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EUROPEAN ATOMIC ENERGY COMMUNITY - EURATOM

EURATOM PARTICIPATION IN FIVE NUCLEAR POWER STATIONS IN THE COMMUNITY

Description of the progress and status of development by May 1963

1963

Industry and Economy
Reactors and Nuclear Power Plants
Brussels

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F O R E W O R D

The subject of the present Report is Euratom's participation in five nuclear power plants in the Community.

At the time of this Report's going to press, the SIMEA plant has reached the full-load stage, the SENN plant will also have reached this stage in the foreseeable future, construction of the SENA plant is well under way, the civil engineering work is being carried out on the KRB plant, while the detailed designing work for the SEP project has got off to a good start.

This Report has two aspects, one being of a permanent and the other of a provisional nature. The former relates to the projects themselves and the method of Euratom's participation, the latter to the status of progress.

The information meetings so far organized by Euratom concerning these projects have afforded still further - and unmistakable - the evidence of the lively interest which is being displayed on all sides in the industrial application of nuclear techniques. It is principally in response to this desire for information that the present Report has been compiled and we trust that it will play its part in stimulating the activity of the Community's nuclear industries.

C O N T E N T S

1. EURATOM PARTICIPATION IN POWER REACTORS

by

E. PARKER and C. RAMADIER

1. EURATOM'S PARTICIPATION IN POWER REACTORS

During the next few years the most important data to be obtained concerning power reactors will be those gained from the operation of the first reactors to be commissioned. Even now, the information obtained presents considerable interest, a fact which is clearly evidenced by the success of the IAEA Conference on Operating Experience with Power Reactors, held in Vienna from 4 to 8 June 1963.

The European Atomic Energy Community, one of whose tasks as laid down by the Treaty of Rome is to further the development of the Community's nuclear industry, has realized the need for gathering data on the operation of power reactors. On a wider basis, it is Euratom's view that the experience and results to be obtained from the design, construction and operation of power plants are extremely valuable for the development of the Community's nuclear technology. It is for this reason that the Community decided to launch a programme of participation in power reactors.

This programme, initiated by the Commission, consisted in awarding contracts worth a total of 32 million EMA u.a., to five of the Community companies engaged on the construction of power plants.

These are as follows:

- the Società Elettronucleare Nazionale (SENN), which is having constructed a 150 MWe net power plant in Italy, equipped with a dual-cycle boiling-water reactor.

- the Società Italiana Meridionale Energia Atomica (SIMEA) which has undertaken the construction of 200 MWe net plant equipped with a natural uranium/graphite/carbon dioxide reactor.

- the Société d'Energie Nucléaire Franco-Belge des Ardennes (SENA), which has undertaken the construction on the Franco-Belgian frontier of a power plant running on a pressurized-water reactor developing 242 MWe net, which can later be stepped up to 266 MWe net.

- the G.m.b.H. Kernkraftwerk RWE - BAYERWERK (KRB), which is building a 237 MWe plant in Bavaria equipped with a dual-cycle boiling-water reactor

- the Samenwerkende Electriciteits-Productiebedrijven (SEP), which has undertaken the construction of a 50 MWe net power plant operating on a boiling-water natural circulation reactor.

For their part, the contract holders communicate to the Commission all the data obtained throughout the period covered by the contract with regard to the design, construction and operation of the plant. This data is supplied either in the form of documents and reports, or is gathered on the spot by engineers seconded to contract holders. The following documents must be submitted by the contract holders:

- design and construction plans
- manuals of operation and equipment
- safety reports
- operational data

The reports contain technical, economic and safety information relating to the plant. Certain reports appear regularly (annually or quaterly), others supply overall data (on completion of the work concerned or of the contract after four years' running). Finally, others contain information of a specialized nature relating, for example, to questions of health and safety, the transportation of fuel elements and accidents and incidents occurring during operation.

The most novel departure embodied in these contracts of participation, however, concerns the clause under which ten engineers can be assigned to each of our contract-holders at the same time. These engineers, who are Euratom officials and engineers employed in industry or by Community bodies, can observe the activities of the staff employed by the contract-holders, becoming integrated with these teams as far as is possible. Visiting scientists or students can also be attached to contract-holders for periods of study or training.

The information, reports and documents thus received can be used by the Commission and communicated to third parties in the Community by the Commission with the agreement of the contract-holders.

Certain information can be published. It is in this way that the first annual reports of SIMEA, SENN and SENA were published.

The first two of these contracts (SIMEA and SENN) were signed at the end of 1961, the third (SENA) in mid-1962, while the last two (KRB and SEP) have just been signed.

In actual practice, our contract-holders have regularly submitted to us the reports and documents required, copies of which can be distributed to Community industries and bodies, and which can be examined at the Euratom head office. They at present make up a total of more than 10,000 pages of text and drawings.

A total of 54 engineers from the six Community countries have been or are shortly about to be seconded to the SENN and SIMEA projects. These qualified engineers carry out on-the-spot studies of the problems which interest them particularly or participate in the work of the construction companies, especially commissioning trials. At the end of their assignment they must draw up a report.

At the present moment Euratom has at its disposal about 30 such reports, constituting an invaluable source of information.

The information obtained depends on the progress achieved by the work being carried out under the five projects. The main items dealt with concern:

Start-up of the SIMEA plant
Pre-start-up tests on the SENN plant
Finally, commencement of work on the SENA plant.

The information thus obtained is distributed by the Commission through the agency of the Member States. Euratom itself has so far held three information meetings for the Community industries and bodies. The last meeting, which was particularly concerned with the results deriving from the commissioning of the SIMEA plant and the initial tests of the SENN plant, was attended by more than 200 people.

The Commission has endeavoured to link up this programme of participation with the development of the Community's nuclear industry. For this reason the bulk of the contribution under this programme, about 17 million EMA u.a., is payable only on condition that the fuel elements of the second reactor cores (the first in the case of the SEP plant) are manufactured in the Community. This is done in order to facilitate the creation and development of a fuel element production industry.

In two cases also, this participation covers the manufacture in Europe of certain reactor components, such as pressure vessels.

Finally, in the case of four projects, it is intended to cover some of the additional fixed overheads incurred during the first three years of operation, 5,400,000 EMA u.a. being earmarked under this head.

In conclusion, mention should be made of the fact that Euratom has embarked on a programme of research and development, both as part of its own activities and under the US/Euratom agreement. This programme is primarily concerned with the development of fuel elements and is centred on the reactors in question. In particular, it is planned to provide the instrumentation of the SENN core and negotiations are also in progress with SENA on the same subject. The SEP has decided to instrument its own reactor. The experiments to be carried out during the operation of the SIMEA plant have recently been decided upon with Euratom's agreement.

It should be stressed also that the Euratom programme of participation in five power reactors now in operation or under construction constitutes a novel form of collaboration between utilities, construction companies and an international organization.

As a result of this collaboration certain projects will be given a definite stimulus and a boost will be given to the development of the Community's nuclear industry, especially with regard to fuel element fabrication. Finally, in the years to come, it will enable extremely valuable and useful information on the construction and operation of five large-scale nuclear plants to be amassed and distributed. This is only possible on a large scale such as that provided by a community of six countries, since the opportunities afforded are considerably greater than those which would be offered by one country acting alone.

2. S.E.N.N., THE GARIGLIANO NUCLEAR POWER PLANT

by

H. NACFAIRE

LIST OF FIGURES

2. S.E.N.N., THE GARIGLIANO NUCLEAR POWER PLANT

2.1. General features

The Garigliano nuclear power plant, which was based on the Dresden plant, is of the boiling-water type with forced circulation and direct dual cycle. This plant has benefited greatly from the experience gained in the operation of the Dresden plant. The reactor, the steam-generating equipment, the reactor auxiliaries and the fuel handling and storage facilities are enclosed in a spherical containment shell. The turbine building contains the turbo-alternator and its auxiliaries, the condenser, the feed water equipment and the plant auxiliaries. The mechanical construction workshop, the building for radioactive waste processing and the administration block with the control room are attached to the turbine building.

The principal features of the plant are as follows:

Gross electrical capacity	160 MW
Net electrical capacity	150 MW
Thermal capacity of reactor	507.8 MW
Reactor pressure	71 kg/cm^2
Secondary heat-exchanger pressure	35 kg/cm^2
Primary steam production	704 tons/hour
Secondary steam production	217 tons/hour
Steam content at core outlet	8.4% wt.
Water mass flow rate in the core	8320 tons/hour
Feed water heating temperature	190°C
Condenser pressure	3.2 cm Hg abs.

2.2. Operating principle of the plant (see Fig. 2.1.)

In dual cycle boiling-water reactors, the primary steam is generated in the core and passes in the form of a steam/water mixture to the steam drum where the steam is separated drom the water before being admitted to the turbine. The steam is required to have a minimum quality of 99.9%. The water from the steam drum is sent by recirculation pumps into the tubes of the secondary steam generators and from there back to the reactor. The steam produced in the secondary steam generators is also admitted to the high pressure part of the turbine but at a lower pressure stage than the primary steam.

Both primary and secondary steam pass the same turbine and go to the main condenser; extraction pumps send the condensate to demineralizers; subsequently the feed pumps pass it to the pre-heaters. The feed water is returned to the primary steam drum and to the secondary steam generators.

In the dual cycle the reactor pressure is maintained at a constant level by the primary control valves. The turbine regulator enables the power to be adjusted within a 30% margin without moving the control rods. In the event of a pressure increase at the turbo-alternator, more secondary steam is admitted to the turbine by the secondary control valves; this in turn reduces the reactor inlet temperature of the water; the increased subcooling effect reduces the quantity of steam bubbles in the core and the reactor power rises to the required level. In the event of a pressure drop the reverse procedure occurs.

2.3. <u>General lay-out</u> (see Figs. 2.2. and 2.3.)

Figs. 2.2. and 2.3. show the general lay-out of the principal components in the spherical containment shell. The main circuits are composed of two circulation loops with two secondary steam generators and two recirculation pumps; the primary steam drum and interconnecting piping can be seen in these figures. The control rods are inserted by a hydraulic mechanism in the lower part of the reactor pressure vessel. Above the primary steam drum an emergency condenser is situated. Fig. 2.3. shows in addition the fuel-element unloading system, which is similar to that used in other boiling-water reactors.

2.4. Reactor Pressure Vessel (see Fig. 2.4.)

The reactor pressure vessel consists of a vertical cylinder with a hemispherical bottom and a flange at the top which is designed to receive a bolted-on head. The vessel is provided with various nozzles of which the principal ones are those which serve for the inlet and outlet of the primary liquid.

There are only two inlets, each having a diameter of 610 mm, an annular diffuser is located at these inlets. Eight tubes, each 406 mm in diameter, are provided for the outlet of the steam/water mixture. The nozzles pass clean through the pressure vessel, to which they are welded both internally and externally.

Longitudinal joints are dispensed with, since forged rings are used, one of the advantages of which is that they allow greater freedom with regard to the positioning of the penetrations as compared with the conventional method of reactor vessel fabrication.

Each of the hemispheres constituting the top and bottom heads is a single forging.

The pressure vessel and structural elements are in carbon steel with a weld-deposited stainless-steel lining 6.3 mm thick. The total wall thickness is 12.7 cm, the inside diameter 3.6 m, the inside height 11.3 m and the weight of the vessel together with its head is 190 tons.

The closure of the top head is conventional; the faces of the flange under the head are clad with stainless-steel except at the hollow O-ring seats, which are of Inconel X. There is no welded joint.

2.5. Primary circuit

Most parts of the primary circuit in contact with the primary liquid are either entirely of stainless-steel or are stainless-steel clad.

The primary circuit tubing is in stainless-steel 304; the straight tubes are of the seamless drawn type and the narrow-radius elbows are forged and welded along the two generatrices.

The reactor pressure vessel and the steam drum are stainless-steel clad. The tube-bundles and the lining of the secondary steam generator primary water shells are of Monel.

The steam drum receives a steam/water mixture containing approximately 8% by weight of steam and has to supply steam with a quality of at least 99.9%; for this purpose, it is equipped with a system of cyclone separators and steam driers. It has a diameter of 2.13 m, an overall length of 19.96 m and a thickness, excluding the cladding, of 77.8 mm.

The two secondary steam generators are similar in design to the vertical main heat-exchangers of pressurized-water plants; they comprise a heat-exchanger section and a steam-separator section. Their overall height is 7.14 m.

The two primary circulation pumps, of 625 kW each, are of the submerged-impeller type; they have a rated capacity of 5920 m^3/hr and a maximum delivery head of 35.6 m. The bearings are in graphite and the shaft is stellited at the bearings.

2.6. Reactor core and fuel elements

The core consists of 208 fuel elements arranged in a square lattice, they are supported by a bottom grid and aligned by means of a guide grid at the top; they are not locked in position and rest on cups attached to the bottom grid.

The core is enclosed in a cylinder 2.75 m high and with an equivalent diameter of 2.91 m. The bottom grid carries the fuel-element load and transmits it via the lower ring of the thermal shield to supports attached to the pressure vessel.

The control-rod guide tubes are arranged under the bottom grid, into which they are crimped.

The thermal shield, 2.54 cm thick, which surrounds the core at a distance of 5 cm from the pressure vessel, serves to support the bottom grid and the top guide lattice as well as to attenuate the neutron flux to which the pressure vessel is exposed.

The control rods are guided within the core by the fuel-element channels.

The fuel elements, the effective length of which is 2.75 m, are made up of 9 x 9 cylindrical rods with an outside diameter of 1.35 cm arranged in a square pitch with sides of 1.73 cm and interconnected by 3 intermediate grids and two end-plates to form a square cluster enclosed in an interchangeable channel, the inside dimensions of which are 15.7 x 15.7 cm. The channel is constructed in stainless-steel 1.52 mm thick or in Zircaloy-2 with a thickness of 2.03 mm. The total weight of a single fuel element is approximately 340 kg.

The fuel consists of uranium-oxide pellets of 98% t.d. with a diameter of 1.19 cm and a height of 2.54 cm. The pellets are contained in Zircaloy-2 tubes 0,762 mm thick, which are filled with helium; at the top and bottom of the rods a space is provided for the fission gases released during irradiation. The uranium enrichment of the standard rods is 2.1% by weight; it should be noted, however, that in order to reduce the flux peaks at the corners of the clusters, the 12 corner-rods have only 1.6% enrichment; similarly, the flux peaks at the rod junctions are reduced by the use of pellets containing 2% by weight of Erbium oxide.

Correct location of the pellets is ensured by Inconel-X springs inside the fuel rods. In the same way, springs at the top and on the outside of the rods allow for the thermal expansion differential. Of the 81 fuel rods in the cluster, 20 are attached to the end-pieces the remainder being freely mounted. Of the above-mentioned 20 rods, 8 are composed of 4 sections each to enable the intermediate grids to be fixed. All the other rods consist of 2 sections.

As a result of the experience gained on the Dresden plant with this type of fuel element, it should prove possible to exceed the burn-up of 12,000 MWd/t stipulated in the contract.

2.7. Control

The reactivity is controlled by means of 89 control rods which are uniformly distributed throughout the core in a 25.4 cm square pitch. Movement of the control rods is effected from the bottom of the reactor vessel by means of a hydraulic mechanism, the total travel of the rods being 2.70 m. The rods, which are cruciform, have a span of 21.8 cm and an effective length of 2.70 m. The active part of a control rod comprised 80 stainless-steel tubes with a diameter of 0.61 cm at the extremety, which are packed with a B_4C powder compacted to 70% of the maximum possible density. The expected average life of a control rod is 8 to 10 years.

The entirely hydraulic control-rod drive mechanisms are enclosed in tubes which form extensions of the reactor vessel an advantage of this arrangement is that it dispenses with sealed penetrations through the reactor vessel. Under normal conditions the insertion and withdrawal speed of the control rods is 7.62 cm/sec; it should be noted that the rods are held in position by teeth located at intervals of 7.62 cm. In the case of a scram the insertion time of the rods for 90% of the total travel must be less than 3.5 seconds, including the response time of the drive mechanism.

Each control rod is connected to a hydraulic accumulator which is maintained at a constant pressure of 100 kg/cm^2; its purpose is to impart a thrust to the control rod in the event of a scram.

The overall efficiency of the control rods in the cold state is 18%. During the initial operating period of the first core, 164 of the 208 fuel elements will be equipped with a stainless-steel channel in order to prevent excessive core reactivity. After about one year's operation, 50% of these channels will be replaced by Zircaloy channels; the other 50% will be replaced after approximately 2 years' operation.

In conjunction with actuation of the control rods by the operator in the control room, there are 80 fission chambers permanently arranged in 20 channels which are installed at regular intervals in the core. These fission chambers determine the distribution of the neutron flux throughout the core and enable the control rods to be so positioned as to prevent the occurrence of high flux peaks. The fission chambers are periodically calibrated by means of activation wires which are inserted near the channels.

In addition to reactivity control by means of neutron-absorbing rods, there is a system for injecting a borate solution into the reactor water. This system is intended to offset any malfunction of the neutron-absorbing rods and to render the cold and xenon poison-free core substantially sub-critical.

The system comprises a tank containing 6,800 litres of a sodium pentaborate solution which is stored at atmospheric pressure. This solution may be injected under the reactor core by two pumps with a capacity of 189 ltr/min each. The system is actuated manually by the operator in the control room, who can start one or both pumps as required.

2.8. Turbo-alternator and condenser

The turbo-alternator is composed of the following units:

(1) a 160 MW, 1500 rpm, 3-stage turbine, the low-pressure section of which is of the double-flow type and the high-pressure section has double admission of dry saturated steam at 67 and 32.3 kg/cm^2.

(2) a 200,000 kVA, 1500 rpm, 12,000 V alternator with a power factor of 0.8 at a hydrogen coolant pressure of 2.1 kg/cm^2.

Moisture bleeds are provided on the turbine and special attention has been given to the elimination of all "pockets" in which radioactive matter could be retained.

With a view to minimizing erosion, the use of saturated steam in the turbine results in a low steam circulation speed and low energy loss in each stage.

Eight bleeds enable the steam to be sent to the condenser after passing through the expanders.

The condenser is of the single-pass type with a de-aerating well. Both the condenser and its auxiliaries are of conventional design, but it should be mentioned that the air ejector is of large capacity in order to extract the hydrogen and oxygen formed by radiolytic decomposition of the water in the reactor. It is provided with horizontal and vertical baffles which ensure that the particles are retained for at least 3 minutes between entering the well and passing to the extraction pumps. (The aim of this is to reduce the activity of substances with a short half-life).

2.9. Loading and unloading of the fuel elements (see Fig. 2.3.)

The fuel-element loading and unloading system is similar to that employed in swimming-pool reactors. With this system the water serves as a protective material, so that when the reactor is shut down the irradiated fuel elements can be handled with very simple tools.

The principal parts of the system, which are totally enclosed in the containment shell, are as follows:

(a) A separate storage basin located near the reactor, which can hold 250 fuel elements. The basin also has a space for the fuel-element transport container, a space for the control rods and a machine for extracting and replacing the fuel-element channels.

(b) A conduit links the pool with the reactor vessel. This conduit is filled with water during the unloading of fuel elements; there is at least 3 m of water above the irradiated elements when these are being transported.

An extension of the conduit receives the reactor vessel head and certain internals which are removed during unloading.

(c) A reactor cooling circuit removes the residual heat 2-3 hours after shut-down and allows the primary temperature to decrease to less than 100°C in 8 hours. This circuit comprises 2 pumps and 2 heat-exchangers, together with the piping and valves which connect them to the primary loop.

(d) Another cooling circuit, serving the basin, enables 20% of the core to be cooled two days after shut-down. This circuit consists of 1 pump, 1 heat-exchanger and the piping and valves linking them with the pool. If it is necessary to unload the entire core into the pool, the circuit described under (c) above can be connected to the pool.

2.10. Purification of the circuit water and chemical processing of radioactive waste

There are three water demineralizer systems in the plant, viz:

(a) A demineralized water make-up system which supplies the circuits and offsets the unavoidable losses occurring in the course of plant operation. The large production capacity of 470,000 litres/day is necessary during start-up, but in normal operation a daily consumption of only 38,000 litres is envisaged.

(b) A condensate demineralization system intended to remove the erosion and corrosion products from the water in circulation and the dissolved or suspended products entrained by the feed water. Owing to the need to reduce the amount of these products to an extremely low level, all the feed water has to be passed across resin beds. The system consists of 4 resin tanks, one of which is a stand-by, located behind a protective wall. It is planned to regenerate the resin in an installation provided specially for this purpose.

(c) A reactor-water clean-up system capable of reducing the impurity content to less than 0.5 ppm. The part played by this system is important for the following reasons:

- the maintenance of good heat-transfer conditions by minimizing deposition on the surface of the fuel elements;
- maximum reduction of the level of radiation resulting from the activation of solid products in the core

This system, which constitutes a bleed from the primary circuit, enables the primary water to be treated at the rate of 52.4 tons/hr; it comprises a pump, a recuperative heat-exchanger and two demineralizer tanks. It is not intended to regenerate these resins initially.

The system for the chemical processing of radioactive waste is installed underground, outside the turbine house and containment shell. After filtration, absorption or evaporation, the radioactive effluents are separated into:

(a) highly radioactive products of small volume, which are stored;

(b) decontaminated products, which are vented to atmosphere or discharged into the water in compliance with the relevant international standards.

2.11. Safety

The safety system must be capable of protecting the plant installations and personnel by rapidly shutting down the reactor.

The system is composed of two independent channels of the "failsafe" type. These two channels have to function simultaneously to bring about a reactor scram. Wherever possible, the two channels are physically separated in order to avoid accidental scrams.

Any of the undermentioned conditions will initiate reactor shut-down and closing of the ventilation shafts:

(a) High pressure in the containment shell.
A substantial difference between the pressures prevailing inside and outside the containment shell indicates a break in the primary circuit.

(b) Low water level in the reactor vessel.

(c) High pressure in the reactor

(d) The closing of 25% of the main steam valves.

(e) Low water level in the steam chest.
It is in fact essential to maintain a water level which is adequate to ensure natural circulation.

(f) A high neutron flux.

(g) A short neutron period.

(h) Turbine trip.

(i) A pressure increase at the condenser.

(j) Failure of the electric power supply.

Conditions (a) and (b) result in the closing of the main penetrations in the containment shell and the operation of a core spray system. This circuit, comprising two independent loops, allows the injection of one million litres of water from the condensate storage tank, at a pressure higher than the design pressure of the containment shell. When the tank is empty, the pumps automatically take suction from the lower part of the containment shell, in which case the water is cooled by auxiliary heat-exchangers. The spherical containment shell, 49 m in diameter, is designed to withstand a pressure of 1.83 kg/cm^2. The permissible leakage rate must be less than 0.5% per day of the air contained in the sphere at a pressure of 2.83 kg/cm^2.

Conditions (c), (d) and (j) likewise bring an emergency cooling circuit into operation. The emergency condenser comprises a water-filled casing and two tube-bundles connected to pipes running from the steam chest. The inlet valves at each tube-bundle are permanently open and the outlet valves are normally closed, so that the tubes are full of condensate water. Opening of the outlet valves causes gravity purging of the bundles and operation by natural circulation. The steam thereby produced is evacuated from the sphere. The 95 m^3 reserve of water in the casing is sufficient for 8 hours operation without replenishment. Replenishment of the water is effected from a reserve tank with a capacity of 1,000 m^3.

