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EUROPEAN ATOMIC ENERGY COMMUNITY – EURATOM

REACTOR CENTRUM NEDERLAND – RCN

NERO DEVELOPMENT PROGRAMME

Report covering the period January 1963 to June 1964

1965

Contract No. 007-61-6 PNIN

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parameters. The incorporation of burnable poison in the fuel is also discussed. Irradiation experiments and corrosion performance tests are other items of the core development programme.

In the system and component development programmes the main emphasis has been on the steam generator performance test installation, on the preparation of pressurizer performance tests and on water quality control. In the design area studies were made of power distribution, burn-up, heat transfer and hydraulics. Heat transfer and hydraulic calculations are being backed up by experiments.

A complete study of dynamics and control has been made which now allows a detailed safety analysis of the reactor system. Shielding work is being done both theoretically and experimentally.

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LIST OF FIGURES

NERO DEVELOPMENT PROGRAMME

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Introduction

The purpose of the NERO Development Programme is to establish a design with specifications for an advanced pressurized water reactor for marine propulsion in sufficient detail to allow for an evaluation of the feasibility to build a prototype reactor plant. The design target is a pressurized water reactor with internal recirculation within the reactor vessel, a thermal rating of 61 MW with a core lifetime of 1200 days at full power, providing slightly superheated steam at a pressure of 40 kgf/cm^2. The fuel consists of uranium-oxide pellets clad in zircaloy. Reactivity control is to be provided by 12 Y-shaped control rods fitting between nearly hexagonal fuel elements, and by burnable poison incorporated in the uranium-dioxide pellets.

Work on the development programme started in 1961, the contract between R.C.N. and Euratom was signed on December 1st, 1961. The work to be performed is divided into 16 different chapters, each of which covers a separate aspect of the development programme. These chapters are listed in Table I.

During the first years of the contract period much effort was put into the building and commissioning of the facilities required for the experimental part of the programme, while on the other hand extensive calculations on the physical and thermal aspects of the core and on the dynamics of the system have been performed. At the end of March 1963 one of the most important experimental facilities, the facility for performing critical experiments, started operating with the attainment of criticality in the first core. Towards the end of 1963 work was started on the preparation of an interim design NEREUS, which will be described in this report, together with some of the results obtained from experimental work in several chapters of the development programme.

Table I

Chapter	I	— Critical experiments
Chapter	II	— Incorporation of burnable poison in UO_2
Chapter	III	— Irradiation experiments
Chapter	IV	— Corrosion and erosion research
Chapter	V	— General design and calculations
Chapter	VI	— Stress analysis of pressure vessel and cores
Chapter	VII	— Steam generator performance test installation
Chapter	VIII	— Control rod drives
Chapter	IX	— Development of special welding techniques
Chapter	X	— Control rod materials and construction
Chapter	XI	— Water treatment
Chapter	XII	— The pressurizer
Chapter	XIII	— Pumps, accessories, etc.
Chapter	XIV	— Heat extraction from the reactor core
Chapter	XV	— Determination of hydraulic parameters
Chapter	XVI	— Shielding

1. GENERAL FEATURES OF THE DESIGN

It is the purpose of the NERO target design to provide an economically attractive reactor installation for ship propulsion. The P.W.R. type reactor was chosen because of the wide experience which has been gained with the operation on ships and the inherent safety features of this type of reactor.

In order to obtain an economically attractive design, however, both the specific construction costs per shp. of the reactor installation and the operating costs have to be reduced considerably.

Taking this into consideration for the target design, main emphasis has been put on the simplicity of the reactor installation, on compactness and ease of operation and on obtaining a long burn-up cycle for the core.

1.1. Interim design NEREUS

The features of the design which contribute towards compactness of the installation are illustrated in Figures 1 and 2 which show the arrangement inside the containment vessel and the engineering diagram of the primary system. The interim design has been given the code name NEREUS in order to distinguish between this interim design and the NERO target design. The containment vessel as shown in Figure 1 has an internal diameter of 9 m. Inside the containment vessel the following components are situated:

— Reactor vessel;

— Unit containing heat exchanger, superheater and primary pump (1, 7);

— Purification system with three ion exchangers (2);

— Pressurizer (3);

— Blow-off tank (4);

— Primary shield tank (5);

— Primary system piping and check valves (6);

— Decay heat pumps (8);

— Ventilation system.

The engineering diagram of the primary system in Figure 2 shows the principle of internal recirculation inside the reactor vessel. Only part of the mass flow through the core (app. 40%) is circulated in the part of the primary circuit outside the reactor vessel itself. From the vessel, the hot coolant flows through the steam superheater, and the steam generator, and is then pumped back to the core through a set of water ejectors which provide the driving force for that part of the coolant flow through the core that is recirculated internally in the vessel. At the outlet of the water jets, that is in the inlet region to the core, the cold feed water and the hot recirculating water are well mixed, and proper temperature as well as the correct velocity distribution are obtained in the bottom plenum. The advantage of the internal recirculation system chosen in this design is that only a portion of the required mass flow in the core needs to be circulated in the outer part of the circuit, thus leading to smaller pipe diameters and therefore allowing a more compact construction. Furthermore, by optimization of the recirculation ratio, the total pumping power required for the mass flow rate through the core is decreased as compared to the conventional set-ups in which the total mass flow is circulated through the heat exchangers and the pumps. Moreover, the internal recirculation system offers a very important safety feature, in that it provides a path for natural convection inside the reactor vessel in case of total loss of pumping power.

As one of the mechanical main problems encountered when working for a compact design of a P.W.R. installation, lies in the high thermal stresses which arise in the short length, thick walled piping of the primary system, the combination of heat exchanger, superheater and primary pump into one unit which eliminates some of the piping, and the reduced flow through the external circuit which allows a decrease in diameter of the piping, make it possible to obtain a more compact installation than with more conventional arrangements.

The situation of the water ejectors inside the reactor vessel is shown in Figure 3, a longtitudinal section of the reactor vessel.

The reactor core consists of 36 nearly-hexagonal fuel elements and 12 subsize elements arranged in hexagonal pattern as shown in the cross section of the reactor vessel (Fig. 4). The full-size fuel element is shown in more detail in Figure 5.

The fuel consists of uranium-dioxide pellets clad in zircaloy. The full-size fuel elements contain 120 fuel rods each, the subsize elements contain either 55 or 48 rods. Reactivity control is provided for by 12 Y-shaped control rods fitting between the fuel elements, and by burnable poison incorporated in the uranium-dioxide fuel pellets in the case of NERO, located in between the oxide pellets in the case of NEREUS.

In Table II the various design parameters of the NERO and NEREUS design are tabulated.

Table II

NERO & NEREUS DESIGN PARAMETERS

	NERO	NEREUS
GENERAL		
Reactor type	Pressurized water	Pressurized water
Ship shaft horse power, max. continuous	20,000	22,000
Reactor power, MW thermal	61	67
Reactor operating pressure, kgf/cm^2	140	151
Steam production, tons/hr	99.7	111.2
Steam pressure, kgf/cm^2	40	40
Steam temperature, °C	295	280
Operational core life, days at full power	1.200	1,200
Refuelling period, years	4	4
PRIMARY COOLING SYSTEM		
Number of cooling circuits	2	2
Design pressure, kgf/cm^2	168	168
Design temperature, °C	350	350
Nominal reactor outlet temperature, °C	314	299
Nominal inlet temperature reactor core, °C	302	288

	NERO	NEREUS
Mass flow through reactor core, kg/sec	945	1,155
Mass flow through fuel elements, kg/sec	900	1,100
External mass flow total, kg/sec	360	440
Recirculation ratio reactor core	2.6	2.6
Pressure drop over reactor core, kgf/cm^2		0.5
Pressure drop over reactor vessel, kgf/cm^2		3.3
Pressure drop over cooling circuit, kgf/cm^2		1.7
Pumping power per pump, kW		260
REACTOR CORE DIMENSIONS		
Core height, active length, mm	1,139	1,327
Average core diameter, mm	1,128	1,128
Number of fuel elements, 120 rods	30	30
Number of fuel elements, 55 rods	6	6
Number of fuel elements, 48 rods	6	6
Diameter of fuel pellets, mm	10.03 ± 0.01	10.03 ± 0.01
Cladding diameter, outside, mm	11.90 ± 0.12	11.90 ± 0.12
Cladding thickness, mm	0.85 ± 0.04	0.85 ± 0.04
Triangular lattice pitch of fuel rods, mm	15.0 ± 0.25	15.0 ± 0.25
Cladding material	Zirconium alloy	Zirconium alloy
Volume ratio H_2O/UO_2, cold, control rods out	1.4344	1.4344
Total weight UO_2, kg	3,953.9	4,600
Number of control rods	12	12
CORE PHYSICS		
Core life at full power, days	1,200	1,200
Average burn-up, MWD/ton UO_2	18,500	16,600
Initial enrichment central zone, %	4.4	3.8
Initial enrichment outer zone, %	4.8	4.2
CORE HEAT TRANSFER AND HYDRAULICS		
$\int k d\Theta$ max., W/cm	30.8	40
Max. heat flux, W/cm^2	106.5	113.5
Total flow cross section, m^2	0.4	0.4
REACTOR VESSEL		
Inside diameter, m	2.0	2.0
Inside height, m	5.5	5.5
Wall thickness cylindrical part, mm	115	115
Material	ASTM A302 Grade B	ASTM A302 Grade B
Cladding	Stainless-304	Stainless-304
STEAM GENERATORS		
Type	Vertical U-tube	Vertical U-tube
Number	2	2
Primary coolant inlet temperature, °C	312	297.5
Primary coolant outlet temperature, °C	283	269.5
Feed water inlet temperature, °C	175	175
Steam temperature, 100% power, °C	250.6	250.6
Total tube surface, m^2	148	200
Tube material	Inconel-600	Inconel-600
SUPER HEATERS		
Type	Horizontal U-tube	Horizontal U-tube
Number	2	2
Heat load per superheater, MW	2.14	1.62
Inlet temperature primary coolant, °C	314	299
Outlet temperature primary coolant, °C	312	297.5
Steam temperature inlet, °C	250.6	250.6
Steam temperature outlet, °C	295	280
Total tube surface, m^2	55.7	43.76
CONTAINMENT		
Configuration	Sphere	Sphere
Inside diameter, m	9	9

1.2. NERO target design

The NERO target design as presently envisaged differs from the NEREUS interim design in some aspects concerning primary system temperatures, mass flow through the core, hot spot factors in the core and the size of the heat exchanger and pressurizer. In general, however, the NERO design features are well represented by the NEREUS figures 1 through 5 which have been discussed in the previous section. Results of experiments which will be performed in the remaining part of the development programme will be used to determine if the operating conditions for the target design as they have at present been postulated can be met or even improved upon.

The values for primary system temperatures and mass flow through the core in the interim design, shown in Table II, were calculated after taking into account a conservative value for the power peaking factor in the core and allowing for subcooled boiling, but no bubble detachment, as calculated by the Bowring criterion. The NEREUS operating conditions are, therefore, somewhat less advanced than those of the target design. In Table II the NERO design parameters are compared with those of NEREUS.

According to present results it is anticipated that continued reactor physics experiments with the critical facility will show that the value for the power peaking factor may be reduced below the value chosen for the interim design. Also a better understanding of the boiling phenomena during heat transfer from the fuel rods to the coolant as a result of the experimental work in the present programme, by means of the high pressure loop, will make it possible to allow for an increased measure of boiling in the hottest channel.

The advantage of boiling in coolant channels lies mainly in the reduction of the hot spot factor, the maximum wall temperature being determined to a large extent by the local saturation temperature, and in the increase of the average coolant outlet temperature. In fact, the larger the relation between the number of channels with a net steam content in the outlet and the total number of channels, the closer is the approach of the average coolant outlet temperature to the saturation temperature set by the operating pressure of the system. If boiling is permitted the mass flow rate through the core can be reduced.

In the NERO design the outlet temperature is 314°C as compared to 299°C in NEREUS, whereas the NERO mass flow is 900 kg/sec as compared to 1100 kg/sec in NEREUS.

Experimental work on pressurizer performance will lead to a better understanding of the factors involved which may result in a reduced size of the required pressurizer.

Finally, it is hoped that investigations in the fields of incorporation of burnable poison, of irradiation of special poisoned fuel, and of corrosion and erosion will allow the present NERO target design parameters to be improved upon.

2. CORE DEVELOPMENT

The first four chapters in the development programme as given in Table I are related directly to core development work. Critical and subcritical experiments are being performed with the critical facility KRITO and the subcritical facility PUK, both erected in the Fermi building of R.C.N. at Petten.

A description of these facilities and of the first series of experiments with KRITO is given in the reports EUR 498.e ("The Critical Facility KRITO", by J.G. Ackers, M. Bustraan, J. Coehoorn, R.J. Heyboer, F. Luidinga, M. Muysken, W.W. Nijs, A. Tas, W.L. Zijp) and EUR 499.e ("The Subcritical Facility PUK", by M. Bustraan and A. Tas).

Work on the incorporation of burnable poison in uranium-dioxide pellets is being performed at the Chemical and Metallurgical Laboratory of R.C.N. at Petten.

A high temperature, high pressure irradiation loop is being built next to the H.F.R. at Petten and will soon be operated for performance testing of fuel rods according to the NERO design.

Experimental work on corrosion of fuel element cladding materials is being performed at the chemical and metallurgical laboratory of R.C.N. at Petten.

2.1. Critical and subcritical experiments

The experimental programme with the critical facility KRITO comprises:

a) Measurements of criticality, reactivity effects and neutron flux or power distributions in order to determine the worth of fuel rods, control rods, void and temperature coefficients, material bucklings, etc.

b) Repeat of the measurements summed up under a) with nuclear poisons in several fuel rods which are placed in different patterns in the core.

c) Kinetic measurements, including noise measurements, to determine the reactor transfer function, mean neutron generation time and effective delayed neutron fraction.

The KRITO core has the same diameter as the NERO core. However, for reasons of economy, use is made of aluminium as canning and fuel box material and the control rods are made of boral. Furthermore, the poison is applied in the form of thin B_4C foils (thickness 0.6 mm) between the UO_2-pellets. This leaves the possibility of changing the concentration in the course of the experimental programme.

The first series of experiments with the first core, KRITO core type I, are reported in EUR 498.e. This core consisted of a single zone core with over 1000 fuel rods, filled to a height of 40 cm with 3.1% enriched uranium-dioxide pellets. Detailed flux mapping in the neighbourhood of water gaps was performed, both for configurations with clean rods and with poisoned rods near the water gaps. Flux mapping in vertical direction was also performed. The results have been compared with a flux pattern calculated with the PDQ code. Additional calculations now being performed, are referred to in para. 4.2 of this report. In Figure 22 a calculated radial power distribution is compared with one measured in the KRITO facility.

In the fourth quarter of 1963 the KRITO core type II (with 3.8% enriched fuel) went critical with about 780 fuel rods. The core was enlarged until an excess reactivity of 3.53% was obtained with a total of 1005 rods. A series of reactivity measurements were performed to determine the worth of fuel rods of 3.1% and 3.8% enrichment, with different amounts of poison in the rods. Figure 6 shows this core as installed in the core vessel. The results of these measurements are being evaluated in order to determine the amount of poison which will be necessary to build up the complete core in two zones with a total of 4218 fuel rods.

In the meantime, a core with 3.8% enriched fuel in the configuration chosen for the subcritical experiments as described in EUR 499.e is being erected in the critical facility, in order to perform a series of criticality measurements required by the Safety Committee of the R.C.N., prior to the subcritical experiments next to the Low Flux Reactor.

2.2. Incorporation of burnable poison in UO_2

The purpose of the incorporation of burnable poison in the fuel for the NERO target design is to provide for reactivity control during the lifetime of the core, in such a way that the excess reactivity required for the burn-up is compensated almost entirely by the burnable poison. Thus the reactivity control to be provided by the control rods can be limited to the effects of temperature and power, and xenon and samarium poisoning.

The burnable poison will be incorporated in the fuel in the form of discrete globular particles to provide for a certain self-shielding effect during burn-up. Extensive calculations have been performed with an analogue computer to determine the required self-shielding factor for different types of poisons and the resulting particle size. The results of these calculations are reported separately as a Euratom publication.

The experimental work on the incorporation of burnable poison started with B_4C as poison material. The first results with this material were negative: it appeared that part of the boron disappeared from the pellet during the high temperature sintering process. As similar work was being performed under the Euratom/United States Agreement for Cooperation, by Combustion Engineering, and the work of Combustion Engineering led to results which did not quite agree with the results obtained by R.C.N., a fundamental investigation into the reactions between B_4C and UO_2 is now underway.

Meanwhile, work has started on the application of two other materials to be used in a lumped form sintered in uranium-dioxide pellets. The first of these is the compound UB_4, which is more stable at high temperatures than B_4C. The second method under consideration is the coating of dysprosium-oxide particles with tungsten before sintering.

As dysprosium-oxide appears not to be stable at high temperature, a coating is required. The requirements for integrity and impermeability of the coating during the lifetime of the particles are, therefore, quite different from those set e.g. for the coating of DRAGON-type fuel particles.

