Document downloaded from:

http://hdl.handle.net/10251/157585

This paper must be cited as:

Hidalga-García-Bermejo, P.; Abarca Giménez, A.; Miró Herrero, R.; Sekhri, A.; Verdú Martín, GJ. (2019). A multi-scale and multi-physics simulation methodology with the state-of-the-art tools for safety analysis in light water reactors applied to a turbine trip scenario (PART I). Nuclear Engineering and Design. 350:195-204. https://doi.org/10.1016/j.nucengdes.2019.05.008

The final publication is available at

https://doi.org/10.1016/j.nucengdes.2019.05.008

Copyright Elsevier

Additional Information

A Multi-scale and Multi-Physics Simulation Methodology with the state-of-the-art tools for safety analysis in Light Water Reactors applied to a Turbine Trip Scenario (PART I)

Patricio Hidalga[1], Agustín Abarca[2], Rafael Miró[1], Abdelkrim Sekrhi[3], Gumersindo Verdú[1]

[1]Institut de Seguretat Industrial, Radiofísica i Medioambiental (ISIRIM)
Universitat Politècnica de València (UPV)
Camí de Vera s/n 46022 València (Spain)

[2]Departament of Nuclear Engineering
North Carolina State University
Raleigh, NC 27695-7909 (USA)

[3]Kern Kraftwerk Leibstadt (KKL)
Eigen, 5325 Leibstadt (Switzerland)

Corresponding author: phidalga@iqn.upv.es

KEYWORDS: Simulation Methodology, LWR safety analysis, Turbine Trip, TRACE, PARCS, COBRA-TF

ABSTRACT

The simulation of transient events is a requirement in the evaluation of the safety of Nuclear Power Plants. The Nuclear Authority request the operators to report the prediction of the evolution of the corresponding safety variables using simulation codes and methodologies that have proved to be validated against real data, whether experiments or plant measurements. Moreover, these simulation codes are used in the engineering work that a Nuclear Power Plant needs for planning a competitive and safe operation strategy. The available resources in simulation tools make possible complex analysis that can be used to predict realistic results. The consequence is the opportunity of making a safe and cost-efficient evaluation of the safety margins. Operators can use these tools for licensing to the Nuclear Authority and for calculation support of the operation of the reactor in whichever considered case.

This paper presents a methodology that takes advantage of different simulation tools to join the capabilities in the Best Estimate (BE) simulation of transients for Light Water Reactors. This methodology works in different steps to account all the physics using the proper scale in a multi-physics and multi-scale approach. An automatic tool manages the data pre- and post-processing the corresponding input and output files. The purpose is to simulate the transient case in a coarse mesh and generate the boundary conditions for a simulation in more detailed scale with a finer mesh in the next step. Therefore, this methodology works generating the corresponding nodal cross section data to be used in coupled 3D thermal-hydraulics and neutron kinetics simulations run with system codes. A channel-by-channel core model is used in order to identify the critical fuel channel. Finally, the boundary conditions of the critical fuel channel are loaded in a pin-by-pin thermal-hydraulic model to perform the definitive Safety Analysis of the target variable, that is selected by the user.

The methodology presented in this paper, is applied to a real fast transient case, a Turbine Trip event of fuel cycle 18 in Kern Kraftwerk Leibstadt, KKL. The results of each step of this methodology have been validated against the available plant data and the selected target safety variable, the Critical Power Ratio at pin level, has been code-to-code verified. The results show good agreement proving the effectivity of this methodology.

1. INTRODUCTION

In the generation of electricity by means of Nuclear Energy, safety is a priority. The operation of a Nuclear Power Plant (NPP) is under the supervision of the corresponding Nuclear Authority of the country where it operates. In addition, several international agencies meet periodically to re-evaluate the safety and deliver recommendations regarding the new challenges that are revealed while the Technology of the Nuclear Power Generation is developed. Due to this, the operators of a NPP are requested to report the Safety Assessment that proves the Nuclear Authority that the operation in certain postulated scenarios is safe. For this purpose, the operators realize and analyze simulations with the state-of-the-art tools. In addition, operators must account the competitiveness of the electric market and the sustainability of a cost-efficient production. Therefore, special effort is put on developing Safety Analysis methodologies that retrieve realistic results that allow a more efficient evaluation of the Safety Margins without compromising the safe operation of the reactor.

This paper presents a multi-scale and multi-physics methodology that joints the capabilities of different state-of-the-art codes for the prediction of safety variables in postulated transient scenarios. For validating the results, the authors use the application case of the Turbine Trip of Fuel Cycle 18 of KKL.

The introduction of this document describes the proposed methodology and the application case. Section 2 shows the results of the first step being the system TH/NK coupled model. Section 3 shows the next step of the methodology which is the analysis in a core defined channel-by-channel TH/NK coupled model. Furthermore, the validation of the results is complemented with the code-to-code verification of the target safety analysis variable corresponding to this type of transient scenario. Lastly, section 4 shows the conclusions and future work.

1.1 Application case and methodology description

The aim of the multi-physics and multi-scale methodology is to take advantage of the state-of-the-art tools available in KKL to perform a Deterministic Safety Analysis with the Best Estimate (BE) approach and complement the results with the corresponding Uncertainty and Sensitivity Analysis.

A BE approach (D'Auria et al., 2006) is a step forward in the simulation analysis techniques. Conversely to Conservative Approach, as the IAEA defines in the Specific Safety Guide No. SSG-2, Best Estimate codes apply the available physics knowledge avoiding conservative assumptions. This means that BE simulations account realistic experimental correlations, numerical methods and also the interaction of the different physics during a transient simulation, for instance the feedback between Neutron Kinetics (NK) and Thermal-Hydraulics (TH) (Jiménez et al., 2016).

1.2 Description and features of the proposed methodology

This methodology combines the multi-scale and multi-physics features. The multi-scale approach (García-Herranz et al., 2017) is realized for combining the efficiency of a macro-scale simulation, due to the use of coarser meshes, and the detail level needed for the evaluation of specific parameters.

The other relevant characteristic of this simulation methodology is the multi-physics (Chanaron et al., 2015) capability for the transient analysis. This feature is necessary as there are different physics involved in the behavior of the NPP that must be evaluated for a given scenario. Accounting not only the different scales of the NPP but also the different physics as well as their interaction gives to this methodology the capability of predicting Best Estimate results.

The proposed methodology is designed to be used in two ways, according to the user needs. On the one hand, the user can apply the full methodology, i.e. perform a step-by-step simulation from the Cross Section Library generation to the thermomechanical analysis at pin level. On the other hand, the user can simulate Stand Alone steps by loading the corresponding boundary conditions. This data flow is allowed by means of MATLAB (MathWorks Inc., 2000) programmed applications (APIs and scripts) that pre- and post-processes the results files retrieving and loading the boundary conditions for the successive steps.

This methodology counts on eight different steps that are to be run hieratically. The steps exchange information from one to another. First of all, a lattice pin-by-pin code performs the deterministic transport calculations in 2D. The resulting data are the Cross Section and neutron kinetics parameters sets for different isotopic compositions (depending of the fuel design, fuel temperature, moderator density, boron concentration, control rod insertion and burnup) of the core in the selected number of energy groups. For this purpose, CASMO (Ekberg et al., 1995) code is in charge of this step. The data sets are formatted and arranged in libraries that the nodal 3D diffusion code will use. In this paper the SIMTAB methodology (Barrachina et al., 2010) is used for this step. The Neutron Kinetics is solved by the 3D Diffusion code PARCS (Downar et al., 2014). The Thermal-Hydraulics at plant level are solved with TRACE (USNRC, 2011), and therefore, the coupled model is designed for TRACE/PARCS. This methodology also uses a coupled channel-by-channel core model for CTF-UPVIS/PARCS (Abarca et al., 2015), where the CTF-UPVIS undertakes the TH calculations and again, PARCS calculates the NK. CTF-UPVIS is the enhanced version of COBRA-TF (Avramova et al., 2008) from Senubio Research Group. Finally, the thermal-hydraulics at pin level are simulated by CTF-UPVIS stand-alone. Figure 1 shows a basic diagram of the data flow through the different steps of the presented methodology.

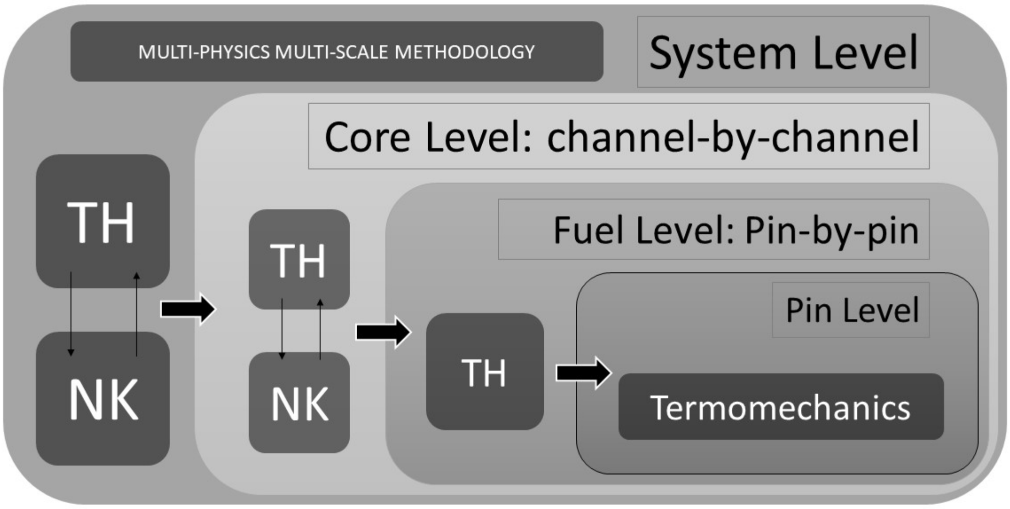

Figure 1.- Diagram of the steps and codes used in the proposed methodology.

