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ANALYSIS OF THE NPP-V1 PRIMARY CIRCUIT FAST COOLDOWN

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SUMMARY

Results of thermal-hydraulic calculations of the NPP-V1 primary circuit fast cooldown during small leakage through openings of diameter 20,32 and 50 mm as well as analyses of cooldown following the steam pipeline break at nominal and null reactor power are given in this paper.

USED ABBREVIATIONS

AFWP - auxiliary feedwater pump
CL - cold leg
DG - diesel generator
EHPIP - emergency high pressure injection pump
FAV - fast-acting valve
FTSV - fast-acting turbine stop valve
HL - hot leg
LOCA - loss of coolant accident
MFWP - main feedwater pump
MSH - main steam header
NC - natural circulation
NPP - nuclear power plant
PC - primary circuit
PCL - primary circuit loop
PCP - primary coolant pump
PTS - pressurized thermal shock
PZR - pressurizer
RC - reactor core
RCSC - reactor core support cylinder
REP - reactor emergency protection
RPV - reactor pressure vessel
SC - secondary circuit
SDVA - steam dump valve to the atmosphere
SDVC - steam dump valve to the condenser
SG - steam generator
TG - turbogenerator
TMCH - top mixing chamber

1. INTRODUCTION

Detailed study of causes, course and consequences of American NPP Three Mile Island-2 accident in the year 1979 has pointed out an importance of transient processes with fast and wide range cooldown of primary circuit with subsequent pressure increase above its nominal value. Such a kind of transient processes is classed as a Pressurized Thermal Shock (PTS) and represents a considerable danger for RPV integrity. Superposition of thermal and mechanical stresses occurs in the RPV wall material, while the presence of undetected material defects cannot even be totally excluded. The PTS processes have got a particularly great importance in connection with finding out a quicker material embrittlement (first of all of the welds) of RPV of the first reactor generations, namely by influence of radiation damage [2]. In consequence of the material brittleness increase the transitional temperature moves from the tenacious failure (fracture) to the brittle one to higher temperature values and thus the so-called "operational window", which is created by limiting conditions for primary circuit pressures and temperatures, decreases.

The RPV of the NPP-V1 first unit and expecially second unit (reactors of the V-230 type) are in worse condition from the point of view of the basic as well as weld metal embrittlement. Therefore more attention was devoted them during their operation and material research.

First of all the primary circuit small-break loss of coolant accidents and accidents with large steam leaks from the SC belong into the group of PTS processes.

Complex study of these processes requires to involve in the solution more technical fields, which include research of material properties, fracture mechanics of bodies, thermal-hydraulic calculations, radiation physics and others. In this report an attention is devoted to thermal-hydraulic calculations only and fast cooldown of a unit is discussed from operational mode point of view only.

2. THERMAL-HYDRAULIC CALCULATIONS

2.1. Small-break LOCA

2.1.1. Characteristics of the process

Small-break LOCAs are worse from the point of view of unfavourable thermal and mechanical stress of the RPV, because during medium and large-break LOCAs the primary circuit pressure decreases quickly and the RPV total stress is smaller. Leakages through small sections are compensable by emergency high pressure injection pumps, which are automatically actuated from the pressurizer pressure and water level decrease. The PC pressure is then maintained higher than its nominal value, while its value depends namely on the EHPIP characteristics ("closing point"). The PCPs remain in operation, flow rate through PCLs is high and influence of the cold water fed by EHPIP on the reactor inlet coolant temperature practically does not occur. Therefore the loss of unit internal consumption supply was chosen as an independent failure, which will occur at the accident rise moment. It was assumed that two PCPs from six PCPs operating (in the PCL No. 2 and 5) are supplied or from the internal consumption supply transformer either from the back-up transformer. The remaining 4 PCPs are supplied from the internal consumption generator. During loss of internal consumption supply two PCPs are practically immediately disconnected and remaining 4 PCPs coastdown with TGs. After loss of two PCPs the flow direction in the appropriate PCLs is reversed within few seconds, but after finishing the coastdown of remaining 4 PCPs the flow rate in all PCLs will set in the original direction and the flow rate depends on the natural circulation intensity only. The NC intensity is proportional to the residual reactor power. Besides the NC has an unfavourable effect which causes the reversal of heat transfer between PC and SC. Scheme of the EHPIP pipe connections to the PZR is given in Fig.a.

The system consists of two identical subsystems, each of which includes three EHPIPs. One subsystem is connected to the suction, the second one to the delivery pipe of the PCPs in all PCLs. Cold water fed by the EHPIPs (~55°C) into the delivery pipe of the PCPs will create a colder layer on the bottom of the pipe, which successively flows into the RPV, where it flows down along the reactor core support cylinder wall and creates the so-called cold tongues. Then water drops lower in the RCSC, heats up from the RPV wall and partially is mixed with hot water which is contained in the RCSC.

Water delivered to the PCP suction successively fills the part of the PCL between the PCP and SG up to the creation of cold water plug what practically will cause stopping the water circulation in this PCL. The hydrostatic pressure of the water seal increases proportionally to the difference of specific weight of cold and hot water, what further supports the flow staguation. Decrease of the NC flow rate causes the coolant temperature increase in the top mixing chamber. After reaching a sufficient warming-up a pressure difference is created, which "will force" the stagnated coolant from the cold leg water seal into the RCSC. Sudden getting of large quantity of cold water into the RCSC can cause a thermal shock on the RPV. Whole process is cyclically repeated, while considerable temperature waves occur at the RPV inlet. Water seal is preferentially created in the failed PCL and in the PCLs with short coastdown of two PCPs.

2.1.2. Definition of initiating event and initial conditions

LOCAs with equivalent leak opening of diameter 20,32 and 50 mm were analysed by the code RELAP5/MOD2. Creation of the leak opening with appropriate diameter in the axis of the horizontal section of the PCL between the PCP delivery and the RPV in the PCL with the PZR was supposed as an initiating event. The initiating event was combined with the loss of unit internal consumption supply at the accident rise moment. Actuation of the REP-1 and loss of PCP were supposed at the same time. Shutdown of both TGs was supposed with the delay 10 sec after the REP-1 actuation. The EHPIPs started with 70 sec delay after signal rise and one AFWP started with delay of 90 sec after signal rise.

The unit operation at the end of fuel campaign, with 6 PCPs, at 102% of nominal power level, at power grid frequency 48 Hz, with corresponding flow rate decrease through the reactor was assumed at the accident rise moment. Geometric and hydraulic characteristics of the NPP-V1 second unit [1] were pessimistically assumed for the calculation.

Assumptions of calculation were chosen with the aim to maximize the temperature effects of cold water to the RPV wall, what was achieved by :

- NC minimization so that the Untermayer formula was used for the reactor residual power calculation providing the following reactor core history :
  - . fuel campaign - 300 days, outage - 84 days, refueling of 119 fuel assemblies, power increase to the nominal value during 0,7 day, then reactor scrammed by the REP-1
- maximization of the EHPIP water flow rate using the characteristics which give the high flow rate
- respecting the unevenness of the cold water injection into the individual PCLs caused by different hydraulic resistances from the EHPIP delivery collector to the PCL
- no interventions of operational personnel into the accident course during whole analyzed time interval were considered.

Nodalizing scheme of the PCL with the leak is given in Fig.b.

2.1.3. Calculation results

Three variants were analyzed which differ by the leakage opening size only (equivalent diameter of 20,32 and 50 mm). Chronology of important events for separate calculational variants is compared in table 1.

The process course for the break diameter 20 mm is briefly described in the following text. Time courses of the most important parameters of the variant with break diameter 20 mm are shown in Fig. 1 -10.

At the time 0.0 sec the break rise and parallely the loss of unit internal consumption supply occur. This results in loss of two PCPs, four PCPs coastdown with the PGs. Both TGs are shutdown at 10th sec by closing the FTSVs. The SDVA is open at 20th sec.

Coolant temperature, PC pressure and PZR water level decrease by coolant leakage and especially after the REP-1 actuation. At the PZR level decrease by 2.56 m four EHPIPs start (92nd sec). Coolant delivery by the EHPIPs is already higher than leakage from the PC at 95th sec and the PZR is successively filled. The PC pressure, after initial short-term, decreases under value of 11 MPa and after starting the EHPIPs it increases to the value of about 13 MPa, at which it is maintained. During successive cooldown of the PC at about 8000th sec a periodical forcing the cold water from the cold leg water seals into the RCSC occurs. Besides the asymmetrical RPV thermal stress occurs at about 3900th sec. It is caused by considerably colder water (by 150°C). With regard to high PC pressure, the thermal shocks caused by this process are particularly danger.

Some parameters are compared for three leakage opening sizes in Fig. 11 to 14. In principle these leakages are compensable by the EHPIP operation. While the PC pressure is continuosly maintained at about value of 13 MPa for the break diameter 20 mm, the PC pressure moves within the range 9.5 - 9.2 MPa for the break diameter 32 mm. Temperature waves with the amplitude about 100°C occur at about 4600th sec. Leakage and filling of water are identical for the break diameter 50 mm at 400th sec. Then the PC pressure quickly decrease and both the PC and SC pressures are identical at 450th sec. As a results of this, the EHPIP water delivery is 3-times higher than for variant with the break diameter 20 mm. It has a considerable influence on the total rate of the PC cooldown. Indication of the temperature wave rise occurs at about 2600th sec, but regarding that the PC pressure is lower at this time than 3,5 MPa, this variant is not so dangerous from the point of view of brittle failure (fracture) than the previous ones.

Table 1. Chronology of important events for separate calculational variants

Time [s] Event	Break opening [mm]		
	Ø20	Ø32	Ø50
Break opening rise	0	0	0
REP-1 actuation	0	0	0
Loss of unit internal consumption supply	0	0	0
Loss of PCPs supply	0	0	0
Shutdown of TGs	10	10	10
First opening of the SDVA	20	20	20
First closing of the SDVA	100	90	85
PC pressure decrease by 0.96 MPa (REP-2)	24	15	9
PZR water level decrease by 2.56m (SHPIP)	92	37	19
PZR water level decrease by 2.7 m (REP-1)	x	41	21
Start of 4 PCPs	92	75	75
First equalization of leakage and filling	95	80	450
Start of 1 AFWP	90	90	90
Second opening of the SDVA	375	x	x
Second closing of the SDVA	625	x	x
Heat transfer reversal at least in one SG	85	80	100
Total heat transfer reversal in SG	2100	120	100
End of analysis	9900	9900	2800

x phenomenon did not occur

2.1.4. Conclusions resulted from analyses

Spectrum of LOCAs with equivalent leakage opening of diameter 20,32 and 50 mm, supposing the loss of unit internal consumption supply at the moment of the leakage opening rise, was analyzed in this report.

Calculational assumptions were pessimistically chosen with the aim to maximize the unfavourable effects of cold water filled by the emergency core cooling system on the RPV wall material.

On the basis of the performed calculations it can be concluded :

1. Filling the cold water from the EHPIPs leads to the unfavourable effects of low coolant temperature on the RPV wall during loss of coolant in all analyzed variants. Three basic effects exist as follows :
    a) Temperature waves with amplitude more than 100°C at the RPV inlet nozzles as a result of periodical forcing the cold water accumulated in the cold leg loop seals. Cold water is here accumulated as a result of the EHPIP operation connected to the PCP suction side.
    b) Creation of the cold tongues (i.e. unevenness of temperature field) on the RCSC walls as a result of the EHPIP operation connected to the PCP delivery side.
    c) Long-term PC cooldown with the possibility of non-conformance with the permissible values of pressure and temperature.
2. As expected the most unfavourable variant is the case of small leakages regarding the high PC pressure during transient process and from the point of view of thermal shock occurrence at high pressure and of total PC subcooling.
3. Calculational results of flow rates, temperatures and pressures in the PC can be used as boundary conditions for evaluation of unfavourable effects of cold water on the RPV material.

4. While managing the accident it is necessary to exclude inadmissibly deep temperature decrease at the PC pressure above 3.5 MPa. For the first approximation, as inadmissibly low temperature can be considered the temperature of brittle failure (fracture). More correct value can be obtained from detailed analysis of influence of cold water on the RPV material.
5. In order to maintain the PC coolant pressure and temperature in the permissible limits, it is possible to use the opening and closing the PZR relief valve, eventually the drainage steam line into the PZR bubbling condenser, switching-on and switching-off of the EHPIPs. These actions are not necessary within the time shorter than 30 minutes after the accident rise. It is necessary to take into account, that the PZR bubbling condenser diaphragm will likely be broken by overpressure. In case of small-break LOCAs the criterion for operational staff intervention should be maintaining the PC coolant subcooled condition (with certain margin), maintaining the PZR water level within controllable limits and exclusion of water temperature decrease at the RPV intel under the brittle fracture temperature limit. Problem of temperature waves can be eliminated by switching-off the EHPIPs connected to the PCP suction side.

But for ultimate conclusions it is necessary to perform a strength analysis respecting the fracture mechanics principles by using the presented result as an input data.

2.2. Large steam leakages from the SC

2.2.1. Characteristics of the process

Complete steam pipeline or the MSH break is assumed in the calculational analyses. After a leakage opening creation the steam, eventually water-steam mixture flows out from both ends of the break by critical flow rate. Thermal power of one or few SGs will suddenly increase, while the heat removal from the PC will be intensified. Primary coolant temperature and pressure decrease cause decrease of primary coolant volume, what results in flow out of water from PZR and in decrease of water level in the PZR. Pressure and water level decrease in the PZR initiates the start of the EHPIPs, which supply cold water (~ 55°C) into the PC in dependence on the hydraulic characteristics. As a result of the EHPIP operation and operation of systems which localize the leakage and decrease the heat removal from the PC (closing the FAVs on the steam pipelines, loss of water feeding into the SGs, decrease of the reactor power), the PC pressure decrease will be stopped, then it will begin to increase and successively the values near the EHPIP characteristics closing point (~ 13 MPa) will be achieved. After some combinations of failures the PC pressure can increase up to the value of the PZR blow-off valve opening pressure (~ 14 MPa). After reaching the closing point the cold water supply into the PC will be stopped. Because the PC is tight, quantity of suppplied water into the PC is considerably lower therm during LOCA, when the EHPIPs operate continuously.