In addition to the metallurgical work, preliminary irradiation experiments to determine the burn-up characteristics of the poison particles experimentally have been started. Pellets of pressed aluminium powder, to provide a neutral matrix for the poison particles, have been irradiated for a relatively short time (one operational period in the High Flux Reactor at Petten) in order to establish the experimental procedures and measuring techniques. After an evaluation of the results, the experimental irradiation will be continued. At present only B_4C, UB_4 and Dy_2O_3 are being considered for irradiation experiments in a neutral matrix.

2.3. Irradiation experiments

The provision of a facility for performance testing of fuel rods under reactor operating conditions was undertaken in the third chapter of the development programme. This chapter also includes the operation of this facility and the post-irradiation examination of the irradiated samples in hot cells.

At present erection of the systems and instrumentation in the basement of the H.F.R. building at Petten is practically completed. Figure 7 shows a view of the facility. Figure 8 shows the flow sheet.

The cooling system is capable of operating at a temperature of about 325°C and a pressure of 140 kgf/cm^2 and allows for a heat production of about 200 kW in the sample rods.

The in-pile section of this loop consists of two adjacent pressure tubes with an inside diameter of 44 mm and a height of 60 cm, in which 6 fuel rods according to the NERO design but with reduced length can be irradiated simultaneously. The in-pile section is at present being subjected to out-of-pile testing at operating pressure and temperature before being installed in the reactor. It will be erected in position in the H.F.R. reactor in the near future.

Operation will start in the second quarter of 1965, first with dummy fuel rods made of stainless steel. Subsequently sub-assemblies with natural uranium will be used, followed by the irradiation of 3.8% enriched uranium-oxide rods clad with zircaloy. The first two sets of uranium-dioxide-bearing elements have been fabricated by Canadian General Electric. Subsequently, elements produced by R.C.N. and Dutch industry will be tested. These elements first pass through a testing programme in the forced circulation, high temperature corrosion loop referred to in section 2.4.

2.4. Corrosion performance tests

The purpose of this part of the programme is to provide an adequate check on the quality and corrosion behaviour of the materials to be used in the reactor core, especially the canning of the fuel rods, the grids in the fuel elements and the control rods.

For the cladding material of the fuel rods a zirconium alloy will be used. Available indications in literature state that the most common zirconium alloy, zircaloy-2, shows increased susceptibility to the uptake of hydrogen and therefore a reduced lifetime in the reactor at canning temperatures above about 320°C. However, there are no records of prolonged tests under these conditions. Several manufacturers are working on improvements in manufacturing methods of zircaloy-2 tubes so as to reduce the problems arising from hydrogen uptake. Also, much work is being done on developing new zirconium alloys by several manufacturers within the Community. The facilities which are available for the Corrosion Group of R.C.N. at Petten will make it possible to subject the newly developed alloys to performance tests at a temperature of 340°C maximum, with closely controlled water quality.

During the first part of 1963 autoclave testing on zircaloy-2 tube specimens from three European manufacturers were performed. At the same time tube samples from these manufacturers were subjected to tests for mechanical defects and wall thickness by the Röntgen Technische Dienst at Rotterdam and for surface roughness, inner and outer diameter variations, ovality and straightness by R.C.N. at Petten. The outcome was that 50% of the tubes delivered by two of the manufacturers and 100% of the tubes delivered by the third manufacturer would have to be rejected on the results of the non-destructive testing.

The cathodic uptake of hydrogen on the outer surface of tube specimens from these manufacturers is still being investigated.

In the meantime sample lots of zircaloy-2 tubes have been ordered from a Swedish and an American manufacturer, both of whom have developed improved manufacturing methods for the tubes. These will be subjected to the same tests that have been performed on the previous samples.

The forced circulation corrosion loop, which has been built by R.C.N. for use in the development programme, is now nearly completed. Figure 9 shows this loop as erected in the Chemical Laboratory building of R.C.N. at Petten. Each of the two test sections of the loop allows for the testing of sub-assemblies of fuel elements with an outside diameter of 80 mm and a length of 1.50 m in water at a temperature of 340°C and a pressure of 175 kgf/cm^2.

In connection with the corrosion investigations, the work on water quality control as described in section 3.3 should be mentioned.

3. SYSTEM AND COMPONENT DEVELOPMENT

Several parts of the development programme are concerned with subjects related to the systems and components of the reactor installation. For example, a steam generator performance test installation is being built. This installation is to be erected at the Technological University at Eindhoven. Also, the development of a control rod drive system suitable for a marine pressurized water reactor is being taken care of.

The application of special welding techniques is the subject of a special sub-programme, which is, however, of very limited size.

Also, the activities concerning the development of the control rod design will not be very extensive as now envisaged.

A separate sub-programme is concerned with water quality control: experimental work in this field is to be carried out in conjunction with the operation of the high pressure, high flux irradiation loop.

Experimental work with a pressurizer test installation is being done in partnership with industry, the Technological University at Delft and with the Royal Navy.

A proposal to develop a canned motor pump for the combined steam generator/superheater/pump unit of the NERO design is at present under discussion.

More details regarding the investigations just mentioned are given in the following paragraphs.

3.1. Steam generator performance test installation

The steam generator performance test installation has been designed and is at present being built to provide a means for measuring the heat transfer capability of compact steam generators and superheaters as well as the performance of these components under dynamic loading.

The main items of the installation as shown on the flow sheet in Figure 10 are the following:

a) A boiler with a capacity of about 6 MW of heat input, which raises the temperature of a circulating quantity of water from 290°C to 314°C as a substitute for the reactor in the primary circuit with temperatures according to the target design.

b) A scale model of steam generator and superheater, designed on a scale of approximately 1/6th of the output of the corresponding equipment in the target design.

c) A circulating pump with a capacity of 218 m^3 per hour for the simulated primary system.

d) A secondary system in which the steam formed in the steam generator on the secondary side and heated up further in the superheater is led to a high pressure condensor, operating at system pressure. The cooling of the condensor is accomplished by spraying water on the outside surface of the condensor, which consists of a set of pressure tubes in open air (see Fig. 23). This enables one to simulate fast changes in load on the secondary side, by regulating the water spray. Besides, this type of operation has the advantage that very little power is required for the feed pump.

Figure 11 shows an isometric view of the installation as it will be erected at the Technological University of Eindhoven. Erection will be completed in the first or second quarter of 1965.

After completion of the testing of the present steam generator unit, the installations will be available for testing of other units as they may be proposed by interested parties within the Community.

3.2. Control rod drive system

The requirements for the control rod drive system for application on marine reactors are more stringent than for land-based power stations, in that gravity cannot be counted on under all conditions in which scram action is required and that dynamic loading of the components due to ship movements must be considered. The design requirements for the control rod drives are to some extent set by the Provisional Rules for the construction of marine reactors by several classification companies.

A set of design specifications for a rack and pinion type drive system was prepared in the first months of 1964. All moving parts are situated inside the pressure envelope of the primary system, which simplifies leakage problems. The driving electro-motor is of the canned rotor type. The reduction gear between motor shaft and pinion shaft, as well as the magnetic coupling, operate in the water of the primary system. The coupling is controlled from outside the pressure envelope.

The set of design specifications has been issued to several industries in the Netherlands to elicit a proposal for further development.

3.3. Water quality control

The purpose of the experimental programme on water quality control is to study the mechanisms of crud formation, crud transport, crud deposition and activity build-up in pressurized water systems, and to develop the instrumentation required to monitor water quality during operation of the reactor.

The experimental work will be performed by the Chemistry Department of R.C.N. at Petten. Water samples to be taken from the forced circulation corrosion loop described in para. 2.4 and the high temperature irradiation loop described in para. 2.3 will be checked for conductivity, pH value, oxygen, hydrogen, nitrogen and ammonia content, soluble inert gases, crud content with analysis of the crud, and, furthermore, the content of chlorides, silicons, carbonates, sodium, iron, chronium, nickel and zirconium. The water samples from the high temperature irradiation loop will be additionally checked for gross γ-activity and for the several radionuclides separately.

3.4. Pressurizer performance

An experimental pressurizer, which was built by an industrial group as an experiment in fabrication of this type of pressure vessel, has been erected at the Technological University of Delft, together with a system which permits the testing of pressurizer performance under simulated operating conditions.

This project was undertaken jointly by the industrial group Neratoom, the Technological University of Delft and the Royal Dutch Navy. The installation has been tested and is in operation. By means of varying gas pressure on the water surface in a second vessel adjacent to the pressurizer, the surges that occur in a pressurizer of a P.W.R. installation are copied. Also included in the installation are a storage vessel for the high temperature water and a heat exchanger to cool down the water flow between the high temperature section and the cylinder in which the gas pressure above a free water surface is being varied.

In its present form, however, the installation is not equipped to simulate a series of consecutive surges such as would be requested for a marine reactor installation during manoeuvring of the ship.

R.C.N. has now joined the project and provides funds to modify the installation such that pressurizer performance under consecutive in and outsurges can be studied.

4. DESIGN WORK AND CALCULATIONS

During the period covered by this report calculations have been performed on several physical, thermal, hydraulic and dynamic aspects of the core and system. Design work during the latter part of 1963 and the first half of 1964 was mainly concerned with the NEREUS interim design. Furthermore the design of test sections for facilities for hydraulic measurements were worked out. Preliminary calculations were performed on the shielding design. An experimental programme on shielding in collaboration with G.K.S.S. at their site at Geesthacht, Western Germany, was started in the second quarter of 1964. A review of existing devices for steam separation in heat exchangers was prepared by industry. Industry also performed preliminary stress calculations for the heat exchanger unit of the NERO design and evaluated this design from the view-point of fabrication. The problems connected with the installation of the reactor in a 65,000-ton tanker were studied together with a ship design consultant. Industry is assisting in preparing the lay-out of the conventional machinery for the engine room.

4.1. Power density factor, hot channel factor and hot spot factor

For the physics and heat transfer calculations on the NEREUS interim design and the NERO target design, the power density factor, hot channel factor and hot spot factor have been defined as follows.

4.1.1. The power density factor F_p is the ratio between the maximum power density in the fuel, W/cm^3, to the average power density, also expressed as W/cm^3.

F_p is the product of three factors:

$$F_p = F_{xy} \times F_z \times F_{op}.$$

The factor F_{xy} is the radial component of F_p as determined from two dimensional physics calculations and measurements in the critical facility. It covers also the irregularities in the core pattern which are a consequence of the design of the fuel elements with boxes, water gaps and control rods.

The factor F_z is the axial component of F_p as determined from physics calculations, also taking into account the flux deformation by control rods.

The factor F_{op} covers deviations and uncertainties arising from manufacturing tolerances, due to variations in enrichment and density of the uranium-dioxide fuel and variations in the pitch, diameter and cladding thickness of the fuel rods.

A second main factor, of importance for heat transfer calculations, is defined as:

$$F_{q''} = F_{xy} \times F_z \times F_{oq''}$$

The factor $F_{q''}$ may be slightly larger than F_p because the uncertainties and variations in the dimensions of the fuel element expressed in $F_{oq''}$ that have an influence on the heat transfer from fuel to coolant may require a larger margin than the values expressed in F_{op}.

4.1.2. The hot channel factor for a certain channel is defined as the ratio between the increase in enthalphy of the coolant when passing through this channel between the fuel rods and the average increase in enthalphy over the core. The quotation for the enthalphy factor is F_H. F_H is the product of two factors: F_{xy} as defined above and F_{oH}. This second factor covers variations in the amount of coolant flowing through the channel dependant on friction factor, hydraulic diameter, length of channel, average coolant density in the channel, pressure drop over the channel coolant flow area and mixing between channels. Thus $F_H = F_{xy} \times F_{oH}$.

4.1.3. The hot spot factor F_T covers the factors influencing the temperature of the cladding in the core. The maximum local temperature of the cladding is obtained by adding to the inlet temperature of the coolant firstly the temperature rise to the point at which maximum cladding temperature is expected in axial direction and secondly the maximum temperature rise from coolant to cladding.

The first temperature rise is expressed as:

$$\Delta\theta_1 = F_H \times \frac{\Delta H(z)}{C_p}$$

The second temperature rise is expressed as:

$$\Delta\theta_2 = F_{xy} \times F_{oq''} \times F_{oa} \times \frac{q''(z)}{\alpha}$$

In this last formula the factors F_{xy} and $F_{oq''}$ have already been defined. The uncertainty factor F_{oa} depends on coolant density, coolant velocity, coolant viscosity, hydraulic diameter, thermal conductivity etc.

The expression $\frac{q''(z)}{\alpha}$ indicates the average heat flux from the cladding to the coolant at the axial distance z above midplane of core divided by the average heat transfer coefficient, α being the heat transfer coefficient between cladding and coolant. The parameter z is to be set at the value where maximum cladding temperature in axial direction is to be expected. The hot spot factor F_t is defined as:

$$F_t = F_{xy} \times F_{oq''} \times F_{oa}$$

The maximum cladding temperature can be calculated by the addition of the two temperature increases θ_1 and θ_2 to the temperature of the coolant at the inlet to the core.

For the calculations for the reactor core of the NEREUS interim design the following values have been derived for the several factors which have been defined above:

$F_{xy} = 2.1$	$F_z = 1.9$
$F_p = 4.39$	$F_{q''} = 4.43$
$F_H = 2.7$	$F_t = 3.6$

These are conservative values, which show little credit for the application of burnable poison in selectively placed fuel rods to obtain a reduction of local flux peaking and for the fact that the excess reactivity to be controlled by the control rods during operation is minimized by the use of burnable poison with the right self-shielding properties. Taking this into consideration it appears feasible to reach lower values for the power, enthalphy, and hot-spot factors for the NERO target design. These had been provisionally set at the following values:

$F_p = 3.2$	$F_{q''} = 3.24$
$F_H = 2.65$	$F_t = 3.26$

4.2. Physics calculations

After completing the preliminary calculations concerning the burn-up behaviour of burnable poison, taking into account variable self-shielding factors, by analogue computer methods, which were mentioned in paragraph 2.2, a start was made with transport calculations of the detailed neutron flux within and surrounding the poison particle which will afford a more exact picture of the depletion of shaped burnable poison particles. Also a system of modifications was prepared to be used with the Fuelcyc digital burn-up code available at Ispra, so that the effect of a single element burnable poison with $1/v$ type cross section can be treated.

A series of calculations using the K-7 Thermos code, which solves the integral transport equation for a heterogeneous unit cell, were performed to investigate the influence of geometric irregularities in the fuel element on the power factor F_p. The series included a calculation for the following cases:

a) Nominal values for all parameters;

b) Minimal thickness of cladding, maximum pitch;

c) Maximum density of UO_2;

d) Maximum enrichment of UO_2;

e) Maximum pellet diameter.

The calculations were performed with ten neutron energy groups.

A total effect of all these variations resulted in a value of 1.173 for the partial factor F_{op} in the power factor F_p and a value of 1.179 for the partial factor $F_{oq''}$ in the heat flux factor $F_{q''}$. The largest contribution was provided by the configuration b). Thus the most effective way to reduce the factors F_{op} and $F_{oq''}$ is to tighten the tolerance on the pitch of the fuel rods in the fuel elements. The effect of tightening the tolerances was to reduce the factor F_{op} from 1.173 to 1.100.

A separate calculation was made concerning the temperature distribution around a small uncooled section of the cladding, such as might occur at the point of contact of a grid. The results of this calculation will be reported separately in the near future.

Finally an economic evaluation was prepared concerning the fuel cycle costs of this reactor while varying the following factors in the core design:

— The average enrichment;

— The core volume;

— Single zone and two zone core;

— The enrichment ratio of a two zone core;

— The maximum to average power density ratio or F_p;

— The radial to average power density ratio or F_{xy}.

In general the qualitative results show that the fuel cycle costs of the reactor are proportional to the total power factor F_p.

Furthermore, small cores with high enrichments are shown to be more attractive than larger cores with lower enrichments for the same lifetime and the same power output.

4.3. Heat transfer calculations and experiments

To perform the heat transfer calculations for the reactor core of the NEREUS interim design the criterion was established that no bubble detachment from the channel walls should occur at the hottest points. This criterion allows for a certain amount of nucleate boiling, but restricts this occurrence in such a way that very little disturbance of the flow in the channel is caused by the boiling.

R.W. Bowring has established a criterion for bubble detachment, which can be applied to the case of a heat flux in the form of a cosine along the axis of the channel, such as is assumed in the reactor core. Applying this criterion it was calculated for the worst case that, taking into account the power factor, heat transfer factor and hot channel factor defined in para. 4.1, an inlet subcooling of 52°C is required to prevent bubble detachment. This results in an inlet temperature to the core of 340°C − 52°C = 288°C, 340°C being the boiling temperature of the coolant at operating pressure. The average temperature rise of the coolant through the core with a mass flow of 1100 kg/sec and a thermal power of 67 MW works out at 11°C, thus the average outlet temperature of the core is 288°C + 11°C = 299°C. It is of obvious advantage to aim at a reduction in the large difference between the average outlet temperature and the boiling temperature of the coolant as calculated for the NEREUS interim design.

