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Out-of-pile Bundle Experiments on Severe Fuel Damage (CORA-Program)

Objectives, Test Matrix and Facility Description

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Contents

Summary

As part of the Severe Fuel Damage Program by the German Nuclear Safety Project, out-of-pile experiments are being conducted at the Kernforschungszentrum Karlsruhe to investigate the damage behaviour of PWR fuel rod bundles under Severe Fuel Damage conditions (CORA-Progam). This report describes the objectives, the test matrix and the CORA-facility.

Out-of-pile Bündelexperimente zur Untersuchung schwerer Kernschäden (CORA-Programm): Testmatrix and Anlagenbeschreibung.

Kurzfassung:

Im Rahmen des vom Projekt Nukleare Sicherheit initierten Programms zur Untersuchung schwerer Kernschäden werden im Kernforschungszentrum Karlsruhe Out-of-pile Experimente für die Untersuchung der Schadensmechanismen an DWR-Brennelementen durchgeführt (CORA-Programm). Dieser Bericht beschreibt die Zielsetzung, die Testmatrix und die CORA-Versuchsanlage.

1. Introduction

LWR safety studies have shown that small break loss-of-coolant accidents and anticipated transient, in combination with the failure of the required safety systems, can lead to overheating of the fuel rods beyond the present design basis accident limits of 1200°C and result in severe fuel damage and fission product release. The TMI-2 accident, on the other hand, has demonstrated that a severe fuel damage transient will not necessarily escalate to an uncontrolled core meltdown accident, as was assumed in earlier risk studies.

Therefore, comprehensive research programs have been initiated in various countries to investigate the relevant damage mechanisms acting with increasing temperature on an uncovered core and to develop models for estimating the damage in the core, when the design limit temperature is exceede, but the transient can be stoped before core meltdown.

In the Federal Republic of Germany, the first experiments on fuel behaviour beyond the design limits were performed at KfK in 1976-78. These tests have shown that the most important phenomena at temperatures beyond 1200°C are the competing effects of cladding oxidation in steam and the interaction between the cladding material and oxide fuel, which can form liquid phases and destroy the rod structure far below the melting point of the fuel.

The objectives of the new Severe Fuel Damage (SFD) Program /1/ which was started by the Project Nuclear Safety (PNS) of KfK are:

- To quantifiy the relevant physical and chemical phenomena, which are already qualitatively known from earlier experiments,
- to develop models describing the extent of fuel damage for cladding temperatures exceeding the design limits, and
- to quantify the safety margins presently existing in the safety systems of operating reactors, and to explore the systems capability of ending a high tempeature transient before it can lead to an uncontrolled core meltdown.

2. SFD-Program description

The overall SFD-progam of KfK is divided into five major areas

- Separate effects tests on the high temperature oxidation of zircaloy and stainless steel and on the interaction between cladding and fuel, including equilibrium phase relations in the U-Zr-O-system /2, 3, 4/.

  (These tests are being performed under idealized, well-controlled conditions and provide the data base and generic understanding of the relevant phenomena which are necessary for the development of computer models.)

- Single rod and bundle experiments with electrically heated fuel rod simulators under realistic coolant conditions (CORA-Program).

  (These tests are used for the assessment of the models and their capability to describe the integral behaviour of fuel rods during severe accidents.)

- Out-of-pile tests on the long term coolability of severely damaged core geometries.

- Annealing tests on the fission gas release from severely damaged fuel.

- Code development concentrating on the relevant phenomena such as high temperature oxidation and fuel/cladding interaction.

  (These models will be integrated into the modular codes system SSYST originally developed for the description of fuel behaviour under LOCA conditions.)

Most of the investigations were begun in 1981 and will be finished in 1986.

During the construction of the CORA-facility, some of the tests, which did not need the special capabilities of the new facility, were run in the NIELS-facility. These included tests for the investigation of the influence of the temperature escalation due to the zircaloy/steam reaction and of the influence of absorber materials on the Severe Fuel Damage behaviour /5-15/.

3. CORA Program Objectives

The purpose of the CORA-Program is to identify and quantify the mechanisms and sequence of events causing severe damage of LWR fuel rods during heatup and flooding. Fuel rod temperatures up to 2000°C will be reached in a series of experiments with single rods and bundles. In particular, the objectives of the CORA program are:

a) to investigate UO_2 dissolution by liquid zircaloy as a function of heatup rate, maximum temperature, and pellet cladding contact pressure. The influence of ballooning, burst, on dissolution will be investigated.

b) to investigate temperature escalation form the exothermal zirconium/water reaction, including the effects of molten zircaloy runoff, oxide thickness, steam starvation and hydrogen. Inherently self-limiting mechanisms are of particular interest.

c) to determine the influence of spacers, absorber material, and control rod guide tubes on fuel rod behaviour and failure, of particular interest are the absorber cladding failure mode and time, the distribution of material through the core, potential blockage formation, and evaporation and aerosol formation.

d) to evaluate the fragmentation of severely embrittled fuel rods during quenching and to characterize the resulting debris. The influence of the oxygen distribution in the fuel and cladding, the presence of steam inside the rod, and the temperature dependence of the fragmentation behaviour will be investigated. Furthermore, the fragmentation behaviour of molten mateial will be examined.

e) to investigate the behaviour of liquid phases in fuel rod bundles and their interaction with steam, including the oxidation and freezing behaviour of (U, Zr, O) melts.

f) to perform out-of-pile reference tests for comparison with in-pile experiments at PBF, ACRR, NRU and PHEBUS.

It should be noted that the specific objectives are closely connected. An initially faster temperature rise rate, for example, produces less cladding oxidation. The thinner oxide layer increases the influence of the exothermic steam/water reaction on temperature escalation. The higher temperatures and lower oxygen content aid in the early melting of zircaloy and increase dissolution of the UO_2 by molten zircaloy. The faster temperature escalation rate also influences the failure of absorber rods and the effect of the absorber materials on fuel rod. The processes influence one another and occur, in part, concurrently.

Key safety issues to the in-vessel termination of a severe fuel damage accident are the energy generated by the metal/steam reaction and the associated hydrogen release, the transient coolability of the core, the long term coolability of a degraded core, and the release and transport of fission products. The objectives of the CORA-program address the oxidation of cladding and structural material and temperature escalation for understanding the extra energy production and hydrogen generation. The relocation, refereezing and fragmentation behaviour of the fuel and absorber rods on quench will provide a technical basis for evaluating the transient coolability of the core.

4. Test matrix

In the following, a compilation of the planned experiments is given. The experiments are intended to answer the questions formulated in the program objectives. The test matrix was developed by discussions of our initial proposal with international SFD experts. The test matrix is given on page 7.

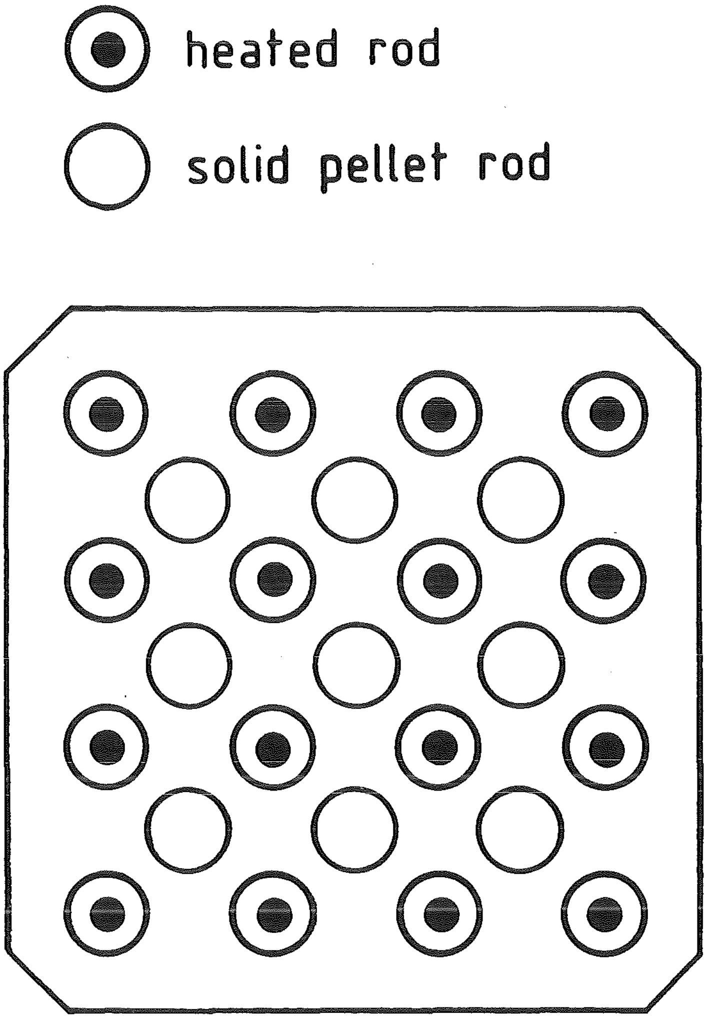
The test matrix contains 15 tests. The emphasis of the program is put on the tests with absorber rods as experiments in the NIELS facility have shown that absorber materials have a crucial influence on the damage behaviour of the bundle. The tests are concentrated at one heatup rate (1°C/sec). The majority of tests are chosen with steam starvation, as this simulates typical severe accident conditions.

The first three tests will be performed as reference tests without absorber material. All the tests of the matrix are planned to be run in steam except for the first test, which will be run in Argon. This test will be used for heat transfer calibration and the study of natural convection in the gap

Tentative Test Matrix for Out-of-Pile Bundle Experiments on Severe Fuel Damage (CORA)

Test No.	Cladding Temperature [°C]	System atmosphere	System pressure [bar]	Rod pressure [bar]	Ballooning and bursting	Test termination	Bundle configuration	Absorber/ Guide tube	Test objectives, remarks
1	< 1850	argon	1	1	none	cooldown in argon	type a	No / No	Scoping test, heat transfer calibration
2	1850	steam-	1	1	none	argon purge	type a	No / No	reference test w/o. absorber material
3	2400	starved	1	1	none	argon purge	type a	No / No	reference test w/o. absorber material
4	> 1200	"	1	1	none	argon purge	type a	1-Ag In Cd / Zry	control rod failure at low system pressure
5	1850	"	1	1	none	argon purge	type a	1-Ag In Cd / Zry	Zry clad melting
6	2400	"	1	1	none	argon purge	type a	1-Ag In Cd / Zry	extensive fuel liquefaction (monotectric melting)
7	2400	"	1	1	none	argon purge	type c	4-Ag In Cd / Zry	different bundle configuration
8	> 1200	"	10	1	none	argon purge	type a	1-Ag In Cd / Zry	control rod failure at elevated system
9	2400	"	10	1	none	argon purge	type a	1-Ag In Cd / Zry	pressure, to be compared with test 4 + test 6
10	2400	steam rich	1	1	none	argon purge	type a	1-Ag In Cd / Zry	influence of steam supply, to be compared with Test 6
11	> 1200	steam-	1	1	none	quench	type a	1-Ag In Cd / Zry	influence of quenching, to be compared with Test 4
12	1850	starved	1	1	none	quench	type a	1-Ag In Cd / Zry	influence of quenching, to be compared with Test 5
13	2400	"	1	1	none	quench	type a	1-Ag In Cd / Zry	influence of quenching, to be compared with Test 6
14	> 1350	"	1	1	none	argon purge	type a	1-Ag In Cd / SS	behaviour of SS guide tube, to be compared with Test 4
15	[a]	"	1	[a]	yes	argon purge	type a	1-Ag In Cd / Zry	influence of internal rod pressure

[a] to be determined   Initial heating rate: 1 °C/s.   State of pellets: as received



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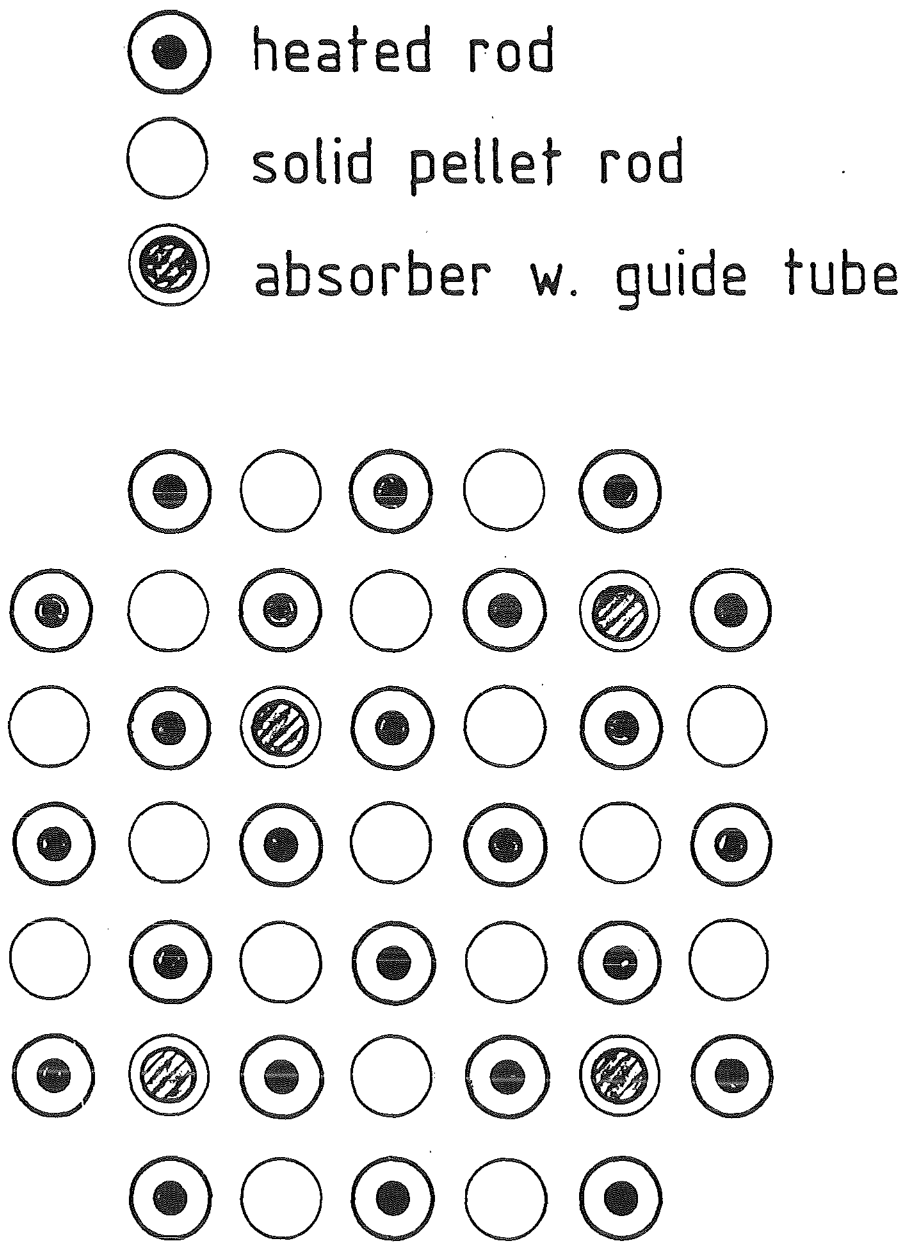

TYPE c TEST CROSS SECTION

between the shroud and radiation shield under conditions of no extra exothermal heating.

