



Original Article

Effect of emergency core cooling system flow reduction on channel temperature during recirculation phase of large break loss-of-coolant accident at Wolsong unit 1

Seon Oh Yu^a, Yong Jin Cho^a, Sung Joong Kim^{b, c, *}^a Korea Institute of Nuclear Safety, 62 Gwahak-ro, Yuseong-gu, Daejeon 34142, Republic of Korea^b Department of Nuclear Engineering, Hanyang University, 222 Wangsimni-ro, Seongdong-gu, Seoul 04763, Republic of Korea^c Institute of Nano Science & Technology, Hanyang University, 222 Wangsimni-ro, Seongdong-gu, Seoul 04763, Republic of Korea

ARTICLE INFO

Article history:

Received 19 April 2016

Received in revised form

22 February 2017

Accepted 31 March 2017

Available online 26 April 2017

Keywords:

External Injection Path

Fuel Channel Cooling

LBLOCA

Loss of ECCS

MARS-KS

PHWR

ABSTRACT

The feasibility of cooling in a pressurized heavy water reactor after a large break loss-of-coolant accident has been analyzed using Multidimensional Analysis of Reactor Safety-KINS Standard code during the recirculation phase. Through evaluation of sensitivity of the fuel channel temperature to various effective recirculation flow areas, it is determined that proper cooling of the fuel channels in the broken loop is feasible if the effective flow area remains above approximately 70% of the nominal flow area. When the flow area is reduced by more than approximately 25% of the nominal value, however, incipience of boiling is expected, after which the thermal integrity of the fuel channel can be threatened. In addition, if a dramatic reduction of the recirculation flow occurs, excursions and frequent fluctuations of temperature in the fuel channels are likely to be unavoidable, and thus damage to the fuel channels would be anticipated. To resolve this, emergency coolant supply through the newly installed external injection path can be used as one alternative means of cooling, enabling fuel channel integrity to be maintained and permanently preventing severe accident conditions. Thus, the external injection flow required to guarantee fuel channel coolability has been estimated.

© 2017 Korean Nuclear Society, Published by Elsevier Korea LLC. This is an open access article under the CC BY-NC-ND license (<http://creativecommons.org/licenses/by-nc-nd/4.0/>).

1. Introduction

Safe operation of nuclear power plants (NPPs) is assured by maintaining basic safety functions such as reactivity control, fuel cooling, and confinement. Among these functions, fuel cooling is the chief function that can maintain the thermal integrity of fuel channels in a pressurized heavy water reactor (PHWR) during event progression of a large break loss-of-coolant accident (LBLOCA). After an LBLOCA occurrence, various materials in the vicinity of the break (e.g., thermal insulation, coatings, and concrete) can be damaged and dislodged. A substantial amount of material produced during the blowdown phase is likely to be transported into the recirculation sump and accumulate on the strainer, forming a debris bed, which can be an important cause of increased pressure head loss across the strainer and can affect the emergency core cooling system (ECCS) performance. Consequently, excessive increase of pressure difference across the strainer can restrict the emergency

coolant flow during the recirculation phase and thus lead to thermal damage of the fuel channels owing to insufficient cooling.

In 1992, there was an actual such event at the Barsebäck-2 NPP in Sweden [1]. During the course of the startup procedure, the reactor power had been less than 1.0% at 30 bar system pressure and corresponding saturation temperature. Prior to the testing of the safety relief valves, on July 28, a safety relief valve on a main steam pipe in the reactor containment dry well opened inadvertently. From equipment nearby, the steam blow ripped off about 200 kg of the surrounding thermal isolation material. About 100 kg of this material was then flushed down into the suppression pool in the wet well. The reactor automatically scrammed, and the core spray and containment spray pumps were initiated, taking suction from the water pool in the wet well. There were five strainers available at the inlet pipes in the pool. However, two of them became clogged by the loose thermal isolation material, and thus the flow decreased. After an hour, the operating personnel observed the high-pressure (HP) difference across the strainers and noticed that one of the pumps showed signs of cavitation. Therefore, they forced a reverse flow through the strainers (back flush

* Corresponding author.

E-mail address: sungjkim@hanyang.ac.kr (S.J. Kim).

operation), hence making them usable again. However, during the entire transient, normal water level had been maintained in the reactor pressure vessel and adequate core cooling had been assured continuously, and the final International Nuclear and Radiological Event Scale rating was 2 (incident). This incident revealed a weakness in the defense-in-depth concept, which under other circumstances could have led to ECCS failure to provide recirculation coolant to the core. Since the above incident, a substantial amount of international cooperative activities has focused on identifying essential parameters and physical phenomena and seeking consensus regarding technical issues considered important for safety and possible paths for their resolution [2–5].

In the event of an LBLOCA, the substantial amount of debris transported to the sump strainer has a tendency to form a bed, which, much like a filter, can increase pressure difference across the strainer. Fig. 1 shows the patterns of debris accumulation on a mesh screen observed in a linear flume experiment [2]. Relatively fine debris accumulated uniformly on the screen is likely to cause a significant pressure head loss across the screen, hindering the establishment of smooth coolant flow. Consequently, the flow restriction caused by the pressure head loss can threaten the safety margin required to assure the successful cooling operation of ECCS after an LBLOCA.

The U.S. Nuclear Regulatory Commission conducted research to estimate the likelihood of the loss of net positive suction head in NPPs. NUREG/CR-6224 [6] documented the results of the U.S. Nuclear Regulatory Commission research and estimated the head loss across the debris bed formed on the strainer according to a semi-theoretical head loss model. Although the head loss correlation of NUREG/CR-6224 has been widely used to estimate debris-induced head loss across strainers, it has been recommended for use only for debris thicknesses less than 10 cm. In order to extend this limit, head loss data were obtained over a wider range of bed thicknesses [7]. Experimental results showed that the correlation of NUREG/CR-6224 conservatively predicted the head loss across debris beds with theoretical thicknesses up to 6 in. A joint research project on debris transport in the sump water flow and particle load on strainers was performed to experimentally investigate the particle transport phenomena in the coolant flow and to develop computational fluid dynamics (CFD) models for simulation of this flow [8]. Lee et al. [9] designed an experimental facility to generate fibrous insulation beds on a strainer and to measure the head loss through the beds.

They analyzed the measured data of head loss, considering the porosity and the fibrous bed thickness, and suggested a head loss correlation that showed good agreement with the experimental data.

In addition, chemical precipitates, which are solid particles formed by chemical reactions among the chemical species dissolved in solution, can form, interact with the fibrous debris bed, and aggravate strainer clogging possibly to an extreme condition, i.e., no coolant flow through the fibrous debris. This phenomenon, known as “chemical effects,” is concerned with the safety issue of long-term cooling. Rouaix et al. [10] have performed studies on the chemical effects that can occur during an LOCA owing to precipitate formation leading to an increase of head loss across strainers and consequently causing loss of safety systems such as ECCS and the containment spray system. Chemical precipitate formation was studied as a function of several physical parameters such as temperature, pH, nature of debris, and debris concentration. It was found that for potential chemical effects, the most important factor is the mass of debris generated; pH and temperature are influential factors on formation of precipitates. Bahn [11] also reviewed research efforts in the USA to resolve the GSI-191 chemical effects and summarized the experimental work of small-scale bench tests, integrated chemical effects tests, and vertical loop head loss tests considering potential chemical effects on the head loss across a debris-loaded strainer under a postaccident condition. Song et al. [12] investigated the chemical and physical effects of containment materials that are used in most Korean NPPs on the mechanism of strainer clogging, such as coating, insulating, and pipe material. They concluded that the silicon fibers deployed as insulating material should be replaced with an alternative material that generates no needle-shaped debris after breakage, and that the gypsum component in the coating should be reduced as much as possible. Considering previous research efforts and the real event, ECCS coolant flow during the recirculation phase is therefore likely to be reduced because of the potentially significant pressure head loss due to inflow and the accumulation of a substantial amount of debris, and its chemical effect, leading to additional increase of pressure head loss across the strainer.

