

Evaluation of reactor safety response to multiple steam generator tube rupture scenarios in a generic PWR simulator

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ABSTRACT

A Multiple Steam Generator Tube Rupture (MSGTR) constitutes a Design Extension Condition (DEC) in pressurized water reactors (PWRs), introducing challenges related to primary-to-secondary mass transport and potential radiological risks. While thermal-hydraulic behavior under single-tube failures is extensively studied, a critical gap remains in the literature concerning transient plant response during prolonged, unmitigated multi-tube ruptures. This study addresses this gap by performing transient simulations using a 1400 MWe Generic PWR (GPWR) simulator. Three MSGTR scenarios of increasing severity were evaluated in a single affected steam generator over a four-hour window: 5 tubes, 10 tubes, and 131 tubes, assuming zero operator intervention. The results indicate that the automatic safety protection systems effectively moderate the accident sequences. The volume control tank provides an initial coolant inventory buffering effect prior to automatic safety injection from the refueling water storage tank. This highly borated water injection introduces negative reactivity, initiating a steady reactor power coastdown. At the time of reactor trip, thermal power had decreased to approximately 3% of nominal power in all three cases, the majority of the remaining power was generated by decay heat. The maximum fuel centerline temperature decreases in strong correlation with this power coastdown, stabilizing at approximately 300°C. However, unmitigated auxiliary feedwater injection leads to complete flooding of the steam generators. To enhance control room diagnostics, a differential level indicator was proposed, demonstrating its effectiveness as an operator aid for rapid identification of the leaking steam generator.

INTRODUCTION

MSGTR in PWR

A steam generator tube rupture (SGTR) constitutes a critical design-basis accident (DBA) in pressurized water reactors (PWRs). This event involves a structural breach of the physical barrier separating the high-pressure reactor coolant system (RCS) from the secondary system, resulting in leakage of primary coolant into the secondary side through the affected steam generator (SG) tubes. Historically, SGTR events represent one of the most frequently occurring transient accidents in operating PWRs, as summarized in Table 1.

The primary-to-secondary side break flow rate serves as the primary factor governing the overall thermal-hydraulic response of the plant, including RCS depressurization and secondary system pressurization. Because an unmitigated SGTR poses severe environmental and nuclear safety risks, particularly the potential release of fission products to the environment, this transient is explicitly included in the Nuclear Energy Agency – Committee on the Safety of Nuclear Installations (NEA-CSNI) validation matrix for benchmarking advanced thermal-hydraulic codes and evaluating nuclear power plant (NPP) designs (Mascari et al. 2024). Consequently, single-tube SGTR transients have been extensively investigated using integral test facilities (Martin et al. 2024) and plant-specific simulations (Israel 2006; Jia et al. 2015; Kang et al. 2006; Rahman et al. 2024; Sui et al. 2020).

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Date received: 04 June 2026

Date revised: 01 August 2026

Date accepted: 13 August 2026

DOI: <https://doi.org/10.54645/202619SupNTA-56>

KEYWORDS

design extension conditions, emergency operation, generic PWR simulator, multiple steam generator tube rupture, steam generator

Table 1: SGTR accidents in PWRs (Abdellatif et al. 2024)

Plant name	Year	Leak rate (gpm)	Causes and location
Point Beach Unit 1	1975	125	Wastage
Surry Unit 2	1976	330	PWSCC in U-bend
Doel Unit 2	1979	135	PWSCC in U-bend
Prairie Island 1	1979	390	Loose parts
Ginna Unit 1	1982	760	Loose parts and tube wear
Fort Calhoun	1984	112	ODSCC at crevice
North Anna Unit 1	1987	637	High cycle fatigue in U-bend
McGuire Unit 1	1989	500	ODSCC in the free span
Mihama Unit 2	1991	700	High cycle fatigue
Palo Verde Unit 2	1993	240	ODSCC
Indian Point Unit 2	2000	150	PWSCC in U-bend
Krško	2008	20	Intergranular stress corrosion cracking

As shown in Table 1, single-tube failure may occur due to tube degradation mechanisms such as outside diameter stress corrosion cracking (ODSCC), primary water stress corrosion cracking (PWSCC), intergranular stress corrosion cracking, fretting and wear, high cycle fatigue cracking, denting, pitting, and wastage (uniform corrosion and thinning of the tube's outer surface due to the secondary water chemistry). Meanwhile, a multiple steam generator tube rupture (MSGTR) represents a significantly more complex transient condition. Historically, probabilistic safety assessments have indicated that the likelihood of simultaneous independent failure of multiple tubes is at least one order of magnitude lower than that of a single SGTR (Kondo 1994). However, common-cause initiators, such as loose parts, provide a credible mechanism for multi-tube damage. For instance, inspections following the 1982 Ginna NPP SG tube rupture revealed that, although only one tube experienced a complete rupture, more than 20 adjacent tubes exhibited severe mechanical degradation due to impact effects. Probabilistic safety assessments further indicate that the frequency of an MSGTR is comparable to that of major DBAs, such as a loss-of-coolant accident (LOCA) (USNRC 1993).

Compared to a single SGTR, an MSGTR accelerates transient progression, drastically reducing the operator action window and heightening the risk of direct radiological release. To evaluate plant resilience against such scenarios and to formalize robust mitigation frameworks, the International Atomic Energy Agency introduced the concept of Design Extension Conditions (IAEA 2024). Unlike rapid large-break LOCAs, MSGTR transients typically evolve over longer timescales, occasionally permitting continued core operation at reduced power configurations. These prolonged transients are highly critical from a fuel performance perspective; the thermal condition of the fuel rod cladding is governed by a gradual power coastdown and localized reactivity feedbacks rather than an instantaneous SCRAM. If mitigation strategies are delayed or ineffective, the rapid primary coolant depletion during an MSGTR can induce a high void fraction in the reactor vessel that might result in core uncover and fuel degradation. This underscores the importance of early operator diagnostics and structured accident management to achieve a stable, safe shutdown state.

While the thermal-hydraulic parameters of MSGTRs have been explored, there remains a distinct literature gap regarding the coupled response of transient plant behavior under these specific prolonged, reduced-power conditions.

Literature on APR1400 and GPWR modeling

Extensive research has been performed to model and simulate severe accidents using the APR1400 reactor architecture. Foundational deterministic, unmitigated best-estimate evaluations of MSGTR sequences were established utilizing the MARS 1.4 system code (Jeong et al. 2002; Jeong and Choi 2003). An APR1400 MSGTR benchmark study by Jeong et al. (2001) established 5 tube ruptures in a single SG as a baseline for localized multi-tube failure. Following the licensing basis of the APR-1400 established in accordance with the United States Nuclear Regulatory Commission (USNRC) SECY-93-087 policy paper (Seo et al. 2012), Shin and Hwang (2018) analyzed the effectiveness of realistic control systems. The APR1400 design specification dictates that the steam generator heat transfer area includes sufficient margin to maintain 100% rated thermal output even if up to 10% of the total tube inventory is plugged as an outcome of maintenance inspections during refueling outages (KEPCO and KHNP 2014). Bang et al. (2022) analyzed accident mitigation strategies under both full-power and low-power/shutdown conditions using MARS-KS. Dynamic operator action windows and required response times have been quantified through comprehensive sensitivity studies via RELAP5 (Bae et al. 2023; Yoon et al. 2021). To handle operational and model uncertainties, Best Estimate Plus Uncertainty (BEPU) methods have been successfully demonstrated by coupling system codes with statistical frameworks like DAKOTA (Dzień and Diab 2024). Advanced compound scenarios have also been evaluated, including deterministic dual-fault analyses capturing a mechanistically coupled Main Steam Line Break (MSLB) that induces SG tube rupture (Hruškovíć et al. 2023). More recently, these thermal-hydraulic databases have been integrated with machine learning meta-models to forecast accident progression (Spisak and Diab 2024).

