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A Study of BWR Mark I Station Blackout Accident with GOTHIC Modeling

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The overall objective of this study is to develop a high-fidelity simulation capability for the analysis of accident progression during Station Black-Out (SBO) scenario in Boiling Water Reactor (BWR) Mark I plant. The motivation is on predicting the effectiveness of containment venting within accident mitigation strategy and how to avoid core melt by choosing a proper time to initiate primary system venting. In this work, a GOTHIC model has been developed to support characterization of reactor safety systems performance and evaluation of the venting strategy. The GOTHIC model provides a seamless coupled simulation of the reactor coolant system and the containment system based on the design of the Peach Bottom-2 plant system. Efforts are also taken on the analysis for the primary system venting strategy. The safe time intervals (so-called "safe venting window") to initiate the primary system depressurization in order to optimize the early cooling strategy by adding fire water and avoid core melt are also studied based on the heat removal capability (HRC) of the reactor vessel coolant. This concept is instructive for the operation of the safety systems during the SBO accident mitigation.

I. INTRODUCTION

I.A. Background

The extending lifetime and uprated power of nuclear power plants have brought great challenges and uncertainties on the performance assessment and prediction during accident sequences, especially for Beyond Design Basis Accidents (BDBA). On March 2011, the seismic event and following tsunami brought an extended SBO accident to the Fukushima Daiichi nuclear power plants in Japan. In Units 1 and 3, hydrogen in the released gaseous mixture accumulated in the reactor building's upper portions, formed a combustible mixture with air and resulted in large explosions with significant damage to the reactor buildings.

Numerous assessments have been performed on how to avoid the Fukushima Daiichi severe accident. Early cooling of the reactor core is required for all water-cooled nuclear plants to avoid core melts during SBO accidents. Isolation Condensers (IC) and Reactor Core Isolation Cooling (RCIC) system are utilized to provide early backup coolant for BWRs. When all the engineered safety systems requiring alternating current electric power are incapacitated during SBO accidents, Safety Relief Valves (SRV), RCIC system and containment venting components play important roles in the accident mitigation. SRV is the major equipment to control the primary system pressure in a BWR, and RCIC system can provide makeup coolant to the reactor vessel when the main steam lines are isolated and normal supply of water to the vessel is lost. The RCIC system was among few of safety systems that still could operate during the Fukushima Daiichi accidents after the tsunami hit the plants. During the extremely extended SBO accident, RCIC system is one of the only two remaining high pressure passive systems able to provide coolant to the reactor vessel before the outside low pressure coolant supply (e.g., fire water) is ready. Actually, the RCIC system performance did contribute to a significant delay of the core uncover and meltdown in Fukushima Daiichi Unit 2 and Unit 3. (Refs. 1 and 2) Another passive coolant makeup system - the HPCI (High Pressure Cooling Injection) system, operates similarly as RCIC system; however, the operation of HPCI would rapidly depressurize the primary system due to its large steam release rate (one order higher than RCIC system), (Ref. 3) which would disable the steam-driven turbines of RCIC and HPCI systems. Therefore, the HPCI system performance and its impact on containment venting strategy will be considered in future study.

Unexpectedly, the long operation of RCIC system still did not prevent the core melt at Fukushima. In the case of Unit 2 and 3, not enough priority was given to assure early core cooling by initiating primary system venting and adding fire water. RCIC system was operated to run almost 20 hours at Unit 3, though meantime, preparation to implement reactor vessel depressurization and fire water addition was disrupted by hydrogen explosion as well as other unforeseen situations. (Ref. 4) Therefore, it was proposed that, no matter whether these backup coolant providers are working or not, provisions like fire water should be initiated to the reactor pressure vessels (RPV) to insure that (1) the reactor core water level is sufficiently high enough to prevent cladding metal-steam chemical reaction producing hydrogen; (2) the containment pressure keeps low enough for primary system depressurization and avoiding radioactivity leakage to the environment.

Thus, the main objective of this study is to analyze the BWR Mark-I SBO accident progression with respect to the effectiveness of containment venting within accident mitigation strategy, and how to avoid core melt by choosing a proper time to implement primary system depressurization and inject fire water before RCIC system loses its capability to maintain the core water level.

I.B. Failure Modes of RCIC System

RCIC system includes a steam-driven turbine, a turbine-driven pump, piping and valves necessary to release high pressure steam into the suppression pool and deliver water to the reactor vessel at accident conditions, as shown in Figure 1.

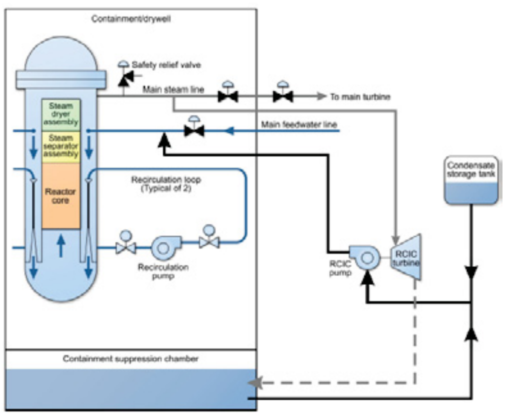


Fig. 1. Schematic of typical RCIC system. (Ref. 5)