During normal power operation of NPPs, if an unplanned reactor trip or abnormal power cutback occurs, operating personnel are required to maintain the reactor in a stable state according to the emergency operating procedure (EOP), and to take diagnostic and

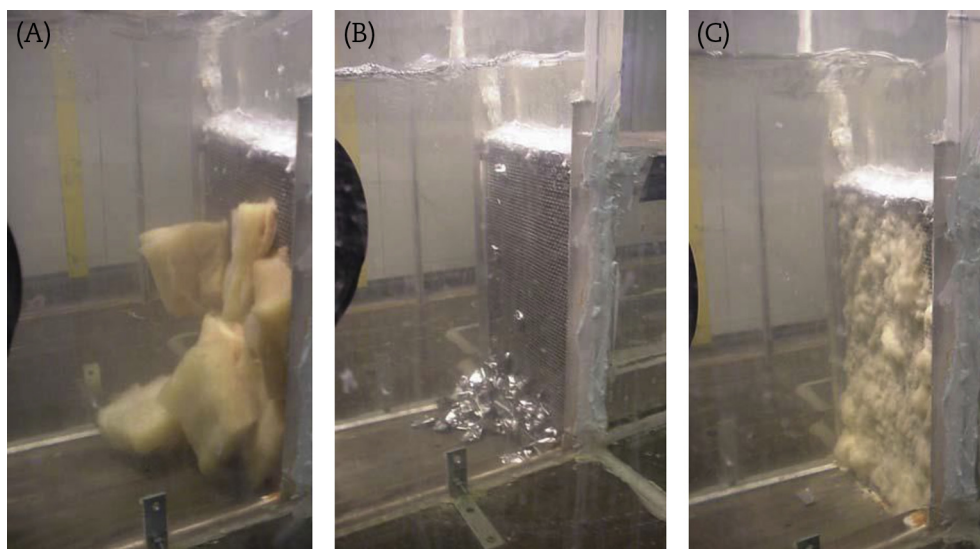


Fig. 1. Photographs of debris accumulation on a mesh screen. (A) Gross fiber blockage (B) Fragmental metal blockage (C) Uniform fiber blockage.

appropriate mitigation actions if necessary. For example, under the EOP for Wolsong unit 1 (the first Korean PHWR plant) in the case of an LBLOCA, intact and broken loops are diagnosed using available plant information such as the pressure and temperature of the reactor outlet headers. For an intact loop, effective cooling is supplied through the operation of the shutdown cooling system, as specified in the EOP. However, for a broken loop, viable cooling can be maintained only by via low-pressure (LP) emergency coolant injection (ECI) through the flow path from the recirculation sump. This may reduce the effective flow area through the strainer equipped in the sump. Eventually, the coolant flow diminishes in the broken loop because of the aforementioned mechanisms. If the pressure head loss across the strainer is detected reliably and found to be within the operational range, sustainable cooling of the broken loop can be expected to be maintained in a more stable manner during the recirculation phase, mitigating the possible adverse consequences. However, when an LBLOCA occurs and is followed by loss of the ECCS during the recirculation phase, the fuel channels may experience severe conditions—namely, if the recirculation pump cannot provide sufficient cooling water to the fuel channels because of the significant pressure head loss caused by the debris accumulation on the strainers and the chemical effects caused by the formation of the chemical precipitates during the recirculation phase, this can lead to serious consequences such as core damage. Thus, a means of alternative cooling, such as external injection, needs to be considered in a timely fashion to prevent severe core damage.

Therefore, a major objective of this study is to use the Multidimensional Analysis of Reactor Safety-KINS Standard (MARS-KS) code to evaluate the effect of the reduced ECCS flow on the adequacy of cooling in a broken loop under a postulated LBLOCA at Wolsong unit 1, one of the CANada Deuterium Uranium (CANDU-6) plants in Korea, during the recirculation phase. Also, feasibility evaluation of the emergency coolant supply through the external injection is conducted to prevent a progression to severe accident conditions.

2. Description of CANDU-6

Fig. 2 schematically illustrates the primary heat transport system (PHTS) of a CANDU-6, a PHWR-type plant. The CANDU-6 is a Canadian-developed nuclear reactor that uses natural uranium for its fuel and heavy water (D_2O) to moderate fast neutrons. A total of 380 fuel channels span the calandria vessel horizontally between the two end shields and are located within two equal figure-of-eight loops that can be isolated from each other under accident conditions. Each fuel channel consists of a pressure tube surrounded by a calandria tube (CT) with a CO_2 annulus gap in between. Each fuel channel contains 12 fuel bundles, which in turn each comprise 37 tubes containing natural UO_2 pellets. Because the fuel channels are located horizontally in the calandria vessel, their failure mode differs from that of the vertical fuel assemblies in pressurized water reactors. Long fuel channels will sag when overheated, dislocating to the lower rows and thus deforming the channel configuration, causing severe core damage. The calandria vessel holds the heavy water moderator (about 200 Mg of D_2O) used to thermalize the fast neutrons produced by fission, and houses reactivity mechanisms that span it vertically in addition to the horizontal fuel channels.

The PHTS is composed of two loops, each serving 190 of the 380 horizontal fuel channels. Each loop contains two pumps, two steam generators, two reactor inlet headers, and two reactor outlet headers in a figure-of-eight arrangement. Feeders connect the inlet and outlet ends of the fuel channels to the inlet and outlet headers, respectively. The two closed loops are generally interconnected with isolation

valves, which are automatically closed when the pressure of the PHTS drops below a set point. Automatic loop isolation reduces the rate of reactor coolant loss in the event of an LBLOCA.

The function of the ECCS is to provide alternative means of cooling the fuel channels in the event of an LBLOCA, in which the normal coolant inventory in the PHTS is depleted enough that fuel cooling is no longer assured. There are three stages of ECCS actuation according to the operating pressure: HP, medium pressure (MP), and LP. As soon as the PHTS header pressure drops below the set point, coolant flows from the HP ECCS tanks into the headers, with the flow depending on the break size. HP ECCS tanks contain sufficient coolant to provide flow for at least 2.5 minutes for all break sizes [13]. The HP injection valves close when a signal indicates a low level in the HP ECCS tanks. The trip valve of the dousing tank opens for MP injection 90 seconds after HP injection starts. The ECCS pump injects the coolant from the dousing tank to the headers when the pump discharge pressure is higher than the reactor header pressure. According to the final safety analysis report (FSAR) of Wolsong unit 1 [13], for example, at least 500 m^3 of the coolant inventory would be needed for MP ECI; however, less is used in conservative accident analyses in general. As the coolant in the dousing tank depletes, the MP ECI is automatically terminated and LP ECI starts. During the recirculation phase, LP ECI is provided by collecting the coolant from the sump and recirculating it into the PHTS via the ECCS heat exchangers; this coolant contains H_2O and D_2O mixed with a substantial amount of debris.