Each step of this methodology is feed with the output values of the previous step. The aim of managing automatically the data flow between steps is to avoid the user effect in the pre and post-processing of the data. In this way, giving the corresponding safety criteria, the critical elements can be tracked.

The state-of-the-art codes that Table 1 shows are used in the framework of the project with KKL and according to the different code user agreements for CTF-UPVIS (COBRA-TF users group) and US Nuclear Regulatory Commission (USNRC) codes for TRACE (CAMP users group). Table 1 shows the current version of the codes that this methodology uses.

Table 1.- Information of the state-of-the-art codes used in the developed methodology.

Code abbreviation	Code version	Developer	Property
CASMO	casmo4-2.05.14	Studsvik/KKL	KKL
SIMULATE	simulate3-	Studsvik/KKL	KKL
SIMTAB	simtab-v37	Senubio (ISIRYM/UPV)	UPV
PARCS	parcs_m16_UPVIS_v1801_ifr	U. MICHIGAN	USNRC
TRACE	Trace-v50p3	ISL and NRC	USNRC
CTF-UPVIS	CTF_UPVIS_v1701_r0_x64_r	Senubio (ISIRYM/UPV)/CTF Users Group	UPV/CTF Users Group

The methodology proposed in this paper will be evaluated against a reference real scenario, namely the TT of Fuel Cycle 18. This transient case is classified as a Postulated Accident according to Chapter 15 (USNRC, 2007) of NUREG 0800 document, and certain safety criteria must be met. The available documentation regarding the Switzer Nuclear Authority (ENSI, 2015), the Nuclear Authority of Switzerland, specifies which figures of merit have to be analyzed. The criteria, defined in Chapter 15.2 (USNRC, 2007) of NUREG-0800 document, can be summarized as follows, as extracted from the ANS (American Nuclear Society) standards:

1) Pressure in the reactor coolant and main steam systems should be maintained below 110 percent of design values.

2) In BWRs the Critical Power Ratio (CPR) must remain above the minimum CPR safety limit, based on acceptable correlations, according to the 95/95 criterion.

In regard of the defined criteria, it is important to know in which of the points this methodology can be applied. These criteria are figures of merit that this methodology can predict. On the one hand, the pressure in the reactor and in the steam lines can be predicted using TRACE/PARCS model. On the other hand, the minimum CPR can be predicted at pin level with CTF-UPVIS. The reader must notice that CTF-UPVIS is version of CTF enhanced at ISIRYM (UPV), belonging to the family of the COBRA-TF codes. According to this, an overview of the results of the methodology is to be detailed in section 3.

1.3 Application case: Turbine Trip event

A TT event is provoked by a malfunction of a turbine or reactor system causing the turbine to trip off the line by abruptly stopping the steam flow to the turbine. The Turbine Stop Valves closure time is faster (~0.10 seconds) than those of the Turbine Control Valves. This fact results in a more severe transient, due to the strong interaction between the neutron kinetics and the thermal-hydraulics in such a short time interval.

The closure of the Turbine Stop Valves causes a sudden reduction in the steam flow resulting in a reactor pressure escalation. If Selected Rod Insertion is not available in the BWR then the SCRAM occurs. The consequent effect results in an increase of the reactor coolant temperature, a decrease in the coolant density and an increase in the reactor coolant pressure. The power peak derived from the pressure surge is mitigated by means of control rods maneuver.

Afterwards, sensible and decay heat is removed through actuation of one or several of the following systems: steam relief system, steam bypass to the condenser, reactor core isolation cooling system and emergency core cooling systems. The event took place in Fuel Cycle 18 in the BWR of KKL and has been an application case for different analysis prior to this work (Hidalga et al., 2013). Table 2 displays the sequence of events.

Table 2.- Turbine Trip Fuel Cycle 18 sequence of events.

Time (ms)	Event	Time (ms)	Event
0	Core flow at 3370.85 kg/s	490	Bypass Valves at ~ 50 %
0	Dome pressure at 71.9 bar	780	Peak Vessel Dome Pressure 7.31 MPa
0	Turbine Trip	1020	Bypass Valves reached maximum opening ~ 82 %
218	Control Valves begin to close	1410	SRI Rod Full-In
260	Bypass Valves begin to open	2030	Peak Steam Flow 2170.7 kg/s
300	Control Valves at ~ 9.125 %	4880	Recirculation FCV at 18 %
300	SRI Channels initiation	9100	Core flow after transient at ~ 1892 kg/s
305	Recirculation FCV closed at 66%	9980	Power Peak at ~ 58 %
440	Bypass Valves at ~ 25 %	29980	Vessel Dome Pressure 6.80 MPa

2. Simulation results of the coupled TRACE/PARCS system model

The simulation of this step uses the Cross Section Libraries generated with SIMTAB. This means that previously the *nemtab* and *nemtabr* libraries have been obtained. After obtaining the Cross Section Libraries, simulations have to be realized in parallel. On the one hand, the TRACE model has to converge to a Steady State initial condition. On the other hand, the coupled model of TRACE/PARCS has to verify the stand-alone model.

2.1 Verification of the coupled steady-state versus the reference code SIMULATE-3

The results of the TRACE/PARCS simulation for the coupled steady state are verified against SIMULATE-3 (Dean et al., 2007). The code-to-code verification is done using the radial and axial power distribution and the *k-effective* parameter as figures of merit. Figure 2 shows the coupling map of TRACE and PARCS. This map shows the three different radial regions of the TRACE CHAN components and the assignment to the PARCS radial node distribution. The CHAN components are labeled as CHAN 26 for the outer region, CHAN 25 for the intermediate region and CHAN 24 for the inner region. Region labeled with 0 corresponds to the reflector nodes.

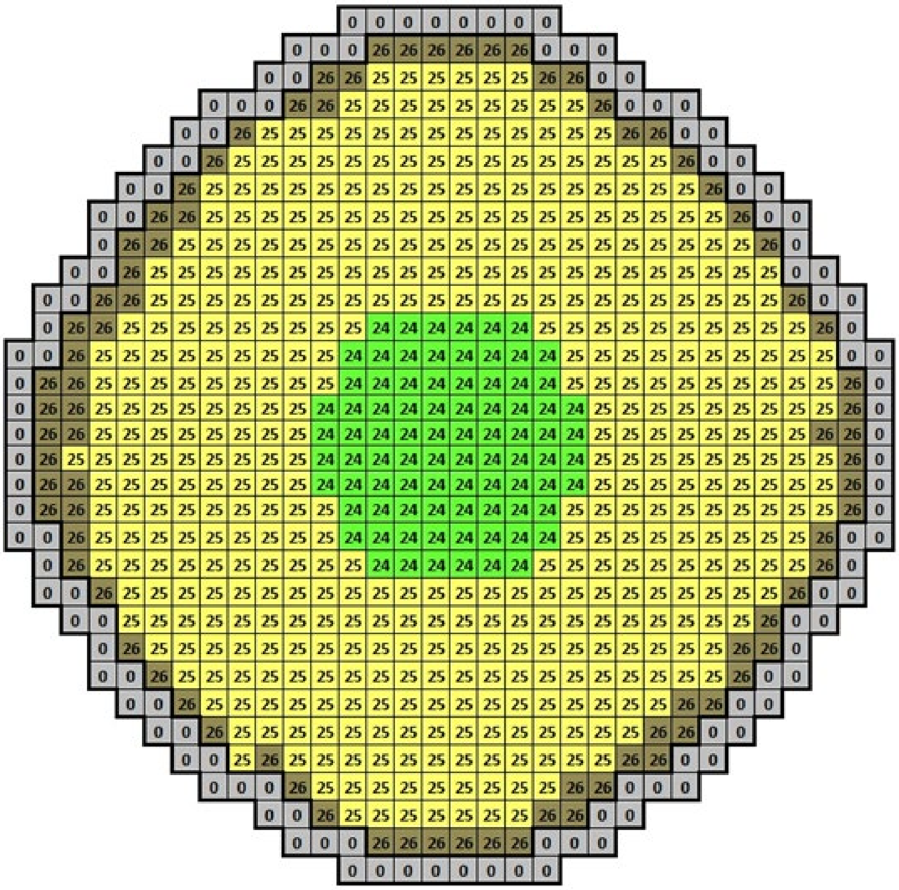

Figure 2.- Coupling Map for the TRACE/PARCS(NEMTAB) model.

Figure 3 shows the comparison of the average axial power profile predicted by TRACE/PARCS using the NEMTAB format libraries, and SIMULATE-3. Moreover, figures 4 to 6 show the axial comparison of the power distribution for the three regions of TRACE against SIMULATE-3. The results of the code-to-code verification are summarized in table 3.

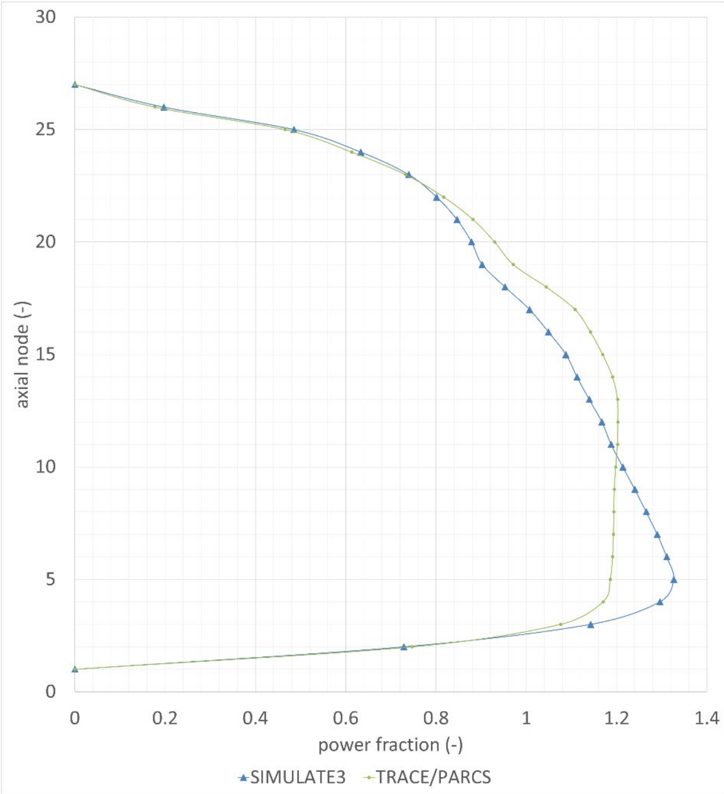

Figure 3.- Core average axial power distribution of TRACE/PARCS (NEMTAB) vs SIMULATE-3.