Automatic localization logic of the steam leakage at the NPP-V1 will not cause a loss of the PCPs and the cold tongues as well as temperature waves rise do not threaten. Therefore, in order to achieve loss of all PCPs besides the accident itself we must also assume loss of internal consumption supply. (At partial loss of the PCPs, water flows in the PCLs with lost PCPs in the reversal direction and pulls down also water from the EHPIPs). If all PCPs fail, the primary coolant in the PCLs flows only by NC effect, while namely reactor residual power represents the "driving" force. The reactor residual power has minimal values in case, when it was long-tern shutdown and then approached the critical state and low power level (approximately 0.1% $N_{nominal}$).

2.2.2. Definition of initiating event and initial conditions

Many calculational variants of accidents with a steam pipeline
or the MSH break were performed in the frame of the NPP-V1 safety evaluation. We will mention here the calculation of steam

pipeline break at nominal as well as at null reactor power, assuming parallel loss of internal consumption supply and failure (no closing of armatures on the feedwater pipeline into the SG with broken steam pipeline. The calculations were performed using the code DYNAMIKA-5 [3].

At the accident rise moment we assumed : reactor power 102% of nominal value, unit operation with 6 PCPs. power grid fregnency 48 Hz, corresponding decrease of flow rate through the reactor.

After loss of the internal consumption supply the REP-1 is actuated, two PCPs are immediatly lost, four PCPs coastdown together with the TGs and all MFWPs are tripped. Drives of armatures will remain without supply till the DG start. Input data on the NPP-V1 second unit were used for the calculation [1].

The NPP unit condition at null power : reactor power 0.1% of nominal value,6 PCPs in operation, PZR pressure 11.3 MPa, coolant temperature in the lower mixing chamber 258°C, MSH pressure 3.9 MPa, TGs are not connected to the MSH, MFWPs are not operating. All PCPs are tripped after loss of the internal consumption supply.

2.2.3. Calculation results

The aim of calculations was to obtain information on time dependence of primary coolant pressures and temperatures at the RPV inlet during the accident course. Calculation results of two variants will be given :

PNC - break of steam pipeline from the SG No.4 in the hermetical area at nominal power; failure (no closing) of armatures on the feedwater line into the SG No.4 is assumed; loss of internal consumption supply; start of four EHPIPs and 1 AFWP.

POC - break of steam pipeline from the SG No.4 inside the hermetical area at null reactor power; loss of internal consumption supply; no closing of armatures on the feedwater pipeline from the AFWP; start of four EHPIPs and 1 AFWP.

Calculation results are given in Fig. 15 - 19 and 20 - 24, respectively.

Initiating event is the complete break of steam pipeline from the SG No.4, which occurs at 20th sec. Opening time of leakage sections is 0.2 sec. Steam, eventualy steam-water mixture flows out from both ends of the break, while both flows are mutually indepedent. Initial accident phase is very violent, actuation of control, protection and safety systems occurs. Signal on the loss of internal consumption supply initiates shutdown of both TGs as well as the REP-1.

Reactor power decreases quickly down to the reactor residual power level (calculation of its time course according the ANS formula). Primary coolant flow rate through the reactor depends on the PCP operation. Quick decrease of reactor power and increased heat transfer from the PC into the SC have considerable effect on the PC coolant temperature change. During first phase of the accident the most important heat removal from the PC is through the SG No.4., which results in coolant temperature decrease at the SG No.4. outlet. After water-steam mixture flow out from the SG No.4. the heat removal will suddenly decrease and outlet temperature will temporarily increase. After water level increase the heat removal from the PC will increase and coolant temperature at the end of 1 hour decrease to value of 188°C.

Changes of coolant temparature will influence the PC pressure and the PZR water level. The pressure will first quickly decrease to the value of 10.5 MPa, but after decreasing the heat removal from the PC and after starting the EHPIP its decrease will be stopped and then it will be increased up to the opening pressure of the PZR relief valves. The PZR water level decreases quickly down to the vaue of 2.2 m after the EHPIP start it will increase. During a short period after break of steam pipeline the saturated steam flows through the break from the SG No.4 side with maximum flow rate approximately 900 kg/s. Water-steam mixture in the SG No.4. will violently boil up, it will fill up whole volume of the SG No.4. and it will flow through the steam pipeline to the break. Maximum flow rate of water-steam mixture reaches appr. 300 kg/s. SG No.4. is quickly emptied, its level "will fall" and steam flows out again through the break, but flow rates are small. From the MSH side the saturated steam flows with minimal flow rate approximately 800 kg/s.

After stopping the steam leakage from the MSH side, the pressure decrease in all SGs (except SG No.4) will be stopped and the pressure increase up to values closely under the opening pressure of the SG safety valves. SG No.4 will be practically depressurized to atmospheric pressure.

After the AFWP start, the "tight" SGs are already pressurized and therefore we supposed that all feedwater from the AFWP flows into the SG No.4. Water level in it increases and heat removal is performed namely through this SG.

Time course of the process at null reactor power is qualitatively the same as at nominal power, but there are some differences here :

- regarding low content of steam under water level in the SG, the filling-up the failed SG by water-steam mixture does not occur after untightness rise
- MFWPs are not in operation, water level in failed SG is maintained by the AFWPs
- decrease of the SG pressures and the PC coolant temperatures is quicker because the reactor power is null
- immediate loss of all PCPs in the case of loss of internal consumption supply
- RC has low residual power and therefore the NC is very weak.

Calculational results are given in Fig. 20 to 24.

After loss of internal consumption supply the loss of all PCPs immediately occurs. Because the RC power is very low, the coolant flow rate in the PCLs and in the reactor is also small. Circulation is weak, namely in the unfailed PCLs, where the flow will be practically stopped from appr. 1000th sec. Flow is maintained through failed PCL only.

After break of steam pipeline, approximately 800 kg/s of saturated steam flows out through the break. Filling-up the SG No.4 by water-steam mixture does not occur, therefore its emptying lasts longer. Coolant temperature at the SG No.4 outlet suddenly decreases until the substantial part of heat transfer surface will be uncovered. Decrease of temperatures in other PCLs is more slower.

The PZP pressure and water level decrease as a result of increased heat removal from the PC. Four EHPIPs start at 666th sec. and pressure and water level begin to increase. The EHPIPs operation supports circulation in the PCLs, but after reaching the closing point (at about 1000th sec.) the water supply into the PC will be stopped.

Inflow of cold water (appr. 55°C) from the EHPIPs into the PCLs with small flow rate will cause sudden water cooldown at the RPV inlet, what represents a considerable shock on the material. Minimal temperature is about 98°C (at the end of one hour), but the temperature can fall even lower unless some measures are undertaken.

2.2.4. Conclusions resulted from analyses

Calculational results confirmed, that measures for the NPP-V1 safety enhancing, performed in the frame of reconstruction have considerably increased resistance of the NPP units against simple failure, especially on the side of feedwater. In order to verify thermal and mechanical stresses of the RPV, such a kind of assumptions was accepted, that they exceed the simple failure principle, while the operational staff intervention into was not considered.

Quantity of supplier cold water will not cause temperature waves at the RPV inlet because the PC remains tight.

If stagnating flow will be set in some PCL (low residual reactor power), the EHPIP, operation can cause creation of cold tongues on the RPV wall. If no measures are undertaken to stop cooldown, the temperature of cold tongue at the RPV inlet can fall down to the values approximately of 60°C. Temperature of cold tongue moves about 100°C at the time of one hour after beginning the accident.

3. LITERATURE

[1] J.Klepáč, J.Mišák, M.Liptáková: Set of Input Data for Innovation of the NPP-V1 Units Safety Analyses.

[2] L.E.Steele: Status of USA Nuclear Reactor Pressure Vessel Surveillance for Radiation Effects, ASTM PCN 04-784000-35.

[3] Calculational Code DYNAMIKA-5. Calculation of Transients of NPPs with WWER. Manual, 8624606, 00306-018101.

[4] P.Matejovič, Z.Bazsó, Ľ.Vranka: RELAP5/MOD2 analyses of the overcooling of the primary circuit in NPP V1 during small and middle LOCAs. NPPRI report 61/92.

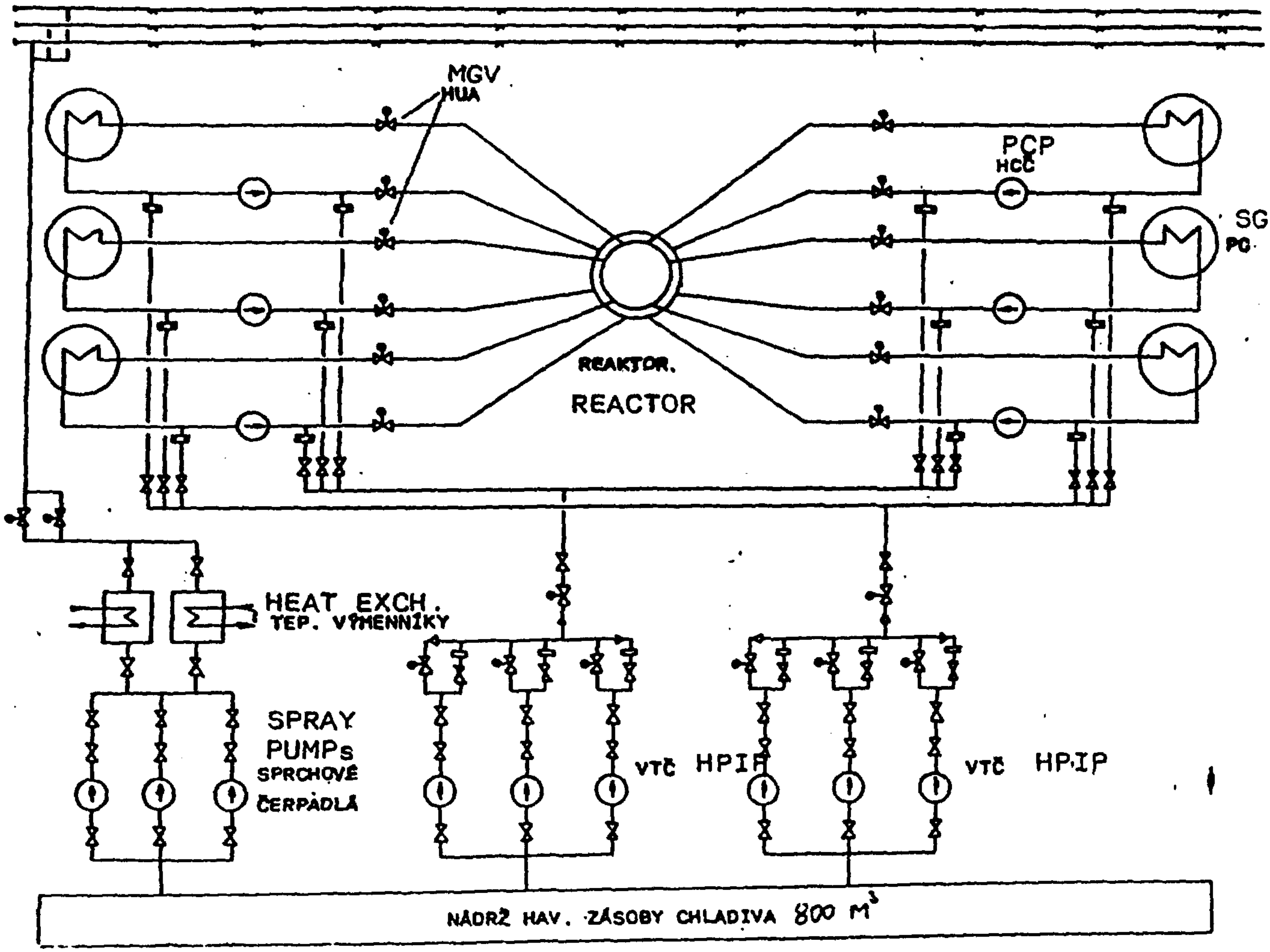
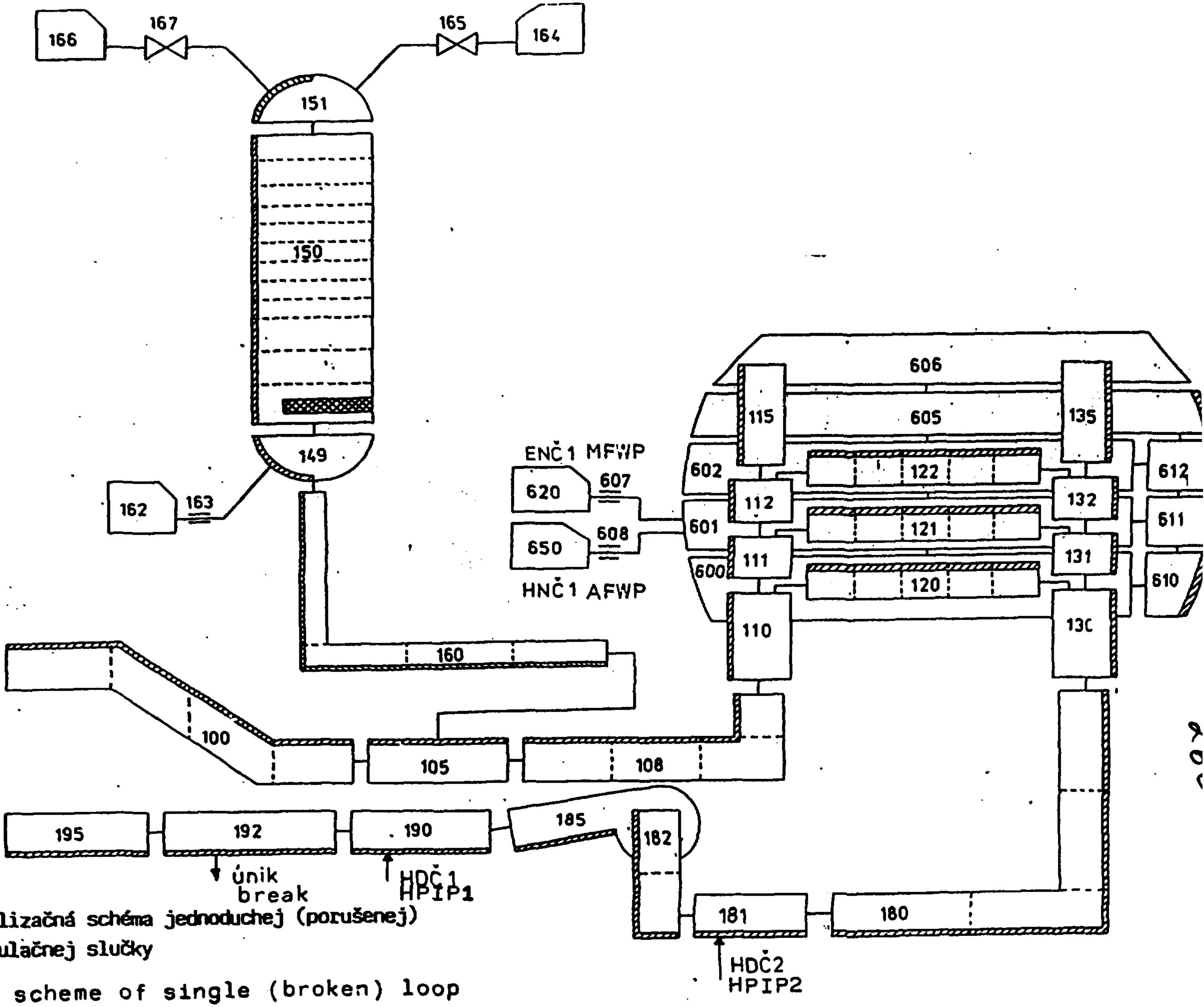