Fig. 2.1. Simplified flow diagram

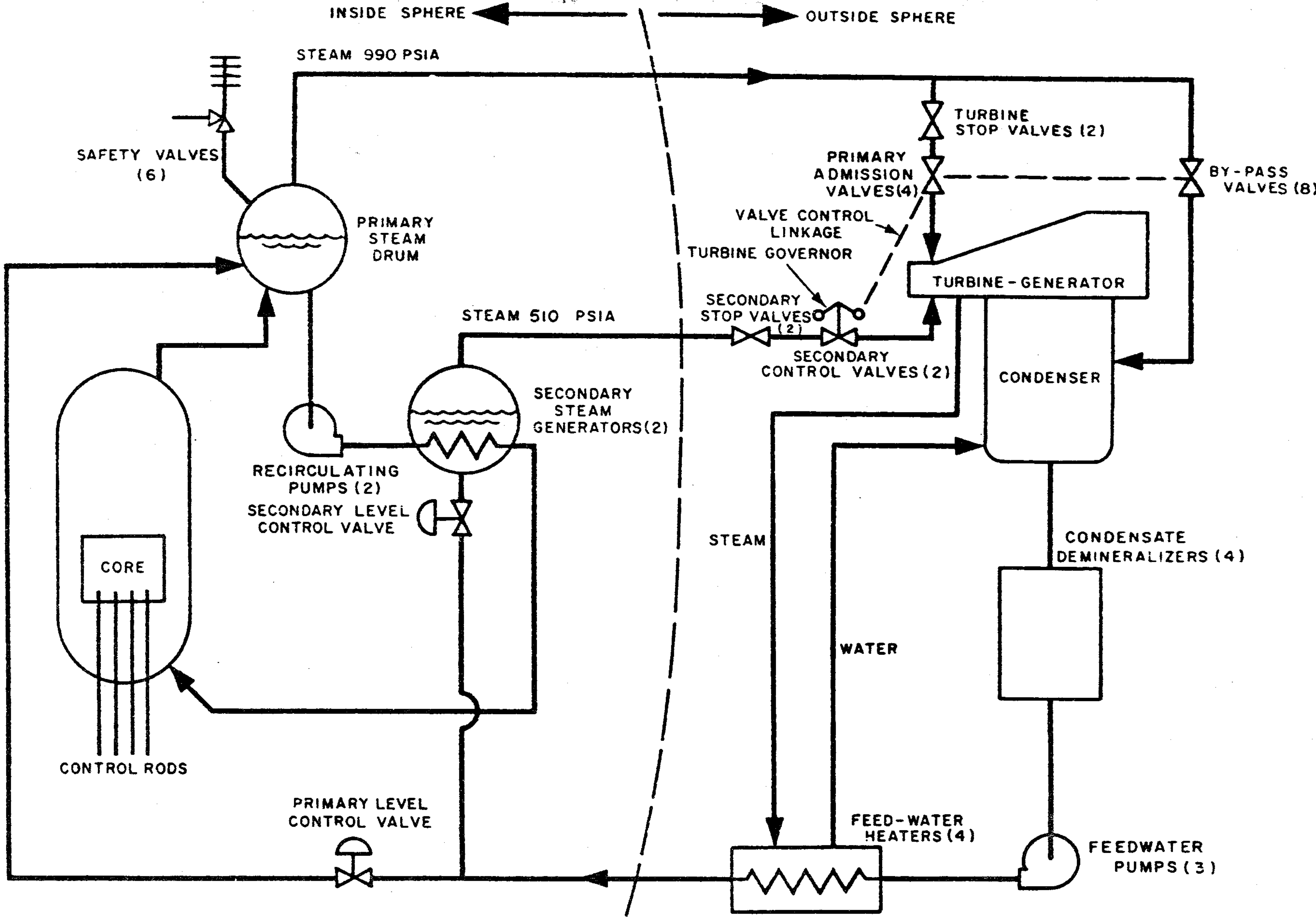


FIG. 2.2

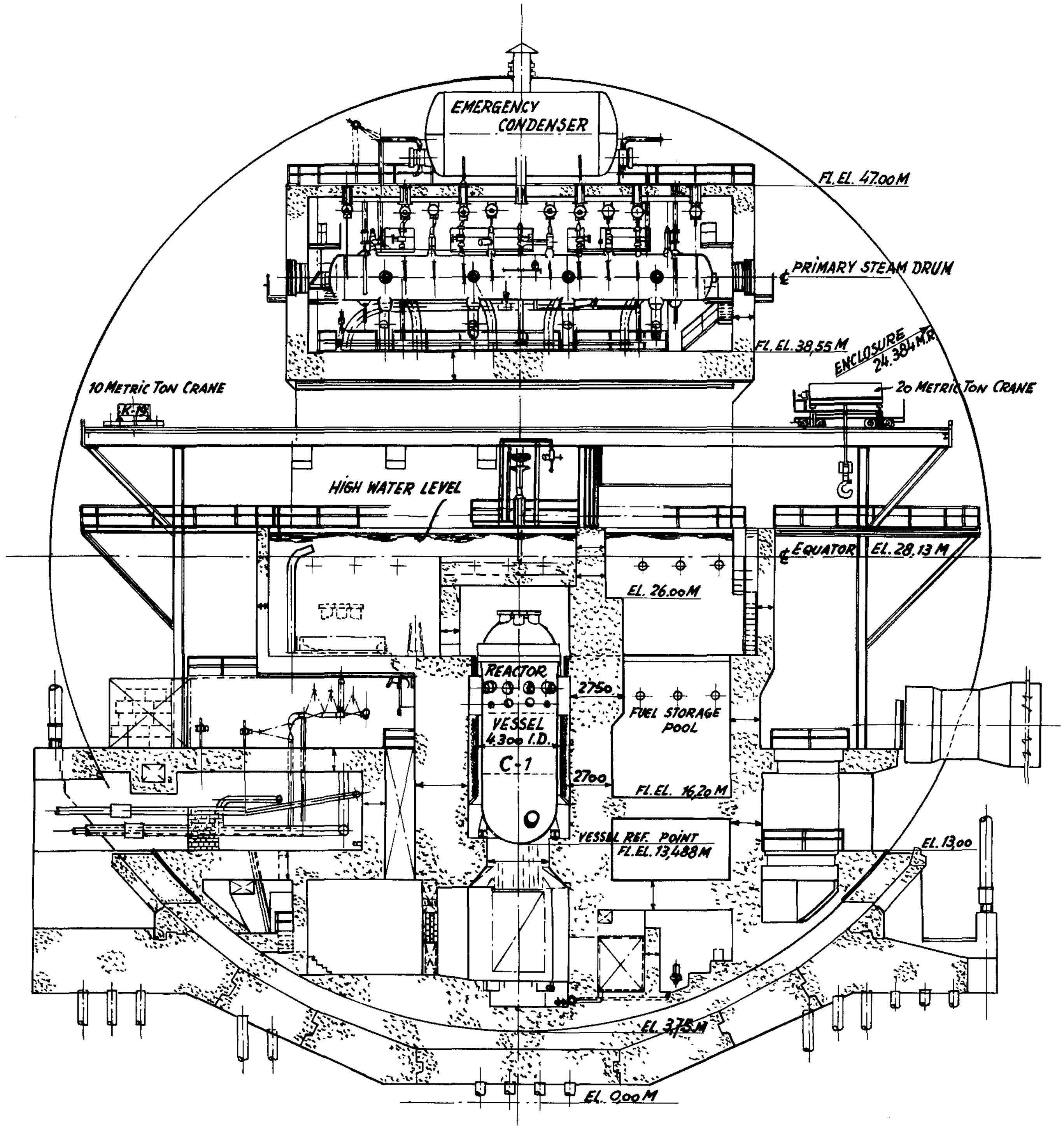


Fig. 2.2. Cross section of containment shell

FIG. 2.3

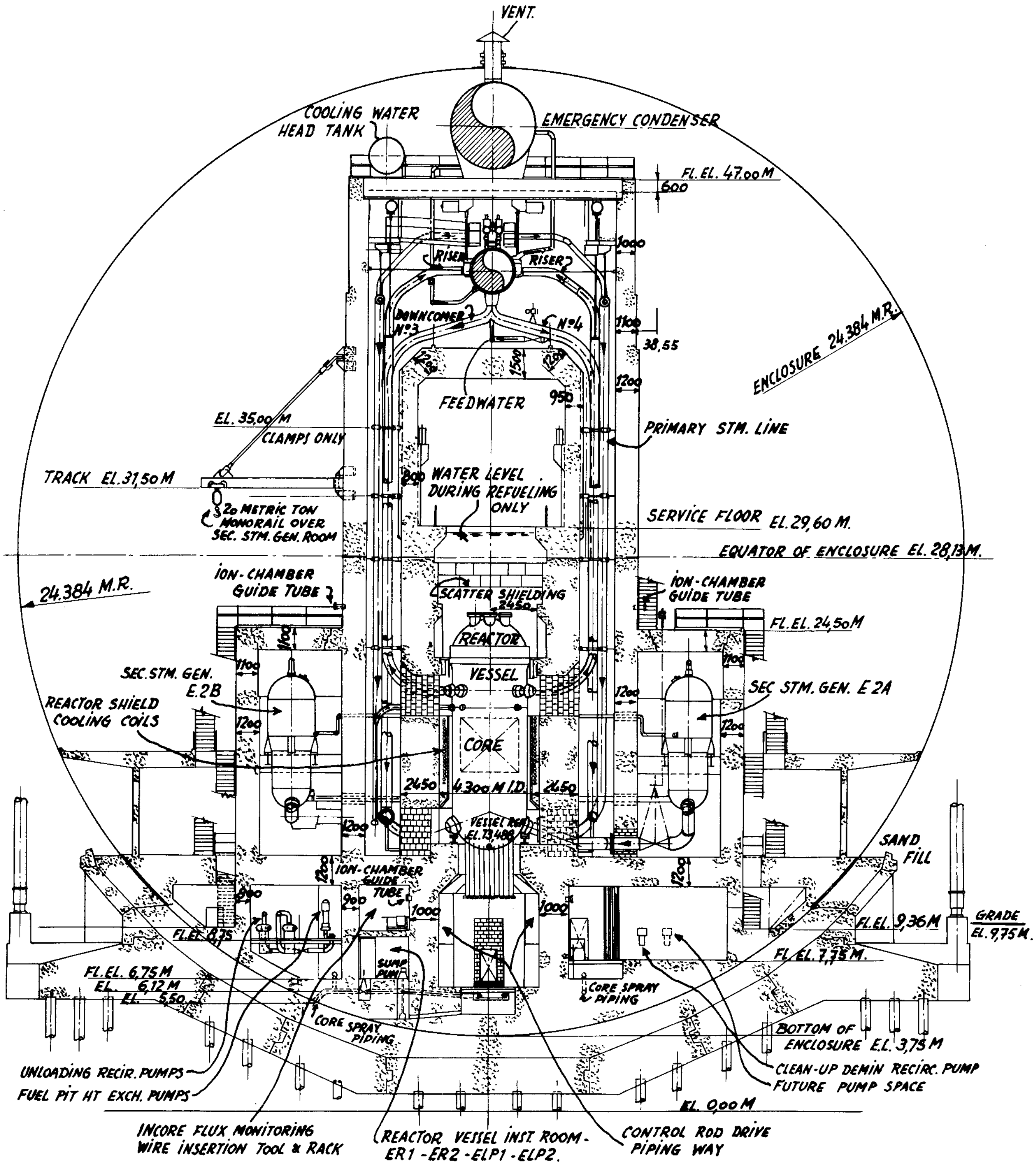
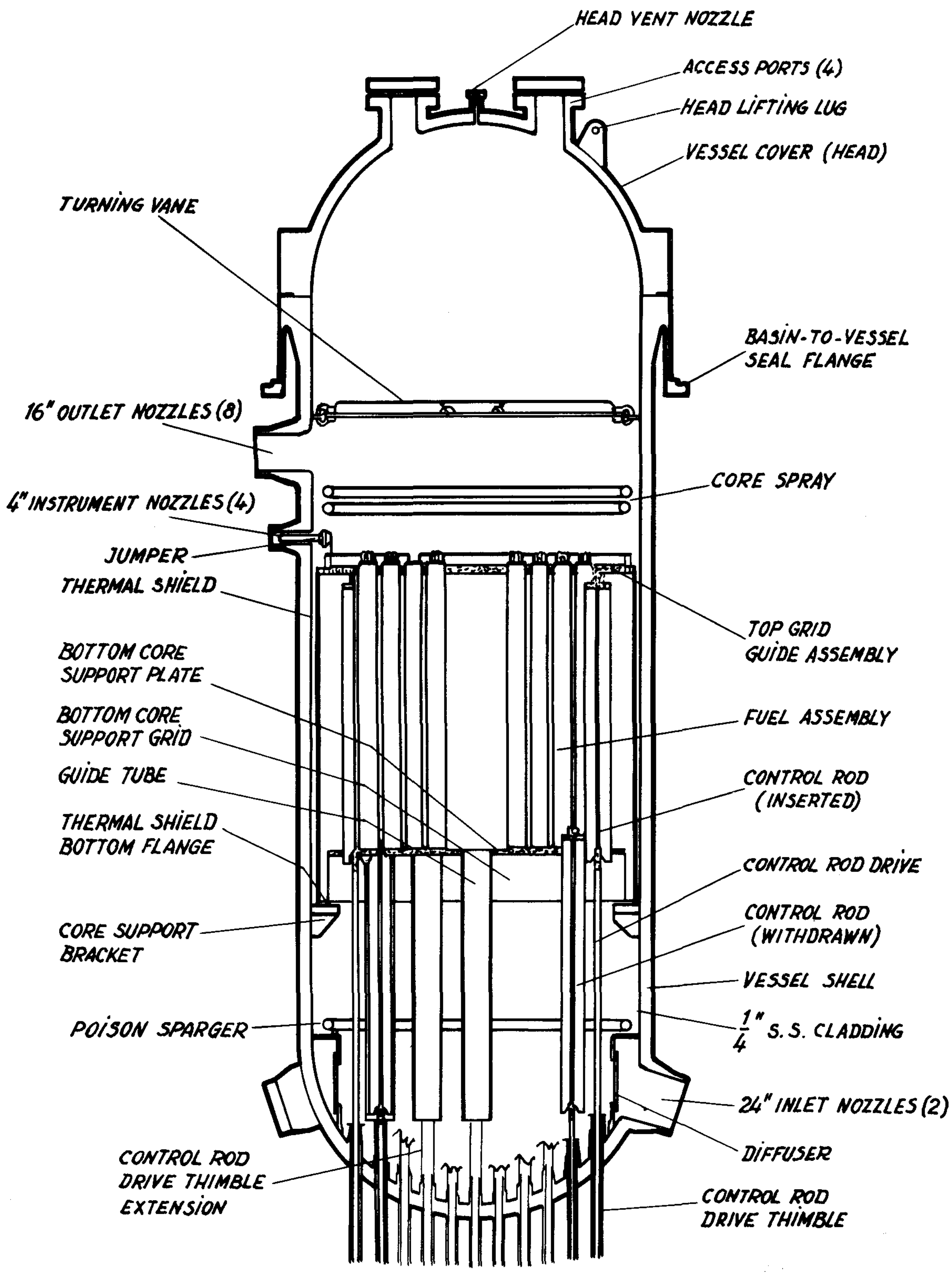


Fig. 2.3. Cross section of containment shell

FIG. 2.4



REACTOR VESSEL ARRANGEMENT

Fig. 2.4. Reactor pressure vessel

3. S.E.N.N., DEVELOPMENT AND REALIZATION OF THE GARIGLIANO NUCLEAR POWER PLANT

by

U. BELELLI and M. SIEBKER

LIST OF FIGURES

3. S.E.N.N., DEVELOPMENT AND REALIZATION OF THE GARIGLIANO NUCLEAR POWER PLANT

3.1. Introduction

By the time this article is published, the first fullsize boiling water reactor power plant in the Community will be approaching commercial operation at the Garigliano river mouth, north of Naples. It may be indicated to briefly recall the history of this project which represents, for the Community itself, an essential corner-stone of nuclear power production at low cost.

The Garigliano plant project went through the various stages peculiar to the history of nuclear energy. At the beginning of 1957, when the need of a new source of energy was being felt all over the world, the Italian power industry recognized that, as a result of the limited availability of primary sources in respect of the growing demand, Italy might relatively soon reach the point where nuclear energy could be produced at competitive costs as compared with convential sources; consequently, in order to acquire the indispensable first-hand experience, it was advisable to build and operate commercial nuclear power plants.

These were the grounds for the launching of the initiative which was to lead to the realization of the Garigliano plant, with the approval and support of the Italian Government and the World Bank.

In October of the same year, invitations to tender were issued to the most qualified manufacturers in the field, and six months later the examination of the submitted bids started. At the conclusion of the examination, backed by the opinion of an international panel of well-known experts, the choice fell on the plant tendered by International General Electric Company, based on the dual-cycle boiling water reactor concept.

It should be borne in mind that this stage was completed as scheduled, notwithstanding the spell of depression which followed the initial euphoria on the possible peaceful applications of nuclear energy after the first International Geneva Conference.

The choice of that type of reactor was based not only on economic considerations, but also on the fact that it would be possible to exploit the experience acquired by the manufacturer with the similar plant which was then being built at Dresden, USA. Actually, the Dresden plant was scheduled to enter commercial operation before the construction of the Garigliano plant would be started, and there was thus a chance for a full-scale demonstration of the excellence and safety of the boiling water reactor concept.

A contract was signed on September 9, 1959 with International General Electric Operations S.A. acting as a Prime Contractor for all the project, save the civil works, the turbogenerator-condenser system and minor auxiliaries. Construction started early in 1960.

Among the first problems to be coped with, great importance was attached to the organization of the work, observance of the construction schedules, and components detail design. With specific reference to the latter, advantage was to be taken of the experience acquired in similar plants, in order to render the Garigliano plant as up-to-date as possible. A description of the improvements over the initial proposal will be given in the following section. However, it is legitimate to state that these changes made it possible to reach the desired goal, and that from a technical standpoint the Garigliano plant can well be considered the prototype of the generation succeeding the Dresden plant.

With regard to the other problems, experience proved the necessity of setting up a sufficiently centralized organization in order to make prompt decisions on any unforeseen difficulties. Specifically, it was found essential to organize the design work so as to avoid undue losses of time among the consulting engineers, the prime contractor and the customer. This aim is best achieved by means of an appropriate contract arrangement, whereby the customer is ensured the direct benefit of the consulting services. Moreover, the observance of the schedules revealed the possibility of reducing construction times further and still maintain some flexibility in the programs. In particular, it was demonstrated that for this type of reactor, contrary to the initial assumptions, it was possible to perform almost all the civil works prior to the arrival of the machinery.

Simultaneously, going beyond the immediate motive initiating the SENN venture, great efforts were made to insert the Garigliano plant in the national grid, and on a broader plane, in the framework of economic, technical and cultural exchanges which represents a tangible result of the Community states' policy.

In fact, the Garigliano plant - Naples transmission line was erected, which is a trunk of high-voltage line scheduled to link Rome and Naples through the nuclear plants of Latina (SIMEA) and Garigliano.

In addition, the Garigliano Plant was the first project to be accepted in the EURATOM-USA Joint Program for Power Reactors and the only one involved in the implementation of the initial stage of the Program. The related contract, signed in July 1961, called for USAEC guarantees on fuel supply and reprocessing in exchange for information on plant construction and operation. The Garigliano Plant was also included in the EURATOM Participation Program for the development of power reactors in the Community countries. In short, this contract, signed in December 1961, provides for a financial coverage by EURATOM up to 3,000,000 AME units in respect of the risks of power production loss during the initial plant operation period and an additional contribution of 4,000,000 AME units in case of fabrication of fuel reloads in Europe. On their part, EURATOM, and thus the Community industries were entitled to receive first-hand information on the design, construction, testing, start-up and operation of the plant. This information includes progress reports, special reports on specific subjects, and above all, presence at the site of a Technical Group composed of EURATOM officials and engineers from the various Community organizations interested in the project.

Finally, we should not forget the particular experience acquired by the European manufacturers in the Community with which about 70% of the total commitment value was placed.
The main items fabricated in Europe were:

BELGIUM	Hanrez:	Butterfly valves for the condenser system
FRANCE	Ateliers J. : Battignolles	Emergency feed pump
	Le Creusot:	Stainless steel plates for the steam drum shell
GERMANY	Mannesmann:	Stainless steel piping
	K.S.B.:	Pumps for special loops
ITALY	Ansaldo:	Turbine (160 MW) and condenser system
	Terni:	180-ton reactor pressure vessel, shaping of containment sphere plates, emergency condenser
	Franco Tosi ATB:	Steam drum
	Termomeccanica:	Demineralizers
	Bonaldi, Tosi, FBM:	Heat exchangers and feedwater heaters (with Monel tubes)
	Dalmine:	Carbon and alloy steel piping
	Gavazzi:	Control rod drive hydraulic system, instrumentation and control panels
	Ansaldo S. Giorgio:	Generator, electric motors, main transformers and exciters
	C.G.E.:	Auxiliary transformers
	Worthington:	Pumps
	S.I.A.C.:	Containment vessel plates
	SENN (in conjunction with Prof. Morandi and Ebasco Services Inc.):	Civil works detail design
	Italstrade:	Civil works

NETHERLANDS	Stork:	Secondary steam generators
	Dikkers:	Primary system main valves
	Rotterdam Drydock:	Reactor internals

3.2. Design development of the SENN project from the offer to the present stage

The initial offer presented by International General Electric Company for the Garigliano plant was based on a design similar to that of Dresden Nuclear Power Plant. Compared to the latter, however, the design presented a number of modifications mainly related to the lower installed capacity (150 MWe versus 180 MWe).
The main differences are:

- The reactor recirculation loops (and thus the secondary steam generators) are two instead of four.
- The plant control range resulting from the adoption of the dual cycle extends from 100% to 75% of rated power, whereas at Dresden it extends from 100% to 55% because of the lower primary/secondary steam ratio.
- As a result of the different features under the two above-mentioned points, the exit quality of the steam leaving the reactor vessel is higher (approximately 8%).
- The fuel assemblies are of a more advanced type and larger in size (each assembly contains 9 x 9 rods instead of 6 x 6).
- About 80% of the fuel assembly channels of the first core are made of stainless steel in order to hold down part of the initial excess reactivity. These channels will be substituted in two steps, by Zircaloy-2 channels after a certain burn-up has been reached.
- A post-incident core spray installation consisting of two independent systems is provided to improve the safety features of the plant under emergency conditions.
- The plant lay-out is different; e.g. the fuel handling and

and storage facilities are located inside the containment sphere rather than in a separate building. This permits easier performance and avoids one of the largest sphere penetrations.

In addition, the equipment air lock is independent from the turbine building.

- There is only one reactor feedwater heater line instead of two, as adopted at Dresden.
- The secondary steam generator tube bundles are made of Monel as this material gives a better performance from the standpoint of stress corrosion under transient conditions.

During the detail design stage, several major modifications were incorporated with a view to commissioning a nuclear power plant whose design was to be as up-to-date as possible with respect to the continuous and rapid evolution of this art.

The changes in the preliminary design may be divided into two categories:

1. changes realized also at Dresden to ensure safe and proper plant operation;
2. changes concerning only the Garigliano plant, and, in some cases, other later General Electric projects.

The first category is relatively well known from various specialistic publications so that only the principal items will be listed here:

- The absorbing part of the cruciform control rod now consists of B_4C powder contained in 80 stainless steel tubes per rod. The 2% boron steel initially provided as an absorber was discarded as at Dresden it proved to be susceptible to stress corrosion. This feature is particularly enhanced at higher exposure levels.
- The control rod drive design was modified considerably to increase its reliability and to preclude the possibility of rod-drive separation. This meant actual deviation from the initial design, reduction

of manufacturing tolerances and new materials specifications, specially with regard to the heat treatment of the steel used for certain parts, which had proved susceptible to stress corrosion.

- The liquid poison system was changed from a hot, pressurized and gravity-operated system to a cold, unpressurized one, in which the poisoning solution (sodium pentaborate) would be injected into the reactor by means of a pump. This modification became necessary to avoid pentaborate precipitation and leakage through the valves. In spite of a longer inspection time, this system still fully serves the purpose of adding a further negative reactivity margin and thus to maintain the reactor subcritical in the extremely improbable event of a coincident failure of several control rods.

- With regard to the fuel failure detection system, the method employing sponges to absorb noble gases from any fuel leaks was discarded even in the design stage as impractical in comparison to other detection techniques. The system initially installed at Dresden was based on the principle of analyzing samples taken from the individual fuel channels through the reactor head. However, because of inconveniences associated with reactor opening and closing operations, this system never worked properly, Therefore, in the Garigliano design the sampling lines leave the reactor laterally as not to be tampered with during head removal operations.

- The fuel pool and reactor canal have been lined with stainless steel plates instead of being painted with special coating compounds. Actually, at Dresden the adoption of these compounds had led to airborne contamination problems during cleaning of these areas; later therefore also the Dresden fuel pool has been provided with a stainless steel surface.

The second category of design changes is also based to a large extent on Dresden experience. The main modifications are as follows:

- It was found advisable to control also the peripheral region to achieve a higher operational flexibility and a better exploitation of the fuel; therefore, the number of control rods was increased from 68 to 89. In addition, the designers realized that by rotating the control rods by 45° without modifying the position of the openings already provided in the reactor vessel bottom, it was possible to achieve a better reactivity control. This meant adopting a different type of lattice (K lattice instead of D lattice); each fuel assembly could thus be surrounded by control blades and, furthermore, the blades themselves could be increased in width by approximately one third. This modification led to a number of consequences. The fuel assemblies had now to contain 9 x 9 instead of 6 x 6 rods and, in order to flatten the flux peaks at the corners of these large assemblies, special corner fuel rods (12 out of 81) with a lower enrichment (1.6% against 2.1% in the standard rods) were used.

