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Research Article

Analyzing Simplified BWR Inherent Safety System using IAEA Generic Boiling Water Reactor Simulator

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Abstract

The inherent safety of boiling water reactors (BWR) has been a vital research topic in the past decades. This study aimed to observe and analyze the simplified BWR inherent safety system incorporated in IAEA Generic BWR Simulator. This simulator represents important features of BWR and provides graphical information and real-time simulation data. The simulated BWR has 1300 MWe power with ABWR-type containment. To analyze its inherent safety system, three conditions are simulated, i.e., normal condition at 100% power, transient condition (feedwater pumps trip), and emergency condition (loss of coolant accident—LOCA). The simulations were performed for up to 30 minutes since the most critical events in all conditions occurred within that time frame. Sequences of transient and emergency conditions were described in detail with the help of an additional screen recorder and time counting software. Results of several parameters in all simulation conditions were compared and analyzed. It was concluded that the simulator could simulate the normal, transient, and emergency conditions and the simplified version of the BWR inherent safety system.

Keywords: *Boiling water reactor; Inherent Safety; Loss of Coolant Accident; IAEA Simulator*

Analisis Sistem Keamanan Bawaan BWR yang Disederhanakan menggunakan IAEA Generic Boiling Water Simulator

Abstrak

Sistem keamanan bawaan Boiling Water Reactor (BWR) menjadi topik penelitian penting dalam beberapa dekade terakhir. Penelitian ini bertujuan untuk mengamati dan menganalisis sistem keamanan bawaan BWR yang disederhanakan yang terdapat dalam Simulator BWR IAEA. Simulator ini mewakili fitur-fitur penting BWR, dan memberikan informasi grafis serta data simulasi real time. BWR yang disimulasikan memiliki daya 1300 MWe, dengan pengungkung reaktor tipe ABWR. Untuk menganalisis sistem keselamatan bawaan, disimulasikan tiga kondisi yaitu kondisi normal pada daya 100%, kondisi transien (pompa air umpan mati) dan kondisi darurat (kecelakaan kehilangan pendingin - LOCA). Simulasi dilakukan hingga 30 menit, karena sebagian besar peristiwa penting di semua kondisi terjadi dalam rentang waktu tersebut. Urutan kondisi transien dan darurat dijelaskan secara rinci dengan bantuan perekam layar tambahan dan perangkat lunak penghitung waktu. Hasil dari beberapa parameter pada *semua kondisi simulasi dibandingkan dan dianalisis. Disimpulkan bahwa simulator dapat mensimulasikan kondisi normal, transien, dan darurat, serta versi sederhana dari sistem keselamatan bawaan BWR.*

Kata Kunci: *Boiling water reactor; Keselamatan bawaan; Kecelakaan kehilangan pendingin; Simulator IAEA.*

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I. INTRODUCTION

Nuclear has become one of the renewable energy sources contributing to fulfilling the world's electricity demand. According to International Energy Agency in their 2020 statistic report, nuclear is the third most significant contributor with a 10.1% contribution after coal (38.0%) and natural gas (23.0%). In 2018, nuclear produced approximately 2700 TWh of electricity resulting from 440 nuclear power plants operating in the world [1]. In reference data series 2019 edition released by International Atomic Energy Agency (IAEA), Pressurized Water Reactor (PWR) and Boiling Water Reactor (BWR) are dominantly used as nuclear power plants operating today [2].

Like any other power plant, nuclear power plants also have several accident risks, such as reactor core damage and radioactive release to the environment that can cause cancer and death [3]. With the advancing research and technology, nuclear reactors operating nowadays are entirely safe because they have inherent safety features. According to Hansson, inherent safety eliminates the accident probability in the first place rather than handling accident indications using additional equipment or procedures [4]. In nuclear reactors, Weinberg introduced the inherent safety concept as the new safety philosophy that covers maximum utilization of physical and chemical properties of the fuel, heat-transfer agent, radioactive waste, and other components [5].