Many factors result in the failure of RCIC system. Firstly, the steam supply system to the RCIC turbine is automatically isolated upon the low primary system pressure, which subsequently disables the steam-driven turbine-pump unit. Although RCIC system operates independently of AC power, only electrical power for the Motor Operated Valves (MOV) is provided by onsite batteries, the exhausted battery energy and mechanical breakdown of the MOVs on the turbine-pump lines may also disable the RCIC system. (Ref. 6) Besides, the RCIC pump may also automatically shut down due to the cavitation. Cavitation is the formation of vapor bubbles in a flowing fluid due to decrease in the local static pressure below the vapor pressure of the pumped liquid. The formation of this vapor and following rapid condensation can produce damage and adversely affect the operation of a centrifugal pump. (Ref. 7)

Net Positive Suction Head (NPSH), which is widely used to describe the concept of cavitation, represents total (or stagnation) pressure at the pump suction point relative to the vapor pressure of the liquid. (Ref. 7) The NPSH margin is defined as the difference between Available NPSH (NPSHA) and Required NPSH (NPSHR).

$$\text{NPSH Margin} = \text{NPSHA} - \text{NPSHR} \tag{1}$$

The NPSH margin should be no less than zero, otherwise, noise, cavitation erosion of pumps parts (mainly the impeller) and pump performance degradation will occur. NPSHR is a property of the pump itself, it corresponds to an acceptable level of pump cavitation, here we choose a conservative value as 8.114 m (26.62 ft). (Ref. 7) The available NPSH (NPSHA) is the head difference available at the pump suction, which can be determined by the following equation,

$$\begin{aligned} \text{NPSHA} &= H_{suf} - H_{v,suc} + \Delta H - H_{loss} \\ &= \frac{P_{suf}}{\rho_{avg} g} - \frac{P_{v,suc}}{\rho_{l,suc} g} + \Delta H - H_{loss} \end{aligned} \tag{2}$$

Where, H_{suf} represents the liquid surface head resulting from the pressure in the air space above (P_{suf}); $H_{v,suc}$ represents the vapor head determined by the saturated pressure of the water temperature at the suction point ($P_{v,suc}$); ρ_{avg} is the averaging density of liquid; $\rho_{l,suc}$ is the liquid density at the suction point; ΔH is the elevation difference between the liquid surface in the pool and the pump suction flange, (m); H_{loss} is the pressure loss in the flow path from fluid suction point to pump suction flange, (m).

P_{suf} consists of noncondensible gas pressure P_{nc} and the vapor pressure determined by the pool surface $P_{v,surf}$; while $P_{v,suc}$ is a function of liquid temperature at the suction point. When wetwell is simulated using a lumped parameter model, then $P_{v,surf} = P_{v,suc}$ and $\rho_{avg} = \rho_{l,suc}$. And Equation (2) could be re-written as,

$$\text{NPSHA} = \frac{P_{nc}}{\rho_l \cdot g} + \Delta H - H_{loss} \tag{3}$$

If strong thermal stratification exists in the suppression pool, the surface temperature could be much higher than the liquid temperature at the suction point, which greatly enlarges the difference between $P_{v,surf}$ and $P_{v,suc}$ and results in a potential large margin for NPSHA. The demonstration that about 30% of margin increase could be obtained when thermal stratification is considered implies that detailed thermal stratification/mixing study should be performed in the BWR Mark I containment SBO analysis in order to accurately simulate the scenarios. (Ref. 8) Thermal stratification study of the suppression pool is left for future work.

I.C. GOTHIC Approach

The study employs GOTHIC, a coarse-mesh CFD-like thermal-hydraulics simulation code that has been developed and used in containment process modeling and

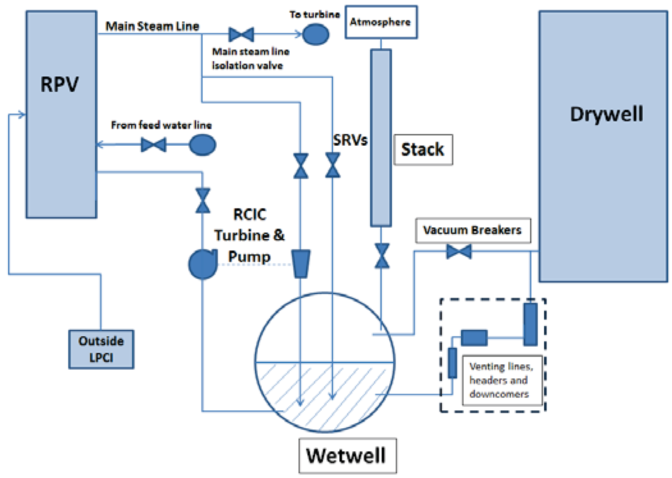


Fig. 2. Schematics of a BWR plant system GOTHIC model for SBO simulations. (Ref. 10)

analysis. (Ref. 9) In the previous study, a demonstration GOTHIC model has been developed for BWR Mark I containment and successfully applied to investigate the performance of reactor safety system and containment venting processes during SBO accident scenario. (Ref. 10) GOTHIC has the capability to simulate the dynamical performance of reactor systems needed for analysis of containment venting strategy during SBO scenarios. It is instructive to note that analysis of SBO scenarios within the context of Risk-Informed Safety Margin Characterization (RISMC) requires consideration of consequences associated with hydrogen combustion and fission product transport, and GOTHIC includes models for both these phenomena. Fukushima Daiichi Unit 1 power plant containment analysis has been performed using GOTHIC, which investigated various aspects of the Fukushima Daiichi event with consideration of the effects of multidimensional modelling and vent heat transfer on the 1F1 event simulation. (Ref. 11)

In this paper, section II describes the GOTHIC model developed for the characterization of reactor safety systems performance and evaluation of the venting strategy. The demonstration SBO scenario is proposed and analyzed in sections III, where the optimized timing to implement primary system depressurization and fire water addition is also discussed.