In order to check whether the safety standards in place when the plant obtained its operating license are sufficient to cover hypothetical unexpected extreme events, a stress test of the Wolsong unit 1 was performed following the severe accident at the Fukushima Daiichi NPPs [14]. As a result, to alleviate the effects of loss of cooling caused by hypothetical failure of ECCS operation, it was decided to install external injection paths for emergency coolant supply in all Korean NPPs. The installation purpose of the external injection path is to provide an alternative safety function for system cooling when normal ECCS operation becomes unavailable during the progression of a severe accident. The major design criteria are to prevent propagation of core melt, maintain the integrity of the calandria vessel, and restore the PHTS coolant inventory. Also included in the criteria are recovering and maintaining the coolant level of the steam generator to allow decay heat removal, and mitigating the release of radioactive materials into the secondary system. The emergency coolant source for external injection is the emergency water system reservoir, with about 4,550 tons of water in each unit; this reservoir supplies the emergency coolant to the PHTS through the ECCS path after topping up the dousing tank via the emergency coolant pump. In addition, the external injection path is designed to supply the PHTS with emergency coolant via a fire truck pump car if emergency coolant pump becomes unavailable because of station blackout.

3. Modeling for evaluation

This study used as a computational tool, the MARS-KS code, which was developed for realistic multidimensional thermal–hydraulic system analyses [15]. The backbones of the MARS-KS code are the RELAP5/MOD3 and the COBRA-TF codes; the former is a system analysis code based on a one-dimensional two-fluid model for two-phase flows, whereas the latter is based on a three-dimensional, two-fluid, three-field model. The MARS-KS code also uses models of PHWR-oriented features, including a fuel channel model (CANCHAN) and component models (Wolsong pump model, header-feeder model, etc.) [16]. The transient responses of the CANDU-6 reactor have also been assessed using the code to evaluate their suitability for modeling the plant and to investigate the

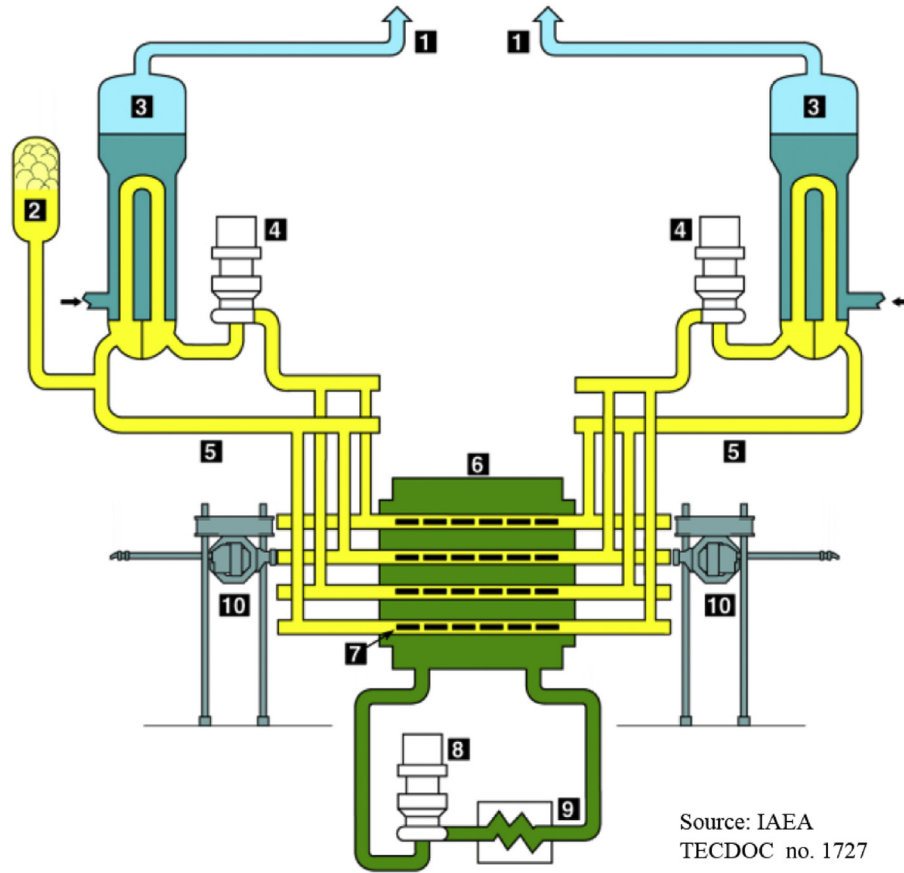


Fig. 2. Schematic primary heat transport system of CANDU-6. Legend: 1, steam line leading to turbines; 2, pressurizer; 3, steam generators; 4, PHTS pumps; 5, inlet/outlet headers; 6, Calandria vessel; 7, fuel channels; 8, moderator circulation pump; 9, moderator heat exchanger; 10, online refueling machines). CANDU, CANada Deuterium Uranium; PHTS, primary heat transport system.

individual effects of the models during LBLOCA progression [17]. In addition, through comparison and validation of the computer codes, which was an International Atomic Energy Agency activity designed to facilitate international cooperative research and to promote information exchange regarding computer codes for thermal hydraulic system analyses, the capabilities of the MARS-KS code have been evaluated via simulation of SBLOCA tests of an RD-14M facility, a full-elevation model of a typical PHTS of CANDU-6, which was built to provide improved understanding of CANDU and to generate integrated experimental data on the thermal hydraulic phenomena in a figure-of-eight test facility under postulated accident conditions [18]. The evaluation included analyses of the code's capability to predict important thermal hydraulic phenomena and comparison with calculated results of other system analysis codes. Consistent with the calculation for the accident analysis, in this study, the standard input deck, which has been validated through audit calculation for LBLOCA analyses [19], is used with slight modifications including simulation modeling of the reduction of LP ECI flow.

Fig. 3 shows the nodalization of the CANDU-6, with 2,064 MW_{th}. The 380 horizontal fuel channels are modeled in a lumped manner by four multiaveraged CANCHAN components (100, 200, 300, and 400) having 12 axial volumes per channel. To represent more specifically the fuel channels considering different relative elevations, flows, and power, one of the average channels in the broken loop (400), in particular, is modeled by seven subchannels (025–085). The reactor core consists of four multiaveraged channels (95 fuel channels in each pass) circulating through two closed loops. Passes 1 and 2 are in the intact loop, which is modeled to be

isolated from the broken one (passes 3 and 4) by closing the loop isolation valve as actuated by the LOCA signal.

Steam generators, each consisting of the major components of steam drum, separator, U-tubes, and downcomer, are modeled to be operated according to the level control logic and the assumptive manual actions for avoiding undesired flooding by continuous auxiliary feedwater supply during the recirculation phase. The main steam safety valves are also modeled to simulate the crash cooling down of the system by means of opening at 30 seconds after the occurrence of a LOCA signal, as designed; this aids heat transport system depressurization, increasing the injected flow.

The HP and MP/LP ECIs of the ECCS are modeled to supply the coolant to eight reactor headers using an accumulator component and time-dependent volume components of the code, respectively. The coolant injected into the PHTS via the ECCS heat exchangers during the recirculation phase is assumed to be at a temperature of 328 K.

