2001 International RELAP5 Users Seminar
Sun Valley, Idaho, USA September 5-7 2001.

Experience with the System Code RELAP5-3D© at NRI Rez.

Input Data Preparation for the 3D Neutronic Model VVER-440/213 of System Code RELAP5-3D© and Its Testing.

Jan HÁDEK
Pavel KRÁL
Jiří MACEK

Nuclear Research Institute Rez, Czech Republic

RELAP5

♥ **RELAP5 – at Nuclear Research Institute (NRI, ÚJV) Rez has been tested on data from experimental facilities - PMK-NVH (Hungary), PACTEL (Finland), RVS-3 (NRI Rez – Czech Republic), and on experimental data from transients and tests performed at NPP Dukovany. The process of RELAP5 testing goes on at NRI Rez .**

RELAP5-3D©

♥ **RELAP5-3D© is the latest in the series of RELAP5 codes** [1]**.**

♥ **The RELAP5-3D© version contains several important enhancements for us over previous versions of the code.**

♥ **The most important feature that distinguishes the RELAP5-3D© code from the previous versions is the fully integrated, multidimensional thermal-hydraulic and neutron kinetic modeling capacity.**

Input model of NPP of VVER-440/213

♥ **The RELAP5-3D© was tested on the input deck of Nuclear Power Plant (NPP) Dukovany with VVER-440/213 reactor (the 1st unit). The VVER-440/213 reactor is water moderated and water cooled reactor of Soviet origin with hexagonal fuel assemblies. Our model of the reactor core consists from 349 fuel assemblies with enrichment 1,6 ÷ 4,4 % of U235.**

♥ **The input deck originates in an „ international “ input deck of NPP Jaslovské Bohunice prepared under IAEA regional program „Safety Assessment of VVER-440/213 “ [3] . This model was further enlarged and modified according to the specification of NPP Dukovany.**

♥ **6 loops model of NPP Dukovany with VVER-440/213. It includes both all realized and expected modifications of NPP Dukovany systems [4]. The nodalization scheme of the input model is presented in Fig. 1 (primary circuit and reactor), Fig. 2 (primary circuit and steam generators) and Fig. 3 (main steam system).**

♥ **Main characteristics of input model:**

Main characteristics of nodalization:

- **6 modeled loops**
- **core region can be modeled with multi-channel hydraulic representation**
- **reactor upper plenum (UP, HSK) and downcomer (DC, SŠR) can be modeled in „ multichannel way “**
- **detailed steam generator (SG, PG) nodalization**
- **detailed description of steam lines (MSL) and main steam headers (MSH, HPK)**
- **detailed description of feed water (FW, NV) system**

Main modeled operational control systems:

- **Reactor power control system (ARM-5S)**
- **Reactor power limiting system (ROM)**
- **Pressurizer (PRZ, KO) heaters and spraying**
- **Primary make-up (TK) and let-down (TE)**
- **Turbine control system (TVER)**
- **Steam bypass valves to main condenser (BRU-C, PSK)**

Main modelled safety systems:

- **Reactor trip signals AZ-1, 2, 3 , 4**
- **Engineered Safety Features Actuation System (ESFAS) signals:**
  “ Small Break ”
  “ Intermediate Break ”
  “ Large Break ”
  “ MSH Rupture ”
  “ Loss of Feed water ”
- **All Technological Protection of Steam Generator (TPSG) signals:**
  “MSL Rupture”
  “MSH Rupture”
- **All Local Protection of Steam Generator (LPSG) signals:**
  High SG level “ + 75 mm ”
  High SG level “ + 100 mm ”
  Low SG level “ - 140 mm ”
- **safety valves at PRZ and SGs**
- **atmospheric dump valves (BRU-A, PSA)**
- **Emergency Core Cooling System (ECCS, SAOZ)**
  High Pressure Injection System (HPIS, VTČ)
  Accumulators (ACCs, TH)
  Low Pressure Injection System (LPIS,NTČ)

♥ **Contemporary size of RELAP5-3D© input file is approximately 2,4 MB.**

♥ **Calculations are performed on workstation Hewlett Packard HP J2240 (2 processors, operational system HP-UX 1020, 1 GB RAM).**

Nodalization of VVER/440-213 core and reactor vessel

- **The reactor core configuration at 60 degree rotational symmetry (Fig. 4.) The core was divided into 12 axial layers (2 layers for unheated lower and upper part, 10 layers for heated core part with total length of 244 cm). The both unheated parts were treated as axial reflectors. The core was surrounded with a hypothetical radial band of reflector cassettes (Fig. 5)**

- **1, 7 and 31 channels model of VVER-440/213 core (Fig. 6 – 8).**

- **1-D reactor down comer nodalization model was used for first calculations. It means that coolant is fully homogenized at the inlet to lower plenum. No models were applied for the description of mixing in the lower and upper plenum.**

Cross section library

- **3 different neutron cross section models built in the code RELAP5-3D©: RAMONA , HWR and GEN models.**

- **USER option**

The user creates his own procedure which computes the set of cross sections for a single node given by the material type in the node determined from the composition maps, the region average properties in the node specified in the zone maps for node, and the control fractions and insertion directions for any control rods associated with the node. The user may also specify the value of up to four additional variables in each node that may affect the neutron cross sections (fuel burnup, Xe and Sm concentration. This subroutine is further linked with the RELAP5-3D© code . The exchange of parameters between user procedure (subroutine userxs.f) and RELAP5-3D© reactor core model can be seen in Fig. 9.

The two-group macroscopic cross section library CSLIBR (see Fig. 9) for homogenized fuel assemblies, control rods and reflectors as well as feedback coefficients and reactor kinetics parameters can be generated from KASSETA or HELIOS neutronic libraries [5], [6] with using of subprogram DYLIE [7]. The VVER-440 control rod model consists usually from four parts - fuel follower (234 cm), steel pellets (10 cm), coupler (30 cm) and absorber part (270 cm). In our first calculations the

control rod model consisted only from fuel follower and absorber part was considered. The volumetric portions of moderator and steel in reflector and control rod assemblies were obtained from the technologic documentation in order to generate corresponding neutronic cross sections from the KASSETA library.

Transient calculations

♥ Shutdown of main coolant pumps (MCPs)

- **shutdown of 1 from 6 MCPs at 100 % of nominal power (P_{nom}) (Fig. 10 ÷ 11)**
- **shutdown of all 6 MCPs at 100 % of P_{nom} (Fig. 12 ÷ 13)**
- **calculations were made both with one- and seven channels model of reactor core [8], [9]**
- **500 ÷ 1000 s steady- state calculation**
- **1000 ÷ 3000 s transient calculation**

The main goal of these calculations was to compare the influence of different reactor core models on the main characteristics of primary and secondary circuit. The calculation results show the good agreement of primary and secondary circuit parameters for both models. Differencies in main core parameters (coolant and fuel temperature) are caused by different nodalization.

♥ The sixth three-dimensional AER dynamic benchmark problem

This benchmark was defined at the 10^{th} Atomic Energy Research (AER) Symposium on VVER Reactor Physics and Reactor Safety in Moscow [10]. It concerns a double ended break of one main steam line in a VVER-440/213. The core is in full power conditions at the end of its first fuel cycle. The control rods belonging to the 6th group of control rods are at position of 175 cm from bottom of the core. Other groups of control rods are in a fully withdrawn position. The initial state conditions of the core in the beginning of the transients were given. The isothermal re-criticality temperature the core is defined to be 210 ^{o}C. It should be achieved by tuning the worth of all control rods. The main geometrical parameters of the plant system and the characteristics of control and safety systems to be considered were given.

The initiating event is a double ended guillotine break in the main steam line of steam generator No. 1 (see Fig. 3). The asymmetric leak causes a different

depressurization of all steam generators with consequent decreasing of primary pressure and coolant temperature. The reactor power scram is caused by power level signal "110 % of P_{nom}" with given time delay. All control rods fall down with exception two stuck rods belonging to the control rod group No. 3 and 4. The turbines are turned-off by closing the turbine isolation valves in the scram time. The pressure and volume control system are in operation and immediately are switched on to correct the system pressure and the pressurizer level. It was postulated that all MCPs remain in operation. MSH pressure signal causes the closing of steam isolation valves and isolation of all SG from the feed water. The culmination of recritical reactor power can be observed as a consequence of primary circuit overcooling. The reactor power excursion is terminated by injection of highly borated water from HPIS.

The detailed six loops model of NPP Dukovany with VVER-440/213 [4] was adopted for the 6th AER dynamic benchmark purposes. The nodalization scheme of the input model is presented in Fig. 1 – 3 .

1-D reactor downcomer nodalization model was used for preliminary benchmark calculations.

Full core neutron kinetic model was connected with seven coolant channel [5] was prepared [11], [12]. Distribution of different coolant channel can be seen in Fig. 7.

The core powers and system events reached during the transient are given in Table 1 and 2. Figures 14 – 19 show behavior of important time functions.

