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Investigations of the VVER-1000 Coolant Transient Benchmark I with the Coupled Code System RELAP5/PARCS

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ABSTRACT

This report presents the investigations performed to validate the coupled code RELAP5/PARCS using data of a pump restart-up test carried out at the Kozloduy nuclear power plant (KNPP).
An integral plant model—including the relevant primary and secondary systems as well as a detailed model of the reactor pressure vessel and of the core—was developed for RELAP5 and PARCS to simulate the pump restart-up test. From the comparison of the thermal-hydraulic data obtained during the tests with the code's predictions, it can be stated that the overall trends of most plant parameters are in a reasonable agreement with the experimental data. The investigations have shown some limitations of 1D thermal-hydraulic coarse models of the downcomer, lower, and upper plenum. Hence, multidimensional thermal-hydraulic models are needed for a more realistic description of the coolant mixing phenomena within the reactor pressure vessel.

FOREWORD

This validation report presents the investigations performed by the Forschungszentrum Karlsruhe GmbH to validate the coupled code system RELAP5/PARCS. The plant data used for this work was distributed in the frame of the OECD/NEA VVER-1000 Coolant Tansient Benchmark. A pump restart-up test was performed at the Kozloduy nuclear power plant (KNPP) in Bulgaria. This type of data is very unique for the validation of the neutron physics/thermal hydraulic coupled codes. International code validation and benchmarking activities are very important to increase the confidence of coupled codes. The CAMP-Program of the US NRC is one of the most important and long lasting activities to foster the improvement of thermal hydraulics and coupled neutronic/thermal hydraulic system codes for light water reactor safety assessment.

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EXECUTIVE SUMMARY

This report presents the validation of the coupled code RELAP5/PARCS using plant data obtained during the commissioning phase of the Kozloduy nuclear power plant unit 6 regarding the pump-restart test.

The measured data were made available to a wider community in the frame of the Organization for Economic Cooperation and Development/Nuclear Energy Agency/U.S. Nuclear Regulatory Commission (OECD/NEA/NRC) VVER-1000 Coolant Transient Benchmark Phase 1. This benchmark includes the following exercises: (1) investigation of the integral plant response using a best-estimate thermal-hydraulic system code with a point kinetics model, (2) analysis of the core response for given initial and transient thermal-hydraulic boundary conditions using a coupled code system with 3D-neutron kinetics model, and (3) investigation of the integral plant response using a best-estimate coupled-code system with 3D-neutron kinetics.

Because the reactor was operated at low power with only three main coolant pumps in operation, complex flow conditions existed within the reactor pressure vessel (RPV) already before the test began. These conditions were characterized by coolant mixing in downcomer and upper plenum caused by the reverse flow through the loop-3 (pump was not in operation). The test was initiated by switching on the main coolant pump of loop-3 leading to a reversal of the flow through the respective loop. After about 15 s the mass flow rate through this loop reaches values comparable with the one of the other loops. During this time period, the increased primary coolant flow causes a reduction of the core-averaged coolant temperature and thus an increase of the core power. Later on, the power stabilizes at a level higher than the initial power.

In this analysis, special attention is paid to the prediction of the spatial asymmetrical core cooling during the test and its effects on the local power distribution within the core. The code's predictions are strongly influenced by the way the analyst models the coolant mixing by means of 1D thermal-hydraulic codes.

An integral plant model—including the relevant primary and secondary systems as well as a detailed model of the reactor pressure vessel and of the core—was developed for RELAP5 and PARCS to describe the key phenomena within the reactor pressure vessel appropriately.

Selected results of these investigations will be presented and discussed. A comparison of the thermal-hydraulic data obtained during the tests with the code's predictions showed that despite the use of the 1D coarse model, the overall trends of most plant parameters are in a reasonable agreement with the experimental data. In addition, multidimensional thermal- hydraulic models are clearly needed for a more realistic description of the coolant-mixing phenomena within the reactor pressure vessel. The coupled code RELAP5/PARCS is able to catch the physics of the investigated test as well as to predict the behavior of the core in case of nonsymmetrical situations in sufficient detail (i.e., at the fuel assembly level that represents a step forward compared to the point kinetics models). But additional model development is necessary for the prediction of local safety parameters based on pin level.

NOMENCLATURE

RELAP5	Reactor loss-of-coolant analysis program
PARCS	Purdue advanced reactor core simulator
VVER	Water-water energy reactor
HFP	Hot full power
HZP	Hot zero power
FA	Fuel assembly
RA	Reflector assembly
MCP	Main coolant pump
BOC	Beginning of cycle
PZR	Pressurizer
PWR	Pressurized water reactor
NPP	Nuclear power plant
EFPD	Effective full power days
NEA	Nuclear Energy Agency
NRC	U.S. Nuclear Regulatory Commission
OECD	Organization for Economic Cooperation and Development

1 Introduction

The Karlsruhe Research Center (FZK) is involved in the overall qualification of computational tools for the safety evaluation of nuclear power plants of different design to improve their prediction capability and acceptability. In this framework, the code RELAP5/PANBOX was qualified within the Organization for Economic Cooperation and Development/Nuclear Energy Agency (OECD/NEA) pressurized-water reactor (PWR) main steam line break (MSLB)-benchmark [San00]. As continuation of this work, partly as a contribution to the international Code Assessment and Maintenance Program (CAMP) of the U.S. Nuclear Regulatory Commission (NRC), the coupled code system RELAP5/PARCS is being validated. The PARCS capabilities for quadratic fuel assembly geometry have been qualified in the frame of both the PWR TMI-1 main steam line break (MSLB) [Kozl00] and the boiling-water reactor (BWR) Peach Bottom Turbine Trip (PBTT) [Bous04] benchmarks. FZK is especially interested in the new PARCS capability [Joo02] for hexagonal geometries. These aspects are important not only for the VVER-type light-water reactor (LWR) but also for innovative reactor concepts. The international OECD/NEA VVER-1000 Coolant Transient Benchmark Phase 1 is an excellent opportunity to validate the overall simulation capability of RELAP5/PARCS regarding both the thermal-hydraulic plant response (RELAP5) using measured plant data and the neutron physics models (PARCS). The Phase 1 of this benchmark is devoted to the analysis of the pump restart test while the other three pumps are in operation and covers following three exercises. Exercise-1 is an investigation of the integral plant response using a best-estimate thermal-hydraulic system code with a point kinetics model. Exercise-2 is an analysis of the core response for given initial and transient thermal-hydraulic boundary conditions using a coupled code system with a three-dimensional (3D)-neutron kinetics model. Exercise-3 is an investigation of the integral plant response using a best-estimate coupled code system with 3D-neutron kinetics. For the analysis of these exercises, the following steps are followed:

- Development of an integral model of the Kozloduy nuclear power plant (NPP) including all major systems for RELAP5.

- Development of a 3D core model for RELAP5/PARCS.

- Integration of the above developed models in one integral coupled model to investigate the plant response.

This report presents both the plant components and the developed models. The performed investigations are described and the main results are discussed in detail by comparing the predictions with the measured data. Final conclusions are drawn and identified points for further investigations are outlined.

2 Short Description of the Reference Plant

The Kozloduy Plant unit 6 was selected as the reference plant for the Benchmark. It is a Russian design VVER-1000 reactor of type W320 with a thermal power of 3,000 MW_{th}. The plant consists of four loops, each one with a horizontal steam generator (SG) and a main coolant pump (MCP) [Ivan02a]. The reactor is equipped with only one turbine. **Figure 1** and **Figure 2** present details of the layout and construction of the plant. As shown in **Figure 3**, the horizontal steam generator is characterized by a large water inventory on the secondary side compared to western-designed vertical steam generators. The reactor pressure vessel design also differs from that of the western pressurized-water reactor (PWR), especially because of the constructive peculiarities in the lower and upper plenum that strongly impact the flow patterns during normal and accidental situations. **Figure 4** shows a vertical cut through the reactor pressure vessel (RPV). The lower plenum consists of an elliptical cone with many perforations that result in a narrowing gap in the direction of the central RPV-point. In addition, 163 support columns are present in the lower plenum where the lower part is a full slab and the upper part is a tube with wall perforations of different sizes. Hence, the flow coming from the downcomer has to pass through very complex flow paths to enter into the core. In the upper plenum, two concentric cylinders with perforations are present where the lower part of the outer cylinder is conic.

As shown in **Figure 5**, the core consists of 163 fuel assemblies (FA) and 48 reflector assemblies (RA). Each fuel assembly has 312 fuel pins that are characterized by a central hole of 0.7 mm diameter. **Table 1** presents the main data about the FA and fuel rod design.

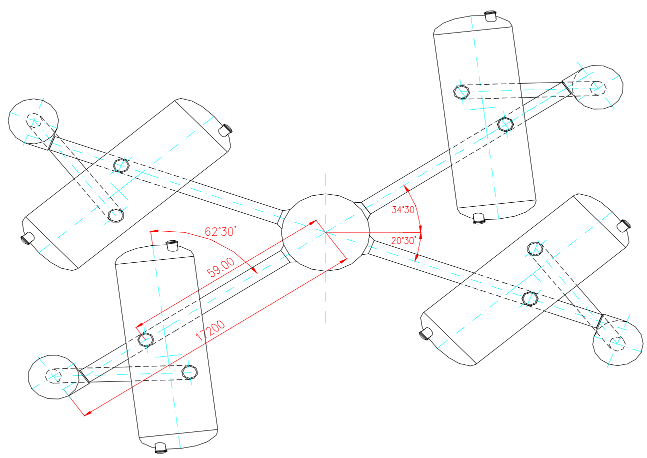


Figure 1. Horizontal Arrangement of the Primary Loops of the VVER-1000 Plant.

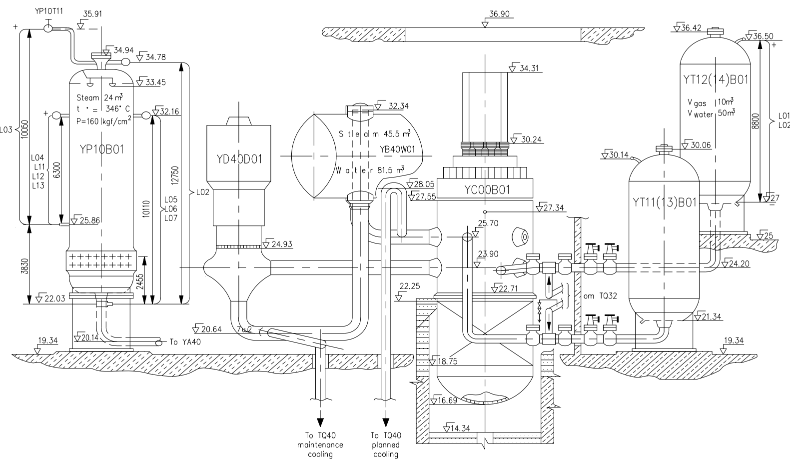
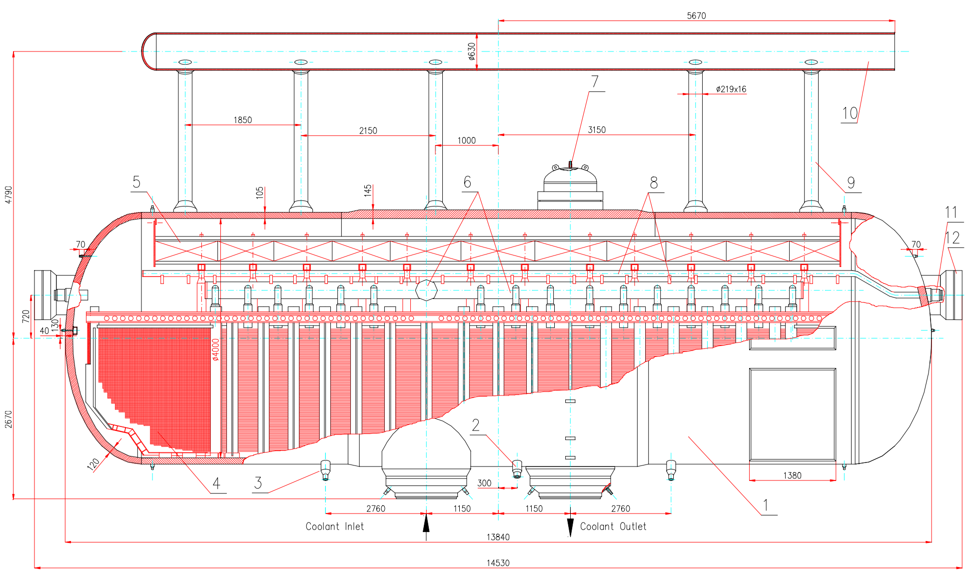


Figure 2. Vertical Arrangement of the VVER-1000 Primary Components**.**



1. Vessel
2. Drainage Nozzle
3. Blow Down Nozzle
4. Heat-Exchange Tubes
5. Separation Units
6. Main Feedwater Spray Unit

7 Gas Removal Nozzle
8 Emergency Feedwater Spray Unit
9 Steam Nozzle
10 Steam Header
11 Emergency Feedwater Nozzle
12 Access Airlock

Figure 3. Typical Horizontal Steam Generator of the VVER-1000 Plant**.**

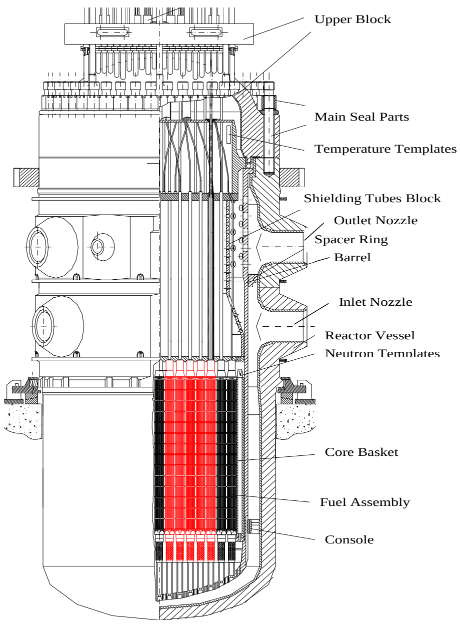


Figure 4. Vertical Cut Through the Reactor Pressure Vessel of the VVER-1000 Plant.

