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Original Article

RCD success criteria estimation based on allowable coping time

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ABSTRACT

When a loss of coolant accident (LOCA) occurs in a nuclear power plant, accident scenarios which can prevent core damage are defined based on break size. Current probabilistic safety assessment evaluates that core damage can be prevented under small-break LOCA (SBLOCA) and steam generator tube rupture (SGTR) with rapid cool down (RCD) strategy when all safety injection systems are unavailable. However, previous research has pointed out a limitation of RCD in terms of initiation time. Therefore, RCD success criteria estimation based on allowable coping time under a SBLOCA or SGTR when all safety injection systems are unavailable was performed based on time-line and thermal-hydraulic analyses. The time line analysis assumed a single emergency operating procedure flow, and the thermal hydraulic analysis utilized MARS-KS code with variables of break size, cooling rate, and operator allowable time. Results show while RCD is possible under SGTR, it is impossible under SBLOCA at the APR1400's current cooling rate limitation of 55 K/hr. A success criteria map for RCD under SBLOCA is suggested without cooling rate limitation.

1. Introduction

Probabilistic safety assessment (PSA) is a methodology to estimate the safety of nuclear power plant quantitatively in 3 levels. Level 1 PSA, level 2 PSA, and level 3 PSA are sequential analyses, where the results of each assessment usually serve as a basis for the PSA at the next level. Among these, level 1 PSA provides insights into design weaknesses and into ways of preventing accidents leading to core damage (CD), which might be the precursor of accidents leading to major releases of radioactive material with potential consequences for human health and the environment. The scope of a level 1 PSA is classified by the cause of the event (i.e. internal and external) and operational conditions of the plant (i.e. full power, low power, and shutdown) [1]. Loss of coolant accident (LOCA) which is induced by a rupture or break in the reactor coolant system (RCS) is one of the most important initiating events in at-power Level 1 internal PSA for pressurized water reactor. When LOCA occurs, operators mitigate the situation using the safety injection system (SIS) to maintain the RCS inventory. Current Level 1 PSA for APR1400 evaluates that there is no way to prevent CD under large-and medium-size break LOCA, which are cases with break sizes larger than 2 inches in diameter if all SISs are unavailable. However in the cases of a small break LOCA (SBLOCA) or a steam generator tube rupture (SGTR), it is evaluated that CD can be prevented with the shut-down cooling system, but only after rapid cool down (RCD) [2]. However, previous research [3] has pointed out a limitation of RCD in terms of initiation time.

APR1400 was used as an example nuclear power plant in this research. The APR1400 is an advanced light water reactor developed in the Republic of Korea. It has a two-loop RCS, with each loop consisting of one steam generator (SG), two cold legs, one hot leg, and two reactor coolant pumps (RCPs). A pressurizer is connected to the hot leg. The SG is a vertical inverse U-tube heat exchanger with an integral economizer, which operates with the RCS coolant in the tube side and secondary coolant in the shell side. About 13,000 tubes are used per SG. The water level in the SG is controlled automatically over the full power operating range by the main feedwater system (MFWS). The main steam system transfers the steam from the SG to the turbine through the steam line. The main steam safety valve (MSSV), the turbine bypass valve, and the atmospheric dump valve (ADV) are installed on the steam line connected to the upper head of the SG. The MSSVs prevent overpressurization of the SG automatically, turbine bypass valve and the ADV are used to de-pressurize the SG by the operator.

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The safety systems and features of APR1400 are designed to be a hybrid system in which active and passive systems perform necessary safety functions. The major safety systems are the SIS, safety depressurization and vent system, in-containment refueling water storage system, the SCS, auxiliary feedwater system (AFWS), and containment spray system. The SIS is comprised of four independent mechanical trains without any tie line among the injection paths and two electrical divisions. Each train has one active safety injection pump and one passive safety injection tank. The SCS is a safety-related system that is used in conjunction with the main steam system and MFWS or AFWS to reduce the RCS temperature in post shutdown periods from the hot shutdown operating temperature to the refueling temperature. The in-containment refueling water storage system is located inside the containment, and the arrangement is made in such a way that the injected core cooling water can return to the in-containment refueling water storage tank (IRWST) which is an annular cylindrical tank along the containment wall at low elevation. The IRWST provides the functions of the normal refueling water storage as well as a single source of water for the SIS, SCS, and containment spray system, and it is also connected to the reactor vessel through four direct vessel injection lines. The safety depressurization and vent system is a dedicated safety system designed to provide a safety grade means to depressurize the RCS in the event that pressurizer spray is unavailable during plant cooldown to cold shutdown and to depressurize rapidly the RCS with four pilot-operated safety relief valves which are installed on the head of the pressurizer. This system establishes a flow path from the pressurizer steam space to the IRWST, and the pilot-operated safety relief valve can perform the functions of both safety and relief valves. The AFWS is a dedicated safety system designed to supply feedwater to the SGs for removal of heat from the RCS for events in which the MFWS are unavailable, and it is an independent 2-division system, one for each SG, and each division has 2 trains. One 100% capacity motor-driven pump, one 100% capacity turbine-driven pump and one dedicated safety related auxiliary feedwater storage tank are included in each division. The containment spray system consists of two trains and takes the suction of its pump on the IRWST to reduce the containment temperature and pressure during the accident occurred in the containment. The containment spray system is designed to be interconnected with the SCS, which is also comprised of two trains. Also, the pumps of these systems are designed to have the same type and capacity in-between [4].