Concurrently, the specific Generic PWR (GPWR) simulator configuration utilized in this study has been validated across different works. These include validation studies of classical system transients, such as station blackout (SBO) accidents with and without failure of the power-operated relief valve (PORV) (Almomani et al. 2023; Zubair et al. 2017), a small-break LOCA combined with a complete safety injection failure (Jeon et al. 2023), and a total loss of feedwater (TLOFW) accident scenario (Chung et al. 2023). The simulator has likewise been utilized for human factor reliability analyses (Jo and Lee 2022; Zubair and Bibi 2025). Furthermore, the high-dimensional data streams from the GPWR have heavily supported data-driven research, including machine learning for online abnormality diagnosis (Cho et al. 2023; Kim et al. 2020; Kim et al. 2025; Lee et al. 2021; Mena et al. 2021; Mena et al. 2022; Zubair and Akram 2023; Zubair and Akram 2024) and LOCA diagnosis (Almomani et al. 2025; Lin and

Ma 2026; Xiao et al. 2024; Zubair et al. 2026), alongside hardware-in-the-loop (HIL) cyber-security testbed deployments (Nam et al. 2025).

Despite this dual body of literature, a distinct gap persists: no published studies have utilized the coupled RELAP5/NESTLE capabilities of the GPWR simulator to comprehensively chart transient plant behavior during unmitigated MSGTR sequences. To address this gap, this study investigates the transient plant behavior within a GPWR subjected to MSGTR accidents across varying severities.

MATERIALS AND METHODS

Simulator facility and architecture

The simulations in this work were executed at the Nuclear Plant Simulator (NuSIM) facility, housed within the Philippine Nuclear Research Institute (PNRI). The NuSIM architecture models a Generic PWR capable of executing comprehensive reactor

operations, such as startup, shutdown, and emergency procedures, for engineering training and high-fidelity reactor power transient analyses. The simulator couples the RELAP5 system code for best-estimate thermal-hydraulic modeling with the NESTLE code for spatial core neutronics kinetics. The integrated suite executes in real time with dynamic graphical visualization inside the 3KEYMASTER environment.

The reference power plant modeled is a 1400 MWe PWR operationally analogous to South Korea's APR-1400 design (IAEA 2023). The primary loop configuration consists of one reactor pressure vessel (RPV), one vertical pressurizer (PZR), and two reactor coolant loops, where each loop comprises one hot leg pipe, two cold leg pipes, two reactor coolant pumps (RCPs), and one vertical U-tube steam generator. The plant overview is integrated within the high-fidelity simulator human-machine interface illustrated in Figure 1. The steady-state operational parameters of the reference APR-1400 design and the GPWR model are consolidated in Table 2. The reactor protection system (RPS) within the simulator engine is detailed in Table 3.

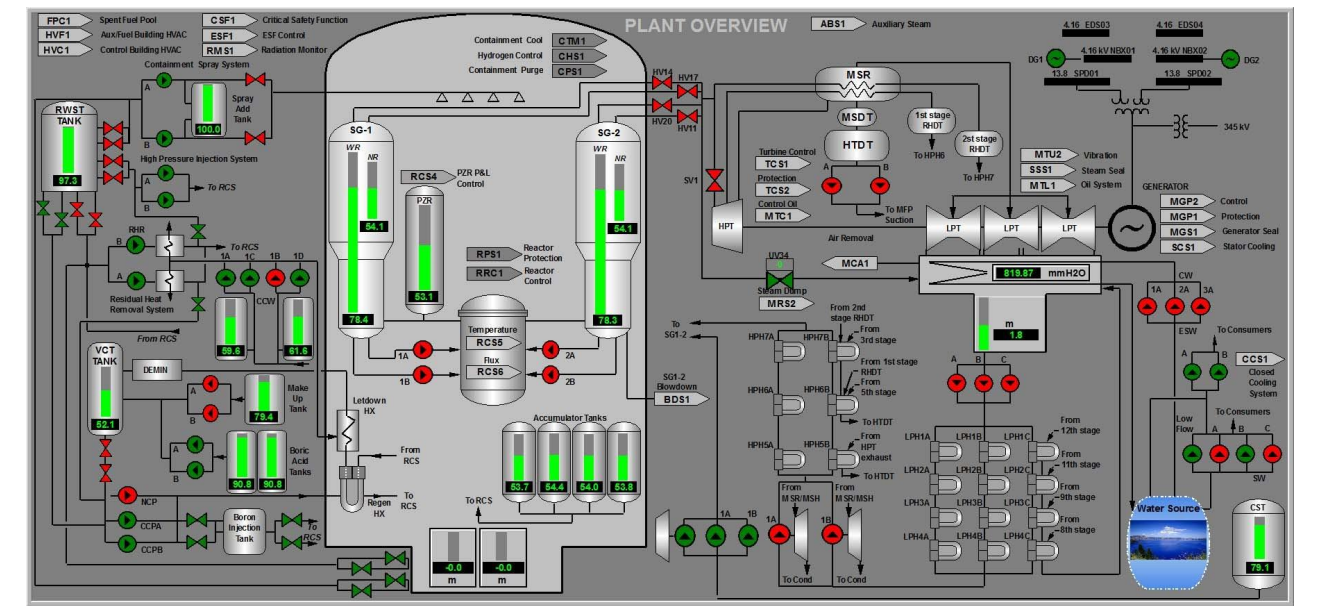


Figure 1: Plant overview of GPWR simulator.

Table 2: Operational parameters of APR1400 and GPWR (Jeon et al. 2023)

Parameter	Nominal Value
Configuration	2 loops, 2 SGs, 4 RCPs
Reactor core power, MWt	3983
Pressurizer pressure, kg/cm ²	158.2
Hot leg temperature, °C	325
Cold leg temperature, °C	292.6
Average temperature rise in vessel, °C	32.4
Average temperature in vessel, °C	309.2

Table 3: Reactor protection system automatic trip setpoints (WSC 2025)

Trip Signal	Value
Variable overpower	109.6%
High local power density (LPD)	656.2 W/cm
Low departure from nucleate boiling ratio	1.24
High pressurizer pressure	165.7 kg/cm ²
Low pressurizer pressure	129.6 kg/cm ²
High containment pressure†	1.24 kg/cm ² a
Low SG1,2 water level (narrow range)	23.5%
High SG1,2 water level (narrow range)	90.8%
Low SG1,2 pressure (narrow range)	67.5 kg/cm ²
Low reactor coolant flow‡	0.90 kg/cm ²

†The suffix "a" denotes absolute pressure (kg/cm²a), used for containment parameters to evaluate total internal pressure relative to a complete vacuum. Other system pressures are conventionally reported in gauge pressure (kg/cm²) because they operate at high pressures where ambient atmospheric variations are negligible.

‡The low reactor coolant flow trip is provided to trip the reactor when the pressure differential across the primary side of either steam generator decreases below a rate limited variable setpoint.

Initiation of MSGTR in GPWR

The steam generators (SGs) transfer heat from the reactor core, carried by the primary coolant, to the secondary side to produce steam for driving the turbine-generator. As shown in Figure 2, the GPWR steam generators are vertical U-tube heat exchangers equipped with an integral economizer, operating with reactor coolant on the tube side and secondary coolant on the shell side. Each steam generator in the GPWR model contains 13,102 U-tubes. In APR1400 steam generator design, these U-tubes were manufactured from thermally treated Alloy 690.

Water level within the SG is determined from differential pressure measurements obtained at two elevations. The simulator engine processes two primary level measurements: the narrow-range (NR) level and the wide-range (WR) level. In both cases, the upper pressure tap is located in the separator area. The lower tap for the NR measurement is positioned just below the U-tube bend area near the top of the downcomer, whereas the lower tap for the WR measurement is located at the bottom of the downcomer. The measured differential pressure, combined with the known distance between the taps, is used to infer the water level. The NR level serves as the primary control variable for feedwater flow regulation and is maintained approximately constant during normal operation by the control system. Significant deviations typically occur only during severe transients, such as rapid depressurization events (Šadek and Grgić 2017).