Schéma systému havarijného doplňovania primárneho okruhu a sprchového systému

Fig.a. Scheme of ECCS and spray system in NPP-V1



Nodalizačná schéma jednoduchej (porušenej) cirkulačnej slučky

Fig.b. Nodalization scheme of single (broken) loop with pressurizer

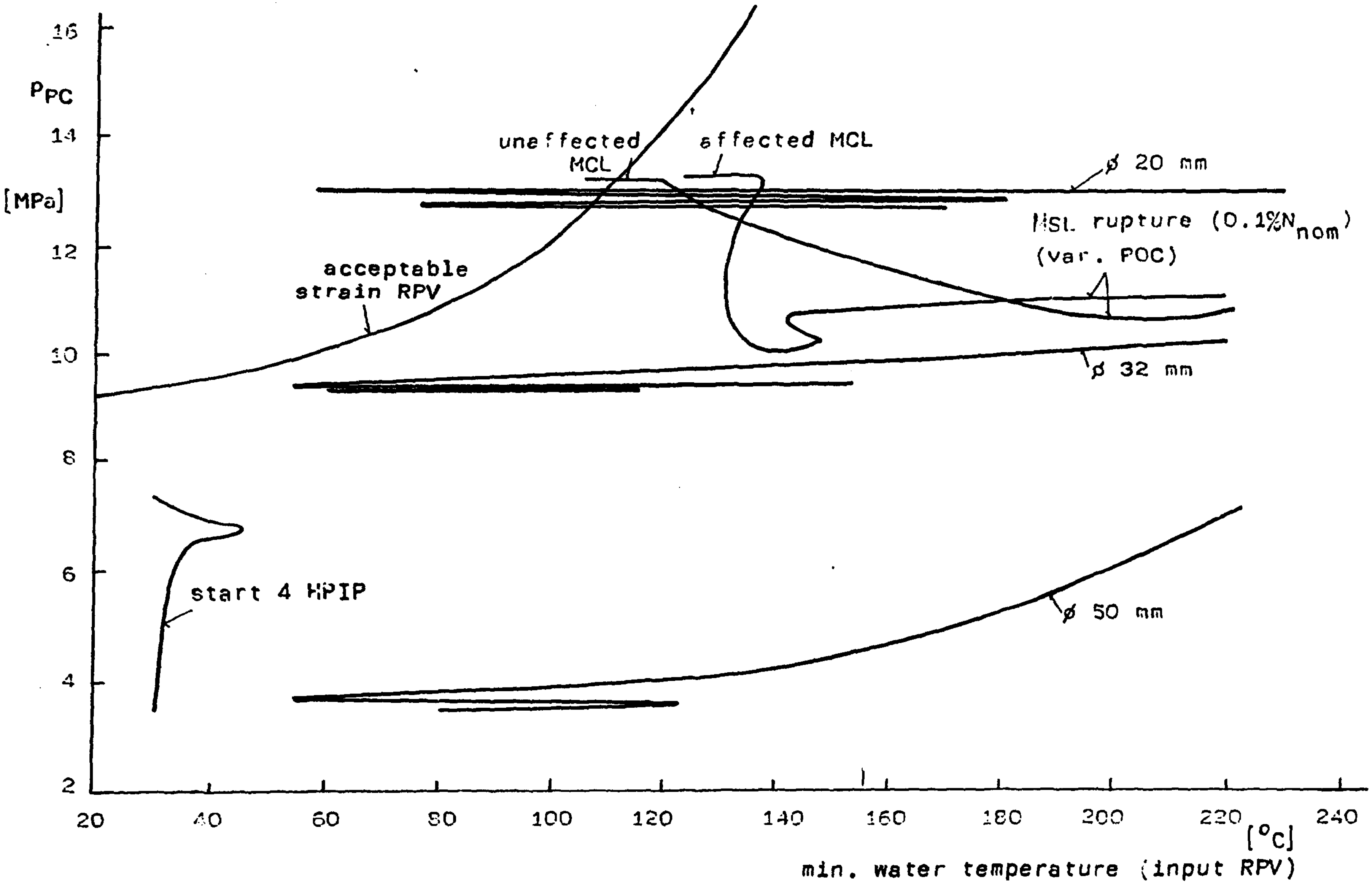


Fig. C : Pressure and coolant temperature

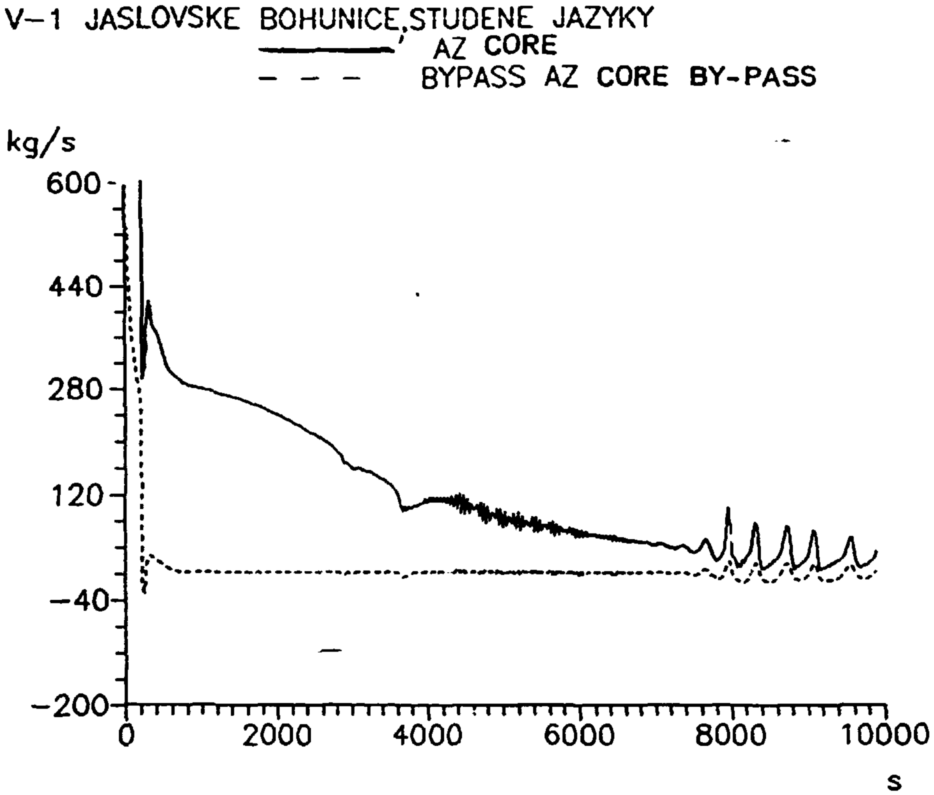


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Fig.1. CORE AND BY PASS MASS FLOW RATE

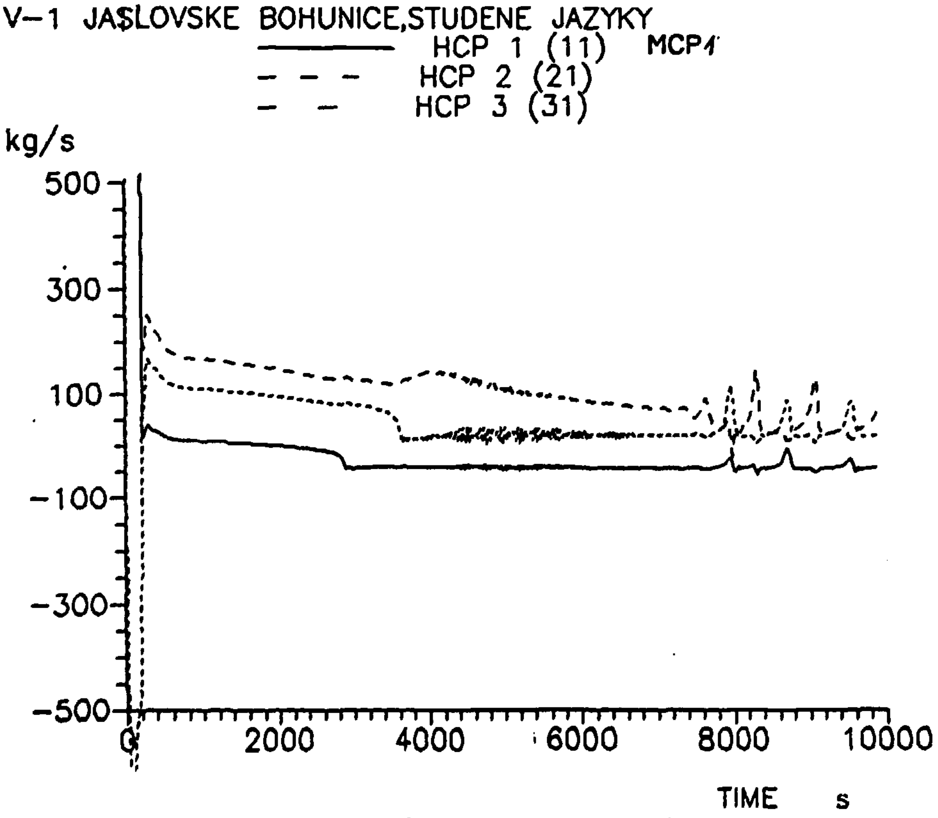

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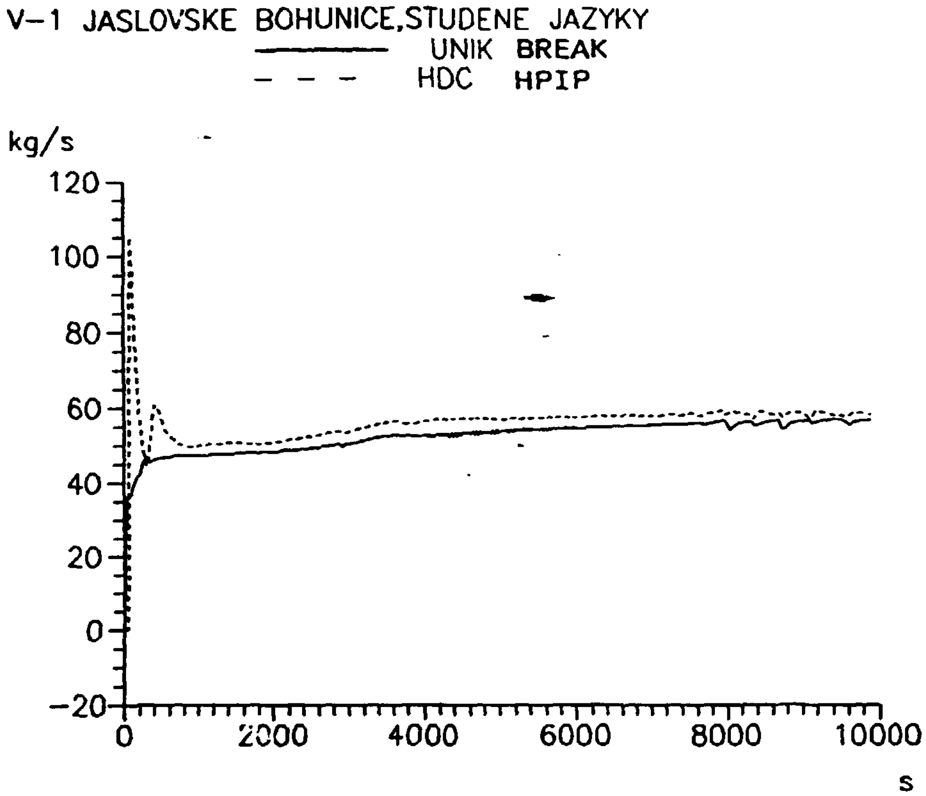
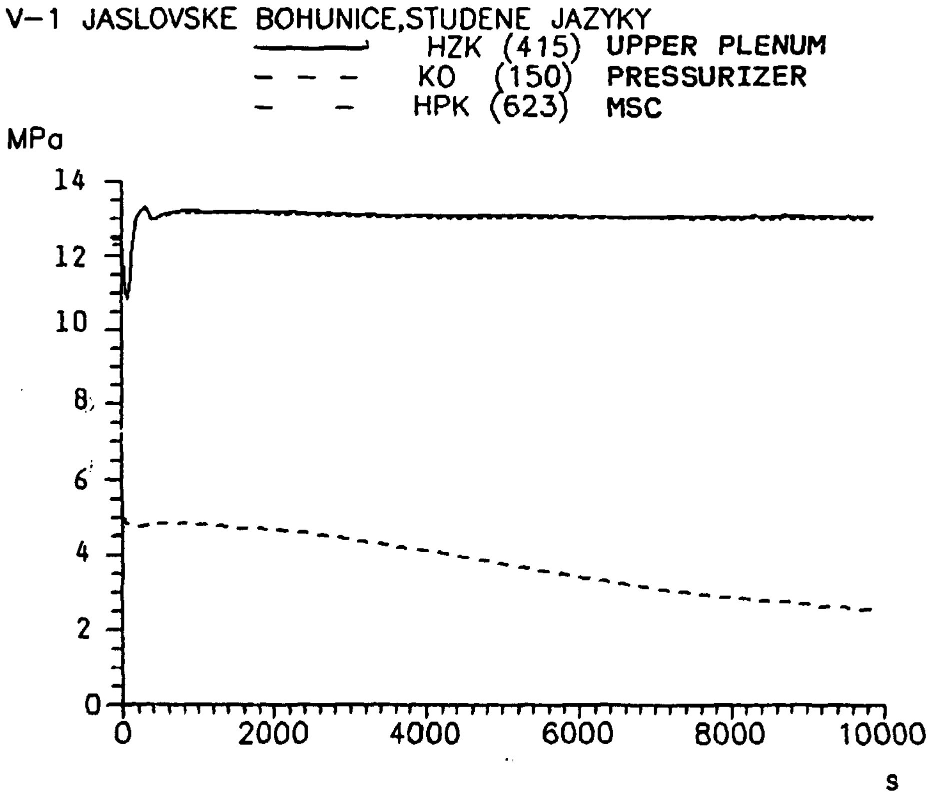
FIG.2. REACTOR VESSEL INLET MASS FLOW RATE