In this work, a RCD success criteria estimation based on allowable coping time under SBLOCA and SGTR when all SISs are unavailable was performed based on time-line (TL) and thermal-hydraulic (TH) analyses. A single EOP flow was assumed in the TL analysis, with the required time for RCD referred from literature. The TH analysis was performed using MARS-KS code, with break size, cooling rate, and operator allowable time considered as variables. APR1400 was used as an example nuclear power plant in this research.

2. Materials and method

2.1. Time-line analysis

2.1.1. Accident scenario

The objective of RCD is to lower RCS pressure and temperature to the entry condition of the SCS using the AFWS and ADV on SGs when the SIS is unavailable in SBLOCA or SGTR. Fig. 1 shows an accident scenario using RCD in an event tree for SGTR [5].

2.1.2. EOP flow

EOP is a plant specific documents, which contains all of the

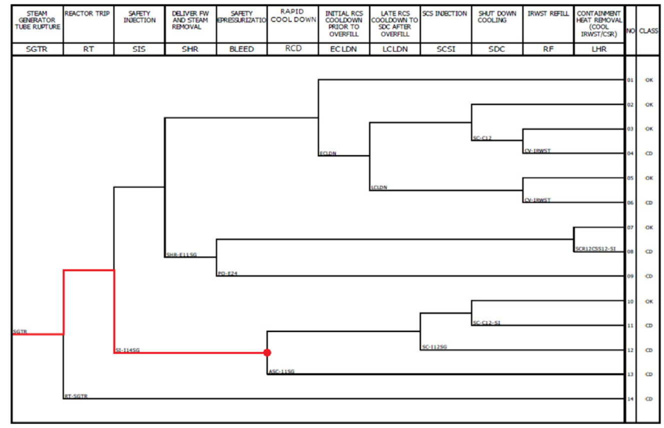

Fig. 1. Accident scenario using RCD in the event tree for SGTR.

processes needed to take the plant from the post-trip state to a safe, stable condition. When an event occurs, operators mitigate the situation to maintain safety functions with the EOP. APR1400 EOP consists of four parts: (a) standard post-trip action (SPTA), (b) diagnostic action (DA), (c) optimal recovery procedure (ORP), and (d) functional recovery procedure (FRP). SPTA is the entry point for the EOP, and operators check all safety functions against acceptance criteria. After SPTA, operators determine the type of event corresponding to the symptom set which causes reactor trip (RT) using a flow chart in DA part. The type of events are grouped into seven classes: (a) RT, (b) LOCA, (c) SGTR, (d) excess steam demand event, (e) loss of all feedwater, (f) loss of offsite power, and (g) station blackout. ORP provides mitigation strategies specifically for each event class, and it contains its own safety function status check (SFSC). Shift supervisor and other three operators mitigate the situation with the ORP until the exit condition is satisfied, meanwhile the shift technical advisor performs the SFSC. If the event cannot be identified or the ORP is not adequate to treat the symptoms, operators mitigate the situation with the FRP. In the first part of FRP, operators determine the priority of safety functions to restore by considering possible success paths with a resource assessment tree and performing the SFSC for each path. There are total eight safety functions: (a) reactivity control, (b) RCS inventory control, (c) RCS pressure control, (d) core heat removal, (e) RCS heat removal, (f) containment isolation, (g) containment temperature & pressure control, and (h) containment combustible gas control. FRP provides actions to restore specifically for each safety function. Following the priority, operators restore all safety functions with FRP. Fig. 2 shows the implementation sequence of APR1400 EOP [6].

After SBLOCA or SGTR occurs, operators initiate the EOP to mitigate the accident situation after automatic RT. In the main control room, the operation team [7] completes the SPTA and diagnoses the initiating event. During the ORP, the shift technical advisor confirms SIS availability with the SFSC. In the case of SIS unavailability following SBLOCA or SGTR, RCS inventory control may be checked in this step. Operators then start to follow the FRP after performing the DA process again. In the FRP-01 process, operators consider mitigation strategies using the resource assessment tree and the SFSC. When SIS is unavailable, operators follow the FRP-04, IC-2 procedure; IC-1 treats the chemical and volume control system, and IC-2 treats the SIS. In step 13, operators initiate RCD at a cooling rate below 55 K/hr [8].

In this EOP, it is assumed that there are no errors of omission or errors of commission throughout the process. In a report from the Korea Atomic Energy Research Institute (KAERI) [9], the time between processes is approximately estimated based on experimental

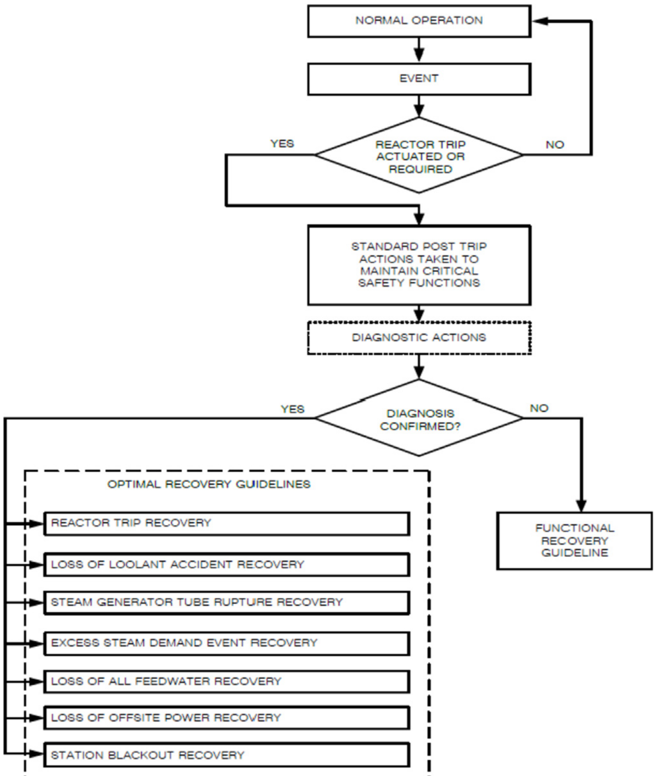

Fig. 2. Implementation sequence of APR1400 EOP.

data from an OPR1000 (predecessor to the APR1400) simulator and human performance analysis data from units 3 and 4 at the Shin-Kori nuclear power plant. According to this report, it takes about 35 min from RT to RCD. Fig. 3 shows the EOP flow and TL analysis results from RT to initiation of RCD in SBLOCA or SGTR when all SISs are unavailable.

