Validation of Coupled RELAP5/PARCS Codes for Main Steam Line Break analysis: a TMI-1 Benchmark Study

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ABSTRACT

The reliable prediction of reactivity transients in Pressurized Water Reactors (PWRs) requires a consistent representation of the strong coupling between reactor core neutronics and plant thermal-hydraulics. The Main Steam Line Break (MSLB) accident constitutes a particularly demanding scenario including the rapid and asymmetric overcooling of the reactor core, combined with the conservative assumption of a stuck-out control rod during the reactor trip. This induces pronounced space-time effects and significant power redistribution phenomena. Such conditions challenge best-estimate analysis tools and provide a meaningful framework for assessing the capability of coupled multi-physics codes. In this work, the coupled RELAP5/MOD3.2.2–PARCS v3.2 code system is validated against the OECD/NEA MSLB benchmark based on the Three Mile Island Unit 1 (TMI-1) Nuclear Power Plant. The benchmark considers a full-power, end-of-cycle core configuration and has been specifically designed to evaluate the performance of advanced system codes in the simulation of reactivity transients involving multi-dimensional neutronic feedback. A detailed RELAP5 Simulation Model (SM) of the reactor coolant system was developed and dynamically coupled with a three-dimensional (3D) PARCS SM employing the benchmark-provided NEMTAB cross-section library. The coupling enables time-dependent evolution of power distributions and feedback parameters, allowing a physically consistent treatment of Doppler, moderator density, and temperature reactivity effects during the transient. The calculated time evolution of core power, reactivity, thermal-hydraulic variables, and steam generator behaviour is compared with benchmark reference results. In parallel, a standalone RELAP5 SM with point kinetics is developed and used to assess the impact of spatial neutronic modelling on the predicted transient behaviour. The results show that the coupled RELAP5/PARCS successfully captures the characteristic features of the MSLB scenario and demonstrates a good agreement with benchmark data for both global plant thermal-hydraulic and core neutronic parameters, with only very minor differences observed in the final phase of the transient, which is not of primary relevance for the analysis. At the same time, the RELAP5 stand-alone simulations also provide excellent results, showing a high level of consistency with the benchmark data and accurately reproducing both the system thermal-hydraulic response and the main neutronic feedback effects. The study confirms that a fully coupled multi-dimensional RELAP5/PARCS simulation is essential for an accurate representation of MSLB-induced space-time effects and for best-estimate safety analyses of PWR reactivity accidents.

Keywords: *PWRs, TMI-1, MSLB, RELAP5, PARCS, Coupling, Multi-physics.*

1 INTRODUCTION

In nuclear reactor safety analysis, the adoption of Best Estimate (BE) methodologies has significantly improved the realism of transient simulations by reducing excessive conservatism while maintaining regulatory compliance. The reliability of BE approaches, however, relies on rigorous validation against internationally recognized benchmarks to ensure that computational tools can accurately reproduce complex multi-physics phenomena.

Within this context, the OECD/NEA Pressurized Water Reactor (PWR) Main Steam Line Break (MSLB) benchmark [1] represents a demanding validation framework. The MSLB scenario is characterized by a highly asymmetric plant response triggered by the rupture of a single steam line. The resulting rapid steam discharge induces strong primary system overcooling, which, through the negative moderator temperature coefficient, introduces a positive reactivity insertion in the core. The situation is further aggravated by the conservative assumption of a stuck control rod during reactor trip, leading to pronounced spatial power redistribution.

Such space-time effects cannot be adequately captured using thermal-hydraulic system codes relying solely on point kinetics approximations, as these assume a fixed spatial power distribution. Therefore, a consistent coupling between one-dimensional system thermal-hydraulics and three-dimensional neutron kinetics is required to achieve a physically representative simulation of the transient.

The objective of this work is to validate the RELAP5/PARCS coupled framework against the OECD/NEA MSLB benchmark data. A progressive validation strategy is adopted, first assessing the standalone thermal-hydraulic and neutronic models separately, and subsequently evaluating the fully coupled configuration in order to quantify the contribution of multi-dimensional neutronic feedback to the overall plant response.

The remainder of the paper is structured as follows: Section 2 presents the standalone RELAP5 analysis, Section 3 describes the PARCS core modelling, Section 4 discusses the coupled simulations, and Section 5 summarizes the main findings.

1.1 Three Mile Island unit 1 nuclear power plant

The benchmark reference plant is based on Three Mile Island Unit 1 (TMI-1) [1], a two-loop Pressurized Water Reactor designed by Babcock & Wilcox with a nominal thermal power of 2772 MWt. The core configuration considered in the benchmark corresponds to end-of-cycle conditions and consists of 177 fuel assemblies surrounded by radial reflectors, with reactivity control provided by eight control rod banks.

The Reactor Coolant System (RCS) is arranged in two primary loops, each including a hot leg, a Once-Through Steam Generator (OTSG), and two cold legs equipped with main coolant pumps. This configuration plays a key role in the development of asymmetric thermal-hydraulic conditions during a steam line break event.

The reactor protection system initiates a trip when predefined power or pressure thresholds are reached. In accordance with the benchmark specifications, the control rod with the highest reactivity worth is assumed to remain fully withdrawn during the scram, while the remaining rods are inserted with specified delays and insertion velocities. The High-Pressure Injection System (HPIS) is also modelled according to benchmark assumptions, although the present study neglects the additional negative reactivity associated with boron injection in order to maintain consistency with the benchmark definition.

2 RELAP5 STAND-ALONE SIMULATION

The RELAP5 [2] standalone model was developed to reproduce the thermal-hydraulic response of the TMI-1 plant under the OECD/NEA MSLB benchmark specifications. The code was employed using its non-equilibrium two-fluid formulation, allowing for unequal phase temperatures and velocities through the solution of separate momentum equations for liquid and vapor phases. Vertical stratification effects and non-homogeneous flow treatment were retained according to the standard RELAP5 implementation. The original RELAP5 critical flow model was adopted for the simulation of the steam line break, ensuring a physically consistent representation of choked flow conditions. Wall friction and form losses, including abrupt and partial area changes, were modeled consistently within the two-fluid framework.

The Primary System (PS) nodalization includes the two identical reactor coolant loops, the Reactor Pressure Vessel (RPV), and the pressurizer. Each loop is composed of a hot leg, a once-through Steam Generator (SG), and two cold legs with associated main coolant pumps. Particular attention was devoted to the core representation, which was discretized into 20 parallel thermal-hydraulic channels, including 18 channels corresponding to the active core region and two channels representing the reflector regions. This nodalization ensures consistency with the benchmark definition and preserves compatibility with the neutronic nodalization adopted in the PARCS model for subsequent coupled simulations. The core thermal power distribution in the standalone configuration was imposed through 18 dedicated heat structures, each associated with an active core channel, as shown in Figure 1.