Test 2 and 3 without absorber rods are reference tests for comparison with later tests with absorber material at the same temperature.

The second group of tests investigate the damage in a bundle with a central absorber rod and its dependance on increasign temperature. To preserve the state of damage reached at maximum temperature, the bundle cool down is by an argon purge.

Test 7 using the maximum bundle size of 45 rods, should show that the bundle of 25 rods used in the other tests is sufficiently large. Tests 8 and 9 investigate the influence of the pressure on the interaction between Zry and UO_2 in the solid state and the failure of the absorber rods. These tests are to be compared with test 4 and 6 at pressure. The minimum pressure is about 1 bar overpressure (2 bar absolute) for reasons of controlling the flow in the test section. The maximum system pressure is 10 bar (limit for rupture disks 12 + 2 bar). To reduce the possibility of breaking the rupture disks which separate the high pressure section from the surge condenser, only a small number of tests will be run at maximum pressure.

Test 10 will be performed in steam rich condition. This test will provide information on the influence of steam supply.

The next group of tests (11-13) will investigate the influence of quenching on the damage behaviour for different maximum temperatures. These tests are to be compared to tests 4-6. Due to safety considerations quenching is only feasible at a low system pressure. This is due to the possibility of vapor explosions which could result in pressure spikes exceeding the design limit of the facility.

The influence of the guide tube material is investigated in test 14. In this test a stainless steel guide tube instead of a zircaloy guide tube is used.

The last test is intended to demonstrate the influence of ballooning and bursting on the SFD-behaviour. In this test the rods are ballooned and burst by internal overpressure.

5. CORA Facility Description

5.1 Introduction

CORA is a facility designed to investigate the behavior of PWR fuel elements under severe fuel damage accident conditions. Electrically heated single rod and bundle experiments up to about 2000 ^{o}C should be possible. The Severe Fuel Damage experiments were begun in the NIELS facility, which was already in operation. Compared to the NIELS facility the CORA facility can use larger and longer bundles. In the CORA facility it is possible to simulate the internal and external pressure of the fuel rods. This makes it possible to investigate the influence of balloon rupture and materials interactions on the SFD behavior. The CORA facility allows quenching of the bundle at the end of the test, so that a realistic investigation of fragmentation is possible. This section of the report describes the experimental and safety requirements of the facility, outlines the main components of the conceptual design, and discussēs the data measurement and instrumentation needs.

5.2 Basic Requirements of the Experiment Apparatus

The primary parameters of the test series are:

- fuel rod heatup rate,
- maximum cladding temperature,
- coolant flow rate,
- system pressure,
- fuel rod internal pressure
- cooldown rate (including quench).

The experimental apparatus must be able to vary these parameters in order to provide a meaningful test matrix. In addition, conventional safety requirements as well as radiological safety requirements must be met.

5.3 Specific Experimental Requirements

The primary experimental parameters include the following specific requirements:

Rod outside diameter:		10.75 mm
Rod Length:		2175 mm
Heated Length:		1000 mm
Bundle Size:	max.	45 rods
Pitch:		14,3 mm (7x7 array)
Number of the heated rods:	max.	24
Fuel simulator: heated rods:		annular UO_2 pellets
unheated rods:		solid UO_2 pellets or absorber rods within guide tube
Heatup rate:		0,5-4 K/s
Rod temperature:	max.	2000 °C
Single rod heating power:		16 kW, 160 W/cm
Bundle heating power:		96 kW, 40 W/cm
System pressure:	max.	12 bar
Rod pressure:	max.	100 bar
Steam flow:		0-12 g/s
Steam temperature at the lower bundle end:		100 - 1000 °C
Cooldown method:		electric power reduction and quench
Lifting speed of the quench cylinder:		0 - 4 cm/s

5.4 Safety Requirements

The experiments will use fuel rod simulators filled with annular or solid UO_2 pellets. The planned experiment conditions will allow the formation of molten cladding/fuel eutectics. The melt can contact the test chamber atmosphere, and therefore, the installation of a leakproof vessel and a ventilated control region around the apparatus is required. Furthermore, the pressure surge in the experimental apparatus due to steam generation during quench must be controlled.

5.5 CORA Conceptual Design

The design of the CORA facility is intended to provide boundary conditions for the out-of-pile fuel rod simulator bundles that are as realistic as possible. Since there are many plausible sequences of events and damage mechanisms within the framework of severe fuel damage, the facility must be very flexible in order to cover all desired experimental possibilities. In particular, the heatup rate and steam supply must be variable with time so that different cladding oxidation rates may be achieved. In addition, quenching of the simulators must be possible. Finally, after the experiments are completed visual and photographic examination of the bundle state must be possible without disturbing the simulators and/or resulting debris. The on-line and posttest data measurements must be extensive and through a variety of measurement techniques, somewhat redundant since data taking at temperatues above 1200°C is difficult at best.

5.5.1 Basic Concepts

A schematic diagram of the main components of the CORA facility is shown in Figure 1. Figure 2 is a scaled axial cross section of the components within the containment (Fig. 11). Numbers in parentheses in the following refer to Figure 1. The central part of the facility is the fuel rod bundle (2). The fuel rod simulators have a heated length of 1000 mm and "cold" ends at the top and bottom of about 500 mm. Different rod arrangements are possible, from single rod tests up to a maximum of 45 rods (a 7x7 configuration omitting the four corner rods). The bundle is enclosed in a zircaloy shroud (Fig. 12).

Simulation of the fission gas pressure inside the rods is achieved by directing inert gas from the fission gas simulating unit (14). The pressure of each rod is individually controlled.

A high temperature radiation shield (1) surrounds the bundle, leaving an annular space. The high temperature shield consists mainly of ZrO_2 and Al_2O_3/SiO_2 fiber ceramics (Fig. 16 + 17).

Below the bundle is the quench unit (3) with a water filled quench cylinder which can be raised around the bundle at hydraulically controlled speed. The cylinder can be filled by the quench water pipe (15). The cylinder is guided by three connecting rods to the bundle lower end, sliding on the seals in its bottom.

The bundle upper end is fixed in the bundle head funnel (23). The bundle head funnel consists mainly of an insulating plate, which provides the pressure seal of the single rods. The plate is connected by a funnel-shaped tube to the surge condenser (4) placed above the high temperature shield. The surge condensor is double-walled, leaving access to the bundle end fittings above the bundle head funnel.

In case of emergency, e.g. if there is too much evaporation during quenching, the volume of the surge condenser serves as a pressure suppression system using the spray water system (10). Normally the surge condenser is physically separated by four rupture disks from the actual high pressure section of the test. The water vapor not consumed in oxidation in the fuel rod bundle is, under normal conditions, condensed in the two vent condensers (5).

The non-condensable gas fraction (H_2) arising from zircaloy oxidation and the amount of argon cover gas added (13) are expanded to atmospheric pressure in the mixing chamber (7). By adding compressed air (16) into the mixing chamber (7), the hydrogen concentration is diluted to ensure that in the pipe leading to the vent air systems (9) the explosion limit is not reached.

The condensate tank (6) accomodates the condensate from the vent condensers and the excess spray water of the surge condenser. Condensed water of the quench unit is collected in the intermediate condensate tank (12). From here the water is moved by pressurized air to the condensate tank.

The vacuum pump (8) can evacuate the high pressure section. In this way it is possible to flush and fill the test chamber with inert gas before the beginning of the experiment. The bundle power (3) is provided by three separately controlled power supplies for three rod groups. This provides maximum experimental flexibility in the number of rods used and allows control of the radial temperature profile in the bundle. The radiation shield is built such that, if necessary, an electrical heating unit (18) already designed can be installed at a later time.

A steam generator provides the steam supply to the bundle (17). This generator is positioned within the containment contrary to the figure 1. The steam can be superheated to about 1000 °C in the superheater (22). The superheater can also be used to heat argon to a temperature of about 1000 °C.

All components that might be contaminated during or after an experiment have been located inside a containment structure (21). The containment serves as both a radiation control area and a pressure tight vessel protecting the local environment from more conventional dangers such as steam explosions. All components not in direct danger of contamination have been located outside the containment.

5.5.2 Experiment Conduct

Figure 3 illustrates the major steps in the conduct of a bundle experiment. During assembly (A) the high temperature shield will be lowered into the housing of the quench funnel, which can be moved to the side. This will allow easy access to the bundle during assembly.

The bundle is assembled from the fuel rod simulators on a work bench outside the containment. The rods are first screwed into the bundle head plate. This plate also provides penetrations for thermocouples which are attached to the fuel rod simulator and shroud surfaces, or are intended for steam temperature measurement. The preconnected bundle is lowered through the surge condensor by crane into its final position.

For the heatup phase (B), the quench funnel housing will be moved back and the high temperature shield raised and fastened into position. The entire apparatus will then be evacuated and filled with inert gas. During the experiment saturated or superheated steam will enter the bottom of the bundle, evenly distributed over the bundle cross section.

For quench experiments (C), the water-filled cylinder will be raised inside the high temperature shield to surround the bundle. The first inspection and photoducumentation is done in phases D and E. This picutre demonstrates an important advantage of the CORA facility. Since the quench cylinder is made of quartz, inspection of the bundle requires only the high temperature shield to be lowered (D). This can be done without disturbing the bundle which is still protected by the surrounding quench cylinder. We can thus investigate the appearance of the bundle at the end of the test in a fully undisturbed state. Protection by the quench funnel is especially convenient for quench experiments, where debris can be formed. In phase E the quench funnel also is lowered. In this state a complete photographic documentation of the bundle will take place, and initial samples from the bundle can be taken. From Figure 2 one can see that the bundle is situated in front of the observer on platform 2.

For transportation and later metallurgical observation, the bundle is filled with epoxy (F). The bundle is disconnected at the upper and lower ends and moved by crane to a room below the facility (Fig. 4), where the cutting, grinding and polishing can be done.

5.6 Main Components

5.6.1 Rod and Bundle Design

The largest possible bundle design consists of 45 rod positions in a 7x7 arrangement omitting the corner positions. The rod to rod pitch is 14.3 mm. Two types of fuel rod simulators can be used: electrically heated rods and unheated solid pellet rods. Absorber rods can also be used.

Figure 5 shows the side view of a heated rod. The heated fuel rod simulator is contained in a standard PWR zircaloy cladding tube. The central part of the internal rod consists of a 6 mm tungsten rod surrounded by annular UO_2 pellets. The length of this central part of the fuel rod simulator is 1000 mm. The tungsten heater is screwed into Mo-electrodes of 250 mm length which fit directly into the zircaloy cladding.

The molybdenum electrodes are connected to copper electrodes, and both are insulated from the zircaloy cladding by a flame-sprayed ZrO_2 layer. The zircaloy cladding is sealed to the copper electrodes by Swage-Lok fittings. To the upper fitting a capillary tube is connected, which allows connection to the pressure reservoir to produce the appropriate pressure within the fuel rod simulator. To the upper and lower copper electrodes flexible copper power cables are connected. The fuel rod simulator is screwed into the bundle flange from the top, producing a hermetic seal.

The unheated rod (Fig. 5b) consists of the original solid UO_2 pellets inside the original zircaloy cladding. The unheated rod is attached to the bundle flange from below. Internal pressurization is provided by a capillary tube connected at the upper end.

There are connections for a maximum of 24 heated rods. The rod to rod spacing is maintained by standard PWR spacers.

The absorber rods conform closely to the standard PWR design. The Ag80In15Cd5-absorber is contained in stainless steel cladding. This rod is positioned inside a guide tube. The proximity to the fuel rods is maintained by the spacers. The guide tube is made of zircaloy or stainless steel, and the spacer is made of zircaloy or Inconel.

Within the confines of the 7x7 array, practically any desired arrangement of rods is possible starting from a single rod. To get the most uniform heating, the heated and unheated rods are arranged alternately. In this arrangement an unheated rod is always surrounded by four heated rods. To minimize the influence of the tungsten heaters solid pellet rods can be concentrated in the inner region. With this arrangement we plan to take advantage of the strong exothermic heating of zircaloy/steam reaction. This can be done by first heating up in argon and then switching to steam.

The bundle head funnel is shown in Figure 6. The rods are screwed into the flange, which serves as a reference point on the bundle. Above the flange, the rod ends are staggered so that the power, internal pressure and instrument lines can be led away from the bundle. These lines are flexible connections.

The lower bundle connections, shown in Fig. 5, are such that the thermal expansion of the rods during heatup is accommodated by guiding the free ends of the rods through bushings. The rods are thus radially, but not axially, constrained.

The lower part of the bundle head funnel is water-cooled by the double-walled chamber seen in Figure. 6. To make the cooling more effective, the space around the rods inside this chamber is filled with water. A double-walled radiation screen is arranged around the flange and the lower part of funnel. Figure 7 gives a top view of the connections from the bundle head, which can hold a maximum of 45 lines for internal pressure, 34 thermo-couples, and 24 power connections.