2.2. Thermal-hydraulic analysis

2.2.1. APR1400 model for MARS-KS code

A TH analysis on RCD under SBLOCA and SGTR was performed using MARS-KS code. Developed by KAERI based on RELAP5/MOD3 and COBRA-TF codes [10,11], MARS-KS code is considered as a best-estimate system code to conduct the TH analyses for Level 1 internal PSA of South Korean nuclear power plants [12].

Before performing the TH analysis, steady-state analysis on APR1400 model for MARS-KS code was done. The MARS nodalization of the APR1400 is depicted in Fig. 4. APR1400 is designed to generate 3983 MWth of thermal output. The design temperature in the hot leg and cold legs are 596.9 K and 563.6 K at the normal operating pressure, 15.51 MPa with 5250 kg/s of flow rate for each cold leg. The design pressure in the SG is 6.84 MPa, and the steam flow rate for each SG is 1130.6 kg/s in the normal condition [13]. The result of steady-state analysis is obtained as shown in Table 1. The maximum error is about 2% compared to the design value of APR1400. Fig. 5 shows the steady-state results of RCS hot leg and cold leg coolant temperature, and Fig. 6 shows the steady-state results of RCS and SG pressure.

AFWS and SCS are safety systems which are used in the scenario that all SISs are unavailable under the SBLOCA or SGTR. Also, MSIS and AFAS are used in this scenario. When the narrow range level of SG is over than 91% or the SG pressure is lower than 5.895 MPa, the MSIS is activated. After the activation of MSIS, main steam isolation valve is closed to isolate SG and MFWS is shut down. When the wide range of SG is lower than 25%, AFAS is activated. AFWS automatically maintains the wide range level between 25% and 40%. SCS initiates to operate when the RCS pressure is lower than 2.758 MPa and coolant temperature in direct vessel injection line is lower than 477.4 K. Conditions of the engineered safety feature actuation signal and properties of safety features used in the TH analysis are described in Tables 2 and 3. RT and turbine trip occur automatically following trip signals. But in the case of RCPs, the operator should turn them off manually according to the condition of RCS pressure and the sub-cooling margin. In the SPTA process, it is indicated that the operator should check the plant condition and determine whether to maintain the RCPs or not; it is assumed here that the RCPs are turned off 6 min after RT at the end of the SPTA process in Fig. 3, following the plant condition. From each RCP, 6.15 MWth of thermal output is generated until the trip. Reactor, turbine, and RCP trip conditions are described in Table 4. AFWS is assumed to activate on a SG with a motor-driven pump of 100% capacity. In case of the SGTR, AFWS is assumed to activate in the intact SG. A single ADV is assumed to be used for the RCD, and the ADV is located on the SG which the AFWS is available. The ADV is opened for a time step at the RCD initiation time. The cooling rate of the RCD is calculated at each time step, based on the RCS hot leg temperature. It is decided that whether ADV is opened or not for

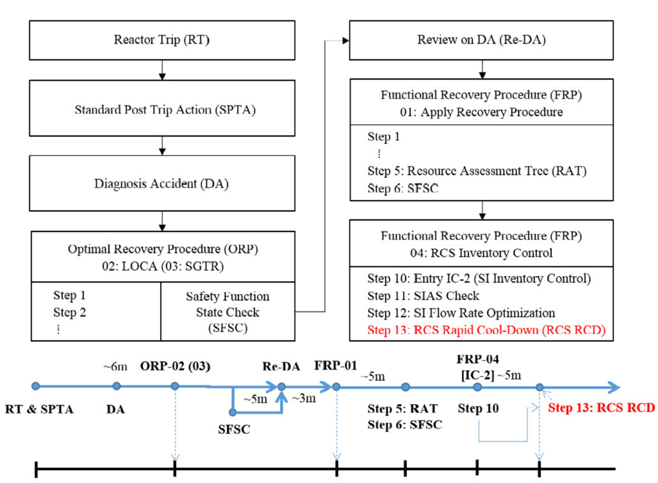

Fig. 3. EOP flow and TL analysis result from RT to initiation of RCD in SBLOCA or SGTR when all SISs are unavailable.

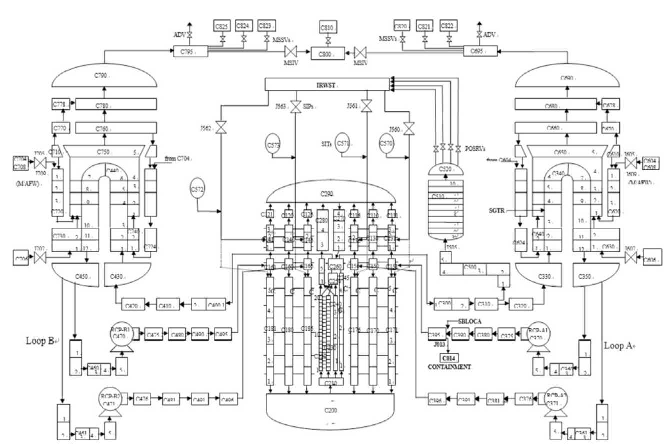

Fig. 4. MARS-KS nodalization for APR1400.

