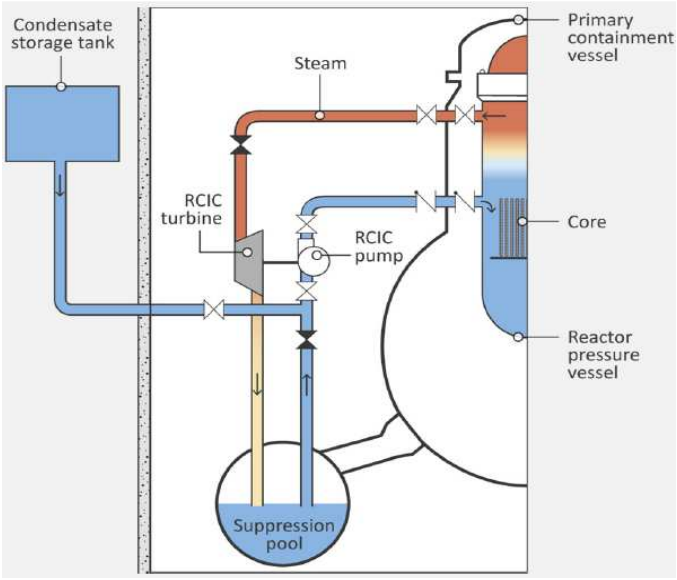

FIG. 5. Reactor core isolation cooling system [1].

There are indications that, after about 68 hours, the RCIC system failed. It was therefore no longer possible to inject water into the RPV because it was at high pressure. It is estimated that the Unit 2 reactor core began to melt about 76 hours after the tsunami. The accident progression at the Fukushima Daiichi nuclear power plant is briefly depicted in *FIG. 6*.

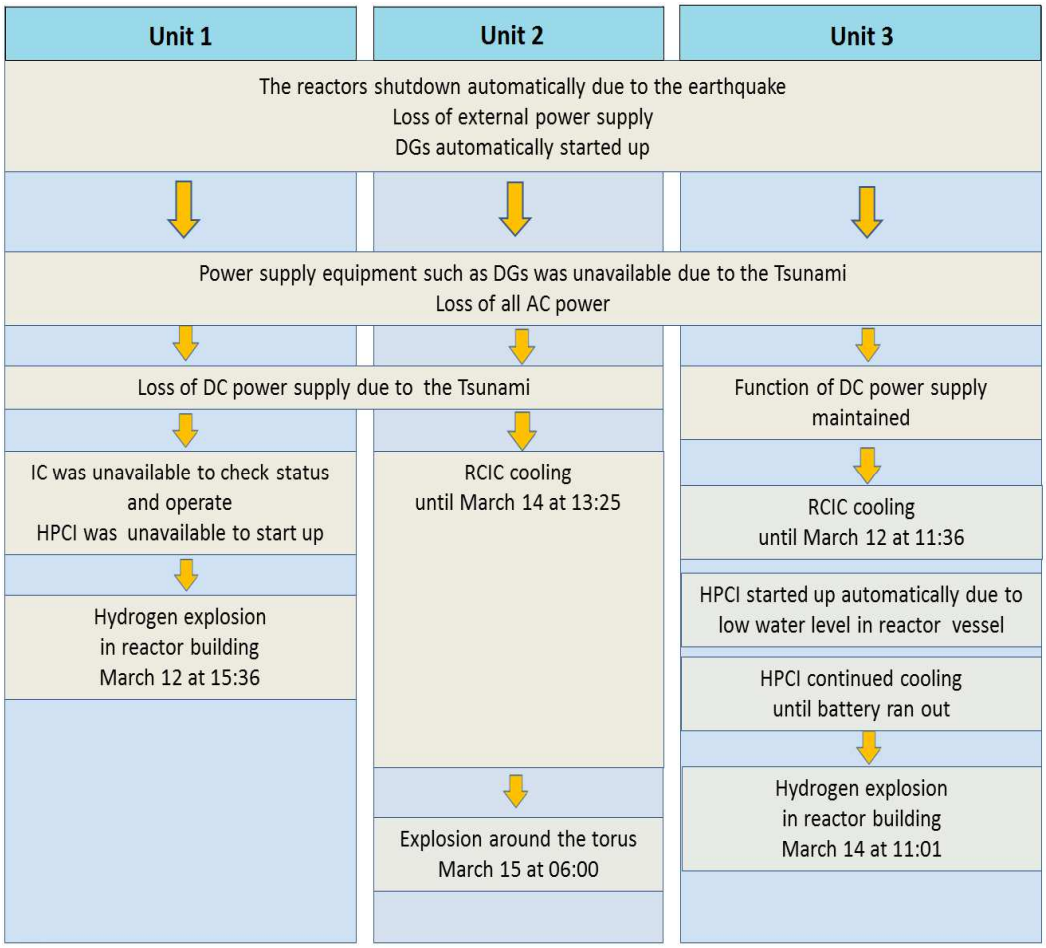

FIG. 6. The accident progressions at the Fukushima Daiichi nuclear plant complex.

Situation in Unit 3 was similar to that in Unit 2 except that some DC power was still functioning after the tsunami. At the beginning the core cooling was provided by RCIC which was manually actuated by operator about 20 minutes after the reactor scram. This RCIC provided injection to the core until it automatically stopped when a high reactor water level was reached. The tsunami then caused the loss of some DC power systems but fortunately the indications for pressure and reactor water level were not affected. An 'on and off' operation for RCIC was used by operator to maintain the water level at about 4 meters above the top of active fuel (TAF). Pressure inside the reactor vessel was controlled by safety relief valves (SRVs). However the condition did not stay long. After 14 hours of continued operation of the emergency HPCI system, the Unit 3 operators became concerned about the reliability and possible failure of the system's turbine powering the injection pump, which was by then operating at low reactor steam pressure. The concern was related to the possibility of turbine damage and the creation of a release path from the reactor vessel. This would result in an uncontrollable release of radioactive steam directly outside the primary containment. This concern was heightened when the turbine did not

automatically stop, as it was designed to, when the reactor pressure decreased below the automatic shutoff pressure. Consequently, the operators decided to stop the HPCI system and instead use the alternative means of injection at low pressure (the diesel driven fire pump). The operators thought this could be achieved without interruption of core cooling, since the reactor pressure was already below that of the diesel driven fire pump and could be kept low by the use of pressure relief valves. The Unit 3 emergency high pressure core injection system was therefore turned off by the operators, who then started their attempts to open the pressure relief valves. However, all attempts to open the pressure relief valves failed, and reactor pressure quickly increased above the level at which the diesel driven fire pump could inject, stopping the cooling of the Unit 3 core about 35 hours after the station blackout. Faced with this setback, the operators tried to return to injection via the emergency HPCI system but were unsuccessful. On 14 March, an explosion occurred in the upper part of the Unit 3 reactor building, destroying the structure above the service floor and injuring workers. In addition to the destruction of the alternative water injection arrangement, the capability to vent the containment in Unit 2 was also lost as a result of the explosion, which affected the previously set up Unit 2 containment venting path. After the explosion, the isolation valve on the Unit 2 vent line was discovered to be closed and could not be reopened [1].

Following the loss of IC, RCIC and HPCI systems in Units 1, 2 and 3, the plant workers prepared to inject water into reactor vessels via fire hoses for emergency cooling. However, situation on the site was far beyond originally estimated conditions in the accident management. Injections of water to the reactor vessels were delayed by on-site difficulties. The reactor and containment pressure was very high so that depressurization and venting were needed before the injection could be performed. Meanwhile some segments of fuel rods were already uncovered, overheated and damaged, which resulted in exothermic reaction between steam and zirconium thus producing hydrogen gas. Despite the efforts of the operators to maintain control, the reactor cores in Units 1–3 overheated, the nuclear fuel melted and the three containment vessels were breached. The high containment pressure caused leakage of the hydrogen and other volatile radionuclides to the reactor building structure. Hydrogen was released from the reactor pressure vessels, leading to explosions inside the reactor buildings in Units 1, 3 and 4 that damaged structures and equipment and injured personnel. Radionuclides were released from the plant to the atmosphere and were deposited on land and on the ocean. There were also direct releases into the sea. Another explosion occurred in Unit 2 around its suppression pool due to the containment overpressure as venting could not be performed in this unit. Units 5 & 6 survived the accident due to the air-cooled EDG of Unit 6.

The Fukushima Daiichi accident demonstrated that extreme natural hazards have the potential to invalidate or impair multiple levels of defence in depth (DiD). The design of the Fukushima Daiichi nuclear power plant provided equipment and systems for the first three levels of DiD: (1) equipment intended to provide reliable normal operation; (2) equipment intended to return the plant to a safe state after an abnormal event; and (3) safety systems intended to manage accident conditions. The design bases were derived using a range of postulated hazards; however, external hazards such as tsunamis were not fully addressed. Consequently, the flooding resulting from the tsunami simultaneously challenged the first three protective levels of DiD, resulting in common cause failures of equipment and systems at each of the three levels. The failure to provide sufficient means of protection at each level of DiD resulted in severe reactor damage in Units 1, 2 and 3 and in significant radioactive releases from these units. A systematic identification and assessment of external hazards and robust protection against these hazards needs therefore

to be considered for all levels of DiD. Furthermore, the accident showed that alternative design provisions and accident management capabilities could still ensure the supply of cooling water to the reactor even if all prime safety systems designed to protect the reactor against accidents were lost [1].

Stress tests were carried out in many IAEA member states with operable NPPs to reassess the design of NPPs against site specific extreme natural hazards, installing additional backup sources of electrical power and supplies of water, and strengthening the protection of plants against extreme external events. There is widespread recognition that everything humanly possible must be done to ensure that no such accident ever happens again. IAEA safety standards embody an international consensus on what constitutes a high level of safety. They were reviewed after the accident by the Commission on Safety Standards. Worldwide operating experience has shown instances where natural hazards have exceeded the design basis for a NPP [1].

3. REVIEW OF ENGINEERED SAFETY FEATURE DESIGN OF SMALL MODULAR REACTORS AND ADVANCED REACTORS

Engineered safety feature (ESF) of NPP is a set of means to protect the public from radioactive fission products in the event of accidents. Its primary functions are to localize, control, mitigate and terminate the consequences of postulated accidents and maintain radiation exposure levels below allowable limits. Various designs and concepts of ESFs are used in different reactors and there are similarities among them. The variations mainly come from the type and safety characteristic of the reactor, power size, availability of passive system and approaches on how to address the accidents. The following is a discussion on the ESF designs of integral PWR type SMRs and advanced large reactors. In general, the ESF consists of several functional systems, i.e. trip system, residual heat removal system, safety injection system, and containment system. Advanced water cooled reactor designs have also added severe accident mitigation features to deal with beyond design basis accidents. With regard to the lesson learned from the Fukushima Daiichi accident, the discussion is focused on water cooled reactor technology. Several water cooled SMRs and advanced reactors are reviewed including integral PWR type SMRs, large PWRs and BWRs. The intention is to gather insights on the capabilities of the engineered safety system design to see if any improvements are needed based on the lessons learned from the Fukushima Daiichi accident.

3.1. TRIP AND SAFETY SHUTDOWN SYSTEMS

Generally, water cooled reactors use control rods as the main reactivity control system to shut down the reactor under normal and emergency conditions. Rods made from neutron absorbing materials are inserted in reactor core using control rod drive mechanism (CRDM) or gravity to cease the nuclear chain reaction. Most of CRDMs in existing designs are located outside the pressure vessel, either on the upperside (as in PWR) or below the vessel (as in BWR), using welded penetrations in the vessel. Their postulated worst case failure causes a single absorber control rod cluster/blade ejection accident, resulting in a small LOCA as well as instantaneous insertion of positive reactivity. A new design is introduced recently for integral PWR type SMRs where the CRDM is placed inside the vessel to eliminate these penetration and the consequences of their failure. This technique inherently removes possibility of rod ejection accident and the consequent LOCA as penetrations in the reactor vessel closure head are eliminated. SMR designs which implement in-vessel CRDM technology, among others, are CAREM25, IRIS, mPower and Westinghouse SMR.

In addition water cooled reactors have a diverse alternate mechanism to ensure fission termination, if the CRDM fails. The mechanism is mostly injection of dissolved boron into primary system by active or passive driving force. SMR designs which implement passive method for their secondary shutdown systems, among others are CAREM25, IRIS and Westinghouse SMR. One of the designs that uses active injection system is SMART.

3.2. RESIDUAL HEAT REMOVAL SYSTEM

3.2.1. Residual heat removal through steam generator and heat exchanger submerged in water pool

Some SMR designs passively remove decay heat through pairing the steam generators (SGs) with heat exchangers (HXs) immersed in a water pool as shown in *FIG. 7*. Steam

produced by decay heat in the SG is routed to the heat exchangers where it is condensed. The condensate flows back to the SG through SG feed water inlet. SMR designs which implement this method, among others, are the passive emergency heat removal system (EHRS) of IRIS, the passive residual heat removal (PRHR) system of SMART and the decay heat removal system (DHRS) of NuScale.

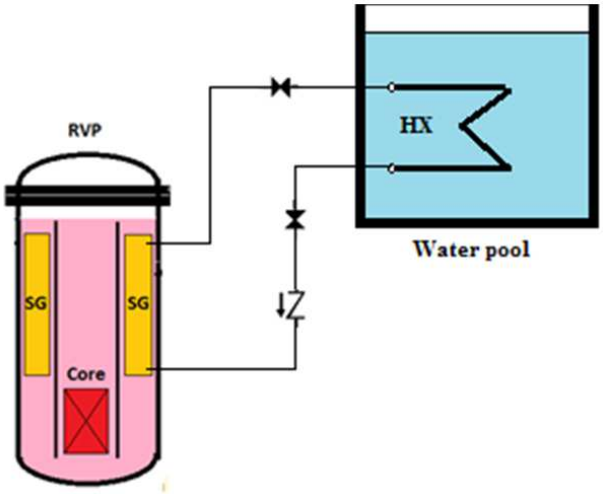

FIG. 7. Emergency heat removal system through steam generator.

3.2.2. Residual heat removal using passively cooled condenser

If a loss of heat sink condition occurs, the continuous reactor core decay heat produces steam after some time. As a result the system pressure increases. The core needs to be cooled down by removing the decay heat. Some SMR and new large NPP designs use passive condensers which are immersed in water pool and connected to the upper dome of the vessel, thus forming a natural circulation loop to cool down the primary system, as shown in *FIG. 8*. When the valves open, the steam goes to the condenser tubes transferring the heat to the water in the pool. As the steam condenses the water in the tubes it returns to the vessel by gravity effect.

The SMR designs which implement passive condenser approach for its residual heat removal system (RHRS) include CAREM25 and NuScale. The same principle is also implemented for the IC of BWR plant.

Passively cooled condenser is also used by large PWR reactor (such as PRHR of AP1000). The hot leg and cold leg of the reactor are connected to the heat exchanger inlet and outlet, respectively. The condenser is submerged in water pool which is located above the reactor vessel to establish a natural circulation path.

3.2.3. Residual heat removal using pump and heat exchanger

Conventional approach for residual heat removal system is to use active means usually consisting of pumps, valves, HXs and related piping. This approach is used by many existing advanced light-water cooled reactors.

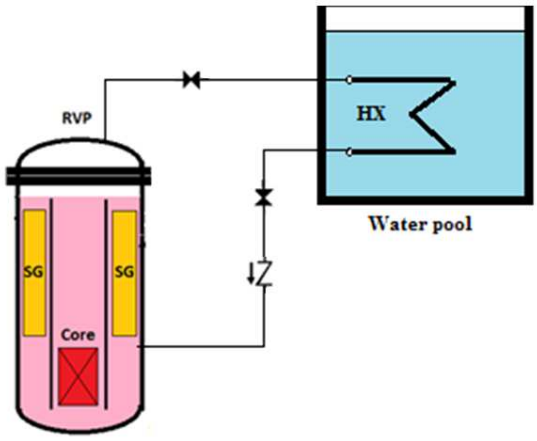

FIG. 8. Residual heat removal through condenser.

3.3. SAFETY INJECTION SYSTEM

3.3.1. High pressure injection system

3.3.1.1. Safety injection using pressurized tank

This injection system has been used in many reactor designs (e.g. AP1000 design) as part of emergency injection system. A typical design is a tank with borated water pressurized with nitrogen or other inert cover-gas. The tank bottom is connected to the reactor vessel through check valve that opens when the pressure of reactor vessel drops below the tank pressure such as during LOCA, as depicted in *FIG. 9*. This injection system provides one time injection of cool water to compensate sudden loss of coolant inventory during the interval in which the active system starts automatically.

Some advanced designs of pressurized tank systems provide longer injection capability, which eliminate the need of forced low pressure injection system, as in accumulator designs of APR1400, APWR and ATMEA1.

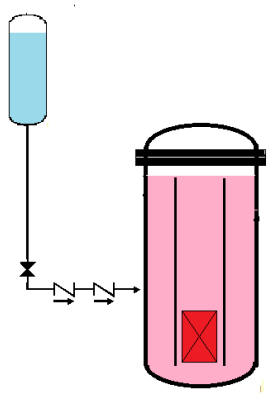

FIG. 9. Pressurized tank (accumulator).

3.3.1.2. Gravity driven safety injection system

Some integral PWR type SMRs uses an elevated borated water tank which use gravitational force to inject emergency borated water into the primary system, as depicted in *FIG. 10*. The top of the tank is connected to the reactor vessel with normally open valve, and the discharge line at the bottom of the tank is isolated from the reactor vessel by a

normally closed valve. During an emergency situation, the bottom valve opens and as a result the borated water flows down to the vessel, simultaneously cooling the core and terminating the fission. The SMR designs which implement this concept, among others, are the emergency boration tank of IRIS and the core make up tank of Westinghouse SMR.

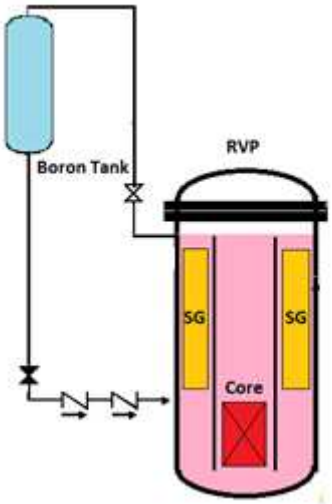

FIG. 10. Gravity driven injection tank (core make-up tank).

3.3.1.3. Injection system using high head pump

A conventional active approach for safety injection system along with secondary shutdown function is used in many existing reactors. This approach requires electric powered pump to inject boron solution from emergency boron tank into the reactor vessel, as illustrated in *FIG. 11*. Advanced reactor designs which use such active system approach are boron injection system of SMART.

3.3.1.4. Turbine driven injection system

In addition to gravity driven system or pressurized tank system, a core injection system is composed using a turbine driven pump. The steam from the reactor pressure vessel drives a pump which transfers water from a storage tank/pool into the reactor vessel. This approach of injection system is widely used in BWR reactors and known as the reactor core isolation cooling (RCIC) system.

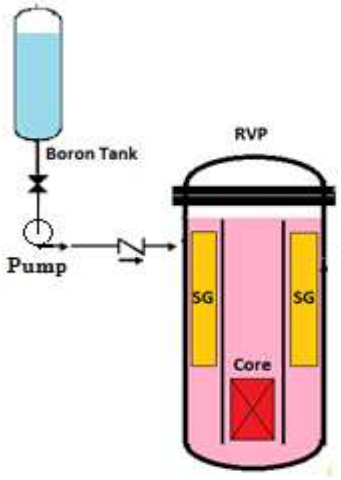

FIG. 11. Boron injection system using high head pump.

3.3.2. Low pressure injection system

3.3.2.1. Low-head pump injection system

An active low pressure injection system is implemented in APR1400, APWR and ATMEA1. In this system, low-head pump is used to inject water from in-containment water pool to the vessel, as shown in *FIG. 12*. Moreover, the high head injection system is useful if the LOCA is small. It develops high pressure and low flow which makes up for the coolant loss rate. If the LOCA is large the pressure in reactor is low and the coolant loss rate is high. Low-head injection system is designed for this. The cooling flow from ultimate heat-sink is usually provided to a heat-exchanger in the low-head injection system. This is because after the initial high head injection, low-head injection system is needed to remove the decay heat.

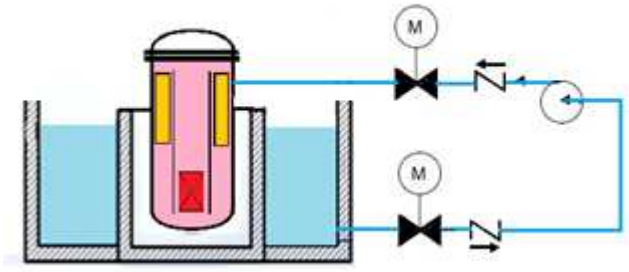

FIG. 12. Low-head pump injection system.

3.3.2.2. Passive low pressure injection system through elevated tank

Some reactor designs use large volume water storage inside the containment. The water storage located above the vessel enables passive injection flow to the core, as depicted in *FIG. 13*. The performance of this kind of system may be limited under core uncovery conditions due to steam produced in core region. Reactor designs that implement this method are in-containment refueling water storage tank (IRWST) of AP1000 and gravity driven cooling system (GDCS) of ESBWR. In AP1000, the flow is controlled by squib valves and the RCS must be depressurized prior to the function of this system. The automatic depressurization system (ADS) depressurizes the primary system using the four stages valves which automatically reduce the pressure to about 0.18 MPa to let the IRWST inject water by gravity.

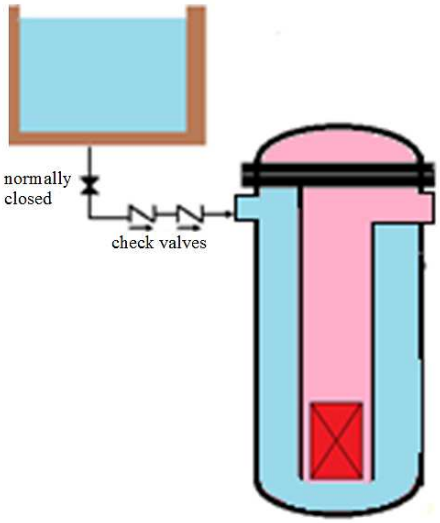

FIG. 13. Passive low pressure injection system.

3.3.2.3. Safety injection using pressurized tank

Pressurized tank is also used for low pressure injection as appears in the Emergency Injection System of CAREM25. The tank containing borated water is connected to RPV with primary function of preventing core uncovery during LOCA. During LOCA, when the pressure in the reactor vessel becomes relatively low (1.5 MPa), the rupture disks separating the accumulator tanks and the RPV would open to refill reactor vessel, as shown in *FIG. 14*.

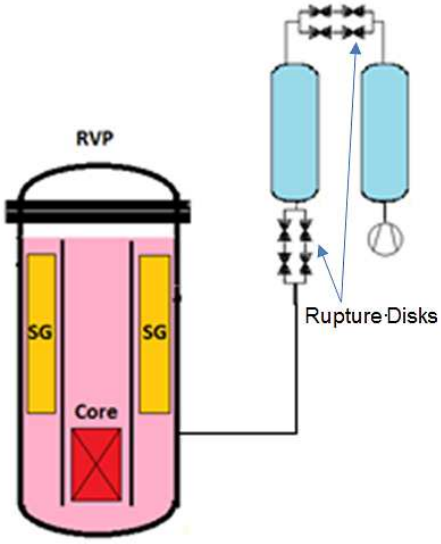

FIG. 14. Safety injection using pressurized tank.

3.3.2.4. Safety injection using recirculation valves

The injection system consists of redundant reactor vent valves and recirculation valves, as illustrated in *FIG. 15*. The system removes the core decay heat by opening of the vent valves. The steam from the reactor goes to the containment. It is cooled and condensed on the inside surface of the containment vessel by the pool water outside. The condensate accumulates in the bottom of the containment. When the level of water raises above the recirculation valves, the recirculation valves open. This establishes a natural circulation

path from the containment vessel to the core. The SMR design that uses this approach is NuScale.

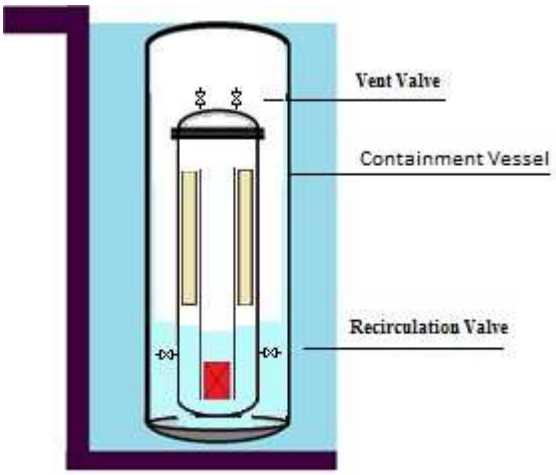

FIG. 15. Safety injection using recirculation valve.

3.4. CONTAINMENT SYSTEM (CONFINEMENT OF RADIOACTIVE MATERIAL)

Containment system has three main functions: (a) Confinement of radioactive substances in operational states and in accident conditions, (b) Protection of the plant against extreme natural hazards and human induced events, (c) Radiation shielding in operational states and in accident [7]. The integrity of the containment must be maintained under all conditions. Therefore the temperature and pressure inside containment must be controlled below design limits. Several methods are implemented to maintain temperature and pressure within design limits in SMR as described in the following subsections.

3.4.1. Pressure suppression containment

Some designs use water pool/tank for carrying out pressure suppression. High temperature steam released from the reactor vessel (through breaks, safety relief valve and ADS) is directed to a suppression pool/tank, as shown in *FIG. 16*. The steam condenses in the pool, thus mitigating the pressure increases in the containment. This type of containment has been used in BWR designs for many years. SMR designs which implement this method for their containment are CAREM25 and IRIS.

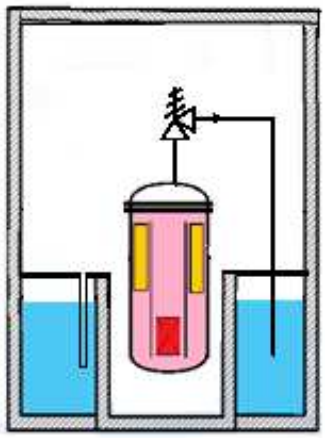

FIG. 16. Containment with pressure suppression pool.

3.4.2. Concrete containment with spray system

One method to control the containment pressure is spraying water into the containment's atmosphere. This approach has been widely used in existing PWR designs where the containment structure is made from concrete. The system uses pumps for injecting water from reservoirs outside the containment in the initial stage from in-containment sumps to the sprayer located in the reactor building dome, as shown in *FIG. 17*. The sprayed water condenses the steam and reduces the containment pressure. SMR design that uses this approach for its containment is SMART.

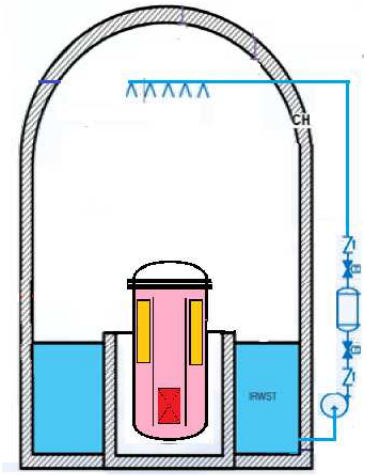

FIG. 17. Large dry concrete containment with spray system.

3.4.3. Submerged metal containment

As the power rating of SMR is relatively small, overall dimension of the core and reactor vessel is also small. A steel containment submerged in a water pool can be used to allow passive containment cooling, as illustrated in *FIG. 18*. This enables steam condensation inside the containment and at the same time heat removal with convection and/or conduction-convection mechanisms through containment wall to the external pool. As the containment is always cooled by pool water, any steam released from the reactor vessel due to an opening of safety relief valves or after LOCA can immediately be condensed in the containment and thus retain the water inventory and radioactive materials inside the containment. The effectiveness of the condensation is enhanced by evacuating the containment atmosphere into deep vacuum for normal operation. The vacuum condition reduces the heat loss and eliminates the need of reactor vessel insulation. It also reduces the possibility of hydrogen explosion during severe accident as oxygen gas is reduced considerably as a consequence. SMR design which implements this concept is NuScale. A similar approach is also used by Westinghouse SMR.

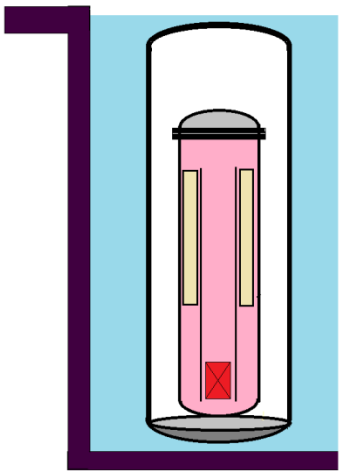

FIG. 18. Submerged metal containment.

3.4.4. Passively cooled large volume metal containment

Variant of metal containment is used by some advanced reactors. A large volume metal containment surrounded by reinforced concrete building is used to withstand the pressure increase during LOCA. The metal containment is passively cooled by air flow or by water spray, as shown in *FIG. 19*. Thus steam will condense at the inner surface of the containment. Containment design of this approach is utilized by AP1000. Similar concept is also used by the mPower where the upper hemisphere of its metal containment is passively cooled by water.

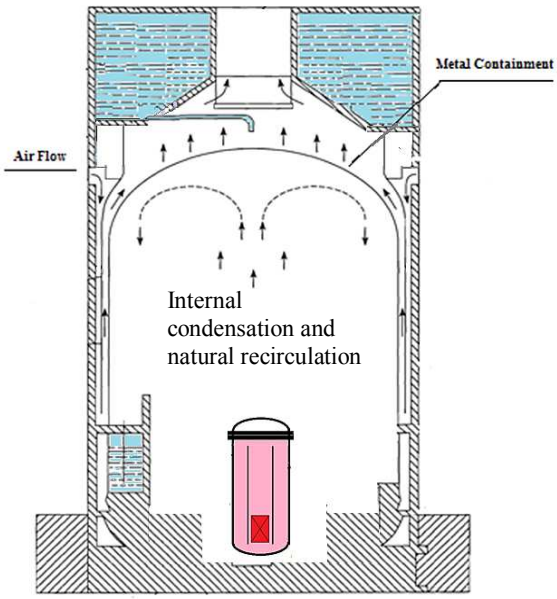

FIG. 19. Passively cooled large volume metal containment.

3.5. SEVERE ACCIDENT MITIGATION FEATURES

3.5.1. In-vessel retention system

Most of the advance reactor designs use strategies to deal with severe accident conditions. One of them is the implementation of in-vessel corium retention feature. In this strategy,

external surface of the reactor vessel is cooled by water induced from in-containment water pool. Reactor vessel cavity is flooded so that lower part of the vessel is always submerged in the water during severe accident, as depicted in *FIG. 20*. In this way the integrity of the vessel is maintained and corium can be retained inside the vessel. This kind of strategy to deal with severe accident is used by many advanced reactor designs such as AP1000, SMART, IRIS, mPower, etc.

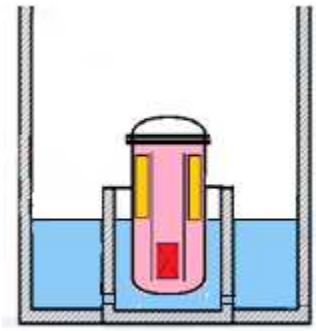

FIG. 20. In-vessel corium retention strategy.

This strategy is usually combined with a containment cooling system, either passive or active, to ensure that the sufficient water inventory can be kept cool for extended period.

3.5.2. Core catcher

In order to prevent the molten core material from escaping the containment building, a specific structure/device is placed below the reactor vessel. It is a space made of heat-resistant concrete ceramic to prevent molten core from penetrating through. Core catcher spreads the molten core and vessel material to decrease thermal density and it is usually designed with a cooling mechanism to keep the molten core stable for extended period, as shown in *FIG. 21*. Advanced reactor designs which implement core catcher include ESBWR and ATMEA1. In the ESBWR, the BiMac core catcher is used.

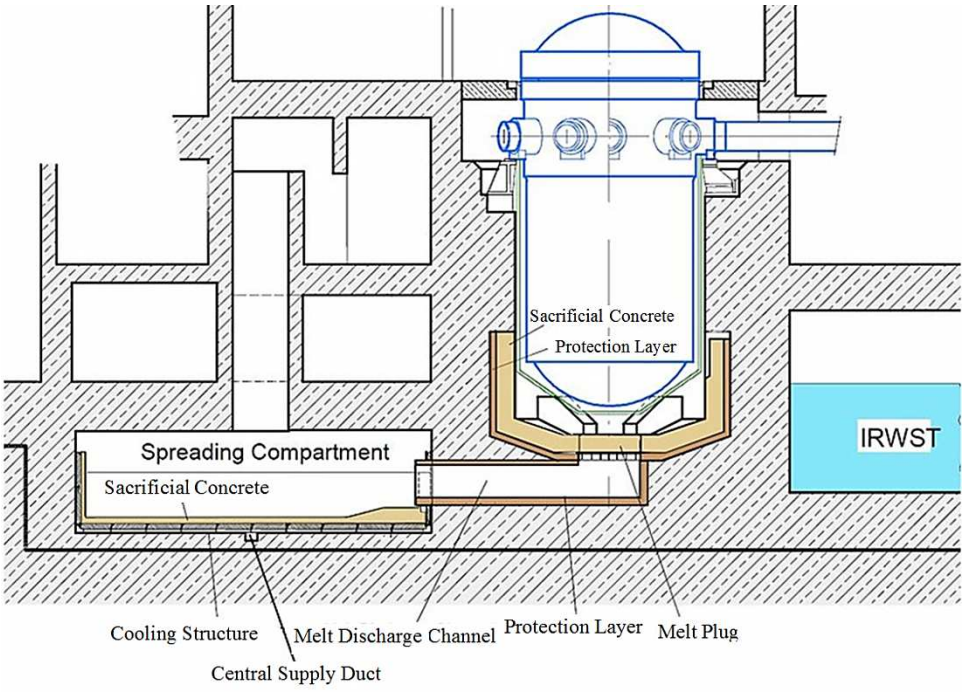

FIG. 21. Core catcher.

3.5.3. Hydrogen control devices

Severe accident is an accident that accompanies nuclear fuel damage. Overheated zirconium alloy in the fuel clad reacts with steam at high temperature to generate hydrogen gas. The oxidation of zirconium is an exothermic reaction which enhances hydrogen generation. The hydrogen may detonate in the presence of the stoichiometric proportion of oxygen. The location of explosion depends on the local hydrogen and oxygen concentration and temperature. It results in high pressure spike more than the containment design pressure. Therefore hydrogen concentration should be reduced and controlled to maintain the integrity of the containment. This is possible if the hydrogen is recombined with oxygen as it is produced, with the use of passive auto-catalytic hydrogen re-combiners (PARs), as illustrated in *FIG. 22* in the containment. New conventional designs use this and as back-fitting in older designs. The SMR and advanced reactor designs which employ such devices, among others, are SMART, mPower, IRIS and CAREM25. Some reactor designs combine the hydrogen control device with pre-inerting of containment atmosphere with nitrogen to remove oxygen, as implemented in the IRIS, ABWR and ESBWR.

