Proceedings of ICONE12
12th International Conference on Nuclear Engineering
April 25-29, 2004, Arlington, Virginia USA

ICONE12-49300

DEVELOPMENT OF THE METHODOLOGY OF THE SAFETY ANALYSIS PERFORMED BY THE COUPLED KIKO3D/ATHLET CODE SYSTEM IN VVER-440 TYPE NPP

Gy. Hegyi, G. Hordósy, A. Keresztúri, Cs. Maráczy, I. Panka, M. Telbisz I. Trosztel

KFKI Atomic Energy Research Institute
Reactor Analysis Laboratory
H-1525 Budapest 114, POB 49, Hungary
ghegyi@sunserv.kfki.hu

ABSTRACT

In the deterministic safety analysis codes are required in order to provide evaluations of potential nuclear plant accidents. In the fields of the core transient behaviour, the computer codes have achieved a high degree of realistic modelling. Nevertheless, some further tools for the investigations of the wide range of physical phenomena in the whole plant transient, such as modeling the ex-core detector signals and the malfunctioning of the emergency control system are unavoidable, too.

The programs and methods used in KFKI-AEKI for safety analysis of VVER-440 NPP are presented. The accident analysis methodology for a boron dilution scenario, in which an inactive coolant loop is started, is shown.

INTRODUCTION

The transient caused by a perturbation of boron concentration and coolant temperature at the inlet of a Russian developed reactor (VVER-440) is analyzed as a part of the modernization (introduction of a new type profiled fuel) and power upgrading (up to 108 %) project. This task is one of the basis cases to be investigated in the safety analysis of the pressurized water reactor (PWR), where the effectiveness of the reactor protection and safeguard systems in preventing the damage of the fuel has to be demonstrated. The accurate modeling of the diluted and cooled slug transport and mixing in the vessel when a Main Circulation Pump (MCP) starts up in a loop which is closed at the beginning is a very demanding task because various complicated processes, such as asymmetric power generation, mixing in the reactor vessel and various protection and conventional automation signals contribute to the scenario. The induced power peak depends on the magnitude and the time schedule of the inserted reactivity. Because of high safety relevance of the question, it has been extensively investigated recently, including experimental tools, too [1-2].

First detailed planning calculations were performed with the thermal hydraulic system code ATHLET and neutron physical code system KARATE-440 to find out the appropriate initial parameter set taking into account the active safety system of the NPP. Finally the most reactive case was analyzed by the KIKO3D/ATHLET coupled system code. On the basis of its results, the stand-alone TRABCO code performs the hot channel calculations. Whereas the investigation is done for safety analysis, conservative assumptions are imposed on reactivity characteristics. Moreover at the core inlet no-mixing is supposed from the unaffected loops. The presented calculations show, how the coupled code system with a detailed description of plant functions and core behaviour can help to understand better the local phenomena in this study as it offers the possibility to evaluate the plant safety in a more realistic and versatile manner.

GENERAL FEATURES OF THE APPLIED CODES AND THEIR COUPLING

The large progress in computer technology enables the direct coupling of such complicate and separate code systems like the 3D neutron kinetics, advanced thermo-hydraulics and plant dynamics. The integration of the analysis increases the accuracy as the conservative boundary conditions at the interfaces can be avoided. Only such types of codes are capable of estimating the feedback effects in a realistic way for instance, in reactivity initiated accidents with strongly asymmetric neutron flux distribution caused by the perturbation in one of the primary circuit loops. Another very attractive feature of the coupled calculations, that the codes can preserve their carefully validated mathematical models and problem dependent data bases.

Recently a coupled neutronics/thermal-hydraulics code system [3-4] was developed in KFKI AEKI, which couples the hexagonal VVER specific space-time kinetics code KIKO3D to the system transient code ATHLET.

The **KIKO3D** code [5-6] has been developed in the KFKI AEKI. Its neutron kinetics model solves the two-group neutron diffusion equations in the homogenized fuel assembly geometry by a sophisticated nodal method. Special, generalized response matrices of the time dependent problems are introduced. The time dependent nodal equations are solved by using the Improved Quasi Static factorization method. The thermo-hydraulics is calculated in separate axial hydraulic channels of the core, each of which relates to one fuel assembly. The conservation equations of mass, energy and momentum are solved for liquid and vapour phases. In order to get an accurate representation of the fuel temperature feed-back, a heat transfer calculation with several radial meshes is done for an average representative fuel rod in each node. The release of prompt and delayed nuclear heat in the fuel is modeled. In the present version of the code, the VVER-440 correlations are used in its thermo-hydraulic module. The input 3D burnup distribution of the core is based on the calculation of the stationary KARATE-440 code system [8]. It is supposed that not too large number of parameters, (e.g. reactivity coefficients, peaking factors) determine the main characteristics and consequences of the transients. The maximum or/and the minimum values of such parameters form the set of the frame parameters, which are also based on the KARATE-440 calculations. Consequently the actual values of the frame parameters must be calculated before each reloading in the core design phase [7]. The user of KIKO3D is supported by special subroutines and input possibilities, which make possible the tuning of the prescribed frame parameters at the initial state of the transients.