Table 1
Comparisons between steady-state results and design value of APR1400.

Parameter	Design value	Steady-state result
Power (MWth)	3983	3983
RCS pressure (MPa)	15.51	15.76
RCS hot leg temperature (K)	596.9	597.1
RCS cold leg temperature (K)	563.6	563.9
Total RCS flow rate (kg/s)	21,000	20,994
SG pressure (MPa)	6.84	6.90
Total Steam flow rate (kg/s)	2261.2	2307.4

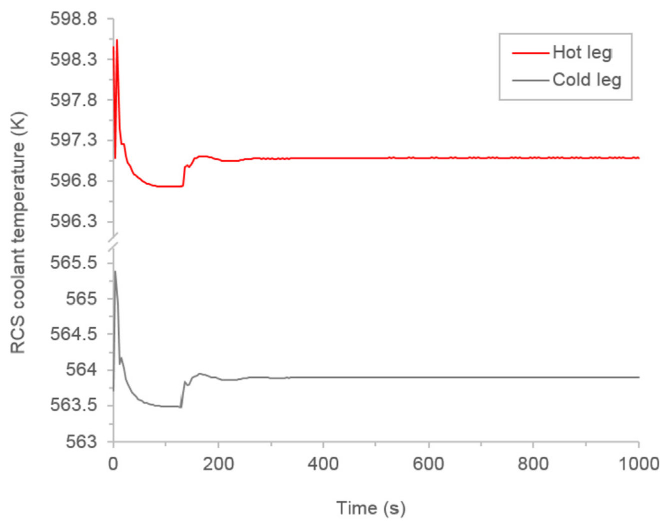

Fig. 5. RCS hot leg and cold leg coolant temperature from steady-state results of APR1400.

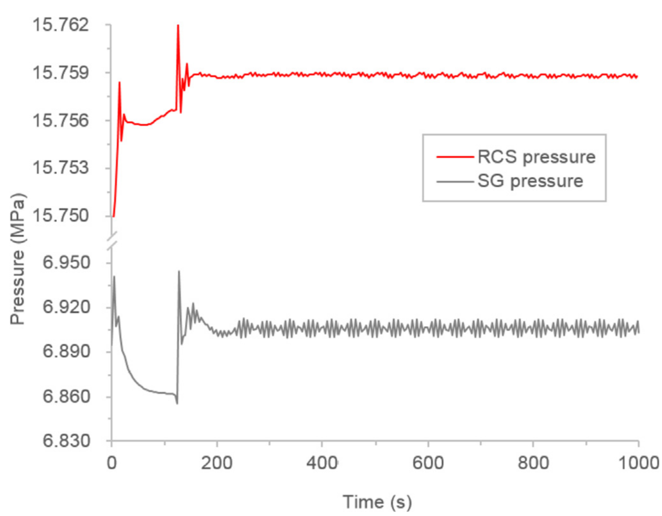

Fig. 6. RCS and SG pressure from steady-state results of APR1400.

Table 2
Conditions of the engineered safety feature actuation signal.

ESFAS	Actuation condition	Logic	Set point		
			Parameter	Value	Unit
MSIS	SG high level	OR	SG NR level	91	%
	SG low pressure		SG main steam pressure	5.895	MPa
AFAS	SG low level	–	SG WR level	25	%

Table 3
Properties of safety features used in the TH analysis.

Safety feature	Parameter		Value	Unit
AFWS	Set point	Time after AFAS	61.45	s
	Initiation	SG WR level	25	%
	Stop	SG WR level	40	%
	Performance	Flow rate	41.07	Kg/s
AFWST	Capacity	–	1,514,000	Kg
MFWS isolation	Set point	Time after MSIS	11.35	s
MSSV	Set point	SG main steam pressure	8.48	MPa
MSIV	Set point	Time after MSIS	6.35	s
RCP	Performance	Additional heat	6.15	MW
SCS	Set point	RCS pressure	2.758	MPa
		RCS coolant temperature	477.4	K
	Performance	Max flow rate	342.23	Kg/s

the next time step by the cooling rate in the previous time step so that the desired cooling rate can be obtained. For a conservative analysis, non-1-E safety systems (e.g. turbine bypass valve, charging pump) and safety injection tank are not considered in this research.

SBLOCA is assumed to occur on the cold leg between RCP and reactor vessel with a round break between 0.5 inch and 2.0 inch diameter following conventional PSA, with a resolution of 0.25 inch. SGTR is assumed to occur on the bottom of the down-comer line. The conventional PSA defines SGTR as a single u-tube guillotine break, but this research considers break size in the range of 0.5 A–2.0 A, with a resolution of 0.25 A; the break size of a single u-tube guillotine break is 1.0 A. It is assumed that a single MSSV stuck open on SGTR occurs when it first opens. CD is assumed to occur when peak cladding temperature reaches 1477 K [14]. If the duration time of the SCS is over 1 h, it is assumed that CD can be prevented.

2.2.2. Base case analysis

Base case analysis was performed to find out the maximum operator allowable time and to check the validity of the accident scenario with various break sizes. Here, two kinds of the base case were analyzed: base case-01 (BC-01) refers to the SBLOCA or SGTR when all safety features are unavailable, and base case-02 (BC-02) refers to the SBLOCA or SGTR when AFWS is only available. In the base case analysis on the pressurizer and the SG are assumed to be available. Fig. 7 shows the resulting average break flow in SBLOCA and SGTR BC-02, and Tables 5 and 6 describe main event timing and maximum operator allowable time for RCD in SBLOCA and SGTR BC-02 with various break sizes. Because the total amount of leakage until CD is almost the same in all cases, average break flow is inversely proportional to CD timing. Accident sequences of BC-01 and BC-02 are same until the initiation of the AFWS. There is no cue for RCD in the EOP of APR1400, so the theoretical maximum operator allowable time is the timing between RT and CD.

