Plant system and required water pool capacity for large scale BWRs with inherently safe technologies

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Received 27 January 2015

Abstract
The Fukushima Daiichi Nuclear Power Plant accident and its consequences have led to some rethinking about the safety technologies used in boiling water reactors (BWRs). We have been developing the following various safe technologies: a passive water-cooling system, an infinite-time air-cooling system, a hydrogen explosion prevention system, and an operation support system to better deal with reactor accidents. These technologies are referred to as “inherently safe technologies”. The passive water-cooling system and infinite-time air-cooling system achieve core cooling without electricity. These systems are intended to cope with a long-term station blackout (SBO), such as that which occurred at the Fukushima facility. Both these cooling systems remove relatively high decay heat for the initial 10 days after an accident, and then the infinite-time air-cooling system is used alone to remove attenuated decay heat. The hydrogen explosion prevention system consists of a high-temperature resistant fuel cladding made of silicon-carbide (SiC) and a passive autocatalytic recombiner (PAR). The SiC cladding generates less hydrogen gas than the currently used zircaloy fuel cladding when core damage occurs, and the leaked hydrogen gas is recombined by the PAR. When a large-scale natural disaster occurs, fast event diagnosis and recognition of damaged equipment are necessary. Therefore, the operation support system consists of event identification and progress prediction functions to reduce the occurrence of false recognitions and human errors. This paper describes the following items: the targeted plant system; evaluation results on the required water pool capacity for 10-day water-cooling; development items for the water- and the air-cooling systems, the hydrogen explosion prevention system and the operation support system.

Key words : Inherently safe technology, Passive safety, Air-cooling system, Hydrogen explosion, Operation support system, BWR

1. Introduction
A severe accident occurred at the Fukushima Daiichi Nuclear Power Station (1F) because of equipment damage and power supply losses due to the tsunami following the great earthquake on March 11, 2011 (IAEA, 2011, Investigation committee on the accident at the Fukushima Nuclear Power Station of Tokyo Electric Power Company, 2012, U.S. NRC, 2012). After the external power supply had been lost, emergency diesel power generators (D/Gs) were immediately started up. The cooling system for decay heat removal was being kept running with electricity from the D/Gs. However, the tsunami that hit the site about one hour after the earthquake flooded the locations where the D/Gs were installed and damaged them, making them inoperable. With the loss of both external power and emergency power supplies, a station blackout (SBO) occurred at 1F. This eventually led to damage of the cores of Units 1 - 3 (Investigation committee on the accident at the Fukushima Nuclear Power Station of Tokyo Electric Power Company, 2012).
Hydrogen explosions also occurred in Units 1, 3 and 4. The hydrogen was considered to have been generated in the primary containment vessel (PCV) by the chemical reaction between zircaloy fuel cladding and water. Although the PCVs of the boiling water reactors (BWRs) were filled with inert nitrogen gas, hydrogen gas leaked from the PCVs and was considered to have exploded (Investigation committee on the accident at the Fukushima Nuclear Power Station of Tokyo Electric Power Company, 2012).

Furthermore, it was shown that there were some issues with operators’ actions and identifications of plant statuses and sensor integrities. For example, reactor water level indicators of Units 1 - 3 were thought to have been showing incorrect values at various times (Investigation committee on the accident at the Fukushima Nuclear Power Station of Tokyo Electric Power Company, 2012).

From these experiences and lessons, we have been developing the following inherently safe technologies for BWRs: a passive water-cooling system, an infinite-time air-cooling system, a hydrogen explosion prevention system, and an operation support system to deal with reactor accidents.

In a series of papers at the ICONE22 conference, we described our detailed considerations of: both the new water-cooling system and the conventional isolation condensers (ICs) and passive containment cooling systems (PCCSs) as a water-cooling system (Ishida, et al., 2014); the infinite-time air-cooling system (Tamura, et al., 2014), the hydrogen explosion prevention system (Ishibashi, et al., 2014) and the operation support system (Kanada, et al., 2014).

In one more paper at ICONE22, we summarized the plant system and introductory information about the above-mentioned developed technologies (Kitou, et al., 2014).