5.6.2 Power Supply

24 of the 45 fuel rod simulators can be heated. The rods can be separately connected to one of three available power systems, each of which can supply a different voltage. By grouping the rods in various ways, we can influence the power distribution of the bundle. The number of rods can be different in the groups. Since the voltages and currents of the individual rods are measured, we know the power input for each rod.

The power input is controlled by the computer. The time sequence is programmed before the test. During the test the programmed power can be overridden and manually controlled. We can reduce the power of each of the three groups in steps or completely. In this way the temperature escalation from the exothermal reactions can be compensated. The electric heating is by direct current which was necessary to avoid eddy currents in the containment. Thus, each power unit consists of a variable transformer, a high current transformer, and a rectifier.

5.6.3 High temperature shield

To keep the heat losses as low as possible, the center bundle is surrounded by a high temperature shield during the test. The vertical and the horizontal cross sections of the high temperature shield are given in figures 8 and 9. The high temperature shield consists mainly of ceramic fiber plates. The two inner rows of plates are ZrO_2, and the two outer ones are Al_2O_3. The fiber ceramics are excellent insulators and have a low density which results in a low heat capacity. Thus the inner surface temperature can be closely follow the bundle temperature. The thermal shock behaviour of the fiber ceramics is also excellent.

The mechanical strength of the high temperature shield is provided by the outer walls of stainless steel. The fiber ceramic plates are attached to the stainless steel cover by ceramic nails. The inner ZrO_2 layer is 38 mm thick, and the outer Al_2O_3 layer 76 mm thick. The distance from the inner insulation surface to the center of the bundle is 152 mm.

The high temperature shield is located within the pressure tube. In the pressure tube a large number of flanges allow access to the bundle. Through these holes and their extension in the temperature shield, the bundle can be inspected during the test. Temperature measurements can also be made by two-color pyrometry using these as viewing ports.

At the lower end of the heated part of the bundle, steam is introduced through flange H22. The insulated ceramic tube, which provides the connection to the bundle shroud, can be withdrawn at the end of the test to allow for quenching.

An important design feature of the CORA facility is the ability to lower the high-temperature shield into the quench unit without disturbing the bundle. The mechanism for raising and lowering the high temperature shield can be seen in Figure 2.

5.6.4 Quench unit

The quench unit is shown in Figure 10 and 11. It is designed to move the water-filled quench cylinder over the heated bundle. It also provides the power connections to the bundle lower end and gives support during raising of the high temperature shield.

The side view of the quench unit in figure 10 shows that the movement of the quench cylinder is accomplished by the hydraulic central quench cylinder lift. The quench cylinder is guided by its bottom plate with the help of three copper rods. Only two of these are shown in figure 10. One can see two seals in the bottom on each of the rods. The quench cylinder guide rods at the upper end are terminated by the bundle lower end support plate. To this support plate are attached the flexible leads of the heated rods, of which 24 is the maximum number.

The quench cylinder guide rods are hollow, so it is also possible to route three thermocouples within each rod to the lower bundle end.

5.6.5 Steam generator and superheater

The steam generator together with the steam superheater is placed inside the CORA-containment. The steam generator consists of two separate components: In an oil heater with an electric power of 24 KW, oil is heated to 300 ^{o}C. In the actual steam generator the steam is produced in a heat exchanger by the heated oil. The steam is introduced into a directly heated coil. The electric power input into this coil amounts to 50 KW. By this a maximum temperature of 1000 ^{o}C can be reached. The superheater and the steam connection line to the high temperature shield are surrounded by an oil heated sheath. The steam generator supplies a maximum flow of 33 g/sec. The maximum temperature is 1000 ^{o}C.

5.6.6 Instrumentation of the bundle

For the operation of the facility a large number of measurements of the temperatures, flow rate, and other bundle and facility parameters are necessary. In this chapter only the instrumentation of the bundle is discussed.

As already mentioned in section 5.6.2 the electric power input of each heated rod is measured by recording the voltage of the group and the individual currents in the data aquisation system. The gas and steam input as well as the gas flow out of the system is measured. With a quadrupol mass spectrometer, the relative concentration of the different gases can be determined.

Thirty two thermocouples can be introduced through the bundle heat plate and nine thermocouples can be introduced to the three guide rods of the quench cylinder for measuring temperature in the bundle . Tantalum/Zry sheathed WRe-thermocouples are to be used. The surface temperature of the bundle can be also determined by two-colour pyrometers. In the insulation of the high temperature shield 32 termocouples are installed which will give a measurement of the axial and radial temperature distribution.
To visually record the bundle behaviour during the test, 10 endoscopes are installed with video cameras and 24/36 mm cameras.

6. Final Remarks

The construction of the CORA facility is finished. All the components have been tested seperately. With a bundle of 24 stainless steel rods with the dimensions of the fuel rod simulators the whole facility was tested up to a bundle temperature of 1000°C. In this way the proper functioning of all the components could be tested.

For the test of the facility up to high temperatures, a bundle of 24 fuel rod simulators of standard dimensions has been installed in which the UO_2 annular pellets were replaced by Al_2O_3 pellets, to avoid contamination of the facility. This will allow minor changes inside the facility to be easily made.

Since the CORA facility is installed inside the hall of the former research reactor FR2, a special license is necessary for the operation of the CORA facility. The license is now promised for the end of August 1986.

After receiving the license, the high temperature acceptance test of the facility will be performed. Following the successful performance of the acceptance test, the program test series will begin following the test matrix. The time required for one test will be about 6 to 8 weeks.

7. Acknowledgement

The authours would particularly like to thank A. Fiege, PNS-PL for the many valuable discussions on the objectives of the program and the test matrix. The design and the construction of the facility was possible only with the contribution of great number of co-workers. We want to express our gratitude to T. Vollmer, F. Brüderle, W. Butzer, J. Hanauer, G. Harbauer, W. Leiling, E. Mackert, N. Paroth, F. Schloß, W. Stahl, R. Töpfer.
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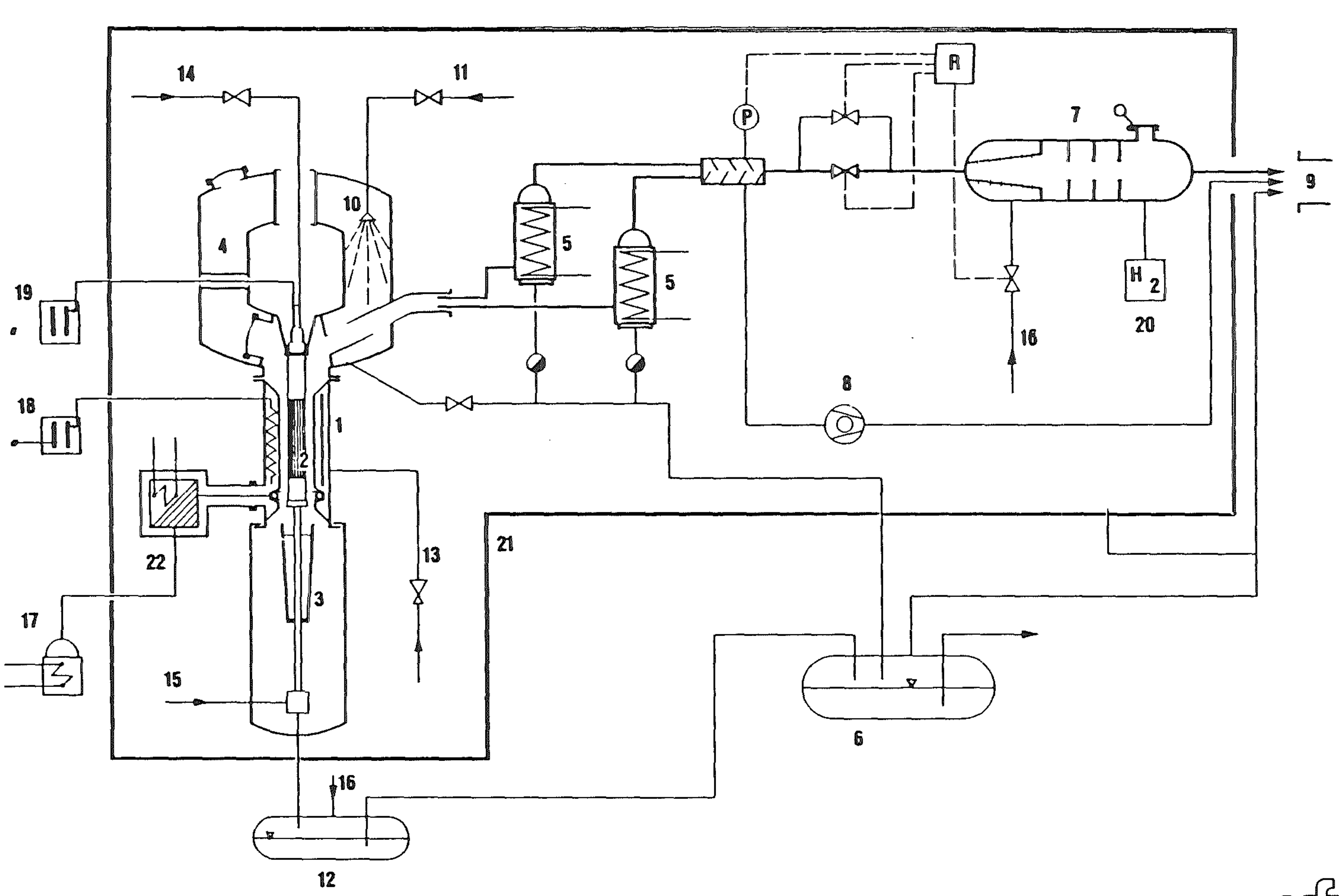
9. List of Figures



1 High temperature shield
2 Fuel rod bundle
3 Quench unit
4 Surge condenser
5 Outlet steam condensor
6 Condensate tank
7 Mixing chamber
8 Vacuum pump
9 Exhaust system
10 Spray system
11 Spray water
12 Intermediate condensate tank
13 Argon inlet
14 Fission gas pressure simulator
15 Quench water
16 Pressurized air
17 Steam generator
18 HT-shield power supply
19 Bundle power supply
20 H_2-monitor
21 Containment
22 Steam superheater
23 Bundle head funnel

