TMI-1 MSLB COUPLED 3-D NEUTRONICS/THERMALHYDRAULICS ANALYSIS: APPLICATION OF RELAP5-3D AND COMPARISON WITH DIFFERENT CODES.

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ABSTRACT

A comprehensive analysis of the double ended MSLB (Main Steam Line Break) accident assumed to occur in the B & W (Babcock & Wilcox) nuclear power plant of TMI-1 (Three Miles Island, unit No. 1) has been carried out at the Dipartimento di Ingegneria Meccanica, Nucleare e della Produzione (DIMNP) of the University of Pisa (Italy) in cooperation with the University of Zagreb (Croatia) and the Texas A&M University (US). The overall activity has been completed within the framework of the participation in the OECD-CSNI/NSC (OECD Committee on the Safety of Nuclear Installations - Nuclear Science Committee) "PWR MSLB Benchmark".

Different code versions have been adopted in the analysis. Results from the following codes (or code versions) are described in this paper:

- Relap5/mod3.2.2, beta version, coupled with the 3-D neutron kinetics Parcs code;
- Relap5/mod3.2.2, gamma version, coupled with the 3-D neutron kinetics Quabbox code;
- Relap5/3D code coupled with the 3-D neutron kinetics Nestle code.

Boundary and initial conditions of the system including those relevant to the fuel status, have been supplied by PSU (Pennsylvania State University) that had a cooperation with GPU (the utility, owner of TMI) and NRC (US Nuclear Regulatory Commission).

The capability of the control rods to recover the accident has been demonstrated in all the cases as well as the capability of all the codes to predict the time evolution of the assigned transient. However, one stuck control rod caused some "re-criticality" or "return-to-power" whose magnitude is largely affected by boundary and initial conditions.

The comparison among the results obtained by adopting the same thermalhydraulic nodalisation and the different 'coupled' code versions is discussed in the present document.

1. INTRODUCTION

The DIMNP of University of Pisa has been engaged in the assessment and application of system codes in the area of safety evaluation of Light Water Reactors, in the last three decades, ref. [1]. Proposals for nodalisation qualification criteria and for an uncertainty method to evaluate the envisaged error of any code prediction, ref. [2] and a tool to quantify the accuracy of a calculation, ref. [3], constitute examples of activities that have been completed. These activities aim at the full use of the thermalhydraulic system codes in the nuclear technology.

The research interest, considering the advancements in numerical methods and computer power has been moved in the last few years toward the investigation of three-dimensional phenomena. The equations relevant in thermalhydraulics, neutronics and structural mechanics plays an important role in this connection, together with the assessment of the coupled models, i.e. thermohydraulics-neutronics, thermohydraulics-structural mechanics.

Several international activities have been completed or are in progress aiming at characterizing the capabilities of calculations in predicting realistically transient scenarios assumed to occur in nuclear power plants, and, definitely to demonstrate the safety of those systems, ref. [4]. The CSNI and the NSC (Committee on the Safety of Nuclear Installations and the Nuclear Science Committee) of the OECD (Organization for Economic Cooperation and Development) are both active in the area of nuclear reactor safety also promoting a variety of ISP (International Standard Problems) and Benchmark. Recently, the two Committees proposed the PWR MSLB Benchmark. The system taken as reference in the study is the B & W TMI-1 Reactor, 2772MWt, about 900MWe power, equipped with two Once Through Steam Generators (OTSG). The general purpose of the activity is to gather a common understanding about the coupling between thermalhydraulics and neutronics, giving emphasis to the 3-D modeling. The considered benchmark is proposed and specified by PSU that had a cooperation with (the utility, owner of TMI) and NRC (US Nuclear Regulatory Commission). The activity was subdivided into 3 main phases:

1. analysis of the transient with a thermalhydraulic system code coupled to a point neutron kinetics model: the purpose of this step is to establish a common understanding of the problem among the participants with main reference to the modeling of the system, including interpretation of the supplied/available information;

2. analysis of the core performance only, when a 3-D neutron kinetic model is coupled with the thermalhydraulic code: the main purpose is to evaluate the hypotheses at the basis of the coupling, e.g. looking at their impact on the results;
3. analysis of the entire system performance with the coupled codes: the purpose is to evaluate the importance of the 3-D neutronics modeling in the considered cases, and to establish realistic and assessed methods (i.e. inclusive of codes, nodalisations and way of code use) suitable for safety, licensing and design analyses.

The Universities of Pisa and Zagreb have already carried out the analysis of the MSLB with the use of coupled codes as Relap5/Parcs and Relap5/Quabbox, respectively.
The same study has been jointly carried out by the University of Pisa and the Texas A&M University with the RELAP5-3D code, supplied by INEEL (Idaho National Engineering and Environment Laboratory) in the framework of bilateral agreements.

The results achieved in Phase 1 of the Benchmark are documented in ref. [5] to [9]. A comparison between the Relap5/Mod3.22 and Relap5-3D is described in ref. [10] and [11].

The purpose of this document is to outline the comparison among results obtained for the Phase 3 of the Benchmark when adopting the same thermalhydraulic nodalisation and different code versions.

2. REACTOR DESCRIPTION AND MAIN BOUNDARY CONDITIONS

A sketch of the reactor is given in Fig. 1. The two Cold Legs (CL) in each loop and the OTSG layout can be observed. The nominal working conditions for the plant are listed in Tab. 1, where for completeness data resulting from the code steady state calculations, are reported too. Superheating at the outlet (i.e. in the steam lines) and 'bypass recirculation' occurring through holes between riser and downcomer (DC) below the FW entrance nozzle, characterize the OTSG.

The list of imposed sequences of main events of the transient Benchmark can be drawn from Tab. 2. The plant status relates to hot full power at the end of the cycle. The main assumptions or relevant information for the transient calculation are as follows:

- assembly relative radial power distribution is given with quarter symmetry;
- axial power is assigned in twenty-four nodes not uniformly spaced;
- the break is assumed to be double ended in one of the two 24" Steam Lines (SL) departing from each OTSG, at the elevation of a 8" cross connection pipe assumed to be broken too, and upstream the MSIV (Main Steam Isolation Valves);
- the PRZ (Pressuriser) is connected to the Hot Leg (HL) of the loop where the SL is assumed to break (broken loop);
- the four reactor Main Coolant Pumps (MCP) are assumed to not trip in order to maximize the potential for reactivity excursion following the MSLB;
- laws for HPIS (High Pressure Injection System) flow versus pressure, of FW (FeedWater) flowrate and of scram reactivity worth versus time, are assigned; related to HPIS, two pumps inject cold water in two cold legs (one per each loop);
- no credit is given to the operation of the PRZ heaters and to the CVCS (Chemical and Volume Control System);
- boron concentration is assumed constant (i.e. notwithstanding the HPIS actuation) and its reactivity coefficient is included in the overall moderator coefficient;
- the containment is assumed as an infinite volume at 0.103 MPa.

Additional details about the plant, the initial conditions and the imposed sequence of main events can be found in refs. [12] and [6].