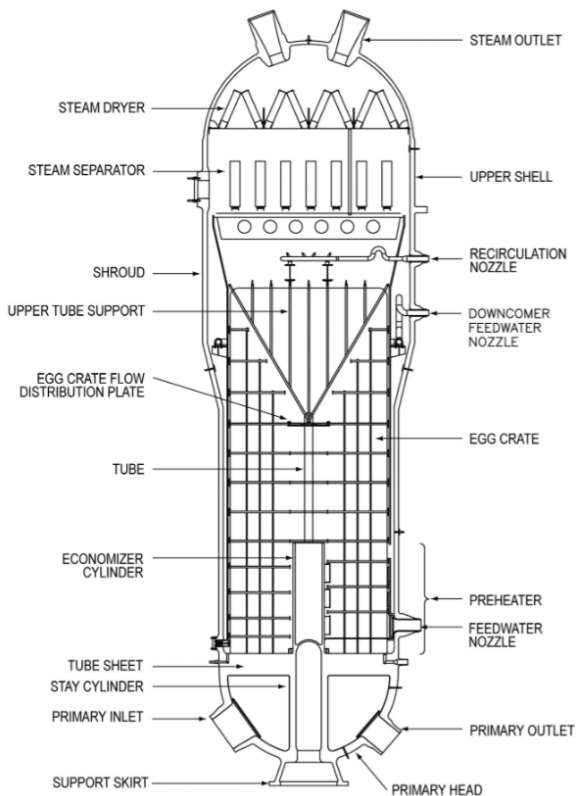


Figure 2: Cross-sectional diagram of the reference steam generator (IAEA 2011).

To evaluate the sensitivity of plant simulator response to accident magnitude, three distinct multi-tube rupture cases were executed inside Steam Generator 1 (SG1):

- Case 1 (Mild MSGTR): Simultaneous rupture of 5 tubes (0.038% of total SG tubes, corresponding to 11.24 cm² aggregate flow area).
- Case 2 (Moderate MSGTR): Simultaneous rupture of 10 tubes (0.076% of total SG tubes, corresponding to 22.49 cm² aggregate flow area).

- Case 3 (Severe MSGTR): Simultaneous rupture of 131 tubes (1.000% of total SG tubes, corresponding to 294.57 cm² aggregate flow area). Case 3 assumes an extreme operational scenario where tube inventory suffers uninspected material degradation or maintenance oversights.

To capture conservative thermal margins and peak fuel conditions, the transients were initiated from a steady-state full rated thermal power (100% nominal capacity) at beginning-of-life (BOL) cycle. The primary loop coolant was modeled with an initial boron concentration of 850.551 ppm. The GPWR simulator modeling architecture evaluates transient thermal-hydraulic, neutronic, and control system responses to specified boundary conditions. It does not incorporate time-dependent material degradation phenomena such as structural corrosion, wall thinning, or fatigue accumulation over time; thus, reactor operating age is not a modeled parameter in these transient simulations.

Tube ruptures were introduced at 20 seconds, and each case was simulated for four hours. To isolate the intrinsic physical behaviors of the coupled core, specifically the raw interaction between reactivity feedbacks, primary inventory depletion, and automatic safety features, no operator mitigating actions were permitted throughout the entire 4-hour transient window. The transient response and performance margins of the nuclear fuel were continuously evaluated using primary and secondary parameters. The primary core metrics include total reactor thermal power and maximum fuel centerline temperature, which directly map the equilibrium between internal heat generation and convective heat removal.

RESULTS AND DISCUSSION

Transient phase progression

The major transient events and chronological sequences for the simulated MSGTR accidents are outlined in Figure 3. The transient across all three cases is structurally divided into four distinct sequential phases: detection, temperature correction, trip actuation, and shutdown protection.

During Phase 1 (Detection), the instantaneous introduction of the tube rupture resulted in a high-velocity leak of radioactive primary coolant into the secondary circuit, localized within the SG blowdown system. Elevated radiation profiles were immediately registered by the process radiation monitors, generating an automatic Blowdown and Sample Process Isolation Signal (BSPIS) to restrict the environmental release of contaminated liquid. The initial drop in RCS pressure induced by the mass inventory leak was counteracted by the automatic activation of the pressurizer (PZR) heaters, which provided a localized thermal buffer that temporarily stabilized the depressurization rate (Figure 4a).

During automatic operation, the pressurizer level control system (PLCS) compares the actual pressurizer level with a programmed setpoint derived from the RCS average temperature (T_{avg}) as shown in Figure 5. Under nominal full-power steady-state conditions ($T_{avg} = 309.2^{\circ}\text{C}$), the controller maintains a programmed level setpoint of 53.5%. As shown in Figure 4b and Figure 6, Cases 1 and 2 maintained pressurizer inventory in alignment with this programmed level-temperature relationship because the makeup flow adequately compensated for the primary-to-secondary leak rate.

Conversely, for larger effective break areas causing an accelerated mass leak rate, the severe break flow rate rapidly overwhelmed the

makeup capacity of the charging system. In Case 3, while the plant was still operating at 100% thermal power ($T_{avg} = 309.2^{\circ}\text{C}$), the SG tube rupture forced the pressurizer water level to drop steeply below the 53.5% programmed setpoint, rapidly triggering a pressurizer low-level alarm at 48% (Figure 4b).

From an operational perspective, the primary-to-secondary leak rate during an MSGTR is not a fixed quantity; it dynamically evolves with the changing differential pressure between the RCS and the secondary side, local subcooling, and fluid density. In the control room, operators monitor this leak rate dynamically through RCS mass inventory balances - specifically via the net charging makeup flow required to compensate for primary inventory loss. Figure 4c illustrates the additional net makeup flow rate supplied by the normal charging system (calculated as total charging flow minus letdown flow). In Case 1, the net makeup flow rate gradually increased and stabilized near 25 gpm towards the end of the simulation, demonstrating that normal charging capacity was well-suited to compensate for the leak rate and maintain pressurizer level. In Case 2, the net makeup flow rate reached 110 gpm prior to reactor shutdown. In Case 3, the break flow rate was substantially higher and progressed rapidly, driving the net makeup flow rate to 180 gpm prior to the reactor trip. Because the automated PLCS adjusts makeup delivery to compensate for inventory loss, this additional water flow, drawn directly from the Volume Control Tank (VCT) as shown in Figure 4d, is directly proportional to the actual SG tube leak rate.