IT - 1983

Fig. 1: **Scheme of the CORA Test Facility**

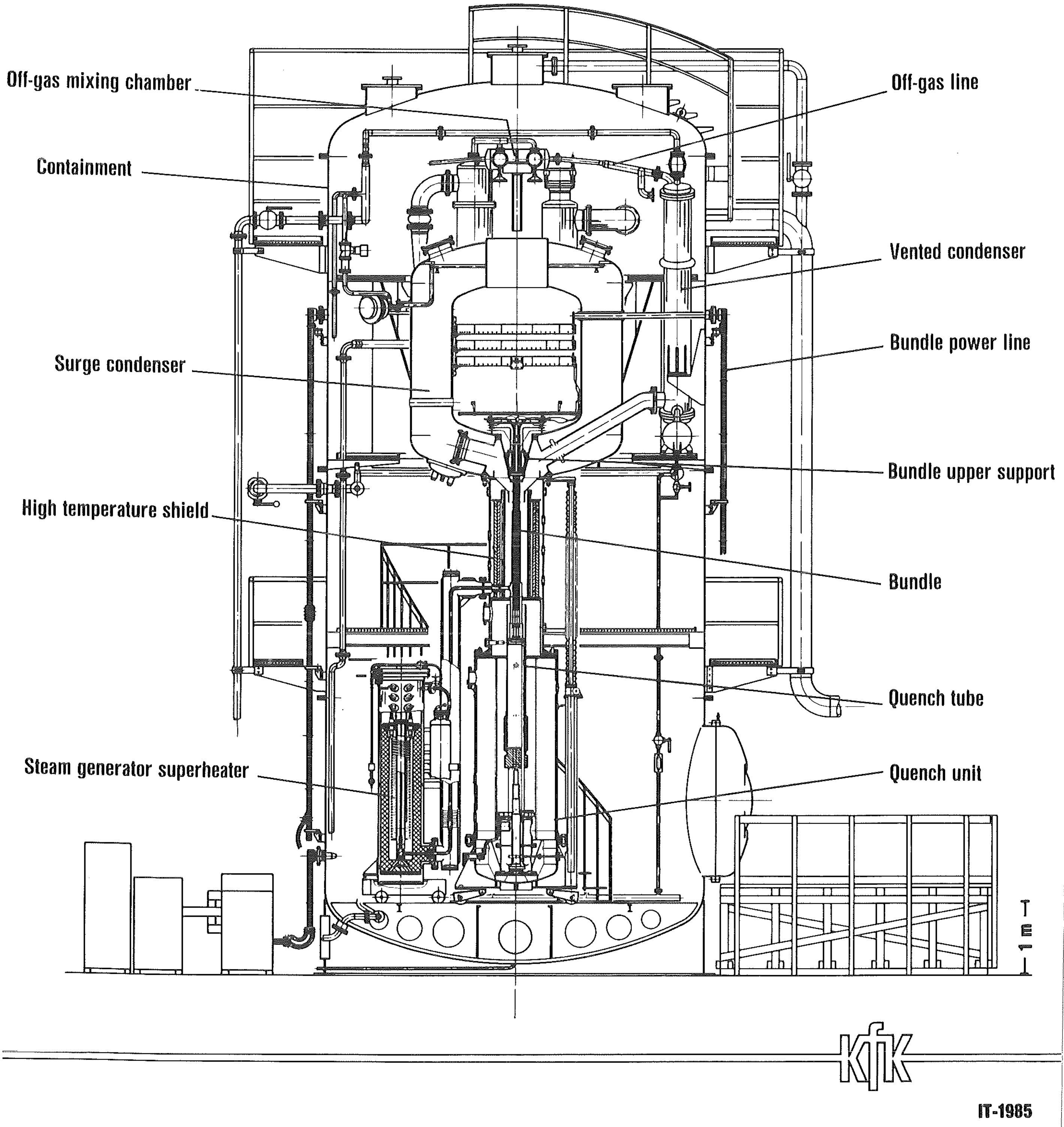


Fig. 2: **SFD-Test Facility CORA**

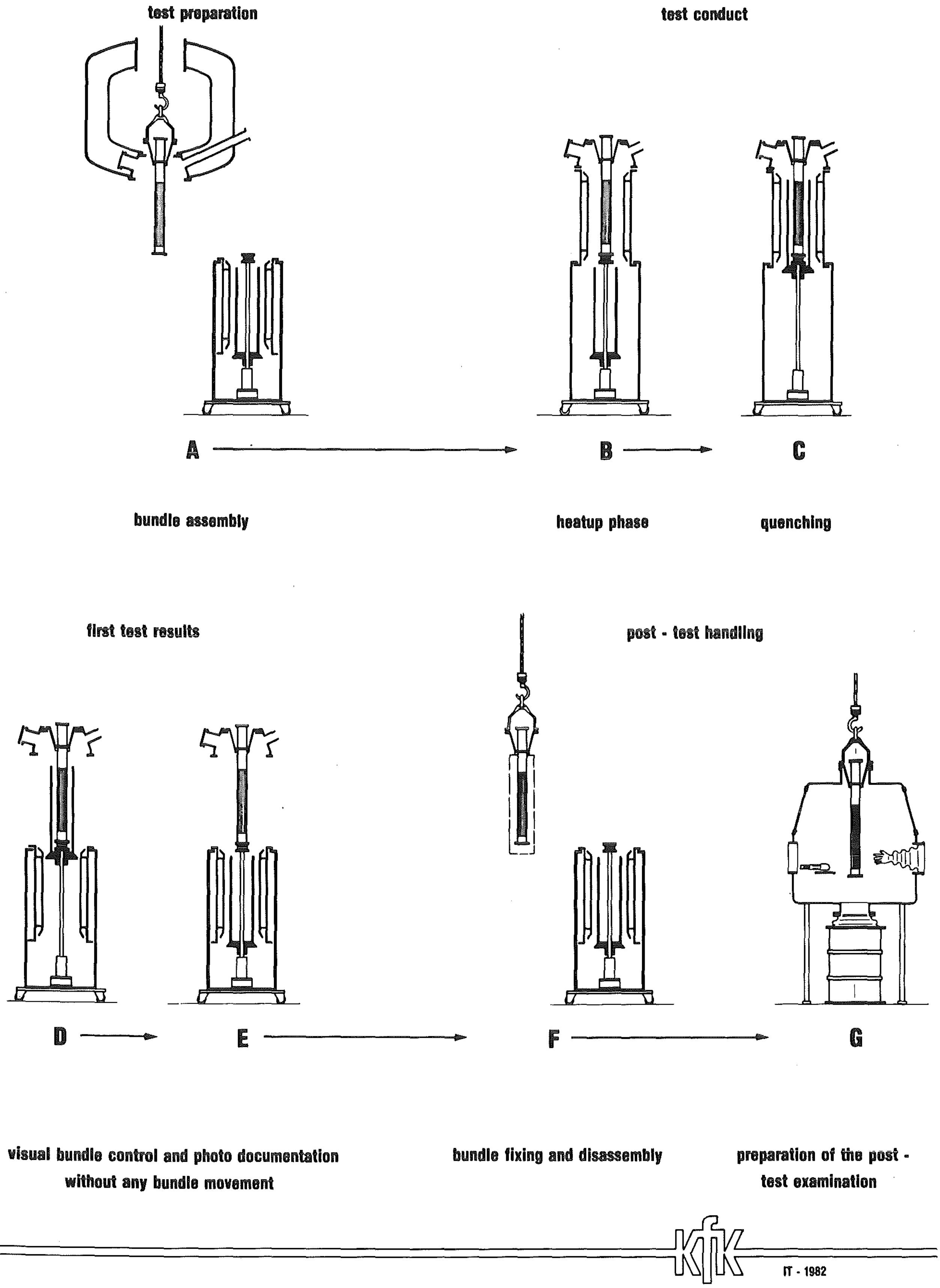


Fig. 3: SFD Test facility CORA, Testing sequence

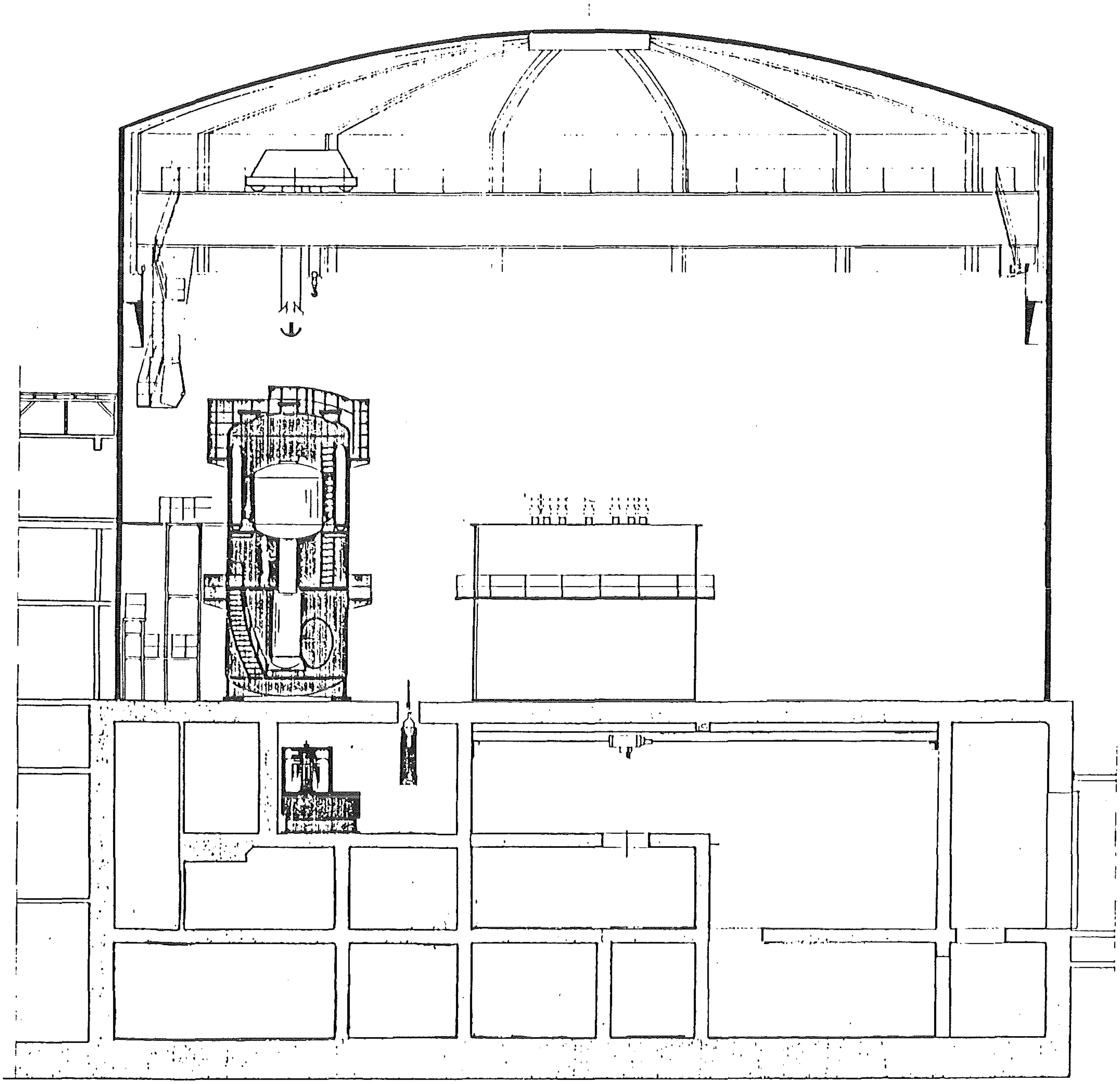


Fig. 4: CORA-Facility in the FR2-Experimental Hall

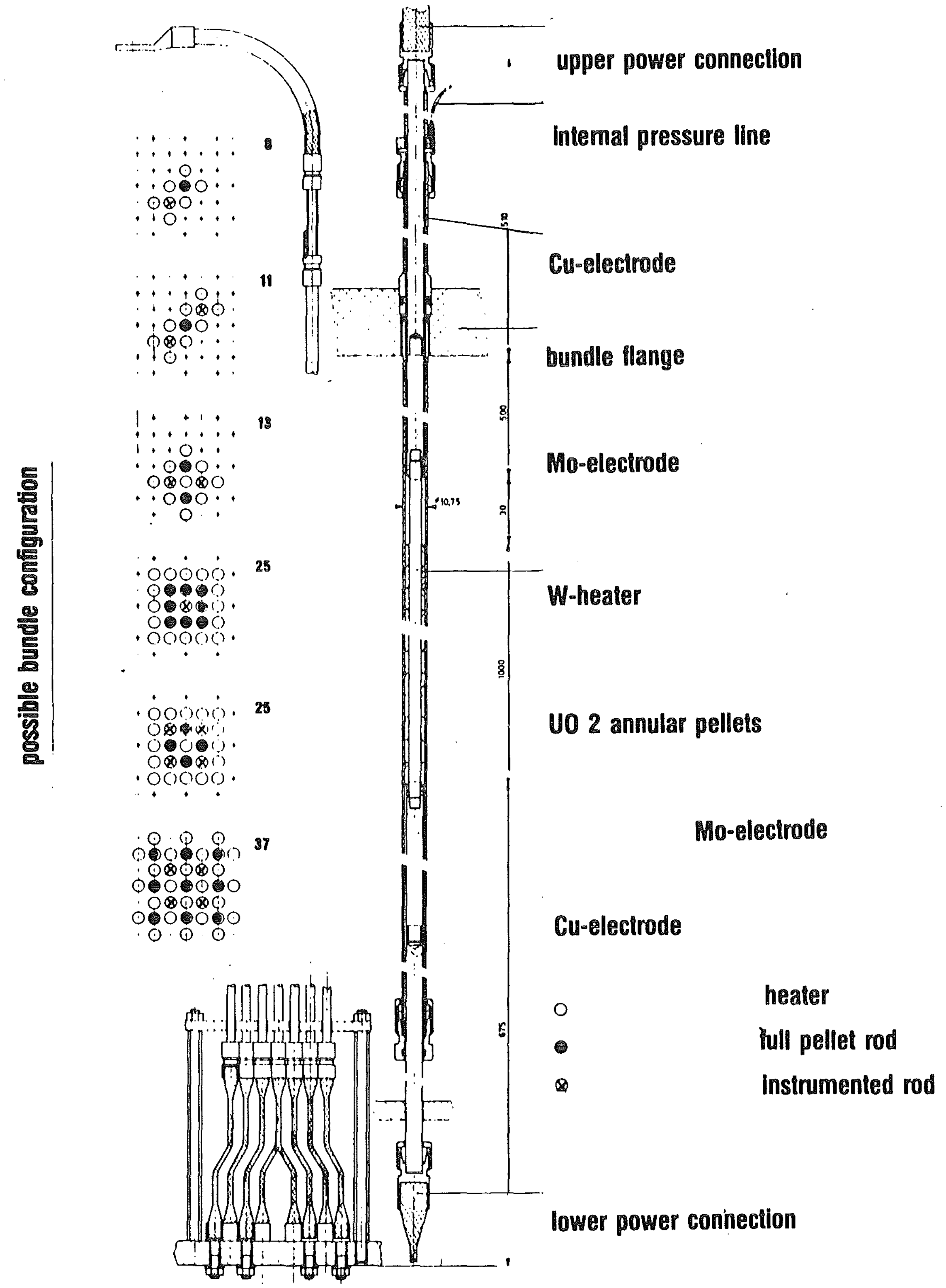


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Fig. 5:
SFD - experiment facility - heated fuel rod simulator