- The amount of uranium in the core was increased by about 10% (that is from 41.4 T to 45.9 T) and the fuel rod diameter was slightly reduced (from 0.563" to 0.534") to decrease the maximum heat flux in the core under overpower (125%) conditions from 375,000 $Btu/hr.ft^2$ to the more conservative value of 315,000 $Btu/hr.ft^2$ and to reduce the maximum fuel centerline temperature, under the same operating conditions, from 4700°F to 4200°F. These modifications permit a considerable improvement of the core working conditions with regard to the probability of fuel failure.

- As far as the fuel element design is concerned, the following improvements were achieved:

  a. The standard rods now consist of only two segments instead of four. Each segment is divided in two communicating sections so that the fission gases may collect at the top and bottom of the fuel rod, that is, out of the active core region.

  b. To reduce flux peaking at the end connectors, the fuel pellets now contain erbium rather than dysprosium as an absorber. In effect, the larger capture cross-section of erbium oxide initially offers greater protection, whereas its effectiveness as a poison decreased with irradiation time much more rapidly thus causing a lower parasitic loss of reactivity.

c. The plate spacers have been replaced by wire spacers which give less flow resistance and a better neutron economy.

- As regards reactor internals, after the trouble experienced at Dresden with the materials exposed to neutron radiation and ambient conditions which may lead to stress corrosion, a ciritical examination was made of the possibility of similar damage being suffered by the stainless steels of the 400 series, which had been proposed for the Garigliano reactor. This initial examination indicated that this type of steel was not likely to create inconveniences under the reactor operating conditions and that the trouble experienced at Dresden was actually of minor importance and imputable to the local conditions. At any rate, since no actual evidence was available on this matter, it was deemed preferable to follow the criterion of maximum safety and replace said steel with an austenitic steel of the 304 series, which had been experimented in operating conditions similar to those foreseen for the Garigliano reactor. Furthermore, calculations on the actual loading conditions led the grid support thickness to be reduced in the new core design from 2 1/2" to 1".

- The better knowledge available on thermal stresses induced by gamma fluxes and on the fast neutron attenuation factor permitted the thermal shield thickness to be reduced from 2" to 1". Another new feature of this shield is that it is now a one-piece cylinder rather than several joined sections.

- The in-core neutron flux detection instrumentation was developed considerably. Detection effectiveness was improved both by increasing the total number of in-core ion chambers from 52 to 80 and by equipping each chamber with a separate indicator. The chambers are arranged four by four in twenty strings which enter the core from the reactor bottom through thimbles that are independent from the fuel assemblies. Thus the chamber strings are not involved in vessel opening operations. This represents an important improvement with respect to Dresden. The design of the fission chambers proper and related connecting cables is also considerably advanced.

- The control rod drive system was further improved as compared to the above-mentioned changes incorporated at Dresden. Separate hydraulic accumulators have been provided for each rod so that all the 89 drives are independent from each other during reactor scram. Besides, the control rod drive hydraulic system was substantially modified to achieve a finer regulation of the pressures effecting the drive movement and to render all the drives completely independent also during normal operation. Calibrated orifices were used for the hydraulic systems such as to permit a satisfactory control of the normal control rod insertion and withdrawal rates as well as of the scram insertion speeds. Control rod coupling is now achieved through a tulip-shaped spud which only requires axial movement of a rod to attach or disconnect it from the locking member in the drive. This represents a considerable advantage inasmuch as it is possible to detach the blade working from either the top or the bottom of the reactor.

  The inner cylindrical parts of the drive are now removable with great ease to allow inspection and facilitate maintenance.

  Another important item is the possibility of performing rod-drive coupling checks from the control room. Indeed the position indicator system is now a direct-reading device with an overtravel switch alarm which allows the mechanical integrity of the individual couplings to be checked from the control room. The position indicators also incorporate thermocouples to measure control rod drive temperatures.

- The reactor pressure vessel fabricated exclusively from forged pieces which results in a material quality and homogeneity not achievable with rolled and welded plates used up to now for vessels of this kind. The elimination of longitudinal weld seams has considerably increased the safety factor of this piece of equipment.

- In the control room the simultaneous joint and complete representation of all the in-core chamber measurements of neutron flux and control rod positions is a new feature extremely useful for the operation of this reactor.

- The radioactive waste disposal system has practically retained all its major features. However, the experience acquired at Dresden permitted an accurate sizing of the equipment and the adoption of new procedures for a more efficient disposal of the contained activity. Also, a few pieces of equipment were relocated to leave sufficient room for eventual future enlargement of this system.

The design development from the preliminary to the final stage which is reflected in this general summary, gives an impressive example of the extent by which certain fundamental plant features had been improved after the offer was presented and even after construction had started. This is in line with the present continuous evolution of nuclear technology and a good indication of the outstanding progress attained in this field in the past few years.

It is known that new nuclear power stations of this type, now at the design stage, will incorporate further improvements such as will reduce the initial financial investment and, consequently, the cost of generated power. However, we feel it our duty to call the reader's attention to the importance of the experience made available by the design and construction of the Garigliano plant, since only by actually coping with new problems it is possible to originate and achieve further possibilities of development. In particular, we should emphasize the experience acquired by all those who worked at the realization of this project and, even more, the vital experience which will be acquired by a first-hand knowledge of the operational problems of a commercial nuclear power plant. The chance of acquiring information on the actual direct and indirect costs incurred with this type of plant, actual technical capability and maintenance problems; the chance of perfecting the juridical relations amongst operators, manufacturers and consultants in order to improve contract arrangements and guarantees on plant supply; the training of an adequate number of technicians and the know-how acquired by the national industry constitute for any nation, in our opinion, essential and irreplaceable means towards the full affirmation of this new source of energy.

3.3. Plant construction and testing

It should be recalled that, pursuant to the contract entered into with the General Electric Company, the latter assumed the qualification of Prime Contractor for the design, construction, erection, start-up and testing of the Garigliano nuclear power plant. SENN provided for the field organization, detail design and construction of the civil works, design and construction of the switchyard, procurement of the turbogenerator-condenser system to General Electric specifications, as well as the installation of a few minor components. IGEOSA procured the services of EBASCO SERVICES Inc., New York, for the design of the various plant sections, save those strictly associated with the nuclear system which were entrusted to APED.

A. Construction schedule

The salient dates of the Garigliano nuclear power plant construction works appear from the following schedule:

November 1959	-	Beginning of works related to land settlement and access roads.
February 1960	-	Completion of the piling for the containment vessel foundation (751 piles, 0.5 meters in diameter, average length 15 meters).
July 1960	-	Completion of station building foundation piling (1138 piles, diameter 0.5 meters, average length 15 meters).
July 1960	-	Completion of containment vessel foundation bowl, whose purpose is to transmit the weight of the containment vessel and internals (50,000 tons) to the piling.
October 1960	-	Beginning of the erection of the 18 columns, guyed to each other, which support the weight of the containment vessel plates (1500 tons) up to the placement of the reinforced concrete for the internal structures. The erection of the sphere proceeds with the following sequence: equator course, lower hemisphere, upper hemisphere.

March 1961	-	Construction of the turbogenerator pedestal. The top slab of this structure, 1050 cu meters in volume, was placed in one pour which continued ininterruptedly for 36 hours.
April 1961	-	Completion of the erection of the containment vessel plates; the welds were 100% X-rayed. Subsequently, the sphere was leak tested with excellent results: actually, the leakage rate at 26 psi internal pressure over 24 hours was 0.02% of the contained air, a value which is definitely below the maximum permissible rate.
September 1961	-	Completion of the concrete pours for the station building and attached access and control building.
October 1961	-	Completion of the turbine crane erection and main condenser shell installation.
December 1961	-	Beginning of mechanical installation with the transfer into the sphere and hoisting into place of the steam drum (diameter 2.3 m; length 20 m, weight 116 tons), performed in less than a day.
February 1962	-	Installation of switchyard equipment completed.
March 1962	-	Arrival at the site and installation of the two secondary steam generators.
April 1962	-	Completion of the 100-m ventilation stack and beginning of the installation of components in the conventional loop, such as condensate extraction pumps, feedwater pumps, mechanical vacuum pump, ejectors, etc.
June 1962	-	Arrival at the site of the reactor pressure vessel and turbine components. The reactor pressure vessel, which is 12 m high, 3.8 m in diameter and weighs 190 tons, arrived with several months' delay over the schedule because of the extreme care with which all the fabrication and testing operations were performed. As a result of the extensive test program, a few minor modifications were made to ensure perfect tightness (reshaping of a main flange, replacement of the first layer of Inconel A weld in the grooves of the "O" rings with material of the same kind but of a lower hardness to reduce tendency to cracking). The delayed delivery of the pressure vessel did not affect the master construction program appreciably as other works - mainly civil works inside the sphere - were carried out in the meanwhile wherever possible without prejudice to the rapid installation of the vessel at its arrival at the site.

August 1962	-	The 89 thimbles for the control rod drives and the twenty guide tubes for the in-core chambers were mounted, aligned and welded in place. Installation of the thermal shield and lower core support plate, both fabricated in Holland, also completed.
September 1962	-	Completion of the installation of all reactor internals, inclusive of those related to the core spray system, liquid poison system and fuel rupture detection system. The primary and secondary steam piping and reactor auxiliaries piping were ready for chemical cleaning. In addition, the last pours were placed in the sphere and the large construction openings in the containment vessel were being closed.
December 1962	-	Construction of the plant was virtually complete and the pre-operational test program had already been started. The main transformer was delivered at the site.
February 1963	-	Delivery at the site of the first fuel elements. The whole core (208 elements plus 10% spares) was transported by plane from New York to the Capodichino (Naples) airport in 10 flights at one week intervals. The shipping containers, each carrying two fuel elements, were transported to the site by means of trucks suitably outfitted for this purpose. It should be pointed out that fuel transportation by air freight has actually proved to be the most practical way. Transportation costs, inclusive of all related charges, from the General Electric shops at San Jose, California to New York and from here to the site totalled approximately 171,000 dollars, that is 3.4 $/kg U.
June 1963	-	On June 5, the plant vent critical with a minimum of 8 fuel elements in Zr-2 channels.

B. Testing

At the end of 1962 the final leak test was conducted on the sphere. This test prevailingly concerned the various penetrations (access locks, and pipe, cable, and ventilation duct penetrations) which had been cut into the containment vessel plates after the initial integrity test. The welds related to these penetrations were subjected to a soap bubble test and the total leakage rate was determined. This measurement was taken according to the same procedure adopted for the initial test, that is, based on comparison between the sphere internal pressure (approx. 1.7 ata) and the pressure of a reference system. The latter was composed inside the sphere in consideration of the disuniform temperature distribution. Sphere internal temperature readings were taken at 36 different points and humidity readings were taken at 7 points. The results of this test were fully satisfactory as they permitted to establish that the magnitude of the leak rate at the internal pressure of 26 psig is by far lower than the maximum permissible daily rate (5‰).

The nuclear steam supply system preoperational test (testing of the individual components; operational tests; thermal expansion checks; pressure drop determination; instrumentation operational tests) had to be repeated because of a few mechanical inconveniences (small leaks, malfunctioning of a few valves). These minor troubles having been eliminated, the test was conducted with fully satisfactory results.

One of the most important items to be tested thoroughly before fuel loading was the control rod drive system. The test was performed in several steps. First the hydraulic circuit was checked-out and flow rates through each individual loop were adjusted without installing the drives proper, i.e. making use of a simulating device. Then the drives were installed and operated in continuous-in and continuous-out motion (normal speed) as well as in scram motion (full speed in). The latter test was performed both on the individual rods and on all 89 rods simultaneously. Finally the step by step (notch by notch) operation was tested; after a small modification on the hydraulic and electric circuitry was made (settling device), also these performance operations worked quite satisfactorily.

In addition to the performance testing the control rod drive system was checked with respect to the possibility of uncoupling of the blades from the drives both from the top of the core and from the bottom of the drives.

A special mention should be made of the turbogenerator tests. For this purpose, two temporary package boilers were installed to supply 55 tons/hr of steam at a pressure of 22 ata in order to proceed with the required tests without having to wait for the nuclear steam. This approach turned out to be extremely useful in avoiding the delays entailed by the remedial action required to overcome the inconveniences which are inevitably encountered during the line-up of the turbogenerator.

In general, it may be stated that all the preoperational tests were performed with quite satisfactory results which demonstrated the perfect working order of the systems and components. In fact, fuel loading started according to the contract schedule and it is quite definite that the plant will start commercial operation by the contract date of October 31, 1963.

3.4. Personnel at the plant

Fig. 2 shows the number of employees of the various firms who took part in the construction, assembly and testing stages of the Garigliano Power Plant. It also includes the General Electric and Ebasco technicians who assisted SENN with the site organization. In this connection it should be mentioned that the personnel who were to operate the plant took up their duties when the mechanical and electrical assembly work began.

Euratom observers were constantly present from April 1962 onwards. The Euratom Technical Group at the plant was officially set up, in line with the terms of the Contract of Participation, in August 1962. This Group consists of three Euratom experts seconded for the duration of the contract and technicians from firms of the European Community interested in this undertaking. The number of technicians seconded to the Euratom Group testifies to the wide interest taken in the problems of erecting, testing and starting up the Garigliano plant. By 30 June 1963 1 Belgian, 3 Dutch, 7 French, 5 Germans and 6 Italians had been sent to the plant for periods ranging from a fortnight to 18 months. The purpose of their stay was to gather information on the designing and construction of various particular plant components, to carry out certain stages of the tests and to acquire a general knowledge of matters relating to the design and operation of a nuclear power plant.

In addition, under a clause in the Contract of Participation, ten students from Community countries had the opportunity of taking a short course at the Garigliano plant.

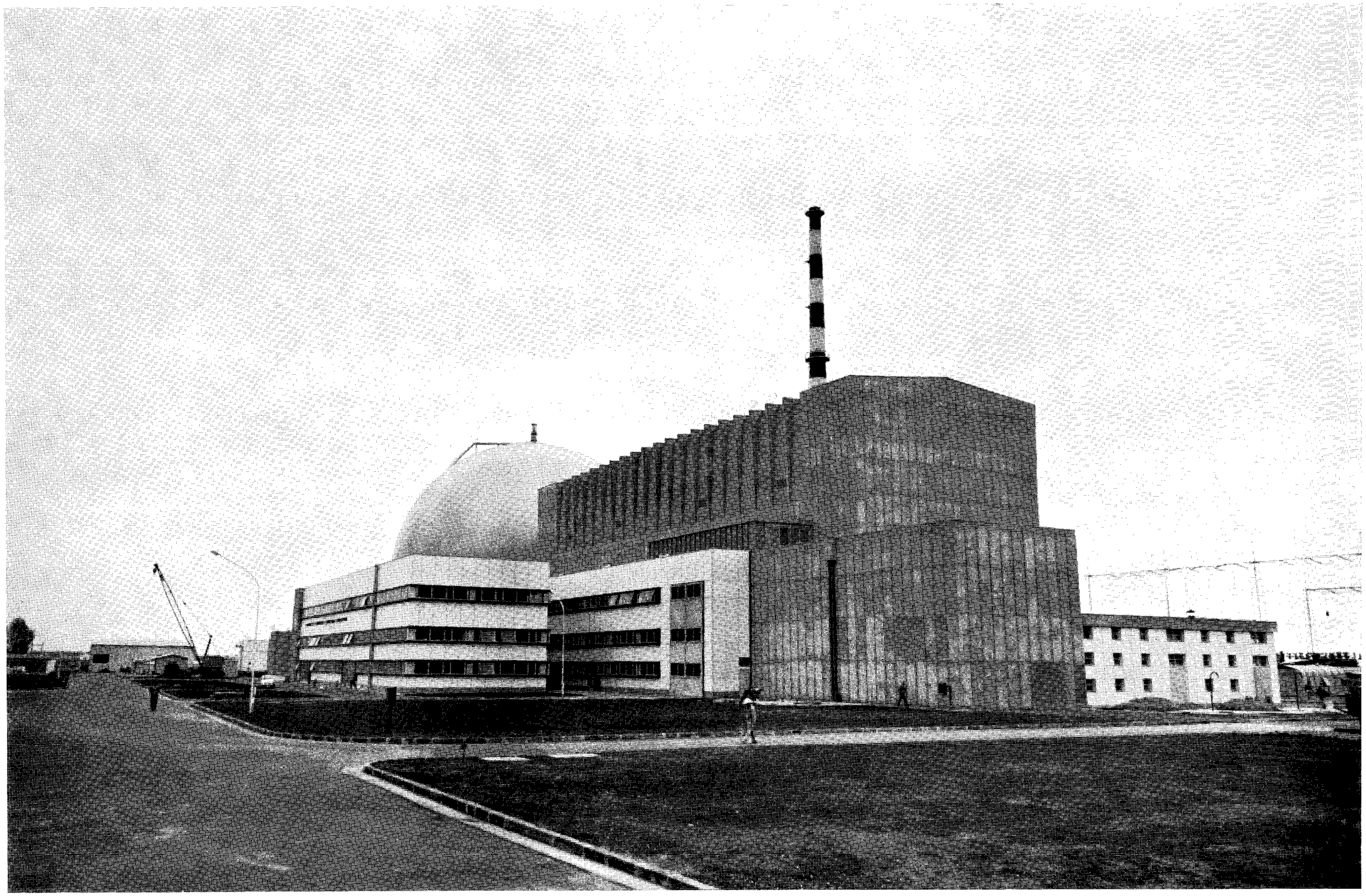


Fig. 3.1

The power station.

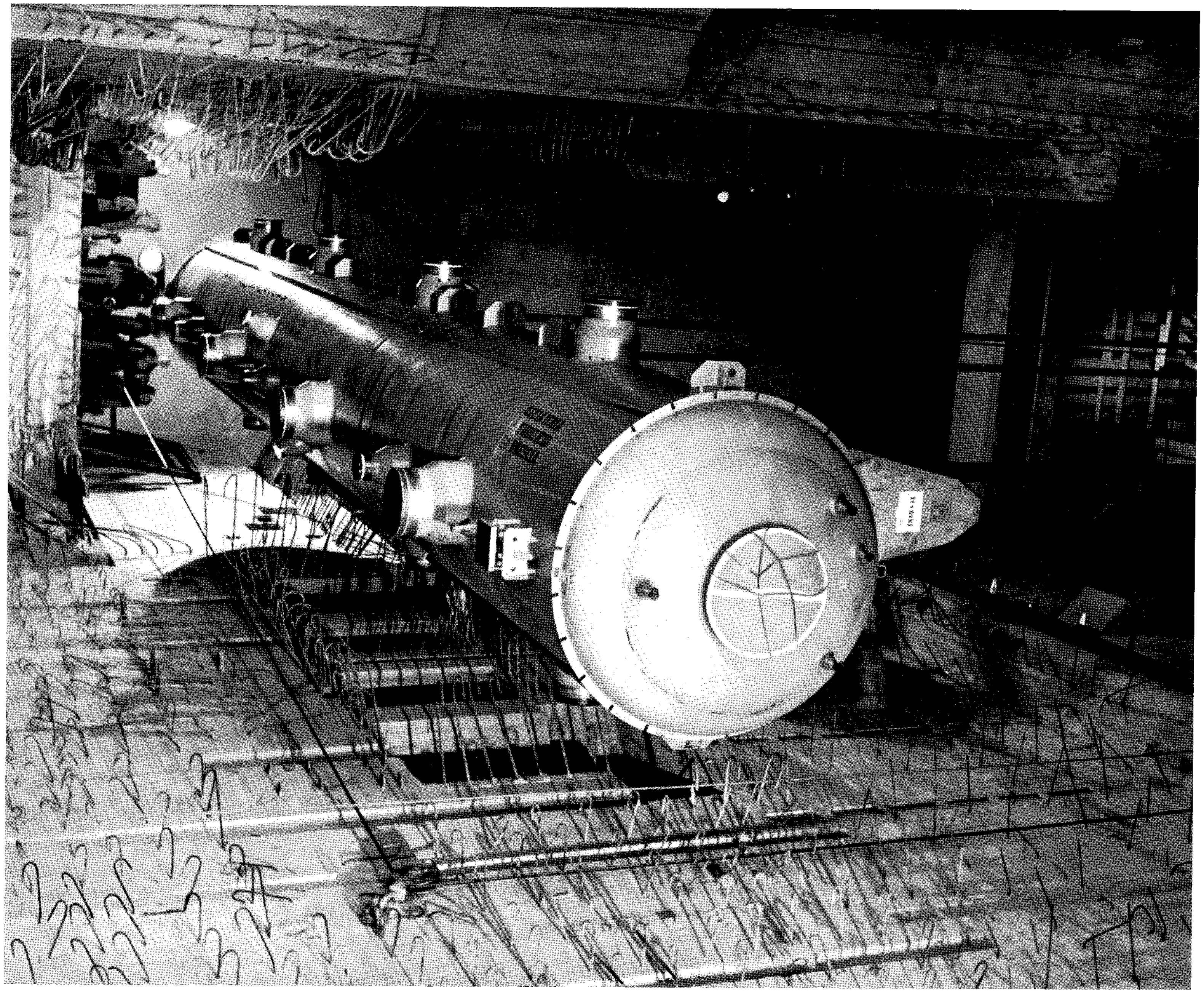


Fig. 3.2

Hoisting of the steam drum.

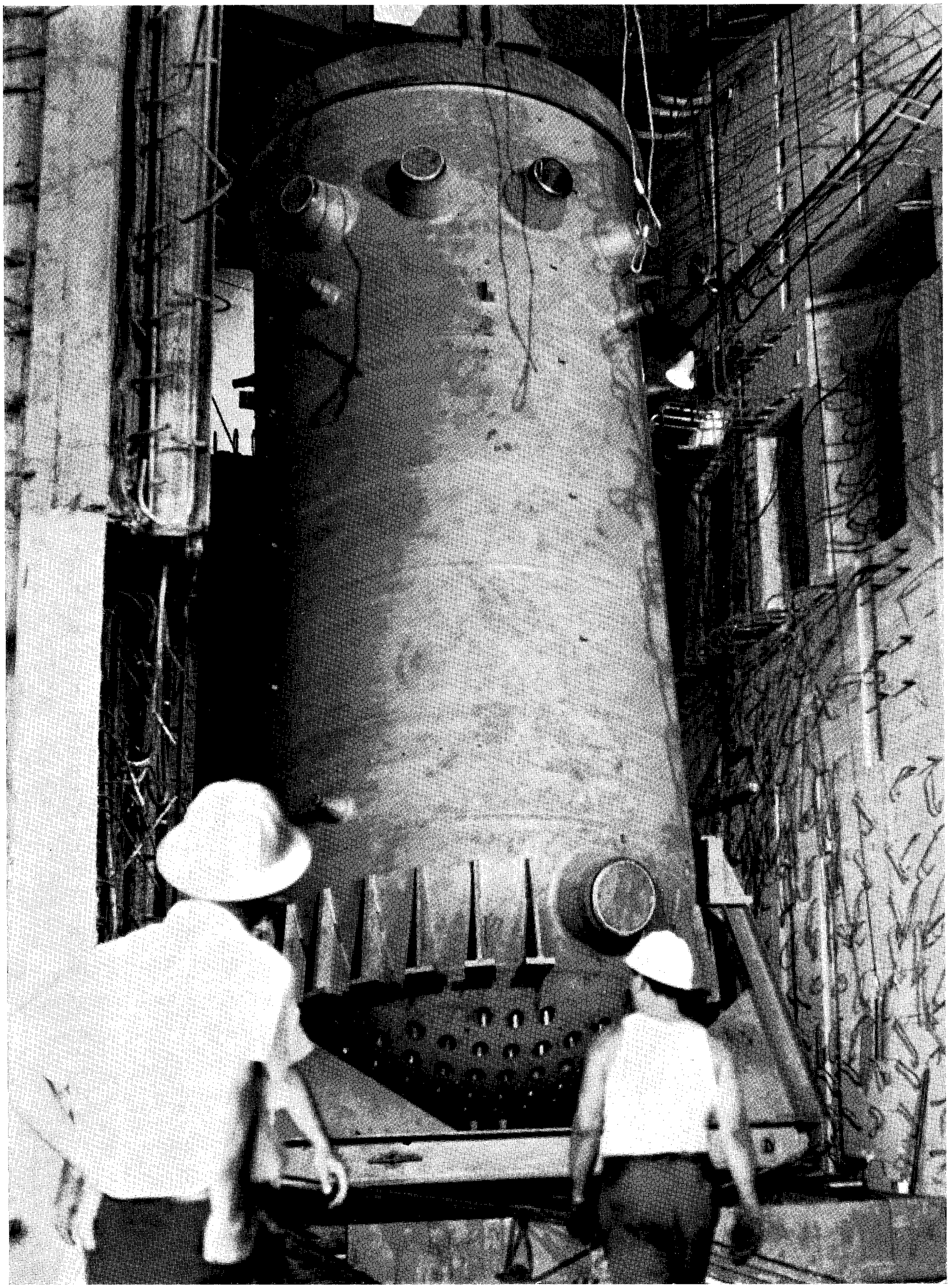


Fig. 3.3

Erection of reactor vessel.

