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Original Article

EVALUATION OF AN ACCIDENT MANAGEMENT STRATEGY OF EMERGENCY WATER INJECTION USING FIRE ENGINES IN A TYPICAL PRESSURIZED WATER REACTOR

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ABSTRACT

Following the Fukushima accident, a special safety inspection was conducted in Korea. The inspection results show that Korean nuclear power plants have no imminent risk for expected maximum potential earthquake or coastal flooding. However long- and short-term safety improvements do need to be implemented. One of the measures to increase the mitigation capability during a prolonged station blackout (SBO) accident is installing injection flow paths to provide emergency cooling water of external sources using fire engines to the steam generators or reactor cooling systems. This paper illustrates an evaluation of the effectiveness of external cooling water injection strategies using fire trucks during a potential extended SBO accident in a 1,000 MWe pressurized water reactor. With regard to the effectiveness of external cooling water injection strategies using fire engines, the strategies are judged to be very feasible for a long-term SBO, but are not likely to be effective for a short-term SBO.

1. Introduction

A state-of-the-art reactor consequence analysis (SOARCA) project was created by the *United States Nuclear Regulatory Commission* (USNRC) to make the best estimates of the offsite consequences of potential severe reactor accidents for two pilot plants: the Peach Bottom Atomic Power Station and the Surry Power Station [1]. A short-term station blackout (STSBO) and a long-term station blackout (LTSBO) were identified as the major groups of accident scenarios for analysis. Both types of scenarios involve a loss of all alternating current (AC) power. The risk management features for the SBO are to be enhanced [2].

In terms of severe accidents caused by an earthquake or tsunami that are beyond expectation, a special safety inspection for operating plants, following the Fukushima accident, has been conducted by the government of Korea to verify that nuclear power plants are adequately designed to

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respond to extreme accidents [3]. The inspection results show that Korean nuclear power plants in operation have no imminent risk for an expected maximum potential earthquake or coastal flooding, based on the up-to-date investigation. However, there is a need to implement long- and short-term safety improvements in order to secure safety for natural beyond-design-basis events [4].

One of the measures to increase the mitigation capability during a prolonged station blackout (SBO) accident is installing injection flow paths to provide emergency cooling water of external sources using fire engines on the steam generators (SGs) or reactor cooling system (RCS). Therefore, it is necessary to develop some guidelines or strategies to cope with an extreme severe accident scenario using the newly installed injection flow paths and fire engines. SOARCA-like analyses, which are limited to accident progression with the exception of offsite consequences, were conducted at the Korea Atomic Energy Research Institute for a typical 1,000 MWe pressurized water reactor. In this paper, an assessment is presented for the mitigative effectiveness of the external cooling water injection strategies using fire engines during a potential extended SBO accident.

A brief outline of the typical 1,000 MWe pressurized water reactor design with special reference to the mitigation capability during an extended SBO accident is provided in this section. The reactor uses pressurized water with a core thermal output of 2,815 MWth. For secondary heat removal, feedwater may be supplied to the steam generators using one of several pumps; for instance, the main feedwater, start-up feedwater, and auxiliary feedwater (AFW). However, the turbine driven auxiliary feedwater (TD-AFW) pumps can be credited as a unique means of supplying feedwater during an SBO event. TD-AFW pumps can provide feedwater until all station batteries, the capacity of which is a minimum of 4 hours, are depleted. The secondary steam can be removed through the main steam safety valves (MSSVs) or atmospheric dump valves (ADVs), which need an operator action in order to be opened [5]. The major design parameters of the reference plant are summarized in Table 1.

The safety injection system of the plants consists of four safety injection tanks (SITs), and high-pressure, and low pressure safety injection pumps. The passive SITs automatically discharge into the reactor coolant system if the RCS pressure decreases below the SIT pressure (4.31 MPa) during the reactor operation. Because the pressure of the RCS is maintained above the SIT injection set point in most transient accident sequences, SIT injection occurs only after depressurization of the RCS, vessel breach, or other induced RCS failure. If secondary heat removal is unavailable owing to failures in either the AFW system or steam removal system, core decay heat must be removed using a feed and bleed operation of RCS to prevent core damage. It is necessary that only the operator aligns a bleed line of the safety depressurization system (SDS) for the feed and bleed operation because the high-pressure safety injection pumps will automatically inject water from the refueling water tank into the RCS once the RCS is depressurized below the shutoff head of the pumps for the feed and bleed operation [5].

New injection flow paths for emergency cooling water into the RCS and SGs were installed as one of the postaction items after the Fukushima accident. The emergency cooling water system consists of a fixed pipe connected from the RCS or SGs to the outside of the containment. A standby valve is installed on the pipe. Following the occurrence of an SBO, movable equipment (e.g., a fire truck hose) can be connected to the pipe hole at the opening of the isolation valve. In many accidents with very hazardous work conditions, the inside of the containment cannot be made accessible or manageable. However, because the emergency cooling water system can be operated from outside of the containment, it has the advantages of high accessibility and maintenance during an accident [6].

2. Analysis methodology

The analyses consider several types of mitigation measures, including those specified in the emergency operating

Table 1 – Major input modeling parameters of the reference plant of 1,000MW pressurized water reactor.