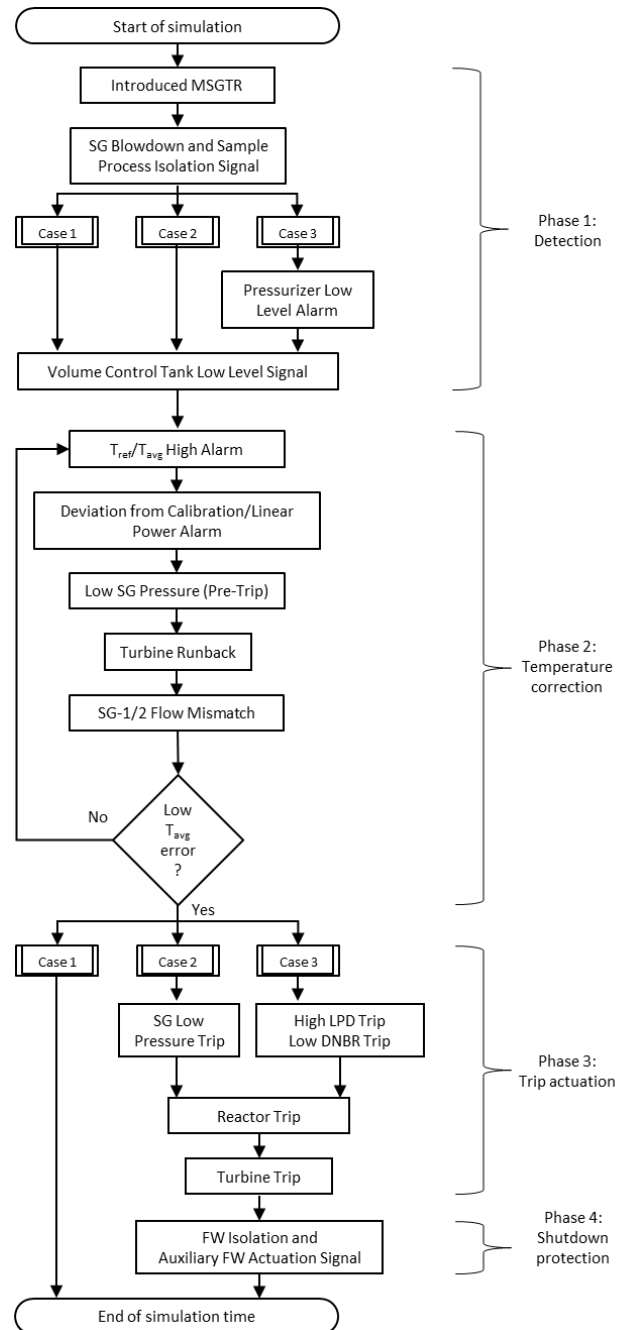


Figure 3: Comprehensive sequence of major protective events and operational phases during the MSGTR transient.

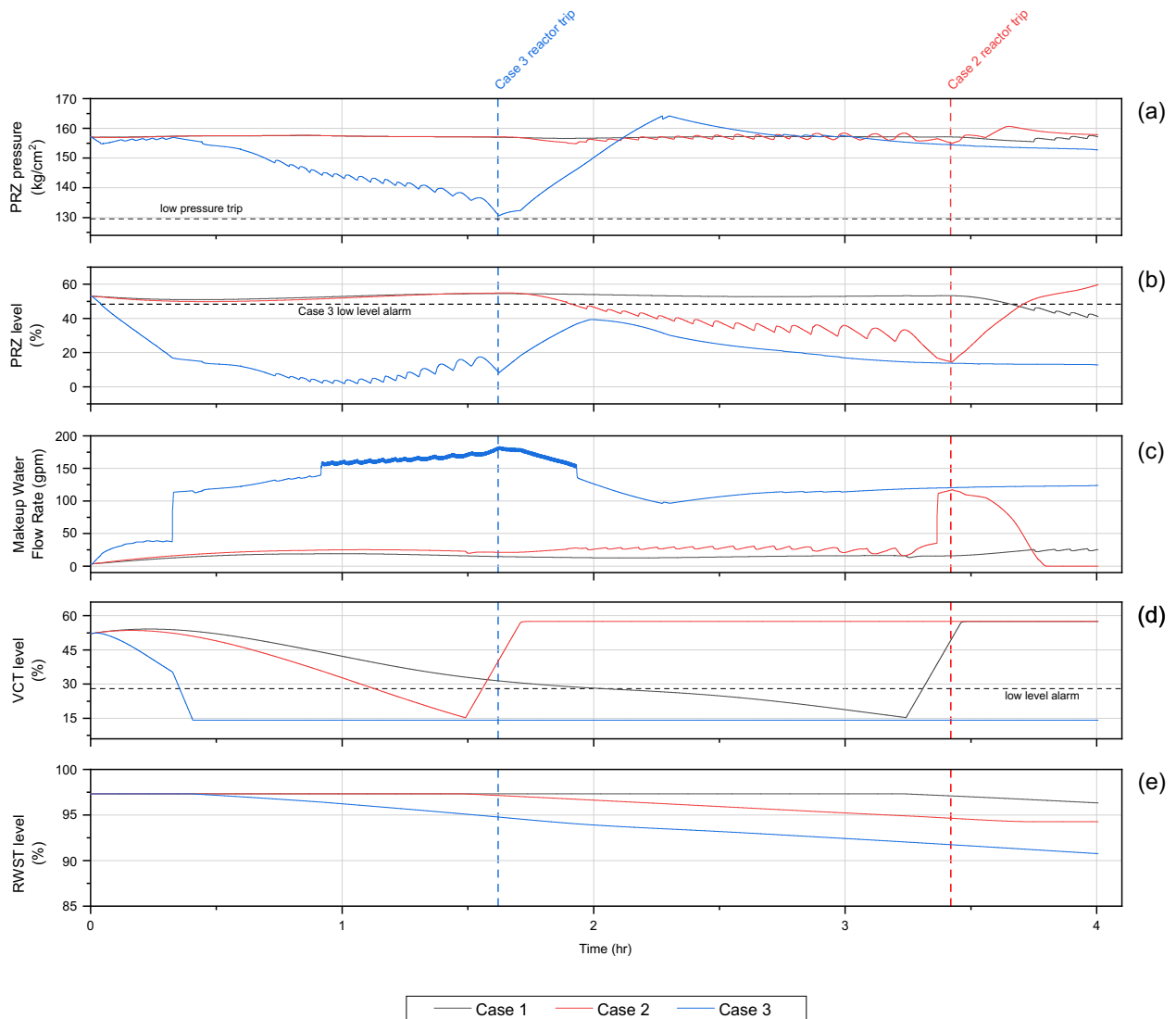


Figure 4: Pressurizer (a) pressure, and (b) level. Makeup water flow rate (c) and levels of (d) VCT and (e) RWST.

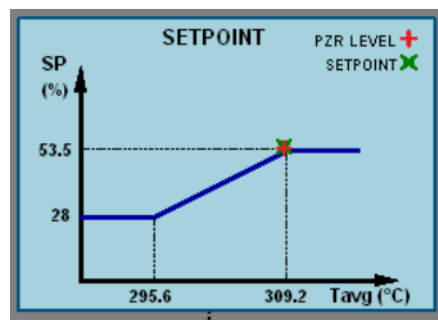


Figure 5: Pressurizer level controller provides a level program based on RCS average temperature.

Phase 2 (Temperature Correction) was initiated when the VCT inventory fell below the 15% low-low level threshold as shown in Figure 4d. This low-low level signal serves as an automatic interlock to protect the centrifugal charging pumps from losing suction head and cavitating. Upon reaching this level, the system automatically realigned the charging pump suction to draw borated water directly from the refueling water storage tank (RWST) as shown in Figure 4e. This automatic injection of borated makeup water introduced negative reactivity into the core, reducing the

reactor thermal power and causing a corresponding decline in the T_{avg} . The deviation of T_{avg} from the programmed turbine reference temperature (T_{ref}) triggered a dual correction response: automatic control rod insertion and the activation of the turbine runback sequence, both of which accelerated the reduction of reactor power and bulk coolant temperature as shown in Figure 6. Throughout this phase, a steam generator flow mismatch and low secondary pressure indicators were observed as the control systems attempted to maintain the T_{ref}/T_{avg} thermal balance.

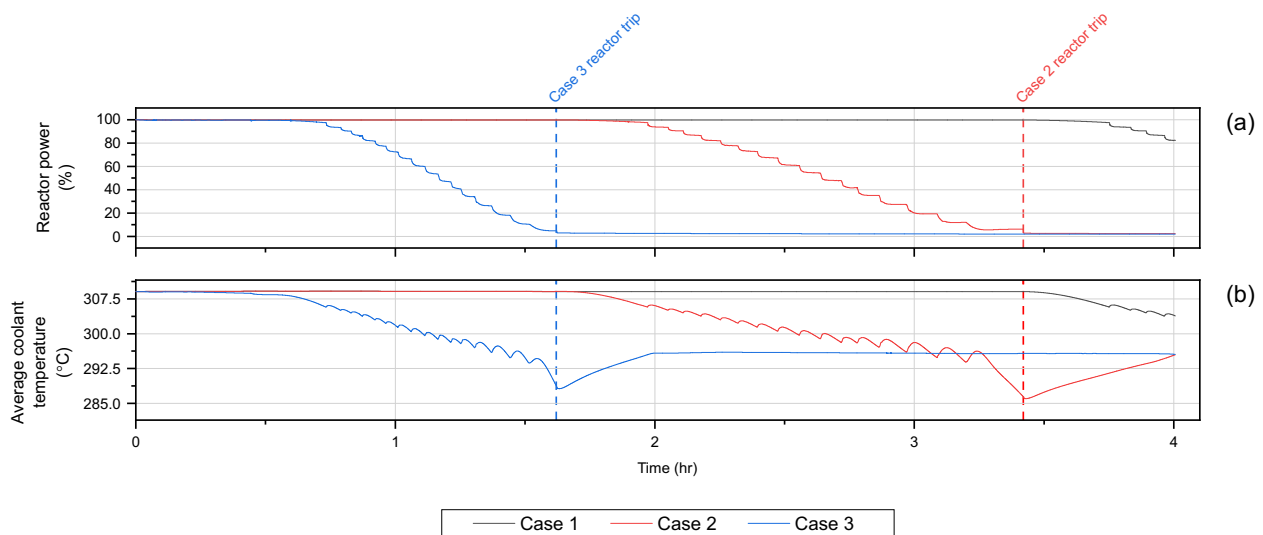


Figure 6: (a) Reactor power coast down, and (b) average coolant temperature during transient.

