Investigating and reassessment of the Bushehr NPP' s SBO Analysis with presenting the last diesel generator recovery time

Authors: Yarizadeh-Bene, Mr. Mehdi, Zangian, Dr. Mehdi, Kargaran, Dr. Hamed, Minuchehr, Dr. A., Zangian, Dr. Mehdi

Date: 2025-02-14T00:00:00+00:00

Abstract

Addressing the issue of station blackout accidents in the NPP is of utmost importance because they can compromise the protection layer of nuclear reactors and significantly impact core integrity. This study assesses the influence of operator actions and their consequences on the evaluation of accidents caused by station blackout (SBO) events in a Bushehr NPP using the Relap5 code. The goal is to determine the optimal recovery time for diesel generators. The operator' s goal during an SBO accident is to manage and control the accident by depressurizing the primary circuit and using the accumulators to postpone the occurrence of core damage. Certainly, trying to activate emergency diesel generators or using transportable equipment, including transportable diesel generators, are some essential strategies for accident control and preventing core damage. But the important point at this stage is how long there is an opportunity to activate the diesel generators or use the transportable equipment to supply coolant for the reactor core and cool the second circuit to prevent core damage. In the case of a SBO accident at an NPP, two critical parameters must be considered for effective control. The first parameter is the selection of an appropriate strategy by the operator to take appropriate measures to reduce the pressure on the primary side of an NPP. The second parameter is knowledge of the time frame for the recovery of diesel generators and the subsequent activation of safety systems to manage and control the accident. According to the presented results, the actions of the operator significantly contribute to preventing SBO accidents. The activation of safety systems in this accident may not be enough to guarantee the safety of the NPP alone. However, the actions of the operator to continuously reduce the pressure of the first circuit, together with the operation of the safety systems, can lead to the establishment of safe conditions in the power plant.

Full Text

Preamble

Investigating and Reassessment of the Bushehr NPP' s SBO Analysis with Presentation of the Last Diesel Generator Recovery Time

Mehdi Yarizadeh-Benea, Mahdi Zangiana*, Abdolhamid Minuchehra, Hamed Kargaran
a Faculty of Engineering, Shahid Beheshti University, Tehran, Iran
b Engineering Department, Shahed University, P.O. Box: 3319118651, Tehran, Iran

Abstract: Station blackout accidents in nuclear power plants represent a critical safety concern as they can compromise reactor protection layers and severely impact core integrity. This study assesses the influence of operator actions on station blackout (SBO) accident evaluation in the Bushehr Nuclear Power Plant using the RELAP5 code, with the goal of determining optimal diesel generator recovery time. During an SBO, the operator' s primary objective is to manage and control the accident by depressurizing the primary circuit and utilizing accumulators to delay core damage. While activating emergency diesel generators or transportable equipment constitutes an essential accident mitigation strategy, the critical question concerns the available time window for such actions to supply coolant to the reactor core and cool the secondary circuit before damage occurs. Two parameters are paramount for effective SBO control: the operator' s selection of an appropriate strategy to reduce primary side pressure, and knowledge of the time frame for diesel generator recovery and subsequent safety system activation. The results demonstrate that operator actions significantly contribute to SBO accident prevention. While safety system activation alone cannot guarantee NPP safety, the combination of continuous primary circuit pressure reduction by operators and safety system operation can establish safe plant conditions.

Keywords: SBO, RELAP, BNPP, Recovery time

Abbreviations:
ACC: Accumulator
AEOI: Atomic Energy Organization of Iran
AC: Alternating Current
LPIS: Low-pressure injection system
BNPP: Bushehr Nuclear Power Plant
DC: Direct Current
MCP: Main coolant pumps
NPP: Nuclear Power Plant
ECCS: Emergency core cooling system
PSD: Pulse safety valves
EDG: Emergency diesel generators
EFWS: Emergency feed water system

EOP: Emergency operating procedures
RPV: Reactor pressure vessel
FSAR: Final Safety Analysis Report
HPIS: High-pressure injection system
LOCA: Loss of Coolant Accident
PWR: Pressurized water reactors
RCP: Reactor coolant pumps
RPV: Reactor pressure vessel
SBO: Station blackout
SG: Steam generators
SRD: Steam relief device

1. Introduction

When a nuclear power plant faces an accident, control room staff must act quickly and follow emergency operating procedures (EOPs) to restore plant safety and stability, having been extensively trained on plant simulators for effective EOP implementation. However, if an accident leads to core damage and possible radioactive release, rendering EOPs invalid, staff must depend on guidelines and technical support for effective situation management (Lee, Yilmaz et al. 2020). Event management thus constitutes a vital element of the defense-in-depth approach, complementing operating procedures developed for normal conditions, anticipated operational occurrences, and accident conditions (IAEA 2008).