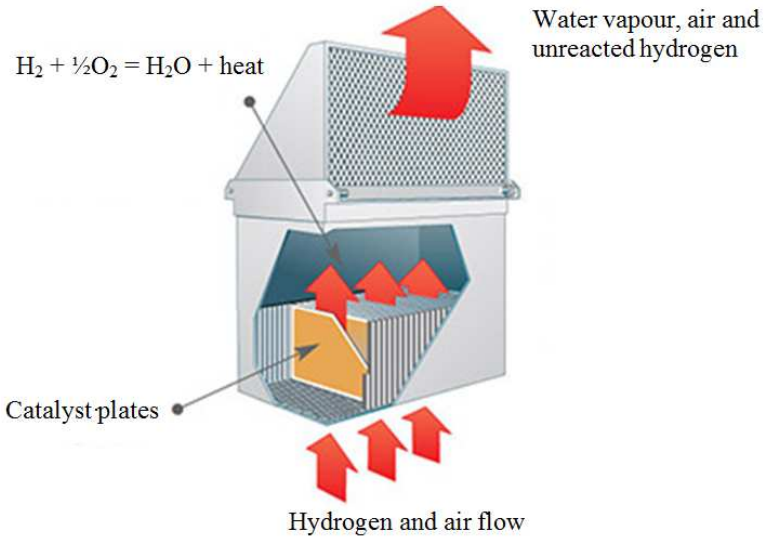

FIG. 22. A typical passive autocatalytic recombiner.

3.5.4. Filtered containment venting system

In the event of accident, especially when the pressure inside containment increases up to a point reaching or beyond its design limit, filtered containment venting is used to prevent its catastrophic overpressure failure. Filtered venting prevents uncontrolled release of entire core radioactivity to the environment. A typical schematic diagram is given in *FIG. 23*. Filtered containment venting systems are employed in most SMR designs.

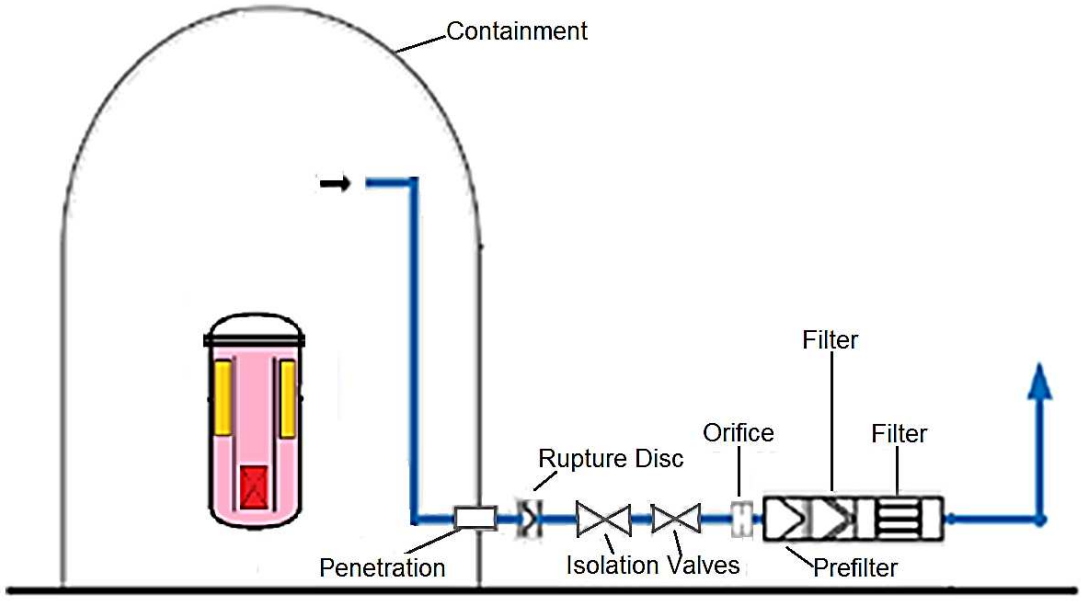

FIG. 23. Filtered containment venting system.

3.6. DEFENCE IN DEPTH IN SMALL MODULAR REACTORS

Defence-in-depth (DiD) is a concept that has been applied to ensure the safety of nuclear installations since the start of nuclear power development. Its objective is to compensate

for potential human and equipment failures by means of several levels of protection. DiD is an established safety philosophy in which multiple lines of defence, safety margins, and compensatory measures are applied to the design, construction, operation, maintenance, and regulation of nuclear plants to prevent and to mitigate accidents and to assure that the adequate protection of public health and safety. Defence is provided by multiple and independent means at each level of protection. The concept of DiD, as applied to all safety activities, whether organizational, behavioral or design related, ensures that they are subject to overlapping provisions, so that if a failure were to occur, it would be detected and compensated for or corrected by appropriate measures. Application of the concept of DiD throughout design and operation provides a graded protection against a wide variety of transients, anticipated operational occurrences and accidents, including those resulting from equipment failure or human action within the plant, and events that originate outside the plant [1]. In SMR designs, as in larger reactor designs, the DiD strategy is used to protect the public and environment from accidental releases of radiation. Nearly all SMRs designs seek to strengthen the first and subsequent levels of defence by incorporating inherent and passive safety features. Certain common characteristics of smaller reactors lend themselves to inherent and passive safety features, such as relatively smaller core sizes enabling integral coolant system layouts and larger reactor surface-to-volume ratios or lower core power densities which facilitate passive decay heat removal. Using the benefits of such features, the main goal is to eliminate or prevent, through design, as many accident initiators and accident consequences as possible. Remaining plausible accident initiators and consequences are then addressed by appropriate combinations of active and passive safety systems. The intended outcome is greater plant simplicity with high safety levels that, in turn, may allow reduced emergency requirements off-site. It should be noted that an approach to maximize the use of inherent safety. Application of the concept of DiD in the design of a plant provides a series of levels of defence (inherent features, equipment and procedures) aimed at preventing accidents and ensuring appropriate protection in the event that prevention fails.

1. The aim of the first level of defence is to prevent deviations from normal operation, and to prevent system failures. This leads to the requirement that the plant be soundly and conservatively designed, constructed, maintained and operated in accordance with appropriate quality levels and engineering practices, such as the application of redundancy, independence and diversity. To meet this objective, careful attention is paid to the selection of appropriate design codes and materials, and to the control of fabrication of components and of plant construction. Design options that can contribute to reducing the potential for internal hazards (e.g. controlling the response to a postulated initiating event), to reducing the consequences of a given postulated initiating event, or to reducing the likely release source term following an accident sequence contribute at this level of defence. Attention is also paid to the procedures involved in the design, fabrication, construction and in-service plant inspection, maintenance and testing, to the ease of access for these activities, to the way the plant is operated and to how operational experience is utilized. This whole process is supported by a detailed analysis which determines the operational and maintenance requirements for the plant.
2. The aim of the second level of defence is to detect and intercept deviations from normal operational states in order to prevent anticipated operational occurrences from escalating to accident conditions. This is in recognition of the fact that some postulated initiating events are likely to occur over the service lifetime of an NPP, despite the care taken to prevent them. This level necessitates the provision of specific

systems as determined in the safety analysis and the definition of operating procedures to prevent or minimize damage from such postulated initiating events.

3. For the third level of defence, it is assumed that, although very unlikely, the escalation of certain anticipated operational occurrences or postulated initiating events may not be arrested by a preceding level and a more serious event may develop. These unlikely events are anticipated in the design basis for the plant, and inherent safety features, fail-safe design, additional equipment and procedures are provided to control their consequences and to achieve stable and acceptable plant states following such events. This leads to the requirement that ESFs be provided that are capable of leading the plant first to a controlled state, and subsequently to a safe shutdown state, and maintaining at least one barrier for the confinement of radioactive material.
4. The aim of the fourth level of defence is to address severe accidents in which the design basis may be exceeded and to ensure that radioactive releases are kept as low as practicable. The most important objective of this level is the protection of the confinement function. This may be achieved by complementary measures and procedures to prevent accident progression, and by mitigation of the consequences of selected severe accidents, in addition to accident management procedures. The protection provided by the confinement may be demonstrated using best estimate methods.
5. The fifth and final level of defence is aimed at mitigation of the radiological consequences of potential releases of radioactive materials that may result from accident conditions. This requires the provision of an adequately equipped emergency control [1].

A relevant aspect of the implementation of DiD is the provision in the design of a series of physical barriers to confine the radioactive material at specified locations. The number of physical barriers that will be necessary will depend on the potential internal and external hazards, and the potential consequences of failures. The barriers may, typically for water cooled reactors, be in the form of the fuel matrix, the fuel cladding, the reactor coolant system pressure boundary and the containment [1]. Design features of pressurized water SMRs contributing to enhancement of Level 1 of defence in depth are summarized in *TABLE 1*; subsequent levels are summarized in *TABLE 2 , 3, 4* and *5* respectively. The approach was taken from IAEA Nuclear Energy Series NP-T-2.2 on Design Features to Achieve Defence in Depth in Small and Medium Sized Reactors (2009) [10].

Applicable Safety Requirement for Defence in Depth in SMRs [11]

IAEA Safety Standards Series on Specific Safety Requirement No. SSR – 2/1 (Rev. 1), entitled Safety of Nuclear Power Plants: Design was issued in 2012. This publication establishes design requirements for the structures, systems and components of a NPP, as well as for procedures and organizational processes important to safety that are required to be met for safe operation and for preventing events that could compromise safety, or for mitigating the consequences of such events, were they to occur.

In the aftermath of the TEPCO's Fukushima Daiichi NPPs accident, this publication was revised with being fully reflected lessons that are to be learned from reports on and studies of the Fukushima accident in the relevant requirements and was approved to be re-issued. This safety requirement will be used primarily for land based stationary NPPs with water cooled reactors designed for electricity generation or for other heat production applications, such as district heating or desalination. This safety requirement

may also be applied, with judgement, to other reactor types, to determine the requirements that have to be considered in developing the design.

Requirement 7 of SSR–2/1 (Rev. 1) establishes the following requirements on application of defence in depth.

"The design of a NPP shall incorporate defence in depth. The levels of defence in depth shall be independent as far as is practicable"

Paragraph 4.9 requires that the defence in depth concept shall be applied to provide several levels of defence that are aimed at preventing consequences of accidents that could lead to harmful effects on people and the environment, and ensuring that appropriate measures are taken for the protection of people and the environment and for the mitigation of consequences in the event that prevention fails.

Paragraph 4.10 requires that the design shall take due account of the fact that the existence of multiple levels of defence is not a basis for continued operation in the absence of one level of defence. All levels of defence in depth shall be kept available at all times and any relaxations shall be justified for specific modes of operation.

Paragraph 4.11 requires the design:

a) Shall provide for multiple physical barriers to the release of radioactive material to the environment;

b) Shall be conservative, and the construction shall be of high quality, so as to provide assurance that failures and deviations from normal operation are minimized, that accidents are prevented as far as is practicable and that a small deviation in a plant parameter does not lead to a cliff edge effect;

c) Shall provide for the control of plant behaviour by means of inherent and engineered features, such that failures and deviations from normal operation requiring actuation of safety systems are minimized or excluded by design, to the extent possible;

d) Shall provide for supplementing the control of the plant by means of automatic actuation of safety systems, such that failures and deviations from normal operation that exceed the capability of control systems can be controlled with a high level of confidence, and the need for operator actions in the early phase of these failures or deviations from normal operation is minimized;

e) Shall provide for systems, structures and components and procedures to control the course of and, as far as practicable, to limit the consequences of failures and deviations from normal operation that exceed the capability of safety systems;

f) Shall provide multiple means for ensuring that each of the fundamental safety functions is performed, thereby ensuring the effectiveness of the barriers and mitigating the consequences of any failure or deviation from normal operation.

Paragraph 4.13 requires that the design shall be such as to ensure, as far as is practicable, that the first, or at most the second, level of defence is capable of preventing an escalation to accident conditions for all failures or deviations from normal operation that are likely to occur over the operating lifetime of the NPP.

TABLE 1. DESIGN FEATURES OF WATER COOLED SMR DESIGNS CONTRIBUTING TO LEVEL 1 OF DEFENCE IN DEPTH

No.	Design features	Design objectives	SMR designs	Relevant safety requirements
1.	Elimination of liquid boron reactivity control system	Exclusion of inadvertent reactivity insertion as a result of boron dilution	KLT-40S, CAREM25, IRIS, IMR, ABV-6M, RITM-200, VK-300, mPower, SMR-160, Flexblue.	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirement 20, Paragraph 4.11 [(a) and (b)] and relevant Paragraphs. [11]
2.	Relatively low core power density	Larger thermal-hydraulic margins	IRIS, CAREM25, NuScale, mPower	
3.	High thermal conductivity of fuel	Relatively low temperature of fuel and High margin to fuel failure	KLT-40S	
4.	Gas pressurizer system	Pressurizer heater potentially unreliable component	KLT-40S, RITM-200	
5.	Integral design of primary circuit with in-vessel location of steam generators	Exclusion of large-break, loss of coolant accidents (LOCA)	CAREM25, IRIS, ACP100, DMS, IMR, SMART, ABV-6M, RITM-200, VK-300, UNITHERM, NuScale, mPower, Westinghouse SMR	
6.	Compact modular design of the reactor unit	Decreased probability of LOCA	KLT-40S, ELENA	
7.	Primary pressure boundary enclosed in a pressurized, low enthalpy containment	Elimination of LOCA resulting from failure of the primary coolant pressure boundary	NuScale	
8.	Leak tight reactor coolant system	Decreased probability of LOCA	KLT-40S	
9.	Internal horizontal, fully immersed pumps	Elimination of pump seizure, rotor lock, and seal LOCA	IRIS, SMART, Westinghouse SMR	
10.	Vertical Canned motor pump	Decreased probability of seal LOCA	ACP100, KLT-40S,VBER-300, mPower	
11.	Flow restriction devices in the primary pipelines	Limitation of the break flow	KLT-40S	
12.	Natural circulation in normal operation	Elimination of loss of flow accidents	CAREM25, DMS, IMR, ABV-6M, VK-300, UNITHERM, ELENA, NuScale, SMR-160	

TABLE 1 (cont.)

No.	Design features	Design objectives	SMR designs	Relevant safety requirements
13.	Steam generator with lower pressure inside the tubes in normal operation mode	Reduced probability of a steam tube rupture	KLT-40S, IRIS	
14.	Steam generator designed for a full primary system pressure	Prevention or downgrading of a steam line break or a feed line break	IRIS	
15.	CRDM in side Reactor pressure vessel	Eliminate control rod ejection accidents	CAREM25, IRIS, Westinghouse SMR	
16.	Large water inventory in core i.e., high primary coolant inventory per unit power, in m^3/MW)	• Large thermal inertia • Reduced requirements of heat removal systems, core uncovery and loss of feed water	IRIS, CAREM25	
17.	Larger surface-to-volume ratio	Facilitates easier decay heat removal	All designs	
18.	Fuel Temperature coefficient	Reduced probability of abnormal super criticality	All designs	
19.	Relatively low linear heat rate of fuel	Higher margin to fuel failure		

TABLE 2. DESIGN FEATURES OF WATER COOLED SMR DESIGNS CONTRIBUTING TO LEVEL 2 OF DEFENCE IN DEPTH

No.	Design features	Design objectives	SMR designs	Relevant safety requirements
1.	Active systems of instrumentation and control	Timely detection of abnormal operation and failures	All designs	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirement 20, Paragraph 4.11 [(a) and (c)] and relevant Paragraphs. [11]
2.	Negative reactivity coefficients over the whole burnup cycle	Prevention of transient over criticality due to abnormal operation and failures. Increased self-control of abnormal operation.	All designs	
3.	A relatively large coolant inventory in the primary circuit, resulting in large thermal inertia	Slow progression of transients due to abnormal operation and failures	CAREM25, IRIS	
4.	High heat capacity of nuclear installation as a	Slow progression of transients due to abnormal	KLT-40S	

TABLE 2 (cont.)

No.	Design features	Design objectives	SMR designs	Relevant safety requirements
	whole	operation and failures		
5.	Implementation of the leak before break concept	Facilitate implementation of leak before break concept	KLT-40S	
6.	Little coolant flow in the low temperature pressurized water containment enclosing the primary pressure boundary	Facilitate implementation of leak before break concept		
7.	Redundant and diverse passive or active shutdown systems	Reactor shutdown	All designs	
8.	Use of digital technology	Proven reliability of I&C system		
9.	Improved human-machine interface		NuScale, mPower, ACP100	

TABLE 3. DESIGN FEATURES OF WATER COOLED SMR DESIGNS CONTRIBUTING TO LEVEL 3 OF DEFENCE IN DEPTH

No.	Design features	Design objectives	SMR designs	Relevant safety requirements
1.	Negative reactivity coefficients over the whole cycle	Prevention of transient over-criticality and bringing the reactor to a subcritical state in design basis accidents	All designs	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirement 20, Paragraph 4.11 [(a) and (d)] and relevant Paragraphs. [11]
2.	A relatively large coolant inventory in the primary circuit	Slow progression of transients in design basis accidents	CAREM25, IRIS	
3.	High heat capacity of nuclear installation as a whole	Limitation of temperature increase in design basis accidents	KLT-40S	
4.	Restriction devices in pipelines of the primary circuit, with primary pipelines being connected to the hot part of the reactor	Limitation of scope and slower progression of LOCA	KLT-40S	
5.	Use of once-through steam generators	Limitation of heat rate removal in a steam line break accident	KLT-40S	

TABLE 3 (cont.)

No.	Design features	Design objectives	SMR designs	Relevant safety requirements
6.	Steam generator designed for full primary	Limitation of the scope of a steam generator tube rupture accident	IRIS	
7.	Soft pressurizer system	Damping pressure perturbations in design basis accidents	KLT-40S, RITM-200	
8.	Self-pressurization, large pressurizer volume, elimination of sprinklers, etc.	Damping pressure perturbations in design basis accidents	CAREM25, IRIS, DMS, SMART, VBER-300, VK-300, UNITHERM, mPower, NuScale, Westinghouse SMR, SMR-160	
9.	Limitation of inadvertent control rod movement by an overrunning clutch and by the limiters	Limitation of the scope of reactivity insertion in an accident with control rod drive bar break	KLT-40S	
10.	Redundant and diverse reactor shutdown and heat removal systems	Increased reliability in carrying out safety functions	All designs	
11.	Insertion of control rods to the core, driven by gravity	Reactor shutdown	KLT-40S, CAREM25, ACP100, IRIS, IMR, SMART, VBER-300, ABV-6M, UNITHERM, RUTA-70, NuScale, Westinghouse SMR	
12.	Insertion of control rods to the core, driven by force of springs	Reactor shutdown	KLT-40S	
13.	Non-safety-grade control rod system with internal control rod drives	Reactor shutdown	IRIS	
14.	Gravity driven high pressure borated water injection device (as a second shutdown system)	Reactor shutdown	CAREM25 and AHWR300	
15.	Injection of borated water from the emergency boron tank at high pressure (as an auxiliary shutdown measure)	Reactor shutdown	IRIS	

TABLE 3 (cont.)

No.	Design features	Design objectives	SMR designs	Relevant safety requirements
16.	Emergency injection system (with borated water), actuated by rupture disks	Reactor shutdown plus prevention of core uncovery in LOCA	CAREM25	
17.	Natural convection core cooling in all modes	Passive heat removal	CAREM25, AHWR-300, DMS, IMR, ABV-6M, VK-300, UNITHERM, ELENA, NuScale, SMR-160	
18.	Safety (relief) valves	Protection of reactor vessel from over pressurization	IRIS, CAREM25, it should be available in all designs	
19.	Long term gravity make-up system	Assures that the core remains covered indefinitely following a LOCA	IRIS and ACP100	

TABLE 4. DESIGN FEATURES OF WATER COOLED SMR DESIGNS CONTRIBUTING TO LEVEL 4 OF DEFENCE IN DEPTH

No.	Design features	Design objectives	SMR designs	Relevant safety requirements
1.	Relatively low core power density	Limitation or postponement of core melting	IRIS, CAREM25, NuScale and mPower	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirement 20, Paragraph 4.11 (a and e) and relevant Paragraphs. [11]
2.	Low heat-up rate of fuel elements predicted in a hypothetical event of core uncovery, owing to design features	Prevention of core melting due to core uncovery	CAREM25 and mPower	
3.	Passive emergency core cooling.	To provide adequate time for accident control	KLT-40S, IRIS, Flexblue, DMS, KLT-40S, CAREM25, VVER-300, VBER-300, AHWR300 Westinghouse SMR, ACP100, SMART, VK-300, NuScale UNITHERM, mPower, SMR-160	
4.	Passive system of reactor vessel bottom cooling	In-vessel retention of core melt	KLT-40S, CAREM25 and Flexblue	

TABLE 4 (cont.)

No.	Design features	Design objectives	SMR designs	Relevant safety requirements
5.	Passive flooding of the reactor cavity following a small LOCA	Prevention of core melting due to core uncovery; in-vessel retention	IRIS, VBER-300, mPower	
6.	Containment building	Prevention of radioactive release in severe accidents; protection against external event impacts	All design	
7.	Containment and protective enclosure or Double containment	Protection against radioactive release in severe accidents and external event (like aircraft crash, missiles)	IRIS, CAREM25, KLT-40S, VVER-300, VK-300, UNITHERM, mPower, NuScale, Westinghouse SMR and SMR-160	
8.	Passive containment cooling system	Reduction of containment pressure and limitation of radioactivity release	KLT-40S, DMS, SMART, AHWR300, NuScale, ACP100, mPower	
9.	Inert containment	Prevention of hydrogen combustion	IRIS	
10.	Reduction of hydrogen concentration in the containment by catalytic re-combiners	Prevention of hydrogen combustion	CAREM25, AHWR300, SMR-160	

TABLE 5. DESIGN FEATURES OF WATER COOLED SMR DESIGNS CONTRIBUTING TO LEVEL 5 OF DEFENCE IN DEPTH

No.	Design features	Design objectives	SMR designs	Relevant safety requirements
1.	Mainly administrative measures	Mitigation of radiological consequences resulting in significant release of radioactive materials	KLT-40S	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirement 20, Paragraph 4.11 [(a) and (f)] and relevant Paragraphs. [11]
2.	Relatively small fuel inventory, less non-nuclear energy stored in the reactor, and lower decay heat rate.	Smaller source term, smaller emergency planning zone (EPZ)	All design	

4. COUNTERMEASURES TO ADDRESS THE LESSONS LEARNED FROM THE FUKUSHIMA DAIICHI ACCIDENT IN THE DESIGN OF WATER COOLED SMALL MODULAR REACTORS

A nuclear power plant (NPP) is constructed and deployed with the main objective to provide reliable electrical power along with the assurance of safety. To guarantee the safety, a plant must be equipped with sufficient capacity and capability of reliable systems in order to deal with various operation modes such as normal operation, shutdown, maintenance and abnormal conditions throughout its design life. Invariably redundant safety systems and barriers are also used to protect the people and environment inside and surrounding the plant and to bring the plant back to safe condition as quickly as possible when unacceptable conditions occur. In addition, diversity is also implemented to avoid functional failure due to either common cause or common mode. General intention of the redundancy and diversity across the systems inside the plant is to ensure that the systems will reliably respond against any deviation or abnormality as designed to prevent and mitigate progression of incident/accident and protect the health and safety of people and the environment.

The plant has numerous systems that shall function properly and reliably as designed. There are interconnected structures, subsystems and components which could raise concerns of complexity. A strategy is needed to organize the components and systems so that their roles are well managed for their success. The safety strategy can be ensured with DiD concept where actions and mitigations are divided into several layers of defence. Plant design is then analyzed using deterministic and probabilistic methods to guarantee the capability of the plant systems in dealing with various events during its lifetime.

According to the IAEA publication 'Defence in Depth in Nuclear Safety, INSAG-10, a Report by the International Nuclear Safety Advisory Group', the DiD concept has two principal strategies: first, to prevent accidents and second, if prevention fails, to limit their potential consequences and prevent any accident progression detrimental to the health and safety of people or environment. Usually DiD is structured in five (5) levels in which should one level fail, the subsequent level comes to play to deal with the reactor conditions and protects the overall system. The five levels of defence are [11]:

1. Prevention of abnormal operation and system failures;
2. Control of abnormal operation and detection of failures;
3. Control of accident within the design basis;
4. Control of severe plant conditions including prevention of accident progression and mitigation of the consequences of severe accidents;
5. Mitigation of radiological consequences to protect people and environment against significant releases of radioactive materials.

The first level of DiD is conservatism in the design, and high quality of construction and operation. The second level of DiD is use of control, limiting and protection systems and surveillance means. The third level of DiD are ESFs to cope with postulated designs basis accidents, such as LOCA. The fourth level of DiD employs complementary measures for accident managements to anticipate the unexpected severe excursions when the capabilities of ESFs are exceeded. The fifth level of DiD are off-site emergency response actions. Each level of DiD should be independent of the others and failure in one level should not impair the functionality of other levels.

As explained in Section 2 on the accident sequence in Units 1, 2 and 3, the Fukushima Daiichi accident has revealed many lessons in dealing with extreme natural hazards for a site with multiple reactors. The accident progression, facts and traces discovered from the fact-finding expert missions, indicated weaknesses in design, accident management and the regulatory system. It is known that the Fukushima Daiichi complex design basis tsunami was far less than what really happened on March 11, 2011. Although the tsunami basis was re-evaluated in 2002 and an updated information such as the Jogan Tsunami was brought into attention, it was neither thoroughly considered nor accurately estimated [12]. This lack of contingency consideration for potential natural hazards led to cliff edge effects to the reactor facilities in the complex during the early phase of the accident. The huge volume of water that came with the tsunami exceeded the height of the protective wall and became a common cause for multiple damages to the facilities. The damages stretched to several functions concurrently such as the damage to oil tanks, the loss of ultimate heat sink, the loss of emergency power supplies due to water flooding in the EDG rooms, and the reduced site accessibility due to the littered stray objects due to tsunami flood.

Multiple impacts from the extreme natural hazards resulted in significant difficulties in managing available resources to cope with the unsafe conditions. Damaged roads slowed down many required actions to respond the emergency situation. Loss of electric power forced the operators to work in darkness when trying to control the reactors. Multiple units with shared resources also hindered the equipment availability to fulfil the simultaneous needs. For example, when the fire engine was asked for by Unit 3 for emergency water injection, it was being used to resolve the problem in Unit 1.

These are a few examples of the lessons learned from the Fukushima Daiichi accident. There are many others which have been revealed and documented. Experts from various organizations collected and identified as many as 94 individual lessons and recommendations on Fukushima Daiichi Accident [13] to [16], which are grouped into integrated lessons learned. These integrated lessons are categorized into 4 main areas as follows:

A. *Design and Siting (section 4.1)*
    1. Strengthen measures against extreme natural hazards and consequential effects
    2. Consider issues concerning multiple reactor sites and multiple sites
    3. Ensure off-site and on-site electricity supplies
    4. Ensure robust measures for reactor core cooling and ultimate heat sink
    5. Ensure design of safety-related structures, systems and components
    6. Ensure measures for prevention and mitigation of hydrogen explosions
    7. Enhance containment venting and filtering system
    8. Ensure hardened instrumentation and cables for safety-related parameters and monitoring equipment
    9. Enhance robustness of spent fuel cooling
    10. Use PSA effectively for risk assessment and management
B. *Accident Management and on-site emergency preparedness and response (section 4.2)*
    1. Ensure on-site emergency response facilities, equipment and procedures

2. Enhance human resource, skill and capabilities

C. *Off-site emergency preparedness and response (section 4.3)*

1. Strengthen off-site infrastructure and capability
2. Strengthen national arrangements for emergency preparedness and response
3. Enhance interaction and communication with the international communities

D. *Nuclear safety infrastructures (section 4.4)*

1. Review and clarify regulatory and emergency response framework
2. Reinforce safety regulatory bodies and legal structures
3. Instill safety awareness and attitude

Some countermeasures were discussed in the technical meetings (see section 1.4) to address the lessons learned from the Fukushima Daiichi event. *Table 1* through *18* summarizes the result of the meetings and the level of Defence-in-Depth being addressed.

4.1. DESIGN AND SITING

4.1.1. Strengthen measures against extreme natural hazards and consequential effects

Historically, extreme natural events that disrupted the operation of NPPs appeared in many different forms such as strong earthquake, high tsunami wave, major flooding, sustained high heat, very strong wind, etc. In some cases, historical occurrences of the event showed a regular pattern, interval and a certain indicative trend of the magnitude with return periods. Although there exists uncertainty, the magnitude of the event can be projected based on historical data and its indicative trend. According to the site circumstances, design needs evaluation to incorporate the impact of the extreme events that have occurred.

Extreme events do not necessarily take place only in a single form. They can be combination of two or more events that come in sequence, thus providing challenges to the reactor facilities. The events may be very closely related as in a cause-effect relation, e.g. tsunami and major flood following a large undersea earthquake, but in some instances different type of natural hazards may occur at the same time. Designers of reactors need to gather historic evidence of the occurrence and strength of the extreme events to be counted in their design or analysis of safety system. However, uncertainty remains large as nature of the occurrence is complex and not completely understood so that the prediction of the strength, location and timing of occurrence is a challenge. Putting a conservative consideration, designers need to consider credible historical data of single events with adequate design margins and probable simultaneous occurrence.

The combination of two or more natural hazards can result in unprecedented impacts to the reactor facility. The Fukushima Daiichi accident which underwent large earthquake and subsequent tsunami revealed several lessons for the design of future reactors. The following are some of the SMR design guidelines based on the lessons learned:

- *Earthquake exceeding the design basis did not cause any known significant damage.* It was revealed that when the large scale earthquake exceeding the design basis occurred, the safety system of the Fukushima Daiichi reactors worked as planned. The reactors tripped and the EDGs automatically functioned as expected to supply

electricity needed by the reactor emergency system. The reactor building had no significant damage and all safety equipment remained intact. This condition shows that the design margin of the facilities was sufficient to withstand a larger earthquake. This approach can be a guideline for SMR designs.

- *Earthquake exceeding the design basis causing loss of off-site power source became the initiating event for accident scenario.*
  Although the earthquake did not cause any significant challenge to the reactor systems, it damaged the infrastructure of electrical grid connected to the station. This led to a loss of off-site power source which became the initiating event for accident that occurred. As designed, loss of off-site power automatically activated the EDGs to supply reactor safety system with electrical power. However, the failure of EDGs after the tsunami induced flooding lead to common cause/common mode failures of all EDGs. A total blackout in the complex resulted in a total loss of emergency core cooling as the DC power was also lost after some time when the batteries drained below their limits. Debris from tsunami and hydrogen explosion created significant logistical difficulties and inhibited response actions. Tsunami spread wreckages and debris on roads and at gates and some areas remained flooded for a long time. Hydrogen explosions damaged the buildings significantly and spread contamination into atmosphere. The debris delayed the arrival of fire engine and emergency teams at the complex and limited the field working time as the radioactivity level rose up rapidly after the explosion. SMR design guidelines should consider some independence, during a practical grace period, e.g. from off-site power, availability of cooling sources, communication and transportation systems, etc.

- *Optimizing for earthquake made some equipment vulnerable to tsunami (e.g., locating EDGs underground).*
  When a strong earthquake shook the Fukushima Daiichi reactor complex, the EDG buildings did not lose their integrity. The buildings were designed and placed conservatively underground to provide better resilience against the earthquake hazards, but this location resulted in their flooding when the tsunami exceeded the height of the protection wall. SMR design should accommodate combined effects of various natural hazards.

- *Extensive tsunami and subsequent hydrogen explosion damage and debris created significant logistical difficulties and inhibited response actions.*
  The large high tsunami waves created much more damages to the site and reactor facilities than the earthquake. The tsunami spread wreckages and debris on the roads and at gates. Some areas remained flooded for a long time. In addition, hydrogen explosions damaged the buildings significantly and spread contamination to the atmosphere. The debris delayed the arrival of fire engine and emergency teams at the complex and hampered the efforts to control the reactor by the emergency team such as delaying the delivery of a fire engine to the complex and limiting the field working time as the radioactivity level rose up rapidly after the explosion.

- *Repeated aftershocks and tsunami threats stopped recovery work on occasions.*
  During emergency recovery the workers efforts were interrupted and they had to take shelter due to repeated tremors. This caused further delays in providing immediate alternate cooling for the reactor leading to more extensive fuel damage.

SMR design should consider subsequent effects of natural hazards that could be beyond the first occurrence.

All the findings from the Fukushima Daiichi accident need to be addressed. The accident basically confirmed the necessity of strengthening measures against extreme natural hazards and their consequential effects. SMRs, which have variety of site options, should carefully include them in the design considerations as they may be deployed in sites such as coastal areas, islands, on floating barges, underground, and near high population zones. Following *TABLE 6* summarises the strengthening measures against natural hazards and consequential effects.

Applicable Safety Requirements against extreme natural hazards and consequential effects [11]

Requirement 17 of SSR–2/1 (Rev. 1) establishes the following requirements on internal and external hazards.

"All foreseeable internal hazards and external hazards, including the potential for human induced events directly or indirectly to affect the safety of the NPP, shall be identified and their effects shall be evaluated. Hazards shall be considered in designing the layout of the plant and in determining the postulated initiating events and generated loadings for use in the design of relevant items important to safety for the plant".

Paragraph 5.17 requires that the design shall include due consideration of those natural and human induced external events (i.e. events of origin external to the plant) that have been identified in the site evaluation process. Causation and likelihood shall be considered in postulating potential hazards. In the short term, the safety of the plant shall not be permitted to be dependent on the availability of off-site services such as electricity supply and fire-fighting services. The design shall take due account of site specific conditions to determine the maximum delay time by which off-site services need to be available.

Paragraph 5.19 requires that features shall be provided to minimize any interactions between buildings containing items important to safety (including power cabling and control cabling) and any other plant structure as a result of external events considered in the design.

Paragraph 5.21 requires that the design of the plant shall provide for an adequate margin to protect items important to safety against levels of external hazards to be considered for design seismic events taking into account the site hazard evaluation, and to avoid cliff edge effects.

Paragraph 5.21a requires that the design of the plant shall provide for an adequate margin to protect items ultimately necessary to prevent large or early radioactive releases in the event of levels of natural hazards exceeding those to be considered for design taking into account the site hazard evaluation.

TABLE 6. STRENGTHENING MEASURES AGAINST EXTREME NATURAL HAZARDS AND CONSEQUENTIAL EFFECTS

Defence in Depth level	Critical issues addressed	Option for counter-measures	Considerations for water cooled SMRs	Relevant safety requirements
Prevention (1)	Natural hazards	Ensure that all types of natural hazards are considered in the design.	• Natural hazards include earthquake, tsunami, external flood, high winds (typhoon, cyclone, hurricane and tornado), forest fire, snow, ice storm, extreme cold weather, dam break, volcano, and sand storm. • The set and magnitude of natural phenomena should be specific to the site. The criteria should include return cycle of the worst event which can be common to all SMR sites in that area. • Entire available history (paleo-science) should be taken into consideration.	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirement 17 and relevant Paragraphs. [11]
Prevention (1)	Magnitude of hazards	Ensure that trends and uncertainties are considered in determining the magnitude of natural phenomena which should be mitigated.	• The type and cause of natural hazards is site specific but it is better to consider large margin (highest in the region) depending on the site characteristics of the plant. • The magnitude should be predicted for a return cycle corresponding to the core damage frequency goal considering trends and uncertainties.	
Prevention (1)	Design adequacy	Confirm design adequacy of structures and components to stand against natural hazards.	• Seismic adequacy of SSCs which provide reactor shutdown, core cooling and decay heat removal, especially RPV integrity, is most important. • Usually SMRs have a small footprint for the containment and reactor building. Some SMRs adopt the seismic isolation option. In the case of re-evaluation of the seismic grade of the site (see NRC recommendation [15]), to some extent the isolators could be redesigned and substituted to allow more flexibility and to upgrade the seismic resistance of the containment and reactor building.	
Prevention (1)	Siting, location	Confirm that the choice of the site takes into account all probable natural hazards along with consequential cascade effects.	• The magnitude and return cycle should consider nearby dams and water sources failing in the worst manner. • Evaluate the system for more flooding from earthen dam breaks. • For a plant with air-cooled ultimate heat sink, the effect of flooding on the air cooling tower should be considered. • In case of underground site (positive solution for seismic), water tightness of safety critical structures, systems and	

TABLE 6 (cont.)