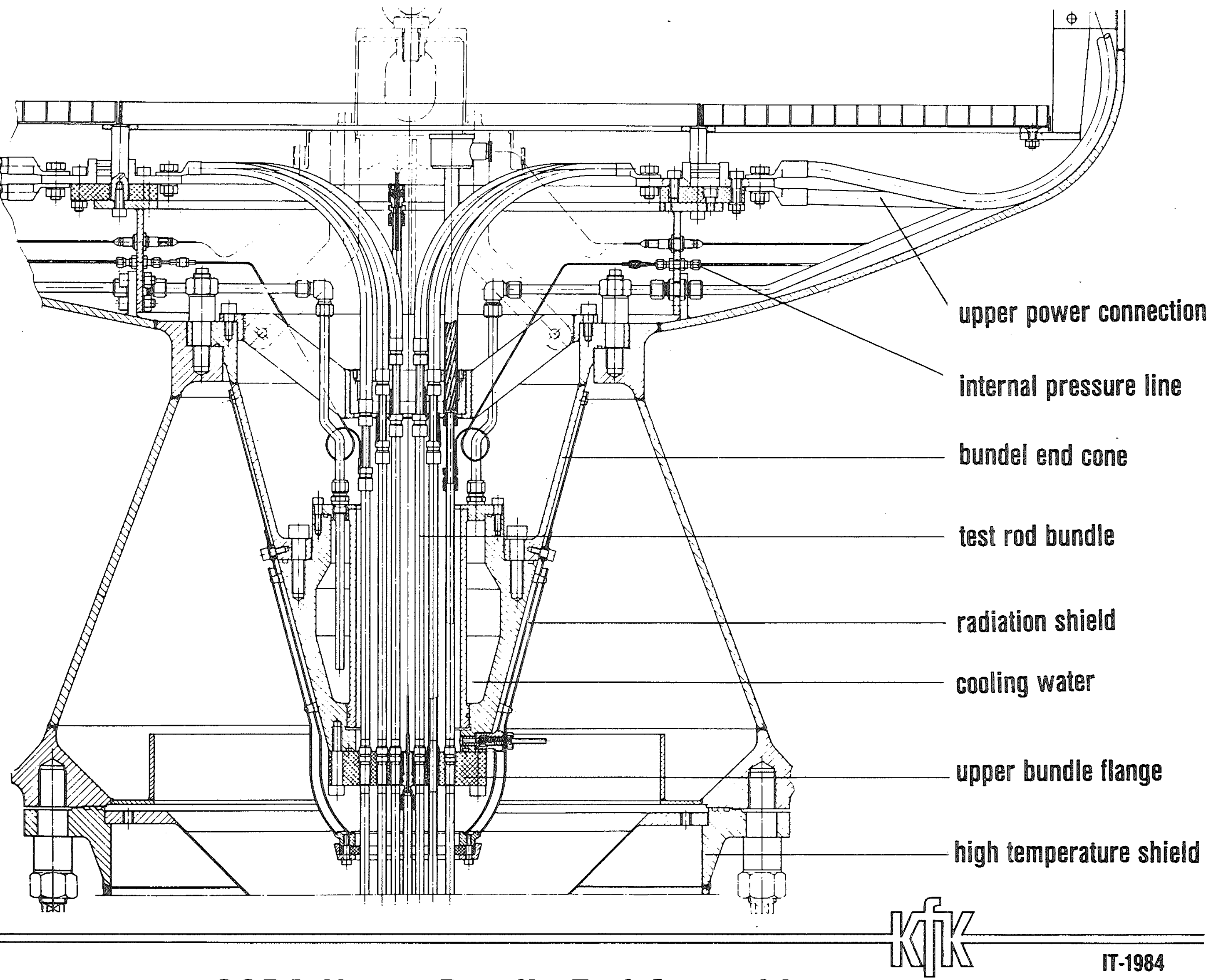

Fig. 6: **CORA Upper Bundle End Assembley**

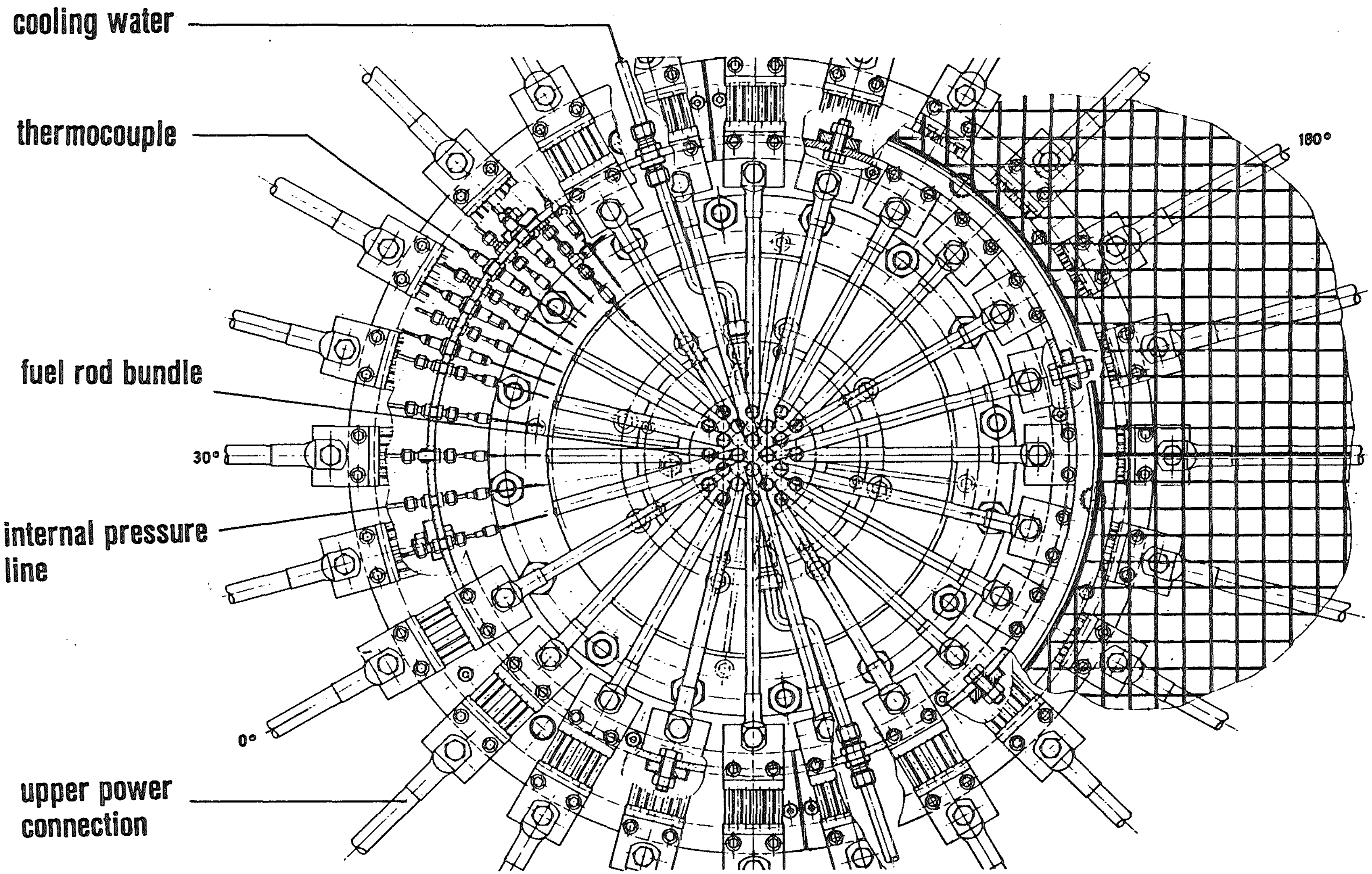

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Fig. 7: **CORA Upper Bundle End Assembly — Top View**