3. ADOPTED CODES AND NODALISATION

3.1 Codes

Three thermal-hydraulic codes and three neutronics codes have been adopted for calculating the OECD/NEA/NSC/CSNI MSLB Benchmark based on a TMI-1 transient.

The thermalhydraulic codes are the US NRC and the INEEL current versions of the Relap5. These are identified as the Relap5/mod3.2 beta and gamma and the Relap5-3D, respectively, refs. [13] and [14].

The 3-D neutron kinetics codes are the Quabbox, the Parcs and the Nestle, refs. [15] to [17]. Quabbox and Parcs are coupled to the Relap5/mod3.2 gamma and beta , respectively. Nestle is coupled with Relap5-3D. Coupling of the involved thermalhydraulics and neutronics software has been done at by (or under the control of) the thermalhydraulic code developers in the case of Parcs and Nestle. Coupling between Quabbox and Relap5/mod3.2 gamma has been completed at University of Zagreb.

In all cases 'officially' released code versions are adopted.

3.2 Nodalisations

<u>Thermalhydraulics</u>

The thermalhydraulic nodalisation of the entire system is given in Fig. 2 and the vessel nodalisation including the core region is shown in Fig. 3.

Eighteen core channels can be seen, representing the regions 1 to 18 of the core proposed by the host Organization of the Benchmark (see below). Two bypass channels can also be seen. Limitations in the maximum number of junctions belonging to a single BRANCH (Relap5 code component) imposed the need to split the lower (LP) and upper (UP) plena into four parts, at least in the zones of connection between LP and UP with the core itself. The DC has been split into four parts, corresponding to the four cold legs of the system. The coolant flowing in each part does not mix with the fluid flowing in the other parts. The UP has been subdivided into two parts because only two hot legs are part of the system. The relative azimuthal positions of the hot and cold legs have been preserved. The passive structures, as well as the core active structures, have been split consistently with the hydraulic nodes. The 'N 12' stuck control rod, ref. [6], is in the vessel quarter pertaining to the broken loop cold leg No. 1.

Main dimensions of the nodalisation can be derived from Tab. 3 where neutronic dimensions are given too.

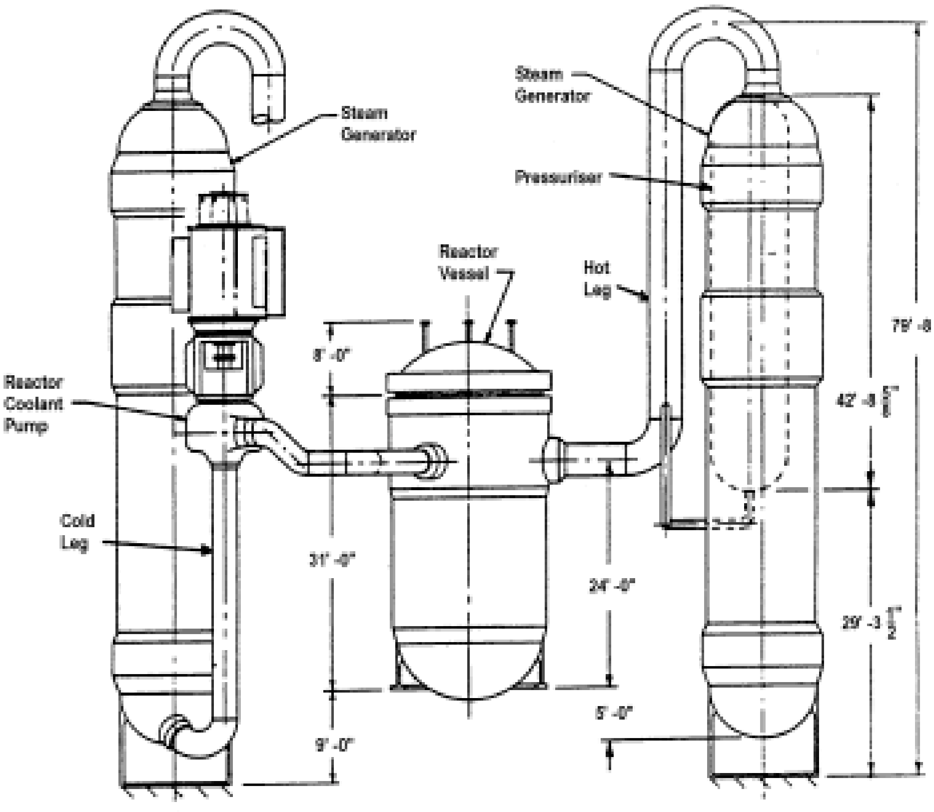


Fig.1 – Sketch of the TMI-1 Plant

QUANTITY	UNITS	*DESIGN VALUE*	*PHASE 1 RELAP5-3D**	*PHASE 3 QUABBOX***	*PHASE 3 PARCS**	*PHASE 3 RELAP5-3D***
Core power	MWt	2772	2772.	2771.	2772.	2772.
CL temperature & subcooling	K / K	563.7 (51.3)	564.7 (52.6)	562	562.7 (55.5)	563 (54.5)
HL temperature & subcooling	K / K	591.4 (23.6)	592.4 (22.3)	592.4	592.4 (23.1)	592.7 (22)
Lower plenum pressure	MPa	15.36	15.29	-	15.39	15.26
Outlet plenum pressure	MPa	15.17	15.16	15.27	15.27	15.13
RCS pressure	MPa	14.96	14.96	14.96	14.96	14.96
Total RCS flow rate	kg/s	17602	17559	-	16941	17383
Core flow rate	kg/s	16052	16003	-	15608	16347
Bypass flow rate (total)	kg/s	1549	1556*	-	1068	761
Pressuriser level	M	5.58	5.58	5.56	5.56	5.58
FeedWater flow per OTSG	kg/s	761.5	761.5	761.5-	761.5	761.5
OTSG outlet pressure	MPa	6.41	6.41 / 6.41	-	6.35 / 6.30	6.41 / 6.41
OTSG outlet temperature	K	572.6	574.8 / 574.8	571/ 570	571.6 / 570.6	568.1 / 568.1
OTSG superheat	K	19.67	21.81	-	19.24	15.13
OTSG DC level°	M	---	7.76 / 7.76	-	7.49 / 7.79	7.30 / 7.56
OTSG bypass flowrate	kg/s	---	107.2	-	108.3	138.4
FeedWater temperature & subcooling	K / K	510.9 (42.1)	510. (43.1)	510.8	510.8 (43.1)	510.8 (43.1)
Core pressure drops	KPa	129 irr. / 200 tot.	140.78	-	132.09	133.54
Initial SG mass inventory	Kg	26000	28094	25600 26800	25532 26810	25216 26342

* steady-state at 200s
** steady state at 100s
Tab. 1 Main Steady State Conditions obtained from the codes compared with design values

EVENT DESCRIPTION	**TIME (s)**
Breaks open	0.0
Reactor trip	6.9
MCP trip	not occurring
Turbine valve closure (start-end)	7.9-11.9
High pressure injection start	46.4
Transient end	100.0