Figure 5 shows the 10 control rod groups (I to X) are arranged in the core symmetrically. The main coolant pumps (MCP-1 to MCP-4) are located asymmetrically with respect to the main axis I-III. The fuel pins are arranged in a triangle within the FA, where the central position is occupied by an empty rod (instrumentation). In addition, the fuel pins have a central hole of about 1.4 mm diameter while the western-type pins are a full slab.

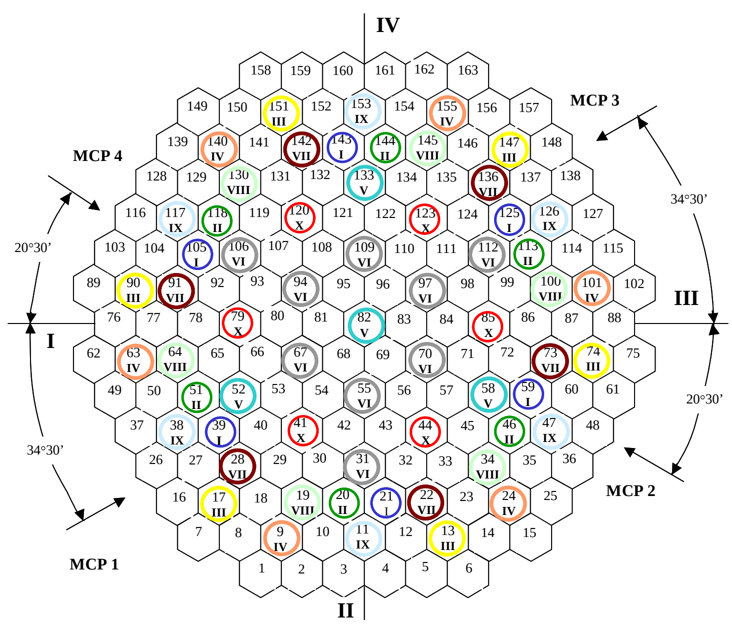


Figure 5. Core Configuration with the Position of the Different Control Rod Groups.

Table 1. Dimensions of the Fuel Rod and Fuel Assembly.

Parameter	Value
Pellet diameter, mm	7.56
Central void diameter, mm	1.4
Clad diameter (outside), mm	9.1
Clad wall thickness, mm	0.69
Fuel rod total length, mm	3837
Fuel rod active length (cold state), mm	3530
Fuel rod active length (hot state), mm	3550
Fuel rod pitch, mm	12.75
Fuel rod grid	Triangular
Number of guide tubes	18
Guide tube diameter (outside), mm	12.6
Guide tube diameter (inside), mm	11.0
Number of fuel pins	312
Number of water rods/assembly	1
Water rod diameter (outside), mm	11.2
Water rod diameter (inside), mm	9.6
FA wrench size, mm	234
FA pitch, mm	236

3 Description of the Scenario

3.1 Pre-Test Plant Conditions

The pump restart test (switch-on MCP-3) was performed during the decommissioning phase (beginning of cycle [BOC]) of the NPP Kozloduy, Unit 6, to investigate the behaviour of the plant. The pre-test phase started reducing the power from 75 percent to about 21 percent by consecutive switching off of MCP-2 and MCP-3. Afterwards, the MCP-2 was switched on some hours before the test. Then the reactor power stabilized at around 27.47 percent of the nominal power (i.e., 824 MW_{th}). The average core exposure amounted to 30.7 effective full power days (EFPD). Before the test, the reactor is operated with only three main coolant pumps (MCPs) at a thermal power of 824 MW_{th} while the MCP-3 is nonoperable. Under such conditions, part of the coolant injected into the downcomer by the three pumps is flowing back through the piping of the affected loop-3. This results in a considerable mixing of cold and hot coolant in the upper plenum.

Table 2 summarizes the main plant parameters of the primary and secondary system just before the test.

Table 2. Measured Plant Data Before the Test with Error Band of Measurements

	Unit	Data	Accuracy
Thermal core power	MW	824	± 60
RCS mass flow rate	Kg/s	13611	± 800
Primary side pressure	MPa	15.60	± 0.3
Sec. side pressure	MPa	5.94	± 0.2
Cold leg temp. loop-1	°K	555.55	± 2
Cold leg temp. loop-2	°K	554.55	± 2
Cold leg temp. loop-3	°K	554.35	± 2
Cold leg temp. loop-4	°K	555.25	± 2
Hot leg temp. loop-1	°K	567.05	± 2
Hot leg temp. loop-2	°K	562.85	± 2
Hot leg temp. loop-3	°K	550.75	± 2
Hot leg temp. loop-4	°K	566.15	± 2
Mass flow rate loop-1	Kg/s	5031	± 200
Mass flow rate loop-2	Kg/s	5069	± 200
Mass flow rate loop-3	Kg/s	-1544	± 200
Mass flow rate loop-4	Kg/s	5075	± 200
PZR water level	m	7.44	± 0.15
Water level in SG-1	m	2.30	± 0.075
Water level in SG-2	m	2.41	± 0.075
Water level in SG-3	m	2.49	± 0.075
Water level in SG-4	m	2.43	± 0.075
DP over core	MPa	0.225	± 0.2
DP over MCP-1	MPa	0.492	± 0.2
DP over MCP-2	MPa	0.469	± 0.2
DP over MCP-3	MPa	0.179	± 0.2
DP over MCP-4	MPa	0.500	± 0.2

3.2 Switch-on of the Main Coolant Pump #3 Test

The test starts by switching on the MCP-3. Immediately, the reverse flow through the loop-3 goes to zero within 15 s leading to an increase of the total mass flow rate through the core. The increased coolant inventory leads to a decrease of the coolant temperature and to an increase of the coolant density. Hence, the core power undergoes initially a rapid increase stabilizing later on at a power level higher than the initial power. Because of the coolant mixing in the downcomer, the mass flow rate of loop-1, -2, and -4 slightly decreases while the mass flow rate of loop-3 greatly increases until around 15 s. Afterwards, the mass flow rates of all four loops are similar. These modifications of the coolant stream influence the heat transfer across the steam generators leading, for example, to an increase of the water level of steam generators 1, 2, and 4 and to nonsymmetrical core cooling. The test is characterized by interactions between the core neutronics and the system thermal hydraulics. The experiment lasted for 130 s., In this time, most of the important primary and secondary thermal hydraulic parameters of the plant were measured.

4 Developed Plant Models and Nodalisation

To study the plant response during the main coolant pump (MCP)-switching-on test with system codes, both the primary and secondary plant systems—including the safety and control systems—need to be represented in the numerical model. All data needed to develop the respective models for PARCS and RELAP5 were taken from the benchmark specifications [Ivan02a]. Based on this information, three models were elaborated—an integral plant model starting with a point kinetics approach (Exercise-1), a three-dimensional (3D) neutron kinetics and thermal-hydraulic core model (Exercise-2), and an integral plant model with a 3D core model (Exercise-3). Details of these models will be described in the following subchapters.

4.1. Integral Plant Model

Figure 6 Nodalization of the Kozloduy Plant (RPV with Two Loops) shows the integral plant model developed for Exercise-1 and Exercise-3, where only two of the four loops are exhibited. This model represents the most relevant primary, secondary, and safety systems of the Kozloduy plant [Metz03]. For Exercise-1, a point kinetics model was implemented. The ***Core*** (volumes 845 and 843) is represented by two parallel volumes, one representing the core average channel and the other one the core bypass. The ***downcomer*** (volumes 108, 208, 308, and 408) is represented in four equal parts, each one connected to one loop so that the complex flow conditions prevailing during the pre-test phase and during the first 15 s of the transient are simulated appropriately. The ***primary circuit*** consisting of the piping system (loop-1: volumes 146, 140, 141, 142, 144, and 145); the pumps (volumes 144, 244, 344, and 444); and the steam generator tubes (SG-1 volumes: 120, 121, and 122) is fully incorporated in the model. In addition, the ***pressurizer (PZR)*** with the four groups of heaters is included in the model. Moreover, the ***make-up and drainage system*** is included in the model. The full data of the Russian-type MCPs are taken from the specifications. Each ***steam generator (SG)*** consists of 11,000 tubes that are horizontally arranged between the hot and cold collector tubes. They are vertically grouped in three units associated to the primary and secondary volumes.

The secondary side of the steam generators is characterized by a complex 3D flow inside this big volume. The back flow in the SG-downcomer (SG-1 volumes: 150, 151, 152, and 109) also is represented in the model (**Figure 6**). The feedwater system (SG-1: volume 190) is simply modelled by a volume providing a constant coolant mass flow with the predefined coolant temperature. No emergency feed water system is considered in the model because these systems are not expected to be activated during the transient. The steam lines (loop-1 volumes: 181, etc.) are in detail modelled including the valves, common header, turbine stop valves and the associated safety steam valves, and the steam dump valve groups. The core fuel pins, the steam generator tubes, the PZR-heaters as well as the walls of all relevant primary and secondary systems (RPV, cold and hot legs, steam generator shell) are considered in the model as heat structure components with its respective heat transfer area, heater diameter, material data, and heat source when available. They are connected to the corresponding fluid volumes via convective boundary conditions. In the point kinetics model, the given neutron physical data characterizing the VVER-1000 fuel like prompt neutron lifetime, effective fraction of delayed neutrons, decay constants of delayed neutrons, axial power profile, moderator and Doppler reactivity coefficients are implemented. The Doppler feedback is calculated using the following Doppler temperature (TDoppler) instead of the volume-averaged fuel temperature in all three exercises:

$$T_{Doppler} = 0.7 \cdot T_{fuel}^{surface} + 0.3 \cdot T_{fuel}^{center} \, .$$

The moderator and Doppler reactivity coefficients for this model were given in the Specification as follows:

- Moderator temperature coefficient (MTC): -4.2652 e-3 $/K
- Doppler temperature coefficient (DTC): -2.2853 e-3 $/K

The effective fraction of delay neutrons (ßeff) amounts 0.007268. The axial power profile was predicted by a 3D-neutronic calculation performed by the benchmark team. **Figure 7** shows the axial power shape at the BOC-conditions before the test.

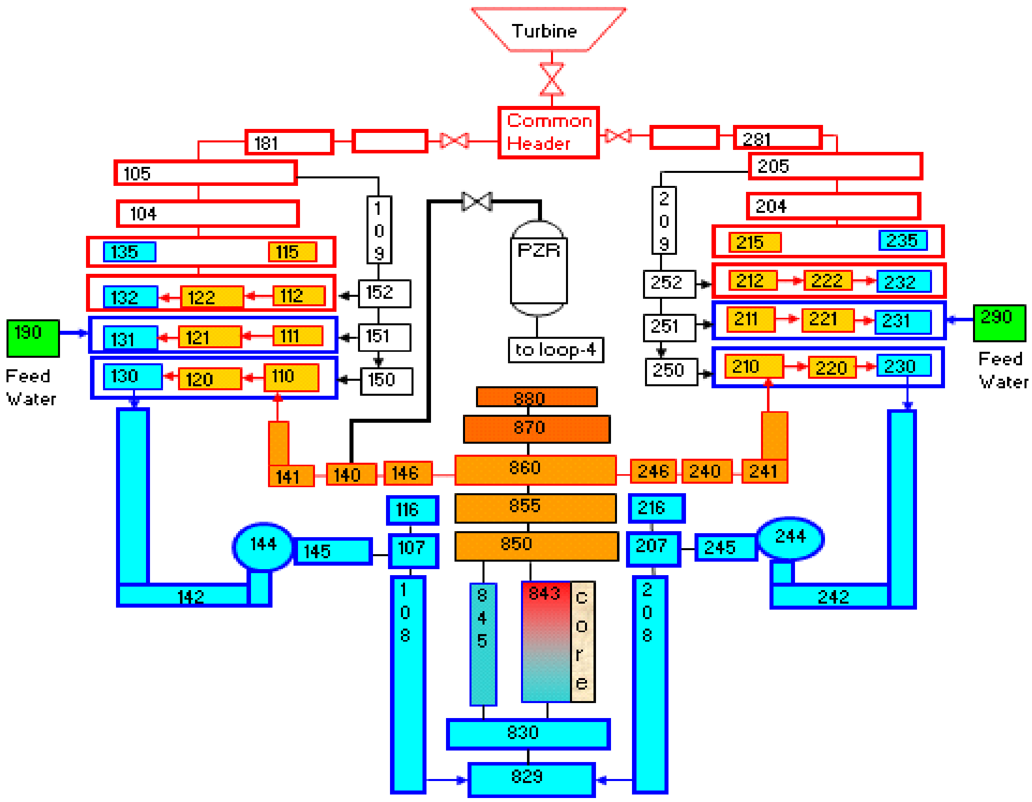


Figure 6 Nodalization of the Kozloduy Plant (RPV with Two Loops)

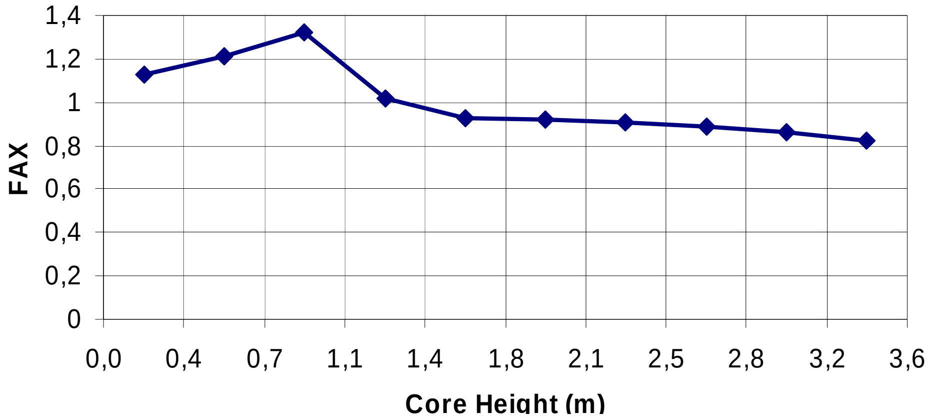


Figure 7. Core-Averaged Axial Power Profile at BOC-Conditions before the test.