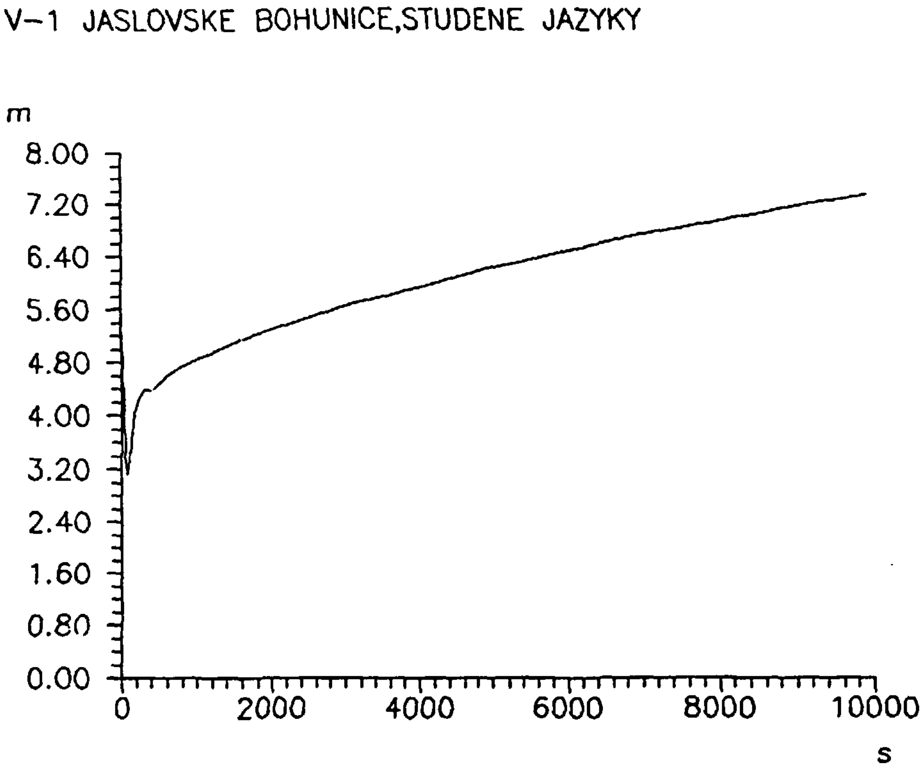
CELKOVY HMOTNOSTNY PRIETOK UNIKOVYM OTVOROM A Z HDC