Previous assessment of the effect of the critical flow models showed that the total accumulation of the break flow does not differ significantly between the Henry–Fauske and Moody models [15]. Thus, the Henry–Fauske model is used to simulate the break flow with the default value for the discharge coefficient, as shown Eq. (1), which is the same model used in the RELAP5 code.

$$G_c^2 = \left\{ \frac{x_o v_v}{\eta P} + (v_v - v_{l,0}) \left[\frac{(1 - x_o) N}{(s_{v,eq} - s_{l,eq})} \frac{ds_{l,eq}}{dP} - \frac{x_o C_{p,v} \left(\frac{1}{\eta} - \frac{1}{\gamma} \right)}{P_t (s_{v,0} - s_{l,0})} \right] \right\}^{-1} \quad (1)$$

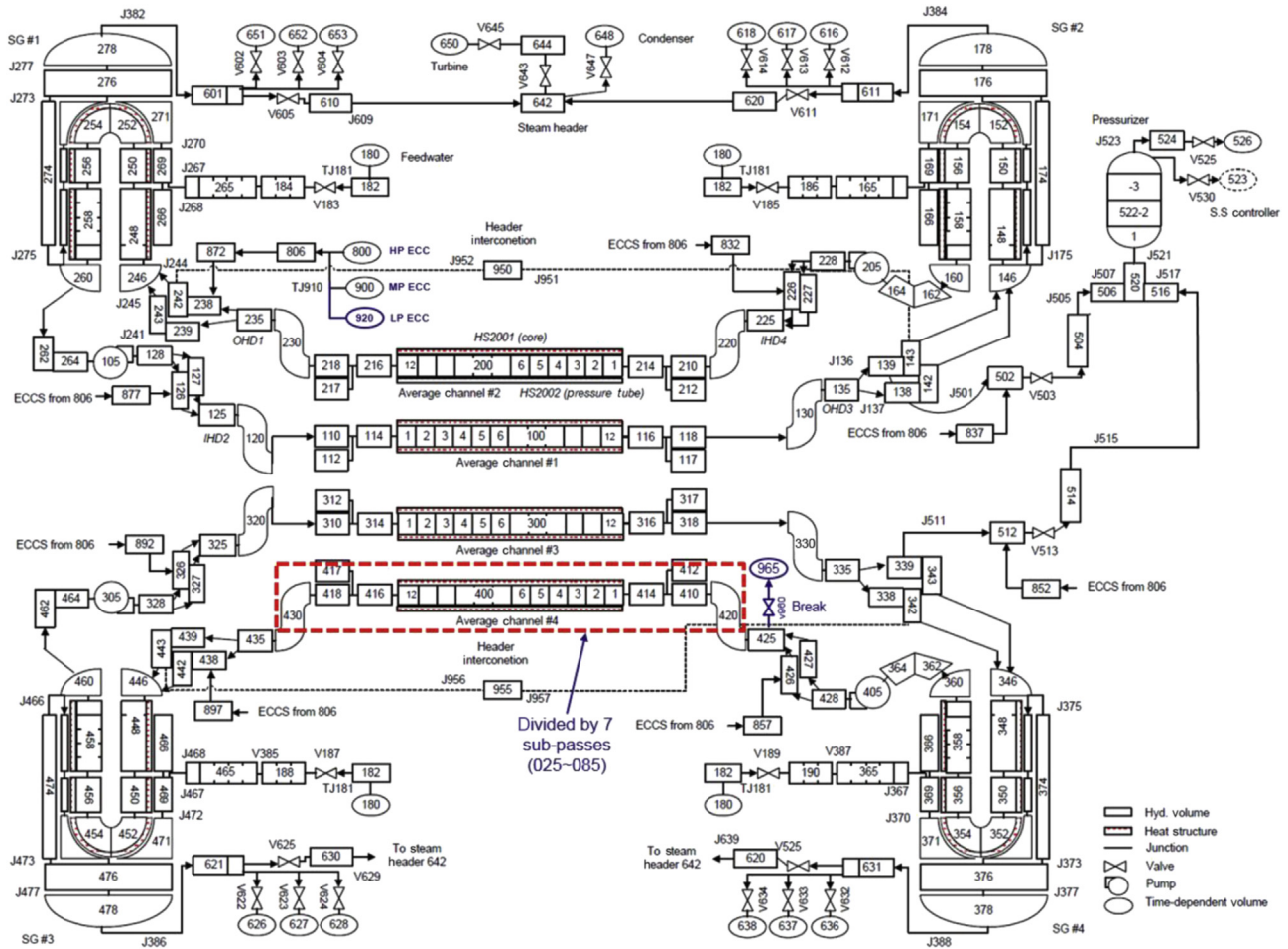


Fig. 3. Nodalization of Wolsong unit 1 for MARS-KS simulation. MARS-KS, Multidimensional Analysis of Reactor Safety-KINS Standard.

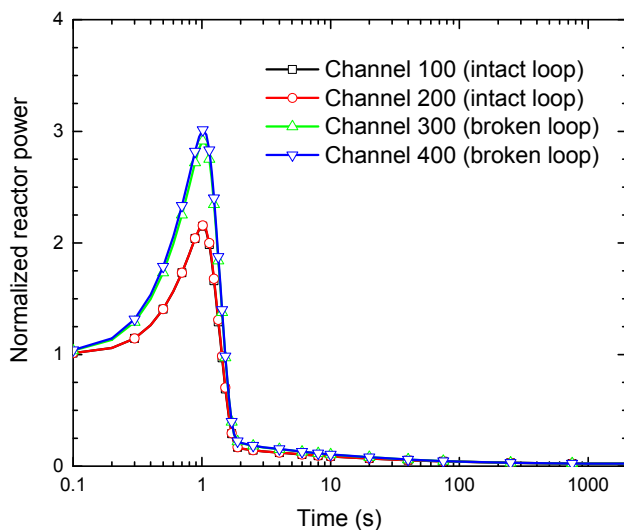


Fig. 4. Trends of reactor power in channels after a 35% reactor inlet header break LOCA. LOCA, loss-of-coolant accident.

where G_c is the critical mass flux, and C_p , P , s , v , x , γ , and η are the specific heat, the pressure, the specific entropy, the specific volume, the quality, the isentropic constant, and the critical pressure ratio (P_t/P_0), respectively. Subscripts o, eq, l, v, and t represent stagnant,

equilibrium, vapor phase, liquid phase, and state at throat, respectively. If the thermal nonequilibrium factor, N defined as $x_{eq,t}/0.14$, is taken to be unity, Eq. (1) becomes close to the equation of the homogeneous equilibrium model. If N equals zero, the solution is approximated to the homogeneous frozen model.

The PHWR has been known to have a positive void coefficient of reactivity for coolant loss, resulting in reactor power rise until reactor shutdown by the overpower signal. However, because of the code limitation in that the reactivity model of the point kinetics is unavailable for simulating instantaneous power pulse by coolant loss from horizontal fuel channels causing a regional overpower signal and thus a reactor trip, the transient reactor power [19] after the LBLOCA occurrence, as shown in Fig. 4, is used in the input deck.

4. Results and discussion

4.1. Steady-state and transient calculations under normal ECCS operation

A steady-state calculation is first performed given 103% reactor power; all major parameters are found to reach steady values successfully. Table 1 shows a comparison of the calculation results with the FSAR values for Wolsong unit 1 [13,19]. Major parametric values predicted by the code are observed to be in good agreement with the FSAR values.