A method is being developed for predicting the mass flow distribution over a large number of parallel channels with individually different power, taking into account that boiling in some of the channels will lead to a redistribution as compared to single phase flow in all channels. This is due to the fact that boiling results in an increase in the friction and acceleration losses if the mass flow rate is kept constant. Therefore, if the pressure drop is fixed, the mass flow rate in the boiling channels is reduced to match the available driving head, but at the same time the degree of evaporation will increase if at constant channel power the mass flow rate is decreased. This will, again, lead to an increased vapour fraction and an increased pressure drop over the coolant channel. At present, a code for the calculations of single channel behaviour has been prepared, whereas an amplification to a multiple channel problem is underway. Furthermore, the development of a digital computer code for the analysis of fluid mixing between adjacent sub-channels and of the so-called "cold wall effect" has been started.

The experimental work will be performed with a test circuit designed for a pressure of 175 kgf/cm^2 and a temperature of 300°C, equiped with a 1 MW fast control power supply. Fig. 12 shows this loop which is erected at the Technological University of Eindhoven. A test section containing 7 heated fuel rods with an artificial "hot spot" on one of the rods and an adjustable channel wall has been fabricated for this loop. Fig. 13 shows the assembly drawing of this test section. The adjustable wall provides a means of checking the calculated "cold wall effect" by systematic variation of the distance between the outer zone of rodlets and the wall of the element.

The first set of introductory experiments in this loop were performed with a test section containing 12 heated rods and 17 unheated rods. This test section failed in the course of the experiments due to insufficient control of the water quality in the loop. The experimental programme is mainly directed towards the investigation of cold wall effects, burn-out phenomena and flow irregularities caused by boiling in the channels.

Concerning the heat transfer in the heat exchangers, a general calculating method for a vertical steam generator with the aid of a digital computer type X-1, has been drawn up. This method has shown good agreement with the results of measurements reported in literature and has been checked on the results of the performance test with a small scale model heat exchanger on the test circuit described in this paragraph.

Also a study was carried out by the industrial group Neratoom to evaluate several types of steam drying equipment and the patent situation in this field. It appears that many different devices have been developed on a practical basis, but there is still a lack of understanding of the fundamental theory involved. A contact has been established with Allis Chalmers, who are carrying out experimental work on this subject under the Euratom/United States Agreement for Cooperation.

4.4. Hydraulic calculations and experiments

The pressure losses in the primary system of the reactor and the required total pump head for the NEREUS design have been calculated in detail. The flow resistance adopted for the fuel element in these calculations was based on experimental results. The hydraulic parameters for the ejectors inside the reactor vessel were taken from literature. An experimental facility to verify these parameters has been built and is now nearly completed. The measurements on flow resistance in the fuel element have been performed with the low temperature loop which is shown in Fig. 14, erected for this purpose at the Laboratory of Aero and Hydrodynamics of the Technological University at Delft. The test section for the loop contained 19 simulated fuel rods in the pattern of the NERO core. Measurements were performed with water under atmospheric pressure up to a temperature of 40°C. This allowed for a Reynolds number of 1.07×10^5 for design operating conditions in the reactor core.

A further study was made concerning the relations between pressure losses and flow rates in the primary system to determine the required operating characteristics for the ejectors while in operation with two external circuits as well as with a single external circuit. The stability of the system as calculated proves to be satisfactory, however, the calculations will have to be checked against experimental results from the ejector test loop.

A series of experiments to determine the flow pattern of the primary coolant inside the reactor vessel is being planned.

4.5. Dynamics and control

Dynamic studies of the reactor and its associated systems have been performed.

In addition to the requirement of a stable operating reactor at constant power levels limits have to be set to the response of the reactor during rapid load changes which occur as a result of manoeuvring the ship.

Flexible manoeuvring sets the following goals to the reactor operation:

1) Power increase: 1.5%/sec.

2) Power decrease: 10%/sec.

Further, the system should be capable of handling the following sucessive load variations on the conditions mentioned under 1) and 2).

3) From 100% to 10% to 100% and

4) From 100% to 10% to 50% to 10% to 50%.

In addition, the system should stand a load variation from 100% to 10% within 2 seconds which occurs when the steam turbine safety valve closes.

Both for the NERO and NEREUS conditions, an initial study of the dynamic behaviour showed that due to the negative temperature coefficient of moderator and fuel, the reactor is a stable system, and that from this point of view an automatic control system would not be necessary. However, as a result of the transient behaviour, the following conditions must also be met:

1) Because the whole system must be designed to withstand the maximum power, the nuclear power overshoot due to load or temperature variations should be limited (less than 10%). The overshoot, in combination with errors in measurement, determines the scram set point.

2) Because the dimensions of the pressurizer depend on the expected surges, a limit has to be set to the in and/or outsurges.

In order to determine how the system could be properly influenced, a study was made of the response of system parameters of the uncontrolled reactor under various transient conditions using an analogue computer.

From the results it could be concluded that the outlet temperature of the reactor would be the best variable to control.

Using the reactor outlet temperature the level variations in the pressurizer are smaller than using the average temperature in the primary circuit or the average moderator temperature. Further, the controller functions more smoothly in this case. The controller reduces the surges in the pressurizer for the NERO target design from 400 kg to 270 kg and for the NEREUS interim design from 420 kg to 300 kg. The reactor transients are shown in Fig. 20 and Fig. 21.

In addition to this work, a construction was developed for the automatic controller. A prototype will be built by the electronic workshop and tested at one of the Petten reactors.

In order to improve the system control an anticipating signal is added. The derivative of the steam generator outlet temperature has proved to be a suitable signal to anticipate load variations. As a follow-up of this work attention was given to the pressure variations in the pressurizer of the NERO and NEREUS reactors due to the changes in primary volume. It was found that a decrease in primary volume results in a pressure decrease and that with an increase in primary volume the pressure might be kept constant, provided the right quantity of surge water is sprayed in top of the pressurizer. To achieve this an adequate control system is to energize continuously all heating elements in the pressurizer and to control the pressure by means of a continuous spray.

In addition to these studies analyses are underway of operational accidents such as loss of flow in the primary system, for instance due to mechanical failure of the pumps or electrical power failure and control rod withdrawal. Calculations on this last point show that the scram mechanism is sufficient to prevent burn-out in the core. Also the situation which occurs after a trip of the main turbine stop valve was analyzed.

Further calculations will be performed on a cold water accident, a start-up accident, trip of one main circulation pump and excessive steam demand. All these analyses ask for intensive analogue computer use. With the existing computer facilities only approximate data can be obtained.

4.6. Design and engineering of systems and components

4.6.1. The engineering diagram for the primary system of the NEREUS interim design is shown in Fig. 2 and mentioned in para. 1.1 of this report. The primary coolant system consists o ftwo closed cooling loops connected in parallel to the reactor pressure vessel. The reactor is a two pass flow internal recirculation reactor. By means of ejectors the pressure vessel inlet flow is mixed with part of the core outlet flow and the combined flow passes through the diffusor tubes of the ejectors to the bottom plenum.

Each loop contains a vertical steam generating unit with integral superheater and canned rotor pump, which is equipped with a check valve, two main gate type isolating valves and associated piping, valves and instruments.

To each loop a decay heat removal pump is connected to provide for a reduced amount of coolant circulation during plant shut down for the removal of decay heat from the core.

Each loop can be isolated completely from the reactor pressure vessel by means of the shut-off valves. A by-pass line connecting the hot and cold leg of a loop, by-passing the reactor pressure vessel, can be used for circulation within the loop when it is isolated from the pressure vessel. This by-pass line is provided with a valve which is normally closed.

The primary coolant system is connected to the following auxiliary systems:

a) The pressurizing system;

b) The reactor coolant purification system;

c) The reactor coolant make-up system;

d) The emergency cooling system;

e) The vent and drain system.

4.6.2. The engineering diagram of the pressurizing system is included in the engineering diagram of the primary system, Fig. 2. It consists of the pressurizer, shown in Fig. 15, containing saturated water topped by a steam volume at the saturation temperature corresponding to the desired pressure. The pressurizer is connected bv a surge line to the reactor pressure vessel outlet header inside of the isolation valve and with a spray line to the inlet header.

In the bottom of the pressurizer electrical heating elements are installed, supplying a constant amount of heat to the water content during all normal plant operations. During steady state operation a constant amount of water is sprayed into the vapour space. The heat supplied by the electrical heating elements compensates the heat losses to the sprayed water and through the insulation on the outside of the pressurizer.

When the reactor coolant contracts and expands during power changes the vapour in the top of the pressurizer will be compressed and expanded resulting in condensation of vapour or flashing of water into steam. The control valve in the spray line, actuated by pressure changes in the vapour space, will increase or decrease the spray quantity at insurges respectively outsurges, so that the pressure changes will be within the required limits for all normal plant operating conditions. The pressurizing system is connected to the emergency cooling system and the blow-off system.

4.6.3. The engineering diagram for the blow-off system is shown in Fig. 16. The blow-off system is provided to release vapour from the pressurizer at an abnormal high pressure of the reactor coolant and associated systems and to blow down water when an abnormal high water level in the pressurizer occurs. The system also serves as a temporary storage for reactor coolant make-up for gaseous and liquid waste.

The system consists of a set of relief, blow-down and isolating valves, a blow-off tank equipped with a cooler and associated piping and instruments.

The vapour and liquid released by the pressure relief and blow-down valves passes through closed piping to a distribution header inside the blow-off tank, which is provided with jets and liners to quench and cool the flow with the water from the blow-off tank.

The absorbed heat is removed by a vertical tube type cooler connected to the component cooling water system, installed in the center of the tank. The vapour space of the blow-off tank is connected to the gaseous waste system, which absorbs the hydrogen and fission gases released from the water. The system pressure is maintained slightly above the atmospheric pressure to prevent any air leakage into the system which might lead to the build-up of an explosive mixture.

The blow-off tank is also used as temporary storage for reactor coolant make-up. The make-up pumps draw water directly from the blow-off tank. Also the vent header is connected to the blow-off tank, while the drain header to the containment drain tank is normally blocked and provided with an open connection to the blow-off tank. By this arrangement the release of radioactive waste is limited, as all possible leakage through the seats of the pressure relief, blow-down, vent and drain valves collects in the blow-off tank to be re-used as reactor coolant make-up. When the water level in the blow-off tank reaches a high level set-point, water is released to the containment drain tank. When the level reaches a low level set-point fresh water is supplied to the tank.

4.6.4. The engineering diagram for the purification system is shown in Figure 17. The purification system is a high pressure system which consists of a regenerative and non-regenerative heat exchanger, a set of three parallel demineralizer/filter units and associated piping, valves and instruments.

Each demineralizer/filter unit consists of a pressure vessel in which a mixed resin bed with inlet and outlet filters are provided. The resin bed and the in- and oulet filters can be removed as a whole unit to facilitate recharging. This method eliminates handling of loose radioactive resins and filters on board the ship, while the time required for recharging the purification system is limited to a minimum. The arrangement of the purification system inside the containment vessel is shown in Figure 18.

The remaining auxiliary systems have still to be worked out in detail.

4.6.5. The vertical steam generating unit with integral superheater and canned rotor pump is shown in Figure 19. The unit consists of a vertical cylindrical section containing the U-tubes of the evaporator section, steam drying equipment and feed water header, and a horizontal cylindrical section containing the superheater and carrying the primary coolant pump. Primary coolant flows into the bottom part of the horizontal section from the left on the drawing and there passes on the outside of the U-tubes of the superheater.

The coolant then passes upwards to the chamber on the right of the upper part of the horizontal section and into the U-tubes of the evaporator. It returns from these U-tubes to the left chamber in the upper part of the horizontal section, which is also the suction chamber for the coolant pump. The coolant is then discharged from the pump casing.

Feed water enters into the feed water heater of the evaporator section. There it is evaporated and passes as steam through the steam drying equipment. Dry steam is led from the top of the vertical section to the upper inlet on the extreme right of the horizontal section, passes through the U-tubes of the superheater and out through the outlet nozzle to the main containment isolating valves.

The steam is routed through the tube side of the superheater in order to improve the overall heat transfer coefficient by improving the heat transer to steam which is now flowing at very high speed.

The tube plate for the vertical U-tubes is formed by part of the upper wall of the horizontal section of the unit. At present the industrial group Neratoom is preparing an evaluation of the design from the standpoint of fabrication. Also, it is drafting a programme to perform stress measurements on a model of the unit, in order to determine the detailed dimensions. This work is being done under Chapter VI of the Development Programme.

4.7. Shielding calculations and experiments

The shielding experiments in the development programme are being performed at the facilities of the G.K.S.S. at Geesthacht. The purpose of these experiments is to verify the calculations and the calculation models which have been used in the design. The calculations on the primary shielding have shown that a configuration in which thick steel plates were used was to be preferred to a configuration using the same thickness of steel, but divided over more plates.

From the calculations made, a preliminary proposal for the primary shield was selected - the reference configuration. Starting with this configuration, the influence of the following variations will be studied:

a) Influence of the positioning of the steel plates;

b) Thickness of the steel plates;

c) Influence of borated steel;

d) Influence of pressure vessel wall thickness;

e) Influence of variation of the water-steel ratio of the thermal shielding inside the pressure vessel.

Measurements will be performed on thermal neutron flux, epithermal flux, fast neutron flux and gamma radiation.

The experimental work was started in the second quarter of 1964.

Only preliminary calculations on the secondary shielding for the NEREUS interim design have been performed to date.

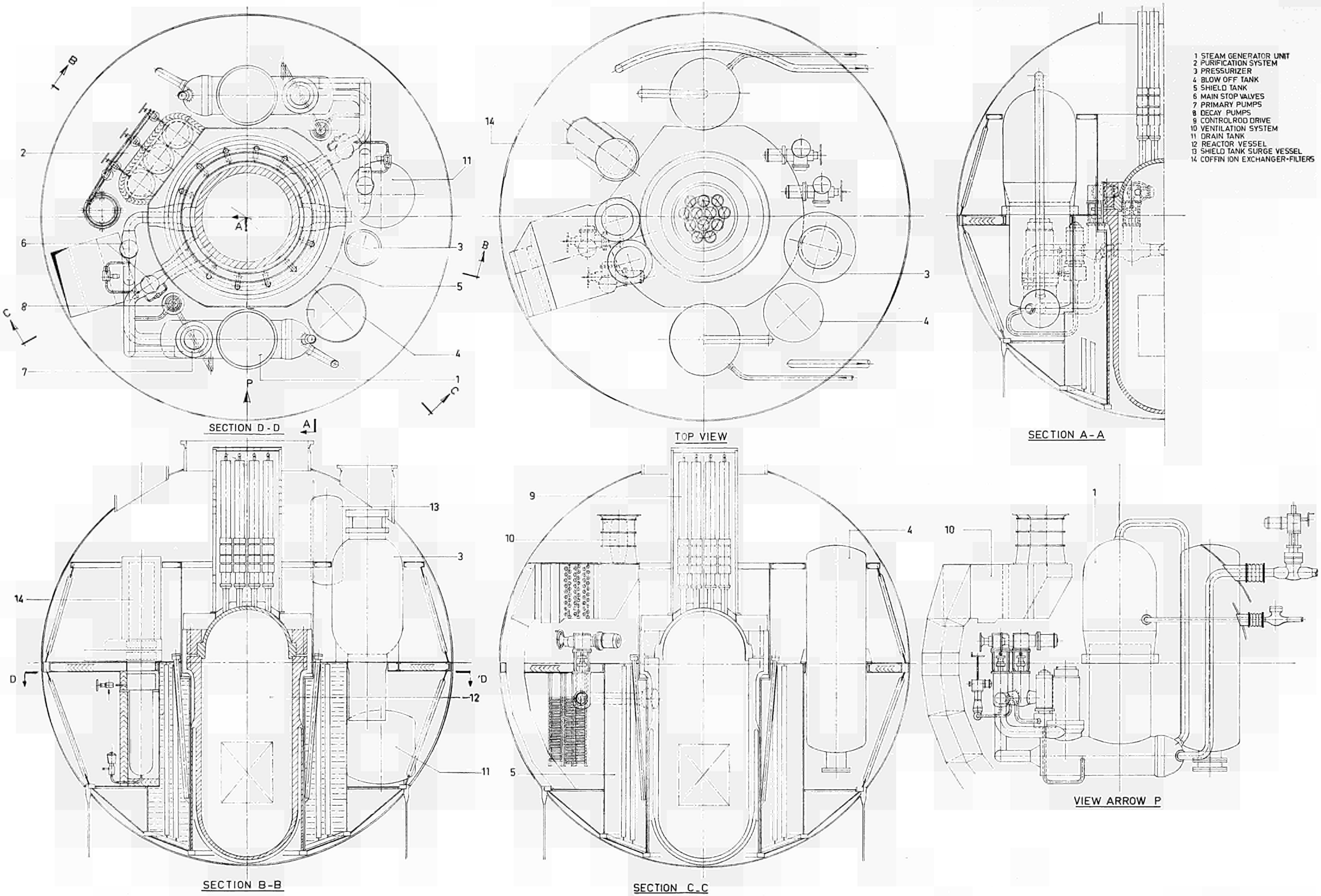


Fig. 1

Arrangement inside containment.

