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DEMONSTRATION OF FULLY COUPLED SIMPLIFIED EXTENDED STATION BLACK-OUT ACCIDENT SIMULATION WITH RELAP-7

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ABSTRACT

The RELAP-7 code is the next generation nuclear reactor system safety analysis code being developed at the Idaho National Laboratory (INL). A number of physical components with simplified two phase flow capability have been developed to support the simplified boiling water reactor (BWR) extended station blackout (SBO) analyses. The demonstration case includes the major components for the primary system of a BWR, as well as the safety system components for the safety relief valve (SRV), the reactor core isolation cooling (RCIC) system, and the wet well. Three scenarios for the SBO simulations have been considered. Since RELAP-7 is not a severe accident analysis code, the simulation stops when fuel clad temperature reaches damage point. Scenario I represents an extreme station blackout accident without any external cooling and cooling water injection. The system pressure is controlled by automatically releasing steam through SRVs. Scenario II includes the RCIC system but without SRV. The RCIC system is fully coupled with the reactor primary system and all the major components are dynamically simulated. The third scenario includes both the RCIC system and the SRV to provide a more realistic simulation. This paper will describe the major models and discuss the results for the three scenarios. The RELAP-7 simulations for the three simplified SBO scenarios show the importance of dynamically simulating the SRVs, the RCIC system, and the wet well system to the reactor safety during extended SBO accidents.

Key Words: **RELAP-7, BWR, Station Blackout, Reactor Core Isolation Cooling.**

1. INTRODUCTION

The RELAP-7 code [1] is the next generation of nuclear reactor system safety analysis code being developed at the Idaho National Laboratory (INL). The RELAP-7 code development is taking advantage of the progress made in the past several decades to achieve simultaneous advancement of physical models, numerical methods, and software design. The RELAP-7 code utilizes the INL's MOOSE (Multi-Physics Object-Oriented Simulation Environment) framework [2] for solving computational engineering problems in a well planned, managed, and coordinated way. The MOOSE framework integrates all the physics into a single fully coupled nonlinear equation system. The nonlinear equation system is solved with Newton's method such as Jacobian free Newton Krylov (JFNK) method. Unlike traditional system codes, all the physics in RELAP-7 are fully coupled and the errors resulted from the traditional operator splitting approach are eliminated.

This allows RELAP-7 development to focus on systems analysis-type physical modeling and gives priority to retention and extension of RELAP5's system analysis capabilities.

The RELAP-7 code development project started in October of 2011. During the first year of the development, the basic software framework, as well as the basic reactor system simulation capabilities for single-phase flow, were established [1]. During the second development year, basic two-phase flow modeling capabilities were implemented in the RELAP-7 code, aimed at demonstrating simulation of a BWR with simplified geometries under extended station blackout (SBO) transient conditions [3].

Reactor systems are very complex and contain hundreds of different physical components. Therefore, it is impractical to resolve the geometry of the whole system. Instead, simplified thermal hydraulic models are used to represent the major physical components and describe major physical processes such as fluid flow and heat transfer. There are three main types of components developed in RELAP-7: one-dimensional (1-D) components, zero-dimensional (0-D) components for setting boundary conditions (BC) for the 1-D components, and 0-D components for connecting 1-D components. 1-D components, such as pipe, heat exchanger, and core channel, describe 1-D fluid flow model and additional heat conduction model. Zou et al. [4] described the single-phase fluid flow model and several 1-D component models developed in RELAP-7. For 1-D single-phase fluid subsonic flow, two boundary conditions such as momentum and temperature or pressure and temperature are needed for the in-flow boundary and one boundary condition such as pressure is needed for the out-flow boundary. 0-D components will provide those boundary conditions for 1-D fluid flow models. Zhao et al. [5] described related 0-D components for simulating the RCIC system and wet well in RELAP-7.

For extended SBO accidents, all the engineered safety systems requiring alternating current electric power are lost. Only those passive components or components only depending on battery power are still available. Among those available systems, safety relief valves (SRV), the reactor core isolation cooling (RCIC) system, the high pressure coolant injection (HPCI) system, and the isolation condenser (IC) system are the most important in term of safety. The SRV is the major equipment to control the primary system pressure in a BWR reactor [6]. High pressure steam is released through SRVs to the suppression pool when the system pressure is higher than the setting point. The RCIC system provides makeup water to the reactor vessel for core cooling when the main steam lines are isolated and the normal supply of water to the reactor vessel is lost. The RCIC system operates independently of AC power, service air, or external cooling water systems. The only required external energy source is from the battery to control the system. It was one of the very few safety systems still available during the Fukushima Daiichi accidents after the tsunami hit the plants and the system successfully delayed the core meltdown for a few days for unit 2 & 3 [7]. Therefore, detailed models for RCIC system components are indispensable to understand extended SBO for BWRs. The HPCI system is also driven by high pressure steam and can be simulated with similar models as for the RCIC system. The HPCI system has one order higher mass flow rate than the design value for the RCIC system. However, the HPCI system is only actuated at much lower water level if the control is still available. The isolation condenser system is composed of a natural circulation loop in which high temperature primary steam flows to the condenser immersed in a water tank, condenses back to water, and flows back to the reactor core. The IC capability is determined by the IC tank water volume, which is typically much smaller than the water volume of the suppression pool. In this paper, both HPCI system and IC system are not simulated.

