Verification and research of SGTR accident based on the PKL test facility

Authors: Zhang, Mr. Dechen, Prof. Xiangbin Li, Xiaojun Qiu, Liu, Prof. Yusheng, Wang, Chengshen, Prof. Xiangbin Li

Date: 2025-09-23T00:00:00+00:00

Abstract

To assess the steam generator tube rupture (SGTR) accident scenario under compromised safety systems (e.g., post-earthquake conditions), as specified in the OECD/NEA/CSNI International Standard Problem (ISP) No. 52 project, a full-scale facility model was developed using the RELAP5 thermal-hydraulic code for numerical simulation. The computational results were systematically validated against experimental data obtained from the PKL test facility. The analysis demonstrates that RELAP5 accurately reproduces the primary progression of the SGTR accident, with the simulated accident sequence and key thermal-hydraulic parameters exhibiting strong agreement with empirical observations. During the early phase of the transient process, depressurization of the primary circuit facilitates secondary-side coolant ingress into the reactor core, thereby maintaining adequate core cooling throughout the initial and intermediate stages. Subsequent depletion of the steam generator inventory results in partial core uncovery, precipitating a rapid escalation in core exit temperature.However, the coordinated actuation of the high-pressure safety injection (HPSI) system, accumulator (ACC), and low-pressure safety injection (LPSI) system effectively preserves core integrity, ultimately enabling a transition to the long-term cooling phase.These findings confirm that adherence to appropriate operational protocols can ensure reactor safety even under degraded safety system conditions.

Full Text

Preamble

Verification and Research of SGTR Accident Based on the PKL Test Facility

Dechen Zhang[1], Xiangbin Li[1] (Corresponding author), Xiaojun Qiu[1], Yusheng Liu[2], Chengshen Wang[1]

[1] School of Nuclear Science and Engineering, North China Electric Power University, Beijing 102206, China
[2] [Affiliation not provided]

Abstract

To assess steam generator tube rupture (SGTR) accident scenarios under compromised safety system conditions (e.g., post-earthquake), as specified in the OECD/NEA/CSNI International Standard Problem (ISP) No. 52 project, a full-scale facility model was developed using the RELAP5 thermal-hydraulic code for numerical simulation. Computational results were systematically validated against experimental data obtained from the PKL test facility. The analysis demonstrates that RELAP5 accurately reproduces the primary progression of SGTR accidents, with simulated accident sequences and key thermal-hydraulic parameters exhibiting strong agreement with empirical observations. During the early transient phase, primary circuit depressurization facilitates secondary-side coolant ingress into the reactor core, thereby maintaining adequate core cooling throughout the initial and intermediate stages. Subsequent depletion of the steam generator inventory results in partial core uncovery, precipitating a rapid escalation in core exit temperature. However, coordinated actuation of the high-pressure safety injection (HPSI) system, accumulator (ACC), and low-pressure safety injection (LPSI) system effectively preserves core integrity, ultimately enabling transition to long-term cooling. These findings confirm that adherence to appropriate operational protocols can ensure reactor safety even under degraded safety system conditions.

Keywords: SGTR, PKL, RELAP5, Numerical simulation

Abbreviations:
SGTR -Steam Generator Tube Rupture
OECD -Organization for Economic Co-operation and Development
CSNI -Committee on the Safety of Nuclear Installations
NEA -Nuclear Energy Agency
ISP -International Standard Problem
RELAP -Reactor Excursion and Leak Analysis Program
HPSI -High Pressure Safety Injection System
ACC -Accumulator
LPSI -Low Pressure Safety Injection System
CET -Core Exit Temperature
SOT -Start of Test
EOT -End of Test

1. Introduction

Steam generator tube failure incidents have been documented during long-term commercial operation of pressurized water reactors (PWRs). In response, the nuclear industry has implemented a range of mitigation strategies to minimize occurrence likelihood, including enhancements in steam generator design, rigorous monitoring of secondary-side water chemistry, and adoption of advanced non-destructive examination techniques. While these improvements have substantially increased steam generator reliability, extreme external events may still induce tube ruptures when mechanical stresses surpass design-basis limits. Consequently, SGTR accidents and their resultant thermal-hydraulic effects on the reactor coolant system remain critical areas of experimental research.