Defence in Depth level	Critical issues addressed	Option for counter-measures	Considerations for water cooled SMRs	Relevant safety requirements
			components (SSCs) and access paths (corridors, stairs, rooms, etc.) should be ensured. Attention should be paid to ensure underground lighting in emergency condition. • In case of locating safety critical SSCs at higher elevations, other resultant hazards should be addressed (e.g., aircraft crash, tornadoes, seismic, etc.). • Plants located in the vicinity of active volcanoes and seas, should consider extreme eruption impact as well as tsunami impact simultaneously.	
Control of accidents within DB(3)	Safety assessment	Include all extreme natural hazards in safety assessment.	• Perform periodic reassessments especially if extreme natural hazards occur in the plant site. • Safety assessment should account for post-accident actions.	
Control of accidents within DB(3)	Safety systems	Verify that protection systems have taken into account all extreme natural hazards.	Assure the reliability of automatic reactor trip on seismic signals.	
Control of accidents within DB(3)	Safety systems	Ensure that *cliff edge* effects are considered and addressed.	*Cliff edge* effects, where an incremental increase in magnitude causes a disproportionate increase in consequences, e.g., due to tsunami exceeding protective walls, can be avoided by use of passive safety systems (e.g., use of air cooling systems without dependence on DGs, etc.), or by incorporating suitable design solutions (water tight rooms, high elevation for critical safety systems, PARs, etc.).	

4.1.2. Consider issues concerning multiple-unit sites

Two or more reactor units can be built in one site. The benefit of placing several units in the same site is not only economical, but also provides the possibility to have electrical system cross connections among the units which are very useful in emergency situation (e.g. the Fukushima Daiichi complex benefited from twin units which allowed temporary cross connection of electrical system between Units 5 and 6). One unit's equipment and staff support could help others when abnormal conditions such as failure of EDGs occur.

However, a site with several units also faces potential major problems. The Fukushima Daiichi accident indicated that multiple reactor units in one site face the followings:

- *Unexpected problems*
  Hydrogen gas produced in Unit 3 due to the interaction of melting core and steam leaked to Unit 4 through shared venting system between the two units. The hydrogen then detonated in Unit 4, damaged the reactor building and distracted the emergency team in dealing with the Unit 1. Such an explosion was never foreseen. The explosion in Unit 4 influenced the emergency team to concentrate more on Unit 4 spent fuel pool, wrongly perceiving that the source of hydrogen explosion was due to uncovered fuel in the spent fuel pool.

- *Unexpected aftershock challenges*
  When multiple reactors are built at a site, unexpected challenges can happen as an accident in one unit may disrupt the operation and accident management of the neighboring reactors. In the Fukushima Daiichi accident when the emergency team was trying to cope with the situation in Units 1 and 2, the explosion in Unit 1 spread radioactive debris in the area around the complex. The explosion damaged the cables and mobile generators that had been installed to provide power to the standby liquid control pumps. The emergency teams had to work in challenging circumstances. They faced a mix of problems from stabilizing the reactor systems to protecting themselves from the radiation due to explosion in other units. The workers had to wear additional protective clothing and stay within time limitations thus limiting their mobility and availability.

- *The need to respond to all units concurrently strained all resources on-site*
  In multiple-units sites, there is a possibility that several reactors undergo concurrent accidents due to a common cause. When these simultaneous accidents occur, the resources to handle all abnormal conditions at the same time become strained. For example, when operators of Unit 3 requested a fire engine to be dispatched to prepare water injection, all of the site fire engines were being used to mitigate the ongoing problem in Unit 1. Earlier, requests for off-site fire engines were unsuccessful because the roads were impassable.

As the above findings reveal, it is recognized that the issues concerning multiple reactor sites and multiple sites must be addressed. The current design of SMRs typically offers multiple reactor modules in one plant which ranges from two (2) to twelve (12) modules. So it is important to consider the issues of countermeasures shown in the *TABLE 7.*

Applicable Safety Requirements concerning multiple-unit sites [11]

Requirement 33 of SSR–2/1 (Rev. 1) establishes the following requirements on safety systems, and safety features for design extension conditions, of units of a multiple unit nuclear power plant.

"Each unit of a multiple unit nuclear power plant shall have its own safety systems and shall have its own safety features for design extension conditions".

Paragraph 5.63 requires that means allowing interconnections between units of a multiple unit nuclear power plant shall be considered in the design to further enhance safety.

TABLE 7. ISSUES CONCERNING MULTIPLE-UNIT SITES

Defence in Depth level	Critical issues addressed	Options for countermeasures	Considerations for water cooled SMRs	Relevant safety requirement
Prevention (1)	Common cause failure	Ensure that the common cause failure and related accident management concerns are considered in the design.	• Multiple unit threat is particularly applicable to modular reactors. Some SMRs are proposed in multiple units. Regulatory body should require safety assessment for all units on the site as a whole. • Some safety related SSC could be interconnected between units in order to supply endangered units with vital assistance under external hazards. Two-unit plant, for example, would be more reliable than one unit. Safety assessment should take into account the suitability of sharing and cross connections, its vulnerability and benefits. • Provide cross connection between units with the reliable isolation capability. • Each module must be capable to cope with each type of accident.	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirement 33 and relevant Paragraphs. [11]
Control of accidents within DB(3)	Safety systems	Ensure that countermeasures can be carried out for a unit where meltdown occurs and accident dose rate increases beyond analysis limits.	• Enhance the containment HYDROGEN control, using cooling, venting and filtering. • Provide shielding, convenient access and remote operations of countermeasures. • Assure crew can execute severe accident management guidelines (SAMG) without exceeding personnel dose.	

4.1.3. Enhance off-site and on-site electricity supplies

Nuclear power plants are both electricity generators and users. They supply large amount of electric energy to the grid as well as rely on it to receive power for essential safety operations, especially during emergency conditions. The off-site power system is the preferred source of power for the plant, particularly for the reactor system and ESFs during normal, abnormal, and accident conditions. The reliability of off-site power is assured by two or more physically independent circuits from the transmission network. The reliability of on-site power is also enhanced with sufficient independence, redundancy and testability of EDGs and batteries. All nuclear power plants are designed to cope with loss of off-site power by tripping the reactor and turbine, and automatically starting the on-site emergency generators to provide vital energy for residual heat removal until normal off-site power is restored.

For existing nuclear reactors the availability of off-site and on-site power supply is crucial as their safety systems mostly rely on active systems which need electrical power to operate. In that regards, the following facts during the Fukushima Daiichi accident further underline the importance of ensuring off-site and on-site electricity supply.

- All EDGs started as designed after off-site power loss caused by earthquake. However, they did not operate for long. When the tsunami hit the complex, seawater filled the DG building and all generators in the operating units were crippled beyond reasonable recovery.
- Elevation of switchboards was low in the building with little margin for flooding. This contributed adversely to the consequence of the accident. A submerged switchboard made electricity power recovery impossible for a long time. Before the Fukushima Daiichi nuclear accident, a complete loss of all switchboards for an extended period was not sufficiently considered. A complete loss of DC power was not addressed either. This made the duration of station blackout (SBO) unpredictable prior to the event.
- It was discovered that a signal generated by loss of all DC power caused IC valves to close on LOCA signal and disabled the passive core cooling system. This shows the important role of DC power in ensuring the establishment of passive system, and that this was essential for the Fukushima Daiichi nuclear plants.
- SMR designers should consider DC power unavailability and ensure core cooling and decay heat removal.

Reliance on active systems for safety functions has been greatly reduced in SMRs designs. Passive cooling systems with a long grace period for DBAs provide improvement over existing reactor designs. However, attention should be paid to the supply of power, especially for severe accident management where the lighting and monitoring systems are needed to cope with the situation.

In order to deal with the above facts, some options of countermeasures to enhance off-site and on-site power are described in *TABLE 8*.

Applicable Safety Requirements on off-site and on-site electricity supplies [11]

Requirement 68 of SSR–2/1 (Rev. 1) establishes the following requirements on design for withstanding the loss of off-site power

"The design of a nuclear power plant shall include an emergency power supply capable of supplying the necessary power in anticipated operational occurrences and design basis accidents, in the event of the loss of off-site power. The design shall include an alternate power source to supply the necessary power in design extension conditions".

Paragraph 6.43 requires that the design specifications for the emergency power supply and for the alternate power source at the nuclear power plant due the requirements for capability, availability, duration of the required power supply, capacity and continuity.

Paragraph 6.44 requires that the combined means to provide emergency power (such as water, steam or gas turbines, diesel engines or batteries) shall have a reliability and type that are consistent with all the requirements of the safety systems to be supplied with power, and their functional capability shall be testable.

Paragraph 6.44a requires that the alternate power source shall be capable of supplying the necessary power to preserve the integrity of the reactor coolant system and to prevent significant damage to the core and to spent fuel in the event of the loss of off-site power combined with failure of the emergency power supply.

Paragraph 6.44b requires that equipment that is necessary to mitigate the consequences of melting of the reactor core shall be capable of being supplied by any of the available power sources.

Paragraph 6.44c requires that the alternate power source shall be independent of and physically separated from the emergency power supply. The connection time of the alternate power source shall be consistent with the depletion time of the battery.

Paragraph 6.44d requires that continuity of power for the monitoring of the key plant parameters, and for the completion of short term actions necessary for safety shall be maintained in the event of a loss of the AC (Alternating Current) power sources.

Paragraph 6.45 requires that the design basis for any diesel engine or other prime mover that provides an emergency power supply to items important to safety shall include:

a) The capability of the associated fuel oil storage and supply systems to satisfy the demand within the specified time period;
b) The capability of the prime mover to start and to function successfully under all specified conditions and at the required time;
c) Auxiliary systems of the prime mover, such as coolant systems.

Paragraph 6.45a requires that the design shall also include features to enable the safe use of non-permanent equipment to restore the necessary electrical power supply.

TABLE 8. ENHANCED OFF-SITE AND ON-SITE ELECTRICITY SUPPLIES

Defence in Depth level	**Critical issues addressed**	**Options for countermeasures**	**Considerations for water cooled SMRs**	**Relevant safety requirements**
Prevention (1)	Equipment readiness	Prepare equipment required to respond to a long term loss of all AC and DC power	The equipment should be conveniently staged, protected, and maintained such that it is always ready for use if needed	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirement 68 and relevant Paragraphs. [11]
Prevention (1)	Equipment location	Consider all hazards in locating electrical equipment.	All scenarios of accident progression, after an initiating internal or external accident, should be considered in locating the electrical equipment safely.	
Prevention (1)	Protection of EDGs (to be used for safety functions)	Protect EDGs from all extreme natural hazards.	• EDGs and associated emergency power supplies should be located at higher elevations considering flooding or their location should be enclosed by water tight and seismically qualified enclosures (e.g., water proof doors and penetrations). • Cooling, combustion air-intake, and exhaust pipes of the EDGs should be protected while ensuring opening of the pipes to atmosphere. • Ensure protection of fuel tank for the EDGs.	
Prevention	Emergency	Ensure electricity	• Use separate redundant DC systems for	

TABLE 8 (cont.)

Defence in Depth level	Critical issues addressed	Options for countermeasures	Considerations for water cooled SMRs	Relevant safety requirements
(1)	power supplies	supplies for safety systems.	various safety functions, e.g., for reactor safety monitoring instruments, power system actuation, valves and motors motive power, etc., with complete electrical and physical separation. • Provide diverse DC power sources (such as diesel driven DC generators) [17]. • Use diversity in power sources functionality (e.g., water cooled EDGs as well as air-cooled EDGs) [17]. • In addition portable generators and batteries should be considered for extreme situations, if those proposed above fail inadvertently.	
Mitigation of severe accident (4)	Complete Power recovery	Ensure quick recovery processes for all power supplies during severe accidents.	Perform analysis to reduce the time to restore off-site and on-site supplies after their loss in severe accidents (e.g., by means of design provisions, administrative actions, and their combination thereof).	
Prevention (1)	Switch-boards	Ensure that the switchboards availability is maintained during all extreme events.	Provide sealed compartments to secure components and power panels in case of flooding [17].	
Control of accidents within DB(3)	Electricity supply	Ensure that operator action and electricity supply are not needed during grace period.	• Usually SMRs rely more on passive safety features to the extent possible, thus on-site and off-site electricity may not be needed for the grace period. DGs could be avoided, at least as safety systems. • The assurance of a suitable grace period (without need of intervention from operators and need of electricity supplies) should be duly considered.	
Control of accidents within DB (3)	Electricity supply	Ensure reliability of valves in passive system.	Provide combination of manual and electrically driven valves for safety functions as well as fail-to-safe features.	
Control of accidents within DB (3)	Electricity supply	Maximize survivability of AC and DC systems and components [17].	• Provide manual AC and DC cross connection capability between adjacent units [18]. • Provide access and connections for prestaged power sources [18].	

4.1.4. Ensure robust measures for reactor core cooling and ultimate heat sinks

Continuous core cooling is crucial in water cooled reactors during normal operation, shutdown conditions and in emergency situations. In normal operation it transfers the core thermal power from the fuel cladding and uses it to generate steam. When the reactor is shut down the core decay heat should be removed to maintain the fuel clad temperature below acceptable limits. During DBAs, e.g. LOCA, emergency core cooling must be provided to prevent core degradation by maintaining the fuel cladding temperature below its melting point. Integrity of fuel clad ensures that the radioactive fission products remain within the first barrier which is the foremost and the fundamental safety requirement. Due to this, a robust design of core cooling systems is essential.

In the early phase of the Fukushima Daiichi accident, loss of off-site power did not interrupt the core cooling systems and they worked as designed with the power supplied by EDGs. Later however, the EDGs were crippled by tsunami and the successive cooling systems degraded and failed. In Unit 1 the indicators in the control room for IC failed. In Units 2 and 3, the steam-driven systems, namely, RCIC and HPCI kept working for a few days. A loss of core cooling systems and failure to provide core cooling by other means lead to severe core damage. Based on these lessons, the following *TABLE 9* recommends options for the countermeasures.

Applicable Safety Requirements for reactor core cooling and ultimate heat sinks [11]

Requirement 47 of SSR–2/1 (Rev. 1) establishes the following requirements on design of reactor coolant systems

"The components of the reactor coolant systems for the nuclear power plant shall be designed and constructed so that the risk of faults due to inadequate quality of materials, inadequate design standards, insufficient capability for inspection or inadequate quality of manufacture is minimized".

Paragraph 6.13 requires that pipework connected to the pressure boundary of the reactor coolant systems for the nuclear power plant shall be equipped with adequate isolation devices to limit any loss of radioactive fluid (primary coolant) and to prevent the loss of coolant through interfacing systems.

Paragraph 6.14 requires that the design of the reactor coolant pressure boundary shall be such that flaws are very unlikely to be initiated, and any flaws that are initiated would propagate in a regime of high resistance to unstable fracture and to rapid crack propagation, thereby permitting the timely detection of flaws.

Paragraph 6.15 requires that the design of the reactor coolant systems shall be such as to ensure that plant states in which components of the reactor coolant pressure boundary could exhibit embrittlement are avoided.

Paragraph 6.16 requires that the design of the components contained inside the reactor coolant pressure boundary, such as pump impellers and valve parts, shall be such as to minimize the likelihood of failure and consequential damage to other components of the primary coolant system that are important to safety, in all operational states and in design basis accident conditions, with due allowance made for deterioration that might occur in service.

Requirement 51 establishes the following requirements on removal of residual heat from the reactor core

"Means shall be provided for the removal of residual heat from the reactor core in the shutdown state of the nuclear power plant such that the design limits for fuel, the reactor coolant pressure boundary and structures important to safety are not exceeded".

Requirement 52 establishes the following requirements on emergency cooling of the reactor core

"Means of cooling the reactor core shall be provided to restore and maintain cooling of the fuel under accident conditions at the nuclear power plant even if the integrity of the pressure boundary of the primary coolant system is not maintained".

Paragraph 6.18 requires that the means provided for cooling of the reactor core shall be such as to ensure that:

a) The limiting parameters for the cladding or for integrity of the fuel (such as temperature) will not be exceeded;

b) Possible chemical reactions are kept to an acceptable level;

c) The effectiveness of the means of cooling of the reactor core compensates for possible changes in the fuel and in the internal geometry of the reactor core;

d) Cooling of the reactor core will be ensured for a sufficient time.

Paragraph 6.19 requires that design features (such as leak detection systems, appropriate interconnections and capabilities for isolation) and suitable redundancy and diversity shall be provided to fulfil the requirements of para. 6.18 with adequate reliability for each postulated initiating event.

Requirement 53: Heat transfer to an ultimate heat sink Requirement

"The capability to transfer heat to an ultimate heat sink shall be ensured for all plant states".

Paragraph 6.19a requires that systems for transferring heat shall have adequate reliability for the plant states in which they have to fulfil the heat transfer function. This may require the use of a different ultimate heat sink or different access to the ultimate heat sink.

Paragraph 6.19b requires that the heat transfer function shall be fulfilled for levels of natural hazards more severe than those to be considered for design taking into account the site hazard evaluation.

TABLE 9. ENSURE ROBUST MEASURES FOR REACTOR CORE COOLING AND ULTIMATE HEAT SINK

Defence in Depth level	**Critical issues addressed**	**Options for countermeasures**	**Considerations**	**Relevant safety requirements**
Control of accident within DB(3)	Safety systems	Confirm the reliability and capability of cooling systems to cool the core after natural hazards occurrences.	• Provide at least two success path to cope with the accident using any combination of passive, active, and manually aligned systems. • Confirm the core injection flow rate of coolant until a balance is reached between decay heat and decay heat removal capability (e.g., providing	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants:

TABLE 9 (cont.)

Defence in Depth level	Critical issues addressed	Options for countermeasures	Considerations	Relevant safety requirements
			safe and reliable depressurisation capability to achieve coolant injection and cool down; providing diverse operating mechanisms for containment vent valves). • Incorporate diversity for alternate water injection capabilities and supplement cooling capability with portable pumps [17]. • Provide diversity of heat sink through use of portable heat removal systems [17] . • Consider any guidance for using seawater, and effects of salinity.	Design Requirements 47, 51, 52, 53 and relevant Paragraphs. [11]
Control of Accidents within DB (3)	SBO	Confirm that SBO can be managed for long time.	• Provide passive cooling systems which are able to function for extended or indefinite period. • Consider portable systems for added margin for long term core cooling.	
Control of accidents within DB(3)	Safety systems	Maximize survivability of reactor cooling capabilities [16].	• The status of all modes of core cooling should be available in control room under all plant conditions [18]. • Provide access and connections for prestaged pumps and alternate water sources [18]. • Provide adjacent SMR units cross connection capability for critical process systems fluids (treated water, raw water, air) [18].	

4.1.5. Enhance design of safety-related structures, systems and components

Reactor design should guarantee that all safety related structures, system and components (SSCs) survive in all accident conditions. They must be accessible through remote control, and if it fails, through manual operation. In the Fukushima Daiichi accident several safety-related SSCs lost functions due to the failure of remote control after loss of all power. Although all control rods were successfully inserted and passive systems such as IC and RCIC started operation as designed at the beginning of the accident, the plant failed to maintain safe shutdown cooling after tsunami. The containment vessel integrity was lost in Units 1, 2 and 3, causing hydrogen gas leakage to the reactor buildings. The operator efforts to control the containment pressure did not succeed because access to some manual valves was hindered due to high radiation levels.

It is believed that if the IC had remained in service, with water makeup to pool, the Unit 1 core damage would have been avoided or at least delayed. Besides, the Units 2 and 3

needed implementation of feed and bleed cooling to avoid core damage but the effort was limited by lack of water supply and power to steam driven pump.

In SMRs such failures of safety related structures, systems and components should be prevented or accommodated with compensatory measures. The following *TABLE 10* explains some considerations.

Applicable Safety Requirements for safety-related structures, systems and components [11]

Requirement 45 of SSR–2/1 (Rev. 1) establishes the following requirements on control of the reactor core.

"Distributions of neutron flux that can arise in any state of the reactor core in the nuclear power plant, including states arising after shutdown and during or after refuelling, and states arising from anticipated operational occurrences and from accident conditions not involving degradation of the reactor core, shall be inherently stable. The demands made on the control system for maintaining the shapes, levels and stability of the neutron flux within specified design limits in all operational states shall be minimized".

Requirement 46 establishes the following requirements on reactor shutdown.

"Means shall be provided to ensure that there is a capability to shut down the reactor of the nuclear power plant in operational states and in accident conditions, and that the shutdown condition can be maintained even for the most reactive conditions of the reactor core".

Requirement 48 establishes the following requirements on overpressure protection of the reactor coolant pressure boundary.

"Provision shall be made to ensure that the operation of pressure relief devices will protect the pressure boundary of the reactor coolant systems against overpressure and will not lead to the release of radioactive material from the nuclear power plant directly to the environment".

Requirement 49 establishes the following requirements on inventory of reactor coolant.

"Provision shall be made for controlling the inventory, temperature and pressure of the reactor coolant to ensure that specified design limits are not exceeded in any operational state of the nuclear power plant, with due account taken of volumetric changes and leakage".

Requirement 55 establishes the following requirements on control of radioactive releases from the containment.

"The design of the containment shall be such as to ensure that any release of radioactive material from the nuclear power plant to the environment is as low as reasonably achievable, is below the authorized limits on discharges in operational states and is below acceptable limits in accident conditions".

Requirement 58 establishes the following requirements on control of containment conditions.

"Provision shall be made to control the pressure and temperature in the containment at a nuclear power plant and to control any build-up of fission products or other gaseous,

liquid or solid substances that might be released inside the containment and that could affect the operation of systems important to safety".

TABLE 10. ENHANCING DESIGN OF SAFETY-RELATED STRUCTURES, SYSTEMS AND COMPONENTS

Defence in Depth level	Critical issues addressed	Options for countermeasures	Considerations or engineered features to be added and/or Modified	Relevant safety requirements
Prevention (1)	SSC design	Protect SSCs from extreme natural hazards.	Provide design solutions to protecting SSCs from all hazards (e.g., flooding, volcanic ash, desert sand).	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirements 45, 46, 48. 49. 55. 58 and relevant Paragraphs. [11]
Prevention (1)	SSC design	Prevent primary containment vessel (PCV) damage caused by elevated temperature.	• Enhance the PCV cooling system [17] (e.g., by passive system, followed by pump supplied spray system for extended service). • Provide diverse cooling systems for containment and provision for connecting portable cooling equipment.	
Prevention (1), Control of accidents within DB (3)	SSC design	Exploit positive features of SMR in the design of safety related SSCs.	• SMRs can implement the safety related SSCs needed to cope with containment vessel integrity, diverse shutdown, core cooling and decay heat removal, with positive features easily exploitable by: reduced decay heat and source term (due to small size); wide use of passive safety features (e.g., air cooling or externally cooling containment vessel, large amount of water per MW(th), compared to large reactors); reduced radiation field (internal shielding); positioning of manually activate safety components/ systems in suitable areas protected from known hazards, and in-vessel core retention and cooling. *Notes:* Once through cooling system should be addressed and possibly eliminated, especially in the case of underground siting. • Plant designs should consider installation of air-cooled EDGs and cross-connections between units to allow sharing of AC and DC power, fresh- and seawater, and compressed air systems during emergencies.	
Prevention (1) and Mitigation of	SSC design	Ensure that design features are flexible to deal with extreme natural hazards	• Locate installed and portable accident mitigation equipment in hardened and protected enclosures fulfilling	

TABLE 10 (cont.)

Defence in Depth level	Critical issues addressed	Options for countermeasures	Considerations or engineered features to be added and/or Modified	Relevant safety requirements
severe accident (4)		including fires, as well as man-made threats (sabotage or terror).	applicable safety requirements. • Portable equipment storage should be off-site, far enough to avoid exposure to the on-site hazards, with transportation plans in place.	
Control of accidents within DB(3)	Redundancy	Ensure that the design covers cases where the primary coping system/function fails. This should be the case for all scenarios.	The design should not depend on any single active or passive system or function. The redundant backup should be diverse, such that if the primary system/function fails for some reason, the backup system/function remains available and it should not fail for the same reason.	
Prevention (1) and mitigation of severe accidents (4)	Connecting ability	Ensure capability to connect safety related SSCs with external equipment.	• Provide redundant exterior "hardened" connections to supply water to reactor with reliable/manual venting for boil off (including de-pressurizing of the reactor/ primary system). • Provide, normally isolated external manual connections manual to be used by portable equipment for: -Cooling and spraying -Filtering and HYDROGEN recombination	
Prevention (1) and mitigation of severe accident (4)	Accessi-bility	Ensure that safety related SSC can be accessed during emergencies.	• One of the lessons learned was that the operators could not access the containment vent valves to open them, because the motor control centre (MCC's) were flooded and valves exceeded their design qualification conditions due to the severe accident condition. • Redundant and diverse systems/equipment should be placed exterior to the main buildings. Shielded/remote-operated manual mechanisms should be provided to allow actuation in case of severe accidents or extensive site damage, to re-establish core cooling. Mechanical remote operation may use hydraulic, pneumatic, fluidic means instead of electrical power to allow more reliable operation with diverse function.	
Control of accidents	Safety	Ensure that all of the safety functions are	• Survivability and reliability of emergency power supply system	

TABLE 10 (cont.)

Defence in Depth level	Critical issues addressed	Options for countermeasures	Considerations or engineered features to be added and/or Modified	Relevant safety requirements
within DB(3)	function	restored during the grace period.	should be enhanced to cope with extreme hazards including flooding. • Use high capacity compact batteries for SBO. Passive safety features should be considered to the maximum extent possible.	
Control of accidents within DB(3)	Leakages	Ensure that leaks from the reactor coolant system are monitored to bring in mitigating efforts quickly.	Monitoring instrumentations and sensors for coolant leak or hydrogen release should be reinforced.	

4.1.6. Ensure measures for prevention and mitigation of hydrogen explosions

High temperature steam and zircalloy cladding chemical interaction, when the overheated core is uncovered produces huge amount of hydrogen gas. Hydrogen is highly reactive when it is from 4 to 90% in the containment air mixture with oxygen. It detonates in this concentration range. Anticipating its presence, BWR designs provide means to control oxygen concentration in the containment; such as by inerting the containment air with nitrogen or installing containment atmosphere dilution system to maintain low oxygen concentration (less than 5%) [19]. During the Fukushima Daiichi accident, ex-containment hydrogen explosions came as a surprise. The explosion damaged the reactor buildings of Units 1, 3 and 4. It was suspected that hydrogen leakages occurred from the primary containment to reactor building through several passages such as top head manhole, top head flange, piping penetration, airlock for personnel, suppression chamber manhole, electric wiring penetration, equipment hatch, vent tubes, etc. Extensive damage and debris from the explosion created significant logistical difficulties and inhibited response actions. The design needs to put more attention on these matters and must ensure measures for prevention and mitigation of hydrogen explosions. In the *TABLE 11* below are options for countermeasures.

Applicable Safety Requirements for prevention and mitigation of hydrogen explosions [11]

Requirement 58 of SSR–2/1 (Rev. 1) establishes the following requirements on control of containment conditions.

"Provision shall be made to control the pressure and temperature in the containment at a nuclear power plant and to control any build-up of fission products or other gaseous, liquid or solid substances that might be released inside the containment and that could affect the operation of systems important to safety".

Paragraph 6.27 requires that the design shall provide for sufficient flow routes between separate compartments inside the containment. The cross-sections of openings between compartments shall be of such dimensions as to ensure that the pressure differentials occurring during pressure equalization in accident conditions do not result in

unacceptable damage to the pressure bearing structure or to systems that are important in mitigating the effects of accident conditions.

Paragraph 6.28 requires that the capability to remove heat from the containment shall be ensured, in order to reduce the pressure and temperature in the containment, and to maintain them at acceptably low levels after any accidental release of high energy fluids. The systems performing the function of removal of heat from the containment shall have sufficient reliability and redundancy to ensure that this function can be fulfilled.

Paragraph 6.28a requires that design provision shall be made to prevent the loss of the containment structural integrity in all plant states. The use of this provision shall not lead to early or to large radioactive releases.

Paragraph 6.28b requires that the design shall also include features to enable the safe use of non-permanent equipment for restoring the capability to remove heat from the containment.

Paragraph 6.29 requires that design features to control fission products, hydrogen, oxygen and other substances that might be released into the containment shall be provided as necessary:

a) To reduce the amounts of fission products that could be released to the environment in accident conditions;

b) To control the concentrations of hydrogen, oxygen and other substances in the containment atmosphere in accident conditions so as to prevent deflagration or detonation loads that could challenge the integrity of the containment.

Paragraph 6.30 requires that coverings, thermal insulations and coatings for components and structures within the containment system shall be carefully selected and methods for their application shall be specified to ensure the fulfilment of their safety functions and to minimize interference with other safety functions in the event of deterioration of the coverings, thermal insulations and coatings.

TABLE 11. ENSURING MEASURES FOR PREVENTION AND MITIGATION OF HYDROGEN EXPLOSIONS

Defence in Depth level	**Critical issues addressed**	**Options for countermeasures**	**Considerations for water cooled SMR**	**Relevant safety requirements**
Prevention and mitigation of severe accident (4).	Hydrogen control	Control hydrogen concentration in containment.	• Include hydrogen re-combiners in containment design. • Implement PAR, ignitor, nitrogen gas to reduce the concentration of oxygen to less than 5%. • Hydrogen monitoring should be provided as well as means for venting at appropriate stage. • Provide early venting mana-gement to prevent accumulation of hydrogen.	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirement 58 and relevant
Prevention	Hydrogen	Reduce hydrogen production during	Consider in the future fuel designs, the replacement of zirconium alloy	

TABLE 11 (cont.)

Defence in Depth level	Critical issues addressed	Options for countermeasures	Considerations for water cooled SMR	Relevant safety requirements
(1)	control	accident by design.	cladding by ceramic compounds, such as silicon carbide (SiC).	Paragraphs. [11]
Control of accidents within DB (3)	Hydrogen control	Prevent core melt accidents.	Provide adequate alternative heat removal systems to assure that core melt can be avoided.	

4.1.7. Enhance containment venting and filtering system

The BWR venting system is part of containment system which plays an important role to control the pressure of dry well and wet well during abnormal conditions. For reactors with Mark I containments, there are several locations of venting as can be seen in *FIG. 24*. During the Fukushima Daiichi accident, the remote venting could not be implemented properly from the control room due to unavailability of electric power, and local manual venting was delayed due to several practical difficulties such as:

- High radiation dose;
- Lack of control air;
- Lack of DC power;
- Procedure approval;
- Lack of command and control;
- Lack of lighting; and
- Communication difficulties.

In addition, there were no alternative power sources for instrumentation and control for over 30 minutes and no SAMG were written for this situation. As venting is an important aspect in coping with the Fukushima Daiichi type accident, the following considerations given in *TABLE 12* are presented to deal with this type of accident.

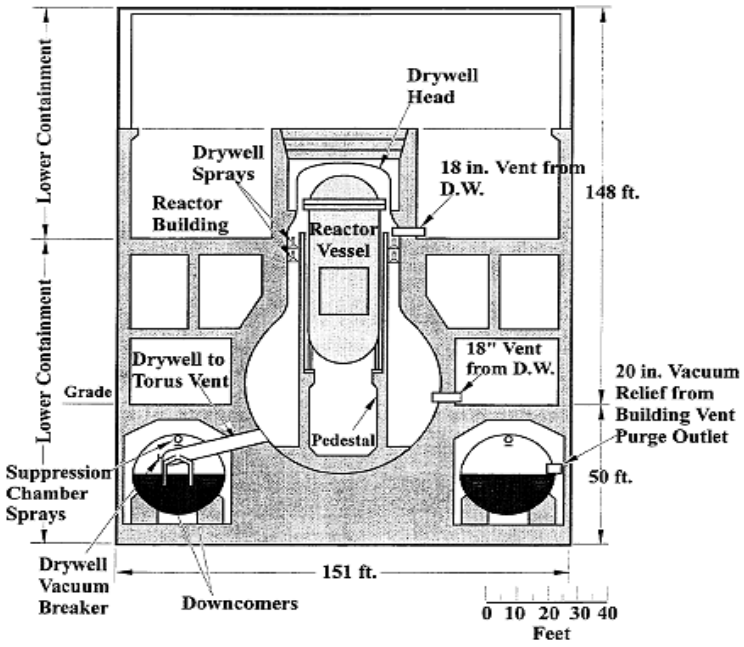

FIG. 24. BWR venting locations (Reproduced courtesy of General Electric company).

Applicable Safety Requirements for containment venting and filtering system [11]

Requirement 73 of SSR2/1 (Rev. 1) establishes the following requirements on air conditioning systems and ventilation systems

"Systems for air conditioning, air heating, air cooling and ventilation shall be provided as appropriate in auxiliary rooms or other areas at the nuclear power plant to maintain the required environmental conditions for systems and components important to safety in all plant states".

Paragraph 6.48 requires that systems shall be provided for the ventilation of buildings at the nuclear power plant with appropriate capability for the cleaning of air:

a) To prevent unacceptable dispersion of airborne radioactive substances within the plant;

b) To reduce the concentration of airborne radioactive substances to levels compatible with the need for access by personnel to the area;

c) To keep the levels of airborne radioactive substances in the plant below authorized limits and as low as reasonably achievable;

d) To ventilate rooms containing inert gases or noxious gases without impairing the capability to control radioactive effluents;

e) To control releases of gaseous radioactive material to the environment below the authorized limits on discharges and to keep them as low as reasonably achievable.

TABLE 12. CONSIDERATIONS TO ENHANCE THE CONTAINMENT VENTING AND FILTERING SYSTEM

Defence in Depth level	Critical issues addressed	Options for countermeasures	Considerations or engineered features to be added and/or modified	Relevant safety requirements
Mitigation of severe accidents (4)	Vent design	Ensure that the vent design is hardened and capable to allow safe depressurization.	• The vent system should be constructed to accommodate a permissible flow of steam/air mixture. The system should be able to reduce pressure inside the reactor before core uncovery (e.g., 1 hour for BWR [21]). • Plant designs should support timely venting of primary containment even with a loss of power and motive force, such as compressed air. • The installation of manual vents in each reactor building may be prudent to allow venting of any hydrogen that may have accumulated.	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirement 73 and relevant Paragraphs. [11]
Mitigation of severe accidents (4)	Radioactive release control	Avoid radioactive release during venting.	Retrofit the vent system with radioactivity filters to reduce the pressure and hydrogen level without releasing large amounts of fission products [22].	
Mitigation of severe accident (4)	Vent design	Ensure each reactor unit has independent venting system.	Provide dedicated vents for each unit, or make provisions to prevent backflow between units [18].	
Mitigation of severe accident (4)	Vent design	Ensure ability to accomplish local manual venting [18].	Provide venting valves that are accessible for manual operation.	

4.1.8. Ensure hardened instrumentation and cables for safety-related parameters and monitoring equipment

In the Fukushima Daiichi accident, after the accident progressed into severe accident, instruments and monitoring equipment were exposed to beyond design basis environment. Some unreliable measurements were observed, such as reactor water level of Unit 1 and suppression chamber pressure of Unit 2. As a result it hampered operators to understand the conditions and respond correctly.

Learning from such experience, *TABLE 13* lists the options of countermeasure to ensure safety parameter and monitoring equipment functions.

Applicable Safety Requirements for instrumentation and cables for safety-related parameters and monitoring equipment [11]

Requirement 59 of SSR–2/1 (Rev. 1) establishes the following requirements on provision of instrumentation.

"Instrumentation shall be provided for determining the values of all the main variables that can affect the fission process, the integrity of the reactor core, the reactor coolant systems and the containment at the nuclear power plant, for obtaining essential information on the plant that is necessary for its safe and reliable operation, for determining the status of the plant in accident conditions and for making decisions for the purposes of accident management".

Paragraph 6.31 requires that instrumentation and recording equipment shall be provided to ensure that essential information is available for monitoring the status of essential equipment and the course of accidents, for predicting the locations of release and the amount of radioactive material that could be released from the locations that are so intended in the design, and for post-accident analysis.

Requirement 62 of SSR-2/1 (Rev. 1) establishes the following requirements on reliability and testability of instrumentation and control systems.

"Instrumentation and control systems for items important to safety at the nuclear power plant shall be designed for high functional reliability and periodic testability commensurate with the safety function(s) to be performed".

