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CODEX-3/1 AND CODEX-3/2 EXPERIMENTS: QUENCHING OF HIGH TEMPERATURE VVER BUNDLES

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CODEX-3/1 AND CODEX-3/2 EXPERIMENTS: QUENCHING OF HIGH TEMPERATURE VVER BUNDLES

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ABSTRACT

Z. Hózer, L. Maróti, I. Nagy, P. Windberg: CODEX-3/1 AND CODEX-3/2 EXPERIMENTS: QUENCHING OF HIGH TEMPERATURE VVER BUNDLES

Experimental investigation of the interaction of high temperature VVER bundle with cooling water under severe accident conditions was carried on the CODEX facility. The test section of the facility included a seven-rod hexagonal bundle with electrical heating. During the tests the bundle was heated up and quenched by cold water. Two experiments were carried out with the same bundle, but with different conditions: preoxidation with quenching at 1150 °C (CODEX-3/1) and quenching of the pre-oxidized bundle at 1500 °C (CODEX-3/2). The tests provided an unexpected result, as very limited temperature increase and hydrogen production was observed.

KIVONAT

Hózer Z., Maróti L., Nagy I., Windberg P.: A CODEX-3/1 ÉS CODEX-3/2 KÍSÉRLETEK: MAGAS HŐMÉRSÉKLETŰ VVER KÖTEGEK VÍZZEL TÖRTÉNŐ GYORS LEHŰTÉSE

A CODEX berendezésen magas hőmérsékletű VVER típusú fűtőelem kötegnek a hűtővízzel való kölcsönhatását vizsgáltuk súlyos baleseti körülmények között. A kísérleti berendezés legfontosabb része az elektromosan fűtött hét rúdból álló köteg. A mérések során a köteget felfűtöttük, majd hideg vízzel elárasztottuk. A két kísérletet ugyanazzal a köteggel, de eltérő körülmények között hajtottuk végre: először a köteg előoxidációját 1150 °C-on követte az elárasztás (CODEX-3/1), majd a már előoxidált köteget 1500 °C-on árasztottuk el (CODEX-3/2). A kísérletek egyik váratlan eredménye a nagyon alacsony hőmérséklet-növekedés és hidrogénképződés volt.

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1. INTRODUCTION

Accidents in a nuclear reactor can lead to core uncovery which results in fuel assembly heat up. If the emergency core cooling system (ECCS) is available, the core can be quenched by water and cooled down to safe conditions. According to the reactor designs, this *reflood phase* takes place when the temperature in the core is not higher then a 800-900°C. In such a case the water quenching is a classical thermal-hydraulic issue, which needs the analysis of heat transfer conditions and quench front progression.

The delay in ECCS availability however can result in higher then 1200°C core temperatures. The high temperature quenching can not be treated as thermal-hydraulic phenomena in the coolant. The coolant-fuel interaction endangers the fuel rod integrity and can initiate the early phase of core degradation.

In the Forschungszentrum Karlsruhe experimental investigation of high temperature quenching of Western design PWR and BWR fuel bundles were carried out on the CORA facility. Tests CORA-12, CORA-13 and CORA-17 showed that the flooding of Zircaloy bundle does not decrease the temperature immediately, but results in a preliminary increase before being quenched even if the electric supply was cut off [1], [2]. The temperature peak was related to a peak in the hydrogen production. The hydrogen production and the temperature increase during the flooding of hot Zircaloy bundles were confirmed by US in-pile tests (LOFT-LP-FP2 and PBF-SFD-ST) as well.

The Soviet design VVER reactors in general terms are similar to Western PWRs, for the fission heat is removed from the reactor core by pressurized water and the fuel assemblies have rod type geometry. However, the differences in core materials - first of all the use of Zr1%Nb cladding instead of Zircaloy-4 - and the hexagonal core geometry may have some effects on the degradation process. The above mentioned experiments were related to PWRs and BWRs and no experimental data were available for VVERs. For this reason two tests was performed in the CODEX facility in order to investigate the quenching of high temperature VVER bundles.

2. THE CODEX FACILITY

The CODEX (COre Degradation EXperiment) integral test facility represents the geometrical arrangement of a VVER-440 reactor fuel assembly and has been constructed of VVER materials. A schematic view of the facility is shown in Fig.1. The basic part of the facility is the test section comprising a seven rod bundle of 600 mm heated length. The rods are arranged in hexagonal geometry, the external diameter of the cladding is 9,13 mm. The six peripheral rods are electrically heated by tungsten bars. The central rod is not heated and it includes a bar for thermocouples. The 3,0 mm diameter bars in the heated rods are surrounded with ring-shaped UO_2 pellets - 3,6% enrichment of U^{235} - and enclosed in industrially fabricated Zr1%Nb alloy cladding.

Pt covered Mo wires of 1,2 mm external diameter in spiral form are used for the connection of the W bars to the electrodes. Water cooling jacket are used to cool the bottom and the top of the electrodes. The rods are filled up with argon, the internal pressure is 1,2 bar. The diameter of the hole in the pellet equals to 3,3 mm, while the external pellet diameter is 7,57 mm.

The bundle is fixed by three spacer grids, which are made of stainless steel. The spacers are located on elevations of 50 mm, 300 mm and 550 mm. The bundle is placed into a hexagonal shroud. The shroud material is Zr2.5%Nb alloy, the same alloy is used for the canisters in the real power plant assemblies. The shroud has no perforations. The relatively large mass of shroud alloy can result in high hydrogen production. This shroud is surrounded by several thermal insulation layers of ZrO_2 (13 mm thickness), Al_2O_3 (92 mm external diameter), stainless steel (98 mm and 106 mm internal and external diameters) and mineral cotton (200 mm thickness). Between Al_2O_3 and stainless steel layers a small air gap is kept to compensate thermal expansion. The cross section of bundle is given in Fig. 2 and the correlation for the calculation of the electrical resistance of the W heaters in Fig. 3.

The test section has inlet and outlet junctions for the coolant at 0 and 650 mm elevations respectively. Two observation windows makes possible the temperature measurements at 300 and 550 mm elevations by pyrometers.

The steam generator and superheater section of the facility provides argon and steam inlet flow for the test section during heating-up and cooling-down phases. The 4 kW superheater and 18 kW steam generator powers produce high coolant temperatures (~600°C) for the experiments.

An additional tank is used for quenching the test section and cooling down the bundle quickly by water. This tank is connected to the lower part of the test section and thus provides a possibility for bottom-up quenching.

The coolant leaving the upper part of the test section flows through a steam condenser and enters the off gas system with filters. The coolant cools down, the steam condenses and the water is collected at the bottom of cooler-condenser unit. The high surface molecule-filters prevent release of aerosol particles, only nonconensable gases are released during the test.

The hydrogen concentration is measured in the off-gas system using a palladium valve system (Fig. 5). The gas mixture with H content enters a volume, which is kept under depression using vacuum system and capillary tube. A thermal-cross type device placed in this volume measures the thermal conductivity of the gas. The increase of thermal conductivity indicates the increase of H concentration in the coolant, for the Pa valve allows only the H to enter the volume. Before the experiment the H measurement system needed thorough calibration.

Bundle type	VVER-440
Number of rods	7
Number of heated rods	6
Number of unhetaed rods	1
Pitch size	12,2 mm
Cladding external diameter	9,13 mm
Cladding internal diameter	7,72 mm
Cladding material	Zr1%Nb
Heater material	W
Heater bar diameter	3 mm
Pellets in heated rods	UO_2
External diameter of UO_2 pellets	7,57 mm
Internal diameter of UO_2 pellets	3,3 mm
Height of pellets	10-13 mm
Heated length with UO_2 pellets	600 mm
UO_2 enrichment	3,6%
Pellets in unheated rod	none
Material of spacer grids	stainless steel 06X18H10T
Number of spacer grids	3
Positions of spacer grids	50 mm, 300 mm, 550 mm
Height of spacer grids	10 mm
Shroud material	Zr2%Nb
Shroud thickness	2 mm
Internal key size of shroud	35,9 mm
Length of shroud	600 mm

Table 1 Main characteristics of the CODEX-3 bundle

The main parameters are collected in the data acquisition system. The values of input voltage and current, coolant flow rate, coolant inlet and outlet temperatures, condensate level, steam generator level, test section level, system pressure, hydrogen concentration and rod temperatures are measured during the test (Tables 2-3). Several high temperature W-Re thermocouples are built into the rods, shroud, insulation layers at different elevations and special windows for optical pyrometers are mounted in the test section in order to provide information on the course of experiments (Fig. 4. and Tables 4-5).

Temperature measurement	TC material/type	Position [mm]	Placement
UH50	Ni	50	central unhetaed rod
UH175	Ni	175	central unhetaed rod
UH300	W2	300	central unhetaed rod
UH425	W2	425	central unhetaed rod
UH550	W2	550	central unhetaed rod
SH50	Ni	50	shroud
SH175	Ni	175	shroud
SH300	W1	300	shroud
SH425	W1	425	shroud
SH550	W1	550	shroud
HTS550	Ni	550	high temp. insulation, steel
HTS175	Ni	175	high temp. insulation, steel
PYR300	proportional pyrometer	300	bundle/observation window
PYR550	integral pyrometer	550	bundle/observation window
TCIN	Ni	0	coolant inlet junction
TCOUT	Pt	650	coolant outlet junction
TSTEAM	Ni		steam generator
TWATER	Ni		steam generator
TAMB	Ni		containment vessel

Table 2 Temperature measurements

parameter	device	placement
Argon flowrate	calibrated valve	gas supply system
Steam flowrate	steam generator el. power	steam generator
H_2 concentration	palladium valve system	gas off system
Steam generator level	DP transducer, 20 mbar/4-20 mA	steam generator
Condensate water level	DP transducer , 60 mbar/4-20 mA	condensate tank
Water level in test section	DP transducer, 100 mbar/4-20mA	test section -100 mm
Voltage	shunt	rail of power supply
Current	el. power	rail of power supply
Gas pressure in rods	P transducer, 2.5 bar/4-20 mA	fuel rods
System pressure	P transducer, 6 bar/4-20 mA	Argon inlet junction

Table 3 Measurements of system parameters

Pyrometer	1.	2.
type	integral radiation measurement	proportional, two-color radiation measurement
position	upper space grid	central spacer grid

Table 4 Pyrometer characteristics

TC	1. (W1)	2. (W2)	3. (Pt)	4. (Ni)
type	W5%Re-W26%Re	W5%Re-W20%Re	Pt-Pt13%Rh	NiCr-NiAl
insulation	BeO	HfO_2	Al_2O_3	MgO_2
shroud	tantalum, niobium	niobium	platinum	stainless steel
external diameter	2.3 mm	2.3 mm	1.5 mm	0.8 mm

Table 5 Thermocouple (TC) characteristics

3. THE EXPERIMENTAL PROGRAMME

In the AEKI an experimental programme was initiated focusing on the high temperature behaviour of VVER fuel and core materials. The interactions of Zr1%Nb cladding with UO_2 pellet, stainless steel spacer and boron steel absorber were studied in small scale separate effect tests [3][4]. On the basis of the experience gained in these tests the CODEX integral test facility was constructed to continue this work under more prototypic conditions.

The VVER experimental programme consisted of four tests. The fifth test is planned to be performed with B_4C control rod. The main parameters of the test matrix are given in Table 2.

First the capabilities of the facility were demonstrated carrying out the CODEX-1 experiment with Al_2O_3 pellets. The test section was heated up with argon, then the electric power was increased. When the rod bundle degradation was indicated by temperature measurements the power was switched off and the section was cooled down by argon. The post-test examination showed that the rod bundle partially damaged, the further melting was stopped in time. So the facility proved to be applicable to the experimental analysis of controlled core degradation processes.

In the second experiment similar procedures were taken, but the Al_2O_3 pellets were replaced with UO_2 [5][6]. The high temperature conditions in steam atmosphere lead to partial damage of the CODEX-2 bundle, and the further core melting was stopped by slow cooling down in inert gas environment.

The CODEX-3/1 and CODEX-3/2 experiments were performed with quick water cooling [7][8]. These tests are described in the present report.

Test	Bundle type	Pellet	Year	Test type
CODEX-1	7-rod VVER	Al_2O_3	1995	scoping test
CODEX-2	7-rod VVER	UO_2	1995	escalation and slow cooling down
CODEX-3/1	7-rod VVER	UO_2	1996	water quench at 1150 °C
CODEX-3/2	7-rod VVER	UO_2	1997	water quench at 1500 °C
CODEX-B4C	7-rod VVER	UO_2,B_4C	2001	control rod relocation

Table 6 Main parameters of CODEX-VVER test matrix

4. THE CODEX-3/1 AND CODEX-3/2 TESTS

The main purpose of the CODEX-3/1 and CODEX-3/2 tests was to investigate the effect of water quenching on the degradation process of a VVER bundle. It was expected to receive similar results as in CORA tests for PWR and BWR bundles. The differences in the bundle geometry and core materials were to be evaluated.

In both experiments water quenching was the cooling down process of the high temperature bundle. However the temperatures and the initial bundle states were different in the two cases. These parameters had an important effect on the final bundle states and the degree of degradation.

4.1 CODEX-3/1: quench at 1150 °C

The first test started with a preheating period, which lasted about 9000 s. During this period 600 °C argon flow was injected through the superheater section with a flowrate of 2 g/s. A stable temperature distribution was established in the facility with an average temperature of 500 °C.

The second period started with power increase (Fig. 27) and with a 300 s delay steam injection was added to the argon flow. The power was increased linearly with a rate of 2 W/s. The steam flowrate was constant and its value was held to 1.5 g/s. This period lasted 1800 s. The maximum power was 3.3 kW and the maximum temperature at the upper part of the facility reached 1150 °C (Fig. 11). The ambient temperature in the containment vessel was 45 °C.

The cooling down period started with switching off the electrical power, stopping steam injection and injecting water from the quench tank (Fig. 26). It took about 90 s to quench the total heated length of the bundle from the bottom and the facility rapidly cooled down (Fig. 7-17).

This test showed only very limited temperature excursion, which was in the magnitude of 10 °C (Fig. 11). The hydrogen concentration measurement indicated some relative peak (Fig. 23), but its value was very low, the estimated value was below 1mg/s.

The test results were in good agreement with the widely used peak cladding temperature criterion (1200 °C): water quenching took place close to but below this limiting value and the process did not lead to severe temperature excursion, the bundle was cooled down without losing the rod-like geometry. Furthermore the later investigation showed, that during this rapid cooling down process the fuel integrity did not damage.

After the first phase the facility was disassembled and the bundle was taken out of the test section for visual investigation. No cladding failure, neither signs of cladding-spacer interaction were observed. The post-test calculation estimated 50 microns oxide layer thickness in the vicinity of highest temperature. The visual observation indicated no cracks in the oxide layer.

4.2 CODEX-3/2: quench at 1500 °C

The second test was carried out with the same bundle, which was pre-oxidized in the first experiments. This test started with a similar preheating period as the first test: using 600 °C argon flow and about 200 W electrical power, a stable temperature distribution was established with an average 500 °C temperature.

The heating up period lasted 2000 s (Fig. 47), the steam injection started with 400 s delay (Fig. 42). The linear power increase rate was also the same as in the first test: 2 W/s. The reached maximum power was higher: 4.2 kW. When the maximum temperature reached 1500 °C (Fig. 32) the power was switched off and the steam injection stopped. The ambient temperature in the containment vessel was 45 °C.

Water quenching was initiated, when the maximum temperature reached 1500 °C. The time period of quenching was about 80 s (Fig. 46.). Water injection initiated some temperature excursion, but the temperature increase was less then 200 °C and the maximum temperature remained below 1700 °C (Fig. 32.). The thermocouples in the shroud showed (Fig. 36-38.) that the Zr oxidation took place on the shroud surface as well. The maximum shroud temperature was measured at the top of the bundle and it was close to 1000 °C (Fig. 37). Measurements at lower elevations showed temperatures for 500-600°C lower. The same trend was seen in the bundle as well: the maximum temperature on the elevation 250 mm (close to the middle of the rods) was only 1000 °C (Fig. 30).

Together with the temperature peak a hydrogen production peak was observed, which indicated the exothermic reaction of Zr oxidation (Fig. 43). The total hydrogen production was about 1 g. In the previous CODEX-2 test, the excursion took place in steam atmosphere, the maximum temperature reached 2400 °C and 26.6 g hydrogen was produced, which was much more then in the case of water quenching of the CODEX-3/2 pre-oxidized bundle.

The quenching was associated with steam production and it resulted in a pressure peak. The maximum system pressure was 1.65 bars (Fig. 48.)

The main parameters are summarized for the two tests in Table 7.