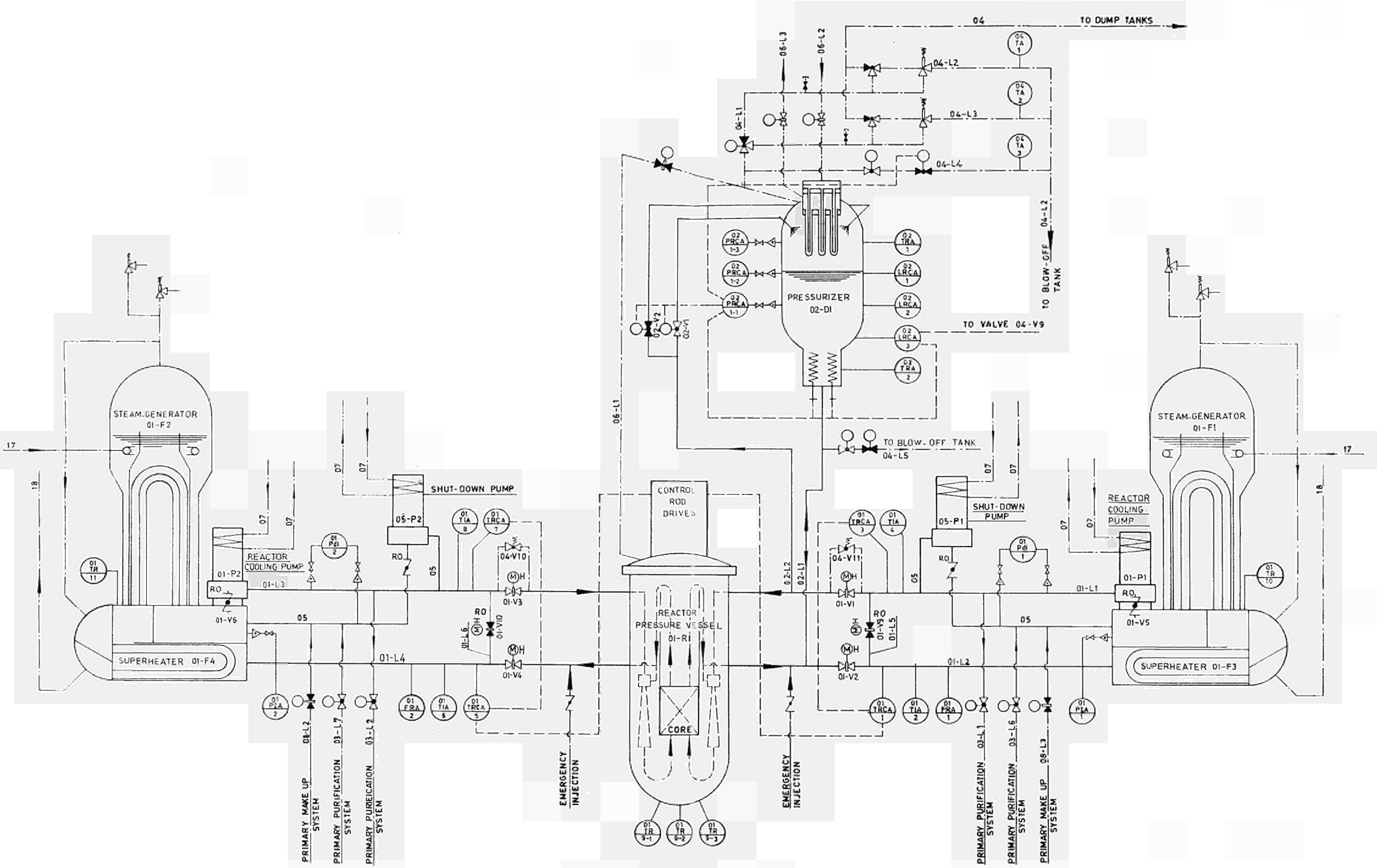


Fig. 2
Engineering diagram primary system.

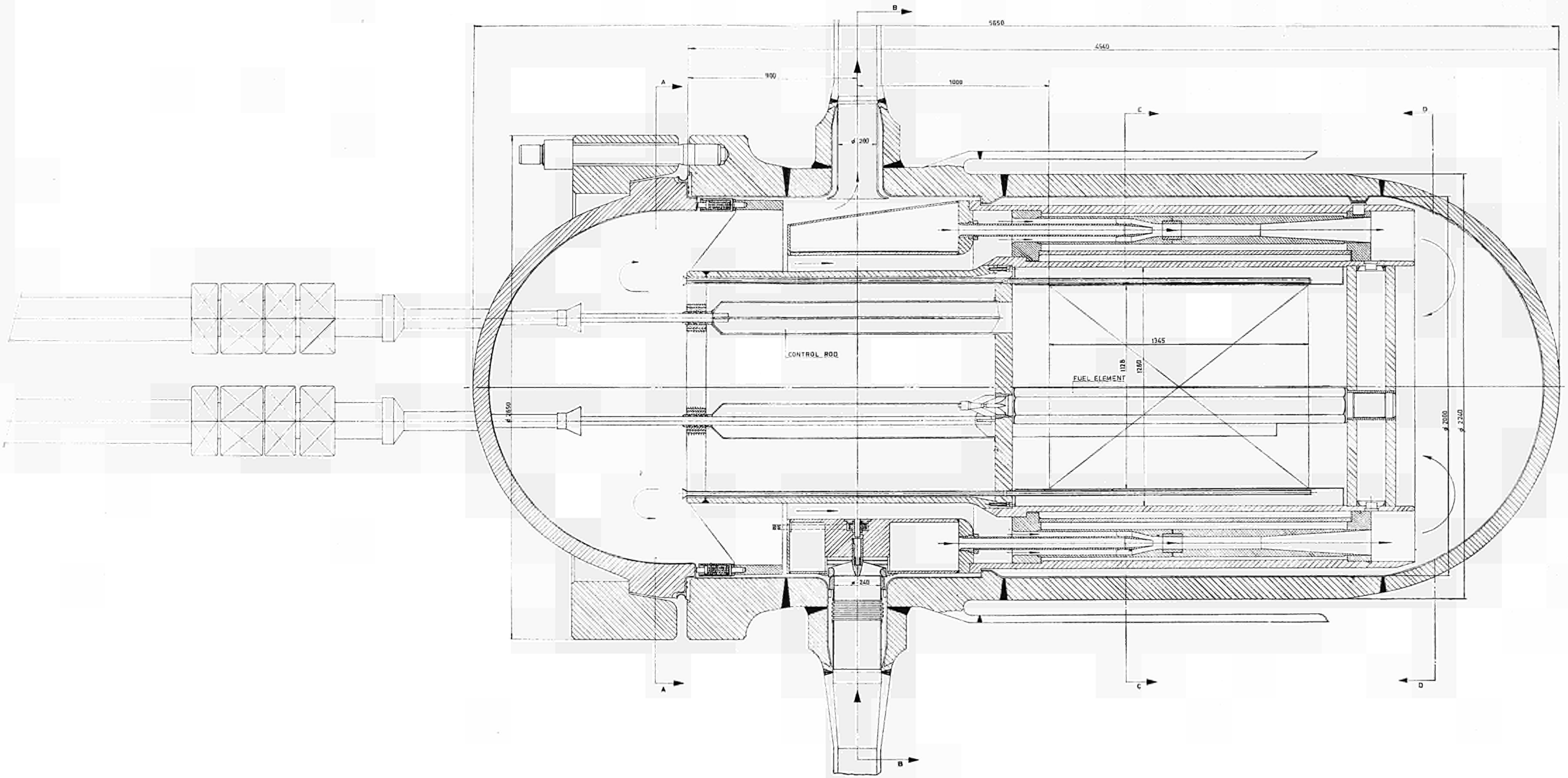


Fig. 3
Longitudinal section of the reactor vessel.

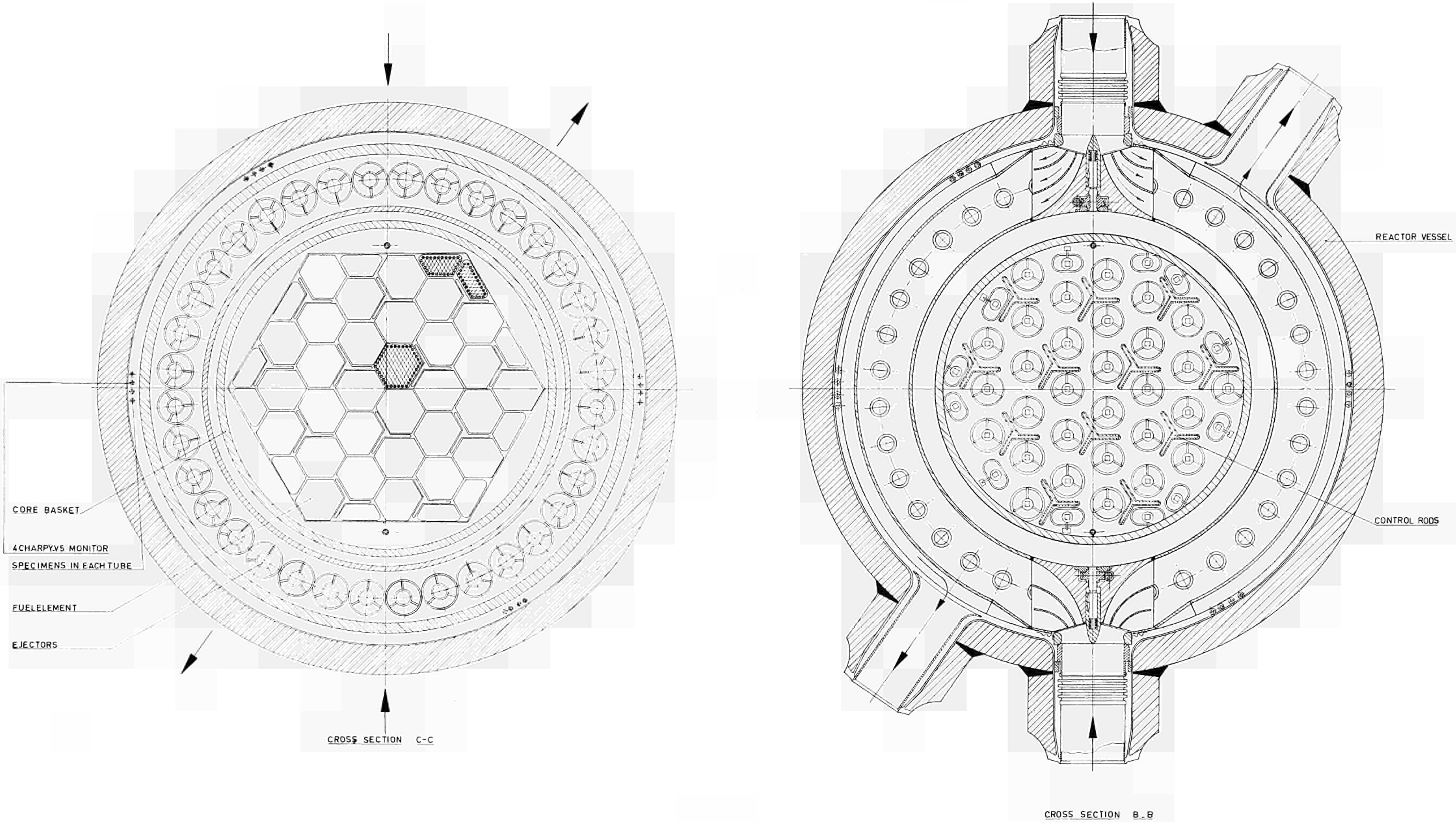


Fig. 4
Reactor vessel cross section.

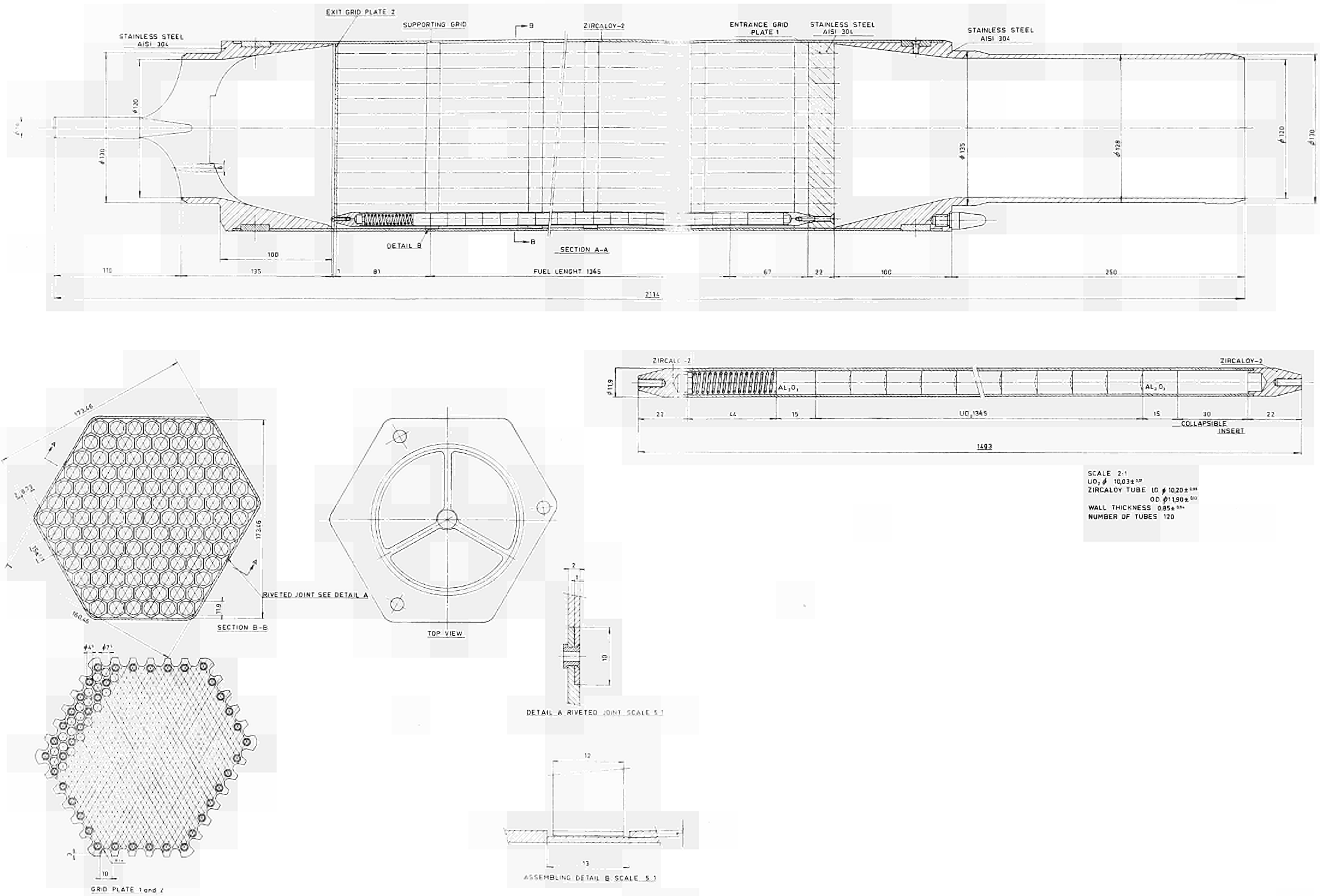


Fig. 5

Fuel element.