Since the 1980s, several researches about BWR inherent safety have been performed. Among them is BWR neutronic design accommodated inherent safety [6], introducing process inherent ultimate safety (PIUS) concept in BWR—which resulted in a more straightforward BWR design that can reduce the probability of core meltdown [7], and inherently safe fluidized-bed BWR, that combined fluidized-bed concept with one of PIUS mechanism [8].

Those researches focus on advancing BWR design by conducting mathematical modeling and experiments. Other methods to study inherent safety are through simulation, building the code from scratch, or using readily available simulators. While the first is challenging and time-consuming, the second option seems to be more effective. Several codes that can simulate the entire BWR plant are BWR plant analyzer [9], BWR-LTAS [10], SIMULATE-3 [11], OSU-ISR [12], POLCA-T code [13], and International Atomic Energy Agency (IAEA) BWR Simulator [14]. BWR Simulator is one of several simulators provided by IAEA, apart from the PCTRAN Pool Reactor Simulator used in the previous study [15].

This study used the IAEA BWR Simulator, which is freely available, easy to use, and has relatively complete BWR features. This analytical study of BWR inherent safety would be a great utilization of this simulator because—to the authors' knowledge, no published BWR studies are using this.

To understand the inherent safety of BWR, we simulated the most occurring accidents, i.e., trip of feedwater pump and loss of coolant accident (LOCA) in the reactor pressure vessel (RPV). Despite several simplifications, both emergency conditions can be simulated by the simulator successfully. In addition, the simulation can represent the accurate simplified responses of the inherent safety system of BWR.

II. METHOD

In this study, we simulated three conditions, i.e., a normal condition with 100% power, a transient condition where all feedwater pumps trip, and emergency conditions of loss of coolant accident (LOCA), with reactor vessel leakage rate of 800 kg/s, accompanied by feedwater pumps trip after 5 seconds. Steps to run the simulator are provided in the manual.

The reactor parameters used in this simulation are shown in Table 1.

Table 1. Reactor Parameters

	Value	Unit
Power plant net output	1300	MWe
Reactor thermal output	3926	MWth
Steam flow rate at normal condition	2122	kg/s
Steam temperature/pressure	287.8/7.07	°C/MPa
Feedwater flow rate at normal condition	2118	kg/s
Feedwater temperature	215.6	°C
Primary coolant flow rate	14,502	kg/s
Reactor operating pressure	7.07	MPa
Fuel Material	Sintered UO_2	

Parameters shown in Table 1 are obtained when the reactor operates at 100% power level with UO_2 as fuel. It produces electricity of 1300 MWe and thermal power of 3926 MWth. To operate at full capacity safely, the steam flow rate is set to 2122 kg/s at 287.8 °C and 7.07 MPa.

There are two machines in the simulator, i.e., CASSIM and Labview. CASSIM is an output calculation machine, while Labview displays CASSIM's calculation results. This simulator cannot be fast-or-back forwarded; thus, the transient conditions are hardly noticeable. To overcome this, we used a screen recorder software such as OBS Studio to capture every moment of simulation results. In addition, time display in the simulator is only available in the Labview graphics, with only start and end times in the left and right parts of the graphs without other details. Therefore, we used additional software to analyze the exact time of every event in the simulation.

The normal condition of 100% power can be performed by running the initial condition (IC) "Full Power". After this IC is opened, the simulator will display reactor condition at 100% power without any additional steps. At this condition, the reactor is always stable and critical. Meanwhile, adding a malfunction after opening IC "Full Power" by pushing the "Malf" button to simulate transient conditions. Choose "Loss of Feedwater – Both FW Pumps Tripped" with a time delay of 0 s. This malfunction will run immediately after clicking the "insert MF" button.

The emergency condition can be simulated the same way as the transient, except choose "Reactor Vessel Medium Size Break LOCA ~ 800 kg/s" in the "Malf" options. After 5 seconds, add another malfunction and select "Loss of Feedwater – Both FW Pumps Tripped" with a time delay of 0 s.