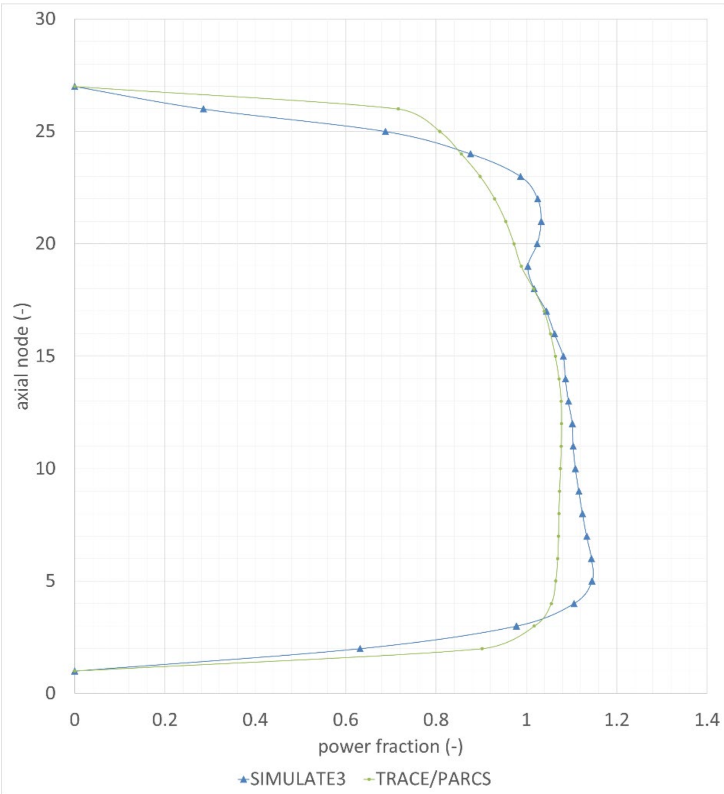

Figure 4.- Inner region average axial power distribution of TRACE/PARCS (NEMTAB) vs SIMULATE-3.

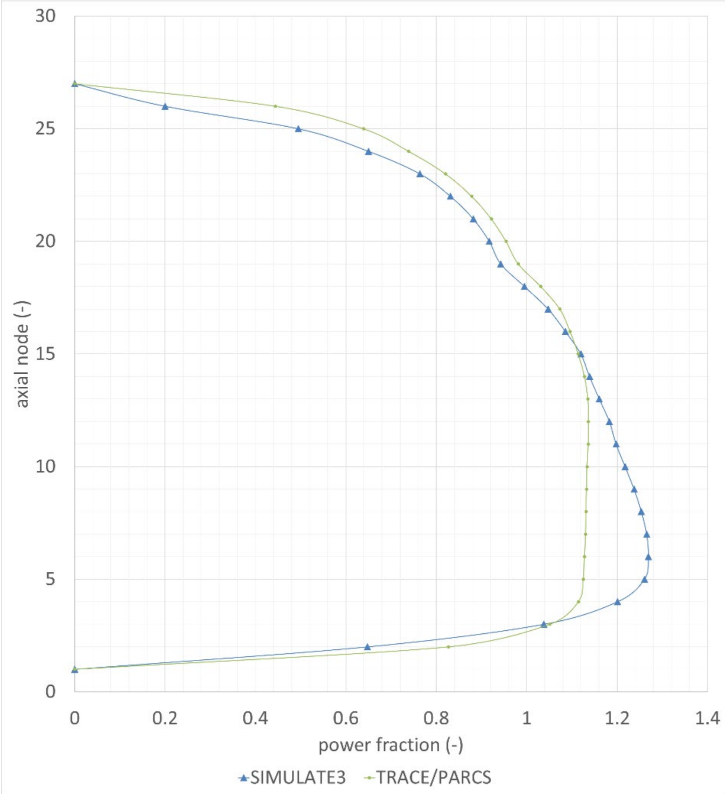

Figure 5.- Intermediate region average axial power distribution of TRACE/PARCS (NEMTAB) vs SIMULATE-3.

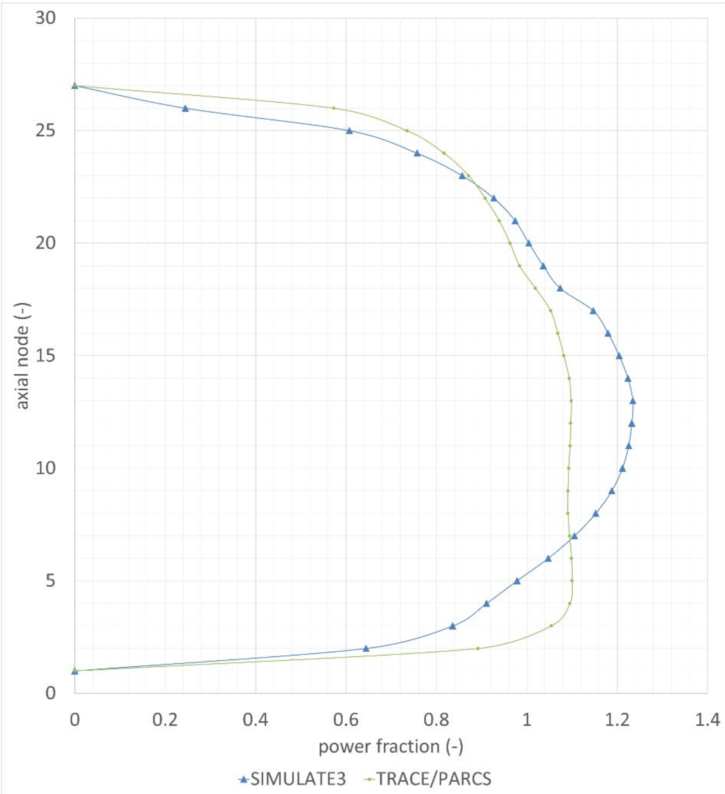

Figure 6.- Outer region average axial power distribution of TRACE/PARCS (NEMTAB) vs SIMULATE-3.

The results of figures 4 to 6 are complemented with the corresponding root mean square (RMS) error. The inner region has 11.45% of error, the intermediate region a 9.72% and the outer region a 13.13%.

Figure 7 shows the radial distribution of the relative error of the average power fraction in comparison with the reference code SIMULATE-3.

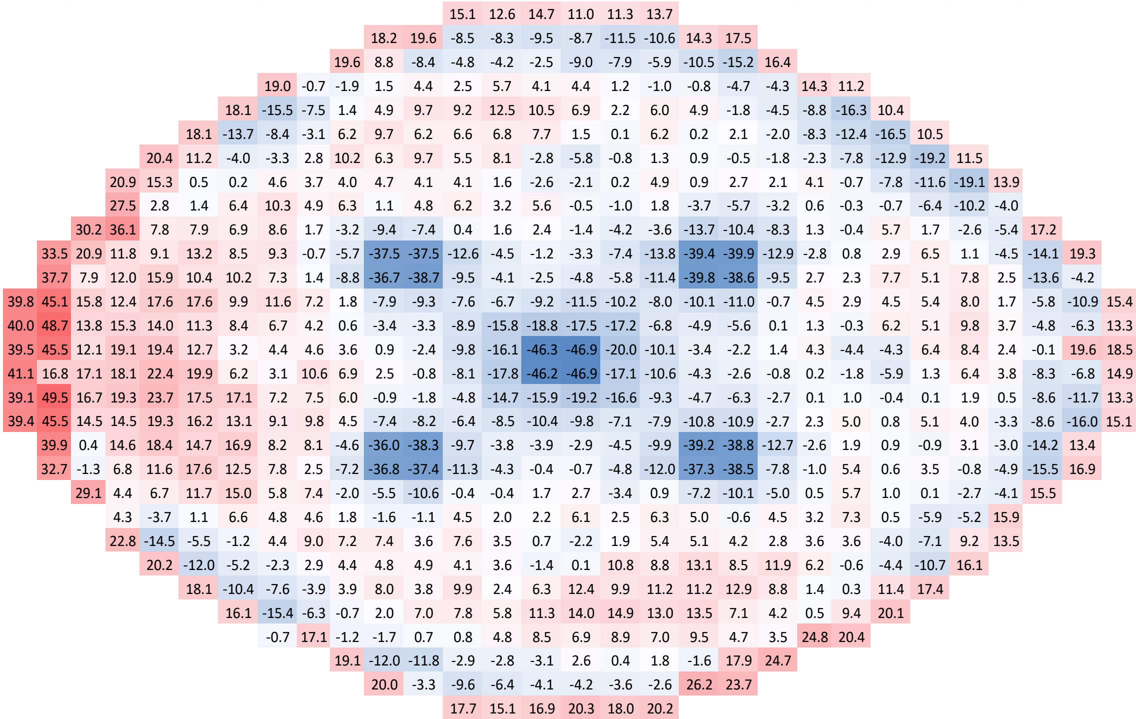

Figure 7.- Radial power distribution for Turbine Trip Fuel Cycle 18 core configuration: TRACE/PARCS (NEMTAB) vs SIMULATE-3.

The results are summarized in table 3.

Table 3.- Summary of the code-to-code verification TRACE/PARCS (NEMTAB) vs SIMULATE-3 for Turbine Trip of Fuel Cycle 18 core configuration.