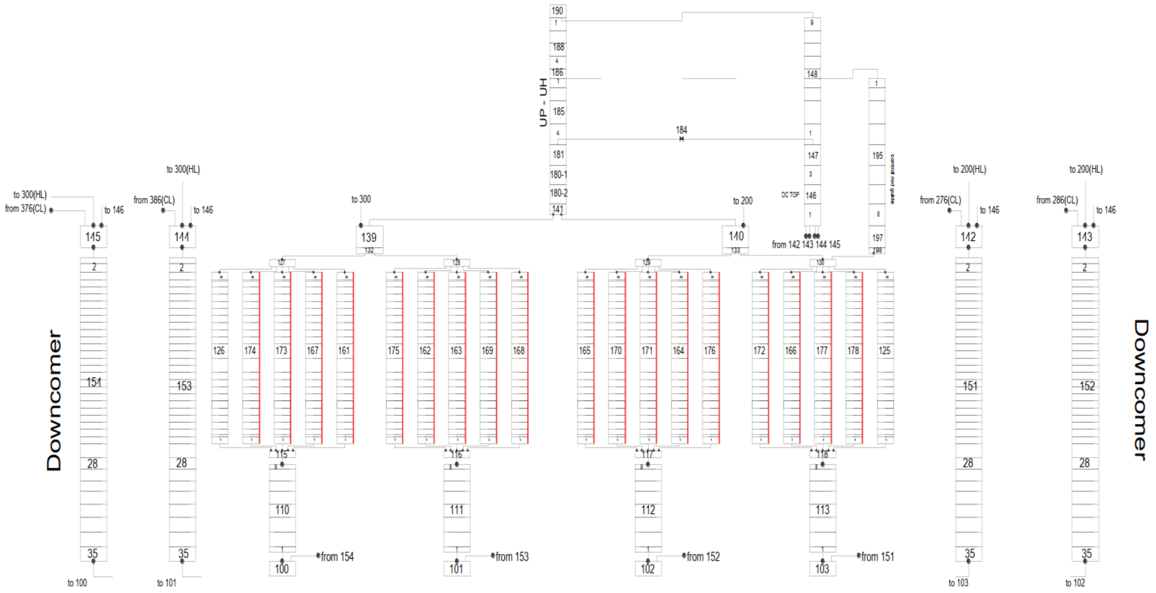


Figure 1: TMI Reactor Pressure Vessel Nodalization

The Secondary System (SS) was fully modeled, distinguishing between the intact and broken loops in order to properly capture the asymmetric behavior induced by the main steam line rupture. All relevant Engineered Safety Features (ESFs) were implemented in accordance with the benchmark specifications.

Prior to transient initiation, a 2000 s steady-state calculation was performed to achieve the initial condition of the plant. The MSLB event was initiated through the opening of the two break valves corresponding to the 24-inch and 8-inch rupture areas, representing the double-ended rupture of a steam line upstream of the cross-connection line. In this configuration, the primary break (24-inch) induces a secondary discharge path through the cross-connection line (8-inch), resulting in two effective break locations that must be simultaneously modeled to accurately reproduce the benchmark scenario. The reactor trip logic follows the benchmark definition: a SCRAM is triggered when either the hot leg pressure decreases below 13.41 MPa or the core power exceeds 114% of nominal power, with a 0.4 s delay, and a total rod insertion time of 2.2 s. The HPIS is actuated when the pressurizer pressure falls below 11.34 MPa, with a 25 s delay, while the pressurizer spray valve opens when the upper plenum pressure exceeds 30 MPa.

The initial steady-state was established at Hot Full Power (HFP) and End-of-Cycle (EOC) conditions, corresponding to 650 effective full power days with a boron concentration of 5 ppm and an average core burnup of 24.58 GWd/MT. Equilibrium xenon and samarium concentrations were assumed. These conditions ensure consistency with the benchmark definition and represent the most limiting configuration in terms of moderator temperature reactivity feedback.

In the standalone configuration, core neutronics were treated using the built-in point kinetics model of RELAP5. To maintain compatibility with the three-dimensional neutronic solution adopted in the PARCS model, the point kinetics parameters were derived from dedicated 3D nodal core calculations under EOC conditions. The reactivity feedback was weighted axially using the core-averaged relative axial power distribution and radially using the assembly-wise relative radial power distribution (quarter-core symmetry), both obtained from the 3D nodal model at HFP conditions. This procedure ensures consistency between spatially resolved neutronics and the lumped kinetics approximation adopted in the standalone analysis.

The implemented reactivity feedback coefficients include the moderator temperature coefficient, the Doppler temperature coefficient, and the relevant delayed neutron kinetic parameters. The effective scram worth and the maximum stuck rod worth were calculated at Hot Zero Power (HZP) conditions, assuming nominal pressure, full coolant flow, and equilibrium xenon distribution representative of HFP operation. A 10% uncertainty was conservatively applied to the calculated rod worth values, and the resulting tripped rod worth was introduced into the point kinetics model.

Although the point kinetics formulation does not explicitly resolve the dynamic spatial redistribution of power during the transient, it preserves the global reactivity feedback mechanisms governing the core behavior. In the present analysis, this approximation proves to be sufficiently accurate to capture the main neutronic features of the transient, demonstrating that the stand-alone model remains a valid and reliable approach for the simulation of MSLB scenarios.

2.1 Results

The RELAP5 standalone results were assessed against the OECD/NEA benchmark solutions [4] by superimposing the present calculation onto the reference envelopes; in the corresponding figures, the red curve denotes the results obtained in this study. The selected parameters allow a comprehensive evaluation of the break discharge behavior, primary system depressurization, asymmetric thermal response, and core neutronic evolution during the MSLB transient.

Figure 2 shows the total break mass flow rate. Following break initiation, a sharp initial peak is observed, corresponding to the rapid depressurization of the broken SG and the establishment of flow at the break location. A second rise appears approximately 30 s into the transient, associated with the progressive reduction of feedwater flow to the broken SG. After that, the break mass flow rate gradually decreases toward zero as the SG inventory is depleted and dryout conditions are reached. The RELAP5 standalone simulation slightly overestimates the magnitude of the initial peak and exhibits minor local deviations during the early phase of the transient. These differences are mainly attributable to modeling assumptions in the two-phase critical flow correlations. In this regard, the default critical flow model discharge coefficients are used for the simulations. Nevertheless, the overall trend, timing of the main features, and long-term behavior are in good agreement with the benchmark solutions.

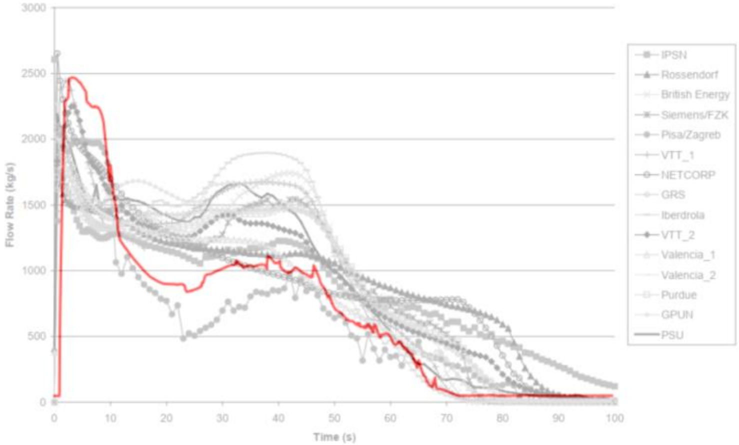


Figure 2: Total Break Flow Rate

The PS pressure response is illustrated in Figure 3, which compares the broken and intact hot leg pressures. The pressure evolution in the two loops is found to be practically identical throughout the transient. This behavior reflects the dominant influence of the global primary system depressurization, which governs the pressure response in a largely uniform manner across the reactor coolant system. The predicted pressures in both the broken and intact loops show good agreement with the benchmark results in terms of both magnitude and temporal evolution, remaining consistent with the reference envelope throughout the transient.