FIG.3.TOTAL BREAK AND HPIP MASS FLOW RATE



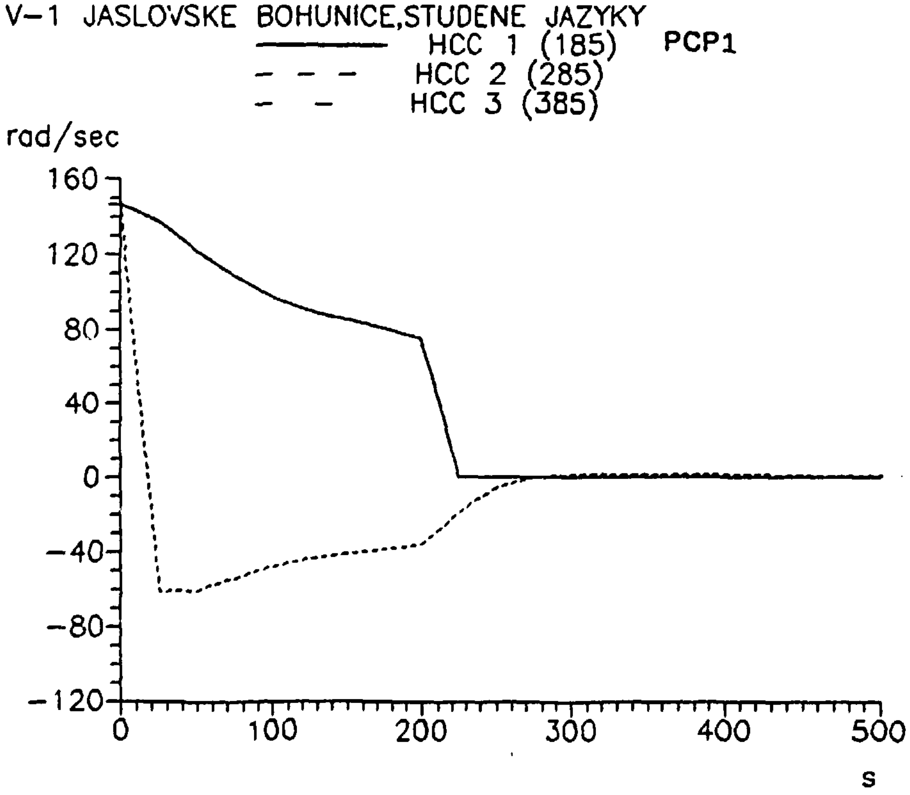
TLAK V HZK,KO A HPK

FIG.4. UPPER PLENUM, PRESSURIZER AND MSC PRESSURE



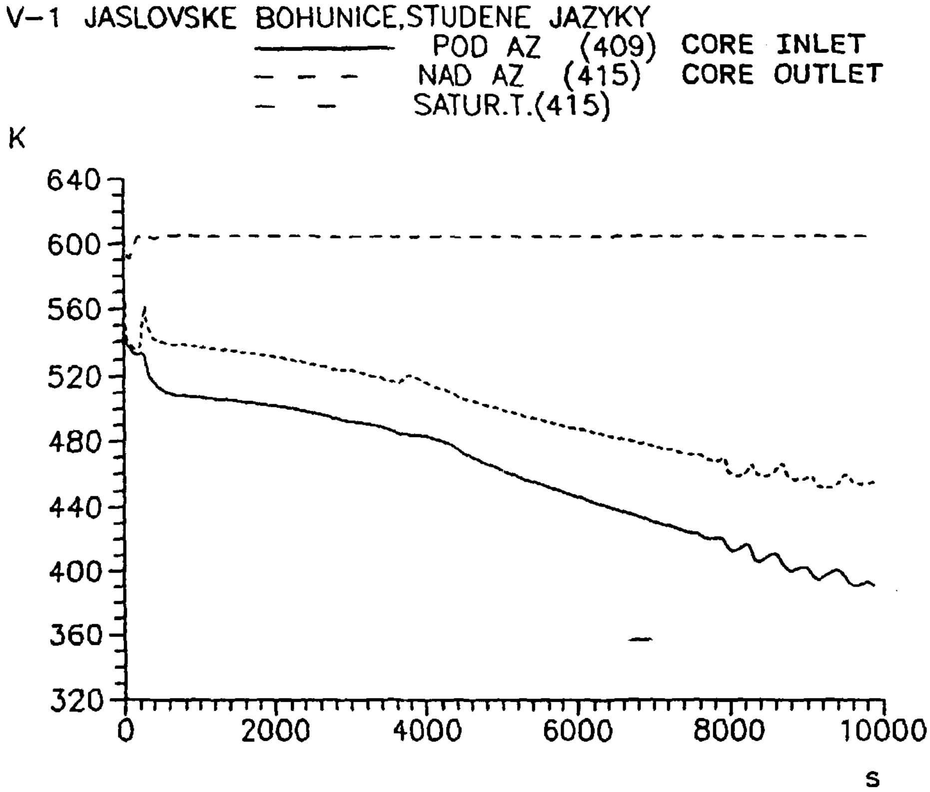
VODNA HLADINA V KO

OBR 5. PRESSURIZER LEVEL



OTACKY HCC

FIG.6. PUMP SPEED



TEPLOTA VODY V OBJEMOCH POD A NAD AZ

FIG.7. CORE INLET AND OUTLET TEMPERATURE

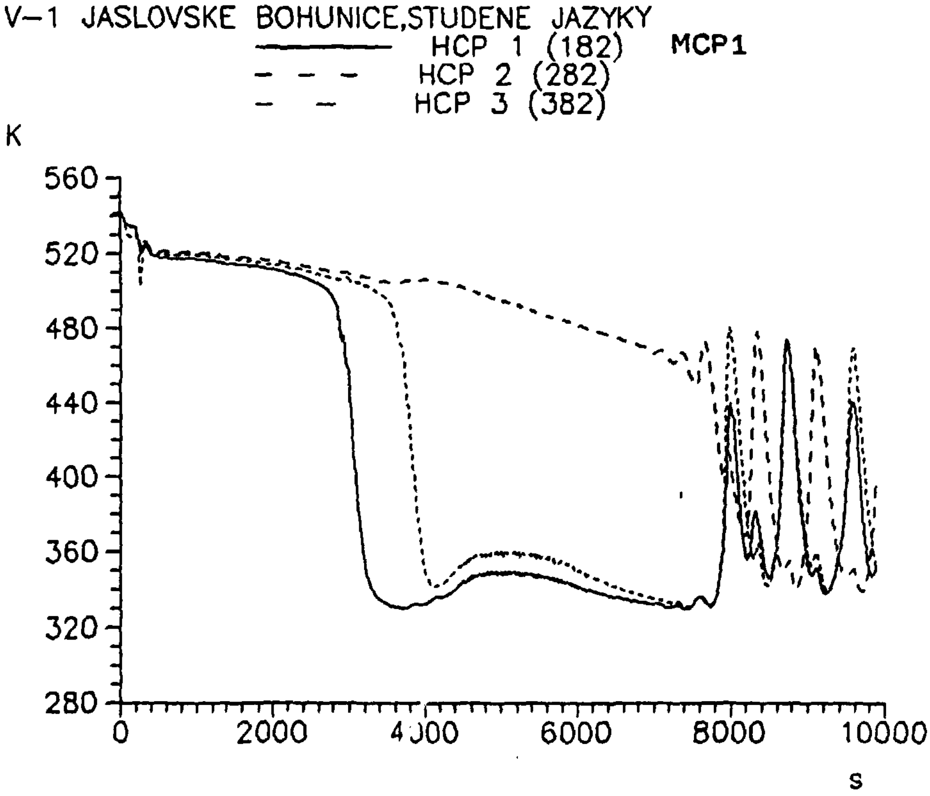


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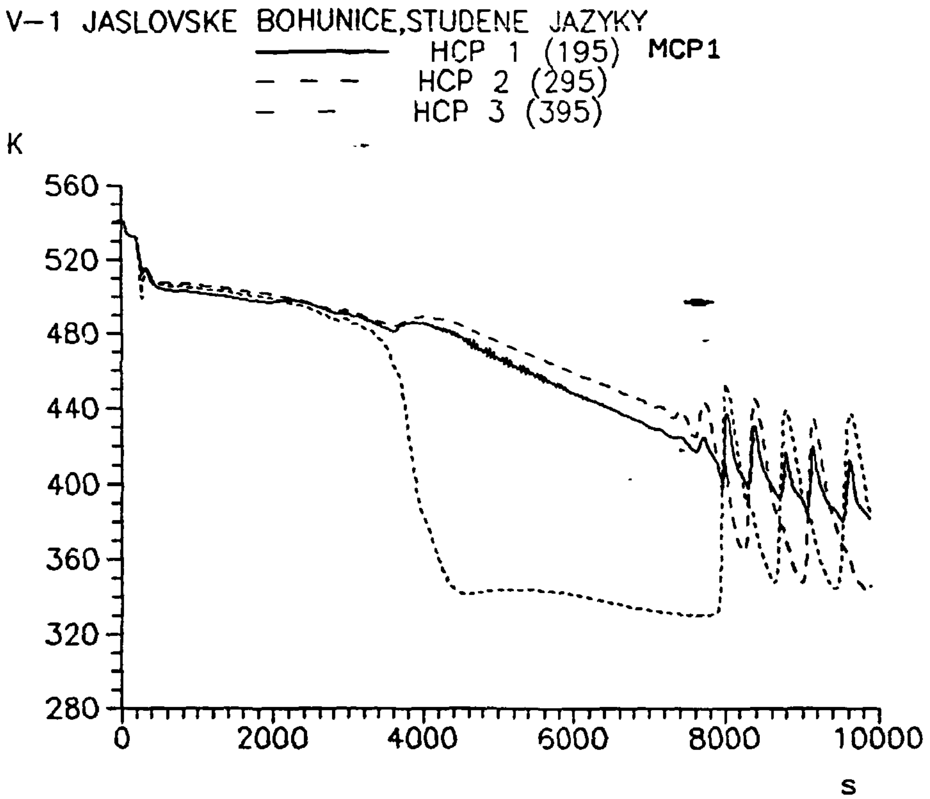
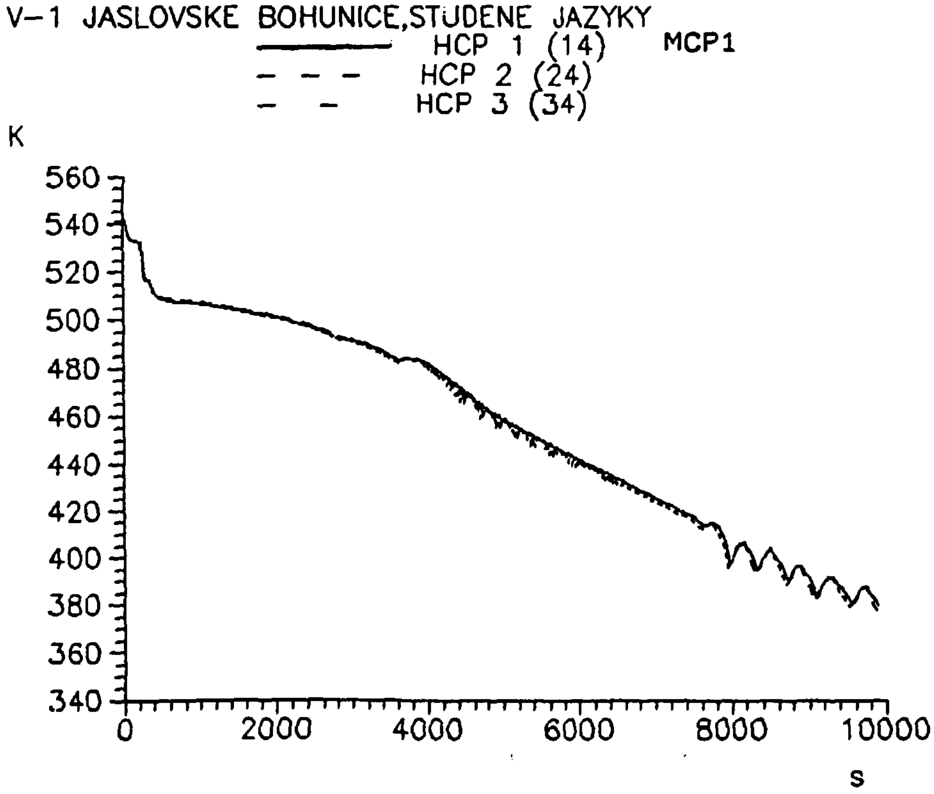
FIG.8. COOLANT TEMPERATURE AT LOOP SEALS



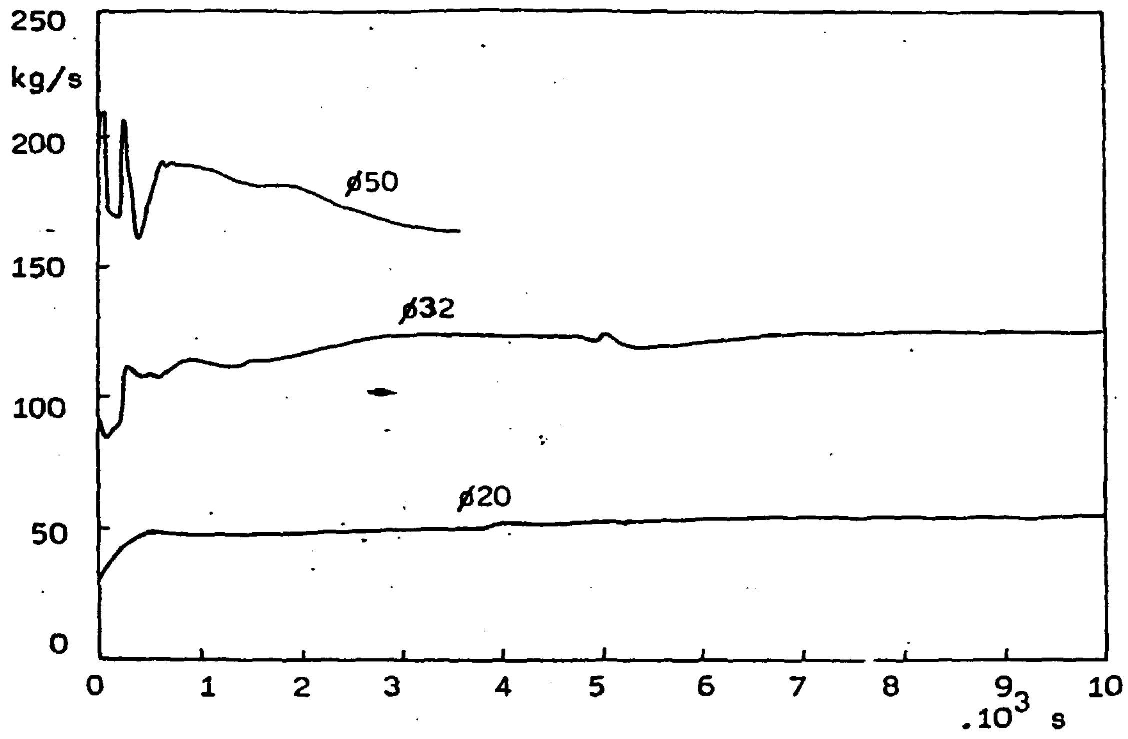
TEPLOTA VODY VO VSTUPNYCH NATRUBKOCH RN

FIG.9. COOLANT TEMPERATURE AT RPV INLET NOZZLES



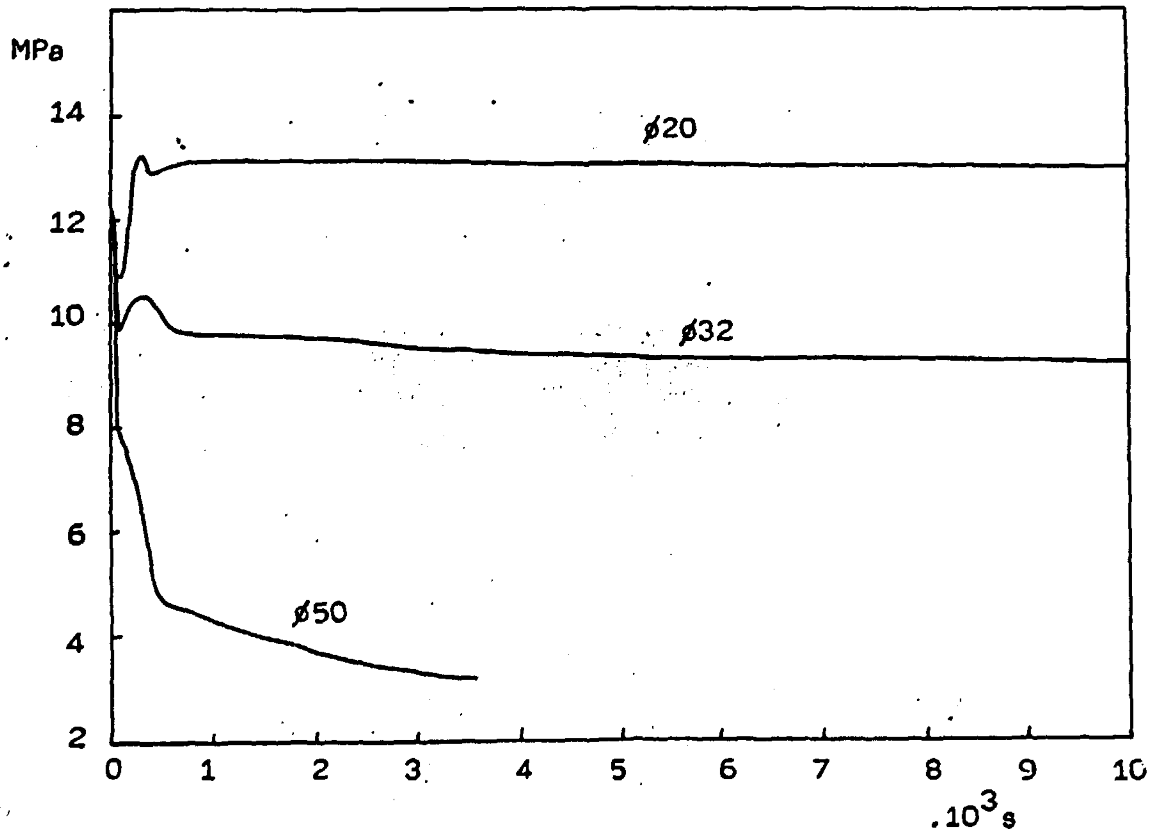
TEPLOTA VODY V SACHTE REAKTORA NA UROVNI KRIT.ZVARU

FIG.10 COOLANT TEMPERATURE IN DC AT CRITICAL WELD ELEVATION



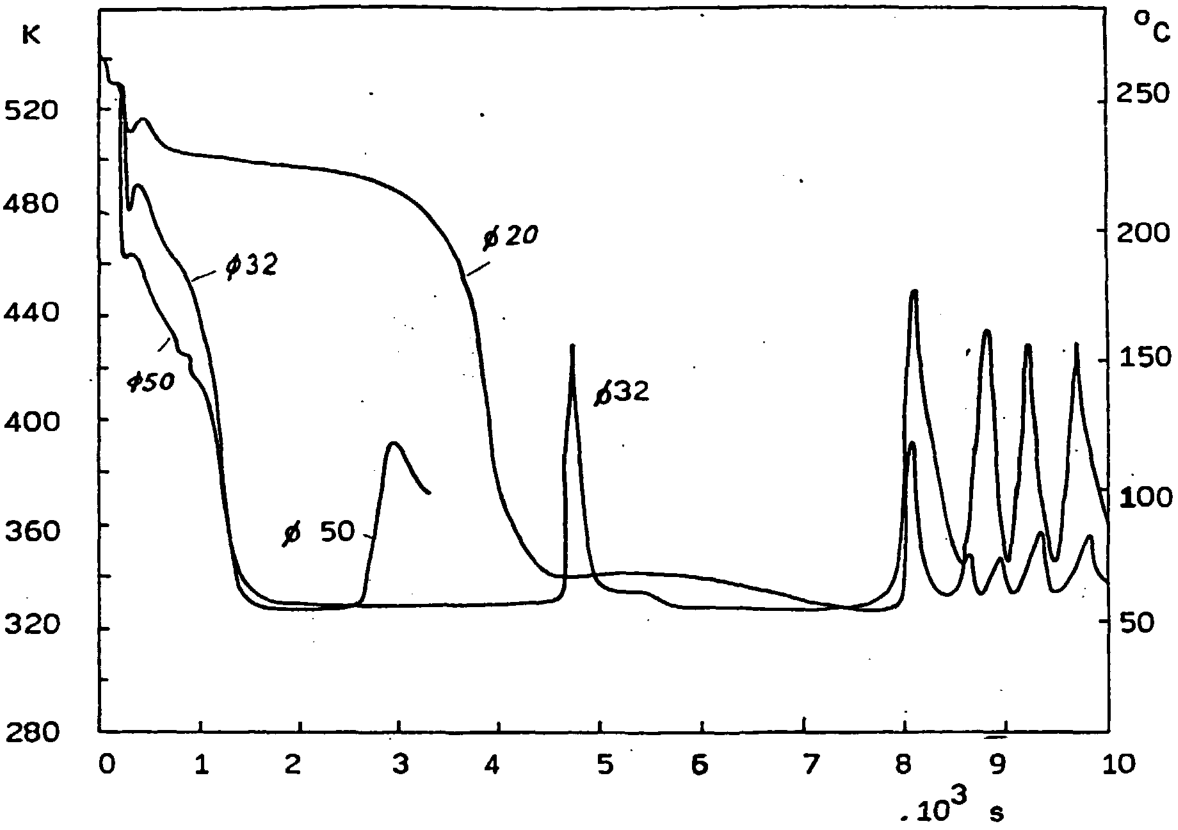
Celkový hmotnostný prietok únikovým otvorom

