AER Benchmark Specification Sheet

1. Test ID: AER-DYN-006

2. Short Description:

The 6th dynamic benchmark concerns a double ended break of one main steam line (asymmetric MSLB) in a VVER-440 plant. The core is at the end of its first cycle in full power conditions. The control rods of group K6 are at position 175 cm from bottom of the core. All other groups of control rods are out of the core. The initial state conditions of the core in the beginning of the transient are given. The isothermal re-criticality temperature of the core is defined to be 210°C. It should be achieved by tuning the worth of all control rods. Otherwise, own best estimate nuclear data are to be used. The main geometrical parameters of the plant and the characteristics of control and safety systems to be considered are given. Otherwise, own input data decks developed for a VVER-440 plant and for the applied codes can be used.

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6. Objective:

The objective is to calculate the behaviour of the core during a re-criticality event using coupled codes, which combine a three-dimensional neutron kinetics code for the core with a thermal hydraulics system code. This benchmark is a logical continuation of the work in the 5th dynamic benchmark. Additional new features include asymmetric operation of the feed water system, effects of coolant mixing in the reactor vessel, and the definition of a fixed isothermal re-criticality temperature for normalising the nuclear data.

7. Rationale for Test Setup:

This benchmark is a continuation for testing the performance of coupled codes. During a steam leak, there is a strong interaction between core behaviour and the thermal hydraulics of the primary and secondary circuits. A number of features in this test are chosen to reduce the problem size and to eliminate extra complications. The asymmetric leak is assumed in one of the main steam lines. Two of the most reactive control rods in the sector of highest over-cooling are assumed stuck at its fully withdrawn position. This requires a full core calculation in consideration of mixing in the lower and upper plenum.

8. Input:

The reference plant for the definition of this benchmark is a VVER-440/213. The definition is based on the assumption that all possible participants of the benchmark have input data decks for the VVER-440 which have been developed according to the needs of their own thermal hydraulics system/neutron kinetic core models. Therefore information about all details necessary for the creation of a new input data deck is not provided. The main geometrical parameters are given to adjust the existing input data decks.

a: Geometry of the primary circuit

Tab. 1: Hot Leg Geometry

Elevation (z) [m]	Length (x) [m]	Diameter (d) [m]
0.0	0.0	0.496
0.0	1.2	0.496
-1.40	5.28	0.496
-1.40	13.52	0.496
-0.45	14.47	0.496
-0.25	14.67	0.800
2.38	17.30	0.800
2.73	17.65	0.550
3.53	18.45	0.496

Tab.: 2 Cold Leg Geometry

Elevation (z) [m]	Length (x) [m]	Diameter (d) [m]
3.53	0.0	0.496
2.73	0.80	0.550
2.38	1.15	0.800
-0.48	4.01	0.800
-0.60	4.13	0.496
-2.92	6.45	0.496
-2.92	15.76	0.496
-1.15	17.53	0.496
-1.15	18.43	0.496
-1.40	18.68	0.496
-1.40	26.28	0.496

The beginning of the hot leg is set to the elevation 0.0m. All elevations provided in the tables 1-3 and 5, 6 are related to this reference point. For checking the geometry of the primary circuit, the most relevant data of the hot leg together with the steam generator (SG) inlet collector are presented in Tab. 1 and of the SG outlet collector together with the cold leg in Tab. 2. It should be kept in mind, that the geometry data in Tab. 1 and 2 are only some key data for tuning the data sets. Details of the geometry are not provided. The 5536 U-tubes have an inner diameter of 13.2mm, an outer diameter of 16.0mm and an averaged length of 9.02m. They are distributed on the collectors from elevation z=0.14m to z=1.96m. The reactor pressure vessel (RPV) elevations are shown in Tab. 3. Tab. 4 contains the water volumes of the main parts of the RPV. The pressurizer is connected to one loop (No. 5 acc. to Fig. 2) by two surge lines of a diameter of 0.21m and a length of 25.37m, each. The connecting point of the surge lines to the hot leg is at x=7.37m (acc. to Tab. 1). The surge lines can be modelled by one two fold line. The lowest elevation of the pressurizer is -1.15m and the highest is 8.85m. The diameter is 2.40m, the whole volume 44.0m^3. The volume control system is

Tab. 3: RPV Elevations

Elevation z [m]	
-9.86	lowest RPV elevation
-6.02	beginning of the unheated core part
-5.38	lower fuel boundary
-2.94	upper fuel boundary
-2.46	end of the unheated core part
3.56	highest RPV elevation

Tab. 4: RPV Volumes

Object	Volume [m^3]
Downcomer	18.8
Lower plenum	23.1
Core region	12.7
Upper plenum including vessel head	40.9

connected to the cold legs of loops 2 and 6 (acc. to Fig. 2) at x= 8.45m (acc. to Tab. 2). The High Pressure Injection System (HPIS) consists of three trains. The trains are connected to the cold legs of loops 1, 3 and 5 at x=21.66m (acc. to Tab. 2).

b: Geometry of the secondary circuit

The SG has an inner diameter of 3.21m (the corresponding elevations are z=-0.025m and z=3.185m). On the top it is connected to the main steam line (MSL). The connecting lines between SG and MSL can be omitted. The elevations and lengths of the MSL are shown in Tab. 5. The steam line isolation valves are located 41.33m from the outlet of the steam generators. The main steam header (MSH) is a pipe with a diameter of 0.425m and a length of 83.40m (Tab. 6). It is directly connected to the MSL (without any small connecting pipes) at x=51.25m (acc. to Tab. 5). The MSH isolation valve is not modelled.

Tab. 5: Main Steam Line

Elevation (z) [m]	Length (x) [m]	Diameter (d) [m]
3.185	0.00	0.425
3.185	9.80	0.425
5.385	12.00	0.425
5.385	26.80	0.425
11.485	32.90	0.425
11.485	66.33	0.425
2.785	75.03	0.425
2.785	83.13	0.425

Tab. 6: Main Steam Header

Elevation (z) [m]	Length (x) [m]	Diameter (d) [m]
11.485	0.00	0.425
12.985	1.50	0.425
12.985	81.90	0.425
11.485	83.40	0.425

c: Leak

The leak is postulated as a double ended break in a main steam line before the steam line isolation valve. The location of the leak is 39.7 m from the outlet of this steam generator, which is connected to core sector No. 1 (see Fig. 2). The leak opens within 0.1 s. It is recommended to use a critical discharge model for the simulation of the leak mass flow rate.

d: Reactor core geometry and material parameters

The core loading pattern with three different fuel enrichments is used (Fig. 1). As follows from Tab. 3 the active core length is 2.44 m. The unheated parts below and above the active core have the same hydraulic diameter and free flow cross section like the core. The main fuel parameters are given in Tab. 7. All types of fuel assemblies have the same heat transfer characteristics. 97.5 % of the total power are released uniformly in fuel pellet, the other 2.5 % directly in the coolant of the respective fuel assembly due to γ-radiation. The gas gap heat transfer coefficient is to be kept constant during the whole transient: 3000 W/ (m^2*K). Radial thermal conductivities and thermal capacities of fuel pellet and cladding are described with best data of each participant. Axial transfer of heat is neglected in fuel pellet and cladding.

Each participant should use own best estimate nuclear cross section and other neutronics related data. The decay heat has to be taken into account.

Tab. 7: Main Fuel Parameters

Fuel assembly pitch	14.7cm
Number of heated pins per assembly	126
Fuel pellet inner diameter	0.14cm
Fuel pellet outer diameter	0.76cm
Cladding inner diameter	0.78cm
Cladding outer diameter	0.91cm
Free flow cross section per fuel assembly	89.0cm^2
Equivalent hydraulic diameter	0.86cm
Fuel density	10.4 g/cm^3
Cladding density	6.25 g/cm^3

e: Heat structure modelling

The following components have to be included in the heat structure modelling: the RPV, the primary coolant pipes, the heat exchanger tubes and the pressurizer with the surge lines.

f: Mixing in the reactor vessel

Each participant should use own models for the description of coolant mixing in the lower and the upper plenum. Both upper plenum and lower plenum mixing are applied to temperature and boron acid concentration. To normalize the different mixing models, the rates of exchange of the coolant belonging to any sector with the adjacent sectors are given. The allocation of fuel assemblies (FA) to sectors is presented in Fig. 2.

Lower plenum mixing

An exchange rate of 30 % of the coolant belonging to any sector with the two adjacent sectors is assumed. That means, 70 % of the coolant of the connected loop goes to the belonging sector, 15 % goes to both of the adjacent sectors.

Upper plenum mixing

The mixing in the upper plenum is normalised to an exchange rate of 10% of the coolant from any sector with adjacent sectors (i.e. coolant transport to both of the adjacent sectors is 5%).

g: Characteristics of considered control and safety systems

In this section the characteristics and set points of control and safety systems to be considered in the calculation are given. Only systems and signals which are mentioned explicitly have to be taken into account. All others should be neglected. The pressurizer has four group of heaters. The power, the activation pressure, and the deactivation pressure are shown in Tab. 8. It is recommended to model the reaching of full heater power after switching-on by a low pass filter with a time constant of 5s. When the collapsed level in the pressurizer measured from the bottom drops below 2.56 m, the heaters are automatically switched-off, also through the same low pass filter.

Tab. 8: Pressurizer Heater Groups

	Power [kW]	Activation Pressure [MPa], measured in the PRZ	Deactivation Pressure [MPa], measured in the PRZ
Group 1	180	12.0	12.1
Group 2	180	11.9	12.0
Group 3	540	11.8	11.9
Group 4	540	11.5	11.8

The volume control system is activated in the following manner: The 1st pump of this system is connected to loop No.2. The pump is started when the pressurizer collapsed level drops over 10cm from the nominal level (L_{nom} = 5.97m). If the 1st pump is working for a period of about 40s (i.e. the difference between actual and nominal level is continued), then the 2nd pump is started, which is connected to loop No.6. The mass flow rate is 1.7kg/s per pump. The supplied water has the same boron concentration like the reactor coolant and enters the reactor coolant system with a temperature of 260^{o}C. By reaching the nominal level the volume control system is switched-off.

The signal of high pressure safety injection (HPIS) is formed and the HPIS valves are opened when one of the following two conditions is fulfilled:

- The upper plenum pressure, measured in the RPV at the elevation of the hot leg outlet nozzle, is less than 9.3MPa, and the hot leg temperature in two or more loops is higher than 255°C.
- The pressurizer collapsed level drops below 2.41m, and the hot leg temperature in two or more loops is higher than 150°C.