Design parameter		Modeling input
Plant type		1,000 MW PWR (2 SG, 2 Hot legs, 4 Cold legs)
Power		2,815 MW_{th}
Coolant inventory	2 Steam generators	134×10^3 kg
	Reactor coolant system	215×10^3 kg
	4 Safety injection tanks	208×10^3 kg
Core Material	UO_2	86×10^3 kg
	Zircaloy	24×10^3 kg
Mitigation system against SBO		TD-AFW with battery power (Minimum battery power: 4 hr)
RCS depressurization system		2 trains of safety depressurization system (62.6 kg/sec/valve at 17.927 MPa)
SG depressurization system		2 atmospheric dump valves (1 ADV/SG) (106.2 kg/sec/valve at 9.308 MPa)
Fire engine capacity	Water flow into SG	0.0 lpm at 13.53 kg/cm^2g (SG pressure) 779 lpm at 1.0 kg/cm^2g (SG pressure)
	Water flow into RCS	1,336 lpm below 13.53 kg/cm^2a (RCS pressure)
Reactor cavity floor area		62.54 m^2
Containment free volume		79,300 m^3
Containment failure pressure		1.236 MPa(g)

ADV, atmospheric dump valves; PWR, pressurized water reactor; RCS, reactor cooling system; SBO, station blackout; SG, steam generator; TD-AFW, turbine driven auxiliary feedwater.

procedures, severe accident management guidelines, and the additional equipment and strategies required by the national actions taken in Korea after the Fukushima accident. One of the post-Fukushima actions to cope with a SBO accident is to supply makeup water using fire trucks into the steam generator or the RCS.

The mitigative measures for secondary heat removal during an SBO accident are atmospheric dump valves and fire trucks when fixed auxiliary feed water systems are unavailable. Meanwhile, the mitigative measures for the water injection into an RCS are SDSs and fire trucks when fixed emergency core cooling systems are unavailable. Even though the SDS still needs AC power, the system is assumed operable during an SBO scenario, which can be achievable by various means, for example, through future design improvements.

For a simulation of an SBO, all emergency core cooling systems, AFW systems except for the TD-AFW, and the containment spray are assumed to be inoperable. The STSBO also involves the loss of TD-AFW systems through the loss of direct current control power or loss of the condensate storage tank, and therefore proceeds to damage the core more rapidly. In the LTSBO, secondary heat removal using atmospheric dump valves and TD-AFW is assumed to be available during 4 hours initially with battery power.

The analyses were performed using a *Modular Accident Analysis Program* (MAAP) computer code version 5.02 [7]. The MAAP code is a system level computer code capable of conducting integral analyses of potential severe accident progressions in nuclear power plants, whose main purpose is to support a Level 2 probabilistic safety assessment or severe accident management strategy development. The code allows operator interventions and incorporates these in a flexible manner, permitting the user to model the operator behavior in a general way. MAAP simulates an accident transient, specifically accounting for system events which occur during the transient, including operator interventions, until a permanently coolable state is achieved or until the containment pressure boundary has failed and the containment building has been depressurized. The code includes models for all of the important phenomena which might occur during accident sequences involving degraded cores. It models thermal-hydraulics and fission product behavior in the RCS, containment, and auxiliary buildings. Models are included for engineered safeguard system logic and performance. To establish that the MAAP5 code is capable of addressing the above purposes and uses, numerous benchmarks have been set, both with respect to individual models and for the integral response of reactor systems. These benchmarks provide insights into the code performance and confidence in the capabilities of MAAP5 to represent individual phenomena as well as the integral response of reactor systems, including the influences of operator actions [7].

In MAAP code, the plant system is divided into two regions: the RCS and the containment. The nodalization of the RCS is not defined by the user, but modeled as fixed nodes. The nodalization schemes for the regions in the RCS are shown in Fig. 1, which represent the reference plant-type design [7]. An important part of this nodalization is that the downcomer, the lower plenum, the core, and the upper plenum are divided into four azimuthal nodes to match the