II. DESCRIPTION OF A SIMPLIFIED BOILING WATER REACTOR PLANT SYSTEM

A BWR Mark I plant system model has been developed using GOTHIC, which includes the major components for the primary system and the safety system, including detailed reactor vessel, RCIC system, SRVs, wetwell, drywell and containment venting components, as shown in Figure 2. The reference design for this model is derived from the Peach Bottom Unit 2, a General Electric-designed BWR-4 Mark I plant, with a rated thermal power of 3293 MW. (Ref. 12)

II.A. Reactor Pressure Vessel (RPV) Model

Figure 3 displays the detailed RPV model developed as part of the BWR plant primary system during steady state. Jet pump is disabled when SBO starts. The main steam line to turbine and main feed water line are isolated following the reactor is shutdown. The make-up coolant is injected into RPV downcomer by RCIC system and flows from lower plenum, through core to the upper plenum. In the meantime, steam is generated in the core region due to decay heat. Two phase coolant flows upwards to separator/dryer, where the liquid mass is separated from vapor and drained downwards to the downcomer while vapor flows into steam dome and then the steam line.

In this simplified BWR model for SBO scenario, the RPV model consists of steam dome, downcomer, lower plenum, core, upper plenum and separator/dryer, as shown in Figure 3. Reactor core region has been simulated using a 6-level 1-D model, six cells have different decay power densities. The thermal conductor model is applied to simulate the core decay heat using a tabulated function available in GOTHIC.

II.B. RCIC Turbine-Pump Unit Model

During the accident, RCIC turbine-pump unit works as the “steam and coolant transporter”. For the simplified SBO simulations, it’s assumed that pure steam flows through the discharging pipelines to the suppression pool and pure water is drawn from the suppression pool into reactor vessel. Turbine and pump are simulated by a fan model and a pump model built in GOTHIC respectively,

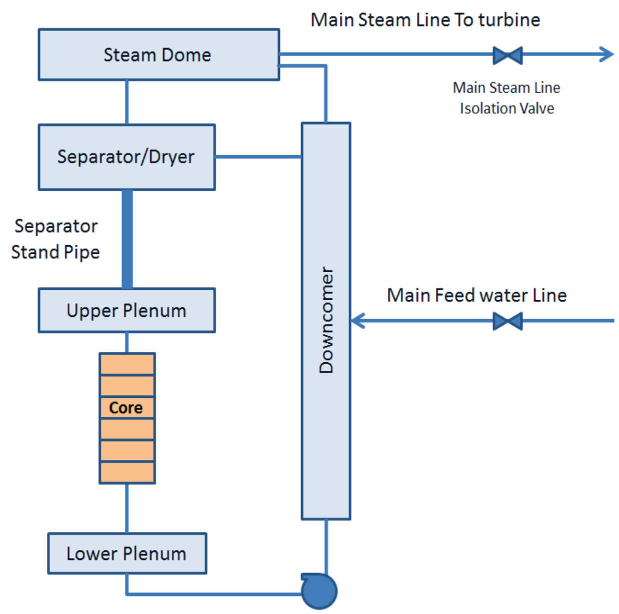


Fig. 3. RPV model during steady state developed using GOTHIC.

it is closed. The operation of MOV depends on the liquid level at RPV downcomer, which opens once it goes down to 9 m, and closes if it exceeds 11 m.

II.C. SRVs

The SRVs and the Automatic Depressurization System (ADS) play an important role in the sequence of events of the SBO scenarios. The SRVs are used to release high-temperature steam into suppression pool to decrease the primary system pressure. Each SRV is located on a steam header attached to the main steam lines leaving the reactor vessel. The Peach Bottom Plant has a total of eleven SRVs and two spring safety valves. The eleven SRVs have different opening and closing pressures; they open/close automatically when the opening/closing pressure in the primary system is reached. (Ref. 12) Data for the SRVs are displayed in Table I. The SRV that opens at the lowest pressure opens at 7.687 MPa and closes at 7.253 MPa. Normally, the steam flow through this SRV is sufficient enough to maintain the reactor vessel pressure in an acceptable range. The same SRV continuously cycles with all the steam releasing to the same location in the suppression pool, which leads to inhomogeneous heating of water in the wetwell. Worse, the risk of this SRV sticking open or close increases. In order to avoid these problems, manual operation of the SRVs is mandated in the Peach Bottom plant-specific operating procedures. (Ref. 12) Manual

with constant mass flow rates (4 kg/s for turbine and 40 kg/s for pump) when the MOV is open, and no flow when

operation of five SRVs available for ADS is applied for the primary system depressurization to an acceptable value for the fire water addition before the RCIC system fails.

II.D. Containment Components

II.D.1. Primary Containment System

The Peach Bottom Atomic Power Station Unit 2 is a BWR-4 with a Mark I containment design, referred to as an inverted light bulb (drywell) and torus (wetwell). The design pressure of drywell and wetwell is 0.485 MPa. (Ref. 12) Drywell is modeled using a lumped volume for the SBO accident analysis. The wetwell consists of gas space and suppression pool, which is the primary coolant source for the RCIC pump line and the main heat sink to condensate the steam from RCIC turbine/SRVs lines. Wetwell is modeled by a subdivided volume with a pseudo-torus geometry using GOTHIC. One 0-D model is applied for the base case study in this work.