4.2. Multidimensional Core Modelling

For Exercise-2 and Exercise-3, a multidimensional core model is needed for both the thermal-hydraulic as well as the neutronic representation of the core for the coupled code system RELAP5/PARCS. This core model is developed for Exercise-2, where only the core behavior is evaluated for given boundary and initial conditions at the core inlet and outlet. Later on this model is fully merged with the integral plant model so that the test can be analyzed with the coupled code system RELAP5/PARCS.

Figure 8 shows that 28 types of fuel assemblies and 2 additional reflector assemblies (radial and axial) are in the core. In addition, the figure shows the enrichment of the pins varies from 2 up to 3.3 percent.

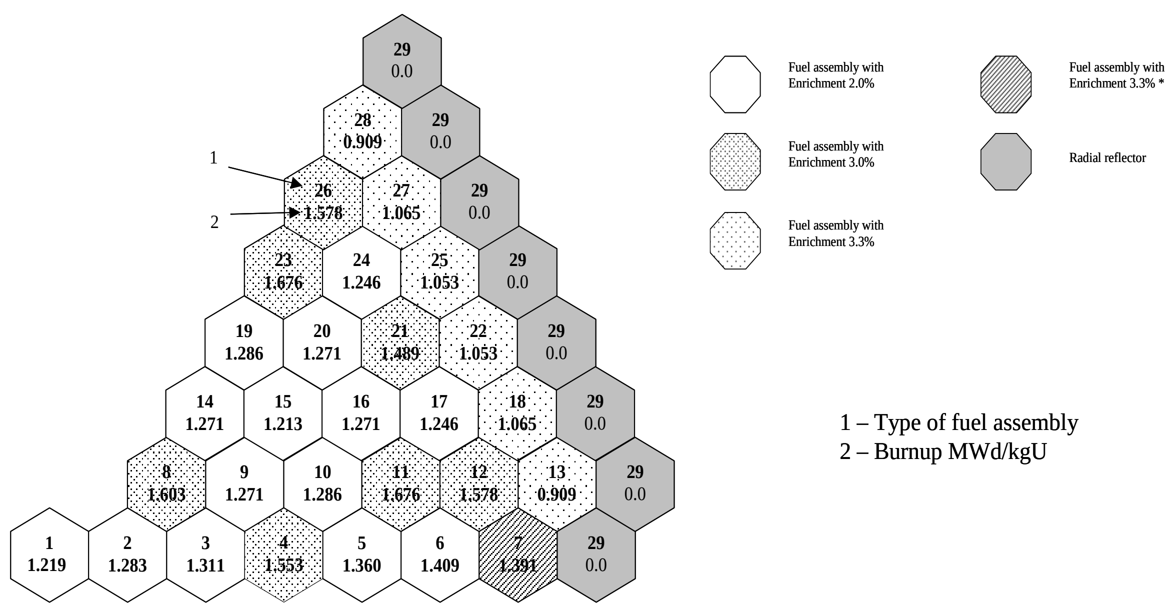


Figure 8. One Sixth of the Core with the Number of FA- and RA-types at BOC.

In the Benchmark Specifications, the 29-FA and -RA-types are subdivided into 12 axial elevations of equal height but of different material composition (10 nodes for the core region and two for the upper and lower reflector). Hence, the neutronic data needed for 3D-core calculations were prepared for a total of 283/260 unrodded/rodded material compositions.

Each one is characterized by unique material properties like enrichment, density as well as burnup, absorber rod history, and spectral history. The Benchmark team delivered these cross-section data to all participants as look-up tables. All neutronic parameters for two energy groups such as diffusion coefficients, scattering, absorption, and fission macroscopic cross sections, assembly discontinuity factors, etc. are given in these libraries. The look-up tables provide additional information about the delayed neutron fractions, decay constants, and neutron velocity. PARCS has several models to read in different formats of cross-section libraries. The nuclear data for the fuel assemblies with absorber rods are delivered in an additional library containing 260 material compositions to account for absorber rods movement.

4.2.1. Neutron kinetics core model

The 3D neutron kinetics model (PARCS) is coupled to RELAP5 via PVM [Bar98]. It solves the time-dependent neutron diffusion equation in two or more energy groups using the triangular polynomial expansion (TPEN) method [Joo02].

Figure 9 shows that, in the core model, eight hexagonal fuel assembly (FA) rings and a reflector assembly (RA) ring are considered to describe the whole core, where each assembly represents a numerical node in the radial plane. **Figure 10** shows that, axially, all assemblies are subdivided into 12 equal nodes, one for the lower and upper reflector and 10 for the fuel zone. A total of 283 material compositions (unrodded) are considered for this problem. The look-up tables are functions of fuel temperature and coolant density. A suitable parameter range of these variables was selected to cover the expected parameter changes for the steady state and during the transient progression. PARCS uses a multidimensional interpolation scheme for the online update of the cross sections during the transient phase in dependence of the actual parameters.

```
                29  29  29  29  29  29  29
             29  29  28  27  25  22  18  13  29  29
           29  13   7  26  24  21  17  12   7  28  29
         29  18  12   6  23  20  16  11   6  26  27  29
       29  22  17  11   5  19  15  10   5  23  24  25  29
     29  25  21  16  10   4  14   9   4  19  20  21  22  29
   29  27  24  20  15   9   3   8   3  14  15  16  17  18  29
 29  28  26  23  19  14   8   2   2   8   9  10  11  12  13  29
   29   7   6   5   4   3   2   1   2   3   4   5   6   7  29
 29  13  12  11  10   9   8   2   2   8  14  19  23  26  28  29
   29  18  17  16  15  14   3   8   3   9  15  20  24  27  29
     29  22  21  20  19   4   9  14   4  10  16  21  25  29
       29  25  24  23   5  10  15  19   5  11  17  22  29
         29  27  26   6  11  16  20  23   6  12  18  29
           29  28   7  12  17  21  24  26   7  13  29
             29  29  13  18  22  25  27  28  29  29
```

Figure 9. Radial Arrangement of the Different FA-Types in the Core (Type 1 to 28).

Axial level=>	1	2	3	4	5	6	7	8	9	10	11	12
assy_type 1	281	1	2	3	4	5	6	7	8	9	10	283
assy_type 2	281	11	12	13	14	15	16	17	18	19	20	283
assy_type 3	281	21	22	23	24	25	26	27	28	29	30	283
assy_type 4	281	31	32	33	34	35	36	37	38	39	40	283
assy_type 5	281	41	42	43	44	45	46	47	48	49	50	283
assy_type 6	281	51	52	53	54	55	56	57	58	59	60	283
assy_type 7	281	61	62	63	64	65	66	67	68	69	70	283
assy_type 8	281	71	72	73	74	75	76	77	78	79	80	283
assy_type 9	281	81	82	83	84	85	86	87	88	89	90	283
assy_type 10	281	91	92	93	94	95	96	97	98	99	100	283
assy_type 11	281	101	102	103	104	105	106	107	108	109	110	283
assy_type 12	281	111	112	113	114	115	116	117	118	119	120	283
assy_type 13	281	121	122	123	124	125	126	127	128	129	130	283
assy_type 14	281	131	132	133	134	135	136	137	138	139	140	283
assy_type 15	281	141	142	143	144	145	146	147	148	149	150	283
assy_type 16	281	151	152	153	154	155	156	157	158	159	160	283
assy_type 17	281	161	162	163	164	165	166	167	168	169	170	283
assy_type 18	281	171	172	173	174	175	176	177	178	179	180	283
assy_type 19	281	181	182	183	184	185	186	187	188	189	190	283
assy_type 20	281	191	192	193	194	195	196	197	198	199	200	283
assy_type 21	281	201	202	203	204	205	206	207	208	209	210	283
assy_type 22	281	211	212	213	214	215	216	217	218	219	220	283
assy_type 23	281	221	222	223	224	225	226	227	228	229	230	283
assy_type 24	281	231	232	233	234	235	236	237	238	239	240	283
assy_type 25	281	241	242	243	244	245	246	247	248	249	250	283
assy_type 26	281	251	252	253	254	255	256	257	258	259	260	283
assy_type 27	281	261	262	263	264	265	266	267	268	269	270	283
assy_type 28	281	271	272	273	274	275	276	277	278	279	280	283
assy_type 29	281	282	282	282	282	282	282	282	282	282	282	283

Figure 10. Axial Discretisation of Each FA/RA-Type in the Core as Used in PARCS

4.2.2. Thermal-hydraulic core model

Figure 11 shows the thermal-hydraulic core model for the coupled calculations consists of 18 parallel channels that are associated with the FA-nodes according to the mapping scheme proposed in the specification. According to this, the whole core is in radial direction divided into 19 thermal-hydraulic channels—18 for the core region and 1 for the radial reflector. All fuel assemblies with the same number are associated to one thermal-hydraulic channel. An additional channel is considered (channel 19) to represent the flow area of the 48 reflector assemblies (RA). **Figure 12** shows that, in axial direction, the parallel channels are subdivided in 12 nodes—the bottom and top nodes for the axial reflector and the remaining 10 nodes for the active core. The additional fluid volumes at core inlet and outlet are also represented because they are needed to define the initial and boundary conditions for Exercise-2. At the core inlet (volume 601 up to 619), the mass flow rate and the coolant temperature are given. The system pressure is defined at the core outlet (volume 800). In the RELAP5-model, 18 ***heat structures components*** representing the 18 groups of FA are modelled. They are linked to the 18 core channels by convective boundary conditions. These heat structures have the same axial nodalization like the corresponding fluid channels. In radial direction each heat structure is subdivided in 7 zones, 4 in the fuel, one gap and two in the cladding material.

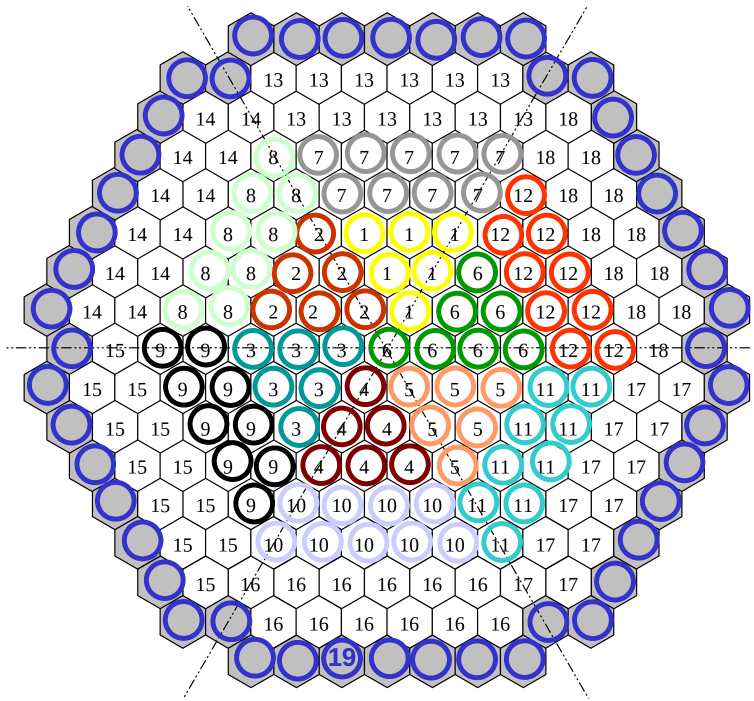


Figure 11. Coolant Channels Numbering for the Mapping Between RELAP5 and PARCS.

The multidimensional neutron kinetics and thermal hydraulics core model was entirely incorporated in the integral plant model to perform the coupled calculations for Exercise-3.

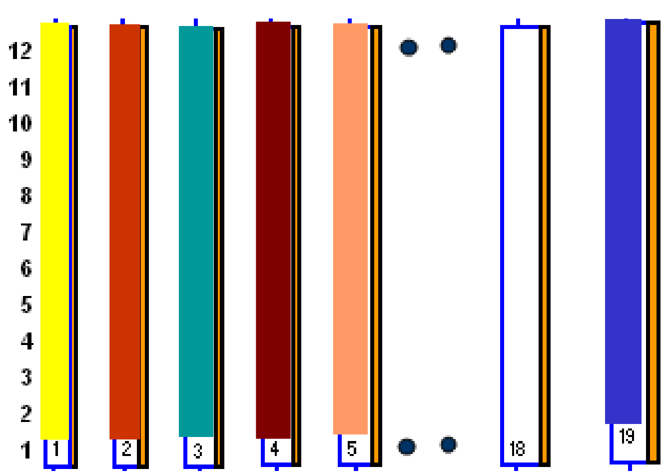


Figure 12. Axial nodalization of the core (RELAP5 part).

5. Performed calculations

According to the main goal of this benchmark, the following investigations were performed:

- Exercise-1: Integral plant simulation with a system code using point kinetics approach (RELAP5/MOD3.3) to demonstrate that the developed plant model is well balanced and that the main steady state plant parameters are well predicted compared to the plant data.

- Exercise-2: Multidimensional core simulation with the coupled system code (RELAP5/PARCS) to test the operability of the coupled code system such as the correctly reading the cross-sections (look-up tables), the appropriateness of the interpolation scheme for the cross-section update, the convergence of both the neutronics, and the thermal-hydraulic model in a coupled calculation, etc.

- Exercise-3: Integral plant simulation with a coupled system code (RELAP5/PARCS) to simulate the overall plant response, especially the space-time core behaviour for the reference scenario.

The numerical simulation of the MCP-switch-on test with the coupled system RELAP5/PARCS was performed on a LINUX platform with PVM-environment including the following steps:

- Run RELAP5-stand alone with the null transient option for some 200 s until stable thermal hydraulic plant conditions are reached.

- Run the RELAP5/PARCS coupled system with the steady state option until the eigenvalue calculation of PARCS converged.

- Then run coupled system code with the transient option restarting from the latter steady-state condition for both codes until 130 s (duration of the test).

In the subsequent chapter, selected results of the performed calculations will be presented.

6. Selected results of the calculations

6.1. Steady state results for integral plant model

As part of the qualification of the initial steady state, attention was paid also to the primary circuit power balance (i.e., the difference between the core power plus main coolant pump (MCP) power and the thermal power transferred over all four steam generators, which amounts around 12 MW. **Table 3** shows the good agreement between data and predictions using RELAP5 and RELAP5/PARCS for the stationary plant conditions before the test demonstrates that the developed integral plant model is appropriate for the subsequent study of the plant response.