The **ATHLET** one dimensional thermo-hydraulic system code developed by the Gesellschaft für Anlagen- und Reaktorsicherheit (GRS) [9-10] has a wide range of application for the analysis of anticipated and abnormal plant transients in PWRs and BWRs. The code structure allows an easy implementation of different physical models, such as thermo-fluiddynamics, neutron kinetics, General Control Simulation Module. A two-fluid, 6-equation model, with completely separated equations for mass, energy and momentum for both phases, taking into account also the non-condensables is included in the last release version. In our calculations the 1.2A version is used. As our Russian type pressurized water reactors differ from other PWRs due to its special construction, a detailed six loop input model was developed for the ATHLET code in our institute. Including the upper head, downcomer and lower plenum, the pressure vessel is divided into six parallel, separate channels. The more detailed description is expected to reflect the asymmetrical behavior of the loops during the modeled transient and to take into account safety systems (ECC systems, control and protection systems) distributed loop by loop. The details of the plant description were also increased and such a way the primary and secondary circuits are modeled by 12 fluid-dynamic systems. Coolant mixing in the downcomer is modeled. Junctions (single junction pipe, see [9]) are introduced among the nodes of the downcomer and lower plenum to fit the cross flow, which are prescribed from the results of the former CFX calculations. Unfortunately this model is not valid in case of the opening a closed loop. The horizontal steam generator was modeled carefully, too. The bundle of the horizontal steam generator is divided into three parts in the primary side and five in secondary side. Three of the nodes contain the U tubes.

Two ways of the coupling have been developed [3]. In the case of the internal coupling ATHLET obtains the heat source from the decay heat model of KIKO3D and the system code models completely the thermal-hydraulics in the primary circuit including the core region. In the second case the two programs run parallel, but the KIKO3D code use their own thermo-hydraulic model in the core. The coupled system code was successfully validated against benchmarks and direct industrial applications [11].

Special attention was paid to the functioning of the emergency reactor protection system using the signal of ex-core detectors. Generally the power level of the core is measured by 3 ex-core detectors positioned in rotational symmetry of 120 degree around the core. Two detector sets can be found around the core, their relative position is 15^{o} rotation and both of them contain one detector, which is very close to the first, affected loop (see Fig 1). Both sets of ex-core detectors can be taken into account. The detector signals were evaluated from the space and time dependent flux in the core, which was calculated by KIKO3D and from predetermined transfer functions between the outgoing current at the core periphery and the detector reaction rate. The scram signal is initiated either if the reactor-period is less than 10 s, or if the signal level reaches the prescribed value.

It turned out, that the initiation of scram is really delayed due to the space dependent detector readings, especially, in that case, when a control rod is stuck during the SCRAM. The following circumstances were chosen in a conservative manner:

1. The scram was initiated if two detectors of three show the high power level, taking account of the 4 % uncertainty of the power set-up,
2. The set of detectors (closest to the 6^{th} loop) from which the higher time delay was gained was chosen as active one during the transient.

As an important tool in the safety calculations, the code system was extended by the capability of the hot channel analysis, which is the axially one-dimensional reactor dynamics TRABCO code, developed by VTT, Finland as a part of transient code SMATRA [7]. SMATRA includes 1D model for two-group neutronics, fuel rod heat transfer, and thermal hydraulics of typical coolant channel. Fuel temperature rise after boiling crisis and clad oxidation are modeled as extreme phenomena. TRABCO core model of SMATRA can be used separately for hot channel analyses on the basis of the output files of the main calculations made with SMATRA 1D or other 3D dynamic code, e.g. KIKO3D. Detailed validation procedure was done for that code package, too. The validation work for SMABRE and SMATRA against Loviisa measurements is described in ref. [7].