The ISP program represents a collaborative multinational framework wherein participating organizations employ standardized experimental facility data to benchmark reactor safety analysis codes, thereby evaluating predictive validity and computational accuracy. Building upon preceding initiatives, OECD/NEA/CSNI endorsed ISP-52, which leverages experimental data from the PKL III J5.1 experiment. As a continuation of the OECD/NEA-PKL4 project research scope, ISP-52 specifically investigates thermal-hydraulic phenomena under design extension conditions, including: two-phase flow dynamics during medium-break loss-of-coolant accidents; temporal evolution of "cold dome" configurations across varied accident sequences; and transient thermal-hydraulic responses during small-break LOCAs occurring at the reactor pressure vessel head. Additionally, the project validates innovative safety technologies through two key assessments: core cooling efficacy under localized flow blockage scenarios at the reactor pressure vessel flow distribution plate, and operational performance of passive residual heat removal systems.

Extensive prior research has established comprehensive understanding of experimental configurations and simulation parameters for the PKL test facility and SGTR accident scenarios. Umminger et al. reviewed PKL experiments on PWR accident thermal-hydraulic behavior, covering LOCA, boron dilution, and main steam line break scenarios, with findings that validated core cooling mechanisms and provided data for code verification, emergency procedure development, and operator training. Al-Yahia et al. simulated passive safety condenser performance during station blackout (SBO) at PKL/SACO using TRACE, finding it effectively removed core decay heat and ensured safety within 72 hours. Bousbia Salah simulated natural circulation experiments at PKL using CATHARE, comparing 1D single/multi U-tube and 3D inlet/inlet-outlet multi U-tube models, with results showing the 1D multi U-tube model provided better overall predictions while 3D modeling better reproduced key physical phenomena. Carlos et al. simulated residual heat removal system (RHRS) failure during refueling shutdown at PKL using RELAP5/MOD3.3, demonstrating the code' s predictive capability for mid-loop conditions while identifying needs for improved modeling of complex flow and boron concentration. Mukin et al. simulated OECD/NEA PKL-3 SBO experiments using TRACE, showing that secondary-side depressurization provided limited cooling during SG drain-out while primary-side depressurization with accumulator injection delayed core overheating. Freixa et al. simulated three SBO scenarios under different boundary conditions at PKL using RELAP5, reproducing thermal-hydraulic phenomena well but exhibiting event sequence deviations attributed to uncertainties in heat loss estimation, pressurizer entrainment, condensation modeling, and boundary conditions. Martinez-Quiroga et al. summarized semi-blind benchmark results from 16 international institutions using system codes for OECD/NEA PKL-4 intermediate-break LOCA experiments, demonstrating reasonable prediction capability for key phenomena like break flow and natural circulation termination while observing discrepancies in complex scenarios involving emergency core cooling bypass, countercurrent flow limitation, and condensation modeling. Freixa et al. simulated IBLOCA experiments from OECD/NEA PKL-4 using RELAP5/MOD3.3 and validated the methodology via best-estimate-plus-uncertainty analysis, accurately predicting key parameters like core uncovery time and peak cladding temperature with most experimental results residing within uncertainty bands. Jonnet et al. simulated a main steam line break accident at PKL using RELAP5/MOD3.2, reproducing key accident phenomena well though calculated minimum coolant temperatures in the core exceeded experimental values and SG-2 depressurization was slower than observed.

Xu et al. simulated the PKL I2.2 IBLOCA experiment from OECD/PKL4 using ATHLET and developed a two-layer sensitivity analysis method, finding the critical flow model exhibited high sensitivity for most response parameters, particularly affecting secondary pressure and peak cladding temperature. Yang et al. simulated SBLOCA superposed with SBO using RELAP5 to investigate primary circuit characteristics under steam bypass discharging system operation, showing the system played a critical role in the initial phase by effectively reducing primary circuit pressure through rapid steam discharge. Bryk et al. performed parametric studies on PKL SG heat transfer mechanisms during RHRS failure under PWR cooldown conditions with nitrogen/steam/liquid multiphase presence, finding coolant inventory variations significantly affected collapsed liquid level distribution and condensation zones, thereby influencing core temperature and pressure stability. Umminger et al. investigated natural circulation recovery after SBLOCA in PKL using S-RELAP5, observing that despite symmetric ECC injection, natural circulation exhibited significant asymmetries across loops. Sui et al. simulated SGTR accidents in HPR1000 using RELAP5, comparing two mitigation strategies and concluding that balancing steam generator overflow risk against radioactive release was essential. Martin et al. simulated SGTR accidents using RELAP5 based on NEA/OECD ROSA-2 LSTF Test 4 and scaled the model with a power-volume scaling tool, identifying the inconsistent length-to-diameter ratio of SGTR components as the primary cause of scaling distortion.