Table 3: Comparison of Predicted and Measured Data for Steady State Conditions**.**

Parameter	**Benchmark Data**			**RELAP5**	**Deviation**	**R5/PARCS**
	Unit	Data	Accuracy	Exercise-1	%	Exercise-3
Thermal	MW	824	± 60	824	0	824
RCS mass	Kg/s	13611	± 800	13577	-0.25	13612
Primary side	MPa	15.60	± 0.3	15.62	0.13	15.62
Sec. side	MPa	5.94	± 0.2	6.105	2.83	6.106
Cold leg	°K	555.55	± 2	555.43	-0.02	555.44
Cold leg	°K	554.55	± 2	554.61	0.01	554.62
Cold leg	°K	554.35	± 2	554.94	0.11	554.95
Cold leg	°K	555.25	± 2	555.16	-0.02	555.17
Hot leg	°K	567.05	± 2	566.18	-0.15	566.19
Hot leg	°K	562.85	± 2	563.71	0.15	563.72
Hot leg	°K	550.75	± 2	550.65	-0.02	550.66
Hot leg	°K	566.15	± 2	565.43	-0.13	565.43
Mass flow	Kg/s	5031	± 200	5021	-0.20	5029
Mass flow	Kg/s	5069	± 200	5036	-0.65	5043
Mass flow	Kg/s	-1544	± 200	-1503	-2.66	-1491
Mass flow	Kg/s	5075	± 200	5034	-0.81	5041
PZR water	m	7.44	± 0.15	7.44	0.00	7.44
Water level	m	2.30	± 0.075	2.305	0.22	2.304
Water level	m	2.41	± 0.075	2.409	-0.04	2.409
Water level	m	2.49	± 0.075	2.439	-2.05	2.439
Water level	m	2.43	± 0.075	2.458	1.15	2.457
DP over core	MPa	0.225	± 0.2	0.2570	14.22	0.255
DP over	MPa	0.492	± 0.2	0.4845	-1.52	0.4825
DP over	MPa	0.469	± 0.2	0.4818	2.73	0.4798
DP over	MPa	0.179	± 0.2	0.1811	1.17	0.1787
DP over	MPa	0.500	± 0.2	0.4824	-3.52	0.4804

It can be seen that the deviation of most parameters is very small for both calculations. Some of the parameters are slightly underpredicted, and others are slightly overpredicted by both calculations. This good agreement between data and predictions for the stationary plant conditions before the test demonstrates that the developed integral plant model is appropriate for subsequent studies of the plant response. A prerequisite for the successful simulation of the plant conditions before the test was the appropriate modelling of the downcomer, lower, and upper plenum of the VVER-1000 reactor that was realized based on in-house CFD-simulations [Boett04].

As can be seen in **Figure 13** and **Figure 14**, the flow conditions in the downcomer and the upper plenum are rather complex. To catch the physical phenomena, the downcomer was subdivided into four parallel volumes that are connected to each other by cross-flow junctions. To assess the flow conditions in the upper plenum, especially the distribution of the reverse flow of about 1,500 kg/s entering into the outer ring between the RPV-wall and the inner perforated shell, an isolated CFX-model was developed for the upper plenum [Boett04]. **Figure 15** shows the predicted flow redistribution in the upper plenum, especially in the outer ring as obtained by a CFX simulation. It can be seen that a considerable part of the cold backflow of loop-3 is flowing sideward in the outer ring to the outlet orifice of loop-2 since loop-3 and loop-2 are located next to each other. A minor part of the coolant of loop-3 is also sideward redirected into the outlet orifice of loop-4. It should be noted that the rest of the cold flow of the loop-3 is entering the inner volume of the upper plenum through the perforations of the inner shell. Based on these results, the fluid volumes representing the outer ring of the upper plenum were connected by cross-flow junction in the RELAP5 model so that the reverse flow of loop-3 can go through the outlet orifices of loop-2 and loop-4. The mixing of a part of the flow of loop-3 in the upper plenum before leaving is also allowed. With these model extensions, the prediction of the stationary plant conditions was improved.

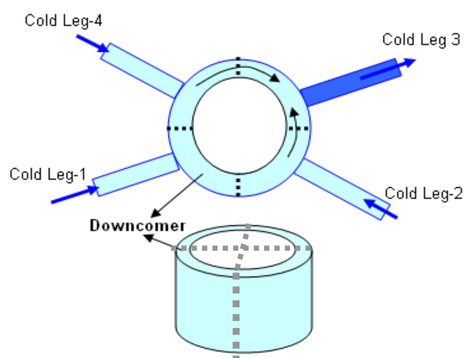


Figure 13. Flow conditions in Downcomer

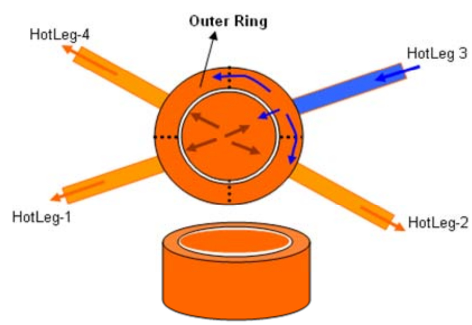


Figure 14. Flow conditions in Upper Plenum

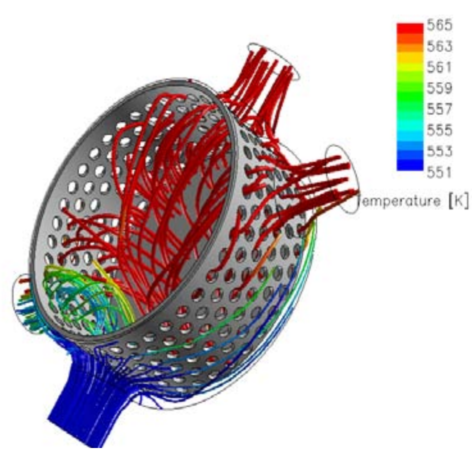


Figure 15. Complex Coolant Mixing in the Upper Plenum Predicted by CFX-5.

6.2. RELAP5/PARCS results for the 3D core model

In the frame of the benchmark Exercise-2, the investigations are focused on the testing of the 3D core model regarding the mapping scheme between the thermal-hydraulic and neutronic nodes as well as the consistence of the nodal cross-sections. In addition, different neutronic parameters like the stuck-rod, shut-down, and tripped rod worth are predicted. Several calculations were performed with RELAP5/PARCS for both the hot zero power (HZP) and hot full power (HFP) conditions of the Kozloduy core using the multidimensional model described above. The initial and boundary conditions as well as the control rod positions used to predict the different reactivity worth of the HZP state are taken from the specifications. The HZP conditions are characterized by a nominal power of 0.1 percent of the total power and fixed feedback thermal-hydraulics conditions (i.e., the moderator density in the core is 767.1 kg/m³ and the fuel temperature amounts 552.15 K).

To assess the developed 3D core models, the effective multiplication factor (k_{eff}) was predicted for different core states defined by different positions of the absorber rod groups. In **Table 4**, neutron physical parameters of the HZP state predicted by RELAP5/PARCS are compared with some data from the specification, where predictions and reference values are in a reasonably good agreement. The reactivity worth for different HZP-core states calculated by the coupled code is compared to the values given in the specifications. Both are close to each other.

Table 4.HZP Results Obtained with RELAP5/PARCS Compared to Reference Values.

Parameters	**RELAP5/PARCS**	**Reference Data**
K_{eff}	0.999669	
Radial power peaking factor	1.4034	
Axial power peaking factor	1.514	
Axial offset	-0.1726	
Ejected rod worth % dk/k	0.078	0.09
Control rod group 10 worth,	-0.69	- 0.61
Tripped rod worth, % dk/k	-7.24	- 7.02

The HFP core state is characterized by beginning of cycle (BOC) fuel conditions with an average exposure of 30.7 effective full power days (EFPD) and a thermal power of 824 MW. For this core state, both steady-state and transient calculations were performed with the coupled code system using the initial and transient boundary conditions given in the specifications. **Table 5** indicates the position of the absorber rod group attained for the HFP state in comparison to the HZP state.

Table 6 summarizes the main neutron physical parameters predicted for the stationary conditions of the HFP by RELAP5/PARCS.

Table 5. Position of the Control Rod Groups for the HFP States (100: out, 0: in)**.**

Core State	**G1-4**	**G5**	**G6-8**	**G9**	**G10**	**G10 EjRod**
Hot Zero Power (HZP)	100	100	100	64	0	0
Hot Full Power (HFP)	100	100	100	100	36	36

Table 6. HFP Results Obtained with RELAP5/PARCS for the Steady State Conditions.

Calculated Parameters	**RELAP5/PARCS**
K_{eff}	1.000425
Radial Power Peaking Factor	1.3471
Axial Power Peaking Factor	1.408
Axial Offset	-0.1734

The RELAP5/PARCS predicted a non-symmetrical axial power distribution for the steady state conditions expressed by an axial offset of 17.34 percent. **Figure 16** compares the predicted core averaged axial power peaking to the one given in the specification that was calculated by the benchmark team (PSU). It can be seen that both curves show the same trends with slight deviations mainly around 1 m and 2 m height.

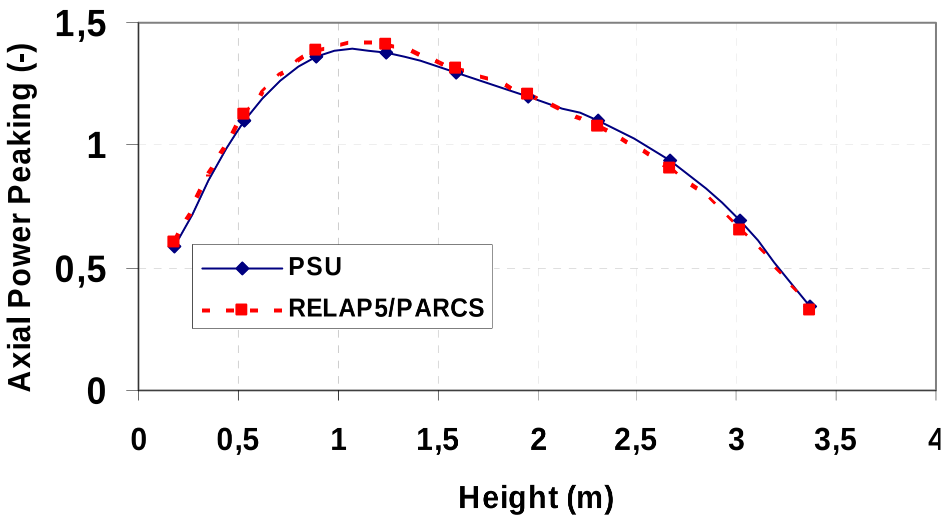


Figure 16. Comparison of the Predicted Axial Power Profile for the HFP Steady State.

6.3. Transient results for the integral plant model

The following subsections present the selected results obtained with both RELAP5/Point Kinetics and RELA5/PARCS and compare them to the plant data. Finally, the subsections present and discuss the detailed results obtained with PARCS.

6.3.1. Global plant response

The transient is initiated by the switch-on of the MCP-3. As a consequence, the mass flow rate of the loop-3 starts to re-invert (**Figure 17**), leading to a continuous increase of the primary coolant mass flow (**Figure 18**). After about 15 s, all loops reached similar mass flow rates, which remain almost unchanged during the transient. As a result, the core averaged coolant temperature decreases some degrees for the first 15 s (**Figure 19**). Like the coolant temperature, the fuel temperature undergoes the same trend during the first 15 s because the cooling conditions of the core have improved. Consequently, the total reactor power increases (**Figure 20**) rapidly until around 15 s because of the moderator and Doppler reactivity feedbacks.

Afterwards, this trend continues until the end of the transient but with a very moderate change rate. The power increase predicted with the point kinetics model (Exercose-1) is higher (5.5 percent of nominal power) than the one predicted with the 3D-neutron kinetics model (Exercise-3, 3.7 percent of nominal power) at the end of the transient. The reason is the use of a constant axial power, peaking Doppler and moderator reactivity coefficients, estimated for the core conditions at the beginning of the test for the whole transient. This leads to an overestimation of the reactivity inserted into the core by the point kinetics approach. On the contrary, PARCS solves a 3D problem with local estimation of the feedbacks by means of using cross-section sets depending on local thermal-hydraulic parameters that represent a more realistic description of the underlying asymmetrical core behaviour. The predictions of Exercise-1 may improve if a more detailed and sophisticated point kinetics model is used for this problem.

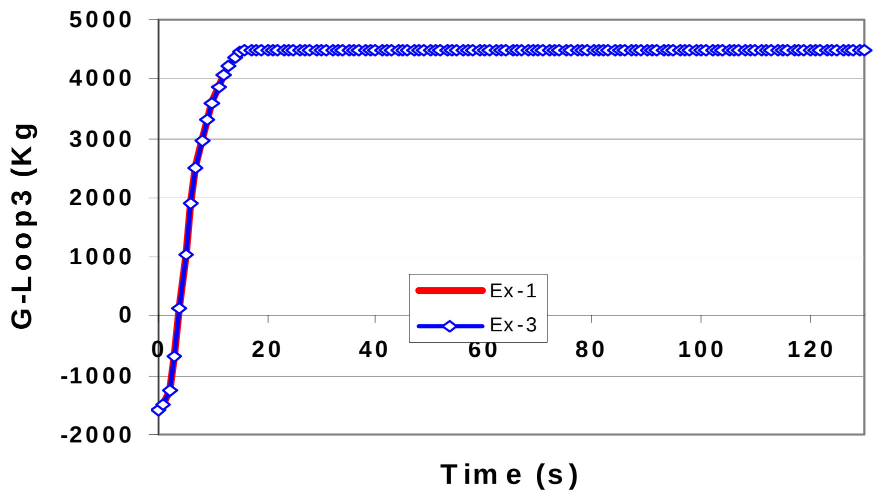


Figure 17. Predicted Reverse Flow of the Loop-3 during the Test.