The hot channel analysis was performed by the TRABCO code. The time-dependent KIKO3D axial power distribution of the most loaded assembly was multiplied by the K_x radial power peaking factor. This factor was determined from the core design limits of the maximum linear heat rate (315 W/cm at EOC) and the maximum pin power (57 kW). Assuming that these limits are valid at 104 % of the nominal power, the maximum value of K_x radial power peaking factor was limited to 1.59 due to the 57 kW pin power limit. As the initial value of K_q and K_z are known from the KIKO3D calculation, K_x is responsible for K_k and the engineering safety factor. While the time dependent power distribution is taken from the KIKO3D calculation, the inlet enthalpy flow, etc. come from the ATHLET results.

The last two models could not run simultaneously with the coupled system code.

STARTUP OF AN INACTIVE COOLANT LOOP WITH LOWER BORON CONCENTRATION, 5 LOOPS IN OPERATION, MSIV CLOSED

As an illustration, the analysis of the startup of one inactive coolant loop with lower boron concentration is presented, which proves the applicability of the KIKO3D/ATHLET coupled code for safety analysis.

Event description

The initiating event may occur in the case of erroneous startup of a long time inactive loop containing diluted boron water. At the beginning five reactor coolant pumps are in operation at the appropriate power (about 80 %) which is just reached after the fast power load. In the primary circuit the boron concentration is increased in order to compensate for the decayed xenon. One loop (the first one) is inactive due to closed main line isolation valves (MLIV) hence there is no reverse flow in that loop. Incorrectly, the reactor coolant pump is started first then the main gate valves are opened. It is assumed that the cold and deborated slug of the starting loop enters its own sector. As a further conservative assumption during scram the most effective control rod is stuck in its upper position in the effected sector, too.

Having been chosen the reasonable conservative scenario, it was analyzed using the KIKO3D/ATHLET coupled thermo-hydraulic system code and 3D neutronics models. The most important issue of the transient is whether the re-criticality due to a strong fast deboration and cooling, combined with an incomplete reactor scram could cause fuel damage. Extending the investigation with hot channel analysis, the results indicate that the cooling of the fuel is not threatened in this transient, so the criteria of Anticipated Operational Occurrences (AOO) are fulfilled for that scenario.

Initial conditions and Input data set

Extensive analysis was done by the KARATE-440 steady state code system and KIKO3D neutron kinetics code to evaluate the suitable core and performance parameters, reactivity coefficients. The thermo-hydraulic system code ATHLET was used to make realistic prediction for the parameters of the closed loop and the nominal parameters for the rest, 5-loop primary system.

For the purpose of determination of the suitable reactor state a 325 full power day equilibrium fuel cycle, using the new profiled fuel (average enrichment is 3.82 %) was planned at a 1485 MW power level, which corresponds to an 8 % power upgrading. The higher thermal power causes some increase in the core inlet temperature T_{in} = 269 ^{o}C, too. The 3D burnup distribution of the core was calculated by the KARATE-440 code. The most important best estimate and enveloping parameters are collected in Table 1. Its first column contains the name and the reactor state (**F**ull **P**ower or **H**ot **Z**ero **P**ower), where the coefficients were calculated. In case of scram 37 control assemblies fall into the core, but in this simulation one of the most effective assembly is supposed to be stuck in its upper position, conservatively.

Table 1: Feedback parameters of the new fuel loading

Parameters\Reactor state	Best estimate value Beginning of Cycle	Best estimate value End of Cycle	Enveloping parameter
Delayed Neutron total fraction at FP [%]	0.596	0.519	0.442
Coolant temperature feedback at FP [pcm/K]	-17.6	-48.9	T<269 ºC: -70.0 T>269 ºC: -35.0
Fuel temperature feedback at FP [pcm/K]	-3.5	-3.5	-2.4
Boron concentration feedback at FP [%/g/kg]	-1.170	-1.397	-1.90
Control rod reactivity worth at HZP [%]	-5.7	-5.6	-5.1

On the basis of Table 1, the transient was initiated at the end of cycle, when the working group position was H_6 = 220 cm, the further scram rods were out of the core H_{1-5} = 250 cm and the critical boron concentration was 0 ppm at full power

(1485 MW). At this boron concentration the first loop was closed and the reactor was shut down. After that xenon transient occurred. It takes about 4 days while the xenon is decayed.

After that the power of the reactor is loaded up again and parallel with it the control group is pulled out (H_6 = 250 cm), to reach higher boron concentration, consequently. In case of five working loop, the maximal power can be the 83 % of the nominal one, and the level of the Emergency Reactor Protection System (ERPS) is 105 % of the nominal power. The switch level of ERPS can be kept until the power is not less than 79 %. Conservatively that power level was chosen for the 5 loop stationary case, which was reached as fast as the regulation of the plant permitted. At the end of this preliminary simulation, the boron concentration in the 5 loop primary system is 369.5 ppm. The incorrect start-up of the earlier closed loop was initiated from that reactor state.