FIG.11. TOTAL BREAK MASS FLOW RATE



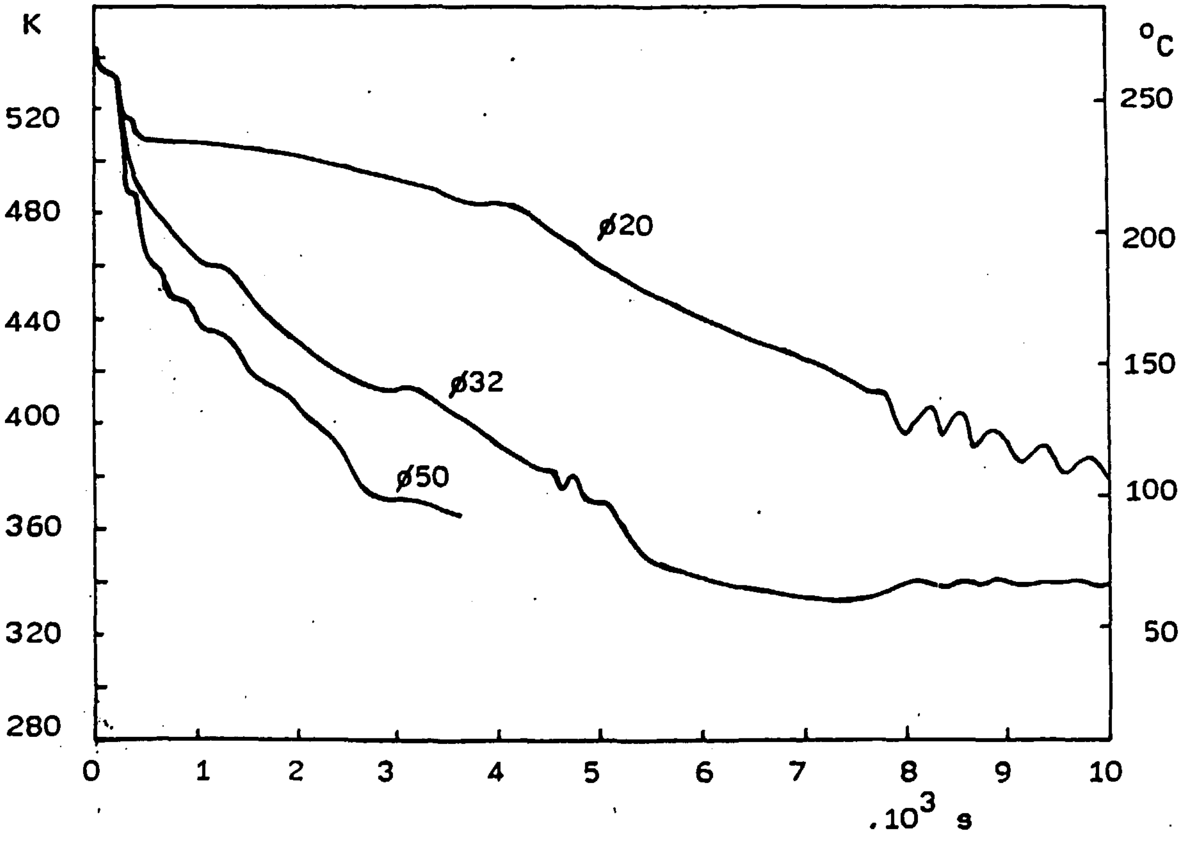
Tlak v HZK

FIG.12. UPPER PLENUM PRESSURE



Teplota vody vo vstupných nátrubkoch TRN

FIG.13. COOLANT TEMPERATURE AT RPV INLET NOZZLESS

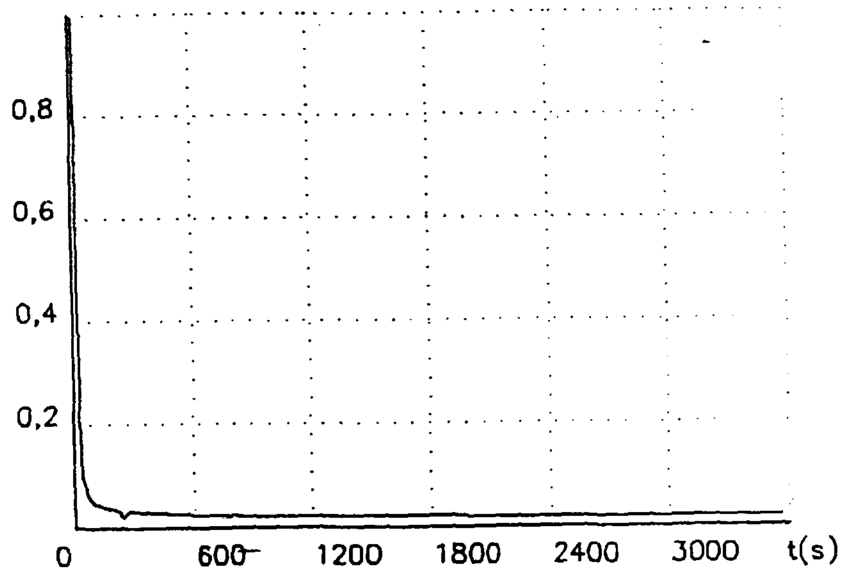


Teplota vody v šachte reaktora na úrovni krit.zvaru

FIG.14. COOLANT TEMPERATURE AT PC AT CRITICAL WELD ELEVATION

FIG.15.

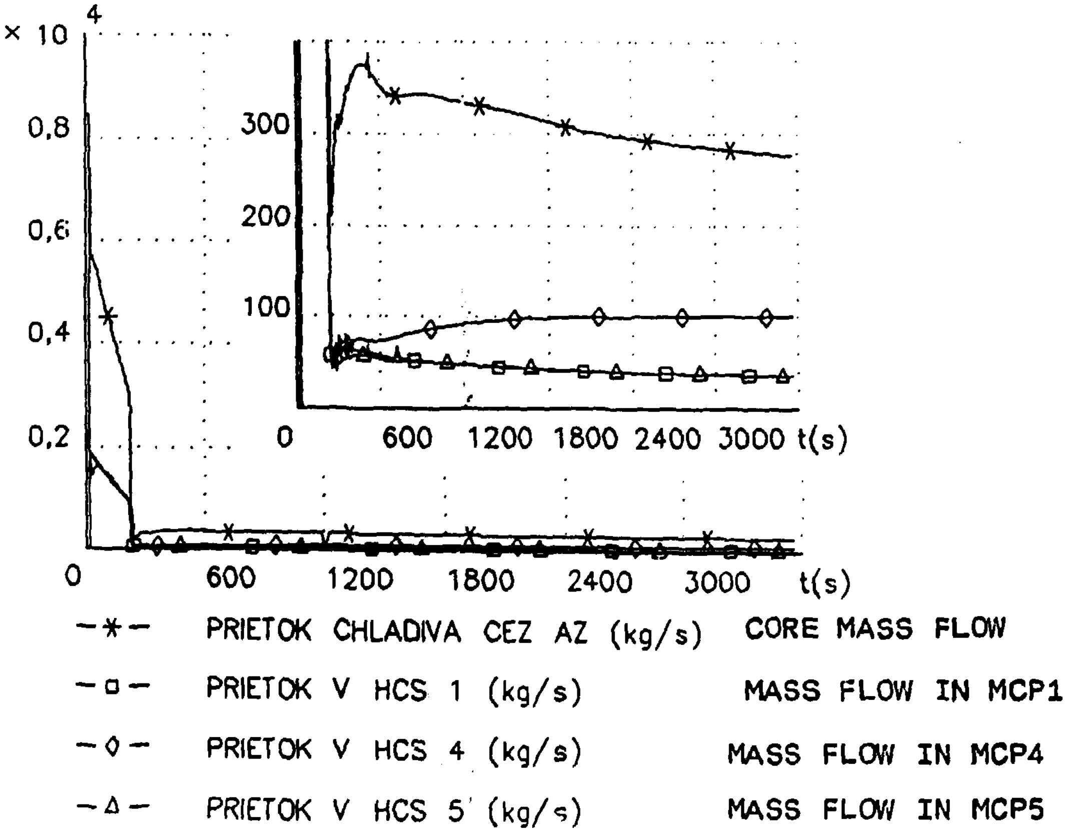
PNC44 :PAR;Nnom;SNVS;RBA4+RAH4 N.;1 HNC;4 HDC;



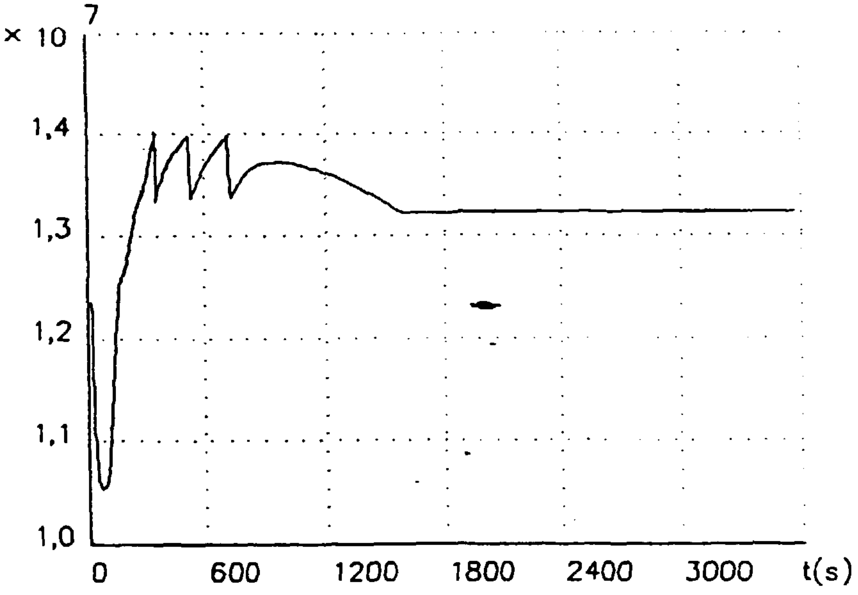
RELATIVNY TEP. VYKON AZ (1)

RATED CORE THERMAL POWER

PNC44 :PAR;Nnom;SNVS;RBA4+RAH4 N.;1 HNC;4 HDC;



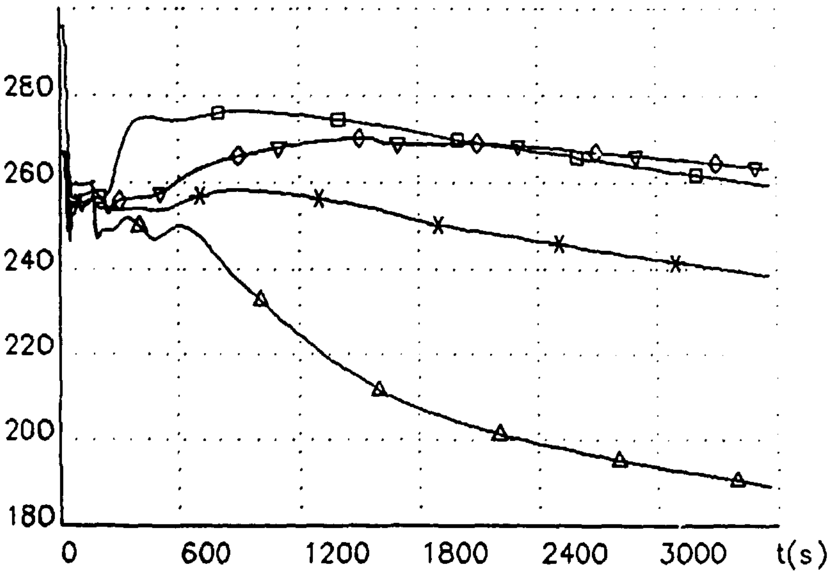
PNC44 :PAR;Nnom;SNVS;RBA4+RAH4 N.;1 HNC;4 HDC;



TLAK V HZK (Pa)

UPPER PLENUM PRESSURE

PNC44 :PAR;Nnom;SNVS;RBA4+RAH4 N.;1 HNC;4 HDC;



-✻- TEPLOTA CHLADIVA V DZK (oC) **LP TEMPERATURE**

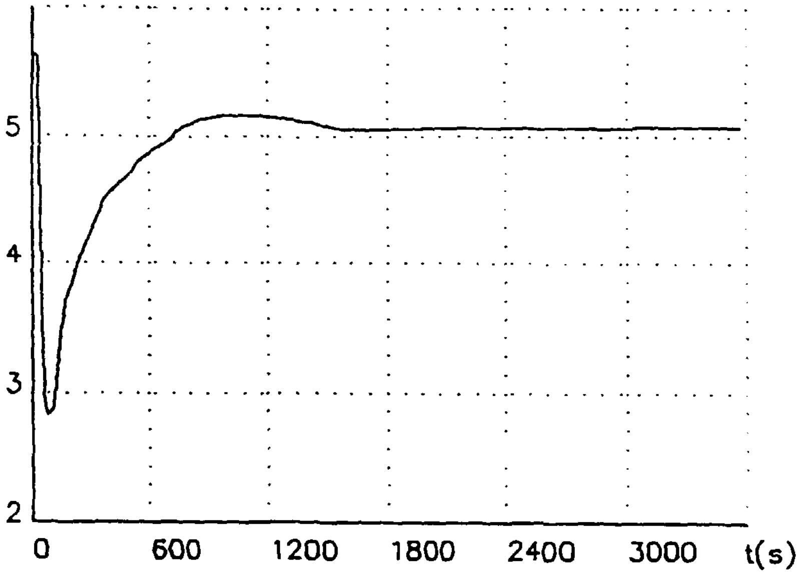
-□- TEPLOTA CHLADIVA V HZK (oC) **UP TEMPERATURE**

-◇- TEPLOTA VODY(HCS 1-VST. TRN) (oC) **RPV INLET (MCP1)TE**

-△- TEPLOTA VODY(HCS 4-VST. TRN) (oC) **RPV INLET (MCP4)TE**

-▽- TEPLOTA VODY(HCS 5-VST. TRN) (oC) **RPV INLET (MCP5)TE**

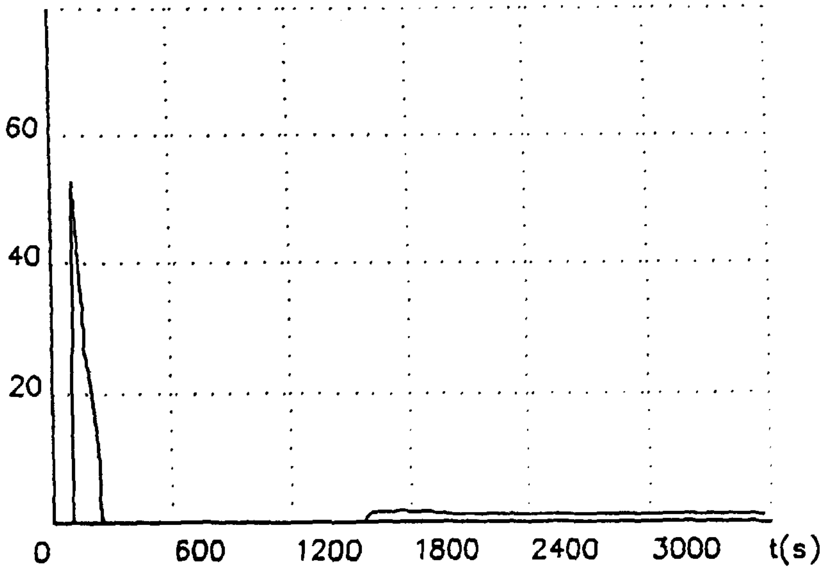
PNC44 :PAR;Nnom;SNVS;RBA4+RAH4 N.;1 HNC;4 HDC;



VYSKA HLADINY V KO (m)

PRESSURIZER LEVEL

PNC44 :PAR;Nnom;SNVS;RBA4+RAH4 N.;1 HNC;4 HDC;