The tsunami that struck the Fukushima nuclear plant triggered a series of events leading to reactor damage, with the primary cause being the loss of both onsite and offsite power during the long-term SBO, which disabled all core cooling systems. Without a heat sink, the reactor core overheated and the fuel structure melted down, while operators faced difficulties using transportable equipment to supply feedwater coolant due to the harsh environment (Mehri, Safarzadeh et al. 2021). A station blackout, where a nuclear power plant loses connection to the external power grid, can potentially trigger severe accidents, as demonstrated by the Fukushima NPP accident following a massive earthquake and tsunami. Consequently, studying SBO causes and consequences in nuclear power plants is crucial (Li, Wang et al. 2014).

Operating under difficult conditions caused by drastic external events or serious accidents necessitates unified collaboration among all emergency staff, including emergency response teams, security personnel, and control room operators (IAEA 2015). The operator' s role is critical, as their awareness of safety system operation affects their performance and behavior in managing and controlling the situation, making effective accident handling training essential. Given the importance of station blackout accidents to NPP safety analysis, this research conducts a deterministic safety analysis of SBO accidents at Bushehr NPP.

A station blackout accident involves loss of all AC power sources, including normal operation power supply systems (offsite grid and turbine generator) and emergency power supplies like diesel generators (Gjorgiev, Volkanovski et al. 2014). SBO can be readily recognized by the loss of all operating systems except the direct current (DC) system (Volkanovski, Avila et al. 2016). The Bushehr NPP features hydro-accumulators in stages 1 and 2 as passive safety systems—tanks filled with water and pressurized gas that can inject water into the reactor core during LOCA events. However, hydro-accumulators can only postpone core damage following SBO occurrence (AEOI 2007).

During an SBO, operators can only rely on passive systems and pressurizer safety valves to control the NPP and postpone core damage. If electricity is restored through auxiliary power source recovery (such as diesel generators or transportable diesel generators), safety systems will activate, enabling operators to control the accident and prevent core damage. The critical question concerns the time window available for operators to prevent accident escalation by connecting auxiliary AC power. This answer indicates the available time to recover diesel generators or use transportable units to prevent severe accident progression. The primary objective of this analysis is to provide operators with a guide that, by knowing this critical time, offers the chance to reduce accident consequences and prevent core damage. This research simulates BNPP station blackout accident scenarios using the RELAP5/Mod3.2 thermohydraulic code, both with and without operator actions, determining thermohydraulic parameters until fuel-clad temperature reaches 1200 °C. The operator' s actions to reduce primary pressure are analyzed for their effect on accident management, enabling use of water stored in ECCS passive safety systems to delay core damage. The results demonstrate that if at least two trains of safety systems are recovered before reaching the third critical point, core damage can be prevented during this accident.

2. Description of Bushehr NPP Reactor

2.1 General Information

The Bushehr NPP is a VVER reactor, the Russian version of PWRs. The reactor core contains 163 hexagonal fuel assemblies generating 3000 MWth of heat. The plant has two circuits: primary and secondary. In the primary circuit, light water serves as both coolant and moderator, facilitating heat transfer to the secondary circuit. The primary side includes a pressure vessel, pressurizer, and four cooling loops with main coolant pumps (MCPs). Steam generators (SGs) act as heat exchangers connecting the secondary and primary circuits, producing steam from primary circuit heat and sending it to the turbine. Figure 1 [Figure 1: see original paper] illustrates the reactor layout and components.

The primary circuit' s ECCS includes a high-pressure injection system, low-pressure injection system, hydro-accumulators (ACCs), and KWU tanks. These passive heat removal systems can inject coolant into the core during loss-of-coolant accidents. The BNPP ECCS features two stages of accumulators

(Pouresgandar, Safarzadeh et al. 2022).