In summary, numerical simulation methodologies for SGTR accident analysis have garnered significant research interest and demonstrated considerable advancement. Nevertheless, coupled parameter interactions between residual primary and secondary circuits under degraded safety system conditions necessitate further systematic investigation. This study develops a comprehensive RELAP5 model of the PKL test facility to investigate SGTR accident progression with restricted safety systems. Through numerical simulation of SGTR scenarios validated against experimental data, this work characterizes transient responses of critical thermal-hydraulic parameters, including system pressures, mass flow rates, temperature distributions, and differential pressures across key components.

2.1 Test Facility Description

The PKL test facility simulates the nuclear steam supply system of a Siemens KWU-designed 1300 MWe pressurized water reactor, with the German Philippsburg Unit 2 nuclear power plant serving as the reference plant. The facility employs scaling ratios of 1:1 in height, 1:12 in diameter, and 1:145 in volume/power. With a maximum power capacity of 2.5 MW (equivalent to 10% of scaled full power), it operates at primary/secondary circuit pressures up to 46 bar and 60 bar, respectively. The electrically heated core comprises 314 fuel rod simulators, each featuring three independently controlled power zones. The PKL test facility comprehensively models the reactor primary coolant system and replicates all relevant safety and auxiliary systems, including HPSI, ACC, LPSI, and emergency feedwater system. Critical secondary-side components such as steam generators, main steam lines, and feedwater piping are also integrated.

2.2 Nodalization Scheme

The overall nodalization of the PKL test facility is shown in Figure 1 [Figure 1: see original paper]. The model simulates the complete primary circuit system, including the pressure vessel and four symmetrically arranged independent loops. Each loop comprises one steam generator, cold leg, hot leg, and associated valves and piping. The surge line connects to the pressurizer at the mid-section of the hot leg of Loop 2. For the secondary side, the downcomer, steam separator, feedwater line, steam discharge line, steam discharge volume, and overpressure protection system are modeled. Regarding safety systems, the HPSI, ACC, and LPSI are modeled; remaining safety systems are excluded as they are assumed unavailable during the simulated accident scenario.

Components in the diagram are classified using alphabetic prefixes: P for pipe components, B for branch components, V for valve components, TDV for time-dependent volume components, and PP for pump components. Elements outlined with bold black borders specifically indicate heat structure placement locations.

The pressure vessel of the PKL test facility features a cylindrical design. Components 103 and 104 represent the inlet ring segments of the pressure vessel. The inlet sections for Loops 1&2 and Loops 3&4 are simplified, consolidating the original four inlets into two. Components 106 and 107 model the downcomer segments. Coolant flows through the downcomers, converges in the lower head 101 and lower support plate 102 sections, and subsequently enters the core section 108 for heating. A core bypass segment 109 is also modeled. The RELAP5 code employs nodalization to model and calculate each control volume, where the number of nodes assigned significantly impacts final calculation results and necessitates sensitivity analysis on nodal division. Following comprehensive computational verification, the optimal number of nodes was determined, with the critical core segment divided into 18 nodes.

The coolant heated in the core flows through the upper unheated core section 111 to the core outlet segment 112, forming the pressure vessel internal loop. The core section 108 incorporates heat loss components to simulate thermal losses during experiments. Bypass lines connect the upper head to the pressure vessel inlet, running parallel to the cold legs of all four loops. These lines feature 2.9 mm orifice plates to model internal flow resistance, with bypass mass flow constituting 0.35% of total core mass flow rate. Due to inlet section simplification, the number of bypass lines was correspondingly reduced to two.

The PKL test facility comprises four identical loops; given this symmetry, only Loop 1 is described in detail. Pipe component 201 models the primary system hot leg. Component 203 models the U-tube section. Component 204 represents the U-tube outlet downcomer, simulating the inlet section at the break location. Components 205 and 206 model the transition section and main coolant pump, respectively. The cold leg, modeled by component 207, connects back to the pressure vessel to complete the primary circuit circulation path.