Paragraph 6.34 requires that design techniques such as testability, including a self-checking capability where necessary, fail-safe characteristics, functional diversity and diversity in component design and in concepts of operation shall be used to the extent practicable to prevent loss of a safety function.

Paragraph 6.35 requires that safety systems shall be designed to permit periodic testing of their functionality when the plant is in operation, including the possibility of testing channels independently for the detection of failures and losses of redundancy. The design shall permit all aspects of functionality testing for the sensor, the input signal, the final actuator and the display.

Paragraph 6.36 requires that when a safety system, or part of a safety system, has to be taken out of service for testing, adequate provision shall be made for the clear indication of any protection system bypasses that are necessary for the duration of the testing or maintenance activities.

Requirement 71 establishes the following requirements on process sampling systems and post-accident sampling systems.

"Process sampling systems and post-accident sampling systems shall be provided for determining, in a timely manner, the concentration of specified radionuclides in fluid process systems, and in gas and liquid samples taken from systems or from the environment, in all operational states and in accident conditions at the nuclear power plant".

TABLE 13. ENSURING HARDENED INSTRUMENTATION AND CABLES FOR SAFETY–RELATED PARAMETERS AND MONITORING EQUIPMENT

Defence in Depth level	Critical issues addressed	Options for countermeasures	Considerations or engineered features to be added and/or modified	Relevant safety requirements
Mitigation of severe accidents (4)	Hardened monitoring system	Ensure post-accident instrumentation for safety related parameters and monitoring equipments [22].	• Provide alternate DC Power for post-accident monitoring system to ensure its availability in case of loss of DC Power along with station blackout. The system and cables should be placed and run in hardened housings to withstand severe accidents [23]. • Enhance the instrumentation in the Reactor pressure vessel to provide the operator with better and diverse monitoring means about the course of a core degradation during postulated design base and sever accidents	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirement 59, 62 and relevant Paragraphs. [11]

4.1.9. Enhanced robustness of spent fuel cooling

In the Fukushima Daiichi accident, hydrogen explosion occured in the reactor building of Unit 4. The failure of spent fuel pool (SFP) monitoring system put the operators in the dark about the actual status. Due to this, after the explosion the emergency team had a consideration that the pool cooling system had failed or lost capability and the fuel assemblies were uncovered leading to overheating and fuel cladding-steam reaction producing hydrogen that caused detonation. It was later learnt that the hydrogen came from a back-flow of Unit 3 as the containment vent exhaust pipings of Units 3 and 4 were connected.

Lack of knowledge about the SFP conditions created confusion and distraction. Efforts were placed to reduce the temperature of the SFP by water spray from above, i.e. using helicopters and fire engines. Investigation showed that the fuel remained covered during the accident and no spent fuel was damaged. This shows the necessity to enhance the robustness of SFP cooling and its monitoring.

Spent fuel contains large inventory of radioactivity. Most nuclear plants store and manage their spent fuels in a pool inside the plants. As the fuels still produce heat during their decaying phase, the cooling system for the pool is important to control the spent fuel temperature. Hence, it is necessary to provide a robust and reliable cooling system for the SFP. Moreover, the water level and temperature of SFP must be constantly monitored as failure of the cooling system could lead to water vaporization and fuel uncovery.

In some water cooled SMR designs, spent fuels are handled in a particular manner depending on where the refueling mode is performed. For example, some SMRs do not require on site SFP as the refueling is performed through a replacement of the whole module and the fuel is taken out in the vendor workshop which may be far away from the site. However, for design with conventional onsite refueling method, the issues of SFP should be considered as proposed in *TABLE 14* below.

Applicable Safety Requirements for spent fuel cooling [11]

Requirement 80 of SSR-2/1 (Rev. 1) establishes the following requirements on fuel handling and storage systems.

"Fuel handling and storage systems shall be provided at the nuclear power plant to ensure that the integrity and properties of the fuel are maintained at all times during fuel handling and storage".

Paragraph 6.64 requires that the design of the plant shall incorporate appropriate features to facilitate the lifting, movement and handling of fresh fuel and spent fuel.

Paragraph 6.65 requires that the design of the plant shall be such as to prevent any significant damage to items important to safety during the transfer of fuel or casks, or in the event of fuel or casks being dropped.

Paragraph 6.66 requires that the fuel handling and storage systems for irradiated and non-irradiated fuel shall be designed:

a) To prevent criticality by a specified margin, by physical means or by means of physical processes, and preferably by use of geometrically safe configurations, even under conditions of optimum moderation;

b) To permit inspection of the fuel;

c) To permit maintenance, periodic inspection and testing of components important to safety;

d) To prevent damage to the fuel;

e) To prevent the dropping of fuel in transit;

f) To provide for the identification of individual fuel assemblies (g) To provide proper means for meeting the relevant requirements for radiation protection;

g) To ensure that adequate operating procedure and a system of accounting for, and control of, nuclear fuel can be implemented to prevent any loss of, or loss of control over, nuclear fuel.

Paragraph 6.67 requires that the fuel handling and storage systems for irradiated fuel shall be designed:

a) To permit adequate removal of heat from the fuel in operational states and in accident conditions;

b) To prevent the dropping of spent fuel in transit;

c) To prevent causing unacceptable handling stresses on fuel elements or fuel assemblies;

d) To prevent the potentially damaging dropping on the fuel of heavy objects such as spent fuel casks, cranes or other objects;

e) To permit safe keeping of suspect or damaged fuel elements or fuel assemblies;

f) To control levels of soluble absorber if this is used for criticality safety;

g) To facilitate maintenance and future decommissioning of fuel handling and storage facilities;

h) To facilitate decontamination of fuel handling and storage areas and equipment when necessary;

i) To accommodate, with adequate margins, all the fuel removed from the reactor in accordance with the strategy for core management that is foreseen and the amount of fuel in the full reactor core;

j) To facilitate the removal of fuel from storage and its preparation for offsite transport.

Paragraph 6.68 requires that the design shall prevent the uncovering of fuel assemblies for reactors using a water pool system for fuel storage in all plant states that are of relevance for the spent fuel pool, so as to practically eliminate the possibility of early or large radioactive releases and to avoid high radiation fields on the site. The design of the plant:

a) Shall provide the necessary fuel cooling capabilities;

b) Shall provide features to prevent the uncovering of fuel assemblies in the event of a leak or a pipe break;

c) Shall provide a capability to restore the water inventory.

The design shall also include features to enable the safe use of non-permanent equipment to ensure sufficient water inventory for the long term cooling of spent fuel and for providing shielding against radiation.

Paragraph 6.68a requires that the design shall include the following:

a) Means for monitoring and controlling the water temperature for operational states and for accident conditions that are of relevance for the spent fuel pool;

b) Means for monitoring and controlling the water level for operational states and for accident conditions that are of relevance for the spent fuel pool;

c) Means for monitoring and controlling the activity in water and in air for operational states and means for monitoring the activity in water and in air for accident conditions that are of relevance for the spent fuel pool;

d) Means for monitoring and controlling the water chemistry for operational states.

TABLE 14. CONSIDERATIONS TO ENHANCE ROBUSTNESS OF SPENT FUEL COOLING

Defence in Depth level	**Critical issues addressed**	**Options for countermeasures**	**Considerations or engineered features to be added and/or modified**	**Relevant safety requirements**
Prevention (1)	Cooling system	Secure the spent fuel pool (SFP) cooling function [16].	• Provide passive cooling system for SFP. • Provide diversity of pool water injection method [16].	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Requirement 80 and relevant Paragraphs. [11]
Prevention (1)	SFP protection	Secure the SFP from external hazards.	Provide robust shielding to avoid and mitigate spent fuel damage.	
Prevention (1)	Fuel Risk	Reduce the spent fuel risk.	Several SMRs solutions use SFP or pit below grade. Due to the size of the core, lesser decay heat and source term can be considered.	

TABLE 14 (cont.)

Defence in Depth level	Critical issues addressed	Options for countermeasures	Considerations or engineered features to be added and/or modified	Relevant safety requirements
Prevention (1)	Integrity	Ensure the integrity and leak tightness of SFP.	• SFP has a large inventory of radio-activity. The coping period for SFPs cooling and shielding is very long because a large inventory is provided. The ability of the pool to maintain leak-tightness & integrity in normal conditions and during natural phenomena should be confirmed in crediting the coping period • Diverse, robust and strengthened external make-up cooling should be provided to further extend this coping time.	
Mitigation of severe accident(4)	Monitoring system	Ensure availability of monitoring system.	Incorporate additional diversified SFP temperature and water level monitoring systems in case of severe accidents that are also displayed in the main control room [17].	
Mitigation of severe accident (4)	Severe accident	Consider severe accident management in SFP design.	Include corium coolability and hydrogen management in design.	

4.1.10. Use of effective probabilistic safety assessment for risk assessment and management

Probabilistic safety assessment (PSA) is an essential tool to practically understand the important part of safety system in NPPs. Generally, the assessment is conducted and completed for a wide range of accident scenarios in NPPs prior to operation. However, the Fukushima Daiichi accident indicated that many accident progression scenarios were not properly taken into account in the PSA although physical consequences of the events were largely in line with previous understanding. Analysis of plant had not previously considered all the failure modes that occurred. It is obvious that some issues related with PSA should be reviewed and reconsidered for the reactor design and more emphasis should be placed on these issues as they could not draw proper attention so far. The following are some countermeasures to address the PSA issues *TABLE 15*:

Applicable Safety Requirements for risk assessment and management [11]

Requirement 42 of SSR-2/1 (Rev. 1) establishes the following requirements on safety analysis of the plant design.

"A safety analysis of the design for the nuclear power plant shall be conducted in which methods of both deterministic analysis and probabilistic analysis shall be applied to

enable the challenges to safety in the various categories of plant states to be evaluated and assessed".

Paragraph 5.70 requires that the design basis for items important to safety and their links to initiating events and event sequences shall be confirmed on the basis of a safety analysis.

Paragraph 5.71 requires that it shall be demonstrated that the nuclear power plant as designed is capable of complying with authorized limits on discharges with regard to radioactive releases and with the dose limits in all operational states, and is capable of meeting acceptable limits for accident conditions.

Paragraph 5.72 requires that the safety analysis shall provide assurance that defence in depth has been implemented in the design of the plant.

Paragraph 5.73 requires that the safety analysis shall provide assurance that uncertainties have been given adequate consideration in the design of the plant and especially that adequate margins are available to avoid cliff edge effects and large or early radioactive releases.

Paragraph 5.74 requires that the applicability of the analytical assumptions, methods and degree of conservatism used in the design of the plant shall be updated and verified for the current or as built design.

Paragraph 5.76 requires that the design shall take due account of the probabilistic safety analysis of the plant for all modes of operation and for all plant states, including shutdown, with particular reference to:

a) Establishing that a balanced design has been achieved such that no particular feature or postulated initiating event makes a disproportionately large or significantly uncertain contribution to the overall risks, and that, to the extent practicable, the levels of defence in depth are independent;

b) Providing assurance that small deviation in plant parameters that could give rise to large variations in plant conditions (cliff edge effects) will be prevented;

c) Comparing the results of the analysis with the acceptance criteria for risk where these have been specified.

TABLE 15. CONSIDERATION OF THE USE OF PSA EFFECTIVELY FOR RISK ASSESSMENT AND MANAGEMENT

Defence in Depth level	**Critical issues addressed**	**Options for countermeasures**	**Considerations or engineered features to be added and/or modified**	**Relevant safety requirements**
Prevention (1)	Cliff edge effect	Use PSA to eliminate cliff edge effect.	PSA can be used to provide insights on which systems are candidates for strengthening to remove cliff edge effects as the magnitude of the natural hazards increases. In using PSA to consider reduction in core damaged frequency (CDF) vs. reduction in large early release frequency (LERF), comparison should be to a CDF criteria (which in turn prevents LERF), there may be sites/cases where preventing large early release, to limit off	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power

TABLE 15 (cont.)

Defence in Depth level	Critical issues addressed	Options for countermeasures	Considerations or engineered features to be added and/or modified	Relevant safety requirements
			site doses, would allow long term mitigation strategies which delay release after core damage.	Plants: Design Requirement 43 and relevant Paragraphs. [11]
Prevention (1)	Success criteria	Ensure that severe accident is considered as a hypothetical possibility.	• Success criteria could be established that recognize the lower probabilities of occurrence for example prevent core damage for more probable events, and prevent LERF in less probable events. • Update the external hazards screening criteria and frequency assessment [24]. • Consider the correlated hazards [24]. • Perform extreme natural hazard impact assessment [24]. • Consider multiple units [24]. • Consider failure possibility for qualified equipment [24].	
Prevention (1), control of accidents, and mitigation of severe accidents (4)	Full scope PSA	Perform full scope PSA.	To determine location of components or various scenarios of accident progress-ion, full scale safety assessment of plant should be implemented. (from 1.3)	
Prevention (1)	Risk informed approach	Consider 'risk-informed' approach in the design.	• SMRs can/must exploit risk-informed approach from the design phase, to increase robustness of the safety level. • A more comprehensive and less optimistic analysis should be performed for all operator actions in order to account for the impact of the external hazards on operator's access, performance and associated human errors [24]. • Risk informed methods are used to make a decision on the design changes.	

4.2. ON-SITE EMERGENCY PREPAREDNESS AND RESPONSE

4.2.1. Ensure on-site emergency response facilities, equipment and procedures

Emergency preparedness program is to ensure that operators of nuclear power plant have enough skills to implement adequate actions needed to protect public health and safety in the event of a radiological emergency. The actions must be supported by appropriate facilities, equipment and procedures. In the Fukushima Daiichi accident, the following facts appeared as important lessons during emergency situation.

It was recognized that the on-site emergency response center (which is located in a seismically isolated building) of Unit 1 was very useful. In addition, non-safety related systems played important roles to block the damage progression. However the loss of power supplies disabled many communication equipments. As a result, outside help or advice was limited and inefficient on-site communication systems delayed and complicated the necessary recovery actions. In addition, loss of light was also a major issue. The ability to physically locate and manually operate valves and equipment in the darkness was very important. In addition, the loss of electricity supply to the heating ventilating air conditioning system (HVAC) also aggravated the accessibility. This suggests that on-site emergency response facilities should be of the same safety class as safety system equipment. It was also found that lack of standard external power connections delayed re-energization.

Procedural sets, including SAMG, did not cover this level of accident. Gross radioactive contamination of the whole site had not been considered in the manuals or procedures. Many actions were conducted in such a difficult situation based on the operator's understanding and judgment on what was occurring. *TABLE 16* lists considerations to deal with this type of accident.

Applicable Safety Requirements for on-site emergency response facilities, equipments and procedures [11]

Paragraph 2.10 of SSR–2/1 (Rev. 1) requires that measures shall are required to be taken to control exposure for all operational states at levels that are as low as reasonably achievable and to minimize the likelihood of an accident that could lead to the loss of control over a source of radiation. Nevertheless, there will remain a possibility that an accident could happen. Measures shall are required to be taken to ensure that the radiological consequences of an accident would be mitigated. Such measures include the provision of safety features and safety systems, the establishment of accident management procedures by the operating organization and, possibly, the establishment of off-site intervention measures by the appropriate authorities, supported as necessary by the operating organization, to mitigate exposures if an accident has occurred.

Requirement 67 of SSR-2/1 (Rev. 1) establishes the following requirements on emergency response facilities control centre on the site.

"The nuclear power plant shall include the necessary emergency response facilities on the site. Their design shall be such that personnel will be able to perform expected tasks for managing an emergency under conditions generated by accidents and hazards".

Paragraph 6.42 requires that information about important plant parameters and radiological conditions at the nuclear power plant and in its immediate surroundings shall be provided to the relevant emergency response facilities. Each facility shall be provided with means of communication with, as appropriate, the control room, the supplementary control room and other important locations at the plant, and with on-site and off-site emergency response organizations.

TABLE 16. ENSURING ON-SITE EMERGENCY RESPONSE FACILITIES, EQUIPMENT AND PROCEDURES

Defence in Depth level	Critical issues addressed	Options for countermeasures	Procedures to be modified and/or improved	Relevant safety requirements
Prevention and mitigation of severe accident(4)	Mitigation Procedures	Emergency and accident response strategies and implementing actions must give highest priority to maintaining core cooling	• Emergency response centers must maintain continuous awareness of the status of core cooling. • Changes to the method of core cooling must be made deliberately and with a clear strategy to establish an alternate cooling method • When there is reason to question the quality or validity of core cooling information, deliberate actions must be taken immediately to ensure a method of cooling is established.	IAEA Safety Standards Series Specific Safety Requirements No. SSR–2/1 (Rev. 1), Safety of Nuclear Power Plants: Design Paragraph 2.10 and Requirement 67 and relevant Paragraphs. [11]
Mitigation of severe accident(4)	Mitigation Procedures	Ensure that mitigation action can be performed by using onsite procedures.	• The mitigation actions which are credited should be incorporated in site procedures, and selected scenarios exercised during periodic drills/exercises including the off-site resources (lesson learned that fire truck connection did not match). • Procedures should continue to be symptom based, to address observed problems, and minimize the burden on diagnosis, and risk of misdiagnosis. • Because the specific sequence of initiation events for beyond-design-basis events is unknown, emergency response strategies must be robust and provide multiple methods to establish and maintain critical safety functions using a defence-in-depth approach. • Optimum accident management strategies and associated implementing procedures (such as emergency operating procedures and accident management guidelines) should be developed through communications, engagement, and exchange of information among nuclear power plant operating organizations and reactor vendors. Decisions to deviate from these strategies and procedures should be made only after rigorous technical and independent safety reviews that consider the basis of the original standard and potential unintended consequences.	

TABLE 16 (cont.)

Defence in Depth level	Critical issues addressed	Options for countermeasures	Procedures to be modified and/or improved	Relevant safety requirements
Mitigation of severe accident (4)	Accident management	Ensure that the accident management/emergency procedure and guideline cover all accident scenario including extreme natural hazards.	• Extreme natural hazards should be integrated with the plant emergency procedure and guidelines (EPGs), SAMGs and EDMGs, (extensive damage mitigation guidelines, i.e. sabotage/-terrorist attacks) to assure the procedures are consistent with each, and the transitions between procedures are rational. Initial & refresher training should be included. • Emergency operating procedure (EOP) to SAMG transition (training + exercise). • Exercise every year for every operating crew.	
Mitigation of severe accident (4)	Accident management	Enhance the clarity of accident management on critical safety functions.	• It is necessary for each nuclear power station to define and provide a list of critical safety functions, such as: 1) Reactor shutdown 2) Core residual heat removal 3) Primary circuit integrity monitoring 4) Containment integrity monitoring 5) Hydrogen concentration in certain phases monitoring, and so forth. 6) Radioactivity monitoring For each of these critical functions, it is necessary to define and describe all possible ways in which personnel and emergency management team can use to understand whether a critical function was satisfied or not. Then, it is necessary to describe all possible means to provide these critical functions. It may be useful to increase the number of technical persons for this management using external help.	
Mitigation of severe accident (4)	Periodic inspection	Ensure the availability of mitigation equipment during accident.	Strengthen the periodic inspection and testing of safety related systems and components and backup systems.	
Mitigation of severe	External	Ensure that safety analyst in the	Provide communication / network lines that allow the headquarters to	

TABLE 16 (cont.)

Defence in Depth level	Critical issues addressed	Options for countermeasures	Procedures to be modified and/or improved	Relevant safety requirements
accident (4)	advise	headquarters is able to supervise and provide adequate advice to on-site workers and engineers.	see/monitor the control room displays and indications.	
Mitigation of severe accident (4)	Emergency equipment	Enhance the communication and the availability of emergency lighting sources.	• Adopt dependable communication equipment. • Adopt dependable lighting equipment. • Independent battery-powered emergency lights in the main control room and key building walkways are needed in the event that normal AC power and DC power are lost.	
Mitigation of severe accident (4)	SAMG	Ensure that a plant specific SAMG has been developed.	Provide plant specific SAMG.	
Mitigation of severe accident (4)	Emergency response	Ensure the availability of monitoring equipment to diagnose and initiate response.	• Install active warning system in remote locations near the source of external natural event or at most sensitive location. • The mitigation plans may involve additional monitoring equipment to speed diagnosis. The mitigation plan can include robust offsite assistance, including transit time to the site. • Provide mobile equipment [25].	
Mitigation of severe accident (4)	Emergency response	Emergency response needs are expected to be minimal due to SMR's small source term and high level of safety.	• Justify that source term is small and level of safety is high. • Justify that emergency response resources needed are minimal.	
Mitigation of severe accident (4)	Emergency response	Ensure that ER facilities incorporate impacts of credible natural disasters, staff loading, radiation effects and duration of site wide emergencies [17].	• Provide prompt access to primary and alternate use points for beyond design basis prestaged equipment [18]. • Provide standard mechanical and electrical interface connections with prestaged onsite and offsite beyond design basis equipment [18].	

4.2.2. Enhance human resource, skill and capabilities

Human capabilities and capacities are very important during emergency situation. Every decision is critical, and any misunderstanding can cause a wrong action that may lead to a

worse condition. In the Fukushima Daiichi nuclear complex, when the loss of power crippled system indications, lack of experience and clear situational awareness in running IC limited the ability to realize that it was not working and removed the chance of recovering the Unit 1. In addition, manual stop of HPCI in Unit 3 prior to confirmation that the alternative system was effective shows lack of situational awareness. These facts indicate that establishment of human capability and capacity can be ensured only through a structured training on a plant specific SAMG.

TABLE 17. ENHANCING HUMAN CAPABILITIES AND CAPACITIES

Defence in Depth level	Critical issues addressed	Options for countermeasures	Program to be implement
Mitigation of severe accidents (4)	Staff skill	Ensure that the plant's staffs have capability for multiple module reactor problems.	Staffing for natural hazards is especially applicable to modular reactors where a response may be needed for multiple module reactors concurrently.
Mitigation of severe accident (4)	Staff skill	Ensure the staff's skill in dealing with accident management.	Severe accident management (SAM) training should be done every year for every crew.
Control of accident within DB (3)	Human factor	Include human factor in the design of the plant to cope with long term SBO.	• Passive safety systems automatically actuated should avoid this concern; Human Factors can be taken into account since SMR design phase. • Human action is not credited before 7 days in an extended SBO scenario. • Portable system as back up.

4.3. OFF-SITE EMERGENCY PREPAREDNESS AND RESPONSE

4.3.1. Strengthen off-site infrastructure and capability

During nuclear accident, all available resources to cope with emergency condition must be delivered timely and without delay. Especially, recovery of a heavily damaged plant is entirely dependent on external equipments and therefore strengthening off-site infrastructure is very important. *TABLE 18* describes options to strengthen off-site infrastructure and capability.

TABLE 18. STRENGTHEN OFF-SITE INFRASTRUCTURE AND CAPABILITY

Defence in Depth level	Critical issues addressed	Options for countermeasures	Infrastructure to be added
Mitigation of severe accident (4)	Command structure	Ensure the availability of command structure in the government in dealing with nuclear emergency.	Ensure governmental agencies with necessary transport and relocation resources are included in emergency plan and participate in drills.
Mitigation of severe accident (4)	Offsite help	Develop infrastructure to cope with emergency with less or delayed offsite help.	• For SMR: less dependence on off-site infrastructure should be implemented by design, using passive safety features. • An SMR based on ice breakers technology is claimed by the designer to be capable to cope with most emergencies without off-site resources or the off-site helps can be significantly delayed.

4.3.2. Strengthen national arrangements for emergency preparedness and response

Preferably, a nuclear reactor accident only affects a limited area around the site with low radiological impact. However when a far distance and wide area are influenced, national response must be engaged. Appropriate facilities and equipment including radiation monitoring post should be prepared adequately and should be able to function in most adverse conditions. In addition, people must also be informed regarding the status of radioactive dispersion, plant conditions and other related parameters through an established and well organized means. During the Fukushima Daiichi accident it was discovered that the preparation of off-site radiation monitoring posts did not consider loss of power condition. The equipment failed to display radioactive dose status due to unavailability of power. Radiation dose chart for the area was unavailable. At that time, news broadcasting was the only available source for plant overview. With lack of reliable radioactivity spread data, the government recommended people to voluntarily evacuate outside the 20 – 30 km range from of the Fukushima Daiichi plant. This condition suggested that System for Prediction Environment Emergency Dose Information (SPEEDI) which uses measurements of radioactive releases, as well as weather and topographical data, to predict where radioactive materials could travel after being released into the atmosphere, has to be used in suggesting evacuation area.

The Fukushima Daiichi accident also tells that the first assistance from outside arrived to location hours after the earthquake. Fire engines were successful in injecting water to systems but limited to low pressure. Plant management might have inadequate information of plant status. The actions taken by the authority during the accident shows that national

emergency preparedness and response plans were insufficient in dealing with multiple infrastructure damage after the earthquake and tsunami. The frequency of emergency preparedness drills involving whole public and stakeholders should be considered for lower frequency and be evaluated accordingly.

Bearing in mind the previous lessons, it is necessary to prepare national arrangement for a nuclear emergency. The *TABLE 19* provides some useful considerations.

TABLE 19. STRENGTHENING NATIONAL ARRANGEMENTS FOR EMERGENCY PREPAREDNESS AND RESPONSE

Defence in Depth level	Critical issues addressed	Options for countermeasures	Infrastructure to be modified and/or improved
Mitigation of severe accident (4)	National team	Establish national team for severe accident management.	Build up the technical support team consisting of designers and operators for decision making (severe accident counter-actions under very harsh environment).
Mitigation of severe accident (4)	Unified control	Harmonize national coordination.	Strengthen the unified control and decision making centre to make it easier and faster to avoid any conflict.
Mitigation of radiological consequences (5)	Public education	Ensure public awareness of emergency situation.	More public education on emergency preparedness is necessary.
Mitigation of severe accident (4)	Direct communication	Established direct communication among decision making people.	It is necessary to organize direct and effective communication among decision making people in the accident management centre. It should not be any intermediate-chain in communication.

4.3.3. Enhance interaction and communication with the international communities

Fukushima Daiichi accident shows the need for international collaboration in dealing with nuclear accident. The radioactive materials spreading across the air and ocean more or less have affected neighbouring countries and environments. In a nuclear accident people shall work together to cope the situation. International communication and contact are needed to let the cooperation succeeds. One country with specific expertise can provide technical help for others that need it. The following are options on enhancing the interaction and communication with international communities.

TABLE 20. ENHANCING INTERACTION AND COMMUNICATION WITH THE INTERNATIONAL COMMUNITY

Defence in Depth Level	Critical Issues addressed	Options for countermeasures	Infrastructure to be added
Mitigation of severe accident (4)	Equipment exchange	Establish emergency equipment exchange agreement with international community.	• Plans should be in place to bring equipment from the closest adjacent nuclear plants, even if these are operated by a different utility/company or in a different country. This should be part of the response plan and included in drills. • Establish a regional resource center for long term cooling needs.
Mitigation of radiological consequences (5)	Information centre	Established information centre for international community.	Report all of the crucial events and stabilization efforts of the plants to international communities.

4.4. NUCLEAR SAFETY INFRASTRUCTURES

4.4.1. Review and clarify regulatory and emergency response framework

Nuclear power complex is a work environment where management and employees are committed to put the safety as first priority. The commitment must be placed in a clear regulatory framework and the documents should contain logical, systematic and coherent regulations for adequate protection including the role and responsibility of relevant organizations.

It was noted that during the Fukushima Daiichi accident lack of clarity on some command and control issues (e.g., failure to report the IC status of Unit 1 to emergency response center) was part of situation which impeded the coping of emergency condition. There was also the decision issue for the evacuation and venting instructions. These problems show that accident mitigation plans were not adequately developed.

Noticing these lessons, *TABLE 21* describes some considerations important to nuclear safety infrastructure.

TABLE 21. CONSIDERATIONS TO REVIEW AND CLARIFY REGULATORY AND EMERGENCY RESPONSE FRAMEWORK

Defence in Depth level	Critical issues addressed	Options for countermeasures	Programs to be modified and/or improved
Mitigation of severe accident (4) and Mitigation of radiological consequences (5)	Regulation	Update all regulations and emergency framework to cover Fukushima Daiichi type accident.	Study of safety standards and regulations to setup the manageable action-items to the existing NPP (SMR).
Mitigation of severe accident (4)	SAMG	Develop plant specific SAMG which covers full spectrum of events.	Add more items in SAMG for the Fukushima type accident conditions.

4.4.2. Reinforce safety regulatory bodies and legal structures

The regulatory body plays an important role in assuring that the public safety receives the highest attention in the use of nuclear energy. Its task is to establish regulations and legal frameworks for wide variety of activities including licensing for design, construction, operation, and many other others. In addition, the regulation must be properly implemented and evaluated in order to be effectively providing beneficial effect.

Findings from the Fukushima Daiichi accident showed that it is important to review and update the existing regulatory and legal structure. Technical requirements, guidelines and safety related criteria must be clarified based on the new laws and regulations. In addition, the updated nuclear regulatory system should ensure clarity of all parties' roles involved in the regulatory framework. In order to reinforce the regulatory body and its legal structure, the following are recommended to establish a condition which confirms the effectiveness of the legal system (see *TABLE 22*).

TABLE 22. REINFORCING SAFETY REGULATORY BODIES AND LEGAL STRUCTURES

Defence in Depth level	Critical issues addressed	Options for countermeasures	Programs to be modified and/or improved
Prevention (1)	Regulatory oversight and legal framework	Strengthen the role of the regulatory body.	• Strong oversight is important to assure that there is effective implementation of design features in construction & installation. This includes oversight of testing/surveillances of the equipment. • The lesson in the US is that there is not a standardized surveillance of the SBO 'Special Event' SSCs. • Independence and expertise of

TABLE 22 (cont.)

Defence in Depth level	Critical issues addressed	Options for countermeasures	Programs to be modified and/or improved
			the regulatory bodies are essential to guarantee the safe operation of nuclear facilities. • Decision making related with the safety regulations should be based on the supreme expertise and experience.

4.4.3. Instil safety awareness and attitude

Safety awareness and attitude must be persistently improved in all stakeholders involved in the use of nuclear energy especially in the operators and workers in nuclear complex. During the Fukushima Daiichi accident, the following positive outlooks were observed when efforts to control the reactor being conducted:

- Operators performed above and beyond call of duty
- Operators stayed within approved dose limits (except for limited inadvertent events)
- Operators/utility had no hesitation in carrying out actions to the benefit of the public but economically detriment to the facility
- Appropriate operational response was not constrained by fear of personal harm
- Operators initiated non-standard procedures early in the event (e.g., fire engines)

However, the enthusiasm of the operators could also be hindered by poor communication techniques which led to misunderstanding and lack of clarity on some command and control issues. This situation indicated that safety awareness, command and control must be addressed.

TABLE 23. CONSIDERATIONS TO INSTILL SAFETY AWARENESS AND ATTITUDES

Defence in Depth level	Critical issues addressed	Options for countermeasures	Procedures to be modified and/or improved
Prevention (1)	Nuclear safety cultures	Establish nuclear safety culture	• Provide consistent and periodic education of nuclear safety culture to all stakeholders including owners, operators, designers, fabricators, and supply chains. • Third party audit of safety culture establishment of an organization is recommended.

5. CONCLUDING REMARKS

New nuclear power plants, including SMRs, are to be designed, sited and constructed consistent with the objective of preventing accidents in the commissioning and operation. Should an accident occur, the plants shall be able to prevent and mitigate possible releases of radionuclides that may cause long term off site contamination.

The Fukushima Daiichi accident unveiled many issues regarding weakness of existing plants particularly regarding the design of ESF in withstanding extreme natural hazards and coping with the challenging emergency situation that could include a simultaneous extended station blackout. The study of accident progression and the lessons learned from this accident recommended the need for NPPs to review and evaluate their ESF designs and sitings, as well as to upgrade emergency preparedness and response, and nuclear safety infrastructure. Accordingly, issues such as the magnitude of natural hazards, multiple reactor units on the same site, common cause failures, hydrogen explosion, all on-site and off-site electric power losses, emergency preparedness and procedures, safety culture and regulatory issues should be addressed and documented in publication.

Many water cooled SMR designs and technologies are under development offering simplified design and flexible deployment options. The issues and lessons learned from the Fukushima Daiichi event are being incorporated into SMR design development. This publication attempts to address concerns by providing technical considerations and options of countermeasures that can be incorporated in water cooled SMRs.

The publication proposes considerations and examples to prevent the occurrence of cascading severe accidents after extreme natural hazards. These proposals are useful for embarking countries and utilities planning to deploy SMRs or advanced reactors. They are also useful for SMR technology developers to enhance the performance of the ESF in their respective reactor designs.

Various design concepts of ESF used by water cooled SMRs, which include the trip systems, residual heat removal systems, safety injection systems and containment systems, were discussed in this publication and they showed both similarities and differences in the approaches to achieve enhanced safety. The features to deal with severe accident conditions from different water cooled SMR designs were reviewed. However, this publication is not at all intended to substitute design review, meant to determine whether the safety requirements are being addressed in the design of water cooled SMR.

Water cooled SMR designs for near term deployment have variations in ESF designs that may come from the reactor attributes such as design and safety characteristics, power level, and type of safety system (active, passive or hybrid) to cope with accidents. They also have the potentialities to duly adapt to and cope with a variety of extreme natural hazards (e.g. Fukushima-like or simultaneous and multiple external hazards). This TECDOC presents and discusses design safety considerations on appropriate and practical countermeasures to incorporate and address the lessons learned from the Fukushima Daiichi accident to enhance the design of engineered safety systems of water cooled SMRs currently under development. Section 4 and the associated Tables in this TECDOC incorporate the details. The following is a summary of the Tables.

- Possibility of simultaneous occurrence of extreme natural hazards, specific to the site with appropriate return cycle (recurrence frequency) should be included in reactor plant site analysis. Accordingly, a feasible degree of independence, with a practical

grace period, from off-site power, availability of cooling sources, communication and transportation systems, etc., should be ensured in the design. (*TABLE 6*)

- Since some SMR designs incorporate multiple-units on the same site, some are designed for deployment underground, barge–mounted (floating unit) or seabed–anchored, which are first of its kind concepts, therefore, designers should consider unprecedented and unexpected accident scenarios, including common cause failures to provide countermeasures which can be carried out on the site if a meltdown occurs. *(TABLE 7)*
- SMR designers should consider electrical power unavailability during extreme natural hazards. The design should ensure core cooling and decay heat removal, e.g., by ensuring survivability of power supply with on-site availability of portable equipment. (*TABLE 8*)
- SMR designs should provide at least one success path in extreme natural hazards, to cope with the accident to cool down the reactor core by passive, or active, or manually aligned systems (such as portable pumps, heat removal systems and fire engines) or suitable combination of these. (*TABLE 9*)
- SMR designs should prevent failure of safety related structures, systems and components, or accommodate those failures with compensatory measures for all extreme natural events. (*TABLE 10*)
    - Water cooled SMRs offer specific features in safety related SSCs to assure containment vessel integrity, diverse shutdown, core cooling and decay heat removal. They include: reduced decay heat and reduced source term (due to small size), the adoption of passive safety features (e.g. external cooling of containment vessel with air or water or both, large amount of water per MW(th) (compared to large reactors) and in-vessel core retention and cooling. (*TABLE 10*)
    - Diverse cooling systems for containment and provision for connecting portable cooling equipment should be provided to prevent primary containment vessel (PCV) failure, e.g., by providing initial containment cooling by passive system(s) followed by portable pump supplied water spray for long term service. (*TABLE 10*)
    - Survivability of emergency power supply system should be assured to cope with extreme natural hazards and extended SBO. SMRs should ensure a reasonable grace period during which the essential safety functions can be restored. This can be achieved by using passive safety features and high capacity compact batteries. (*TABLE 10*)
- SMRs should avoid combustible mixture of hydrogen and oxygen inside and outside the containment, by adopting re-combiners or inerting the containment. (*TABLE 11*)
- Containment vent system of SMR should be capable of preventing catastrophic failure of containment by reducing pressure at appropriate rate. The venting system should ensure full capability of filtering particulates to reduce off-site dose to the minimum. *(TABLE 12)*
- SMR designs should ensure DC power availability for post accident monitoring system even in case of extended SBO. The system and cables should be placed and run in hardened and perforated housings to withstand severe accidents. (*TABLE 13)*

- SMR designs should ensure the integrity and leak tightness of spent fuel pool (SFP), considering extreme natural hazards. Provision and facility for cooling water replenishment and monitoring in emergency conditions should be taken into account in the pool design. (*TABLE 14*)
- The design process of SMRs can take advantage of a risk-informed approach for safe design that concurrently uses deterministic and probabilistic safety tools and analysis, to provide robust and optimised safety features and eliminate cliff edge effects. (*TABLE 15*).
- Requirement 42 of IAEA Safety Requirement – Safety Standards Series No. SSR–2/1 (Rev. 1) establishes the following requirements on safety analysis of the plant design.