3. Description of the RELAP Model

The RELAP5/MOD3 code is a thermal-hydraulic tool for simulating light-water reactor systems, first released in 1979 for INEL analysts studying LOFT and semi-scale experiments. It subsequently became popular for analyzing commercial and experimental reactors and their scaled models. RELAP5 (Fletcher and Schultz 1992) serves as the best-estimate code for simulating thermal-hydraulic behavior of light water reactors, with primary application in thermal-hydraulic analysis of loss-of-coolant accidents in PWR reactors. It can analyze various accidents and transients affecting the reactor cooling system, primarily for PWRs but also capable of handling research reactors, boiling water reactors (BWRs), and heavy water reactors. RELAP5 is trustworthy and versatile software for studying reactor safety and performance.

The code has numerous applications, including regulation validation, calculation verification, incident management method evaluation, user guideline assessment, and test design. It can simulate various thermal-hydraulic transients in any nuclear or non-nuclear system containing steam, water, gas, and dissolved substances, serving as a framework for nuclear power plant analysis. RELAP5 is versatile software for analyzing system performance during transients and reactor coolant system behavior (Fletcher 1999).

RELAP5 employs a two-fluid model for simulating two-phase flow in nuclear power plants. The break flow models include the Ransom-Trapp model, initially used in RELAP5 but deficient and poorly matching experimental data; the Henry-Fauske model, introduced as an alternative that performed better for various test cases and eventually became the main modeling tool; and choking model shortcomings including two-phase critical flow underestimation at low pressure when slip ratio isn’ t forced to unity, subcooled break flow prediction errors of 40-50% for thin orifice plates near saturation, and issues with thin orifice plate configurations at low pressure and low-quality conditions.

For two-phase flow modeling, RELAP5 uses a non-homogeneous and non-equilibrium hydrodynamic model formulating temporally averaged volume and flow parameters (Fletcher and Schultz 1992). A non-homogeneous two-phase model models the BNPP in RELAP, supplemented by the abrupt area model for breaks and reactor vessel inlet/outlet nozzles. The choking model is used in break modeling and safety valves, while the cross-flow model is applied in steam generator, fracture, and reactor vessel nozzle modeling. The BNPP model in RELAP5 studies plant behavior under different scenarios and is validated by comparing results with plant data for steady-state and transient conditions (Mirzamohammadzadeh, Hadad et al. 2022).

The VVER-1000 RELAP5 model consists of several components: the reactor core and vessel divided into two sections with different power distribution coefficients (Kr); a pressurizer with three pulse safety valves (PSDs) discharging into the same relief tank, with one control valve opening above 18.1 MPa and two safety valves opening above 18.6 MPa, all requiring DC power; safety components including the high-pressure injection system, eight boron supply tanks, and LPSI with a control volume system model; and a secondary side inlet control valve comprising four steam generators with safety valves, four BRU-A and BRU-K valves, a steam pipeline, and turbine.

Figures 2 and 3 present nodalization of the BNPP primary and secondary sides. The average core channel is modeled using pipe component 30, while the hot rod temperature in the hot channel is determined based on the maximum power peaking factor of the hot fuel assembly, modeled using pipe component 35. Table 1 contains hydrodynamic volume details for reactor components.

The pressurizer (910) connects to the cold leg of the third loop through pipe 915 and valves 916 and 917, which inject water into the pressurizer when pressure increases. The pressurizer is constantly connected to the hot leg of the second loop through pipe 905, allowing primary side volume fluctuations to transfer to the pressurizer. On the primary side of the horizontal steam generator, hot fluid enters through the hot collector, flows through horizontal pipes exchanging heat with the lower temperature and pressure secondary side, and exits through the cold collector after cooling.

The VVER-1000/V-446 passive ECCS includes Stage 1 and Stage 2 hydro-accumulators. Stage 1 accumulators (numbers 85, 87, 86, 88) connect to the reactor pressure vessel’ s upper and lower plenums, while Stage 2 accumulators (numbers 91-98) connect to the hot and cold legs of main coolant pipelines. Hot primary fluid enters steam generator tubes through the hot collector, with tubes modeled as five parallel channels of different cross-sectional areas but equal length, each represented as an individual pipe. Cross-sectional areas are proportional to tube numbers in each channel. The primary side main cooling line has branches at entrance (120) and exit (140), with downstream branches (141-145 and 121-125) connecting to main coolant channels (pipes) and two upper branches (126 and 146) connecting to the emergency gas output system. The primary side total water volume is approximately 420 m^3. The steam generator secondary side is modeled using hydrodynamic components representing each flow channel, with the downcomer section transferring water from the steam-water separator (volume 508) to facilitate natural circulation. Adjacent hot and cold channels connect through transverse connections.