The HPIS pumps begin to provide highly borated water to the circuit 180s after formation of the HPIS signal. It is postulated, that two out of three trains of HPIS are enabled. These trains are connected to loops No.3 and No.5. The water from the HPIS has a boron concentration of 40g/kg and a temperature of 55^{o}C. The mass flow rate of one train depends at a linear rate from the pressure at the connection point and is defined in Tab. 9.

Tab. 9: HPIS Mass Flow Rate for One Train

Pressure [MPa]	Mass flow rate [kg/s]
0.1	31.7
13.16	0.0

The feed water system has to be modeled in the following manner:
The feed water and steam flow in the initial state should be adjusted to the full power conditions. After the initiation of leak, the feed water mass flow rate to the steam generator of the defect loop (SG-1) is increased to 300kg/s during 20s (linear characteristic). After this increase, the mass flow rate is constant. After the initiation of leak, the regulation of feed water mass flow rate to the other steam generators (SGs in the intact loops) is continued by level control of SG. The controlling value is fixed on a collapsed level (2.015 ± 0.10) m. If the pressure in the main steam header drops below 3.0 MPa, the isolation of feed water to all steam generators is realized during a period Δt=30s by a linear characteristic. In the event of scram, the feed water temperature decreases linearly from 220°C to 160°C during 50s and then remains constant.
The reactor protection (scram) is activated after reaching a reactor power level of 110% of the

nominal value N_N (nominal power is 1375 MW). The time delay of starting the following control rod drop is 0.5s. The velocity of control rod insertion is 25.5 cm/s.
Until the scram initiation, both turbines work at the full power level with the constant mass flow rate of the initial state. The reactor scram signal immediately initiates the turning-off of turbines by closing the turbine isolation valves during 0.5s.
The closing of the steam isolation valves is initiated, if the pressure in the main steam header drops below 3.0 MPa. The steam isolation valves are closed during a period Δt=2.5s.

h: Initial conditions

Burn-up

Because the benchmark calculation will be performed for the end of the first fuel cycle (EOC) conditions, a burn-up calculation for the first loading of the VVER-440 core is required. This calculation should be made at a power level of 1375MW until the critical boron concentration reaches the value zero. During the burn-up calculation all control rod groups are fully withdrawn. Their position will not change. The thermal hydraulic conditions during the burn-up calculation correspond to the conditions of the initial state before the transient. The burn-up distribution obtained at the end of this calculation should be used in the transient calculation.

Initial neutronic conditions

At the beginning of the transient the reactor is at full power level (1375 MW). All control rod groups are fully withdrawn with the exception of control rod group K6 which is at the position 175 cm from the bottom of core. This is different from their position during the burn-up calculation. The locations of control rod groups are presented in Fig.3. Xe and Sm concentrations are assumed in equilibrium (state with partly inserted K6 rods). No boron acid is in the coolant. The multiplication cross section $\nu\Sigma_f$ are divided by the initial k_{eff} to obtain a critical state at the beginning. For the purpose of realisation of the isothermal re-criticality temperature T_{recrit} = 210°C an adaptation of neutronic data is necessary. A modification of the efficiency of the control rods by adopting the cross section data of absorber (e.g. Σ_r, Σ_a) is recommended. Other neutronic data will not be given.

Initial thermohydraulic conditions

The following thermohydraulic input data are given:

Primary circuit

Upper plenum pressure:	12.25 MPa
Core inlet temperature:	267.4°C
Core mass flow rate (including 3% bypass):	9300kg/s
(The core mass flow rate without the 3% bypass is entirely available for fuel cooling.)	
Pressurizer collapsed level (measured from the bottom):	5.97m

Secondary circuit

Pressure at SG outlet:	4.63MPa
Feed water temperature:	220°C
Feed water mass flow rate to single SG:	124.5kg/s
SG collapsed level (measured from the bottom):	2.015m

i: Scenario of the transient calculation

The initiating event is a double ended break in the main steam line of steam generator No.1. The connection of the defect loop to the core is presented in Fig.2. The beginning of the leak opening refers to the time t=0s. The asymmetric leak causes a different depressurization of all steam generators. The activation of the reactor scram is caused by the corresponding power level signal with the indicated time delay. The stuck rods belonging to group K3 (position 240) and K4 (position 125) are located in sector No.1 (see Fig. 3). A fully withdrawn position is assumed. Resulting from the scram, the turbines are turned-off by closing the turbine isolation valves. According to the steam header pressure signal, the steam isolation valves are closed and all SG are isolated from the feed water. The leak causes an overcooling of the primary circuit and the pressure decreases. The pressure and volume control systems are in operation and immediately will be switched on to correct the system pressure and the pressurizer level. It is postulated, that all main coolant pumps (MCP) remain in operation. The HPIS begins to provide highly borated water to the circuit with the indicated delay time after reaching the corresponding actuation points. The calculation should be continued until the highly borated water from the HPIS enters the core and terminates the power excursion. It is recommended to perform the calculation until at least 400s after the leak opening.

9. Hardware and Software Requirements:

For the calculation of this benchmark, an average work station is necessary. On such a computer, the computation time is about ten hours.

10. Output:

a: Requested Results

1. “Single key parameters”

- Isothermal temperature coefficient at T=210 ^{o}C and zero power after tuning (all control rods are fully inserted, except the indicated stuck rods), in pcm/K
- Total control rod worth of all control rods except the indicated stuck rods at hot zero power (the core inlet temperature is 260 °C) before and after tuning, in pcm

(The other relevant initial conditions for the determination of these two values should be taken from **8. h**)

- Total core power [MW]:
  - at the beginning of the transient t=0.0s
  - at the first power maximum
  - at the second power maximum
  - at 20 seconds after start of HPIS
- Total prompt fission power [MW]:
  - at the beginning of the transient t=0.0s
  - at the first power maximum
  - at the second power maximum
  - at 20 seconds after start of HPIS

- 3D power peak factor F_Q (with information about positions according to Fig.1 and core layer):

- at the beginning of the transient t=0.0s
- at the first power maximum
- at the second power maximum
- at 20 seconds after start of HPIS

2. Spatial nuclear power distribution (" Power distribution")

The normalized two-dimensional assembly-wise power distributions and the axial distribution generated by radial averaging of the 3D normalized power distribution (for each of ten equal core layers) are to be given. The nodes with absorber material belongs to the core volume in any normalization procedure. This normalization has to be applied also for the determination of the power peaking factors F_Q. The values of the assembly powers are to be provided according to the numbering used in Fig. 1 (row-wise from left to right and from bottom to top).

The power distributions are to be given at following times:

- t=0.0s, initial state
- time of first power maximum (before SCRAM)
- time of second power maximum
- at 20 seconds after start of HPIS

3. "Table of events" (switching-off/on, closing and opening of the different systems ...)

Time (s)	**Event**
0.0	Begin of leak opening
0.1	Leak is fully open
	...
x.x	SCRAM value reached
	...
x.x	End of calculation

4. "Time functions 1"

- Total nuclear power of the core [MW] (FPOW)
- Total prompt fission power of the core [MW] (PFPOW)
- Total power transferred to coolant [MW] (THPOW)
- Reactivity [pcm] (REAC)
- Upper plenum pressure, measured at the hot leg outlet elevation [MPa] (PUP)
- Pressurizer collapsed level measured from the pressurizer bottom [m] (CLPRZ)
- Maximum fuel pellet centerline temperature [^{o}C] (TFMAX)
- Averaged fuel temperature in the core [^{o}C] (TFAVE)

(The fuel temperature should be averaged over all nodes of the active core including the absorber parts of inserted control assemblies. In the absorber parts, the fuel temperature is assumed to be equal to the coolant temperature.)

- Primary circuit mass flow rate [kg/s] (MFPC)
- Pressure in the main steam header [MPa] (PMSH)

5. "Time functions 2"

- Cold leg outlet coolant temperature of loop 1 [^{o}C] (TCL1)
- Cold leg outlet coolant temperature of loop 2 [^{o}C] (TCL2)
- Cold leg outlet coolant temperature of loop 3 [^{o}C] (TCL3)
- Cold leg outlet coolant temperature of loop 4 [^{o}C] (TCL4)
- Cold leg outlet coolant temperature of loop 5 [^{o}C] (TCL5)
- Cold leg outlet coolant temperature of loop 6 [^{o}C] (TCL6)
- Averaged core inlet coolant temperature of sector 1 [^{o}C] (TIN1)
- Averaged core inlet coolant temperature of sector 2 [^{o}C] (TIN2)
- Averaged core inlet coolant temperature of sector 3 [^{o}C] (TIN3)
- Averaged core inlet coolant temperature of sector 4 [^{o}C] (TIN4)
- Averaged core inlet coolant temperature of sector 5 [^{o}C] (TIN5)
- Averaged core inlet coolant temperature of sector 6 [^{o}C] (TIN6)

6. “Time functions 3”

- Averaged core inlet boron concentration of sector 1 [g/kg] (CBIN1)
- Averaged core inlet boron concentration of sector 2 [g/kg] (CBIN2)
- Averaged core inlet boron concentration of sector 3 [g/kg] (CBIN3)
- Averaged core inlet boron concentration of sector 4 [g/kg] (CBIN4)
- Averaged core inlet boron concentration of sector 5 [g/kg] (CBIN5)
- Averaged core inlet boron concentration of sector 6 [g/kg] (CBIN6)
- Full HPIS mass flow rate [kg/s] (MFHPI)

7. “Time functions 4”

- Leak mass flow rate [kg/s] (MFLEA)
- Leak steam mass flow rate [kg/s] (MFST)
- Leak liquid mass flow rate [kg/s] (MFLI)
- Leak mass flow rate from SG-1 side [kg/s] (MFLEA1)
- Leak steam mass flow rate from SG-1 side [kg/s] (MFST1)
- Leak liquid mass flow rate from SG-1 side [kg/s] (MFLI1)
- Leak mass flow rate from MSH side [kg/s] (MFLEA2)
- Liquid mass flow rate at SG-1 outlet [kg/s] (MFLISG)
- Liquid mass flow rate in the MSL-1 at 20.0m after the SG-1 outlet [kg/s] (MFLIMSL)

8. “Time functions 5”

- Secondary side collapsed level in SG-1 [m] (CLSG1)
- Steam pressure at SG-1 outlet [MPa] (PSG1)
- Secondary side collapsed level averaged over all intact SG [m] (CLSG2)
- Steam pressure at SG outlet averaged over all intact SG [MPa] (PSG2)
- Total power transferred to secondary side in SG-1 [MW] (POWSG1)
- Total power transferred to secondary side averaged over all intact SG [MW] (POWSG2)
- Power transferred to secondary side in the U-tubes part 1 of SG-1[MW] (UT1)
- Power transferred to secondary side in the U-tubes part 2 of SG-1[MW] (UT2)
- Power transferred to secondary side in the U-tubes part 3 of SG-1[MW] (UT3)
- Power transferred to secondary side in the U-tubes part 4 of SG-1[MW] (UT4)
- Power transferred to secondary side in the U-tubes part 5 of SG-1[MW] (UT5)

9. “Time functions 6”

- Mass inventory of the broken SG [kg] (MASSBR)
- Turbine mass flow rate [kg/s] (MFTUR)

b: Files, Format

Each type of the described output data should be preceded by the keyword given in the heading, and each power distribution additionally by the time for the distribution. The time functions should be presented with a time resolution of at least 1s. It is recommended, to use a finer output during the power peaks. The functions UT1-UT5 require the presentation of the power transferred to the secondary side in one fifth of the U-tubes (division over the height) beginning with the lowest part. Therefore they are requested optionally, only if the nodalization of the SG allows such type of presentation.