The main sense of NRI Rez participation at AER benchmark project is the preliminary verification of RELAP5-3D© system code solution against the results received by other best estimate three-dimensional neutronic codes coupled with NPP system codes - HEXTRAN/SMABRE (VTT Espoo Finland), DYN3D/ATHLET (FZ Rossendorf, Germany), BIPR8/ATHLET (KI Moscow, Russian Federation) and KIKO3D/ATHLET (KFKI Budapest, Hungary).

Table 1 : Core power

parameter	value	Time [s]
Total core power [MW]		
- at the beginning of transient	1376,909	0,0
- at the first power maximum	1520,678	11,5
- at the second power maximum	119,357	229,0
- at 20 second after start HPIS	80,323	261,5
Total prompt fission power [MW]		
- at the beginning of transient	1286,069	0,0
- at the first power maximum	1428,769	11,5
- at the second power maximum	79,206	229,0
- at 20 second after start HPIS	41,841	261,5

Table 2 : Table of events

Time [s]	Event
- 1000	- begin of calculation stabilization at required parameters
0	- double ended guillotine break of MSL1
0,1	- break is fully open
5	- activation of 2nd group of PRZ heaters
8,5	- start of 1st make-up pump
11	- activation of 3rd group of PRZ heaters
11,6	- reactor SCRAM (from 110% power)
11,6	- turbines FAV closing
19	- activation of 4th group of PRZ heaters
48,5	- start of 2nd make-up pump
59,5	- deactivation of all group of PRZ heaters
61	- low PRZ level 2,41 m
74,5	- low MSH pressure 3 MPa: → closing of all MSL FAV → isolation of FW lines to all SGs
229	- culmination of recritical reactor power (P_{max} = 8,68 % P_{nom})
241,5	- start of HPSI (180 s delay after „low PRZ level“, working 2 HPIS trains, injection to loops-3,5)
532	- activation of all group of PRZ heaters
699	- deactivation of 4th group of PRZ heaters
723	- deactivation of 3rd group of PRZ heaters
751,5	- deactivation of 2nd group of PRZ heaters
786	- deactivation of 1st group of PRZ heaters
1000	- end of analysis

Conclusions

- **Accurate tuning of initial steady state. The relatively accurate estimate of initial distribution of coolant temperature, energy releasing, heat structure temperature and coolant mass flow rate in coolant channels is necessary in order to reach the stabilization of steady-state before the transient calculation.**

- **Multi-channels reactor core representation has to be tested in the future.**
  1 channel → 7 channels →31 or more channels representation.

- **Testing of multidimensional (3-D) thermal-hydraulic reactor pressure vessel nodalization and comparison with 1-D nodalization model should be performed.**

- **Cross sections library should be prepared in a more general form.**

- **Influence of Xe and Sm concentration on neutron cross section should be included in USER option.**

- **In USER option for neutron cross section data doesn't exist transfer of volume feedback weighting factors and heat structure feedback factors from input deck into common xsuser.h, which is called in external subroutine userxs.f.**

References

[1] RELAP5-3D© Code Manual, Volume I: Code Structure, System Models and Solutions Methods, Idaho National Engineering and Environmental Laboratory, Lockheed Martin Idaho Technologies Company, Idaho Falls, Idaho 83415, INEEL-EXT- 98-00834, February 1999

[2] NESTLE 5.0 – Code System to Solve the Few-Group Neutron Diffusion Equation Fixed-Source Steady- State and Transient Problems, RSIC, P.O. Box 2008, Oak Ridge, TN 37831-6362, December 1996

[3] Engineering Handbook for the Safety Analysis of WWER-440 Model 213 Nuclear Power Plants. Revision 1. IAEA TC/RER/004-A042, April 1994

[4] P. Král, J. Krhounková, L. Denk, Z. Parduba: Thermal Hydraulic Analyses of LOCA with Break Size D50 for PTS Evaluation, ÚJV Rez, November 1999 (in Czech)

[5] I. Tinka: Data Libraries for VVER-440 Standard Fuel, Report Energoprojekt, 221-6-920322, Praha 1992 (in Czech)

[6] I. Tinka: Data Libraries for VVER-440 Control Rods and Reflectors, Report Energoprojekt, 221-6-920555, Praha 1992 (in Czech)

[7] J. Hádek: Description of the Code Dylie for Automatized Preparation of the DYN3D Code Neutronic Input Data, Report ÚJV Rez, ÚJV 11051 R,T, December 1997 (in Czech)

[8] J. Hádek, P. Král: 3D Neutron Kinetics Model Input Data Preparation of Computing Code RELAP5-3D© for VVER-440/213 , Report ÚJV Rez, ÚJV 11 554 R,T, December 2000 (in Czech)

[9] J. Hádek: First Experience with the System Code RELAP5-3D©. Input Data Preparation for the 3D Neutronic Model VVER-440/213 of System Code RELAP5-3D© and Its Testing, AER Working Group D Meeting on VVER Safety Analysis, Budapest, Hungary, 7 – 9 May 2001

[10] S. Kliem, A. Seidel, U. Grundmann: Definition of the Sixth Dynamic AER Benchmark Problem, 10th AER Symposium on VVER Reactor Physics and Reactor Safety, Moscow, Russia, 18 – 22 September 2000

[11] J. Hádek, P. Král, J. Macek: Preliminary Results of the Sixth Three-Dimensional AER Dynamic Benchmark Problem Calculation. Solution of Problem with DYN3D and RELAP5-3D© Codes, AER Working Group D Meeting on VVER Safety Analysis, Budapest, Hungary, 7 – 9 May 2001

[12] J. Hádek, P. Král, J. Macek: Interim Results of the Sixth Three-Dimensional AER Dynamic Benchmark Problem Calculation. Solution of Problem with DYN3D and RELAP5-3D© Codes, 11th Symposium of AER on VVER Reactor Physics and Reactor Safety Analysis, Csopak, Hungary, 24 – 28 September 2001

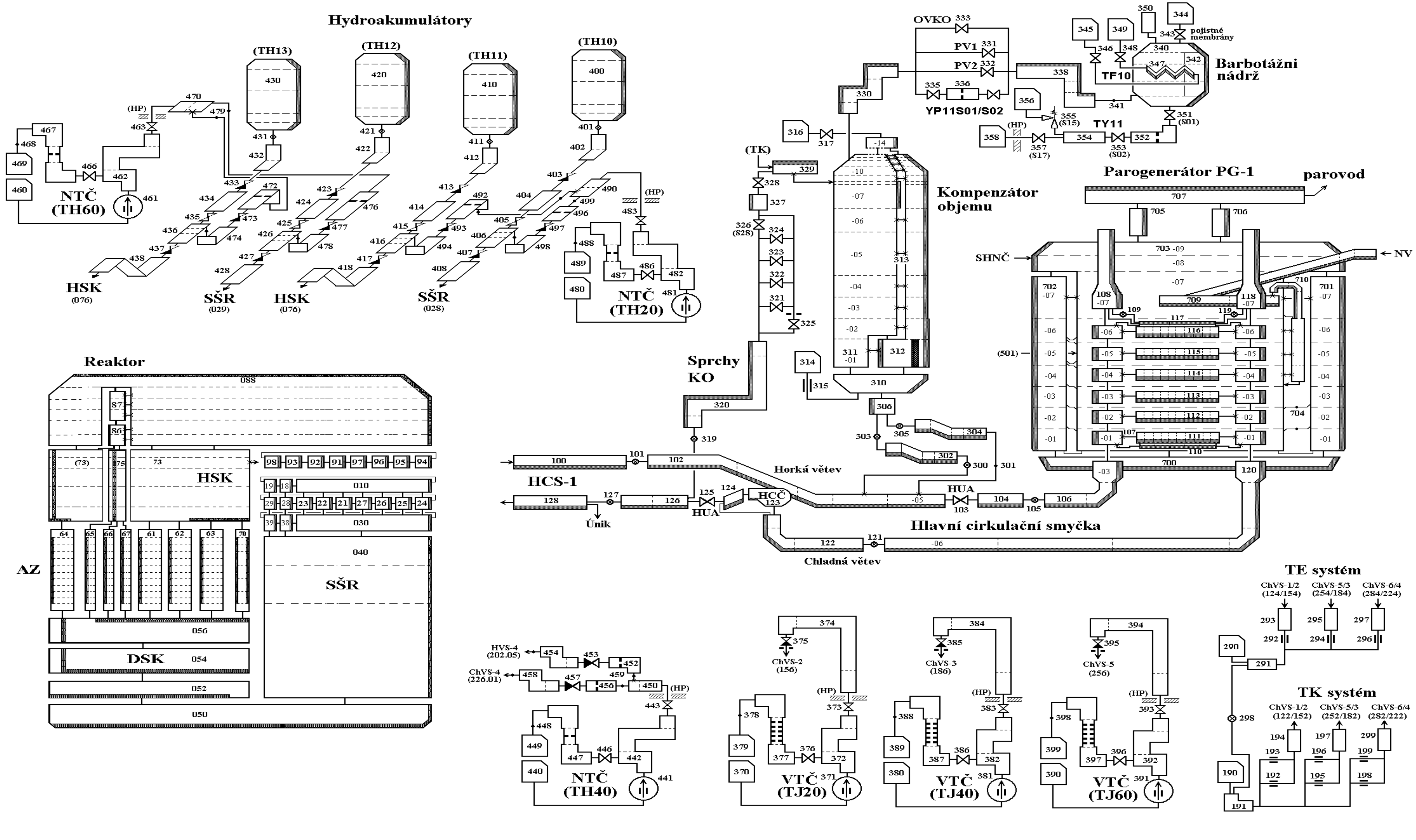


Fig. 1 : Nodalization of part of VVER-440/213 primary circuit and reactor with seven channels core