Phase 3 (Trip Actuation) began when the integrated control system identified a low T_{avg} thermal error that exceeded normal operational limits. The specific trip mechanisms and core behaviors diverged sharply based on the accident severity. In Case 1, the reactor power decreased gradually, causing a slow, controlled reduction in maximum fuel temperatures over several hours, with a projected reactor trip deferred near the sixth hour. In Case 2, power coastdown occurred more rapidly, resulting in an earlier reduction in the maximum fuel temperature. As the coolant inventory in the RCS decreased, the reactor experienced an automatic SG low-pressure trip at approximately 3.4 hours. In Case 3, the severe rupture generated localized power peaking and reduction in the thermal-hydraulic margin. This activated simultaneous High Local Power Density (LPD) and Low DNBR Trips at approximately 1.6 hours to protect the fuel rods from structural material degradation and cladding dryout. Immediately following the reactor trip in Cases 2 and 3, an automatic turbine trip was executed. Larger ruptures accelerated the reactor trips due to the increased rate of coolant mass loss. At the time of reactor trip, the reactor power had decreased to approximately 3% of nominal power in all three cases, the majority of the remaining power was generated by decay heat.

Phase 4 (Shutdown Protection) marked the long-term stabilization of the core, focusing on maintaining inventory control and managing decay heat removal. As the core shifted to subcritical conditions, the Feedwater Isolation Signal (FWIS) was

automatically initiated once T_{avg} fell below 295.5°C, closing all feedwater isolation, regulating, bypass, and chemical addition valves, while tripping the main feedwater pumps to prevent overcooling. Concurrently, the Auxiliary Feedwater (AFW) system activated, drawing water from the Condensate Storage Tank (CST) to ensure reliable secondary heat removal throughout the remainder of the 4-hour transient.

Fuel rod behavior during the transient

The transient behavior and performance margins of the nuclear fuel elements were monitored continuously across the four-hour window. Figure 7 illustrates the dynamic response of the maximum fuel centerline temperature across the three evaluation cases. The simulation demonstrated that the maximum fuel temperature decreased in close correlation with the reactor power coastdown. This coupling indicated that the transient thermal response of the fuel rods during an MSGTR was governed primarily by internal power reduction driven by negative reactivity feedbacks. Following the automatic reactor trip in the moderate and severe cases (Cases 2 and 3), the fuel temperature stabilized at approximately 300°C, indicating effective decay heat removal. By limiting the peak fuel temperatures, the system maintained conservative safety margins, completely mitigating the risks of localized core uncover or high-temperature material degradation under these severe design extension conditions.

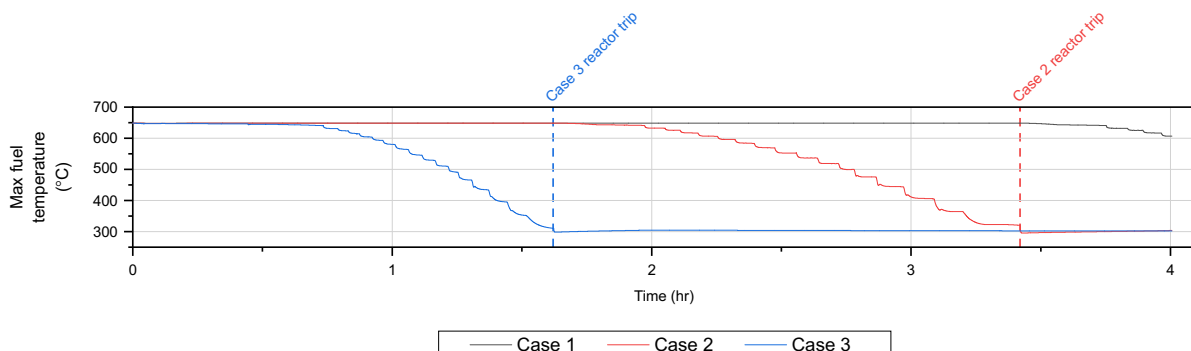


Figure 7: Transient evolution of the maximum fuel temperature under varying MSGTR severities.

Secondary-side vulnerabilities and operator diagnostic aids

Given that the simulation was run without operator intervention during the safe shutdown of the reactor and fully relied on the automatic action of the simulator, it is worth emphasizing the

importance of operator analysis during the post-trip mitigation strategies of the accident. In standard emergency operating guidelines, identifying the specific leaking steam generator is paramount for accident isolation. Thermally, a primary-to-

secondary tube rupture should cause the shell-side pressure of the affected steam generator (SG1) to rise above that of the intact unit (SG2). However, the raw pressure trends displayed on the simulator's control panels provided negligible diagnostic utility, as the visible differences between the two units remained entirely indistinguishable during the power operation phase as shown in Figure 8. To resolve this, a differential pressure indicator ($\Delta p_{SG} =$

$p_{SG2} - p_{SG1}$) was introduced, where p_{SG1} and p_{SG2} are the pressures of SG1 and SG2, respectively. However, even this metric demonstrated minimal variance and maintained a positive bias for SG2, only shifting to a negative differential after the reactor trip when auxiliary feedwater flow patterns altered the secondary shell dynamics as shown in Figure 9.

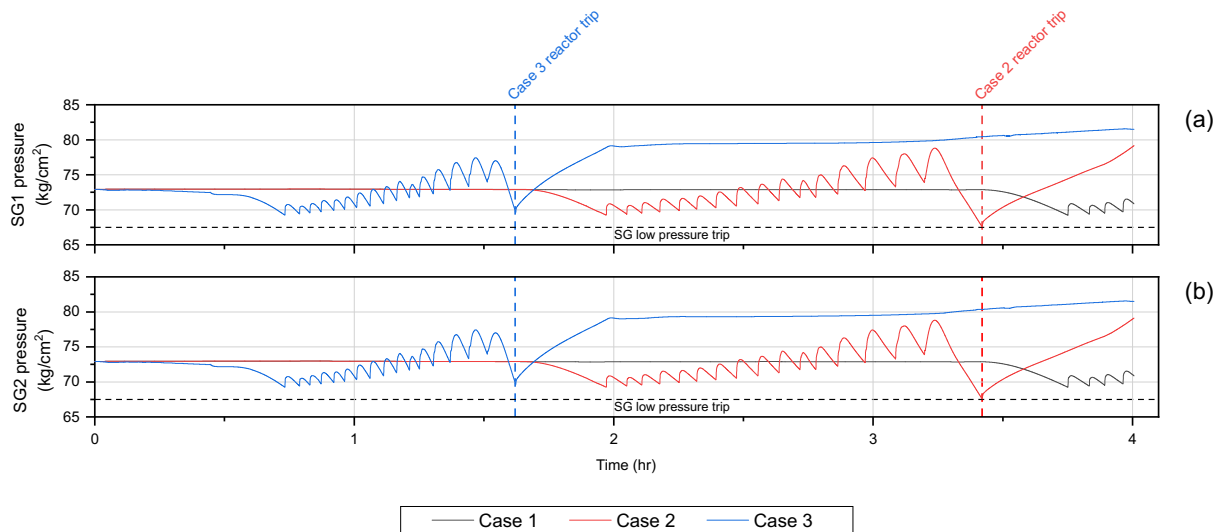


Figure 8: Pressure trends of SG1 and SG2 are indistinguishable.