The data arrays of all time functions should contain the time (in s) and the values of the requested quantities (in the given order) for successive time points. The first point t=0.0s corresponds to the leak opening. Each data array with time functions should contain a heading line with the keyword ”TIME” in the first column and the abbreviations for the provided quantities given above in the other columns.

All output should be given in one file.

11. References

[1] P. Siltanen: AER working group D an VVER safety analysis – Minutes of the meeting, Rossendorf, 10 – 12 May 1999

[2] S. Kliem: Requirements to a new dynamic AER benchmark, AER working group D meeting, Rossendorf, 10 – 12 May 1999

[3] S. Kliem, A. Seidel, U. Grundmann: “Definition of the 6th Dynamic AER Benchmark - Main Steam Line Break in a NPP with VVER-440“, Proc. 10th Symposium of AER, pp. 749-762, KFKI Atomic Energy Research Institute, Budapest (2000)

[4] S. Kliem: “Comparison of the updated solutions of the 6th Dynamic AER Benchmark - Main Steam Line Break in a NPP with VVER-440 “, Proc. 13th Symposium of AER, pp. 413-444, KFKI Atomic Energy Research Institute, Budapest (2003)

[5] S. Danilin, M. Lizorkin, S. Nikonov: “The new solution of the AER sixth benchmark problem with ATHLET/BIPR8KN code package “, Proc. 12th Symposium of AER, pp. 333-348, KFKI Atomic Energy Research Institute, Budapest (2002)

[6] A. Hämäläinen, R. Kyrki-Rajamäki: “HEXTRAN-SMABRE Calculations of the 6th Dynamic AER Benchmark - Main Steam Line Break in a VVER440 NPP“, Proc. 13th Symposium of AER, pp. 445-458, KFKI Atomic Energy Research Institute, Budapest (2003)

[7] J. Hadek, P. Kral: “Final Results of the 6th Three-Dimensional AER Dynamic Benchmark Problem Calculation. Solution with DYN3D and RELAP5-3D Codes”, Proc. 13th Symposium of AER, pp. 459-488, KFKI Atomic Energy Research Institute, Budapest (2003)

[8] A. Seidel, S. Kliem: “ Solution of the 6th Dynamic AER Benchmark using the coupled code DYN3D/ATHLET“,Proc. 11th Symposium of AER, pp. 251-267, KFKI Atomic Energy Research Institute, Budapest (2001)

12. Recommended Solution:

There is no unique reference solution to this test. It is a pure comparison of solutions by different codes and data libraries.

13. Summary of Available Solutions:

Results are available from the following organizations:

Tab 10: Participants of the calculations

Organization	Code	Reference
RRC Kurchatov Institute Moscow (Russia)	BIPR8/ATHLET	[5]
VTT Processes Espoo (Finland)	HEXTRAN/SMABRE	[6]
Nuclear Research Institute Rez (Czech Republic)	RELAP5-3D	[7]
KFKI AEKI Budapest (Hungary)	KIKO3D/ATHLET	[4]
Forschungszentrum Rossendorf (Germany)	DYN3D/ATHLET	[8]

In this section, an overview of the main results of the benchmark is given. The whole comparison can be found in [4].

Fig. 4 and 5 show the time course of the total core power. After leak opening, the core power increases in all solutions. The power level of 110 %, necessary for the activation of the reactor scram is reached at different time points in the various solutions (see Tab. 11). First of all, the scram value is reached in the VTT-calculation, followed by the AEKI and the NRI-calculations within less than 3 s. The scram value in the FZR-calculation is reached about 5 s after the NRI-calculation. More than 20 s later, the scram value is reached in the KI-calculation.

The power increase after leak opening is determined by two factors, by the amount of overcooling and by the moderator temperature coefficient. The average core inlet temperature is shown in Fig. 6. The decrease of this average temperature from leak opening until reaching the scram value is in all calculations nearly the same. The different isothermal moderator coefficients, determined for a coolant temperature of 210 °C and inserted control rods, seem not to have a big influence onto the power behaviour until the reactor scram. That means, that the spreading of the time of reaching the scram value is caused by differences in the rate of overcooling. It can be clearly concluded from Fig. 6, that the higher the overcooling rate the faster the scram value is reached. The time delay for reaching the scram value in the KI-calculation is connected with the fact, that shortly after leak opening, the overcooling is stopped for a certain time, the average core inlet temperature rises again. Only after some 20 s the overcooling continues and the scram value is reached, what causes the reactor scram. The reason for such a different from the other calculations behaviour is in the secondary circuit, namely in the behaviour of the intact steam generators.

Fig. 7 shows the heat transfer summarized over all six steam generators. The higher the heat transfer from primary to secondary side, the higher is the overcooling. Three solutions show more or less the same behaviour, the VTT-calculation yields a higher level of about 20 %.The higher heat transfer in the VTT-calculation corresponds with the temperature curve (Fig. 6) and is also responsible for the fastest upper plenum pressure decrease (Fig. 8). In all mentioned calculations, the heat transfer rises until the scram. In the KI-calculation, already at t = 5s, the heat transfer starts to decrease.

The leak mass flow rate is shown in Fig. 9. The explanation for a lower mass flow rate in the VTT-calculation can be seen in Figs. 10 and 11, where the leak steam and liquid mass flow rate are shown. Nearly no liquid is going through the leak in the VTT-calculation, while the other four calculations show a significant liquid mass flow rate through the leak until t = 30 s. A very high amount of liquid is ejected in the KI-calculation, what could be the reason for the different pressure behaviour.

Tab. 11: Table of events (time in s)

Event	FZR	VTT	AEKI	NRI	KI
Leak opening	0.0	0.0	0.0	0.0	0.0
Start 1st make-up pump	13.4	3.1	5.6	7.5	
SCRAM value reached	19.5	11.1	13.9	14.2	39.9
PRZ Heater group 1 on	20.3	5.8	7.6		28.2
PRZ Heater group 2 on	24.5	7.3	13.9	9.0	30.7
PRZ Heater group 3 on	27.0	8.8	17.3	16.0	33.8
PRZ Heater group 4 on	31.5	14.0	22.4	22.5	47.6
PRZ Heater groups off	51.5	5.27	51.3	66.5	
PRZ-Level <2.41m	54.1	55.9		69.0	71.3
HPIS Signal	54.1	55.9	54.2	69.5	71.3
Start 2nd make-up pump	53.6		45.6	47.5	
Pressure in MSH <3.0MPa	89.0	55.9	64.4	99.5	98.7
Begin of HPIS supply	234.1	235.9	234.2	249.5	251.3
Second power maximum	272.0	253.0	225.8	234.5	256.5
End of calculation	400.0	398.0	400.0	1000.0	400.0

Tab. 12 contains the requested key parameters of the different calculations.

Tab. 12: Comparison of key parameters

Parameter		**FZR**	**VTT**	**AEKI**	**NRI**	**KI**
Total core power [MW]	T=0.0s	1351.0	1376.8	1374.4	1375.0	1375.0
	1st power maximum	1521.4	1516.2	1503.5	1518.6	1532.3
	2nd power maximum	43.5	58.3	130.3	209.8	87.1
Total fission power [MW]	T=0.0s	1265.0	1283.3	1278.5	1284.8	1278.5
	1st power maximum	1433.0	1421.2	1405.8	1426.9	1436.5
	2nd power maximum	10.3	18.5	98.7	163.1	48.7
3D power peak factor (FA;Layer)	T=0.0s	1.5164 (28;4)	1.5302 (36)	1.639 (183;3)	-	1.534 (28;3)
	1st power maximum	1.5680 (183;4)	1.5567 (183)	1.683 (183;3)	-	2.683 (183;3)
	2nd power maximum	10.6746 (222;10)	12.1425 (222)	11.947 (144;9)	-	10.194 (144;9)
Isothermal MTC	pcm/K	-35.1	-29.3	-29.8	-30.1	-41.3
CR worth before tuning	pcm	6583	5670	5397	5250	6612
CR worth after tuning	pcm	4228	3571	3571	3543	4155

The reactor scram stops the pressure decrease in the intact steam generators in all calculations. After a short increase, the pressure drops until the isolation of the intact steam generators. The level in the intact steam generators remains nearly constant. That is due to the working level control system in these steam generators. The level in the broken steam generator (Fig. 12) shows nearly the same behaviour in all calculations (considering the “Riser”-value of the VTT-calculation). The overfeeding during a short time interval before the steam generator isolation is reflected in a level increase during this time in all calculations.

When the main steam header pressure (Fig. 13) decreases down to a value of 3.0 MPa, the main steam isolation valves and the feed water valves are closed. In such a way, five steam generators will be fully isolated. As can be seen from Fig. 13, the main steam header pressure decreases during the first phase of the transient in all calculations. After scram, the pressure increases for a short time period, in the KI-calculation already before the reactor scram. Later, the pressure decreases nearly with the same gradient in all calculations. Due to the differences in the maximum value reached after scram, a considerable spreading in the time of reaching the set point of steam generator isolation (3.0 MPa) is observed. According to Tab. 11, this spreading is from t = 55.9 s (VTT) to t = 99.5 s (KI). The isolation of the steam generators stops the heat transfer in the intact steam generators.

After the isolation of the intact steam generators, the overcooling continues only in the steam generator of the broken line. The further decrease of the cold leg temperature of the intact loops is connected only with the coolant mixing in the lower and upper plenum. The decreasing level and inventory of the broken steam generator lead to an aggravation of the heat transfer conditions. At a certain level, the heat transfer tubes are no more covered by water and the heat transfer is almost fully stopped. This is the case at about t = 180 s in the NRI- and AEKI-calculations. The others follow later. During that time, the core power begins to rise in all calculations (Figs. 4 and 5). The power increase in the AEKI- and the NRI-calculations starts earlier. In these calculations maximum values of more than 100 MW are reached (130.3 MW - AEKI and 209.8 MW - NRI). In the remaining three calculations, the maximum values are lower (FZR – 43.5 MW, KI – 87.1 MW and VTT – 58.3 MW).