The feedwater system is simplified in the model, utilizing time-dependent volume 221 to simulate feedwater flow. Feedwater combines with water separated by the steam separator in the upper ring header at component 218. This mixture flows through the secondary-side downcomer 219 into the steam generator cylinder section 215; for modeling purposes, the two original downcomers were consolidated into one. The combined feedwater is heated within the steam generator cylinder section before entering the steam separator 216. Following separation, steam flows into the upper steam dome 217. Accumulated steam discharges through steam line 223, passes through steam discharge valve 226, and enters steam discharge volume 227. A time-dependent volume controls the secondary-side pressure boundary in the simulation. A secondary-side steam-generator safety-relief valve 224 and relief-discharge volume 225 are installed on the opposite end of the steam discharge line to ensure the steam generator operates without overpressure during normal conditions. A steam generator bypass heater 232 is modeled at the lower section of the steam generator cylinder to compensate for thermal losses during operation.

The pressurizer connects to the mid-section of the Loop 2 hot leg via surge line segment 341. The pressurizer vessel body is modeled by component 343, with its upper head section 344 connected at the top. The upper head connects to the primary circuit pressure relief system, comprising safety valve 351 and relief valve 353. Pressure relief line 348 routes steam from the upper head to prevent primary circuit overpressurization; this pressure relief system also enables depressurization during accidents to ensure core safety. A pressurizer bypass heater 361 is modeled in the lower section to compensate for operational heat losses, which is shut down during accident transients.

For this specific accident scenario, the auxiliary feedwater system, chemical and volume control system, and secondary-side cooling system were isolated due to experimental constraints. Only the HPSI, ACC, and LPSI were activated to explore the test facility’ s safety performance under restricted safety system availability. Each loop contains two ACCs originally connected to both cold and hot legs; however, during this accident, the ACC connected to the hot leg was unavailable. Consequently, the model exclusively incorporates the cold leg ACC vessel 601, with injection routed through upstream piping 602, isolation valve 603, downstream piping 604, and converging line 606. The HPSI 608 and LPSI 612 connect to the converging line via their respective isolation valves 609 and 615, enabling integrated injection into the cold legs.

2.3 Break Model

ISP-52 is developed based on the PKL J5.1 RUN 2 experiment, which simulated a scenario of SGTR occurring in all four steam generators. Specifically, RUN 2 assumed a 4A-class break in each affected SG, corresponding to a double-ended guillotine break of two heat transfer tubes per steam generator. The study simulated a break size of 3.7 mm, located at the primary-side outlet and secondary-side shell bottom of all four steam generators. Since break configurations were identical across steam generators, nodalization for only one break (Loop 1) is presented as shown in Figure 2 [Figure 2: see original paper]. The break system consists of motor-operated valve and pipe components; during accident conditions, the valve activates to connect the primary and secondary sides. As illustrated, component 204 represents the primary-side outlet section of the SG, 203 denotes the U-tube bundles, and 215 indicates the secondary-side shell of the SG. Component 618 refers to the motor-operated valve that simulates multiple SGTRs, with its outlet connected to the bottom of the SG shell.

3.1 Steady-State Calculation Results

The RELAP5/MOD3.4 code was first employed to calculate the steady-state process based on initial conditions. As demonstrated in Table 1 , computational results exhibit strong agreement with experimental parameters. The RELAP5 input deck (excluding break and safety injection systems) confirms stabilized primary and secondary side parameters, with consistent heat transfer rates between both circuits. Computed temperatures and power correspond closely to experimental initial values, with relative deviations remaining within acceptable limits, validating the model’ s reliability and operational realism.

3.2 Transient Calculation Results

In the PKL J5.1 RUN2 test, partial activation of safety systems was implemented as follows: (1) Secondary-side pressure limited to 43 bar; (2) Secondary-side auxiliary feedwater system became inoperative post-accident; (3) Primary loop depressurization initiated via PRZ safety valves after 1000 s; (4) HPSI injection triggered when CET exceeded 350°C; (5) ACC injection activated at primary pressure below 23.6 bar; (6) LPSI injection engaged at primary pressure below 12.4 bar; (7) Post-rupture valve opening, core power progressively declined with its profile shown in Figure 3 [Figure 3: see original paper]; (8) SG bypass heaters deactivated after SG emptying; (9) PRZ bypass heaters shut down following accident initiation. The sequence of events during the accident is summarized in Table 2 . After accident initiation, core power evolved according to Figure 3, dropping to approximately 550 kW at 2500 s and further declining to 350 kW around 12500 s. Beyond 12500 s, the power reduction rate decelerated, reaching 287 kW at termination.