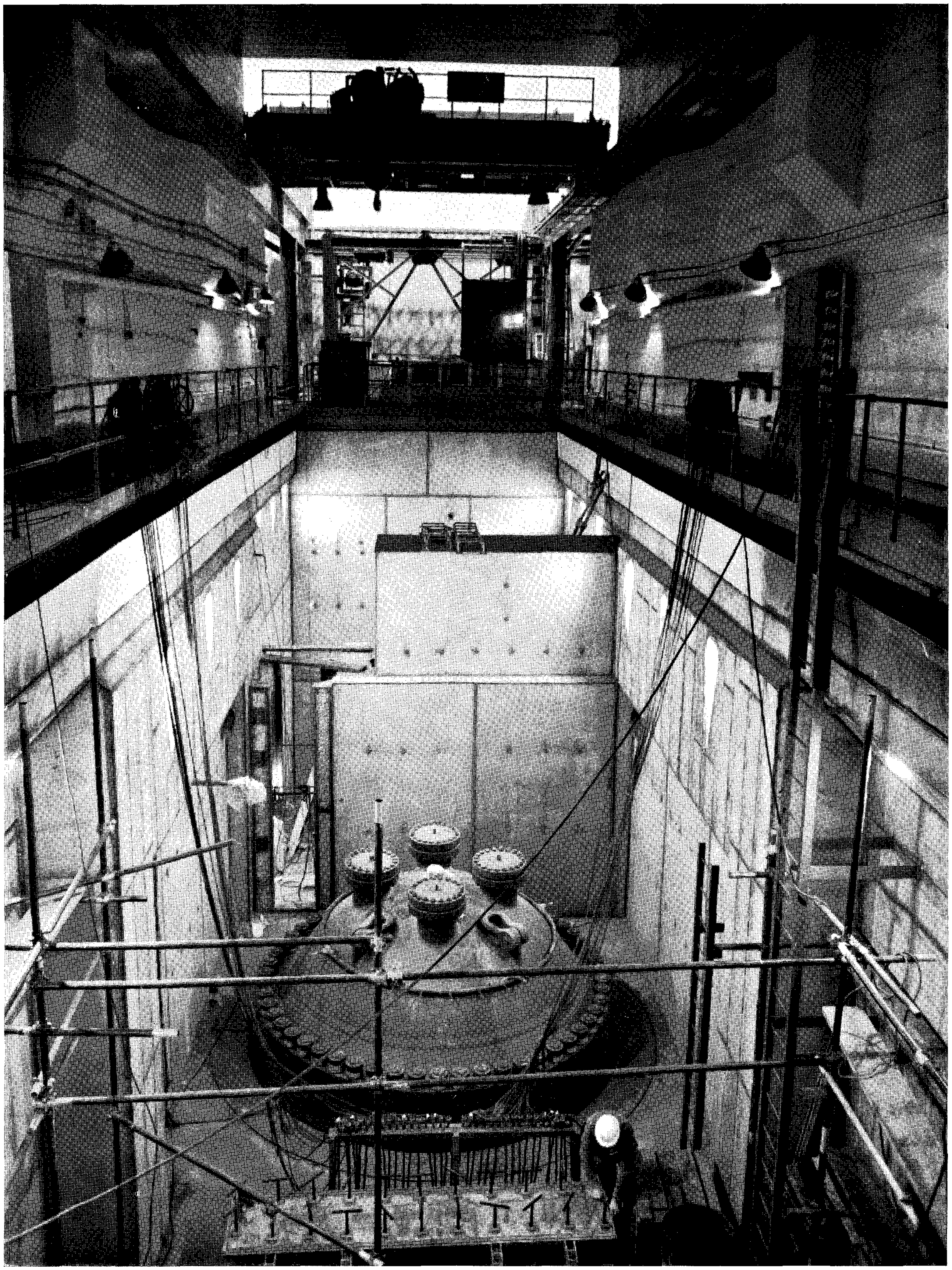


Fig. 3.4

Reactor cover.

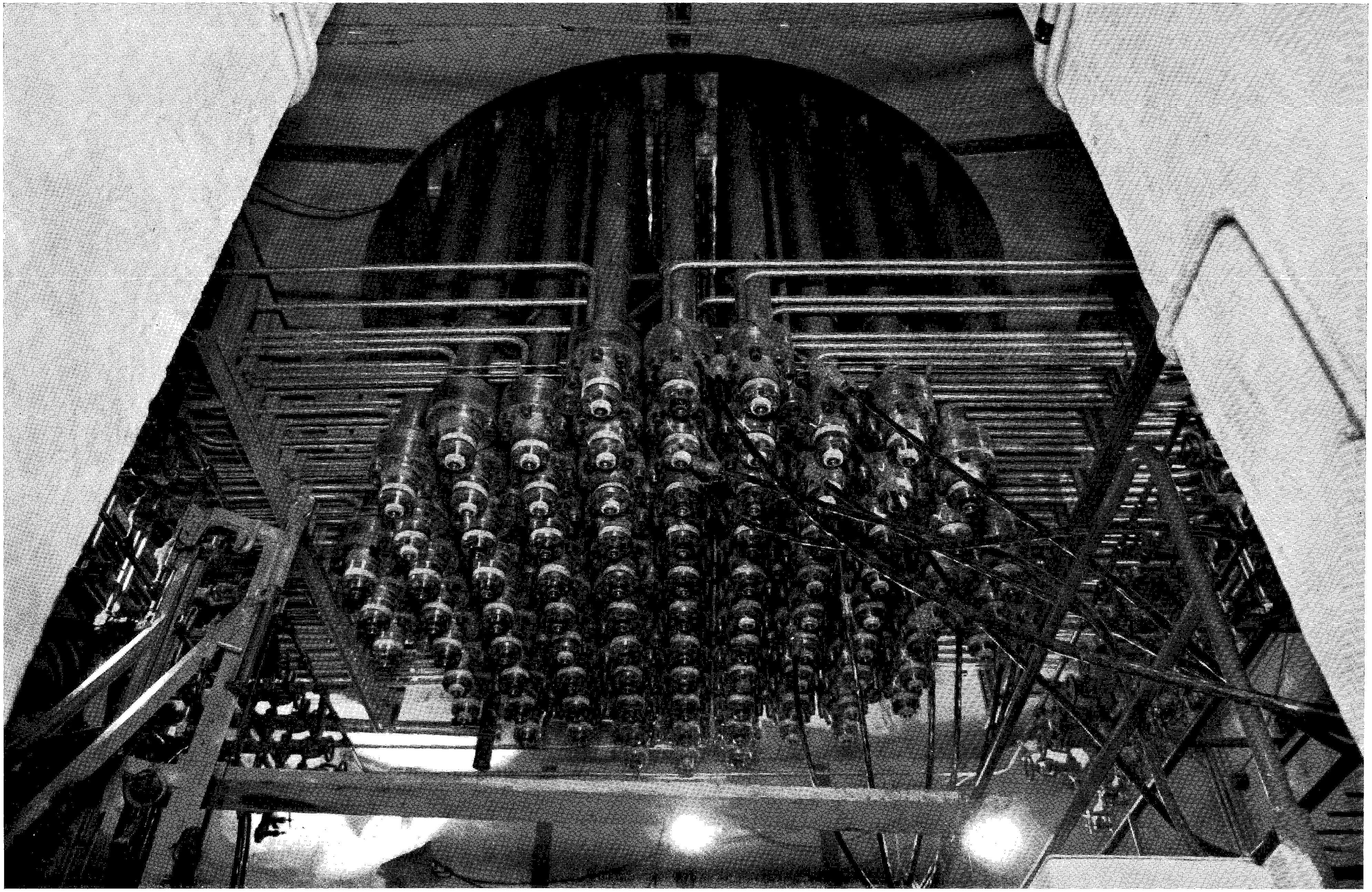


Fig. 3.5

Control rod drives.

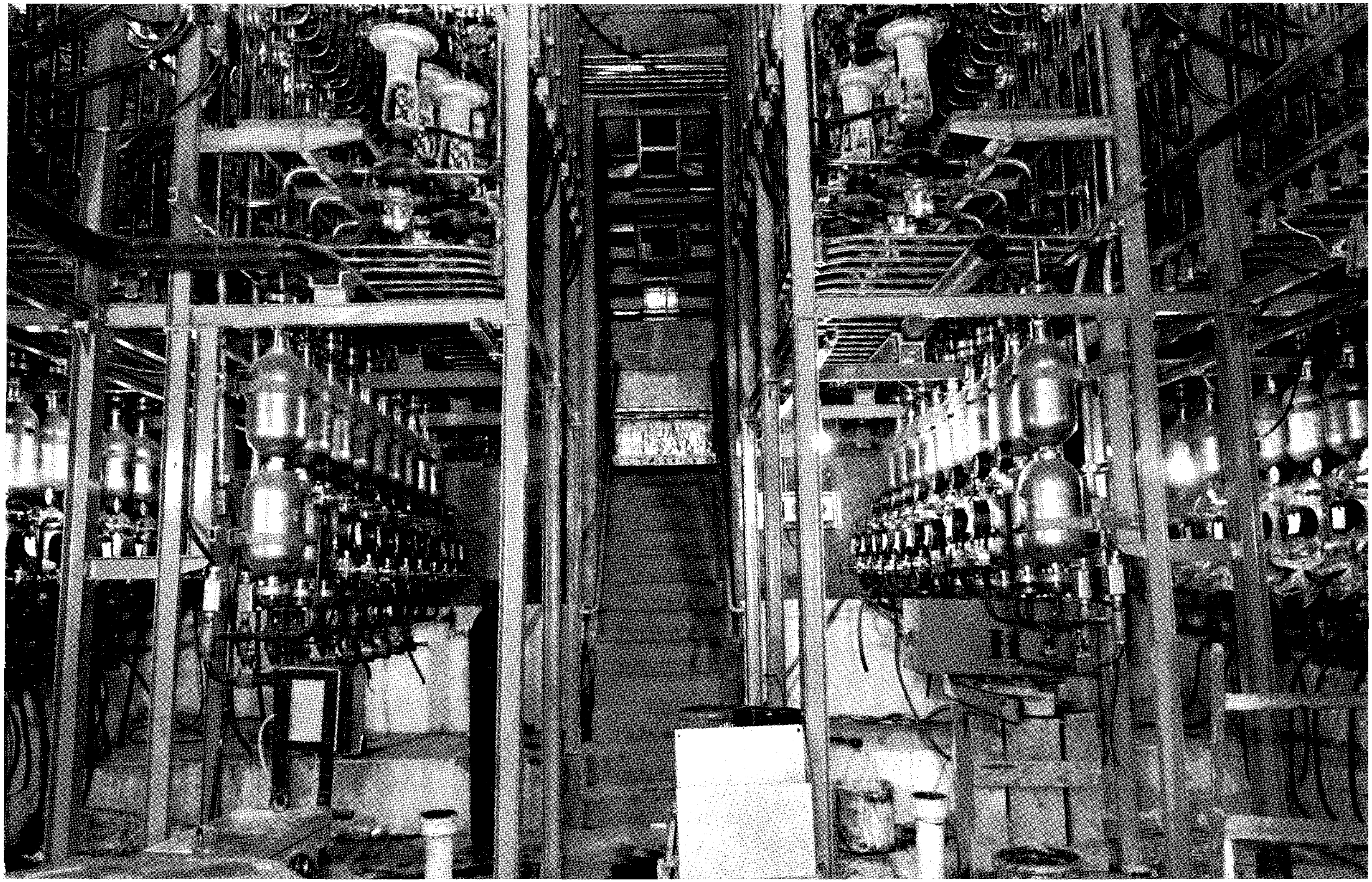


Fig. 3.6

Hydraulic system of control rod drives.

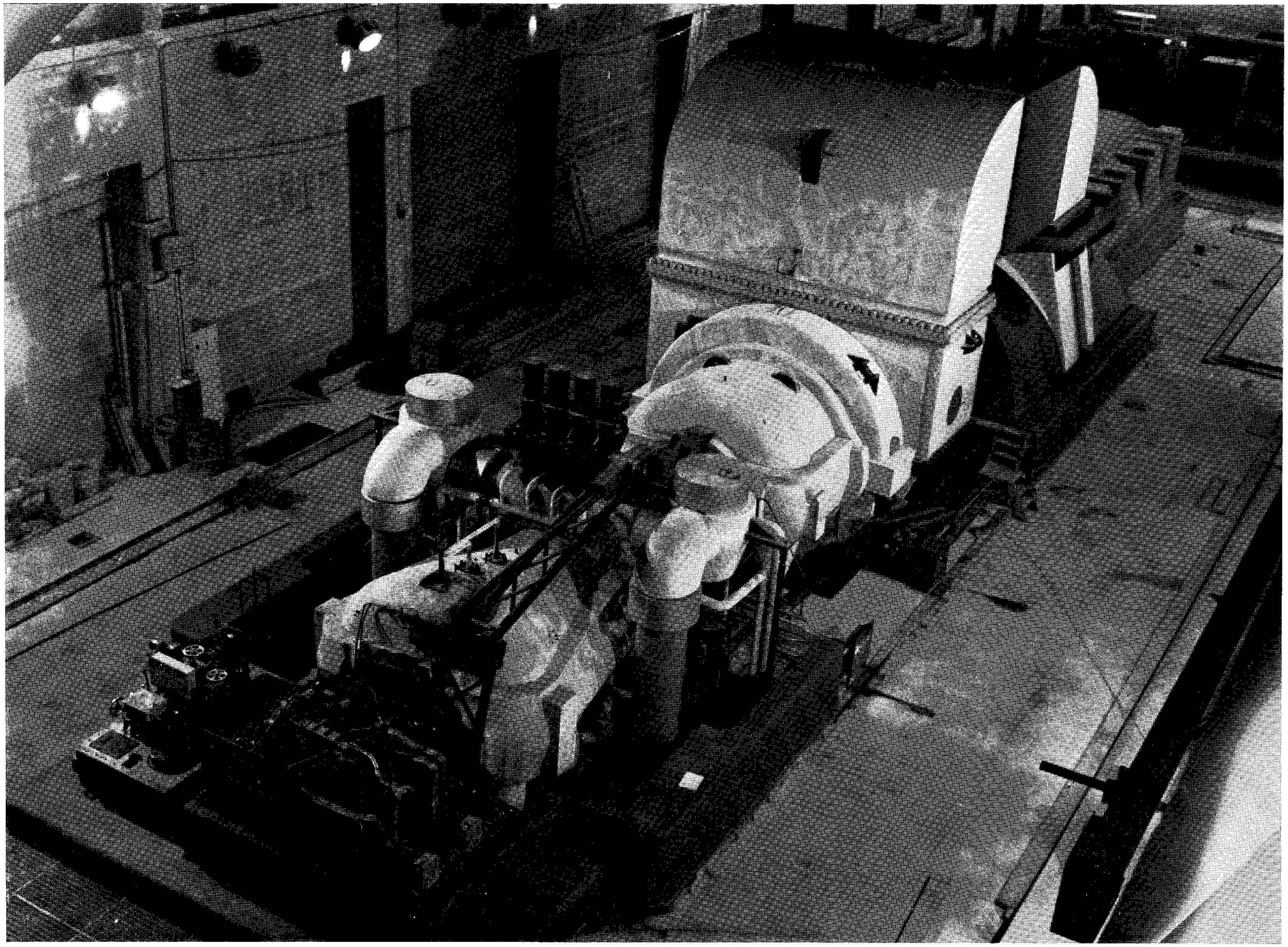


Fig. 3.7

Turbo alternator.

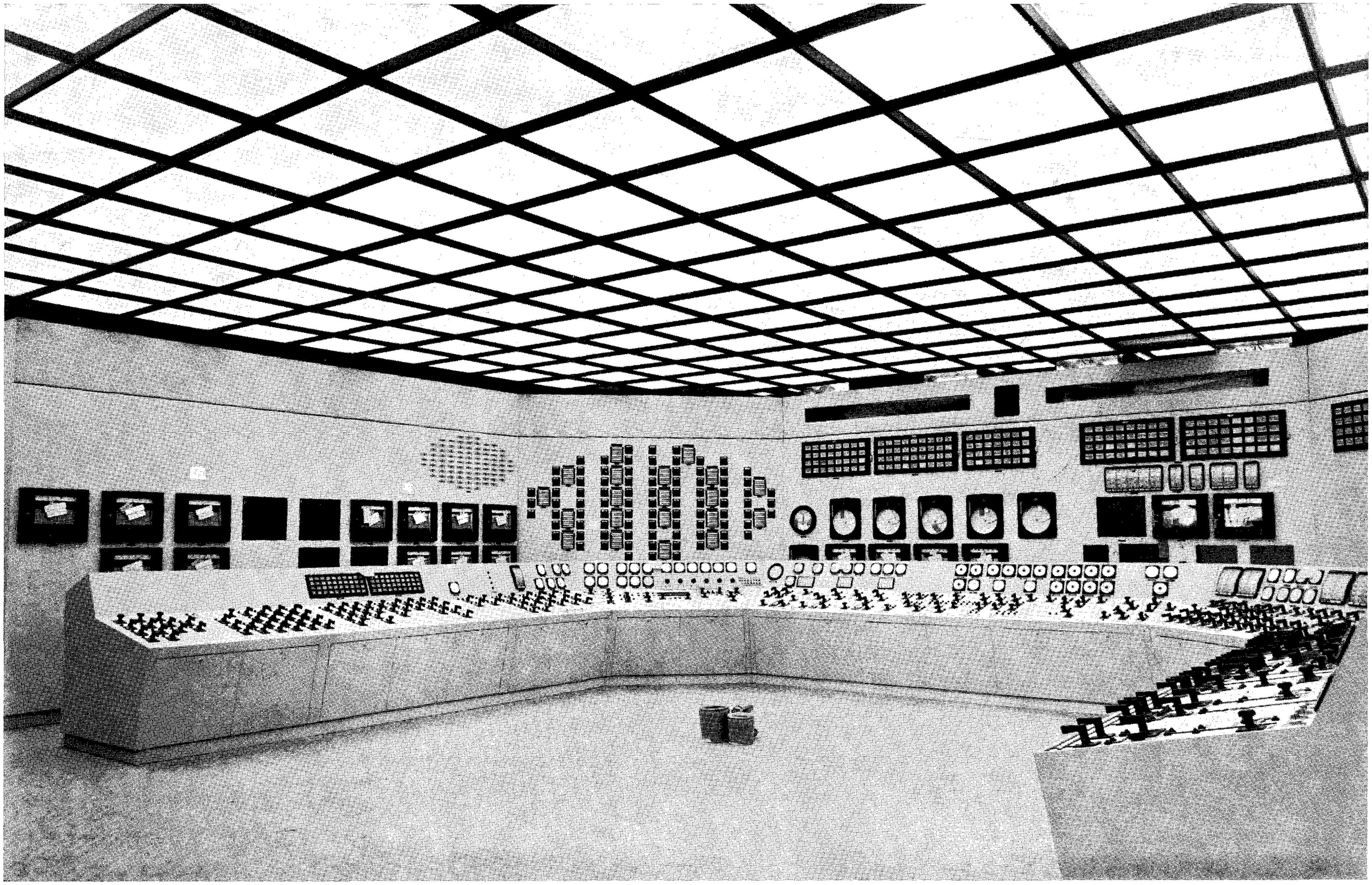


Fig. 3.8

Control room.

4. S.I.M.E.A., THE LATINA NUCLEAR POWER PLANT

by

P. MONTOIS

LIST OF FIGURES

4. S.I.M.E.A., THE LATINA NUCLEAR POWER PLANT

4.1. General features

The nuclear power plant at Latina, 70 kilometers south of Rome, is the first Italian plant for the production of electricity from nuclear fission to reach operation.

Work on the plant started during the autumn of 1958 and the reactor reached criticality on 27 December 1962; the nuclear tests at very low power have been completed at the end of March 1963 and the power run-up has started in April 1963. The first turbo-generator set was put on the grid on 12 May 1963.

The design of the nuclear part of the plant and the supply of the more specifically nuclear equipment were entrusted to the British NPPC group, who in the course of the work amalgamated with the AEI/John Thompson group, as a result of which The Nuclear Power Group was born.

Whereas NPPC provides the overall guarantees on the commissioning and operating characteristics of the plant, the Italian AGIP NUCLEARE was responsible for the design of the conventional part of the plant, the adaptation of the project to local requirements, the placing of orders for equipment to be supplied by Italian manufacturers, and the general management of the work.

Operation of the plant is in the hands of SIMEA, now a member Society of the ENI group, but which will shortly be incorporated in the national enterprise ENEL under the recent Italian law on the nationalization of the electricity supply industry.

The plant, which has a net electrical capacity of 200 MW, is equipped with a natural-uranium, graphite-moderated, carbon-dioxide-cooled reactor, six heat-exchangers and three turbo-alternator sets.

The project was based on the design used for the Bradwell power plant in England, and the principal elements of the two plants are identical.

It has, however, been possible to step up the power to 200 MW, as compared with 150 MW per reactor at Bradwell, by increasing the absolute pressure of the coolant from 10.5 to 13.8 atmospheres without any appreciable change in the dimensions or the other operating parameters.

The reactor is housed in a spherical steel vessel about 20 m in diameter and with a thickness of 9 cm, except at the top, which carries the reinforcements for the standpipes, and in the region of the supporting columns, where the thickness is 11 cm.

An even temperature distribution throughout the vessel and a substantial reduction in the thermal stresses is ensured by insulation, which is applied internally in the upper part and externally on the lower part. In normal operation, the maximum and minimum temperature of the steel are 215 and 180°C respectively.

The vertical graphite stack is 14.20 m in diameter and 9.40 m high. It contains 2,929 fuel-element channels and 108 control-rod channels. It consists of vertical blocks keyed together in such a way that the dimensional variations occurring as a function of temperature and irradiation do not affect the linearity of the channels. The Wigner energy stored in the graphite could, if necessary, be released by raising the inlet temperature to 300°C during low-power operation. The blowers and circuits are designed to withstand these particular operating conditions.

The fuel elements, of which there are 8 in each channel, are rods of natural uranium clad with a magnesium alloy (Magnox). The cladding is provided with spiral fins and longitudinal "splitters", which serve both to stiffen the element and to improve the heat-exchange coefficient (polyzonal system).

The respective maximum permissible temperatures are 480°C at the cladding and 620°C at the centre of the rod.

The average burn-up envisaged is 3,000 MWd/ton.

Flux flattening is achieved by inserting steel absorbers into interstitial channels of the central zone, thus avoiding the need to reduce the number of channels in the lattice which are available for the fuel and for control purposes.

The reactor is automatically controlled by sectors. Four rods in the central sector and two rods in each of the eight peripheral sections are actuated by the temperature variations measured in the sector concerned. In this way two functions are performed: the gas temperature at the channel outlet is maintained constant and the spatial stability of the flux is ensured.

The bulk rods, which are arranged in two groups of 46 and 22 rods respectively, move as a curtain. The first group controls the negative shut-down reactivity and the temperature effect, while the second group controls the poison effect.

Fuel-element loading and unloading are effected during operation by a single machine which travels on the pile cap and is capable of performing all the necessary operations. It enables the fuel of about 7 channels to be renewed in a day. A second machine is provided as a stand-by.

The heat-exchangers have been slightly modified with respect to those at Bradwell. Although having a 33% higher heat-exchange capacity, they are slightly smaller, the improvement being obtained by the use of the finned tubes for the superheaters. The secondary cycle comprises two steam circuits, viz. a high-pressure circuit (53.2 atm.) and a low-pressure circuit (13.8 atm.).

The six axial blowers are driven by electric motors fed at variable frequency by auxiliary generators. Furthermore, emergency electric motors fed by diesel generator sets enable the blowers to continue operating in the event of a general electrical failure, in order to ensure an adequate supply of coolant to evacuate the residual heat of the reactor after a shut-down.