Tab. 2 –Imposed sequence of main events for the OECD/CSNI/NSC TMI-1 MSLB Benchmark

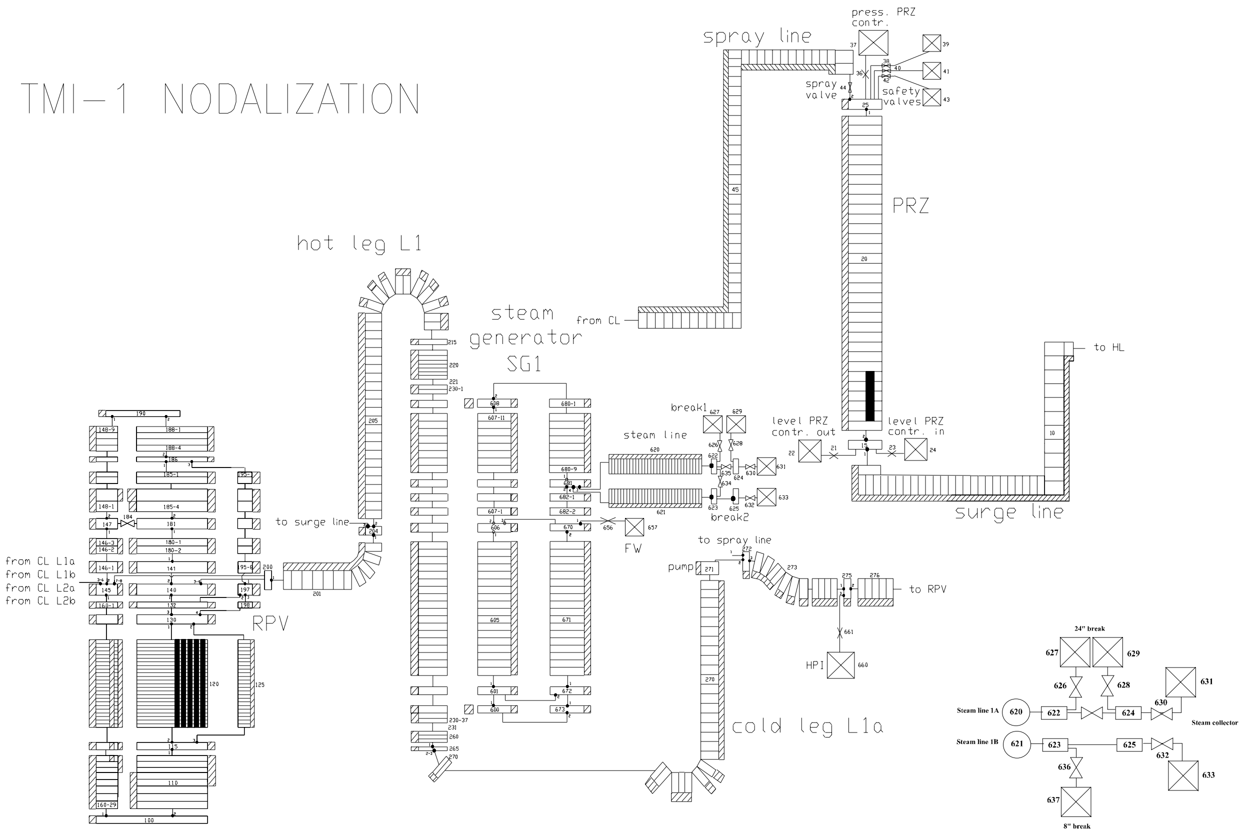


Fig. 2 – Sketch of the nodalisation for the TMI-1 MSLB Benchmark (entire system)

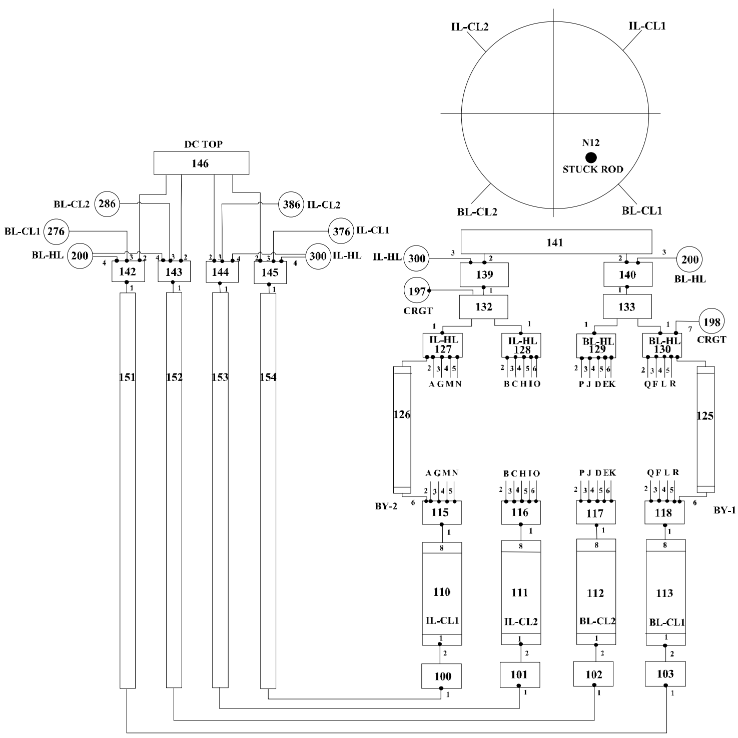


Fig. 3. Outline of Relap5 nodding scheme for the vessel downcomer, the upper and the lower plenum adopted in the present study.

No	QUANTITY	VALUE
1	Total number of hydraulic nodes	1499
2	Total number of mesh points for conduction heat transfer	15700
3	Total number of slabs	26
4	Total number of neutron kinetics nodes	4602
5	Number of parallel hydraulic stacks in the core region	18
6	Number of parallel core bypass regions	2
7	Number of elements in each core stack (hydraulics, heat conduction and neutronics)	26

Tab. 3 – Main dimensions of the coupled Relap5/mod3.2-Parcs input deck developed for the TMI-1 Nuclear Power Plant.

3-D neutronics

Each of the 177 fuel assemblies has been modeled by considering the radial maps given in Fig. 4 (ref. [12]). Fuel Assemblies (FA, 177) and Reflector Assemblies (RA, dashed zones, 64) are shown in Fig. 4 and 5.

Twenty-nine different FA types are selected assuming a 1/8 core symmetry. For each of the 29 groups of FA, 26 axial subdivisions are foreseen. Some of the axial subdivisions are the same as shown in Fig. 4. Therefore, the total number of cross sections sets needed for the FA would be 177*26=4602. Additional sets of cross section would be needed for the RA. The total number of adopted cross section sets results to be 438 (ref. [12]), including FA and RA. Cross section sets have been derived by CASMO (or a CASMO type) code in each FA and RA type. An effort was needed to modify the format of the cross section values to make these consistent with the requirements of the adopted 3-D neutronics codes.