The maximum fuel temperature in the second power peak (Fig. 14) is the highest in the AEKI-calculation, although the highest secondary power peak is observed in the NRI-calculation. The reduced number of thermal hydraulic core channels, used in the NRI-calculations is mainly responsible for smoothing the influence of this effect onto the maximum fuel temperature.

Fig. 15 shows the normalized axial power distribution (radially averaged) at the moment of first power maximum. All calculations show a typical full power distribution with a maximum in the lower part of the core. The agreement between the four provided solutions is good. After the scram and during the overcooling, a redistribution of the core power can be observed in all calculations. The maximum of the core power in the moment of second power maximum (Fig. 16) moves from the lower to the upper part of the core in all calculations. At this time point, the differences between the single calculations are higher. In the VTT-, FZR- and AEKI-calculations, the maximum of the normalized power distribution is nearly 1.5 or higher. In the KI-calculation, the power distribution is more flat, the maximum value is only about 1.2.

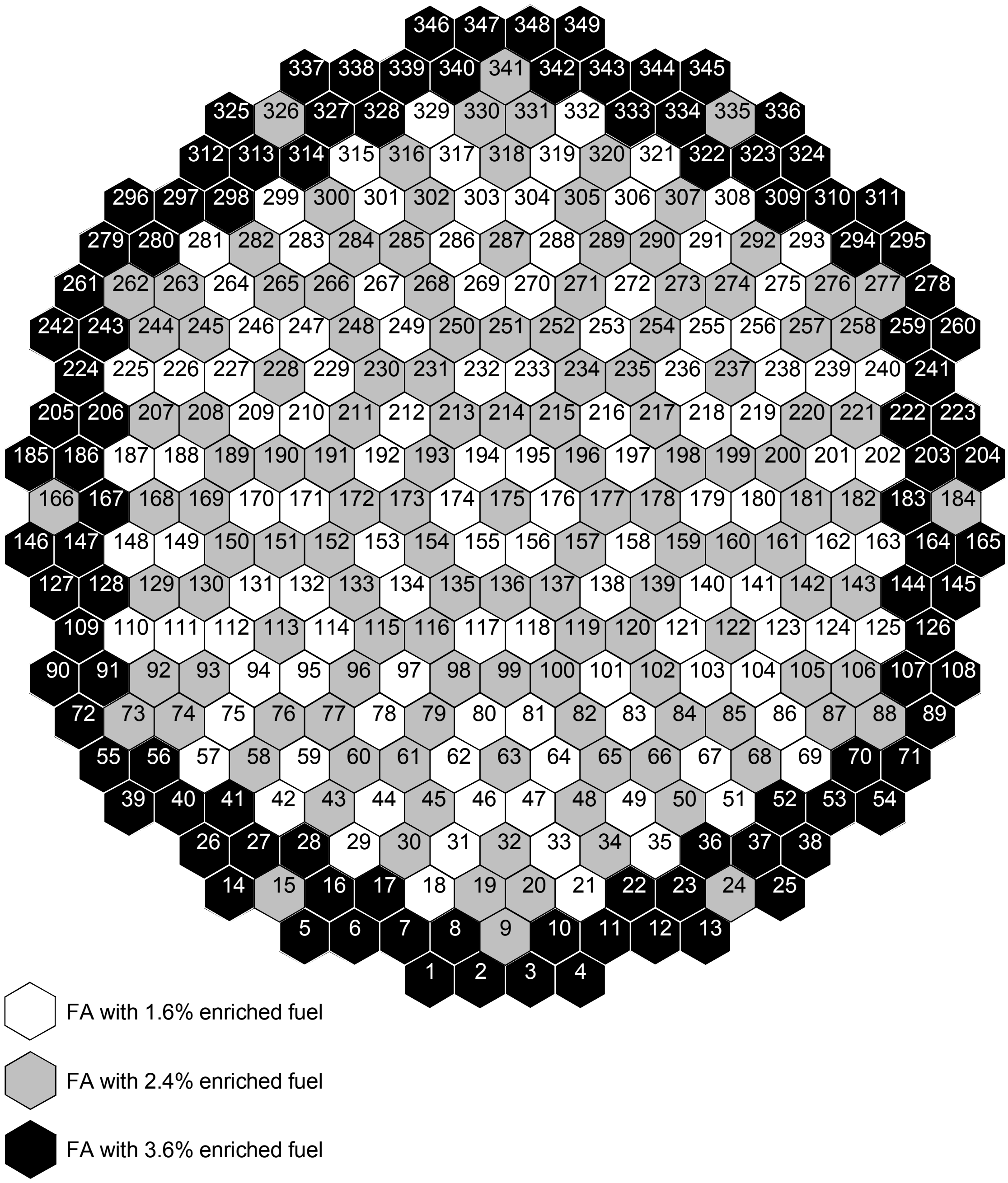

Fig. 1 Horizontal map of the VVER-440 reactor core for this benchmark.