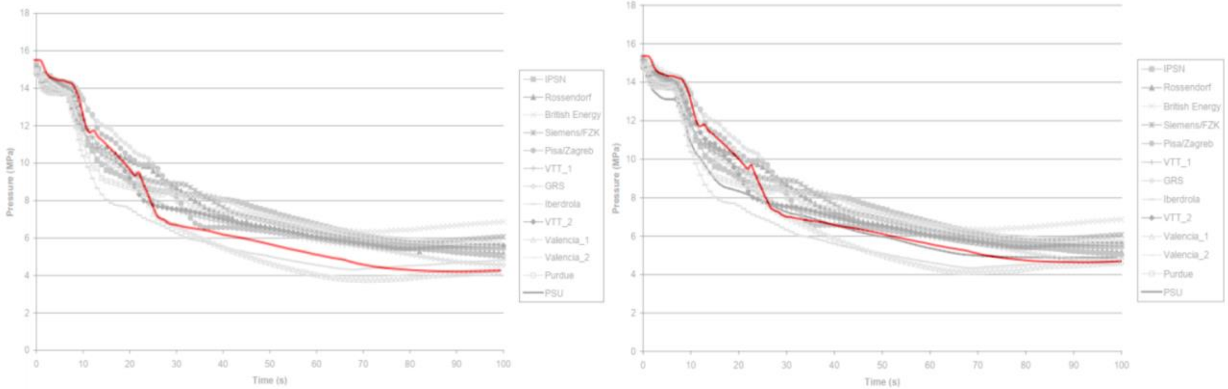


Figure 3: Broken (left) and Intact (right) Loop Hot Leg Pressure

The global PS behavior is further characterized by the pressurizer pressure evolution, shown in Figure 4. As the overcooling in the broken loop propagates through the core, the reduction in coolant temperature results in fluid contraction and a rapid decrease in RCS pressure. The subsequent activation of the HPIS contributes to stabilizing the pressure evolution and moderates the depressurization rate. The standalone results capture both the rapid initial pressure drop and the subsequent stabilization phase with satisfactory agreement relative to the benchmark calculations.

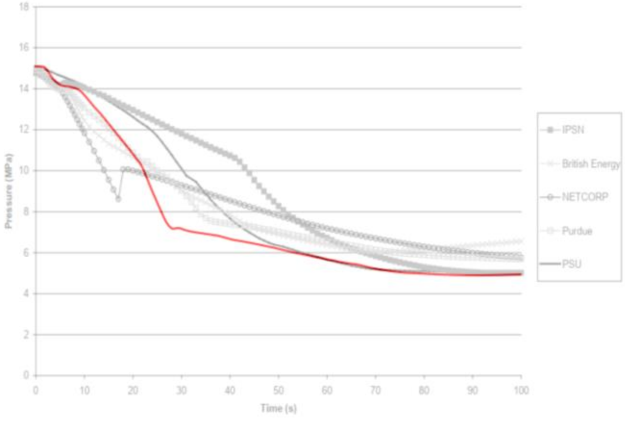


Figure 4: Pressurizer Pressure

The thermal asymmetry of the system is highlighted in Figure 5, which reports the broken and intact cold leg temperatures. The depressurization of the broken SG enhances heat removal and produces a marked overcooling of the associated loop. The broken loop cold leg temperature decreases progressively, while the intact loop temperature exhibits a milder response. The interaction between thermal-hydraulic and neutronic feedback is evident during the early phase of the transient: the decrease in moderator temperature introduces positive reactivity, causing a temporary increase in core power that partially offsets the cooling effect. The temperature evolution in both the broken and intact loops is in good agreement with the benchmark results, with the standalone simulation accurately reproducing the asymmetric behavior of the system.

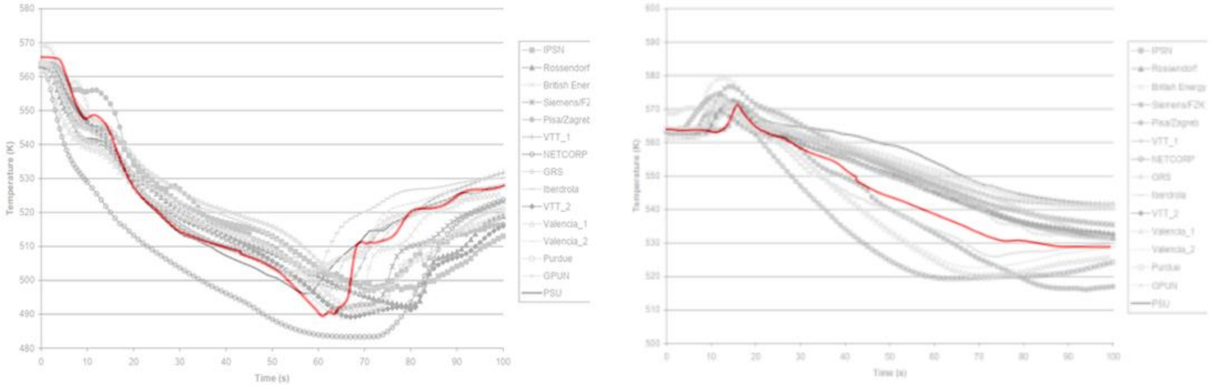


Figure 5: Broken (left) and Intact (right) Loop Cold Leg Temperature

The neutronic response of the core is presented in Figure 6, which shows the total core power evolution. Immediately after break initiation, the inflow of colder moderator into the core region increases coolant density and inserts positive reactivity, resulting in a rapid power excursion. The reactor trip is triggered when the power exceeds the benchmark-defined threshold (3160 MW), leading to a prompt insertion of negative reactivity and a sharp decrease in core power. The RELAP5 standalone results demonstrate good agreement with the benchmark envelope in terms of peak magnitude and general trend. A return to power at approximately 60 s consistent with the benchmark results is noted. Minor discrepancies in the exact timing of the power peak and reactor trip can be attributed to the inherent simplifications of the point kinetics approach, which relies on core-averaged quantities and does not account for spatial flux redistribution. Although the reactivity feedback coefficients are consistently defined according to the benchmark specifications, their application in a lumped formulation leads to slight differences in the predicted transient evolution.