In the coupled calculation the 3D core behavior is modeled by the KIKO3D code, where the above described burnup distribution, boron concentration, core flow and the conservative feedback values were set up (see Table 1). Due to the very asymmetric perturbation, all the 349 assemblies of the core were calculated. The total length of the active core was divided into 10 axial layers. The loose type (internal) coupling option was used in the coupled code so the ATHLET gave the feedback data to the reactor kinetic calculation. The assemblies were grouped into 27 ATHLET super channels as it is shown in Fig. 1 and summarized in Table 2. Special attention was paid to the simulation of the ERPS using the signal of ex-core detectors and the malfunctioning of the emergency control system. During the scram the most effective control assembly from the first sector was stuck at its upper position (27^{th} super channel, see Fig. 1). The orientation of the ex-core detector sets used by the ERPS system (D11, D12, D13 & D21, D22, D23), can be seen in Fig. 1, too. It is supposed conservatively that the scram is not initiated from the signal "reactor-period less than 10 s", but from the signal "level reaches the prescribed value" (see above). The detectors, in the vicinity of the first loop give fast and large signal, which will be qualified as false one. In this way the two distant detectors initiate the scram, which causes some delay to the set point comes from the level of the average power (Fig. 2).

The plant behavior was modeled by the ATHLET code. Preliminary calculations were performed to find out the behavior of the closed inactive (first) loop with open main loop isolation valves (MLIVs). As the MLIVs were closed no mixing of the water of the stagnant loop occurred to the active five ones and there was no reverse flow in this loop. On the other side there was a limited cooling process through its steam generator (SG). The loop was inactive more than a day (it is an overestimated value as the MLIV of a loop is not permitted to be closed so much time). As the SG of the first loop was not closed on the secondary side (main steam isolation valve, MSIV is open), it was fed up by water with 221 ^oC up to the nominal level but above it the steam of the joint steam generators existed. Internal circulation occurred in both sides of the SG, which caused a limited cooling in the closed loop. Finally the average temperature of the stagnant flow was about 250 ^oC. On the other hand, the two steam domes of the secondary side differ as the closed loop cooled one of them. The restart of the loop was modeled on the basis of some measured plant data. The MLIVs are opened 5 second after the pump in this loop was started. According to the plant data the discontinuous opening of the MLIVs lasted ca. 125 s in four periods. Keeping the cross section of the flow used in the measurement, parametric studies were done to open the MLIV prior to the run-up or even the startup of the MCP and to change the opening time of the MLIVs. In the final scenario the interval was cut down to provide a sudden change in the parameters of coolant flowing from the first loop to the core.

Summary of the results

The analysis has been done with the KIKO3D/ATHLET coupled code in two steps. First, on the basis of the preliminary calculations the initial values were achieved. The sequence of the calculation procedure and the corresponding core parameters are summarized in Table 3.

Table 3: Sequence of the calculation procedure

Time(s)	Events
0-100	Steady state: 6 loops, N=1173.2 MW, T_{in}=269 ^oC, C_{Boron}=2.11 g/kg, H_6=215 cm
100-160	First MCP is stopped, MLIVs are closed
160-380	In the closed loop: T_{in} = 250 ^oC, C_{Boron} = 0.0 g/kg are set up
380-400	Steady state: 5 loops, N=1173.2 MW, T_{in}=269 ^oC, C_{Boron}=2.11 g/kg, H_6 = 250 cm

The simulation of transient, which describes the combined effect of the stuck control rod and the above mentioned scenario for the opening the MLIVs started at 400 s. The case of the highest perturbation is presented. The MCP started to work 8 s beforehand opening the MLIVs. The time dependent flow-rate can be seen in Fig. 3 for all the six loops. The startup events as well as some important parameters of the transient are summarized in Table 4.