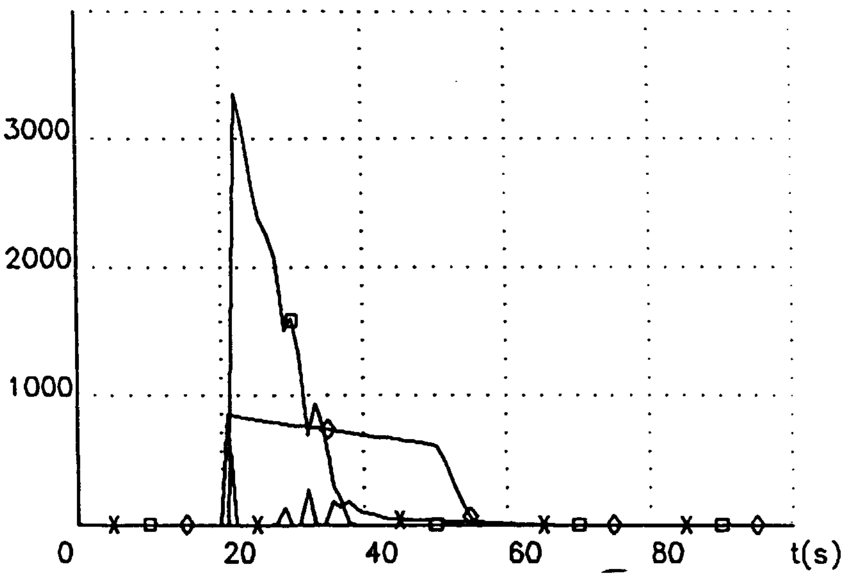


CELKOVY PRIETOK VODY OD SAOZ (kg/s)

TOTAL ECCS MASS FLOW

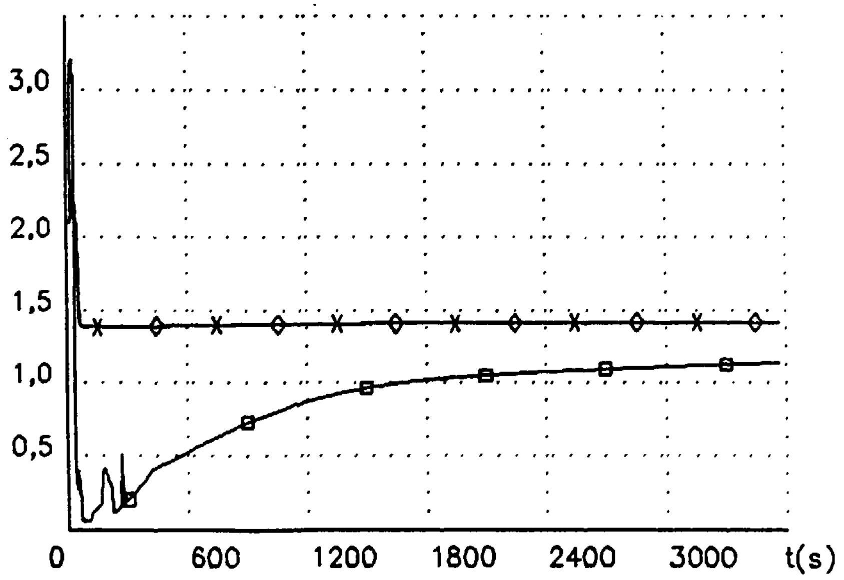
FIG 18.

PNC44 :PAR;Nnom:SNVS;RBA4+RAH4 N.;1 HNC;4 HDC;



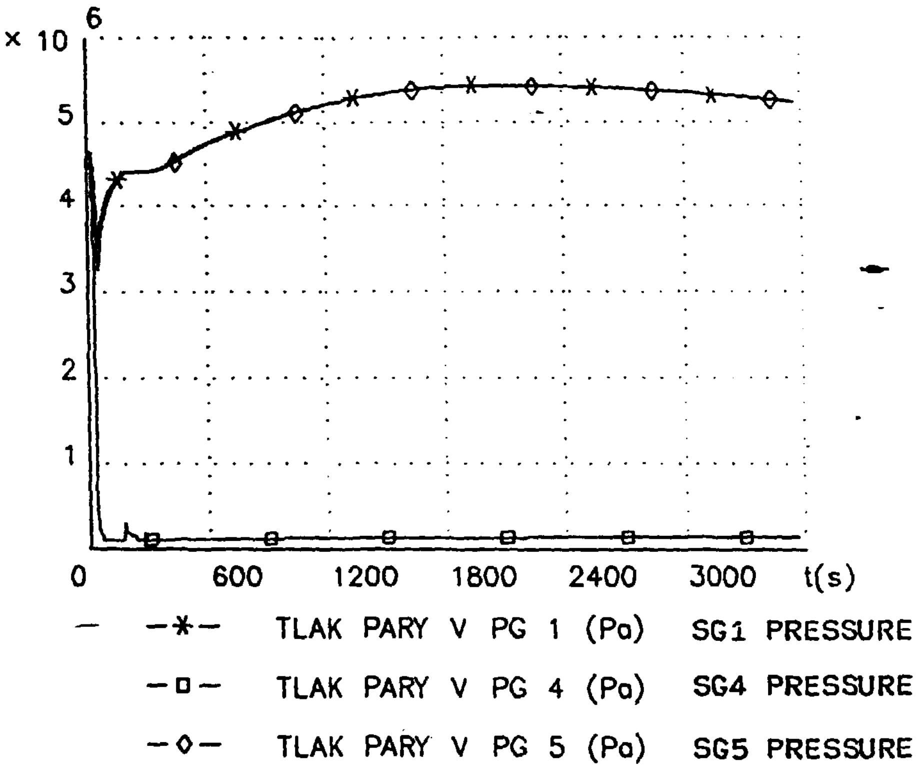
—*— PRIETOK PARY NET OD PG 4 (kg/s)
STEAM MASS FLOW FROM SG4
—□— PRIETOK ZMESI NET OD PG 4 (kg/s)
STEAM-WATER FLOW FROM SG4
—◇— PRIETOK PARY NET. OD HPK 2 (kg/s)
STEAM MASS FLOW FROM MSC2

PNC44 :PAR;Nnom;SNVS;RBA4+RAH4 N.;1 HNC;4 HDC;

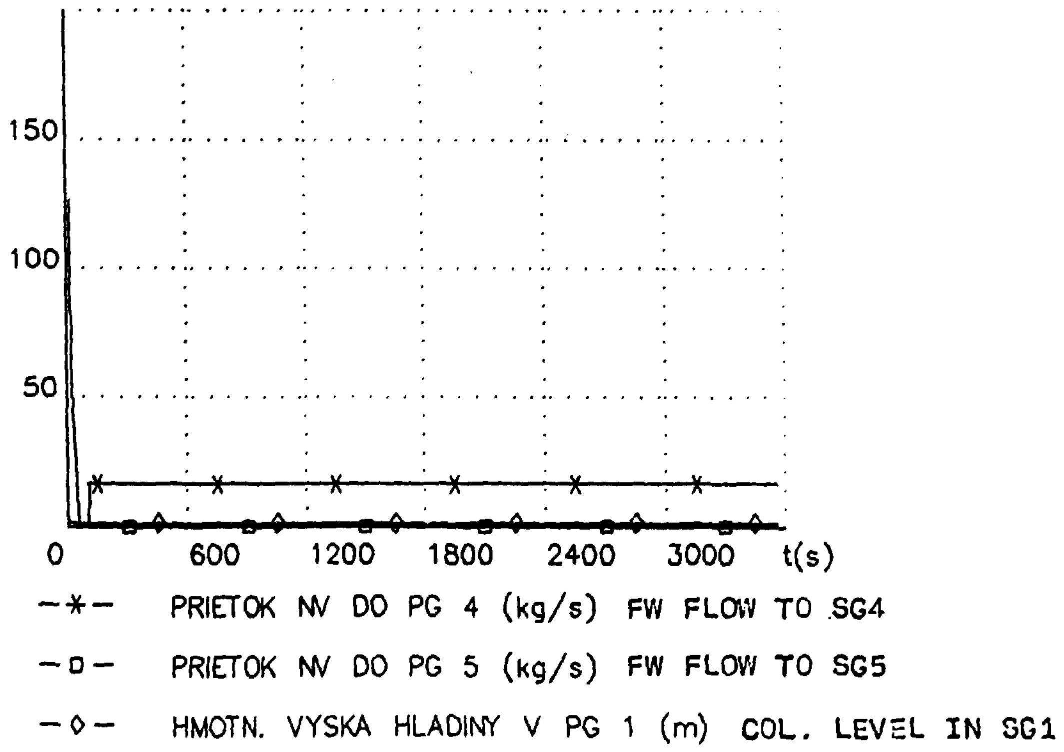


—*— VYSKA HLADINY V PG 1 (m) SG1 LEVEL
—□— VYSKA HLADINY V PG 4 (m) SG4 LEVEL
—◇— VYSKA HLADINY V PG 5 (m) SG5 LEVEL

FIG. 19.

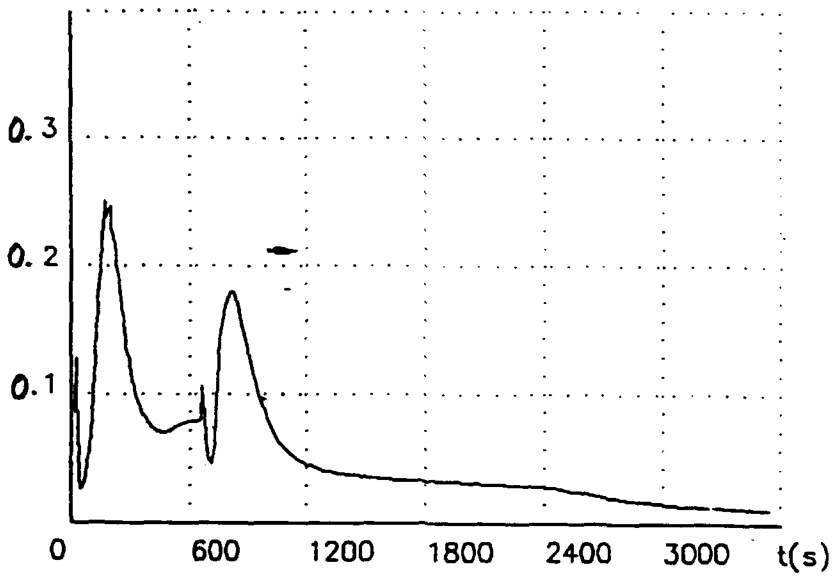


PNC44 :PAR;Nnom;SNVS;RBA4+RAH4 N.:1 HNC;4 HDC;



PNC44 :PAR;Nnom;SNVS;RBA4+RAH4 N.:1 HNC;4 HDC;

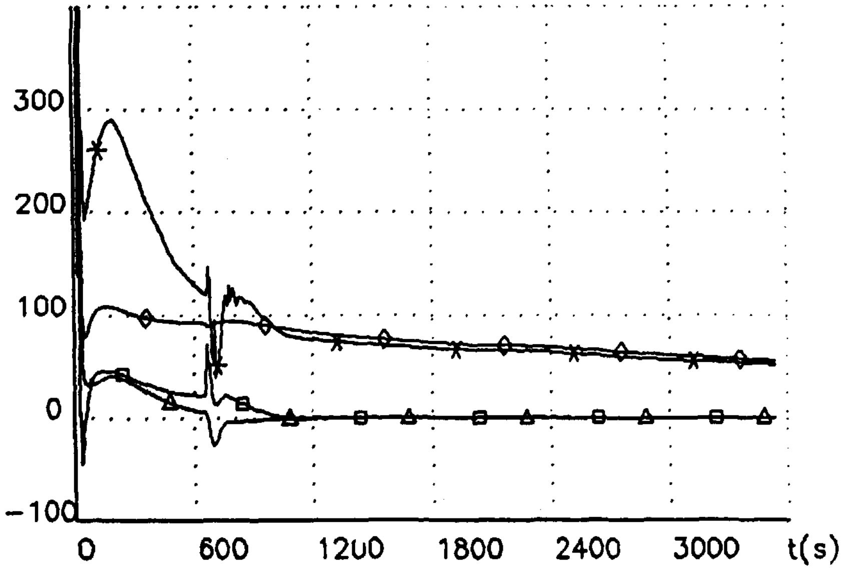
P0C44: PAR;MKV;6 HCC;1 HNC;SNVS;4HDC;



RELATIVNY TEP. VYKON AZ (%)

RATED CORE THERMAL POWER

P0C44: PAR;MKV;6 HCC;1 HNC;SNVS;4HDC;



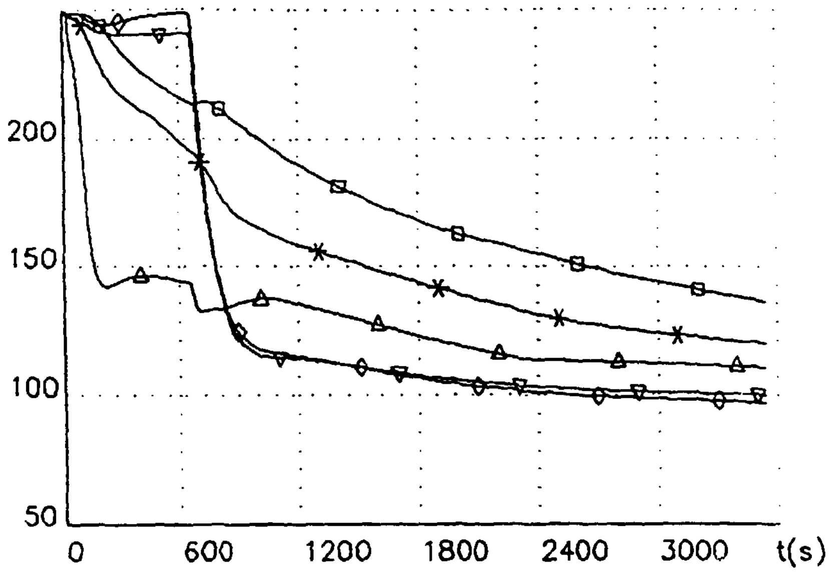
—*— PRIETOK CHLADIVA CEZ AZ (kg/s) CORE FLOW RATE

—□— PRIETOK V HCS 1 (kg/s) FLOW IN MCP1

—◇·– PRIETOK V HCS 4 (kg/s) FLOW IN MCP4

—△— PRIETOK V HCS 5 (kg/s) FLOW IN MCP5

POC44: PAR;MKV;6 HCC;1 HNC;SNVS;4HDC;



-*- TEPLOTA CHLADIVA V DZK (oC) LP TEMPERATURE

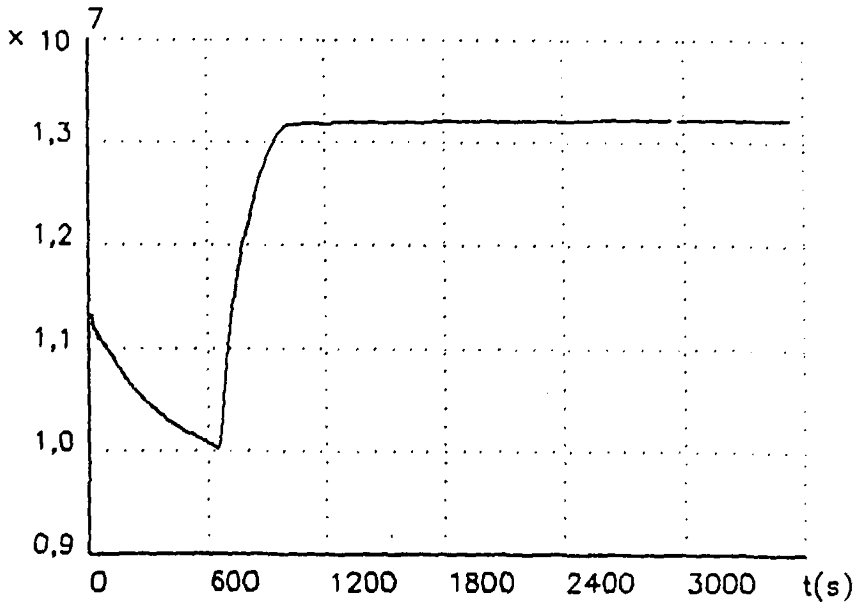
-□- TEPLOTA CHLADIVA V HZK (oC) UP TEMPERATURE

-◇- TEPLOTA VODY(HCS 1-VST. TRN) (oC) RPV INLET (MCP1) TEMP.