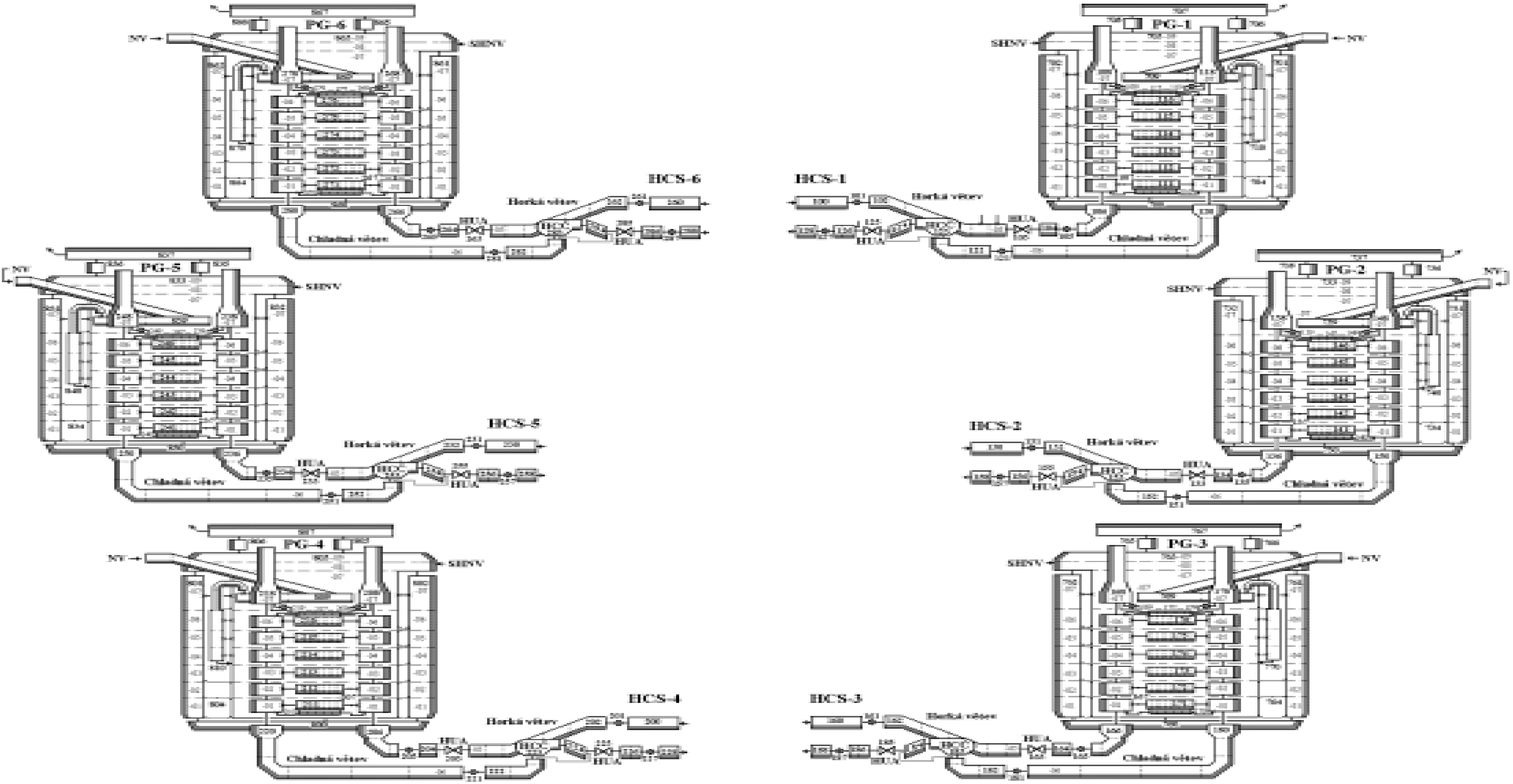


Fig. 2 : Nodalization of primary circuit and steam generators

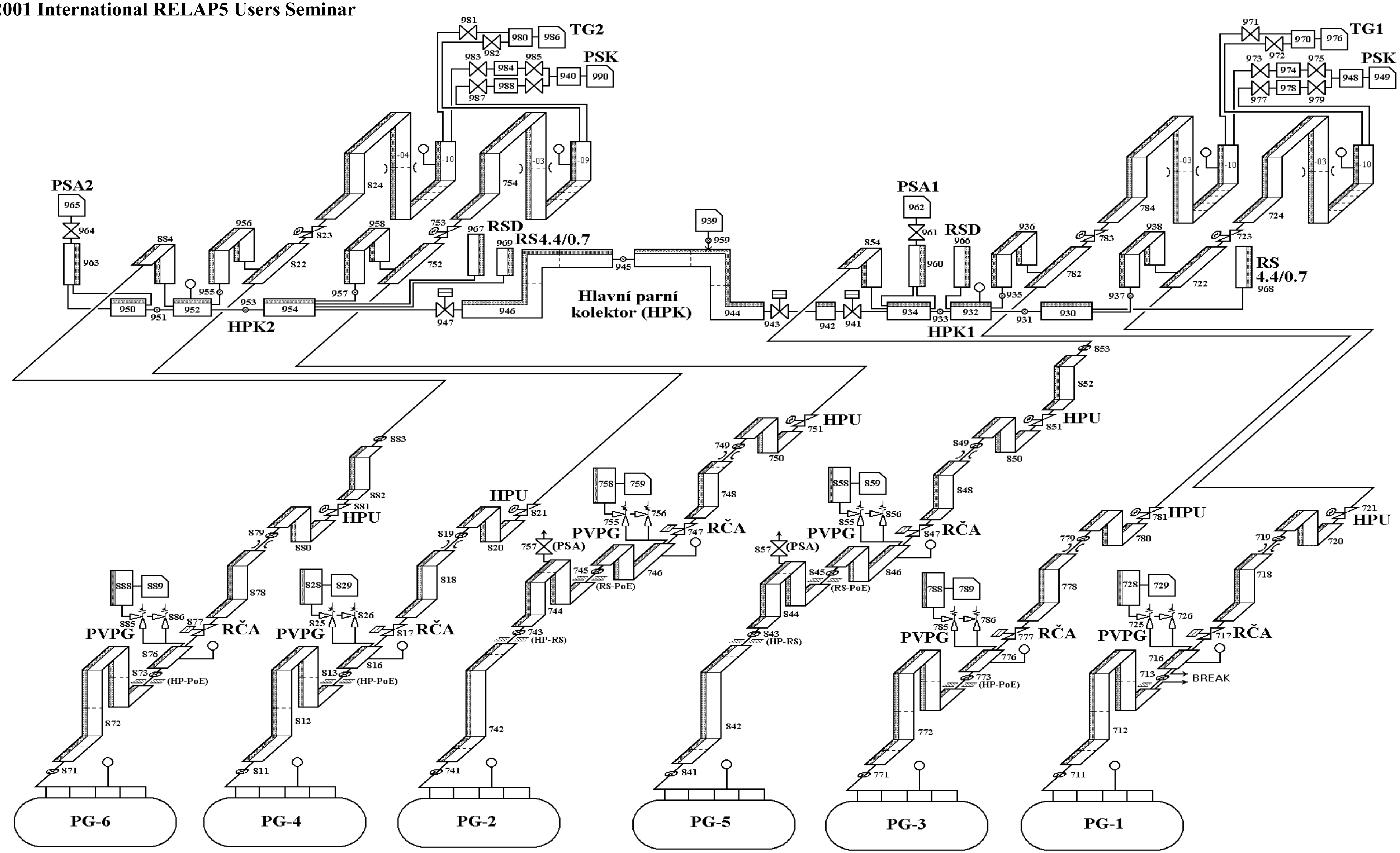


Fig . 3: Main steam system (MSS) nodalization

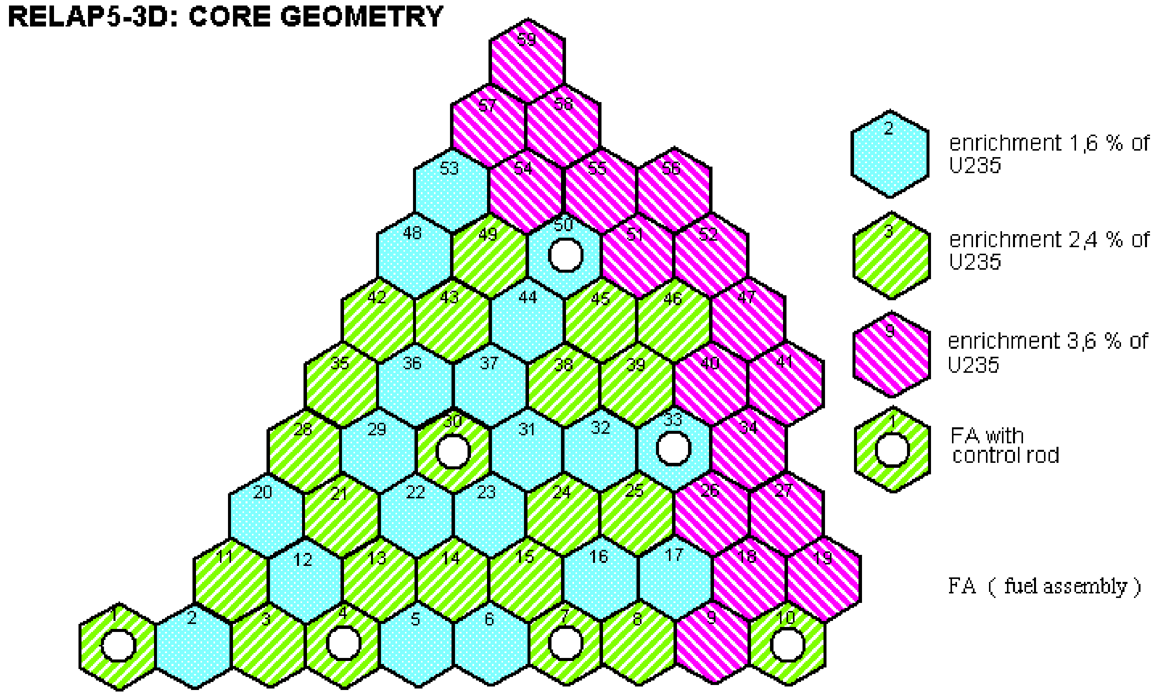


Fig. 4 : Distribution of fuel assemblies in VVER-440 reactor core, 60 degree rotational symmetry

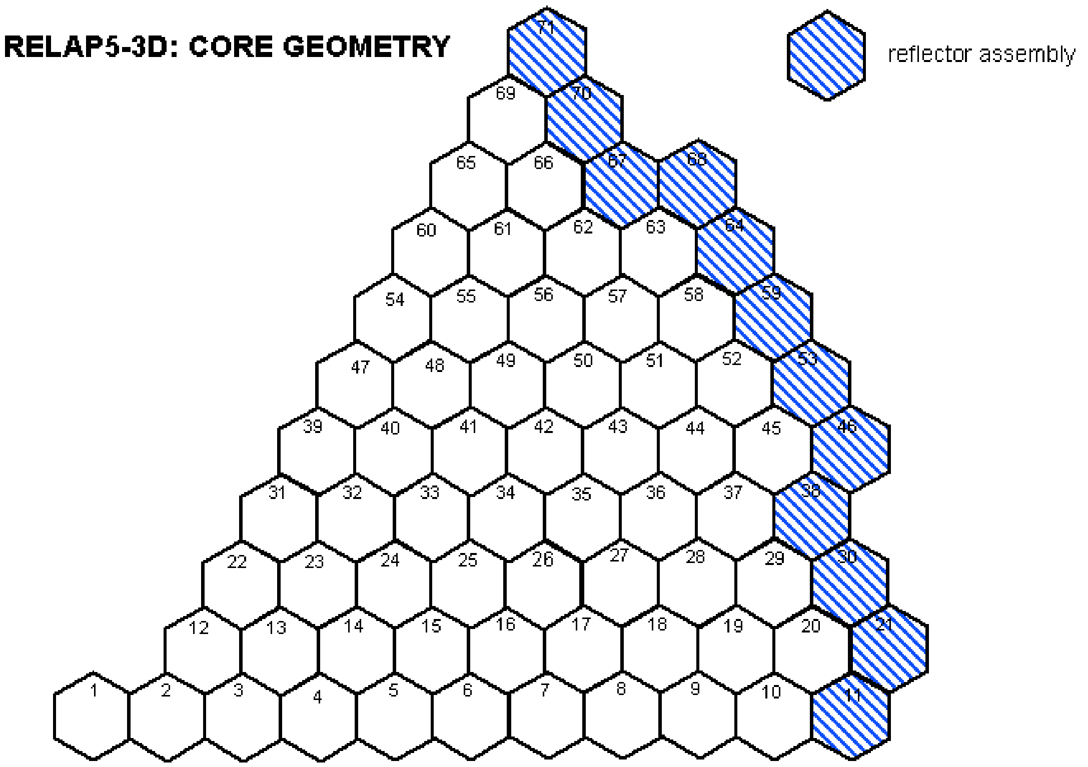


Fig. 5 : VVER-440 reactor core with reflector assemblies, 60 degree rotational symmetry