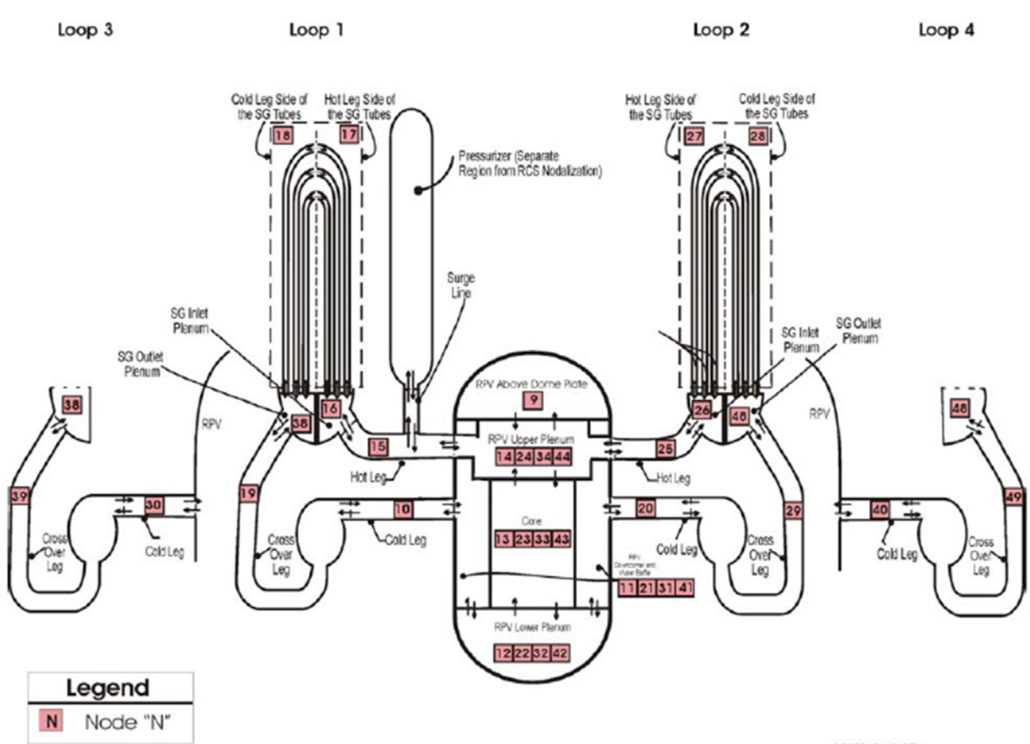

Fig. 1 – Reactor coolant system nodalization of Modular Accident Analysis Program 5.02 for a reference plant. RCS, reactor cooling system; SG, steam generator.

number of coolant loops. Hence the downcomer/lower plenum nodes are 11, 21, 31, and 41/12, 22, 32, and 42. The azimuthal nodalization allows the modeling of: (1) the pressure-driven flow; and (2) the turbulent mixing between adjacent nodes within these subregions of the vessel when the nodes are water-solid and the pump flow is asymmetric, such as when one or more coolant pumps are tripped but the others remain operating. The nodalization of the reactor core is shown in Fig. 2, where a total of 91 nodes are defined for this analysis.

The containment is modeled not as a fixed compartmentalized structure but as an interconnection of compartments and flow paths. However, the code is not sensitive to the number of compartments. Six compartments are defined for this analysis: (1) reactor cavity; (2) lower compartment including steam generators; (3) upper compartment; (4) annular compartment; (5) containment dome compartment; and (6) emergency core cooling system sump.

Several assumptions were made in the current analysis. These assumptions are either embedded in the code as models with input control parameters, or they are assumed in the present analysis. One of the code models relevant to this analysis is a hot leg rupture model. During a postulated severe accident, counter-current flow in the hot legs and steam generators during high-pressure sequences is an important and uncertain phenomenon. Hot gases coming from the core flow along the top of the hot leg, enter the inlet plenum of the steam generators, and form a plume that rises toward the SG tube bundle. This natural circulation phenomenon affects the heat-up and eventual creep rupture of the hot leg and the steam generator tubes during high-pressure sequences [7]. The default model of the temperature induced hot leg or SG tube rupture, which is similar to an independently developed model [8], was employed in this analysis. RCS components under stress at high temperatures will undergo irreversible strain known as material creep. When the strain is large enough, the component can rupture. Rupture of the RCS components due to material creep may be predicted by the application of the Larson-Miller parameter method. The method may also be applied for cases of time-varying temperature by considering the fractional contribution to rupture during consecutive intervals. MAAP employs the method and calculates the steel wall stress-strain creep rupture of the reactor vessel lower head, the surge line, the hot leg, and the steam generator tubes.

There are assumptions that the cooling water injection rate into the steam generator or RCS by a fire engine is calculated from the system pressure, which is referred in the utility document [9]. The injection flow started when the pressure of the SG secondary side decreases below 13.53 kg/cm^2g, and the flow rate reaches a design flow of 779 lpm at the SG pressure of 1.0 kg/cm^2g. And, the flow rate of the cooling water injection