Main feedwater supply to the steam generator is modeled using boundary volumes and connections 540 and 541, while emergency feedwater supply uses boundary volumes and connections 542 and 543. The simulation includes emergency cooling systems such as accumulator stage 1 and KWU accumulators, which inject water into the reactor core and hot/cold legs during emergencies. The secondary side model considers the steam generator, vapor pipeline, generator power supply, emergency feedwater system (EFW), and BRU-A and BRU-K safety systems based on BNPP information. The VVER-1000 RELAP5 model is first validated under steady-state conditions. Table 2 shows main safety protection system actuation conditions, while Table 3 compares RELAP5 model simulation results with FSAR reference data under steady-state conditions, validating the code input model by comparing basic nuclear reactor power plant parameters under steady conditions with design document data (AEOI 2007).

4. Definition of Accident Scenarios

A station blackout occurs when offsite power is lost (LOOP) and emergency diesel generators (EDGs) fail to start (Tatu and Kim 2017), representing a design extension condition (DEC) (IAEA 2010). Initiating events involve failure of diesel generators, normal power supply sources, HPIS, LPIS (ECCSs), EFWS, and spray systems. Various factors can cause these failures, including natural disasters (earthquakes, tsunamis), military attacks, human errors, or inadequate maintenance (Amirsoltani, Pirouzmand et al. 2022). The only remaining power sources are DC supplies from class 1E batteries requiring operator action. During SBO, decay heat in the reactor core dissipates through natural convection within the primary side and transfers to steam generators on the secondary side. If batteries are unavailable or depleted and AC power is not restored, secondary cooling fails and core damage occurs (Tatu and Kim 2017).

This scenario assumes the primary coolant boundary remains intact—a conservative but unrealistic assumption taken to study the worst-case scenario (AEOI 2007). During station blackout, total AC power supply failure trips RCPs, main and auxiliary secondary side feedwater equipment, primary side makeup and blowdown systems, and pressurizer system power supply links. BRU-K valves become isolated, vapor flow to the turbine is interrupted, and turbine stop valves close while a scram signal is generated.

During initial accident stages when steam generators evaporate on the secondary side, BRU-A valves automatically open to release excessive steam into the atmosphere, managing pressure and preventing damage. As SG evaporation continues and they eventually dry out, the primary side experiences increased coolant pressure and temperature, leading to core heat-up (H. Ebrahimgol 2023). Eventually, primary side pressure reaches pressurizer safety valve setpoints, providing emergency overpressure protection. However, with normal safety systems nonfunctional, this primary coolant loss leads to core and reactor vessel damage from continued heat buildup.

The SBO accident sequence has three critical points: first, BRU-A valve activation and steam loss from steam generators, reducing primary side heat removal and damaging SG tubes while raising primary coolant temperature; second, core coolant level drop and fuel rod uncovery; and third, fuel-clad temperature reaching 1200 °C. Reactor cooling system behavior and thermohydraulic parameter analysis continues until clad temperature reaches 1200 °C, at which point core cooling becomes unsteady, leading to core heat-up and fuel-clad temperatures exceeding 1200 °C. Pressure reduction and accumulator water addition strategies have consequences such as increased hydrogen production. Safety system engineering specifications are presented in Table 4 .

4.1 SBO Accident Analysis in the Absence of Operator Actions (Scenario Version 1)

This scenario assumes no personnel actions or additional safety system failures. Primary and auxiliary secondary circuit feedwater systems, reactor primary circuit coolant pumps, volume control and chemical systems, and pressurizer are disconnected due to complete AC power source loss. The BRU-K valve is isolated and vapor flow to the turbine is interrupted. Table 5 shows the chronological progression of events during the station blackout accident without operator measures. In this scenario, PSD valves function passively without operator involvement, opening automatically when pressure reaches activation setpoints without requiring battery electric power.