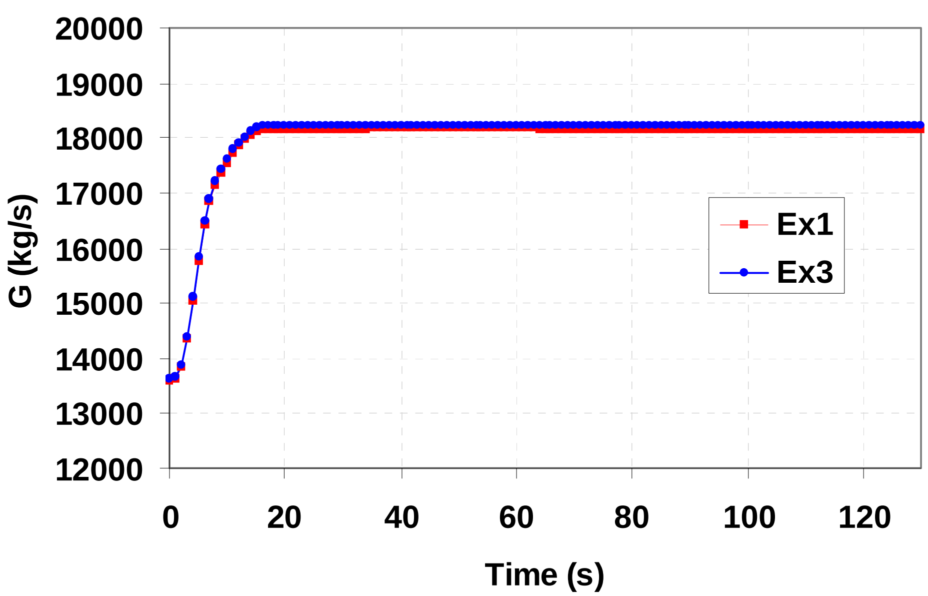


Figure 18. Predicted Change of the Total Primary Mass Flow Rate During the Test.

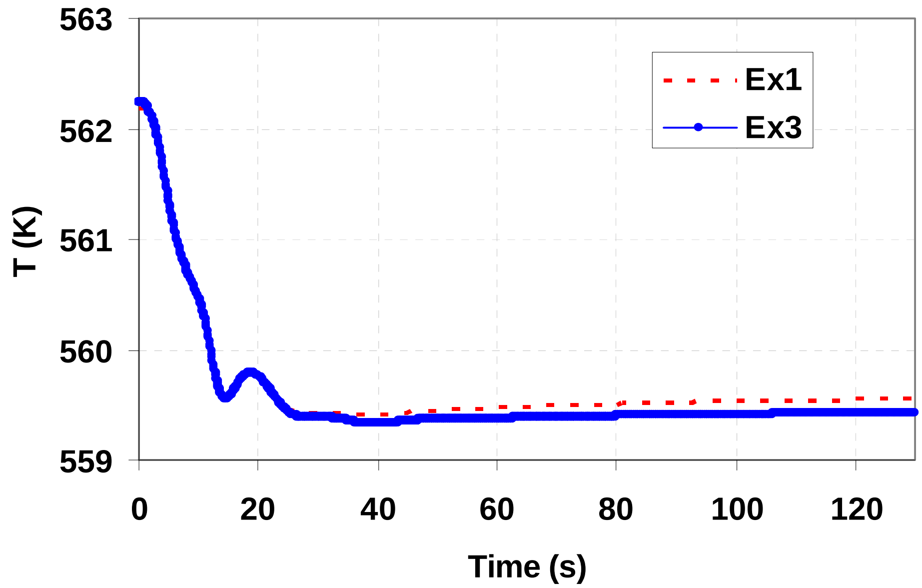


Figure 19. Predicted Core Averaged Coolant Temperature During the Test.

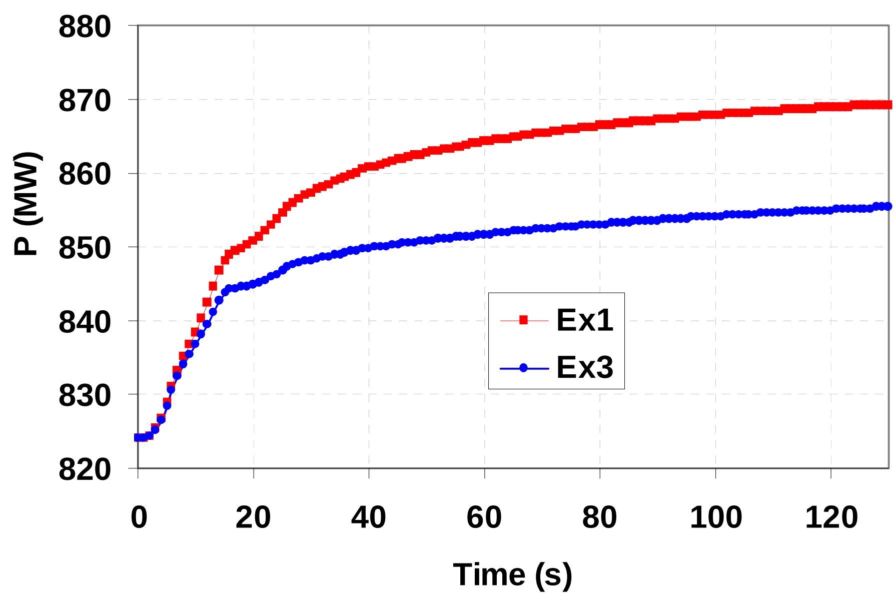


Figure 20. Predicted Power Increase for Exercise-1 and Exercise-3.

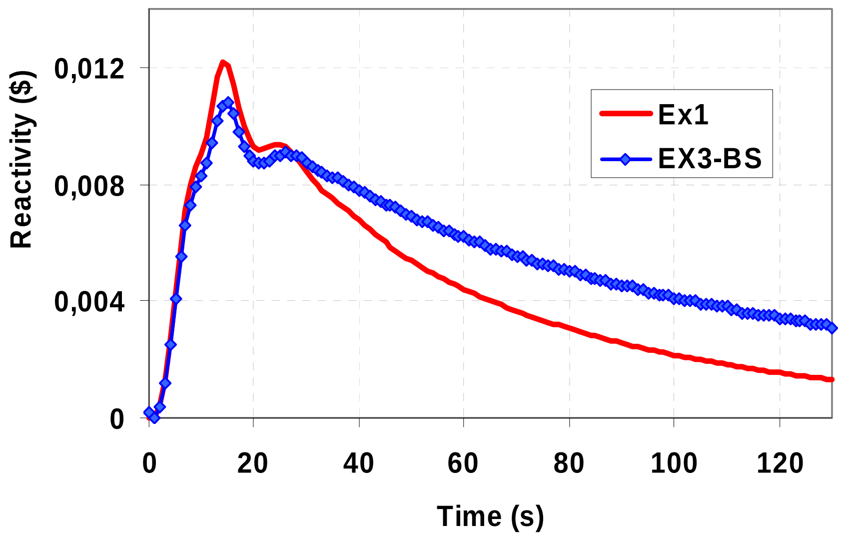


Figure 21. Predicted Reactivity Increase for Exercise-1 and Exercise-3.

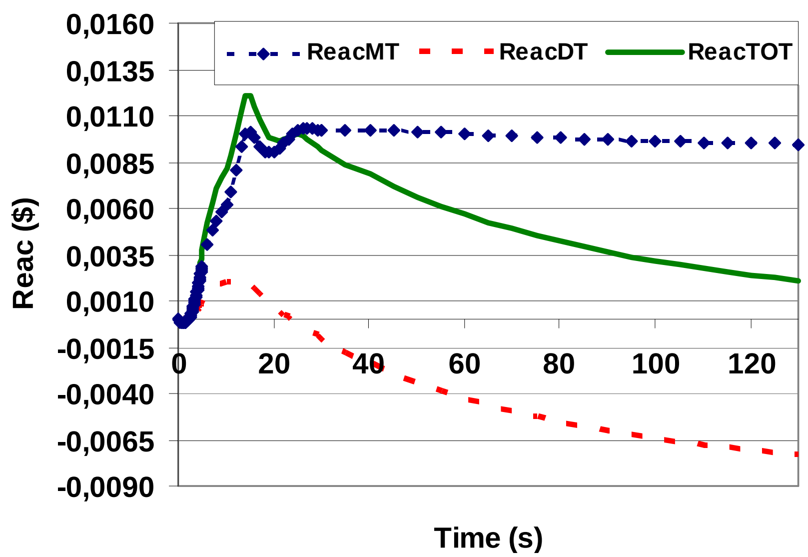


Figure 22. Predicted Reactivity with Main Contributors.

6.3.2. Code predictions versus experimental data

During the MCP switch-on test, several parameters of the plant were measured and its error bands were estimated (see **Table 2**). To show the quality of the RELAP5/PARCS predictions, selected parameters are chosen and compared to available data. **Figure 23** compares the pressure of the upper plenum predicted by the codes to the measured values.

The couple code is able to predict the initial time and decrease rate of the pressure during the first 10 s. **Figure 24** shows that, later on, the contraction of the primary system coolant is overpredicted by the simulations, which results in a faster decrease of the PZR level (around 55-80 s in the transient), and both predictions and measured PZR level are very close to each other. The primary to secondary heat transfer is almost constant in the calculation, while a slow but steady cooldown of the primary system is observed in the measured data.
Figure 25 and **Figure 26** show the changes of coolant temperature of the loop-1 for both cold and hot legs predicted by the codes in comparison with the measurements data.

It must be noted that the changes of the coolant temperature are moderate and smaller than the error band of the temperature measurement devices. The overall trend of the measured data can be reproduced by the calculation. The predictions tend to estimate a larger variation of the coolant temperature than one shown by the data. In **Figure 27**, the measured pressure drop over the MCP of loop 3 is compared with the values predicted by the code. The agreement is quite good for the whole transient. On the secondary side, the measured variation of the water level in the steam generator 1 is also compared with the predicted one in **Figure 28**. Although prediction and data start from around the same level, the code overestimates the heat transferred to the secondary side for the first 15 s. Then, the primary-to-secondary heat transfer is underestimated until the end of the transient.

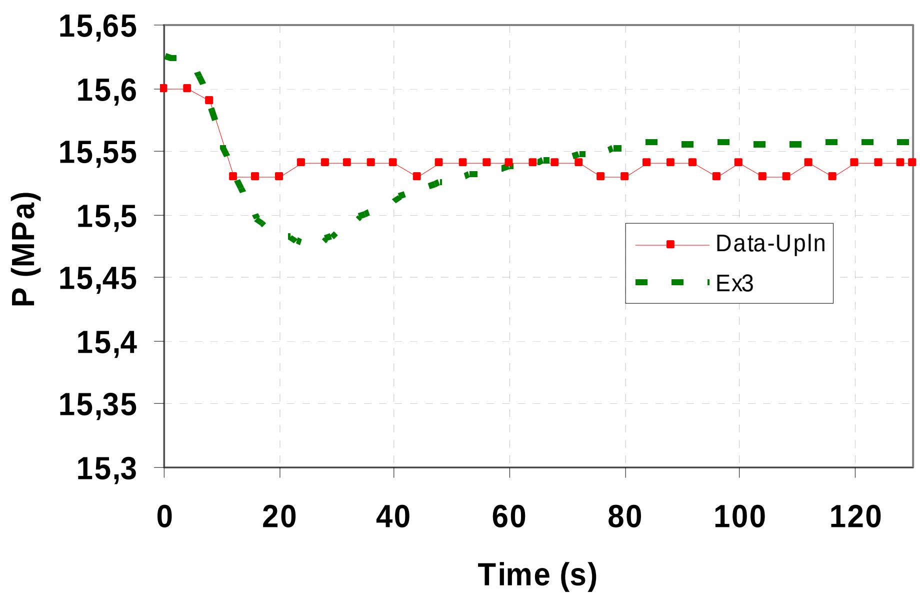


Figure 23. Comparison of Predicted and Measured Pressure in Upper Head.

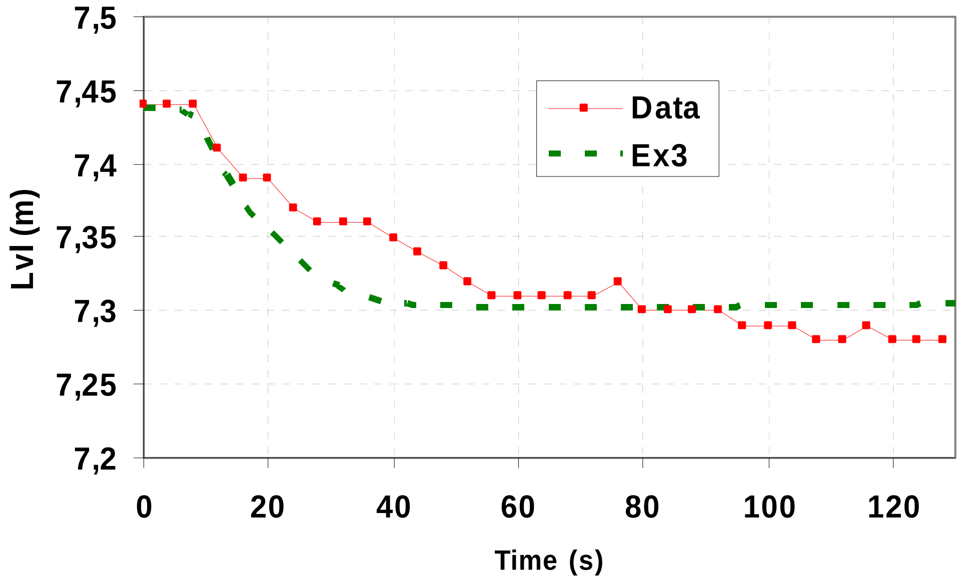


Figure 24. Comparison of the Predicted and Measured PZR Water Level.