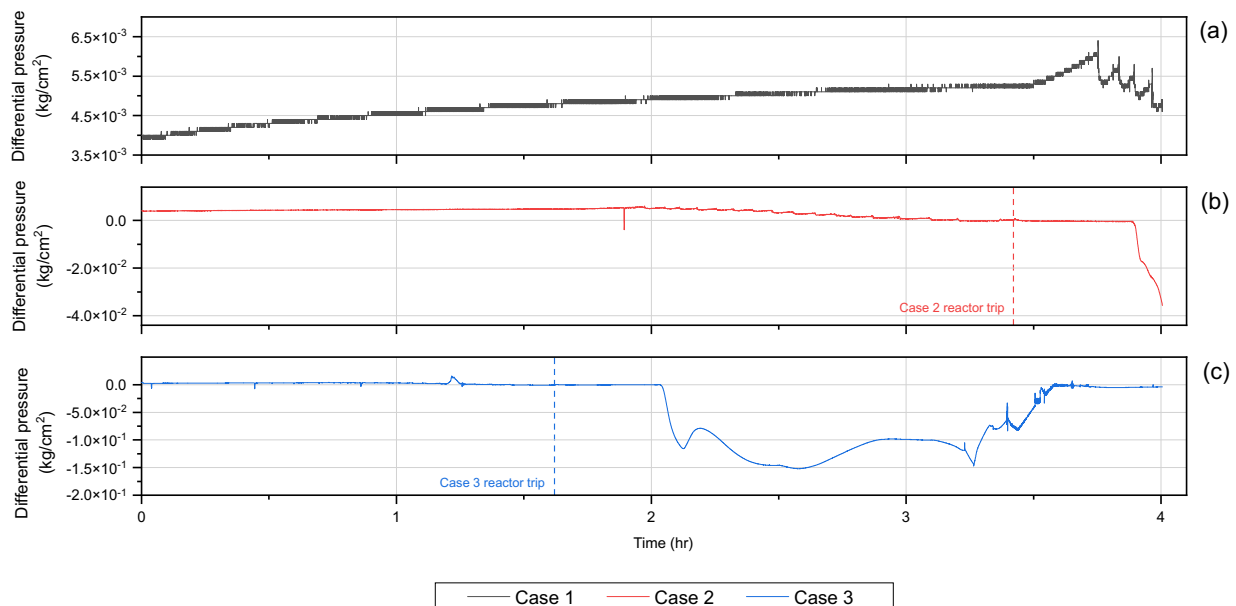


Figure 9: Differential pressure of SG1 and SG2.

A more reliable diagnostic indicator is the shell-side water level, which rises in the leaking steam generator due to continuous primary mass transport. While standard NR level trends exhibited minimal detectable differences during the initial phases of the transient as shown in Figure 10, the unmitigated long-term operational sequence resulted in an uncontrolled level surge that eventually saturated the narrow-range indicator at 100%. This complete water flooding caused liquid coolant to spill directly into the main steam lines. This secondary consequence introduced operational risks, including potential downstream equipment damage. This severe flooding scenario was driven directly by the

continuous, unmitigated injection of the auxiliary feedwater system, which remained active without operator actions.

To address this diagnostic limitation and provide an early indicator for leaking SG isolation, a differential level metric ($\Delta l_{SG} = l_{SG2} - l_{SG1}$), where l_{SG1} and l_{SG2} are the water levels in SG1 and SG2, respectively, was proposed as illustrated in Figure 11. The detailed inset plots capturing the first 0.1 hours (6 minutes) of the transient demonstrated that the differential level dropped into the negative domain immediately following accident initiation across all three cases. This negative deviation provided a diagnostic signature that indicated an accelerated water level increase within SG1 relative to the intact SG-2. Implementing this differential level algorithm within advanced control room safety parameter display systems

would enable operators to rapidly identify, isolate, and terminate feedwater flow to the damaged steam generator.

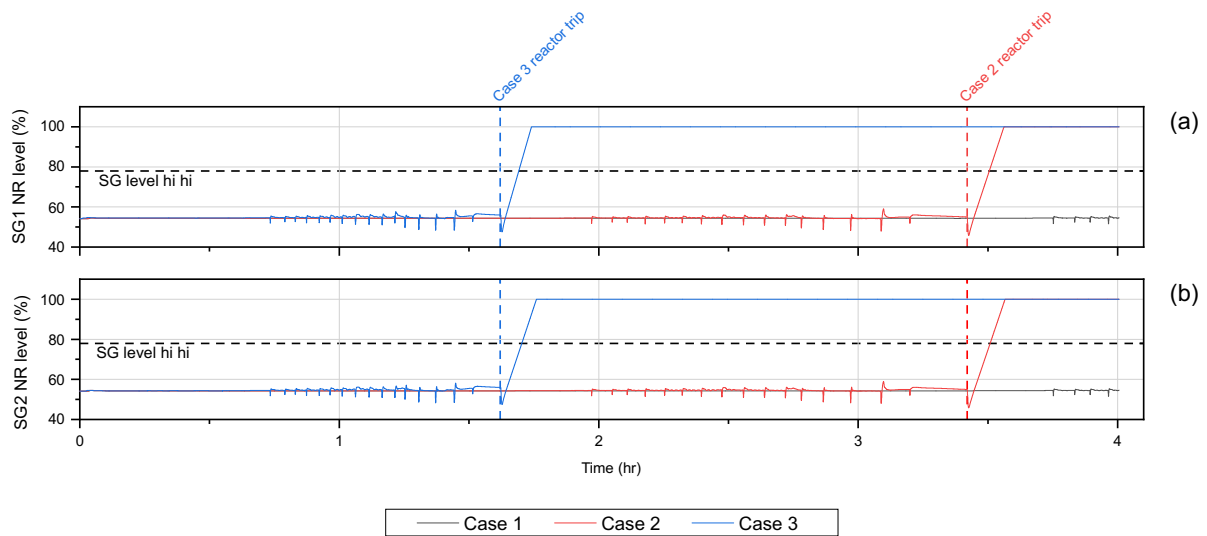


Figure 10: Steam generator NR level.

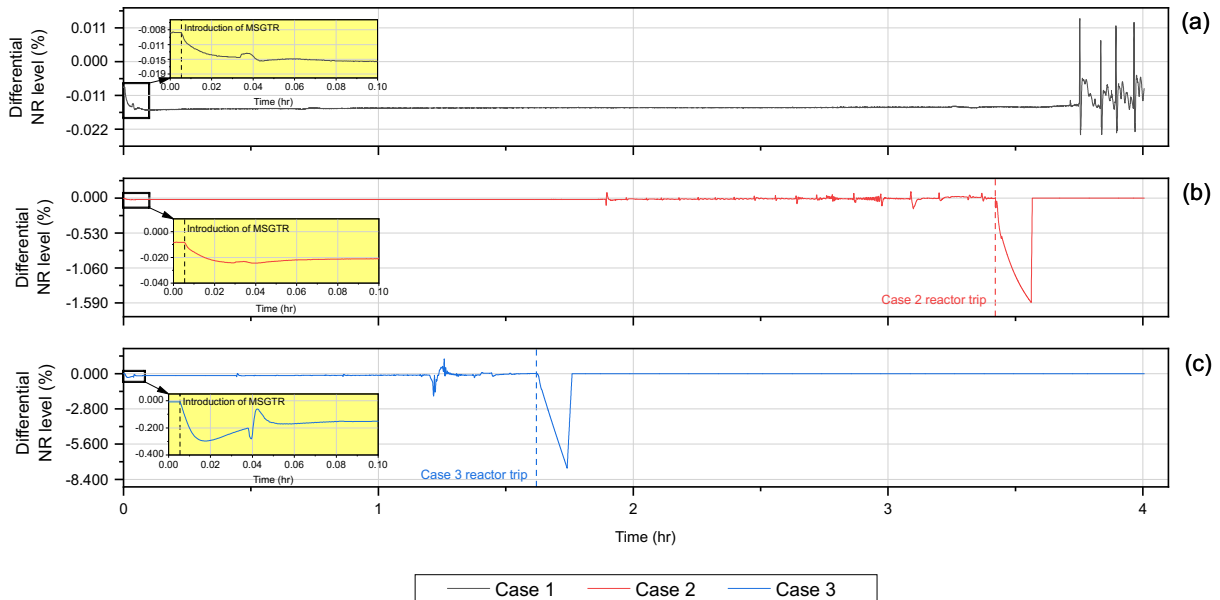


Figure 11: Differential NR level response with early-transient inset plots.