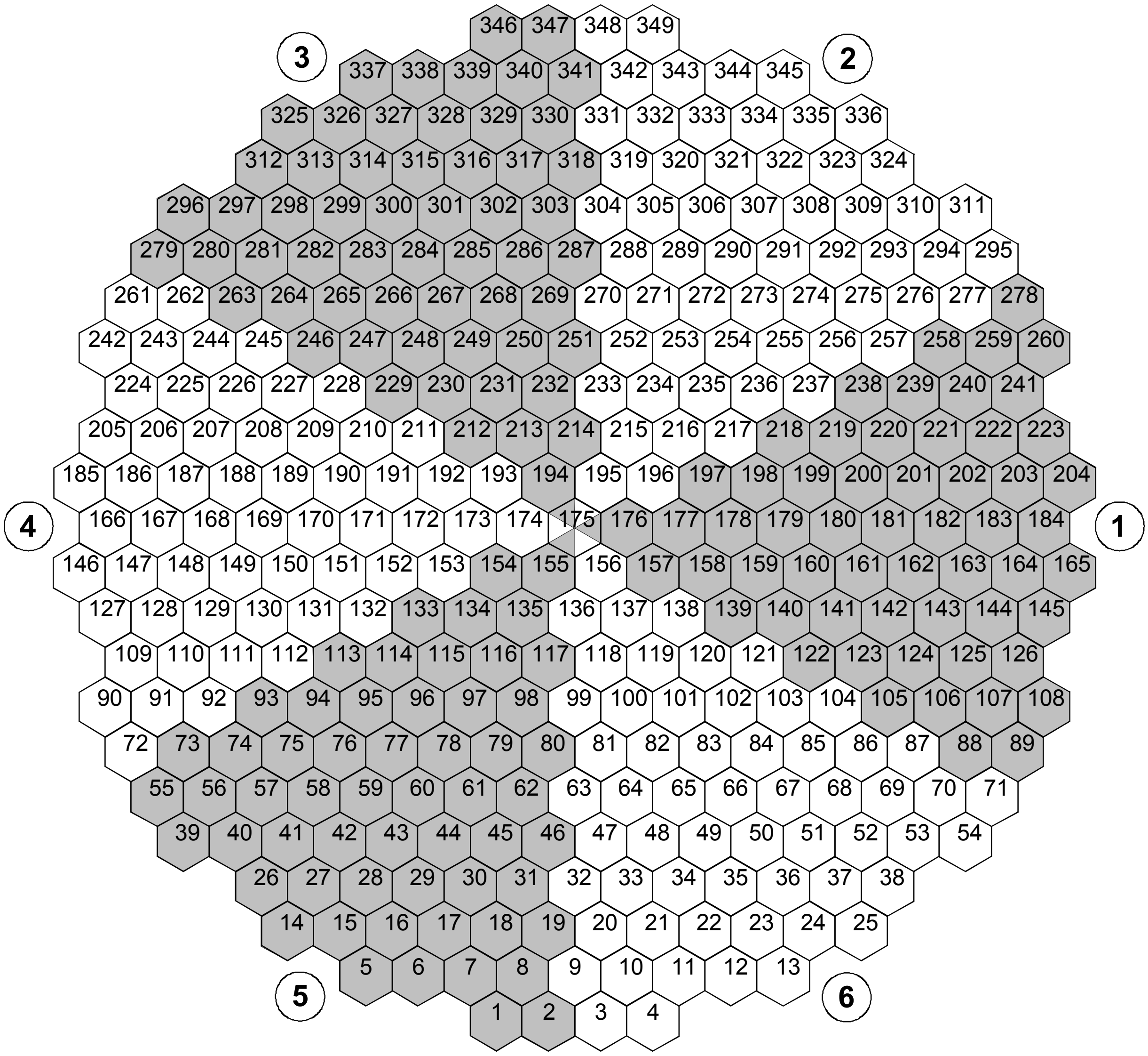

Fig. 2 Allocation of FA to sectors of VVER-440 reactor core

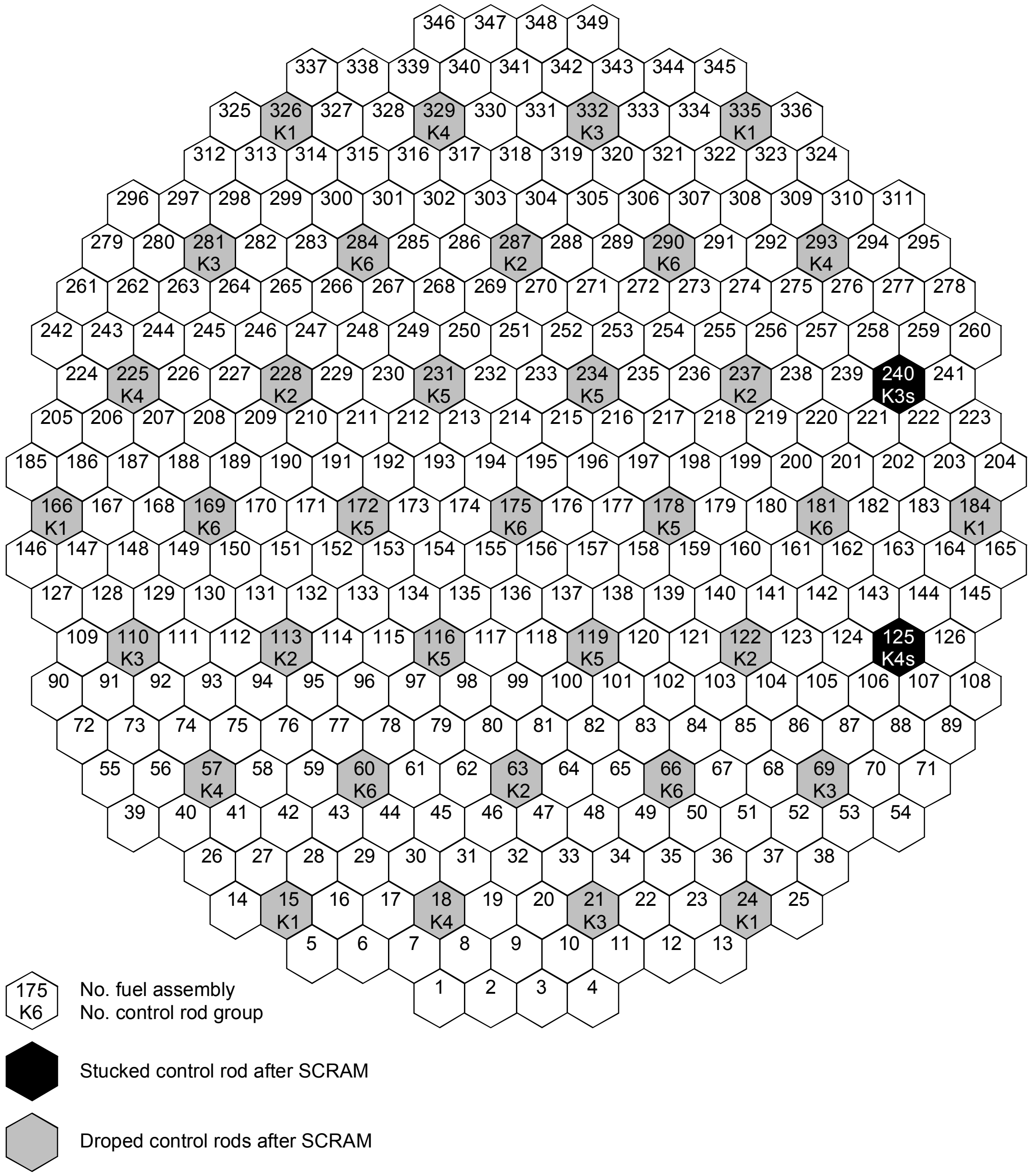

Fig. 3 Location of control rod groups in VVER-440 reactor core

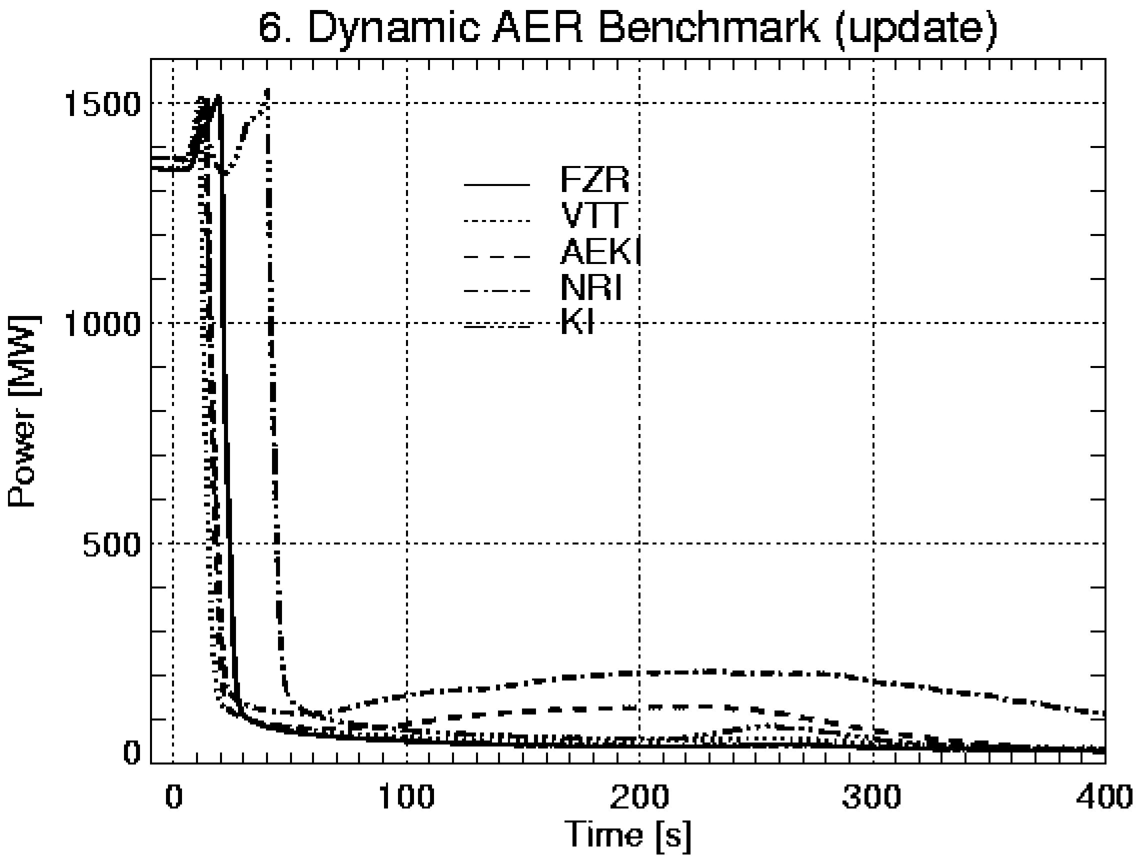

Fig. 4 Total core power

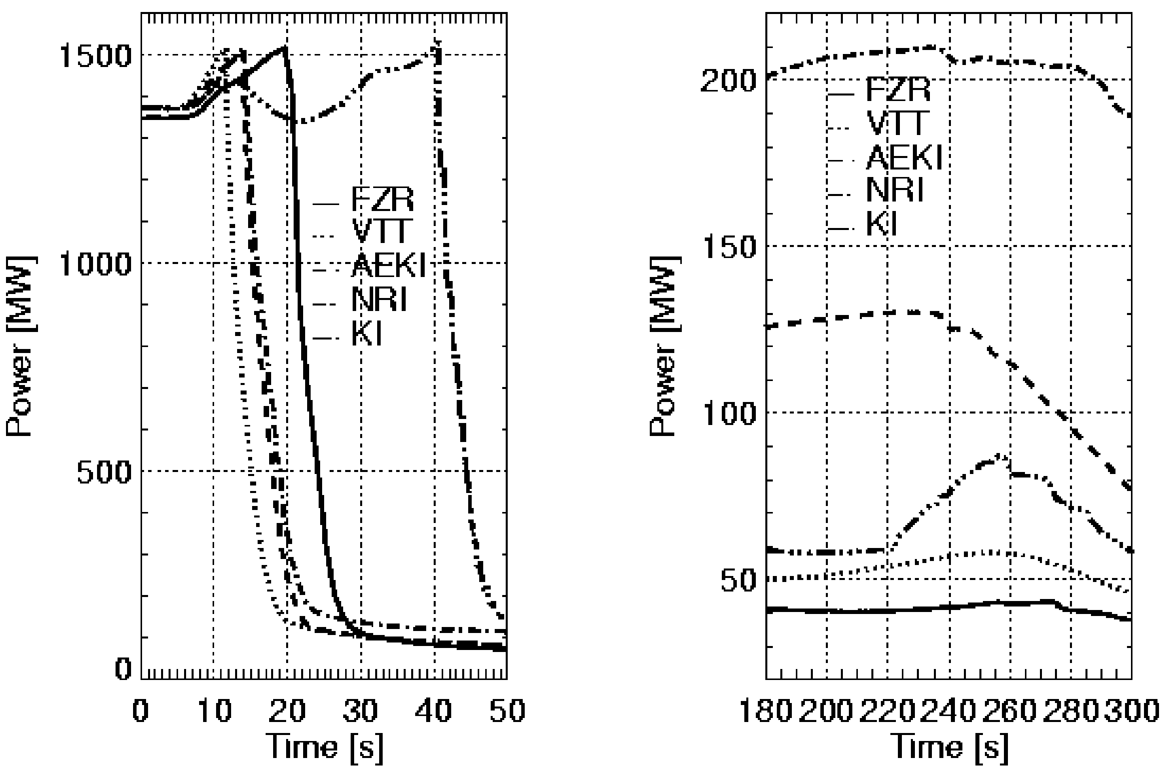

Fig. 5 Total core power (zoom)