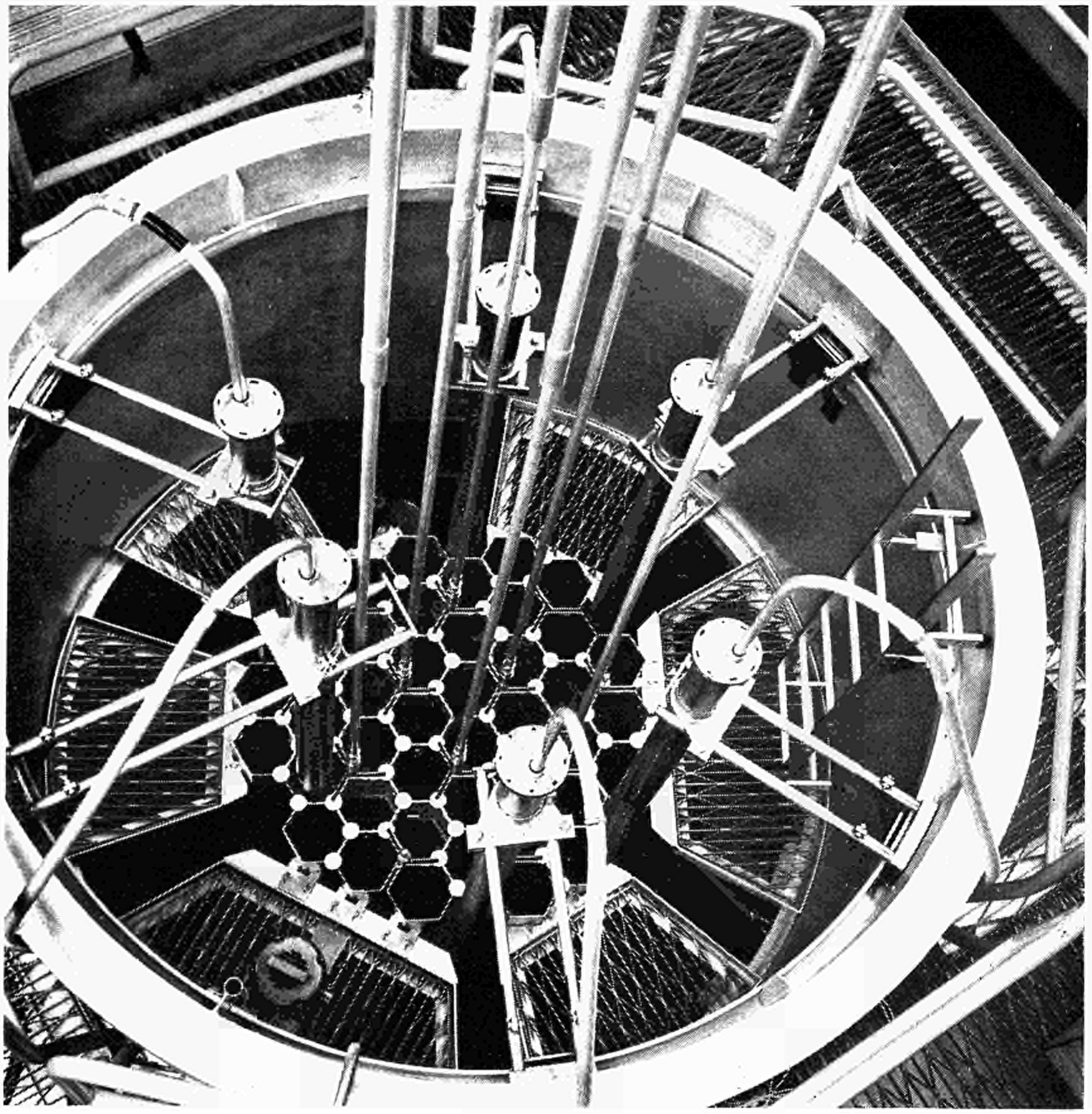


Fig. 6
KRITO, top view.

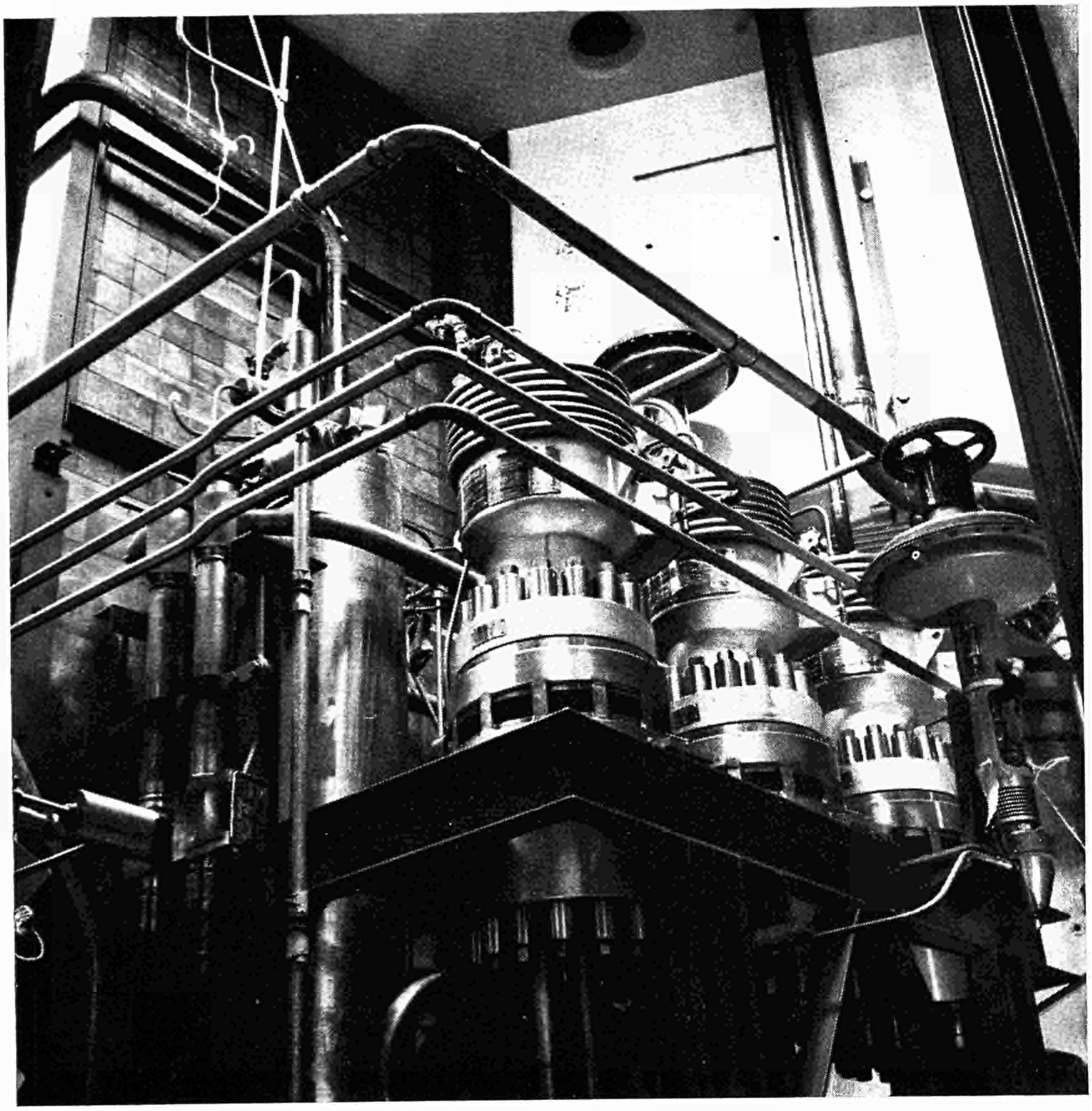
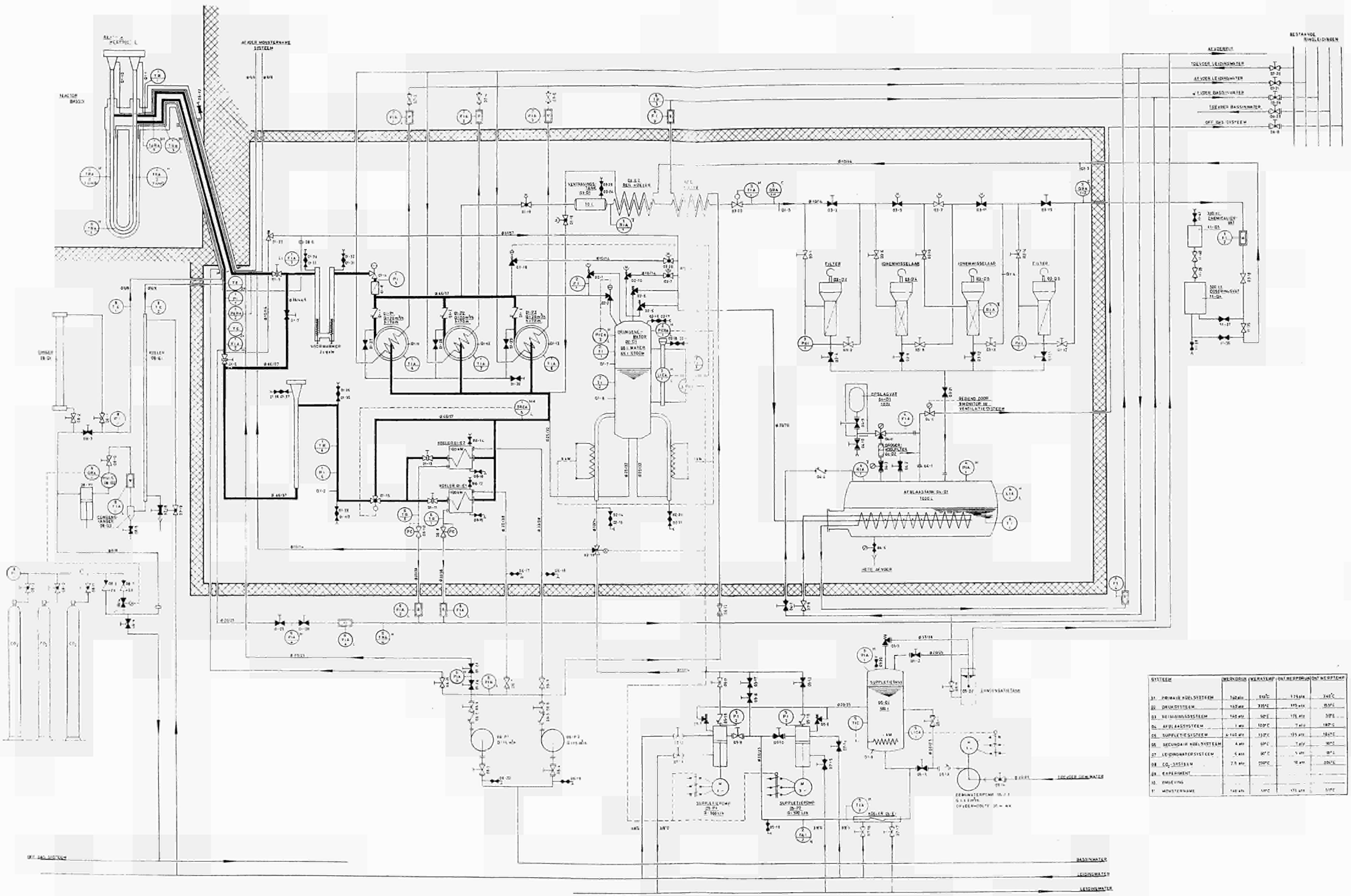


Fig. 7

High pressure irradiation loop, general view.



SYSTEEM	WERKDRUK	WERKTEMP	ONTWERPDRUK	ONTWERPTEMP
01 PRIMAIR KOELSYSTEEM	140 atu	310°C	175 atu	340°C
02 DRUKSYSTEEM	140 atu	335°C	175 atu	355°C
03 REINIGINGSSYSTEEM	140 atu	50°C	175 atu	[illegible]
04 AFBLAASSYSTEEM	1 atu	100°C	7 atu	[illegible]
05 SUPPLETIESYSTEEM	4-140 atu	130°C	175 atu	[illegible]
06 SECUNDAIR KOELSYSTEEM	4 atu	60°C	7 atu	[illegible]
07 LEIDINGWATERSYSTEEM	5 atu	90°C	5 atu	[illegible]
08 CO_2-SYSTEEM	7,5 atu	250°C	10 atu	[illegible]
09 EXPERIMENT				
10 OMGEVING				
11 MONSTERNAME	140 atu	50°C	175 atu	[illegible]

FIG. 8

High pressure irradiation loop - flow sheet.

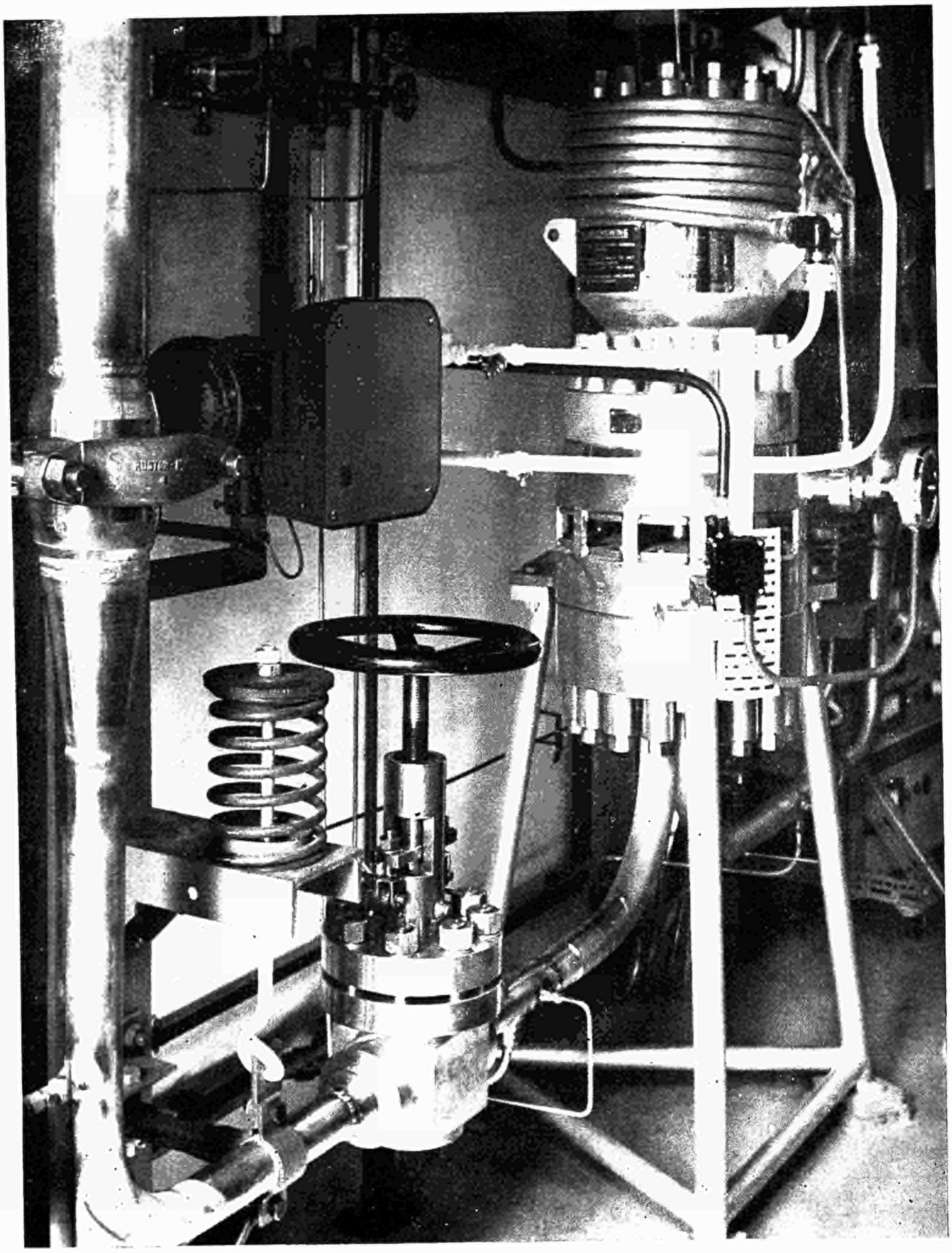


FIG. 9

Corrosion loop, general view.

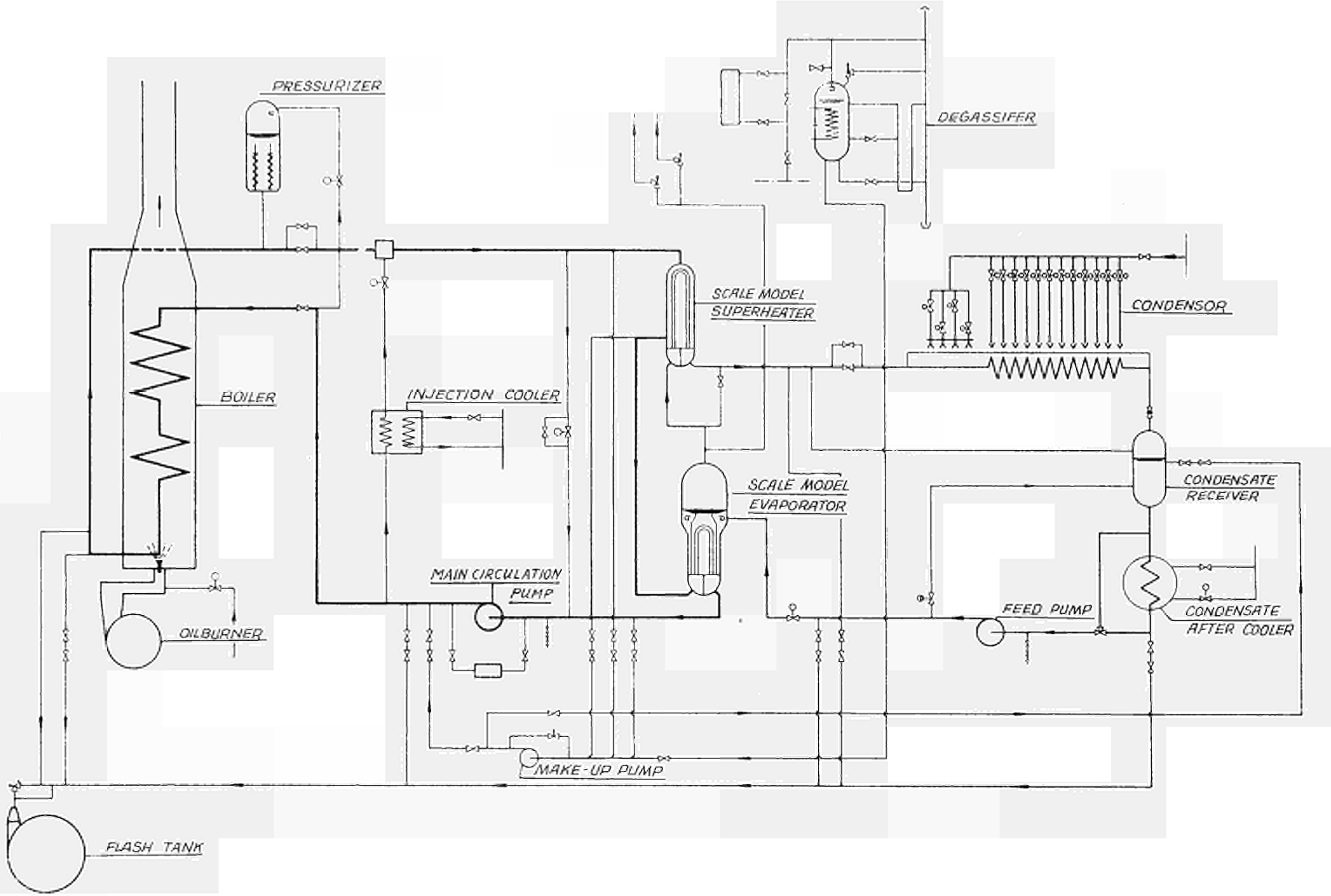


FIG. 10
Heat exchanger test installation - flow sheet.