In the SBLOCA BC-01, RT and turbine trip occur by the pressurizer low-pressure signal without reference to the break size. The RCS pressure rapidly decreases and the SG pressure rapidly increases, thereby inducing MSSV and main steam isolation valve operation, as well as MFWS shutdown after RT. The sub-cooling margin on the hot leg drops below 15 K before RT, and it is maintained at 0 K during the whole accident sequence, allowing operators to turn off the RCP in the SPTA process. Residual heat causes an increase of the RCS pressure, on the other hand, the RCS coolant leakage causes the RCS pressure drop by removing the residual heat. When break sizes are below 1.0 inch, pilot-operated safety relief valves on the pressurizer operate automatically because of the lack of residual heat removal via leakage. If the break size is 1.5 inch, it is balanced between heat removal via leakage and residual

Table 4
Trip conditions of reactor, turbine, and RCP.

Safety feature	Trip condition	Logic	Set points			Delay time (s)
			Parameter	Value	Unit	
Rx	PZR low pressure	OR	PZR pressure	12.48	MPa	1.15
	PZR high pressure		PZR pressure	16.39	MPa	0.85
	SG low level		SG WR level	45	%	1.25
	SG high level		SG NR level	91	%	1.15
	SG low pressure		SG main steam pressure	0.5895	MPa	1.15
TBN	Rx trip	–	Time	–	–	0.14
RCP	Low sub-cooling margin	AND	Hot leg	15	K	–
	Rx trip		Time	–	–	360

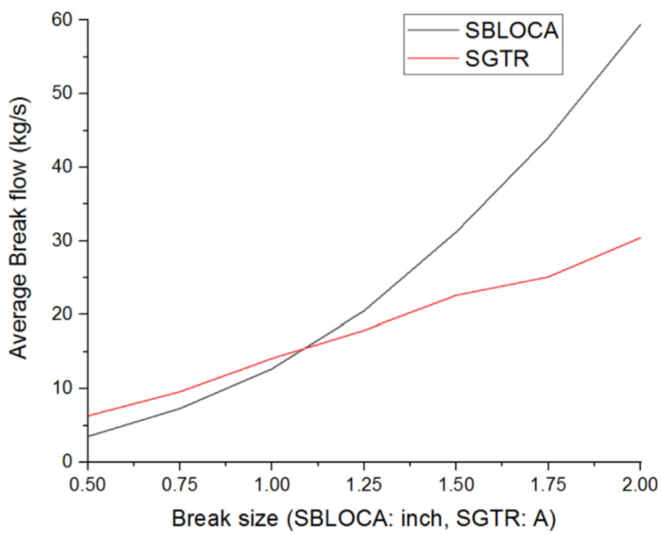

Fig. 7. Average break flow in SBLOCA and SGTR BC-02. The units of break size for SBLOCA and SGTR are inch and A each. For example, SGTR with 1.0 A break size is equal to a single u-tube guillotine break.

Table 5
Main event timing in SBLOCA BC-02 (AFWS only available).

Break size (in)	0.50	0.75	1.00	1.25	1.50	1.75	2.00
Main event 1)[1] (s)	646	271	147	92	63	46	3s
Main event 2)[2] (s)	1652	678	371	237	163	118	89
Main event 3)[3] (s)	1673	699	390	255	179	136	108
Main event 4)[4] (s)	1738	762	453	319	245	200	171
Main event 5)[5] (s)	1743	767	458	324	250	205	176
Main event 6)[6] (s)	2012	1038	731	597	523	478	449
Main event 7)[7] (s)	5340	5075	4740	4737	4413	4506	–
Main event 8)[8] (s)	64,942	31,195	18,111	11,124	7303	5237	3986
CD – Rx trip (min)	1054	509	296	181	119	85	65

Table 6
Main event timing in SGTR BC-02 (AFWS only available).

Break size (A)	0.50	0.75	1.00	1.25	1.50	1.75	2.00
Main event 1) (s)	635	408	299	236	196	166	143
Main event 2) (s)	1713	1094	799	646	533	454	395
Main event 3) (s)	1719	1099	805	652	538	459	400
Main event 4) (s)	1798	1178	882	729	615	536	477
Main event 5) (s)	1803	1183	887	734	620	541	482
Main event 6) (s)	2073	1454	1159	1006	893	814	755
Main event 7) (s)	7873	8507	10,010	10,867	–	–	–
Main event 8) (s)	36,465	23,687	16,019	12,658	10,006	8966	8179
CD – Rx trip (min)	579	377	254	200	158	142	130

[1] Sub-cooling margin on HL < 15 °C.
[2] Rx & TBN trip by PZR low pressure signal.
[3] MSSV first open (in case of SGTR, stuck open).
[4] MSIV by MSIS.
[5] MFWS off by MSIS.
[6] RCP trip.
[7] AFWS initiation.