The RCIC system, as shown in Figure 1 [8], consists of a turbine and a turbine-driven pump,

pipings and valves necessary to deliver water to the reactor vessel at accident conditions. The turbine is driven by high-temperature and high-pressure steam and is designed to rapidly accelerate from standby to the full load condition within a pre-prescribed time period. The turbine exhaust steam is routed to the suppression pool. The turbine-driven pump supplies makeup water from the condensate storage tank or the suppression pool, to the reactor vessel via the feedwater piping. The wet well of a BWR containment consists of the suppression pool and the gas space above it. The suppression pool is the alternate source of water for the RCIC pump and it condenses steam from the turbine exhaust or from safety relief valves.

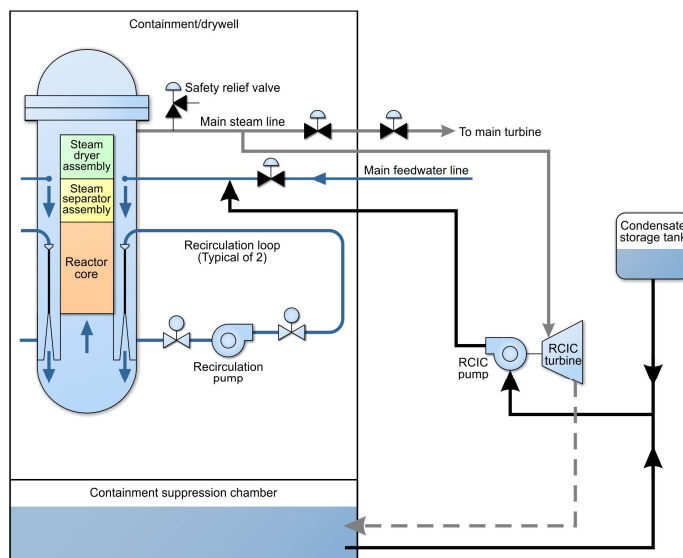


Figure 1. Reactor Core Isolation Cooling (RCIC) system (Credit of U.S. NRC [8]).

Many researchers have used different codes to study SBO scenarios for different reactors. U.S. NRC sponsored SOARCA (State-of-the-Art Reactor Consequence Analysis) project used the MELCOR code models to perform a realistic evaluation of SBO accident progression, source term, and offsite consequences for the Peach Bottom Nuclear Power Station [9]. Park and Ahn [10] compared SBO accident progressions for typical pressurized water reactor (PWR), BWR, and pressurized heavy water reactor (PHWR), based on the analysis of the MAAP code for PWR/BWR and ISAAC simulation for the PHWR. Chen et al. [11] analyzed the LOCA (Loss of Coolant Accident) combined SBO accident using the TRACE code and evaluated alternate mitigation strategy using the turbine driven pumps and residual steam for the emergency operational procedures (EOPs) and the severe accident mitigation guidelines (SAMGs). Ortiz-Villafuerte et al. [12] simulated a severe accident scenario for a BWR using RELAP5/SCDAPSIM Mod 3.4, with an emphasis on the behavior of the pressure-suppression pool during the event. A simplified model of a wet well and dry well was added to the primary loop model of a BWR to determine if conditions for containment venting are reached. Hirano et al. [7] presented a TRAC-BF1 code model for long-term station blackout accident analysis at Fukushima Daiichi unit 2 reactor. The authors also used a simplified stand-alone model, CVBAL, specially developed for estimation of accident conditions inside the pressure containment vessel from observed pressure and temperature trends based on rough assumptions.

In spite of all the efforts, there are still areas needing further improvement in order to better understand SBO phenomena and improve severe accident mitigation strategies. One example is the RCIC system behavior during Fukushima Daiichi accidents. It is generally assumed [9] that loss of

the DC power would result in overfilling the steam line and allowing liquid water to flow into the RCIC turbine, where it is assumed that the turbine would then be disabled. This behavior, however, was not observed in the Fukushima Daiichi accidents after the tsunami hit the plants, where the Unit 2 RCIC pump functioned without DC power for nearly three days. Moreover, all these analyses often replace dynamical physics with given boundary conditions, which are based on limited knowledge and assumptions. For example, in the TRAC model for Fukushima accidents [7], the RCIC was actuated under the assumption that the steam flow rate to the RCIC turbine and the injection flow rate from the RCIC pump were both balanced with the reactor decay heat power. In MAPP models [10], the flow rate through RCIC system is through user input tables. As the next generation system analysis code, RELAP-7 code tries to include all the dynamical models for key physical components and solves all the physics in strongly coupled way. Because of short development time, RELAP-7 is still under rapid development and cannot resolve all the Fukushima thermal hydraulics issues. The simulation cases presented in this paper are for demonstration purposes. However, the successful demonstration of the simplified SBO simulations with RELAP-7 shows the potential to further improve the models for safety analysis by including more detailed physical processes in the near future.

2. DESCRIPTION OF RELAP-7 PHYSICAL COMPONENTS

As discussed in the previous section, physical components are basic units in the RELAP-7. Table 1 lists the major components developed to demonstrate BWR SBO transient analysis. It is noted that the two-phase flow model used in this paper is the homogeneous equilibrium model that is a reduced subset of the seven-equation model. The full seven-equation, two-phase model has been implemented into RELAP-7 and the results have been demonstrated with a few components. However, it needs further development to be able to perform BWR transient simulations for the complete component set. The detailed model descriptions for physical components can be found in references [1].