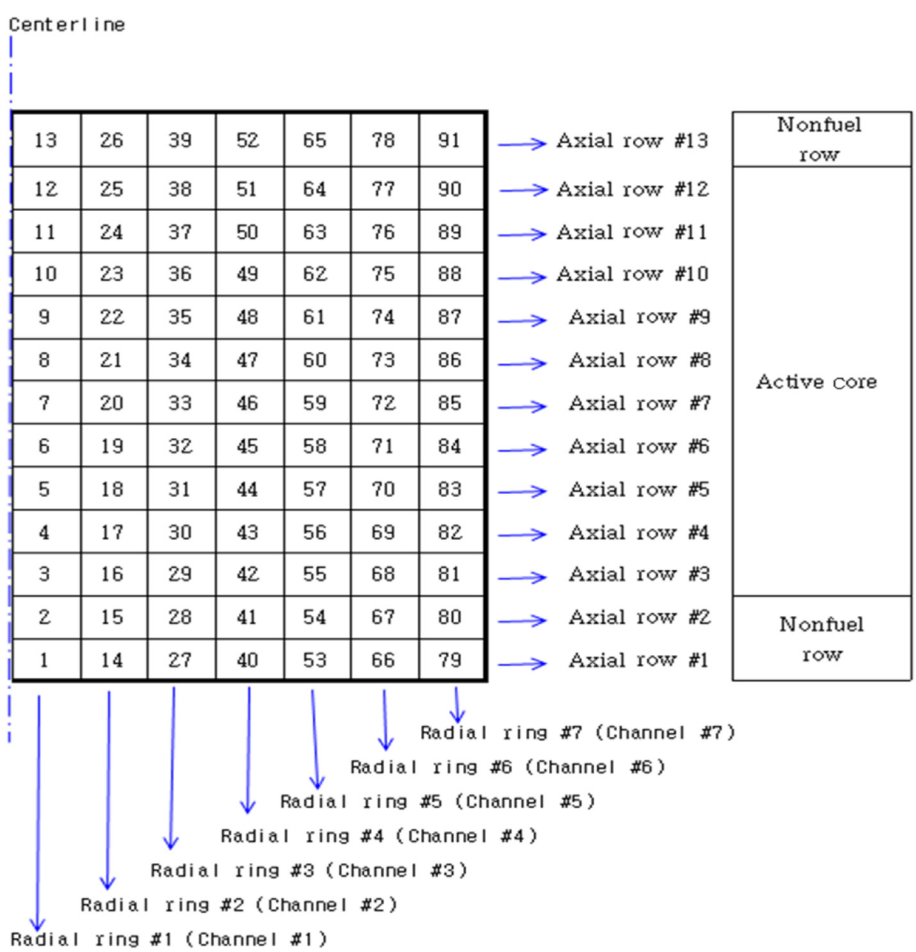

Fig. 2 – The definition of reactor core node for the analysis.

into RCS is 1,336 lpm when the RCS pressure decreases to below 13.53 kg/cm^{2}a.

3. Analysis results

3.1. Results of STSBO

For an STSBO, one unmitigated base case and two mitigation strategies were analyzed. There were no mitigative actions in the unmitigated base case, while two mitigation strategies included the strategies of the cooling water injection into SGs or RCS. The calculations were performed over 72 hours from the accident's initiation.

3.1.1. *Unmitigated base case*

Figs. 3–6 show plots of the plant time parameter variables for an STSBO unmitigated base case. Following a simultaneous loss of off-site and on-site AC power, a reactor scram occurs immediately due to a loss of power to the control rod drive mechanism. Since the STSBO involves a loss of TD-AFW systems through the loss of direct current control power, and has no other mitigative operator actions or power recovery, the heat is removed from the secondary side only by the SG water inventory and MSSVs until the inventory is depleted. Engineered safety features such as high-pressure safety injection and low-pressure safety injection are not available. The only water available to cool the core on the primary side is the initial reactor coolant system inventory.

Following the reactor trip, the pressure in the RCS decreases slowly for about 1 hour until the SGs dry out (Fig. 3). The RCS pressure then increases because of the continued addition of decay heat to the water up to the pressurizer safety valve (PSV) set point (17.2 Ma). PSVs start the cycling of opening/closing at 1.07 hours (Fig. 4). The inventory of the RCS is then lost as a water phase or a two-phase mixture through the PSVs. The water level in the vessel (Fig. 5) continues to decrease owing to the loss of RCS inventory through the PSVs after the pressure reaches the PSV set point and the core becomes uncovered at 1.99 hours.

The uncovered region of the fuel then heats up, owing to insufficient water/steam flows, to the onset temperature of

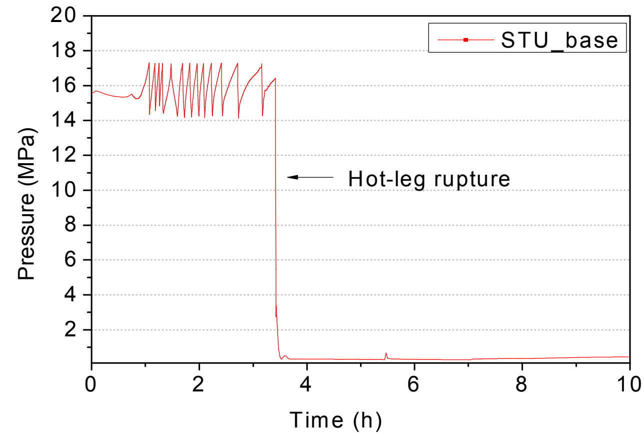

Fig. 4 – Pressure in reactor cooling system for short-term station blackout base case.

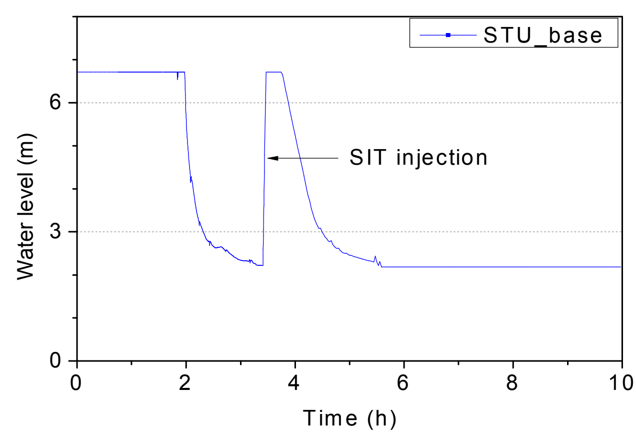

Fig. 5 – Water level in reactor cooling system for short-term station blackout base case. SIT, safety injection tanks.