Figure 4 [Figure 4: see original paper] shows the total mass flow rate at the break, where positive flow indicates fluid transfer from primary to secondary circuit and negative flow signifies reverse flow from secondary to primary circuit. Figure 5 [Figure 5: see original paper] shows primary and secondary pressures. Following break valve opening, higher primary pressure drove coolant leakage from primary to secondary circuit, with mass flow rate increasing rapidly to peak at about 0.75 kg/s. Continuous coolant loss through the break valve reduced primary pressure while the main steam system maintained constant secondary pressure at 43 bar. As the primary-to-secondary pressure differential progressively narrowed, the break mass flow rate exhibited corresponding decline.

Figure 6 [Figure 6: see original paper] shows void fraction and volume equilibrium quality distribution in the core. Figure 7 [Figure 7: see original paper] shows CET and core collapsed level. Figure 8 [Figure 8: see original paper] presents fuel cladding temperature distribution in the core. At 1000 seconds, opening of the PRZ depressurization valve triggered primary pressure reduction, during which the break mass flow rate showed good agreement with experimental data. However, around 2500 s, as break flow had just reversed and failed to fully compensate for primary coolant loss, void fraction and volume equilibrium quality in upper core nodes 10-18 gradually increased, leading to partial core uncovery and decline in core collapsed level, which induced fluctuations in break mass flow. As the pressure differential between primary and secondary sides narrowed, secondary-side water continuously migrated into the primary circuit, facilitating core reflooding. Under this transient condition, maximum CET stabilized at ~280°C, remaining below the HPSI activation threshold. Cladding temperature peaked at approximately 400°C, still within operational safety margins. The core collapsed level reached its first minimum at 3750 s due to intense steam generation within fuel assemblies.

Subsequent continuous secondary-side fluid influx resulted in a simulated mass flow rate increase of ~0.35 kg/s. At ~15000 s, break flow diminished to zero following SG depletion. By ~25000 s, LPSI initiation coupled with reduced PRZ depressurization flow induced significant fluid reflux from primary to secondary side, thereby amplifying simulated break mass flow rate. As primary circuit pressure continued decreasing, secondary-side water refluxed into the primary side. Around 14700 seconds, the SGs progressively emptied. This SG emptying terminated coolant replenishment to the primary system, causing void fraction and volume equilibrium quality in upper core nodes (10-18) to increase again. Consequently, core collapsed level began declining, leading to inevitable fuel uncovery with minimum level reaching approximately 3 m. This resulted in CET rise accompanied by rapid increase in fuel cladding temperature in upper core nodes (10-18), peaking at about 600°C, which remained within safe operating limits.

Figure 9 [Figure 9: see original paper] presents mass flow rate and pressure differential across the PRZ depressurization valve. Figure 10 [Figure 10: see original paper] shows PRZ collapsed level. At 1000 seconds, the PRZ depressurization valve opened, and due to high primary circuit pressure, mass flow rate through the valve increased rapidly, reaching a peak of approximately 0.55 kg/s (simulated value: 0.9 kg/s). As illustrated in Figures 5 and 10, opening of the PRZ depressurization valve caused rapid pressure drop in the primary circuit. During the early accident phase, low PRZ collapsed level facilitated significant coolant transfer from primary circuit to PRZ, resulting in rapid rise of PRZ collapsed level to approximately 10 m. As PRZ collapsed level stabilized, primary circuit pressure decline trend gradually ceased. Around 17000 seconds, PRZ collapsed level increased due to HPSI injection. At this stage, simulated PRZ collapsed level stabilized at approximately 10 m due to flow rate discrepancy in ACC injection. Around 22500 seconds, fluctuations in PRZ depressurization valve mass flow rate were observed, attributed to safety injection system introduction causing significant coolant influx into the PRZ. Around 25000 seconds, PRZ depressurization flow rate decreased due to reduction in pressure differential across the valve. When pressure differential approached zero, mass flow rate through the PRZ depressurization valve also decreased to zero. At this stage, simulated PRZ collapsed level rose again due to HPSI injection, eventually stabilizing at approximately 13 m.