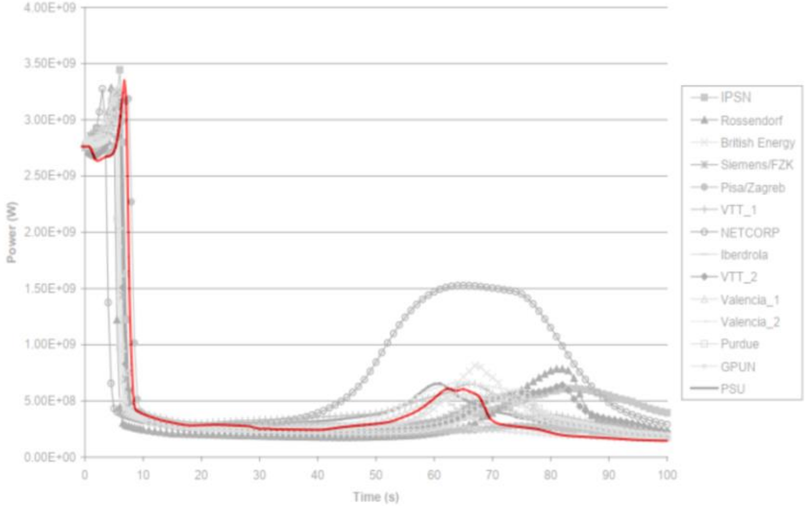


Figure 6: Total Core Power

Finally, the average fuel temperature is shown in Figure 7. The fuel temperature follows the core power behavior and reflects the Doppler feedback mechanism. During the initial power excursion, the increase in fission rate leads to a corresponding rise in fuel temperature. After scram, the insertion of negative reactivity causes a rapid power reduction and a subsequent decrease in fuel temperature. It should be noted that the benchmark does not explicitly define the methodology for calculating the average fuel temperature; therefore, in the present work, this quantity is approximated by considering the temperature of a representative heat structure located in the active core region at approximately mid-height. The standalone simulation predicts somewhat higher peak fuel temperature values compared to most benchmark participants. This difference is mainly associated with variations in spatial averaging and post-processing definitions among the submitted results. Despite these quantitative differences, the overall temporal evolution and stabilization trends remain consistent with the benchmark reference.

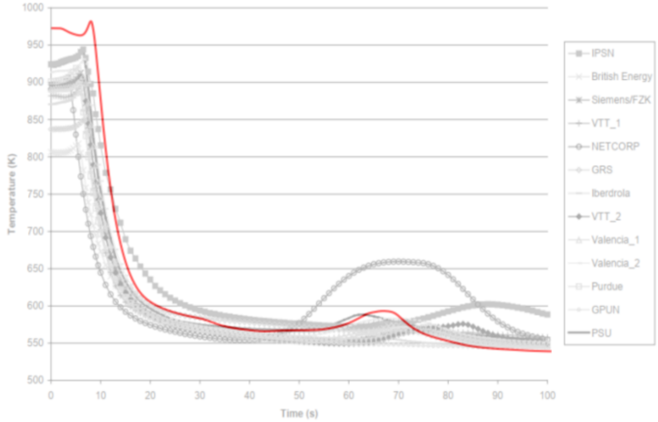


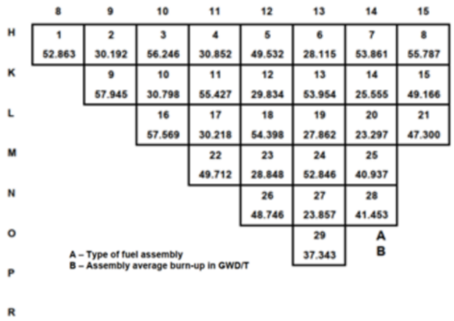
Figure 7: Average Fuel Temperature

Overall, the RELAP5 stand-alone model reproduces the thermal-hydraulic and neutronic behavior of the MSLB transient with a high level of agreement with the benchmark data. The results demonstrate that the model is capable of capturing both the system response and the main reactivity feedback mechanisms, despite the use of the point kinetics approximation. The strong consistency with the benchmark confirms that the stand-alone approach provides an accurate and reliable representation of the transient, supporting its validation within the framework of the present analysis.

3 PARCS STAND-ALONE SIMULATIONS

A three-dimensional full-core neutronic simulation was performed using PARCS [3] in order to analyze the spatial power behavior of the TMI-1 core under the OECD/NEA MSLB benchmark specifications. The objective of this standalone calculation is to assess the intrinsic neutronic response of the EOC configuration using a detailed 3D nodal model prior to the fully coupled RELAP5/PARCS transient analysis.

The core is represented according to the benchmark geometry, consisting of a 17×17 fuel assembly lattice surrounded by radial reflector regions. Since the geometric configuration has already been described in the introductory section, only the discretization strategy is summarized here. The active core region is axially subdivided into 24 nodes, consistently with the thermal-hydraulic discretization adopted in the RELAP5 model, and two additional nodes are included to represent the top and bottom axial reflectors. Radially, a full-core model is employed in order to preserve the asymmetric effects expected during the steam line break scenario. To represent the EOC configuration, 30 distinct fuel assembly types are defined within the core model, as specified by the benchmark data derived from one-eighth core symmetry and shown in Figure 8. These assembly types differ in terms of U^{235} enrichment, number of burnable absorber rods, and burnable poison depletion history. Their spatial distribution reflects the burnup map of the core at EOC conditions and ensures consistency with the reference benchmark configuration.



	8	9	10	11	12	13	14	15
H	1 52.863	2 30.192	3 56.246	4 30.852	5 49.532	6 28.115	7 53.861	8 55.787
K		9 57.945	10 30.798	11 55.427	12 29.834	13 53.954	14 25.555	15 49.166
L			16 57.569	17 30.218	18 54.398	19 27.862	20 23.297	21 47.300
M				22 49.712	23 28.848	24 52.846	25 40.937	
N					26 48.746	27 23.857	28 41.453	
O						29 37.343	A B	
P								
R								

A – Type of fuel assembly
B – Assembly average burn-up in GWD/T

Figure 8: Fuel Assembly-type Map with Burnup

The few-group cross-section libraries provided within the benchmark were originally supplied in NEMTAB format and were converted into PMAX format for compatibility with PARCS. The cross sections are parameterized as functions of fuel temperature, coolant density, and reflector temperature, thereby enabling the representation of Doppler and moderator density feedback effects within the nodal framework. Assembly Discontinuity Factors (ADFs) are included in the dataset to preserve interface flux continuity and heterogeneous transport effects at assembly boundaries. It is noted that the generation of reflector cross sections provided in the benchmark accounts for the presence of adjacent fuel assemblies, ensuring realistic treatment of neutron leakage.

In order to simplify reflector feedback modeling, the benchmark prescribes specific assumptions. For the radial reflector, an average fuel temperature equal to 600 K is adopted in both the initial steady-state and transient simulations, while the coolant density is set equal to the inlet coolant density. For the axial reflector regions, different conditions are applied: in the bottom reflector, the fuel temperature and coolant density are set equal to the inlet coolant conditions of the corresponding thermal-hydraulic channel, whereas in the top reflector they are set equal to the outlet coolant conditions. These assumptions provide a consistent yet computationally efficient representation of axial and radial leakage effects. The reactor is initialized at Hot Full Power and End-of-Cycle conditions with a boron concentration of 5 ppm. The control rod configuration follows the benchmark specification, with seven operational banks explicitly modeled in PARCS. The reactivity associated with the APSR bank is already embedded within the material compositions provided in the benchmark dataset and therefore does not require explicit representation as a separate movable group. In order to reproduce the limiting shutdown margin scenario, the control rod located at position N12 is modeled as a distinct group and assumed to remain fully withdrawn throughout the calculation, representing the maximum stuck rod condition. The decay heat contribution is modeled using the ANS-79 standard formulation.