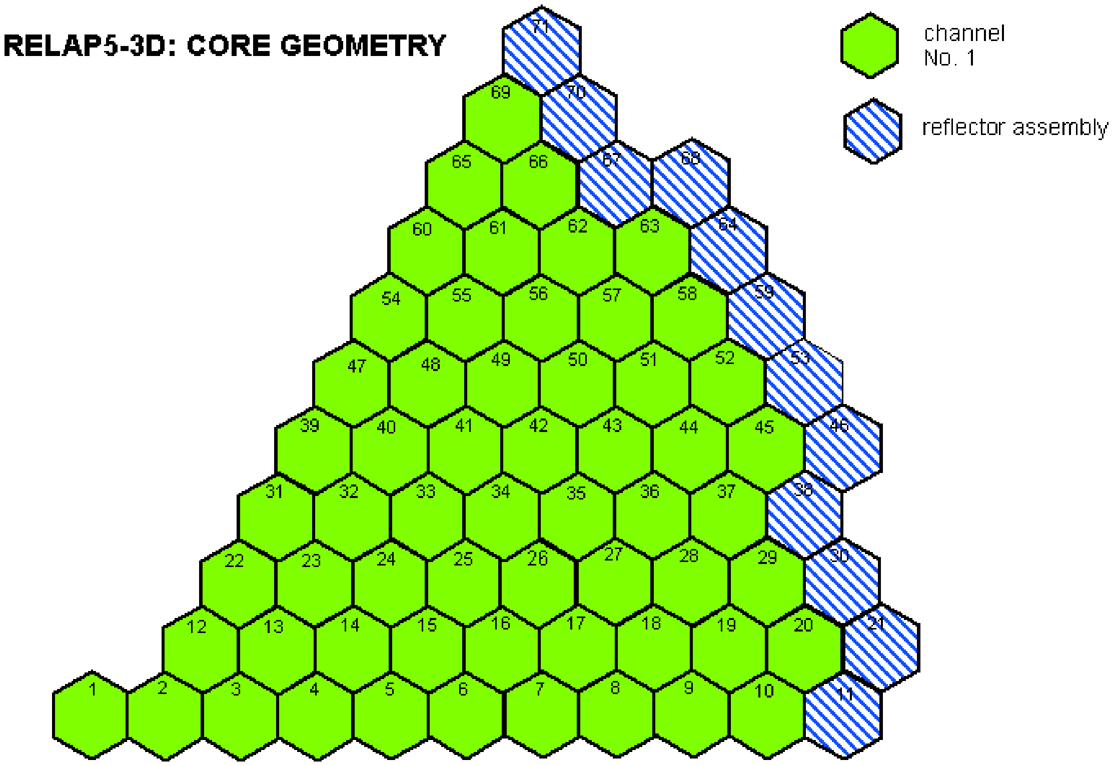


Fig. 6 : Reactor core configuration with 1+ 1 coolant channels

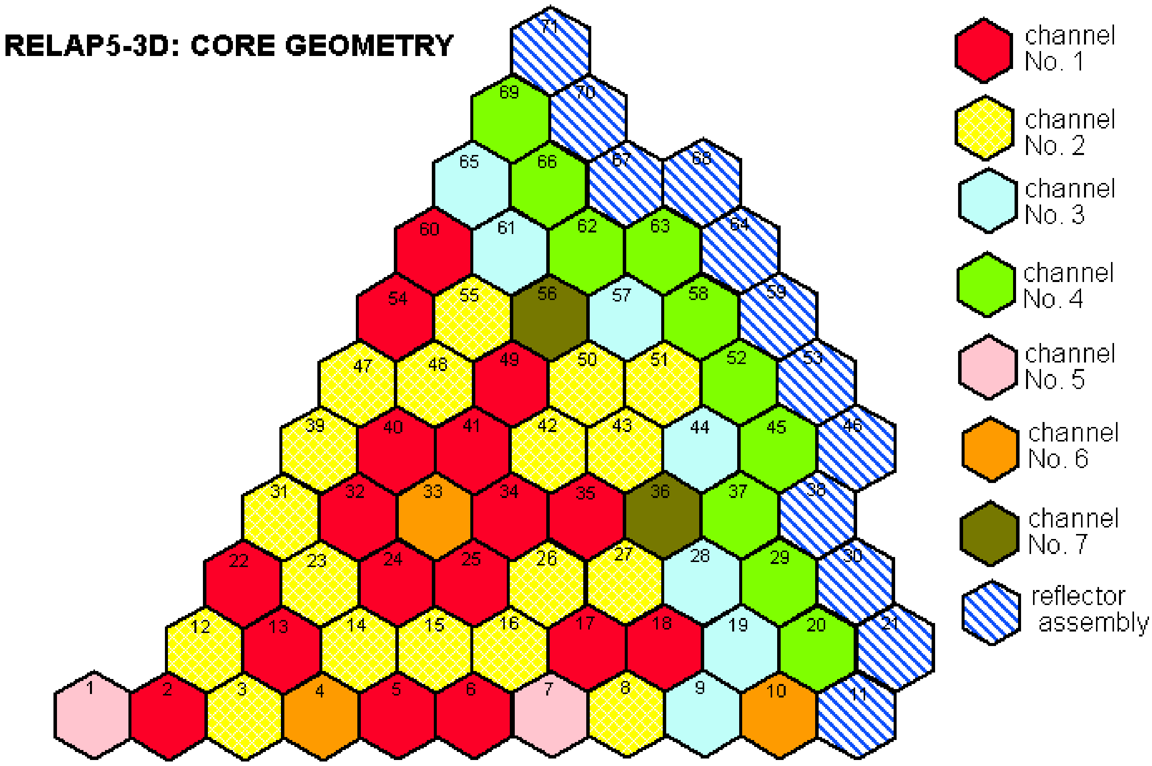


Fig. 7 : Reactor core configuration with 7 + 1 coolant channels

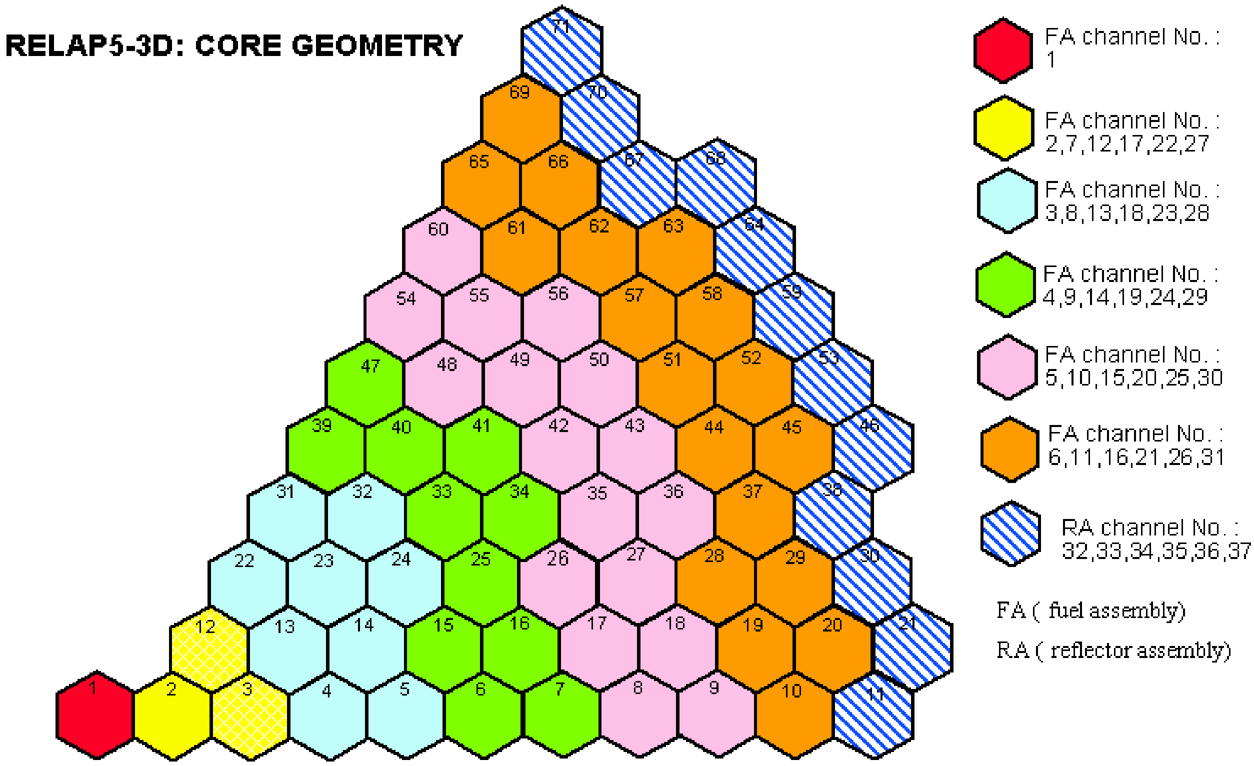


Fig. 8 : Reactor core configuration with 31 + 6 coolant channels