Figure 11 [Figure 11: see original paper] shows SG collapsed level. From Figures 4 and 11, collapsed level of SG began decreasing after accident initiation. Around 15000 seconds, SG collapsed level dropped to zero, resulting in complete SG depletion and causing break mass flow rate to reach zero. At approximately 25000 seconds, as PRZ depressurization valve flow rate decreased, significant coolant from primary circuit redirected back to SG through break valve, leading to simulated break mass flow rate of 0.4 kg/s. This coolant reflux caused SG liquid level to rise again, reaching approximately 3.4 m by 27500 s.

Figure 12 [Figure 12: see original paper] shows total mass flow rate of HPSI across four loops. When CET exceeded 350°C, HPSI initiated injection into each loop' s cold leg at 0.12 kg/s. As seen in Figure 7(b), injected cold water induced local pressure drop due to condensation effects at injection point, causing decline in core collapsed level. Observations from Figures 5, 6, and 7 indicate HPSI injection triggered core quenching, resulting in localized violent evaporation while continuous water injection caused temporary pressure recovery; however, primary circuit pressure gradually declined. In the simulation, initial ACC injection flow rate was relatively high, which accelerated primary circuit pressure drop and consequently diminished visibility of pressure recovery phenomenon.

Figure 13 [Figure 13: see original paper] shows ACC liquid level and total mass flow rate. As HPSI activation caused pressure drop in primary circuit and reached ACC trigger threshold, ACC initiated operation and injected water into primary circuit, peaking at 0.9 kg/s. Figures 5 and 7 indicate combined loss from primary-side injection of HPSI and ACC compensated for coolant depressurization, sustaining core collapsed level. The cumulative effect of steam pressure in the core overwhelmed condensation effect from coolant injection, causing pressure recovery in primary circuit that resulted in ACC injection flow blockage. When primary pressure again fell below ACC pressure threshold, ACC resumed injection with average mass flow rate of about 0.15 kg/s. In the simulation, peak ACC injection mass flow rate is higher than experimental value at about 1.7 kg/s because ACC shares the same injection pipeline with HPSI and LPSI, creating mutual influence during continuous injection process. To maintain continuous ACC injection, larger initial ACC pressure is required, which causes liquid level to rapidly decrease during initial injection and results in high flow rate peak. In later simulation stages, due to HPSI and LPSI injection, ACC injection was blocked and its flow rate decreased to zero.

Figure 14 [Figure 14: see original paper] shows total mass flow rate of LPSI across four loops. As primary pressure continued decreasing, LPSI began operation at 26475 seconds with injection rate of 0.07 kg/s. All loops exhibit symmetric water injection pattern, replenishing primary circuit water inventory and ensuring system enters long-term cooling phase, thereby maintaining overall reactor safety. In the simulation, due to faster primary circuit depressurization rate compared to experiment, LPSI injection time was advanced to some extent.

4. Conclusion

This paper presents a RELAP5 simulation of the multiple SGTR accident conducted on the ISP-52 PKL test facility, with comparative analysis focusing on accident sequences, key parameters, and critical phenomena. The main conclusions are as follows:

(1) RELAP5 effectively reproduced the primary accident progression of the multiple SGTR scenario in ISP-52, demonstrating consistency with experimental data in both event chronology and key thermal-hydraulic phenomena.

(2) Following accident initiation, the main feedwater system and pressurizer bypass heaters tripped. Coolant underwent primary-to-secondary leakage and countercurrent flow. Core temperature rose rapidly after steam generator depletion. Subsequent staged activation of HPSI, ACC, and LPSI maintained core integrity throughout the transient process, proving that residual safety systems can ensure reactor safety under constrained

conditions. Simulation results confirm that primary-side depressurization effectively prevents large-scale radioactive coolant release to the secondary side while enabling reverse inventory transfer from secondary to primary systems, thereby mitigating potential core uncovery accidents.

In this study, RELAP5 modeling employed simplified representations of core components, U-tubes, and secondary circuit systems, introducing inherent limitations. Subsequent research will refine modeling for the core and U-tubes, including implementation of multichannel core models to evaluate the impact of channel quantity on simulation accuracy.

Acknowledgments: This study is part of the ISP project under the WGAMA task framework, with particular acknowledgment to support from NEA/OECD.

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