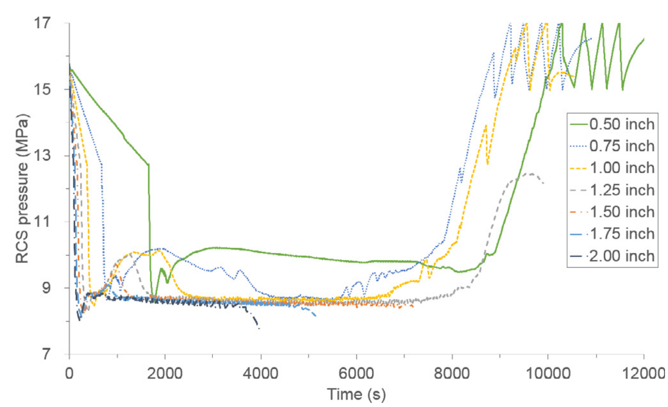

Fig. 8. RCS pressure over time in the SBLOCA BC-01 (no safety feature).

heat from the core, so that the RCS pressure is maintained at the pressure of the SG which the MSSV operates until CD. The RCS pressure decreases until CD in case that break sizes are above 1.75 inch. Fig. 8 shows the RCS pressure over time in the SBLOCA BC-01. The TH analysis on the SBLOCA BC-01 can be verified by the previous research which the TH analysis on the same case for the OPR1000 was conducted [12]. In the SBLOCA BC-02, the AFWS operates when break sizes are below 1.75 inch because remaining heat after the heat removal via the RCS coolant leakage is removed by the SG, otherwise, the AFAS is not activated in case that break size is 2.0 inch. The RCS pressure decreases after the initiation of the AFWS and it is maintained at the SG pressure which is shown in Fig. 9.

The accident sequence of the SGTR BC-01 is similar to the accident sequence of the SBLOCA BC-01 like in Tables 5 and 6 In the SGTR BC-01 without reference to the break size, the RCS pressure rapidly decreases and the SG pressure rapidly increases thereby inducing the MSSV and main steam isolation valve operation, as well as the MFWS shutdown after the RT which is induced by the pressurizer low-pressure signal. At the timing of the first open of MSSV, a single MSSV stuck open occurs on the damaged SG. The RCS pressure decreases because the damaged SG pressure decreases to atmospheric pressure. The minimum RCS pressure is lower with larger break sizes because the damaged SG pressure decreases faster with larger break sizes. In the case of 2.0 A SGTR, the RCS pressure decreases to the minimum 3 MPa. After all the secondary coolant in the damaged SG is exhausted, the RCS pressure increases to the pressure of the intact SG which the MSSV operates normally. When break sizes are below 0.75 A, the RCS

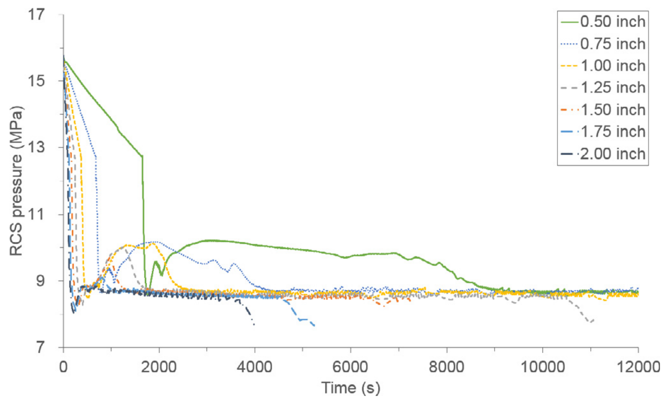

Fig. 9. RCS pressure over time in the SBLOCA BC-02 (AFWS only available).

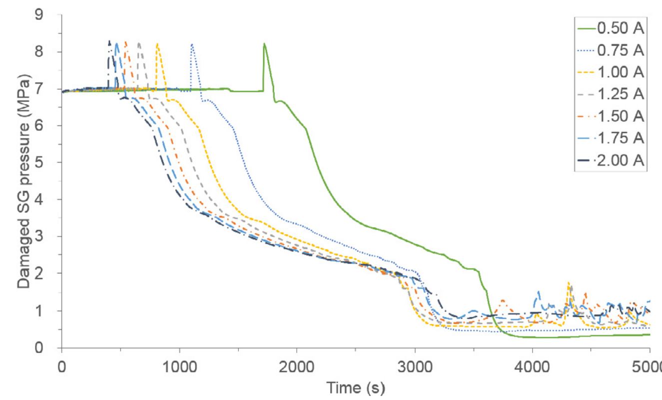

Fig. 11. Damaged SG pressure over time in the SGTR BC-01 (no safety feature).

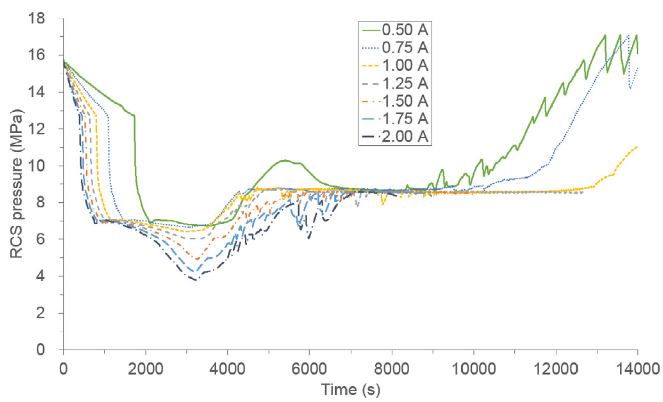

Fig. 10. RCS pressure over time in the SGTR BC-01 (no safety feature).