Table 1. Major components developed to perform BWR SBO analysis

Component Name	Descriptions	Dimension
Pipe	1-D fluid flow within 1-D solid structure with wall friction and heat transfer	1-D
CoreChannel	Simulating flow channel and fuel rod thermal hydraulics, including 1-D fluid flow and fuel rod heat conduction for either plate type or cylinder type of fuel	1-D
TimeDependentVolume	Time Dependent Volume to set pressure and temperature boundary conditions as constants or time functions, no unknowns	0-D
TimeDependentJunction	Time Dependent Junction to set velocity and temperature boundary conditions as constants or time functions, no unknowns	0-D
VolumeBranch	Multiple in and out 0-D junction with volume, which provides form loss coefficients (K) at each connection	0-D
Pump	Simple pump model to provide a head and reverse flow form loss coefficients (K), for either isothermal flow and non-isothermal flow; pump can be driven by user input head or through a driving component which provides shaft work	0-D

IdealPump	A one-in and one-out junction model with given mass flow rate to simulate an ideal pump behavior	0-D
SeparatorDryer	Separating steam and water with mechanical methods, 1 in and 2 outs	0-D volume
Downcomer	Large volume to mix different streams of water and steam and to track the water level	0-D
Valve	Simulate valve open and close behavior for incompressible flow with user given trigger time and response time; Abrupt area change model is used to calculate form loss coefficient	0-D
CompressibleValve	Simulate valve open and close behavior for compressible flow, including choking; can be used as SRV	0-D
Turbine	A simplified dynamical turbine model to simulate a gas/steam turbine	0-D
WetWell	0-D volume to simulate suppression pool and its gas space	0-D
Reactor	A virtual component that allows users to input time dependent thermal power for CoreChannel model	0-D

3. DESCRIPTION OF A SIMPLIFIED BWR SBO SIMULATION MODEL

A simplified BWR plant system model has been built for the demonstration SBO simulations, based on the parameters specified in the Organization for Economic Cooperation and Development (OECD) turbine trip benchmark problem [14]. The reference design for the OECD BWR Turbine Trip benchmark problem is derived from the Peach Bottom-2, which is a General Electric-designed BWR-4 nuclear power plant, with a rated thermal power of 3293 MW.

Figure 2 shows the schematic of the simplified BWR plant system model for RELAP-7 simulations. The reactor vessel model consists of the downcomer model, the lower plenum model, the reactor core model, the upper plenum model, the separator/dryer model, the steam dome model, the main steam line model, the feedwater line model, the primary pump model, the SRV model, the RCIC turbine model, the RCIC pump model, and the wet well model. A core channel model was used to describe the reactor core. Each core channel represents thousands of real cooling channels and fuel rods. For the sake of simplicity, only one core channel was used to represent the entire core and bypass flow was ignored. The lower plenum, upper plenum and steam dome are modeled with volume branch models. External to the reactor vessel, the main steam line is connected to the steam dome. A time dependent volume is attached to the main steam line to provide the necessary boundary conditions for the steam flow. A feedwater line is connected to the downcomer component. A time dependent volume is attached to the feedwater line to provide the necessary boundary conditions for the feedwater. The safety system includes SRVs, the RCIC turbine and RCIC pump, the containment wet well and dry well. Valves are placed at various locations to provide the flow control functions of the plant system. Notably missing from this simplified BWR model are the jet pumps and the recirculation loops that allow the operator to vary coolant flow through the core and change reactor power. Instead, for this case study, a pump model is used to represent the functions of the jet pumps and the recirculation loops.

The following provides more detailed information on the geometry and parameters used for the model. The Peach Bottom-2 reactor core consists of 764 fuel assemblies. The initial cycle was selected as the reference design cycle for the simulations done in this paper. In the initial cycle, 7×7 fuel rod lattice type assemblies with no water rods were loaded. The active core height spec-

ified is 3.66 m. The fuel assembly and fuel rod geometry data were taken from reference [14]. The major parameters required to build the simplified BWR plant system configurations also were obtained from reference [14] and some key data are shown in Table 2.

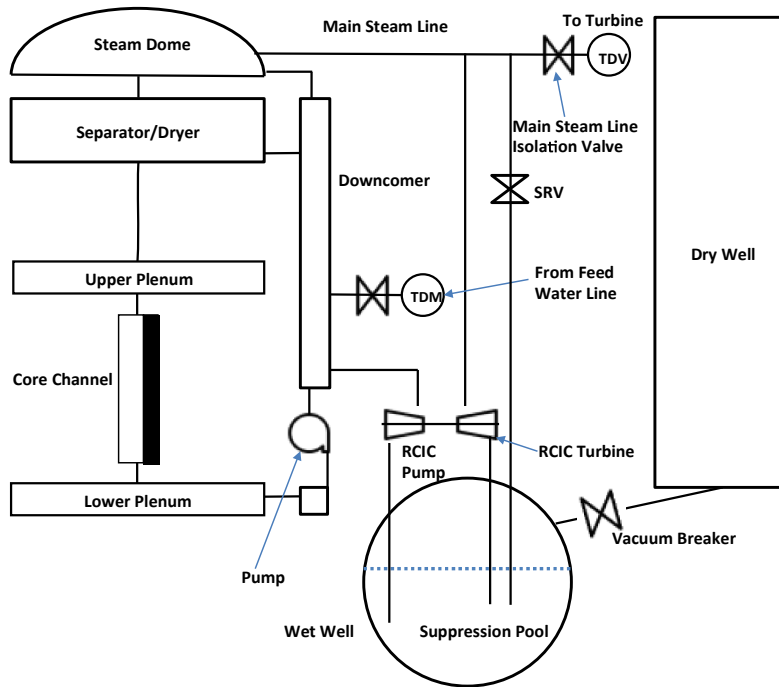


Figure 2. Schematics of a simplified boiling water reactor plant system model.