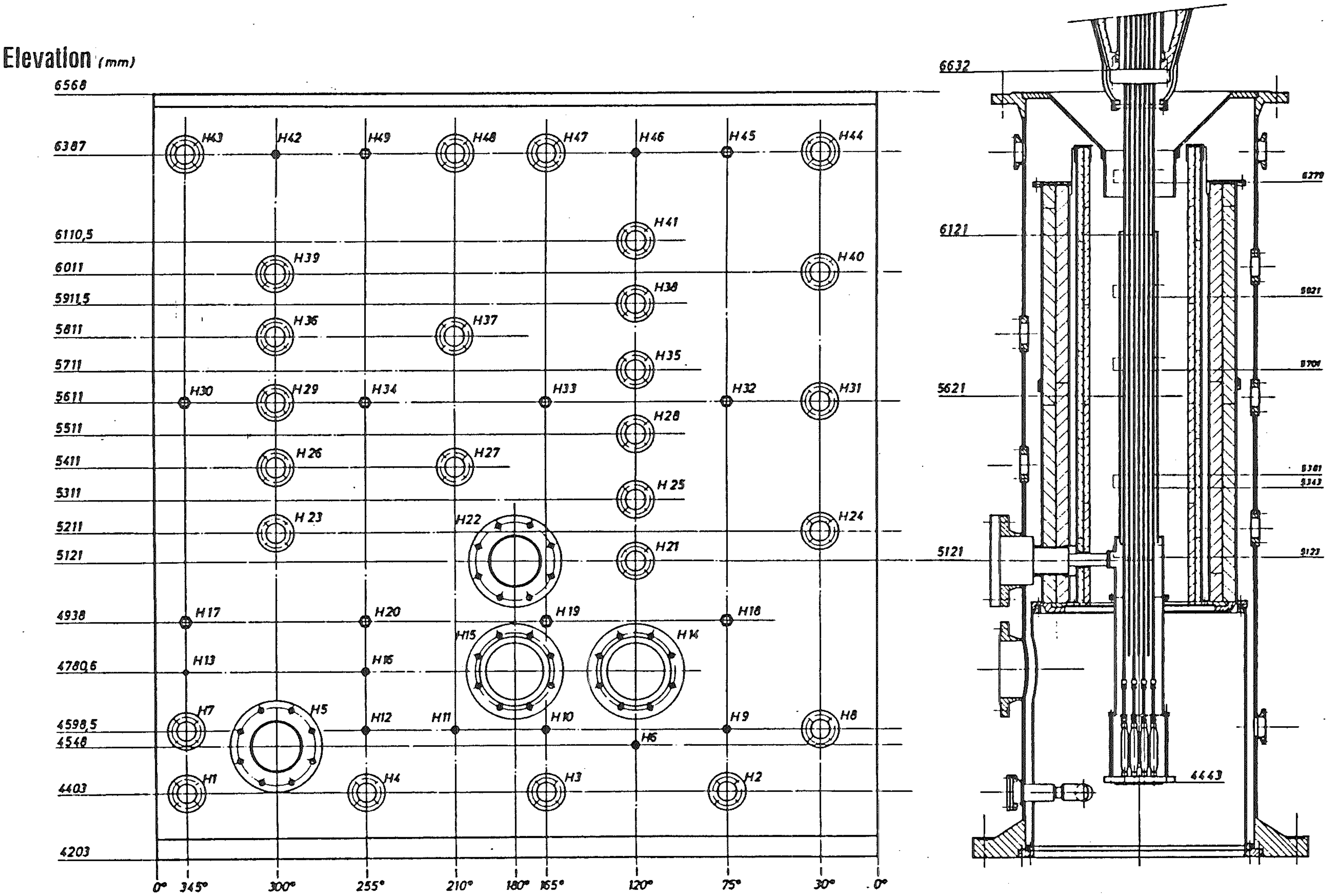

Fig. 8: **CORA Bundle and High Temperature Shield Assembly**

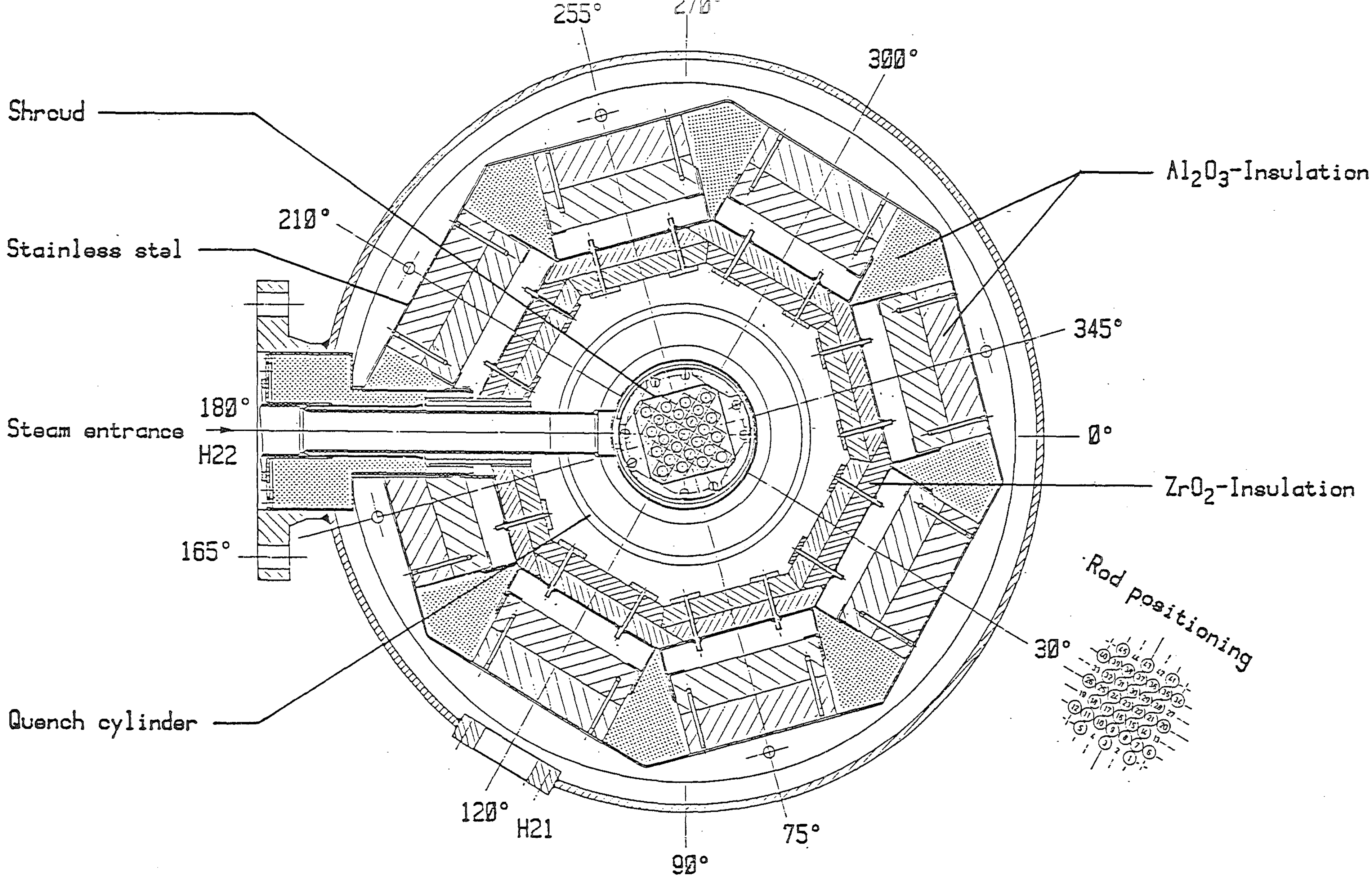

Fig. 9 : Cross section of high-temperature-shield

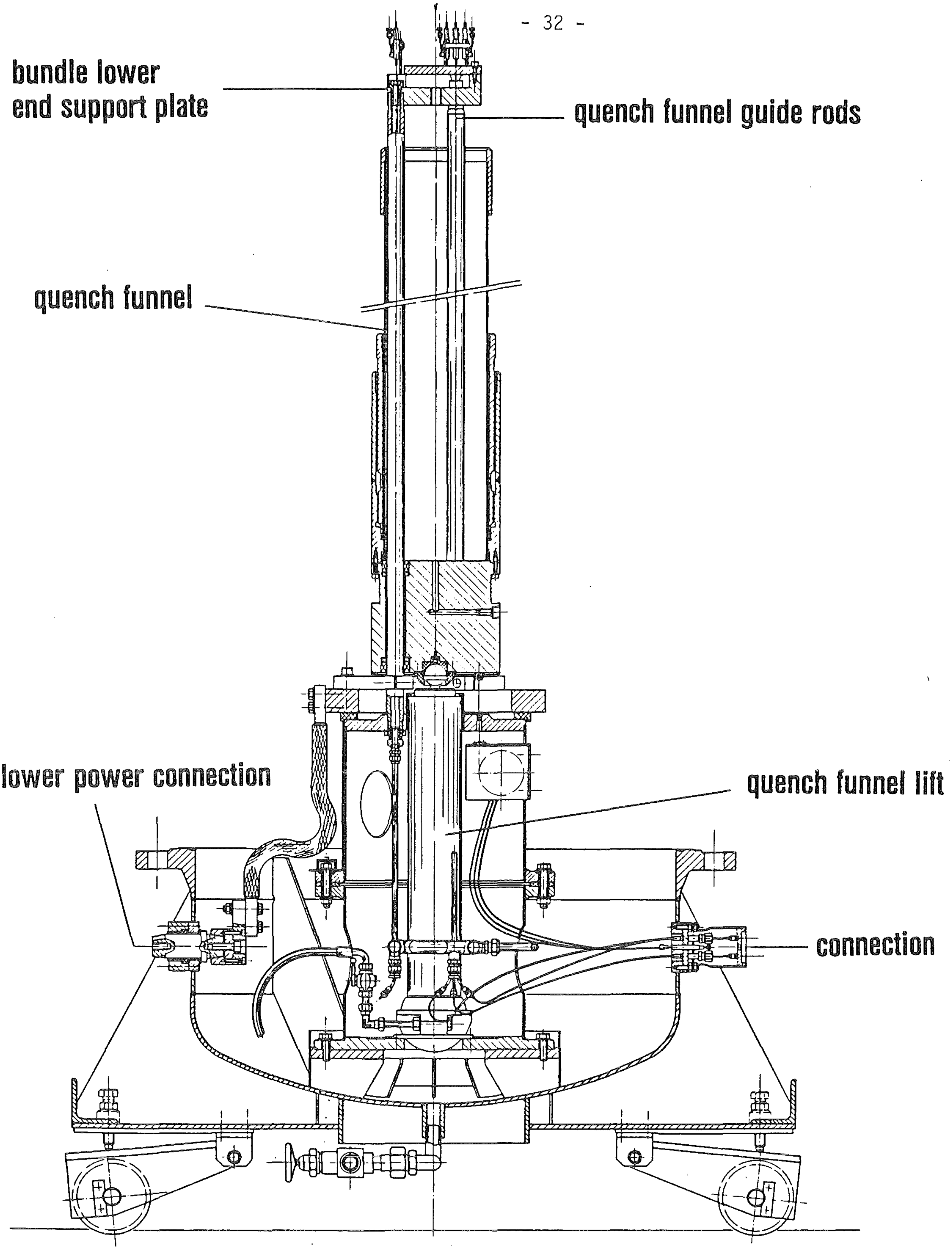

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Fig. 10: **Quench Unit**

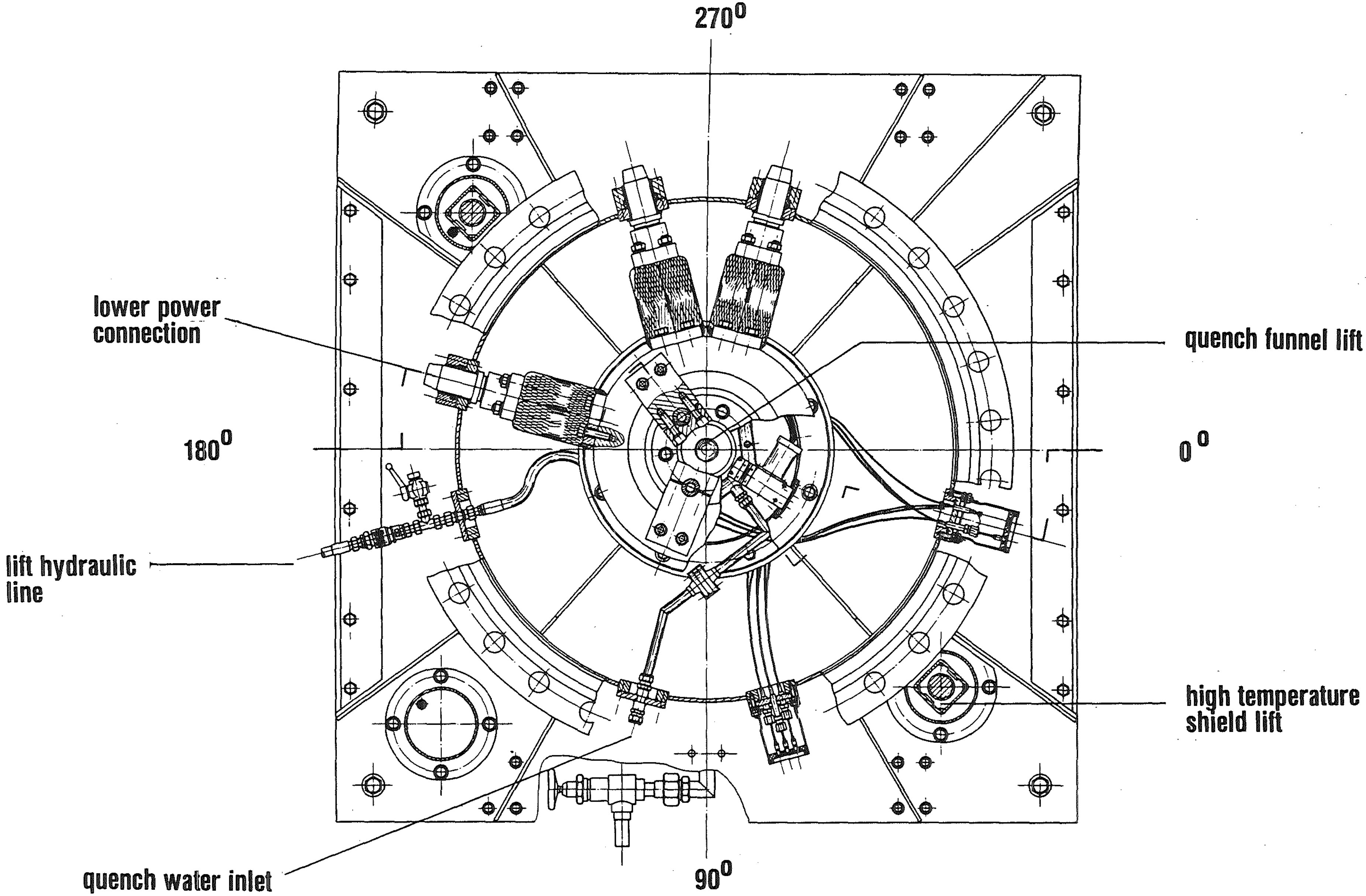

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Fig. 11: **CORA Quench Unit Top View**

Fig. 11a:

CONTAINMENT AND HIGH TEMPERATURE SHIELD WITH QUENCH UNIT OF CORA

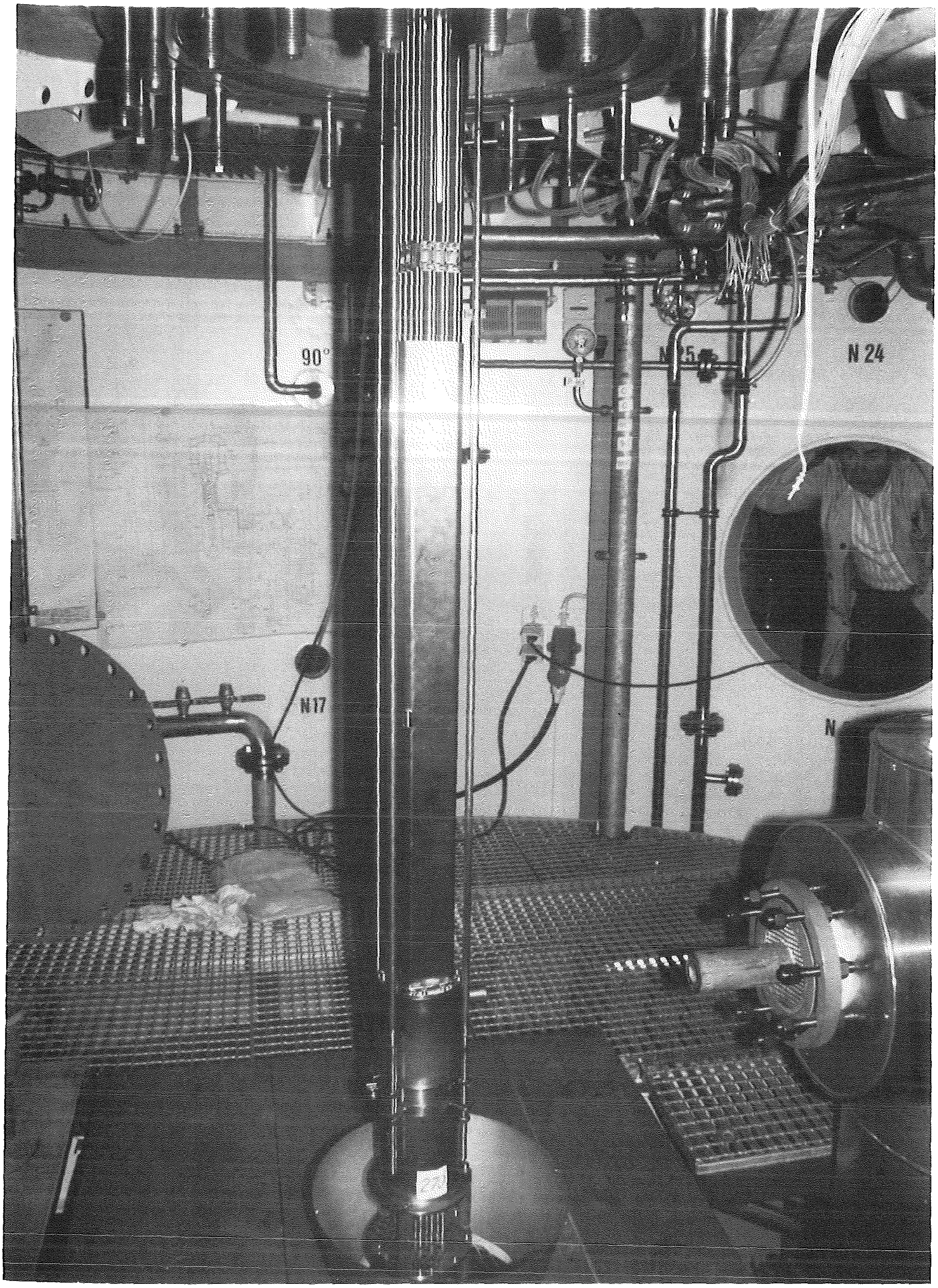

Fig. 12:

CORA-bundle with high temperature shield removed (type a)