CONCLUSION

This study utilized a 1400 MWe Generic PWR simulator to systematically chart the transient plant performance during an unmitigated Multiple Steam Generator Tube Rupture (MSGTR) accident of varying severities.

The plant safety systems effectively moderated the transients. The pressurizer mitigated pressure fluctuations, while the volume control tank mitigated changes in coolant inventory. Simultaneously, the refueling water storage tank supplied borated water, introducing negative reactivity and reducing reactor power and average coolant temperature. At the time of reactor trip, reactor power had decreased to approximately 3% of nominal power in all three cases, the majority of the remaining power was generated by decay heat.

Across all evaluated break sizes, the maximum fuel temperature remained closely correlated with the reactor power coastdown. The

fuel thermal response was governed primarily by negative reactivity feedback associated with automatic borated water injection, resulting in a gradual temperature decrease and long-term stabilization at approximately 300°C.

The operational timeline and trip mechanisms exhibited a strong dependence on the total number of ruptured tubes. Mild cases (5 tubes) experienced a prolonged power decay over several hours, moderate cases (10 tubes) triggered an automatic secondary low-pressure trip at approximately 3.4 hours, and severe cases (131 tubes) induced rapid, simultaneous High LPD and Low DNBR trips at approximately 1.6 hours to prevent fuel degradation.

Leaving the safety systems unmitigated resulted in complete shell-side flooding of the steam generators due to continuous auxiliary feedwater injection, introducing operational risks, including potential damage to downstream equipment. To resolve standard control panel diagnostic limitations, a differential level indicator

was proposed to be an effective operator aid for the rapid identification of the leaking steam generator.

In summary, the integrated simulator model demonstrates that the plant automatic protection systems possess excellent inherent resilience for maintaining core cooling and fuel stability under severe Design Extension Conditions without immediate operator intervention.

ACKNOWLEDGMENTS

The author would like to acknowledge the funding support for the NuSIM project provided by the Department of Science and Technology – Grants-in-Aid (DOST-GIA) Program. The author gratefully acknowledges Ms. Rothana Hor for her valuable assistance and insights in the creation of the manuscript.

CONFLICTS OF INTEREST

The author declares that there is no conflict of interest.

USE OF AI

OpenAI GPT-5.5 Instant was used for language editing and text organization only. The author developed all scientific content, including the methodology, analysis, and conclusions. AI was not used to generate, modify, or manipulate data, figures, or results.

CONTRIBUTIONS OF INDIVIDUAL AUTHORS

RF Menor Jr: Conceptualization, Methodology, Software, Formal Analysis, Data Curation, Validation, Writing – Original Draft, Writing – Review & Editing, Visualization.

REFERENCES

- Abdellatif HH, Bhowmik PK, Arcilesi D, and Sabharwal P. Accident event progression, gaps, and key performance indicators for steam generator tube rupture events in water-cooled SMRs: A review. *Progress in Nuclear Energy* 2024; 168:105021. <https://doi.org/10.1016/j.pnucene.2023.105021>
- Almomani B, Ababneh A, and Zubair M. Study on EOPs for PORV-Break LOCA in Generic PWR Simulator. *Nuclear Technology*. 2023; 209(2): 214-227. <https://doi.org/10.1080/00295450.2022.2133507>
- Almomani B, Khuwailah BA, and Alam SB. Machine Learning-Based Classification and Regression Approach for Early Detection of Large Break Loss-of-Coolant Accident Conditions. *Arabian Journal for Science and Engineering*. 2025; 50(22): 18971-18991. <https://doi.org/10.1007/s13369-025-10190-1>
- Bae JH, Jeon SS, Bang J, and Hwang DH. Operator actions evaluation for MSGTR accident mitigation. *Annals of Nuclear Energy* 2023; 182: 109623. <https://doi.org/10.1016/j.anucene.2022.109623>
- Bang J, Choi GH, Jerng DW, Bae SW, Jang S, and Ha SJ. Analysis of steam generator tube rupture accidents for the development of mitigation strategies. *Nuclear Engineering and Technology* 2022; 54(1): 152–161. <https://doi.org/10.1016/j.net.2021.07.032>
- Cho SG, Choi J, Shin JH, and Lee SJ. Multi-Abnormality Attention Diagnosis Model Using One-vs-Rest Classifier in a Nuclear Power Plant. *Journal of Nuclear Engineering* 2023; 4(3): 467–483. <https://doi.org/10.3390/jne4030033>
- Chung SI, Lee JH, Jeon WJ, Choi YK, and Choi JS. Reviewing the possibility of TLOFW accident scenario using GPWR. In: *Transactions of the Korean Nuclear Society Autumn Meeting*. 2023.
- Dzień S, and Diab A. Safety evaluation of Multiple Steam Generator Tube rupture accident using the best estimate plus uncertainty approach. *Nuclear Engineering and Design* 2024; 422. <https://doi.org/10.1016/j.nucengdes.2024.113150>
- Hrušković J, Park Y, Lee DY, Jeon SS, and Hwang DH. Analysis of a MSLB-induced Steam Generator Tube Rupture Accident for APR1400. In: *Transactions of the Korean Nuclear Society Autumn Meeting*. 2023.
- International Atomic Energy Agency (IAEA). Assessment and Management of Ageing of Major Nuclear Power Plant Components Important to Safety: Steam Generators, TECDOC Series – 1668. Vienna, Austria: IAEA, 2011.
- International Atomic Energy Agency (IAEA). Advancing the State of the Practice in Uncertainty and Sensitivity Methodologies for Severe Accident Analysis in Water Cooled Reactors of PWR and SMR Types, TECDOC Series – 2031. Vienna, Austria: IAEA, 2023.
- International Atomic Energy Agency (IAEA). Design Extension Conditions and the Concept of Practical Elimination in the Design of Nuclear Power Plants, Safety Standards Series No. SSG-88. Vienna, Austria: IAEA, 2024. <https://doi.org/10.61092/iaea.la1m-dy8m>
- Israel S. EPR: Steam Generator Tube Rupture analysis in Finland and in France. In: *Eurosafe - Towards Convergence of Technical Nuclear Safety Practices in Europe*. 2006.
- Jia B, Jing J, Qiao X, and Zhang C. Calculations for AP1000 nuclear power plant SGTR accident based on RELAP5/mod 3.3 program. *Journal of Computational Methods in Sciences and Engineering* 2015; 15(1): 81–90. <https://doi.org/10.3233/JCM-150522>
- Jeon WJ, Lee JH, Chung SI, Choi YK, and Choi JS. Analysis of SBLOCA with SI Fail using GPWR and review of the possibility of using GPWR. In: *Transactions of the Korean Nuclear Society Autumn Meeting*. 2023.
- Jeong JH, and Choi KY. Effects of tube rupture modeling and the parameters on the analysis of multiple steam generator tube rupture event progression in APR1400. *Nuclear Engineering and Design* 2003; 224(3): 313–336. [https://doi.org/10.1016/S0029-5493\(03\)00132-8](https://doi.org/10.1016/S0029-5493(03)00132-8)
- Jeong JH, Chang KS, Kim SJ, and Lee JH. Best-Estimate Analysis of MSGTR Event in APR1400 Aiming to Examine the Effect of Affected Steam Generator Selection. *Journal of the Korean Nuclear Society* 2002; 33(4): 358–369.
- Jeong JH, Kim SJ, Chang KS, and Lee JH. Analysis of MSGTR Events for APR1400 by Means of Best Estimate Thermal-Hydraulic System Code. In: *Proceedings of the Korean Nuclear Society Spring Meeting*. 2001.