Table 2. Major component parameters for the simplified BWR plant configuration.

Component Name	Volume (m ³)	Area (m ²)	Axial Elevation (Top) Relative to the Bottom of the Vessel (m)
Lower Plenum	61.48	11.64	5.28
Reactor Core	28.55	7.80	8.94
Upper Plenum	26.99	14.36	10.82
Separator Stand Pipe	10.69	3.93	13.54
Separator Dryer (S/D)	19.30	10.27	15.42
Steam Dome	178.19	26.19	22.32
Downcomer	201.30	15.0	15.52
Wet Well Water Space	3570	892.5	-12 (bottom)
Wet Well Gas Space	3570	892.5	-4 (top)
RCIC Turbine/Pump	N/A	N/A	-3

4. SBO SIMULATION SCENARIOS AND RESULTS

Three scenarios of RELAP-7 SBO accident simulations were performed on the simplified BWR system model described in the previous section. Scenario I represents an extreme station blackout accident without any external cooling and cooling water injection. The system pressure is controlled by automatically releasing steam through SRVs. Scenario II includes the RCIC system but

without the SRVs. The RCIC system is fully coupled with the reactor primary system and all the major components are dynamically simulated. The third scenario includes both the RCIC system and the SRVs to provide a more realistic simulation.

The RELAP-7 input files were built for all three scenarios and all cases were first run such that the plant system reaches steady state with a rated thermal power of 3293 MW. Subsequently, the cases continuously run to perform the transient simulations of the SBO scenarios. The control is realized through RAVEN [15], the software package providing graphic user interface, control, uncertainty qualification, and probability risk analysis capabilities for RELAP-7. Reactor scram was assumed to occur at SBO initiation. Therefore, the heating source comes from the decay heat of the fuel in the reactor core. The sinusoidal power density distribution in the axial direction was used in both the steady state and SBO transient simulations.

4.1. Simulation Results for Station Blackout Scenario I – Without Safety Injection

In this scenario, it is assumed that all the safety coolant injection systems fail to start and function properly when SBO occurs. Consequently, only the SRVs automatically open and close periodically to control the primary system pressure and discharge the high temperature and high pressure steam into the suppression pool. Although highly simplified, this case is quite similar to what happened in Fukushima Daiichi Unit 1. Unit 1 had no RCIC system, while the isolation condenser system was believed either not functioning or only available for a very short period of time during the accident. Therefore, the Unit 1 reactor core reached a fuel failure temperature only a few hours into the accident.

Figure 3 shows the evolution of the downcomer water level during the scenario I SBO simulation. The downcomer water level indicates the water inventory within the reactor vessel. The relative height for the reactor core top fuel to the bottom of downcomer is 6.84 m. Without makeup water from the safety injection system, the water level in the downcomer gradually decreases as the SBO accident progresses. The decreasing water level results in a less driving head to drive the coolant through the reactor core and to transport the heat out of the reactor core. Consequently, the natural circulation capability is degraded. The oscillations of the water level are due to the pressure oscillations as shown in Figure 4, which are caused by the periodic SRV opening and closing actions. Pressure oscillations cause the oscillations of the core void fraction as shown in Figure 5. When the core average void fraction approaches 1, dryout happens. The peak clad temperature (PCT), as shown in Figure 6, begins to increase. When the SRVs open, the residual water in the downcomer and lower plenum enters the core again; the steam cooling effect [10] also reduces the PCT. But finally at around 4000 s, the core is full of steam and the PCT rapidly increases. Within 1000 s, the fuel clad damage temperature is reached. The simulation is stopped. The timing for PCT to reach 1200 K is quite close to the similar case studied by the SOARCA project, which is around 4300 s for the fuel at the core middle plan [9].