Graphs in these simulations are displayed in real-time and were recorded with screen recorder software. Then, the resulting videos were analyzed to investigate the reactor's response to the changes. Every second, the sequence of events was observed until the reactor reached a steady-state or returned to its normal condition.

BWR Safety System

BWR has emergency core cooling systems (ECCS) to cool down the reactor core under LOCA like any other light water reactor. It consists of two high-pressure systems, i.e., high-pressure coolant injection (HPCI) system and automatic depressurization system (ADS), and two low-pressure systems, i.e., low-pressure coolant injection (LPCI) and core spray (CS) system. A complete explanation of the ECCS integrated performance and its work can be found in [16].

In case of severe core accidents, such as core meltdown, the reactor containment prevents the release of radioactive materials into the environment. In general, containment consists of a steel dome head and several major components needed to protect gaseous and particulate fission products inside the reactor building.

IAEA BWR Simulator

Figure 1 shows an initial display when the simulator is used. It contains several information, such as the simulator's name, IAEA's logo, BWR diagram, the developer's identity, and IC selection. The developer provides several IC options, and users can make their own additional IC if necessary.

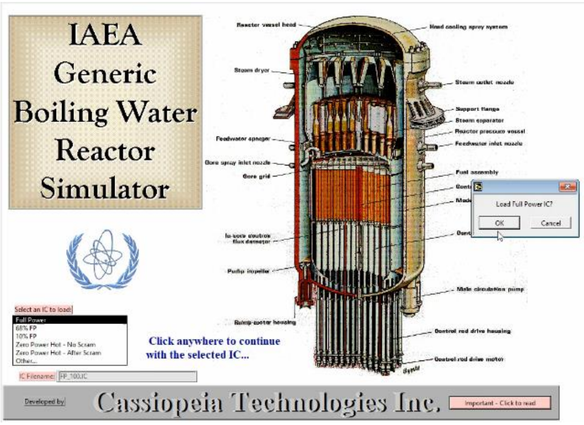


Figure 1. The First Display of BWR Simulator

After one IC is selected, the simulator's view is shown in Figure 2. The main one is the BWR Plant Overview, which offers essential reactor parameters and their indicators. The numbers shown in the view are the initial conditions of the reactor's parameter according to the selected IC. When the simulation is running, eight other displays can be viewed. Each display explains a particular part of the reactor in detail.

Figure 3 shows indicators on the simulator, which represent the real BWR indicators. The indicator's light will turn red or yellow when the reactor reaches a particular condition. The reactor engineer designs these turning on and off indicators to warn the operator when the reactor condition passes beyond the save level. A detailed explanation of the limits of every indicator can be found in the BWR Simulator manual.