	CODEX-3/1	CODEX-3/2
Date	28 November 1996	29 January 1997
Argon flowrate	2 g/s	2 g/s
Steam flowrate	1.5 g/s	1.5 g/s
Heat up period	1800 s	2000 s
Maximum electrical power	3.3 kW	4.2 kW
Temperature of quenching	1150 °C	1500 °C
Period of quenching	90 s	80 s
Maximum temperature	1158 °C	1643 °C

Table 7 Main parameters of the CODEX-3/1 and CODEX-3/2 tests

5. EXPERIMENTAL RESULTS

The results of the CODEX-3/2 test were in some contradiction with similar high temperature quench tests performed earlier on the CORA facility. The German experiments with comparable reflooding rates normally lead to higher temperature escalation and resulted in partial melting of the cladding. There is an obvious difference between the CODEX-3/2 and the CORA tests: in the present case the high temperature quenching took place on a preliminary pre-oxidized bundle.

The CODEX-3/1 and CODEX-3/2 tests were carried out in November 1996 and January 1997. In February 1998 the QUENCH-01 test was performed at FZ Karlsruhe with a preliminary pre-oxidized bundle [9]. In this test the bundle appeared to quench steadily during reflood with no evidence of temperature excursion or excess hydrogen production. These facts are in good agreement with CODEX-3/1 and CODEX-3/2 test data and confirm that the reason for the unexpected behavior of bundle during quenching was not the application of VVER materials and geometry, but the protective role of oxide layer on the external surface of cladding material.

The experimental data of both tests were collected for code validation purposes into a database, which covers 3615 s for CODEX-3/1 and 3815 s for CODEX-3/2 of the test phase and the cooldown phase with 1 s frequency. The parameters are listed in Tables 8-9 and the plotted in Fig. 7-28 for CODEX-3/1 and Fig. 29-48 for CODEX-3/2.

Name	Definition	Unit
UH50	unheated rod temperature at 50 mm	°C
UH175	unheated rod temperature at 175 mm	°C
UH300	unheated rod temperature at 300 mm	°C
UH425	unheated rod temperature at 425 mm	°C
UH550	unheated rod temperature at 550 mm	°C
SH50	shroud temperature at 50 mm	°C
SH175	shroud temperature at 175 mm	°C
SH425	shroud temperature at 425 mm	°C
SH550	shroud temperature at 550 mm	°C
HTS550	steel temperature at 550 mm	°C
HTS175	steel temperature at 175 mm	°C
PYR550	pyrometer temperature at 550 mm	°C
TCIN	coolant inlet temperature	°C
TCOUT	coolant outlet temperature	°C
TSTEAM	steam temperature in steam generator	°C
TWATER	water temperature in steam generator	°C
H2	hydrogen flowrate	mg/s
LSG	evaporated water volume in steam generator	l
LCOND	condensed water volume in condensate tank	l
LEVEL	water level in test section	mm
POWER	total electrical power of the heater rods	W
PABS	system pressure	bar

Table 8 List of parameters available in the CODEX-3/1 experimental database

Name	Definition	Unit
UH50	unheated rod temperature at 50 mm	°C
UH300	unheated rod temperature at 300 mm	°C
UH425	unheated rod temperature at 425 mm	°C
UH550	unheated rod temperature at 550 mm	°C
SH50	shroud temperature at 50 mm	°C
SH175	shroud temperature at 175 mm	°C
SH300	shroud temperature at 300 mm	°C
SH425	shroud temperature at 425 mm	°C
SH550	shroud temperature at 550 mm	°C
HTS550	steel temperature at 550 mm	°C
HTS175	steel temperature at 175 mm	°C
TCIN	coolant inlet temperature	°C
TCOUT	coolant outlet temperature	°C
STEAM	steam flowrate	g/s
H2	hydrogen flowrate	mg/s
LSG	evaporated water volume in steam generator	l
LCOND	condensed water volume in condensate tank	l
LEVEL	water level in test section	mm
POWER	total electrical power of the heater rods	W
PABS	system pressure	bar

Table 9 List of parameters available in the CODEX-3/2 experimental database

6. POST-TEST EXAMINATION

After the CODEX-3/1 experiment the bundle remained intact and for this reason no cross sections were prepared from the bundle. The visual observation showed no sings of cladding failure or bundle damage. The thickness of oxide layer on the fuel rods was estimated about 50 microns.

After the CODEX-3/2 experiment the bundle was filled up with epoxy and horizontal cross sections were cut and polished. According to the metallographical and optical microscope observations the upper part of the bundle strongly was oxidized, the cladding became brittle and the bundle damaged. The typical phenomena of rod failure were fragmentation and mechanical break-up as results of thermal shocks and embrittlement. No signs of cladding or pellet melting were observed. The lower part of the bundle remained intact and the lower temperatures resulted only in a few micron thick oxide layer. The typical oxide layer thickness in the upper part was between 100-200 microns.(Fig. 50.)

The upper spacer grid disappeared and the fuel rods fragmented in the upper part of the bundle and pieces of the cladding and fuel relocated to lower positions. The central spacer grid and the bundle structure below 300 mm elevation remained intact (see cross section in Fig. 49).

In the cross section at 425 mm elevation (Fig. 51) eight rods can be seen, one of them obviously is a fragmented piece of the rod and relocated together with the tungsten heater bar from an upper elevation.

The temperature in the bundle was high enough for the pellet-cladding interaction. Fig. 52 shows the cladding at 425 mm elevation with oxide layers on both internal and external surfaces. The internal oxide layer contains inclusions of metallic uranium.

Small size debris particles were accumulated in the narrow space between the central and the left hand side rods at elevation 425 mm (Fig. 53). The fragmentation of the brittle cladding with a thick external oxide layer can be observed in Fig. 54.

7. CONCLUSIONS

Two water quenching experiments were carried out on the CODEX facility with high temperature VVER bundles.

- In the CODEX-3/1 experiment quenching took place at 1150 °C, which resulted in the oxidation of the bundle without damage.
- The pre-oxidized bundle was used in the CODEX-3/2 experiment, where the quenching was initiated at 1500 °C. Limited temperature increase and hydrogen production were observed, the upper part of bundle damaged due to thermal stresses and embrittlement.

The basic conclusion of the CODEX-3/2 experiment was that the oxide layer on the external surface of the cladding can play a protective role during water quenching and to some extent can limit the cladding oxidation, hydrogen production and so the degradation process.

The CODEX-3/1 and CODEX-3/2 experiments provided important information on the behaviour of VVER fuel rods under severe accident conditions [10] and the experimental database is available for model development and code validation purposes.

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[8] L. Maróti, I. Nagy, P. Windberg: Quenching of Highly Oxidized VVER Type 7-Rod Bundle, 3rd Quench Workshop, Karslruhe, 3-5 December 1997

[9] P. Hofmann, S. Horn, W Leiling, A. Miassoedov, D.Piel, L.Schmidt, L.Sepold, M. Steinbrück: Experimental Results of Test QUENCH-01, Presented at the *Fourth International QUENCH Workshop, Karlsruhe, 6-7 October 1998.*

[10] Z. Hózer , K. Trambauer , J. Duspiva, "VVER-Specific Features Regarding Core Degradation", NEA/CSNI/R(98)20, 1998