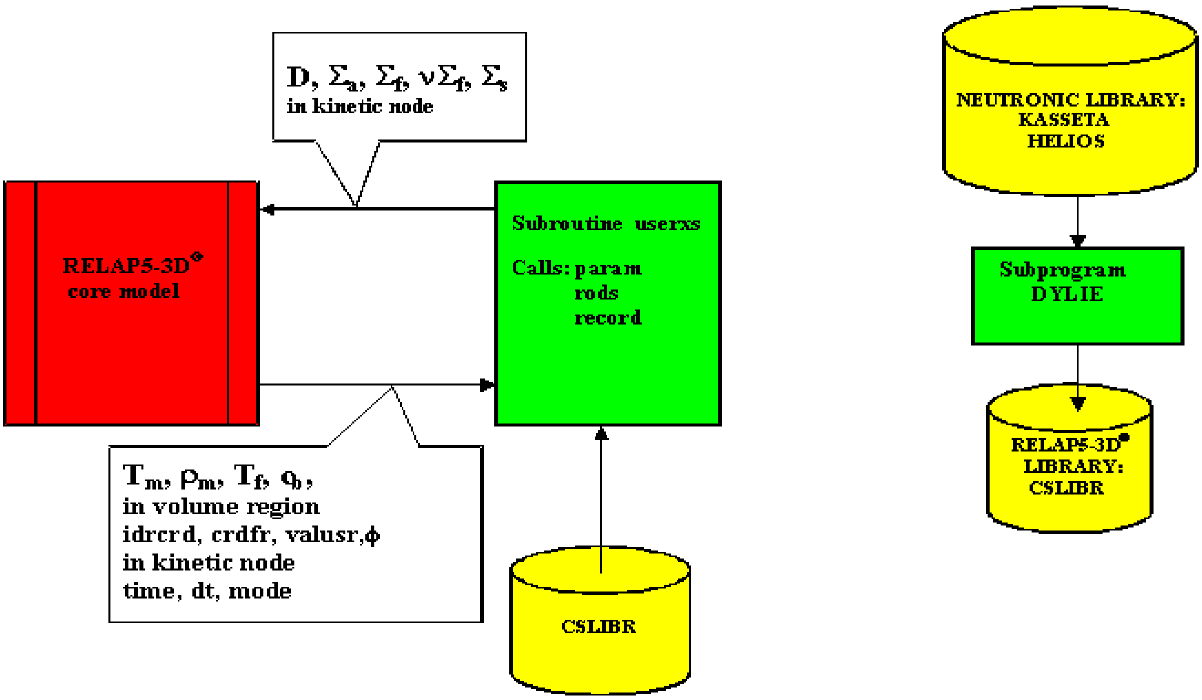


Fig. 9 : Neutronic data preparation of and exchange of parameters in USER option

Fig. 10 : 1 a 7 channel model - shutdown of 1/6 MCPs
fission and total reactor power

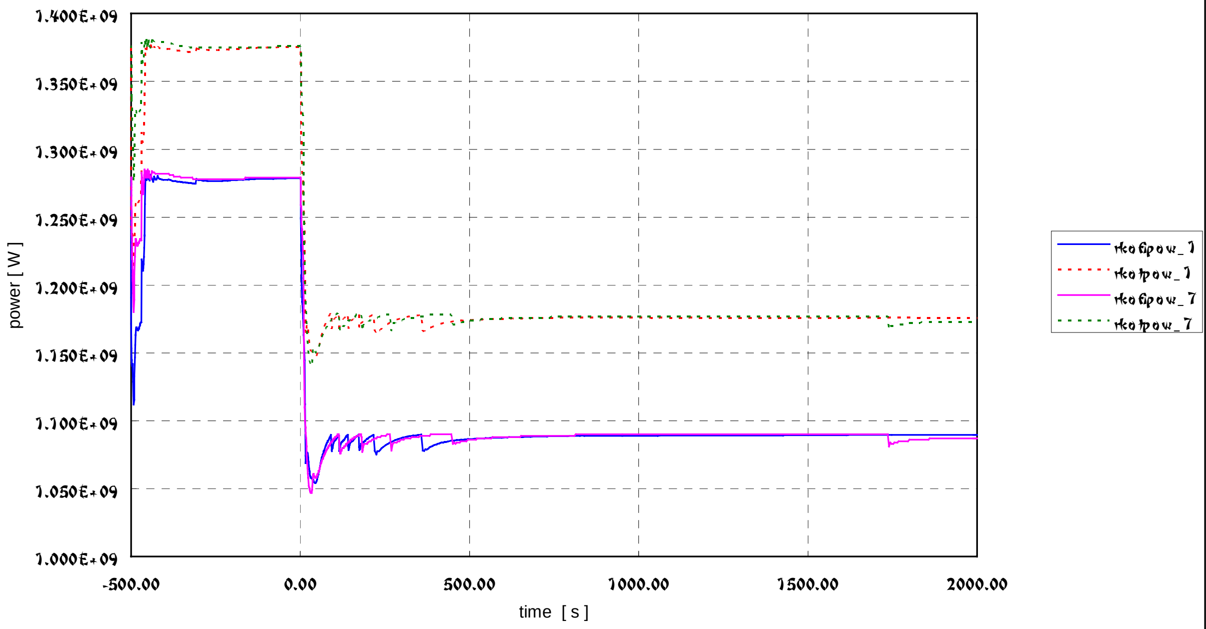


Fig. 11 : 1 a 7 channel model - shutdown of 1/6 MCPs
reactor, core (1st channel) and bypass mass flow rate