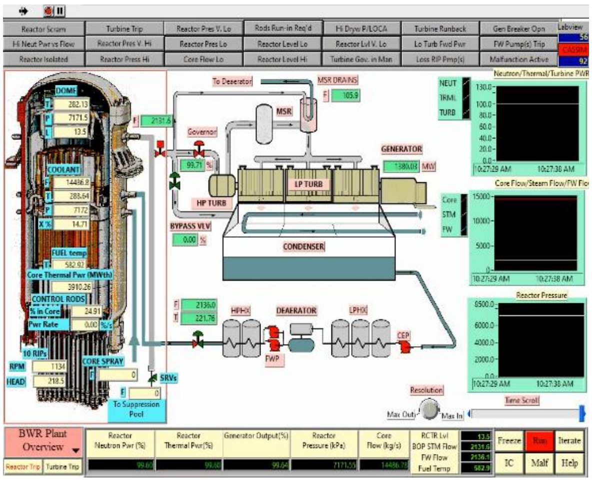


Figure 2. BWR Plant Overview Display

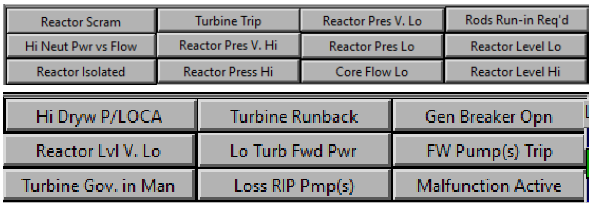


Figure 3. Simulator Signs Display

IAEA BWR Simulator was made by Cassiopeia Technologies Inc. (CTI) to educate students, study the reactor operations, and give a general overview of how the reactor responds to the accidents. This simulator describes BWR realistically, with several simplifications. Therefore, IAEA does not recommend responses given by simulators--particularly incidents--to be used as a base of safety analysis in the real events [17]. It is also important to note that this simulator cannot simulate severe core accidents. Its responses might only give a general overview in the first stage of the after-accident scenario.

BWR simulated in this simulator provided an active safety system. The reactor can react automatically without operator intervention, and its process utilizes auxiliary power sources (such as AC power, diesel, pumped cooling water, etc.). Therefore, the simulation will be run assuming that the power generation system never loses electricity sources.

This simulator uses a general BWR design with 1300 MWe and ABWR (Advanced BWR) containment-type power. Several simulated systems in the simulator are reactor pressure vessel (RPV) and dome, containment, reactor auxiliary system, turbine generator, feed water, extraction steam, and reactivity and control. The simulator's front view and its signs displays are shown in Figures 1 and 2, respectively.

The simulator's working principle and detailed explanation of each component of the BWR—especially its safety systems can be found in BWR Simulator Manual, published by IAEA [17].

III. RESULTS AND DISCUSSION

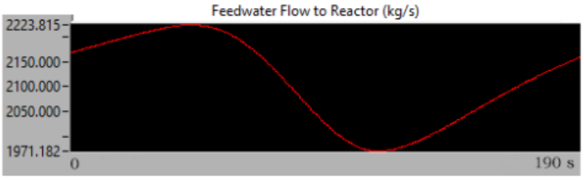
The simulation results of normal, transient, and emergency conditions are shown in Figures 4-8. Graphs of the same parameter are joined in one figure to see the difference between conditions. The graphs of reactor water level, reactor power, and reactor dome pressure at normal conditions are not shown because they remain constant during the simulation. The detailed analysis of each condition is discussed in the following subsections.

Normal Condition (100% FP)

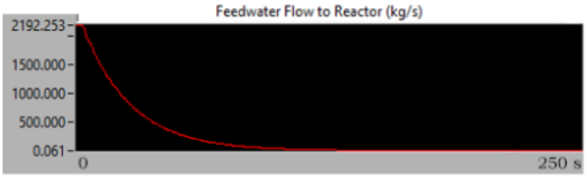
In this condition, the reactor operates at full power with 1300 MWe output, and all reactor components work normally. Table 2 shows the reactor parameters under normal condition, as the addition of parameters in Table 1.

Table 2. Reactor Parameters at 100% FP

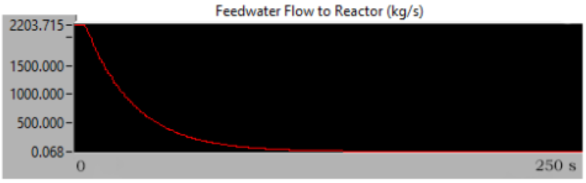
	Value	Unit
Reactor water level	13.5	m
Fuel Temperature	582	°C
Control rods in the core	24.91	%
Core Flow	14,490	kg/s
Governor valve	99.92	%



(a)

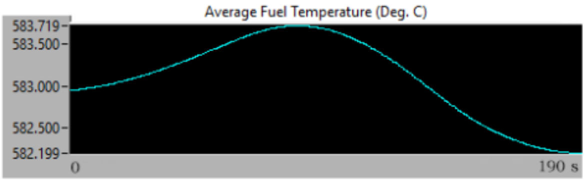


(b)

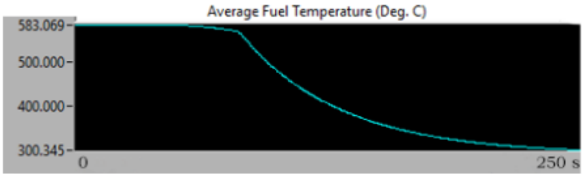


(c)