A one-way vacuum breaker is used to relieve the negative pressure difference between drywell and wetwell. It will open when the wetwell pressure exceeds the drywell pressure by 3.45 kPa and close when the pressure difference goes below 3.45 kPa. Three other lumped volumes are created as venting lines, venting header and venting downcomers in wetwell, as illustrated in Figure 2. The heat transfer between wetwell and drywell venting system is also modeled using three thermal conductors.

II.D.2. Containment Venting Strategy

Previous analyses indicate that hydrogen explosions during the Fukushima Daiichi accident are caused by an unintended leak of the steam, non-condensable gases and hydrogen mixture from the containment into the reactor building. (Ref. 1) When the pressure from the reactor vessel was relieved into the wetwell, the temperature of the water and wetwell pressure started to increase. However, there was no way to cool the suppression pool without electrical power. In order to keep the integrity of primary containment, the decision was made to initiate the containment venting. Typically, the venting would perform through the ventilation system to the standby gas treatment system and out from the site stack. Actually, the gaseous mixture with non-condensable gases and hydrogen escaped into reactor building, a secondary containment structure. In unit 1 and 3, hydrogen accumulated in the upper portions of the reactor building, mixed with air and formed a combustible gaseous mixture that resulted in large explosions. In unit 2, it appears that hydrogen accumulated in the reactor building closer to the suppression pool and the following explosion caused damage to the reactor building. Containment is the last defense-in-depth barrier for fission products. Containment venting during accidents could lead to release of fission products to the environment. However, high temperature and subsequent high pressure in the containment threatens the containment integrity, with potential for uncontrolled leakage of gas mixture.

TABLE I. The Peach Bottom Safety Valve Characteristics

Valve NO.	Opening Pressure (MPa)	Closing Pressure (MPa)	Rated Flow (kg/s)	Rated Pressure (MPa)	Rated Density (kg/m^3)	ADS
1	7.687	7.253	109	7.722	40.8	Y
2	7.708	7.101	109	7.722	40.8	Y
3	7.722	7.184	109	7.722	40.8	Y
4	7.756	6.991	109	7.722	40.8	N
5	7.763	7.053	110	7.791	41.2	Y
6	7.791	7.322	110	7.791	41.2	Y
7	7.797	7.184	110	7.791	41.2	N
8	7.825	7.246	110	7.791	41.2	N
9	7.846	7.391	111	7.860	41.6	N
10	7.860	7.115	111	7.860	41.6	N
11	7.867	7.308	111	7.860	41.6	N

The decision of containment venting should reach the optimized balance between the high-temperature-and-high-pressure threat on the containment integrity and potential for uncontrolled release of radioactive gas mixture. This paper is trying to study the safe venting window for primary system depressurization and the optimized criteria of containment venting initiation, then making efforts to support the decision making on both the primary system and containment venting strategy. In the demonstration SBO scenario case implemented, a venting stack is modeled in GOTHIC to simulate venting from the wetwell to the atmosphere. The venting is assumed to start when the wetwell pressure exceeds 80% of the design pressure and stops when the pressure decreases to 70% of the design value.

III. CHARACTERIZATION OF SBO SCENARIO

III.A. Description of the Scenario

The central idea of the demonstration SBO scenario study is to explore the safety benefit obtained by the performance of the RCIC system during the SBO accident. The RCIC system is expected to keep functioning as long as possible to provide coolant for the reactor vessel.

As discussed above, the steam supply system to the RCIC turbine could be automatically isolated upon low primary system pressure. The risk of mechanical breakdown and exhaustion of battery energy for the motor operated valves is expected to rise due to frequently open and close. According to the experience of Fukushima Unit 3, the MOV battery energy is enough for RCIC system to work for 20 hours, (Ref. 13) here it is assumed that the MOV fails due to exhaustion of battery energy after 20 hours. Besides, pump impeller blades may be eroded and damaged due to cavitation. In order to avoid the pump damage due to cavitation, manual operation of SRVs will be performed when NPSH margin decreases to zero in the demonstration SBO scenario. The following primary system depressurization instantly disable the RCIC steam-driven turbine-pump unit.

The fire water is also assumed to be ready to inject water into the reactor vessel (downcomer) if the primary system pressure goes below to 414 kPa, which is a normal nozzle pressure for fire truck pumps (Ref. 14). Therefore, depressurization of reactor vessel is needed to enable fire water by manually opening the SRVs. In order to decrease the time of core uncovering, the SRVs are expected to be opened immediately when the RCIC system tends to fail (when the NPSH margin reduces to 0), representing the longest time for the normal performance of RCIC system. Peak cladding temperature (PCT) is the key parameter to determine whether the core is damaged or not in this scenario. The real limiting temperature for the fuel cladding (Zr) is 1100°C, because beyond this temperature, exothermic reaction (Zirconium oxidation) contributes to cladding degradation, and intense hydrogen production. (Ref. 15) The major events are described as below,

1. At time = 0 second, reactor is shut down and decay heat is generated;
2. At t = 1 s, the primary system starts to be isolated with the feedwater line valve closed in 1 s and main steam line isolation valves closed in 3 s;
3. When NPSH margin decreases to 0, cavitation is expected to occur. Then manually open five SRVs and depressurize the reactor vessel to 414 kPa for fire water. The steam in reactor starts to be vented into suppression pool and RCIC system fails.
4. Fire water starts and pumps water into RPV with mass flow rate as 12.6 kg/s; (Ref. 14)

5. When cladding surface temperature in any cell of core (6 cells in total) reaches 1100°C, Zr-steam reaction starts, and generates hydrogen, which results in core damage.