  "A safety analysis of the design for the nuclear power plant shall be conducted in which methods of both deterministic analysis and probabilistic analysis shall be applied to enable the challenges to safety in the various categories of plant states to be evaluated and assessed".

  - *Paragraph 5.70 requires that the design basis for items important to safety and their links to initiating events and event sequences shall be confirmed on the basis of a safety analysis.*
  - *Paragraph 5.71 requires that it shall be demonstrated that the nuclear power plant as designed is capable of complying with authorized limits on discharges with regard to radioactive releases and with the dose limits in all operational states, and is capable of meeting acceptable limits for accident conditions.*
  - *Paragraph 5.72 requires that the safety analysis shall provide assurance that defence in depth has been implemented in the design of the plant.*
  - *Paragraph 5.73 requires that the safety analysis shall provide assurance that uncertainties have been given adequate consideration in the design of the plant and especially that adequate margins are available to avoid cliff edge effects and large or early radioactive releases.*
  - *Paragraph 5.74 requires that the applicability of the analytical assumptions, methods and degree of conservatism used in the design of the plant shall be updated and verified for the current or as built design.*
  - *Paragraph 5.76 requires that the design shall take due account of the probabilistic safety analysis of the plant for all modes of operation and for all plant states, including shutdown, with particular reference to:*

    a) Establishing that a balanced design has been achieved such that no particular feature or postulated initiating event makes a disproportionately large or significantly uncertain contribution to the overall risks, and that, to the extent practicable, the levels of defence in depth are independent;

    b) Providing assurance that small deviation in plant parameters that could give rise to large variations in plant conditions (cliff edge effects) will be prevented;

    c) Comparing the results of the analysis with the acceptance criteria for risk where these have been specified.
- Plant operators, utility management and governmental support organizations should coordinate and prepare for extreme emergencies by establishing improved command

and communication chains and support logistics, hold periodic exercises and drills, and storing backup equipment, on-site and off-site, supported with periodic inspection as for other safety systems. (*TABLE 16* and *TABLE 18)*

- Provide design solutions for a grace period of seven or more days, without operator intervention. In addition, operator training for SAMGs should be provided, including analysis of multiple unit plants scenarios where needed. (*TABLE 17*)
- National team for severe accident management shall be established for nuclear power plants, including SMRs. Technical support team should be established consisting of designers and operators for decision making for severe accident counter-actions under harsh environment. More communication and education on emergency preparedness are necessary to ensure public awareness of emergency situation. (*TABLE 19*)
- Procedure and plans should be in place to bring equipment from adjacent SMR units or other power plants. Reciprocal support from international community should be established through formal agreements for emergency equipment and expert assistance. A regional resource centre should also be established. (*TABLE 20*)
- Water cooled SMRs could be first-of-a-kind, hence plant specific SAMG which cover full spectrum of events shall be developed by incorporating means to cope with extreme natural events. (*TABLE 21*)
- The role of the regulatory body should be reinforced. Particularly for embarking countries, independence and expertise of the regulatory bodies is essential to guarantee the safe operation of nuclear facilities, including SMRs. (*TABLE 22*)
- Establishing nuclear safety culture that incorporates lessons learned from the Fukushima Daiichi type accident is essential. Consistent and periodic education on nuclear safety culture to all stakeholders including owners, operators, designers, fabricators, and supply chains, should be provided. (*TABLE 23*)

It is also commented that there will be a need of IAEA role in SMR design safety review to provide advice regarding the design's ability to meet the IAEA Fundamental Safety Principles. It is also suggested that the IAEA should develop relevant safety standards to incorporate SMR specific design features and special condition, as the current safety standards are applicable primarily for land based stationary nuclear power plants with water cooled reactors designed for electricity generation.

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DEFINITIONS OF TERMS

Engineered safety features	Features of the third level in defence in depth to control postulated incidents or accidents in order to prevent them from progressing to severe accidents or to mitigate their consequences, as appropriate.
Small modular reactor	Advanced nuclear reactors with electric power of up to 300 MW(e), built as modules in a factory setting then shipped to sites as demand arises, aiming for the economy of multiple by shortening construction schedule.
Extreme natural hazards	Extreme events unconnected with the operation of a facility or the conduct of an activity that could have an effect on the safety of the facility or activity. Typical examples of natural hazards for nuclear facilities include earthquakes, tornadoes, and tsunamis.
Integral PWR	A type of water cooled SMR adopting the principle of pressurized water reactor that integrates the components within the reactor coolant system, such as steam generators and pressurizer to be within the reactor pressure vessel, i.e., in the same compartment with the core assembly and eliminates the need of large bore piping network, with the objective to enable modularization and enhance safety performance.
Cliff edge effect	In a nuclear power plant, an instance of severely abnormal plant behaviour caused by an abrupt transition from one plant status to another following a small deviation in a plant parameter, and thus a sudden large variation in plant conditions in response to a small variation in an input.
Design basis accident	Accident conditions against which a facility is designed according to established design criteria, and for which the damage to the fuel and the release of radioactive material are kept within authorized limits.
Severe accident	Accident conditions more severe than a design basis accident and involving significant core degradation.

Isolation condenser system

The IC system transfers residual and decay heat from the reactor coolant to the water in the shell side of the heat exchanger resulting in steam generation. The steam generated in the shell side of the heat exchanger is then vented to the outside atmosphere. The system employs natural circulation as the driving head from the reactor steam side, through the IC tubes, and back to the reactor.

The IC system is automatically initiated if a high reactor pressure condition is sustained for 15 seconds. The time delay prevents unnecessary system initiation during turbine trips. Also at most plants, the IC system automatically initiates on a low vessel water level to aid in reducing reactor pressure for small line breaks. The IC system is designed to provide core cooling regardless of whether electrical power is available.

Reactor core isolation cooling system

The RCIC system is a steam-driven single train standby system for safe shut down of the plant. The system is not considered part of the emergency core cooling system (ECCS), and does not have a loss of coolant accident (LOCA) function. The RCIC system is designed to ensure that sufficient reactor water inventory is maintained in the vessel to permit adequate core cooling. This prevents the reactor fuel from overheating in the event that the reactor is isolated from the secondary plant.

Residual heat removal system

The RHR system is designed to remove residual heat from the reactor core after shutdown, and during and after appropriate operational states and accident conditions.

Hardened Containment vent

A reliable, hardened vent that can remove heat and pressure before potential damage to a reactor core occurs. This not only helps preserve the integrity of the containment building, but can also help delay reactor core damage or melting.

Defence in depth

A hierarchical deployment of different levels of diverse equipment and procedures to prevent the escalation of anticipated operational occurrences and to maintain the effectiveness of physical barriers placed between a radiation source or radioactive material and workers, members of the public or the environment, in operational states and, for some barriers, in accident conditions.

The objectives of defence in depth are:

(a) To compensate for potential human and component

failures;

(b) To maintain the effectiveness of the barriers by averting damage to the facility and to the barriers themselves;

(c) To protect workers, members of the public and the environment from harm in accident conditions in the event that these barriers are not fully effective.

INSAG defines five levels of defence in depth:

(a) Level 1: Prevention of abnormal operation and failures.

(b) Level 2: Control of abnormal operation and detection of failures.

(c) Level 3: Control of accidents within the design basis.

(d) Level 4: Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents.

Risk-informed approach

A "risk-informed" approach to regulatory decision making represents a philosophy whereby risk insights are considered together with other factors to establish requirements that better focus licensee and regulatory attention on design and operational issues commensurate with their importance to health and safety.

Grace period

The period of time during which a safety function is ensured in an event with no necessity for action by personnel. Typical grace periods range from 20 min to 12 hrs. The period of grace may be achieved by means of the automation of actuations, the adoption of passive systems or the inherent characteristics of a material (such as the heat capacity of the containment structure), or by any combination of these.

Ultimate heatsink

A medium into which the transferred residual heat can always be accepted, even if all other means of removing the heat have been lost or are insufficient. This medium is normally a body of water or the atmosphere.

Accident management

The taking of a set of actions during the evolution of a beyond design basis accident:

a) To prevent the escalation of the event into a severe accident;

b) To mitigate the consequences of a severe accident;

c) To achieve a long term safe stable state.

The second aspect of accident management (to mitigate the consequences of a severe accident) is also termed severe accident management.

Passive autocatalytic recombiner

In the unlikely event of a LOCA, large amounts of hydrogen could release in the reactor containment, leading to a high concentration of explosive gas which might ultimately affect the integrity of the containment. Passive Autocatalytic Recombiner (PAR) is a hydrogen reduction system. This advanced safety system requires neither operator action nor a power supply. Based on the principle of catalytic oxidization, the PAR features a gas treating capacity of up to 1500 m^3/h per unit. The hydrogen concentration can then be kept below explosive limits — even under severe accident conditions.

Emergency preparedness

The capability to take actions that will effectively mitigate the consequences of an emergency for human health and safety, quality of life, property and the environment.

Automatic depressurization system

System to depressurize the reactor (and keeps it depressurized) to allow emergency core injection system or other low pressure refill.

Primary containment vessel

The primary vessel as method or physical structure designed to prevent or control the release and the dispersion of radioactive substances.

Although related to confinement, containment is normally used to refer to methods or structures that perform a confinement function, namely preventing or controlling the release of radioactive substances and their dispersion in the environment.

ABBREVIATIONS

ACS	Atmospheric Control System
ADS	Automatic Depressurization System
AM	Accident Management
BDB	Beyond Design Basis
BWR	Boiling Water Reactor
CCF	Common Cause Failure
CDF	Core Damage Frequency
CMT	Core Make-up Tank
COPS	Containment Overpressure Protection System
CRDM	Control Rod Drive Mechanism
CSS	Core Spray System
CST	Condensate Storage Tank
CV	Containment Vessel
DB	Design Basis
DBA	Design Basis Accident
DC	Direct Current
DG	Diesel Generator
DiD	Defence in Depth
DHRS	Decay Heat Removal System
DVI	Direct Vessel Injection
ECCS	Emergency Core Cooling System
EBS	Extra Borating System
EDG	Emergency Diesel Generator
EDMG	Extensive Damage Mitigation Guidelines
EHRS	Emergency Heat Removal System
EOP	Emergency Operating Procedure
EPG	Emergency Procedure Guidelines
ESF	Engineered Safety Feature
FA	Fuel Assembly

FDA	Fukushima Daiichi Accident
FMCRD	Fine Motion Control Rod Drive
FWIV	Feed Water Isolation Valve
GDCS	Gravity Driven Cooling System
HPCF	High Pressure Core Flooder
HPCI	High Pressure Coolant Injection
HPCS	High Pressure Core Spray
HVAC	Heating, Ventilating, and Air Conditioning
HX	Heat Exchanger
IC	Isolation Condenser
IRWST	In-containment Refuelling Water Storage Tank
LERF	Large Early Release Frequency
LOCA	Loss of Coolant Accident
LPCI	Low Pressure Coolant Injection
LPFL	Low Pressure Flooder Mode
MCC	Motor Control Center
MSIV	Main Steam Isolation Valve
NPP	Nuclear Power Plant
NSSS	Nuclear Steam Supply System
PAR	Passive Autocatalytic Re-combiner
PCV	Primary Containment Vessel
PCCS	Passive Containment Cooling System
PSA	Probabilistic Safety Assessment
PRHR	Passive Residual Heat Removal
PWR	Pressurized Water Reactor
PXS	Passive Core Cooling System
RCIC	Reactor Core Isolation Cooling
RCS	Reactor Cooling System
RHRS	Residual Heat Removal System
RPS	Reactor Protection System
RPV	Reactor Pressure Vessel

RV	Reactor Vessel
SAMG	Severe Accident Management Guideline
SBO	Station Blackout
SCS	Shutdown Cooling System
SFP	Spent Fuel Pool
SG	Steam Generator
SGTS	Standby Gas Treatment System
SIS	Safety Injection System
SLC	Standby Liquid Control
SMR	Small and Medium Sized Reactor or Small Modular Reactor
SRV	Safety Relieve Valve
SSC	Structures, Systems and Components
TAF	Top of Active Fuel
UHS	Ultimate Heat Sink

ANNEX I

Technical overview of Engineered Safety Features of water cooled reactors

I.1. Boiling water reactors (BWRs)

I.1.1. GE BWR/4

The BWR reactor is the second most common type of reactor used for electricity generation. The reactor core is cooled by water, which boils and turns into steam in the upper part of the core. After passing through moisture separator and dryer, the steam is directed to a turbine to rotate electric generator. The first BWR was developed by the Idaho National Laboratory and General Electric in the mid-1950s. Since then, incremental development of features has been conducted covering modification of containment design, steam-dryer, general layout of the reactor building, reactor control and safety system and elimination of heat exchanger. One of BWR series is the BWR/4 which is one of the units suffered from tsunami and flooding in the Fukushima Daiichi nuclear complex.

The BWR/4 steam supply system mainly consists of a reactor pressure vessel – which contains the core, steam dryer and separator– that is connected to pipes to allow re-circulation pumps to circulate the coolant water inside the vessel. In addition, a set of

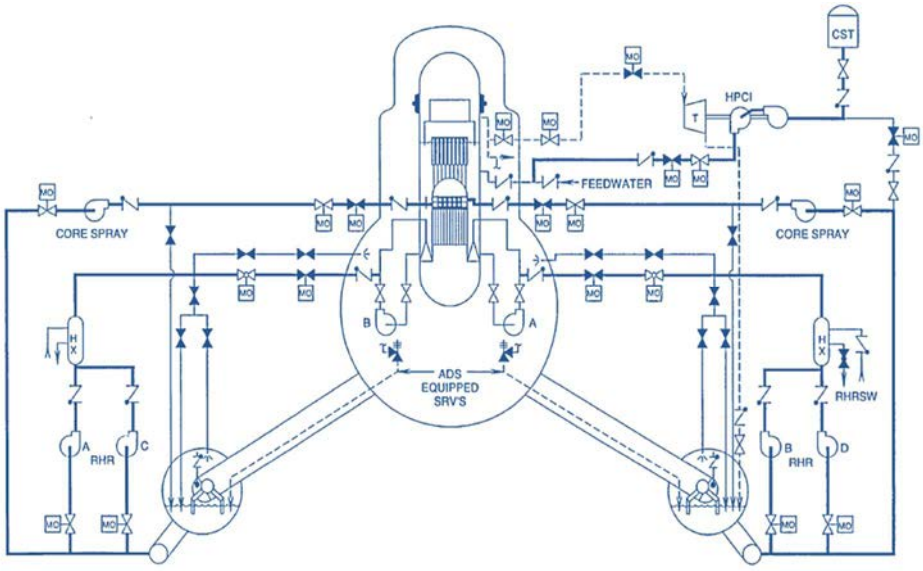


FIG. I-1. BWR/4 emergency core cooling system [20].

control and safety systems to ensure safe operation and a containment system which is intended as barrier for radioactive release during accident and as protection against external hazards are provided. The safety system is made up of redundant and multiple equipment to guarantee the intended functions. Some of their functionalities are to provide emergency core cooling, to control and maintain containment's pressure and temperature below its design limit, and to provide additional control for core reactivity. In BWR/4 the emergency core cooling function is facilitated by the low pressure coolant injection (LPCI) mode of residual heat removal (RHR) system, RCIC system, HPCI system, automatic depressurization system (ADS), and core spray system (CS) as can be seen in *FIG. I-1*. Meanwhile the other two functions are provided by the containment spray system (CSS) and standby liquid control (SLC) system.

I.1.1.1. Residual heat removal

The Residual Heat Removal System of BWR/4 consists of two separate piping loops. Each loop contains two pumps, heat exchanger and related piping and valves. RHR system is a multiuse system which has 5 working modes.

The dominant mode is the LPCI. The LPCI mode operates automatically to restore and maintain the fuel clad temperature below 1200°C. In this mode, the RHR pumps take water from the suppression pool and discharge it to the reactor vessel via recirculation system discharge piping.

I.1.1.2. Reactor core isolation cooling system

When the condensate and feed water system is not available or the main steam line is isolated, core cooling is provided by the RCIC system as shown in *FIG. I-2*. The system consists of a steam turbine driven pump and associated valves and piping capable of supplying water to the reactor vessel at operating conditions. The turbine is powered by steam produced from decay heat in the core, which flows through a steam line to the RCIC turbine and then discharges to the suppression pool.

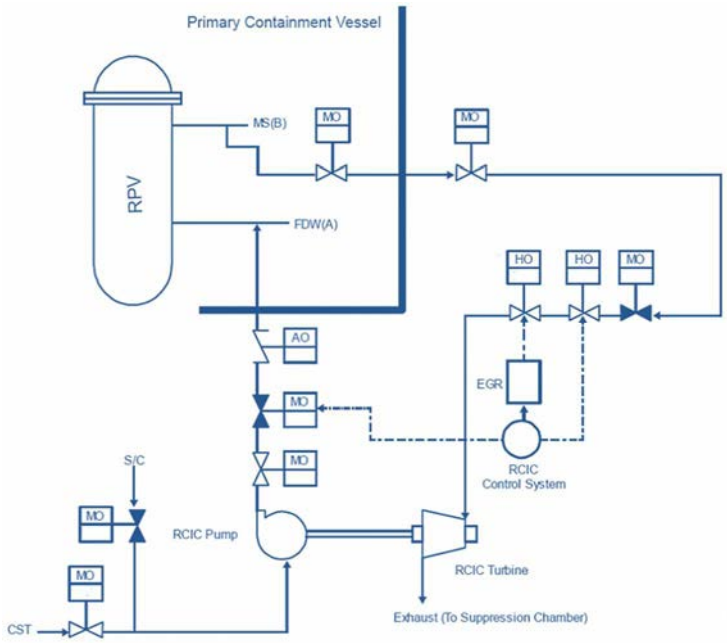


FIG. I-2. BWR/4 RCIC system [6].

I.1.1.3. High pressure coolant injection system

The HPCI has at least 3 functions, e.g. to maintain reactor vessel water inventory for core cooling on small breaks LOCAs, to help depressurize the vessel in order to allow the low pressure ECCS to inject water on intermediate break LOCAs and to back up the RCIC under isolation conditions where it supplies high pressure make up coolant to the reactor under such condition. The system has a functional interface with the RCIC and shares a suction line from the condensate storage tank as can be seen in *FIG. I-3*.

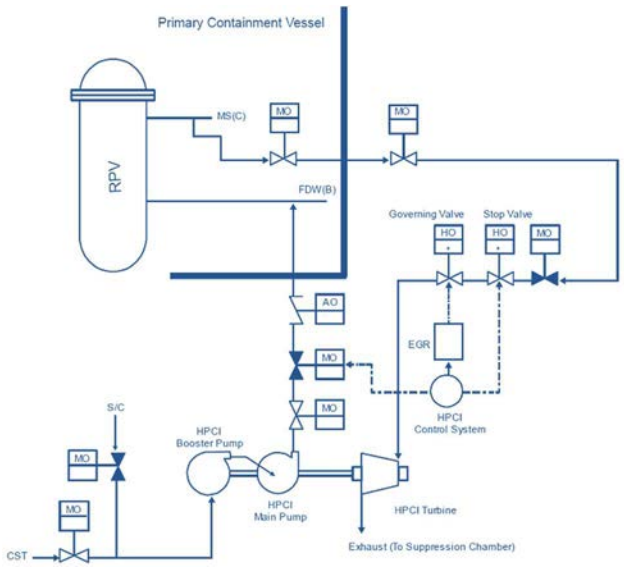


FIG. I-3. HPCI system of BWR/4.

I.1.1.4. Automatic depressurization system

The role of ADS is to depressurize the reactor vessel during a small break LOCA. It consists of automatically activated pressure relief valves. When the valves are opening, the pressure inside vessel decreases to a level which permits water injection from the low pressure core spray system to the core.

I.1.1.5. Core spray system

The main purpose of the core spray system (CS) is to provide a low pressure spray cooling to the reactor core under LOCA conditions. There are two independent pumping loops of CS, each consists of a set of spray nozzles located on independent ring spargers put within the core shroud above the fuel assembly (FA). The nozzles are arranged to provide uniform coolant flow to the fuel assemblies. The system provides enough cooling water and can be powered by emergency power system. The schematic diagram of CS is shown in *FIG. I-4*.

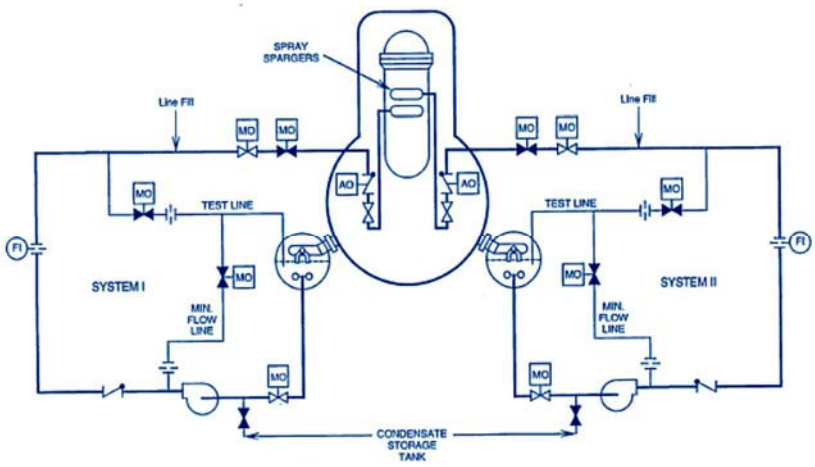


FIG. I-4. BWR/4 core spray system [20].

I.1.2. GE BWR/5

BWR/5 is one of the units built in the Fukushima Daiichi nuclear complex (Unit 6). This reactor was in outage period when the complex was struck by large tsunami wave. As the reactor location is higher than others, this reactor was not challenged as hard as the other three units being in operation at that time.

For dealing with the design basis accident (DBA), the safety feature of this reactor is provided by emergency core cooling system as shown in *FIG. I-5*, which consists of high pressure core spray system (HPCF), ADS, low pressure core spray system (LPCS), and LPCI mode of residual heat removal [19]. These systems are elaborated in more detail in the following.

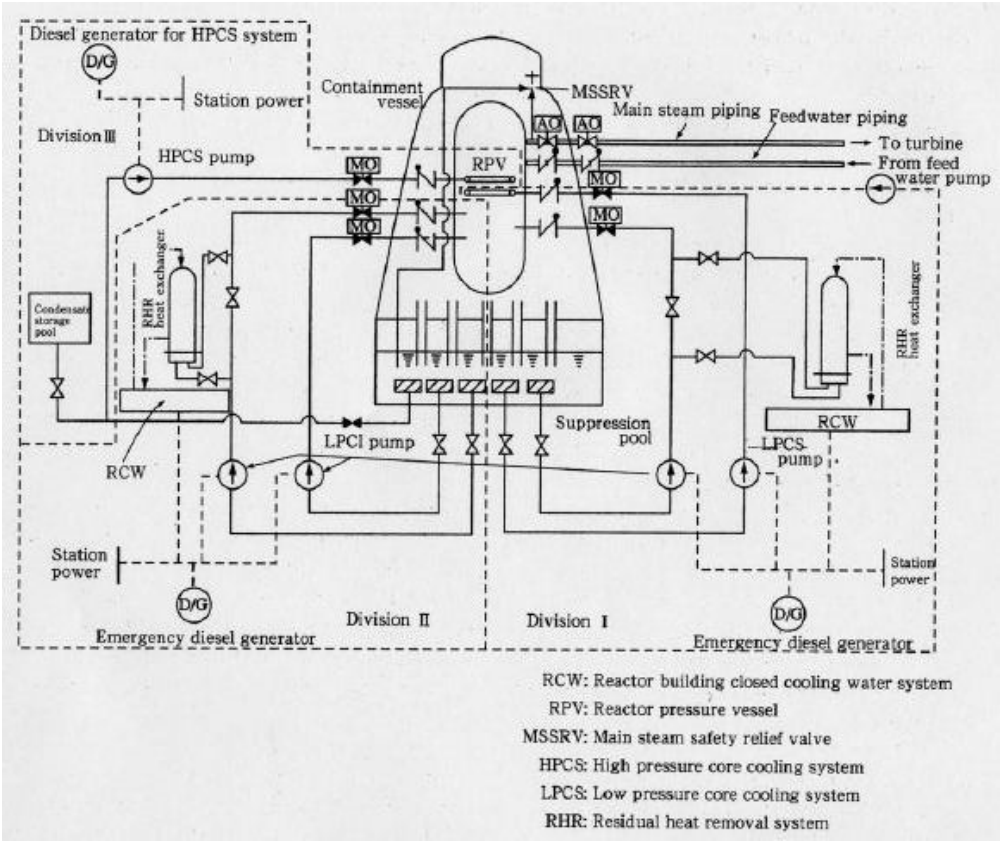


FIG. I-5. ECCS of BWR/5 with Mark II containment [19].

A.1.2.1 High pressure core spray (HPCS) system

The HPCS system is a single loop system consisting of a suction shutoff valve, one motor drive pump, discharge check valve, motor operated injection valve, minimum flow valve, full flow test valve to the suppression pool, two high pressure flow test valves to the condensate storage tank, discharge sparger and associated piping and instrumentation. It is designed to provide high pressure emergency core cooling following small, medium or large line breaks. The system takes water from the condensate storage tank or suppression pool and delivers the water to the sparger positioned on the upper core shroud. The HPCS pump starts automatically on high pressure in the drywell signal or on the low water level-2 in the reactor vessel. Power for this system comes from a standby power system diesel generator.

A.1.2.2 Low pressure core spray (LPCS) system

The LPCS system purpose is to remove decay heat generated by the fuel bundle following a postulated LOCA. The system takes suction from the suppression pool and discharges the water through core spray sparger ring located on top of the fuel assemblies. There is only single loop LPCS which consists of a suction shutoff valve, one motor driven pump, discharge check valve, motor operated injection valve, minimum flow valve, full flow test valve to the suppression pool, discharge sparger and associated piping and instrumentation. This system is actuated either on high pressure in the drywell or a low reactor water level signal.

A.1.2.3 LPCI Mode of RHR System

The RHR of BWR/5 is a multi-function system. It has five operational modes and one of them is the low pressure coolant injection mode which is dominant for this system. The RHR system consists of three separate piping loops, labeled A, B, and C. Loops A and B each has a pump and two heat exchangers. Loop C is not equipped with a heat exchanger and used merely for LPCI mode.

The LPCI mode initiates automatically on either on low (level-1) reactor vessel water level or high pressure in the drywell. During LPCI operation, the RHR takes water from the suppression pool and delivers to the reactor vessel inside the core shroud via dedicated penetrations. The injection will restore and maintain the fuel clad temperature below 1200°C.

I.1.3. ABWR

The standard ABWR plant design was certified and licensed in the United States of America, followed by Japan and Taiwan, China. The developer also produces an adapted design to meet European requirement which has net power output of about 1,600 MW(e) (4300 MW(th)). This plant is designed to have 60 years life time, refueling interval of 18 – 24 months and availability greater than 90%. The overall plant system of the ABWR is shown in *FIG. I-6*.

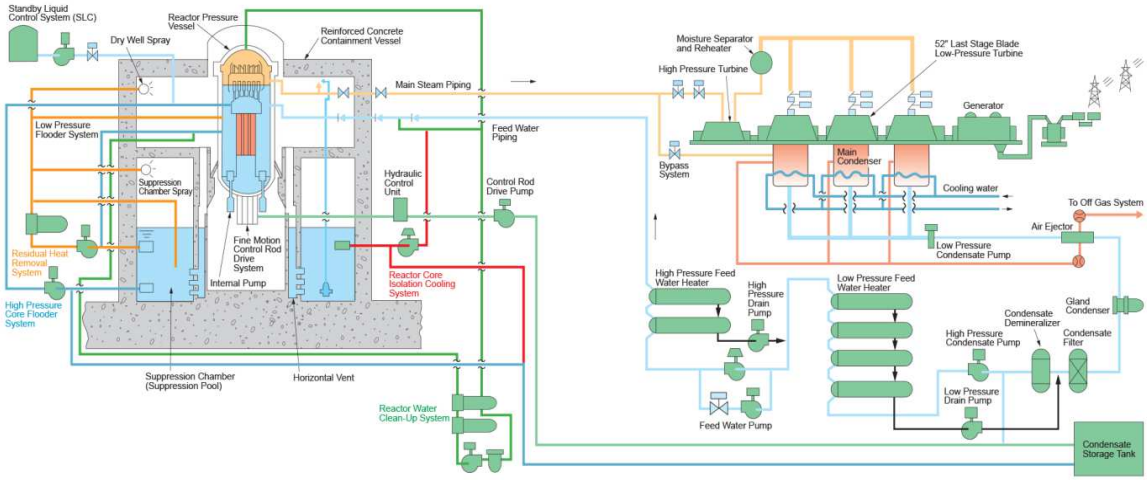


FIG. I-6. ABWR plant system (Reproduced courtesy of GE Hitachi Nuclear Energy) [17].

The design of ABWR represents an evolutionary development for the BWR type reactor. The improvement includes numerous modifications such as: addition of reactor internal pumps (RIP) where each has a nominal capacity of 7700 m^3/h, use of an electro-hydraulic Fine motion control rod drive (FMCRD) for control rod adjustment capabilities which allow fine position adjustment using an electrical motor without losing its reliability, use of full digital reactor protection system (RPS) which provides a high degree of reliability and reduction in complexity for safety condition detection and response (with redundant digital backups and redundant manual backups), application of total digital reactor controls with operator task-based control room system which allow the operator to easily and rapidly control plant operations and processes, improvement of the containment, improvement in Emergency core cooling system, etc. The safety feature is described in more detail below [26].

Emergency core cooling system

The emergency core cooling system (ECCS) of ABWR has been improved in many areas compared to the previous generation of BWRs, providing a higher level of defence-in-depth against accidents, contingencies, and incidents. The overall system is divided up into 3 divisions; where each division has capability to terminate the limiting fault/design basis accident (DBA) prior to core uncovery, even in the event of loss of offsite power and loss of proper feed water. The ECCS consists of high pressure core flooder (HPCF), reactor core isolation cooling, residual heat removal system and automatic depressurization system. The configuration of ABWR's ECCS is shown in *FIG. I-7*.

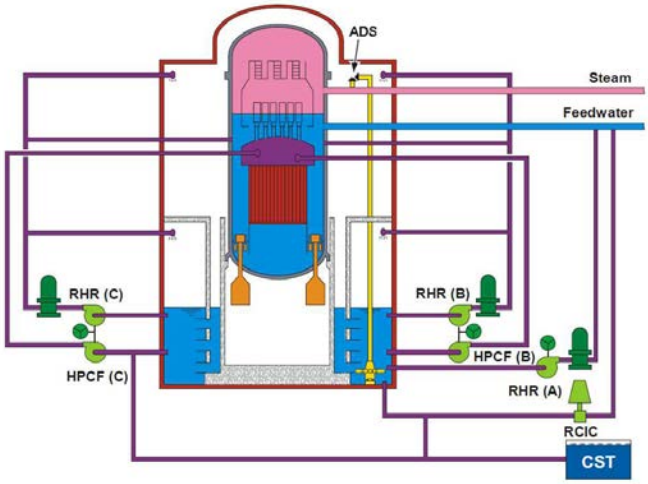


FIG. I-7. ABWR ECCS [25].

High pressure core flooder (HPCF)

The HPCF provides several functionalities: first, it is to maintain the RPV inventory during a small break LOCA which does not depressurize the reactor vessel. Secondly, it serves as a backup for the RCIC system in response to transients. The system is provided in two divisions and can be powered by EDG in case auxiliary power is unavailable. Both HPCF systems take their suction from the condensate storage pool (primary source) or from the suppression pool (secondary source). The discharge capacity is 182 m^3/hour at rated pressure. This system is automatically activated by either high pressure in the dry well or low water level in the reactor vessel. The HPCF of ABWR is shown in *FIG. I-8*.

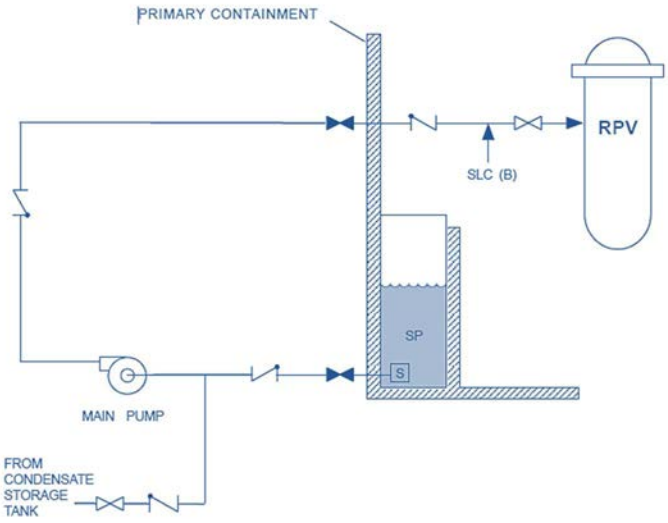


FIG. I-8. ABWR HPCF [26].

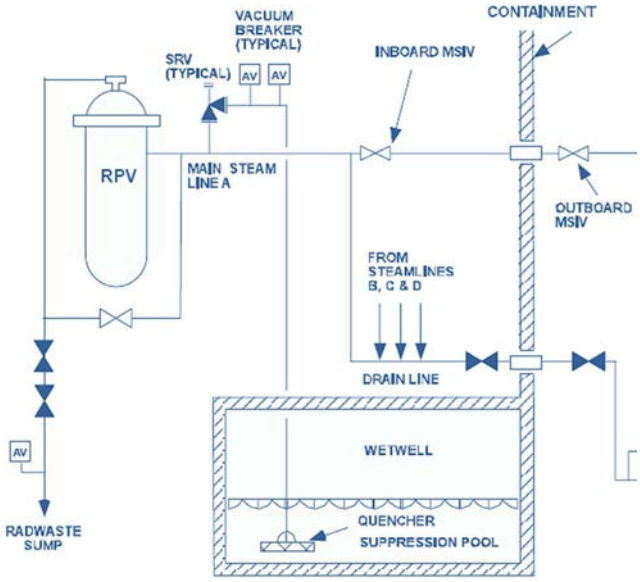


FIG. I-9. ABWR ADS [26].

Automatic depressurization system (ADS)

The ADS is automatically initiated when several combination of signal occurs:

1. After a short delay when RPV low water level signal is present concurrently with high drywell pressure signal.
2. After a longer delay of only RPV low water level signal is present.

The system can also be initiated through manual initiation signal, which concurrent with positive indication of RHR or HPCF pumps is running. The ADS consists of 8 safety relief

valves (SRV), where 2 SRVs on each Main Steam Line and each SRV blow-downs to quencher in suppression pool, as shown in *FIG. I-9*.

Reactor core isolation cooling (RCIC) system

The main function of the RCIC system is to deliver makeup water to the RPV and to maintain adequate water level in the vessel for the events such as: vessel isolated and maintained at hot stand by, loss of AC power and plant shutdown with loss of normal feed water, by using steam-driven high pressure pump which capable to supply water of about 181.7 m^3/hour. The water supply is taken either from suppression pool or condensate storage tank as depicted in *FIG. I-10*.