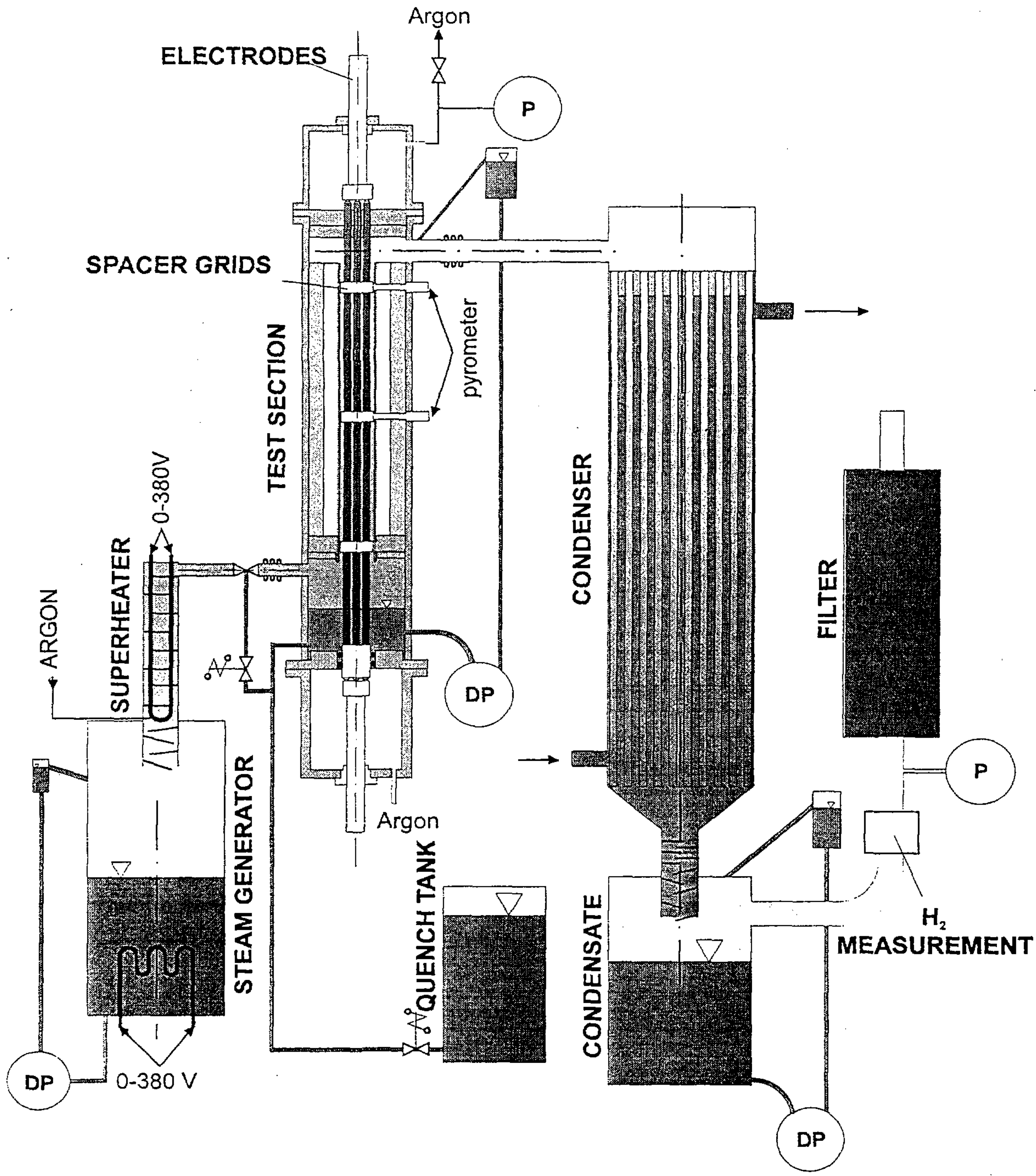


Fig. 1 Scheme of the CODEX facility

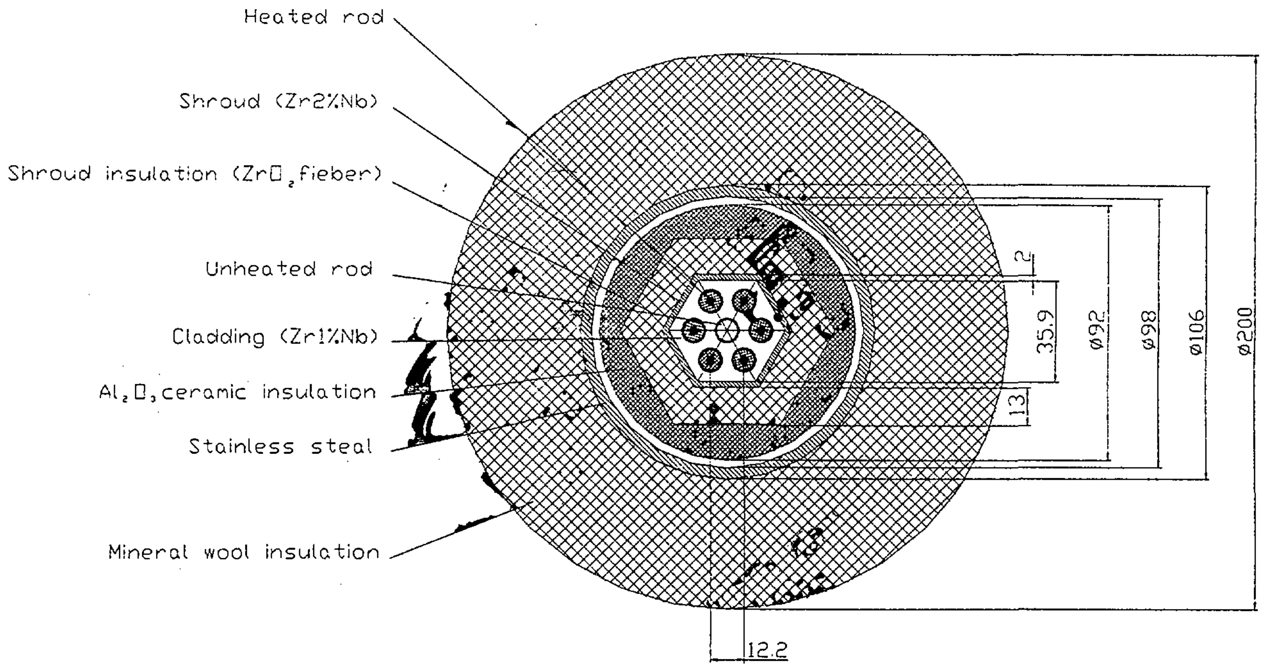


Fig. 2 Horizontal cross section of the CODEX-3 bundle

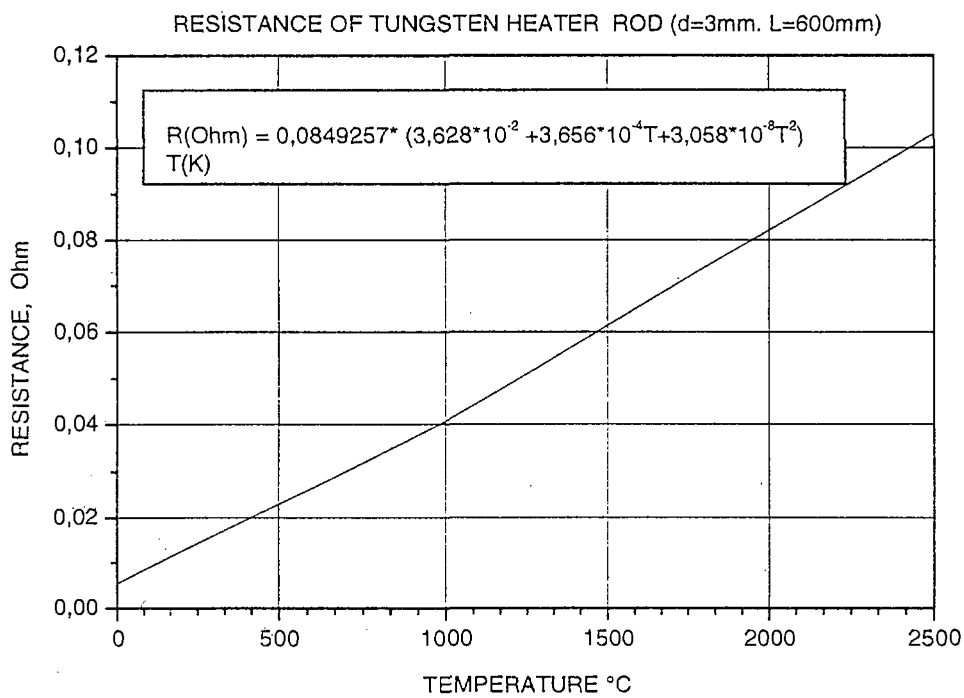


Fig. 3 Electrical resistance of the W heater as function of temperature

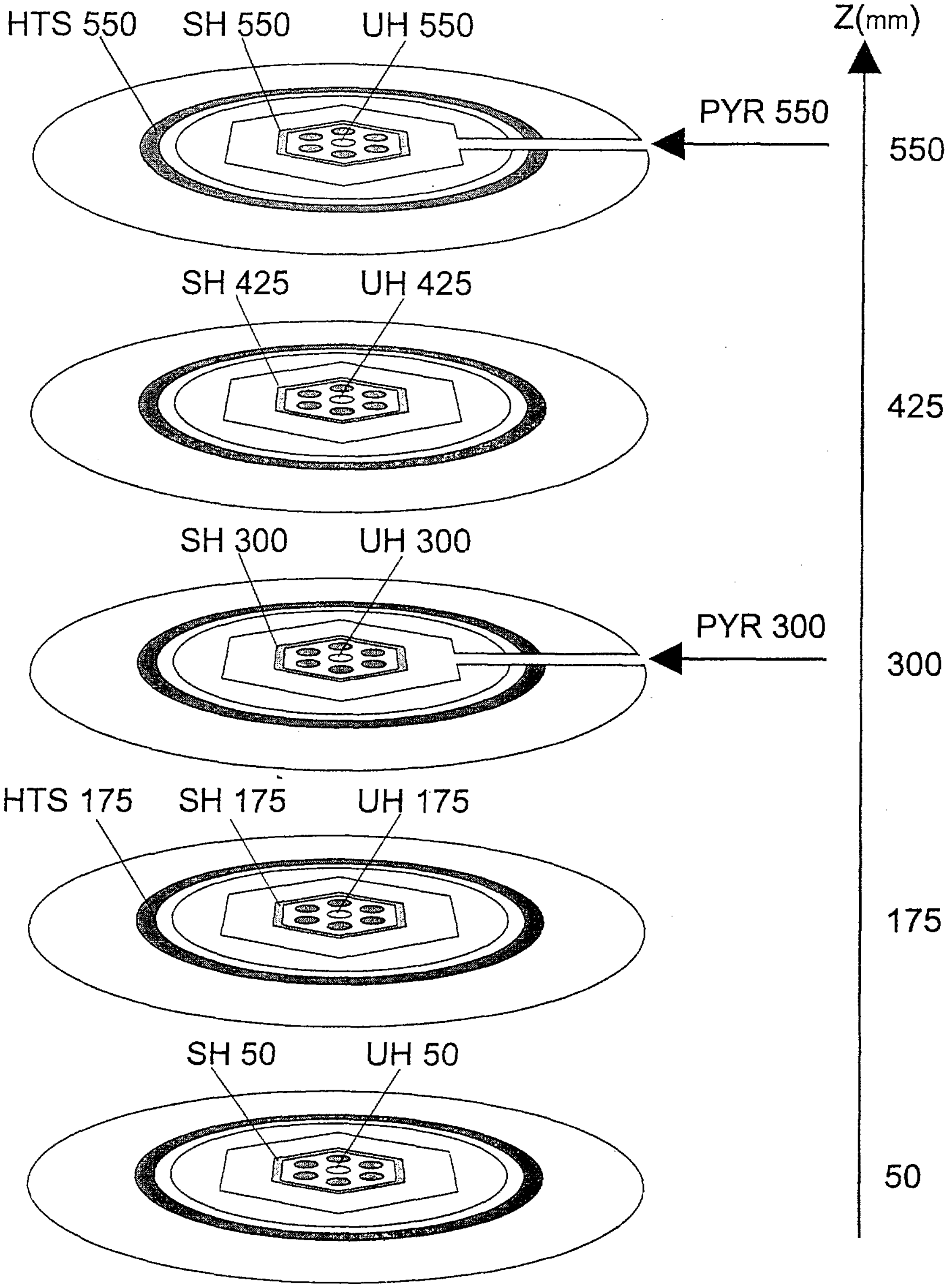


Fig. 4 Locations of thermocouples and pyrometers

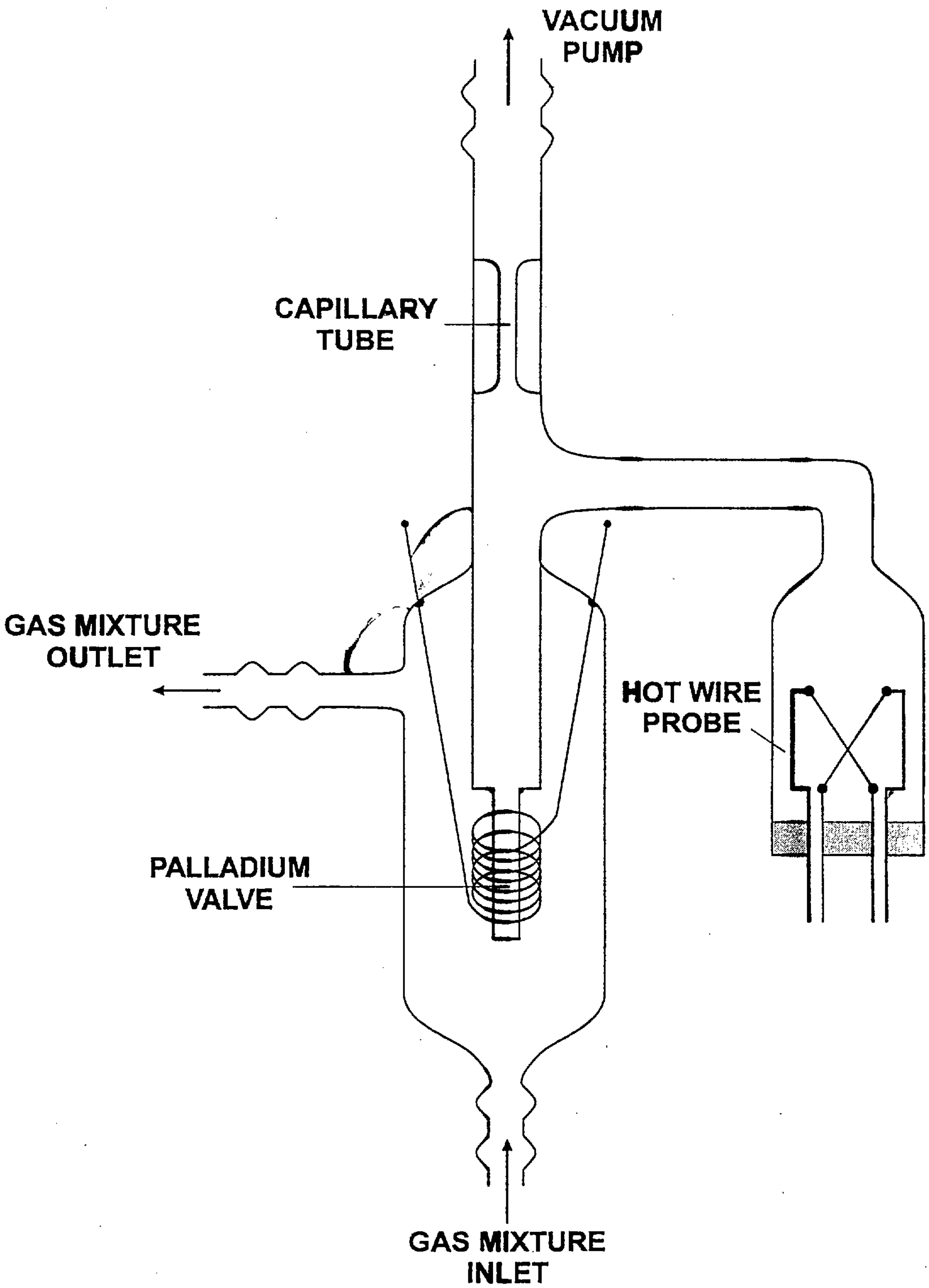


Fig. 5 H concentration measurement

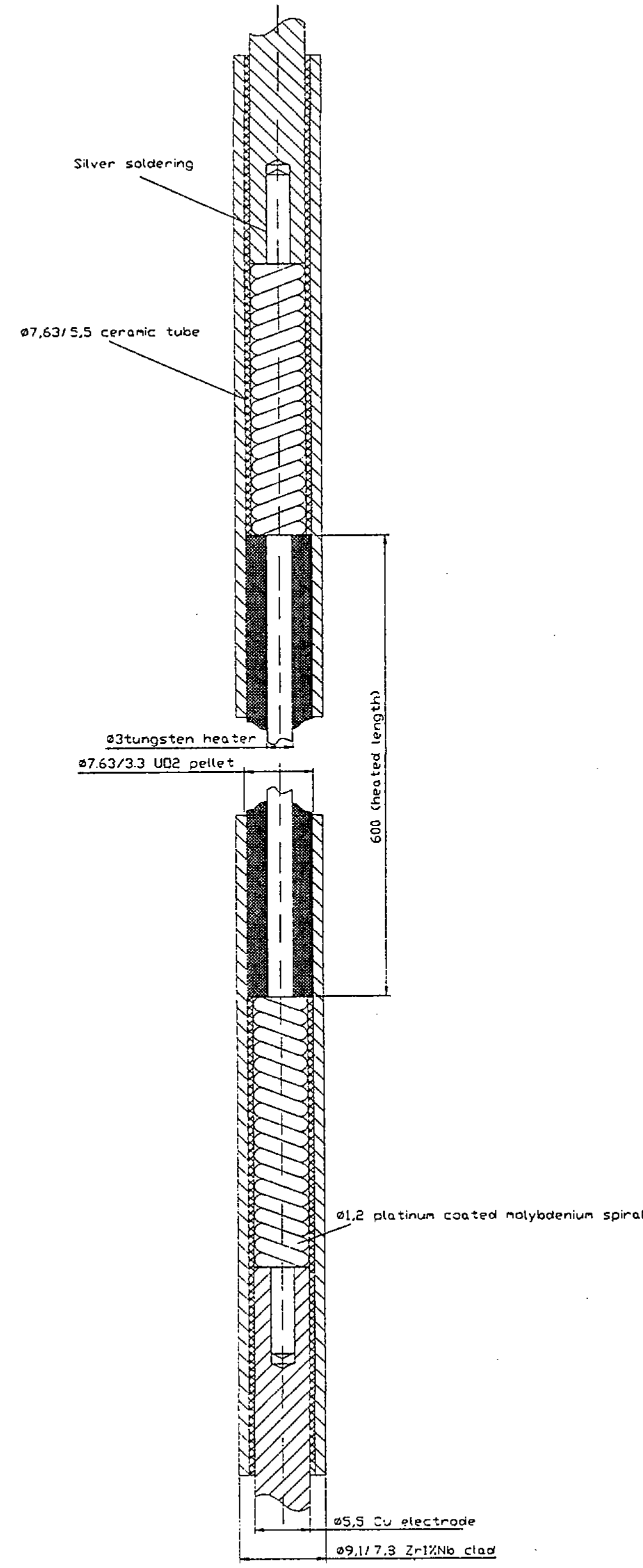


Fig. 6 Heated rod of the CODEX bundle

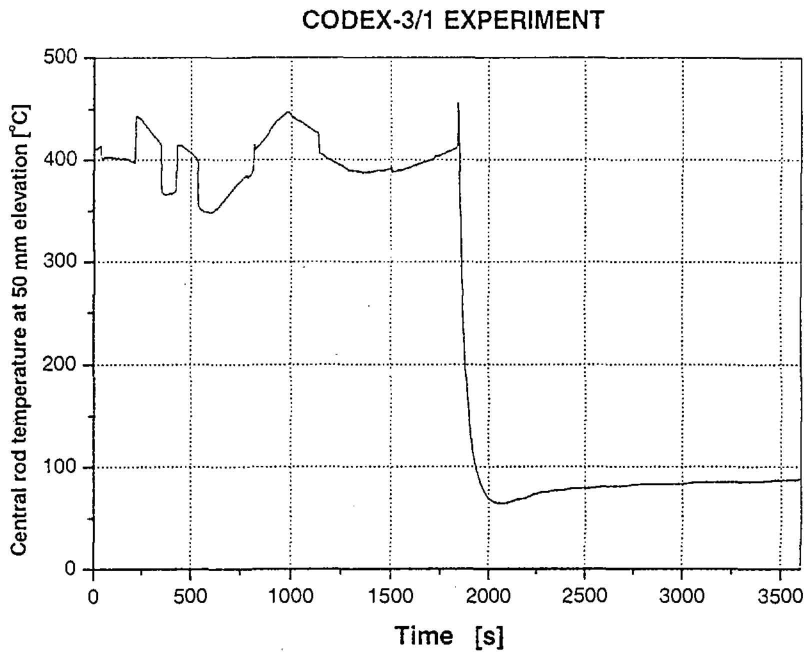


Fig. 7 UH50: unheated rod temperature at 50 mm

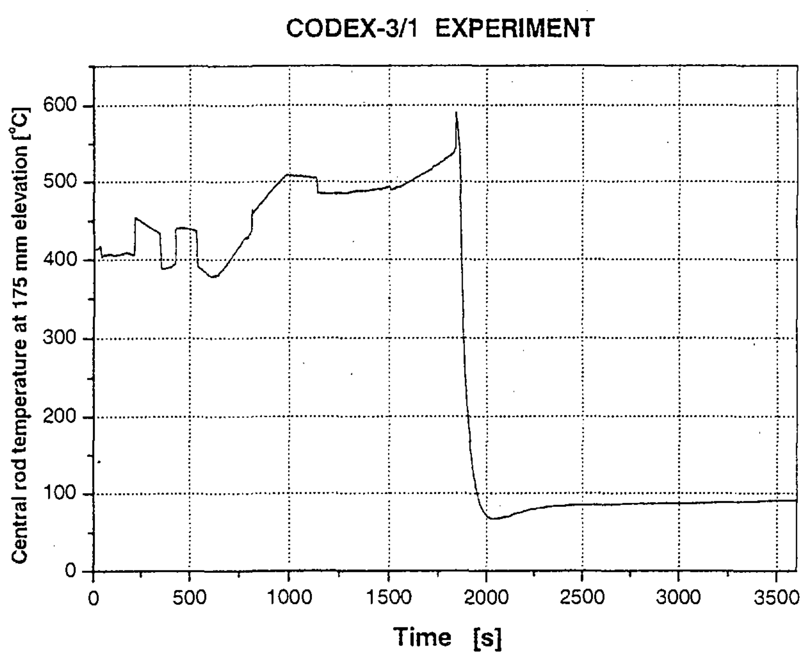


Fig. 8 UH175: unheated rod temperature at 175 mm

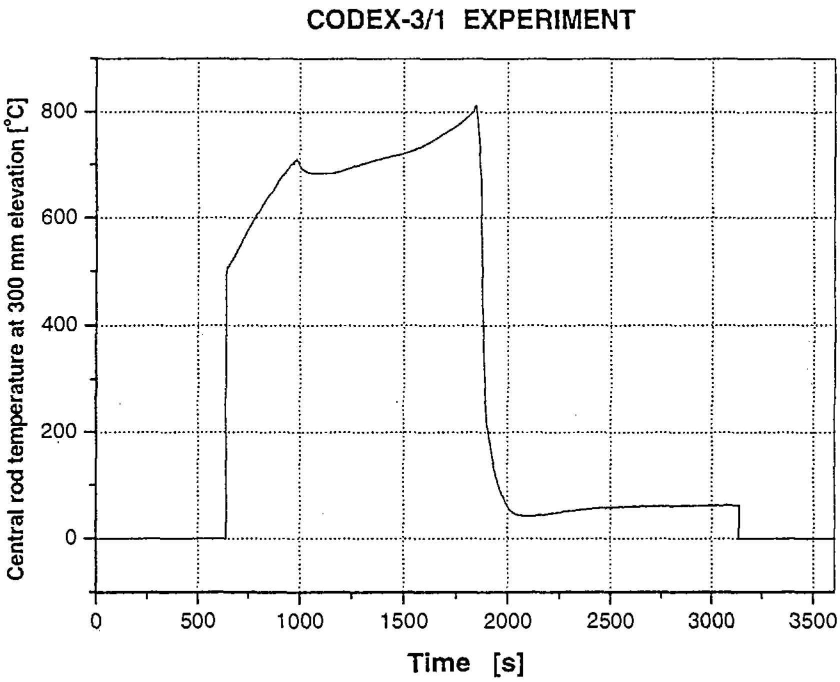