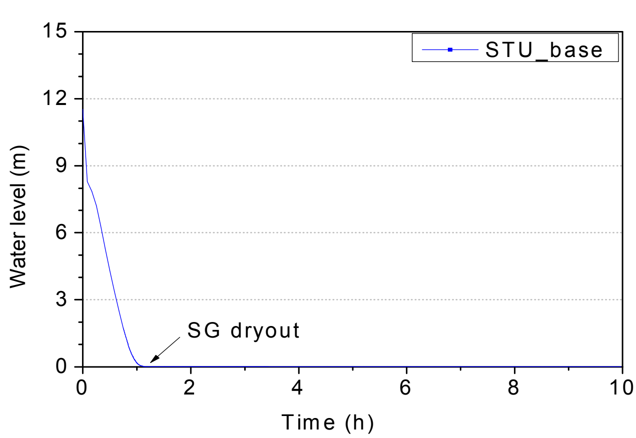

Fig. 3 – Water level in the steam generator for short-term station blackout base case. SG, steam generator.

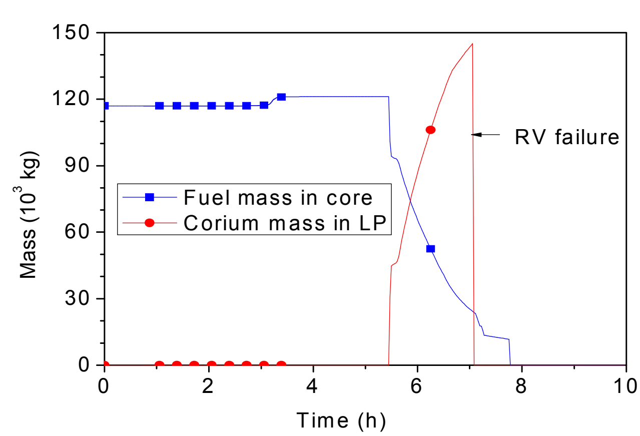

Fig. 6 – Fuel mass in core and corium mass in lower plenum for short-term station blackout base case. LP, lower plenum; RV, reactor vessel.

the zircaloy-steam reaction, and then quickly rises to the fuel melting temperature at 3.16 hours (Table 2). As the core melting spreads, a number of hot gases are generated in the core. A natural circulation of hot gases through the RCS loop results in a temperature-induced hot leg rupture. In this case, the hot leg ruptures at about 3.42 hours, and the inventory of the safety injection tank is injected as soon as the hot leg has ruptured. Molten corium relocates into the lower plenum at 5.45 hours and a vessel eventually fails at 7.06 hours (Fig. 6). When a temperature induced rupture occurs, the vessel failure is delayed by the injection of the safety injection tank inventory resulting from the depressurization of the RCS. The major event occurrence times predicted by the MAAP code during an STSBO transient are summarized in Tables 2 and 3 (Sequence ID: STU-base).

3.1.2. *Mitigated Case 1: Cooling water injection into SGs*

Four sensitivity cases were analyzed for an evaluation of the SG injection strategy depending on the number of opening ADVs and the opening time of the ADV. The assumptions for the sensitivity cases and the calculation results of the timing of key events for the SG injection using fire trucks are summarized in Table 2.

Following the dry out of the SGs, the RCS pressure increases and the PSV is opened at 1.07 hours. If one ADV is opened at the time of the PSV opening and the cooling water is injected through a fire engine, it successfully cools down the reactor core and the core is not uncovered (Table 2, Sequence ID: STM1-1ADV-PSV05). When one ADV is opened at 1 hour after the PSV is opened, the core is uncovered at 2 hours but no further accident progression, such as a core melt, is predicted (Table 2, Sequence ID: STM1-1ADV-PSV60). Meanwhile, when one or two ADVs are opened at 2 hours after the PSV opening time, then a hot leg rupture and reactor vessel failure are inevitable (Table 2, Sequence ID: STM1-1ADV-PSV120, STM1-2ADV-PSV120).

3.1.3. *Mitigated Case 2: Cooling water injection into RCS*

Four sensitivity cases were analyzed for an evaluation of the RCS injection strategy depending on the number of SDSs and the opening time of the SDS. The assumptions for the sensitivity cases and the calculation results of the timing of key events for the RCS injection using fire trucks are summarized in Table 3.

As discussed earlier, after the SGs have dried out, the PSV is opened at 1.07 hours in the STSBO sequence. The calculation results show that if one SDS starts to depressurize the RCS as early as the PSV opening time, the system pressure reaches the point where the fire engine can provide an emergency cooling water into the RCS before the hot leg rupture or reactor vessel failure (Table 3, Sequence ID: STM2-1SDS00). If the RCS depressurization starts with two SDS systems within 2 hours after the PSV opening, the temperature induced hot leg rupture and reactor vessel failure can be prevented (Table 3, Sequence ID: STM2-2SDS60, STM2-2SDS120).