Figure 4. Feedwater Flow in Normal (a), Transient (b), and Emergency (c) Conditions



(a)



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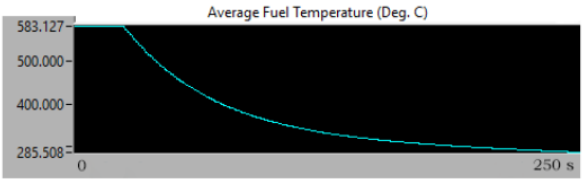
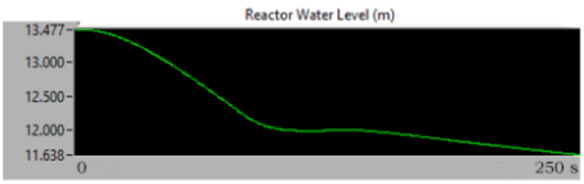
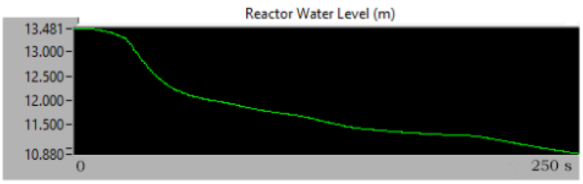


(c)

Figure 5. Average Fuel Temperature in Normal (a), Transient (b), and Emergency (c) Conditions

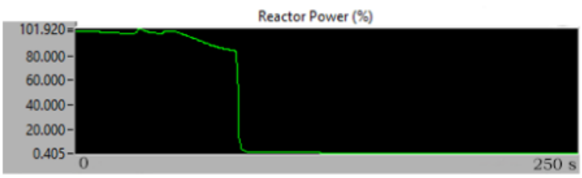


(a)

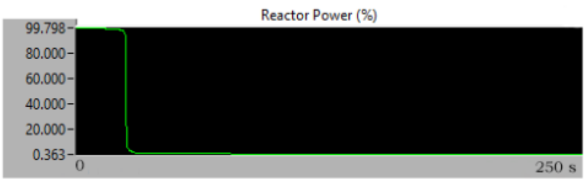


(b)

Figure 6. Reactor Water Level in Transient (a) and Emergency (b) Conditions

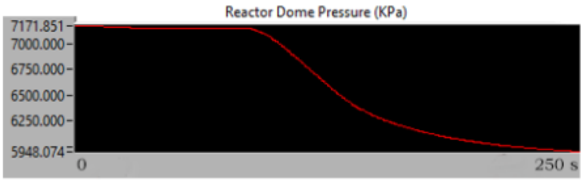


(a)

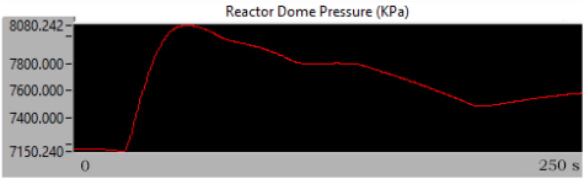


(b)

Figure 7. Reactor Power in transient (a) and emergency (b) conditions



(a)



(b)

Figure 8. Reactor Dome Pressure in transient (a) and emergency (b) conditions

Graphs at normal conditions (part (a) in Figures 4 and 5) show an oscillation, known as the "density wave" oscillations [18], resulting from the constant changes of water-void fraction in the reactor core. This continuous change occurs because vapor escapes the reactor core, followed by freshwater from feedwater. As we know, water and void have different abilities in reducing neutron energy [19]. Void fraction change in moderator will affect the thermal neutron population in the reactor core and reduce fission reaction.