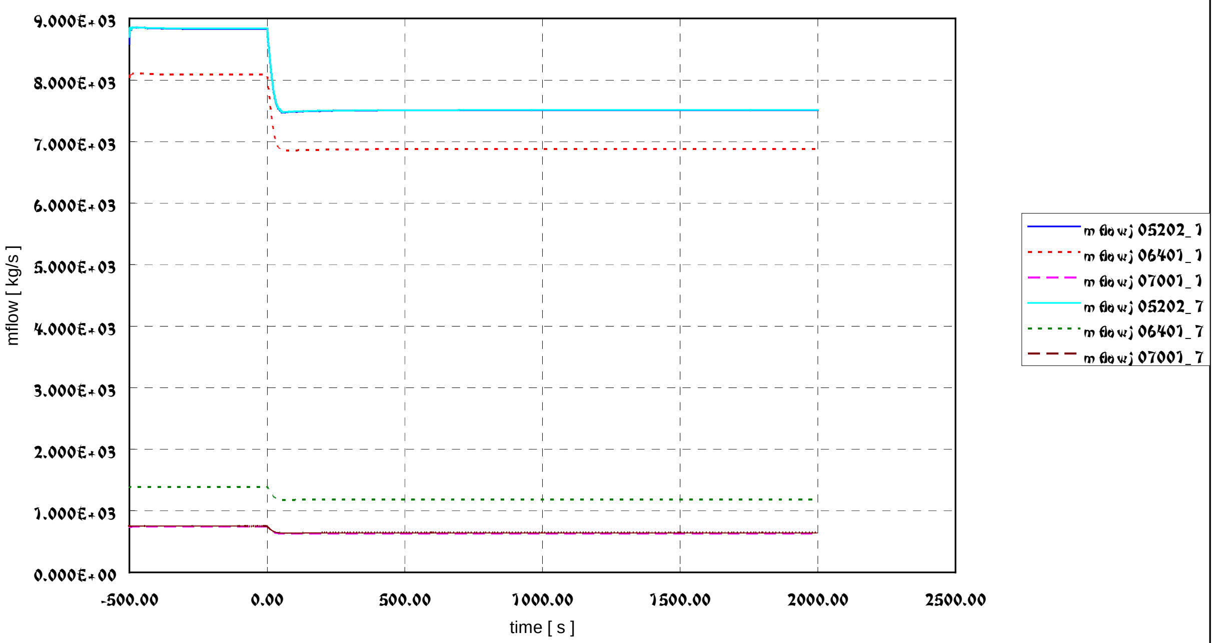


Fig. 12 : 1 a 7 channel model - shutdown of all MCPs
fission and total reactor power

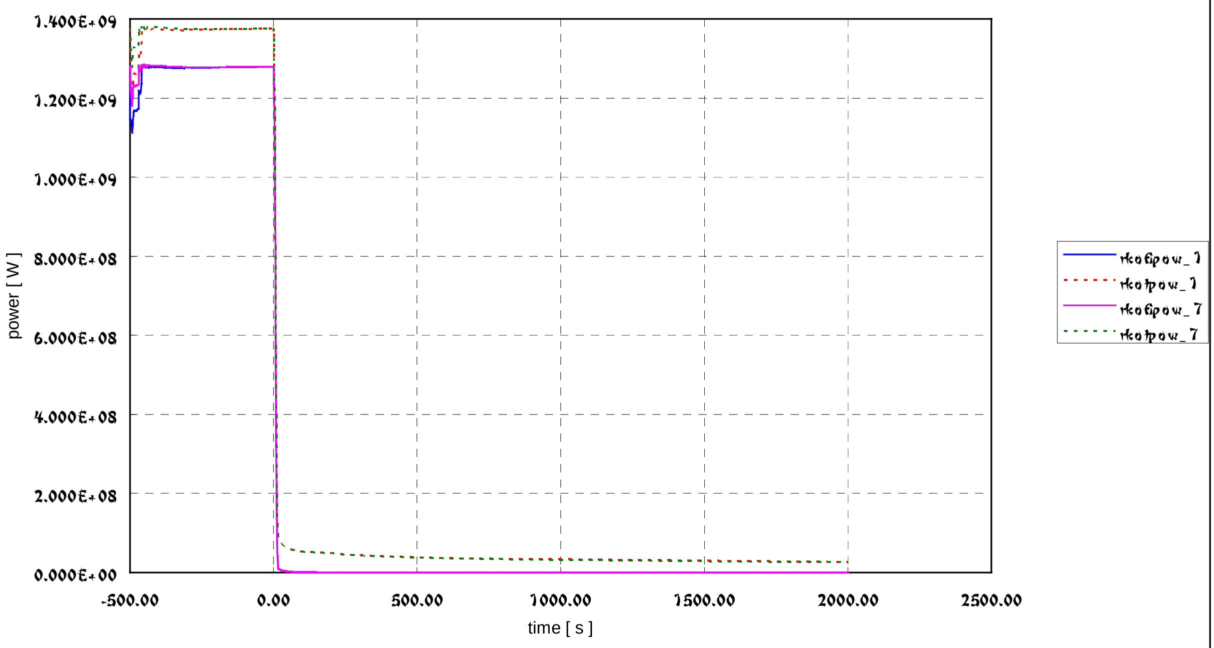


Fig. 13 : 1 a 7 channel model - shutdown of all MCPs
reactor, core (1st channel) and bypass mass flow rate

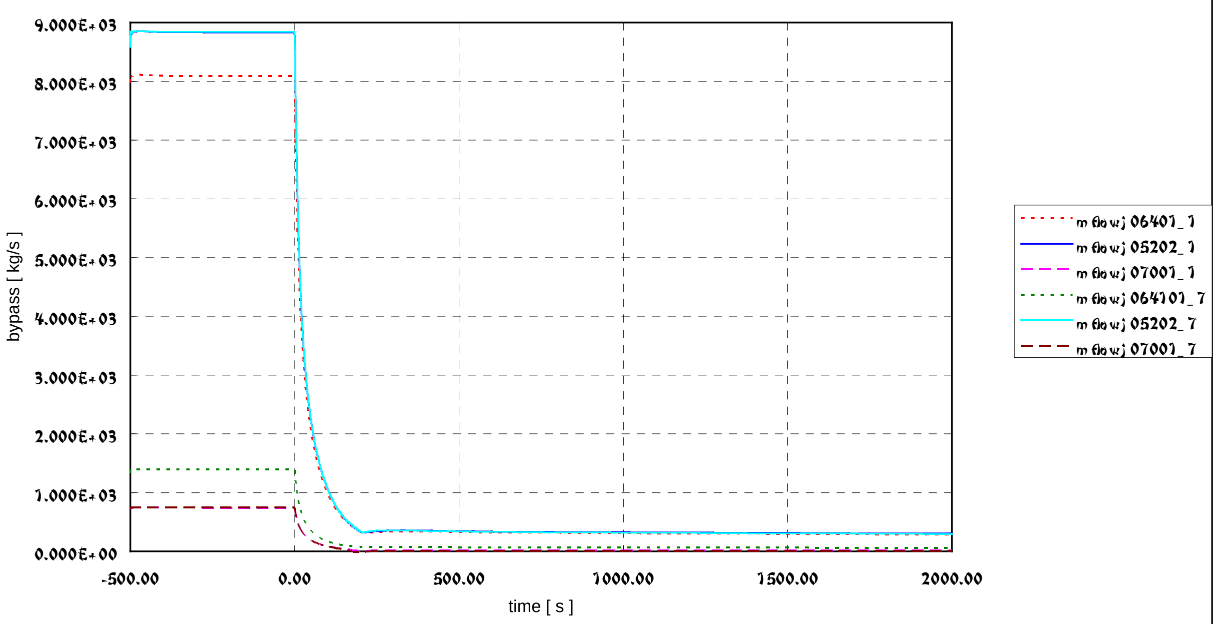


Fig. 14: Parts of core nuclear power
(MSL1 break - ÚJV preliminary calculation for AER, RELAP5-3D, 2/7/2001