In the same Fig. 4, the individual fuel assembly power (1/4 core power symmetry) is reported in the condition End Of Cycle Hot Full Power (EOC HFP) together with the core average axial relative power distribution utilised for the 0-D neutron kinetics input deck. Both of these are taken from ref. [6]. Power of the individual fuel elements is affected by the burnup, by the fuel type and by the position in the core. Maximum and minimum relative power result to be 0.439 and 1.285, Fig. 4.

The digits 1 to 18 (Fig. 4 and 5) relate to groups of 'homogeneous FA' from the thermalhydraulics point of view. All the RA constitute one group again from the thermalhydraulic point of view. The information at these last statements is not used for setting up the 3-D neutronics input deck.

4. PROCEDURES

4.1 Procedure

A detailed, step-by-step procedure has been adopted in order 'to keep under control' the achieved results, as documented in ref. [19]. The starting point for the step-by-step procedure is constituted by the Relap5 calculation adopted for the phase 1 of the TMI-1 MSLB Benchmark. This is the standard thermalhydraulic code calculation where 1-D thermohydraulics is coupled with the 0-D neutron kinetics.

Three main steps constitute the procedure for a 'consistent' use of the 3-D neutron kinetics:

- I 1-D thermohydraulics of the core and 3-D neutronics (to check consistency among results obtained from the application of 0-D and 3-D neutron kinetics).
- II 'Fictitious' 2-D core thermohydraulics and 3-D neutronics (perfect mixing in lower plenum).
- III 'Fictitious' 2-D core and vessel thermohydraulics and 3-D neutronics (no mixing in the vessel).

For Relap5-3D, the calculations carried out correspond to Phase I and Phase III of the MSLB Benchmark.

The Relap5 nodalisation adopted for the step I is the same as used for the Relap5 stand-alone code, suitable for the 0-D neutronics. The example sketch of the nodalisation adopted for the steps II and III can be derived from Fig. 2, although the input deck is needed for a complete understanding of the nodalisation features. It must be noted that no effort has been made to qualify the coupled nodalisation or to optimise it

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

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

Assembly relative radial power distribution (quarter symmetry)

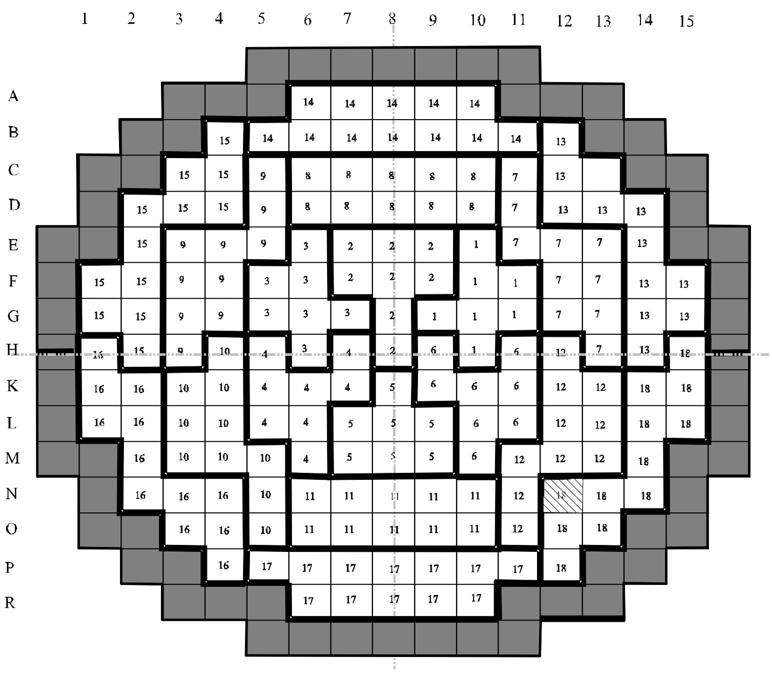
Core Centre

0.918	1.253	1.057	1.285	1.031	1.248	0.805	0.439
1.253	1.023	1.270	1.051	1.278	1.048	1.124	0.496
1.057	1.270	1.039	1.278	1.022	1.254	1.051	0.476
1.285	1.053	1.278	1.048	1.273	0.952	0.767	
1.031	1.282	1.022	1.271	1.035	1.093	0.580	
1.248	1.043	1.254	0.952	1.093	0.740		
0.805	1.121	1.051	0.767	0.580			
0.439	0.493	0.475					

Core average axial relative power distribution

Bottom

0.8008	0.98178	1.05563	1.06437	1.05347	1.03940	1.02745	1.01800	1.00775	1.00160	0.99907	0.99798
0.99785	0.99857	1.00041	1.00391	1.00980	1.01896	1.03230	1.05048	1.05834	1.03893	0.94526	0.79778



TH CHANNELS

A : 1
B : 2
C : 3
D : 4
E : 5
F : 6
G : 7
H : 8
I : 9
J :10
K :11
L :12
M :13
N :14
O :15
P :16
Q :17
R :18

Fig. 4 – Identification of fuel and reflector assemblies on a radial plane and characterisation of 29 different fuel assemblies in the assumed 1/8 core symmetry (ref. [12]).

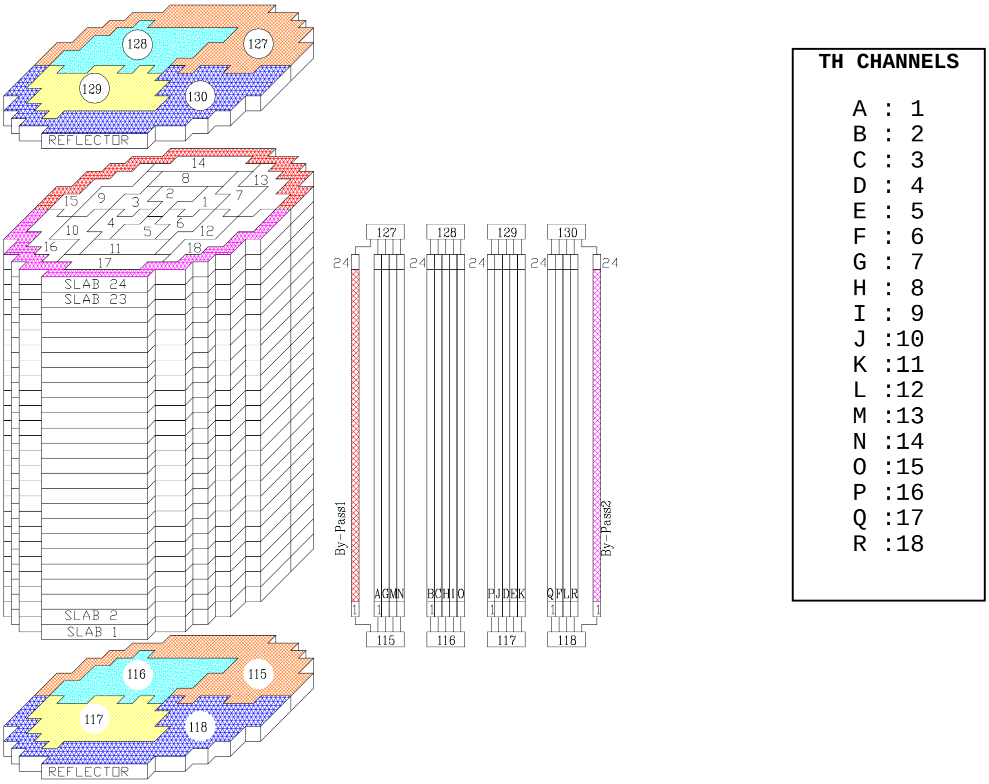


Fig. 5 – Zone figure and hydraulic structure of the core

4.2 List of Input data set

The list of calculations considered in the present study is given in Tab. 4.