The system automatically initiates at RPV water level signal and its capacity is sufficient for make up on loss of feed water without support from any other make up system. The RCIC is considered as part of ECCS. In the event of LOCA, the RCIC is designed to be able to pump water to the vessel from full pressure down to about 150 psig. The steam generated by reactor decay heat is then directed to the wet well suppression pool. To maintain the pool temperature within acceptable limit, the RHR heat exchanger is used.

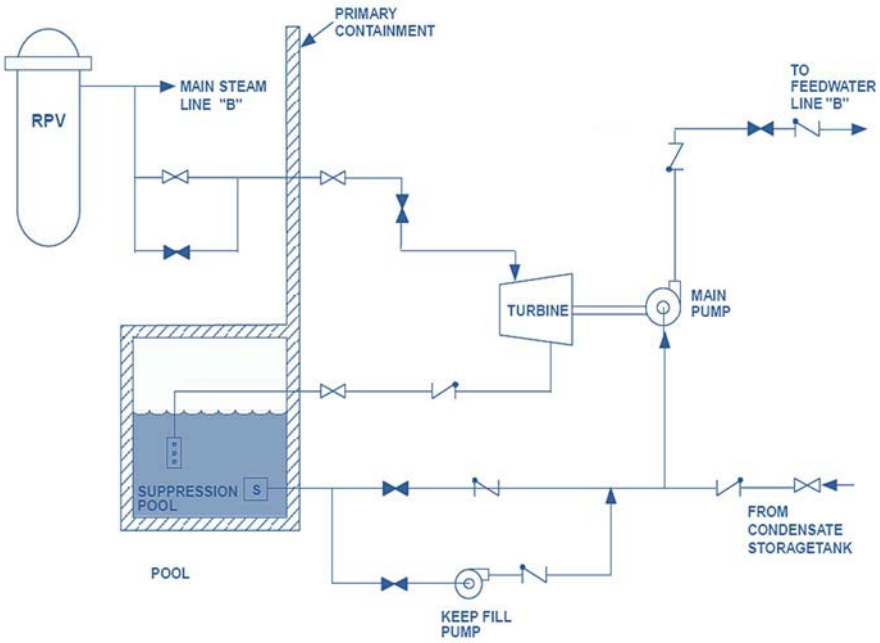


FIG. I-10. ABWR RCIC [26].

Severe accident features

In addition to ECCS, other important safety systems are available in ABWR such as the standby gas treatment system (SGTS), the atmospheric control system (ACS), the flammability control system, and the standby liquid control system (SLCS) which are essential in case of severe accident.

The residual heat removal (RHR) system

This system is intended to remove residual heat during normal plant shutdown, reactor isolation and LOCA. It has 3 motor driven low pressure pumps which capable to deliver water of 954 m^3/hour when vessel is depressurized. Single pump operating ensures no core

damage. The RHR system has six working modes, each with specifics purposes, as shown in *FIG. I-11.*

Safety-related modes:

1. Low pressure flooder system (LPFL) consists of 3 individual loops which provide water supply for core cooling to compensate the water loss due to any cause including LOCA. This mode is automatically initiated by a low water level in reactor vessel or high pressure in the drywell (LOCA signal) or can also be manually actuated.
2. Suppression pool water cooling mode consists of 3 individual loops which automatically initiates on high suppression pool temperature. During normal state, this mode cools the pool below 49°C. The water is taken from suppression pool, and then the flow goes to the heat exchanger and is returned to the pool again. The mode can also be manually actuated.
3. Primary containment vessel spray cooling mode (consists of 2 loops). When manually actuated, this mode sprays the water from suppression chamber pool into drywell and wet well. The sprayed water (which becomes hot) will return to the suppression chamber through vent pipes after the drywell water level reach the pipe level. When the hot water from the dry well reaches the suppression pool it increases the pool temperature which is then cooled by the RHR system heat exchanger before used for spraying again.

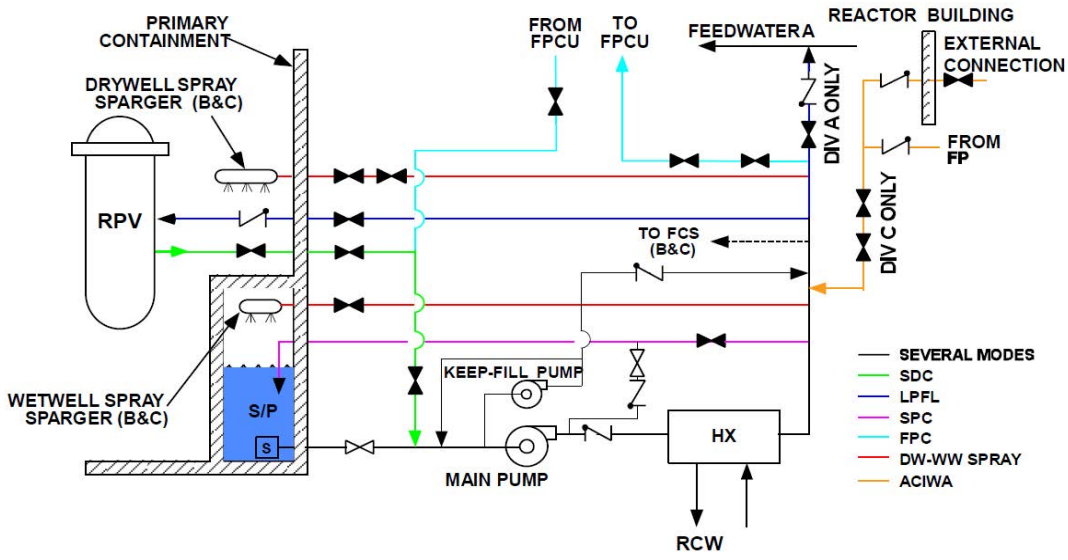


FIG. I-11. ABWR RHR [26].

Non – safety:

4. *Reactor shutdown cooling mode*

This mode is to allow refuelling activities and equipment maintenance. In this mode, the RHR system cools down the reactor coolant below 60°C within 20 hours after the shutdown. The suction is taken from the RPV and the flow goes to the RHR heat exchanger and returned to the RPV via LPFL injection lines.

5. *Supplemental fuel pool cooling (3 loops)*

Three loops are used to provide supplemental cooling for the fuel pool. This mode will only be used if the fuel pool cooling system (FPCU) is unable to maintain the fuel pool water temperature below allowable limit.

6. *AC independent water addition (1 loop)*

This mode uses RHR loop C to provide a means for introducing water from fire protection (FP) through RHR loop into RPV, drywell spray header or wet well spray header. This mode provides a manual cooling system to prevent core damage when all ECCS are lost.

7. *The standby gas treatment system (SGTS)*

This system is to treat and discharge either primary or secondary containment air to the plant stack. It is capable of removing more than 99% of elemental iodine or methyl iodide. Following a LOCA, the system automatically takes suction from the secondary containment and maintains a negative pressure of about 6 mm water and during refueling operation this system processes the secondary containment atmosphere. The SGTS consists of two 100% capacity divisions as can be seen in the *FIG. I-12*. Each division has filter train and two fans. The filter train consists of moisture separator, main electric heater, primary HEPA filter, charcoal absorber and secondary HEPA filter.

8. *Atmospheric control system (ACS)*

This system is intended to maintain an inert atmosphere within the primary containment and always active during all plant operating mode except during plant shutdown for refueling and maintenance. When activated, the system capable to reduce the concentration of oxygen to lower than 3.5% by volume in less than 4 hrs. The ACS is also equipped with a containment overpressure protection system (COPS) which is designed to reduce the containment pressure whenever the containment integrity is challenged by overpressure following an accident.

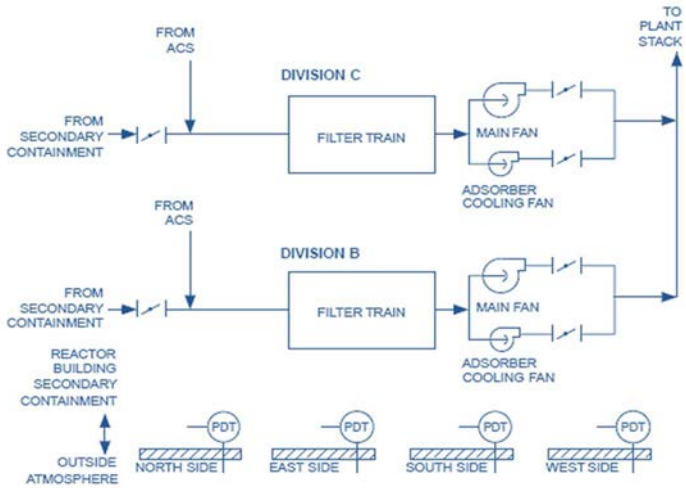


FIG. I-12. ABWR SGTS [26].

9. *Flammability control system (FCS)*

The FCS is designed to control the buildup of combustible mixture of hydrogen and oxygen inside the primary containment below the flammability limit. It is comprised of

thermal hydrogen and oxygen re-combiner units which are located in two different divisions and manually initiated based on hydrogen and oxygen level indications.

10. Stand by liquid control system (SLCS)

The SLCS is a backup reactivity controller to maintain the core sub- criticality when it cools down. The system can be either automatically or manually initiated. It consists of boron solution tank, pumps, valves and associated piping. The system is connected to the reactor vessel through HPCF line.

I.1.4. ESBWR

ESBWR is a BWR designed by GE Hitachi Nuclear Energy. The reactor has a rated power of about 4,500 MW(th) (1600 MW(e)) and its construction time can be completed within 36 months. The distinctive feature of this reactor is the use of natural forces for its normal and accident conditions which eliminates the need for safety grade diesel generators. Natural circulation is used for in-vessel recirculation during normal operation. Hence, no recirculation pump is needed to bring the water from the steam separator to down comer. In addition, passive safety feature is implemented to response the design basis accidents and severe accidents involving core melt. The passive safety system consists of isolation condenser system (ICS), Emergency core cooling - gravity driven cooling system (GDCS) and passive containment cooling system (PCCS). With all these functions, no operator action is required for 72 hours during the design basis accident [11]. The configuration of these systems is shown in *FIG. I-13*.

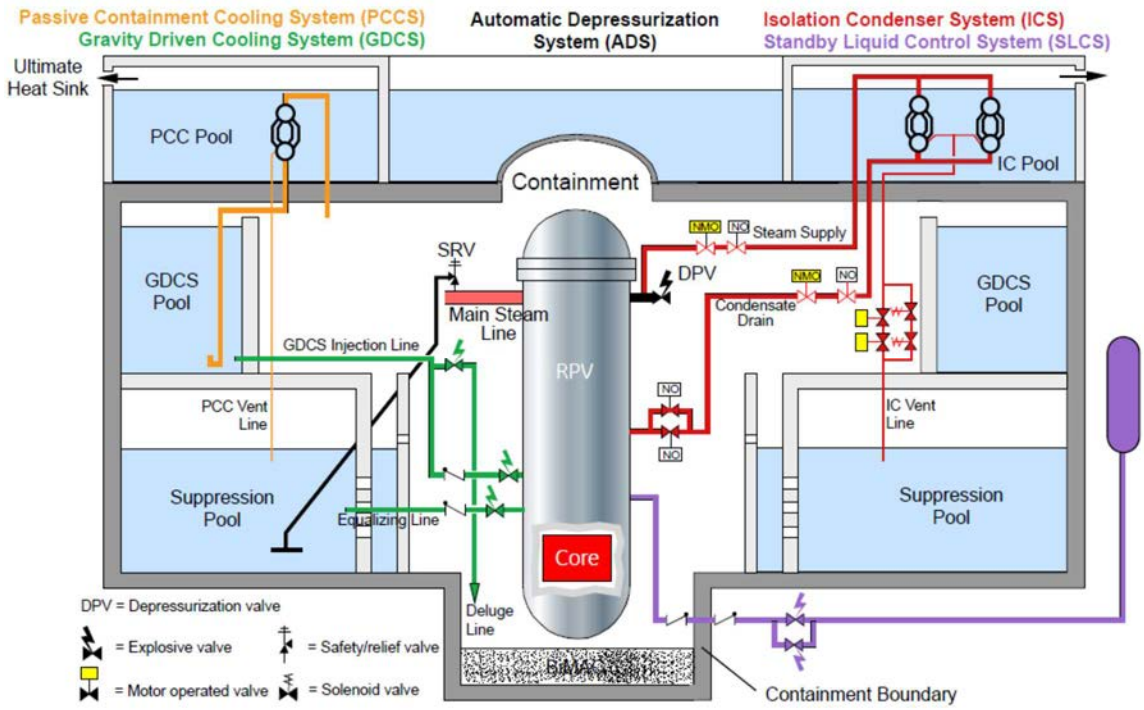


FIG. I-13. ESBWR safety system configuration (Reproduced courtesy of GE Hitachi Nuclear Energy) [27].

I.1.4.1. Isolation condenser system (ICS)

The IC system is composed of four independent loops where each loop has a heat exchanger submerged in the IC pool and the pool is vented to the atmosphere. The system is designed to passively remove decay heat after any reactor-isolation during power operations or when the normal heat removal system is unavailable. The steam carrying decay heat from the core is condensed inside the tube of the heat exchanger and the heat is transferred to the water in the IC pool through heating or evaporation. The heat transfer between the tubes and the surrounding water is accomplished through natural circulation. The cooling process in ICS removes the decay heat and stops further increase of the steam pressure. Eventually this maintains the RPV pressure below the safety relief valve set point.

The operation of the ICS can be manually initiated by operator or through automatic actuation signals of: high reactor pressure, MSIV closure and RPV water level signal. It starts its function by opening the condensate return valve so that the condensate occupying tubes drain into the reactor and the water steam surface boundary moves downward below the lower headers. The IC pool has capacity about 72 hours to remove decay heat without refilling. Replenishment of the pool inventory can maintain the heat removal indefinitely.

I.1.4.2. Emergency core cooling – gravity driven cooling system (GDCS)

The GDCS consists of four identical safety-related divisions in which each comprises three subsystems. The first subsystem is intended for “short term” cooling. In this subsystem each division connects water in GDCS pool located in the upper elevation of the containment to the RPV. The flow in each division is controlled by pyrotechnic-type ECCS injection valves, which has characteristic to provide permanent open flow path to the vessel after actuation. Normally, its actuation is completely automatic. However as a backup capability it can also be manually opened by operator. Once it opens the valves cannot be closed afterward. Each division of this subsection supplies water to two injection nozzles on the RPV.

The GDCS in combination with ADS provides emergency core cooling system for ESBWR. After low water level signal is accepted and the reactor has been depressurized by ADS, the GDCS will inject large volume of water into the reactor. The flow of water is driven passively by gravitational force.

The second subsystem is the equalizing line that links suppression pool water to the RPV and provides a long term inventory control function for the reactor vessel and the core cooling. Each division feeds one injection nozzle and is controlled by pyrotechnic-type ECCS valve. The nozzle is located at lower elevation than those nozzles of the first subsystem.

The third subsystem is a deluge line, which is used to flood the lower drywell region with GDCS pool water in the event of a postulated core melt sequence that causes failure of the lower vessel head and lets the molten fuel to reach the lower drywell floor. The deluge valves open when very high temperature in the lower drywell indicating a severe accident is detected. The inventory of the three GDCS pools is sufficient to flood the lower drywell cavity to a level equal to the top of the active fuel, providing RPV cooling when the postulated severe accident resulting core melt occurred.

I.1.4.3. Passive containment cooling system (PCCS)

PCCS consists of six low pressure loops that each is totally independent of one another and containing a steam condenser placed in PCC pool water. The pool is vented to the atmosphere and position above and outside the ESBWR containment. This PCC system is designed to maintain the containment below its pressure limit for design basis accidents such as a LOCA. The steam line inlet is located in drywell area surrounding the RPV. When LOCA occurs, the pressure difference created between the containment drywell and the suppression pool during a LOCA bring the steam enters the PCCS loop, and condenses on the tube side of the condenser and transfers heat to the pool water. The other end of the condenser is connected to the GDCS and the suppression pools which by gravity the condensate will move down to be collected in GDCS providing water supply to that system and the non-condensable gas is directed to the wet-well. On the PCC pool side, heat transferred from steam heats up the pool water and later evaporates the water. Without inventory refilling to this pool, the PCCS loops are capable to limit the containment pressure for at least 72 hours. Since there is no valve for each loop, PCCS operation requires no sensing and power activated devices for operation. In other words this system is always in "ready stand by" mode.

I.2. Pressurized water reactors (PWRs)

I.2.1. AP1000

The AP1000 is a Westinghouse-designed pressurized water cooled reactor with a rating of net electrical power to the grid of 1000 MW(e) and a core power of 3400 MW(th). Its design is intended to achieve a high safety and performance record based on proven technology and passive safety systems. The safety system relies on natural driving forces such as pressurized gas, gravity, natural circulation flow, and convection. As a result the number of component and system is reduced in the design and that shortens construction time and reduces the total cost. The AP1000 passive safety related system and function include the passive core cooling system, passive containment cooling system, main control room emergency habitability system, containment isolation function, passive 1E dc power system, passive containment sump water pH control and passive cooling of 1E instrumentation and control areas by the plant structure.

Passive containment cooling system (PCS)

The containment vessel of AP1000 is a free standing cylindrical steel vessel with ellipsoidal upper and lower heads and surrounded by reinforced concrete building as illustrated in *FIG. I-14*. The containment has two functionalities; to contain the release of radioactivity after postulated design basis accident and to function as safety related ultimate heat sink. Its construction and materials allow transferring the heat associated with accident to the surrounding environment [29].

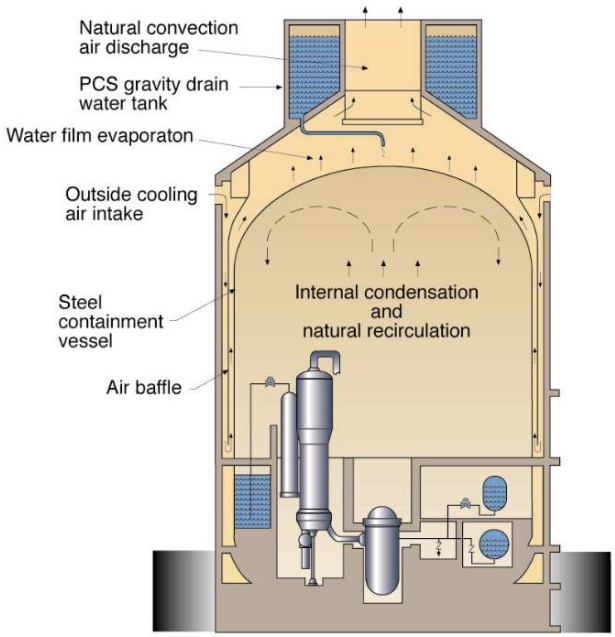


FIG. I-14. The Containment and PCS of AP1000 [17].

The purpose of the PCS is to reduce and maintain the containment temperature and pressure below a maximum value following a postulated design basis accident. It passively removes heat from the containment atmosphere and also acts as the safety related ultimate heat sink. The steel vessel provides a medium of heat transfer from inner side to the external. Steam released to the containment atmosphere condenses on the inner surface of the containment and the external side is cooled by air flow natural circulation. In case of accident, the cooling on the external surface is aided by evaporation for water which is drained by gravity from a tank located on the top of containment building. With this feature, benefit also comes to the reduction of fission product leakage. As the containment pressure decreases the driving force of leakage also diminishes due to the differential pressure reduction between containment atmosphere and external environment [28].

Containment isolation system

The primary objective of the containment isolation system is to assure that fluid lines penetrating the containment boundary are isolated in the event of an accident. This system minimizes and prevents the escape of fission products from postulated accident to the environment. It preserves the containment boundary integrity while allowing the channel of fluid through containment boundary during normal and emergency conditions. The AP1000 containment isolation is significantly improved over that of conventional PWR where the number of penetration has been greatly reduced. In addition, the normally open penetration is reduced by 60 percent.

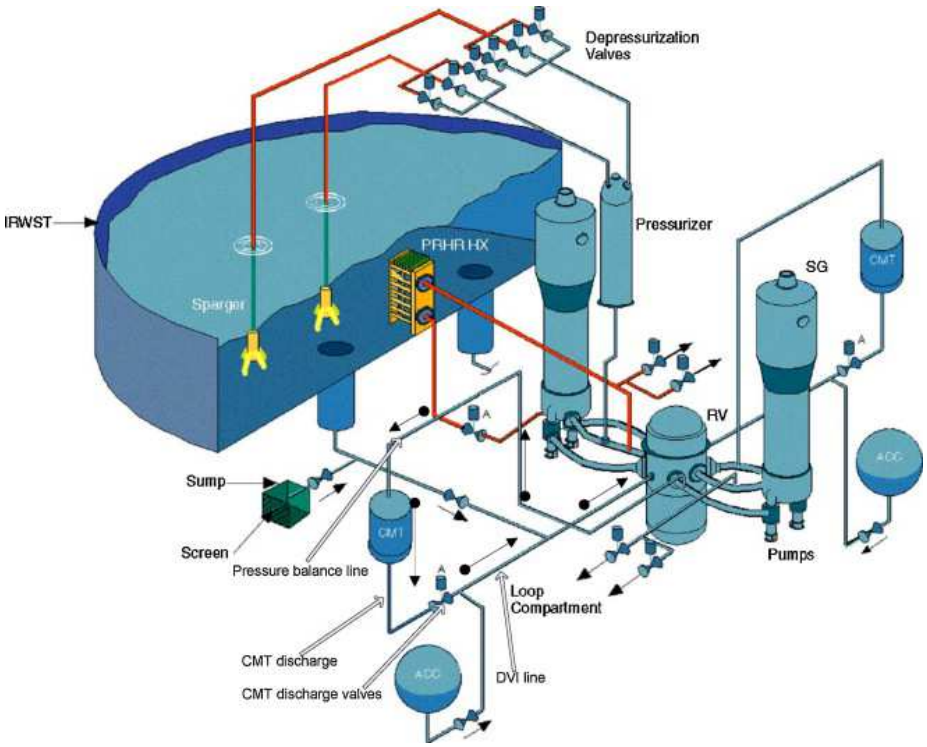


FIG. I-15. The PXS of AP1000 [30].

Passive core cooling system (PXS)

PXS is designed to provide emergency core cooling after the postulated design basis. During the transient it provides safety function of core residual heat removal, safety injection and depressurization. The system has three passive water sources to maintain core cooling through safety injection, i.e., the core make up tanks (CMTs), the accumulators, and in-containment refueling water storage tank (IRWST) as shown in *FIG. I-15*. They are directly all connected to two nozzles on the reactor vessel. For long term low pressure injection, water is provided by the IRWST which flow gravitationally as it is located in containment just above the RCS loop. The flow is controlled by squib valves and to be able to serve the function the RCS must be depressurized. The ADS depressurizes the primary system using the four stage valves which automatically reduce the pressure to about 12 psig (0.18 Mpa) to let the IRWST inject water by gravity.

The PXS has a full capacity passive residual heat removal heat exchanger (PRHR HX) which is connected to the RCS through inlet and outlet of the loop 1. The PRHR HX is capable to protect the plant against loss of feed water and feed water or steam line breaks with no operator action required. The capacity of the PRHR HX is sufficient to maintain the RCS temperature within subcooled and acceptable pressure. The heat exchanger is immersed inside the IRWST so the core heat which goes to the heat exchanger through reactor coolant natural circulation is transferred into water of IRWST. The water volume is sufficient to absorb decay heat for more than one hour before it boils up and evaporate. The evaporated steam is vented from IRWST to the containment which is then condensed on the inner surface of the steel containment vessel. The condensate then drain back to the IRWST through gravity.

Safety system to cope with severe accidents

Severe accident management feature of AP1000 is achieved through the retention of molten core debris inside the reactor vessel (in vessel retention). When the postulated severe accident occurred where the core is uncovered and melting, the reactor cavity is flooded by the IRWST water. This action submerges external surface of the vessel and thus provides outside cool down and maintain the integrity of the vessel. The molten core is prevented from relocating into the containment basement and as a result the ex-vessel hydrogen explosion and core-concrete interaction can be avoided. This feature provides high confidence that containment failure and radioactive release to the environment will not occur due to severe accident.

I.2.2. ATMEA1

ATMEA1 is a 3-loops PWR type plant with electric generation capability about 1100 MW(e) that jointly developed by two nuclear plant suppliers: the Mitsubishi Heavy Industry and AREVA. The reactor is designed using proven technologies to achieve higher thermal efficiency, better availability, and high level of safety. In addition, its deployment is adaptive to various site conditions and be able to provide flexible operability in response to customer needs. The general plant layout inside containment is shown in *FIG. I-16*.

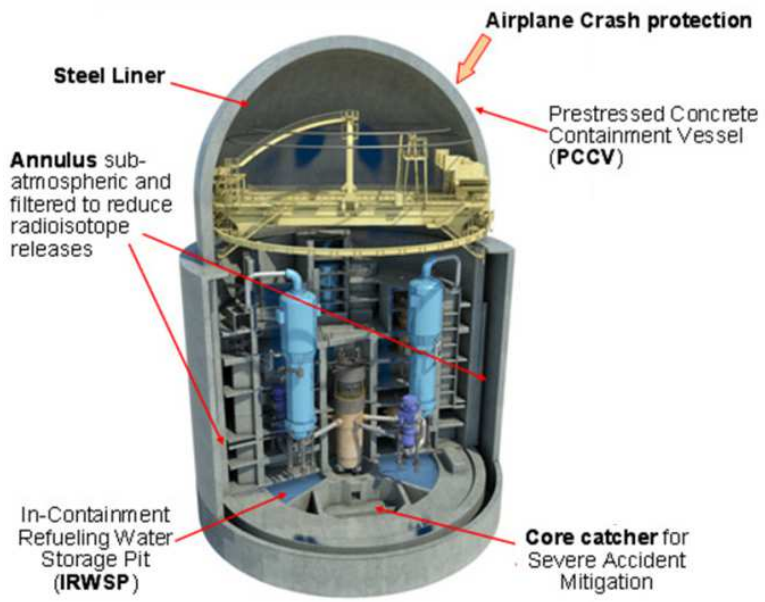


FIG. I-16. ATMEA1 plant system [33].

Typical safety system design of ATMEA1 consists of 3-trains active systems which are individually installed to the three reactor loops. Each train is adequate to mitigate accidents, to ensure safe shut-down and to perform residual heat removal of the overall reactor. The train is also complemented with passive features that are provided by advanced accumulators and in in-containment water storage. Besides, the system is arranged so that it is fully independent from other safety system and from the operational system. The trains are installed into dedicated areas, called ‘divisions’ where each division is physically separated from the others (by walls, floors, etc.) so spreading of internal hazards from one division to another can be avoided. Emergency power sources are also segregated, making them less susceptible to a common cause failure. In order to deal with

station blackout (SBO) additional alternative AC power system (AAC) or diversity in EPSs is provided. The safety system configuration to cope with the Design Basis Event is shown in *FIG. I-17*. The system includes safety injection system, advanced accumulator, and RHRS/CS [30] – [32].

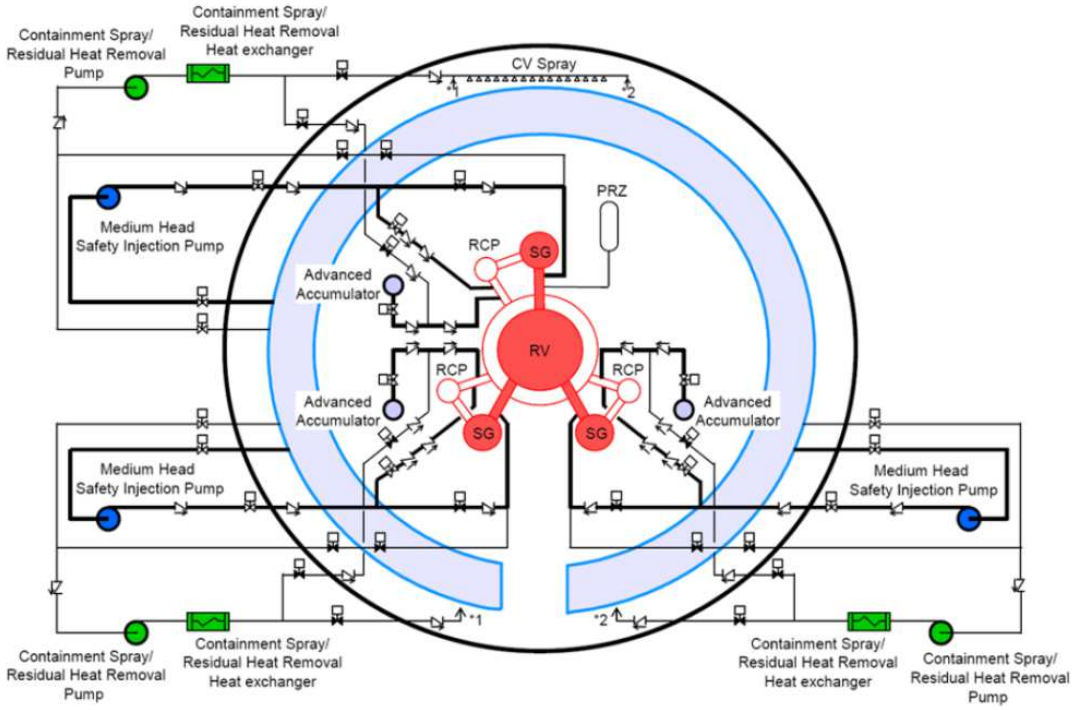


FIG. I-17. General arrangement of the ATMEA1 safety system [31].

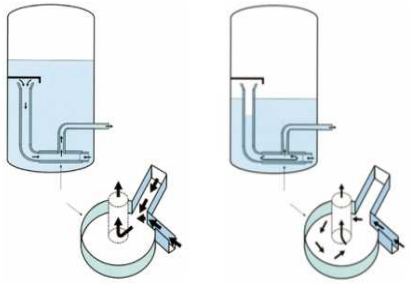


FIG. I-18. Advanced accumulator of ATMEA1 [32].

Advanced accumulator

The use of advanced accumulator in ECCS provides longer injection time to the reactor vessel post LOCA initiation. The design allows two rates of flow going out from the accumulator tank. At the beginning of blow down phase, the flow injection rate is high and then at subsequent core re-flooding phase (when water level inside the accumulator tank is lower), the flow damper switches the injection flow rate passively into low head injection. This prolongs the injection time and at the same time integrates function of low head injection system. Consequently the low head injection system (LHIS) is no longer needed

and hence provides economic benefits for the capital cost. The basic design of the advance accumulator is shown in *FIG. I-18*.

Safety injection system (SIS)

The primary function of SIS is to maintain the reactor coolant inventory following a LOCA or a main steam line break (MSLB). It consists of three identical independent trains that each is capable to inject and recirculate medium head emergency coolant to different cold legs of the RCS. Each train has a medium head safety injection (MHSI) pump and an advance d accumulator pressurized by nitrogen. The suction line is connected to IRWSP via a series of screen which protect the MHSI pumps from debris being entrained with IRWSP fluid.

In-containment refueling water storage pit (IRWSP)

The IRWSP is a water source that located at the bottom of the containment. This placement has the following benefits: i.e. in the event of LOCA the requirement to switch over from injection mode to recirculation mode after the tank is empty can be avoided and in the event of a core melt it provides water for corium cooling. In its design, the area between IRWSP and containment wall is filled with concrete to avoid water losses during an accident.

Containment spray system (CSS)/Residual heat removal system (RHRS)

The CSS is to provide containment spray injection in order to reduce and maintain the reactor building pressure and temperature within acceptable level in the event where high temperature steam is released to the containment atmosphere such as LOCA or MSLB.

The RHRS is used to perform normal shutdown cooling, maintain safe shutdown state and refuelling conditions.

The CSS and RHRS share the same injection pump and heat exchanger. The system consists of three independent trains with a separate suction connection to the IRWSP. Each train is powered by independent emergency buses and backed up an emergency power supply. They are all located in a separate division so ensure the protection against external and internal hazards.

Extra borating system (EBS)

The EBS comprise of two identical trains. Each is equipped with a boron tank, high pressures 100% capacity pump, test line and injection lines to the RCS. This system is to maintain the core subcritical for safe shutdown and also can be used to deal with ATWS in the beyond design basis event.

Emergency feed water system (EFWS)

Following a loss of normal feed water in the AOO and DBE conditions, the EFWS provides water to the SG to ensure the removal of the heat from the RCS. The system has three independent trains where each consists of a water storage pool, pump, control valves, isolation valves, piping and instrumentation and is powered by a separate electrical train. One EFWS train is placed in each division of the safeguard buildings (SB), which provides physically separate protection in coping with external and internal hazards. The EFWS has sufficient capacity to perform its required function even in a failure of one EFW train.

When a common mode failure of all EPS occurred, one of EFW pump can be powered by an alternative AC power source.

Safety system to cope with severe accidents

The ATMEA1 design addresses severe accident through several approaches. The approaches are mainly based on prevention of core melt and loss of containment which would lead to a large radioactive material early release. The approaches are:

- Prevention of high pressure core melt by using reliable decay heat removal system and primary system overpressure protection.
- Primary system discharge into containment during total loss of secondary side cooling.
- Corium spreading using sacrificial protective material which has cooling system to protect the basemat.
- Prevention of hydrogen detonation by using catalytic hydrogen re-combiners.
- Controlling the containment pressure by using a dedicated severe accident heat removal system (SAHRS).
- Collection of all leaks in an annulus and prevention of bypass of the confinement.

Countermeasure for external hazards

The ATMEA1 plant design has provisions for the impact of large commercial airplane crash by using single reactor building with high strength concrete and thicken wall (Pre-stressed concrete containment vessel). This feature complies with the regulation of European countries and also with the expected regulation by US NRC in future. With this the safety facilities are protected by segregation or bunkerization to secure the required safety functions.

ANNEX II

II. Review of Engineered Safety Features and Fukushima Action Plans of Small Modular Reactors

II.1. CAREM25

Central Argentina de Elementos Modulares (CAREM25) is a national SMR development project based on LWR technology coordinated by the Argentine National Atomic Energy Commission (CNEA) in collaboration with leading nuclear companies in Argentina with the purpose to develop, design and construct advanced small nuclear power plants with economic competitiveness and high level of safety. CAREM25 is deployed as a prototype plant that generate 31 MW(e) to validate the innovations for future commercial version of CAREM that will eventually generate an electric output of 150 MW(e).

CAREM25 is an integral type PWR based on indirect steam cycle with distinctive features that simplify the design and support the objective of achieving a higher level of safety compared with current NPP designs. Some of the design characteristics of CAREM25 are: integrated primary coolant system, self-pressurization, core cooling by natural circulation and in-vessel hydraulic control rod drive mechanisms. Another important characteristic are the passive safety systems actuating during a grace period of 36 hours, where the released energy is stored inside containment. Due to this and to the presence of additional safety features with external coolant supply, SBO is intrinsically included into the design basis of CAREM25, which strength the design in coping with extreme natural hazards.

The reactor vessel internals and major design characteristics of CAREM25 are shown in *FIG. II-1* and *TABLE II-1*.

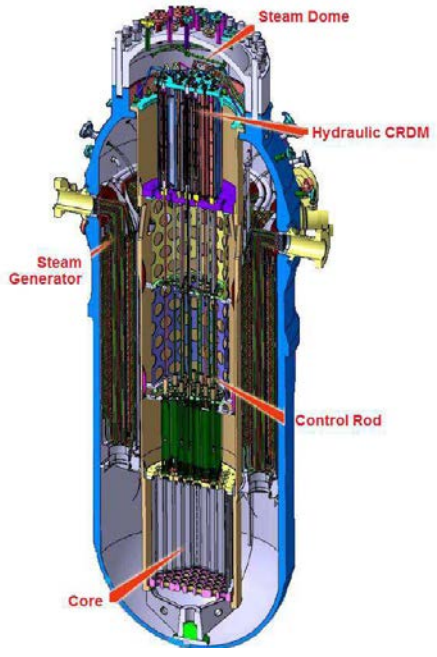


FIG. II-1. Reactor vessel of CAREM25 (Reproduced courtesy of CNEA, Argentina) [34].