Density wave oscillations--a dynamic instability--are the most common fluctuations in the two-phase natural circulation loops. The mechanisms associated with density wave oscillations are the delay in propagation of perturbations and feedback effects on the initial parameters of the concerned system [20]. The change in the inlet flow rates affects the void generation, hence the two-phase mixture density. The propagation of the transition from the inlet to the outlet with delay changes the pressures drop across the channel and intensifies disturbance in the flow rate. This flow rate feedback in the loop is manifested as density wave oscillations in the loop [21, 22].

Several studies about density wave oscillations have been performed previously. Paul's study of nonlinear dynamics of density wave oscillations of a two-phase flow found that the main cause of this phenomenon is the effects of different non-uniform axial heat flux profiles [23].

Suwoto et al. [24] have performed a similar study using the IAEA BWR simulator. They lowered the reactor power level from full power until shutdown. The simulator can perform the decreasing power level smoothly, although the density wave oscillations affect the shutdown process. As at the start-up, Shanbin Shi et al. have studied the startup process of The Purdue NMR (Novel Modular Reactor) BWR-type small modular reactor. They recommend startup procedures to eliminate flow instabilities of NMR during the startup [25], which also can be applied in a full-type BWR.

Transient Condition

According to the ANS Annual meeting in 1997, from 1988 until 2013, the loss of feedwater has been the second-largest incident in US's NPP. The total incident was 202; 68 occurred in BWR while 134 others were in PWR [26]. This shows that the loss of feedwater in BWR is one of the important incidents to be analyzed.

This study simulated the transient condition when all feedwater pumps trip. BWR safety system in this simulator responded effectively during the transient condition. Thus, it did not lead to other accident sequences. It reduced water level in RPV, but there were no significant parameter changes that could damage the reactor core. Graphic representations of this condition are shown in part (a) of figures 5 and 6. In detail, the sequences of the transient event resulting from the simulator are as follows.

a. (0s) "FW Pump(s) Trip" sign is on as soon as feedwater pumps are being disabled. The feedwater flow rate began to decrease.
b. (77s) Reactor water level decreases to 12.3 m, leading to reactor scram. "Reactor Scram", "Rods Run-in Req-'d", and "Reactor Lvl V.Lo" signs are automatically on. Neutron power decreases to 0%, while thermal power and generator output decrease slowly. Coolant to core flow reduces to 13,282 kg/s, with a fuel temperature of 544 °C.
c. (105s) "Reactor Pres Lo" sign is on. Parameter decrease occurs, i.e., reactor pressure of 6,862 kPa, fuel temperature of 447 °C, coolant flow of 9,108 kg/s, and reactor water level of 12.0 m.
d. (126s) The "Turbine Runback" sign is on. Thermal power decreased to 41.95 MW. Meanwhile, the generator output is still 78 %.
e. (129s) "Reactor Pres V.Low" is on. More decreasing parameter values: reactor pressure of 6,468 kPa, coolant temperature of 280 °C, fuel temperature of 389 °C, the core flow of 6,995 kg/s, and generator output of 77.84%.
f. (193s) "Rods Run-in Req'd" sign is off. Core flow, fuel temperature, coolant temperature, pressure, thermal power and generator values are 5,138 kg/s, 318 °C, 276 °C, 6,049 kPa, 18.68 % and 34.35 %, respectively.
g. (252s) Feedwater flow reach 0.0 kg/s.
h. (324s) "Loss RIP Pmp(s)" sign is on, with the reactor water level of 11.4 m; the core flow of 4,967 kg/s, fuel temperature of 291 °C, reactor pressure of 5,887 kPa, the head pump of 0 kPa, and pump rotation of 313 rpm.
i. (329s) "Core Flow Lo" sign is on, and the core flow decrease to 3,999.5 kg/s. The Governor valve is open by 15%, and the coolant temperature is 274 °C.
j. (10min, 20s) The water level reaches the lowest level at 10.9 m.
k. (10min, 43s) The core spray is on.
l. (12min, 12s) Water level started to increase with the core flow of 3,453 kg/s.
m. (23m) Simulation was terminated because the simulator successfully overcame the transient condition, and almost all parameters are not changing anymore (steady). The water level at this time is 11.8 m (from 13.0 m at the beginning), with a fuel temperature of 284 °C, coolant temperature of 272 °C, the core flow of 1,600 kg/s, thermal power of 4%, and generator output of 1.82% (25.23 MWe).