- Jo W, and Jun Lee S. Bayesian belief network-based human reliability analysis methodology for start-up and shutdown operations in nuclear power plants. *Annals of Nuclear Energy* 2022; 179: 109403. <https://doi.org/10.1016/j.anucene.2022.109403>
- Kang D, Lee S, and Chung B. SGTR Event Analysis of APR1400 on the Effects of Multi-D Modeling Compared with 1D Modeling using MARS Code. In: Transactions of the Korean Nuclear Society Spring Meeting. 2006.
- KEPCO, and KHNP. APR1400 Design Control Document Tier 2: Chapter 1 Introduction and General Description of the Plant APR1400-K-X-FS-14002-NP Revision 0. 2014.
- Kim JM, Lee G, Lee C, and Lee SJ. Abnormality diagnosis model for nuclear power plants using two-stage gated recurrent units. *Nuclear Engineering and Technology* 2020; 52(9): 2009–2016. <https://doi.org/10.1016/j.net.2020.02.002>
- Kim JM, Shin JH, and Koo SR. Data-Driven Behavioral Clustering for Feature Reduction in Nuclear Power Plant Operational Time Series. In: Transactions of the Korean Nuclear Society Autumn Meeting. 2025.
- Kondo S. Lessons learned for PSA from the SGTR incident at Mihama, unit 2, in 1991. *Reliability Engineering and System Safety* 1994; 45(1–2): 57–65. [https://doi.org/10.1016/0951-8320\(94\)90076-0](https://doi.org/10.1016/0951-8320(94)90076-0)
- Lee G, Lee SJ, and Lee C. A convolutional neural network model for abnormality diagnosis in a nuclear power plant. *Applied Soft Computing* 2021; 99: 106874. <https://doi.org/10.1016/j.asoc.2020.106874>
- Lin Y, and Ma Q. Data-Driven Fault Diagnosis Method for Nuclear Power Plants LOCA Based on GADF-MC-CNN Enhanced with Memory-Based Causal Multi-Head Attention Mechanism. SSRN. Preprint posted online 2026. <https://doi.org/10.2139/ssrn.6818903>
- Martin K, Martinez-Quiroga V, and Freixa J. Full scope scaling analysis and recommendations for PWR SGTR scenarios. *Nuclear Engineering and Design* 2024; 418: 112864. <https://doi.org/10.1016/j.nucengdes.2023.112864>
- Mascari F, D'Auria F, Bestion D, Lien P, Nakamura H, Austregesilo H, Moon SK, Reventos F, Umminger K, Reyes JN, and Rohatgi US. OECD/NEA/CSNI state-of-the-art report on scaling in system thermal-hydraulics applications to nuclear reactor safety and design (The S-SOAR). *Nuclear Engineering and Design* 2024; 416: 112750. <https://doi.org/10.1016/j.nucengdes.2023.112750>
- Mena P, Borrelli RA, and Kerby L. Nuclear Reactor Transient Diagnostics Using Classification and AutoML. *Nuclear Technology*. Published online 2021. <https://doi.org/10.1080/00295450.2021.1905470>
- Mena P, Borrelli RA, and Kerby L. Expanded analysis of machine learning models for nuclear transient identification using TPOT. *Nuclear Engineering and Design*. 2022; 390: 111694. <https://doi.org/10.1016/j.nucengdes.2022.111694>
- Nam K, Kwon K, and Kim A. HINT-Sec: Hardware-in-the-loop nuclear power plant testbed for cyber security. *Progress in Nuclear Energy*. 2025; 180: 105600. <https://doi.org/10.1016/j.pnucene.2024.105600>
- Rahman SMT, Joarder MSA, Hashan AM, Rahman SMM, and Ali RH. Holistic Approach to Safety and Operational Stability: Analyzing VVER-1200 Reactor Dynamics in SGTR and AC Power Loss. *Journal of Power and Energy Engineering* 2024; 12(09): 102–130. <https://doi.org/10.4236/jpee.2024.129007>
- Šadek S, and Grgić D. Operation and Performance Analysis of Steam Generators in Nuclear Power Plants. In: Murshed SMS, and Lopes MM. eds. *Heat Exchangers - Advanced Features and Applications*. InTech, 2017: 167–191. <https://doi.org/10.5772/66962>
- Seo YH, Jeong WS, and Moon YT. Equipment Survivability Assessment in APR1400. In: Transactions of the Korean Nuclear Society Autumn Meeting. 2012:734–735.
- Shin BS, and Hwang SH. Best Estimate Analysis for MSGTR Accidents in APR1400 Considering Control Systems. In: NENE 2018 - 27th International Conference Nuclear Energy for New Europe. 2018:318.1–318.8.
- Spisak S, and Diab A. AI-Assisted Forecasting of a Mitigated Multiple Steam Generator Tube Rupture Scenario in a Typical Nuclear Power Plant. *Energies* 2025; 18(2): 250. <https://doi.org/10.3390/en18020250>
- Sui D, Fan R, Lu D, Shang C, and Wei Y. Analysis of single failure criteria and operator action effect for different SGTR mitigation strategies of HPR1000. *Annals of Nuclear Energy* 2020; 136: 107035. <https://doi.org/10.1016/j.anucene.2019.107035>
- U.S. Nuclear Regulatory Commission (USNRC). SECY-93-087 Policy, technical, and licensing issues pertaining to evolutionary and advanced light- water reactor (ALWR) designs. 1993.
- Western Service Corporation (WSC). 3KEYMASTER Generic PWR Simulator: training guide. 2025.
- Xiao X, Qi B, Liang J, Tong J, Deng Q, and Chen P. Enhancing LOCA Breach Size Diagnosis with Fundamental Deep Learning Models and Optimized Dataset Construction. *Energies*. 2024; 17(159): 1-20. <https://doi.org/10.3390/en17010159>
- Yoon BK, Song MJ, Park CE, and Lee MH. Analysis of Multiple Steam Generator Tube Rupture with Emergency Operating Guidelines for APR1400. In: Transactions of the Korean Nuclear Society Virtual Autumn Meeting. 2021.
- Zubair M, and Akram Y. Utilizing MATLAB machine learning models to categorize transient events in a nuclear power plant using generic pressurized water reactor simulator. *Nuclear Engineering and Design*. 2023; 415: 112698. <https://doi.org/10.1016/j.nucengdes.2023.112698>
- Zubair M, and Akram Y. Enhancement in the safety and reliability of Pressurized Water reactors using Machine Learning approach. *Annals of Nuclear Energy*. 2024; 201: 110448. <https://doi.org/10.1016/j.anucene.2024.110448>
- Zubair M, and Bibi A. Enhancing safety of nuclear power plant by human error detection and identification through AI techniques. *Results in Engineering*. 2025; 27: 106616. <https://doi.org/10.1016/j.rineng.2025.106616>
- Zubair M, Ababneh A, and Ishag A. Station black out concurrent with PORV failure using a Generic Pressurized Water Reactor simulator. *Annals of Nuclear Energy* 2017; 110: 1081–1090. <https://doi.org/10.1016/j.anucene.2017.08.023>

Zubair M, Samir Abushady M, and Hussain Alzarooni A.
Classification of LOCA events in pressurized water reactors
using neural networks. Annals of Nuclear Energy. 2026; 227:
111909. <https://doi.org/10.1016/j.anucene.2025.111909>