To eliminate the effect of the mixing and the numerical diffusion the outlet of the loops were transformed directly to the core inlet. It was prepared on the artificial way, by the help of LEAK & FILL technique in ATHLET code. It means that junctions were defined at the outlet of the loops and at the upper parts of the lower plenum and by this way the flow of the loops was directed just under the core. The time shift of the transformation was 4.4 s due to the flow-rate and the distance between the leak and fill junction. The parameters of the flow at the inlet of core-sector could be calculated as a weighted sum of the original parameters and parameters of the time dependent flow of the opening loop. In case of boron concentration the original value entering the core and the resulted one entering the segment of core can be seen in Fig. 5

and 6. The inlet temperature evaluated similarly way can be seen in Fig. 4. The asymmetric effect of the slug can be seen in the power distribution and in the transferred heat to the secondary side, too (see Fig. 7-8). Due to the high reactivity insertion the signal level of the ex-core detectors increased and the scram signal is initiated. After the conservatively chosen 0.5 s time delay the scram was initiated and the core was shut down safely. The most loaded assembly of the core was chosen for hot channel analyses. The calculation was made by the TRABCO code, which used the time dependent flow, the inlet enthalpy, and outlet pressure of the assembly from the ATHLET calculation and the axial power from KIKO3D. Boiling crisis did not occurred, the DNB ratio (Gidropress correlation) was higher than 2.0 (see Fig. 9-10)

Table 4: Chronology of events

Time(s)	Events
392.0	MCP No. 1 is switched on
400.0	Opening MLIVs in the first loop
405.0	MLIV in the cold leg of the first loop is opened up to 35 %
409.92	SCRAM signal is initiated
410.42	36 control assemblies (CA) fall into the core
414.9	Pressuriser heater 2. group switches on, then the 3. 4. & 5. group by 0.5 s delay
419.92	The valves of 1. and 2. turbine are closed
422.42	36 CA reach their lowest positions
423.8	Steam dump valves to the condenser opening
432.7	$P_{pressuriser}$ < 113 bar
478.0	MLIV is fully opened in the cold leg of the first loop

CONCLUSION

The main objective of this work is to show the status of the code package based on the best estimate coupled code system ATHLET/KIKO3D and to present a safety analysis of an inadvertent startup of an inactive loop with lower boron concentration in a VVER-440 NPP. The origin of the conservative parameters used in the final calculation was introduced. The detailed input deck of ATHLET comprising 6-loop nodalisation was successfully used in the preliminary calculations and in the KIKO3D/ ATHLET coupled code, too. The cooling and the strong dilution increase the reactivity resulting in increasing power level especially in the effected sector. Due to the use of time dependent signals of the ex-core detectors the SCRAM is delayed. Investigating the DNBR value by TRABCO code, dangerous hot spot was not found. The DNB results are acceptable according to the AOO criteria.

REFERENCES

[1] H. M. Prasser, G. Grunwald, T. Höhne, S. Kliem, U. Rhode and F.P Weiss, "Coolant Mixing in a Pressurized Water Reactor: Deboration Transients, Steam-line Breaks and Emergency Core Cooling Injection", Nuclear Technology, Vol. 143 July 2003, pp.37-55.

[2] M. Antila, P. Siltanen, R. Kyrki-Rajamäki, "Study of Core Response in Reactivity Accidents due to Local Dilution of Boric Acid Concentration", ENC’94 International Congress – Atoms for Energy Lyon 2-6 October 1994 Transactions Vol. II. Pp 105-111.

[3] A. Keresztúri, Gy. Hegyi, Cs. Maráczy, I. Trosztel, H. Austregesilo, F. Cester, S. Langenbuch, K. Velkov, The Coupled ATHLET-KIKO3D, The Neutronic and Thermohydraulic Models, Verification Results, User’s Guide, Garching-Budapest, 1997.

[4] Gy. Hegyi, A. Keresztúri, I. Trosztel, S. Langenbuch, W. Horche, K. Velkov, "Improvement of Plant Transient Analysis for VVER by Coupling KIKO3D with ATHLET", ICONE-6, San Diego CA USA, May 10-14, 1996.

[5] A. Kereszturi: KIKO3D - a three-dimensional kinetics code for VVER-440. Transactions of the ANS Winter Meeting, Washington, 1994

[6] A. Keresztúri, Gy. Hegyi, Cs. Maráczy, I. Panka, M.Telbisz, I. Trosztel and Cs. Hegedus, “Development and validation of the three-dimensional dynamic code - KIKO3D, Annals of Nuclear Energy 30 (2003) pp. 93-120.

[7] A. Keresztúri, at al., "Development, Validation and Application of Tools and Methods for Deterministic Safety Analysis of RIA and ATWS Events in VVER-440 Type Reactors", CSNI Workshop on Advanced Thermal-hydraulic and Neutronic Codes: Current and Future Applications, 10-13 April, 2000.