TABLE II-1. DESIGN CHARACTERISTIC OF CAREM25. [34], [35]

System/ Component	Design value
Core	Power : 100 MW(th)
Primary System	Pressure: 12.25 MPa Core inlet temperature: 284°C Core outlet, riser, and dome temp. ~ saturation = 326°C Mass flow rate: 410 kg/s
Secondary System	12 identical 'mini helical' steam generator 'once-through' type, secondary system in the tube side. Secondary pressure: 4.7 MPa. Superheated steam : +30°C (290°C) Tube of similar length to equalize pressure-loss and superheating
Fuel	PWR type FA with low enriched UO_2 Enrichment: about 3.5% Refueling cycle: 14 months
Reactor Vessel	Material: SA508 Grade 3 Class 1 Lining material: SS-304L Height: 11 m Inner Diameter: 3.16 Wall thickness: 0.123 m

Design internalization of Defence in Depth concept

Defence in Depth (DiD) concept was internalized in the design of CAREM25 since the conceptual engineering. It is the base for structures, systems and components safety classification, which allows a clear assignation of design rules and requirements to systems important to safety.

The applied DiD concept is based on Western European Nuclear Regulators Association (WENRA) proposal and include clarification on multiple failure events, severe accidents and independence between levels.

The adopted approach in CAREM25 is schematically presented in *FIG. II-2* and briefly describe here:

- **Level 2:** The objective is the control of abnormal operation and failures associated with anticipated operational occurrences (AOO), by means of enhanced process and control systems (EPCS).
- **Level 3**: The objective is the control of events to avoid radiological releases and prevent escalation to core melt conditions.

Design goals: to avoid fuel damage, to avoid DNB during LOCA events, to keep the core covered and to keep the RPV and containment pressure below design limits.

Passive safety features of CAREM25 makes possible to have a grace period (36 hours) in which neither operator action nor electrical power supply is required to ensure the fulfillment of the fundamental safety functions. Based on this grace period, two different plant states are distinguished at this level from the safety point of view:

- Plant Safe State: corresponding to the plant state reached after the actuation of the safety system during the grace period (initial stage after initiating event).
- Final Safe State: corresponding to the plant state reached with active safety systems, which operate after the grace period to carry the plant to conditions equivalent to cool shutdown.

Also, in order to address the clear differences between single events and multiple failure events without core melt, this level is divided in two sublevels according to WENRA:

- *Sub-level 3a:* control of Postulated Single Initiating Events (PSIE). During the initial stage (grace period) this is done by means of the Passive Safety Systems of the Main Line of Protection, to reach the Plant Safe State. During the final stage, after the grace period, the control is done by means of the active Final Safe State Systems, in order to reach Plant Final Safe State.
- *Sub-level 3b*: control of postulated multiple failure events (PMFE). During the initial stage (grace period), and in case of failure of systems of the Main Line of Protection, this is done by means of the passive Safety Systems of the diverse line of protection, to reach the plant safe state. During the final stage, after the grace period, the control is done by means of the active Final Safe State Systems, in order to reach Plant Final Safe State. In case of failure of the final safe state systems, the extension of plant safe state systems (external supply) of this sublevel actuate for the purpose of extending the plant safe state.

- **Level 4**: The objective is the control of postulated core melt accidents (PCMA) to limit off-site releases, by means of the severe accident mitigation systems.

Design goals: to retain the corium inside RPV, to avoid hydrogen detonations and to limit iodine releases.

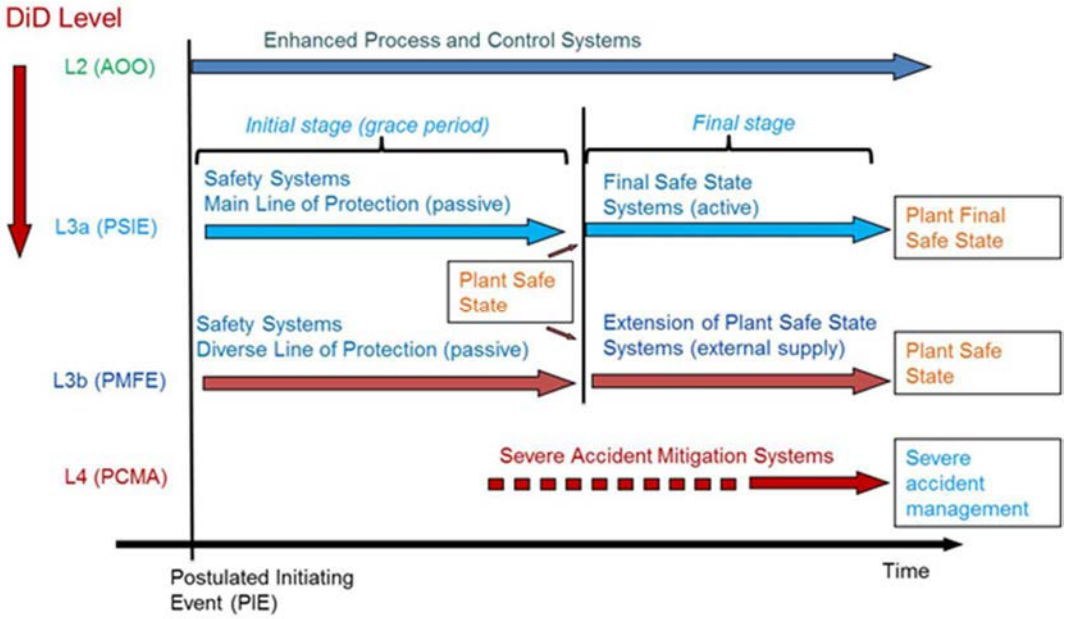


FIG. II-2. Defence in Depth internalization in CAREM25.

Engineering safety features:

A short description of the most relevant engineering safety features of CAREM25 is given below, grouped according to their corresponding level in the DiD approach explained before. Some of these systems are graphically represented in *FIG. II-3.*

- **DiD Level 3:**

INITIAL STAGE (GRACE PERIOD)

Sublevel 3a: **Safety Systems - Main Line of Protection**

- *First reactor protection system:*
  The first reactor protection is four channels redundant system that demands the actuation safety systems of the main line of protection, in order to fulfill the fundamental safety functions after the occurrence of a postulated single initiating event. It is actuated when safety systems set-points are reached for selected plant parameters.
- *First shutdown system:*
  The first shutdown system is provided by dropping neutron absorbing elements into the core through a hydraulic mechanism. In keeps the reactor subcritical in cold shutdown conditions without the need of boron injection. The hydraulic mechanism used by the control rod drive system drops control rods when the flow is interrupted, so any malfunction of any powered part of the hydraulic circuit (i.e., valve or pump failures) causes immediate shutdown. In addition, the whole control rod drive system is located inside the RPV so large LOCA possibility is eliminated.
- *Passive residual heat removal system (PRHRS):*
  This system is a simple, passive, reliable and redundant apparatus that works by means of natural circulation. Its main function is to depressurize the RPV by removing the decay heat generated after the occurrence of a postulated single initiating event. It consists of two redundancies; each of them includes emergency condensers, connecting pipes to the RPV and inlet-outlet valves. The condensers are immersed in cold water pools inside containment building and compose of parallel, horizontal u-tubes with two

common headers. The top header connects to the reactor vessel steam dome and the lower header is connected to the reactor vessel at a position below the reactor water level. During normal operation, the valves in the steam line are always open, while the outlets valves are normally closed. When the system is initiated, the outlet valves open automatically and the steam from the reactor primary system will flow into the tubes bundle and will condense on the tubes inner surface. The condensate returns to the reactor vessel establishing a natural circulation loop. During the condensation, the heat of the steam is transferred to the water pool, providing cooling for the primary system. On the other side, the water in the pool will boil and evaporate. The evaporated water is then routed to the suppression pool of containment for condensation.

- *Passive safety injection system (PSIS):*
  This system consists of two redundant borated water accumulators connected to the RPV with primary function is to prevent core un-cover in case of LOCA. During LOCA, when the pressure in the reactor vessel becomes relatively low (2 MPa), the rupture disks separating the accumulator tanks and the RVP will break. As a result, the core will be flooded and the injected water provides uncover prevention for the grace period. Should the area of LOCA is very small with failure of the steam generators heat removal, the PRHRS is also activated to help the primary system depressurization.

- *Pressure suppression containment:*
  The containment pressure suppression pool represents the main heat sink after the occurrence of a postulated single initiating event during the grace period. The steam released into the containment in case of LOCA is intended to be condensed in the suppression pool water, as well as the steam generated by the actuation of the PRHRS or the steam discharge through the RPV safety valves if requested.

Sublevel 3b: **Safety systems - Diverse line of protection**

- *Second reactor protection system:*
  The second reactor protection system is in charge of performing the initiation of the second shutdown system of the diverse line of protection. This system is redundant and diverse with respect to the first protection system.

- *Second shutdown system:*
  The second shutdown system is a gravity driven injection system that injects borated water in the core at reactor pressure when the Second reactor protection system detects the failure of the first reactor shutdown system. This system consists of two redundancies, each of them comprise a tank located in the upper part of the containment as can be seen in *FIG. II-3*. When the system is actuated, the valves connecting the tanks and the RPV (steam and discharge lines) will open. The capacity of single tank is sufficient to provide complete shutdown of the reactor.

- *RPV safety valves:*
  The RPV safety valves actuates in case of failure of the PRHRS, for the purpose of limiting the RPV pressure.

- *RPV depressurization system:*
  This system is manually actuated in case of failure of the PRHRS, after the operation of the RPV safety valves. Its main function is to depressurize the RPV in order to allow the actuation of the PSIS.

FINAL STAGE

Sublevel 3a: Final safe state systems

- *Low pressure injection system:*
  This system injects water into the depressurized RPV by active redundant means, after the grace period, with the purpose of allowing the actuation of the residual heat removal system. This is an active redundant system.
- *Residual heat removal system:*
  After the actuation of the low pressure injection system, this system is in charge of primary heat removal by active redundant means.
- *Suppression pool cooling system:*
  This system removes heat from the suppression pool by active redundant means, in order to depressurize the containment and to keep the suppression pool water subcooled.
- *Drywell containment spray:*
  The drywell containment spray system is required, in addition to the suppression pool cooling system, to depressurize the containment to the final safe state in case of LOCA. Due to the air accumulation inside the suppression pool chamber after a LOCA the suppression pool cooling system alone is not capable of depressurizing the containment until the final safe state and the spray system is needed. This is an active redundant system.
- *Components cooling system and ultimate heat sink:*
  The components cooling system represents the ultimate heat sink for the active heat removal systems of the final safe state systems. This is an active redundant system.
- *Emergency electrical power supply:*
  This system is comprised by redundant non-1E diesel generators and gives support to the final safe state systems.

Sublevel 3b: Extension of plant safe state systems

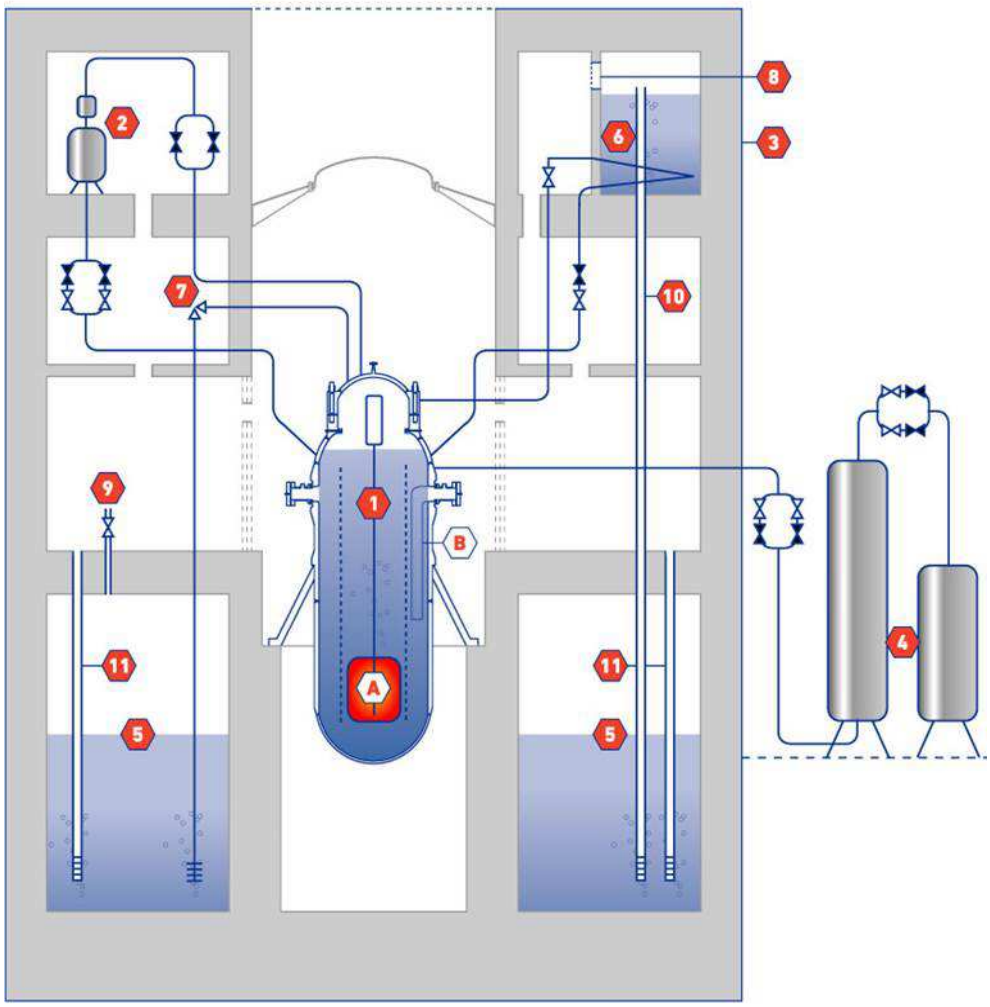
These systems are required in case of very extreme events as a SBO longer than the grace period (failure to recover off and on site electricity supply) or common cause events that affect some or all the final safe state systems.

- *RPV water injection system by external means:*
  In case of failure of the active low pressure injection system, after grace period, this system injects water into the depressurized RPV through off-site fire engines in order to keep the core covered. Diverse water sources are considered.
- *PRHRS pool water injection by external means:*
  In case of failure of the active residual heat removal system, after grace period, this system is capable of injecting water into the pools of the PRHRS through off-site fire engines for the purpose of extending the operation of that system. Diverse water sources are taken into account. Complementary to this system that injects water, one of the following systems is required in order to fulfill the heat removal safety function.
- *Suppression pool cooling (heat exchanger with external water supply):*
  In case of failure of the suppression pool cooling system, after grace period, this system is capable of removing heat from the suppression pool using off-site fire engines to pump the water needed for its operation. Diverse water sources are considered.

- *PRHRS chamber cooling (heat exchanger with external water supply):*
  In case of failure of the active residual heat removal system, after grace period, this system is capable of removing heat by condensing the steam of the PRHRS chamber using off-site fire engines to pump the water needed for its operation. Diverse water sources are considered.

- **DID Level 4: Severe accident mitigation systems**

- *Hydrogen control system:*
  This system comprises of passive autocatalytic re-combiners installed at different containment positions for limiting hydrogen concentration, in order to avoid possible deflagrations or detonations that could damage the containment.
- *In-vessel corium retention:*
  This system removes the heat generated in the corium located in the lower RPV head during the late phase of a severe accident by cooling the external surface of the reactor vessel. This is done by submerging the lower part of the vessel in water that is injected through off-site fire engines. Diverse water sources are considered. This system allows maintaining the integrity of the vessel and the corium inside it.
- *Iodine suppression pool retention (pH increase):*
  This system is intended to keep dissolved in the suppression pool water, significant quantities of the iodine released during a severe accident, to limit off-site releases. This is done by injecting an alkaline solution into the suppression pool water in order to increase its pH, which prevents the formation gaseous iodine.



(A) Reactor core
(B) Steam generators
(1) First shutdown system
(2) Second shutdown system
(3) Reinforced concrete containment
(4) Passive safety injection system (PSIS)
(5) Pressure suppression pool
(6) Passive residual heat removal system (PRHRS)
(7) Primary depressurization system
(8) PRHRS chamber relief devices
(9) Suppression pool chamber relief valves
(10) PRHRS chamber relief ducts
(11) Drywell relief ducts

FIG. II-3. Some relevant engineering safety features of CAREM25 (Reproduced courtesy of CNEA, Argentina).

II.2. SMART

General Description

The System-integrated Modular Advanced Reactor (SMART) is an integral type SMR with a rated power of 330 MW(th) (100 MW(e)) for multi-purpose applications. It contains

major primary components such as pressurizer, steam generators and reactor coolant pumps in a single reactor pressure vessel. The integral arrangement of the reactor coolant system removes large bore pipe connections, resulting in the elimination of the large break loss of coolant accident (LBLOCA) from the design bases events.

The reactor vessel has dimension of 6.5 m in diameter and of 18.5 m in height. The large volume in-vessel pressurizer controls the system pressure at a nearly constant level over the entire operating conditions. Eight (8) helically coiled once-through steam generators produce 30°C superheated steam under normal operating conditions. Four (4) canned motor reactor coolant pumps inherently prevent coolant leakage associated with pump seal failure. The low power density design with low enriched (< 5 w/o) UO_2 fueled core provides a thermal margin of greater than 15% to accommodate any design basis transients. Reactivity control is achieved using control rods and soluble boron, and burnable poison rods are introduced for uniform power profile. Four (4) channel control rod position indicators contribute to the simplification of the core protection system and to the enhancement of the system reliability. The safety of the SMART is assured by the sensible combination of the passive and active engineered safety systems together with severe accident mitigation features. The general arrangement of SMART reactor and its basic design information are provided in *FIG. II-4* and *TABLE II-2* respectively. SMART obtained the standard design approval on July 4th, 2012 from the Korean nuclear regulatory authority. Currently, design upgrade program is underway to incorporate full passive safety system to the existing design.

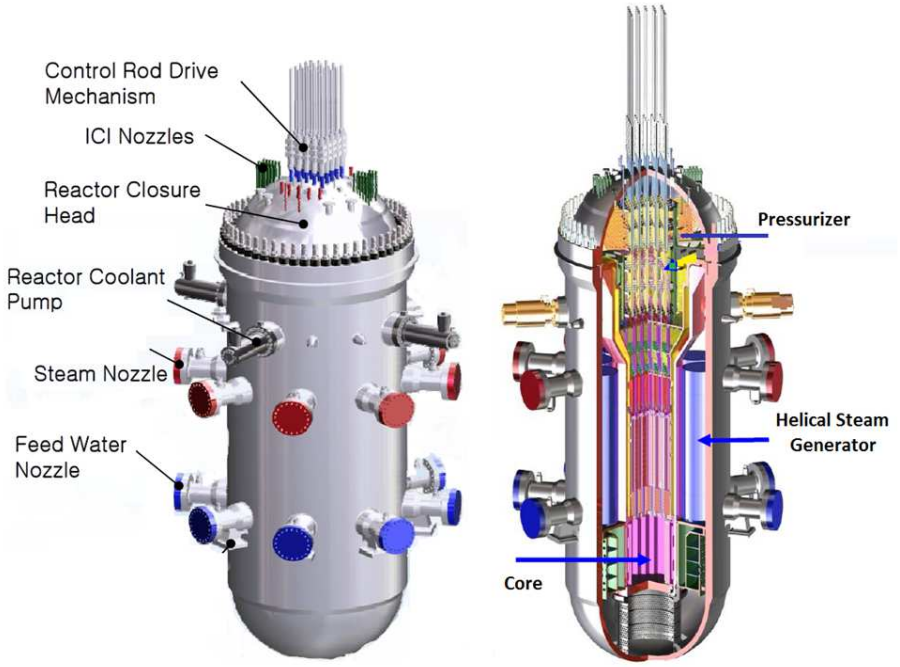


FIG. II-4. Reactor system configuration of SMART (Reproduced courtesy of KAERI) [36].

TABLE II-2. *Basic design information of SMART*

General Information	
Reactor type	Integral PWR
Power (MW(th)/MW(e))	330/100
Design life time (yr)	60
Fuel and Reactor Core	
Fuel type	17x17 Square FA
Fuel material	UO_2 ceramic (< 5.0 w/o)
Active core length (m)	2.0
Refuelling cycle (months)	36
Reactor Coolant System	
Design pressure (MPa)	17
Design temperature (^oC)	360
Core outlet temperature (^oC)	323
Core inlet temperature (^oC)	296
Minimum flow rate (kg/s)	2090
Steam pressure (MPa)	5.2
Steam temperature (^oC)	298
Primary Components	
Steam generator (8)	Helically coiled, once-through type
Steam generator tube material	Inconel Alloy 690
Reactor coolant pump (4)	Glandless canned motor pump
Control rod drive mechanism (25)	Magnetic-jack type

Engineered safety features

The safety approach for design and operation of SMART is based on the defense-in-depth philosophy. Multiple barriers such as fuel pellet, cladding, reactor coolant pressure boundary, and containment prevent radioactive release to environment. Multiple and diverse systems are designed to remove heat for the protection of those barriers. Safety systems of SMART, a sensible mixture of proven technologies and advanced design features, are designed to function automatically on demand. They consist of shutdown cooling system, passive residual heat removal system, passive safety injection system,

reactor overpressure protection system, passive containment cooling system, and severe accident mitigation system.

The reactor can be shut down under any circumstances by inserting control rods or boron injection. The passive safety system of SMART maintains the plant in a safe shutdown condition following design basis accidents such as LOCA or non-LOCA transient events without safety grade AC power or operator actions. Passive residual heat removal system prevents over-heating and over-pressurizing of the primary system in case of emergency situations where normal steam extraction or feed water supply is unavailable. It removes the decay and sensible heat by natural circulation of a two-phase fluid. The core is maintained undamaged without any corrective action taken by the operator for at least 72 hours. The reactor overpressure is prevented through the opening of the pressurizer safety valve. The containment building is resistant to any kind of seismic activity and can withstand possible air-crash incident. A schematic diagram of the safety systems of the SMART is shown in *FIG. II-5*. *TABLE II-3* provides summary of the ESFs adopted in the SMART.

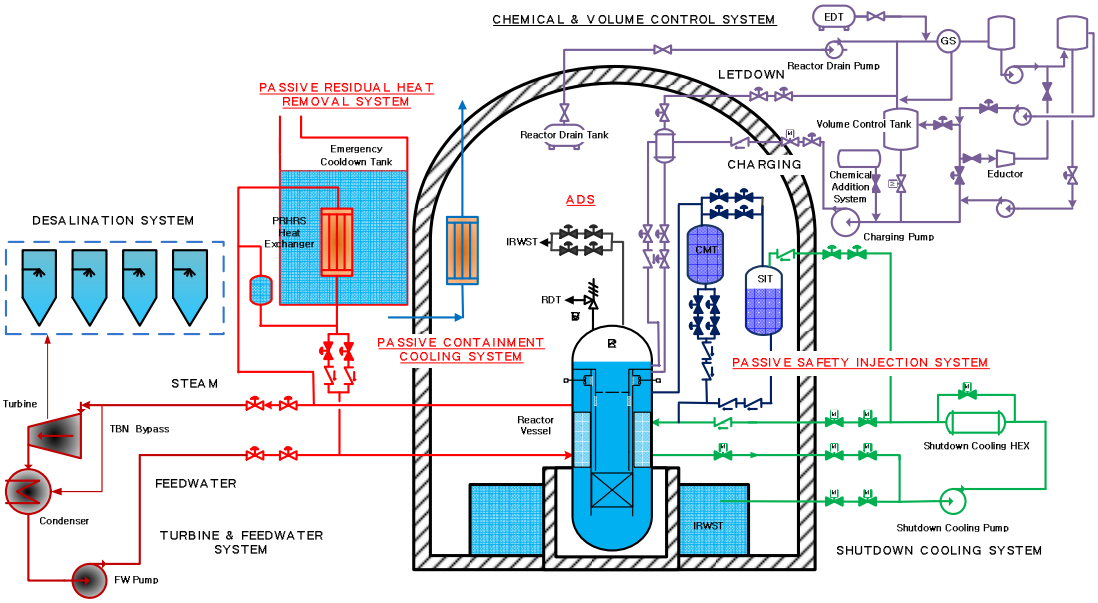


FIG. II-5. Schematic diagram of passive safety systems of SMART (Reproduced courtesy of KAERI).

TABLE II-3. SUMMARY OF ENGINEERED SAFETY FEATURE FOR SMART.

System	No. of Trains	Capacity/Train	Remark
Shutdown cooling system	2	100%	
Passive residual heat removal system	4	50%	
Safety injection system	4	100%	
Containment spray system	2	100%	
Safety depressurization system	2		

TABLE 26 (cont.)

System	No. of Trains	Capacity/Train	Remark
Passive auto-catalytic re-combiner	(12)	200%	
Basic safety requirements: • Core Damage Frequency $< 10^{-6}$ / RY • Large Radioactivity Release Frequency $< 10^{-7}$ / RY • Core Thermal Margin > 15 %			

II.2.1. Shutdown cooling system (SCS)

The SCS cools down the reactor from the safe shutdown temperature to the refuelling temperature, and maintains the RCS refuelling condition for extended period. These roles are defined as non-safety function in the passive nuclear plants. The relief valve, which is installed at the inlet pipe of SCS, provides a low temperature overpressure protection (LTOP) function of the RCS. In addition, the SCS is designed to provide the safety injection tank (SIT) refilling function which is required after 72 hours following the design basis accidents. The SCS consists of four mechanically independent trains for the SIT refilling function. Each train is composed of a pump, valves, pipes, and monitoring instrumentation. Two trains share a heat exchanger.

II.2.2. Passive residual heat removal system (PRHRS)

After the reactor is shutdown, when the normal decay heat removal mechanism utilizing the secondary system is not operable by any reason, the PRHRS brings the RCS to the safe shutdown condition within 36 hours after accident initiation and to maintain the safe shutdown condition for at least another 36 hours, therefore total 72 hours without any corrective action by operator for the postulated design basis accidents. The safety function of PRHRS is maintained continuously for a long term period when the emergency cool down tank (ECTs) is replenished periodically.

The PRHRS consists of four independent trains with a 33.3% capacity each, and each train is composed of an ECT, a heat exchanger and a makeup tank. In the design of PRHRS the possibility of loss of one train by a single failure is eliminated. Each train of PRHRS has a pair of check valves and isolation valves, both of which are installed on parallel lines. Therefore even if one valve is failed, the whole train is still in operation. And also to remove the possibility of common mode failure, diversity of actuator is provided. Two kinds of different isolation valves are adopted, which are air-operated and electro-hydraulic valves. Therefore, a single failure is no longer an issue in the design of PRHRS and three out-of-four trains are enough to remove the residual heat after an accident occurs.

II.2.3. Passive safety injection system (PSIS)

Passive safety injection systems provide emergency core cooling for at least 72 hours following postulated design basis accidents without operator actions or AC power. Emergency core cooling is performed through the four (4) core make-up tanks (CMTs) and four (4) safety injection tanks (SITs). Core cooling inventory is maintained through passive safety injection of CMTs and SITs.

The four (4) CMTs with full of borated water provide makeup and boration functions to the RCS during early stage of LOCA and non-LOCA. The top and bottom of CMT are connected to the RCS through the pressure balance line (PBL) and the safety injection line (SIL), respectively. Each SIL is isolated by 2 x 2 parallel closed valves, which meets single failure criteria. Each PBL is normally open to maintain pressure of the CMT at RCS condition. This arrangement enables the CMT to inject water to the RCS by gravity when the isolation valves are open. The isolation valves of the CMT injection line are signaled to open by pressurizer low pressure, containment high pressure and low steam line pressure, etc.

The four (4) SITs are filled with borated water and nitrogen gas at atmospheric pressure. They provide makeup water to the RCS at low pressure conditions for about 72 hours in case of LOCA. The arrangement of the SIT is similar to that of CMT except that the isolation valves are installed at PBL. The isolation valves of the SIT injection line are open when the RCS pressure reaches the set point pressure.

II.2.4. Automatic depressurization system (ADS)

The ADS lowers RCS pressure so that gravity head injection from SIT becomes available in case of LOCA. The ADS can be manually opened for total loss of feed water (TLOFW). In addition, the ADS can provide RCS depressurizing for easy connection of shutdown cooling system (SCS). The ADS consists of two (2) trains with two (2) stages each. One train of ADS is operated when the water level of CMTs reaches the set point level, and another train is operated by set point level of SITs.

II.2.5. Reactor overpressure protection system (ROPS)

The reactor overpressure protection system (ROPS) consists of two pressurizer safety valves (PSVs) installed on the top of the reactor vessel head assembly. The purpose of the ROPS is to reduce the reactor pressure when a postulated design basis accident related with a control system failure occurs. Should the primary system pressure increases over the predefined set point, the safety valves open rapidly to discharge the steam into the reactor drain tank (RDT).

II.2.6. Passive containment cooling system (PCCS)

The safety functions of PCCS are to reduce containment pressure and temperature for main steam line break (MSLB) or LOCA, and to remove fission products from the containment atmosphere following LOCA. The PCCS consists of four independent trains with a 33% capacity each, and each train is composed of a heat exchanger inside, connecting pipe and valves. The PCCS dissipates the heat from the containment atmosphere to the environment through the heat exchanger by natural circulation of working medium.

II.2.7. Severe accident mitigation system (SAMS)

The reactor pressure vessel is located in a cavity that can be flooded when a severe accident occurs. The design allows the SAMS to fill the air gap under the RPV with water from the in-containment refuelling storage tank (IRWST). This function provides an external cooling to the vessel which then prevents the egress of the corium out of the RPV during severe accident (in-vessel retention). Water in the IRWST floods the cavity by gravity.

The combustible gas control in the containment of SMART is performed by the passive auto-catalytic re-combiners (PARs) and containment purge system. Twelve (12) PARs installed in the containment, designed with 200% capacity, prevent the accumulation of hydrogen without external power supply.

Fukushima Action Plan

Ever since the Fukushima Daiichi accident, mitigation measures and facilities to cope with severe accidents cause by extreme natural hazards have become the key safety issues to nuclear power plants. Fukushima Daiichi accident showed that securing the continuous cooling capability of reactor core and spent fuel pool is essential to maintain the nuclear power plant safe.

After the Fukushima Daiichi accident, the Korean nuclear regulatory authority formed a task team to perform a through safety audit on all the nuclear plants and facilities under operation, construction, or development. As a result, fifty (50) improvement orders (action items) were derived to enhance the safety level against Fukushima Daiichi type extreme hazards including earthquake, tsunami, and flooding. Among them, ten (10) items are applied to SMART while other items are not directly relevant to the SMART design. *TABLE II-4* shows the post-Fukushima action items applied to the standard design of SMART.

TABLE II-4. POST-FUKUSHIMA ACTION ITEMS APPLIED TO SMART.

No.	Action Item	Resolution
1.	Automatic Reactor Shutdown following an Earthquake > 0.18g	Resolved in SSAR[1)]
2.	Strengthen Aseismic Design for Main Control Room Panel	Resolved in SSAR
3.	Provide Water tight Door and Drain Pump	Resolved in SSAR
4.	Secure Mobile Generator and Connection Points	Resolved in SSAR
5.	Improve Design Requirements of Alternate AC – capacity, diverse cooling, fuel supply	Resolved in SSAR
6.	Fix up Extra Transformer Anchor Bolt	To be resolved in PSAR
7.	Prepare Measure to Cool-down Spent Fuel Pool	Resolved in SSAR
8.	Prepare Anti-Flood & Recovery for Final Heat Removal	Resolved in SSAR
9.	Provide Passive Hydrogen Control Device (PAR)	Resolved in SSAR
10.	Provide External Injection Path on Safety Injection Line	Resolved in SSAR

[1)] Standard Safety Analysis Report

It is clear that maintaining continuous and proper core cooling capability after shutdown of a reactor is very important. A series of realistic simulation showed that the SMART PRHRS working on the secondary side of steam generators effectively removes decay heat, and maintains the reactor in a stable condition for 20 days without external power and operator action. This grace time can be extended continuously when the PRHRS emergency cool down tanks (ECTs) are periodically replenished.

For the hydrogen control, assuming 100 % oxidation reaction of fuel zircaloy cladding with steam after severe accident, relatively large containment volume of SMART limits the

average hydrogen density in the containment atmosphere below 5 volume %. Twelve (12) PARs effectively remove the accumulation of the hydrogen, preventing possibility of hydrogen explosion.

II.3. IRIS

IRIS is a medium sized PWR which applies an integral reactor coolant system layout in its design. It offers a formation of single or multiple modules in one site, each having power rating of 1000 MW(th) (about 335 MW(e)). The IRIS reactor vessel houses all nuclear steam supply system components such as the nuclear fuel and control rods, coolant pumps, steam generators, pressurizer, control rod drive mechanism and steel reflector. Because of that, the size of IRIS reactor vessel is larger than the traditional loop-type PWR. Its dimension has a diameter about 6.2 m and an overall height of 22.2 m. Some other design parameters are presented in *TABLE II-5*.

Inside the reactor pressure vessel, the primary coolant moves upward through the core, the riser region, and the place between the extended core barrel and RV inside wall where the reactor coolant pumps is located. There are 8 pumps employed to circulate the primary coolant. The direction of the coolant flows after the pumps is downward, going into each corresponding helical coil steam generator module. The coolant continues down through the annular down comer region and then back to the core. The configuration of the reactor internal can be seen in *FIG. II-6*.

The IRIS design provides multiple level of defence. It also applies a very basic level of DiD i.e., elimination of accident initiator**,** in addition to the traditional method using barriers, redundancy, diversity, etc. This implementation is known as “safety by design” approach.

To deal with any postulated accidents, the IRIS includes the following passive systems where natural gravitational force is used in their operation instead of active components such as pumps, fan coolers or sprays and other supporting systems [38], [39].

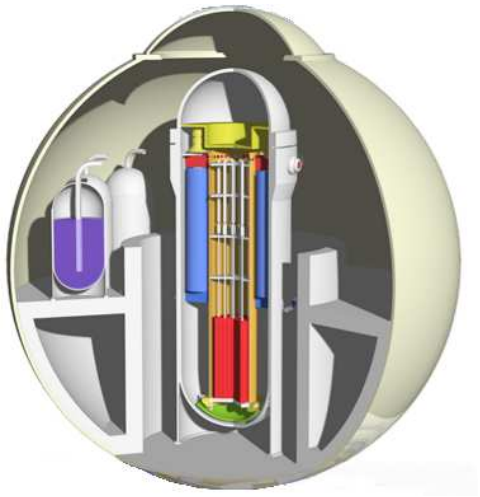


FIG. II-6. IRIS reactor vessel and containment [38].

TABLE II-5. IRIS MAJOR DESIGN PARAMETERS [39].

System/Component	Design value
Core	Power = 1000 MW(th)
Nuclear steam supply system	Integral RCS Steam temperature and pressure = 317°C , 5.8 MPa Feed water temperature and pressure = 224°C , 6.5 MPa
Reactor coolant system	Flow rate = 4700 kg/s Operating pressure = 15.5 MPa Core inlet/outlet temperature = 292/330°C
Fuel	Enrichment = 4.95% Cycle length = 40 – 48 months Average burn up = 60000 MWd/tU
Reactor Vessel	Inner diameter = 6.21 m Wall thickness = 285 mm Total height = 21.3 m
Steam generator	8 vertical, helical coil tube bundle, once through, superheated SGs. Thermal capacity (each) = 125 MW(th)
Reactor coolant pump	8 spool type, fully immersed pumps Head = 19.8 m
Primary containment	Pressure suppression, steel, spherical geometry 25 m diameters. Design pressure and temperature = 1300 kPa, 200°C

II.3.1. Passive emergency heat removal system (EHRS)

The passive EHRS consists of four independent subsystems which operate in natural circulation for removing heat from the primary system to the refueling water storage tank (RWST) located outside the containment. Each subsystem has a horizontal U-tube heat exchanger immersed in the RWST and is connected to the steam generator feed and steam lines as depicted in *FIG. II-7*. The steam flowing-in from the SG is condensed inside the EHRS heat exchanger and the condensate is returned back to the SG by gravitational force.