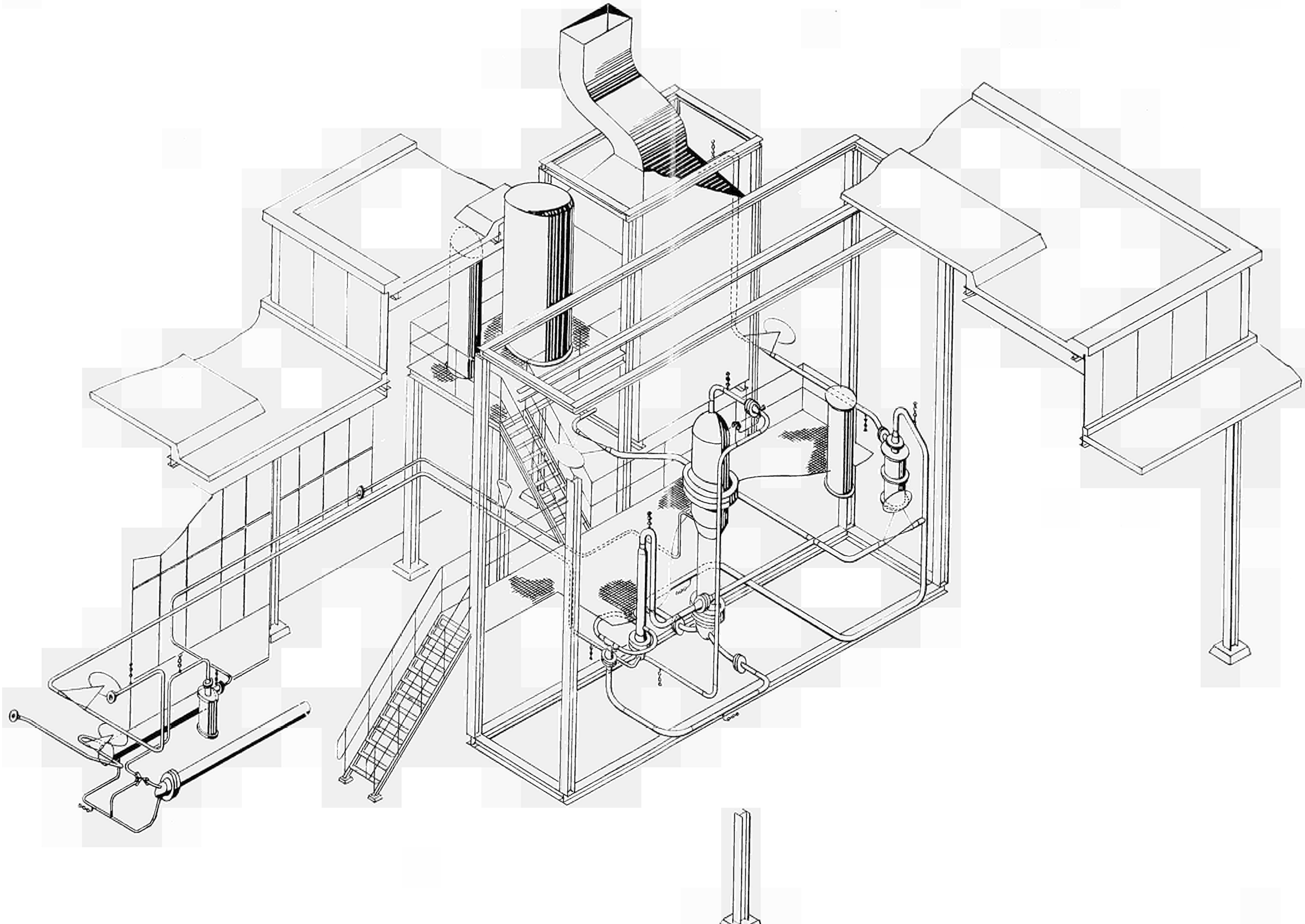


Fig. 11

Experimental facility - steam generator.

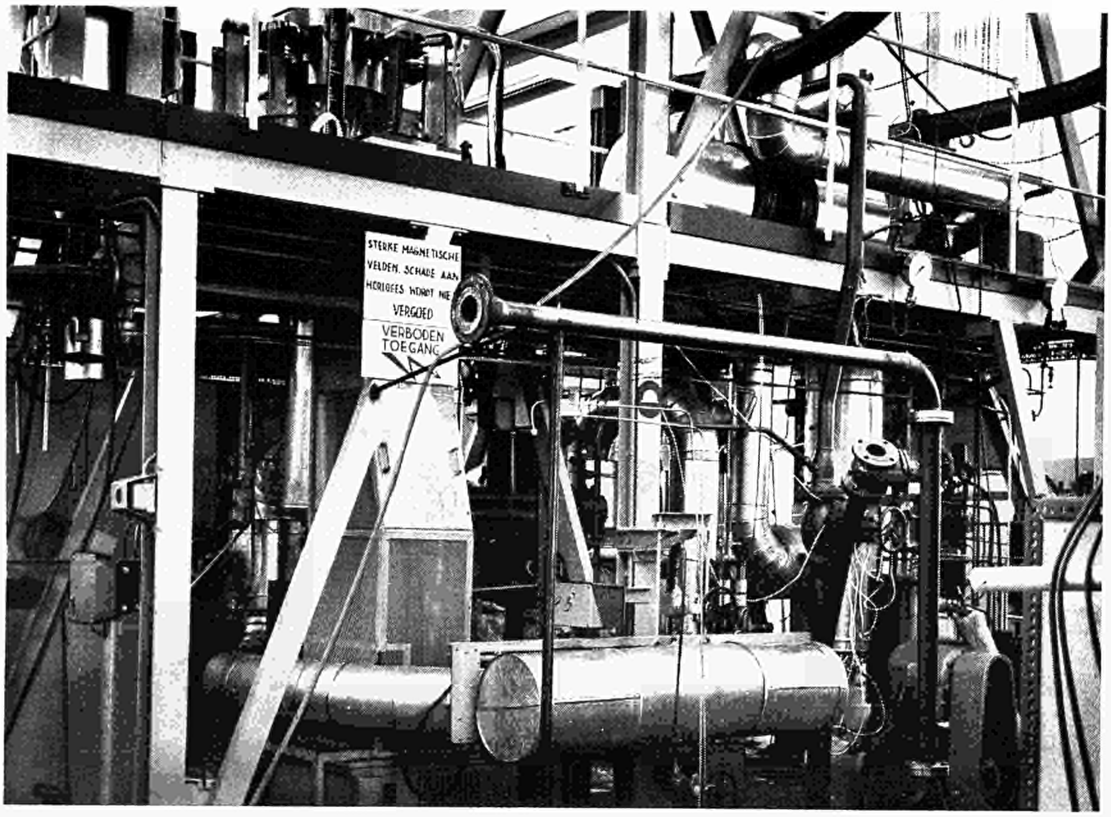


FIG. 12

High pressure heat extraction loop.

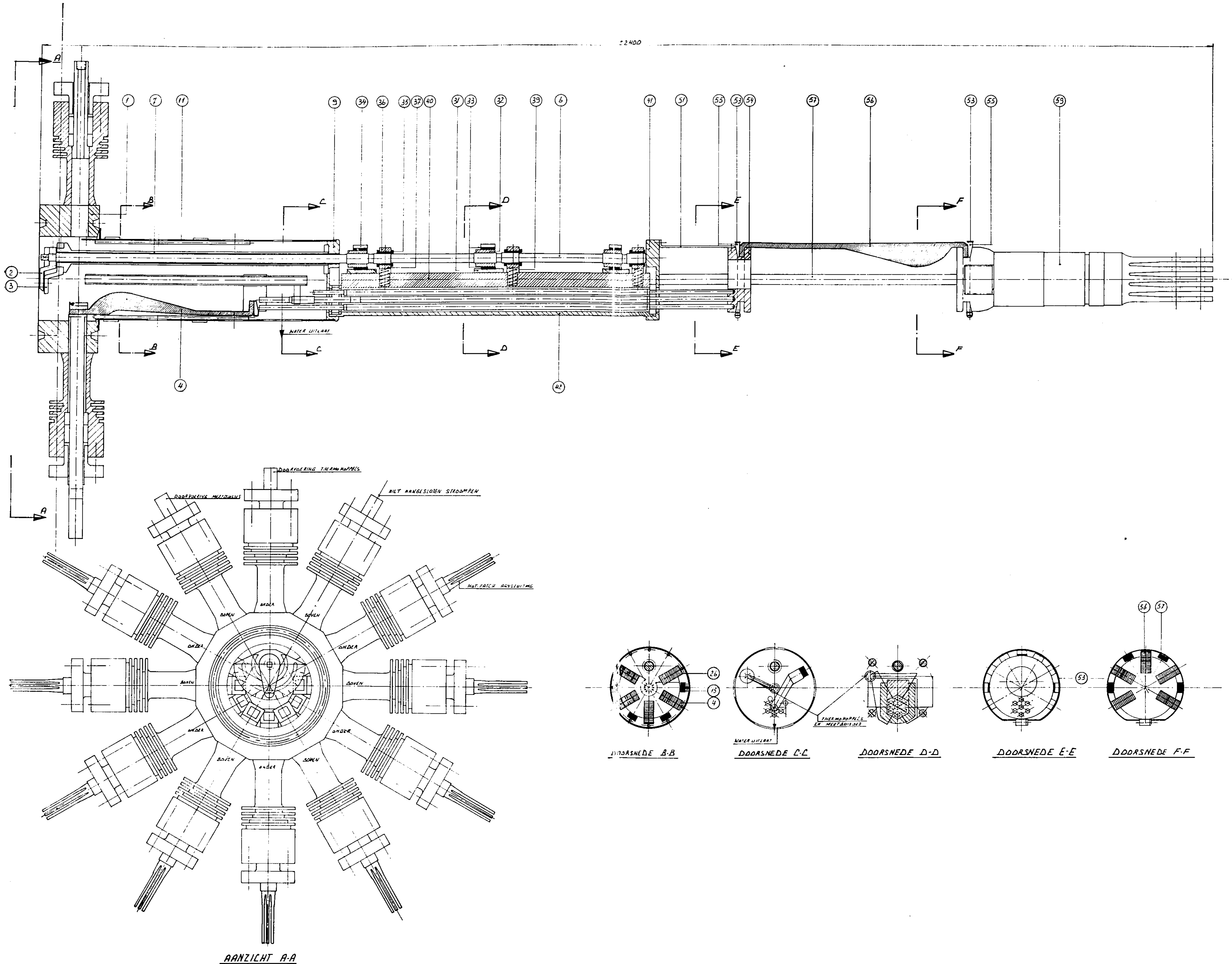


Fig. 13
Assembly heat extraction test element.

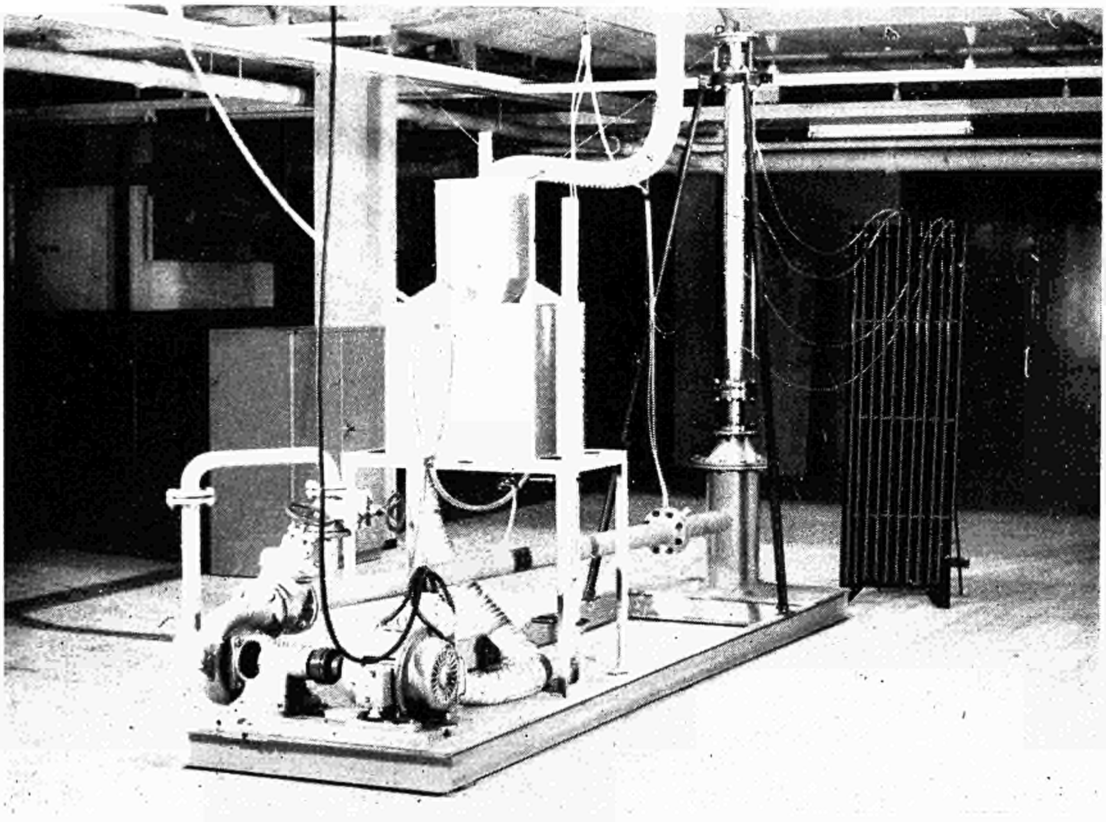


Fig. 14

Hydraulic test assembly.