4.1.1 Simulation Results for Scenario Version 1 Parameter variations throughout the accident are illustrated in Figures 4-9. Principal parameters are assessed against FSAR (AEOI 2007) and previous scholarly work (Jabbari, Hadad et al. 2019). Figure 4 [Figure 4: see original paper] shows core power decreasing with decay heat level when the unit loses off-site power and the reactor scrams. Primary pressure slightly decreases as inherent coolant circulation on the primary side cools both primary and secondary sides after RCPs stop, as shown in Figure 5 [Figure 5: see original paper]. This causes water evaporation and heating in the secondary circuit, with SG water drainage shown in Figure 6 [Figure 6: see original paper]. Steam discharges from the SG into the atmosphere through the BRU-A valve, maintaining SG pressure between 7.15 and 6.27 MPa. Figure 7 [Figure 7: see original paper] shows intermittent BRU-A valve operation with periodic closing and opening. As evaporated feedwater is not replaced by makeup water, SG level and BRU-A mass flow rate fall until the SG empties.

As the initial critical point, primary side pressure reaches pressurizer PSD valve actuation level when coolant heats up, causing primary coolant loss and core heat-up. Reducing coolant inventory in the core leads to fuel rod dryout—the second critical point. The third critical point occurs at 7300 s when the hot rod heats up, as illustrated in Figure 8 [Figure 8: see original paper]. Figure 9 [Figure 9: see original paper] shows pressurizer level variation during the accident. Without management measures, the core and reactor vessel will suffer damage at high pressure.

4.2 Station Blackout Accident Analysis with Consideration of Operator Actions (Scenario Versions 2 and 3)

This scenario analyzes operator actions in accident handling to reduce primary pressure. Total AC power supply failure trips reactor coolant pumps (RCPs), primary and auxiliary secondary circuit feedwater equipment, and makeup and blowdown systems. Table 6 shows the chronological sequence for scenario version 2, and Table 7 shows the sequence for scenario version 3 with system and device actuation.

The operator' s role in this accident scenario involves reducing primary coolant circuit pressure by opening pressurizer PSD and gas removal valves, continuing pressure reduction until hydro-accumulator activation, and allowing automatic accumulator injection into the primary side to cool the reactor once pressure is sufficiently lowered. A decrease in primary side pressure indicates successful pressurizer PSD valve opening by the operator. These actions are necessary for primary pressure reduction and potential use of water inventory in ACCs and Bushehr accumulators for core cooling during SBO. However, core cooling remains unsteady, leading to core heat-up and fuel rod cladding temperatures exceeding 1200 °C.

4.2.1 Simulation Results for Scenario Version 2 Figures 10–15 show BNPP analysis results during a station blackout accident with accident management scenario version 2. The reactor scrambles after losing power sources, with core power decreasing with decay heat level as shown in Figure 10 [Figure 10: see original paper]. Natural coolant circulation after RCPs disconnect slightly lowers primary circuit pressure and evaporates SG water inventory for 3000 s, as shown in Figure 11 [Figure 11: see original paper]. Secondary circuit pressure cycles between BRU-A opening and closing set points, shown in Figure 12 [Figure 12: see original paper]. After steam generator drainage, primary coolant heats up and pressure rises to the pressurizer PSD valve opening setpoint of 18.1 MPa, shown in Figure 13 [Figure 13: see original paper].

Coolant loss within the core and primary side arises from PSD valve opening. During the blackout accident at 5000 s, the operator opens one PSD and gas removal valve (shown in Figure 13) to lower primary side pressure and use ACC water resources for core cooling. Primary side pressure drops to ACC actuation pressure, with ACCs supplying boron solution to the reactor vessel when primary circuit pressure reaches 5.88 MPa. Hot rod temperature increase occurs at 7500 s, as shown in Figure 14 [Figure 14: see original paper]. Figure 15 [Figure 15: see original paper] shows pressurizer level changes during the accident.