The geometries of components are shown in Table II. The steady state of Peach Bottom Unit 2 during normal set as the initial condition of the assumed SBO accident. (Ref. 16) Pressure, temperature and liquid fraction in each component are listed in Table III.

TABLE II. Major component parameters for the simplified BWR plant configuration

Component Name	Elevation Relative to the Vessel Bottom (m)	Free volume (m^3)	Flow Area (m^2)
Lower Plenum	0	61.48	11.64
Reactor Core	5.28	28.55	7.80
Upper Plenum	8.94	26.99	14.36
Separator Stand Pipe	10.82	10.69	3.93
Separator/Dryer	13.54	19.30	10.27
Steam Dome	15.42	178.19	26.19
Downcomer	2.10	201.30	15.00
Drywell	-8.635	4777	1571
Wetwell Air Space	-6.93 (Top)	4380	-
Wetwell Water Space	-16.38 (Bottom)	2753	-

TABLE III. The steady state during normal operation and initial condition for SBO in each component of GOTHIC model

Component	Temperature (C)	Pressure (MPa)	Liquid Fraction
Lower plenum	276	7.3084	1
Core	303	7.2326	0.6
Upper plenum	303	7.1567	0.25
Separator/Dryer	304	7.0537	0.2989
Steam dome	305	7.0330	0
Downcomer	286	7.0537	0.8942
Drywell	63	0.108	0
Wetwell	33	0.1	0.386
Venting lines	63	0.108	0
Venting header	63	0.108	0
Venting Downcomer	50	0.108	0.4643

III.C. Results and Discussions

Figure 4 represents the NPSH margin variation during the assumed SBO accident. When NPSH margin decreased to zero at nearly 3000 s, the RCIC system failed and five SRVs were manually opened to initiate the primary system depressurization and fire water injection. A sharp drop on pressure was following due to amount of steam was vented from reactor vessel into suppression pool through the manually opened SRVs. The steam was condensed by water and led to the increase of water level, liquid/gas temperature and wetwell pressure, as shown in Figure 5 and Figure 6. However, the vacuum break between wetwell and drywell also automatically opened due to the pressure difference exceeded the opening limit. Gas mixture including steam and air escaped into drywell through the vacuum breaker, which resulted in the decrease of noncondensible gas pressure and NPSHA according to Equation (3).

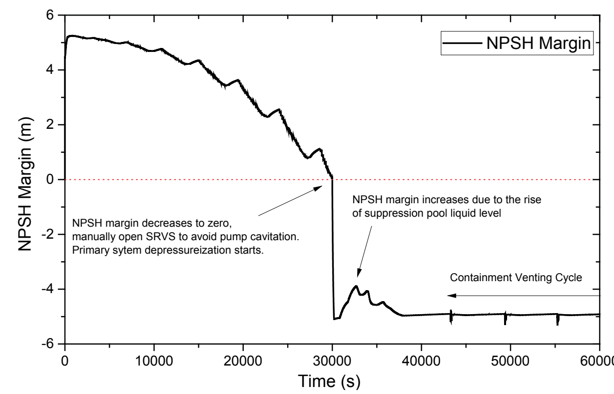
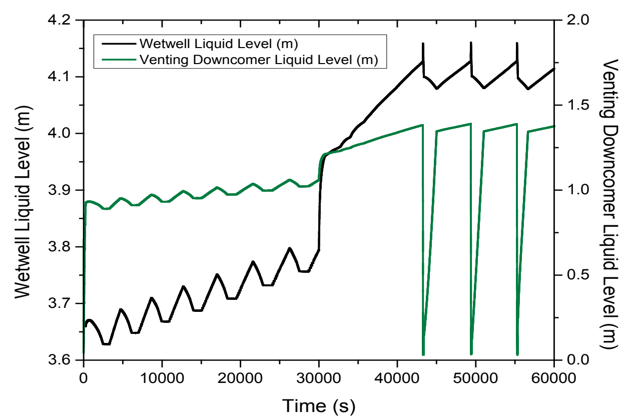
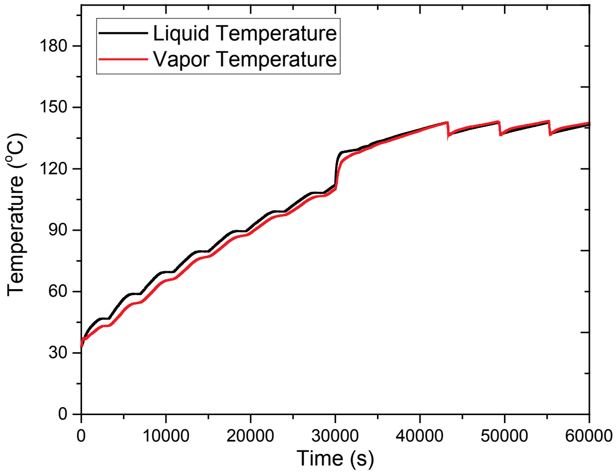


Fig. 4. NPSH margin variation during the assumed SBO accident



(a)



(b)

Fig. 5. (a) Liquid levels in wetwell and venting downcomers , (b) Liquid and gas temperature in wetwell

The subcooled coolant bulk in the suppression pool condensate amounts of the steam coming through the manually opened SRVs. This made the rise of liquid level in wetwell, and then NPSH margin increased after a short-time drop. When the wetwell pressure reached the opening point of containment venting, the gas mixture started to escape out from wetwell through stack to the environment after about 38000 s. The mass flow rate of containment venting is displayed in Figure 6. As the key parameter for venting strategy, the variation of NPSH margin provides plenty information about the venting process.