Axial power distribution RMS Error (%)	Radial power distribution RMS Error (%)	*k-effective* absolute error (pcm)
10.39	6.83	-201.70

In view of the results, the steady state simulation is considered converged with sufficient level of agreement. The radial profile shows a slight discrepancy in the radial relative error of the code-to-code verification. The analysis of the authors reveals two limitations in TRACE/PARCS with NEMTAB libraries regarding the verification against the results of SIMULATE-3. First of all, the *nemtabr* file contains the Cross Section Data for a unique controlled composition in the whole core. The modeling of the core in KKL considers a high detail of control rod definition, where there exist more than one compositions, among other aspects. SIMTAB is designed to account only one control rod worth, therefore, the channels affected by other compositions of control rods, other than the one considered in the *nemtabr* library will show the corresponding discrepancy due to the difference in the absorption cross section data. Figure 7 confirms the position of the inserted control rods that use the composition not considered by *nemtabr*. Moreover, the reader must notice that PARCS is working with the same input data for the NK calculations, i.e. the Cross Section Data from SIMULATE-3 and the thermal-hydraulic boundary conditions, but in the coupled model, the TH is being handled by TRACE. Furthermore, the core model of TRACE is designed lumping the fuel channels in three different radial nodes. There is an inner channel component in TRACE gathering the 84 inner fuel assemblies. A second TRACE channel lumps the intermediate ring of the core, collapsing 479 fuel assemblies. Lastly, a peripheral ring gathers the surrounding 85 fuel assemblies. The coupling of both radial distributions between TRACE and PARCS heads to an averaged use of the thermal-hydraulic variables used for the definition of the Cross Section sets at each neutronics node. The result is the discrepancy observed in the code-to-code verification, added to the different solution of the TH between TRACE and SIMULATE-3. The advantage is the optimization in computational cost.

The main objective of this code-to-code verification is to have a reference of the NK solution, which shows good agreement. The agreement of the prediction of the TH will be confirmed by the validation against real plant data.

The model of TRACE code used for the simulation includes a 3-channel vessel model. The stand-alone models of both TRACE and PARCS have been previously tested and the results have been presented in different scientific congresses (Hidalga et al., 2018). Due to this, there will be no further detailed explanations about the model.

The converged Steady State simulation of TRACE was achieved in 9955 time steps with a computer cost of 1819 seconds in a Linux machine with an Intel Core i7-3770 CPU at 3.40 GHz.

2.2 Validation of the transient results against plant data

The results of the transient simulation with the TH/NK coupled model is validated against the available plant data. The validation of the model assesses the Best Estimate capabilities of the proposed methodology that can be confirmed with the good agreement in the prediction of the results as figures 8 and 9 show. These figures show respectively the power prediction of the core and the pressure evolution.

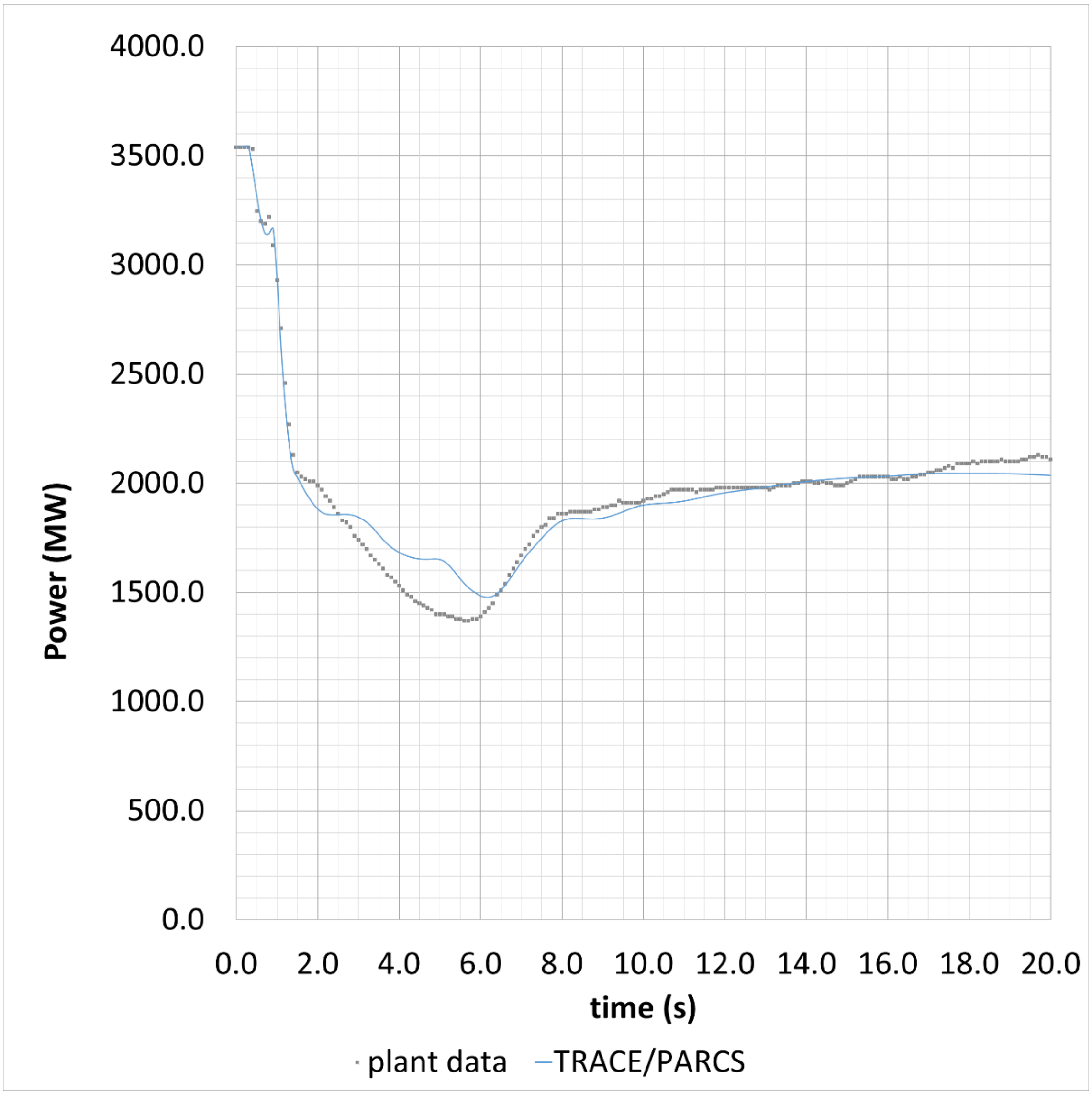

Figure 8.- Total core power evolution: TRACE/PARCS vs Plant Data.

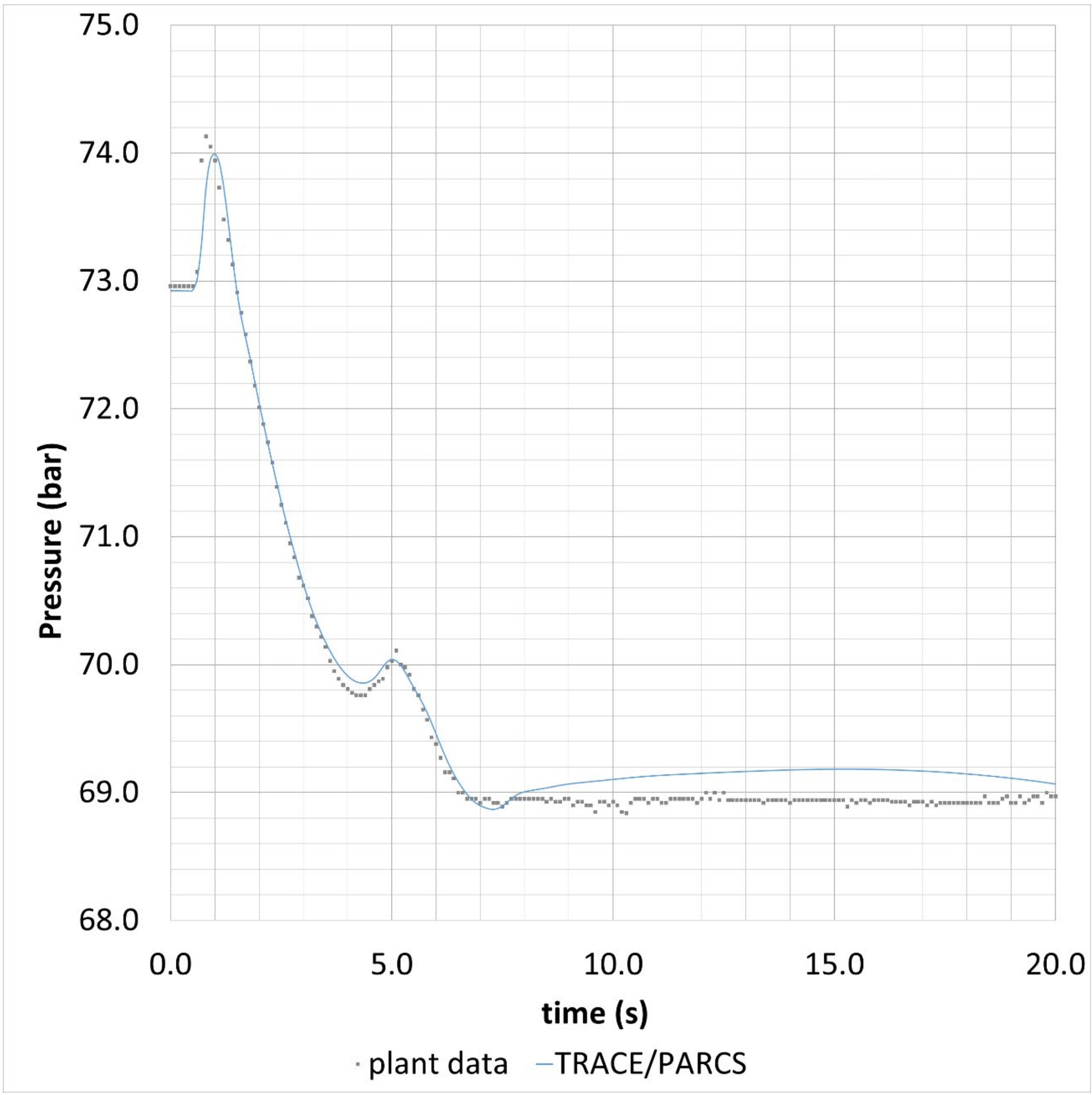

Figure 9.- Dome pressure evolution: TRACE/PARCS vs Plant Data.