Fig. 9 UH300: unheated rod temperature at 300 mm

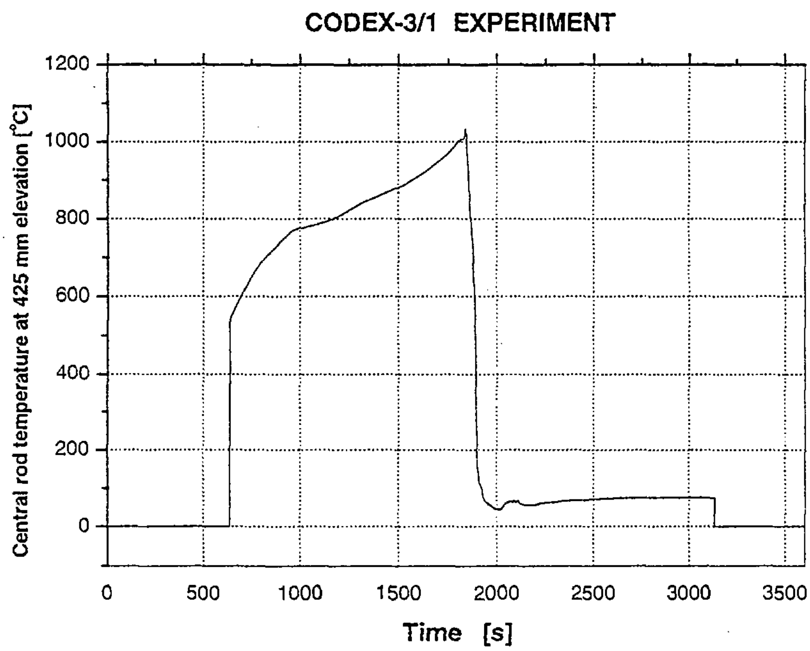


Fig. 10 UH425: unheated rod temperature at 425 mm

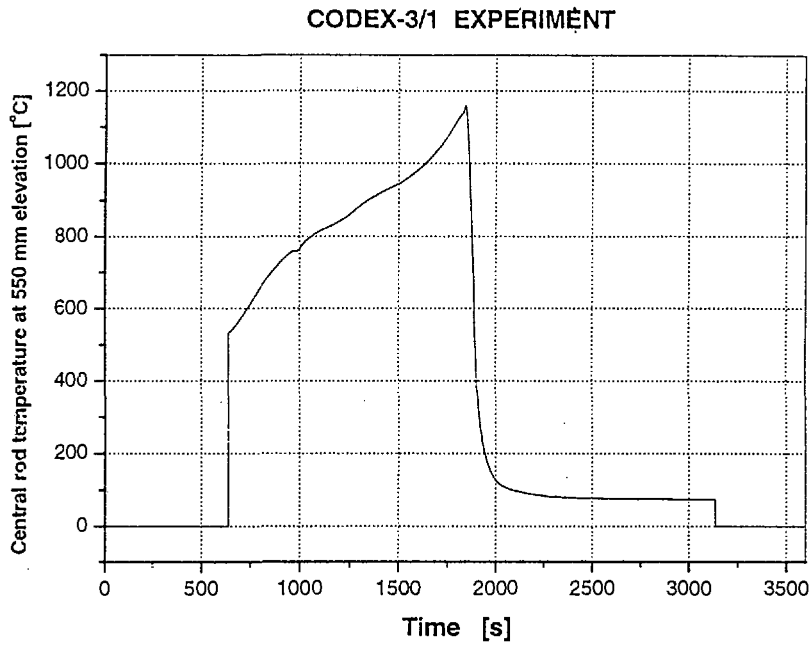


Fig. 11 UH550: unheated rod temperature at 550 mm

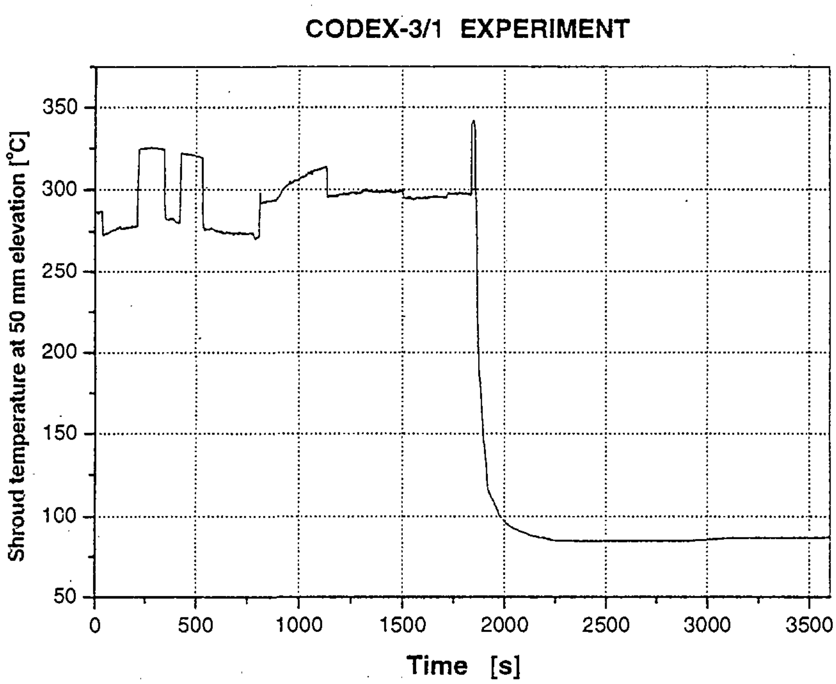


Fig. 12 SH50: shroud temperature at 50 mm

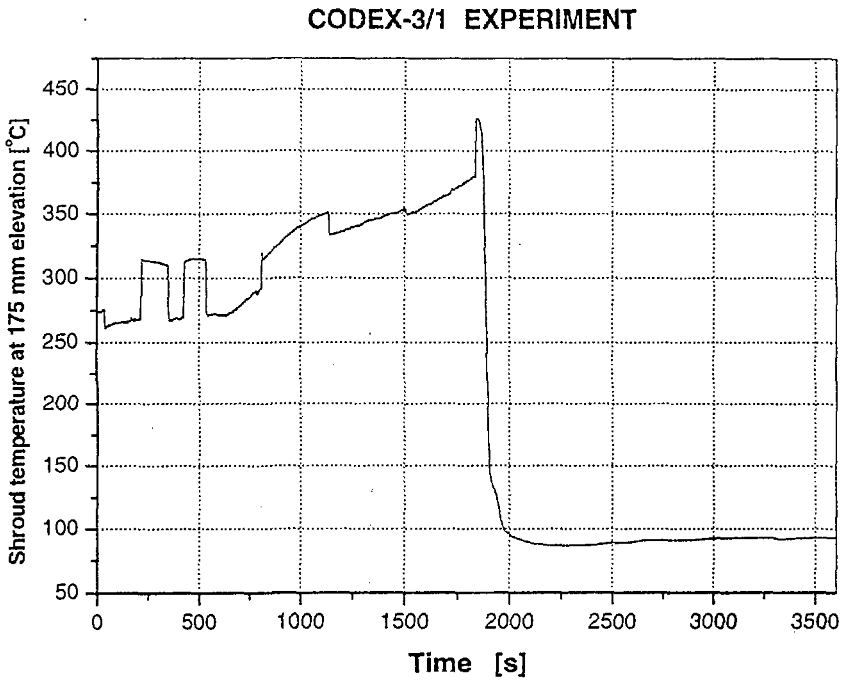


Fig. 13 SH175: shroud temperature at 175 mm

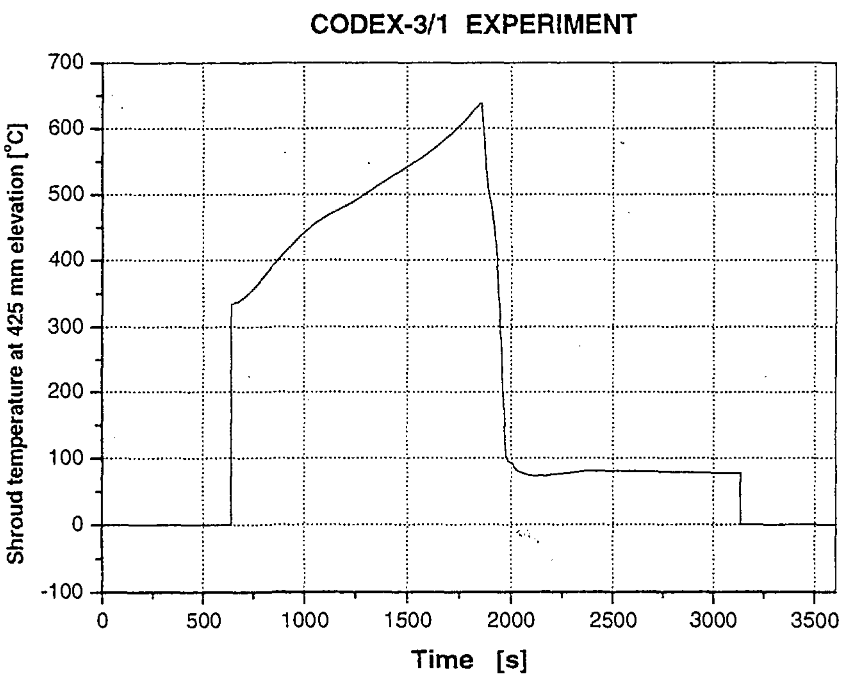


Fig. 14 SH425: shroud temperature at 425 mm

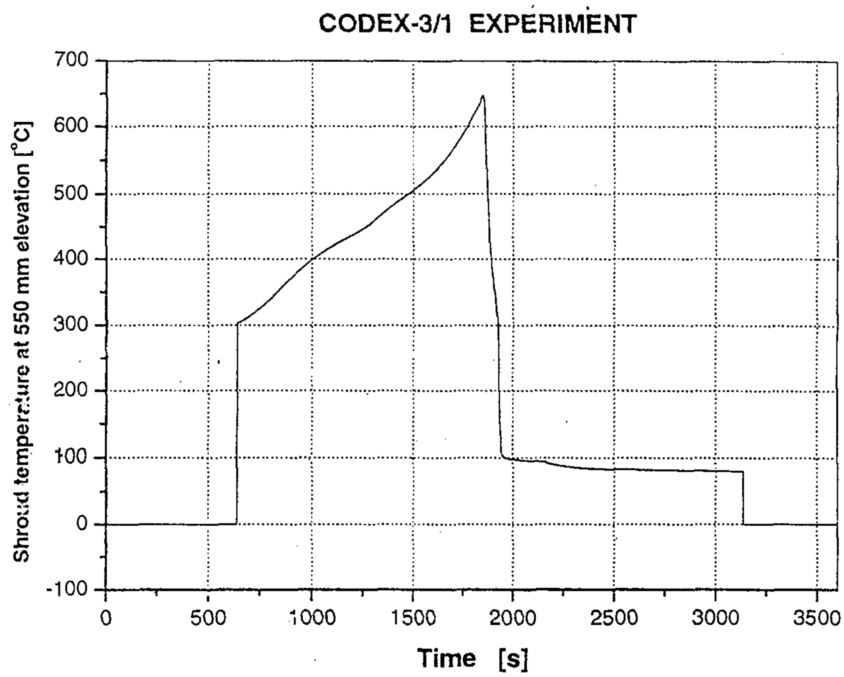


Fig. 15 SH550: shroud temperature at 550 mm

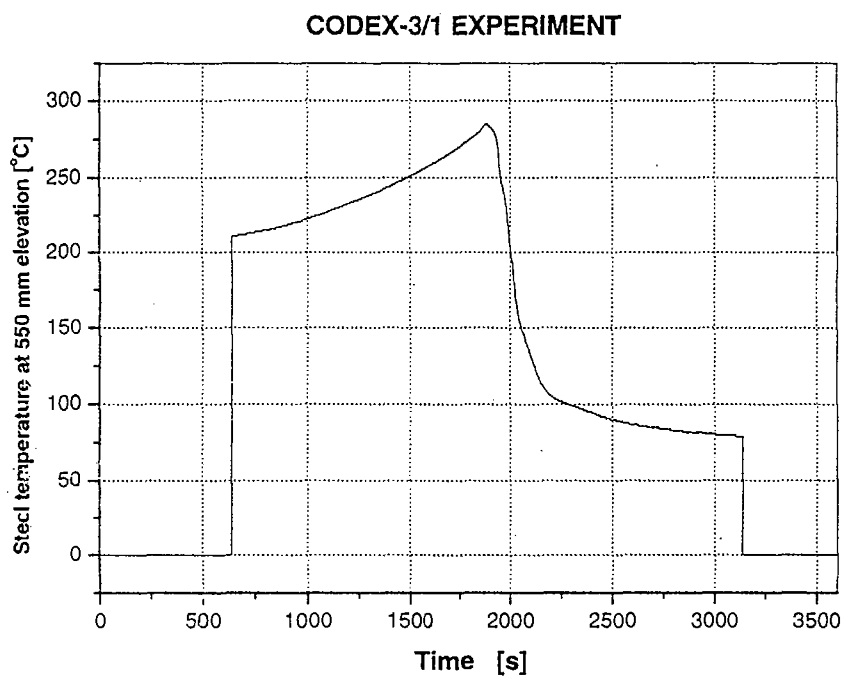


Fig. 16 HTS550: steel temperature at 550 mm

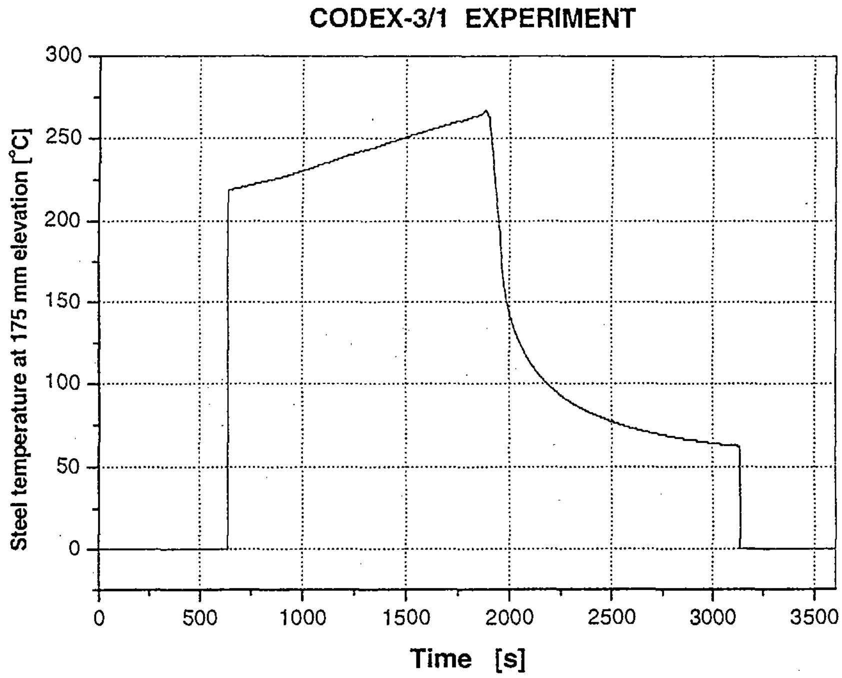


Fig. 17 HTS175: steel temperature at 175 mm