LBLOCA transient is then initiated with the postulated 35% reactor inlet header break; this is simulated by quick opening of the

Table 1
Values of major parameters in the steady-state calculation.

Parameter (unit)	FSAR	MARS-KS	Difference (%)
Reactor inlet header pressure (MPa)	11.2	11.26	−0.54
Reactor inlet header temperature (K)	540.0	538.50	0.28
Reactor outlet header pressure (MPa)	10.0	10.01	−0.10
Reactor outlet header temperature (K)	584.0	583.72	0.05
Channel flow (kg/s)	2,062.0	2,100.85	−1.88
Averaged power for each channel (MW)	527.9	527.88	0.00
Total feedwater flow (kg/s)	1,061.0	1,062.88	−0.18
SG drum pressure (MPa)	4.7	4.77	−1.49
SG drum temperature (K)	533.0	534.14	−0.21
Steam flow (kg/s)	1,076.0	1,060.16	1.47

FSAR, final safety analysis report; MARS-KS, Multidimensional Analysis of Reactor Safety-KINS Standard.

break valve (v960 in Fig. 3) connecting the reactor inlet header of channel 400 and the containment under atmospheric pressure and is modeled using the time-dependent volume (965 in Fig. 3) component. The break size is obtained by adjusting the cross-sectional area of the broken reactor inlet header. The simulated break leads to substantial discharge of coolant at high temperature and high pressure from the PHTS, resulting in depressurization and possible core voiding. This can deteriorate the cooling conditions in the fuel channels and abruptly increase the power, leading to higher fuel temperature. However, the PHTS pressure continually decreases because of the reactivity control by reactor trip and the crash cooling provided through the main steam safety valves of the secondary system, as designed. In addition, the system pressure is kept stable under normal operation of the ECCS. So, it is estimated that in the normal ECCS condition, the major equipment is properly operated according to the designed control logic, as shown in Table 2, after the transient starts.

4.2. Effect of LP ECI flow reduction on fuel channel integrity

Table 3 shows the calculation matrix used for evaluation of the effect of the sensitivity of LP ECI flow reduction, which is modeled by varying the flow area of the LP ECI valve. To provide a conservative calculation, flow reduction is assumed to be initiated with the LP ECI signal. Fig. 5 shows the decreasing patterns of LP ECI flow rate according to the reduced flow area of the valve, indicating that the coolant inventory necessary for cooling the PHTS is decreasing, which implies that reasonable modeling has been achieved.

Table 2
Major sequences of LBLOCA in Wolsong unit 1.

Time (s)	Major sequence	Control logic
0.00	35% inlet header break	Break valve opens
9.05	Reactor trip LOCA signal	LOCA + 0.0 s Max [min(outlet headers), min(inlet headers)] ≤ 5.25 MPa
14.06	Loop isolation Turbine governor trip PHT pump trip	LOCA signal + 0.0 s LOCA signal + 5.0 s
29.64	HP ECI signal (open)	Max[min(outlet headers), min(inlet headers)] ≤ 3.62 MPa
39.05	SG crash cooling down (MSSV open)	LOCA signal + 30.0 s
119.64	MP ECI signal (open)	HP ECI start + 90 s
194.06	AFWP operation	LOCA signal + 185 s
313.96	HP ECI signal (close)	Accumulator inventory ≤ 5.47 m ³
638.32	MP ECI signal (close)	MP ECI stop
	LP ECI signal (open)	(amount of MP ECI > 200 m ³)

AFWP, auxiliary feedwater pump; ECI, emergency coolant injection; LBLOCA, large break loss-of-coolant accident; HP, high pressure; LP, low pressure; MP, medium pressure; MSSV, main steam safety valves; PHT, primary heat transport.

Table 3
Calculation matrix used for sensitivity evaluation.

1. LP ECI flow conditions	2. Flow area (m ²)	3. Area ratio (%)
4. Normal operation	5. 0.044	6. 100
7. Partial reduction	8. 0.01–0.040	9. 22.8–91.1
10. Complete reduction	11. 0	12. 0

LP ECI, low-pressure emergency coolant injection.

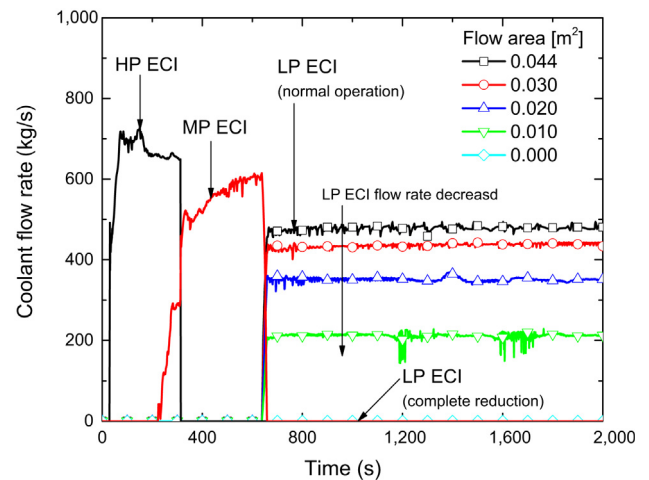


Fig. 5. Modeled ECCS flows versus time, including various scenarios during recirculation phase. ECCS, emergency core cooling system; ECI, emergency coolant injection; HP, high pressure; LP, low pressure; MP, medium pressure.

A sensitivity evaluation is performed to assess whether the thermal integrity of the fuel channel can be threatened by varying the flow area of the LP ECI valve, which simulates the reduction of flow area by the inflow of debris into the strainer. In this study, it is assumed that sufficient LP ECI flow and thus appropriate cooling can be achieved when the void fraction is held at zero in the fuel channels; however, when the void fraction is greater than zero, corresponding to incipience of boiling in the fuel channels, undesirable effects on the thermal integrity of the fuel channels can be anticipated, such as pressure tube/CT contact leading to CT dryout, sag and collapse of fuel channels, and consequent fuel melting after it reaches high and severely fluctuating temperatures.

Fig. 6A shows the predicted temperature of the fuel channel in the midplane of channel 200, located in the intact loop. In the case in which the flow area has been reduced to 0.02 m² (45.5% of nominal flow area) after an LBLOCA occurs, a slight temperature increase of approximately 35 K is observed at 1,250 seconds; thereafter, however, virtually no further temperature rise occurs. In the more severe case in which the flow area is completely blocked, corresponding to the LP ECI flow rate of 0.0 kg/s, irregular fluctuations of temperature are observed starting at 1,500 seconds. In this period of sporadic fluctuation, a maximum temperature rise of approximately 80 K is observed to occur at 1,600 seconds, a rise sufficient to cause incipience of boiling unless external intervention is taken. During practical operation with operator actions guided by the EOP, however, cooling and depressurization of the intact loop can be effectively maintained by the operation of the shutdown cooling system after the intact loop is successfully isolated from the broken loop. Thus, it is expected that proper cooling can be achieved in the intact loop during the recirculation phase.