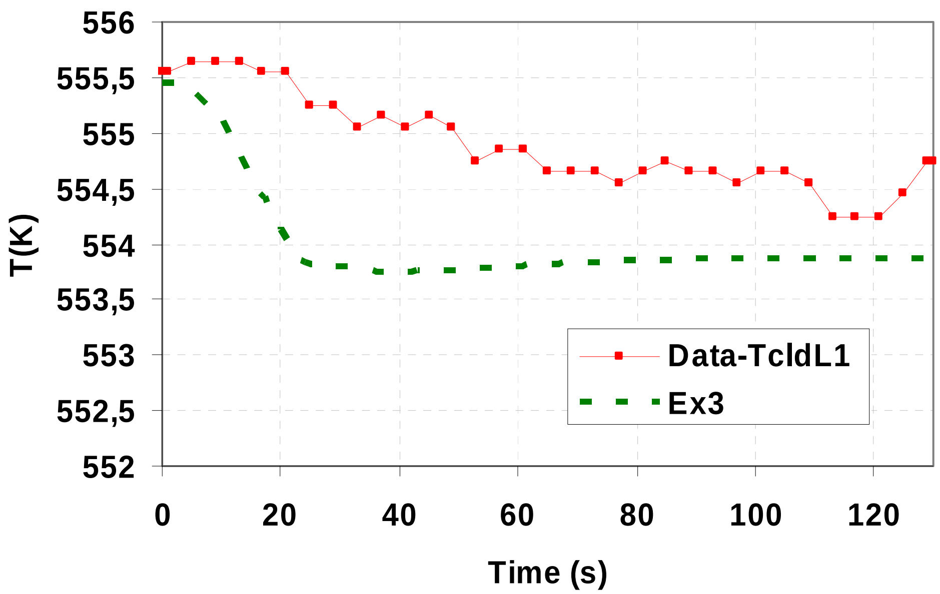


Figure 25. Comparison of Predicted Cold Leg-1 Coolant Temperature with Data

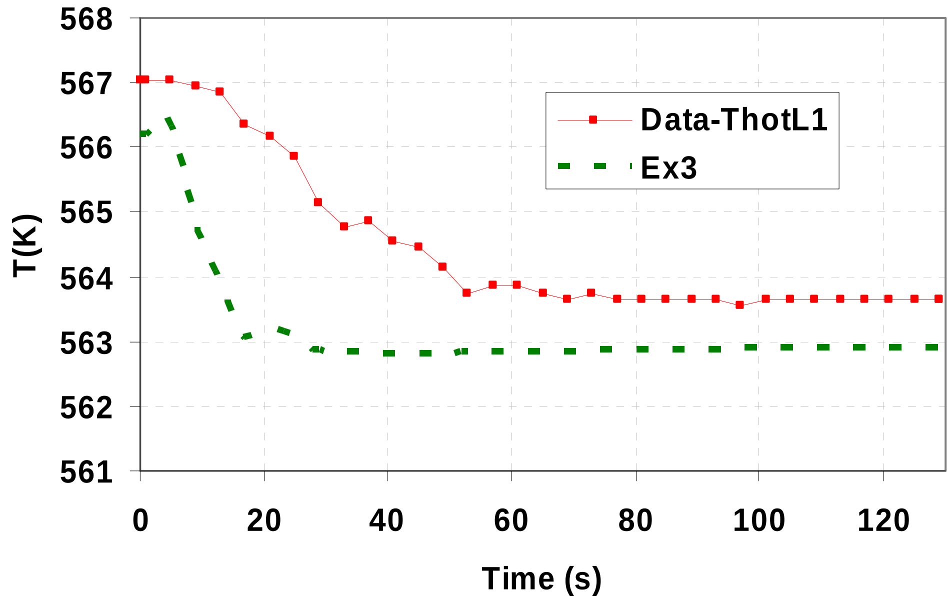


Figure 26. Comparison of Predicted Hot Leg-1 Coolant Temperature with Data.

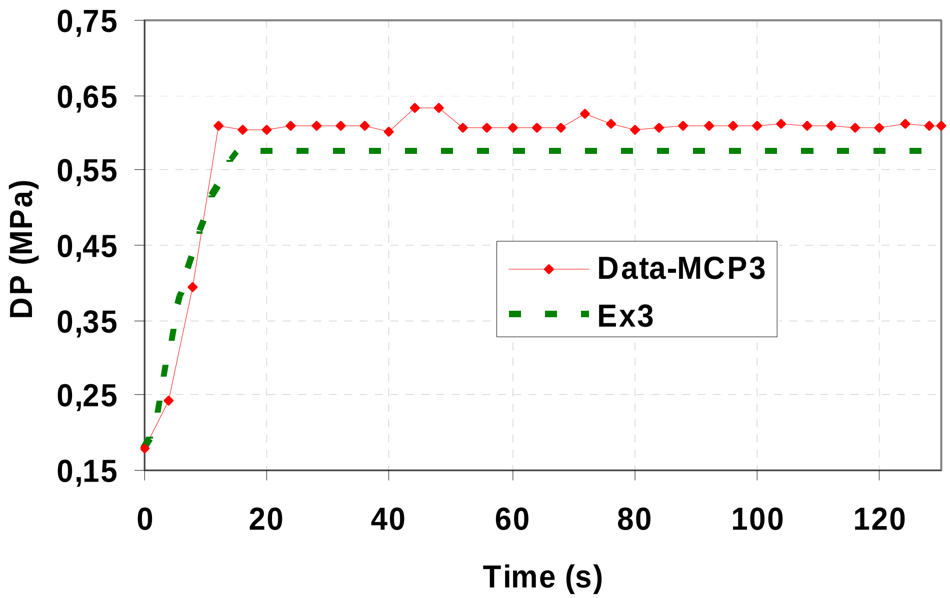


Figure 27. Comparison of Predicted Pressure Drop over MCP-3 with Data.

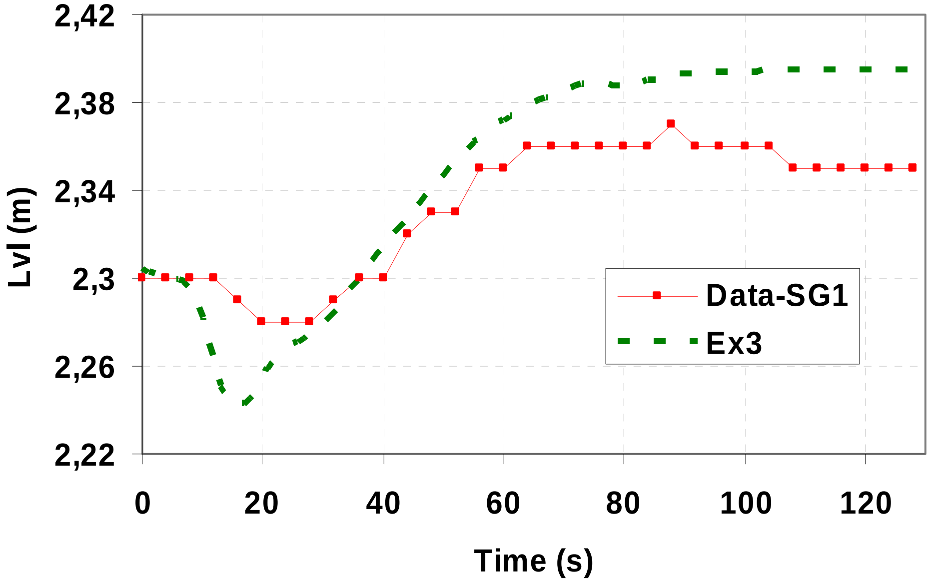


Figure 28. Comparison of Measured and Predicted Water Level of the SG-1.

Figure 29 to **Figure 32** compare the measured data of selected parameters to the predictions of both the point kinetics model (Exercise-1) and the 3D-kinetics models. The figures show that both simulations calculated very similar qualitative global trends. Because the predicted power by the point kinetics and 3D kinetics is different, other parameters like the collapsed liquid level in the pressurizer are quantitatively different for both simulations.

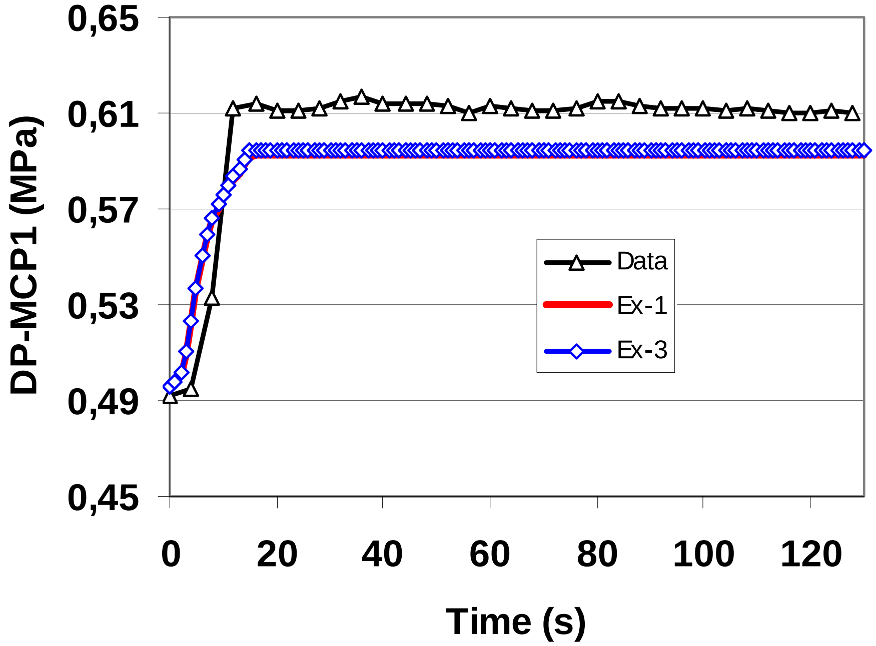


Figure 29. Comparison of Predicted Pressure Drop over MCP1 with Data.

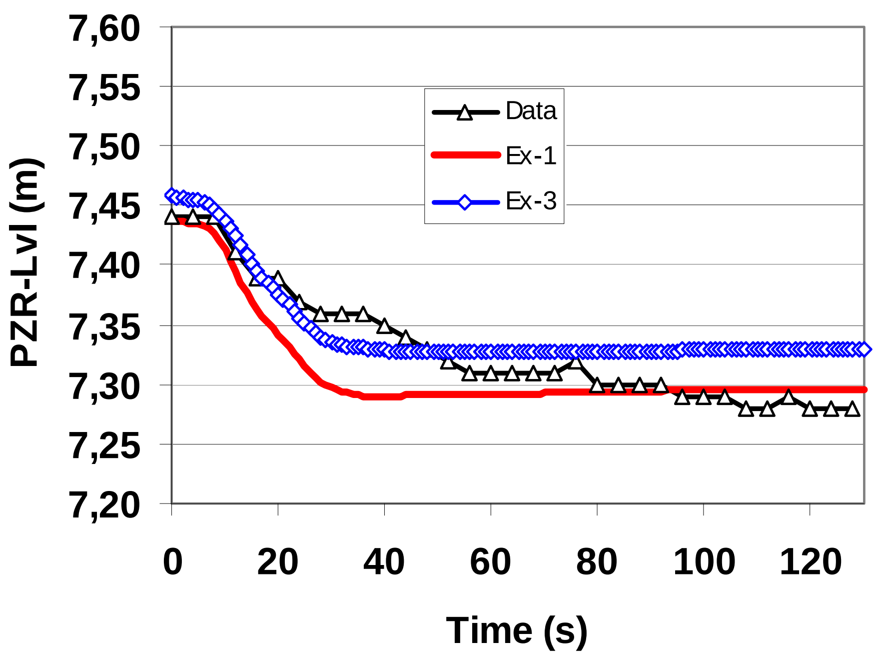


Figure 30. Comparison of Predicted Collapse Liquid Level with Data.

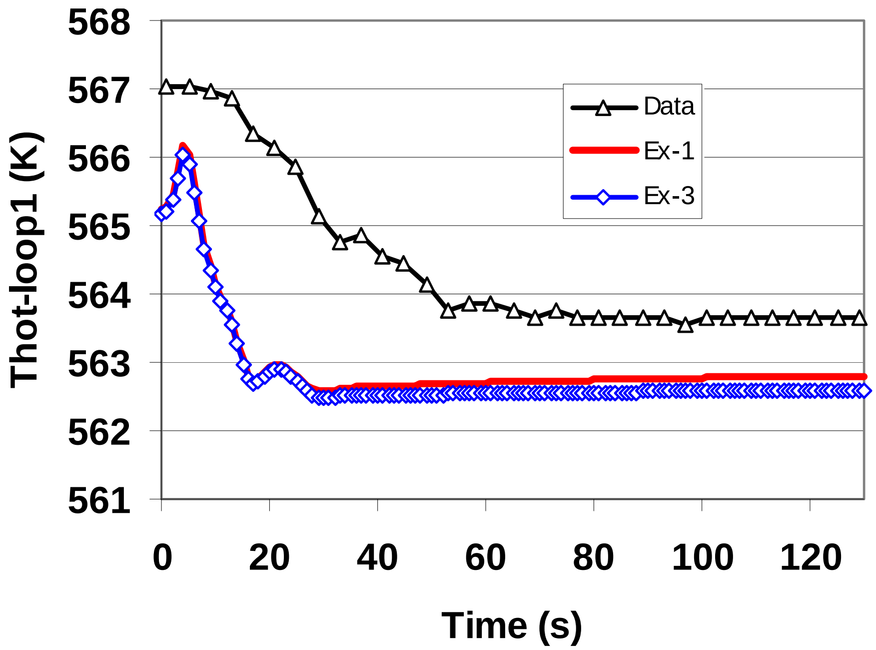


Figure 31. Comparison of Predicted Hot Leg-1Temperature with Data.