4.2.2 Simulation Results for Scenario Version 3 Figures 16–21 show BNPP analysis results during a station blackout accident with accident management scenario version 3. Principal parameters are assessed against FSAR (AEOI 2007) and previous scholarly work (Jabbari, Hadad et al. 2019). Core power decreases with decay heat level, as shown in Figure 16 [Figure 16: see original paper]. Similar to scenario version 2, natural coolant circulation after RCPs disconnect slightly lowers primary side pressure (shown in Figure 17 [Figure 17: see original paper]) and evaporates SG water inventory for 3000 s (shown in Figure 18 [Figure 18: see original paper]). Secondary side pressure cycles between BRU-A opening and closing set points, shown in Figure 19 [Figure 19: see original paper]. After steam generator drainage, primary coolant heats up and pressure rises to the pressurizer PSD valve opening setpoint of 18.1 MPa, shown in Figure 17. Coolant loss in the vessel and primary side arises from PSD valve opening. During the station blackout at 5000 s, the operator opens three pressurizer PSD and gas removal valves (shown in Figure 17) to lower primary side pressure and use ACC and KWU accumulator water resources for core cooling.

Primary side pressure falls to accumulator triggering point, and boron solution is supplied from there. ACCs supply boron solution to the reactor vessel when pressure reaches 5.88 MPa. Water level rises above fuel assemblies when accumulators empty. After accumulator emptying and subsequent coolant level reduction in the core, fuel rods dry out and fuel-clad temperatures increase, as shown in Figure 20 [Figure 20: see original paper]. Figure 21 [Figure 21: see original paper] shows pressurizer level changes during the accident.

5. Ascertaining the Latest Time for Diesel Generator Recovery

Previous scenarios examined SBO accidents without emergency AC power system return, emphasizing the role of water accumulators. This section discusses modeling simultaneous SBO and LOOP accidents with emergency power restoration by NPP emergency diesel generators. To model emergency power restoration, systems recovered after power connection are modeled by RELAP5 code, including EFW systems and HPIS and LPIS emergency injection systems.

This section aims to provide guidance for plant operators and personnel. The SBO accident resulted from a LOOP accident without emergency diesel generator activation. As previously shown, the operator' s goal during SBO is to manage and control the accident by depressurizing the primary circuit and using accumulators to postpone core damage. While lowering primary side pressure and adding accumulator water has consequences such as increased hydrogen production, constraining operator actions, activating emergency diesel generators or transportable equipment remains among the most important measures for accident control and core damage prevention.

If personnel successfully activate diesel generators or transportable units, two critical points must be considered: the equipment operation time and the number of successfully activated units. The key question concerns the time window available to activate diesel generators or use transportable equipment to inject coolant into the reactor core and cool the secondary circuit to prevent core damage. Subsequently, the number of activated units substantially affects accident mitigation.

This section compares effective equipment activation times under two different assumptions about the number of activated units.

5.1 Determining Diesel Generator Recovery Time Assuming All Safety Systems Are Activated (4 Trains)

The first step examines effective activation time when all equipment in both circuits is successfully activated for all three modeled scenarios. Figures 22-24 show results of successful activation of all safety systems on primary and secondary sides at different times in scenario versions 1-3. According to these figures, successful activation of all emergency diesel generators and subsequent activation of all trains of emergency core cooling and emergency feedwater systems can prevent core damage up to 7500 seconds for versions 1 and 2, and up to 7800 seconds for version 3 after the accident, after which the accident cannot be controlled.

Figures 25-33 show results of successful activation of all safety systems on secondary and primary sides at 7500 s for versions 1 and 2, and at 7800 s for version 3 after the accident. The emergency feedwater system in BNPP provides deionized water to steam generators during power loss triggering automatic reactor shutdown. The EFW system and main BRU-A valve help dissipate remaining heat through SGs. This scenario assumes all safety system trains are recovered and all four feedwater trains operate, with SG water level rising to 2.2 meters due to fluid injection, as shown in Figures 25-27.

Emergency feedwater injection into steam generators leads to subsequent primary pressure decrease, as depicted in Figures 28-30. In version 1, activation of all four safety system trains causes an unexpected pressure increase reaching pressurizer PSD valve activation setpoint after a period of decrease. This pressure increase occurs because before safety system activation, primary circuit pressure was at PSD pressurizer valve activation level, and heat transfer between circuits ceased due to dry SGs. Consequently, heat produced in the reactor core without secondary circuit heat exchange causes coolant boiling and two-phase conditions. However, with HPIS implementation in the primary circuit and EFW in the secondary circuit, primary circuit pressure decreases due to established thermal transfer. This pressure decrease, combined with safety system injection into the primary circuit, reduces reactor core coolant temperature —effectively, primary circuit pressure reduction lowers cooling fluid saturation temperature in the reactor core. After some time, primary circuit pressure increases again due to several factors, including water injection through safety systems increasing primary circuit pressure. The initial thermal transfer between circuits reduced pressure while safety system injection increased coolant amount in the reactor core. As a result, pressure reduction decreased reactor core coolant saturation temperature. Since pressure and temperature reduction are not operator-controlled, coolant saturation temperature reduction from pressure decrease and thermal transfer between circuits causes boiling and two-phase conditions in the cooling liquid, while safety system injection increases primary circuit pressure.