3.2. *Results of LTSBO*

For an LTSBO, one unmitigated base case and two mitigation strategies were analyzed. In an LTSBO, AFW is delivered to

Table 2 – The assumptions and calculation results of key events for the steam generator injection cases using fire truck in short-term station blackout.

Sequence ID	Assumption: ADV open for SG depressurization		Calculation results (event summary, hr)								
	No. of ADV	Opening time	PSV open	ADV open	SG makeup	Core uncovery	Core melt	Hot leg rupture	SIT injection	Corium relocation into LP	RV failure
STU-base	N/A	N/A	1.07	N/A	N/A	1.99	3.16	3.42	3.43	5.45	7.06
STM1-1ADV-PSV05	1	at PSV open + 5 min	1.07	1.16	1.18	No uncovery	No melt	No rupture	10.33	No relocation	No failure
STM1-1ADV-PSV60	1	at PSV open + 1 hr	1.07	2.07	2.10	2.00	No melt	No rupture	8.41	No relocation	No failure
STM1-1ADV-PSV120	1	at PSV open + 2 hr	1.07	3.07	3.10	2.00	3.13	3.49	3.50	6.49	7.96
STM1-2ADV-PSV120	2	at PSV open + 2 hr	1.07	3.07	3.10	2.00	3.13	3.90	3.90	7.14	9.27

ADV, atmospheric dump valves; LP, lower plenum; PSV, pressurizer safety valve; RV, reactor vessel; SG, steam generator; SIT, safety injection tanks.

Table 3 – The assumptions and calculation results of key events for the reactor cooling system injection cases using fire truck in short-term station blackout.

Sequence ID	Assumption: SDS open for RCS depressurization		Calculation results (event summary, hr)								
	No. of SDS	Opening time	PSV open	SDS open	Core uncovery	Core melt	Hot leg rupture	SIT injection	RCS makeup	Corium relocation into LP	RV failure
STU-base	N/A	N/A	1.07	N/A	1.99	3.16	3.42	3.43	N/A	5.45	7.06
STM2-1SDS00	1	at PSV open	1.07	1.10	1.74	2.22	No rupture	2.21	6.15	5.42	No failure
STM2-2SDS60	2	at PSV open + 1 hr	1.07	2.10	1.99	No melt	No rupture	2.31	6.10	No relocation	No failure
STM2-2SDS120	2	at PSV open + 2 hr	1.07	3.10	1.99	3.04	No rupture	3.27	6.71	6.46	No failure
STM2-2SDS180	2	at PSV open + 3 hr	1.07	4.10	1.99	3.16	3.42	3.43	3.46	No relocation	No failure

LP, lower plenum; PSV, pressurizer safety valve; RCS, reactor cooling system; RV, reactor vessel; SDS, safety depressurization system; SIT, safety injection tanks.

steam generators using the TD-AFW pump. Decay heat can be removed from the secondary side by TD-AFW and ADVs during the 4 hours of battery power supply. There is no mitigative action using a fire engine in the unmitigated base case, while two mitigation strategies include the strategies of the cooling water injection into steam generators or into RCS using fire engines. The calculations are performed for 144 hours from the accident initiation.

3.2.1. *Unmitigated base case*

Figs. 7–10 show plots of the plant time parameter variables for the LTSBO unmitigated base case. The major event occurrence times predicted by the code are summarized in Tables 4 and 5 (Sequence ID: LTU-base).

Following the reactor trip, the RCS pressure and water temperature decreases over a four-hour period. During that time interval, the AFW system supplies water to the SGs (Fig. 7) and the ADV is opened. The ADV is opened manually by the operator and the heat transfer rate is controlled to maintain the RCS water temperature between 563 K and 570 K [10]. At 4 hours into the accident, the TD-AFW pumps

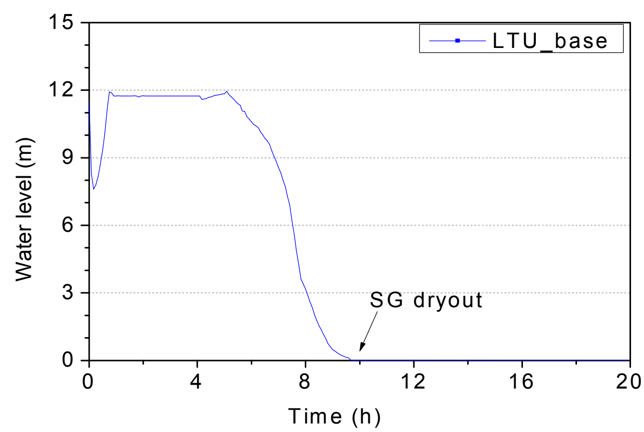

Fig. 7 – Water level in the steam generator for long-term station blackout base case. SG, steam generator.