The condensers of the turbo-alternator units are cooled by sea water. The cooling-water intake is located 750 meters offshore so as to prevent silting-up. Two reinforced-concrete pipelines laid on the sea bed connect the intake to the shore, whence the water flows through an open conduit to the pumping station.

The automatic control system is designed to enable the plant to operate at between 20% and 100% of its maximum power.

4.2. Principal construction stages

Work on the site started in November 1958. The foundations of the reactor building were laid between February and July 1959. The "Goliath" crane covering the entire reactor building, which was to serve for the erection of the containment shell and the heat-exchangers, was erected during the same period. At the end of the summer of 1959 the first pouring of concrete took place for the reactor's biological shield, which reached the level of the pile cap in September 1960.

The first plates for the pressure vessel arrived on the site in August 1959. The assembly, welding and stress-relieving operations were completed 15 months later in December 1960.

Hydraulic testing was carried out in February 1961, external insulation between March and April 1961 and internal insulation between September and October 1961. From then on the inside of the pressure vessel was maintained under clean conditions control for both the personnel and material.

Stacking of the graphite started on 19 February 1962 and was completed at the beginning of June, the time taken being determined by the phasing of graphite deliveries and not by the actuel erection work. The summer of 1962 saw the installation of the equipment located on the upper face of the stack, viz., charge pans, thermocouples for measuring the gas temperature at the outlet of the channels, piping for the detection of cladding ruptures, etc.

Installation of the first fuel-element loading and unloading machine started on 20 May 1962 and was completed in September of the same year; that of the second machine started in February 1963 and was completed in April 1963.

Erection of the heat-exchanger shells was carried out between March and September 1960. This was followed by the internal assembly work and installation of the tube-banks, which were completed at the beginning of 1962. Installation of the main piping for the primary circuit took from March 1960 to August 1962.

Construction of the turbine house was carried out concurrently with that of the reactor building. Installation of the main turbo-alternator sets and their auxiliaries started in June 1960 and was finished in July 1962.

The construction work proper was completed on 1 November 1962 with the exception of the pool for the storage of irradiated fuel elements and the processing of radioactive waste, which was ready at the beginning of 1963, and of the finishing operations.

The first fuel-supply contract has been concluded with the UKAEA. It provides for 400 tons as the initial charge, one and a half year's consumption, and a reserve of 30 tons. Between 1 August and 15 December 1962, 300 tons were delivered to the site.

4.3. Testing and start-up

The start-up of the plant was preceded by a large number of tests and adjustments, which can be divided chronologically into the following four groupes:

4.3.1. Unit tests

As assembly work on the various items of equipment reached completion, individual tests were carried out in order to ensure that the assembly work had been performed in accordance with the specifications of the contract, to effect the necessary adjustments, and to guarantee that each piece of equipment was capable of fulfilling its function.

This phase of testing naturally commenced well before completion of the construction work. Among the principal operations in this phase, mention can be made of the separate testing of the six closed-loop cooling circuits comprising the blower, the lower tube-banks of the heat-exchangers and the heat-exchanger by-pass line; this involved in particular a dynamic test in the hot state, the circuit being filled with CO_2 at 300°C and 14 kg/cm^2 gauge.

As a result of these tests it was possible to detect two weak points and to carry out the necessary modifications before start-up of the reactor. These modifications related on the one hand to the mounting of the distributors for the blowers and on the other hand to the arrangement and support of the tube-banks located at the level of the outlet of the heat-exchanger by-pass lines.

4.3.2. Combined tests before loading of the fuel

These tests are carried out when the various units and circuits required for the functioning of the reactor have been completed and individually tested. Their purpose is to confirm, before the fuel is loaded, that the plant is capable of operating under load. They are, limited by the steam generation facilities available on the site. The combined tests constitute the transition stage between the construction work and the start-up proper; they provide the operating personnel with initial training in the operation of the plant.

These tests, which were successfully carried out between 1 November and 8 December 1962, consisted mainly in the following activities:

- leak-tightness testing of the entire circuit with compressed air at 15.5 kg/cm^2 gauge. The various sections of the circuit had previously been tested with air at 19 kg/cm^2 gauge (equivalent to one and a half times the nominal pressure). The leakage rate was measured at the nominal operating pressure.

- preliminary testing, in cold air and at the operating pressure, of the various elements: blowers, valves and auxiliaries of the primary circuit, control-rod actuation mechanisms and circuits, apparatus for flux measurement and the detection of cladding ruptures, safety circuits, and CO_2 blow-down system.
- drying of the graphite by blowing air which had been heated and dried in a provisional drying installation.
- filling of the circuit with CO_2
- a second CO_2 test programme, in the hot state and at operating pressure, on the various elements listed above plus the fuel-element loading and unloading machine
- the blow-down, vacuum test and purge with air of the primary circuit.

This test phase, which had been scheduled to take 6 weeks, was actually completed in slightly less time, thanks to the introduction of a number of simplifications as compared with the original programme.

4.3.3. Fuel loading - nuclear measurements in air at very low power

The loading of the first charge of fuel into the reactor differs considerably from the normal fuel replacement procedure by means of the operating equipment. It consists in fact in loading the reactor with 24,000 new fuel elements in the least possible time, whereas at the rate permitted by the fuel element charge-discharge machine this operation would have taken several months. The initial loading was therefore carried out chiefly by hand in the following four stages:

- Transfer of the fuel elements in cases from the store to pile cap of the reactor

- Unpacking and inspection

  The fuel elements arrive on the pile cap in their transport packaging, which consists of steel cases, each containing 20 fuel elements separately packed in a polyethylene bag and a strong paper bag and kept in place by pads of horsehair and rubber. After unpacking, a visual inspection is carried out of the welds, the surface, the fins and the end-pieces.

- Lowering of the fuel elements from the pile cap to the charge pan of the stack via three standpipes, which are equipped for this operation with electric winches and baskets, each holding 2 elements.

- Loading into the channels

  On the charge pan, teams of workers take the fuel elements from the baskets and insert them one at a time into the channels with the aid of grabs and nylon ropes.

  At each stage of the operation, the identity of the element is checked by reference to a number engraved on a splitter.

  Loading started on 24.12.62 and was completed on 14.1.63. It was done in 2 stages. The critical size was obtained after the loading of 420 channels, and a preliminary series of measurements was carried out with 432 channels loaded. This first phase of the loading took 55 hours, including the stoppages for measurements in the course of loading.

The loading of 2,497 channels to bring the small pile up to full size was then effected in less than 10 days. Altogether, 10.5 channels were loaded on average per hour, the maximum rate maintained during a 2-hour period being 22 channels per hour. The work was carried out continuously by shifts of 26 men.

The nuclear measurements at very low power were as follows:

- Measurements on the small pile (with 432 channels loaded) including flux distribution plotting by the irradiation of manganese detectors, and measurement of the reactivity-pressure coefficient. These measurements, which were reduced to the minimum, took only one day. Their primary object was the determination of the axial and radial buckling.

- The following measurements on the pile loaded to full scale

(a) Determination of the built-in reactivity of the pile, both non-flattened and when flattened with a radius equal to the flattening radius calculated for the pile in the reactivity-equilibrium condition.

The measurement on the flattened pile was repeated at three temperatures, viz. $20^{\circ}C$, $95^{\circ}C$ and $170^{\circ}C$, in order to ascertain the overall reactivity temperature coefficient of the reactor in the non-irradiated and non-poisoned state.

(b) Flux distribution charts for the two above-mentioned conditions, microscopic distribution and deformations in the vicinity of irregularities: empty channels, absorbers, control rods.

(c) Measurement and adjustment of the gas coolant flow per channel

This adjustment was carried out in cold air at atmospheric pressure, with the six blowers rotating at their nominal speed. Each channel is equipped with an adjustable gagging orifice at the level of the reactor diagrid. The adjustment of the channels in their entirety is carried out by successive approximations, the flow at the channel outlet being measured by means of anemometers. The adjustment of the total number of fuel channels was finally obtained with a standard deviation of 0.5% from the required value as calculated, the operation having taken a total of 19 days.

(d) Calibration of the control rods

The complete calibrations of the various control-rod groups were carried out in the cold state. As a result the following determinations were possible:

- overall shut-down reactivity of the 12 normal safety rods;
- overall shut-down reactivity of the 28 direct-acting safety rods (rods equipped with a release mechanism actuated by a rapid rise in pressure);
- calibration curves for the two groups of bulk rods and of all the sector rods at different positions of the bulk rods;
- local criticality conditions for each successive raising of the rods in a given reactor zone.

Finally, by analysing the results it was possible to establish a linear law governing the variation of the reactivity pressure coefficient as a function of the zero-pressure of the pile.

The measuring techniques employed were as follows: for the rod positions normally corresponding to the supercritical state of the reactor, poisoning in air and measurement of the reactivity-pressure coefficients; for the intermediate points of the calibration curves, measurements of the equilibrium pressures. On the other hand, the measurements in the sub-critical zone were performed by the rod drop method, causing the rods to drop starting from a critical condition and analysing the flux attenuation curve during the following 90 seconds.

Furthermore, two of the previous calibrations relating to the sector rods and to one of the two bulk-rod groups, were repeated in the hot state in order to evaluate separately the effect of temperature on the efficiency of the two types of control rods (grey rods and black rods).

The control-rod calibrations gave satisfactory results which were in good agreement with the calculated values. They took a total of 20 days.

(e) Determination of the sensitivity of the cladding burst detection system

This determination was effected by loading the reactor with non-clad sheets of an enriched-uranium/aluminium alloy and maintaining the reactor at a constant power of 100 kW for ten hours. In this way it was possible to demonstrate the correct operation of the measuring equipment and to determine its sensitivity.

The nuclear tests at very low power in air were completed at the end of March, that is to say three months after first criticality.

In general, all the test proceeded in accordance with a programme which had been very carefully drawn up by NPPC. Each test was the subject of a procedure which was checked down to the last detail and approved before execution of the test by the operator and the inspection authorities.

This method proved to be extremely efficient, the planned programme being carried out in its entirety within the time schedule adhered to.

4.3.4. Power run-up

This last phase commenced on 2nd April. It was carried out in a number of stages and at the beginning of July, power levels were 600 MW thermal and 160 MW electrical.

At each stage a record was made of the operating stable conditions of the reactor: gas and fuel temperature distribution, heat balance, reactivity balance, radiation survey, etc...
Tests were carried out on the charge-discharge machine and the flux scanning equipment.

During this first power operation period, fuel temperatures were kept below the maximum operating limits. These limits will only be reached when the procedure of reactor temperature survey is definitely settled.

On the whole, power raising was performed with only minor troubles, generally not due to the reactor and which were rapidly corrected.

At the end of June, the net electrical production since the beginning was 69 million kWh.

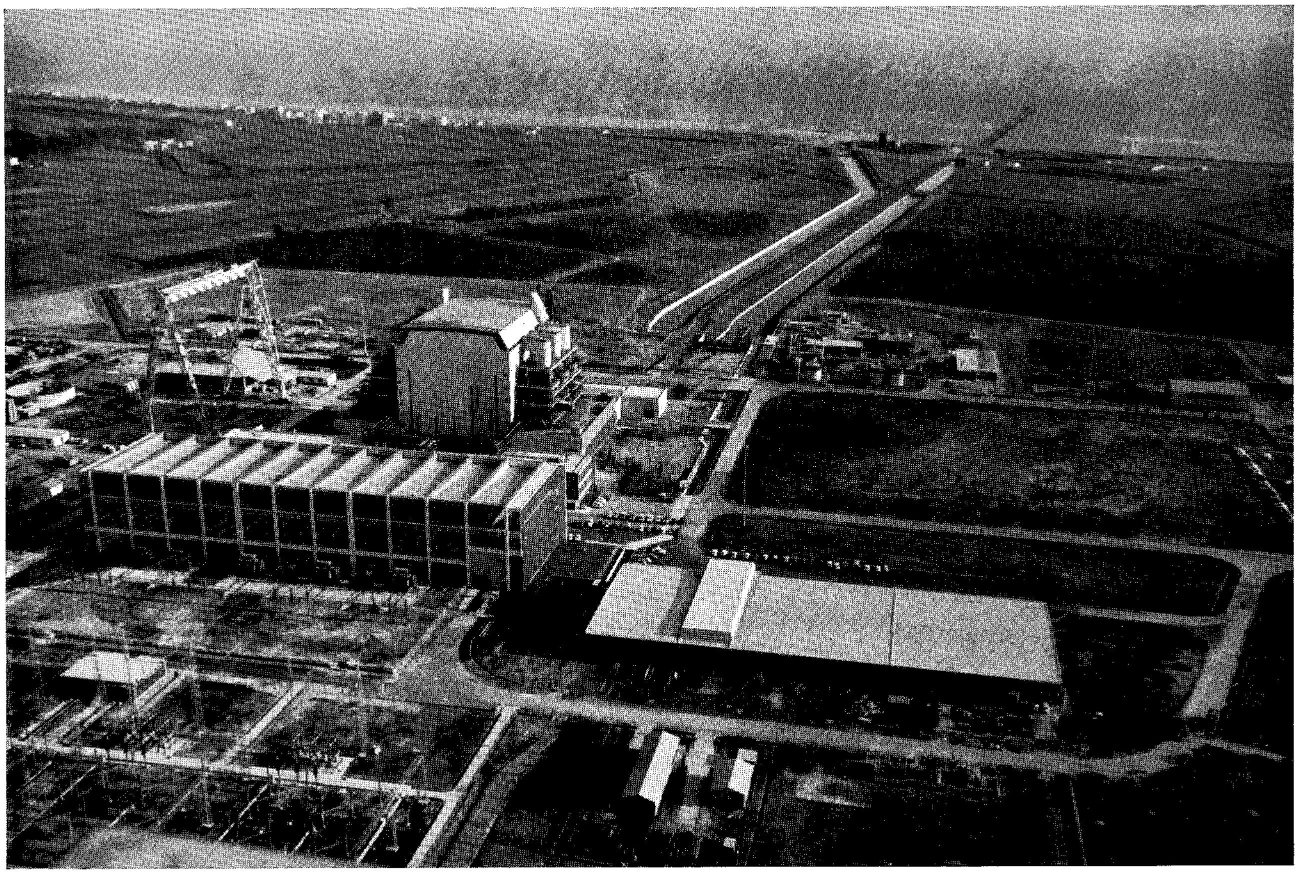
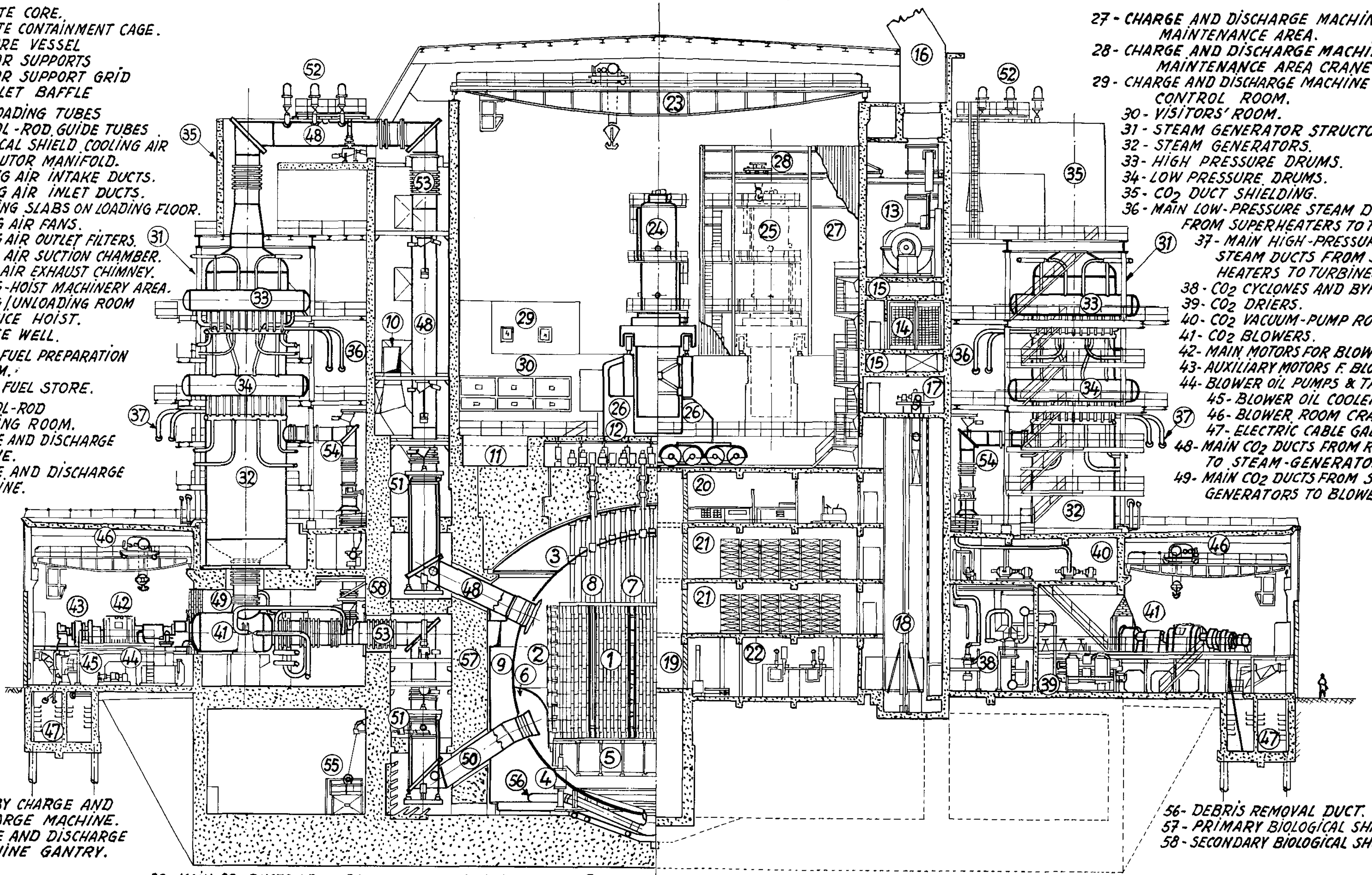


Fig. 4.1

Survey of site.

FIG. 4.2

VERTICAL CROSS SECTION OF REACTOR



1- GRAPHITE CORE.
2- GRAPHITE CONTAINMENT CAGE.
3- PRESSURE VESSEL
4- REACTOR SUPPORTS
5- REACTOR SUPPORT GRID
6- CO_2 INLET BAFFLE
7- FUEL-LOADING TUBES
8- CONTROL-ROD GUIDE TUBES
9- BIOLOGICAL SHIELD COOLING AIR DISTRIBUTOR MANIFOLD.
10- COOLING AIR INTAKE DUCTS.
11- COOLING AIR INLET DUCTS.
12- SHIELDING SLABS ON LOADING FLOOR.
13- COOLING AIR FANS.
14- COOLING AIR OUTLET FILTERS.
15- COOLING AIR SUCTION CHAMBER.
16- COOLING AIR EXHAUST CHIMNEY.
17- LOADING-HOIST MACHINERY AREA.
18- LOADING/UNLOADING ROOM SERVICE HOIST.
19- SERVICE WELL.
20- FRESH FUEL PREPARATION ROOM.
21- FRESH FUEL STORE.
22- CONTROL-ROD HANDLING ROOM.
23- CHARGE AND DISCHARGE CRANE.
24- CHARGE AND DISCHARGE MACHINE.
25- STANDBY CHARGE AND DISCHARGE MACHINE.
26- CHARGE AND DISCHARGE MACHINE GANTRY.
27- CHARGE AND DISCHARGE MACHINE MAINTENANCE AREA.
28- CHARGE AND DISCHARGE MACHINE MAINTENANCE AREA CRANE.
29- CHARGE AND DISCHARGE MACHINE CONTROL ROOM.
30- VISITORS' ROOM.
31- STEAM GENERATOR STRUCTURES.
32- STEAM GENERATORS.
33- HIGH PRESSURE DRUMS.
34- LOW PRESSURE DRUMS.
35- CO_2 DUCT SHIELDING.
36- MAIN LOW-PRESSURE STEAM DUCTS FROM SUPERHEATERS TO TURBINES.
37- MAIN HIGH-PRESSURE STEAM DUCTS FROM SUPERHEATERS TO TURBINES.
38- CO_2 CYCLONES AND BYPASS.
39- CO_2 DRIERS.
40- CO_2 VACUUM-PUMP ROOM
41- CO_2 BLOWERS.
42- MAIN MOTORS FOR BLOWERS.
43- AUXILIARY MOTORS F. BLOWERS.
44- BLOWER OIL PUMPS & TANKS.
45- BLOWER OIL COOLERS.
46- BLOWER ROOM CRANE.
47- ELECTRIC CABLE GALLERIES.
48- MAIN CO_2 DUCTS FROM REACTOR TO STEAM-GENERATORS.
49- MAIN CO_2 DUCTS FROM STEAM-GENERATORS TO BLOWERS.
50- MAIN CO_2 DUCTS FROM BLOWERS TO REACTOR.
51- BUTTERFLY VALVES ON MAIN CO_2 DUCTS.
52- CO_2 SAFETY VALVES
53- EXPANSION JOINTS ON MAIN CO_2 DUCTS.
54- CO_2 BYPASS DUCT.
55- BUTTERFLY-VALVE REMOTE CONTROL.
56- DEBRIS REMOVAL DUCT.
57- PRIMARY BIOLOGICAL SHIELD.
58- SECONDARY BIOLOGICAL SHIELD.

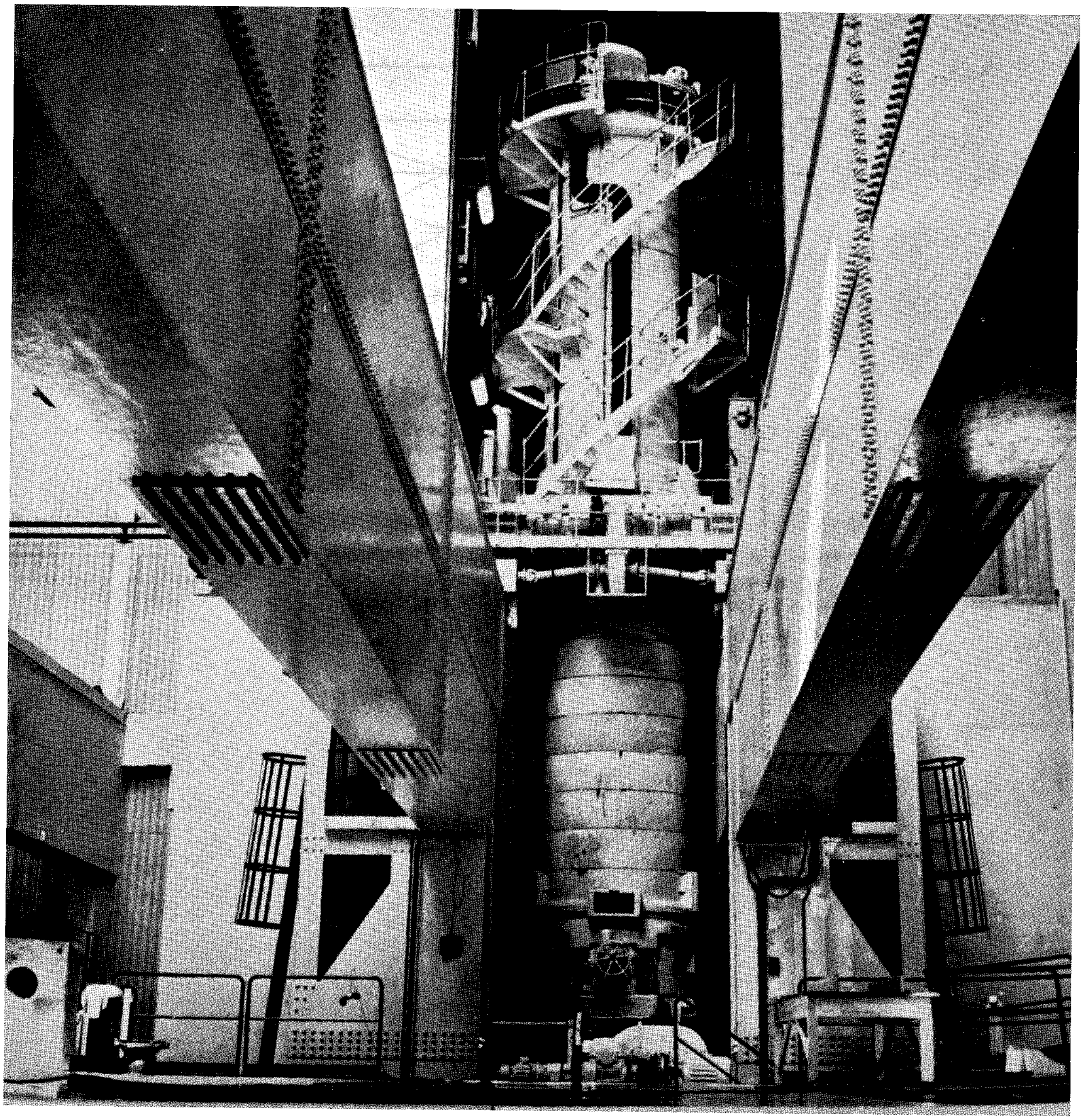


Fig. 4.3

Fuel charge- and discharge machine.