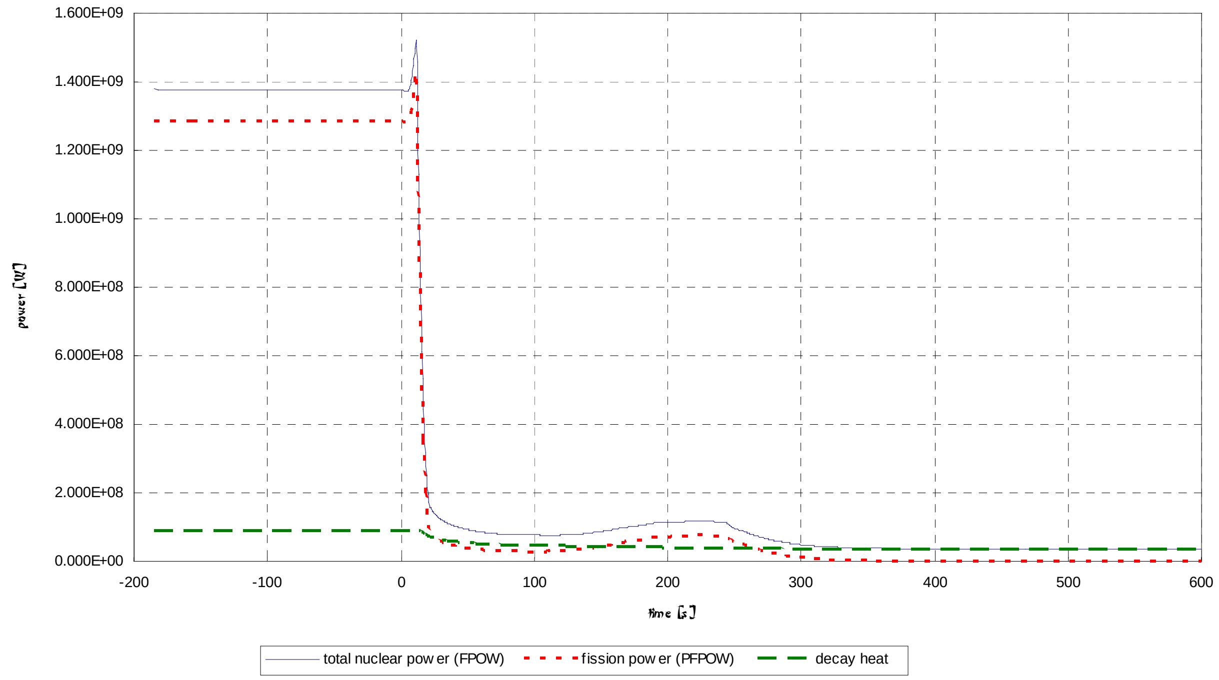


Fig. 15: Reactor temperatures
(MSL1 break - ÚJV preliminary calculation for AER, RELAP5-3D, 2/7/2001

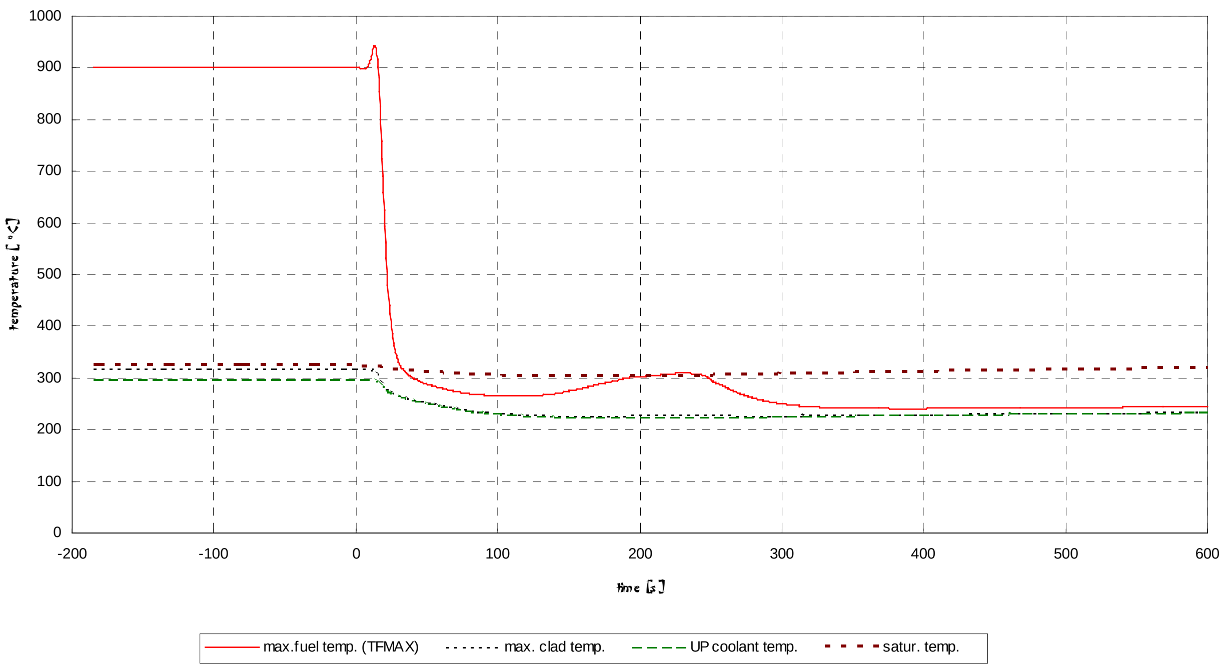


Fig. 16: Reactor inlet temperatures
(MSL1 break - ÚJV preliminary calculation for AER, RELAP5-3D, 2/7/2001

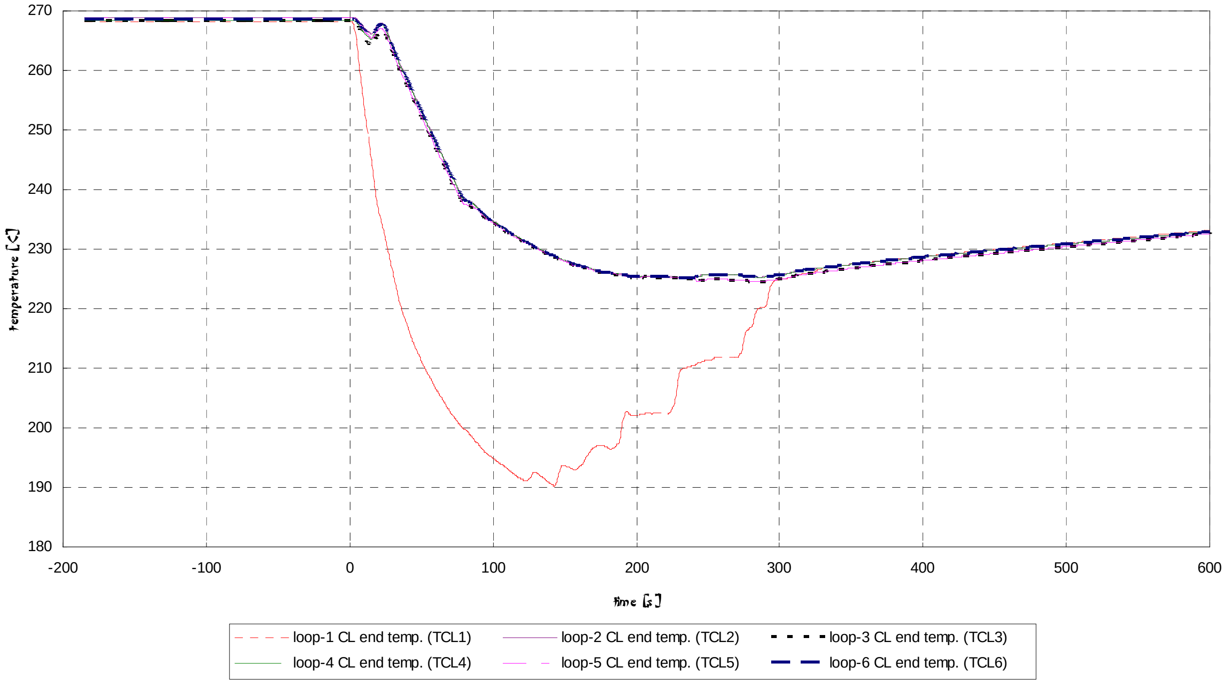
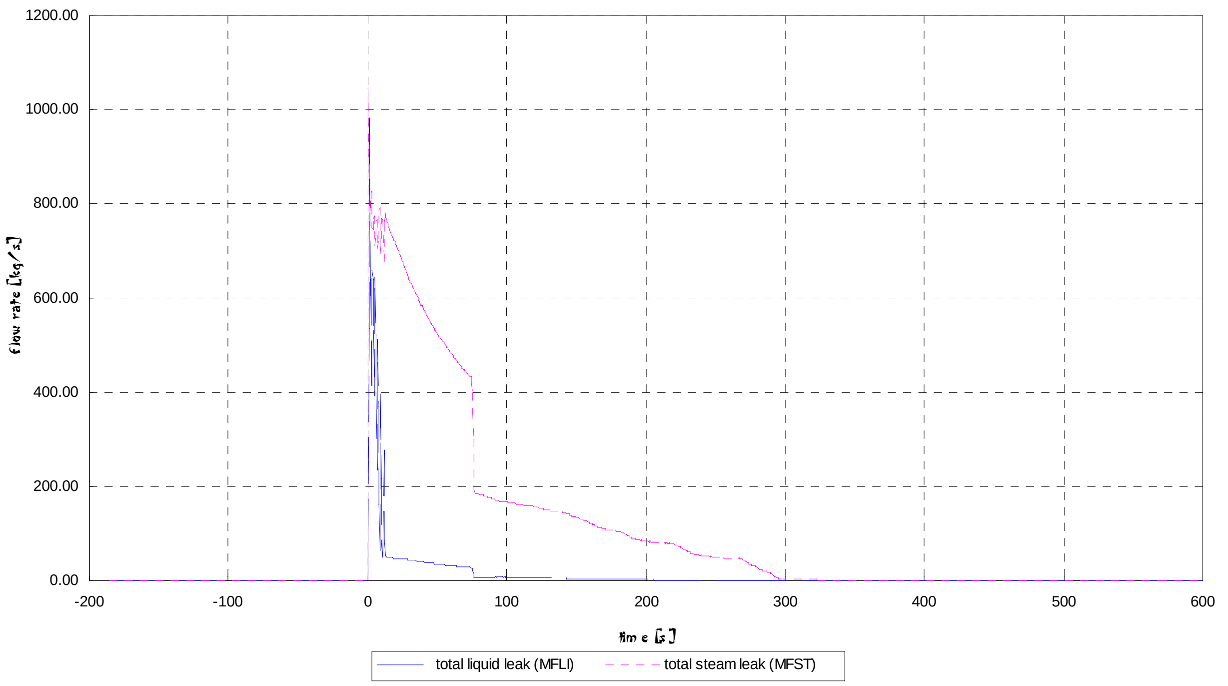