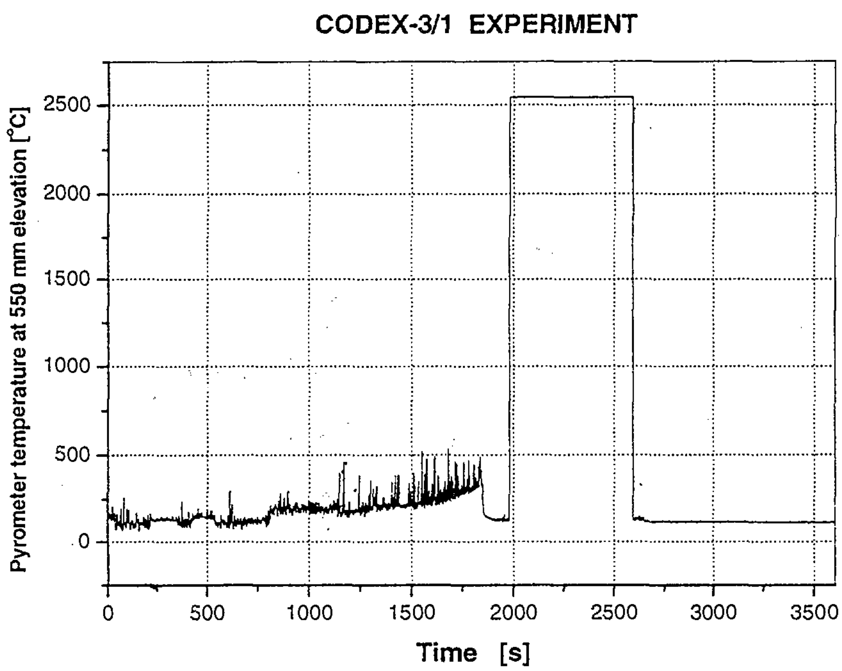


Fig. 18 PYR550: pyrometer temperature at 550 mm

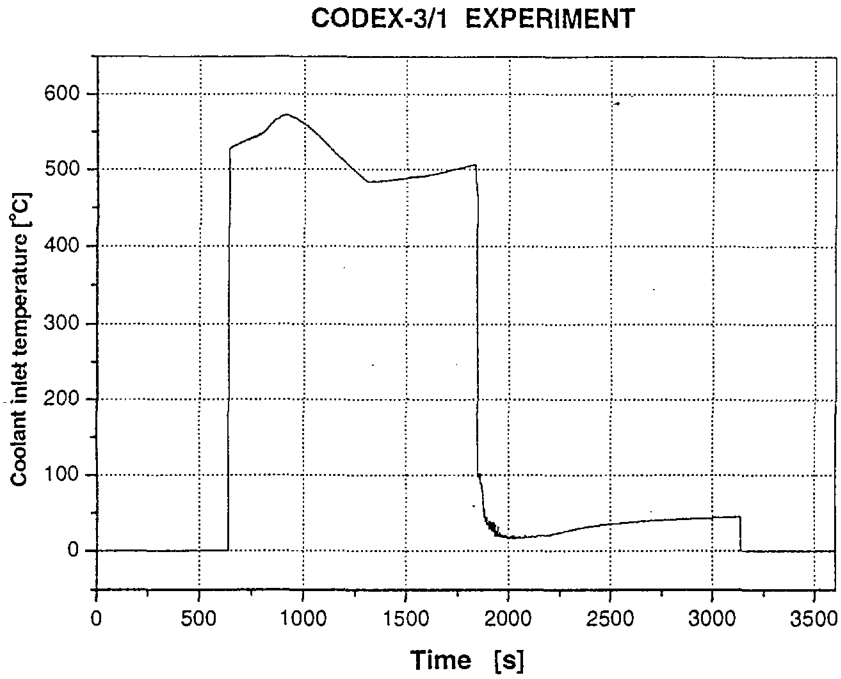


Fig. 19 TCIN: coolant inlet temperature

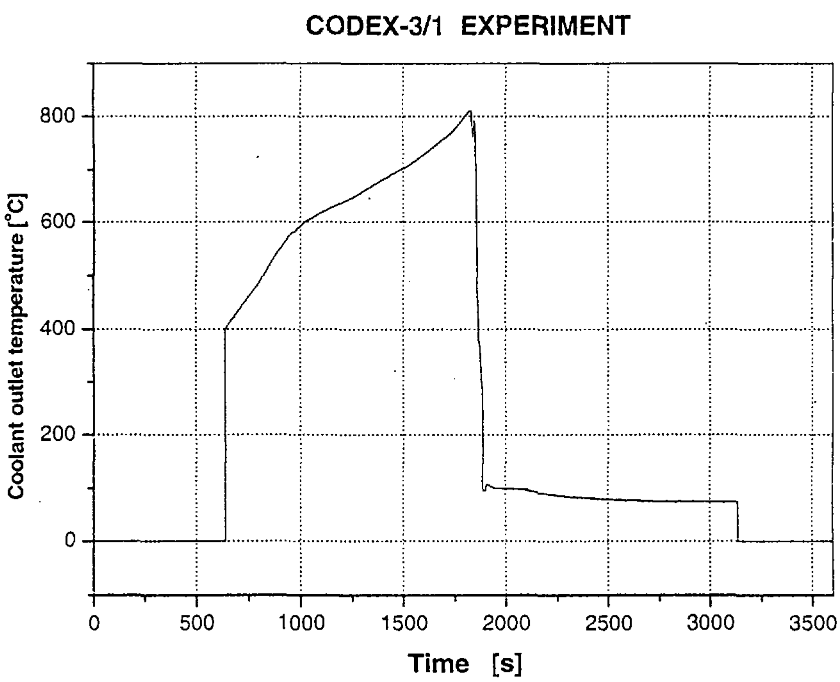


Fig. 20 TCOUT: coolant outlet temperature

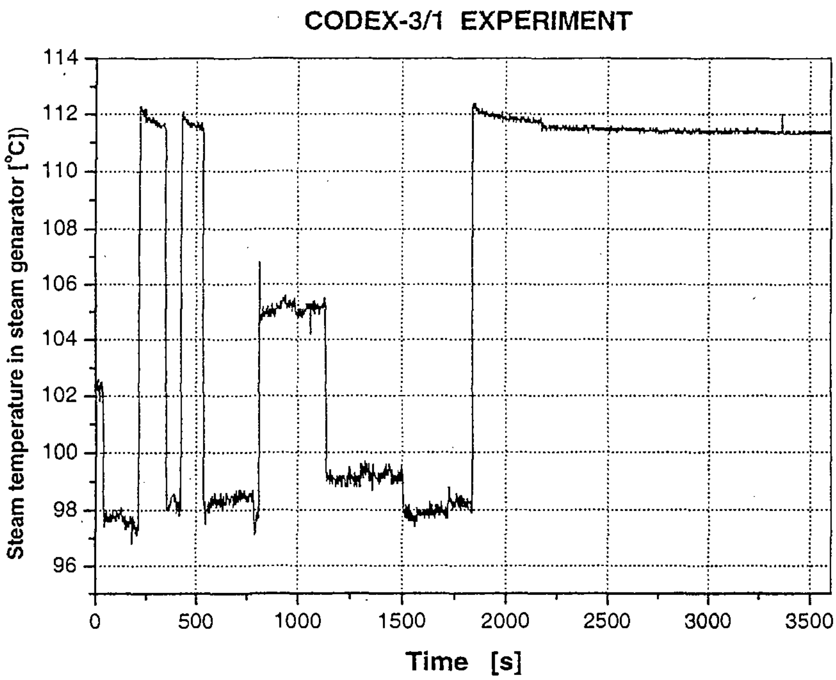


Fig. 21 TSTEAM: steam temperature in steam generator

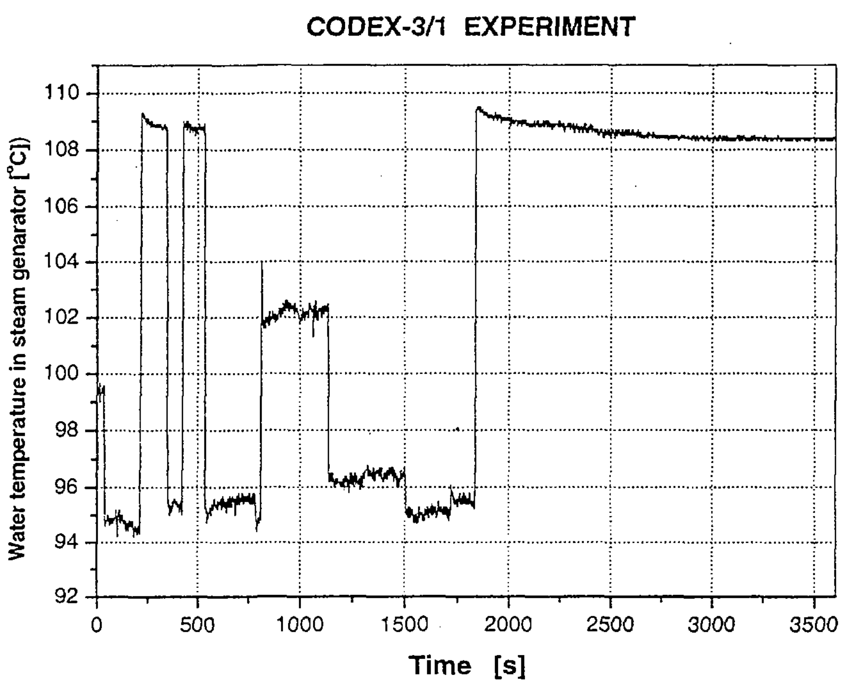


Fig. 22 TWATER: water temperature in steam generator

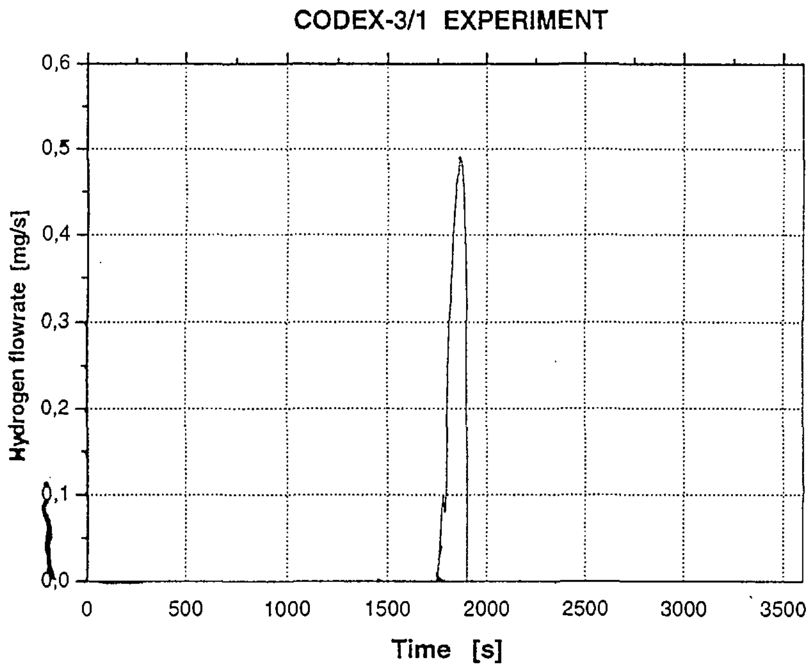


Fig. 23 H2: hydrogen flowrate

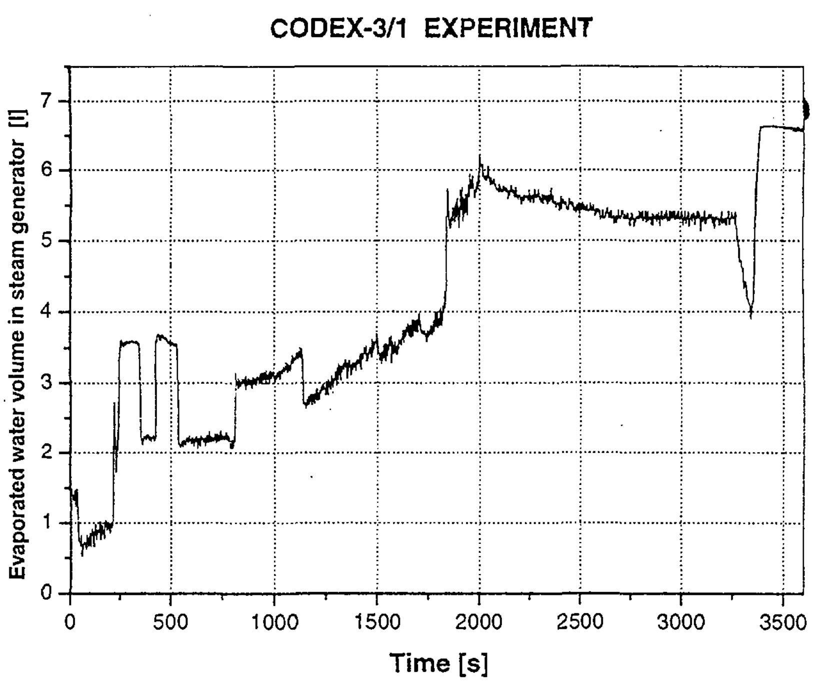


Fig. 24 LSG: evaporated water volume in steam generator
(based on water level measuerement)

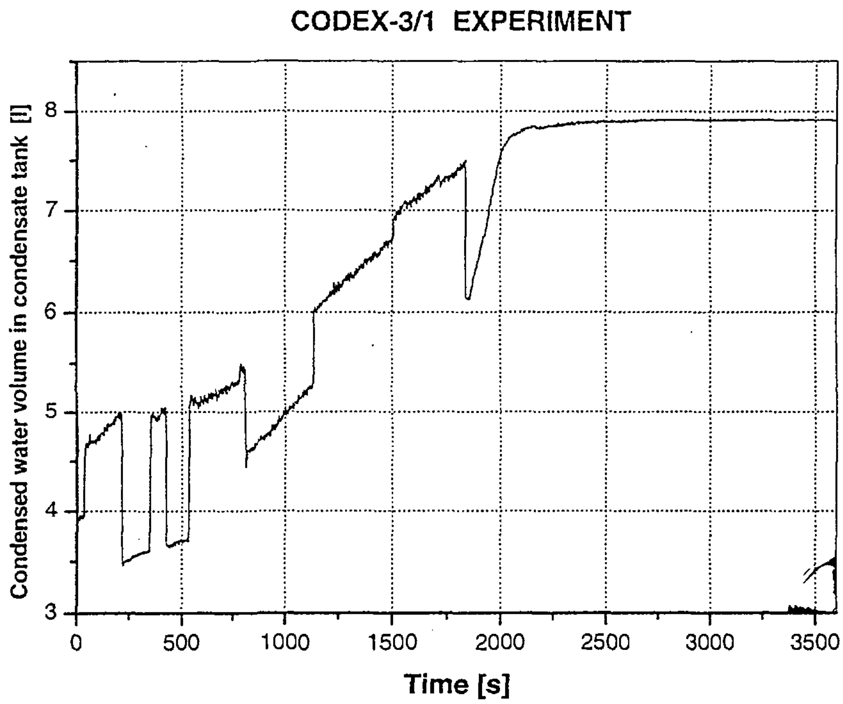


Fig. 25 LCOND: water volume in condensate tank
(based on water level measuerement)