Run No.	Thermal Hydraulic code	Neutron kinetic code	Neutron Kinetics	Core Model
1	Relap5/Mod3.2 beta	-	0D	1 channel
2	Relap5-3D	-	0D	1 channel
3	Relap5/mod3.2 beta	PARCS	3D	18 channels
4	Relap5-3D	NESTLE	3D	18 channels
5	Relap5/mod3.2 gamma	QUABBOX	3D	18 channels

Tab. 4 - List of calculation considered in the present study

The following should be noted:

- complex nodalisation must be developed to run 3-D neutron kinetics code coupled to thermalhydraulic code. These may consist of up to seventy thousands ‘lines’, 1.6MB of text file in the case of Relap5-3D;
- huge resources are needed for the development of each nodalisation and for a suitable evaluation of results (long lasting activity), Consideration of Quality Assurance in the process may reveal non-feasible;
- feedback (automatic, unavoidable), as expected, occurs between neutronics and thermalhydraulics. The problem derives from achieving steady-state conditions before transient initiation. For instance, differences in

predicting entrainment in secondary side (affected by the interfacial drag, a model 'easily' changed in different code versions) causes differences in steam generator removed power, therefore differences in primary system temperatures, definitely differences in core fission power result. These differences can be set to zero by proper procedures for achieving steady state. Initial mass in steam generators is also affected by the interfacial drag: different transient behavior of the entire system can be predicted;

- averaging processes and lumping hydraulic and thermal nodes to neutronic nodes depend upon user choices and partly upon the features of the interface between the thermalhydraulic and the 3-D neutronic code.

As a consequence of the above, the processing of the same input information within each of the code runs, may give (slightly) different steady state results. Limited attempts were made to achieve a 'converged' set of boundary and initial conditions.

5. RESULTS

The documentation related to runs No. 1 and 2 has been entirely reported in Ref. [11]. Only the relevant time trends that characterize the transient have been included into the present paper (Fig. 6 and 7). The results obtained with Relap5/mod3.2 and with Relap5-3D have been compared in order to evaluate the influence of the code on the transient. Exactly the same input deck file has been adopted with both codes. The results show that at the steady state most of the values are similar or exactly the same, except the initial SG secondary side mass inventory and the OTSG outlet temperature (see also Tab. 1). The consequences on the calculation of the transient are mostly connected with the return to power phenomenon (see fig. 6, where the total power versus time in represented). Although the first power peak has similar features for the two code runs, the second power peak predicted by the Relap5-3D is higher and shows a well defined shape compared to the Relap5/Mod3.2 calculation that is characterized by two peaks. Moreover, the core power after the second peak converges to a similar value at a similar time in both calculations.

Concerning runs No. 3, 4 and 5, the documentation of coupled thermalhydraulic 3-D neutron kinetics calculation includes for both steady state and transient periods:

- relevant time trends (quantity reported as a function of time)
- spatial distributions taken at the end of the steady state
- 3-D graphical representations
- tables of steady conditions

Short video-clips can also be used and facilitate the overall interpretation of the overall system behavior (e.g. evolution of core power).

All of the above has been used to document the calculation of runs No. 3 and 4, (10b in ref. [17]).

Steady state conditions

The demonstration of a thermohydraulic and neutronic stable steady state before the initiation of the transient calculation is a necessary condition to achieve reliable results. Differences in thermalhydraulic code versions models affected the steady state results. The reasons already mentioned in the previous section prevented the achievement of a unique set of initial conditions, although the same thermalhydraulic input deck was adopted.

The parameters calculated by the codes concerning the primary circuit are very close to the design values at steady state (see Tab. 1). The main differences are related to the OTSG superheat and residual mass.

Axial power distributions are reported in Figs. 8 and 9 for Relap5-3D and Relap5/Parcs codes, showing a good similarity in term of values and shape. The axial distribution proposed by the Organizers of the MSLB is very close to the distribution predicted by both Relap5-Parcs and Relap5-3D coupled codes.

Phenomenology predicted by the three coupled codes runs (overall system behaviour)

The overall system performance can be derived from the set of Figures 10 to 15 related to code runs 3, 4 and 5 of Tab. 4.

The most relevant outcome from the study is constituted by the prediction of the core power and of the primary system pressure, Figs. 12 and 10, respectively (see also Figs. 11, 13). The core power excursion is controlled by the assumed scram logic; the power does not overpass 120% of the initial value. The primary system pressure decreases to a value that allows the actuation of the HPIS.

The fast depressurization of the broken steam generator (loop 1) can be observed in Fig. 11. The fluid mass in the broken SG does not decrease in the early phase of the transient (Fig. 14): this comes from the assumption of increased FW mass flow rate (assigned in the calculation) that compensates break flows. Core mass flow rate increases during the transient, owing to improved Main Coolant Pumps (MCP) efficiency caused by liquid cooling. In the intact loop CL, fluid temperature becomes higher than in the HL of the same loop, this is connected with the heat transfer reversal in the steam generator.

The present results are basically within the dispersion bands obtained from the envelope of results predicted by other participants to the Benchmark, ref. [20]. In this connection it should be noted that we did not consider the recommendation, coming from the Benchmark proponents, to neglect passive structure masses in the secondary side of SG in order to overestimate the cooling potential of the MSLB event. This actually causes a delay in the power excursion that can be estimated in a few seconds, related to the case when passive structures are not modeled.

Three-dimensional core behavior

3-D graphical representations of core related parameters are limited to run No. 3 and 4 (Figs. 16 to 18).

One-by-one core channel nodding has been adopted for the neutronics calculation, but eighteen core channels have been distinguished in the thermohydraulic calculation, as already mentioned. The thermalhydraulic calculation is also characterized by the radial nodding (i.e. layout of the eighteen core channels) depicted in Fig. 5. Therefore the reproduced 3-D imagines of the core related quantities are affected by both the aforementioned nodding.

All the radial pictures are reported in the same position as in Fig. 3 where the intact and the broken loops are situated at the North and the South, respectively and the stuck rod is located in the South-East quarter.

3-D distributions can be reported for each time step from each calculation. Clearly this is well beyond the scope of the present document. The time when the second core power peak occurs has been selected for the 'transient representation' of 3-D effects. Larger spatial non uniformities for core related quantities are expected than what observed for the steady state owing to the discontinuity introduced by the presence of the stuck control rod in the South-East zone of the core, Figs. 16, and 17.