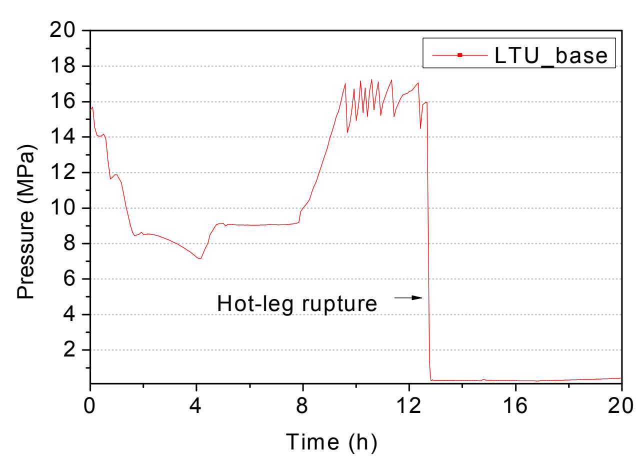

Fig. 8 – Pressure in reactor cooling system for short-term station blackout base case.

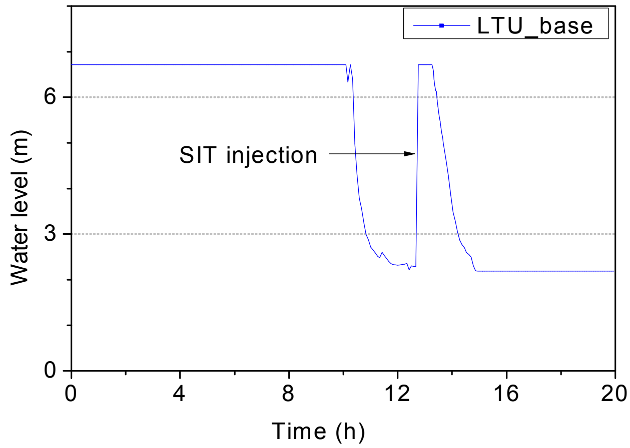

Fig. 9 – Water level in reactor cooling system for long-term station blackout base case. SIT, safety injection tanks.

fail to deliver feed water and the ADV is closed. The RCS pressure is maintained for 4 hours as the MSSVs are opened (Fig. 8).

The RCS pressure increases from about 8 hours as the water level of the SGs decreases (Fig.7), and reaches the PSV set point (17.24 MPa). The PSVs start the open/close cycle at 9.6 hours (Fig. 8, Table 4). The water level in the vessel (Fig. 9) continues to decrease owing to the loss of RCS inventory through the PSVs, and the core becomes uncovered (water level decreases to 6.1 m) at 10.2 hours (Table 4). The uncovered region of the fuel then heats up and quickly rises to the fuel melting temperature at 12.3 hours (Table 4). A natural circulation of hot gases through the RCS loop results in a temperature induced hot leg rupture at 12.7 hours. The inventory of four safety injection tanks is injected into the RCS after the hot leg rupture (Fig. 9). The molten core material relocates into the lower plenum and the reactor vessel eventually fails at 16.9 hours (Fig. 10, Table 4).

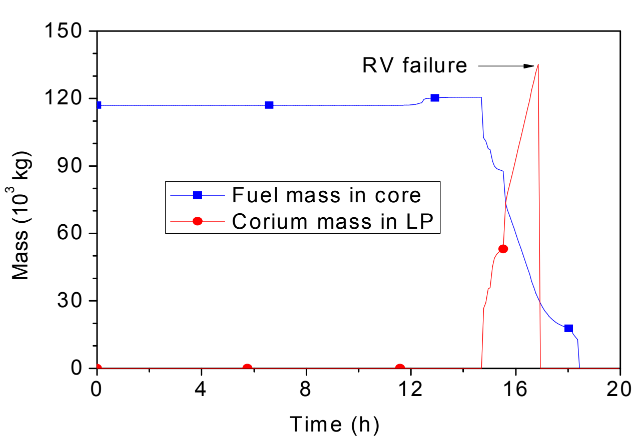

Fig. 10 – Fuel mass in core and corium mass in lower plenum for long-term station blackout base case. LP, lower plenum; RV, reactor vessel.

Table 4 – The assumptions and calculation results of key events for the steam generator injection cases using fire truck in long-term station blackout.

Sequence ID	Assumption: ADV open for SG depressurization		Calculation results (event summary, hr)								
	No. of ADV	Opening time	PSV open	ADV open	SG makeup	Core uncovery	Core melt	Hot leg rupture	SIT injection	Corium relocation into LP	RV failure
LTU-base	N/A	N/A	9.6	N/A	N/A	10.2	12.3	12.7	12.7	14.7	16.9
LTM1-1ADV-PSV05	1	at PSV open + 5 min	9.6	9.7	9.8	No uncovery	No melt	No rupture	10.8	No relocation	No failure
LTM1-1ADV-PSV60	1	at PSV open + 1 hr	9.6	10.6	10.7	10.3	No melt	No rupture	16.4	No relocation	No failure
LTM1-1ADV-PSV180	1	at PSV open + 3 hr	9.6	12.6	12.7	10.3	12.4	12.7	12.7	16.7	18.5

ADV, atmospheric dump valves; LP, lower plenum; PSV, pressurizer safety valve; RV, reactor vessel; SG, steam generator; SIT, safety injection tanks.