A different tendency is shown in the predicted temperature of the fuel channel in the midplane of channel 300, located in the broken loop as shown in Fig. 6B. When the flow area is reduced to 0.03 m² (68.3% of nominal flow area), neither a dramatic

temperature rise nor irregular temperature fluctuation is observed. Likewise, no incipience of boiling is observed. Unlike the temperature behavior in the intact loop, however, when the flow area is further reduced to 0.02 m^2 , boiling is predicted to occur in the fuel channel. In addition, the beginning of irregular temperature fluctuation is predicted to occur much earlier when the flow area is further reduced. In the case in which a substantial amount of debris is assumed to induce a dramatic reduction of the flow area to less than or equal to 0.01 m^2 (22.8% of the nominal flow area),

insufficient LP ECI flow leads to significant temperature rise in the fuel channel, followed by termination of the MP ECI. Subsequently, large-amplitude temperature fluctuations occur, which may lead to conspicuous deficiency of the subcooling margin. Thus, the dryout of the fuel channel may be expected, with undesirable effects on the thermal integrity of the fuel channel.

Fig. 6C shows the predicted temperature of the fuel channel in the midplane of channel 045 (one of the subchannels of channel 400) under various conditions of reduced LP ECI flow rate. According to the assumption that the break is located at the reactor inlet header of channel 400, the decrease in coolant density due to depressurization and boiling, followed by the increase in power owing to the positive void coefficient, is most pronounced in the fuel channels of the broken loop downstream of the break. Thus, significant temperature rise can be expected to occur in channel 400 soon after the break. However, under the normal operation of the ECCS, following the reactor trip and crash cooling, the temperature of channel 400 decreases. A similar trend can be observed for channel 300, with a difference in degree. When the flow area is reduced to 0.03 m^2 , no fluctuations or dramatic rises in temperature are observed, similar to the results for channel 300. In the case in which the flow area is further reduced to 0.02 m^2 , however, some temperature fluctuation is observed to occur during the period of 900–1,500 seconds after the reactor trip, during which a maximum temperature rise of approximately 120 K is estimated to occur at about 1,400 seconds. In addition, no substantial temperature fluctuation is observed after 1,500 seconds. For a worse case of reducing the flow area to less than half its nominal value, however, disturbances and irregular increases in temperature are observed. Under the LP ECI flow rate of zero, relatively frequent temperature fluctuations are observed, and the temperature rapidly increases starting at about 1,300 seconds. It should be noted that the simulation continues until a point 2,000 seconds after the reactor trip. Thus, if the operators take no additional actions to mitigate the event, the temperature of the fuel channel is likely to increase continuously and lead to severe damage of the fuel channels.

Through the results discussed so far, it has been found that during the recirculation phase, cooling of the fuel channel in the broken loop is feasible if the flow area remains above approximately 70% (corresponding to flow of 90%), which facilitates sufficient LP ECI, as shown in Fig. 7. If the flow area is reduced further,

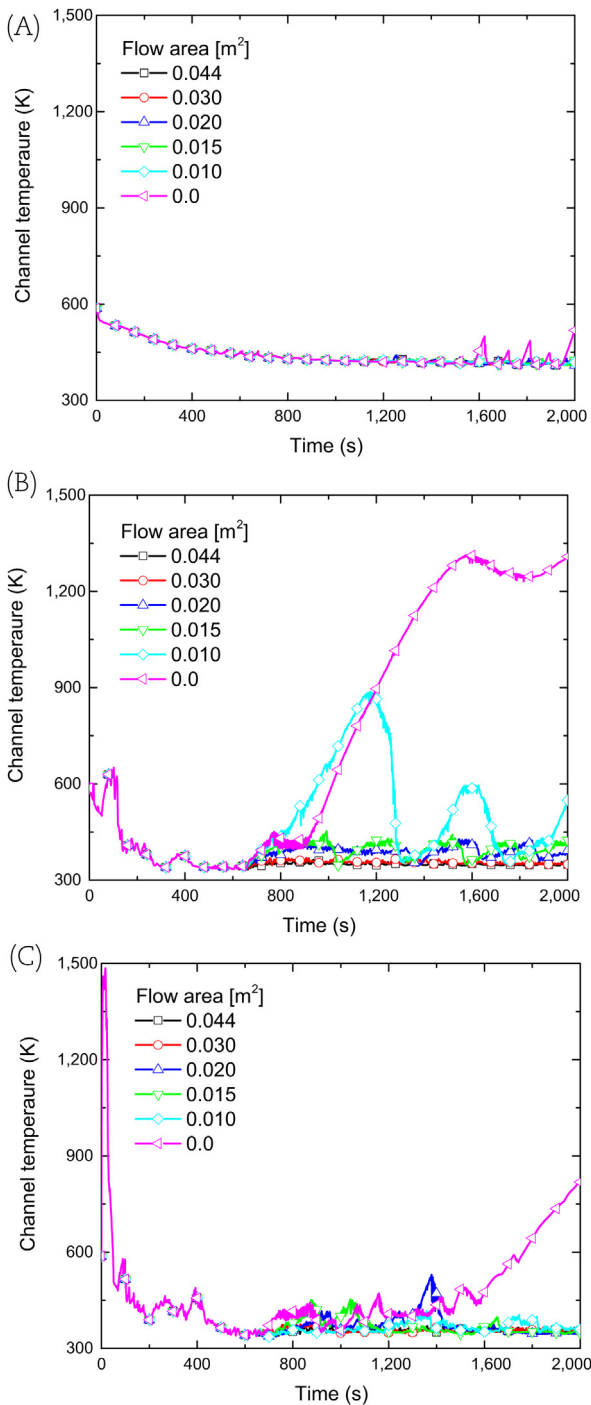


Fig. 6. Effects of various levels of reduced flow area during the recirculation phase. (A) Channel 200 in the intact loop. (B) Channel 300 in the broken loop. (C) Channel 045 in the broken loop.

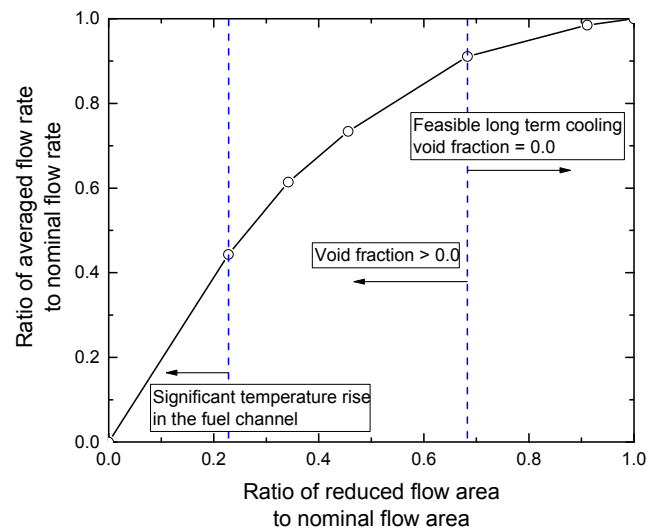


Fig. 7. Regimes of reduced flow area/rate of the LP ECI having different implications for cooling. LP ECI, low-pressure emergency coolant injection.

however, incipience of boiling is expected, which would impair the thermal integrity of the fuel channels. In addition, if the LP ECI flow area is further reduced, to below approximately 25% (corresponding to the flow of 45%), excursions and frequent fluctuation of temperature in the fuel channels are likely unavoidable, and impairment of the fuel channels can be anticipated. Also, there can be the severe fluctuations in differential pressure between the upstream and downstream of the strainers because of the greatly reduced flow area, causing drastic fluctuation and then reduction of LP ECI flow during event progress.