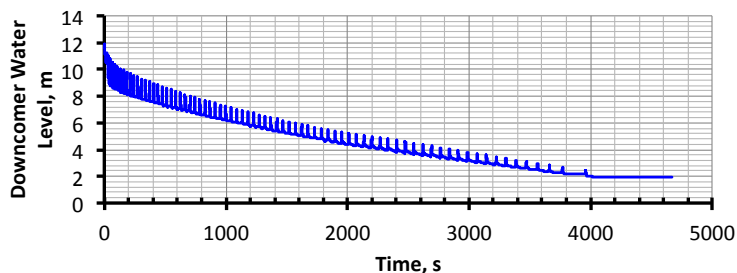
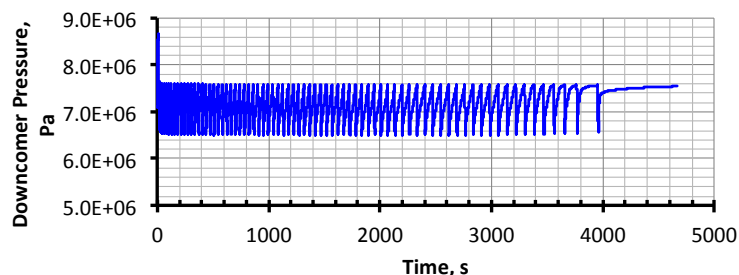
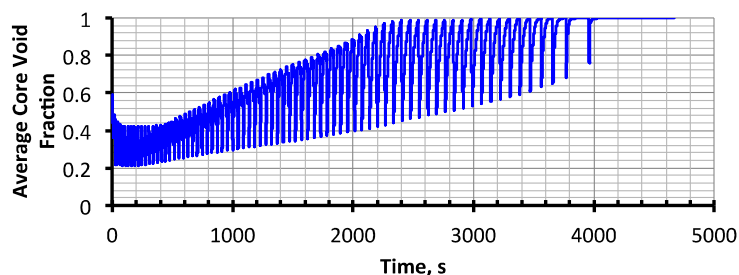
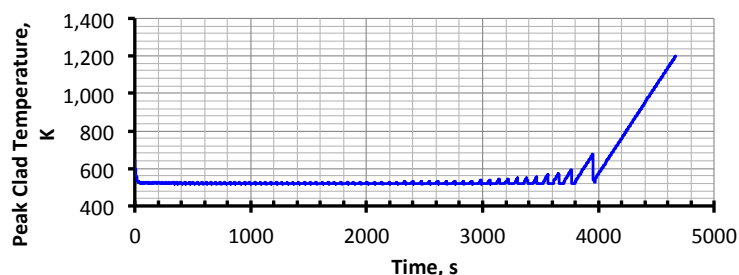


Figure 3. Downcomer water level during scenario I SBO transient.**Figure 4.** Downcomer pressure during scenario I SBO transient.**Figure 5.** Average core void fraction during scenario I SBO transient.**Figure 6.** Peak fuel clad temperature during scenario I SBO transient.

4.2. Simulation Results for Station Blackout Scenario II – With RCIC

In this simulation, the RCIC system is turned on and off for three periods until assumed battery energy (2 hours) is exhausted. Then the turbine is kept on to simulate the pressure release and the makeup water through the RCIC pump is shutdown. The reactor water level gradually decreases due to the loss of the reactor vessel water inventory to the suppression pool. Dryout happens after the downcomer water level becomes very low. The PCT rapidly increases after dryout. The simulation stops when the PCT reaches 1200 K.

The simulation sequence for the scenario is summarized as follows:

- At $t = 0$ s, SBO initiating event occurred:
 - Reactor scrammed
 - Decay heat was turned on
 - Primary pump coast down started with half-time of 1 s
- At $t = 1$ s:
 - Feedwater line valve began to close and became fully closed at $t = 2$ s
 - Main steam isolation valve began to close and became fully closed at $t = 11$ s

- From $t = 10$ s to 7270 s, the RCIC system was turned on and off for three periods with a transition time of 10 s between all the changes of status:
 - 1st period of turning on RCIC system for 30 minutes (same nominal mass flow rate at 40 kg/s for steam release through RCIC turbine as water injection through RCIC pump)
 - 1st period of turning off RCIC system for 15 minutes
 - 2nd period of turning on RCIC system for 30 minutes (nominal mass flow rate at 40 kg/s)
 - 2nd period of turning off RCIC system for 15 minutes
 - 3rd period of turning on RCIC system for 30 minutes (nominal mass flow rate at 20 kg/s)
- From $t = 7270$ s and on, maintained pressure release through turbine with nominal mass flow rate at 20 kg/s and shut off makeup water supply through the RCIC pump
- At $t = 22937$ s, simulation stopped when the PCT approached 1200 K.

Time evolutions of key system parameters are shown in Figure 7 through Figure 13. Figure 7 shows the RCIC turbine shaft work during the transient. The first two RCIC on/off periods are clearly identifiable from the figure. The shaft work (or turbine power) is dynamically determined by the plant operational conditions, including inlet stagnation pressure and temperature and outlet pressure. When the turbine nominal mass flow rate is ramped up (using the turbine control valve), both the shaft work and real mass flow rate increase quickly. The reactor vessel pressure decreases with time (as shown in Figure 8) due to the release of steam through the turbine into the suppression pool. The turbine power decreases along with the decreasing reactor vessel pressure. The turbine model gives results that are consistent with the expected dynamical behavior. This is a major improvement over current SBO simulations, where a given mass flow rate is typically used to simulate the turbine behavior. During extended SBO accidents (as demonstrated in the Fukushima Daiichi accidents), major instruments are not available. The turbine mass flow rate and power should not be guessed, but should be directly simulated through dynamical models such as the one presented in this paper.

Figure 8 shows the reactor vessel pressure at the downcomer (same as at the steam dome). Generally, when the RCIC system is on, the vessel pressure would decrease with time due to pressure release through the RCIC turbine and cold water injection through the RCIC pump. When the RCIC system is off, the pressure would increase with time. After the makeup water injection is turned off, the reactor vessel pressure increases initially then begins to decrease because the decay heat power drops to a lower level with time. It should be noted that it is not an efficient way to control reactor pressure through the RCIC system. Instead, the pressure relief valve should be relied on to rapidly reduce the system pressure as shown in the scenario I.