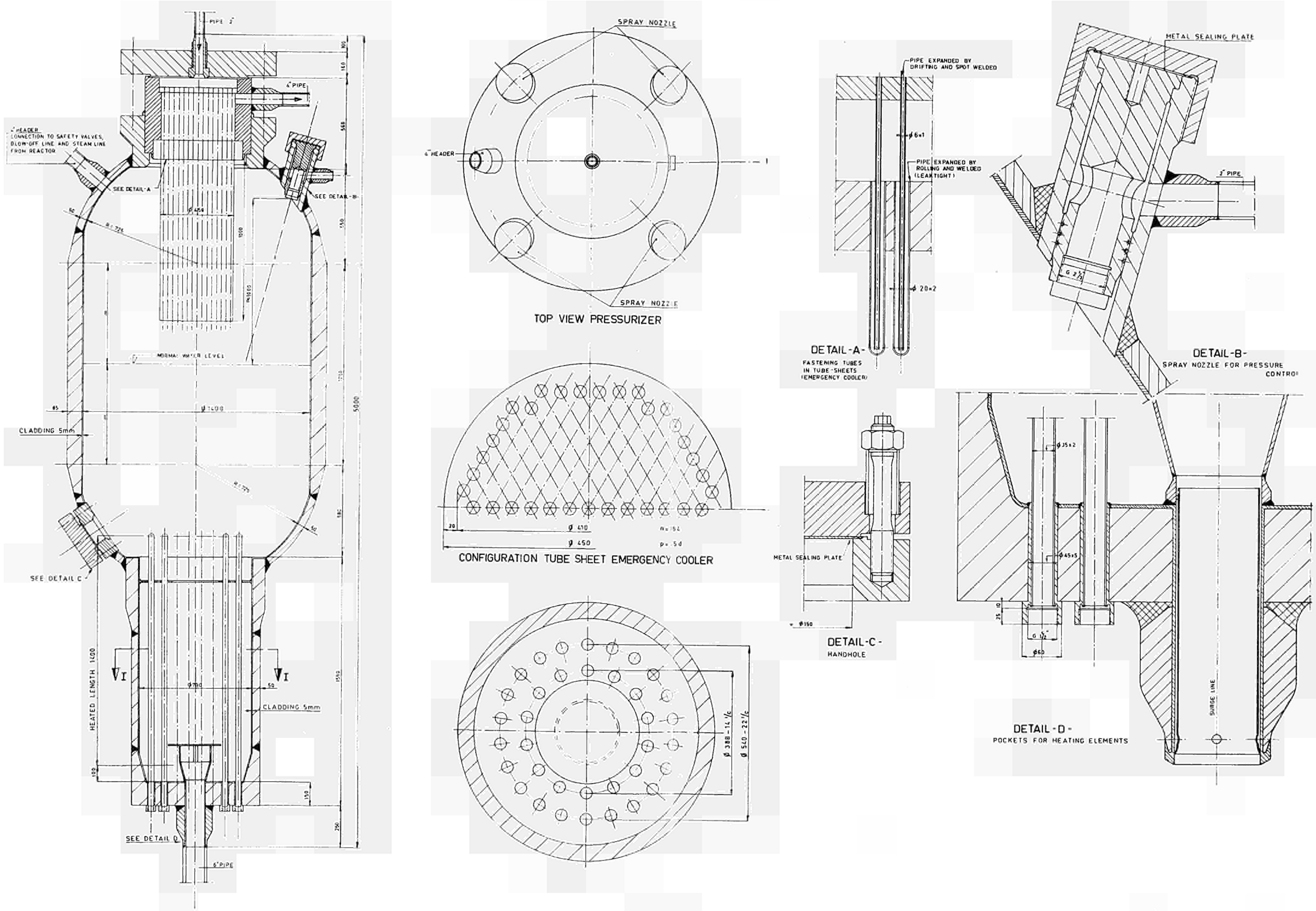


Fig. 15

NEREUS pressurizer.

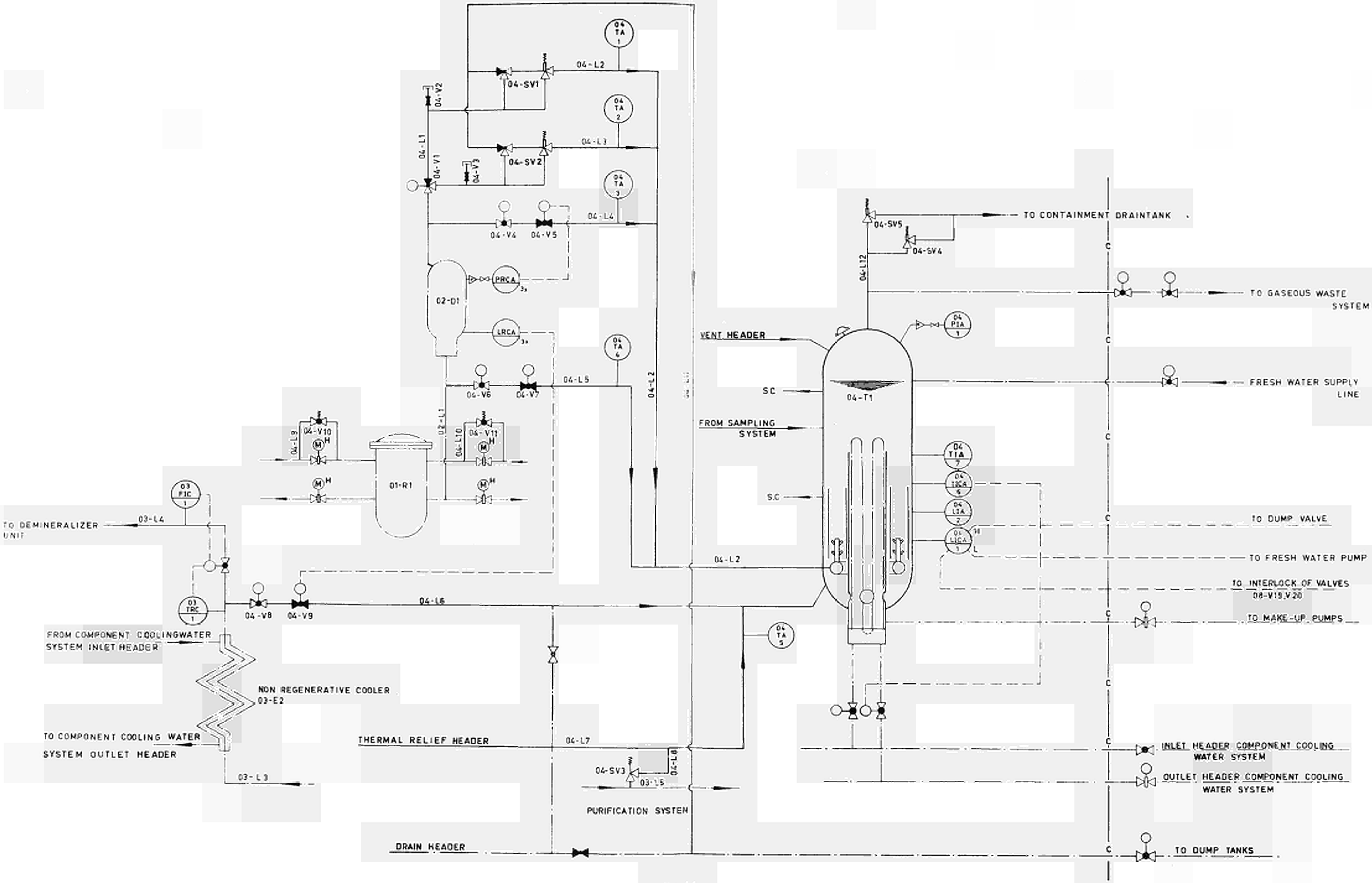


Fig. 16
NEREUS blow-off system.

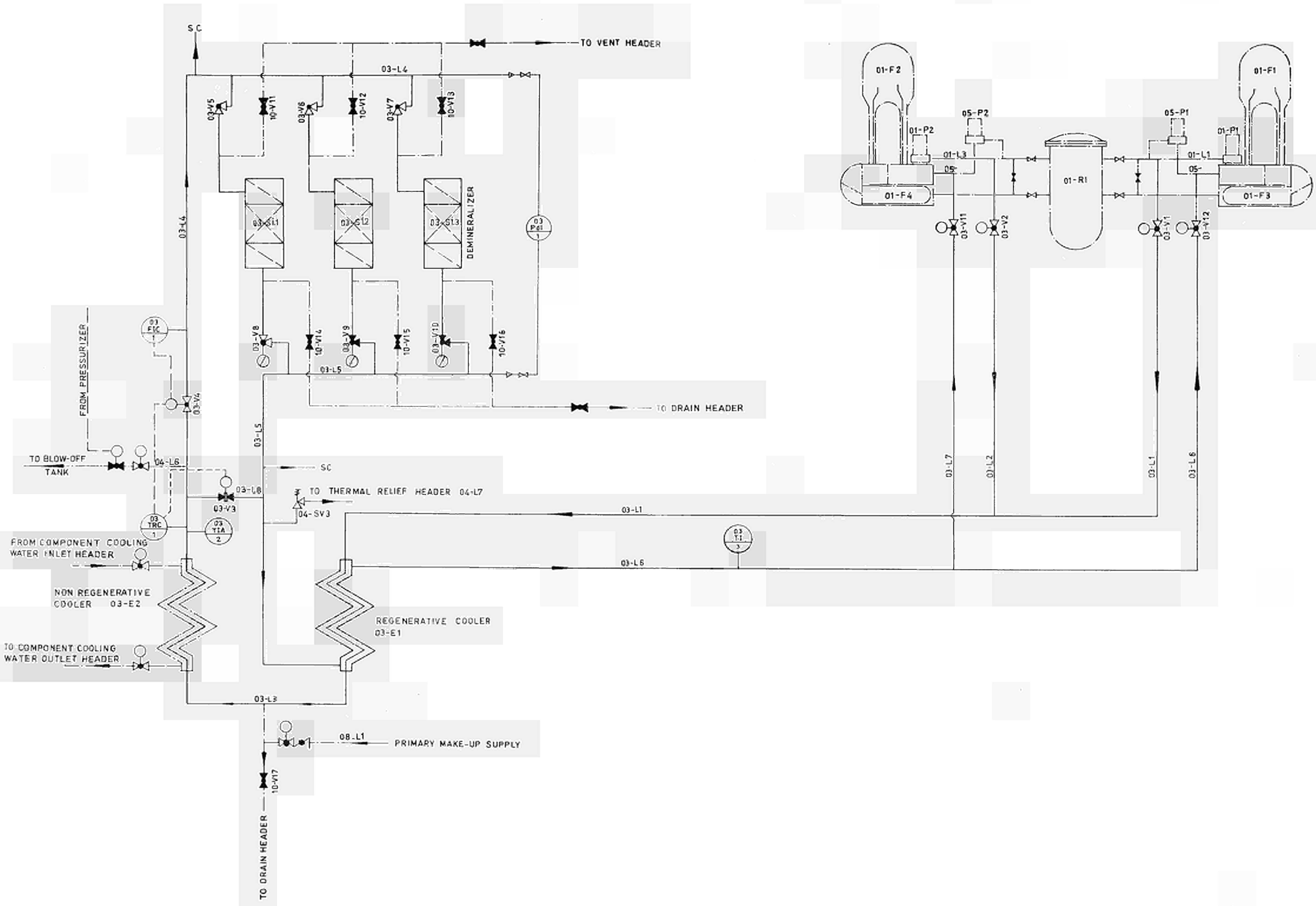


Fig. 17

Primary purification system.

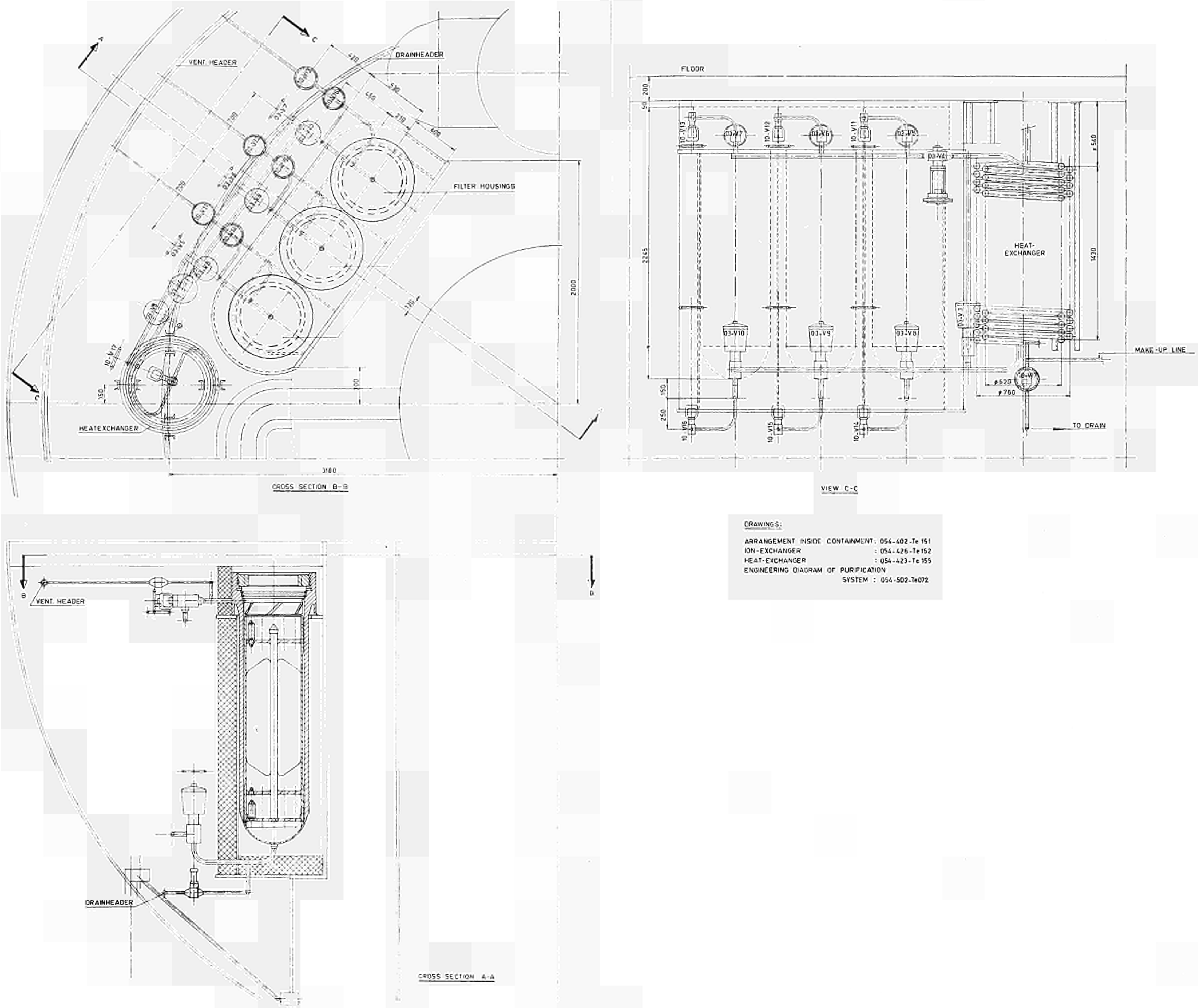
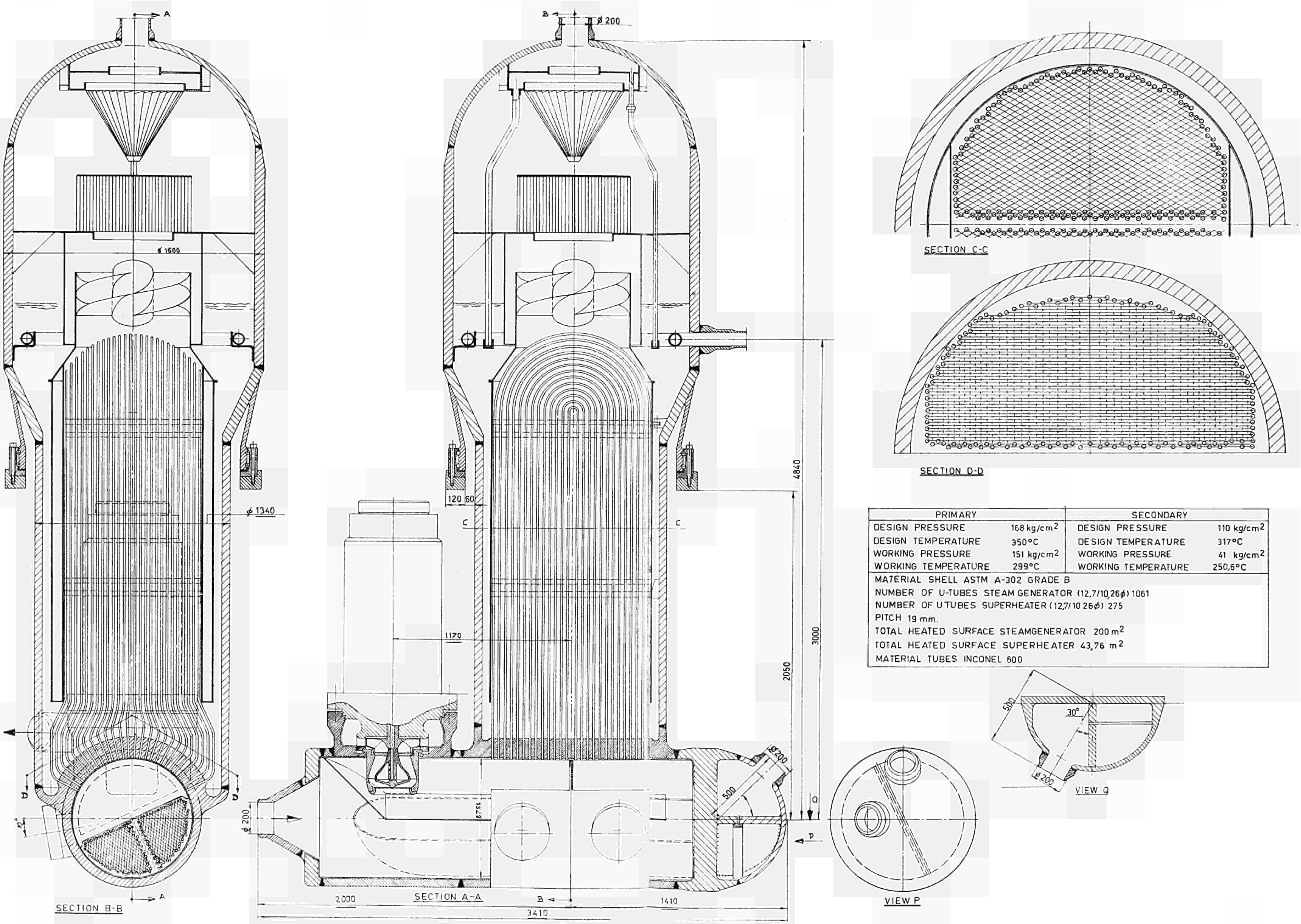


FIG. 18
Assembly NEREUS purification system.