3.1 Results

The steady-state solution obtained with PARCS standalone was validated against the benchmark reference values [1]. A comparison between the calculated results and the benchmark data is summarized in Table 1. The Doppler Temperature Coefficient (DTC) at HFP EOC conditions is predicted to be −2.73 pcm/K, compared to the benchmark value of −2.57 pcm/K, indicating a slightly stronger Doppler feedback in the present model. The prompt neutron lifetime is calculated as 0.2×10^{-4} s, in close agreement with the benchmark value of 0.18445×10^{-4} s.

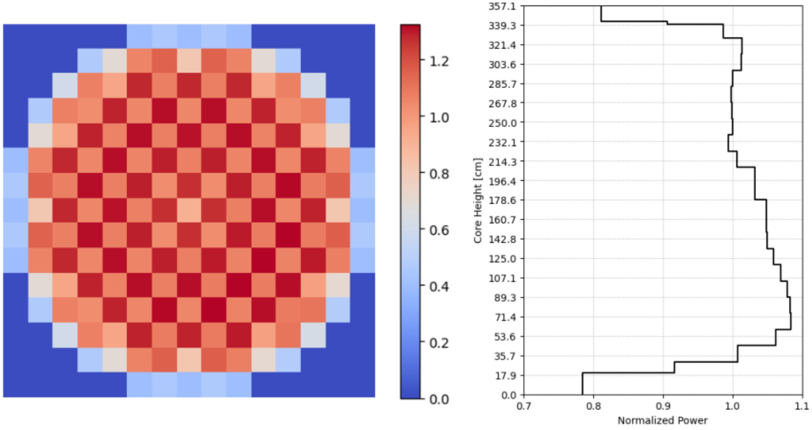
Table 1: Neutron Kinetics Parameter Comparison

Parameter	Benchmark value	PARCS SM results
HFP EOC DTC, [pcm/K]	-2.57	-2.73
HFP EOC prompt neutron lifetime	0.18445e-4	0.2e-4
EOC TRW – V1 [%dk/k]	4.526	4.485
EOC TRW – V1 [%dk/k]	3.040	2.996

For the evaluation of the tripped rod worth (TRW), the benchmark defines two calculation scenarios, referred to as V1 and V2, which differ in the cross-section treatment of control rods during scram. In the first scenario (V1), the control rod absorption cross sections are modified in order to reproduce the tripped rod worth predicted by point kinetics models without altering the initial HFP steady-state solution. This approach reflects the current licensing practice and typically leads to 3D calculations that do not predict a significant return-to-power phenomenon. In the second scenario (V2), an alternative control rod cross-section library is adopted without adjustment to match point kinetics results. In this case, three-dimensional models are expected to predict a return-to-power behavior, thereby providing a more stringent test for spatial kinetics capabilities.

Although these scenarios are primarily defined for coupled transient calculations, the corresponding TRW values are also evaluated in the standalone PARCS steady-state configuration for consistency. In the present work, the calculated TRW values are 4.485% Δk/k (V1) and 2.996% Δk/k (V2), compared to benchmark values of 4.526% Δk/k and 3.040% Δk/k, respectively. The close agreement confirms that the control rod reactivity insertion is accurately represented under both benchmark definitions.

The radial and axial normalized power distributions at EOC are shown in Figure 9. At these conditions, the radial power distribution appears significantly flattened across the core. The assembly power distribution closely follows the burnup map, with lower power levels corresponding to higher burnup regions due to fuel depletion effects. This behavior is consistent with the expected EOC configuration, where reactivity is more uniformly distributed compared to Beginning-of-Cycle (BOC) conditions.



a) Radial

b) Axial

Figure 9: Normalized Power Distribution

A similar flattening effect is observed in the axial power distribution. While at BOC the power peak is typically located near the core mid-plane, at EOC the higher fuel depletion in the central region reduces the mid-plane peak, leading to a two-peak axial shape located in the upper and lower regions of the core. The lower axial region exhibits a slightly higher peak compared to the upper region due to the partial insertion of control rods, which introduces additional neutron absorption in the upper part of the core. The predicted radial and axial power shapes are fully consistent with the benchmark reference distributions.

Overall, the PARCS standalone model accurately reproduces the steady-state neutronic characteristics of the TMI-1 core at EOC conditions, including global reactivity parameters, shutdown margin metrics under both V1 and V2 definitions, and spatial power distributions. This validated three-dimensional representation provides a robust foundation for the subsequent transient and coupled neutronic–thermal-hydraulic analyses.

4 RELAP/PARCS COUPLED SIMULATIONS

The strongly asymmetric thermal-hydraulic conditions induced by a MSLB scenario challenge the validity of spatially lumped neutronic approaches. In the standalone RELAP5 calculation, core neutronics were modeled using a point kinetics formulation, which preserves global reactivity feedback but does not allow dynamic spatial redistribution of power. However, during a MSLB transient, localized overcooling in the broken loop can generate non-uniform moderator density variations across the core, potentially leading to significant three-dimensional power distortions. For this reason, a fully coupled RELAP5/PARCS simulation was performed to assess the impact of spatial neutronic effects on the system response.

In the coupled configuration, RELAP5 provides the time-dependent thermal-hydraulic boundary conditions, including coolant temperature, density, and flow distribution, to the PARCS three-dimensional nodal solver. PARCS, in turn, computes the instantaneous power distribution and reactivity feedback, which are transferred back to RELAP5 as updated volumetric heat sources in the core thermal-hydraulic channels. This bidirectional data exchange is performed at each time step, ensuring a consistent treatment of Doppler and moderator density feedback under transient conditions. The coupling strategy preserves the detailed 20-channel core thermal-hydraulic representation in RELAP5 and the 3D nodal discretization in PARCS, thereby enabling a physically consistent simulation of asymmetric core cooling. Compared to the standalone configuration, the coupled approach allows local moderator overcooling in the broken loop to directly influence the spatial flux distribution and the subsequent reactivity evolution. The coupled simulation was performed under the same initial conditions and benchmark assumptions adopted for the standalone calculations, including EOC core configuration, boron concentration of 5 ppm, and the limiting stuck rod scenario. The results are evaluated through comparison with the OECD/NEA benchmark reference solutions and with the RELAP5 standalone point kinetics calculation, in order to quantify the influence of spatial neutronic effects.

4.1 Results

4.1.1 Comparison with OECD/NEA Benchmark

The coupled RELAP5/PARCS results were evaluated against the OECD/NEA MSLB benchmark envelope [5] for both V1 and V2 scenarios in order to assess the capability of the fully coupled model to reproduce the asymmetric neutronic and thermal-hydraulic response of the TMI core. The comparison focuses on the break discharge behavior, primary system pressure evolution, loop temperature asymmetry, and core power dynamics.

Following break initiation, the total mass discharge exhibits the characteristic sharp initial peak associated with rapid depressurization and choked flow conditions in the broken steam line, as shown in Figure 10. A second increase occurs during the feedwater ramp-down phase before the flow gradually decreases as the broken steam generator inventory is depleted and dryout conditions are reached. For both scenarios, the initial discharge peak is slightly overestimated compared to the benchmark envelope, and minor local deviations are observed during the early transient phase. These differences are primarily attributable to variations in critical flow modeling assumptions and two-phase discharge correlations. Despite these localized discrepancies, the overall trend and timing of the main discharge features are in strong agreement with the benchmark reference solutions.