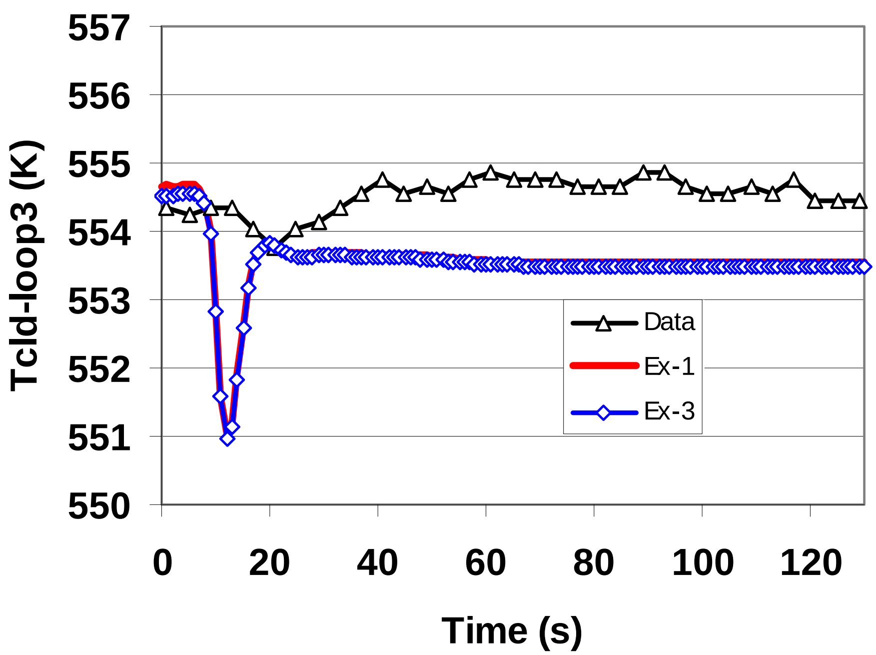


Figure 32. Comparison of Predicted Cold Leg-3 Temperature with Data.

6.3.3. Multidimensional core behaviour

The use of coupled codes with 3D neutron kinetics models allows a more detailed analysis of the core response compared to the point kinetics. **Figure 33** shows the core averaged axial power peaking predicted by PARCS for three time windows during the transient. The figure shows that for the basic scenario only a very moderate variation of the power peaking occurs. A similar trend was observed when the core averaged radial power profile was analyzed. **Figure** 34 **to Figure** 37 show that minor changes of the local radial power profile occur at different time windows. The maximal relative radial peak power changed slightly from 1.3450 at 0.0 s to 1.3473 at 15 s.

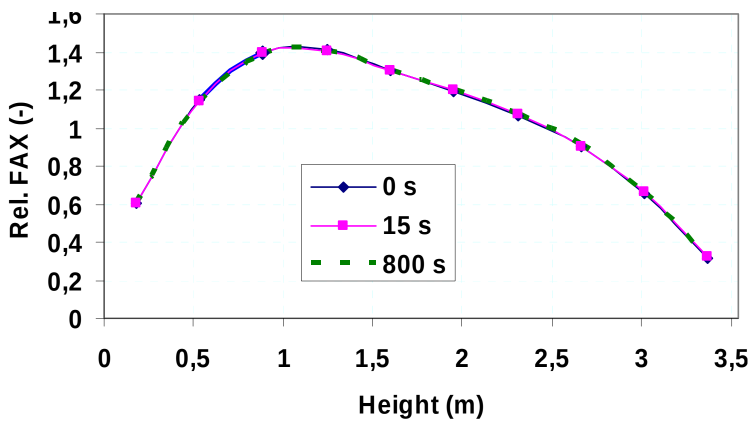


Figure 33. Predicted Core Averaged Axial Power Profile for Three Times

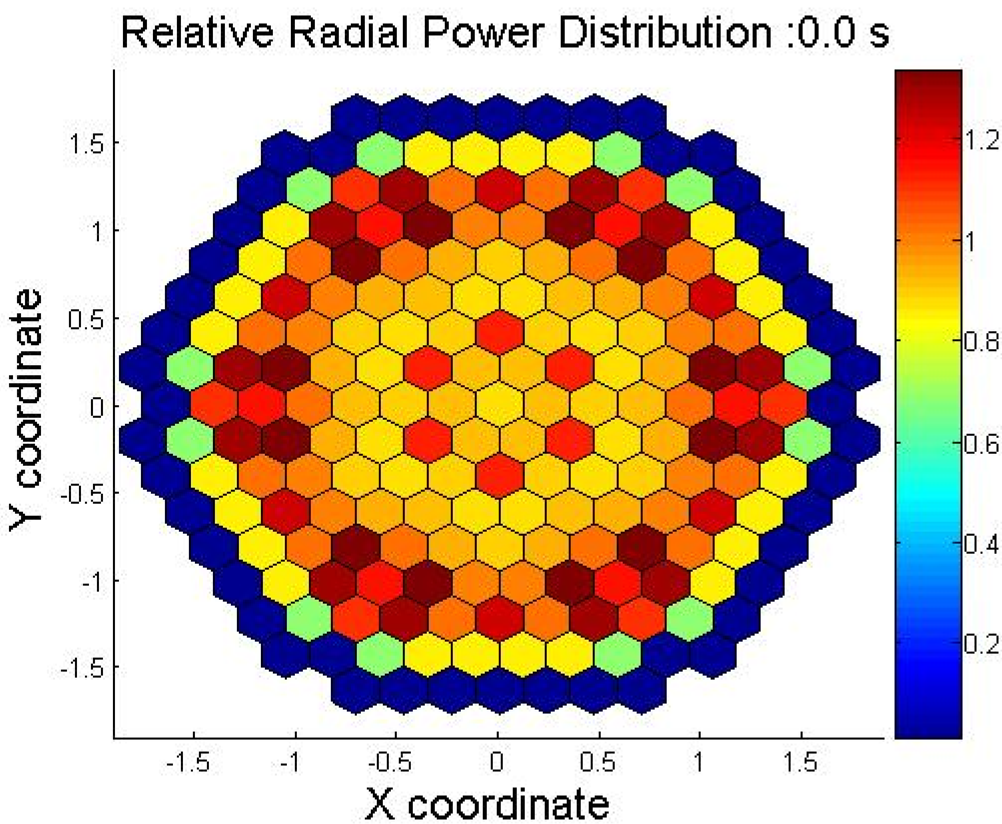


Figure 34. Predicted 2D Rel. Radial Power per FA at Steady State Conditions.

0.6925 0.8496 0.8604 0.8607 0.8503 0.6929

0.6929 1.1086 1.2882 1.0255 1.2275 1.0262 1.2892 1.1087 0.6926

0.8502 1.2891 1.1454 1.3435 1.0033 1.0037 1.3449 1.1455 1.2883 0.8497

0.8607 1.0261 1.3448 1.0237 0.9359 0.8939 0.9363 1.0238 1.3437 1.0256 0.8605

0.8604 1.2274 1.0037 0.9362 0.9205 0.8761 0.8762 0.9206 0.9360 1.0034 1.2276 0.8608

0.8495 1.0254 1.0033 0.8938 0.8761 0.9008 1.1334 0.9009 0.8764 0.8940 1.0038 1.0263 0.8503

0.6925 1.2881 1.3434 0.9359 0.8760 1.1333 0.9119 0.9120 1.1339 0.8765 0.9364 1.3450 1.2893 0.6930

1.1086 1.1453 1.0236 0.9204 0.9008 0.9118 0.8713 0.9122 0.9012 0.9209 1.0239 1.1455 1.1088

0.6929 1.2891 1.3447 0.9362 0.8761 1.1333 0.9118 0.9119 1.1335 0.8762 0.9361 1.3437 1.2883 0.6926

0.8502 1.0261 1.0036 0.8938 0.8760 0.9008 1.1334 0.9008 0.8762 0.8939 1.0034 1.0256 0.8497

0.8607 1.2274 1.0033 0.9359 0.9205 0.8761 0.8761 0.9205 0.9363 1.0038 1.2276 0.8605

0.8604 1.0254 1.3435 1.0237 0.9362 0.8938 0.9359 1.0237 1.3449 1.0262 0.8608

0.8495 1.2881 1.1453 1.3447 1.0037 1.0033 1.3435 1.1454 1.2892 0.8503

0.6925 1.1086 1.2891 1.0261 1.2275 1.0255 1.2882 1.1086 0.6929

0.6929 0.8502 0.8607 0.8604 0.8496 0.6925

Figure 35. Predicted Rel. Power in the Core at Steady State Conditions.

0.6969 0.8542 0.8644 0.8647 0.8549 0.6974

0.6973 1.1153 1.2946 1.0285 1.2308 1.0292 1.2957 1.1154 0.6970

0.8549 1.2956 1.1491 1.3458 1.0026 1.0030 1.3472 1.1492 1.2948 0.8543

0.8647 1.0292 1.3470 1.0224 0.9324 0.8893 0.9328 1.0225 1.3459 1.0286 0.8645

0.8644 1.2307 1.0029 0.9327 0.9145 0.8693 0.8694 0.9146 0.9326 1.0028 1.2310 0.8648

0.8542 1.0284 1.0026 0.8892 0.8693 0.8927 1.1233 0.8928 0.8696 0.8895 1.0032 1.0293 0.8550

0.6969 1.2946 1.3457 0.9324 0.8692 1.1232 0.9023 0.9024 1.1237 0.8697 0.9330 1.3473 1.2958 0.6974

1.1153 1.1490 1.0223 0.9144 0.8926 0.9022 0.8612 0.9026 0.8931 0.9149 1.0226 1.1493 1.1155

0.6973 1.2956 1.3470 0.9327 0.8692 1.1232 0.9023 0.9023 1.1234 0.8694 0.9326 1.3460 1.2948 0.6970

0.8548 1.0291 1.0029 0.8892 0.8692 0.8927 1.1232 0.8927 0.8694 0.8894 1.0027 1.0286 0.8543

0.8646 1.2307 1.0026 0.9324 0.9144 0.8693 0.8693 0.9145 0.9328 1.0031 1.2309 0.8645

0.8644 1.0284 1.3457 1.0223 0.9327 0.8893 0.9324 1.0224 1.3471 1.0292 0.8648

0.8542 1.2946 1.1490 1.3470 1.0030 1.0026 1.3458 1.1491 1.2957 0.8549

0.6969 1.1153 1.2956 1.0292 1.2308 1.0285 1.2947 1.1154 0.6973

0.6973 0.8549 0.8647 0.8644 0.8542 0.6969

Figure 36. Predicted Rel. Radial Power in the Core 15 s After the Transient Initiation.

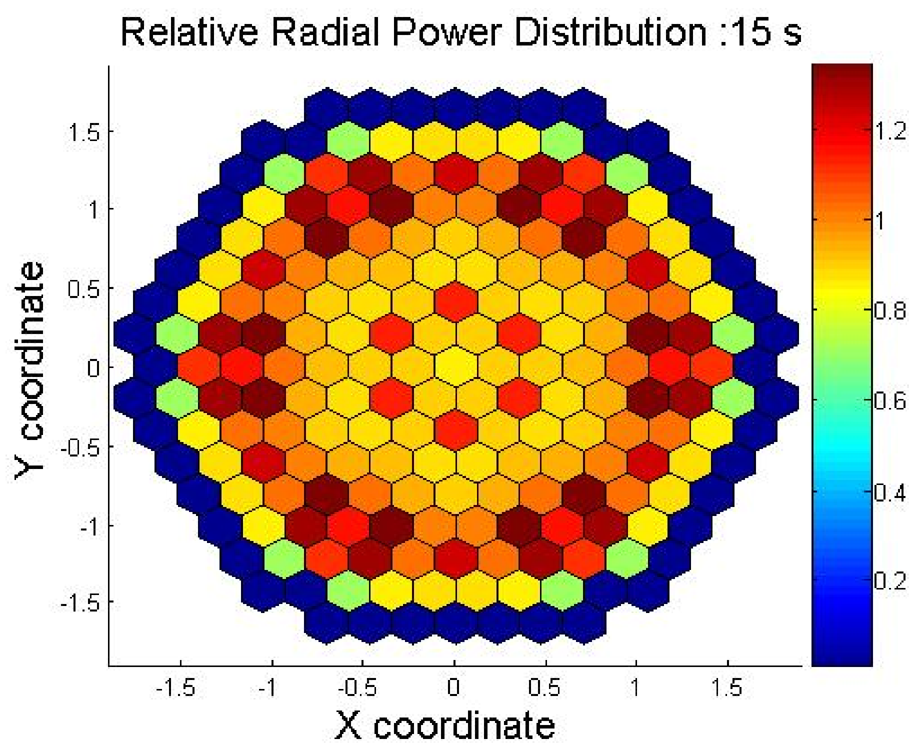


Figure 37. Predicted Rel. Radial Power in the Core at 15 s After the Transient Begins**.**

7. Results for the extreme scenario

Since the local power distribution during this transient was very moderate, an additional scenario (called extreme scenario) was defined in the Specifications to better check the capabilities of coupled codes. For this extreme scenario the ejection of a control rod located in the affected core sector corresponding to the loop-3 (FA#123) was assumed to happen at 13 s. In **Figure 38** a comparison of the power evolution during the transient predicted by RELAP5/PARCS is given. It can be seen that due to the assumptions the power increase of the extreme scenario is very pronounced compared to the one of the basic scenario mainly due to the ejection of the control rod group # 10 at 13 s which has a reactivity of 0.95 $. This power increase was stopped mainly by the Doppler and also by the moderator reactivity coefficient. Both the fuel temperature and the moderator density started to increase after the control rod #10 was ejected out of the core. At the time of the highest power core reactivity amounted 0,157 $ and the maximal relative power radial/axial power was around 1.605/1.372, see **Figure 39**. These values are much higher than for the basic scenario. A higher reactivity insertion due to assumption of control rod ejection at 13 s was predicted by RELAP5/PARCS than for the basic scenario. The detailed 3D radial power distribution for the extreme scenario is shown in **Figure** 40 to **Figure** 42. Here the fuel assemblies with the highest power can be identified very clearly.