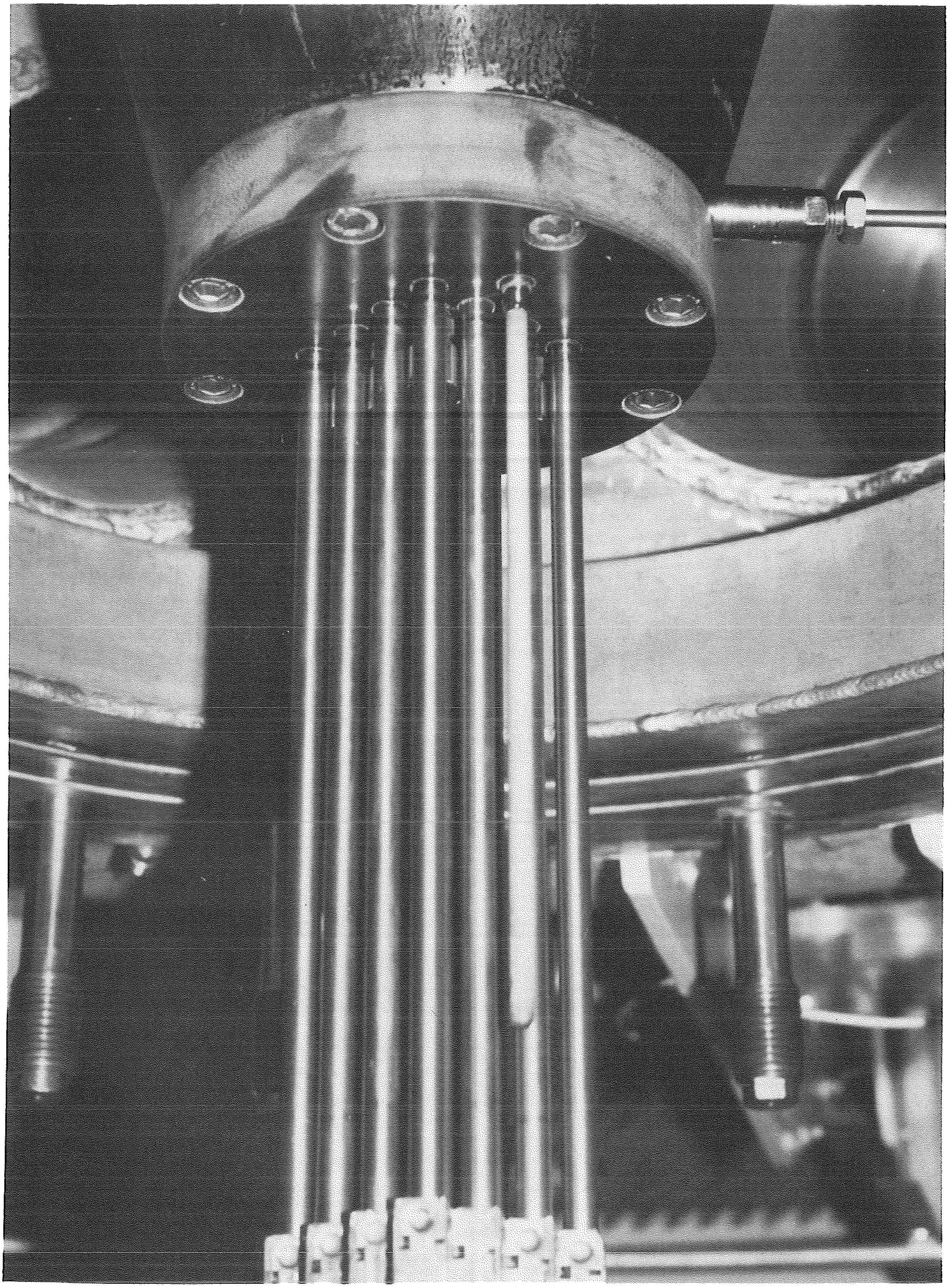

Fig. 13: BUNDLE UPPER END

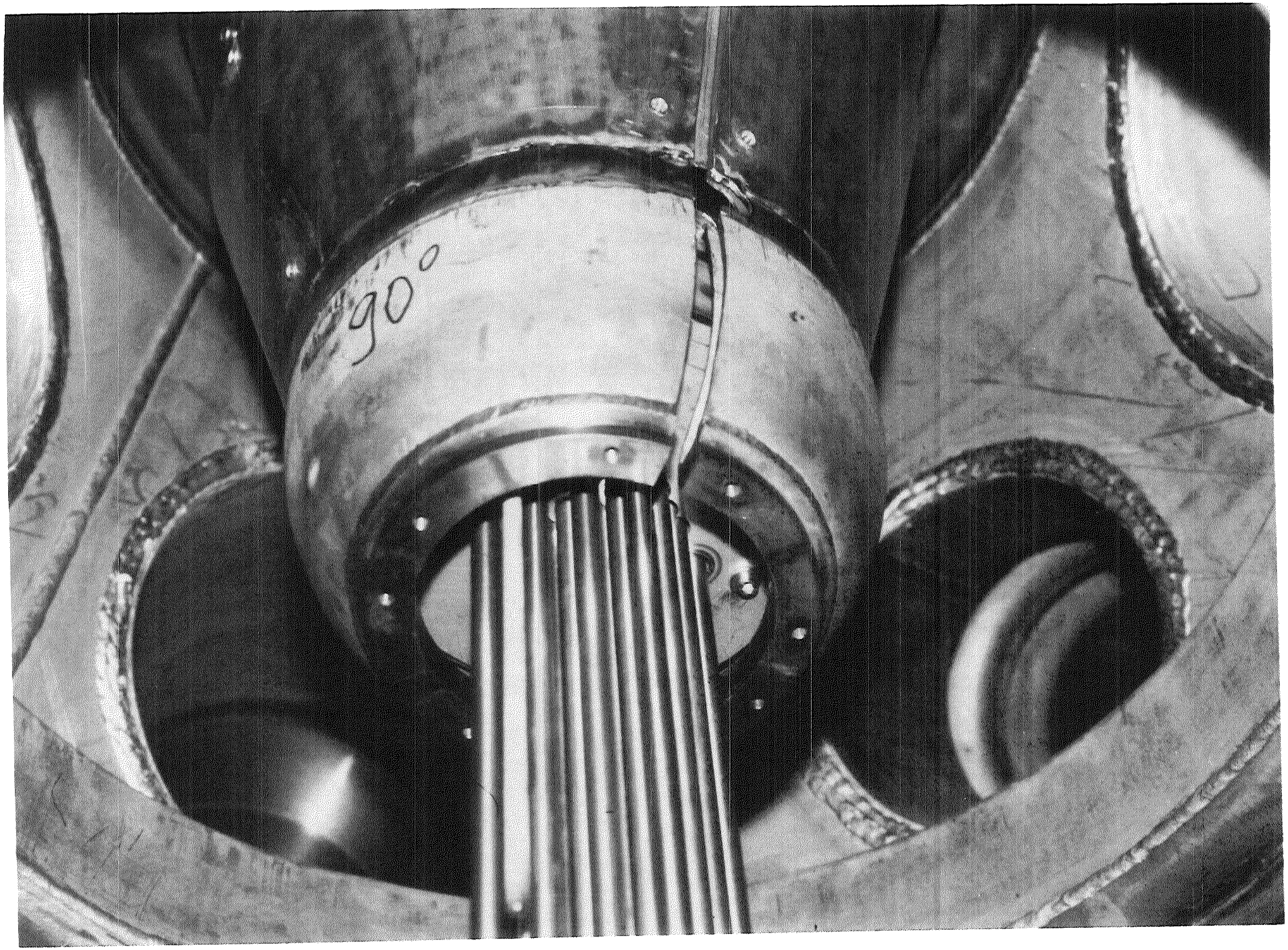

Fig. 14: BUNDLE UPPER END WITH RADIATION SHIELDS FOR HEAD PLATE

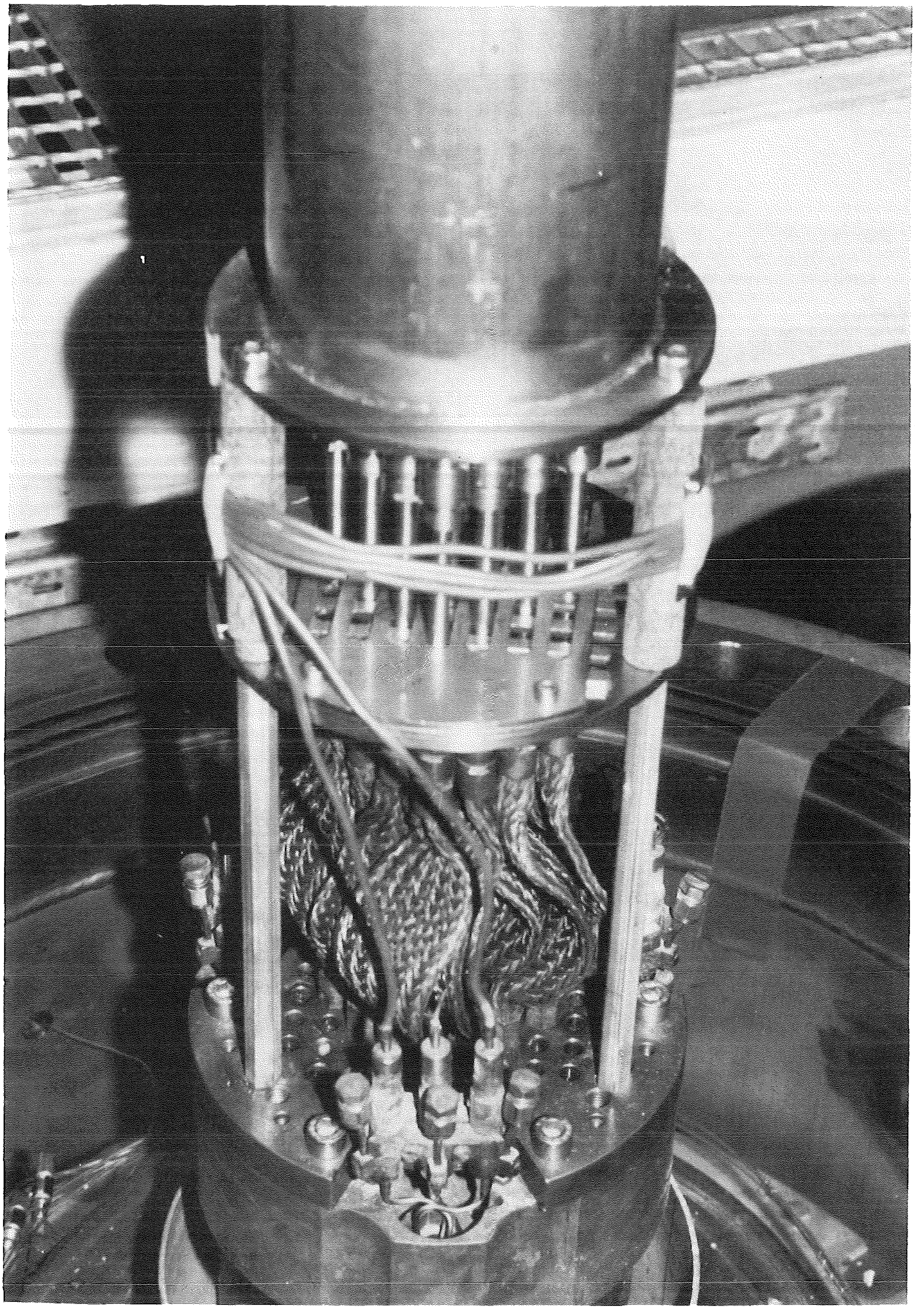

Fig. 15: BUNDLE LOWER END

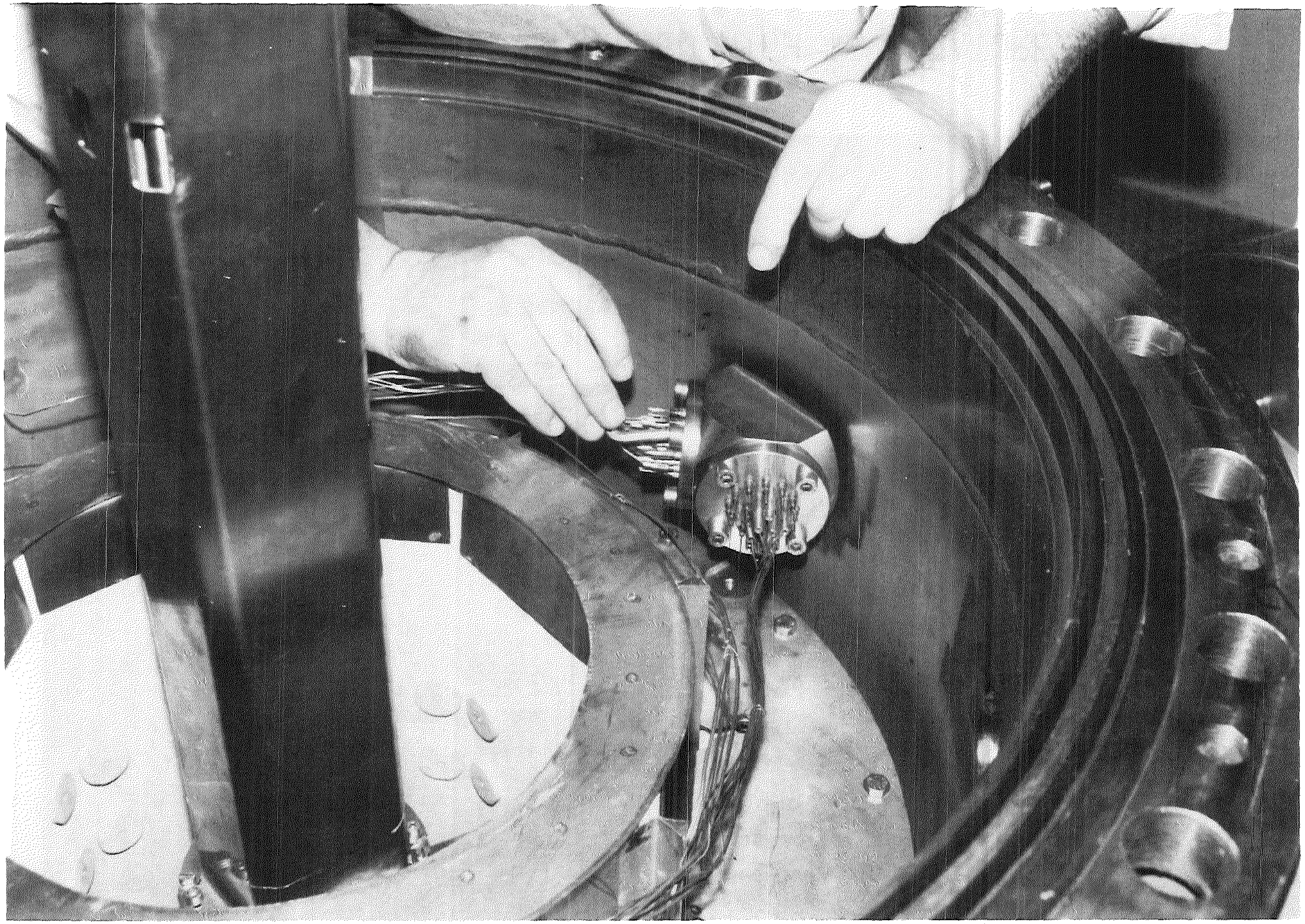

Fig. 16: BUNDLE INSIDE HIGH TEMPERATURE SHIELD