Actually, peak regions in the South-East zone of the core can be seen in the distributions of total power in both code runs. More detailed results from ref. [19] and [21] show non uniformities in radial distributions of core power at middle core elevation, of coolant temperature and of fuel temperature at the center of the rod. Complex 3-D effects have been observed in the quantity "mass flowrate / hydraulic channel power". However, no attempt has been made to interpret this result.

6. CONCLUSION

A comprehensive analysis has been carried out during a four years period (1997- 2000) in the framework of the participation of the Universities of Pisa and Zagreb to the OECD/CSNI NSC & PWG2 TMI-1 MSLB Benchmark. Additional analysis carried out with Relap5-3D have been performed with Texas A&M University in 2001.

The Benchmark has been subdivided into three phases. The end product of the activity is constituted by the results related to the 3-D performance of the TMI-1 core following the Main Steam Line Break event, i.e. phase 3 of the Benchmark. The capability to use a 3-D coupled thermohydraulic/neutronic code constitutes an important result of the performed activity. Conclusions are drawn hereafter in relation to the comparison between predictions of two version of Relap5 code, INEEL and US NRC versions, and of three coupled thermalhydraulics 3-D neutronics codes, i.e. Relap5/mod3.2 beta – Parcs, Relap5/mod3.2 gamma – Quabbox and Relap5-3D – Nestle.

Results predicted by three coupled codes, , adopting the same thermalhydraulic nodalisation are qualitatively similar. However, differences in quantitative terms have been found. This is mainly the case of the 2^{nd} peak in core power that characterizes the 'return-to-power' phenomenon. General reasons for differences have been identified as follows:

- user choices when defining the coupling between neutronic and thermalhydraulic model in the case of differences between Relap5/mod3.2 beta – Parcs and Relap5/mod3.2 gamma – Quabbox predictions,
- thermalhydraulic models affecting initial conditions in the steam generators in the case of differences between Relap5/mod3.2 beta – Parcs and Relap5-3D – Nestle,

The huge resources needed for completing the analyses made difficult a deeper investigation for the causes of the detected.

7. REFERENCES

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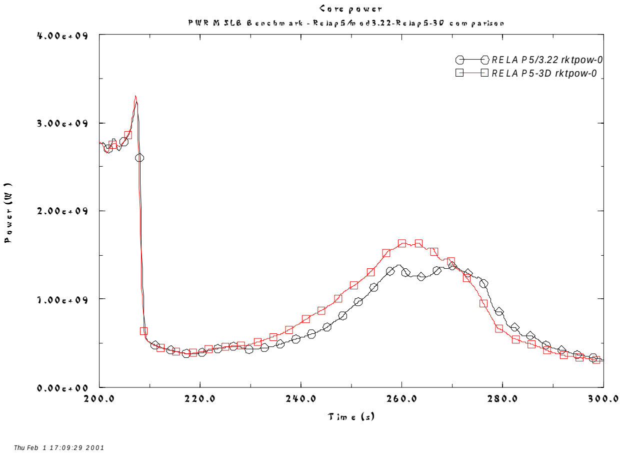

Fig. 6 – Core Power Comparison for Runs No.1 and 2 obtained with Relap5-3D and Relap5/mod3.2

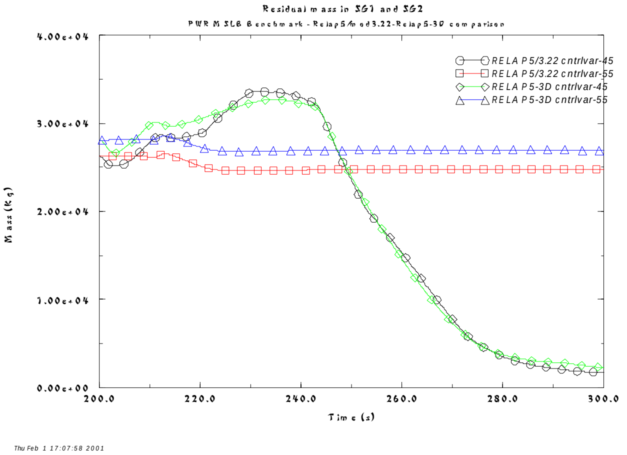

Fig. 7 – Residual mass in SG1 and SG2 for Runs No. 1 and 2 obtained with Relap5-3D and Relap5/mod3.2

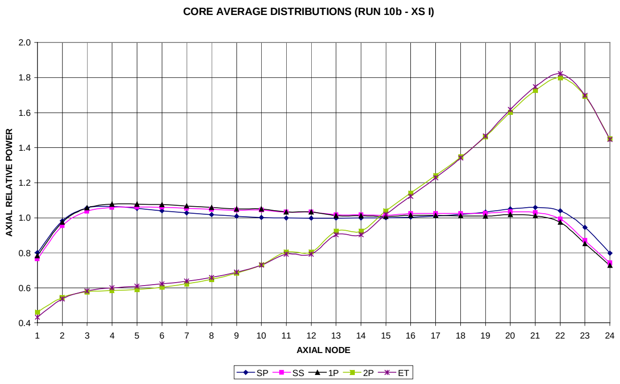

Fig. 8 – Axial total power distribution for the TM1-1 MSLB calculated by the code run No. 3. Specified power (SP), power calculated at steady state (SS), during the first and the second power peak (1P and 2P, respectively, see below) and at the end of the transient (ET), are reported.

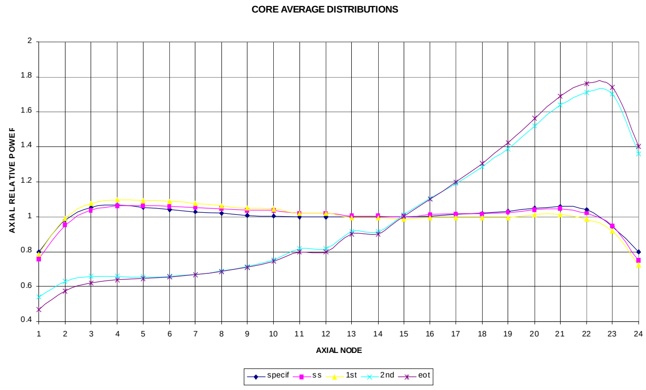

Fig. 9– Axial total power distribution for the TM1-1 MSLB calculated by the code run No. 4. Specified power (SP), power calculated at steady state (SS), during the first and the second power peak (1P and 2P, respectively, see below) and at the end of the transient (ET), are reported.