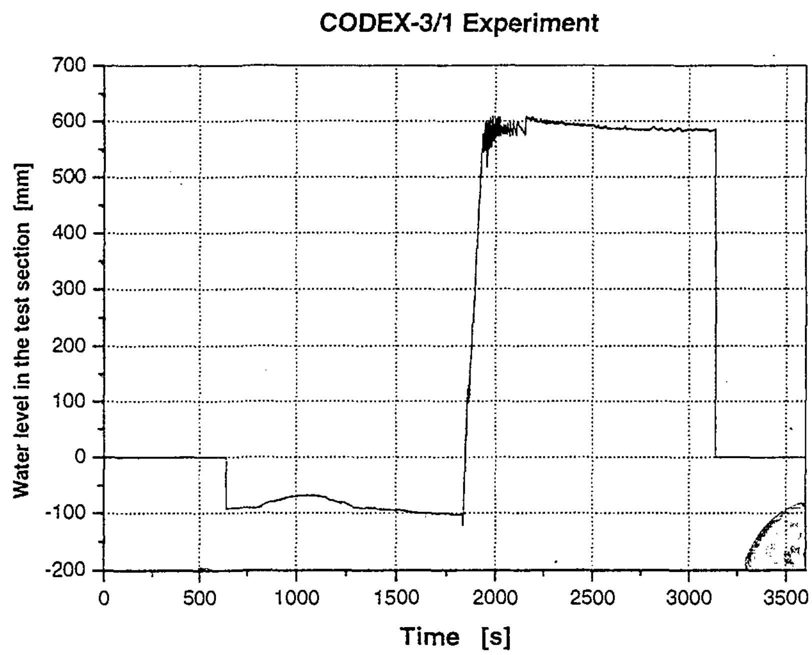


Fig. 26 LEVEL: water level in test section

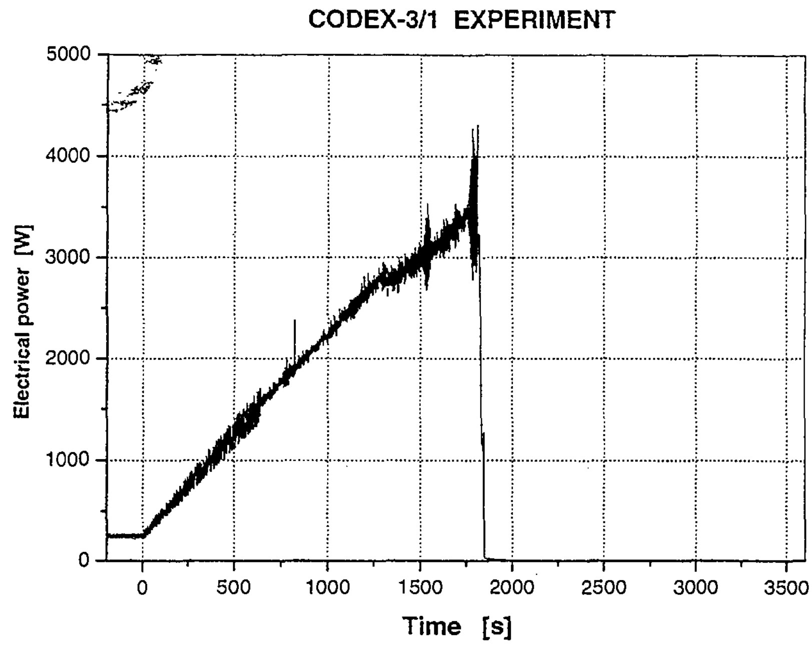


Fig. 27 POWER: total electrical power of the heater rods

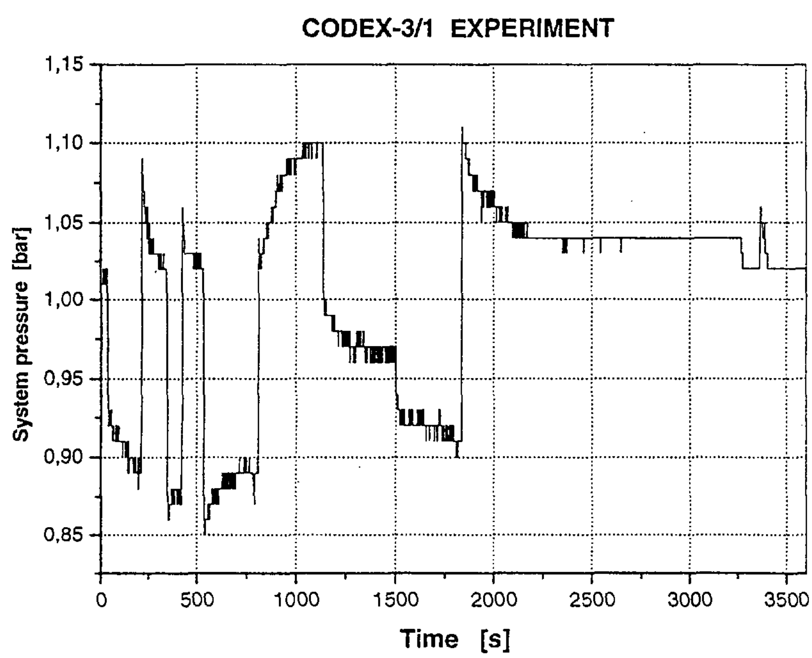


Fig. 28 PABS: system pressure

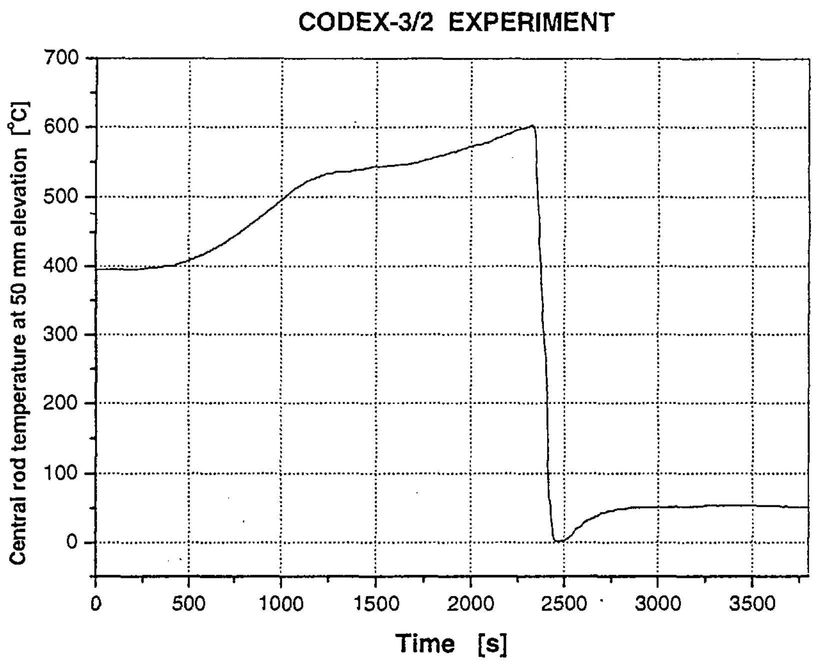


Fig. 29 UH50: unheated rod temperature at 50 mm

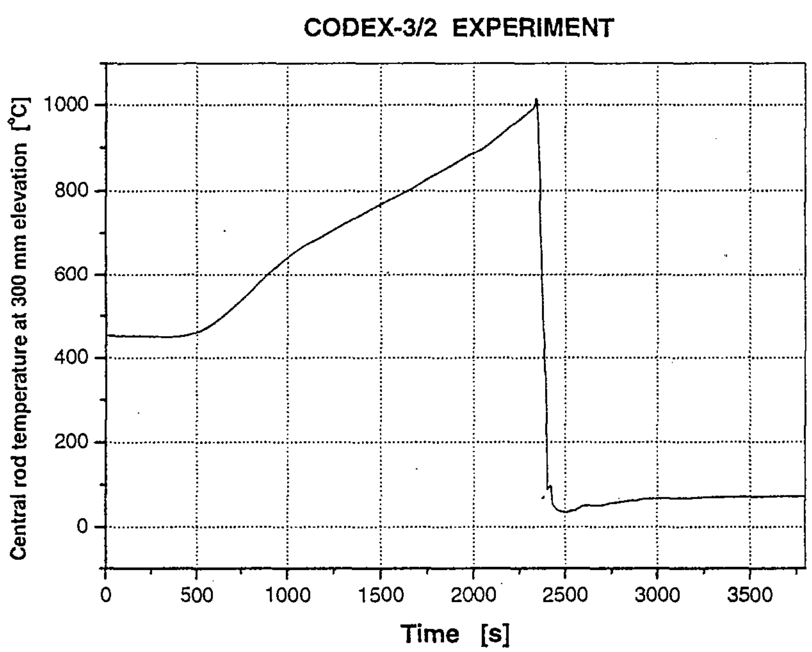


Fig. 30 UH300: unheated rod temperature at 300 mm

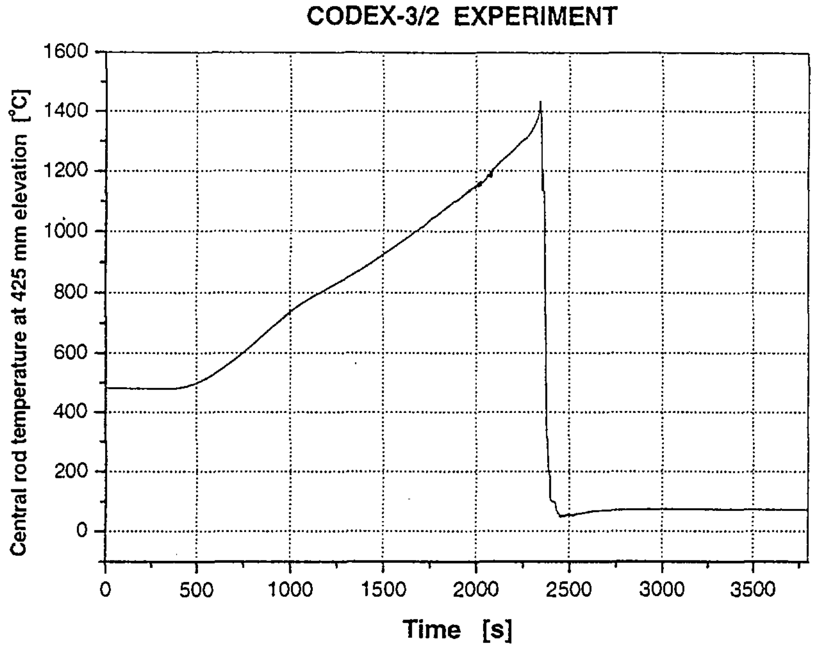


Fig. 31 UH425: unheated rod temperature at 425 mm

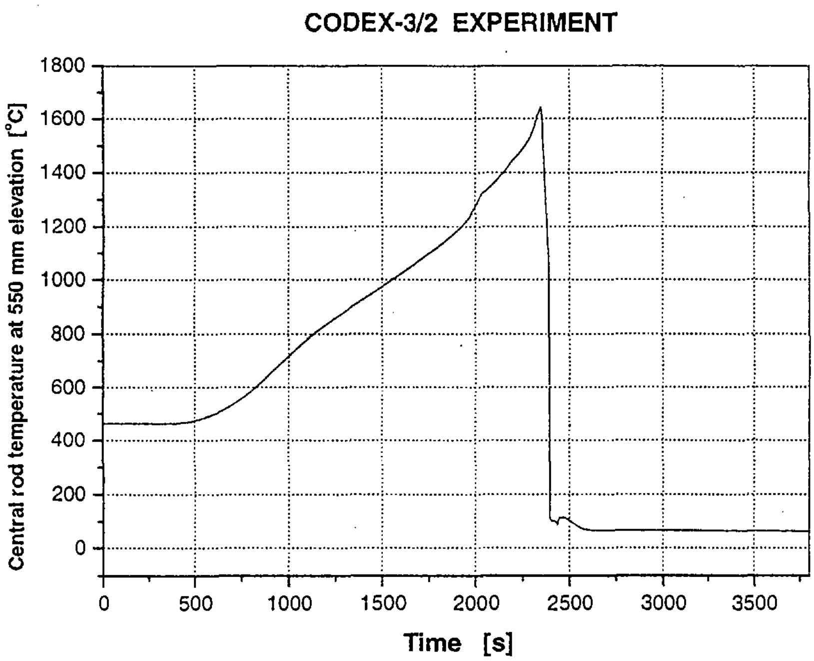


Fig. 32 UH550: unheated rod temperature at 550 mm

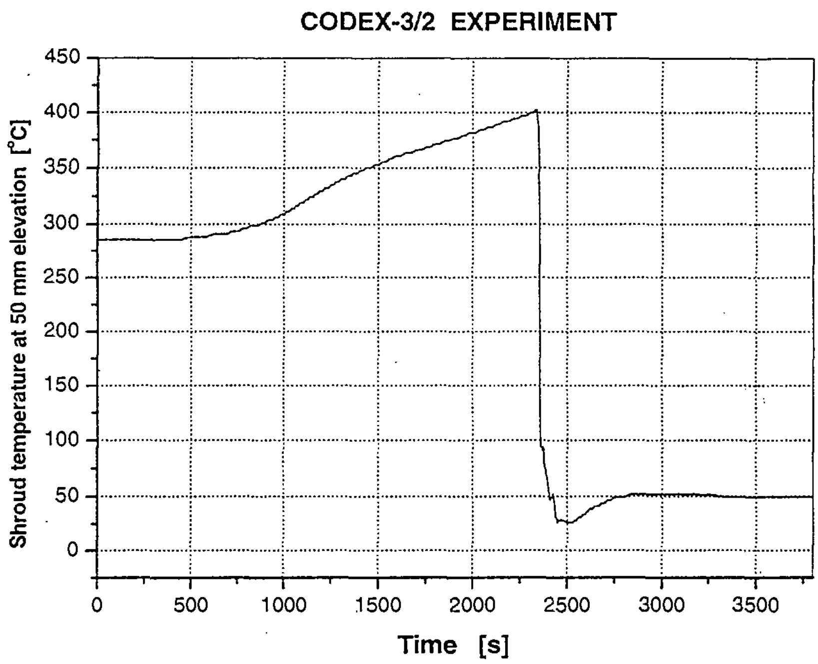


Fig. 33 SH50: shroud temperature at 50 mm

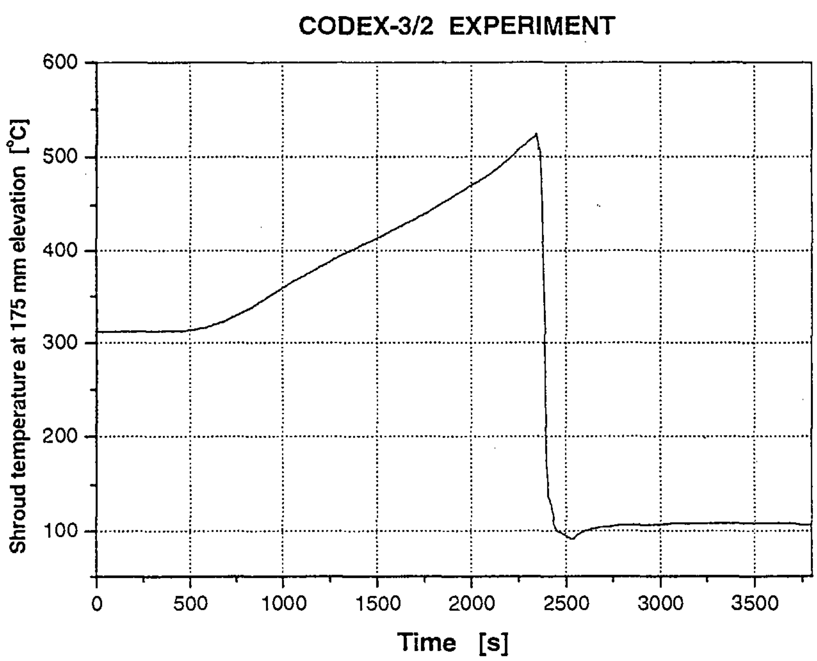


Fig. 34 SH175: shroud temperature at 175 mm

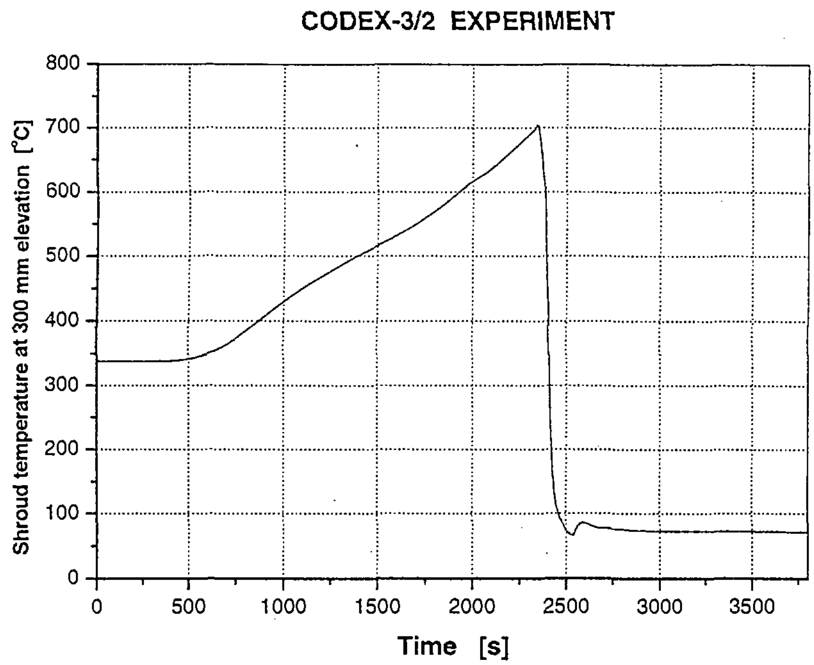


Fig. 35 SH300: shroud temperature at 300 mm