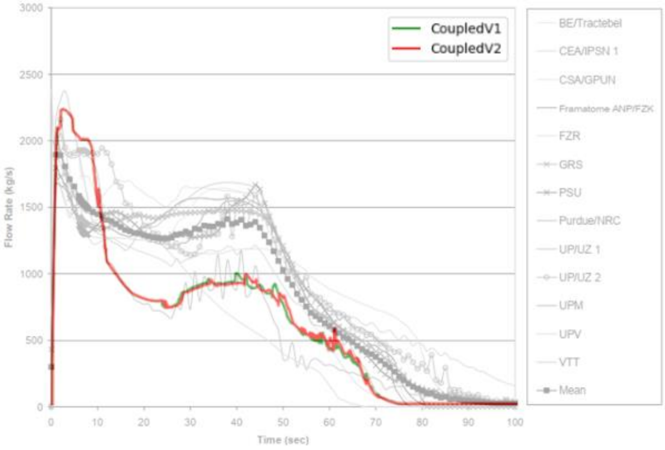


Figure 10: Total Break Flow Rate

Figure 11 represent the depressurization of the primary system captured by the coupled simulation. The predicted hot leg pressures remain within the benchmark dispersion band throughout the transient. In the later phase, differences between the two scenarios become apparent. The V2 case, characterized by a reduced negative scram reactivity insertion, exhibits slightly higher-pressure levels during the final period of the transient. This behavior is directly linked to the partial return-to-power phenomenon and the associated increase in core heat generation.

Conversely, the V1 scenario maintains lower pressure levels due to the absence of a secondary power rise. Overall, the pressure evolution for both scenarios demonstrate good agreement in magnitude and temporal progression with the benchmark results.

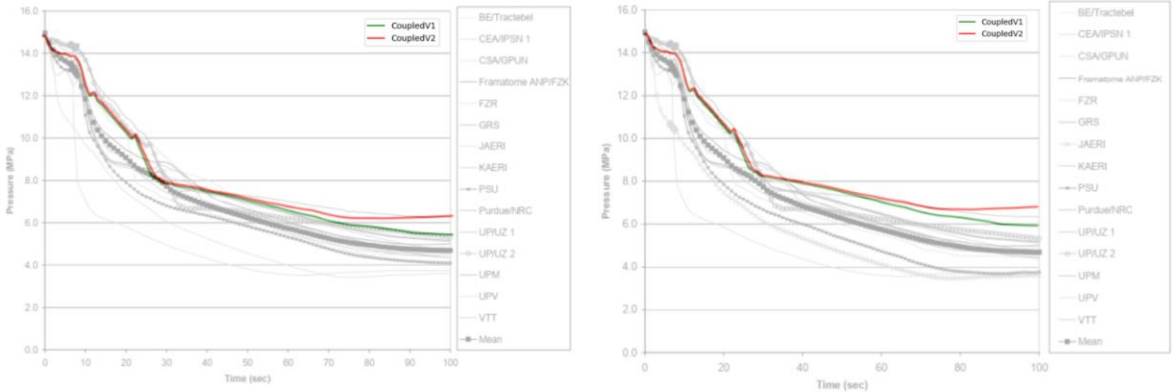


Figure 11: Broken (left) and Intact (right) Loop Hot Leg Pressure

The thermal response of the primary system reflects the strong coupling between neutronics and thermal-hydraulics, as shown in Figure 12. Immediately after break initiation, enhanced heat removal in the broken steam generator leads to rapid overcooling of the associated loop. The broken cold leg temperature decreases significantly, while the intact loop exhibits a more moderate response. This early behavior is consistent across both scenarios and aligns with the benchmark trends. In the second half of the transient, however, the scenarios diverge due to differences in core power evolution. In V2, the partial return-to-power introduces additional heat generation in the core, progressively offsetting the overcooling effect of the broken steam generator. As the steam generator inventory approaches depletion and its cooling capacity diminishes, the broken loop temperature increases. The intact loop temperature approaches a quasi-steady value following the isolation of the intact steam generator. In contrast, the V1 scenario exhibits a slight temperature increase in the final phase of the transient, although less pronounced than in the V2 case. The coupled calculations show full agreement with the benchmark temperature trends during the transient, accurately reproducing both magnitude and temporal evolution.

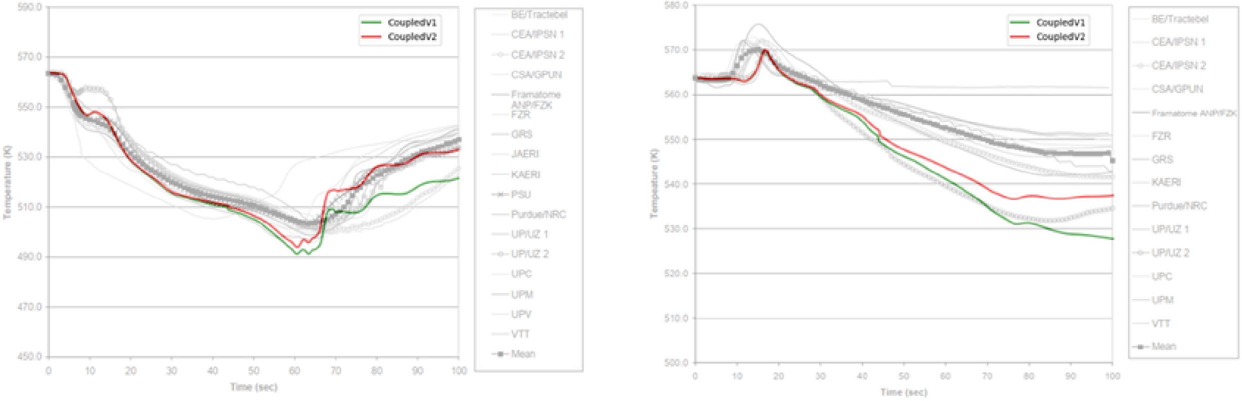


Figure 12: Broken (left) and Intact (right) Loop Cold Leg Temperature

The core power response highlights the fundamental distinction between the two benchmark scenarios, as illustrated in Figure 13. In both cases, the inflow of overcooled moderator from the broken loop into the core region introduces positive reactivity, producing a rapid power excursion immediately after break initiation. The reactor trip is triggered when the power reaches 114% of nominal value, leading to a prompt insertion of negative reactivity and a sharp power decrease. In the V1 scenario, no significant return-to-power is observed following scram, in agreement with the majority of benchmark participants and consistent with current licensing-oriented assumptions. In contrast, the V2 scenario predicts a second power rise due to the combination of sustained moderator overcooling and the reduced magnitude of negative scram reactivity assumed in this case. Although the second peak remains lower than the upper bound of the benchmark envelope, its timing and qualitative behavior are consistent with the expected return-to-power phenomenon defined for V2. The dispersion observed among benchmark submissions in the post-scram phase—particularly during the potential return-to-power interval—can be attributed to differences in predicted total reactivity evolution, moderator density feedback, and Doppler temperature response. Within this context, the present coupled results fall within the benchmark band and capture the scenario-dependent behavior in a physically consistent manner.