Therefore, based on the above results regarding the detailed analysis of reasonable operating ranges of the major system parameters that facilitate sufficient and stable cooling in the broken loop, the degree of reduction of ECCS flow can be monitored by measuring the pressure drops along the ECCS paths and the vibrations of the ECCS pumps. Correspondingly appropriate corrective measures can be taken to avoid deviation of the preset operating ranges and thus to avoid progress to more severe states.

4.3. Evaluation of feasibility of cooling through the external injection path

Additional transient calculations are carried out to evaluate the feasibility of coolant supply through the newly installed external

injection path. It is assumed that prior to entering into severe accident conditions, normal recirculation operation via ECCS is unavailable because of the serious pressure head loss caused by the potential accumulation of a substantial amount of debris on the strainers following the LBLOCA. In several scenarios with differing LP ECI failure times, sensitivity analyses are performed to confirm whether appropriate cooling of fuel channels can be achieved given the emergency coolant flow. The LP ECI is conservatively assumed to completely fail, and the failure of the LP ECI is modeled by completely closing the LP ECI valve. Additionally, the supply of emergency coolant through the external injection path is modeled via a time-dependent volume component simulating the emergency water system reservoir; the coolant supply is achieved by opening the valve connected to the reservoir when the LP ECI fails. In practice, this measure is hypothesized to be feasible by means of operating the valve arrangement for external injection of the emergency coolant after the operators notice that the ECCS has entered an abnormal state.

Fig. 8 shows the temperature variations of the fuel channels in the broken loop for various types of emergency coolant flow through the external injection path. When the LP ECI fails entirely, the fuel channels experience sharp increases and severe fluctuations in temperature because of the still-high decay heat. However, it is observed that the fuel channel can be cooled by the external

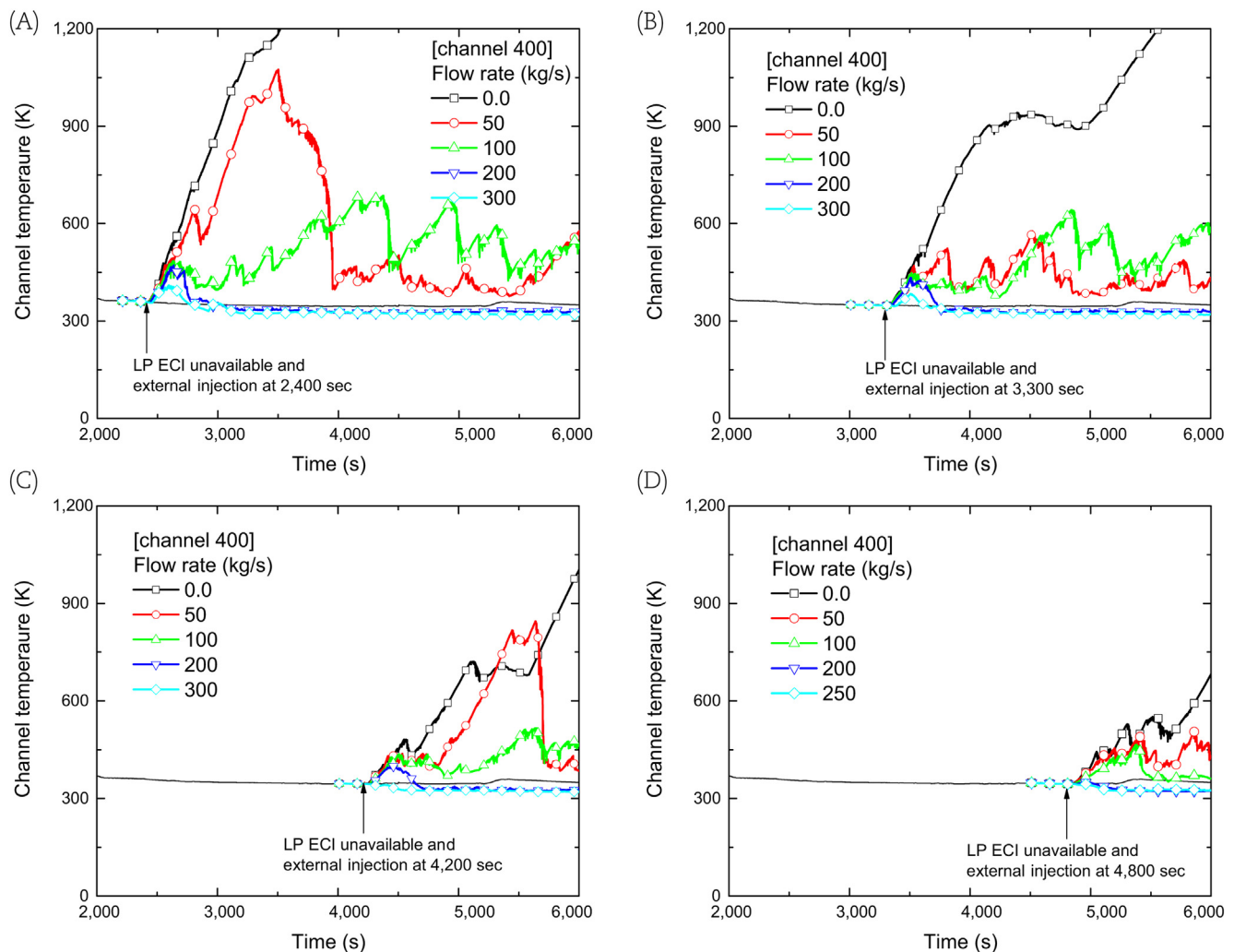


Fig. 8. Effects of various external emergency coolant flows on fuel channel temperature in the broken loop, modeled for various times at which the LP ECI becomes unavailable. (A) 2,400 seconds (B) 3,300 seconds (C) 4,200 seconds (D) 4,800 seconds. LP ECI, low-pressure emergency coolant injection.

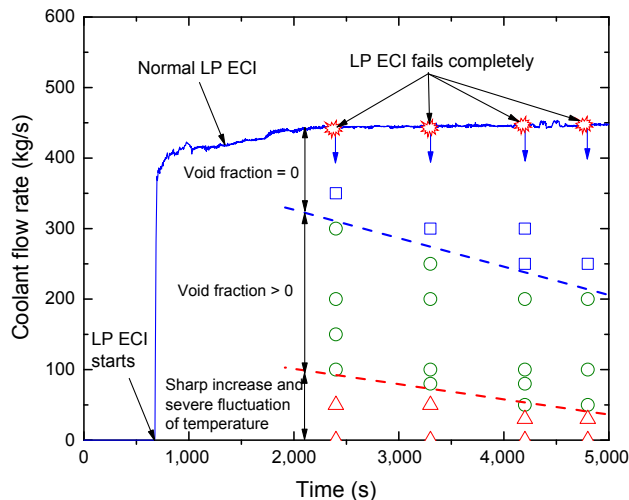


Fig. 9. LP ECI flow during operation, including four scenarios in which the LP ECI system fails. For each failure time, shapes indicate the sufficiency of cooling corresponding to various levels of available coolant flow through the external injection path: squares indicate void fraction = 0, circles indicate void fraction > 0, and triangles indicate void fraction > 0 combined with sharp increases and severe fluctuations in temperature. LP ECI, low-pressure emergency coolant injection.

emergency coolant supply, given sufficient flow. Also, because the time at which the LP ECI becomes unavailable has been modeled and found to be earlier, proportionally more emergency coolant is necessary for sustainable cooling owing to the much larger decay heat generated in the fuel channel.