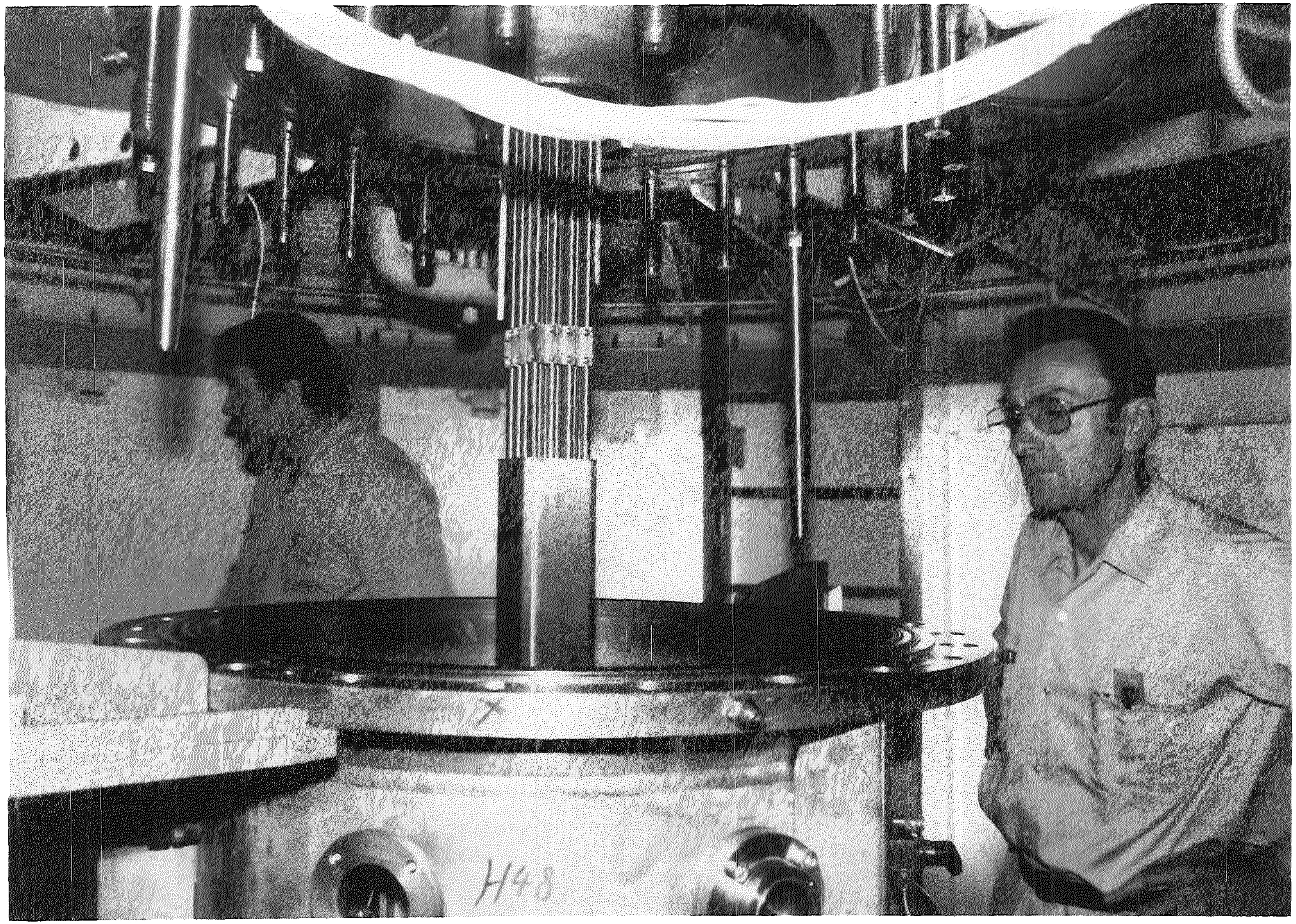

Fig. 17: PARTIALLY LOWERED HIGH TEMPERATURE SHIELD

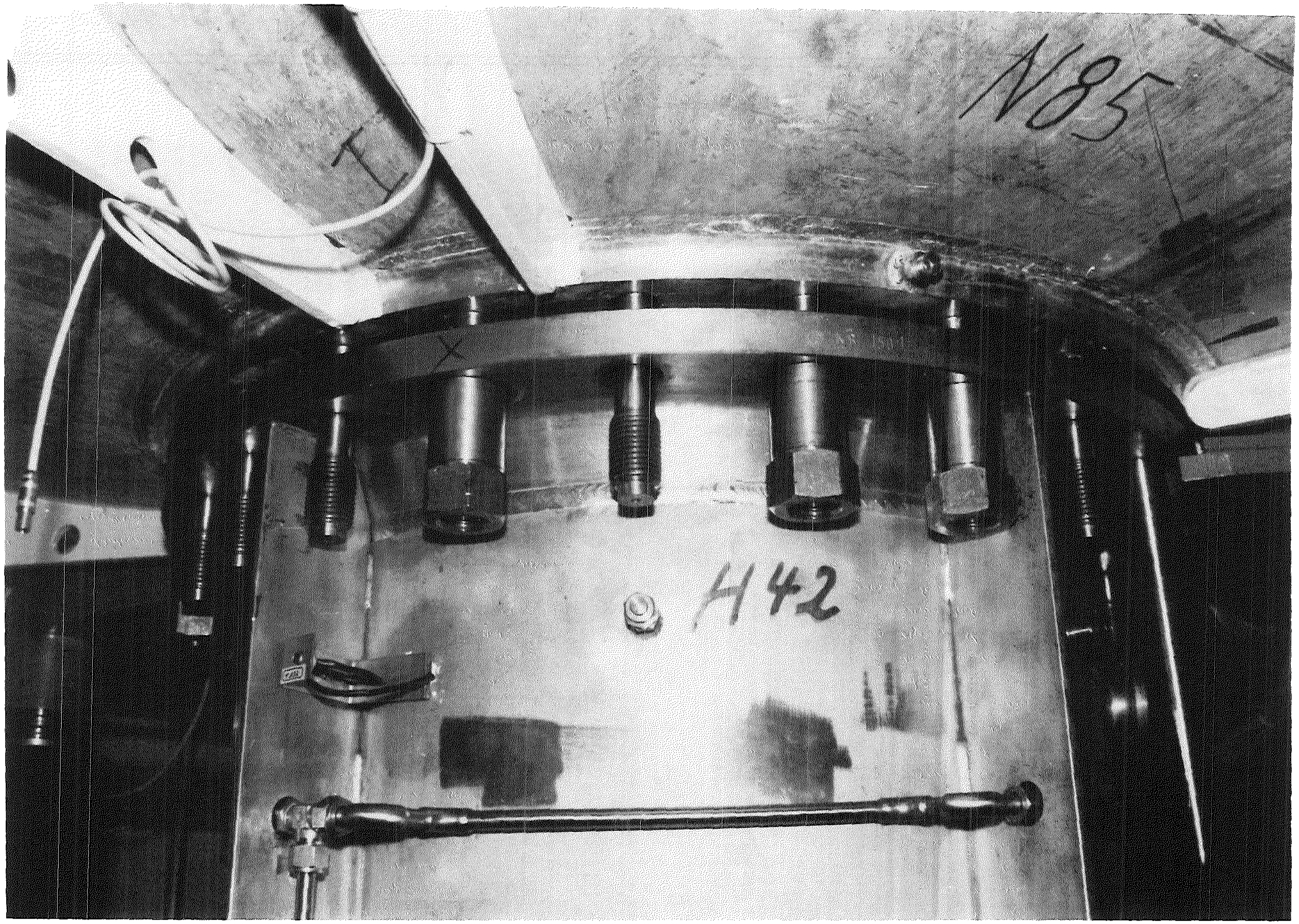

Fig. 18: UPPER END OF HIGH TEMPERATURE SHIELD

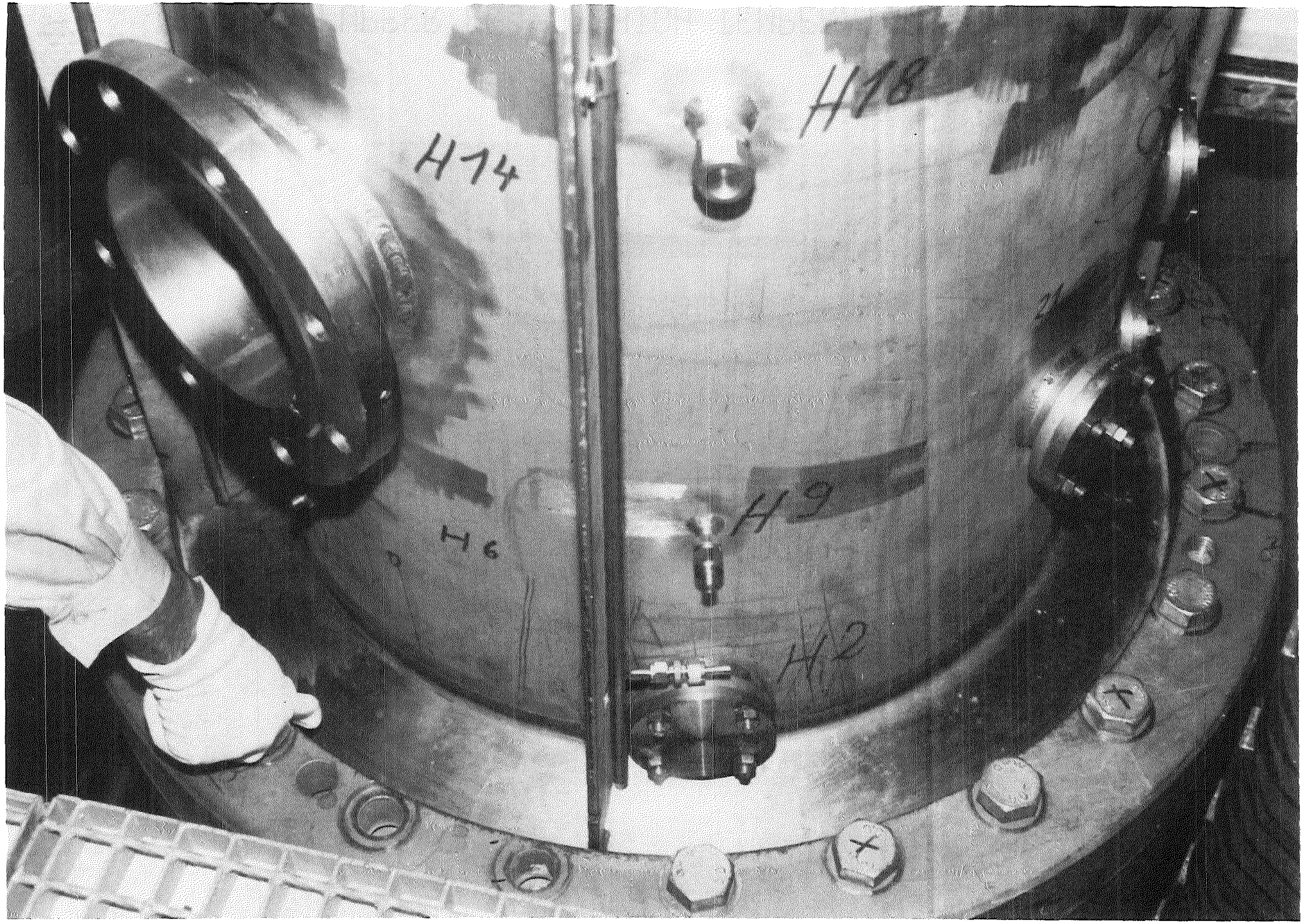

Fig. 19: LOWER END OF HIGH TEMPERATURE SHIELD

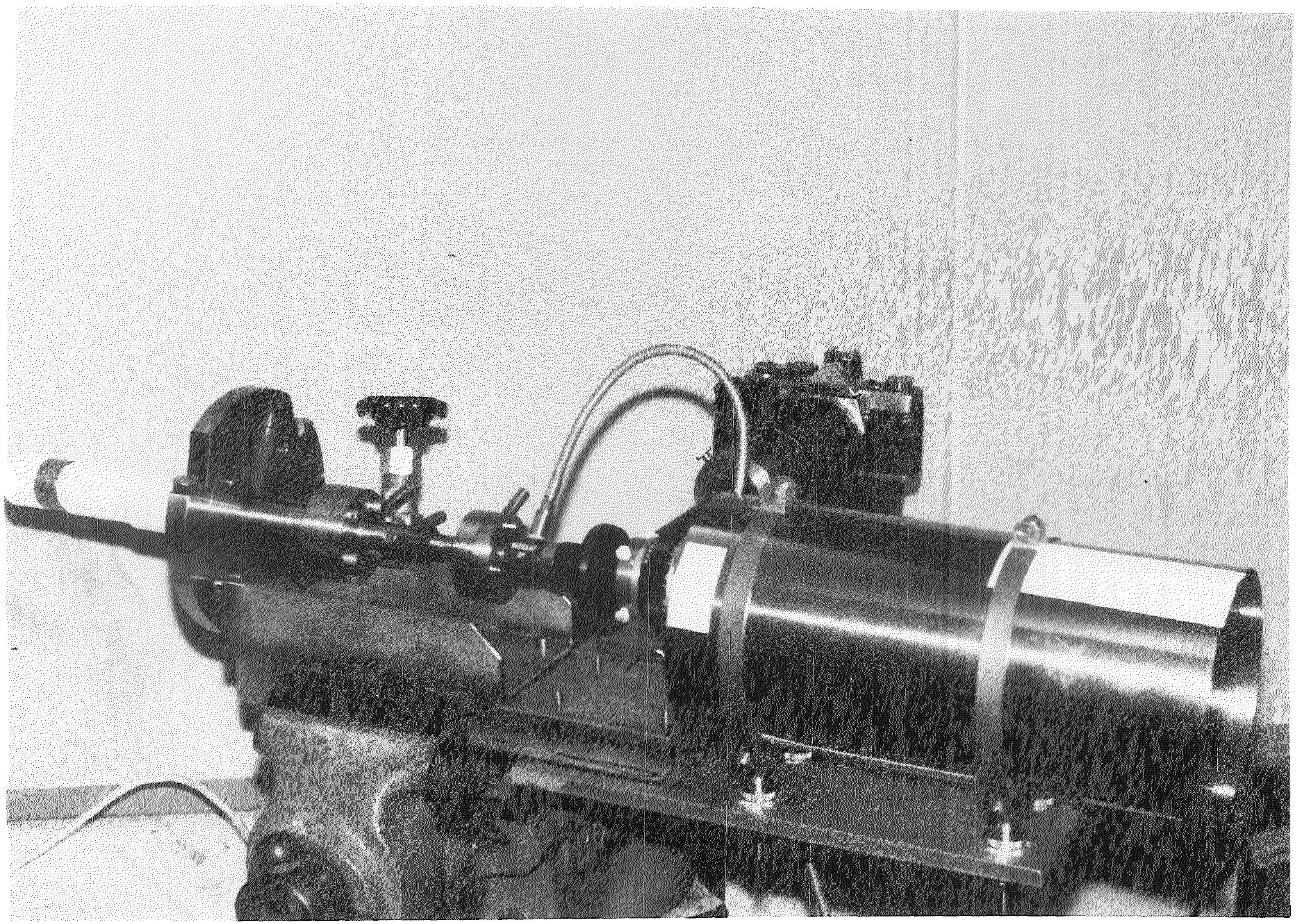

Fig. 2Ø: ENDOSCOP WITH 24 MM VIDEO CAMERA