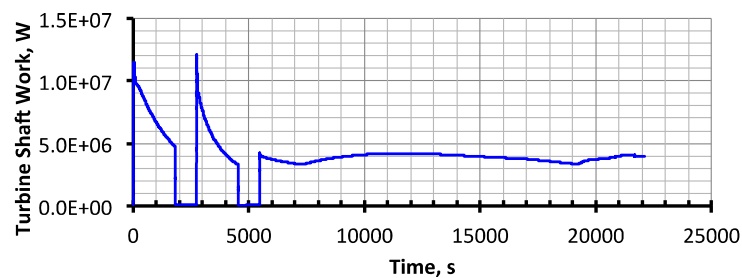


Figure 7. RCIC turbine shaft work during scenario II SBO transient.

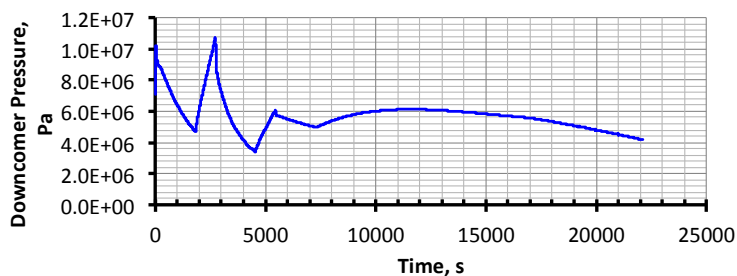


Figure 8. Pressure at downcomer during scenario II SBO transient.

Figure 9 shows the downcomer water level during the transient. With the RCIC system fully functioning, the water level can be well maintained within a couple of meter range. When the safety water injection is lost, the level begins to steadily drop.

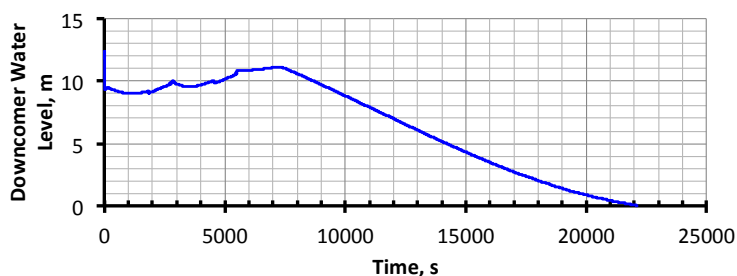


Figure 9. Downcomer water level during scenario II SBO transient.

Figure 10 shows the PCT during the transient. The PCT decreases when the RCIC system is on and increases during off time. When the RCIC system makeup water injection stops, the PCT changes slowly with the system pressure until dryout occurs. Then the PCT increases very quickly and reaches 1200 K at about 22000 s.

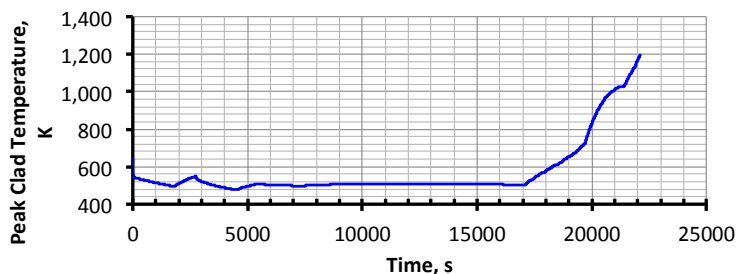


Figure 10. Peak clad temperature during scenario II SBO transient.

Figure 11 shows the wet well suppression pool average water temperature. This is an important parameter to determine the available net positive suction head (NPSHa) and the availability and performance of the RCIC pump. The water temperature rose by 41 K from the steady state value of 300 K. The conservative value for the pool temperature limit (373 K) is the boiling temperature at atmospheric pressure. Figure 12 shows the wet well suppression pool level evolution during the transient. The initial level is set at 4 m and the final level is computed to be 4.72 m, which represents a nearly 20% increase due to steam injection into the suppression pool and its heat up. Figure 13 shows the wet well gas pressure, which also is an important parameter to determine the NPSHa value. The gas pressure increases by about 40% from the initial value. In this simulation the vacuum breaker is assumed to be closed.

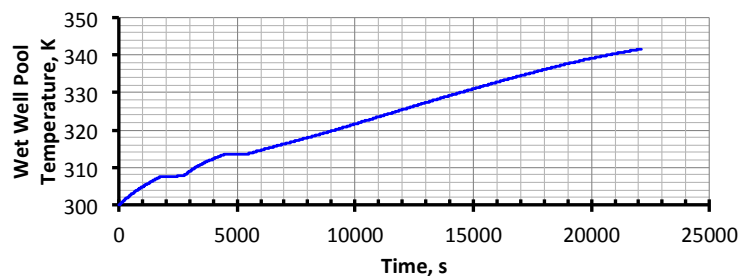


Figure 11. Wet well pool temperature during scenario II SBO transient.

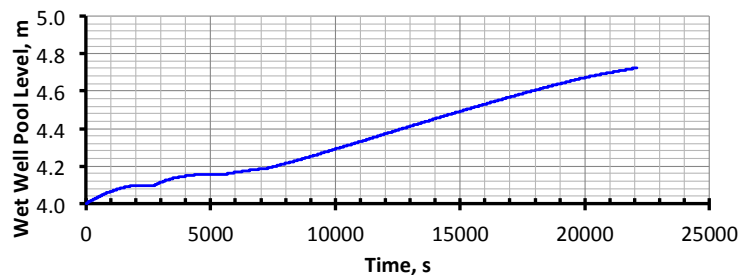


Figure 12. Wet well water pool level during scenario II SBO transient.