Fig. 17: Leak mass flow rate
(MSL1 break - ÚJV preliminary calculation for AER, RELAP5-3D, 2/7/2001



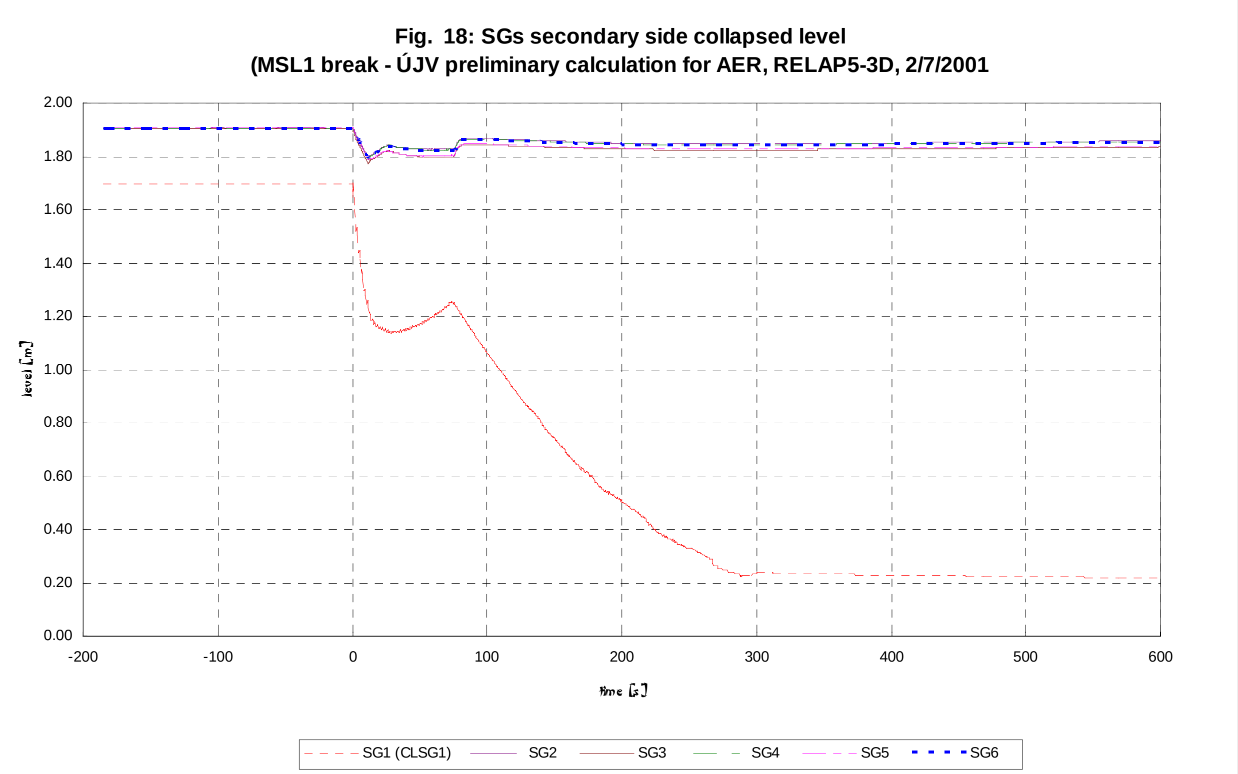


Fig. 18: SGs secondary side collapsed level
(MSL1 break - ÚJV preliminary calculation for AER, RELAP5-3D, 2/7/2001
2.00
1.80
1.60
1.40
1.20
1.00
0.80
0.60
0.40
0.20
0.00
level [m]
-200
-100
0
100
200
300
400
500
600
time [s]
SG1 (CLSG1)
SG2
SG3
SG4
SG5
SG6

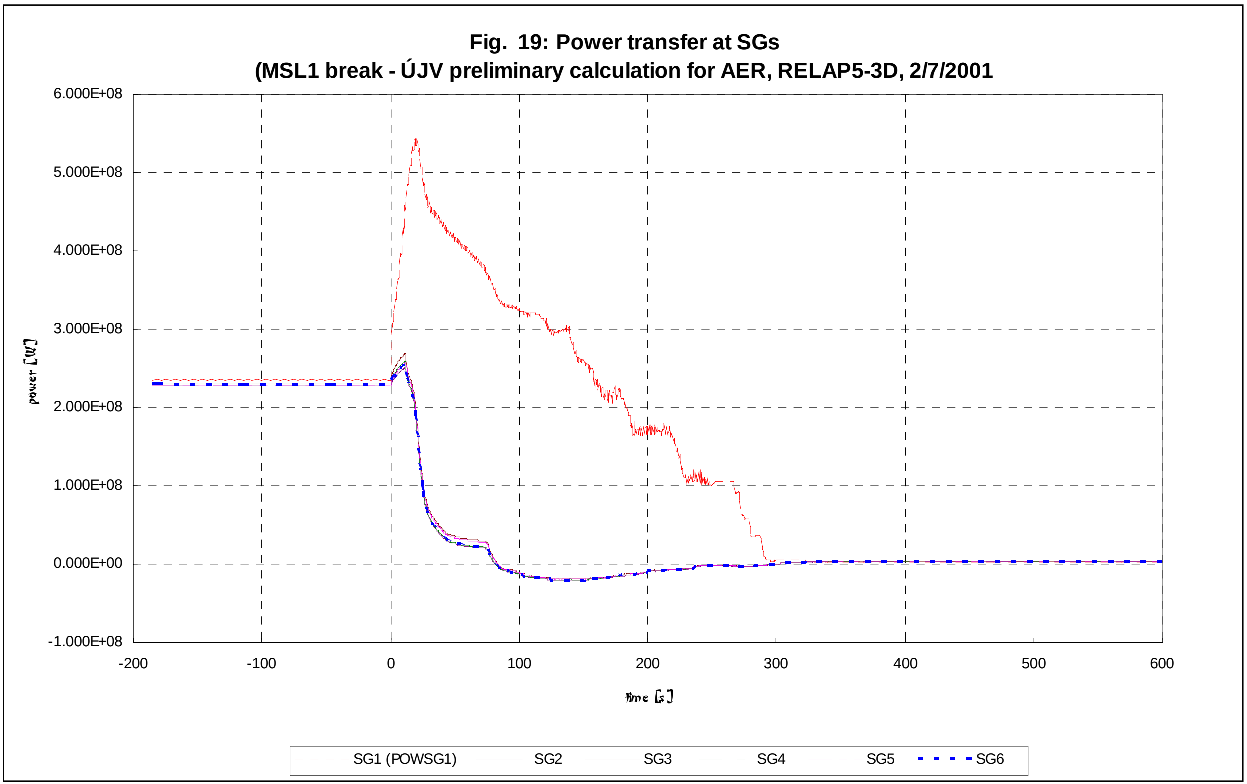


Fig. 19: Power transfer at SGs
(MSL1 break - ÚJV preliminary calculation for AER, RELAP5-3D, 2/7/2001
6.000E+08
5.000E+08
4.000E+08
3.000E+08
2.000E+08
1.000E+08
0.000E+00
-1.000E+08
power [W]
-200
-100
0
100
200
300
400
500
600
time [s]
SG1 (POWSG1)
SG2
SG3
SG4
SG5
SG6