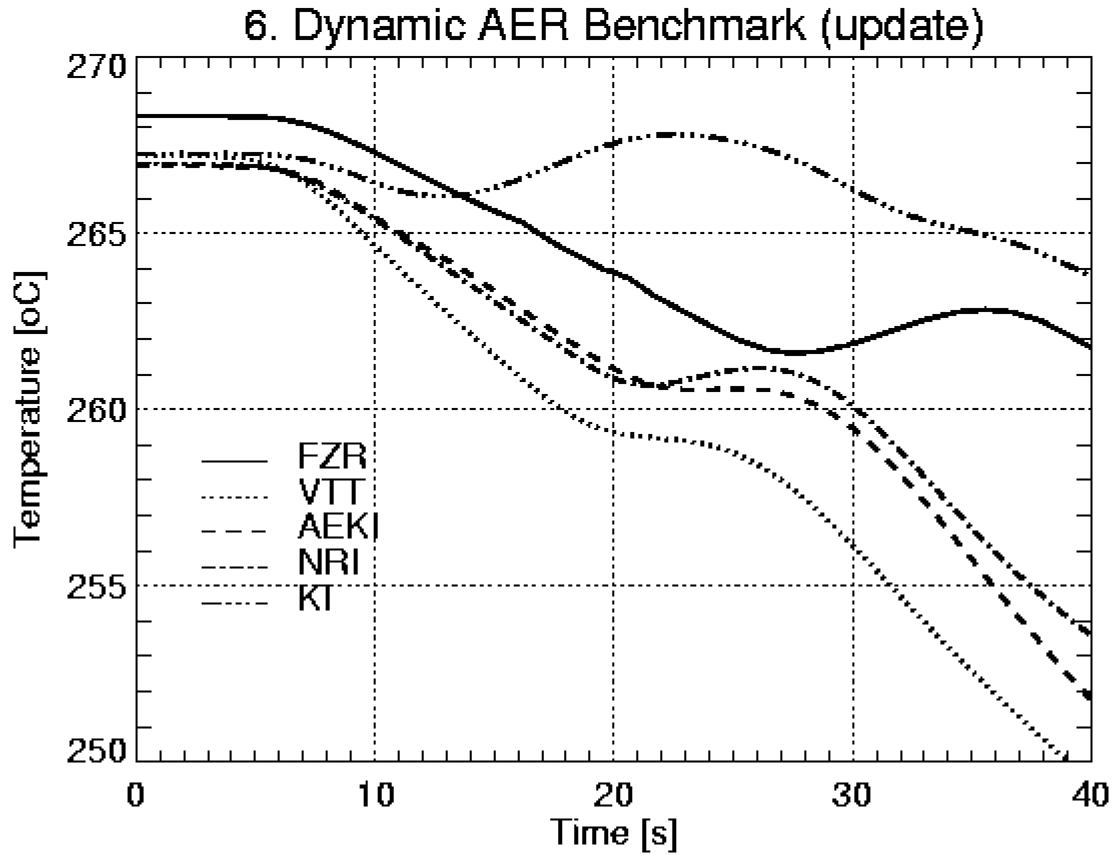

Fig. 6 Average core inlet temperature

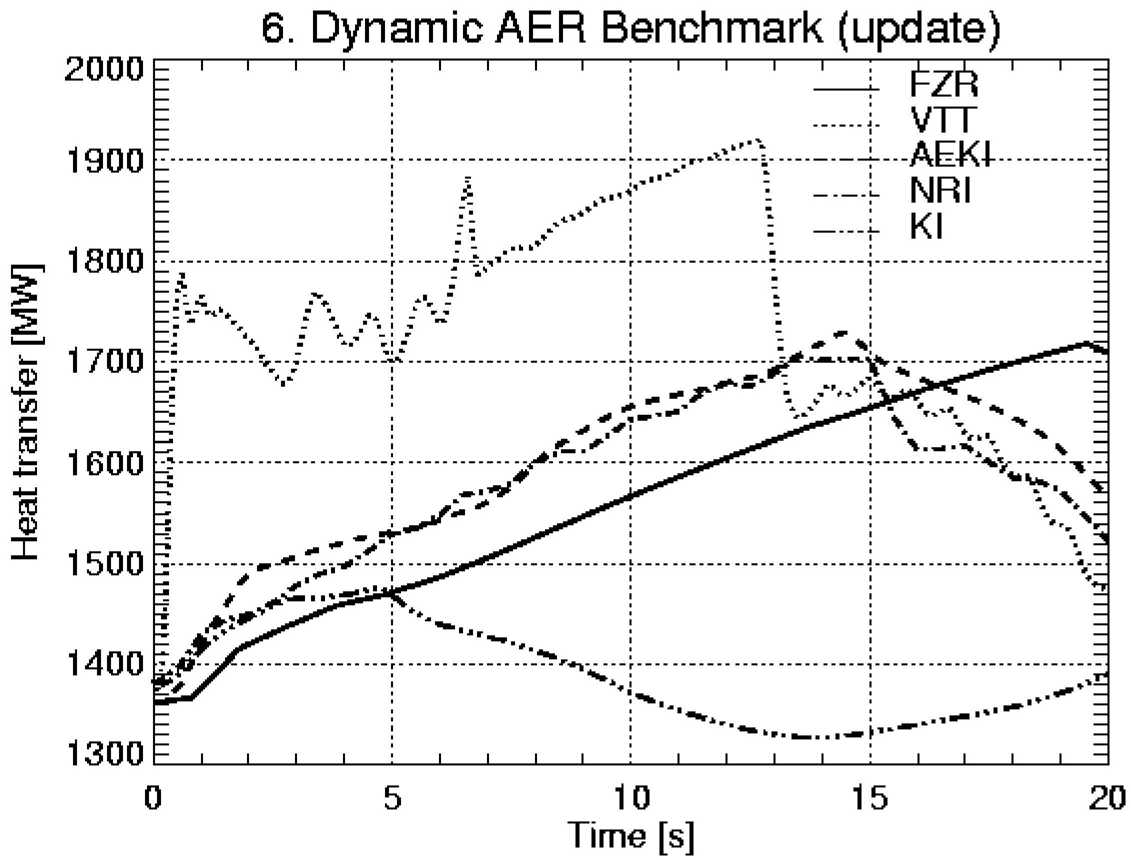

Fig. 7 Summary heat transfer in all steam generators

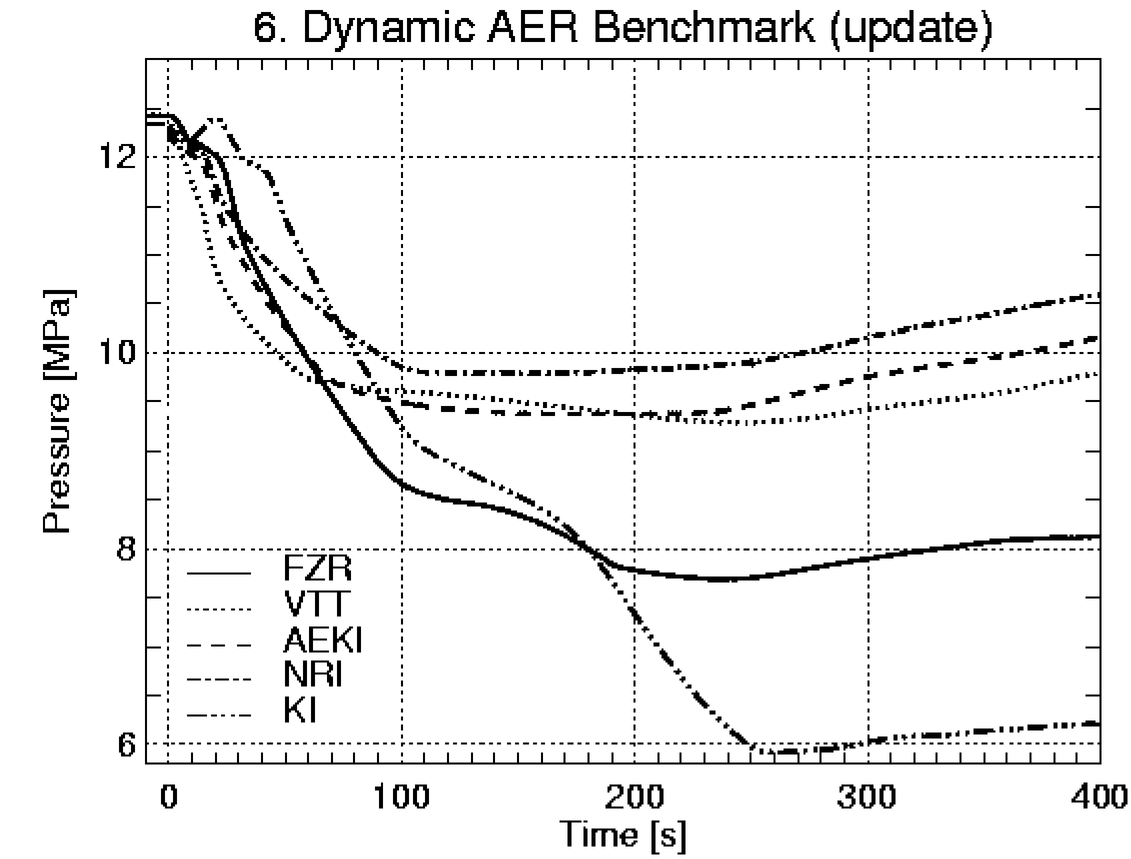

Fig. 8 Upper plenum pressure

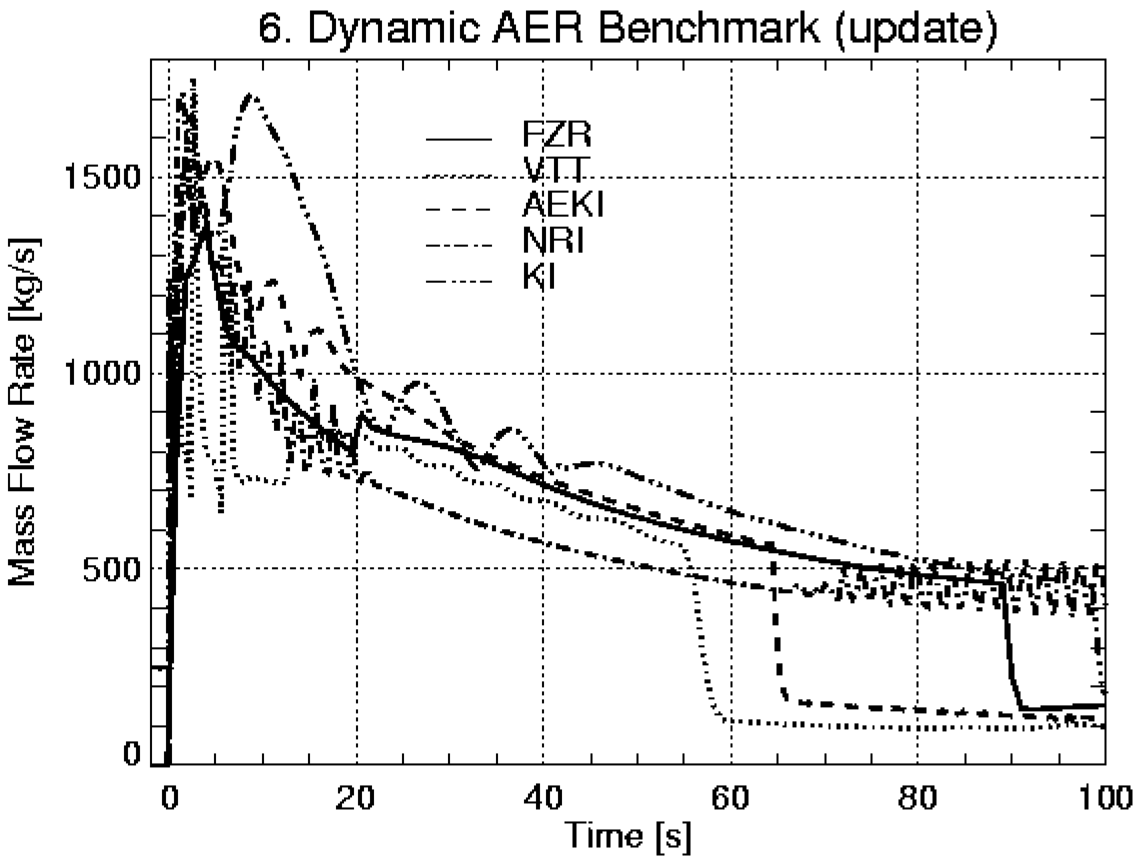

Fig. 9 Leak mass flow rate (zoom)