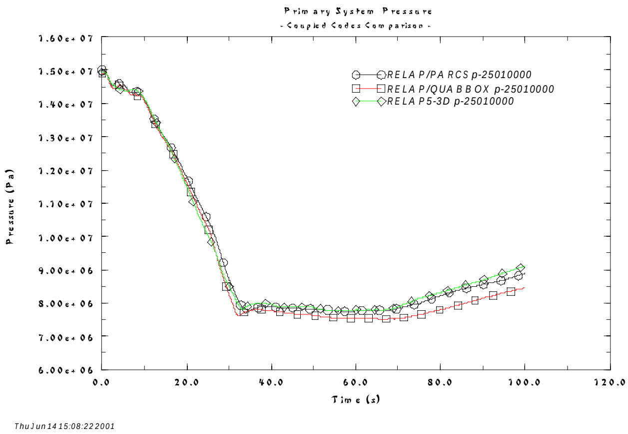

Fig. 10 – Application of coupled thermalhydraulics 3-D neutronics codes to the TMI-1 MSLB: Primary System pressure

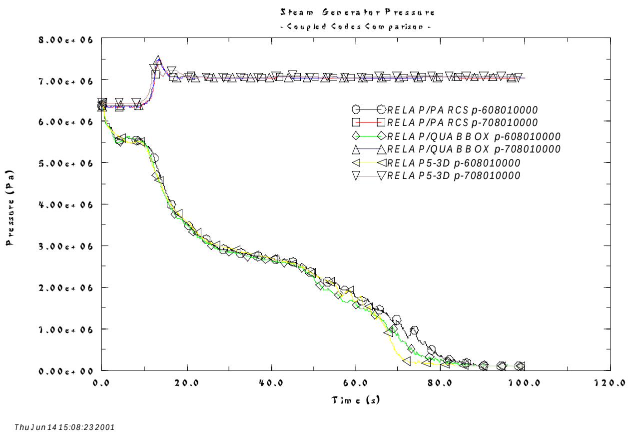

Fig. 11 – Application of coupled thermalhydraulics 3-D neutronics codes to the TMI-1 MSLB: Steam Generator pressure.

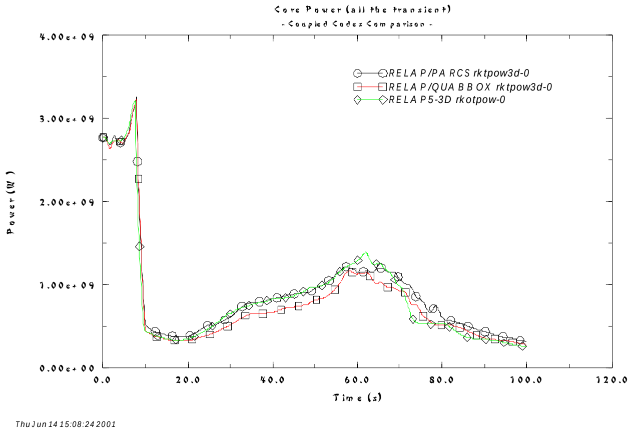

Fig. 12 – Application of coupled thermalhydraulics 3-D neutronics codes to the TMI-1 MSLB: Core Power (all the transient).

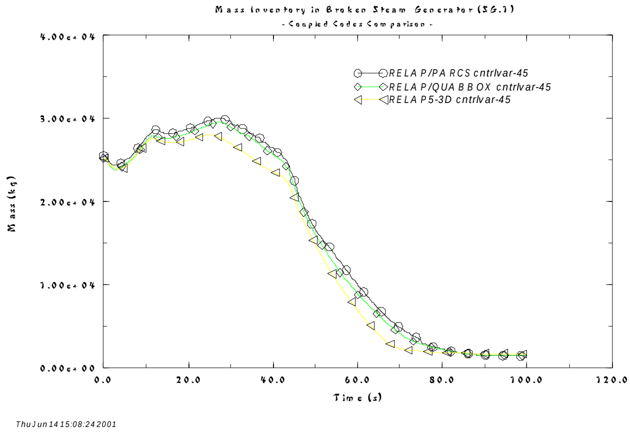

Fig. 14 – Application of coupled thermalhydraulics 3-D neutronics codes to the TMI-1 MSLB: Mass inventory in Broken Steam Generator (Steam Generator No. 1).

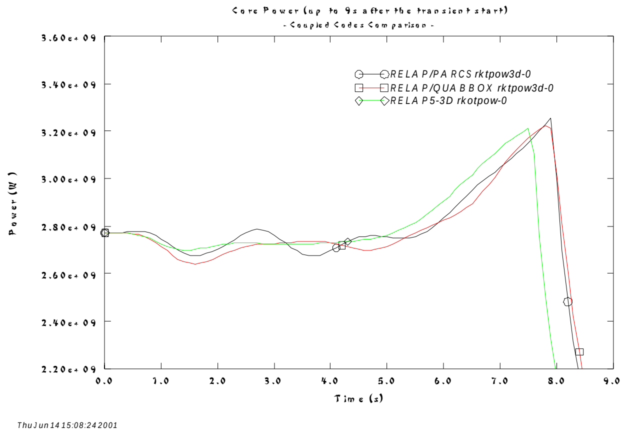

Fig. 13 – Application of coupled thermalhydraulics 3-D neutronics codes to the TMI-1 MSLB: Core Power (up to 9 s after the transient start).

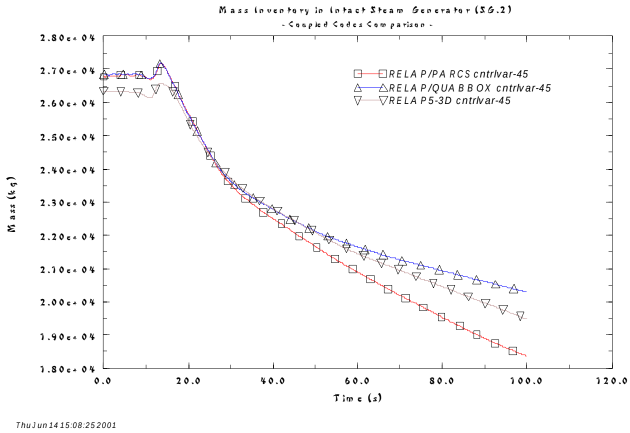

Fig. 15 – Application of coupled thermalhydraulics 3-D neutronics codes to the TMI-1 MSLB: Mass inventory in Intact Steam Generator (Steam Generator No. 2).