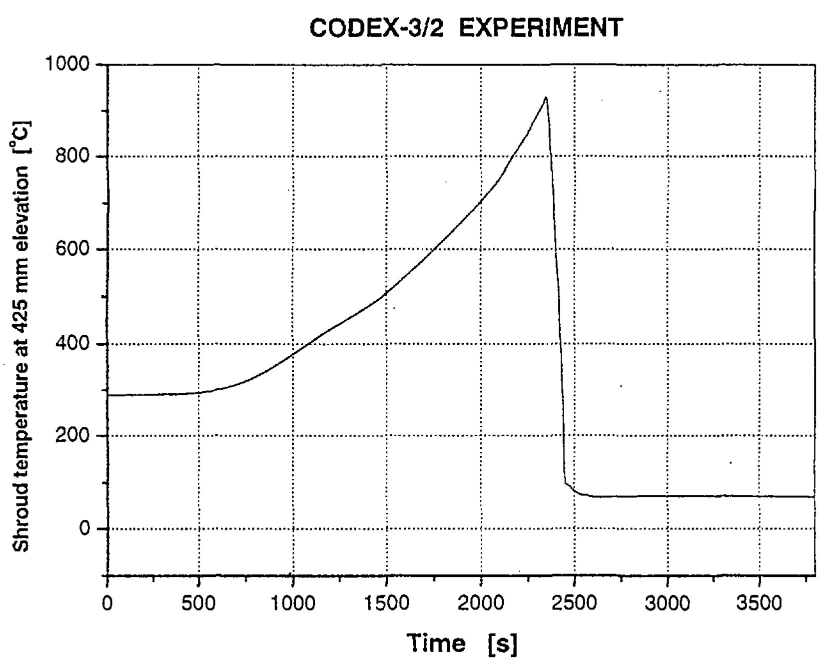


Fig. 36 SH425: shroud temperature at 425 mm

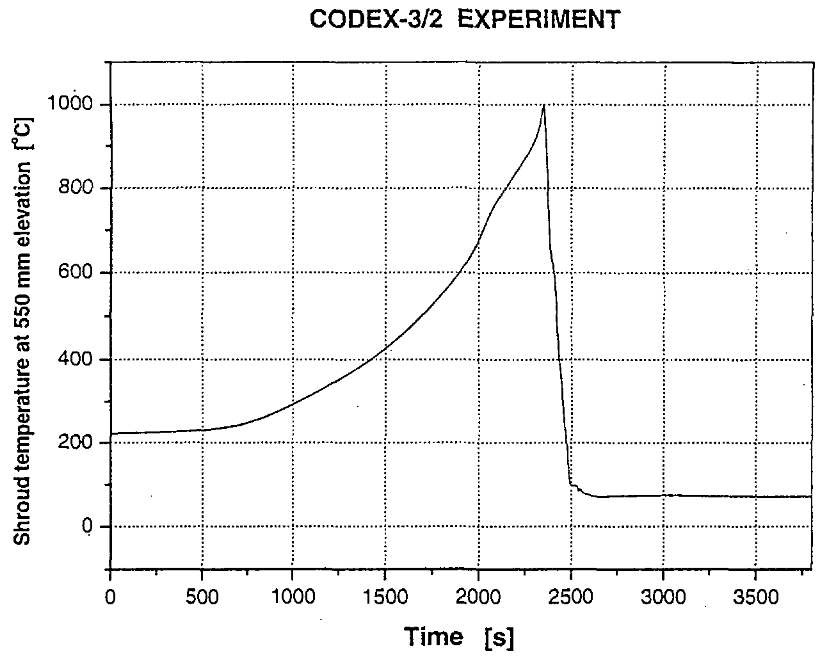


Fig. 37 SH550: shroud temperature at 550 mm

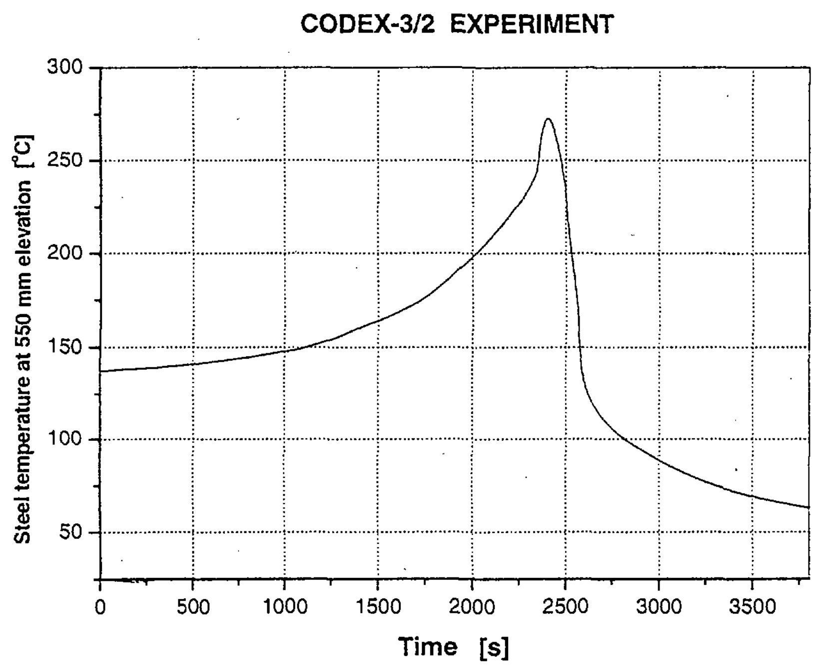


Fig. 38 HTS550: steel temperature at 550 mm

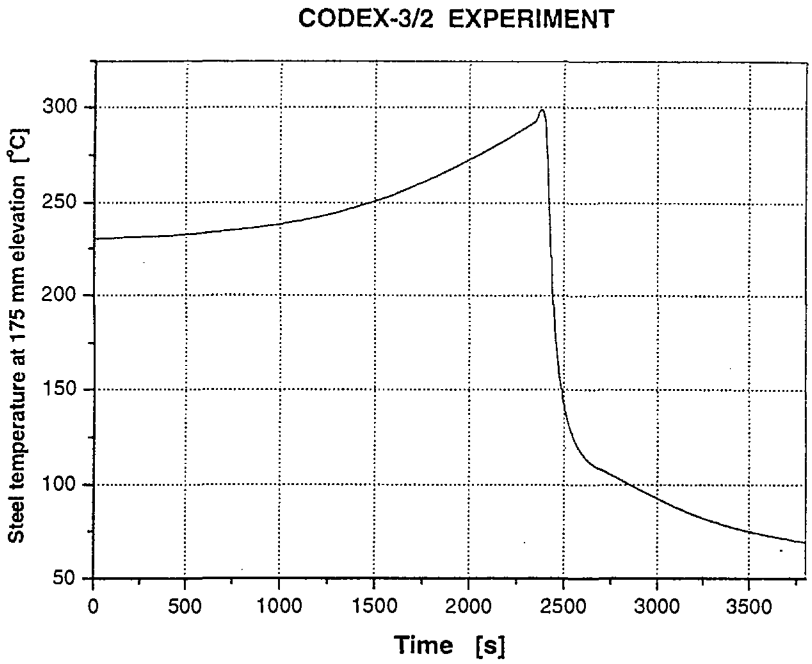


Fig. 39 HTS175: steel temperature at 175 mm

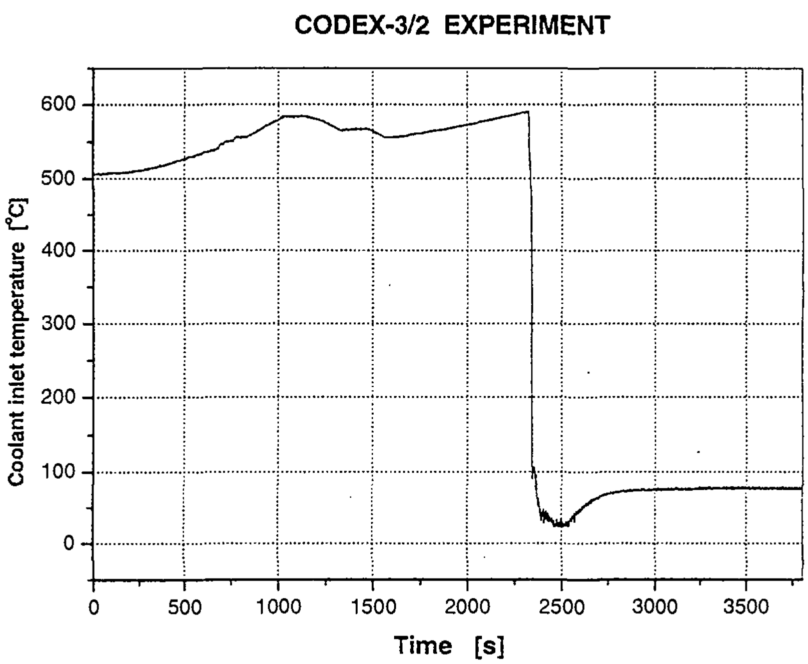


Fig. 40 TCIN: coolant inlet temperature

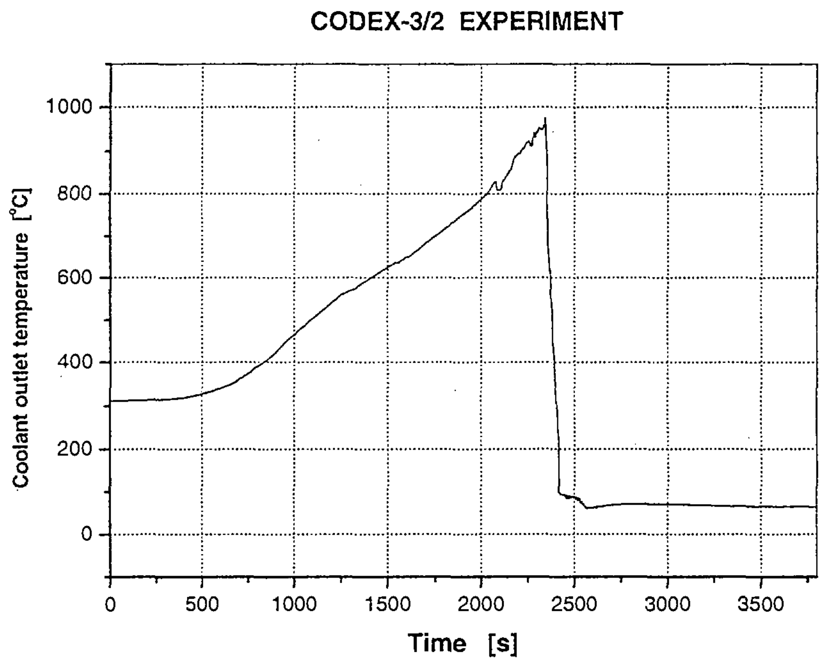


Fig. 41 TCOUT: coolant outlet temperature

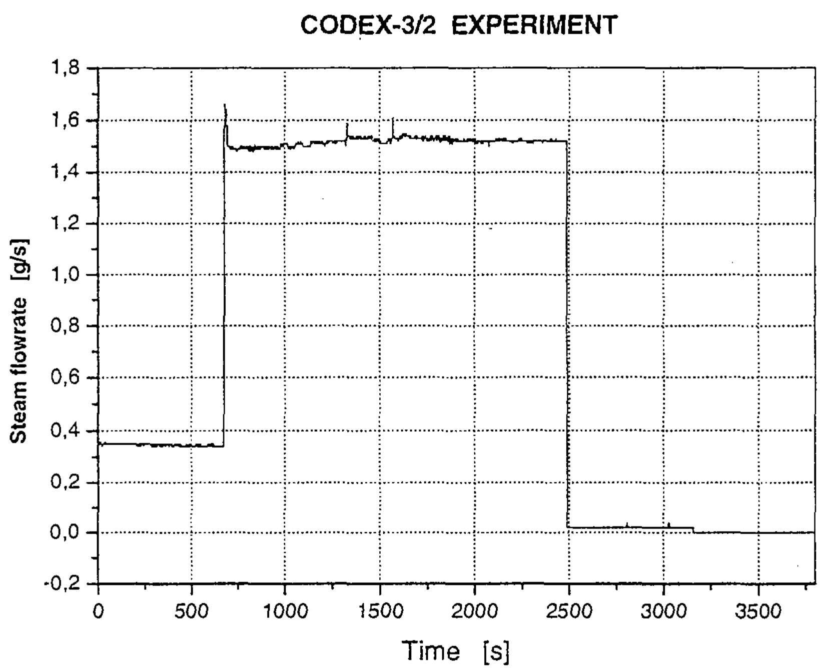


Fig. 42 STEAM: steam flowrate

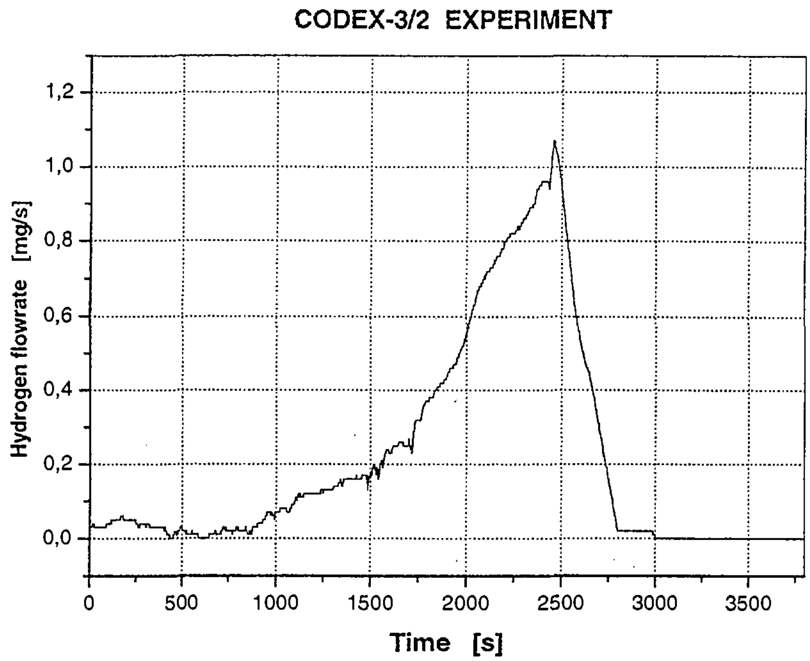


Fig. 43 H2: hydrogen flowrate

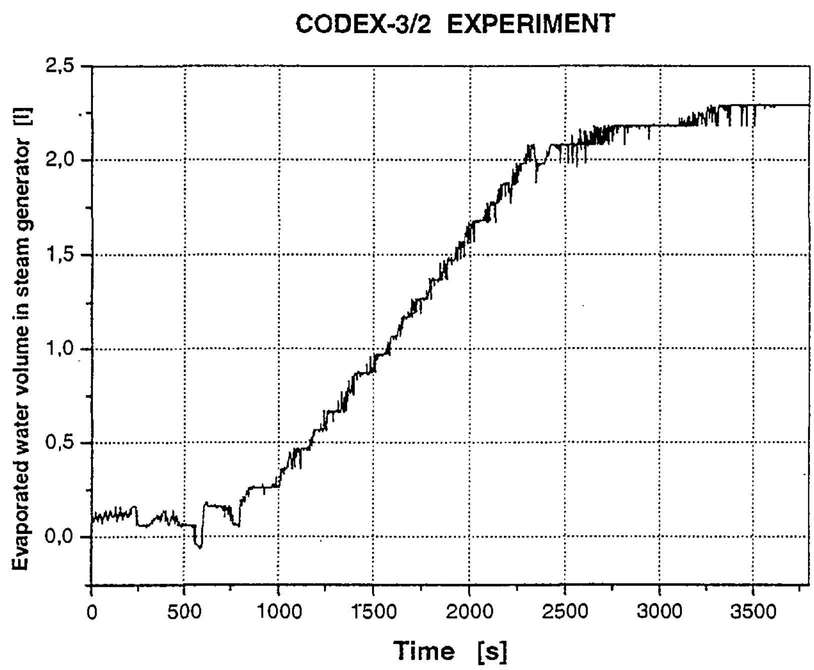


Fig. 44 LSG: evaporated water volume in steam generator
(based on water level measuerement)

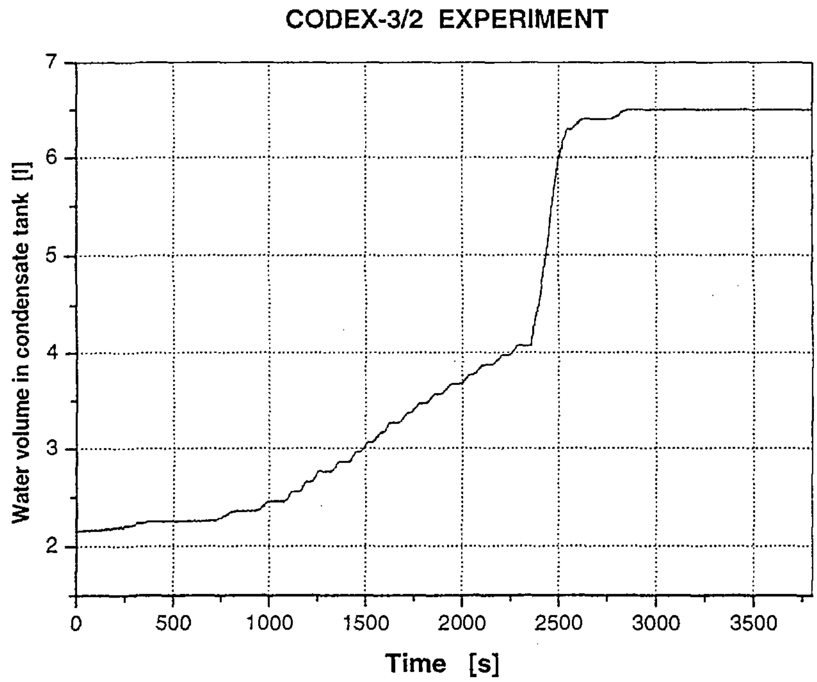


Fig. 45 LCOND: water volume in condensate tank
(based on water level measuerement)