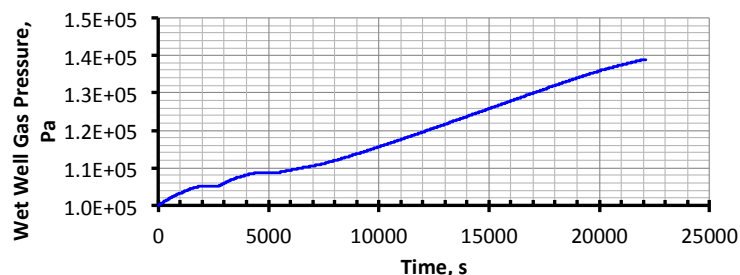


Figure 13. Wet well gas pressure during scenario II SBO transient.

4.3. Simulation Results for Station Blackout Scenario III – With SRV and RCIC

In this scenario, the RCIC system is assumed to be available for the first 2 hours and the same manual control as for the scenario II is used. When the RCIC system cannot limit the pressure below the SRV high pressure setting point, SRVs will open to release pressure. After about 2 hours, only SRVs are available to control the system pressure through releasing steam into the suppression pool. This case therefore is more realistic. The simulation is terminated when the PCT reaches 1200 K.

Time evolutions of key system parameters are shown in Figure 14 through Figure 22. Figure 14 shows RCIC turbine shaft work, which is quite similar as in the scenario II for the first two hours. Figure 15 shows the mass flow rate through the RCIC turbine. Although in the control logic, the nominal mass flow rate for the turbine is set to be 40 kg/s for the two opening periods, the actual mass flow rate is not constant but follows the dynamical conditions at the turbine inlet and outlet, such as pressure and temperature. The nominal mass flow rate can be obtained only under the design condition.

Figure 16 shows the system pressure. Comparing to Figure 8 for the scenario II, the high pressure

is limited by the SRV high pressure setting point. After the RCIC system is disabled, the pressure trend is quite similar to that in scenario I as shown in Figure 4. The cycling of SRVs causes the pressure oscillations and maintains the pressure within a narrow band.

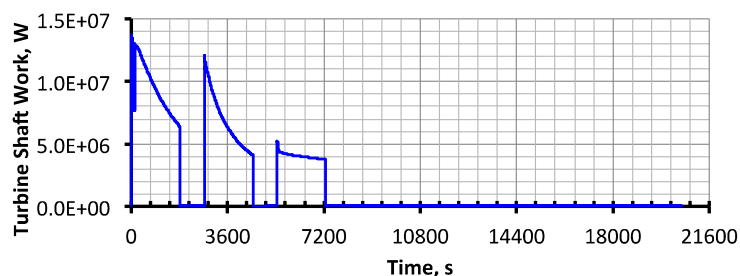


Figure 14. RCIC turbine shaft work during scenario III SBO transient.

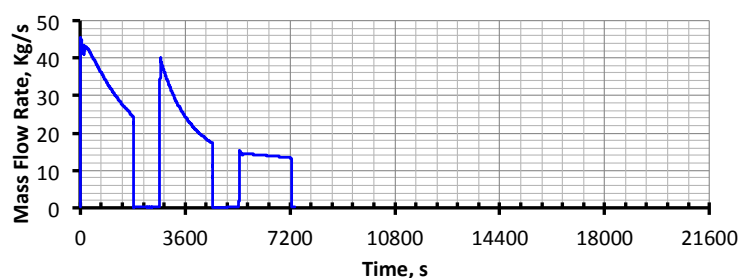


Figure 15. RCIC turbine mass flow rate during scenario III SBO transient.

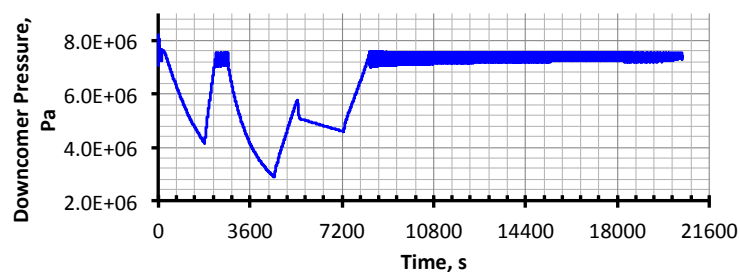


Figure 16. Pressure at downcomer during scenario III SBO transient.

Figure 17 shows the downcomer water level during the transient. The water level drops to a lower level than that in the scenario II due to additional discharge through SRVs in the first two hours, but in general similar behaviors can be observed. Figure 18 shows the average void fraction in the reactor core. Note that the core is flooded with water during short periods of time during the transient.

Figure 19 shows the PCT during the transient. The PCT reaches 1200 K at around 20600 s, which is earlier than the corresponding fuel damage time (22000 s) in scenario II.

Figure 20 shows the wet well pool temperature during the transient. The trend and the final temperature are very close to scenario II. The wet well pool level and the gas space pressure are shown in Figure 21 and Figure 22, respectively. The final statuses of the wet well for scenario III and II are very close.