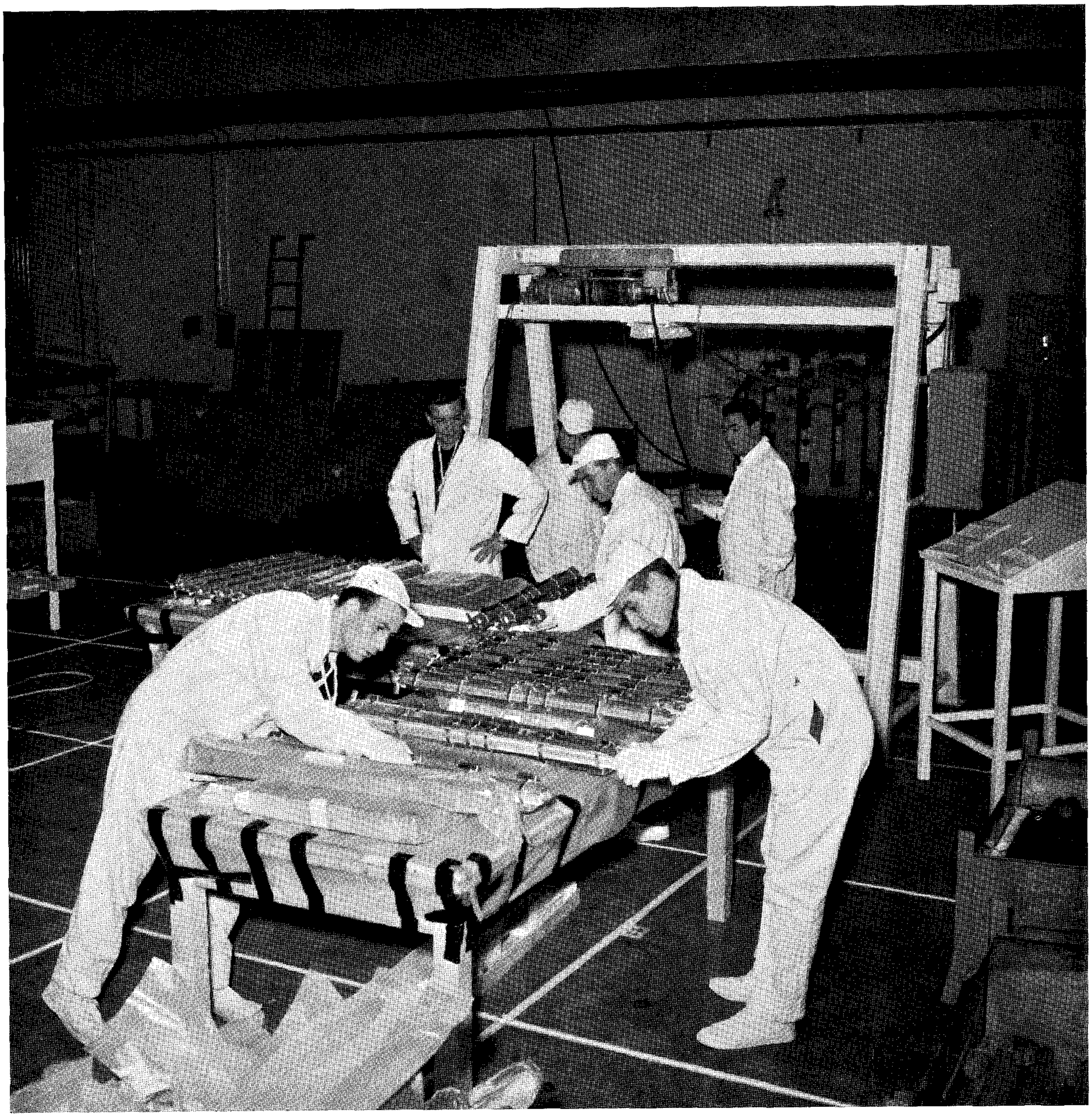


Fig. 4.4

First fuel loading.

5. S.E.N.A., THE NUCLEAR POWER PLANT IN THE ARDENNES

by

J. EHRENTREICH and W. KAUT

LIST OF FIGURES

5. S.E.N.A., THE NUCLEAR POWER PLANT IN THE ARDENNES

5.1. Introduction

The nuclear power station in the Ardennes is a large PWR-type electric plant which is being constructed on the river Meuse near the village of Chooz in France. The plant is owned and will be operated by "Société d'Energie Nucléaire Franco-Belge des Ardennes (S.E.N.A.)", which is a company formed by "Electricité de France" (a public corporation in France) and "Centre et Sud" (a group of Belgian utility undertakings). The member companies of "Centre et Sud" are:

Société d'Electricité de Sambre-et-Meuse des Ardennes et du Luxembourg (Esmalux),

Union Intercommunale des Centrales Electriques du Brabant (Interbrabant),

Sociétés Réunies d'Energie du Bassin de l'Escaut (Ebes),

Société Intercommunale Belge de Gaz et d'Electricité (Intercom).

This group represents practically all electricity production and distribution enterprises in Belgium.

The necessary design and construction studies for the plant are being carried out for S.E.N.A. by:

B.E.N. (Bureau d'Etudes Nucléaires) Sofina Electrobel Traction Electricité	in Belgium
Région d'Equipement Thermique No. 1 de l'EDF Région d'Equipement Thermique Nucléaire de l'EDF	in France

A technical committee coordinates the studies of the above-mentioned organisations and those of the constructor. The design of the nuclear power plant and the manufacture of its equipment are being handled by the A.F.W. constructor group, which includes A.C.E.C., Framatome and Westinghouse Electric Corporation.

Besides the firms which constitute Framatome in France the Belgian firms Cockerill - Ougrée, Seraing and Métallurgie et Mécanique Nucléaires (M.M.N.), Brussels, are also members of the A.F.W. constructor group.

The manufacture of parts will be equally distributed between Belgian and French companies. Some of the main parts and their manufacturers are listed below.

Reactor pressure vessel	Société des Forges et Ateliers du Creusot (S.F.A.C.)
Pressuriser	S.F.A.C.
Primary pipes	S.F.A.C.
Primary valves	Ateliers de Constructions Electriques de Charleroi (A.C.E.C.)
Primary pumps	A.C.E.C. Forges et Ateliers de Constructions Electriques de Jeumont (F.A.C.E.J.)
Steam generators	Cockerill - Ougrée
Turbine	S.F.A.C., Rateau, Cockerill - Ougrée
Alternator	A.C.E.C. Le Matériel Electrique SW, Paris (SW)
Condenser	Cockerill - Ougrée
Control rods	Westinghouse Electric Corp.
Control-rod drive mechanisms	A.C.E.C., SW
Fuel elements:	
First charge	Westinghouse Electric Corp.
Subsequent charges	Métallurgie et Mécanique Nucléaires, Bruxelles (M.M.N.) and Compagnie d'Etudes et de Recherches de Combustibles Atomique (C.E.R.C.A.)

The civil-engineering work will be carried out by a Franco-Belgian group consisting of Compagnie Industrielle des Travaux, Société Générale d'Entreprises, Société Centrale d'Entreprises and Entreprise Générales et Matériaux.

5.2. Construction

The nuclear power plant will be of mixed-type construction. The components of the reactor cooling system and the auxiliaries are housed in two caves excavated out of the hillside along the river Meuse. Two access galleries, 120 m. long, lead to these caves. Biological shielding on the outside will be provided by the rock. The machine house and the buildings for the electrical plant and control room and for the operating and administrative staff will be situated on the far side of the hill, on the right bank of the Meuse. There will also be buildings for waste disposal, ventilation, workshops and stores.

Some of the principal dimensions of the essential buildings are as follows:

	Width (m)	Height (m)	Length (m)
Reactor cave	18.5/19.5	37.3	41
Auxiliary cave	15/16	35.5/29/ 15.5	46.5
Reactor access gallery	max. 9 8	max. 9 6	120
Auxiliary access gallery	6	5	116
Machine house	35	30/35	70
Building for electrical equipment and control room	35	21	23.5

The contract with the A.F.W. constructor group for the plant design and the manufacture of equipment was signed in September 1961. The various mechanical and electrical parts of the plant are now being built in the manufacturers' workshops in Belgium and France.

The contract for the civil-engineering work was concluded in January 1962 and at the same time the on-site construction work was started. The right bank of the Meuse was prepared first, so that work could begin on the excavation of the galleries leading to the two caves. The problem of landslides from the mountain-side above the gallery entrances had to be overcome before excavation was possible. Loose boulders made the excavation itself difficult. It was necessary to line the gallery entrances with concrete and to prop the excavated sections. Excavation of the galleries has now been completed and that of the caves is being continued (Figs. 1 and 2). Excavation work is still very difficult; for protection against falling rocks, bolts have to be placed not only in the gallery vault but also in the vault of the cave.

Foundation-piling for the machine house and the building for the electrical plant is proceeding and the concreting of the bridge over the Meuse is being prepared.

Construction work was held up by the extremely cold winter of 1962/1963. Excavation work and preparation of the machine-house were brought to a stand-still in February 1963 by a strike.

The erection of the mechanical and electrical parts will start in mid-1964.

The start-up of the plant is scheduled for the middle of 1966.

5.3. Description

The nuclear power plant consists primarily of a large pressurised-water reactor of advanced design. The plant capacity is 905 MWth (266 MWe net).

The core contains 120 fuel assemblies arranged in a square pattern. UO_2 fuel of low U_{235} enrichment is contained within stainless-steel tubes in a stainless-steel assembly. Cruciform rods are provided for reactor control.

The reactor cooling system removes heat from the core and converts it into steam for driving the turbo-generator. Water is circulated through 4 loops at an average temperature of 284°C and a pressure of 140 atmg with a total mass flow rate of 19×10^6 kg/h.

Auxiliary systems are provided for cooling services, chemical and volume control, sampling and other functions. Dry saturated steam is produced in the steam generators at 34 atmg, the total output from the 4 units being 1570 t/h. The steam flows to a single turbo-generator unit which has 4 stages, viz. 1 high-pressure and 3 low-pressure stages.

5.3.1. Reactor core

The reactor core consists of 120 fuel elements (see Fig. 3), each containing 200 fuel rods; it is nearly cylindrical and has a diameter of 2.5 m. and a height of 3 m. The core therefore contains 40 tons of uranium with 3.5% U_{235} enrichment. Demineralised water at a pressure of 140 atmg acts as coolant and moderator.

The rods in each fuel assembly are held in place within the lattice by spacer grids located at intervals along the assembly. Spring clips will prevent lateral movement of the rods, while permitting them to expand axially. Nozzles at the top and bottom of the bundle and a perforated stainless-steel can welded to the grids and surrounding the bundle complete the assembly. Portions of the outer rows of fuel rods will be omitted at the corners of the bundle to form slots for the blades of the offset-cruciform control rods. The core is provided with a form-fitting austenitic stainless-steel baffle which will surround the cluster of assemblies and confine the coolant flow to the fuel-containing region.

5.3.2. Reactor cooling system

The reactor cooling system (Fig. 4) transfers heat from the reactor core to the secondary system to produce steam for the turbine. All the system equipment is located in the reactor cave. The system consists of 4 identical heat-transfer loops connected in parallel to the reactor vessel, each containing a circulating pump and a steam generator. The system also includes the pressuriser, the pressuriser relief tank, the interconnecting piping and the instrumentation for operational control.

The reactor vessel (Fig. 5), which contains the core, is a welded cylindrical container with a hemispherical bottom-head and a flanged and gasketed removable top-head. The main cooling water enters the reactor vessel assembly through 4 inlet nozzles and leaves through 4 outlet nozzles. All the internals are suspended from the support ledge near the top of the reactor vessel flange and are held in place by the vessel head, which exerts a downward pressure on the core retaining ring. The fuel assemblies are held between the 2 core-support plates, which are perforated to accommodate the assembly extensions.

The vessel head flange has provision for seal welding and the vessel flange has two monitored leak-off connections for detecting any malfunction of the two in-series gaskets. The inside surfaces of the vessel are clad with stainless steel.

The internals, which are made of stainless-steel, are:

a 3"-thick cylindrical thermal shield,
a lower core-support assembly,
an upper core-support assembly.

The vessel will be made from low-alloy carbon steel, designated type 1.2 Mo 7. The closure bolts are forged from SAE-4340. The main reactor vessel characteristics are as follows:

inside diameter of the vessel	3200 mm
main wall thickness	174 mm
hemispherical bottom-head thickness	95 mm
hemispherical top-head thickness	140 mm
overall height	11.3 mm

The 4 steam generators are vertical shell and U-tube evaporators with integral moisture-separating equipment. The reactor coolant flows through the inverted U-tubes, entering and leaving through the nozzles located in the hemispherical bottom head. Steam is generated on the shell side and flows upwards through the moisture separators to the outlet nozzles at the top of the vessel.

The steam generators are primarily of carbon steel, the heat transfer tubes are of austenitic stainless-steel, and the internal surfaces of the coolant heads and nozzles are clad with austenitic stainless-steel. The main steam generator characteristics are:

outside diameter of shell,	upper	3,095 mm
	lower	2,288 mm
overall length		14,096 mm

The 4 reactor coolant circulating pumps are identical canned-motor, single-speed units with the centrifugal pump impeller mounted on a common vertical shaft. All parts of the pumps and valves in contact with the coolant are of austenitic stainless-steel.

The pressuriser maintains liquid and vapour in equilibrium for pressure-control purposes. Steam is generated by electric heaters inserted through the vessel wall near the bottom head. A spray nozzle is located at the top of the pressuriser. A small continuous flow is normally maintained through the spray line. A much larger spray flow is used during pressure transients.

The pressuriser relief tank condenses and cools the discharge from the pressuriser safety and relief valves. Discharge from smaller relief valves in the vapour container is also piped to this tank. The tank normally contains water and nitrogen atmosphere. Steam is discharged below the water level so as to condense and cool the steam by mixing it with water. The tank is equipped with a spray system and drain which are operated automatically to cool the tank after a discharge. The tank is protected by rupture discs against any steam discharge exceeding the design value. The tank is of carbon steel, with a corrosion-resistant coating on the wetted surfaces.

The reactor coolant piping, fittings and valves are of stainless steel.

5.3.3. Chemical and volume control system

The chemical and volume control system allows a certain degree of reactivity control and maintains the proper water inventory in the reactor coolant system, reduces the quantity of fission- and corrosion-product impurities and maintains the proper concentration of corrosion-inhibiting chemicals in the reactor coolant. The system is also used for filling and pressure-testing the reactor coolant system. All the system equipment is located in the auxiliary cavern with the exception of the regenerative heat-exchanger, the drain cooler and the let-down valves, which are located in the reactor cavern.

The residual-heat exchangers of the auxiliary coolant system and the deborating demineralisers of the waste-disposal system are also incorporated in this system.

During operation of the plant, reactor coolant is withdrawn from the cold leg of loop C of the coolant system and returned to the cold legs of loops B and D.

The reactor coolant entering the chemical and volume control system from loop C is first cooled on the shell side of the regenerative heat-exchangers, after which the coolant pressure is reduced by passing the coolant through one let-down valve. The cooled low-pressure water then leaves the reactor cavern and flows to the auxiliary cavern. Here the water enters the residual-heat removal loop of the auxiliary coolant system. The temperature of the water is further reduced on the tube side of the residual-heat exchangers. The coolant then leaves the residual-heat removal loop and passes through one of the mixed-bed demineralisers. Ionic impurities are removed in the demineraliser and the coolant passes on through the coolant filter and a spray nozzle into the volume control tank. The atmosphere in this tank contains hydrogen, which dissolves in the reactor coolant. The fission gases are removed from the coolant by venting this tank. The coolant thereupon flows to the charging pumps, which raise the pressure above that in the reactor coolant system.

The chemical and volume control system piping is of austenitic stainless steel.

5.3.4. Neutron shield system

The neutron shield system protects the primary concrete shielding from excessive thermal gradients and dehydration caused by radiation from the core and heat leakage from the reactor vessel. In conjunction with other neutron shielding it reduces component activation and biological hazards to negligible levels by reducing the neutron flux from the reactor. All the system equipment is located inside the reactor cave, with the exception of the filling line valves. This system includes the neutron shield tank, a surge tank, 8 coolers, valves, piping and instrumentation.

The neutron shield tank is a vapour container which constitutes the neutron shield; it is located in the annular space between the reactor vessel and the primary concrete shield. When the plant is in operation, heat is absorbed by the water in the neutron shield tank and transferred to component cooling water, which is circulated by the auxiliary coolant system through 8 tubular heat-exchangers inserted in the top of the neutron shield tank.

The neutron shield tank is open to the reactor cave and is not pressurised.

5.3.5. Auxiliary coolant system

The auxiliary coolant system provides cooling for the :

reactor coolant pumps
sample heat exchanger
neutron shield tank
reactor coolant let-down flow
drain cooler
reactor coolant system
spent fuel pit

This system consists of 3 loops, viz.

the component cooling loop
the residual-heat removal loop
the spent fuel pit cooling loop

The component cooling loop removes heat from the reactor coolant pumps, the sample heat-exchanger, the neutron shield tank, the drain cooler and the residual-heat exchangers. These units are arranged in parallel flow circuits. When water is circulated through these heat sources, heat is transferred to the component cooling water, which in turn is cooled by river water. The component cooling loop thus serves as an intermediate system between the reactor coolant and the river water cooling system.

The residual-heat removal loop removes heat from the reactor core and reduces the temperature of the reactor coolant system during the second phase of plant shut-down. During the first phase of the plant shut-down the temperature of the reactor coolant system is reduced by transferring heat from that system to the secondary system.

The residual-heat exchangers are also used for removing heat from the reactor coolant let-down flow during operation of the chemical and volume control system.

The spent fuel pit cooling loop removes residual heat from spent fuel stored in the spent fuel pit. The spent fuel is placed in the pit during refuelling operations and is stored there until it is shipped to a reprocessing plant.

5.3.6. Sampling system

The sampling system provides fluid samples for laboratory analysis, by means of which the chemistry and radio-chemistry of the reactor coolant are evaluated.

5.3.7. Waste-disposal system

The waste-disposal system receives, processes, stores and/or disperses all the radioactive liquid, solid and gaseous wastes produced. Any material dispersed to the environment will comply with the relevant regulations.

5.3.8. Safety injection system

The safety injection system adds borated water to the reactor core in the event of loss of coolant. This water serves to cool the core and prevents overheating of the cladding.

5.3.9. Fuel handling system

In order that the reactor may be refuelled without risk to the personnel, facilities are provided for the under-water removal of fuel assemblies, for transferring the assemblies from the reactor cave to a water-filled spent fuel storage pit located in the auxiliary cave, for storing spent fuel assemblies under water for a sufficient time to allow them to decay, and for removing the shipping cask from the spent fuel storage pit and placing it on a car for transport from the auxiliary cavern.

5.3.10. Radiation shielding

The purpose of radiation shielding is to provide biological protection for plant personnel wherever a potential radiation hazard exists. Radiation shielding is provided mainly by:

- the neutron shield tank (see 3.4);
- the primary shield, which takes the form of a reinforced-concrete structure immediately adjacent to and surrounding the exterior of the neutron shield tank;
- the auxiliary shielding, which is provided in the reactor and auxiliary caves around those items of equipment which may become radioactive in the course of operation and which might present a radiation hazard in the auxiliary cavern during operation in the reactor and auxiliary caves during a plant shut-down;
- the fuel-handling shielding, which provides protection during the removal and transfer of spent fuel assemblies and control rods from the reactor vessel to the spent fuel pit.

The shielding design will comply with the relevant regulations.

5.3.11. Ventilation system

The ventilation system will maintain the radioactivity in the air of the various plant buildings and in the surrounding atmosphere at a reasonable level. This system comprises various circuits which ensure:

the constant ventilation of the reactor cavern;

the constant cooling and circulation of the air inside the reactor cave;

the temporary ventilation of the reactor cavern at very high displacement rates, in order to effect purging in a few hours before the arrival of the workers;

the constant ventilation of the tanks containing fuel elements;

the constant ventilation of the rooms in the auxiliary basement;

the constant ventilation of the storage buildings and of the buildings for the treatment of radioactive waste;

the supply of fresh air to the underground rooms.

5.3.12. Reactor control system

The reactor is regulated by a set of control rods. The complete control-rod and drive assembly consists of:

the absorber section
the fuel-bearing control-rod follower
the drive-shaft and shock-absorber assembly
the magnetic jack drive mechanism

The absorber section is a silver-indium-cadmium extrusion shaped as an offset cruciform to fit between the fuel assemblies. The fuel-bearing follower fitted to the lower end of each absorber section is constructed of stainless-steel tubes filled with UO_2 pellets and assembled into an offset cruciform geometry by means of cross-straps welded to spacers in the fuel tubes.

Each control rod is attached to a drive-shaft shock-absorber assembly by means of a split collet-type coupling. The magnetic jack drive mechanism provides the motive power for moving a control rod as and when required for regulating the reactor power. The mechanism also provides a signal indicating the position of the control rod.

The drive is a magnetic-jack, latch-type mechanism. In this type of mechanism, magnetic fields set up by coils outside the pressure housing exert forces on the pole pieces inside the pressure housing to engage and move the mechanism drive shaft.

The reactivity can be controlled independently of the reactor control rods by means of the chemical and volume control system. This will be done by the introduction of boric acid dissolved in water. This system controls the boron concentration in the reactor primary system.

The reactor will normally be operated by an automatic control system at power levels in excess of approximately 10% of full power. This system evaluates the highest hot leg temperature and the highest cold leg temperature. The average of these two is computed and combined with a rate of change of Tavg to form an error signal. This error signal is compared with a Tavg reference signal. If the error signal exceeds the permissible dead band, rod insertion is initiated. If the error is less than the dead band, rod withdrawal is initiated.

5.3.13. Secondary circuit and turbo-generator

The plant includes a single turbo-alternator set with a gross output of 288 MW.

The turbine, which runs on saturated dry steam at a pressure of 34 atmg and has a speed of 3000 rpm, is a 4-stage unit with one high-pressure and three low-pressure stages. Between the high- and low-pressure stages the steam is dried by live steam from the reactor.

The condenser is cooled by water from the Meuse. The turbine drives a single 3-phase alternator, which is cooled by pressurised hydrogen.

The rest of the equipment in the secondary circuit is identical with that of a thermal power plant.