[8] Cs. Hegedus, Gy. Hegyi, G. Hordósy, A. Keresztúri, M. Makai, Cs. Maráczy, F. Telbisz, E. Temesvári, P. Vértes, "The KARATE Program System", PHYSOR 2002, Seoul, Korea, October 7-10,2002.

[9] G. Lerchl, H. Austregesilo, ”ATHLET Mod 1.2 Cycle A, User's Manual”, Code Doc. Package, GRS-P-1/Vol.1, Rev.1, March 1998.

[10] V. Teschendorff, H. Austregesilo, G. Lerchl: Methodology, Status and Plans for Development and Assessment of the Code ATHLET, Workshop on Transient Thermal-Hydraulic and Neutronic Codes Requirements, Annapolis, Nov. 5-8, 1996.

[11] S. Mittag, S. at al.: Validation of coupled neutron kinetic/thermal-hydraulic codes, Part 2: Analysis of a VVER-440 transient (Loviisa-1); Annals of Nuclear Energy, 2002, Vol 29, pp 255-269.

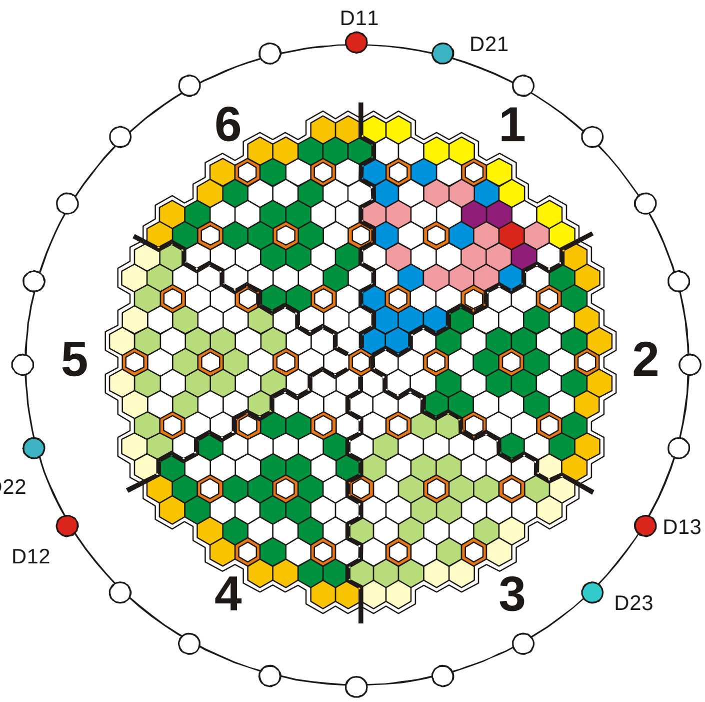

Fig. 1: Nodalization of the core into super-channels and the positions of the ex-core detectors

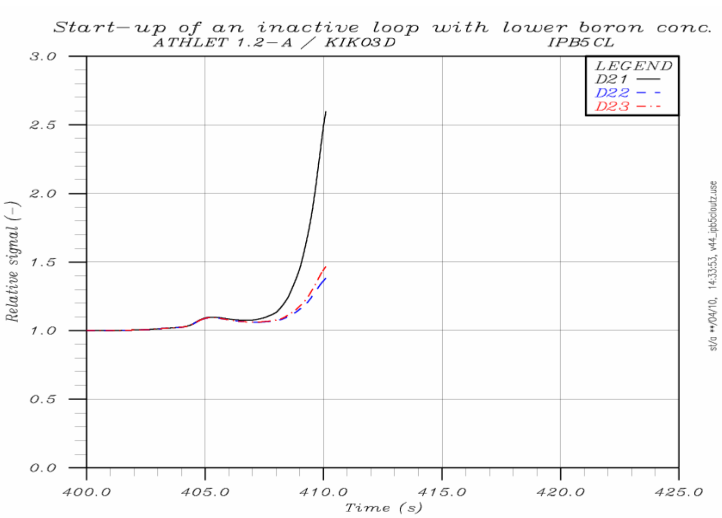

Fig. 2: Signals of the ex-core detector set

Table 2: Core pattern with the 27 ATHLET super-channels

No.	Marker	Number of heated rods
1		3150
2		2394
3		1008
4		777
5		3024
6		2520
7		1008
8		777
9		3024
10		2520
11		1008
12		777
13		3024
14		2520
15		1008
16		777
17		2898
18		2520
19		1134
20		777
21		1008
22		1764
23		1890
24		1512
25		378
26		651
27		126 -stuck assembly