Figure 5(a) denotes the liquid level variation in the wetwell and venting downcomers during the assumed SBO accident. The rising part and falling part of previous fluctuations of wetwell liquid level before 30000 s resulted from the performance of RCIC pump and SRVs. The RCIC pump injected the water from suppression pool to reactor vessel and the SRVs vented large amount of steam into the wetwell. The later fluctuations during containment venting cycle implied that the water from venting downcomers flowed back into suppression pool due to the pressure change in wetwell and drywell. Figure 5(b) displayed the variations of liquid temperature and gas temperature. In the beginning the subcooled water obtained heat from the injected steam from the SRVs and RCIC turbine line and started boiling when saturation temperature was reached.

From Figure 6 the variations of wetwell gas space pressure and mass flow rate through containment venting could be observed. During the containment venting cycle, containment venting components show dynamic behaviors. The containment pressure was maintained in a relatively safe range (70%~80% of design pressure). The amount of vented gas mixture is an important parameter for containment venting strategy since radioactive products and explosive gases may escape with steam and air into the environment.

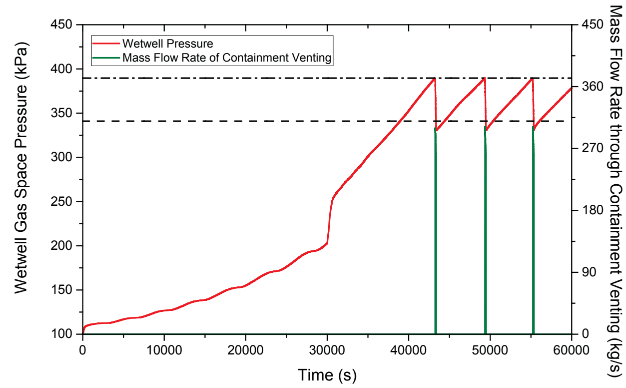
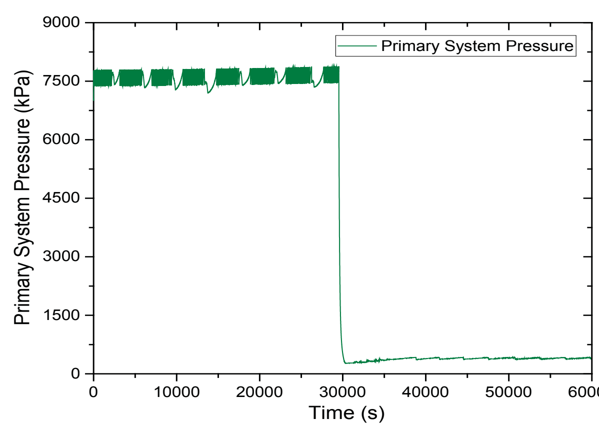
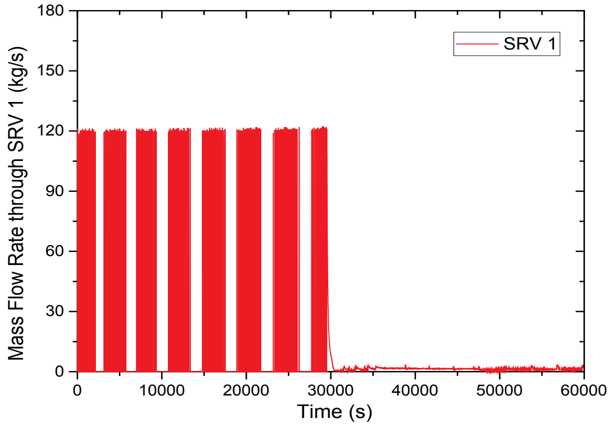


Fig. 6. Dynamic behaviors of containment venting components during the assumed SBO accident

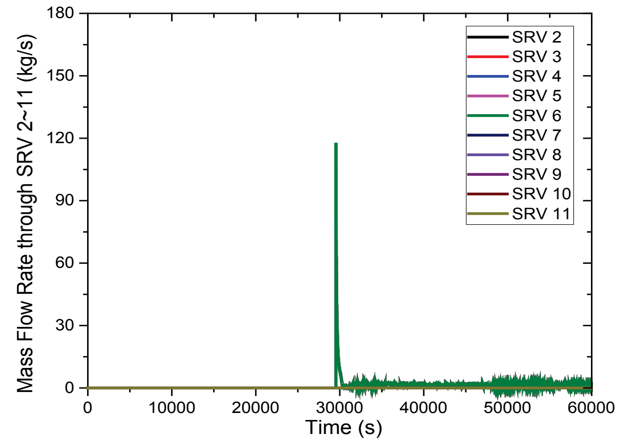
Figure 7 implies the performance of the 11 SRVs and their relation with primary system pressure. In the beginning, the primary system pressure was well controlled by the opening and closing of the SRVs. Then the RCIC system failed and 5 SRVs were manually opened at about 30000 s, the pressure suddenly dropped below the opening point of fire water, and then kept at a low stable state. Figure 7(b) and 7(c) compare the mass flow rate of each SRV. It is obvious to find that only the SRV with lowest opening pressure was working through the whole scenario. It could well maintain the vessel pressure. Another four manual-controlled SRVs only started to work after the primary system venting decision was made. The activity of the rest SRVs was not observed.