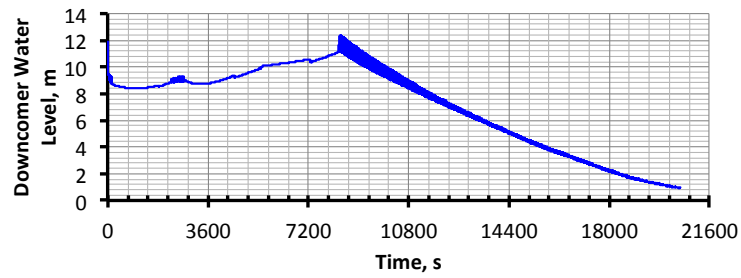


Figure 17. Downcomer water level during scenario III SBO transient.

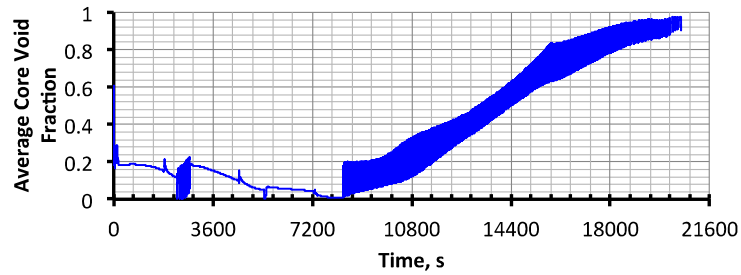


Figure 18. Average core void fraction during scenario III SBO transient.

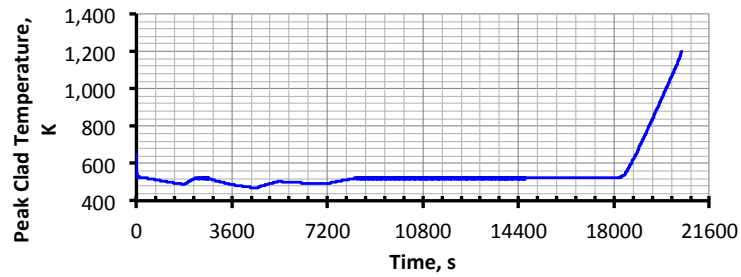


Figure 19. Peak clad temperature during scenario III SBO transient.

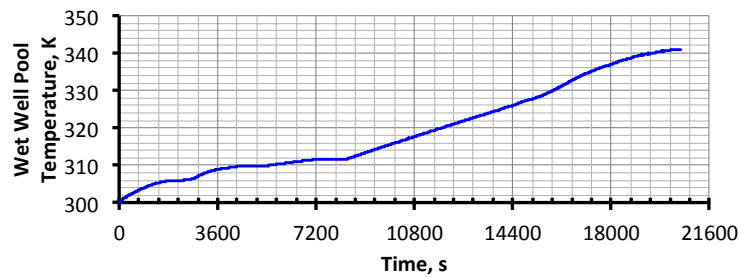


Figure 20. Wet well pool temperature during scenario III SBO transient.

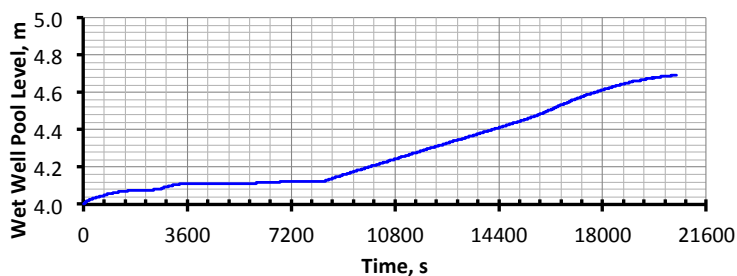


Figure 21. Wet well pool level during scenario III SBO transient.

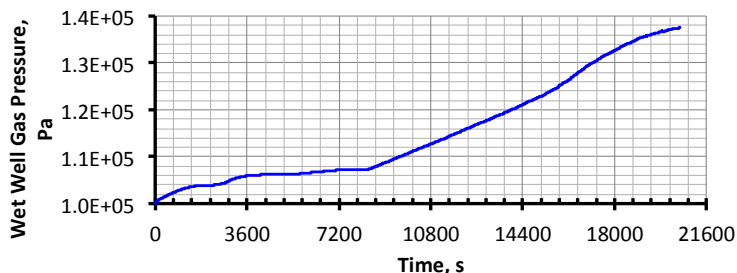


Figure 22. Wet well gas space pressure during scenario III SBO transient.

5. CONCLUSIONS

To fully understand the complex system behavior during extended SBO for a BWR reactor, it is important to include detailed dynamical models for safety-important components and systems in the system analysis codes and models. By fully coupling all these dynamical models together at the system level, the complex interaction between different physics and physical components can be better revealed and predicted. Thus, the key figures of merits for safety such as PCT can be more accurately computed. The demonstration RELAP-7 simulations for three simplified SBO scenarios show the importance of the safety relief valves, the RCIC system, and the wet well system to the reactor safety during extended SBO accidents. The ultimate purpose of simulation for reactor systems is to establish confidence for reactor safety and improve economy while enhance safety through improving safety system design and accident management procedures. Due to short development time, RELAP-7 code currently is still not matured enough to be used as safety analysis tool. However, the successful demonstration of simulations of these simplified SBO scenarios shows the potential to further improve the models for safety analysis by including more detailed physical processes in the near future.

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