The simulation revealed that the maximum pressure achieved in the dome is of 7.40 MPa according to the TRACE/PARCS model prediction. On the other hand, the maximum pressure detected in the reactor vessel by the instrumentation is of 7.41 MPa. The simulation ends with a pressure ramp with an increase of 0.1 MPa approximately. This can be justified with the valves movement of the model. The opening and closure times of the Turbine Bypass Valves and Turbine Control Valves are defined as a forcing function in the TRACE model. Core analysis with CTF-UPVIS/PARCS model

3. CORE ANALYSIS WITH CTF-UPVIS/PARCS MODEL

The simulation with TRACE/PARCS is used to retrieve the core boundary conditions that will be applied for this step. In this case, the evolution of the core pressure will be introduced into the CTF-UPVIS/PARCS coupled model (Abarca et al, 2017) model to perform a Best Estimate simulation of the core in more detailed level i.e., channel-by-channel. This step will allow tracking the critical fuel channel in order to simulate the subsequent steps achieving a more detail level for each step.

The CTF-UPVIS model is a core model defined channel-by-channel. For the model, 648 fuel channels are defined, plus an additional channel working as the core bypass. Each of the channels is connected to the corresponding channel of the PARCS model. The model accounts the feedback between the thermal-hydraulics and the neutron kinetics for each fuel channel. This model has been previously tested and validated and the results have been published in different scientific congresses (Hidalga et al., 2017). For this reason, no further details of the model will be commented.

This step of this methodology must realize the corresponding steady-state simulation of the stand-alone models as for the previous step. The following subsections will show the results of the code-to-code verification.

3.1 Initial conditions for CTF-UPVIS/PARCS core model simulation

The procedure to verify the initial conditions of the CTF-UPVIS/PARCS core model is equivalent to the previous step. First of all, the proposed methodology evaluates the NK part by means of verifying the CTF-UPVIS/PARCS coupled steady state results against the reference code SIMULATE-3. Prior to that, the Stand Alone CTF-UPVIS core model must be simulated, in order to achieve converged initial conditions for the beginning of the transient simulation. The converged Steady State simulation of CTF-UPVIS was achieved in 11232 time steps with a computer cost of 3793.0 seconds with an Intel Core i7-3770 at 3.40 GHz in a Linux machine.

The procedure to verify the initial conditions of the coupled steady state CTF-UPVIS/PARCS is similar to the verification of TRACE/PARCS model. CTF-UPVIS thermal-hydraulic model is defined channel-by-channel. The authors expect better results in this code-to-code verification due to the use of a more detailed scale for the analysis of this fast transient. Figures 10 and 11 show the comparison of the axial and radial power distribution of SIMULATE3 and CTF-UPVIS/PARCS. Table 4 contains the summary of results as for table 3.

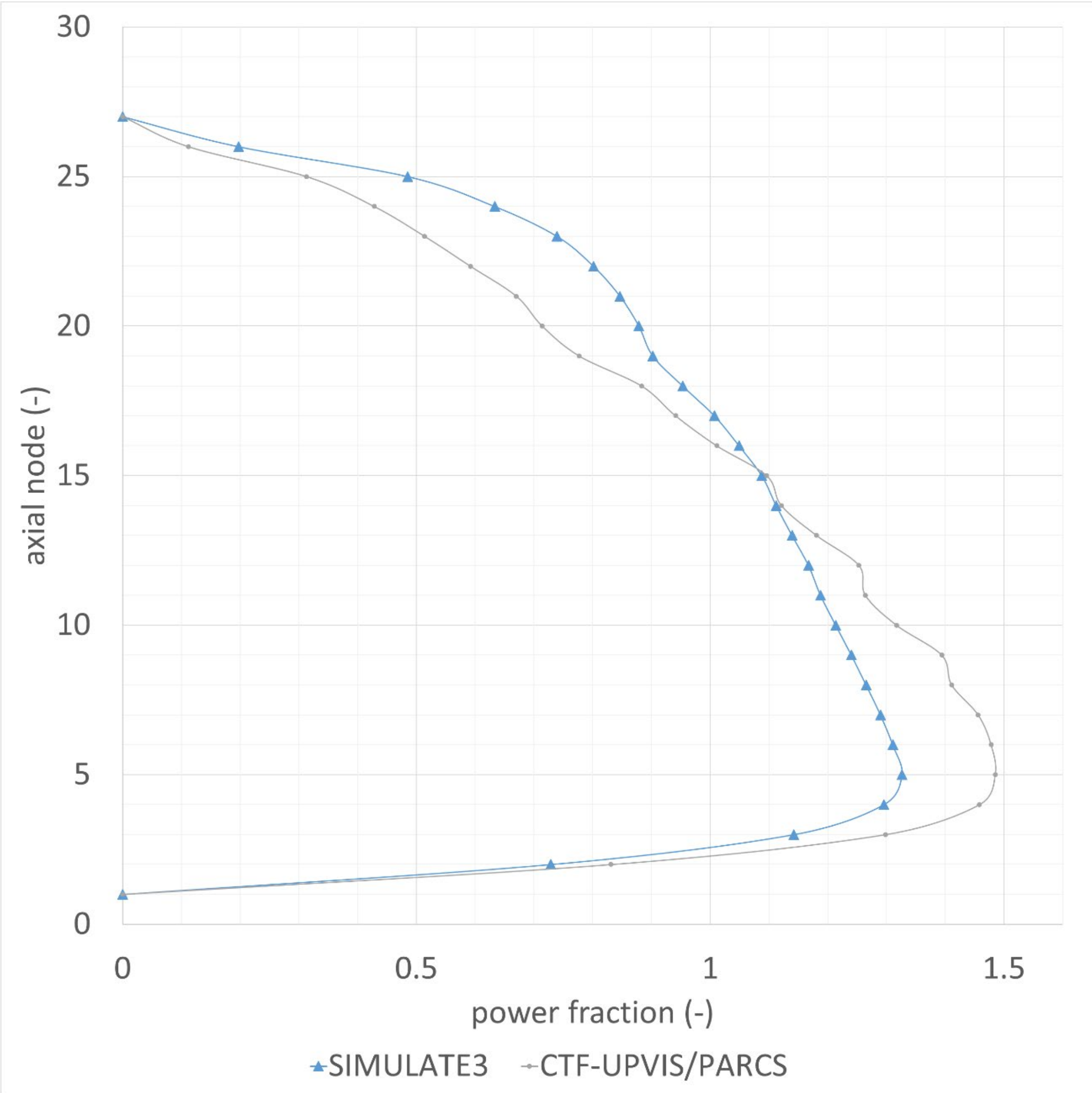

Figure 10.- Axial power distribution for Turbine Trip Fuel Cycle 18 core configuration: CTF-UPVIS/PARCS (NEMTAB) vs SIMULATE-3.

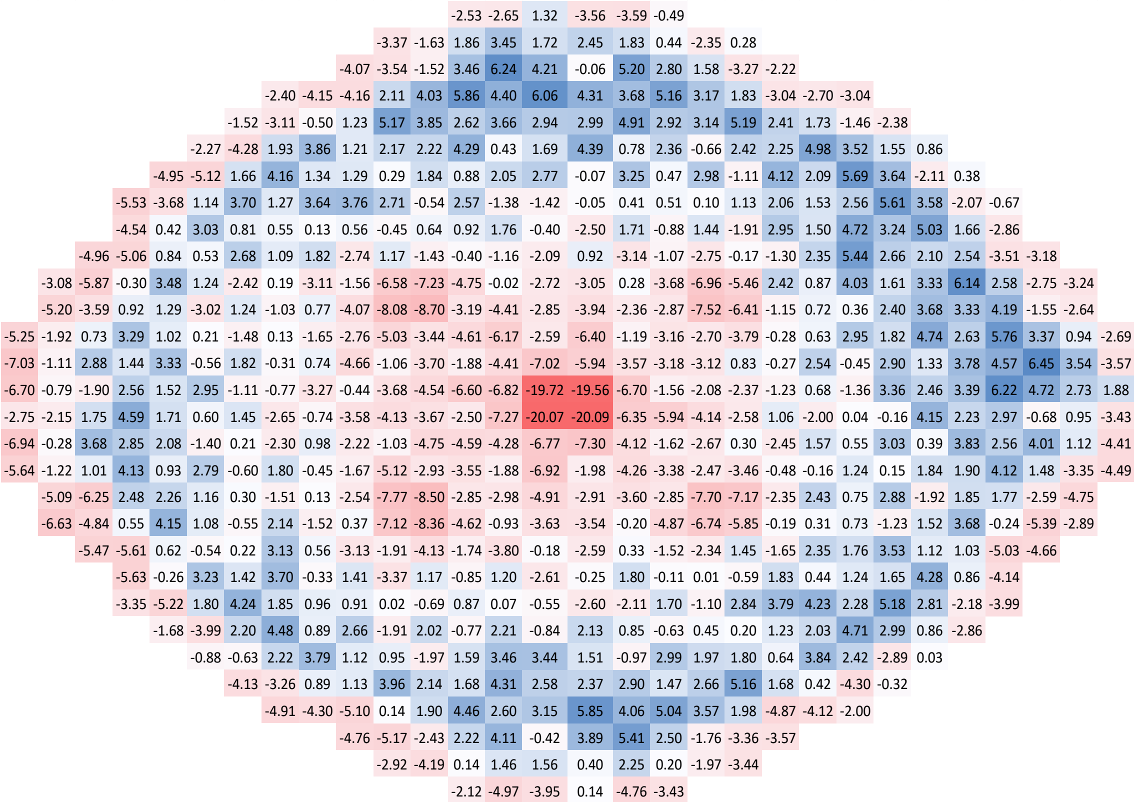

Figure 11.- Radial power distribution for Turbine Trip Fuel Cycle 18 core configuration: CTF-UPVIS/PARCS (NEMTAB) vs SIMULATE-3.