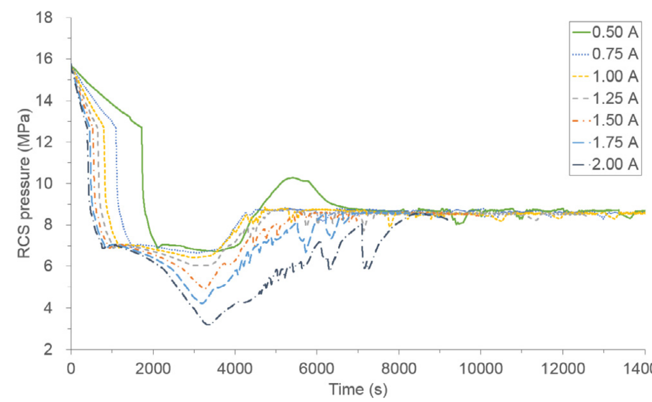

Fig. 12. RCS pressure over time in the SGTR BC-02 (AFWS only available).

pressure increases to the operation pressure of pilot-operated safety relief valves on the pressurizer before CD. If break sizes are below 1.25 A, the RCS pressure increases to the intact SG pressure and it is maintained. Figs. 10 and 11 show the RCS and the damaged SG pressure over time in the SGTR BC-01. In the SGTR BC-02, the AFWS operates when break sizes are below 1.25 A. The RCS pressure is maintained at the intact SG pressure after the initiation of the AFWS which is shown in Fig. 12. In case of the SGTR BC-02, the AFAS is activated later than in the SBLOCA BC-01 because of heat removal by secondary coolant in the damaged SG.

3. Results and discussion

3.1. Operator allowable time

In the current EOP of APR1400, the maximum cooling rate is indicated as 55 K/hr to prevent thermal shock on the RCS [15], and the minimum cooling rate is not indicated. Multiple thermal-hydraulic analyses were performed in this work to obtain operator allowable times according to break size and cooling rates under both SBLOCA and SGTR, with conditions given in Section 2.2.2. The cooling rate was considered from 40 K/hr to 55 K/hr, with a resolution of 5 K/hr. Fig. 13 shows the RCS hot leg temperature in the size of 2.00 inch SBLOCA when RCD is performed. The cooling rate of RCD is 55 K/hr, and initiation time is 30 min after RT. In this case, CD occurs at 4687 s before the initiation of SCS. Figs. 14 and 15 illustrate the operator allowable times according to break size and cooling rates under SBLOCA and SGTR, respectively. Such times drastically decrease for larger break sizes because the theoretical maximum operator allowable time and efficiency of secondary heat

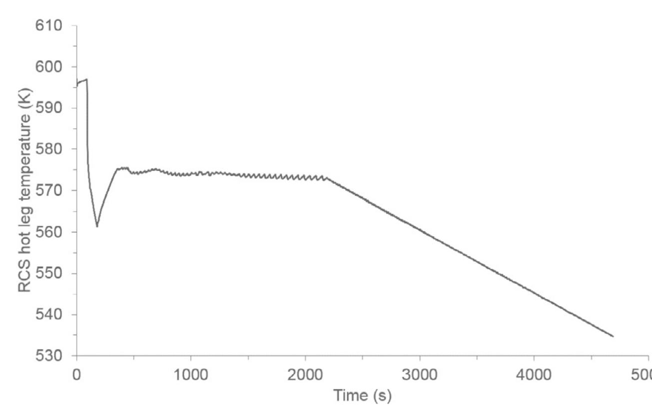

Fig. 13. RCS hot leg temperature in SBLOCA (2.00 inch) when RCD (Cooling rate of RCD: 55 K/hr, Initiation time: 30 min after RT) is performed.

removal also decrease with larger break size. Results demonstrate that CD can be prevented at the 55 K/hr cooling rate over the whole range of SBLOCA and SGTR. But if the TL analysis is considered like in Fig. 14, CD occurs for 1.75 inch and 2.00 inch SBLOCA even with the 55 K/hr cooling rate.

3.2. Success criteria map for RCD

It was concluded that CD cannot be prevented in SBLOCA by RCD if the results of the TL analysis are considered; however, this does not imply that RCD is theoretically impossible under SBLOCA. In the

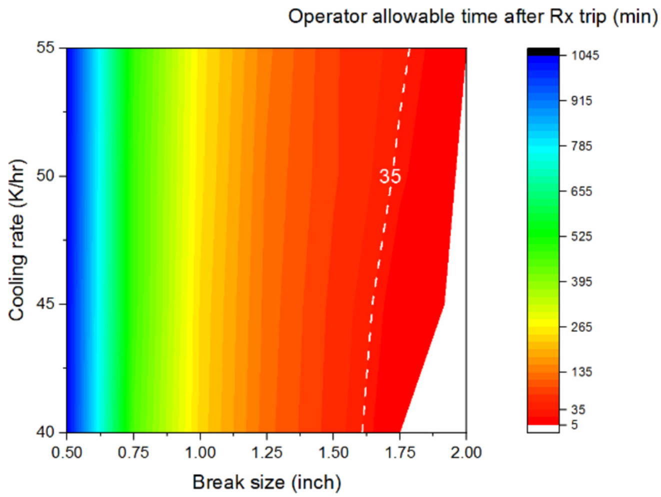

Fig. 14. Operator allowable times according to cooling rate and break size in SBLOCA. The result of the time-line analysis is plotted as the white dotted line, indicating the minimum initiation time.