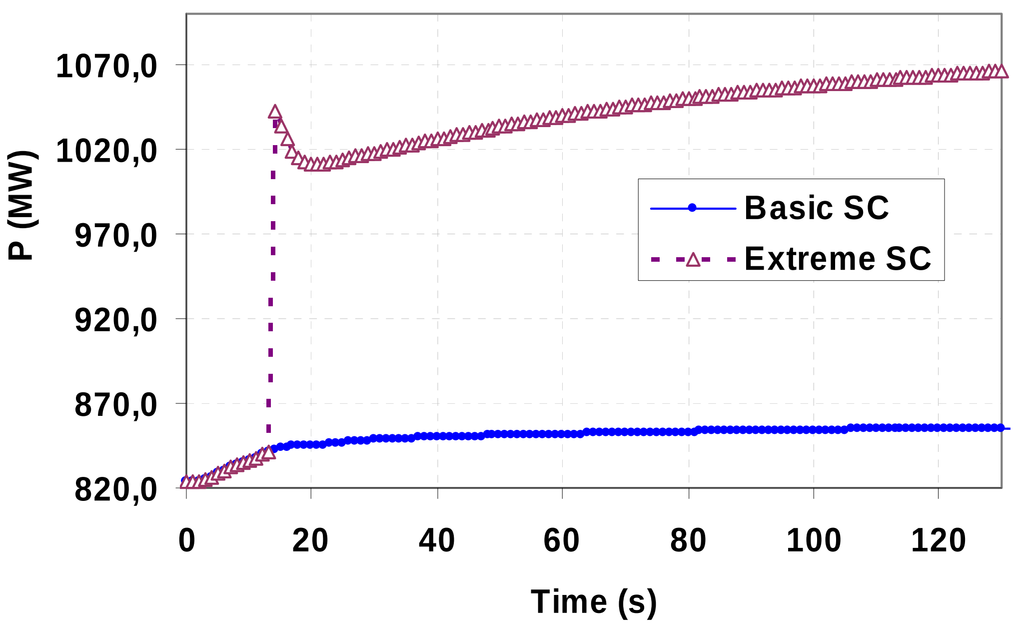


Figure 38. Comparison of Predicted Total Power Change During the Transient.

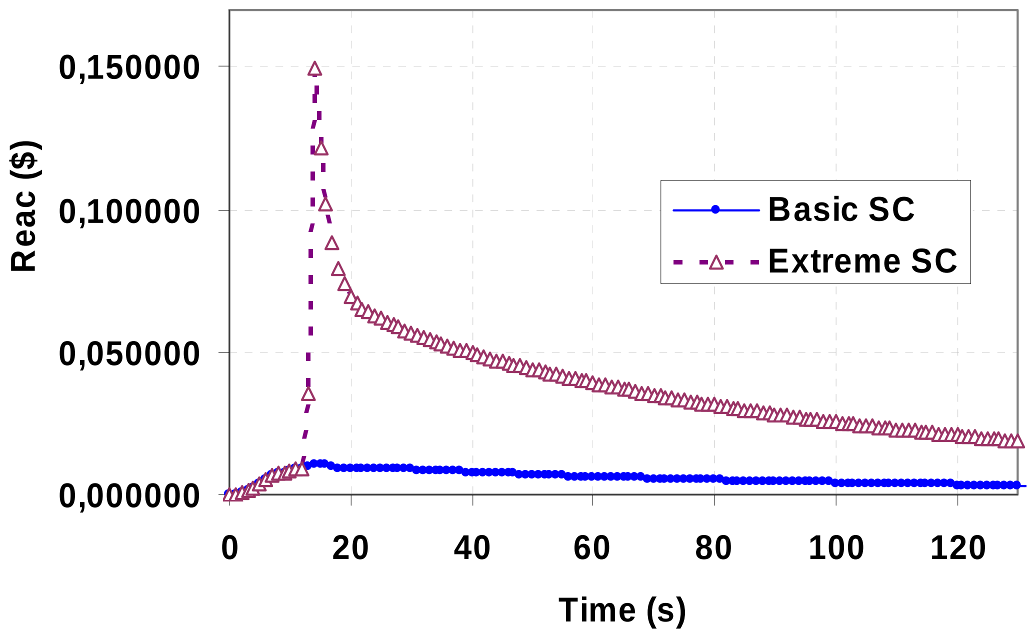


Figure 39. Comparison of the Predicted Core Reactivity for the Two Scenarios.

0.7090 0.8882 0.9292 0.9607 0.9731 0.8012

0.6874 1.1175 1.3255 1.0876 1.3560 1.1750 1.5016 1.2913 0.8009

0.8314 1.2773 1.1566 1.3969 1.0920 1.1570 1.6051 1.3594 1.5008 0.9727

0.8262 0.9970 1.3295 1.0380 0.9943 1.0275 1.1863 1.3036 1.6042 1.1747 0.9608

0.8119 1.1686 0.9689 0.9236 0.9364 0.9668 1.0937 1.4797 1.1864 1.1572 1.3567 0.9299

0.7907 0.9591 0.9466 0.8586 0.8679 0.9399 1.2759 1.1148 1.0946 1.0282 1.0931 1.0887 0.8891

0.6395 1.1916 1.2473 0.8769 0.8406 1.1264 0.9495 0.9996 1.2771 0.9679 0.9952 1.3988 1.3268 0.7094

1.0182 1.0523 0.9430 0.8553 0.8608 0.8960 0.8843 0.9498 0.9404 0.9368 1.0379 1.1563 1.1171

0.6329 1.1772 1.2276 0.8569 0.8127 1.0722 0.8790 0.8961 1.1268 0.8683 0.9235 1.3283 1.2763 0.6869

0.7730 0.9309 0.9098 0.8132 0.8060 0.8407 1.0723 0.8609 0.8410 0.8589 0.9688 0.9964 0.8308

0.7783 1.1076 0.9043 0.8457 0.8359 0.8061 0.8128 0.8554 0.8774 0.9472 1.1689 0.8261

0.7747 0.9215 1.2075 0.9209 0.8460 0.8132 0.8567 0.9430 1.2487 0.9599 0.8123

0.7630 1.1564 1.0279 1.2088 0.9047 0.9095 1.2265 1.0523 1.1927 0.7915

0.6214 0.9948 1.1573 0.9222 1.1077 0.9303 1.1763 1.0181 0.6399

0.6217 0.7636 0.7750 0.7780 0.7724 0.6325

Figure 40. Predicted Relative Radial Power Profile for the Extreme Scenario.

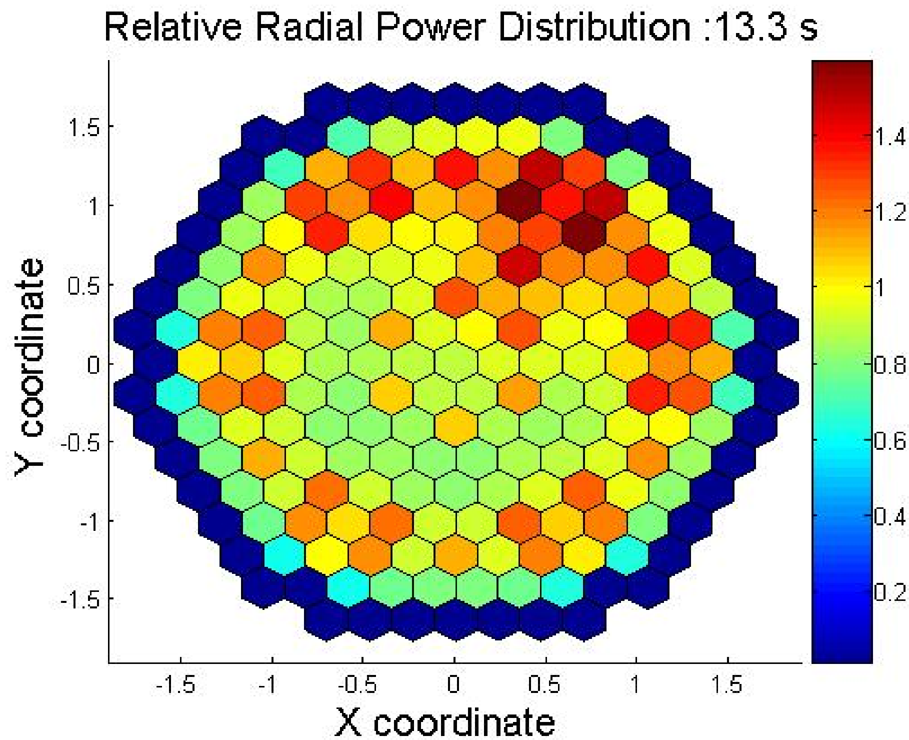


Figure 41. Predicted Relative Radial Power per Fuel Assembly.

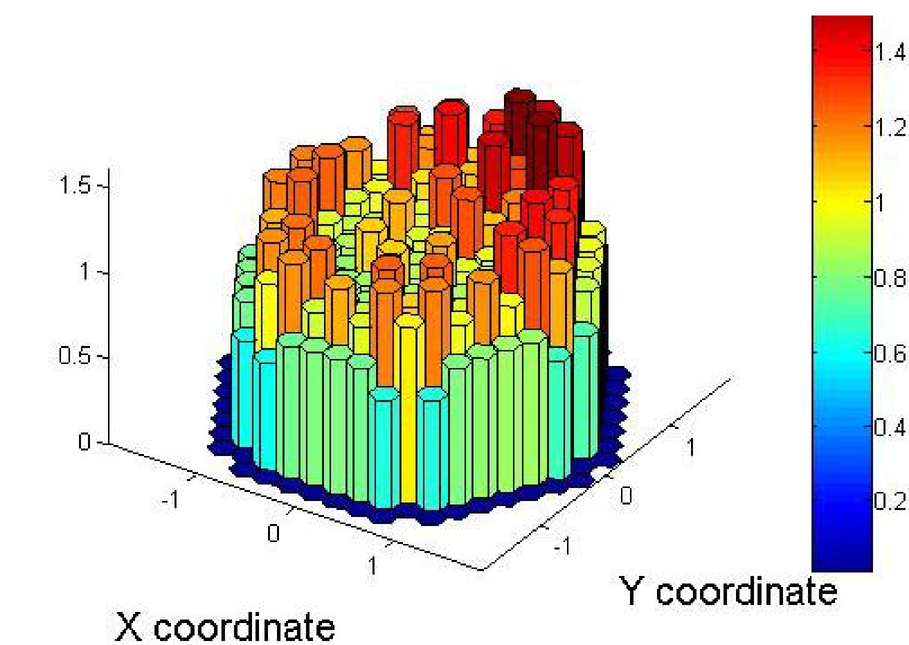


Figure 42. Predicted Relative Radial Power per Fuel Assembly at 13.3 s.

8. Conclusions

The Phase 1 of the V1000 Coolant Transient was analyzed with different computational tools using both point kinetics and three-dimensional (3D) kinetics models coupled to system codes. It was demonstrated that the developed integral plant model as well as the multidimensional core model are appropriate to describe the main plant and core response in case of an MCP-restart transient. The comparison of the predicted plant conditions with the plant data for the stationary reactor conditions showed a good agreement.

For the transient phase, most of the predicted parameters show trends that are qualitatively in good agreement with the experimental data measured during the test. The predicted reactivity worth for the different states of the cold zero power is close to the ones given in the specifications. Also, the axial power-peaking factor for the steady state hot full power estimated by RELAP5/PARCS agreed well with the reference solutions.

From the multidimensional detailed results for the core, it is apparent that the nonsymmetrical spatial power perturbation for the investigated test is rather moderate. But the analysis of the extreme scenario showed that the coupled system RELAP5/PARCS is able to predict core conditions with pronounced power distortions.

In general, the performed investigations clearly illustrate the capability of coupled codes systems with 3D neutron kinetics models as promising simulation tools to predict local hot spots within the core at a fuel assembly level. It can be stated that the time-dependent neutron diffusion solution for hexagonal geometries of PARCS works quite well in connection with the thermal hydraulic part (RELAP5) and that the RELAP5/PARCS works stable and fast enough under both Linux and Windows platforms.

Finally, the analysis of the MCP switch-on test showed the limitations of 1D thermal-hydraulic models of RELAP5 to describe the coolant mixing process. Hence, a more realistic description of such transients may only be possible using 3D thermal-hydraulics (3D coarse mesh, subchannel, or CFD codes) models coupled with the multidimensional neutron kinetics models. This kind of investigation is envisaged for the Phase 2 of this benchmark (V1000CT-2).

Although PARCS is able to predict the detailed axial and radial peaking factor within the core where each fuel assembly is considered as a computational node, the current coupled codes are not appropriate to predict the local fuel pin power within the fuel assembly with the maximal radial power. Note that exactly these local parameters like the pin power, maximal cladding, and fuel temperature are very important to assess the safety margins of any reactor design. Here additional model developments are necessary to extend the prediction capability of safety analysis codes aiming to implement transport and subchannel codes in the transient solution scheme.
Summarizing the following statements can be made based on this analysis:

- The investigations demonstrated the capability of coupled code systems to simulate complex transients with tight feedbacks between the system thermal-hydraulics and the core neutronics.

- Both RELAP5/Point Kinetics and RELAP5/PARCS are able to predict the overall plant response in case of the MCP-3 restart physical sound.

- RELAP5/Point Kinetics tends to overpredict the reactivity insertion (power increase) partly due to the use of fixed axial power distribution for the whole transient time and by using fixed reactivity feedbacks coefficients (DTC, MTC).

- Although the basic scenario is mild in terms of reactivity perturbation, coupled codes like RELAP5/PARCS permit the gain of detailed spatial information about hot spots in the core (here at fuel assembly level).

The following areas for further improvements were identified:

- Implementation of an automatic mapping scheme between the thermal-hydraulics and neutronics nodes is needed for RELAP5/PARCS similar to the one of TRACE/PARCS. This capability would minimize errors and considerably reduce the tedious work of generating the mapping manually.

- Development of 3D plotting and movies capabilities to represent the PARCS and also the RELAP5 results obtained for detailed core models.

- Correction of the PARCS output regarding important fuel-assembly-related parameters like axial power profile, reactivity contribution by moderator, and fuel temperature changes, specifically for hexagonal fuel assembly geometries.

- Implementation of an option in PARCS that allows the user to select the plot frequency. This is very important because the output and restart file of PARCS can become very large analysing real nuclear power plant problems (in the order of five or more gigabytes).

- Currently, PARCS auxiliary output files for hexagonal geometries are fixed to MSLB and, in addition, the data of many of them are useless. Modifications of these files in a more general sense (i.e., case-independent) is desirable:

  - MSLB_TMMAP.DAT (Assembly averaged coolant temperature map at each time step)

  - MSLB_TFMAP.DAT (Assembly averaged fuel temperature map at each time step)

  - MSLB_RPDMAP.DAT (Assembly averaged relative radial power map at each time step)

  - MSLB_KINFMAP.DAT (Assembly averaged Kinf map at each time step)

  - MSLB_DMMAP.DAT (Assembly averaged coolant density map at each time step)

- Improvement of the convergence between the thermal-hydraulic (RELAP) and the neutron kinetics solution (PARCS).

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