Table 4 summarizes the results of the coupled steady state simulation with CTF-UVIS/PARCS

Table 4.- Summary of the code-to-code verification CTF-UPVIS/PARCS (NEMTAB) vs SIMULATE-3 for Turbine Trip of Fuel Cycle 18 core configuration.

Axial power distribution RMS Error (%)	Radial power distribution RMS Error (%)	*k-effective* absolute error (pcm)
13.60	3.44	129.00

The verification step reveals that the power distribution of CTF-UPVIS/PARCS model shows better agreement with SIMULATE-3 results since the scale of the thermal-hydraulics is similar, i.e. channel-by-channel core model. The CTF-UPVIS model accounts, as SIMULATE-3, the 648 fuel channels. On the other hand, as both models use the same Cross Section data, whether in the TRACE/PARCS model or in the CTF-UPVIS/PARCS model, due to the source of the *nemtab* and *nemtabr* libraries, the results of the comparison of the *k-effective* parameter is similar in both TRACE/PARCS and CTF-UPVIS/PARCS model. As in section 2.2 the overall agreement of the model will be confirmed against real plant data, while this code-to-code verification is used as a reference for the NK calculations.

3.2 CTF-UPVIS/PARCS/NEMTAB Transient Simulation

The results of the transient simulation with the NK/TH coupled model is validated against plant data and compared with the TRACE/PARCS model results. The resulting comparison is shown in figure 12.

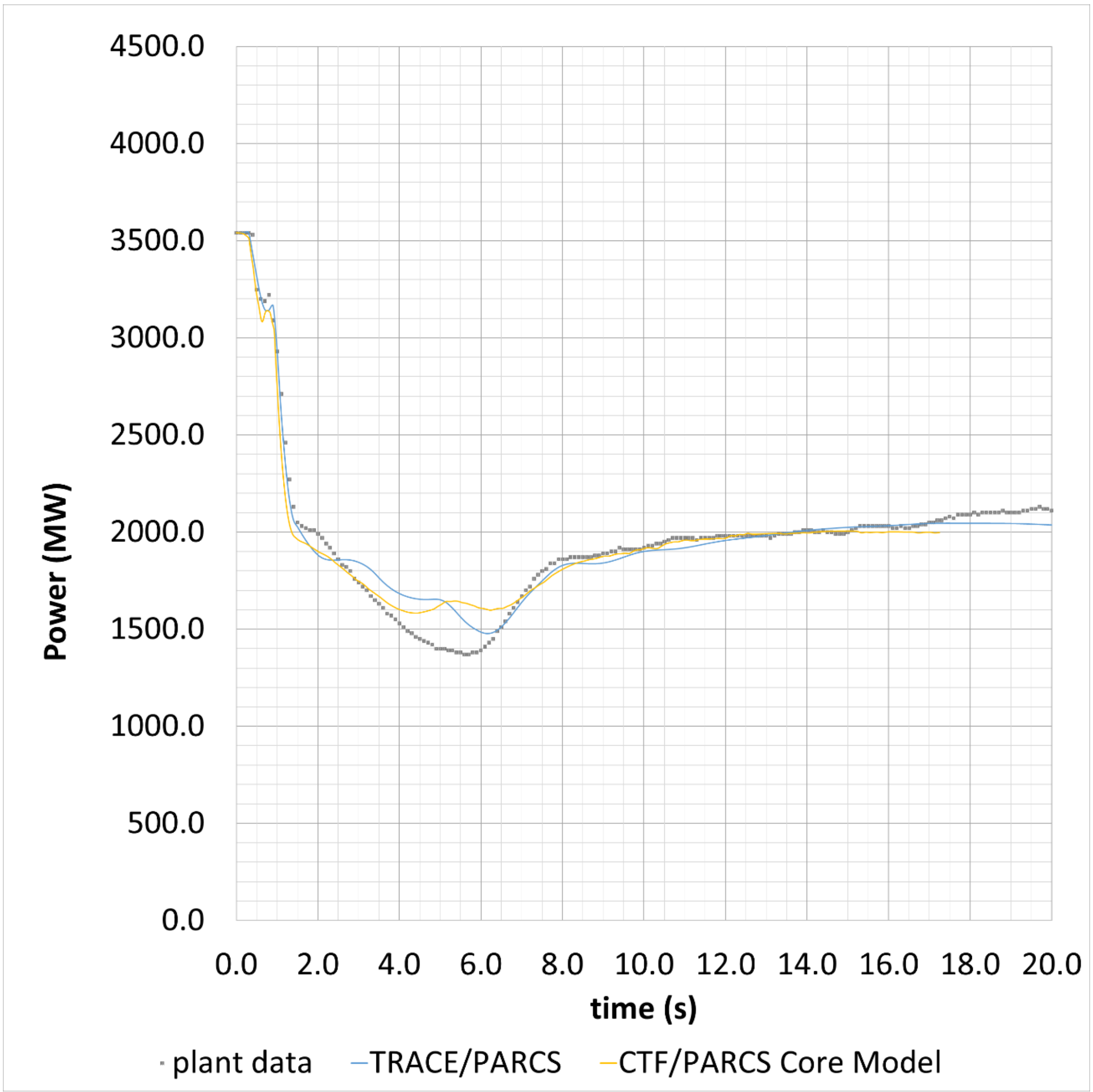

Figure 12.- Total core power evolution: CTF-UPVIS/PARCS vs Plant Data vs TRACE/PARCS.

The pressure evolution is a forcing function retrieved from the previous step, so there is no need to perform the safety assessment of the pressure acceptance criteria. Nevertheless, the scale of this simulation allows obtaining the critical fuel channel based on the minimum CPR criterion.

As figure 12 shows, the CTF-UPVIS/PARCS model shows good agreement in the power prediction. The results of the CTF-UPVIS/PARCS channel-by-channel core model are better than the 3-Channel TRACE/PARCS model. This confirms that a more detailed scale lead to more accurate results.

3.3 Safety Analysis criteria for the CTF-UPVIS/PARCS simulation

The objective of applying step by step the multi-scale and multi-physics methodology is to achieve sufficient detail level. According to the approaches for the safety analysis commented in the introduction the Best Estimate procedure would be done by introducing the corresponding uncertainties in this level and analyzing the propagation until the final step.

On the other hand, the figure of merit as the Acceptance Criteria sets for the fuel integrity is the minimum Critical Power Ratio (CPR). The CPR is the rate between the power level at which the heat transfer through the fuel cladding is compromised in a way that can damage the fuel rod and the predicted power during the transient. Hence, when the CPR is approaching the unit, the fuel cladding integrity is being compromised.

On the other hand, the figure of merit as the Acceptance Criteria sets for the fuel integrity is the MCPR. The CPR is the rate between the power level at which the heat transfer through the fuel cladding is compromised in a way that can damage the fuel rod and the predicted power during the transient. Hence, when the CPR is approaching the unit, the fuel cladding integrity is being compromised.

This criterion is used to track the critical fuel channel. By means of an application developed in MATLAB© programming language, the output files of CTF-UPVIS/PARCS are post-processed. Figure 13 shows the evolution of the MCPR for the most critical fuel channel of the core model.

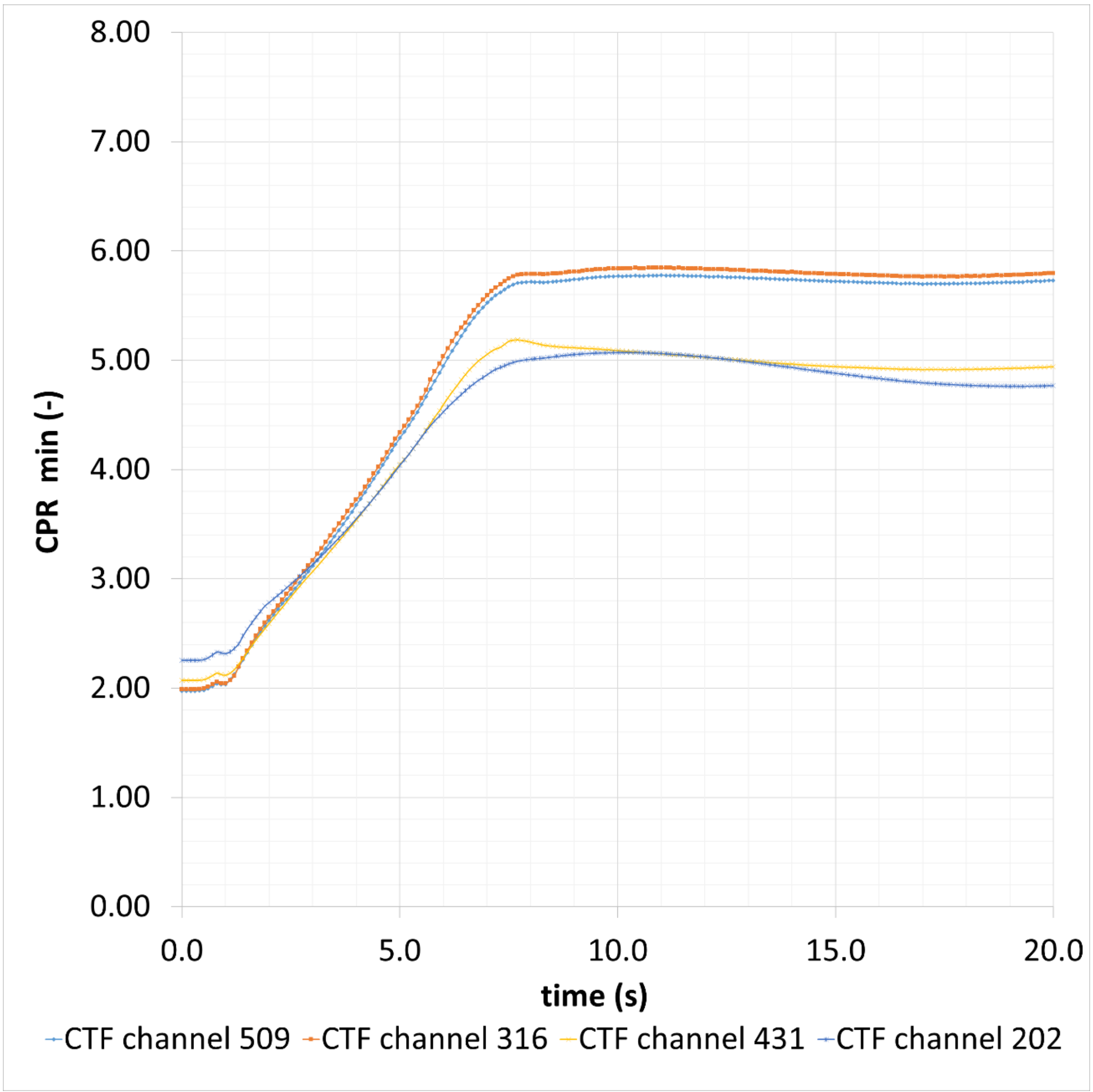

Figure 13.- Evolution of the MCPR predicted by the CTF-UPVIS/PARCS core model and the critical fuel channels detected.