To improve reactor safety, development work on some types of reactors with “inherently safe technology”, for example a gas turbine high temperature reactor (GTHTR) and an integral inherently safe light water reactor (I2S-LWR), has been done before now. The safety features of the GTHTR are negative reactivity coefficients, decay heat removal system using air-cooling and high-temperature resistant fuel (Kunitomi, et al., 2011). The I2S-LWR cannot encounter a large break loss of coolant accident (LOCA) and it achieves infinite-time decay heat removal using air-cooling (Petrovic, 2014).

We defined “inherently safe technology” for BWRs as infinite-time passive cooling, because LOCA events for BWRs are milder than those for pressurized water reactors (PWRs) and negative reactivity coefficients are already achieved. We think that infinite-time passive cooling can be achieved by combination of a passive water-cooling and an infinite-time air cooling systems. High-temperature resistant fuel which is one of the safety features of the GTHTR is included as a part of our hydrogen explosion prevention system. Furthermore, we added the operation support as a part of the inherently safe technologies from the lessons of Fukushima Daiichi nuclear accident.

In the present paper, we review the main points of the plant system and the developed technologies and describe evaluation results about the required water pool capacity to achieve infinite-time core cooling without electricity using the water-cooling and the air-cooling systems.

2. Plant system

Table 1 summarizes major plant characteristics of the target plant, and Fig. 1 presents a schematic view of the plant system. In the present study, the plant thermal power is assumed to 4700 MW as a large scale BWR with an electric power of about 1600 MW. The infinite-time air-cooling system for a large scale plant is generally more difficult to realize than that for a small power reactor. So, the developed air-cooling system for the large scale plant is easily applied to small power plants.

The plant has a passive water-cooling system and an infinite-time air-cooling system to handle the situations caused by a long-term SBO. As the passive water-cooling system, we have considered an IC, a PCCS, and a new water-cooling system. We will choose at least one water-cooling system from among them. ICs and PCCSs are used in the economic simplified BWR (ESBWR) (GE Nuclear Energy, 2006). For design basis accidents, the plant has other active or passive safety systems such as an emergency core cooling system (ECCS). The PCV has a suppression pool (S/P) similar to the current advanced BWR (ABWR) and ESBWR.

The water pool capacity for ICs and PCCSs of the ESBWR is about 3 days. The new water-cooling system has been developed as an optional system to deal with long-term water-cooling. In the new water-cooling system, heat exchangers and water pools can be located at a lower level, i.e. underground, because the condensate can be transferred.
to the S/P using the differential pressure between the RPV and the S/P. The lower level location of the heavy-weighing water pools improves the seismic design of the reactor building (R/B) and allows construction of larger water pools and that makes it easier to supply water to them even in an accident such as caused by a long-term SBO

<table>
<thead>
<tr>
<th>Table 1</th>
<th>Major plant characteristics</th>
</tr>
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<tbody>
<tr>
<td>Thermal power (MW)</td>
<td>4700</td>
</tr>
<tr>
<td>Reactor pressure (MPa)</td>
<td>7.2</td>
</tr>
<tr>
<td>Steam temperature (°C)</td>
<td>287</td>
</tr>
</tbody>
</table>

Fig. 1  Schematic view of the plant system

Since the new water-cooling system and the PCCS do not have a function to supply water to the RPV unlike the ICs, water is supplied to the RPV by a reactor core isolation cooling system (RCIC) like in the ABWR or an alternative feedwater system which is installed outside the R/B. The operation period of RCICs was designed as 8 hours, however, the RCICs worked in 1F Unit 3 after the incident for 1 day. In addition, a similar coolant injection system, such as the high pressure coolant injection system (HPCI), worked under automatic control for about 36 hours in 1F Unit 3. We think that a longer-term RCIC operation is possible if the RCIC design is improved, for example having a larger battery capacity for the RCIC controllers. In this study, we assumed that a long-term RCIC operation or alternative feedwater supply up to 10 days was possible. The period of “10 days” was decided considering that the electric power supply was not fully recovered in 9 days in 1F nuclear accident. We estimated from the operation results of the HPCI in 1F Unit 2 that the necessary battery capacity was about 6 times the current battery capacity. Recently, an improved RCIC pump system which needs much less control battery capacity has been developed (Matsuura, et al., 2013). So, the necessary battery capacity can be decreased using the improved RCIC pump system.