(a) Primary system pressure variance



(b) Mass flow rate variance through SRV 1



(c) Mass flow rate variance through SRV 2-11

Fig. 7. Performance of the SRVs and their relation with primary system pressure

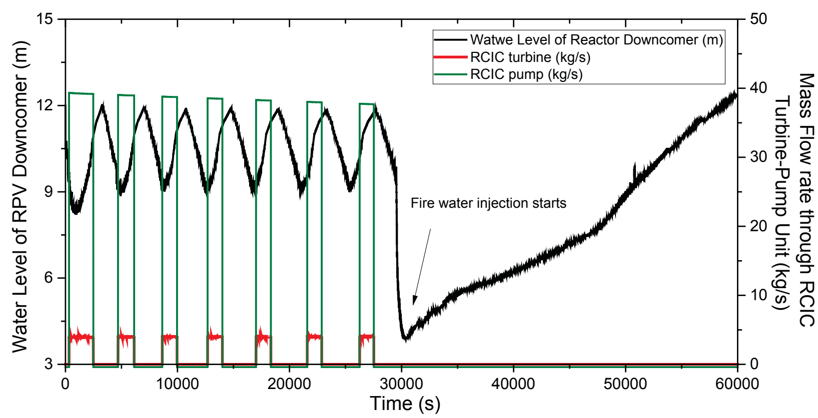


Fig. 8. Performance of RCIC turbine-pump unit and the variation of reactor downcomer liquid level

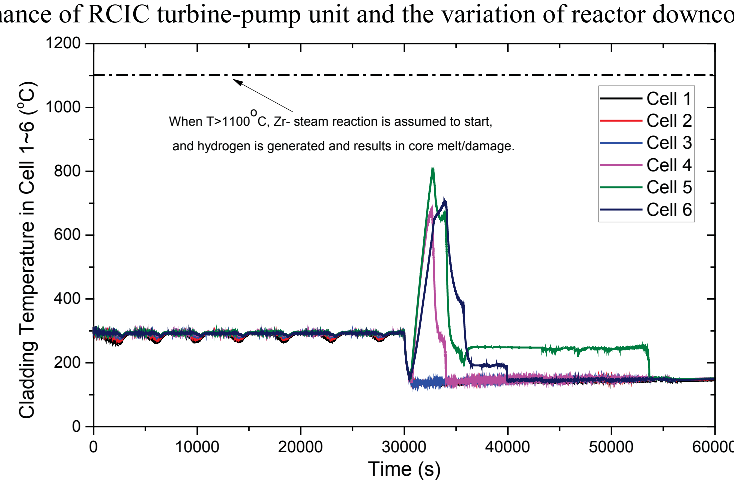


Fig. 9. Distribution and variation of cladding surface temperature during the accident

Figure 8 exhibits the performance of the RCIC turbine-pump unit during the assumed SBO accident. It shows that the RCIC system could keep the vessel liquid level in a relatively safe range in which the core could be fully covered while the RPV would not be flooded. The cladding surface temperature of the core has been shown in Figure 9. The distribution of temperature is obtained and measured using 1D core model. The cladding surface temperatures did not exceed 1100 ^{o}C, the assumed trigger point for metal-steam reaction. No hydrogen was supposed to be generated in this case. It is reasonable that the most dangerous part is located in cell 5, since cell 5 has relatively high power density and would become uncovered relatively early. It is “easier” for cell 5 to reach the dry-out condition.

The results in the above figures prove that the reactor cooling system and containment system of this GOTHIC model performed as expected during the assumed SBO accident. In this case, core was uncovered for almost 10000 seconds and almost got damage due to overheating. This means that even mitigation systems perform well, the core damage may still happen. As discussed in the beginning, NPSH margin determines when to initiate the primary system depressurization and also strongly influences when the containment venting starts. By performing the demonstration scenario, we maximized the safety benefit of RCIC system and suppression pool as the final heat sink. However, it was proved not a good choice since core experienced a long-time uncovering.

Therefore, “when is the best time to initiate the primary system depressurization and initiate fire water injection” is now of most interest and importance. The cladding surface temperature depends on how fast the coolant can remove the decay heat. The concept, Heat Removal Capability (HRC), is applied to represent the capability of liquid in the reactor vessel to remove heat by phase change from subcooled liquid to saturated steam, and could be calculated by,

$$\boldsymbol{HRC\ (J)\ = \sum_i \rho_{l,i} \cdot V_i \cdot (1 - VF_i) \cdot (h_{st,i} - h_{l,i})} \quad (4)$$

Where, i represents the number of control volume in reactor vessel, V is the total volume, VF is the void fraction, $\boldsymbol{h_{st}}$ is the enthalpy of saturated steam and $\boldsymbol{h_l}$ is the enthalpy of the subcooled liquid.