When a feedwater pump trip occurs, the water supply to cool down the core is decreased. Coolant available inside the core receives great heat from the fuel. However, BWR operated at saturation temperature. It is shown by the increase of void fraction instead of the growth of coolant temperature.

Due to the "negative void reactivity" characteristic of BWR [19], the increase of void fraction decreases its fission reaction; thus, the fuel temperature decrease. Loss of feedwater supply is followed by the continuous loss of vapor in the turbine and coolant level in RPV decrease. Loss of vapor is also meant the reduction of the reactor's pressure.

When the reactor scram, heat in the core only results from decay heat. This decay heat is hot enough to change water to vapor. Thus, water continuously changes to vapor even though its temperature is decreasing. Due to the decreasing pressure, the saturation temperature is also reduced. Following this, reactor core temperature is continuously decreasing due to the decrease of decay heat.

Meanwhile, the reactor water level keeps decreasing because the water inside the core changes to vapor. After reaching L2, or 11.43 m, ECCS is automatically on. ECCS supplies the water inside the core, and its level slowly increases to almost normal condition. The reactor will go completely back to normal when shutdown.

If the reactor's water level doesn't fall beyond the top of active fuel—TAF, the reactor will not be damaged. In Kuosheng's BWR/6 reactor (reactor with thermal power of 2896 MWth) simulation, the water level will not fall to TAF during feedwater trip if there is water injection with a minimum of 12.11 kg/s [27]. In this simulator, the reactor received additional water from core spray at 180 kg/s, much more than the minimum limit in Wang's study.

Emergency Condition

As shown in parts (b) of figures 5 and 6, emergency conditions caused a huge rapid loss of the coolant reactor. However, similar to the above, the simulator's safety features can also overcome the emergency situation. The sequences in this condition are described as follows:

- (0s) Malfunction "Reactor Vessel Medium Size Break LOCA~800 kg/s" is activated.
- (5s) Malfunction "Loss of Feedwater" is added.
- (24s) "Hi Dryw P/LOCA", "Reactor Scram", and "Rods Run-in Req'd" signs are on.
- (26s) Core Spray is on.
- (29s) "Reactor Isolated" sign is on.
- (30s) Valve that controls the release of vapor is closed, leading to increasing reactor pressure increase. This also leads to a decrease in generator output.
- (36s) SRVs (safety/release valves) are on generator output decrease to 38 MW.
- (38s) "Reactor pressure Hi" is on.
- (40s) "Reactor pressure V.Hi" is on, with reactor pressure of 7,882 kPa, the reactor water level of 12.5 m, the core flow of 10,859 kg/s, and the fuel temperature of 499 °C.
- (37s) "Reactor Lvl V.Lo" is on, water level decrease to 12.3 m with the core flow of 9,730 kg/s.
- (56s) Pressure increases continuously until it reaches 8,080 kPa, and then decreases.
- (60s) "Turbine Runback" and "Lo Turb Fwd Power" signs are on, generator output decreases to 3.35 MW.
- (81s) "Turbin Trip" sign is on due to reducing reactor water level to 11.9 m.
- (86s) "Gen Breaker Opn" sign is on.
- (100s) Parameter values are fuel temperature of 355 °C, coolant temperature of 280 °C, reactor pressure of 7,867 kPa, reactor water level of 11.7 m, generator output of 0%, reactor thermal power of 997 MW (25%), and core flow of 4,902 kg/s.
- (150s) "Core flow Lo" and "Loss RIPS Pmp(s)" signs are on, with a reactor level of 11.4 m and a core flow of 3,119 kg/s.
- (9min, 20s) Reactor water level reaches its lowest level at 6.9 m. Parameter values are decreasing: fuel temperature to 230 °C, core flow to 727 kg/s, reactor pressure to 7,559 kPa, and coolant temperature to 209 °C.
- (9min, 42s) "Reactor pres Lo." sign is on, with the pressure of 6,834 kPa.
- (9min, 51s) "Reactor Pres V.Lo." sign is one, pressure decrease to 6,456 kPa.
- (30min) Simulation is terminated with the following parameter values: Reactor flow of 845 kg/s, fuel temperature of 162 °C, the reactor water level of 13.9 m, reactor pressure of 1,129 kPa, coolant temperature of 151 °C, and core thermal power of 140 MW.