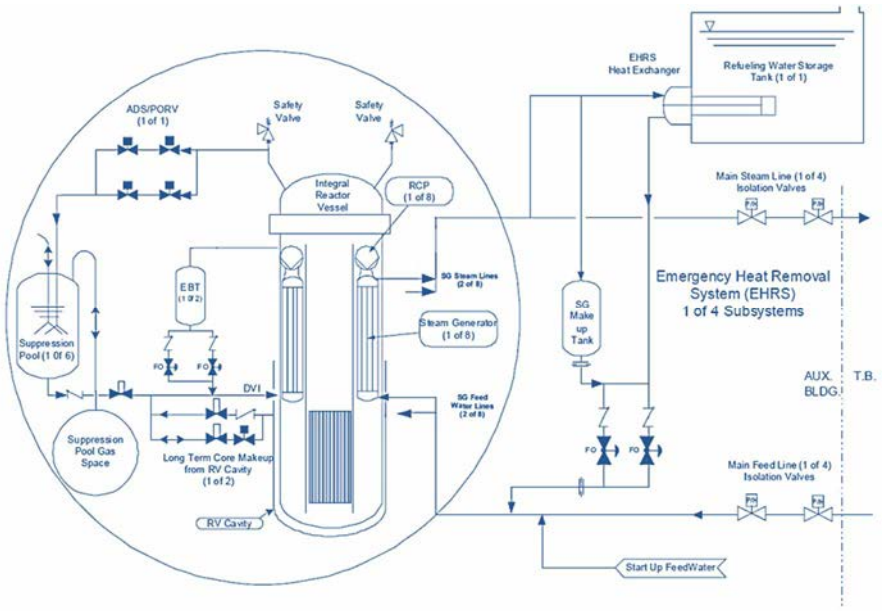


FIG. II-7. Schematic diagram of the IRIS's passive safety system [38].

A single EHRS is designed to sufficiently remove decay heat load from the reactor core when heat removal capability of the secondary system is lost. In addition to this core cooling function, the EHRS also provides the primary system post–LOCA depressurization function (depressurization with no loss of mass) by condensing the steam produced by the core immediately inside the reactor vessel. This role potentially reduces the break flow while transferring the decay heat to the environment.

II.3.2. Emergency boration tank

The emergency boration tank is intended as a diverse means to control core shutdown in addition to the control rods. Two full-systems of emergency boration are available inside the containment, capable to deliver borated water to the RV through the direct vessel injection (DVI) line when the system is actuated. Besides, the tanks also serve as a limited gravity feed makeup water to the primary system.

II.3.3. Automatic depressurization system (ADS)

The ADS consists of one leg with two parallel 4 inch lines, each with two normally closed valves. The downstream line of the valves is directed to the pressure suppression system pool tank and the upstream line is connected to the pressurizer steam space. The ADS principal function is to assist the EHRS in depressurizing the reactor vessel when/if the reactor vessel coolant inventory falls below a definite level and to ensure that the reactor vessel and the containment pressures are equalized in timely manner so the loss of coolant can be limited and the core uncover is prevented after the postulated LOCAs event occurred [38].

II.3.4. Containment pressure suppression system (CPSS)

The CPSS is a system to limit the peak containment pressure after the most limiting blow down event. This system consists of six water tanks and one common tank to store non-condensable gas. Each water tank is associated to the containment atmosphere through a vent pipe. The pipe going to the tank is a submerged sparger so that steam released to containment following loss of coolant or steam/feed line break can be condensed inside the tank. In the meantime, the non-condensable gas mixed in the steam will be collected in the

non-condensable storage tank. The containment pressure suppression system is capable to limit the pressure of the containment to less than 1.0 MPa (130 psig) which is very much lower than the containment design pressure. Besides, the water inside the suppression tank can also be used for reactor vessel injection in the event of LOCA as this water level provides an elevated source that can be driven by gravity to enter reactor vessel through DVI line.

II.3.5. Long term gravity makeup system (LGMS)

The design of IRIS containment comes with a cavity in lower containment (RV cavity) that is specifically planned to be able to collect liquid and any condensate from the containment in the event of LOCA. Following a LOCA, the water floods the cavity and at a definite elevation forms a gravity head which sufficient to provide coolant make up to the reactor vessel through DVI line. The water in cavity also ensures that the RV surface is wetted and cooled so the integrity of the vessel is maintained in the event of postulated core damage and the corium can be retained inside the vessel. In additions, LGMS also provides a path for gravity injection to the coolant system from the CPSS.

II.4. mPower

The mPower is an advanced integrated PWR reactor which has a rated power output of approximately 530 MW(th) and can be operated up to 4 years between refueling. The size of reactor vessel is approximately 83 feet long by 13 feet in diameter.

Launched by the Generation mPower LLC, the reactor design incorporates several inherent safety features and employs proven and standard technology with simpler and smaller components. Its primary system and secondary loop flows are shown in *FIG. II-8*. All the mPower's nuclear steam supply system components such as steam generator, pressurizer, core and CRDM are arranged inside the reactor vessel and the circulation of the primary coolant is powered by canned motor pumps attached at the upper part of the vessel around the pressurizer. As a result, the reactor coolant does not leave the vessel and the possibility of large LOCA in primary system like in conventional reactors is eliminated. Only small penetrations for a water level instrumentation tap, pressurizer sprays line, and for letdown, purification and make up that are connected to the vessel. These penetrations are placed well above the top of the core so it ensures that the reactor will remain be covered by water during a LOCA accident. The design of the reactor also includes large water inventory in the vessel which also guarantees that during the blow down phase in response to the LOCA there is still enough water to cover the core.

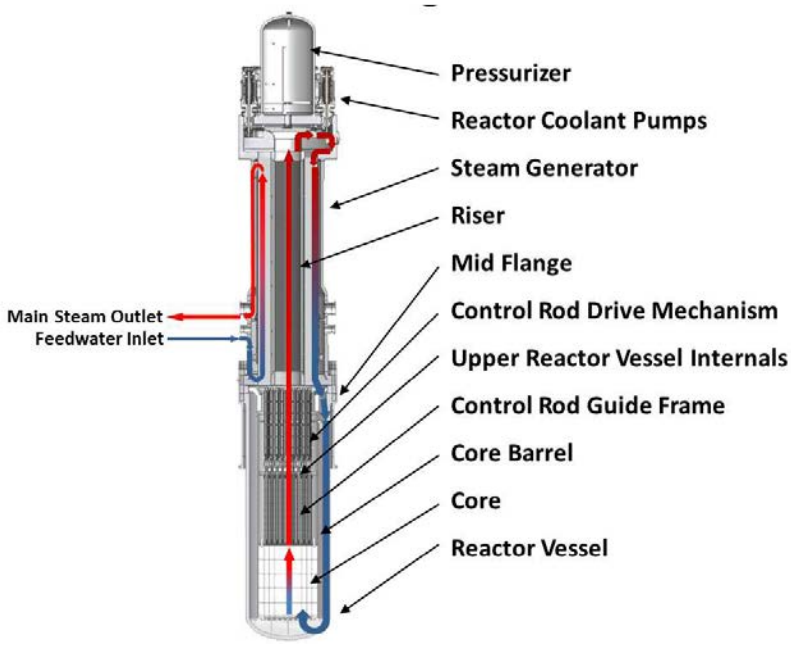


FIG. II-8. The mPower reactor diagram (Reproduced courtesy of B&W Generation mPower) [40].

The possibility of LOCA is also reduced through the application of new design of CDRM in that all parts are put inside the reactor vessel. With this technique there is no differential pressure between the mechanism and the control rod so the possibility of rod ejection is eliminated and the consequential LOCA through CRDM is removed.

In addition, the design of the core also provides low power core density which in conjunction with large water inventory improves operating margin and longer operator response time. The mPower has passive safety system that serves for the removal of decay heat, depressurization of reactor vessel, injection of low-pressure coolant and flooding of the reactor cavity. The safety system relies only on the gravity and natural circulation to remove decay heat and maintain the reactor safety. It does not require EDG to power the system. The designer claims that the large water tank inside the containment ensures the availability of on-site cooling for about 7 days, as depicted in *FIG. II-9*.

To cope with radioactive released during severe accident, the mPower's containment is located inside building with seismic category I and flood resistant structures and placed below grade level which has capability to confine the radioactive material inside the building. It is a metal, leakage free vessel and has sufficient volume to limit internal pressure for all design basis accidents. It is also passively cooled. The containment environment is suitable for human occupancy during normal operation; simultaneous refueling and NSSS equipment inspections. Passive hydrogen re-combiners are also installed to prevent hydrogen build up during core melt condition. The hydrogen re-combiners work automatically without the need of electrical power or operator action. All the safety system work together in protecting the reactor core and preventing the release of radioactive materials to the environment for at least 72 hours without operator action after the accident occurs.

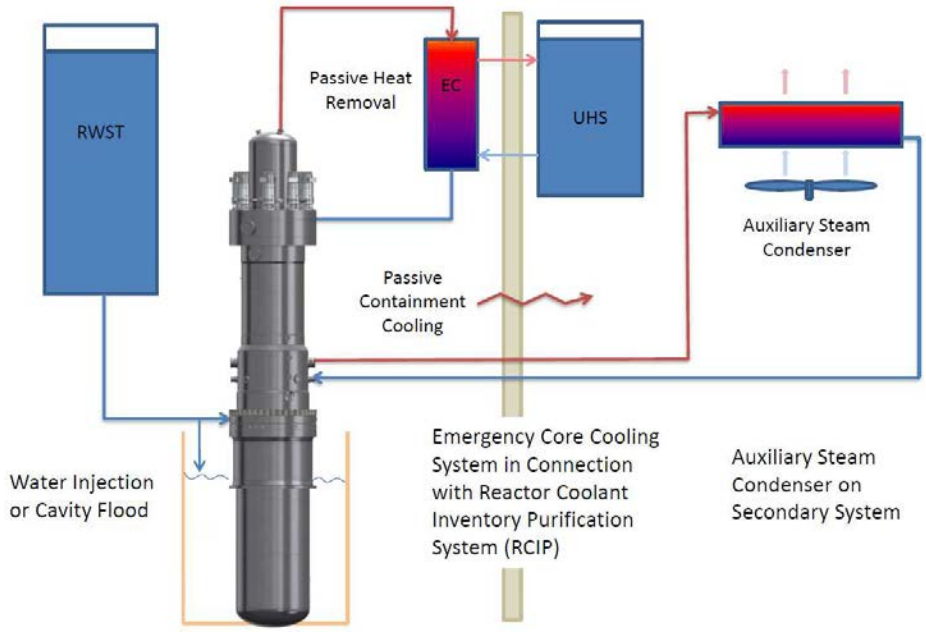


FIG. II-9. Decay heat removal strategy of the mPower [40].

II.5. NuScale

A single NuScale Power Module (NPM) is designed to produce 50 MW(e) (gross) and 160 MW(th) (gross). The 12-NPM power plant is expected to produce electric output as 570 MW(e) (net, nominal). NuScale, an integrated PWR system, is designed by taking benefits of existing knowledge and application of practical design tools of light water cooled PWR technology. The module of NuScale is entirely assembled in factory and can be delivered to the intended site by rail, truck or barge so increases its economics and deployment flexibility. The size of reactor vessel is approximately 19.2 m long by 2.8 m in diameter and is enclosed in a steel containment that is 25 m long by 4.6 m in diameter. A complete single module system and some basic plant parameters can be seen in *FIG. II-10* and *TABLE II-6.*

The RPV for NuScale contains all the nuclear steam supply system (NSSS), i.e. reactor core, helical coil steam generator and pressurizer. The steam generator is made up of two independent sets of tube bundles with separate feed-water inlet and steam outlet lines. A superheated steam is generated inside the tubes that boil the feed water injected by pump. The pressure inside the reactor vessel is controlled by a set of pressurizer heaters positioned in the upper head of the vessel.

TABLE II-6. THE NUSCALE BASIC PLANT PARAMETERS

Reactor Core	
Thermal power rating	160 MW(th)
Operating pressure	8.72 MPa
Fuel	UO_2 (<4.95% enrichment)
Refuelling intervals	24 – 48 months
Dimensions	19.2 meters x 2.8 meters (height x diameter)
Weight	264 tonnes
Containment	
Dimensions	25.0 meters x 4.6 meters (height x diameter)
Weight	303 tonnes
Power Conversion Unit	
Number of reactors	One
Electrical output	>50 MW(e) (gross)
Steam generator number	Two independent tube bundles
Steam generator type	Vertical helical tube
Steam cycle	Superheated
Turbine throttle conditions	3.1 MPa
Steam flow	71.3 kg/s
Feed water temperature	149°C

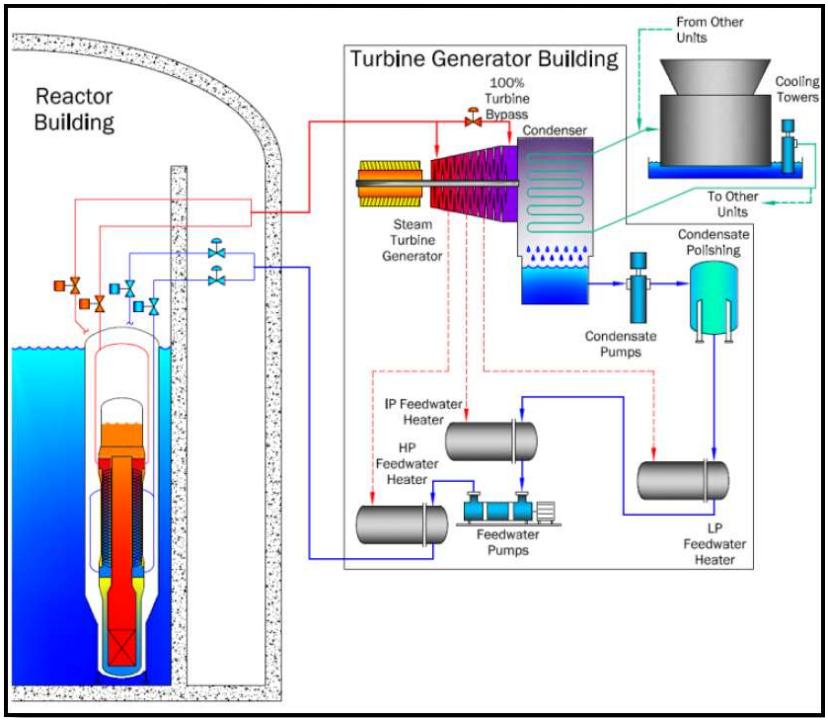


FIG. II-10. Schematic diagram of single NuScale unit [41].

To deal with any accident conditions and to maintain a stable long term core cooling under such conditions, including severe accident and its mitigation, NuScale plant design establishes a complete set of ESF consisting of high pressure containment vessel, passive decay heat removal and containment heat removal system, and severe accident mitigation system. These systems are briefly described in the following subsections [41] - [43].

II.5.1. High pressure containment

The containment of NuScale plant is dedicated for three key safety functions i.e., to contain radioactivity release in case of the occurrences of postulated accident, to guard the reactor pressure vessel and its internals from external exposures, and to be an interfacing medium for decay heat removal following an accident or normal reactor shutdown. These features distinguish NuScale's containment from current containment designs, as shown in *FIG. II-11*.

The containment vessel of NuScale is made of a steel cylinder having an outside diameter of about 15 ft and an overall height about 65 ft. This containment casings reactor pressure vessel, control rod drive mechanisms and other related components and it is placed under water in the reactor pool which provides stable cooling for the containment vessel. Under LOCA conditions this placement allows a passive heat sink for the heat removal. The containment vessel is designed to withstand the high temperature and pressure (5.5 MPa or 800 psia) of any design basis accident as well as the environment of the reactor pool [41]. The equilibrium pressure between reactor and containment following any LOCA is always below containment design pressure.

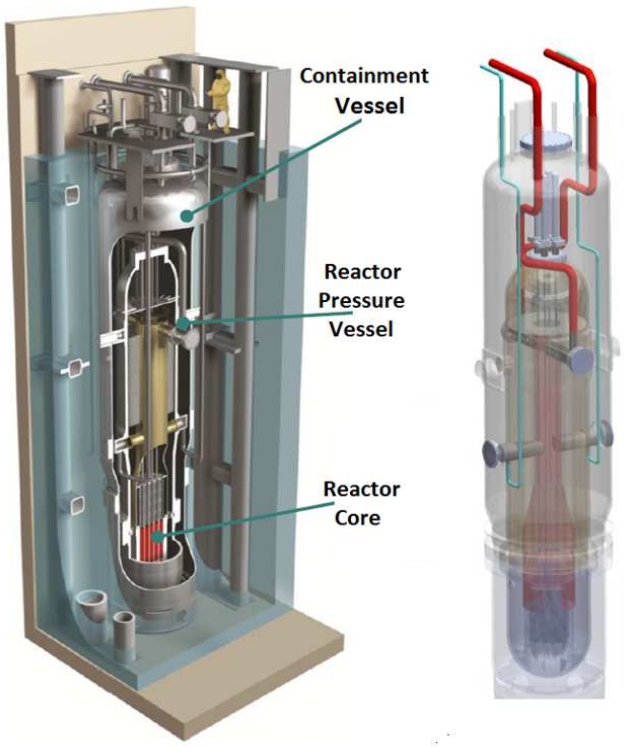


FIG. II-11. Containment vessel of NuScale [41].

Under normal operating conditions, the containment is maintained at deep vacuum so that it provides barrier against heat loss from the reactor vessel. With this method, the reactor vessel does not require surface insulation and the potential of sump screen blockage is eliminated. Moreover, the deep vacuum increases steam condensation rates when safety valve exhausts steam into this space. By maintaining the vacuum condition, the corrosion and humidity problem is also reduced and the creation of a combustible hydrogen mixture in the event of severe accident is prevented. Furthermore, due to its relatively small size,

the equilibrium pressure between the reactor and the containment vessels in the event of a small break LOCA is achieved within a few minutes.

II.5.2. Passive safety systems

NuScale module has two redundant passive safety systems to bring the decay heat from reactor core into containment pool, as illustrated in *FIG. II-12* and *FIG. II-13*. These systems are decay heat removal system (DHRS) and emergency core cooling system (ECCS). Both do not require external power to actuate. The DHRS provides secondary side reactor cooling for non-LOCA events when normal feed water is not available. Each module has two trains of closed loop, two-phase natural circulation cooling system in which each capable to remove 100% decay heat load from the core and to cool the coolant system. Every single loop has a passive condenser (decay heat removal heat exchanger) submerged in the reactor pool. When actuation signal is received, the DHR valve will open. This allows water from the condenser to travel to the helical steam generator tube bundles to take heat generated within the core and to cool the reactor coolant as it changes to steam. The steam then moves back to the condenser where it is condensed by reactor pool water.

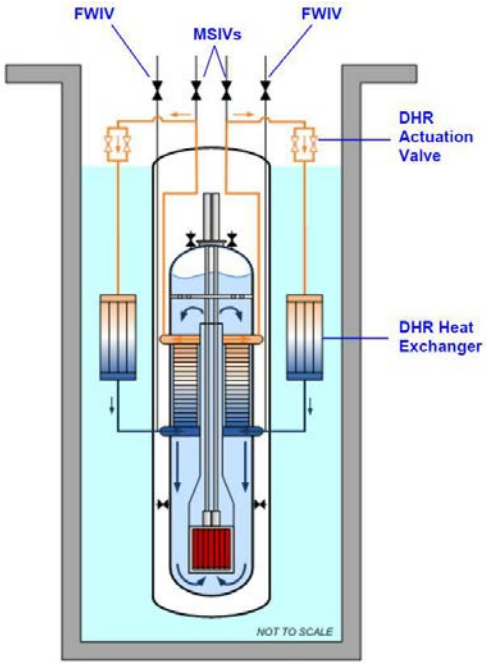


FIG. II-12. Schematic of the decay heat removal system of NuScale [41].

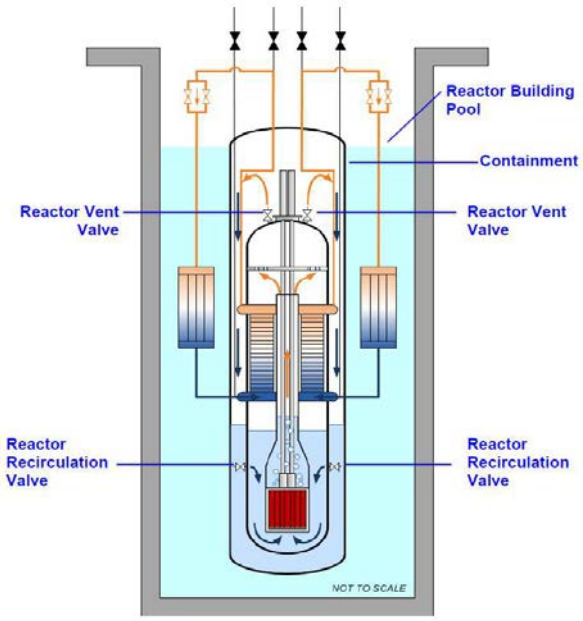


FIG. II-13. Schematic of the ECCS of NuScale [41].

The ECCS operates in the event of a LOCA or loss of the main feed water flow in combination with the loss of the DHR system. It consists of two independent reactor vent valves and two independent reactor recirculation valves. ECCS removes the core heat by opening the vent valve located in lines exiting the top of the reactor pressure vessel. The opening lets the steam from the reactor to the containment which is then cooled and condensed on the inside surface of the containment vessel by the pool water. The condensate water then accumulates in the lower containment region. When the level of water in the region rises above the top of recirculation valves, the recirculation valves open. This establishes a natural circulation path from the core to the containment.

In an accident where external heat sink to cool the reactor pool water is not available, the inventory in the reactor pool is large enough to cool the reactor decay heat for at least 72 hours without addition. After 72 hours, the pool water will start to boil off and finally after 30 days the cooling for containment will be provided by passive natural convection air cooling that is adequate for long term decay heat removal.

The ECCS valves passively open upon loss of power. With this fail-safe nature, cooling pathways are always available to remove decay heat and the reactor can be safely cooled with no AC or DC power and no operator action.

II.5.3. Severe accident mitigation design features

The NuScale plant design applies substantial severe accident mitigation features. As with other conventional light water designs, the barrier to prevent potential release of radioactive fission products to the environment in NuScale Plant is provided by the fuel pellet, cladding, reactor vessel and containment. However, NuScale also implements additional defence-in-depth by providing the containment cooling pool, the stainless steel lined containment pool structure, biological shield and reactor building to further reduce the potential for severe accident releases. Furthermore, its design offers important severe accident mitigation features such as the small fuel inventory in each module that reduces significantly the source term, the deep vacuum containment that eliminates the need for combustible gas control inside containment, the immersed containment steel in the pool which reduce the possibility of molten concrete coolant interaction and the ability to

equilibrate the pressure of reactor and containment which decrease the potential of a high pressure “corium” melt ejection. With all these combinations the NuScale plant design offers substantial advantages for emergency planning and response.

II.6. Westinghouse SMR

The Westinghouse SMR is an integral PWR that capable to supply a thermal output of 800 MW(t) and an electric output of about 225 MW(e) as a stand-alone unit. The core is made up of 89 item of 17x17 robust fuel assemblies (RFA) with active length of 8 ft. Eight axial-flow seal-less pumps mounted to the shell of the reactor vessel just below the closure flange provide the reactor coolant flow through the fuel assemblies necessary to operate the plant, as shown in *FIG. II-14*. The upper internals of the reactor support 37 control rod drive mechanisms (CRDMs) used to control reactivity. The reactor module employs an advanced evolution of a straight tube steam generator with a steam separating drum located outside of the containment vessel. The entire plant is designed for fully modular construction with all components shippable by rail, truck, or barge. Construction period predicted can be completed within an 18-24 month project schedule [44].

The plant is equipped with passive safety features derived from the AP1000 plant design. Configuration of the passive cooling system is shown in *FIG. II-14*. The main components of the passive core cooling systems are a high pressure steel containment vessel, four core makeup tanks (CMTs), an in-containment pool (ICP) and associated ICP tanks, an automatic depressurization system (ADS), an outside containment pool (OCP) and two ultimate heat sink (UHS) tanks. Connected to the CMTs are passive residual heat removal heat exchangers. Altogether, these components serve as the protection required to mitigate the various initiating faults. The safety systems is designed to safely shut down the nuclear reaction, remove decay heat following shutdown, guarantee that the reactor core remains covered with water to maintain effective cooling, and provide long term cooling and shutdown.

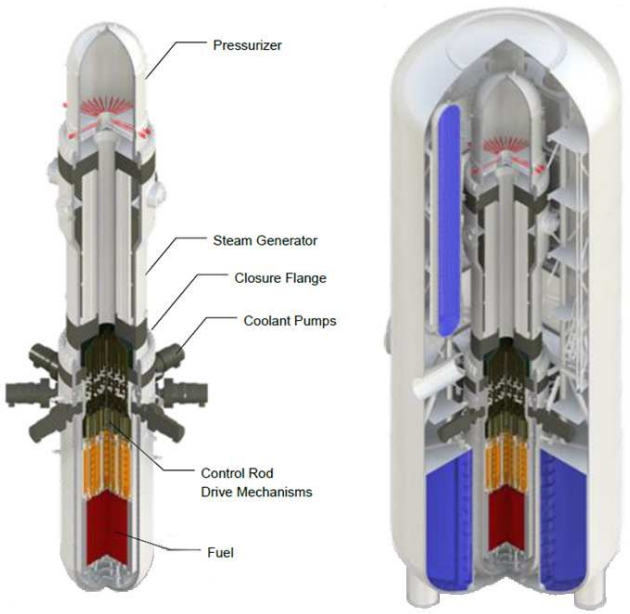


FIG. II-14. The Westinghouse SMR [45].

II.6.1. Reactivity control system

Westinghouse SMR relies on gravity driven control rods and borated water injection to control the reactor core's reactivity. The control rods will be rapidly inserted and shutdown the reactor when the protection system sends a signal to de-energize latches holding the rod out of the core. In an unlikely event that the control rods do not fall into the core or event occurs while at a shutdown condition, highly borated water is injected from CMTs. This alternate reactivity control method is performed passively through gravity-driven liquid flow. This CMT water supply is also intended for long term reactivity control of the plant.

II.6.2. Decay heat removal

The Westinghouse SMR has a unique approach to remove the core's decay heat. It uses the steam generator and large water volume in the steam drum to provide immediate residual heat removal during most of accident scenarios. The steam drum provides a gravity driven water supply to the steam generator tube and should water is heated up and became steam in the SG tubes, the steam will be going back to the steam drum again. Inside the steam drum droplet is separated and collected and sent again to the SG tube. This provides a natural circulation cooling through SG and steam drum. If power supply is available, pump is available to maintain the inventory of the steam drum and provide water supply to the steam generator tube.

For an extended station blackout condition, the plant utilizes heat exchanger to perform safety grade passive decay heat removal. The heat exchanger is integrated into each of the four CMTs. The configuration on how the heat exchanger is combined with the CMT and how the CMT is connected to the vessel can be seen in *FIG. II-15.* As shown, the bottom of CMT is connected via valves to DVI line and the upper side of CMT in connected with balanced line to the upper side of the pressure vessel. During SBO, the water inside steam drum would be empty after sometime and this will trigger the opening of the CMT valves. Upon opening of the valves, the cold water falls into the RCS beginning a natural circulation cooling loop. In the meantime, the heat exchanger within the CMTs allows for heat transfer to a secondary loop of cooling water.

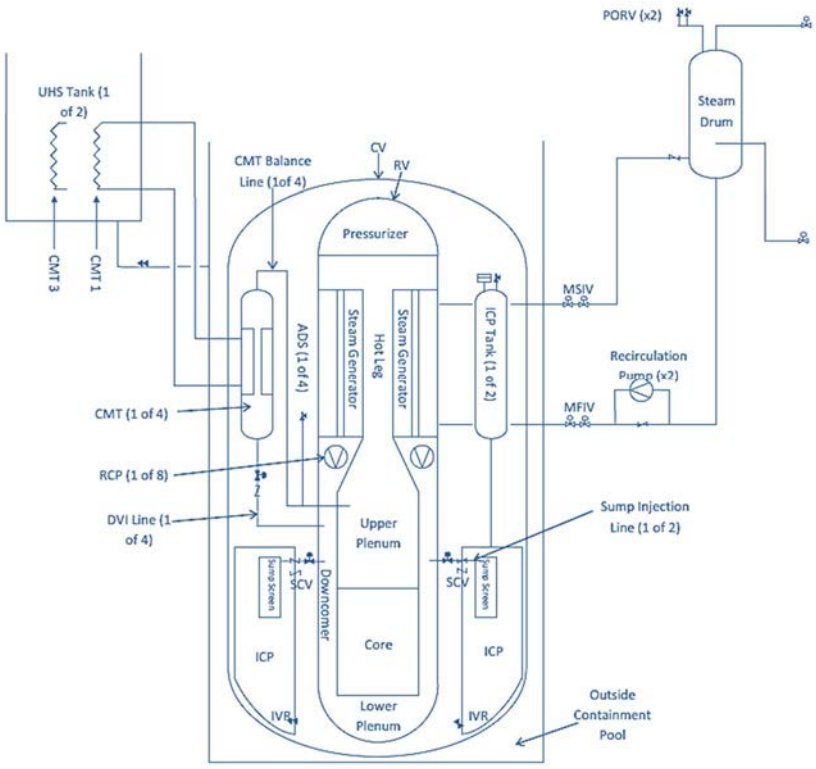
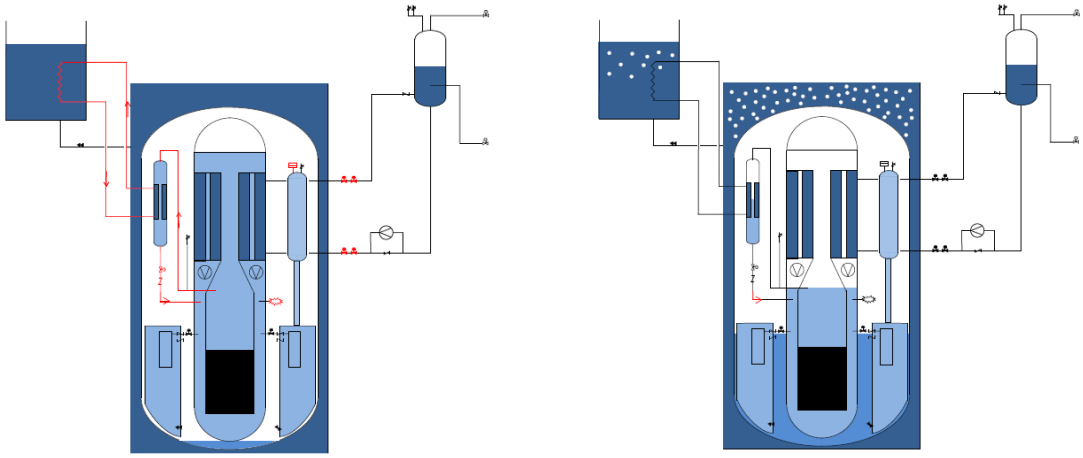


FIG. II-15. Reactor coolant and passive core cooling systems [46].

The secondary side of each CMT is connected through a closed loop of piping to a heat exchanger that sits in one of two UHS tanks. Each UHS tank is designed to accommodate decay heat removal from the core and spent fuel pool for at least 72 hours. When combined with the water in the OCP, seven days of decay heat removal capability is available. The two UHS tanks are physically separated to prevent an external event from compromising both tanks. Connections to each UHS tank allow for the addition of water to extend the decay heat removal indefinitely.

II.6.3. Safety injection

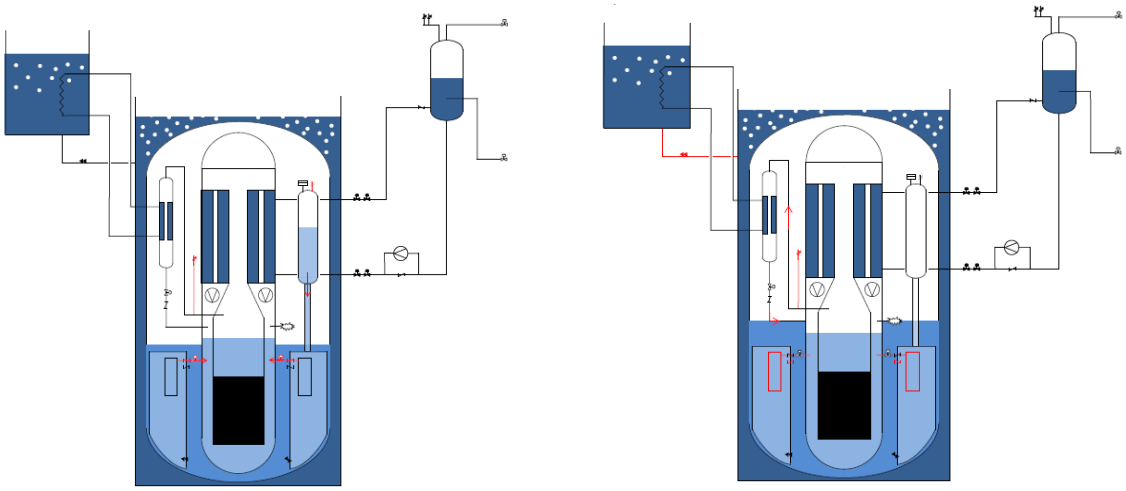
When LOCA occurs, the RCS's inventory decreases and pressure drops. The water level will keep decreasing until the protection system set point is reached. Following that, the reactor trips and the isolation valves between the SG and steam drum close. On the other hand, the valves below the CMT open. With the CMT valves open, the highly borated and cold water would flow into the RCS and the hot water from the vessel would enter the CMT, as depicted in *FIG. II-16 a*.



a. LOCA blow down *b. CMT Natural Circulation / Draining*

FIG. II-16. Loss of coolant accident in Westinghouse SMR [46], [47].

Some of the inventory going out from the vessel during LOCA will condense in the containment and collect at the bottom. As the containment pressure increase, the disk in the ICP will rupture to equalize the ICP and containment pressures. As more water inventory released from the RPV and more heat transferred through the CMT, the water in the OCP and UHS tank begins to boil. At the time the RCS level reaches the elevation of the CMT balance lines, steam enter the lines and breaks the natural circulation of liquid water. This makes the CMT drains cooling water to the RPV as can be seen in *FIG. II-16 b*. At the same time, the protection system signal is generated to open the ADS valves attached at the line of CMT, the valves at the top of the ICP tanks and the valves into the RPV. This opening allows the RCS to equalize with the containment. As the pressure goes to equilibrium, the head in the ICP tanks is high enough to inject water through check valves into RPV, as shown in *FIG. II-17 a*.



a. ADS actuation / ICP tank injection *b. Long term core cooling*

FIG. II-17. Safety injection and long term cooling [46], [47].

II.6.4. Long term cooling

As a result of draining the ICP tank and CMT, the water level in containment sump is high enough to be able to deliver water flow into the vessel. Inside the vessel, the water heats up and evaporate due to decay heat in the core. Some steam exits the vessel through ADS valves and some other condenses in CMT heat exchanger which then the condensate goes back to the vessel again. This process continues indefinitely as long as condensation on the containment wall still happens. During this period, the OCP water is boiling. The water evaporates and its level gradually drops until the level where the valves connecting the OCP with UHS tanks automatically open, as depicted in *FIG. II-17 b*. The UHS tank then refills the OCP. The capacity of each UHS tank is able to remove decay heat for about 72 hours. Connections to each UHS tank allow for the addition of water to maintain water in the pool indefinitely.

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ISBN 978–92–0–100716–2
ISSN 1011–4289