As the results show, the global MCPR is predicted for fuel channels 509, 431, 316 and 202 at different transient times. Due to the nature of the transient case, the MCPR is achieved at the beginning of the simulation and is increased in short time due to the power decrease. Table 5 shows the summary of the resulting MCPR criterion in the CTF-UPVIS/PARCS simulation whilst figure 14 shows the radial location of the critical fuel assemblies.

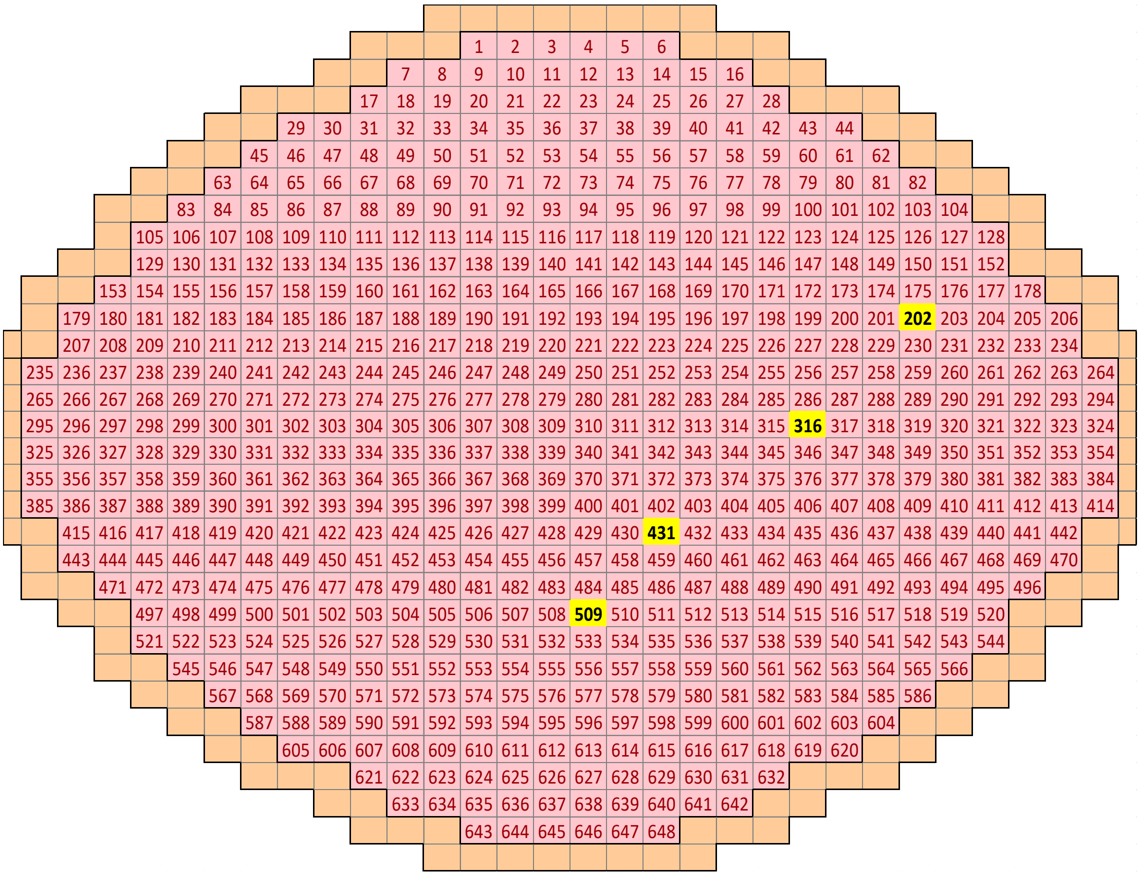

Figure 14.- Location of the affected channels with minimum CPR along the transient for the CTF-UPVIS/PARCS core model.

At the beginning of the TT fuel channel 509 shows the lower CPR. At the end of the transient, when the reactor is stabilized, fuel channel 202 shows the MCPR.

Table 5.- Summary of MCPR prediction in the CTF-UPVIS/PARCS core model.

Channel	MCPR (-)
509	1.97
316	1.99
413	2.07
202	2.25

Channels 509 and 316 experiment similar MCPR at the beginning of the transient. Due to this, the critical case is assumed for channel 509 that will be implemented in the following steps of this methodology.

In this paper, the multi-scale and multi-physics methodology will be applied only for one single fuel channel. The fuel channel selected is the 509 due to the aforementioned reasons. In the following subsection the next step is explained and simulated. A more extended analysis would consider applying this methodology to every fuel channel that has endured the MCPR. Table 5 shows very small difference between the MCPR of the critical channels (specially for channels 509 and 316) and a more detailed scale analyzed with a BE tool such as CTF-UPVIS can reveal the MCPR in other fuel channel other that 509 if such small differences exist. Fuel level analysis with CTF-UPVIS pin-by-pin model

This step moves to next level of detail. The results from the previous step are loaded as boundary conditions for this step. The fuel analysis is realized using the thermal-hydraulic code CTF-UPVIS where the boundary conditions are introduced as temporal forcing functions from the previous step. To perform this step, it is necessary to account the database of KKL. According to the available map of the core it is possible to select which fuel model corresponds to the channel 509. The consulted data reveals the fuel type. The resources of this multi-scale and multi-physics methodology provide this step with the corresponding fuel assembly input deck corresponding to the fuel type that has been designed in the data base of fuel models that KKL can use. The selected model has been designed in the framework of a project between ISIRYM/UPV and KKL, and the validation and verification are documented in the corresponding internal report.

The boundary conditions are retrieved from the output files of the previous step and loaded in the input deck automatically. This is done by means of an interface developed in MATLAB© programming language. The simulation of the pin-by-pin model accounts several details including an advance design of the water rods. In addition, the average burnup level is introduced in the fuel pins giving a BE approach to the simulation.

The results of the simulation with CTF-UPVIS and the corresponding post-processing tool allow tracking the critical fuel rod according to the MCPR criterion. Figure 15 shows the MCPR predicted in the fuel model input deck of CTF-UPVIS. The result is compared with the CPR predicted in the previous step, i.e. a coarser scale.

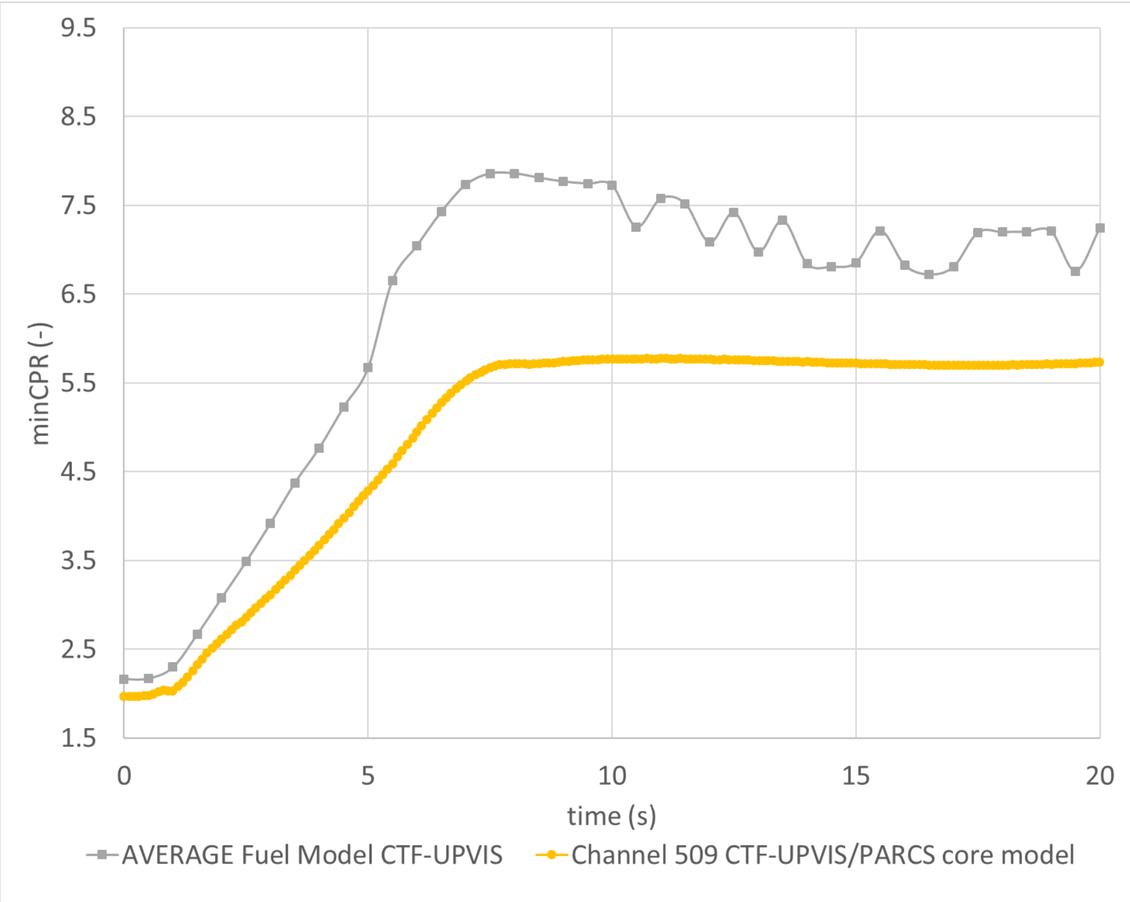

Figure 15.- Comparison of the averaged MCPR of the pin-by-pin model and the fuel channel of the CTF-UPVIS/PARCS core model.