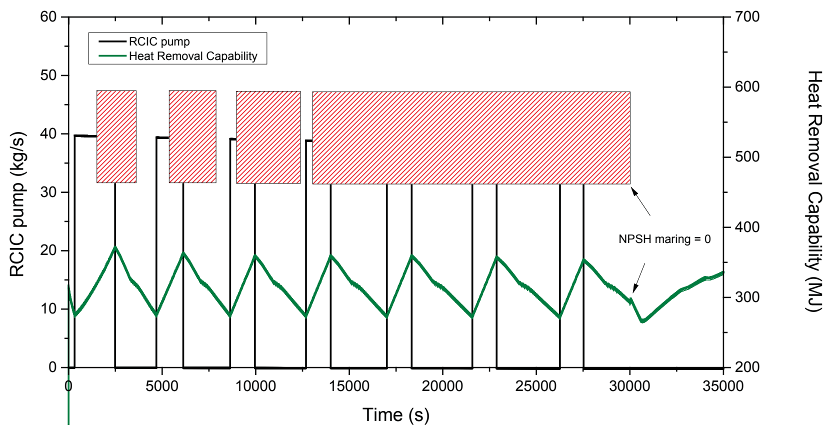


Fig. 10. Variation of heat removal capability and the “safe venting windows” to initiate primary system depressurization

The variation of HRC is displayed in Figure 10. HRC exists at a relatively low value when NPSH margin=0. The heat removal condition of the core is strongly influenced by decay heat and HRC. Therefore, we tested the cases in which primary system venting is initiated at the end of every RCIC pump working cycle, where HRC has the highest value, all the results show that the peak cladding temperature (PCT) is much lower than the damage point. It means primary system venting decision is safe to be made at these time points.

Then the “safe venting window” in each working cycle could be quantified by calculations. A prediction that the “safe venting window” in later cycle should be wider than the former ones since the decay heat is decreasing is proved by result shown in Figure 10 and Table IV. Once the objective nuclear power plant (NPP) is determined, these safe venting windows could recommend on the operation of primary system venting if extended SBO happens. Even if the monitor for reactor vessel liquid level fails, the best time for venting operation could also be executed based on these safe venting windows.

TABLE IV. The ranges of the safe venting windows for primary system depressurization and fire water injection

Window NO.	Starting Time (s)	Ending Time (s)
1	1900	3600
2	5300	7850
3	8850	12650
4	12800	29995 *

*: NPSH margin decreased to 0 at t = 29995 s in the demonstration SBO scenario.

IV. CONCLUSIONS AND DISCUSSIONS

In this paper, a GOTHIC model has been developed to support characterization of reactor safety systems performance and evaluation of the venting strategy during SBO scenario in BWR Mark I plant. The safe time intervals ("safe venting window") to initiate the primary system depressurization in order to optimize the early cooling strategy by adding fire water and avoid core melt are studied based on the heat removal capability (HRC) of the reactor vessel coolant. This concept is instructive for the operation of the safety systems during the SBO accident mitigation. Insights from the simulation and analysis are discussed, forming the technical basis to extend the current model for simulation of scenarios involving hydrogen transport and radioactive isotopes release through containment venting strategy in the near future. The development created a confidence in the technical basis needed to develop a framework of robustness assessment and improvement that enables the massive simulation of GOTHIC for the application of the Risk Informed Safety Margin Characterization (RISMC) methodology on the BWR Mark I plant Station Black-Out (SBO) accident analysis.

In this study, RPV depressurization is assumed to be initiated if the makeup water is lost when RCIC system fails due to the factors mentioned above. However, the factors that may initiate the RPV depressurization according to Emergency Operating Procedures (EOPs) are:

1. Heat Capability Temperature Limit (HCTL) exceeds the limit value;
2. Pressure Suppression Pressure (PSP) exceeds limit value;
3. Drywell temperature exceeds limit value;
4. Suppression pool water level is too low;
5. Makeup water (from RCIC, e.g.) is lost and water injection is needed to keep core covered. [17]

where (5) is the only considered "trigger" in this study. However, One could argue (cited as personal communication) that (1) may occurred, because the water in suppression pool is close to saturation before RCIC system fails (pump cavitation). And the result of the demonstration case shows that liquid temperature in suppression pool is close to saturation temperature. So RPV depressurization may occur earlier than the simulation in the demonstration case if following EOPs.

However, EOPs improvement deserves much more attention since existing EOPs are not adequate enough for extended SBO accident. "The traditional system of Emergency Operating Procedures (EOPs) and Severe Accident Management Guidelines (SAMG) failed to protect core and containment, and severe core damage resulted, followed by devastating hydrogen explosions and, finally, considerable radioactive releases." [18] If intentionally not follow EOPs (HCTL), or unintentionally not follow due to the loss of DC power for control signals, RPV depressurization will be initiated by the failure of RCIC system, later than EOPs requirement, like the demonstration case. The benefit is that SP could be used as heat sink longer and more time could be saved to prepare fire water injection. However, the risk is that containment may fail due to high temperature and tiny leakage may occur in the containment wall. Besides, researchers from Taiwan also proposed Ultimate Response Guidelines (URGs) for compound BDBAs. Once any designed power and water is made available, reactor depressurization, raw water injection will be activated simultaneously by any available power supply, earlier than EOPs requirement. [19]

Therefore, it is necessary to investigate these risks and explore the benefits from not following existing EOPs for the strategy of RPV depressurization (e.g., HCTL) during accident. Comprehensive study should be performed based on the assessment of probability and consequences of alternative scenarios which do not follow EOPs. This can be the future work.

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