As water level rises in SGs and heat transfer between circuits increases, secondary circuit pressure increases to BRU-A valve performance limits, as shown in Figure 28 [Figure 28: see original paper]. In version 2, emergency feedwater injection leads to primary pressure decrease, as depicted in Figure 29 [Figure 29: see original paper]. This version considers operator primary side pressure reduction, with safety systems such as HPIS and loop fluid injection increasing pressure to about 4 MPa before gradual decrease. In version 3, the operator reduces primary circuit pressure to first and second-stage accumulator activation thresholds, causing slight pressure rise from safety systems and fluid injection, followed by slow pressure decrease due to heat transfer between circuits, shown in Figure 30 [Figure 30: see original paper].

As water level rises in steam generators and heat transfer between circuits increases, version 1 shows secondary circuit pressure increase due to primary circuit pressure rise (Figure 28), as shown in Figure 31 [Figure 31: see original paper]. In versions 2 and 3, operator actions to lower primary circuit pressure cause steady pressure decrease in both primary and secondary circuits, with continuous downward pressure trends shown in Figures 32-33.

5.2 Determining Diesel Generator Recovery Time with Two-Train Safety Systems Activated

The second step evaluates effective equipment activation time when only two equipment trains are successfully activated under conservative assumptions. Figures 34-36 show results of successful two-train activation and safety times in scenarios 1-3. According to results, successful activation of two emergency diesel generator trains and subsequent activation of two emergency core cooling and emergency feedwater system trains can prevent core damage up to 7500 seconds for versions 1-2 and up to 7800 seconds for version 3 after the accident, after which the accident cannot be controlled.

Steam generator water level rises to 2.2 meters due to fluid injection, as shown in Figures 37-39. This scenario considers two-train safety system recovery with two emergency feedwater system trains operating.

Primary pressure decreases following EFW injection into SG, as depicted in Figures 40-42. In version 1, without operator action, safety system activation such as HPIS and circuit fluid injection increases primary circuit pressure. However, with only two active safety system trains limiting primary circuit injection, and with two EFW systems and heat exchange active, pressure increase from injection and saturation temperature cooling reduction is limited. In version 2, primary pressure decreases following EFW injection, as shown in Figure 41 [Figure 41: see original paper], with operator primary loop pressure reduction, safety system injection, and effective heat transfer causing gradual pressure decrease. In version 3, operator primary circuit pressure reduction to first and second-stage accumulator activation thresholds causes slight pressure rise from safety systems and fluid injection, followed by slow pressure decrease from heat transfer, as shown in Figure 42 [Figure 42: see original paper].

As steam generator water level rises and heat transfer between circuits increases, version 1 shows secondary circuit pressure increase to BRU-A valve performance limits, as shown in Figure 43 [Figure 43: see original paper]. In versions 2 and 3, operator primary circuit pressure reduction actions cause steady pressure decrease in both circuits, with continuous downward trends shown in Figures 44–45.

6. Conclusion

The Fukushima accident highlighted the importance of complete deterministic safety assessment for nuclear power plants to ensure resilience against accidents and design extension conditions. This research investigated SBO accidents in BNPP, which cause the highest core damage frequency, using a validated RELAP5 model. The first step simulated station blackout scenarios with and without operator actions. To delay heating onset, the main operator strategy is reducing initial side pressure to reach ACC and KWU accumulator activation points (version 3), though this results in lower primary side pressure than scenario version 1, potentially leading to low-pressure core damage.