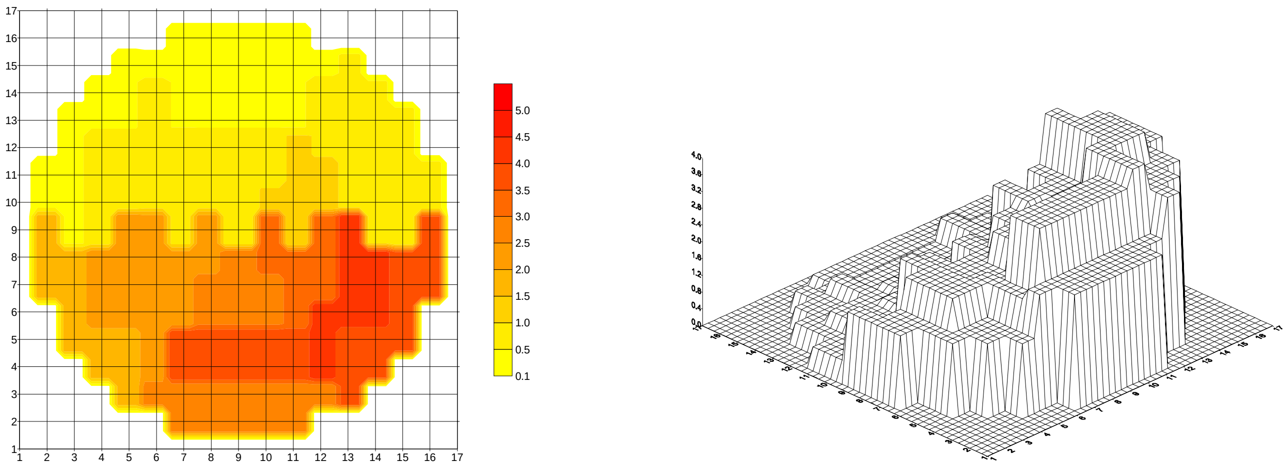

Fig. 16 – Radial power distribution for the TMI-1 MSLB calculated by the code run No. 3 at the time of the second core power peak. (Relap5/Parcs).

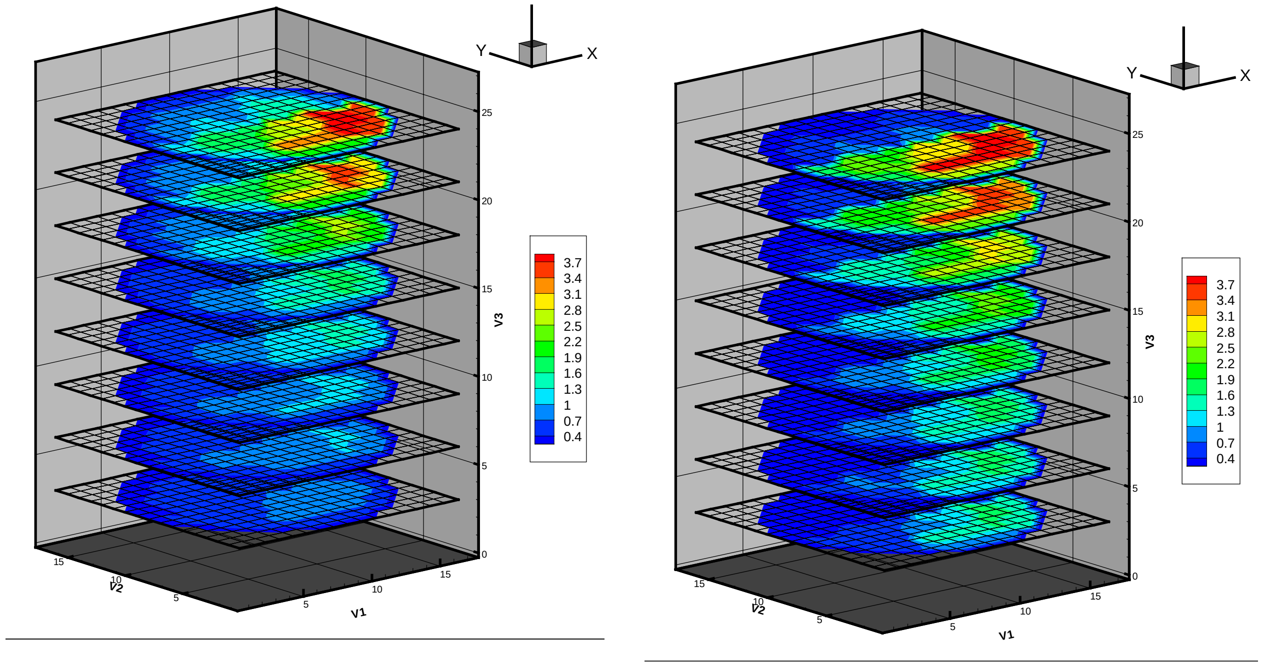

Fig. 17 – Radial power distribution for the TMI-1 MSLB calculated by the code run No. 4 at the time of the second core power peak and at end of transient. (Relap5-3D).

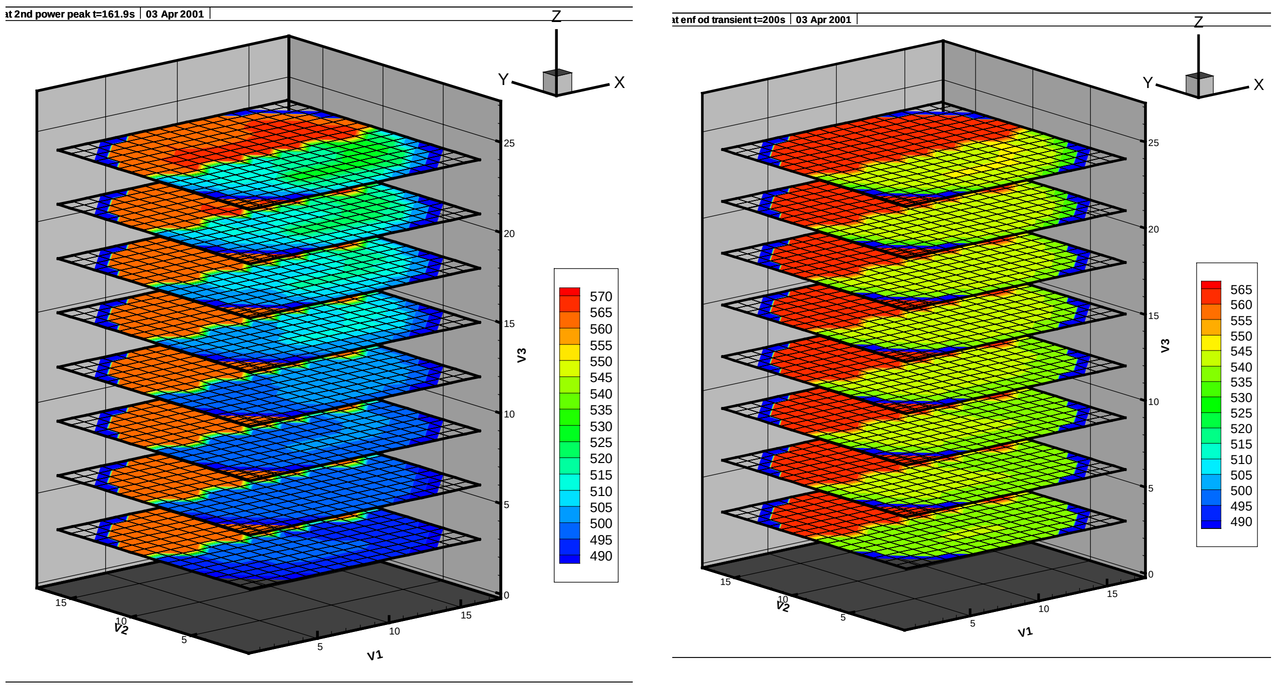

Fig. 18 – Radial moderator temperature distribution for the TMI-1 MSLB calculated by the code run No. 4 at the time of the second core power peak and at end of transient. (Relap5-3D).