Based on sensitivity analyses for various assumed times of total failure of the LP ECI, a flow of emergency coolant supply sufficient to maintain fuel channel integrity is determined, as shown in Fig. 9. Under the most severe condition, for example, when LP ECI fails 2,400 seconds after the LOCA signal, it is predicted that an emergency coolant supply of more than approximately 300 kg/s is required to completely suppress the incipience of boiling in the fuel channel, a condition corresponding to appropriate cooling by the external injection path. However, when the emergency coolant flow ranges from 100 to 300 kg/s, the void fraction in the fuel channels is estimated to be greater than zero, indicating incipience of boiling in the fuel channels. If the external coolant flow is dramatically reduced to below approximately 100 kg/s, sharp increases and severe fluctuations of temperature in the fuel channels can be anticipated, with subsequent fuel melting. For later LP ECI failures, less external flow is required to maintain the coolability of the fuel channel. It should be emphasized that these analyses assume the complete failure of the LP ECI; if the LP ECI were instead partially available, the emergency coolant flow required for feasible cooling of the fuel channels and maintenance of fuel channel integrity is reduced accordingly.

In summary, emergency coolant supply through the external injection path can be considered one means of alternative cooling that can maintain fuel channel integrity and thus prevent or delay progress to severe damage conditions. Because this measure is an emergency procedure, timely action needs to be taken to utilize it. The present study is expected to provide an important technical basis regarding pressure head loss and alternative cooling under reduction condition of ECCS flow during the recirculation phase of LBLOCA.

5. Conclusions

In this study, using the MARS-KS code, adequacy of cooling in fuel channels at the Wolsong unit 1 has been evaluated under the

postulated conditions of reduced flow of ECCS during the recirculation phase of an LBLOCA. The reduction of the ECCS flow can be caused by the significant pressure head loss arising from the inflow of a substantial amount of debris released by the high-energy break flow and its chemical effects. In the intact loop, although incipience of boiling is expected to occur in the fuel channel, effective cooling can be achieved by operating the shutdown cooling system according to the EOP. In the broken loop, however, if the flow area for LP ECI is reduced to below approximately 70% of its nominal value, incipience of boiling is expected. Additionally, if the flow area is dramatically reduced to below approximately 25%, impairment of the fuel channels is anticipated, with frequent fluctuations and sharp increases in the fuel channel temperature.

Additional transient calculations have been performed to evaluate the feasibility of emergency coolant supply through the external injection path under the condition in which the LP ECI is entirely unavailable for cooling because of a postulated significant pressure head loss. Provided that this measure is used in a timely manner and that the available flow is sufficient, it should be possible to maintain fuel channel integrity and thus to prevent or delay entry into severe accident conditions.

It is expected that sustainable cooling of the broken loop can be maintained at a more stable level by taking appropriate corrective actions reflecting the results of the present study.

Conflicts of interest

All authors have no conflicts of interest to declare.

Acknowledgments

This work was supported by the Nuclear Safety Research Program through the Korea Foundation Of Nuclear Safety, and granted financial resources by the Nuclear Safety and Security Commission of the Republic of Korea (number 1305006) and by the National Research Foundation of Korea grant funded by the Korean government (Ministry of Science, ICT, and Future Planning; number NRF-2017M2A8A4018213).

References

- [1] International Atomic Energy Agency, Clogged Pump Suction Strainers in the Wet Well Pool. 1992 [Internet]. [cited 2016 Apr 19]. Available from: <https://www-news.iaea.org>.
- [2] USNRC, Knowledge base for the effect of debris on pressurized water reactor emergency core cooling sump performance, NUREG/CR-6808 (2003) 1-1–1-11.
- [3] OECD/NEA, Debris Impact on Emergency Coolant Recirculation, Workshop Proceedings, Feb. 25–27, 2004, Albuquerque, USA.
- [4] OECD/NEA, Proceedings of the CNRA/CSNI International Workshop on Taking Account of Feedback on Sump Clogging, NEA/CSNI/R, 2009, 14, January, 2010.
- [5] OECD/NEA, Update Knowledge Base for Long-term Core Cooling Reliability, NEA/CSNI/R(2013)12, November, 2013.
- [6] G. Zigler, J. Brideau, D.V. Rao, C. Shaffer, E. Souto, W. Thornas, Parametric study of the potential for BWR ECCS strainer blockage due to LOCA-generated debris, NUREG/CR-6224 (1995) 1-1–2-7.
- [7] B.G. Park, J.W. Park, C.H. Kim, Experimental study of debris head loss through a pressurized water reactor recirculation sump screen after LOCA, Nucl. Eng. Des. 241 (2011) 2462–2469.
- [8] E. Krepper, G. Glover, A. Grahn, F.P. Weiss, S. Alt, R. Hampel, W. Kastner, A. Seeliger, CFD-modeling of insulation debris transport phenomena in water flow, Nucl. Eng. Des. 240 (2010) 2357–2364.
- [9] S. Lee, Y.A. Hassan, S.S. Abdulsattar, R. Vaghetto, Experimental study of head loss through an LOCA-generated fibrous debris bed deposited on a sump strainer for Generic Safety Issue 191, Prog. Nucl. Energy 74 (2014) 166–175.
- [10] S. Rouaix, C. Laurent, Y. Armand, J.-M. Mattei, M. Liska, D. Galuskova, Y. Vicena, B. Soltész, Precipitate formation contributing to sump screens clogging of a nuclear power plant during an accident, Chem. Eng. Res. Des. 86 (2008) 633–639.
- [11] C.B. Bahn, Chemical effects on PWR sump strainer blockage after a loss-of-coolant accident: review on U.S. research efforts, Nucl. Eng. Technol. 45

- (2013) 295–310.
- [12] H. Song, S.A. Par, M. Choi, J.Y. Par, M.W. Kim, J.W. Jung, Y.T. Kim, Examination of chemical and physical effects on sump screen clogging of containment materials used in Korean plants, *Ann. Nucl. Energy* 69 (2014) 51–56.
 - [13] Korea Hydro and Nuclear Power Company, Chapter 6. Safety Systems and Chapter 15. Accident Analyses in Final Safety Analysis Report of Wolsong Unit 1, 2016.
 - [14] Korea Hydro and Nuclear Power Company, Stress-Test Report of Wolsong Unit 1, 2013 [Internet]. Available from: <http://voc.kins.re.kr/stresstest/> [in Korean].
 - [15] Korea Atomic Energy Research Institute, MARS code manual, in: Code structure, System models, and Solution methods, Vol. I, 2013. KAERI/TR-2812/2004.
 - [16] USNRC, Improvement of RELAP5/MOD3.3.3 Models for the CANDU Plant Analysis, NUREG/IA-0189, 2000.
 - [17] Korea Atomic Energy Research Institute, MARS code manual, in: Developmental Assessment Report, Vol. IV, 2009. KAERI/TR-3042/2005.
 - [18] IAEA, Comparison of Heavy Water Reactor Thermalhydraulic Code Predictions with Small Break LOCA Experimental Data, IAEA-TECDOC-1688, 2012.
 - [19] Korea Institute of Nuclear Safety, Audit calculation for Wolsong unit 1 large break LOCA, 2011. KINS/HR-1129.