Based on deterministic safety analysis, diesel generator recovery time and the number of activated diesel generators and subsequent safety systems play crucial roles in accident control and management. Two scenarios were considered: first, assuming all diesel generators and safety systems activate; second, assuming two diesel generator trains (or portable generators) and two safety system trains. In both scenarios, if diesel generators are recovered before 7800 seconds and the operator reduces primary side pressure (scenario no. 3), the accident can be maintained, controlled, and core damage prevented. The proposed scenarios were reevaluated to ensure no fuel rod damage and adequate core cooling. Under the most conservative assumption with only two system trains recoverable, the operator can prevent core damage by taking necessary measures and decreasing primary circuit pressure within the system recovery time window. Additionally, SBO accident management guidelines for BNPP are expanded based on safety analysis. According to results, accident management and control require operator measures to reduce first circuit pressure using PSD valves (as outlined in scenario no. 3), with operation of at least two safety system trains preventing reactor core damage.

References

AEOI (2007). "Final Safety Analysis Report (FSAR)." Atomic Energy Organization of Iran.

AEOI (2007). "Final safety analysis report of BNPP' s VVER-1000 reactor, chapter 15. accident analysis."

Amirsoltani, M., A. Pirouzmand and M. Nematollahi (2022). "Development of a dynamic event tree (DET) to analyze SBO accident in VVER-1000/V446 nuclear reactor." Annals of Nuclear Energy 165: 108786.

Fletcher, G. D., Schultz, R.R (1999). "RELAP5/MOD3.3 code manual." Idaho National Engineering Laboratory Idaho.
Gjorgiev, B., A. Volkanovski, D. Kančev and M. Čepin (2014). "Alternative off-site power supply improves nuclear power plant safety." Annals of Nuclear Energy 71: 304-312.
Hosseini, S. A., A. S. Shirani, M. Zangian and A. Najafi (2020). "Re-assessment of accumulators performance to identify VVER-1000 vulnerabilities against various break sizes of SB-LOCA along with SBO." Progress in Nuclear Energy 119: 103145.
IAEA (2008). "APPROACHES AND TOOLS FOR SEVERE ACCIDENT ANALYSIS FOR NUCLEAR POWER PLANTS."
IAEA (2010). "Development and Application of Level 1 Probabilistic Safety Assessment for Nuclear Power Plants." IAEA
IAEA (2015). "Severe Accident Management in the Light of the Accident at the Fukushima Daiichi Nuclear Power Plant."
Jabbari, M., K. Hadad and A. Pirouzmand (2019). "Re-assessment of station blackout accident in VVER-1000 NPP with additional measures following Fukushima accident using Relap/Mod3. 2." Annals of Nuclear Energy 129: 316-330.
Lee, J. H., A. Yilmaz, R. Denning and T. Aldemir (2020). "An online operator support tool for severe accident management in nuclear power plants using dynamic event trees and deep learning." Annals of Nuclear Energy 146: 107626.
Li, L., M. Wang, W. Tian, G. Su and S. Qiu (2014). "Severe accident analysis for a typical PWR using the MELCOR code." Progress in Nuclear Energy 71: 30-38.
Mehri, A., O. Safarzadeh and S. Talebi (2021). "The station blackout accident in a dual- cooled annular fuel of a VVER-1000 reactor with application of portable pumps for mitigating the accident." Annals of Nuclear Energy 152: 107964.
Mirzamohammadzadeh, M. H., K. Hadad and A. Pirouzmand (2022). "Failures of pressure innovative HPIS regulation system enhancing reactor safety in SBLOCA for WWER- 1000/V446 NPP." Annals of Nuclear Energy 169: 108949.
Pouresgandar, M., O. Safarzadeh and S. Talebi (2022). "Evaluation of advanced accumulator in a VVER-1000 reactor in loss of coolant accident." Annals of Nuclear Energy 170: 108988.
Tatu, A. and T. Kim (2017). "Extending SBO coping capability: An improved auxiliary feedwater system." Progress in Nuclear Energy 95: 40-47.
Volkanovski, A., A. B. Avila, M. P. Veira, D. Kančev, M. Maqua and J.-L. Stephan (2016). "Analysis of loss of offsite power events reported in nuclear power plants." Nuclear Engineering and Design 307: 234-248.
H. Ebrahimgol, M. A. (2023). "Evaluation of filtered containment venting system (FCVS) effects on severe accident of a WWER1000 plant." Annals of Nuclear Energy 181: 109522.

Note: Figure translations are in progress. See original paper for figures.