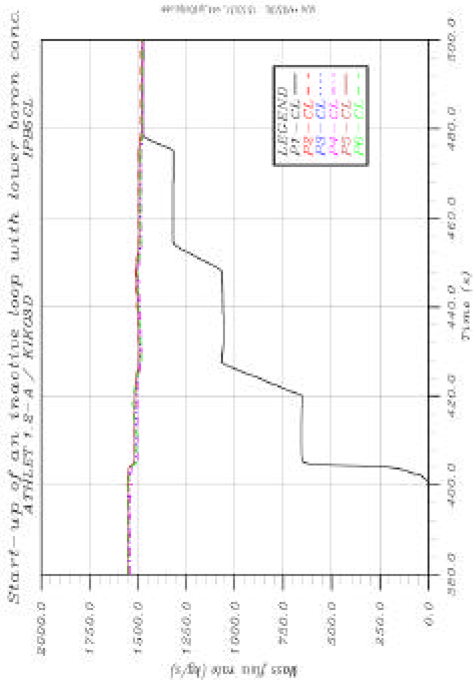

Fig. 3: Mass flow rate in the six loops of primary circuit

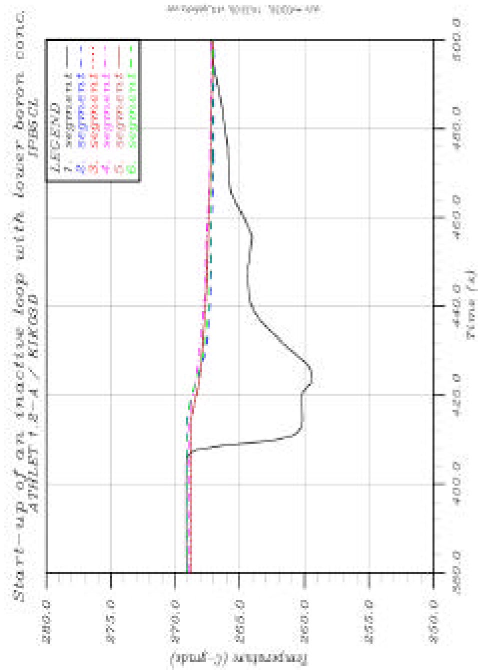

Fig. 4: Coolant temperature at the inlet of the core segments

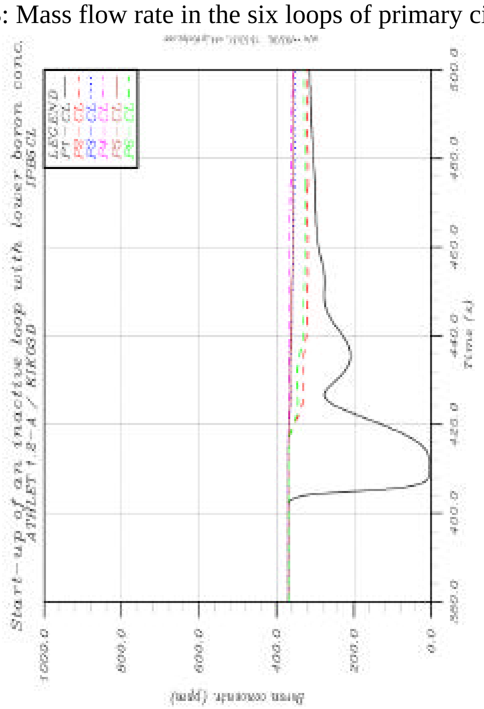

Fig. 5: Boron concentration at the six inlets of vessel

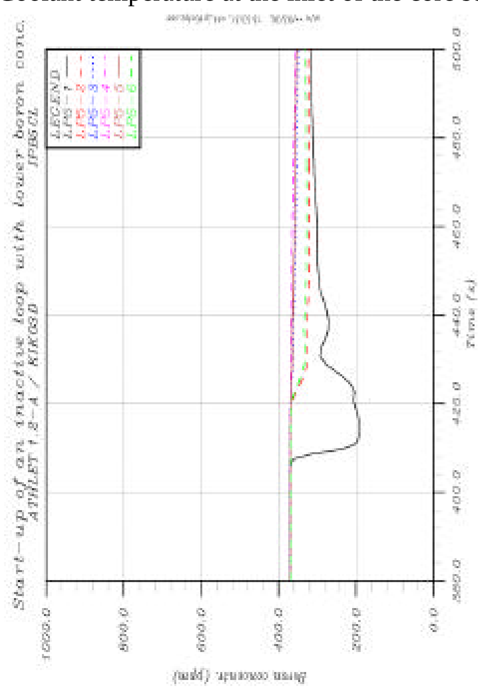

Fig. 6: Boron concentration at the inlet of the core segments