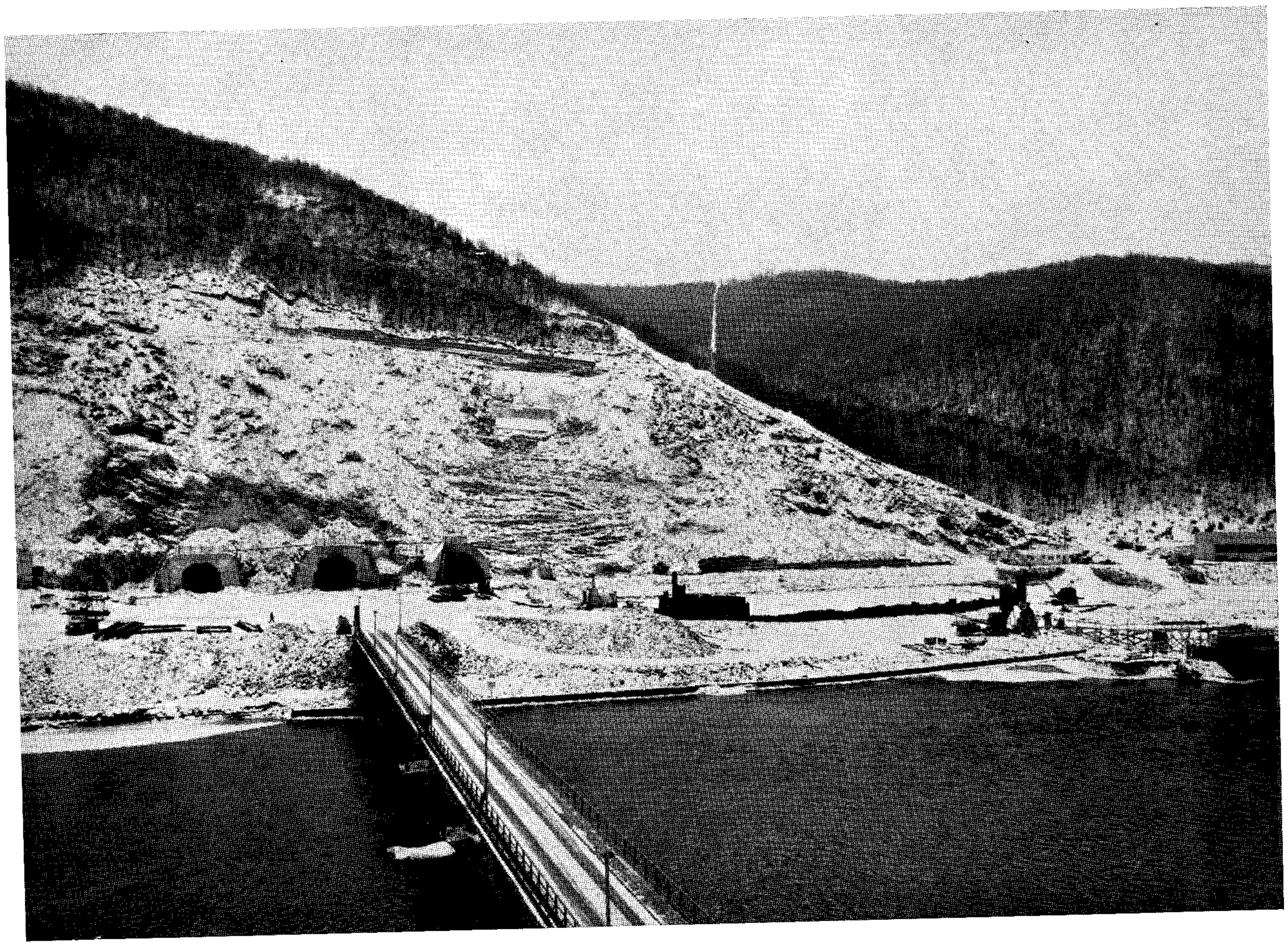
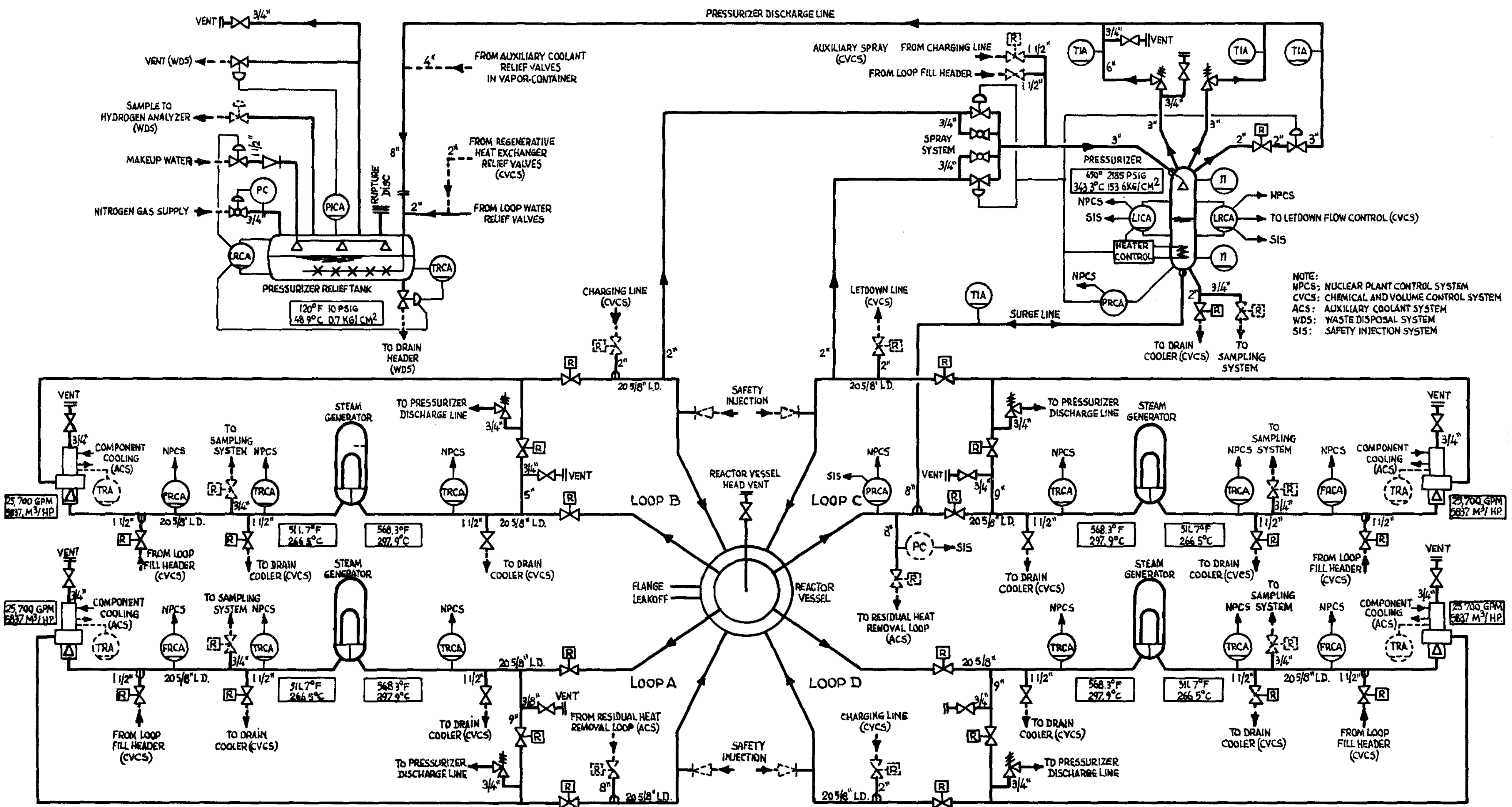


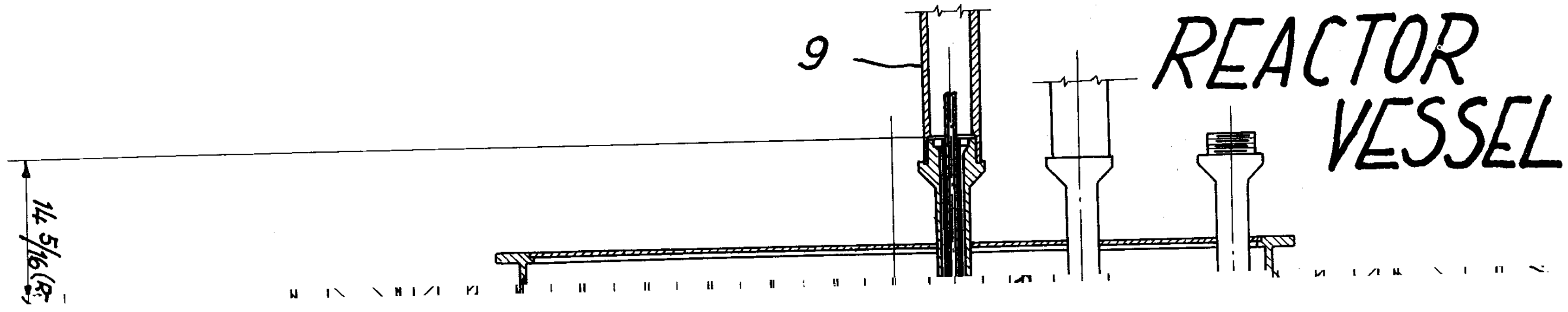
Fig. 5.1

General view of the right bank of the river Meuse (Febr. 1962).



FLOW DIAGRAM FIG. 5.4

Fig. 5.4. Flow diagram



9
REACTOR VESSEL
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6. K.R.B., THE R.W.E.-BAYERNWERK NUCLEAR POWER PLANT

by

J. DESFOSSES, W. KAUT and J.S. TERPSTRA

LIST OF FIGURES

6. K.R.B., THE R.W.E.-BAYERNWERK NUCLEAR POWER PLANT

6.1. Description of the project

After participating in the construction of the nuclear power plants SIMEA and SENN in Italy, and the SENA plant in the Ardennes, the European Atomic Energy Community (EURATOM) on 29.3.1963 signed an agreement to take part, under the joint US/Euratom Nuclear Power Plant Programme, in the construction of the KRB nuclear power plant. The other contracting party and constructor of the installation is the Kernkraftwerk R.W.E.-Bayernwerk G.m.b.H. of Gundremmingen, Kreis Günzburg in Bavaria.

The main supplier is a group of firms consisting of International General Electric Operations Ltd. (IGEOSA), Geneva, Switzerland, Allgemeine Elektrizitätsgesellschaft (AEG), Frankfurt-am-Main, and Hochtief A.G., Essen. Under the terms of the Joint Nuclear Power Plant Programme the plant is to be commissioned by 30.12.1965 at the latest.

Euratom will contribute to the costs to a maximum of 8 million EMA u.a. In return, apart from receiving the information specified in the agreement, Euratom has also the right to assign to KRB a limited number of persons who will work together with KRB personnel on the designing, construction and operation of the plant.

The plant has a rated power of 801 MWth and a net electric power of 237 MW. The boiling-water reactor is light-water cooled and moderated, with forced circulation and a double direct cycle; the fuel is enriched uranium dioxide.

The plant will be built in Gundremmingen, Kreis Günzburg, on the Danube, which will supply the water required as coolant.

The reactor building, a cylinder of about 30 m diameter and height 60 m, will accommodate the whole nuclear steam generating plant with its auxiliary systems, and the fuel storage tank.

Adjacent to it are the turbine-house and the operations building. The chosen site has sufficient room for three more plants with the same output.

The reactor core contains 368 fuel elements each of 36 rods, with a total uranium weight of 51 t. A non-uniform enrichment of the fuel elements was chosen with the aim of flattening out peaks in the flux distribution; thus for each element in the first core the enrichment is 2.72% in 31 rods and 1.86% in the remaining 5. In contrast to the boiling-water reactor of SENN in Italy and the Dresden plant in the USA, which use zircaloy as their fuel-cladding material, the KRB fuel will be clad in 0.305 mm stainless steel. A burn-up of 16,500 MWd/t is guaranteed.

The chosen power density of 36.9 kW/l represents a notable advance on the corresponding figures for the Dresden and SENN plants which are 28.9 kW/l and 28.6 kW/l respectively. A further difference from these plants is that the steam is separated from the primary water coolant inside the reactor pressure vessel.

The turbine has an output of 250 MWe at 1500 rpm. For this it requires a supply of fresh steam of about 1025 t/hr at 70.3 atm. abs. direct from the reactor, and a secondary supply of about 450 t/hr at 32.7 atm. abs. from the secondary steam-generators, which is fed into the turbine at an intermediate stage. Particular attention must be paid to water drainage from the turbine. The water separated during expansion in the wet steam range is removed from the turbine at six points together with the steam extracted for the preheating of the feed water. In addition, water-separators are mounted in the change-over lines between the high pressure and the low pressure sections.

At rated load the plant attains an overall efficiency of 29.6%. The cost of construction is about 280 EMA u.a./kW, assuming a net power of 237 MWe. On the basis of a load factor of 7000 hr/year, the power cost works out at 9.57 EMA u.a./MWh.

6.2. Present status of activities

The contract for the construction and commissioning of the RWE-Bayernwerk nuclear power plant was confirmed by the contractors AEG, IGEOSA and Hochtief A.G. in November 1962, while work has already started on the site preparation in October of the same year. The bottoms of the construction pits had been provided with complete frost protection before the onset of the bad weather so that the work on the foundations could be resumed as soon as the warmer weather returned in March. The frost merely caused an initial six-week delay with the construction of the reactor building, but this was partly offset by extra shifts in the erection of the containment shell and it will probably be possible to make up for the delay entirely. The construction of the other buildings - particularly the turbine building - and the sewers in the vicinity of the plant is proceeding according to schedule. The rail connection has been effected and the access roads are fully usable.

Of the heavy plant components, orders have been placed for the reactor building containment shell, the reactor pressure vessel, the circulation pumps, the heat exchangers and the large transformers: these items are now being manufactured. Wherever necessary, the approval of the Technische Überwachungsverein, the expert body acting for the licensing authorities, has been obtained for these. The orders are about to be placed for most of the other components of the primary and secondary circuits.

Work on the detailed design has been started, partly in conjunction with the manufacturers alone and partly in collaboration with the Technische Überwachungsverein. Instances of items already being dealt with are: problems relating to the selection of appropriate materials, the capacity and construction of the waste processing installation, power consumption, measurement and control, regulation systems and procedures, reactor shielding, lighting, etc. The turbine design has been more or less settled.

Furthermore, the sponsors of the project are carrying out on their own account preliminary work connected with environmental radioactivity monitoring. The monitoring system must be functioning, in line with the principle of "demonstrable safety", before the beginning of zero-power operation. For this purpose, orders have been placed for an observation tower over 100 m high, as well as for a considerable proportion of the metering devices required for the recording of local weather conditions and for monitoring environmental radioactivity.

The sponsors likewise placed orders, on their own account, for the telephone and communications equipment. Work is under way on the designing of the line construction and the adjacent 220 kV switch-gear.

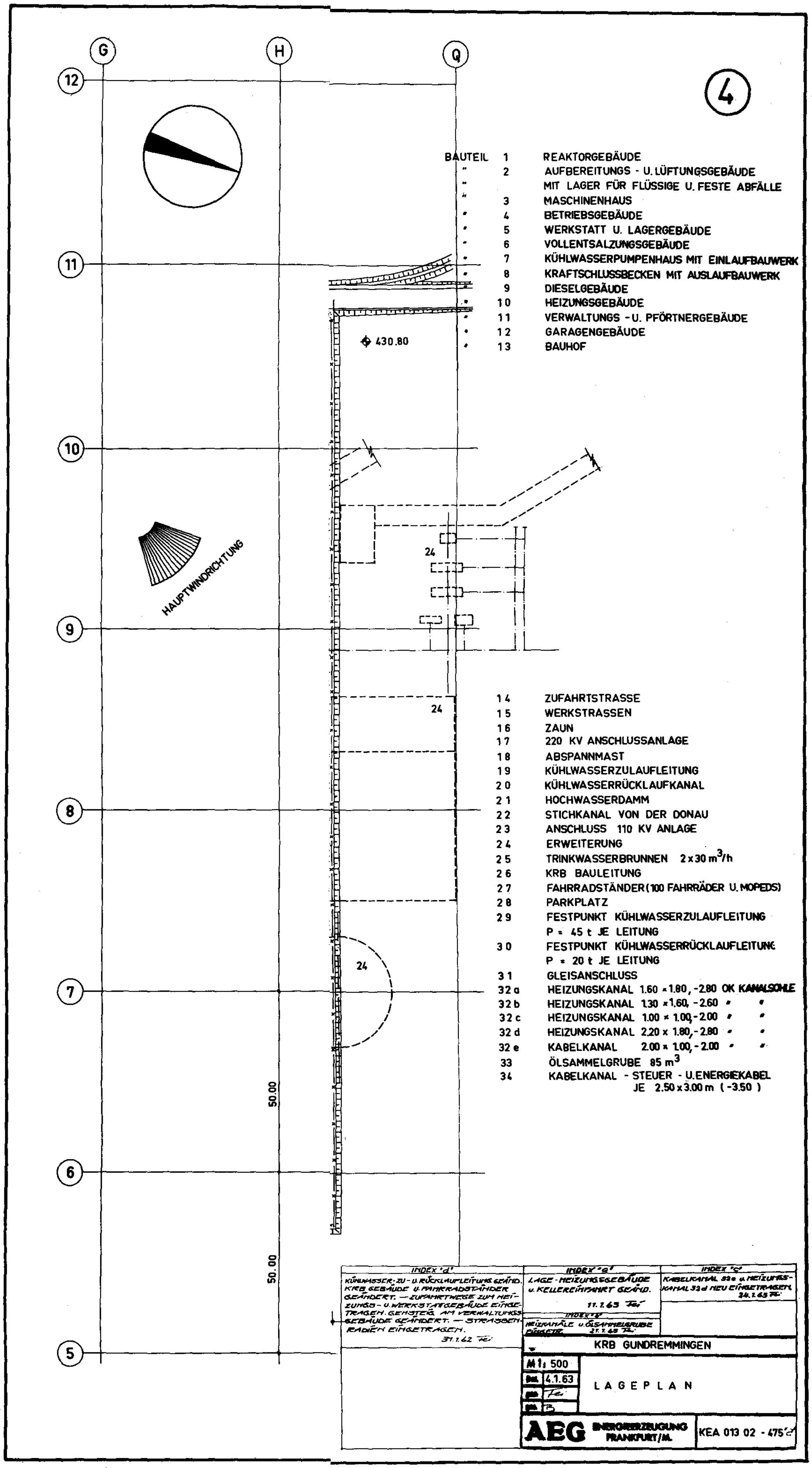
The authorities have been provided with a provisional safety report dealing mainly, in line with the status of the planning work, with the fundamental design criteria and aimed at the granting of a construction license under the licensing procedure governed by nuclear legislation. The description of the individual pieces of equipment and circuits with which the design requirements are to be fulfilled, which must be provided before the operating license is granted, will be supplied as the detailed design work progresses.

The Technische Überwachungsverein and the Reactor Safety Commission are still in the process of studying the provisional safety report before issuing the expertise requested by the licensing authorities. The study has not yet progressed sufficiently for a definitive and formal construction license to be granted under the Atom Law. To date, no factors have emerged according to which the design of the structures would appear inadequate.

6.3. Technical data

Thermal power	905 MW
Electrical power gross	250 MW
Electrical power net	237 MW

Overall plant efficiency	29.6%
Power density	36.9 kW/l
Reactor working pressure	71.3 atm. abs.
Primary steam supply	1025 t/hr
Primary steam pressure	70.3 atm. g.
Secondary steam supply	450 t/hr
Secondary steam pressure	32.7 atm. abs.
Coolant mass flow rate	12300 t/hr
Fuel	UO_2
Enrichment	2.72 or 1.86%
Number of fuel elements	368
Number of fuel rods per element	36
Total weight of uranium	51 t
Cladding material	stainless steel
Guaranteed burn-up	16500 MWd/t
Number of control rods	89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" 2 AUFBEREITUNGS - U. LÜFTUNGSGEBÄUDE
MIT LAGER FÜR FLÜSSIGE U. FESTE ABFÄLLE
" 3 MASCHINENHAUS
" 4 BETRIEBSGEBÄUDE
" 5 WERKSTATT U. LAGERGEBÄUDE
" 6 VOLLENTSALZUNGSGEBÄUDE
" 7 KÜHLWASSERPUMPENHAUS MIT EINLAUFBAUWERK
" 8 KRAFTSCHLUSSBECKEN MIT AUSLAUFBAUWERK
" 9 DIESELGEBÄUDE
" 10 HEIZUNGSGEBÄUDE
" 11 VERWALTUNGS - U. PFÖRTNERGEBÄUDE
" 12 GARAGENGEBÄUDE
" 13 BAUHOF
14 ZUFAHRTSTRASSE
15 WERKSTRASSEN
16 ZAUN
17 220 KV ANSCHLUSSANLAGE
18 ABSPANNMAST
19 KÜHLWASSERZULAUFLEITUNG
20 KÜHLWASSERRÜCKLAUFKANAL
21 HOCHWASSERDAMM
22 STICHKANAL VON DER DONAU
23 ANSCHLUSS 110 KV ANLAGE
24 ERWEITERUNG
25 TRINKWASSERBRUNNEN 2 x 30 m³/h
26 KRB BAULEITUNG
27 FAHRRADSTÄNDER (100 FAHRRÄDER U. MOPEDS)
28 PARKPLATZ
29 FESTPUNKT KÜHLWASSERZULAUFLEITUNG
P = 45 t JE LEITUNG
30 FESTPUNKT KÜHLWASSERRÜCKLAUFLEITUNG
P = 20 t JE LEITUNG
31 GLEISANSCHLUSS
32 a HEIZUNGSKANAL 1.60 x 1.80, -2.80 OK KANALSOHLE
32 b HEIZUNGSKANAL 1.30 x 1.60, -2.60 " "
32 c HEIZUNGSKANAL 1.00 x 1.00, -2.00 " "
32 d HEIZUNGSKANAL 2.20 x 1.80, -2.80 " "
32 e KABELKANAL 2.00 x 1.00, -2.00 " "
33 ÖLSAMMELGRUBE 85 m³
34 KABELKANAL - STEUER - U. ENERGIEKABEL
JE 2.50 x 3.00 m (-3.50)
INDEX 'd'
KÜHLWASSER- ZU - U. RÜCKLAUFLEITUNG GEÄND. KRB GEBÄUDE U. FAHRRADSTÄNDER GEÄNDERT. — ZUFAHRTWEGE ZUM HEIZUNGS - U. WERKSTATTGEBÄUDE EINGETRAGEN. GEHSTEIG AM VERWALTUNGSGEBÄUDE GEÄNDERT. — STRASSENRADIEN EINGETRAGEN.
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INDEX 'c'
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24.1.63 Fe
KRB GUNDREMMINGEN
M 1:500
4.1.63
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B
LAGEPLAN
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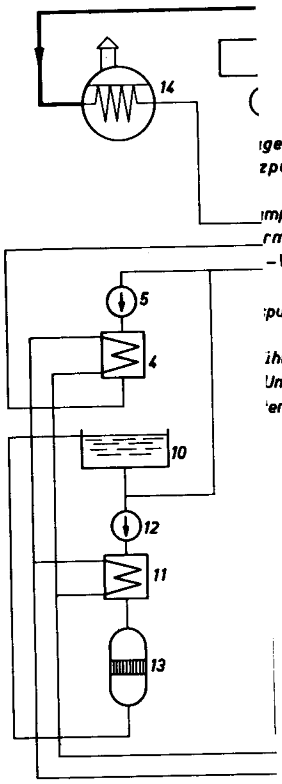

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'er

17 Hotwell
18 Kondensatpumpen
19 Treibdampfkondensator
20 Kondensatreinigung
21 Druckerhöhungspumpe
22 Speisewasservorwärmer
23 Speisepumpen
24 Zusatzwasseraufbereitung
25 Zusatzwasserbehälter
26 Zusatzwasserpumpen
27 Hauptkühlwasserpumpen
28 Vergiftungslösungsbehälter
29 Vergiftungslösungspumpen
30 Nebenkühlwasserpumpen
31 Reaktorgebäudesumpf
32 Notkühlwärmetauscher
33 Dampfstrahlluftsauger
34 Vorwärmerkondensatpumpen
35 Zwischenkühler
36 Kondensatvorratsbehälter
37 Regelstabantriebspumpe

250 KRB

Systemschaltplan

AEG KEA

Zchng.Nr. 1007-01-4 a

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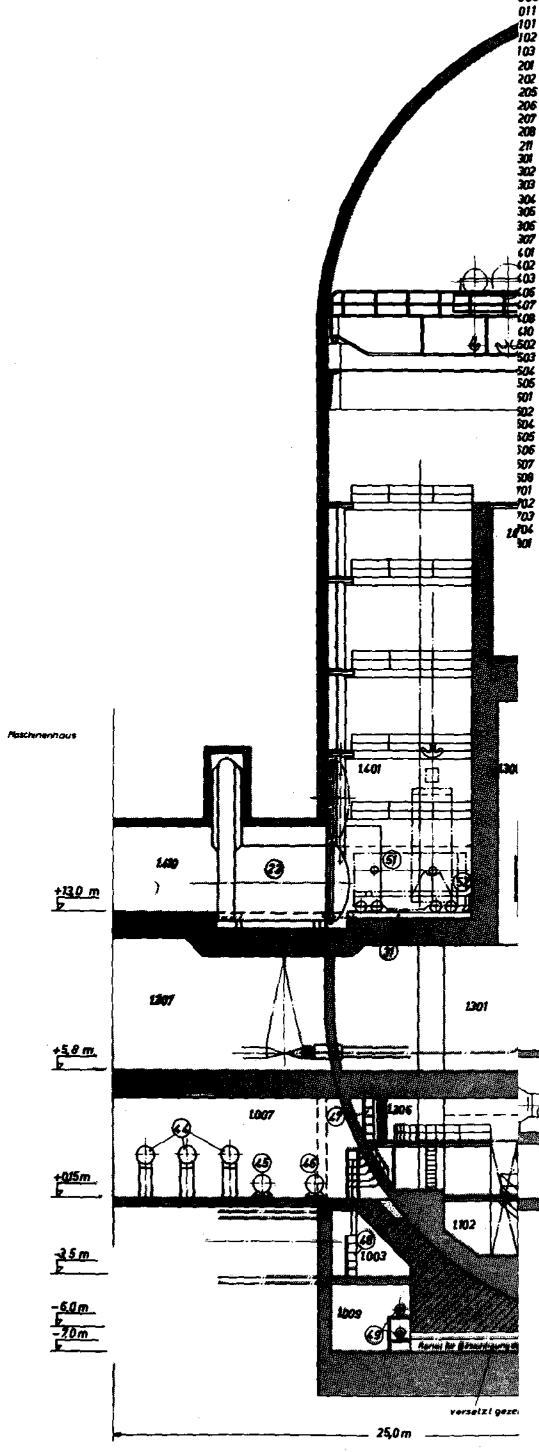