PRIMARY		SECONDARY	
DESIGN PRESSURE	168 kg/cm^2	DESIGN PRESSURE	110 kg/cm^2
DESIGN TEMPERATURE	350°C	DESIGN TEMPERATURE	317°C
WORKING PRESSURE	151 kg/cm^2	WORKING PRESSURE	41 kg/cm^2
WORKING TEMPERATURE	299°C	WORKING TEMPERATURE	250,6°C
MATERIAL SHELL ASTM A-302 GRADE B			
NUMBER OF U-TUBES STEAM GENERATOR (12,7/10,26ϕ) 1061			
NUMBER OF UTUBES SUPERHEATER (12,7/10 26ϕ) 275			
PITCH 19 mm.			
TOTAL HEATED SURFACE STEAMGENERATOR 200 m^2			
TOTAL HEATED SURFACE SUPERHEATER 43,76 m^2			
MATERIAL TUBES INCONEL 600			

Fig. 19

Steam generator/superheater/pump unit.

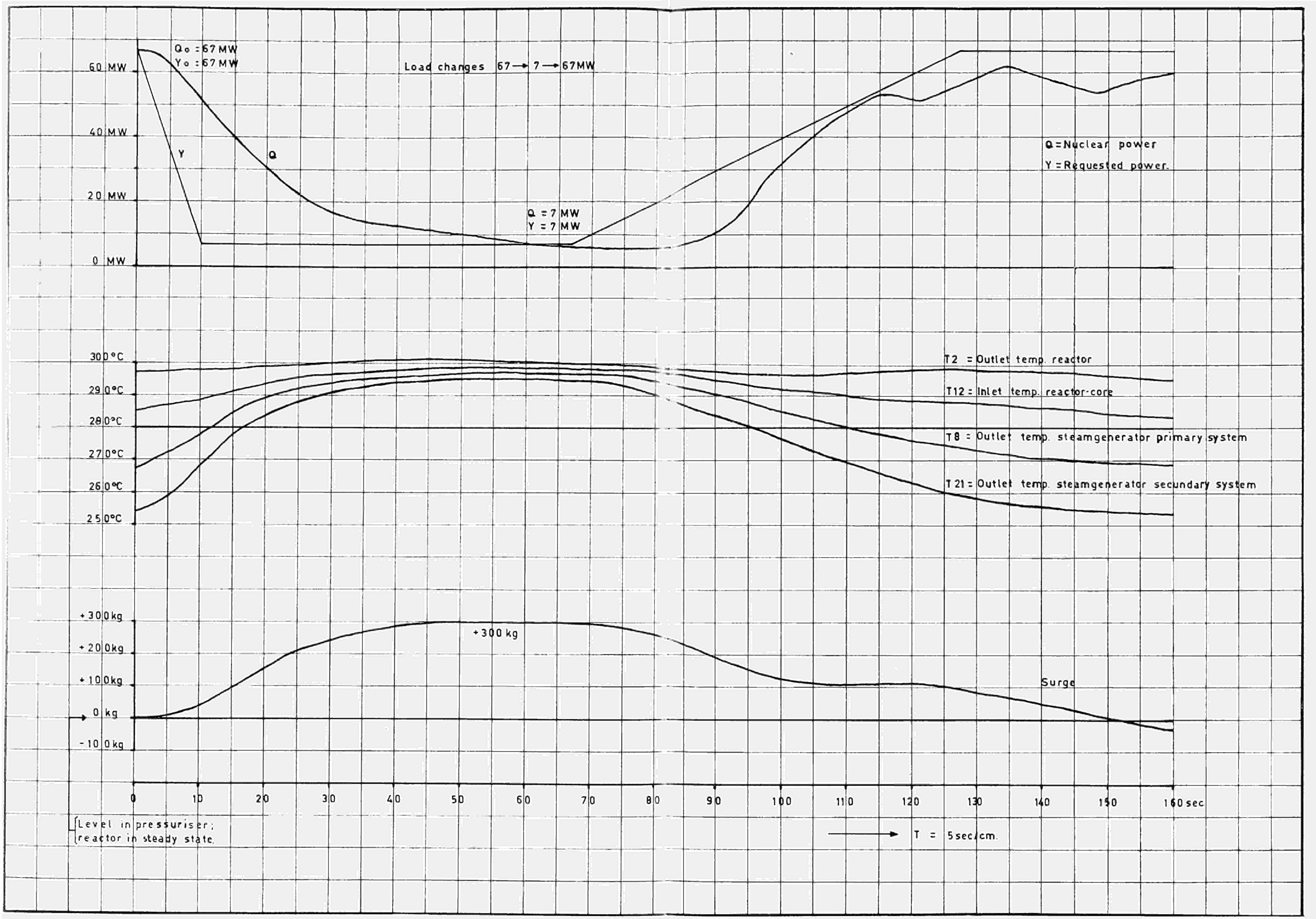


Fig. 20
NEREUS reactor load following characteristics 67, 7, 67 MW.

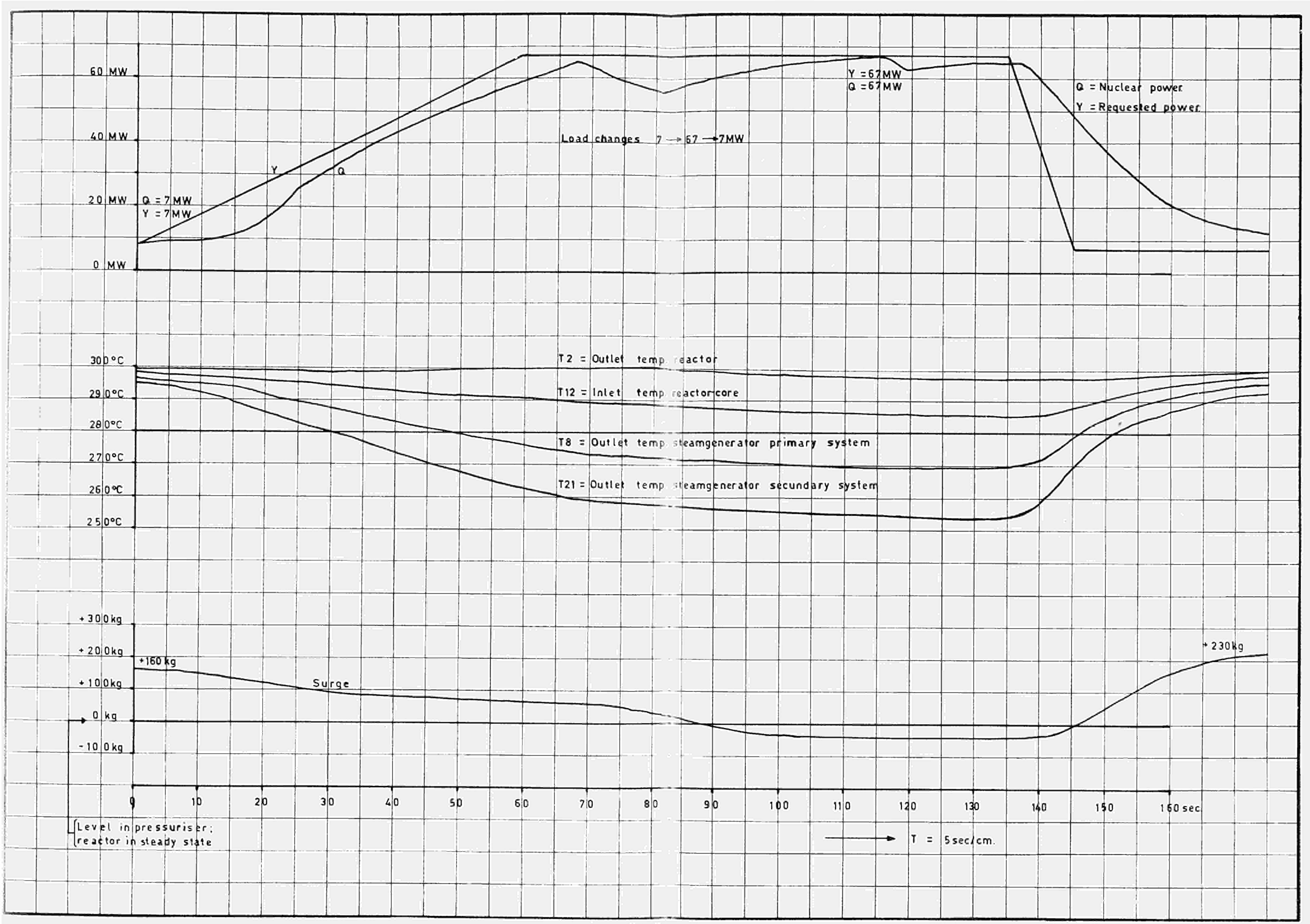


Fig. 21

NEREUS reactor load following characteristics 7, 67, 7 MW.

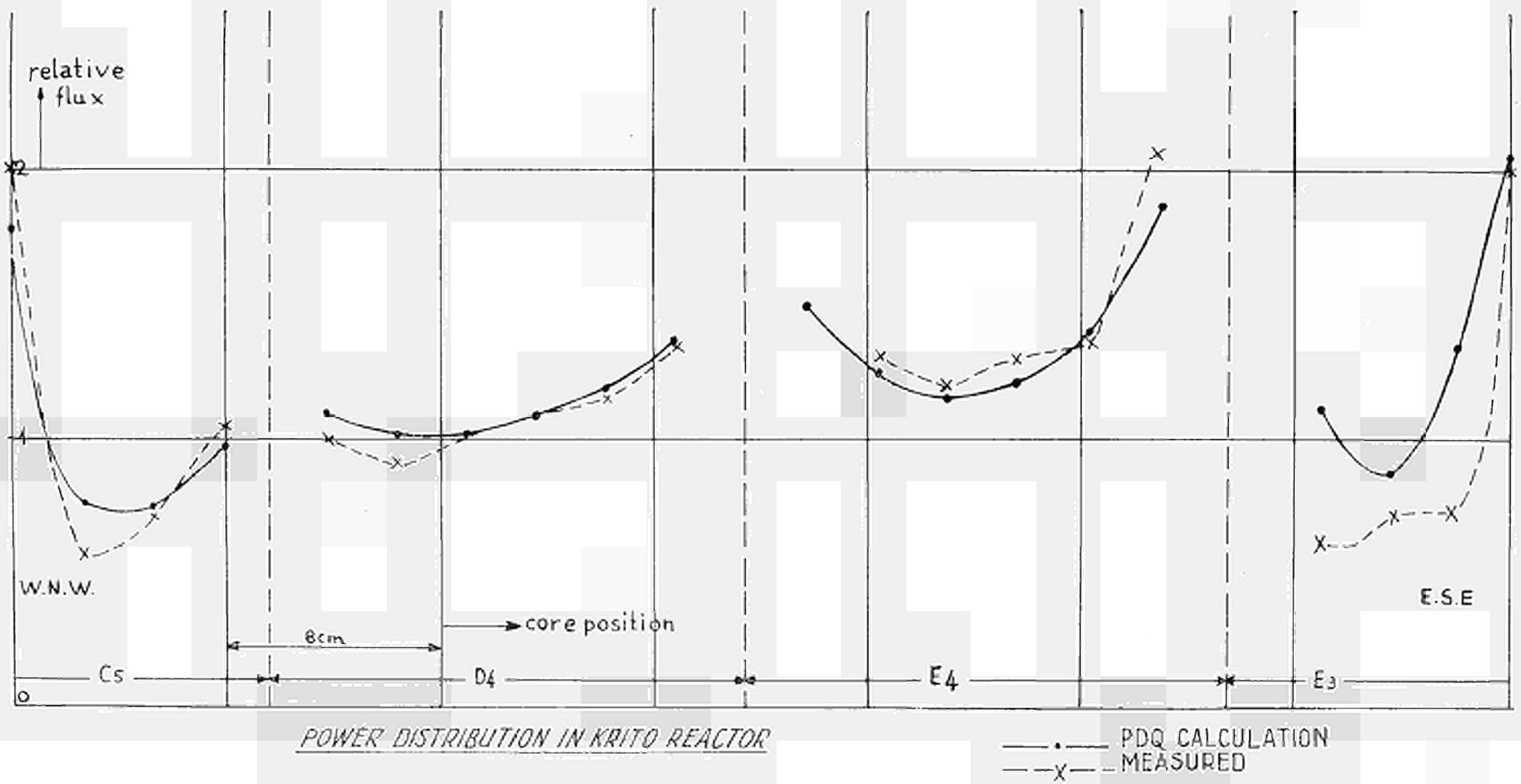


Fig. 22 Power distribution in KRITO reactor.

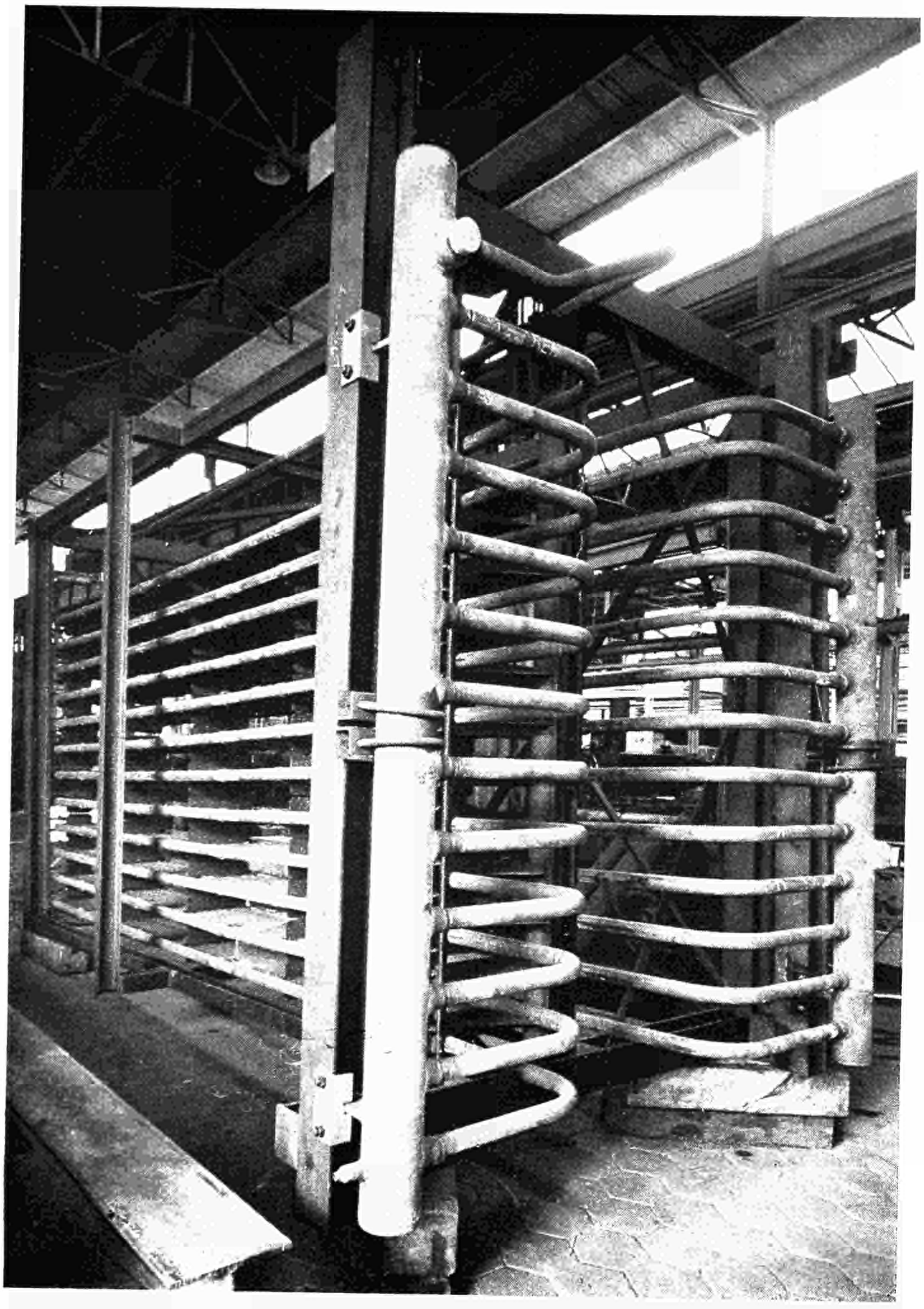


Fig. 23

Fast control condensor of steam generator test installation.

To disseminate knowledge is to disseminate prosperity — I mean general prosperity and not individual riches — and with prosperity disappears the greater part of the evil which is our heritage from darker times.

Alfred Nobel