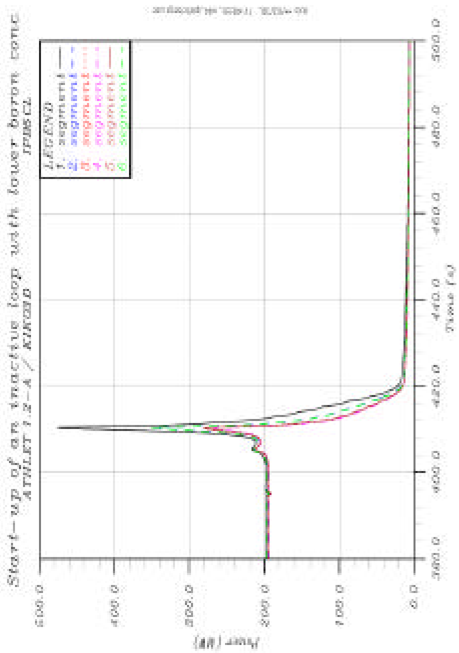

Fig. 7: Power of the six core segments

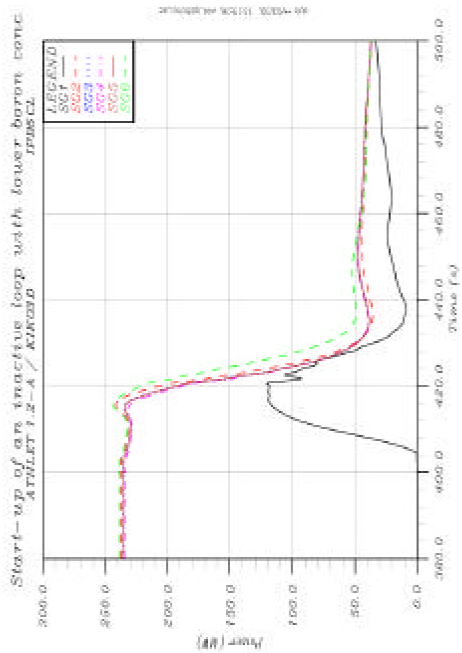

Fig. 8: Heat transferred to the secondary side in the steam generator

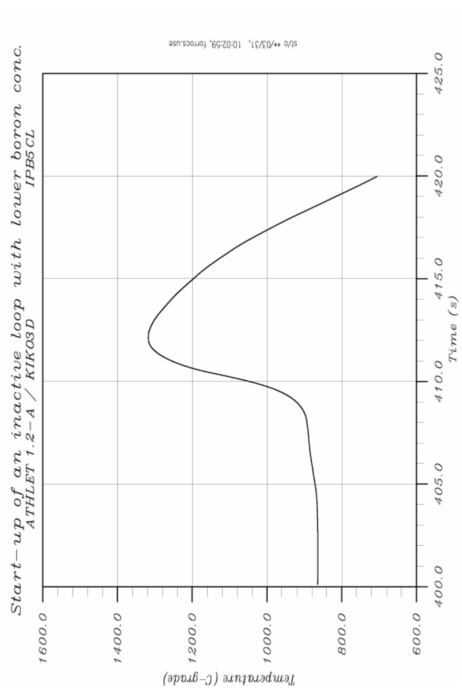

Fig. 9: Max. of fuel centerline temperature in the most loaded fuel rod

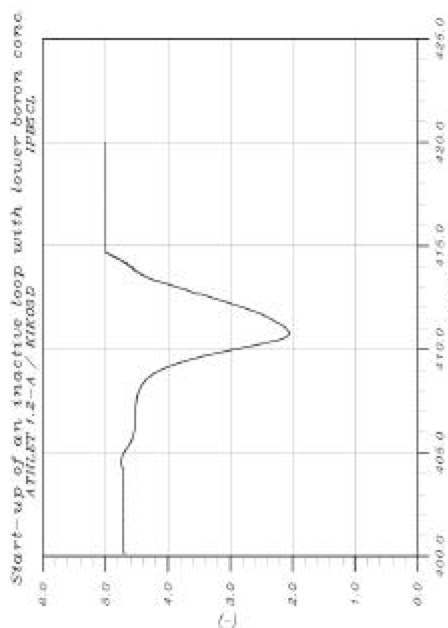

Fig.10: Minimum of DNB ratio in the most loaded fuel rod (Gidropress)