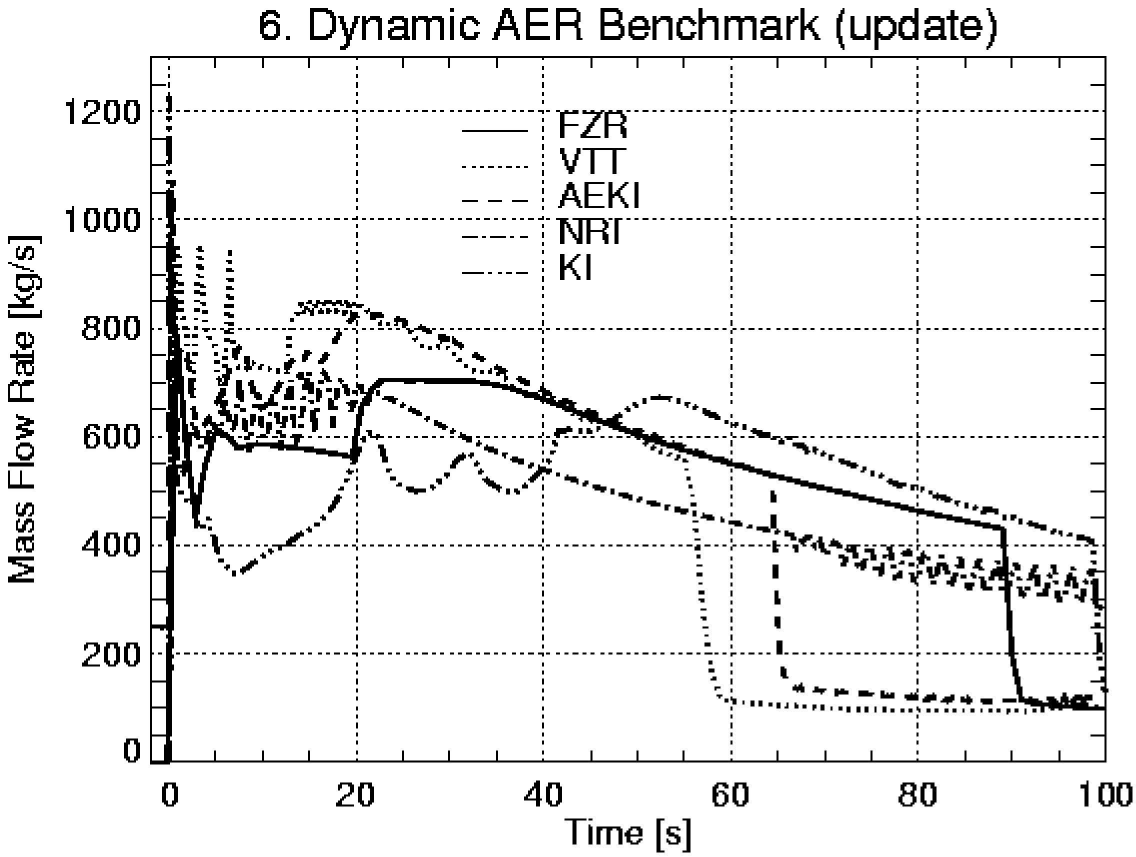

Fig. 10 Leak steam mass flow rate

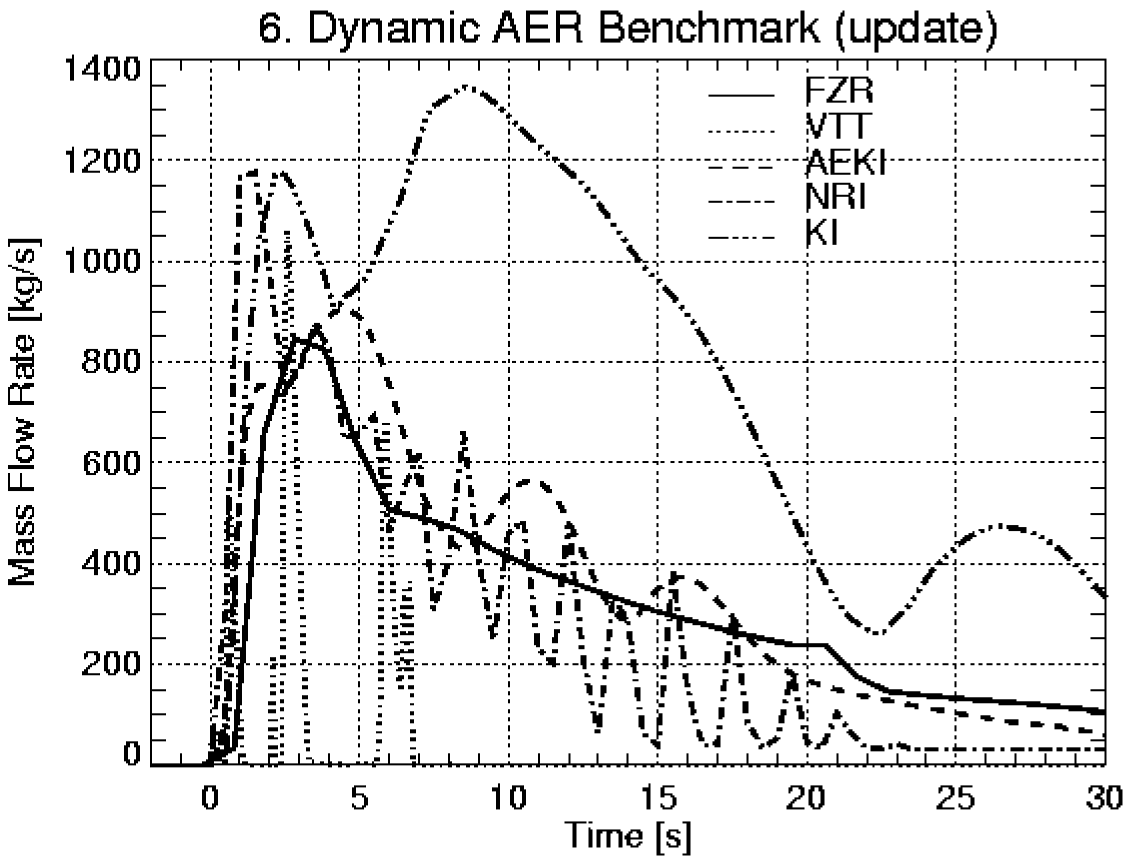

Fig. 11 Leak liquid mass flow rate

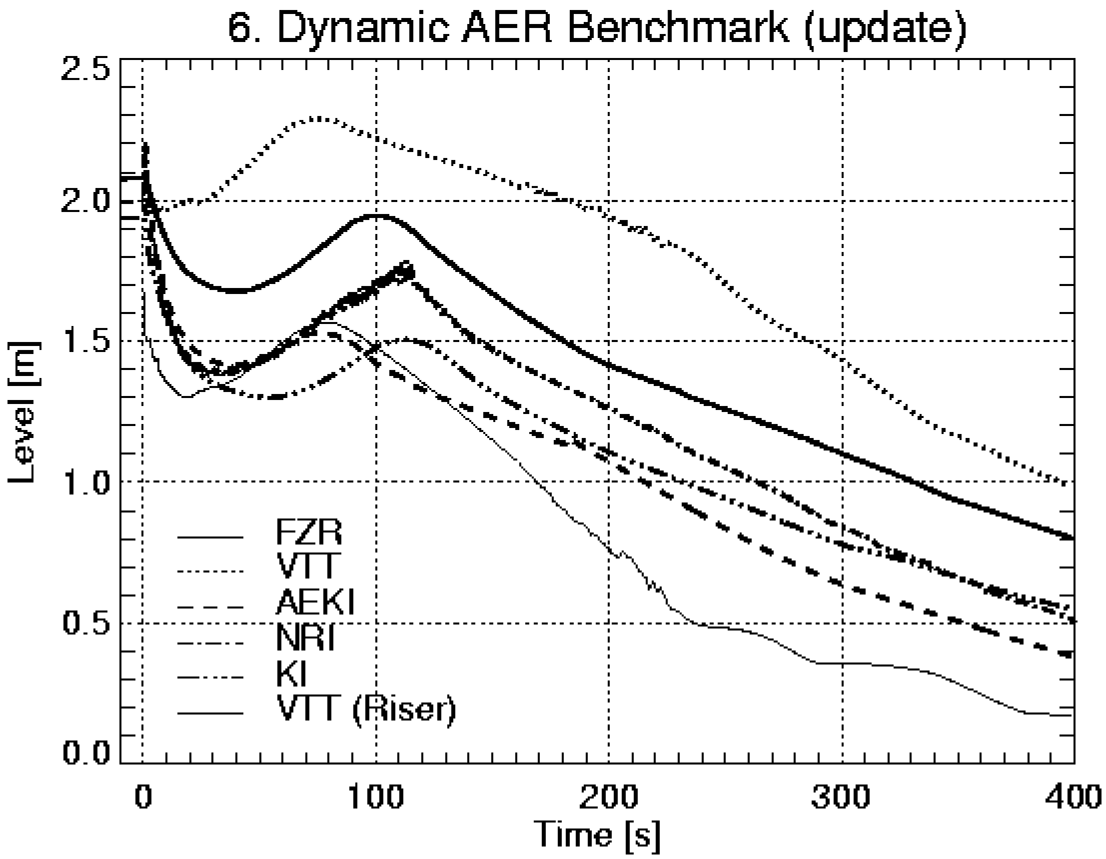

Fig. 12 Collapsed level in steam generator 1 (broken line)