001 Ringraum Fundament
002 Kabelkanal(E-Technik)
003 Kabelkanal(M-Technik)
004 Raum für Kabelverteilung(E-Technik)
005 Rohrleitungskanal (Abfallaufb.)
007 Raum für Hilfsysteme
009 Raum für Gebäudenotkühlsystem
011 Ringraum Fundament
101 Raum für aktiven Sammeltank
102 Raum für Steurstabantriebsystem
103 Raum für Leerlaufpumpen u. Sumpfpumpen
201 Raum für Incor-Messungen
202 Bedienungsraum
205 Raum für Kabelverteilung(E-Technik)
206 Treppenhaus
207 Raum für Rohrleitungen u. Lüftungskanäle
208 Raum für Steuerstabantriebe
211 Rohrkanal für Zu- u. Ablufleitungen
301 Rohrkanal Maschinenhaus
302 Raum für Zwangumlauf
303 Raum für Zwangumlauf
304 Raum für Zwangumlauf
305 Bedienungsraum
306 Ringspalt Reaktor
307 Rohrkanal Maschinenhaus (ausserhalb Reaktorgebäude)
401 Treppenhaus u. Bedienungsraum
402 Brennstoffschadenfeststellungssystem
403 Brennstoffschadenfeststellungssystem
406 Raum für Hilfsysteme
407 Kabelboden
408 Brennstofflagerbecken
410 Zugangmaterialschleuse
502 Raum für Reaktorinstrumentierung
503 Raum für Reaktorinstrumentierung
504 Raum für Schaltanlagen(E-Technik)
505 Brennstoffbeckenkühlsystem
601 Treppenhaus u. Bedienungsraum
602 Raum für Primärreinigungssystem
604 Raum für Ventile
605 Rohrkanal
606 Raum zur Lagerung neuer Brennstoffelemente
607 Lagerbecken für Abscheiter u. Trockner bei Brennstoffwechsel
608 Raum zum Fluten bei Brennstoffwechsel
701 Treppenhaus u. Bedienungsraum
702 Raum für Ionisationskammerantriebe
703 Raum für Ionisationskammerantriebe
704 Raum für Leerlaufsystem
901 Hauptbedienungsraum

1 Notkondensator
2 Arbeitsbühne
3 Kühlwasserbehälter
4 Vergiftungslösungsbehälter
5 Aufzug
6 Reaktorgefäß
7 Leerlaufkühler
9 Vergiftungspumpen
10 Ionisationskammerantriebe
13 Nichtregenerativwärmetauscher
14 Regenerativwärmetauscher
15 Lagergestelle für neue Brennstoffelemente
16 Ionisationskammerrohre
17 Brennstofflagerbeckenkühler
18 Brennstofflagerbeckenumwälzpumpen
19 Schildkühler
20 NS-Anlage
23 Materialschleuse
25 E-Kabelkanal
26 Sekundärdampferzeuger
27 Zwangumlaufpumpe
28 Absperrschieber
29 Umwälzpumpenkühlsystem
30 Schleuse
31 Dekontiminationsbecken
32 Hängelaufkatze 3t
33 Hängelaufkatze 25t
34 Regelstabantriebe
35 Regelstabantriebsystem
36 fahrbare Arbeitsbühne
37 Aktiver Sammeltank
38 Aktive Sammeltankpumpen
39 Flutbehälter
40 Raum für Incor-Messungen
41 Antriebe für Incor-Messungen
42 Leerlaufpumpen
43 Sumpfpumpe
44 Zwischenkühler
45 Zwischenkühlpumpe
46 Druckerhöhungspumpe
47 Steur- u. Koaxkabelverteilung
48 Steur- u. Koaxkabelverteilung
49 Gebäudenotkühlwärmetaucher
51 Transportbehälter für verbrauchte Brennstoffelemente
52 Transportwagen
67 Lagerregale für Brennstoffelemente, Steuerstäbe
68 Kanäle
69 Abstreifmaschine
70 Kran 75t

Zchng. Nr.: 1007-02-37
1007-02-38

250 KRB

Reaktorgebäude
Längsschnitte

1:100

AEG
KEA

Zchng. Nr. 1007-02-36

7. S.E.P., THE DUTCH NUCLEAR POWER PLANT PROJECT

by

J.S. TERPSTRA

LIST OF FIGURES

7. S.E.P., THE DUTCH NUCLEAR POWER PLANT PROJECT

7.1. Introduction

The contract between Euratom and Samenwerkende Electriciteits Productiebedrijven (S.E.P.) of Arnhem was signed at Brussels on Tuesday, 2 April 1963. This contract governs Euratom's participation in the construction and operation of a nuclear power plant to be built by S.E.P. This nuclear power plant, the first in the Netherlands, will presumably be situated in the municipality of Doodewaard in the Betuwe district, where S.E.P. has an option on a site in the unembanked alluvial land along the right bank of the river Waal (see location map). Among the many advantages which this location offers are the certainty of a constant supply of cooling water, good building land, a central position in relation to the Dutch national grid, moderate population density, etc. It is hoped that operation of the plant can be commenced in the spring of 1968.

The design of the nuclear part of the plant is being carried out, with the assistance of S.E.P., by IGEOSA (International General Electric Operations S.A.) of Geneva, with whom the relevant contract has already been concluded. One of the main stipulations of this contract is that both Dutch firms and the nuclear industry of the Euratom countries shall be allowed to collaborate on a large scale in the construction of the plant and the manufacture of its components, including the fuel elements and control rods. S.E.P. itself will thus be acting as a structural engineering enterprise and will obtain from General Electric only the necessary know-how for the design and start-up.

The signing of these contracts means that the plans for the building of an atomic power plant, which have been under study at S.E.P. for many years, have finally taken on a highly promising concrete form.

7.2. Purpose

As already stated, the express desire of S.E.P. in embarking upon this project is to stimulate the initiative of the European nuclear energy industry. Furthermore, the reactor core will be equipped with a large number of additional measuring instruments in order to enable extensive experiments to be carried out in the fields of nuclear physics and hydraulics. Thus, besides the parameters which are customarily measured in a reactor of this type, a constant watch will be kept on the neutron-flux distribution, the energy spectrum of the neutrons and the kinetic/hydraulic behaviour of the coolant. All these measurements are directed in particular toward the development of better and cheaper fuel elements with a high burn-up rate.

Moreover, by carrying out the entire construction itself, S.E.P. will obtain an excellent overall view of the cost breakdown among the various reactor components.

7.3. Reactor

The main element of the plant is a single cycle boiling-water reactor with a thermal capacity of 165 MW. This heat is removed by light boiling water which circulates through the fuel elements by natural convection. The saturated steam thus produced (at a pressure of 70 atm.abs.) is led into a turbine which drives a 50 MWe generator. The exhaust steam which condenses in the condenser is preheated to 125°C as feed water and pumped back into the reactor. The quantity of steam produced is 256 tons/hour. Of the electricity generated, 2.5 MW is used for driving the auxiliary machinery, so that the remaining 47.5 MW can be fed into the national grid.

The core is made up of 156 fuel elements, each of which comprises 36 fuel rods. Four of these elements are equipped with the instruments mentioned above. The fuel rods consist of slightly enriched (2.2 to 2.5%) uranium dioxide pellets with a stainless-steel or Zircaloy cladding. The cooling water also serves as a moderator and reflector for the neutrons. All the fuel elements are provided with a "chimney" to promote circulation of the coolant. The core also contains 37 cruciform control rods of stainless boron steel, inside which are 60 neutron-absorption tubes packed with B_4C granules. One of the control rods will be equipped with a special drive, enabling oscillating movements for the study of the transient behaviour of the reactor. The total weight of uranium in the core is 13,000 kg and the specific power is 30 kW per litre of core volume. The estimated average burn-up rate of the fissile material is 13,000 MWd per ton of uranium.

The cylindrical reactor vessel has an internal diameter of 3 m, a length of 11 m and a wall thickness of 10 cm. In addition, the inner wall is lined with a stainless-steel jacket 12 mm thick. The weight of the vessel, including the cover, is 120 tons. Besides the actuel reactor core, together with its supporting structure, the vessel also contains a steam/water separator and a steam drier. The reactor building is equipped with a pressure-suppression system, which in the event of an accident condenses any steam escaping from the reactor.

7.4. Euratom - S.E.P. contract

The construction cost of the power plant is estimated by S.E.P. at N.fl.95,000,000. The maximum share of Euratom in the construction and operating costs will be N.fl.18,100,000, the break-down of which is as follows:

- N.fl.1,400,000 as an allowance towards the losses incurred during the three-year initial operating period;
- N.fl.12,000,000 for reactor components manufactured in the European Community;
- N.fl.4,700,000 for the fuel elements, insofar as they are manufactured by European firms within the European Community.

In return for this financial aid S.E.P. will place at Euratom's disposal the knowledge acquired during construction and operation of the plant. Furthermore, Euratom has the right to second to S.E.P. members of its own staff and any third parties it wishes to invite. A liaison committee will also be set up in which Euratom and other interested parties will discuss the technical and economic aspects of the project with S.E.P. experts.

7.5. Status of activities

The project has been divided into three phases, after each of which a decision is made, whether to continue or not.

The first phase, comprising the study of the preliminary project, has already been terminated. During the second phase, which is now being carried out, a detailed study of the plant will be made. Phase three comprises the construction period.

Following the signing of the contracts with General Electric and Euratom, S.E.P. has made a vigorous start on the solution of the technical problems which are to be cleared during the second phase. These problems concern for example:

- the cladding of the fuel elements
- the pressure vessel
- the steam separating and drying devices
- the charge and discharge machine
- the shieldings
- the conception of the station buildings

The latter point poses a particular problem with regard to the situation of the plant.

In San José, California, a team of S.E.P. engineers is at present working with the engineers of General Electric on the further elaboration of the nuclear part of the power plant. Another scene of great activity is the SEP/KEMA head office at Arnhem, where every effort is being made on the finalisation of the plans.

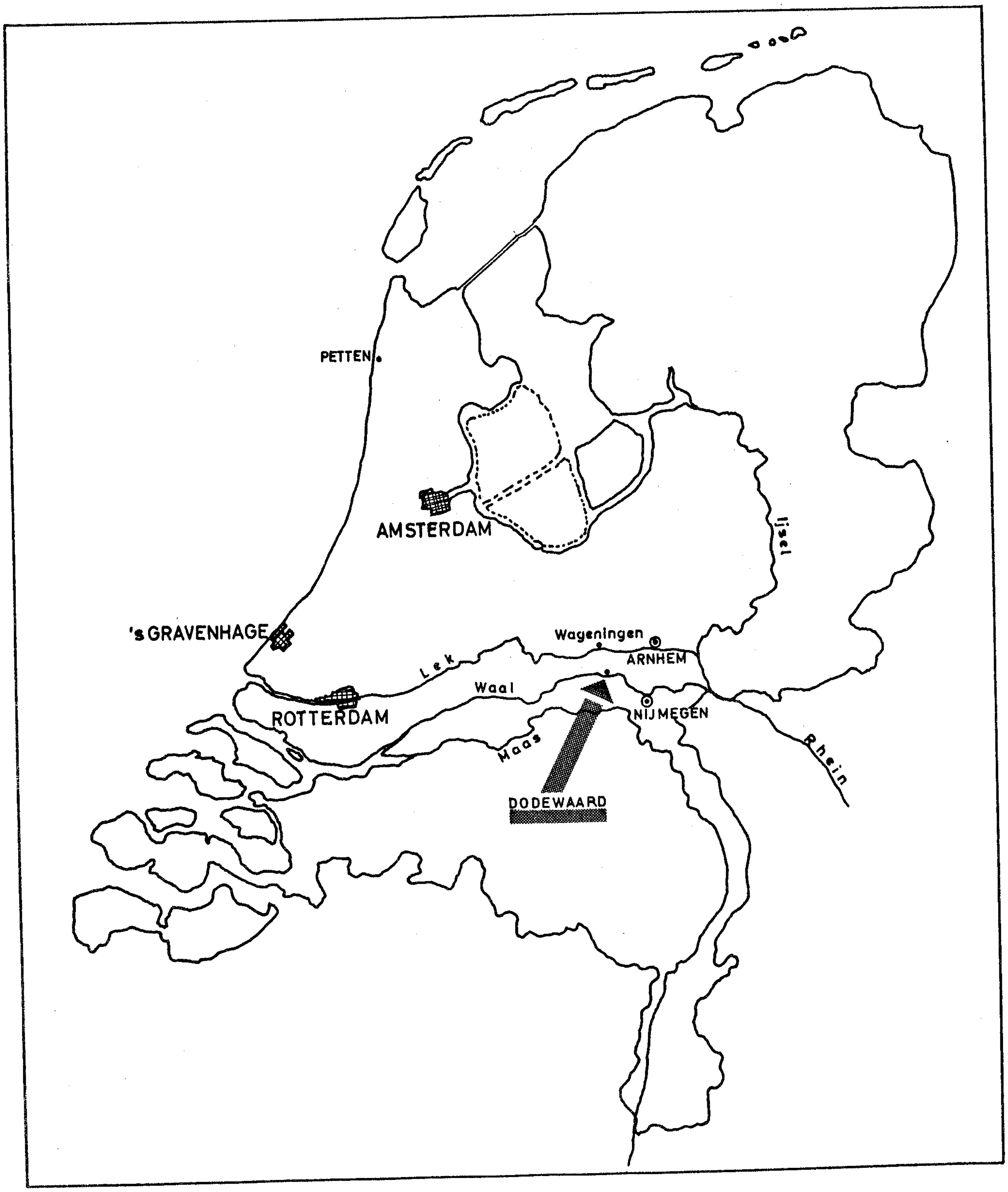
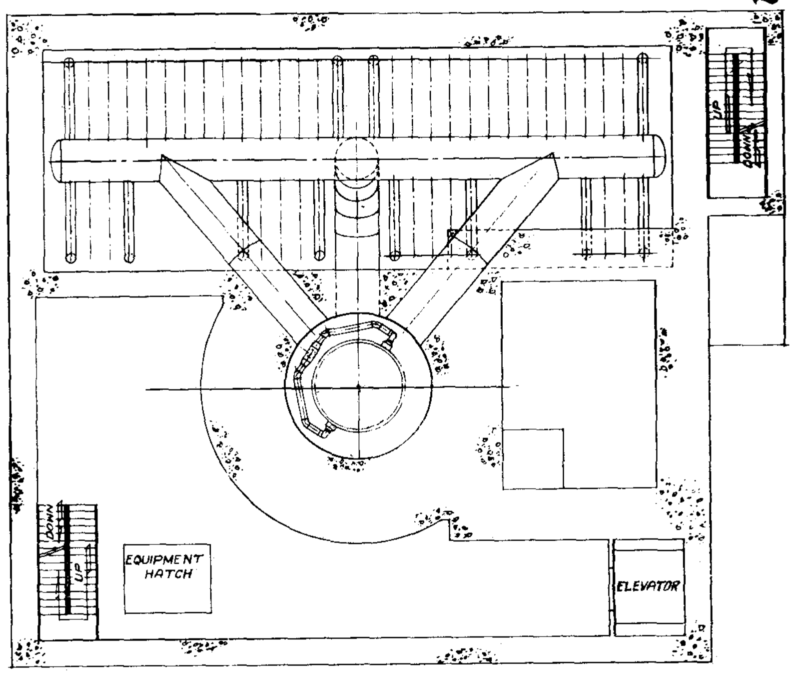
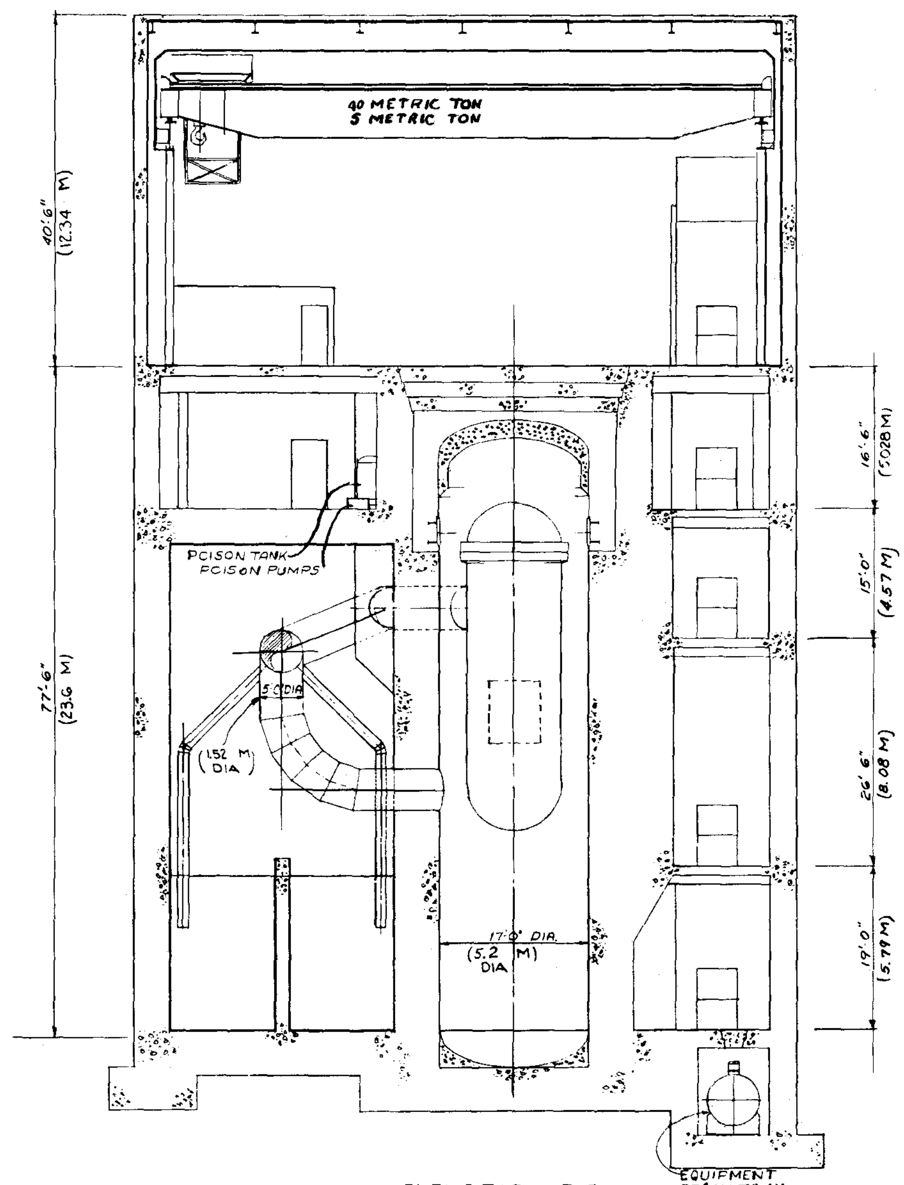


Fig. 7.1. Location map

2

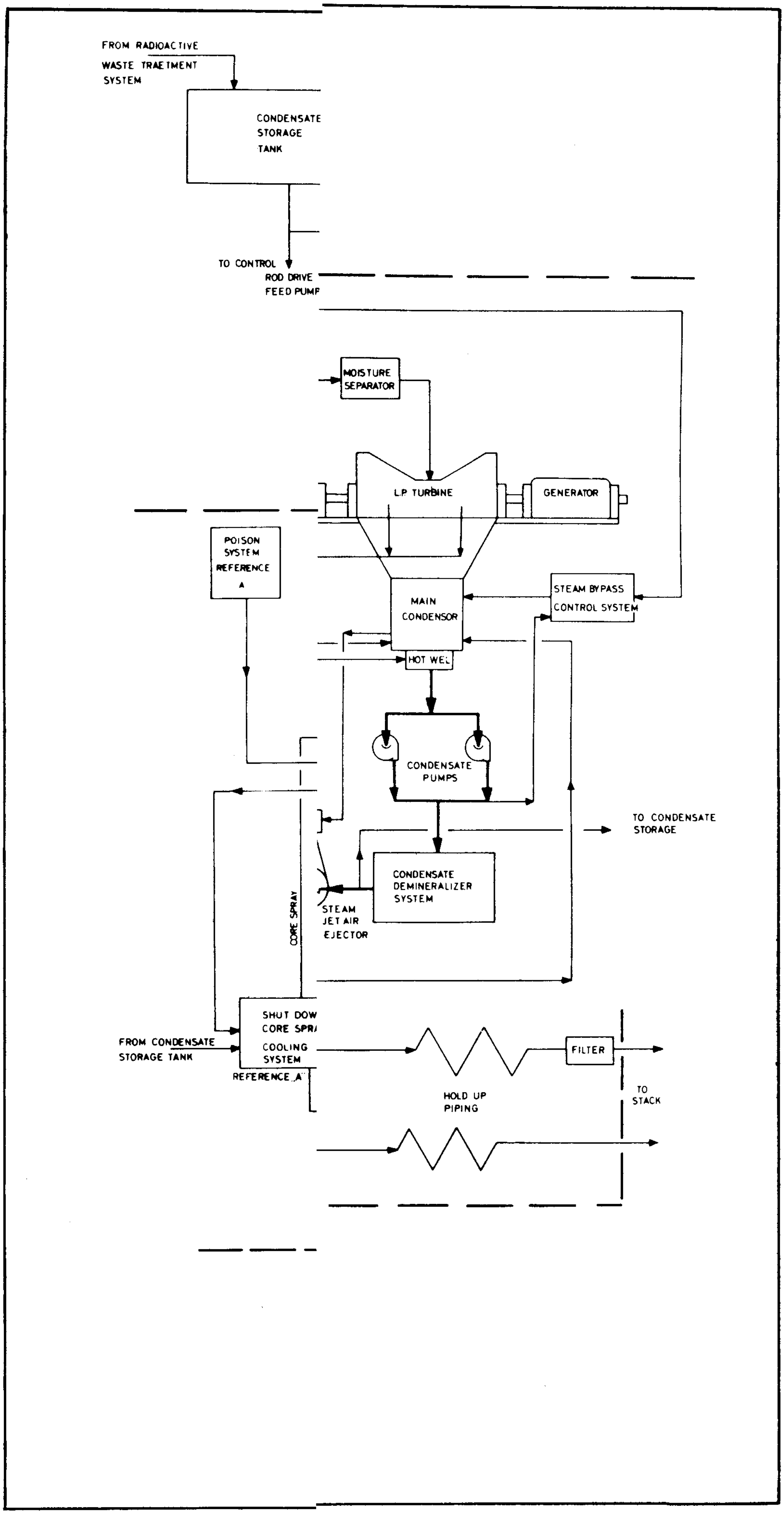


PLAN No3



ELEVATION B-B

Fig. 7.3. Reactor building



FROM RADIOACTIVE
WASTE TRAETMENT
SYSTEM
CONDENSATE
STORAGE
TANK
TO CONTROL
ROD DRIVE
FEED PUMP
MOISTURE
SEPARATOR
L.P TURBINE
GENERATOR
POISON
SYSTEM
REFERENCE
A
MAIN
CONDENSOR
STEAM BYPASS
CONTROL SYSTEM
HOT WEL
CONDENSATE
PUMPS
TO CONDENSATE
STORAGE
CONDENSATE
DEMINERALIZER
SYSTEM
STEAM
JET AIR
EJECTOR
CORE SPRAY
SHUT DOW
CORE SPR
COOLING
SYSTEM
REFERENCE A
FROM CONDENSATE
STORAGE TANK
FILTER
HOLD UP
PIPING
TO
STACK