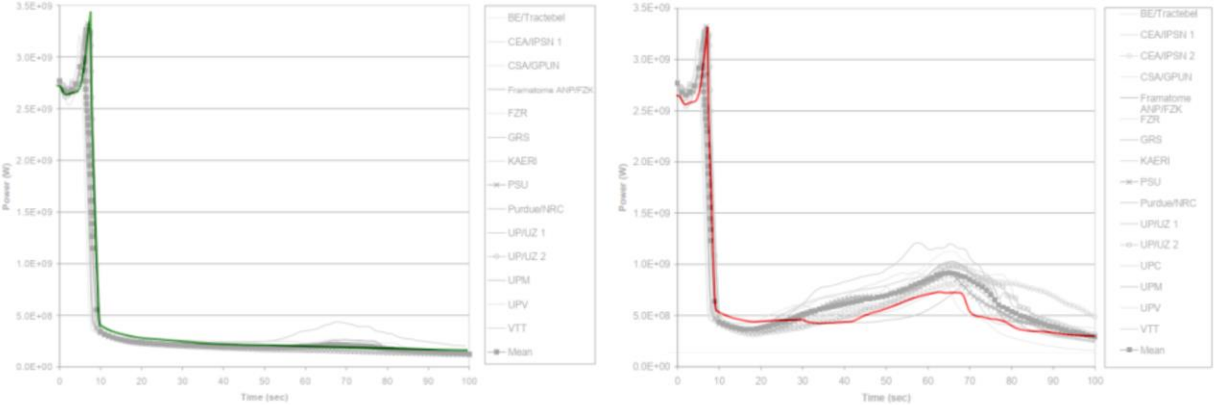


Figure 13: Core Power for V1 (left) and V2 (right) scenario

Overall, the RELAP5/PARCS coupled model demonstrates a robust capability to reproduce both the global system response and the scenario-specific neutronic behavior prescribed by the OECD/NEA benchmark. The agreement obtained for both V1 and V2 configurations confirms the consistency of the implemented coupling strategy and validates the integrated treatment of spatial power redistribution and thermal-hydraulic feedback under strongly asymmetric transient conditions.

4.1.2 Comparison with RELAP5 Stand-Alone

In order to quantify the added value of the three-dimensional neutronic treatment, the coupled RELAP5/PARCS results were compared with a RELAP5 stand-alone calculation employing point kinetics. The assessment is primarily based on scenario V2, while V1 results are included in the figures to provide a complementary reference for the coupled behavior. The comparison aims to identify the impact of spatial kinetics on both the early neutronic response and the subsequent thermal-hydraulic system evolution.

The most significant discrepancy between the two approaches is observed in the core thermal power evolution, as shown in Figure 14. In the coupled RELAP5/PARCS simulations, the break-induced overcooling generates a strong, spatially localized positive reactivity insertion in the sector of the core hydraulically connected to the broken steam generator. This effect leads to a rapid and pronounced power increase prior to reactor trip. In the stand-alone calculation, the power peak is slightly lower in magnitude and occurs marginally earlier (by approximately 0.5 s) compared to the coupled case. These differences can be attributed to the point kinetics approximation, which neglects spatial flux redistribution and assumes uniform reactivity feedback across the core. As a result, the local effects induced by asymmetric moderator cooling are smoothed out, leading to small discrepancies in both the timing and magnitude of the power excursion. The return-to-power phenomenon is also observed in the stand-alone calculation. While the peak value is comparable to that predicted in the coupled V2 scenario, the duration of the power excursion is noticeably shorter, resulting in a narrower peak. This behavior is a direct consequence of the point kinetics model, which cannot capture the sustained spatial redistribution of power within the core. In the coupled approach, local feedback mechanisms and spatial effects allow the overcooling-induced reactivity insertion to persist longer in specific regions, thereby prolonging the return-to-power phase. In contrast, the stand-alone model, based on core-averaged quantities, leads to a faster stabilization of reactivity and a more rapid decay of the power excursion.

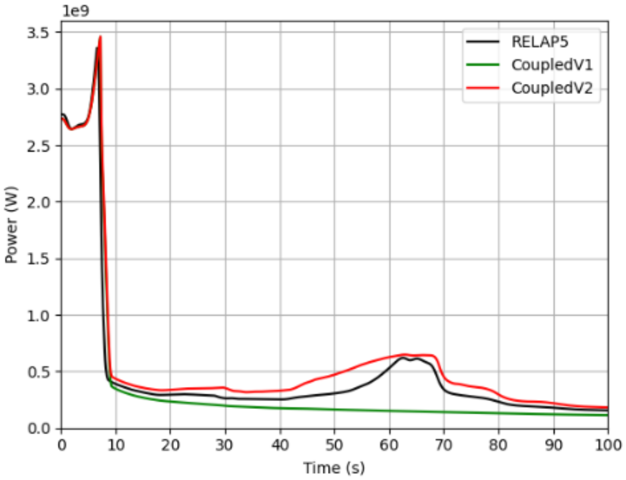


Figure 14: Core Thermal Power

In terms of total break mass flow rate, all simulations exhibit the expected rapid increase immediately after break initiation (t = 0 s), corresponding to the sudden depressurization of the broken SG, as shown in Figure 15. The stand-alone and coupled calculations display an almost perfectly overlapping behavior throughout the entire transient, with negligible differences observed. This strong agreement is expected, as the break discharge is primarily governed by thermal-hydraulic conditions—such as local pressure, fluid properties, and critical flow modelling—rather than by the details of the neutronic treatment. As a result, the influence of spatial kinetics on this parameter remains limited, leading to a consistent prediction of the discharge dynamics across all approaches.

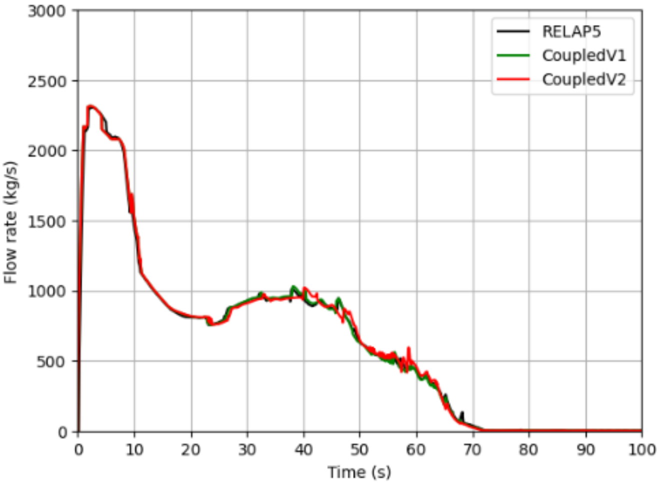


Figure 15: Total Break Flow Rate

A similar behavior is observed for the hot leg pressures in both the broken and intact loops, as can be seen in Figure 16. During the first approximately 10 seconds of the transient, all three cases show a nearly perfect overlap in the primary pressure evolution for both the broken and intact loops, indicating a consistent prediction of the initial system response. In the subsequent time interval (approximately 10–30 seconds), small deviations begin to emerge, with the stand-alone calculation predicting slightly lower pressure values compared to both coupled scenarios. In the final phase of the transient, the stand-alone results tend to lie between the V1 and V2 solutions, as a consequence of the behaviour of the core power. These differences can be attributed to the point kinetics approximation, which smooths the reactivity feedback over the entire core and does not capture localized effects associated with asymmetric cooling.