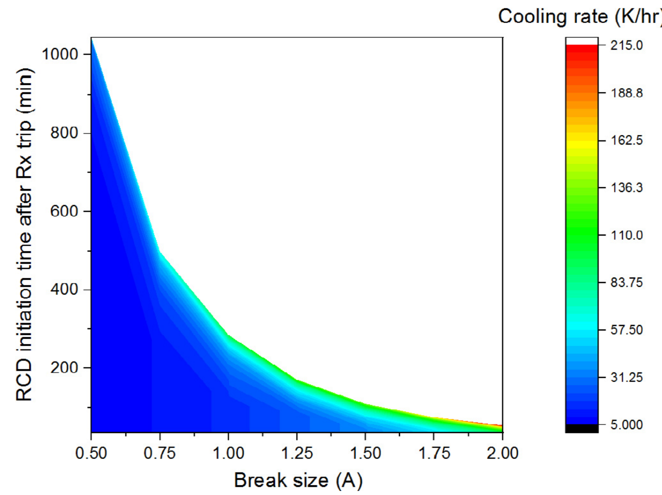

Fig. 16. RCD success criteria map in SBLOCA.

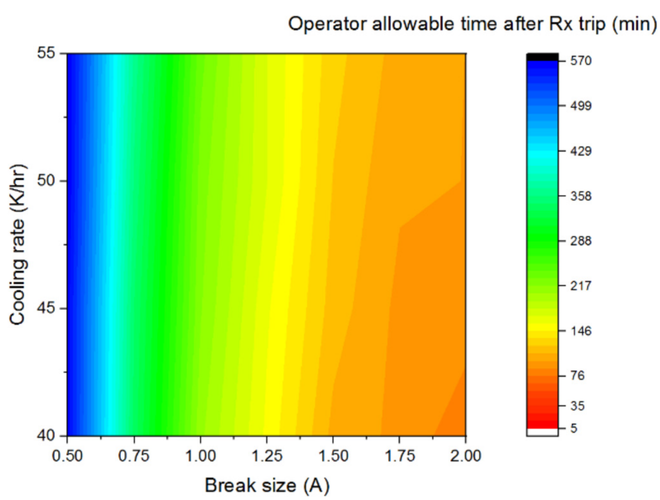

Fig. 15. Operator allowable times according to cooling rate and break size in SGTR.

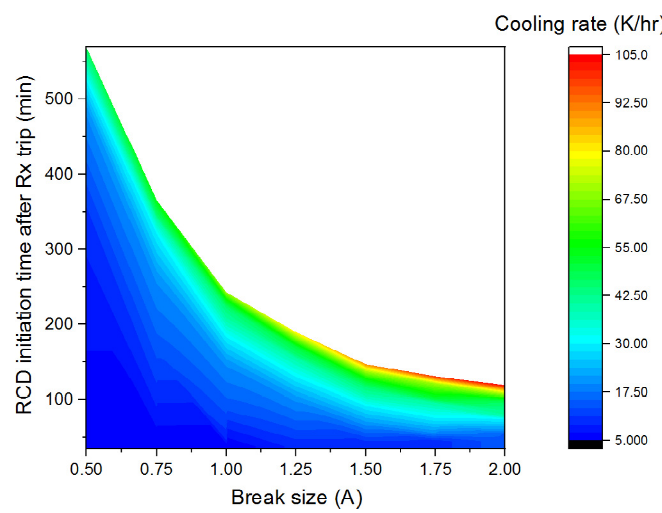

Fig. 17. RCD success criteria map in SGTR.

EOP of a Westinghouse-type nuclear power plant, for example, RCD is divided into ordinary cool down and aggressive cool down strategies. The former is performed in a power cutback or reactor shut down during normal operation, and it has the same 55 K/hr cooling rate limitation as RCD in the EOP of APR1400. On the other hand, aggressive cool down is an emergency strategy performed in SBLOCA and SGTR when all SISs are unavailable, but it has no cooling rate limitation [15]. To apply this no-limit condition to the present case, success criteria maps were established, which include the required cooling rates without limitation according to RCD initiation time after RT, based on multiple TH analyses. The minimum required cooling rates for RCD initiation times from 30 min to 5 min before CD were obtained, with a resolution of 5 min. Then, the minimum cooling rate at 35 min after RT was found, to set a range with the minimum cooling rate at 30 min before CD. The maximum operator allowable times were then obtained in this range of minimum cooling rates, at a resolution of 5 K/hr. Figs. 16 and 17 show the success criteria maps for RCD in SBLOCA and SGTR, respectively. Because CD cannot be prevented in 5 min even if one ADV is fully opened in the 2 inch SBLOCA case, 10 min before CD is considered as the last RCD initiation time. The maximum required cooling rates are 215 K/hr and 105 K/hr for SBLOCA and SGTR, respectively. As shown in Fig. 16, CD can be prevented by RCD considering the TL analysis in SBLOCA if there is no cooling rate limitation for RCD.

4. Summary and conclusion

In this research, a RCD success criteria estimation based on allowable coping time under SBLOCA and SGTR when all SISs are unavailable was performed based on TL and TH analyses. It was determined that RCD is possible under SGTR, but in the case of SBLOCA, it is impossible. In the TL analysis, it was assumed that there were no errors in the accident diagnosis, and that the operators consider RCS inventory control as the priority among all safety functions. Therefore, a single EOP flow was considered. Referenced time values were used between processes to obtain minimum operator required times. If possible, other EOP flows and statistic time values should be used for future TL analyses to obtain more realistic results. It was assumed that RCD was done with AFWS and ADV as a constant cooling rate in the TH analysis. But in the real situation, the constant cooling rate is hard to be achieved because operators perform RCD using ADV manually. More realistic results can be obtained with the more realistic model of operator performance for RCD. Additional researches about the cooling rate limitation of RCD [16] and EOP process may be required. If the RCD cooling rate limitation is maintained, the range of SBLOCA break sizes needs to be updated following the analyses of further accident scenarios for better PSA results. But if the RCD cooling rate limitation is removed like the aggressive cool down in Westinghouse-type nuclear power plant, the success criteria maps which are suggested in this research can be utilized in the success criteria analysis for APR1400.

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