Although the long-term RCIC operation or the alternative feedwater system can maintain core cooling, these systems do not have a function of decay heat removal from the PCV. The steam generated by decay heat is discharged into the suppression pool in the PCV through safety relief valves, and it leads to an increase of PCV pressure due to the increasing suppression pool temperature (which is a result of the increasing saturated steam pressure). The new water-cooling system can remove decay heat from the PCV inside, and discharge decay heat to the atmosphere by evaporating water pool coolant to avoid overpressurization of the PCV. Because of this, the suppression pool temperature can be maintained at a low level. Although the condensate storage tank which is installed outside the PCV is used as the water source instead of the suppression pool for the RCIC operation during SBO events in current ABWRs because the suppression pool temperature increases with time, the suppression pool can be used as the water
source if the water-cooling system is used with the RCIC operation. This is another merit of the new water-cooling system because the suppression pool is located in a strongly constructed PCV.

The air-cooling system has air-cooling heat exchangers outside the PCV. They are located between an operation floor wall and a cover wall. The air flows in from the bottom of the air-cooling heat exchangers, and it is heated there by the heat exchangers. The heated air is exhausted from the exhaust opening at the top of the cover wall. Since the flow channel of the air is separated by the cover wall from the outer air, the air flows by natural circulation.

Figure 2 shows an image of operation terms of the water-cooling and the air-cooling systems. For the initial 10 days after the reactor scram, both the water-cooling and the air-cooling systems remove relatively high decay heat, and then the air-cooling system is used alone to remove attenuated decay heat after 10 days.

We set the heat removal capacity using both the water-cooling and the air-cooling systems to 40 MW which corresponds to the decay heat at 5 hours after the reactor scram. Since we designed these passive systems for only severe accidents, SBO events within 8 hours will be handled using ECCSs for design basis accidents.

![Figure 2: Expected operation terms of water-cooling and air-cooling systems](image)

Although we assumed that 10 day water-cooling was possible, this exceeded the operation term of existing water-cooling systems: ICs and PCCSs of ESBWR. So, we evaluated the required water pool capacity for 10 days water-cooling.

Figure 3 shows the changes of the integrated decay heat and the required water pool capacity with time for a 4700 MW thermal power reactor. The integrated decay heat $Q_{\text{Integral}}$ [GJ] was evaluated by the following equation:

$$Q_{\text{Integral}} = \int_0^{T_{\text{End}}} Q'(t) \cdot dt$$  

where, $Q'(t)$ is the decay heat [GW], $t$ is time [s], and $T_{\text{End}}$ is the operation term of the water-cooling system [s]. The decay heat was evaluated using the ORIGEN-2 code. In the actual calculation, we evaluated the decay heat every hour, and then $Q_{\text{Integral}}$ was calculated using the values. The required water pool capacity $M_{\text{Pool}}$ [ton] in Fig. 3 was calculated using Eq. (2) based on the following assumptions.

1) Pressure in the water pool is atmospheric pressure
2) Initial water temperature in the water pool is 40 °C
   (Water enthalpy $H_W = 168 \text{ kJ/kg} = 0.168 \text{ GJ/ton}$)
3) Discharged steam is saturated steam
   (Steam enthalpy $H_S = 2675 \text{ kJ/kg} = 2.675 \text{ GJ/ton}$)
4) Heat removal by the air-cooling system is NOT considered
\[ M_{\text{Pool}} = \frac{Q_{\text{Integral}}}{H_S - H_W} \] (2)

The integrated decay heat for 10 days was about 15500 GJ, and the required water pool capacity was about 6200 tons. In addition, we must consider some design margins and the ineffective pool volume in actual plants. Estimations of the design margins and the ineffective pool volume will be considered in a future study because they strongly depend on the plant and the pool design.

The ESBWR has the water pool that can be used by ICs and PCCSs for 3 days (72 hours). The estimated pool capacity obtained using the same equations was about 2650 tons. The decay heat of the ESBWR was assumed to be proportional to the plant thermal power, and the thermal power of the ESBWR was assumed to 4500 MW. So our required pool capacity of 6200 tons significantly exceeds the existing designs.