The simulation of this step allows tracking the critical fuel pin regarding the MCPR. It can be observed that the MCPR is higher with a more detailed scale. The results of the application of this methodology will be discussed in next section, as well as the necessary ongoing work for concluding the remaining steps of the methodology.

As the results show, the minimum CPR is predicted for fuel channels 509, 431, 316 and 202. Due to the nature of the transient case, the minimum CPR is achieved at the beginning of the simulation and is increased in short time due to the power decrease. Table 5 shows the summary of the resulting minimum CPR criterion in the CTF-UPVIS/PARCS simulation.

Figures 9 and 10 show the evolution and location of the minimum CPR along the transient. At the beginning of the TT fuel channel 509 shows the lower CPR. At the end of the transient, when the reactor is stabilized, fuel channel 202 shows the minimum CPR.

Channels 509 and 316 experiment similar minimum CPR at the beginning of the transient. Due to this, the critical case is assumed for channel 509, that will be implemented in the following steps of this methodology.

In this paper, the multi-scale and multi-physics methodology will be applied only for one single fuel channel. The fuel channel selected is the 509 due to the aforementioned reasons. In the following subsection the next step is explained and simulated. A more extended analysis would consider applying this methodology to every fuel channel that has endured the minimum CPR. Table 6 shows very small difference between the minimum CPR of the critical channels (specially for channels 509 and 316) and a more detailed scale analyzed with a BE tool such as CTF-UPVIS can reveal the minimum CPR in other fuel channel other that 509 if such small differences exist. The extended application of the proposed methodology is not reported since the intention of this document is to present the general lines of this methodology.

4. CONCLUSIONS AND ONGOING WORK

The results presented in this paper show the capabilities of a multi-scale and multi-physics Methodology to evaluate at different scales the Safety Variables suggested by the USNRC. Within the different steps of the methodology it has been possible to evaluate the minimum Critical Power Ratio from a coarser scale to a finer one, applying a Best Estimate approach by means of Best Estimate codes such as TRACE and CTF-UPVIS and the coupled Thermal-Hydraulics and Neutron Kinetics.

The proposed methodology accounts the avoiding of the user effect by means of realizing the corresponding pre- and post-processing of data automatically using MATLAB© based application that track the critical element to be analyzed in the next step, i.e. the fuel channel and the fuel pin, according to the stablished safety criterion, in this case, the minimum CPR.

The results showed good agreement in the validation against plan data, therefore, the proposed simulation tools have proven to be of significant relevancy. Moreover, the use of a fine mesh i.e. the channel-by-channel core model, not only shows very good agreement predicting the power evolution but also allows tracking the critical fuel assembly according to the MCPR criterion.

The completion of the methodology will be presented on a continuation paper. The ongoing work consists of the analysis at pin level of the critical fuel pin that will be located in the pin-by-pin CTF-UPVIS model. Furthermore, the suite FRAPCON/FRAPTRAN can be used in order to account the effect of the burnup history in the gap conductance, covering the limitations of the thermal-hydraulic code CTF-UPVIS. Moreover, the Best Estimate approach of this methodology is complemented undertaking the corresponding Uncertainty and Sensitivity Analysis to the results of the last step of this methodology. This evaluation will define uncertainty of the margins of the prediction of the minimum CPR using the 95/95 criterion as the USNRC requires.

5. ACKNOWLEDGEMENTS

The authors of this paper acknowledge the interest and economical and technical support of KKL to undertake the development of this project. This acknowledgement is extended to the collaboration of KKL in the research tasks by sharing plant measurements to validate the methodology and to use their software resources to generate useful data for the code-to-code verification.

REFERENCES

Abarca, A., Miró, R., Verdú, G., Bermejo, J. A. Analysis of Thermal-Hydraulic Fluctuations in Trillo NPP with CTF/PARCS v. 2.7 Coupled Code. ASME. ASME J. of Nuclear Rad. Sci. 2016;2(2):021023-021023-5. http://doi.org/10.1115/1.4031660

Abarca, A., et al. "Verification of CTF/PARCSv3.2 Coupled Code in a Turbine Trip Scenario" Proceedings of ICAPP 2017, Japan (2017).

Adu, S., Horvatović, I., D 'auria, F. Application of Best Estimate Plus Uncertainty in Review of Research Reactor Safety Analysis. Nuclear Technology and Radiation Protection. 30. https://www.researchgate.net/publication/276274654_Application_of_Best_Estimate_Plus_Uncertainty_in_Review_of_Research_Reactor_Safety_Analysis

Avramova, M., Ivanov, K. CTF - A Thermal-Hydraulic Subchannel Code for LWRs Transient Analyses User's Manual. Department of Mechanical and Nuclear Engineering Pennsylvania State University University Park, PA 16802.

Barrachina, T., Miró, R., Verdú, G., Collazo, I., González, P., Concejal, A., Ortego, P., & Melara, J. (2010). Turbine trip transient analysis in peach bottom NPP with TRAC-BF1 code and Simtab-1D methodology. Proceedings of SNA + MC2010: Joint international conference on supercomputing in nuclear applications + Monte Carlo 2010 Tokyo, (p. 1630). Japan

Chanaron, B. et al., Advanced multi-physics simulation for reactor safety in the framework of the NURESAFE project. Annals of Nuclear Energy. 84, 166-177. https://doi.org/10.1016/j.anucene.2014.12.013

Dean, D., Rempe, K., Umbarger, J. SIMULATE-3 Advanced Three-Dimensional Two-Group Reactor Analysis Code. Studsvik Scanpower, Inc. 2007.

Downar, T., et al. "PARCS v3.0 U.S. NRC Core Neutronics Simulator USER MANUAL" Department of Nuclear Engineering and Radiological Sciences University of Michigan Ann Arbor, MI, USA (2014).

Ekberg, K., Forssén, B., Knott, D., Umbarger, J. A., Edenius, M. CASMO-4 User's Manual. Studsvik of America, Inc. Studsvik Core Analysis AB. 1995.

ENSI-G20/e "Reactor Core, Fuel Assemblies and Control Rods: Design and Operation" SWITZERLAND 2015.

Fourth COBRA-TF (CTF) User's Group Meeting - CTF-4. Erlangen, Germany, 2017.

Garcia-Fenoll, et al. REA 3D-dynamic analysis in Almaraz NPP with RELAP5/PARCS v2.7 and SIMTAB cross-sections tables. Progress in Nuclear Energy. 53, 1167-1180. https://doi.org/10.1016/j.pnucene.2011.07.012

García-Herranz, N. et al. Multiscale neutronics/thermal-hydraulics coupling with COBAYA4 code for pin-by-pin PWR transient analysis. Nuclear Engineering and Design. 321, 38-47, https://doi.org/10.1016/j.nucengdes.2017.03.017

Hidalga, P., et al. "Evaluation of a Bepu Methodology with State-Of-The-Art Tools in a Non-Realistic Turbine Trip with ATWS for a BWR" BEPU 2018, Italy.

Hidalga, P., et al. "Integrated methodology for the analysis of fuel integrity in high burnup fuel assemblies using CTF-UPVIS, FRAPCON and FRAPTRAN codes" NURETH-17, CHINA 2017.

Hidalga, P.,et al. "TRACE Coupled with PARCS Benchmark Against Leibstadt Plant Data During The Turbine Trip Test" INAC, Brazil 2013.

IAEA, 2010. Deterministic Safety Analysis for Nuclear Power Plants. Specific Safety Guide No. SSG-2. International Atomic Energy Agency, Vienna.

Jiménez Escalante, J. et al. Application of the ATHLET/COBRA-TF thermal-hydraulics coupled code to the analysis of BWR ATWS. Nuclear Engineering and Design. 321, 318-327, https://doi.org/10.1016/j.nucengdes.2016.10.001

Papadopoulos, P., Sekhri, A. Development of a Coupled TRACE/PARCS Model for KKL and Benchmark Against the Turbine Trip Test. USNRC International Agreement Report. April 2017. https://www.nrc.gov/docs/ML1709/ML17096A273.pdf

The MathWorks Inc., 2000, Simulink/Dynamic System Simulation for Matlab.

USNRC, Fall 2017 CAMP Meeting. Thermal-Hydraulics Codes News vol. 18, 2018.

USNRC, NUREG-0800, Chapter 15 Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants: LWR Edition - Transient and Accident Analysis. 2007. https://www.nrc.gov/docs/ML0707/ML070710376.pdf

USNRC, NUREG-0800, Chapter 15.2.1-15.2.5 Standard Review Plan for the Loss of external load; Turbine trip; loss of condenser vacuum; closure of main steam isolation valve (BWR); and steam pressure regulator failure (PWR) USA. 2007. https://www.nrc.gov/docs/ML0703/ML070300702.pdf

USNRC, "Regulatory Guide 1.105: Setpoints for Safety-Related Instrumentation" USA. 1999. https://www.nrc.gov/docs/ML9935/ML993560062.pdf

USNRC. TRACE V5.0 USER'S MANUAL Volume 1: Input Specification. Division of Safety Analysis Office of Nuclear Regulatory Research U. S. Nuclear Regulatory Commission Washington, DC 20555-0001. 2011.