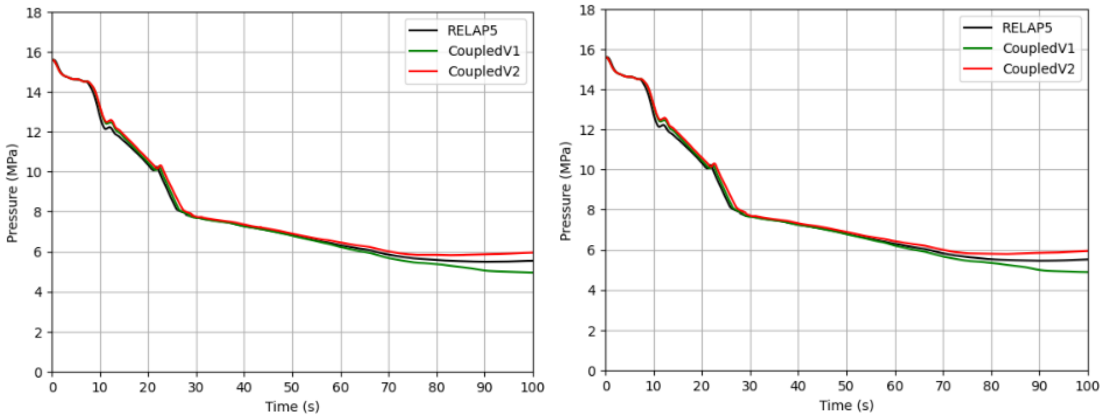


Figure 16: Broken (left) and Intact (right) Loop Hot Leg Pressure

Figure 17 illustrates the temperature evolution in the cold legs of both loops, showing a behavior consistent with that observed for the pressure response. During the initial phase of the transient (approximately the first 20 seconds), all simulations exhibit an almost perfect overlap, confirming that the early thermal response is consistently captured across the different approaches. As the transient progresses, small differences begin to develop and gradually increase over time, with the stand-alone results systematically lying between the V1 and V2 coupled scenarios. This behavior is linked to the point kinetics approximation, which does not resolve the spatial redistribution of power induced by asymmetric moderator cooling. In the coupled simulations, local feedback effects lead to a more pronounced differentiation between the two scenarios, particularly in the later stages of the transient. In contrast, the stand-alone model, based on core-averaged quantities, produces an intermediate thermal response. Despite these differences, the overall temperature evolution remains in good agreement, indicating that the primary system behavior is largely governed by thermal-hydraulic dynamics.

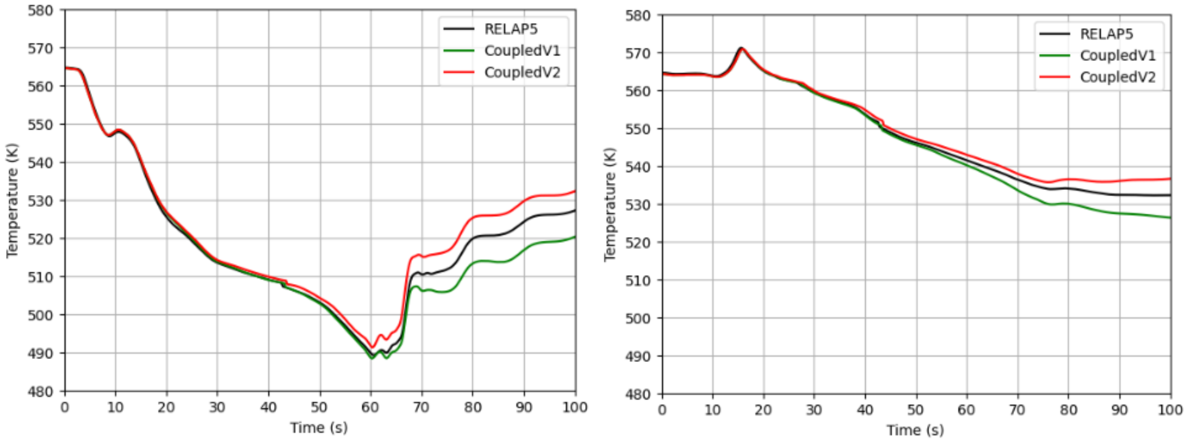


Figure 17: Broken (left) and Intact (right) Loop Cold Leg Temperature

From a global system perspective, the stand-alone model proves capable of reproducing the overall transient progression with a high degree of accuracy, even when employing the point kinetics approximation. The comparison shows that the stand-alone results are highly consistent with those obtained from the coupled V2 scenario, which is adopted as the reference case. This indicates that, despite the inherent simplifications of point kinetics, the model is able to capture the main neutronic phenomena governing the transient with good fidelity. Nevertheless, the coupled RELAP5/PARCS approach provides a slightly more accurate representation of the transient, particularly in resolving spatial effects associated with asymmetric moderator cooling and localized reactivity feedback. While the differences between the two approaches remain limited, the three-dimensional coupling ensures a more detailed description of power redistribution within the core. Overall, both approaches demonstrate a strong capability in predicting the system response, with the coupled model offering an incremental improvement in physical consistency.

5 CONCLUSION

This study presented a comprehensive analysis of the TMI-1 MSLB transient using RELAP5 stand-alone, PARCS stand-alone, and fully coupled RELAP5/PARCS simulations, with systematic comparisons against the OECD/NEA benchmark and between modeling approaches. The RELAP5 stand-alone calculations reproduced the thermal-hydraulic system response with results fully consistent with the benchmark, accurately capturing the break flow rate, primary pressure evolution, and loop temperature behavior throughout the transient. The use of point kinetics, despite its simplifying assumptions, proved sufficient to represent the main neutronic feedback mechanisms, leading to results in strong agreement with the reference solutions. The PARCS stand-alone simulations highlighted the importance of three-dimensional neutronic modeling in capturing the core response to asymmetric moderator overcooling. The results demonstrated the sensitivity of the power evolution to control rod reactivity assumptions and confirmed the capability of the 3D model to reproduce both the no return-to-power (V1) and enforced return-to-power (V2) scenarios defined in the benchmark. The fully coupled RELAP5/PARCS calculations provided a physically consistent description of transient, showing results in agreement with benchmark data for all the analyzed parameters. The coupling approach successfully captured the interaction between spatial power redistribution and thermal-hydraulic feedback, as well as the scenario-dependent neutronic behavior.

Overall, both the RELAP5 stand-alone and the coupled RELAP5/PARCS models demonstrated excellent agreement with the benchmark results, confirming their reliability and robustness for the simulation of MSLB transients. The comparison between the two approaches indicates that, while the coupled model provides a more detailed representation of spatial effects, the stand-alone model remains capable of delivering accurate and consistent predictions of the system response. Therefore, both modeling approaches can be considered fully validated within the framework of the present benchmark analysis.

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