Our concept, however, uses the air-cooling system with the water-cooling system to remove the decay heat. We think the concept can lead to reduction of the required water pool capacity. The required water pool capacity using the air-cooling system \( M_{\text{Pool\_with\_AC}} \) [ton] was evaluated by the following equation:

\[ M_{\text{Pool\_with\_AC}} = \frac{\left( Q_{\text{Integral}} - \dot{Q}_{\text{AC}} \times T_{\text{End}} \right)}{H_S - H_W} \] (3)

where, \( \dot{Q}_{\text{AC}} \) is heat removal capacity [GW] by the air-cooling system, and the operation term of the water-cooling system \( T_{\text{End}} \) was assumed to \( 8.64 \times 10^5 \) s (10 days).

Figure 4 summarizes the relationship between the required water pool capacity used with the air-cooling system \( M_{\text{Pool\_with\_AC}} \) and the heat removal capacity by the air-cooling system \( \dot{Q}_{\text{AC}} \). Since \( \dot{Q}_{\text{AC}} \) is defined as the minimum required heat removal capacity during the initial 10 days, \( \dot{Q}_{\text{AC}} \) is treated as a constant value in this study. The required water pool capacity decreases linearly with the increasing air-cooling capacity. If the heat removal capacity of the air-cooling system was 10 MW (0.01 GW), the required water pool capacity was decreased to 2700 tons which is almost the same as that of the estimated ESBWR water pool capacity of 2650 tons. So, we set the target value of the air-cooling heat removal capacity to 10 MW. In addition, the decay heat of 10 MW corresponds to the decay heat at 10 days after the reactor scram. So, the attenuated decay heat after 10 days can be removed using the air-cooling system.
alone.

Because of this, the conventional ICs or PCCS also can be used as the water-cooling system for 10-day cooling, if the air-cooling system is used with them. If the new water-cooling system is adopted, longer-term water-cooling operation without the air-cooling system or a smaller capacity air-cooling system is possible because the water pool capacity is easily expanded.

![Diagram](image_url)

Fig. 4 Relationship between required water pool capacity used with the air-cooling system and heat removal capacity by the system

The hydrogen explosion prevention system (Ishibashi, et al., 2014) consists of using a high-temperature resistant fuel cladding made of silicon-carbide (SiC) cladding and the passive autocatalytic recombiner (PAR). The SiC cladding generates less hydrogen gas than the currently used zircaloy fuel cladding when core damage occurs. Less hydrogen gas generation reduces not only the risk of a hydrogen explosion but also the risk of hydrogen leakage from the PCV because the increase of PCV pressure due to the hydrogen gas can be mitigated. If the hydrogen leaks from the PCV to the R/B, such as an operating floor, the leaked hydrogen gas is recombined by the PAR. Since the PCV is filled with inert nitrogen gas to prevent hydrogen explosions, the PARs are set outside the PCV.

The plant also has an operation support system (Kanada, et al., 2014). When a large-scale natural disaster occurs, fast event diagnosis and recognition of damaged equipment are necessary. Therefore, the operation support system has event identification and progress prediction functions to reduce occurrence of false recognitions and human errors. The operation support system also has a function of sensor integrity diagnosis. The sensor integrities are evaluated by correlating redundant sensors with design information of the plant.

3. Development items

3.1 Passive water-cooling system

The passive water-cooling system works without electricity for 10 days after the reactor scram to remove a relatively large amount of decay heat from the core. The main development item of the system is the heat exchanger. To design the heat exchanger, we carried out heat transfer tests.

The tests were conducted using full-scale U-shaped single tubes with three diameter sizes in a wide range of pressure and inlet steam velocity conditions. The inner diameters of the tubes were 22.2mm, 28.4mm and 35.5mm. The heat transfer data were obtained at system pressures of 0.2 to 3.0 MPa and inlet steam velocities of 5 to 56 m/s. The test range of pressures and inlet velocities includes thermal hydraulics conditions for the PCCS and some of them can be
extrapolated to IC conditions. We also confirmed thermal hydraulics conditions to achieve our new concept of the water-cooling system. The detailed test conditions and results were summarized in the reference (Ishida, et al., 2014).