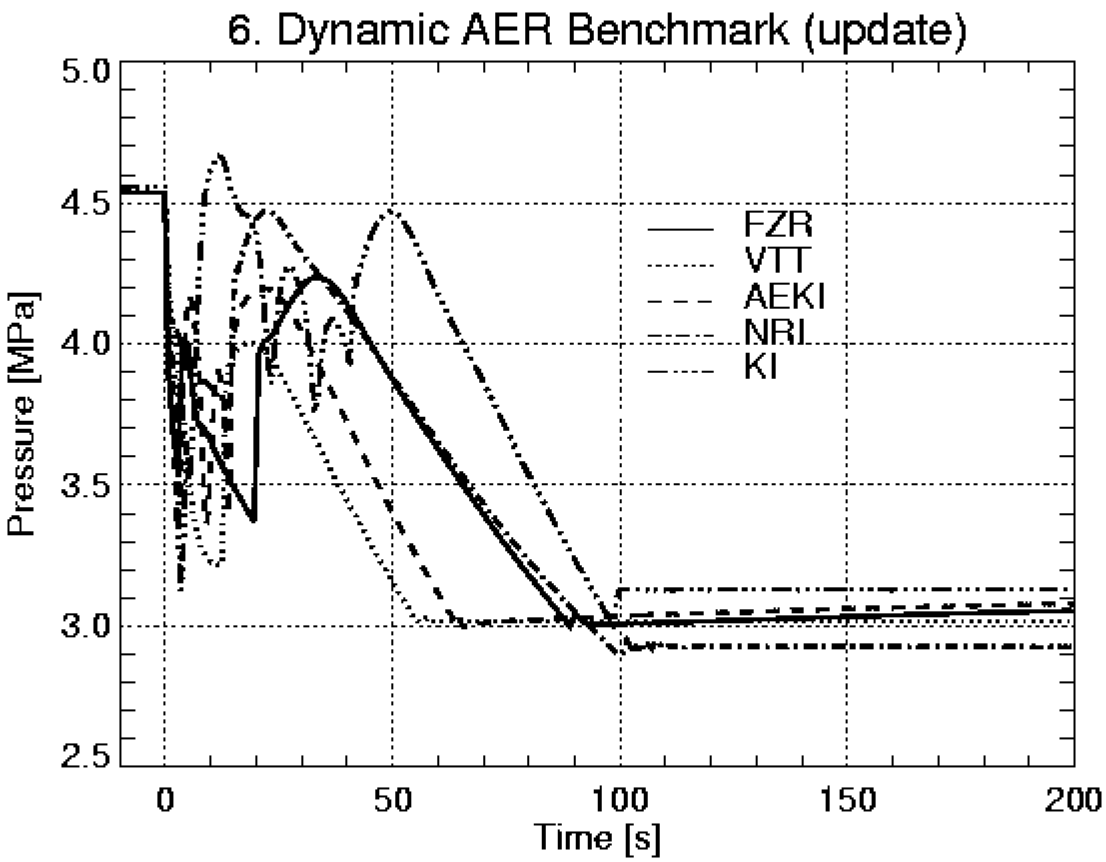

Fig. 13 Pressure in main steam header

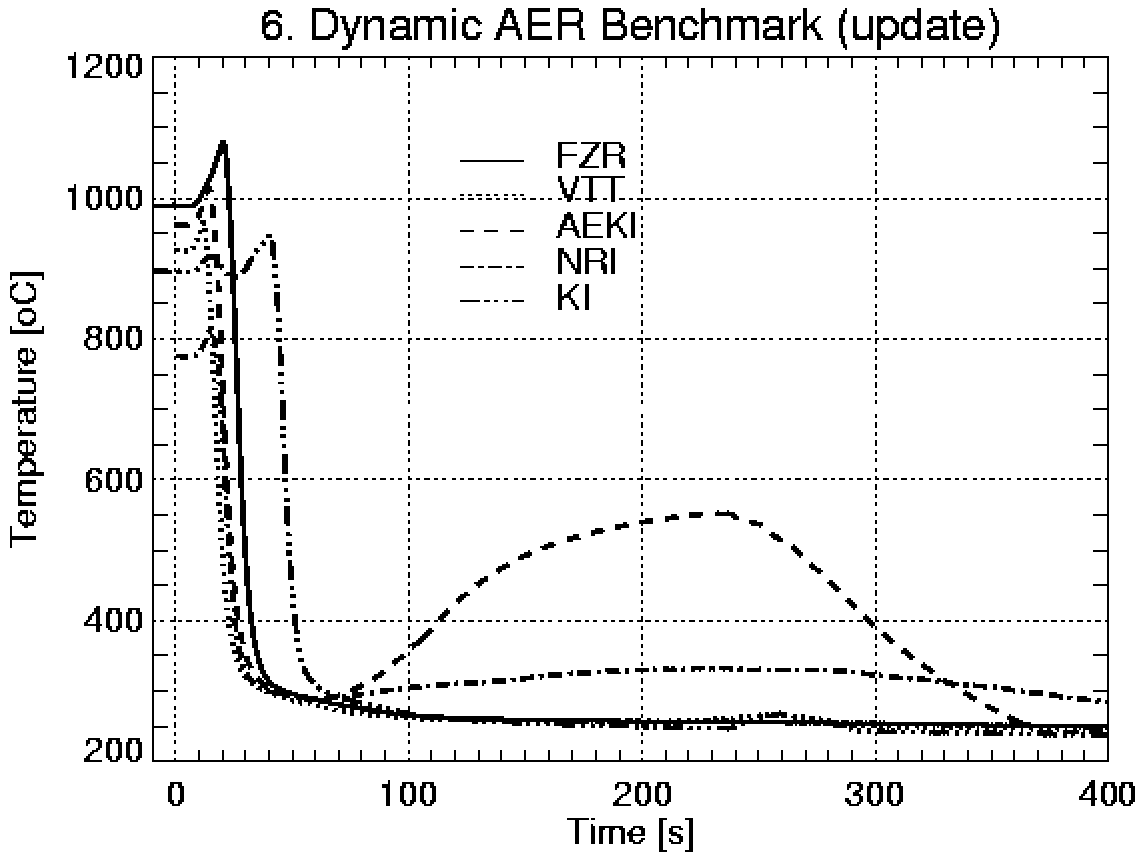

Fig. 14 Maximum fuel temperature

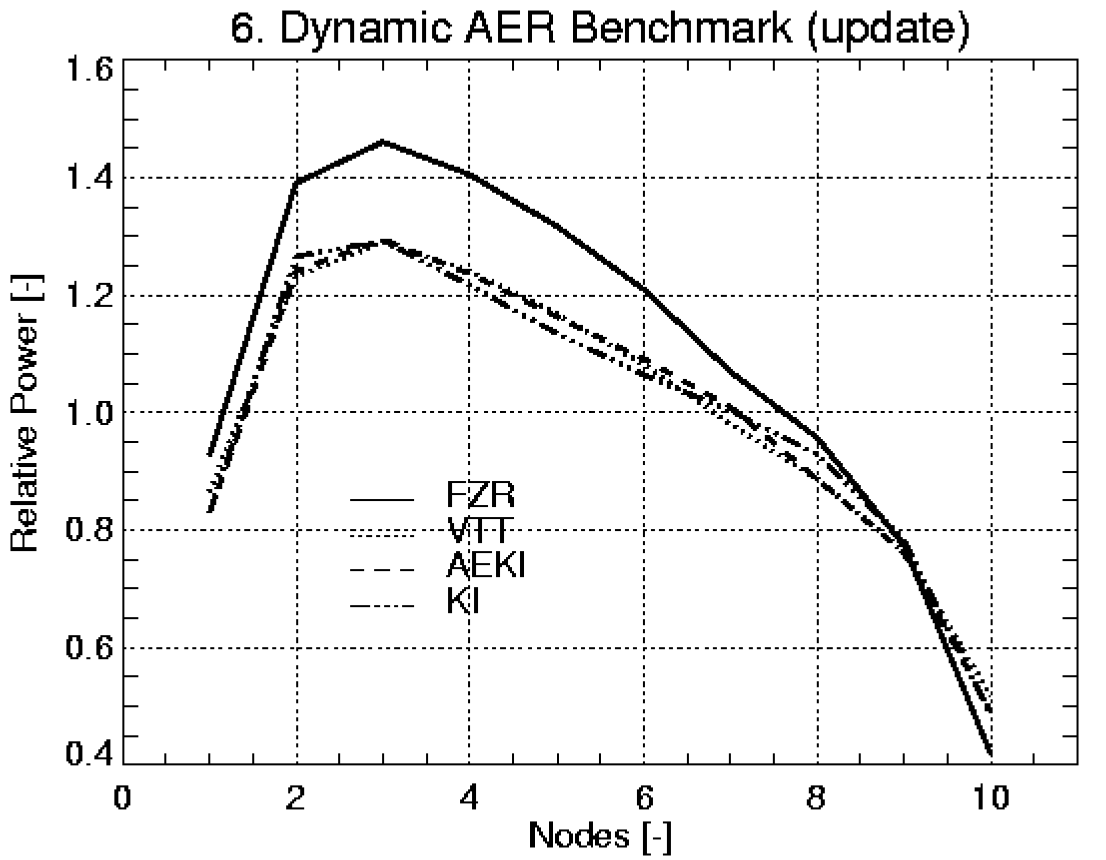

Fig. 15 Axial power distribution at the time of first power maximum

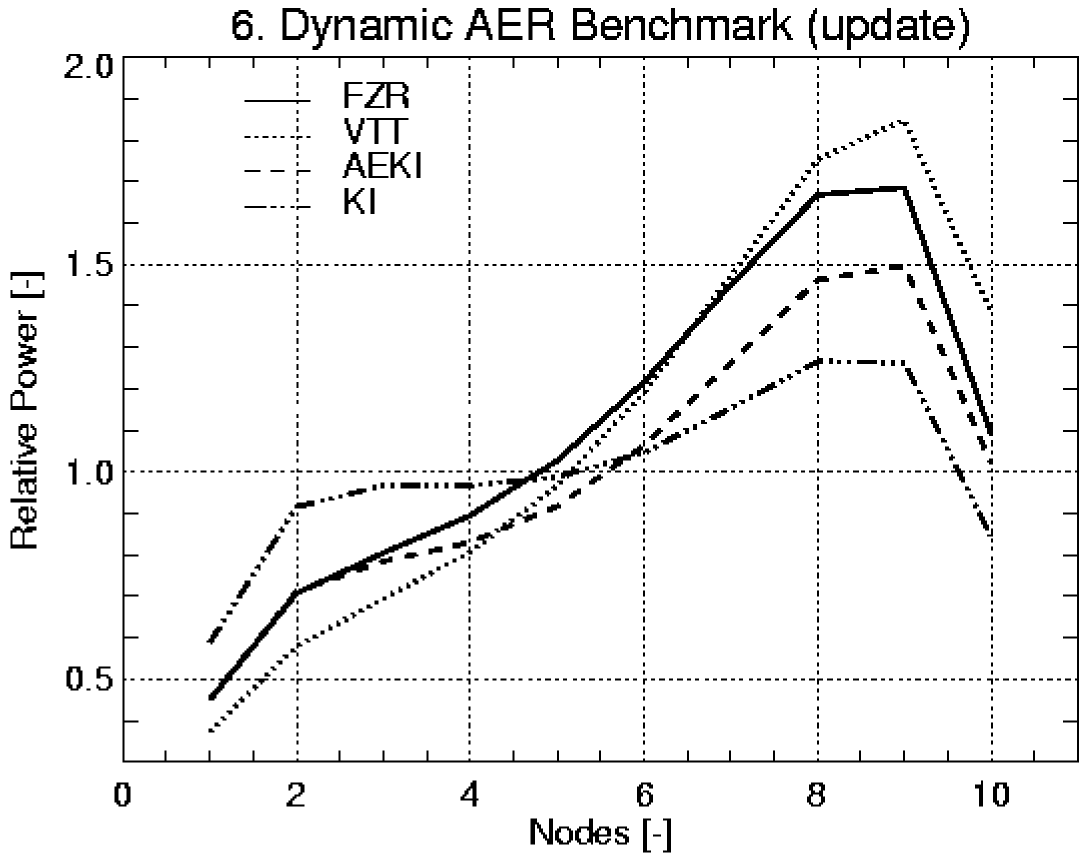

Fig. 16 Axial power distribution at the time of second power maximum