Table 5 – The assumptions and calculation results of key events for the reactor cooling system injection cases using fire truck in long-term station blackout.

Sequence ID	Assumption		Calculation results (event summary, hr)								
	SDS open for RCS depressurization		PSV open	SDS open	Core uncovery	Core melt	Hot leg rupture	SIT injection	RCS makeup	Corium relocation into LP	RV failure
	No. of SDS	Opening time									
LTU-base	N/A	N/A	9.6	N/A	10.4	12.5	12.7	12.7	N/A	15.3	17.0
LTM2-1SDS00	1	at PSV open	9.6	9.6	10.0	16.1	No rupture	10.4	18.6	106.0	No failure
LTM2-2SDS120	2	at PSV open + 2 hr	9.6	11.6	10.4	No melt	No rupture	11.8	12.7	No relocation	No failure
LTM2-2SDS180	2	at PSV open + 3 hr	9.6	12.6	10.4	12.5	No rupture	12.8	14.2	No relocation	No failure
LTM2-2SDS300	2	at PSV open + 5 hr	9.6	14.6	10.4	12.5	12.7	12.7	12.7	No relocation	No failure

LP, lower plenum; PSV, pressurizer safety valve; RCS, reactor cooling system; RV, reactor vessel; SDS, safety depressurization system; SIT, safety injection tanks.

3.2.2. *Mitigated Case 1: Cooling water injection into SGs*

Three sensitivity cases were analyzed for the evaluation of the SG injection strategy depending on the number of opening ADVs and the opening time of the ADV. The assumptions for the sensitivity cases and the calculation results of the timing of key events for the SG injection using fire trucks are summarized in Table 4.

Following the SGs dry out, the RCS pressure increases and the PSV is opened at 9.6 hours. If one ADV is opened at the time of PSV opening and the cooling water is injected through a fire engine, it successfully cools down the reactor core and the core is not uncovered (Table 4, Sequence ID: LTM1-1ADV-PSV05). When one ADV is opened at 1 hour after the PSV is opened, the core is uncovered at 10.3 hours, but no further accident progression such as a core melt is predicted (Table 4, Sequence ID: LTM1-1ADV-PSV60). Meanwhile, when one ADV is opened at 3 hours after the PSV opening time, a hot leg rupture and reactor vessel failure are inevitable (Table 4, Sequence ID: LTM1-1ADV-PSV180).

3.2.3. *Mitigated Case 2: Cooling water injection into RCS*

Four sensitivity cases were analyzed for an evaluation of the RCS injection strategy depending on the number of SDSs and their opening time. The assumptions for the sensitivity cases and the calculation results of the timing of key events for the RCS injection using fire trucks are summarized in Table 5.

As discussed earlier, after the SGs have dried out, the PSV is opened at 9.6 hours in an LTSBO sequence. The calculation results show that if one SDS starts to depressurize the RCS as early as the PSV opening time, the system pressure reaches the point where the fire engine can make-up emergency cooling water into the RCS, before the hot leg rupture or reactor vessel failure (Table 5, Sequence ID: LTM2-1SDS00). If the RCS depressurization starts with two SDS systems within 3 hours after the PSV opening, the temperature-induced hot leg rupture and reactor vessel failure can be prevented (Table 5, Sequence ID: LTM2-2SDS120, LTM2-2SDS180). In the case of a 5-hour delay of the SDS opening, the reactor vessel is prevented even though the hot leg rupture has already occurred (Table 5, Sequence ID: LTM2-2SDS300).

4. Summary and conclusions

This paper illustrates an evaluation for the effectiveness of external cooling water injection strategies using fire trucks during a potential extended SBO. The strategies of emergency water injection into the SG and RCS are included. In addition, the STSBO and LTSBO sequences are considered. The time, when the depressurization with the ADV of the SG secondary side or with the PSV of the RCS is initiated, is focused on in this study, which might be a key feature for a successful strategy implementation.

The analysis results lead to the summary that the SG or RCS depressurization should be carried out before about 2 hours from the accident initiation to prevent severe core damage for the STSBO, and that the depressurization should be carried out before about 10 hours from the accident initiation to prevent core damage for the LTSBO.

The USNRC performed a SOARCA to develop a body of knowledge regarding the realistic outcomes of severe reactor accidents. The availability of the external cooling water injection time was assessed to occur at 3.5 hours [11], the time of which include the following: (1) initial plant status assessment by operators, (2) attempt to start an emergency diesel generator manually, (3) manning and operation of the onsite technical support center and offsite emergency operations facility, (4) decision-making of the technical support center and emergency operations facility for the recommendation of operator actions; and (5) operator's assessment and implementation of recovery actions.

With regard to the effectiveness of external cooling water injection strategies using fire engines in an OPR-1000, it can be concluded from the above discussion, that the strategies are judged to likely be ineffective for the STSBO. However, the strategies are very feasible for LRSBO based on the emergency response time assessed by the SOARCA project. In addition, the operation of a TD-AFW system is the key important mitigation measure for the successful implementation of the strategy.

Conflicts of interest

All authors have no conflicts of interest to declare.

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