### 3.2 Infinite-time air-cooling system

For the initial 10 days after the reactor scram, the infinite-time air-cooling system is used with the water-cooling system. After that, the air-cooling system is used alone to remove relatively small decay heat; operation of this system does not require an electricity supply. The heat removal capacity of the air-cooling system was set to 10 MW considering the decline of the decay heat for 10 days.

One of the most important development items of the system is improvement in the heat transfer performance of air-cooling to allow the air-cooling heat exchanger to be made smaller. To achieve this, we have been developing the air-cooling enhancing technologies for the infinite-time air-cooling system by using a micro-fabrication surface, turbulence-enhancing structures, and heat transfer fins. To evaluate the performance of these air-cooling enhancing technologies, we conducted heat exchange tests using a single tube test section. The detailed test conditions and results were summarized in the reference (Tamura, et al., 2014).

From the test results, we have currently estimated that the necessary number of heat exchanger tubes for 10 MW air-cooling is a few thousand with 8 m length and 24 mm inner tube diameter. In this evaluation, we adopted an intermediate loop for the air-cooling system to avoid the influence of non-condensable gas generation in the RPV, and we assumed the atmospheric temperature was 50 °C and the reactor pressure was 0.4 MPa which is the assumed PCV design pressure. We think that the above-mentioned heat exchanger tubes can be arranged around the operation floor (reactor building) of 4700 MW thermal power class large scale BWRs. However, the heat exchanger design and accident scenarios which should be considered have not been fixed, yet. We will present our detailed heat exchanger design of the air-cooling system at a later time.

### 3.3 Hydrogen explosion prevention system

The hydrogen explosion prevention system consists of a high-temperature resistant SiC fuel cladding and PARs. Since the PARs have already been installed in some light water reactors, we think that there are no significant development items for them. So, we have mainly been developing the SiC fuel cladding. The SiC fuel cladding replaces the zircaloy cladding currently used in BWRs in a countermeasure to decrease the hydrogen generation. Less hydrogen generation decreases the risk of hydrogen leakage from the PCV. The leaked hydrogen gas is recombined by the PARs.

We have been developing a three-layered SiC fuel cladding concept which consists of an inner metallic layer to prevent fission product leakage, SiC/SiC composite substrate, and an outer environmental barrier coating to provide corrosion and oxidation resistances. To develop the SiC fuel cladding, we conducted steam oxidation tests for the SiC material, and we evaluated the merits of adopting the SiC cladding by a numerical severe accident analysis. The details about our SiC fuel cladding concept and evaluation results were summarized in the reference (Ishibashi, et al, 2014).

### 3.4 Operation support system

The operation support system is intended to reduce occurrence of false recognitions and human errors by identifying accident events and predicting the progression of plant behavior. To meet these purposes, we have been developing three main functions of the system: sensor integrity diagnosis, accident event identification, and plant simulation.

The sensor integrity diagnosis function judges the sensor integrities by correlating redundant sensors with the design information of the plant. The accident event identification function extracts a few candidate accident events using alarms and sensor signals. The plant simulation function predicts future plant behaviors on the basis of the identified accident events. The details about the operation support system were summarized in the reference (Kanada, et al., 2014).
4. Conclusions

We have been developing the following inherently safe technologies for BWRs: a passive water-cooling system, an infinite-time air-cooling system, a hydrogen explosion prevention system, and an operation support system for use during severe accidents.

The passive water-cooling system and the infinite-time air-cooling system can remove the decay heat without electrical power in the event of a long-term SBO. Although we assumed water-cooling for 10 days, we confirmed that 10-day water-cooling could be achieved using the water pool with the pool capacity for 3 days of cooling if the air-cooling system of 10 MW heat removal capacity was used with the water-cooling system. To realize the systems, we have been conducting water-cooling tests and developing air-cooling enhancing technologies.

The hydrogen explosion prevention system consists of SiC fuel cladding and PARs. We have been developing the three-layered SiC fuel cladding based on results of steam oxidation tests for SiC material and a numerical severe accident analysis.

The operation support system is planned to reduce occurrence of false recognitions and human errors. We have been developing three main functions for this system: sensor integrity diagnosis, accident event identification, and plant simulation functions.

The details about each system were summarized in papers for ICONE22.

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