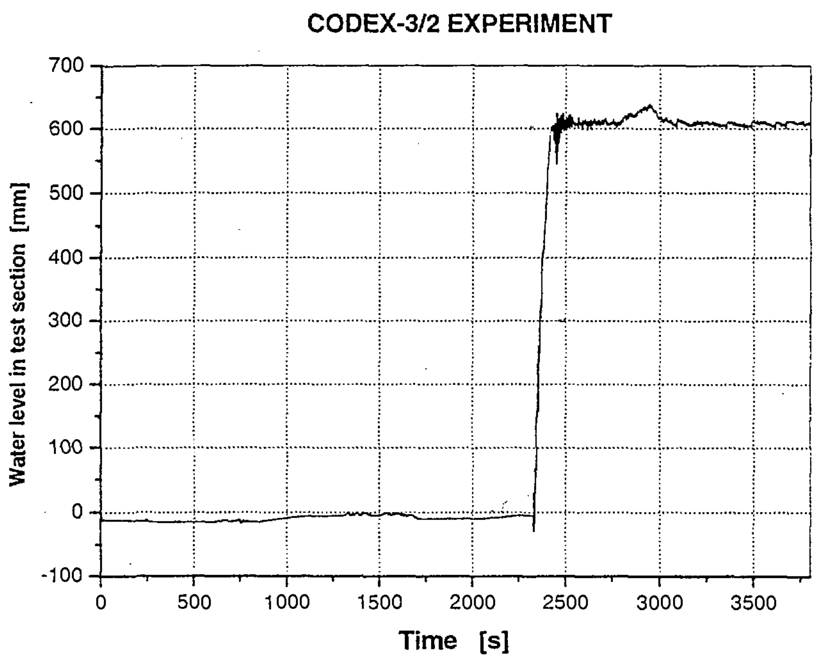


Fig. 46 LEVEL: water level in test section

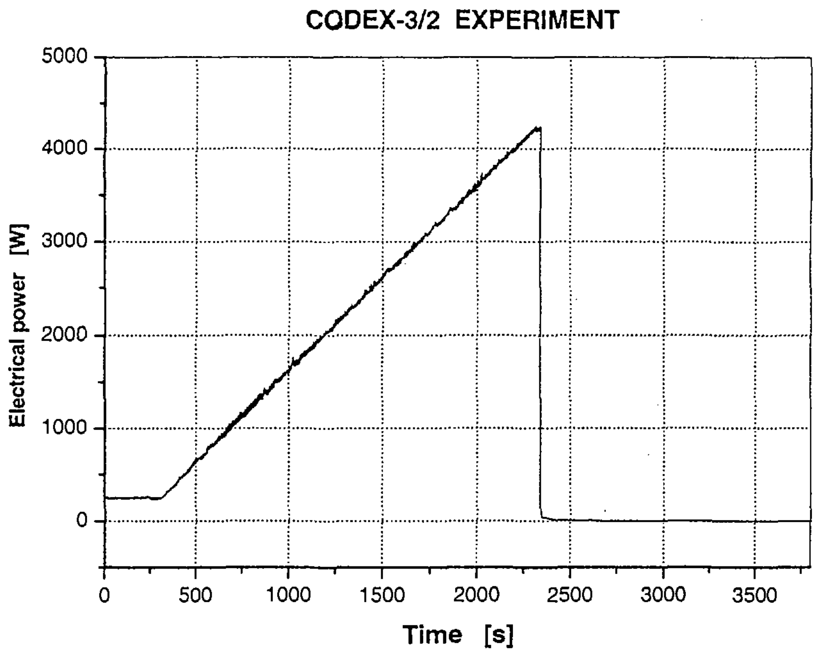


Fig. 47 POWER: total electrical power of the heater rods

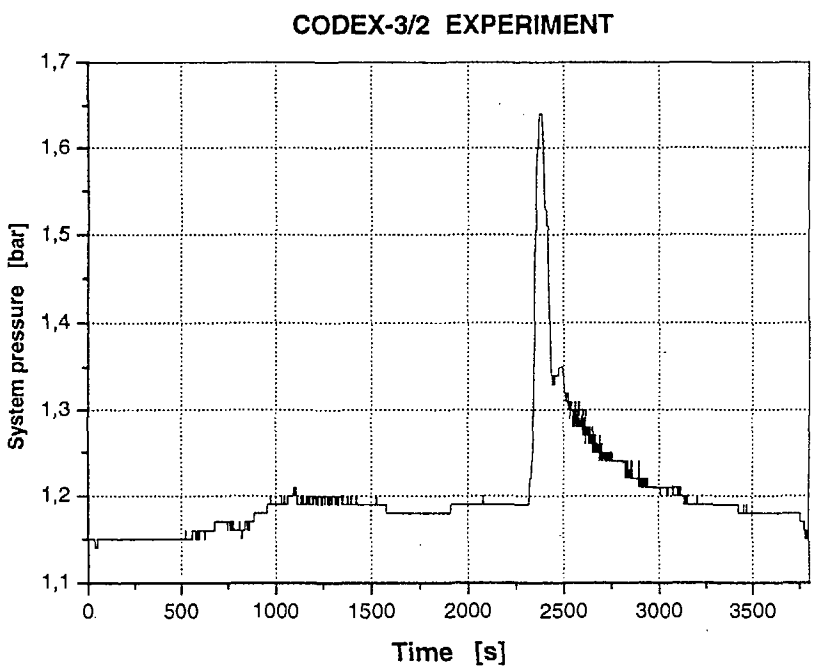
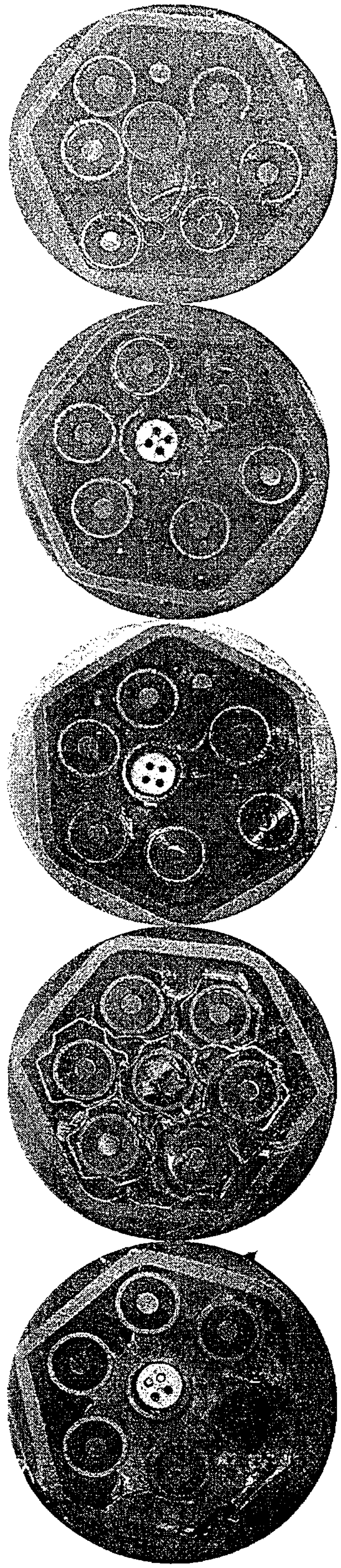


Fig. 48 PABS: system pressure



555 mm

480 mm

425 mm

300 mm

10 mm

Fig. 49 Cross sections of the CODEX-3 bundle

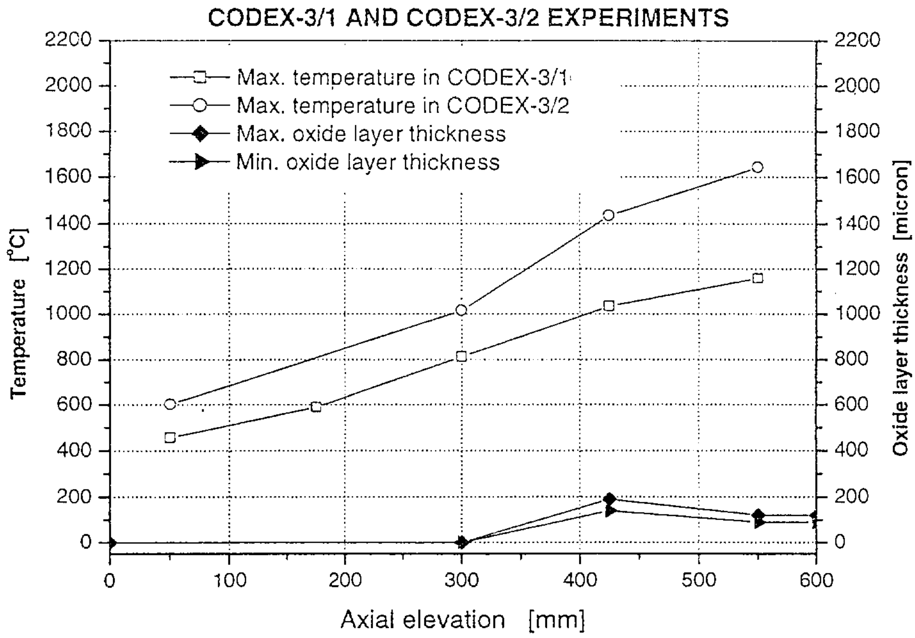


Fig. 50 Oxide layer thickness and temperatures in the CODEX-3 bundle

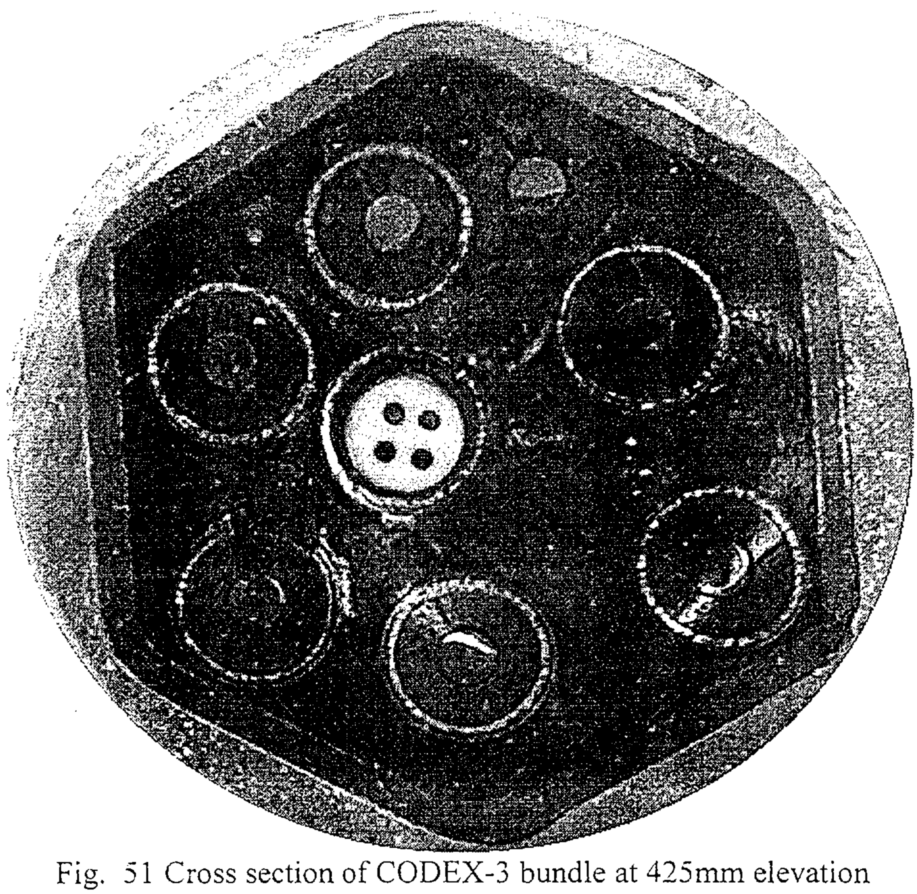


Fig. 51 Cross section of CODEX-3 bundle at 425mm elevation

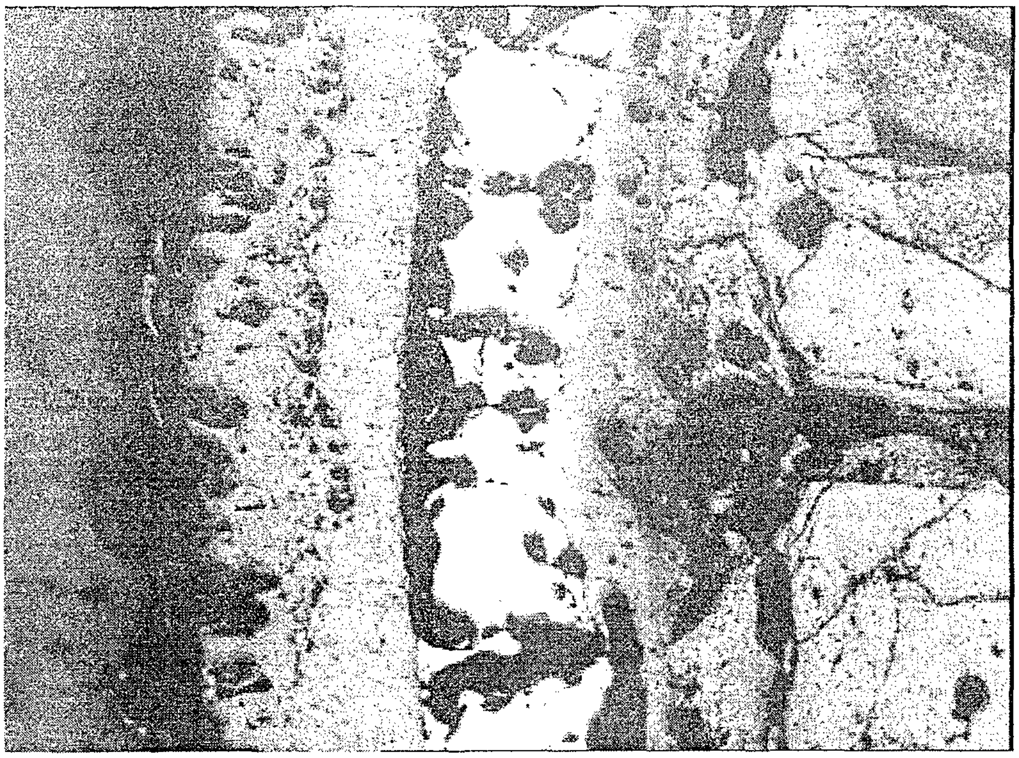


Fig. 52 Cross section of CODEX-3 bundle at 425mm elevation, signs of pellet-cladding interaction

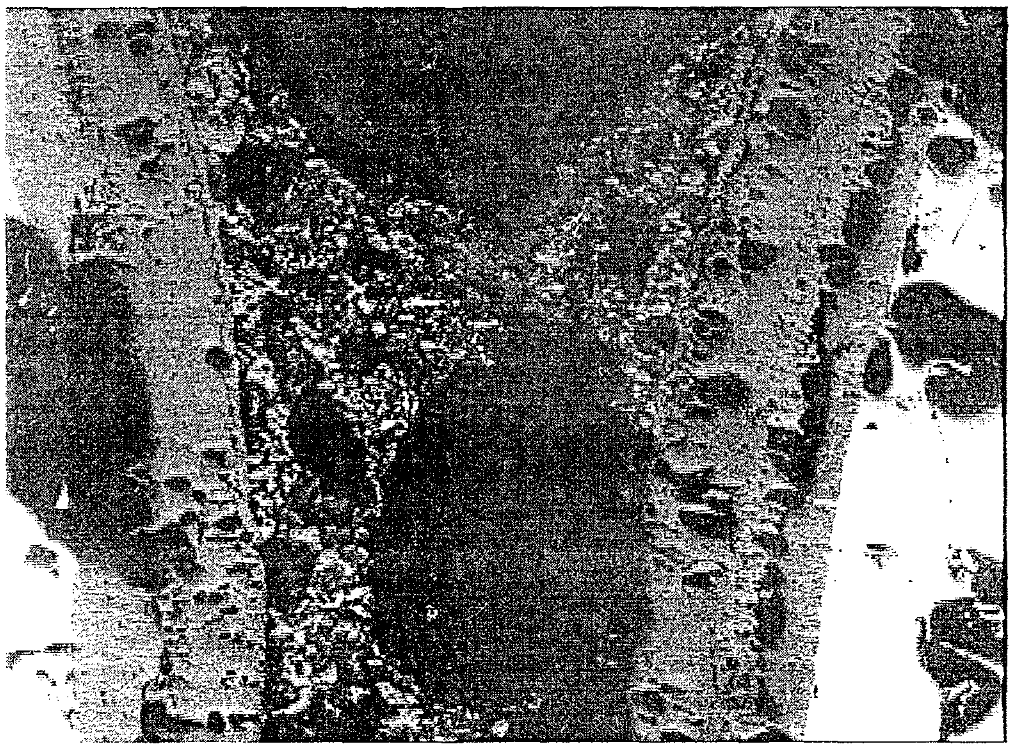


Fig. 53 Cross section of CODEX-3 bundle at 425mm elevation, debris in the place of contact of two claddings

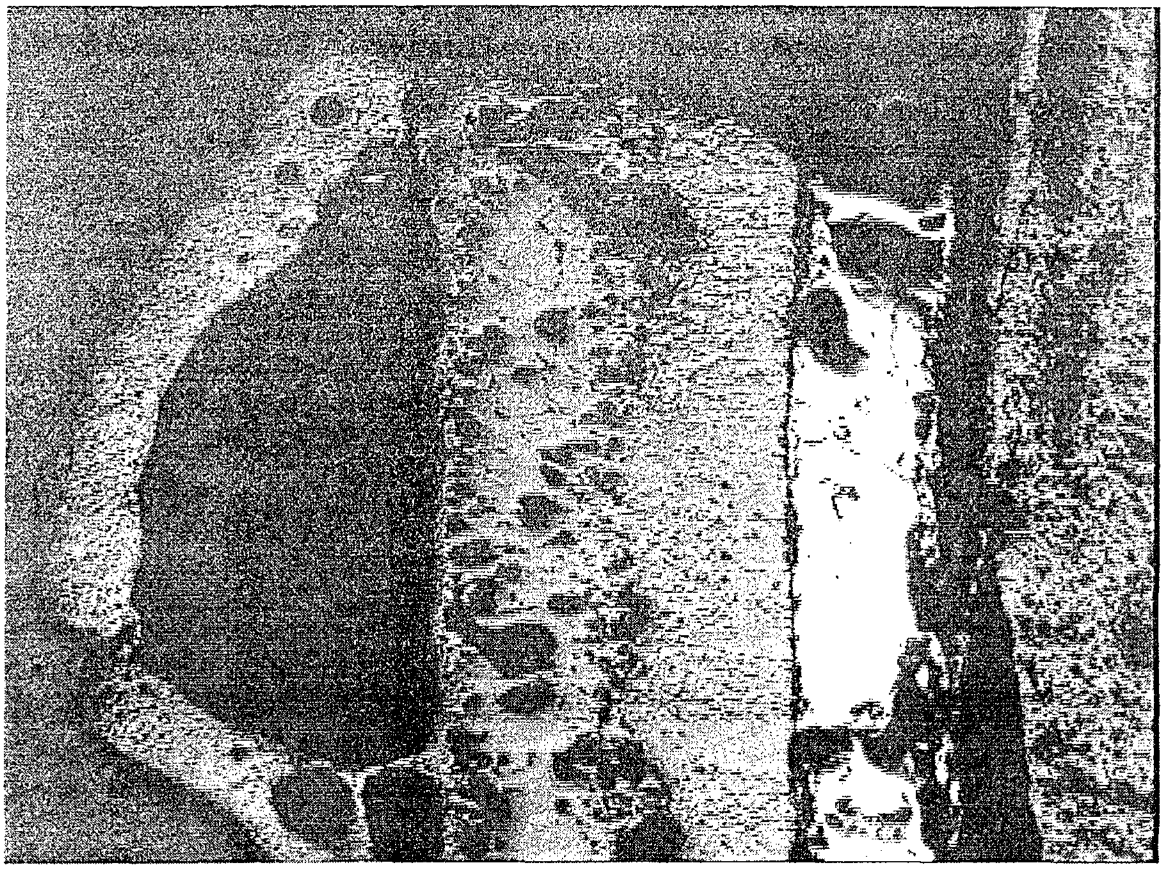


Fig. 54 Cross section of CODEX-3 bundle at 425mm elevation,
segment of the fragmented cladding

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KFKI-2000-02/G *Z. Hózer, L. Maróti, I. Nagy, P. Windberg:* CODEX-3/1 AND CODEX-3/2 EXPERIMENTS: QUENCHING OF HIGH TEMPERATURE VVER BUNDLES

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Készítette: Prosperitás Nyomdaipari Kft
Budapest, 2000. október