-△- TEPLOTA VODY(HCS 4-VST. TRN) (oC) RPV INLET (MCP4) TEMP.

-▽- TEPLOTA VODY(HCS 5-VST. TRN) (oC) RPV INLET (MCP5) TEMP.

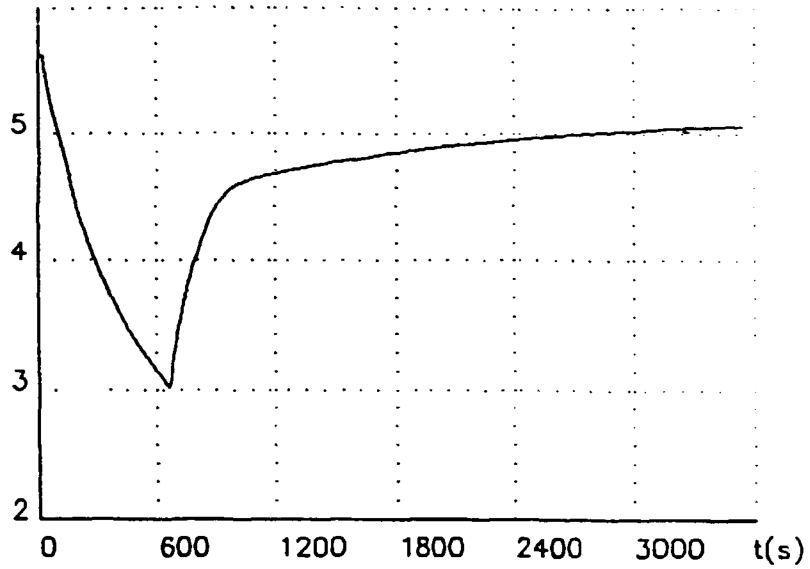
POC44: PAR;MKV;6 HCC;1 HNC;SNVS;4HDC;



TLAK V HZK (Pa)

UPPER PLENUM PRESSURE

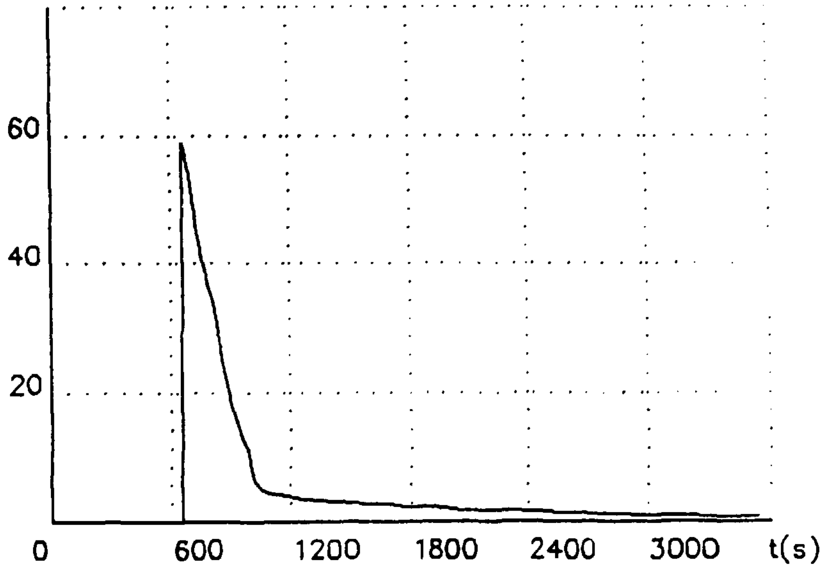
POC44: PAR;MKV;6 HCC;1 HNC;SNVS;4HDC;



VYSKA HLADINY V KO (m)

PRESSURIZER LEVEL

POC44: PAR;MKV;6 HCC;1 HNC;SNVS;4HDC;

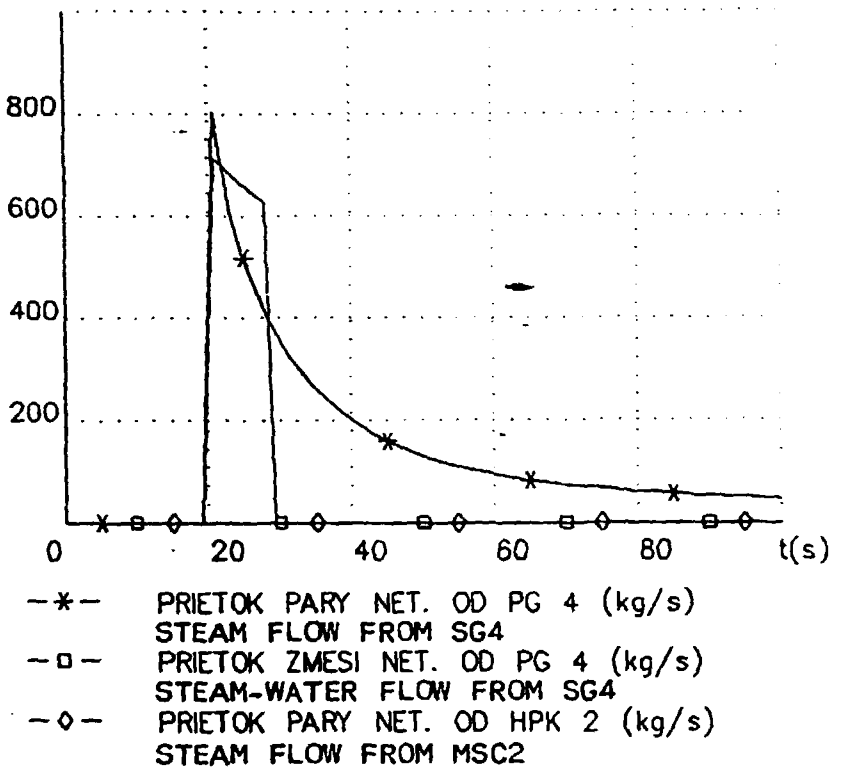


CELKOVY PRIETOK VODY OD SAOZ (kg/s)

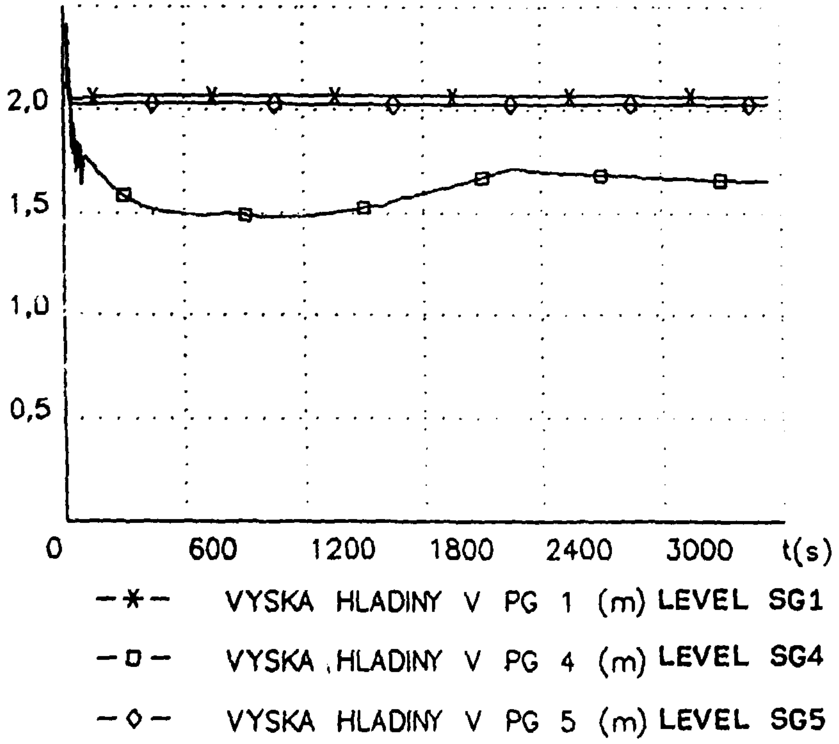
TOTAL ECCS MASS FLOW RATE

IG. 23

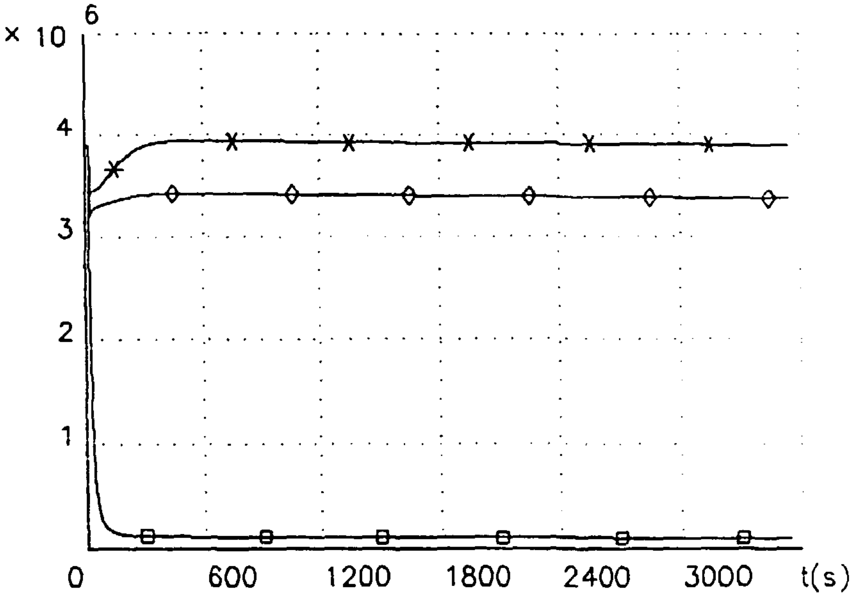
POC44: PAR;MKV;6 HCC;1 HNC;SNVS;4HDC;



POC44: PAR;MKV;6 HCC;1 HNC;SNVS;4HDC;



P0C44: PAR;MKV;6 HCC;1 HNC;SNVS;4HDC;

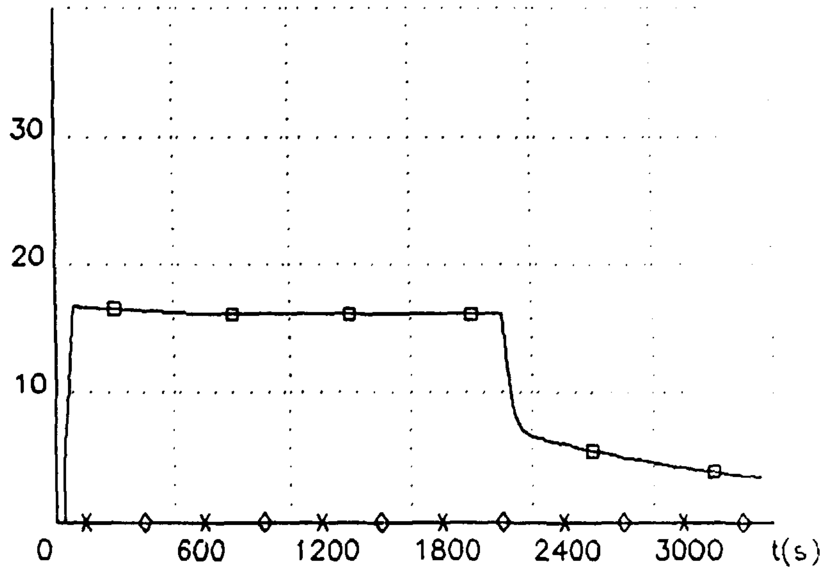


—✱— TLAK PARY V PG 1 (Pa) **SG1 PRESSURE**

—□— TLAK PARY V PG 4 (Pa) **SG4 PRESSURE**

—◇— TLAK PARY V PG 5 (Pa) **SG5 PRESSURE**

P0C44: PAR;MKV;6 HCC;1 HNC;SNVS;4HDC;



—✱— PRIETOK NV DO PG 1 (kg/s)
FW FLOW TO SG1

—□— PRIETOK NV DO PG 4 (kg/s)
FW FLOW TO SG4

—◇— PRIETOK NV DO PG 5 (kg/s)
FW FLOW TO SG5