This simulator can detect LOCA from the increasing pressure of the drywell. When LOCA occurs, water flows from RPV to the containment system due to the lower pressure (7 MPa inside the RPV and 100 kPa in the containment's drywell). This water leak leads to increased pressure in the drywell, and the simulator will detect it as LOCA when the drywell's pressure reaches more than 114.6 kPa [14]. When this happens, the valve connecting RPV to the turbine is closed to avoid more vapor and coolant loss. This also increases the reactor's pressure because water coolant that evaporates to vapor has a higher temperature and bigger volume. If the valve is not closed, the reactor will suffer more coolant loss; the pressure will decrease; thus, most of the high-temperature water will change to vapor and escape from RPV.

In an emergency condition, reactor pressure can increase to 8,080 kPa. However, this increase is not dangerous because the RPV was designed to hold pressure until 8.62 MPa or 8,620 kPa. On the other hand, the reactor water level can decrease to only 6.9 m. With the fuel rod's height of 9 m measured from the same origin, 2.1 m of fuel is not covered by coolant. Nonetheless, fuel temperature is not increasing because BWR protection systems such as HPCF (high-pressure core flooder) and RCIC (Reactor Core Isolation Cooling System) work effectively when LOCA is detected.

These emergency simulation results are slightly different from other studies. According to Mindaugas study, if the top of active fuel is uncovered for 18 seconds, it can rupture due to a ballooning [28]. Javier also showed that this emergency system could not stop the evolution of core degradation during a LOCA design basis accident progressing to a severe accident [29].

IAEA defines seven stages of Reactor Accidents. The three first stages are design basis accidents, while the other four are severe accidents. The three first stages of a reactor accident are [30]:

1. Boil down of coolant and fuel heat up
2. Clad balloon and rupture
3. Clad oxidation and temperature transient.

This study produced slightly different results because, in this simulation, the reactor was hardly in the first stage. Fuel heats up did not occur due to the good functioning of the inherent safety system.

Despite the relatively good simulation results, this study has several limitations. There is very limited literature about BWR Simulator used in this study, and this simulator is a generic one—not specified in a typical BWR reactor. Thus, it is difficult to compare it with the real commercial BWR. In addition, users also cannot change parameters set in the simulator to make it as close as possible to the real reactor's parameters.

As the basic safety analysis, this study can be useful for students or reactor operators training to understand the principal safety systems of BWR before they operate the real one. Especially, they can appreciate several transient and emergency conditions and how the reactor reacts in those situations.

IV. CONCLUSION

The simulator "IAEA Generic Boiling Water Reactor" can simulate several reactor conditions and its inherent safety systems. These systems can detect changes in reactor components' potential to endanger the reactor. In addition, they also respond automatically without operator intervention and can overcome transient and emergencies without damaging the reactor. The simulator can simulate the reactor in a quite detailed manner. In FP conditions, the simulator can show density wave oscillation. In addition, the simulator provides other 16 malfunctions (of 18) that can be simulated and analyzed. The simulator can also be operated manually, if necessary.

Due to several simplifications, this simulator cannot reproduce a more complex emergency condition due to several simplifications, such as the Fukushima accident. Because, unlike in the Fukushima accident, the simulator assumed that the electricity source was always available. This simulator cannot simulate severe conditions such as the fuel meltdown in Fukushima. From the simulator's simulation results, it can be concluded that—generally, this simulator can successfully represent the inherent safety system of BWR and can be used as a valuable learning resource.

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