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Reactor Group

CONCEPTUAL DESIGN AND ASSESSMENT OF A HELIUM COOLED 2500 MWE MOLTEN SALT FAST REACTOR WITH INTEGRATED GAS TURBINE PLANT

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Technical Assessments and Studies D

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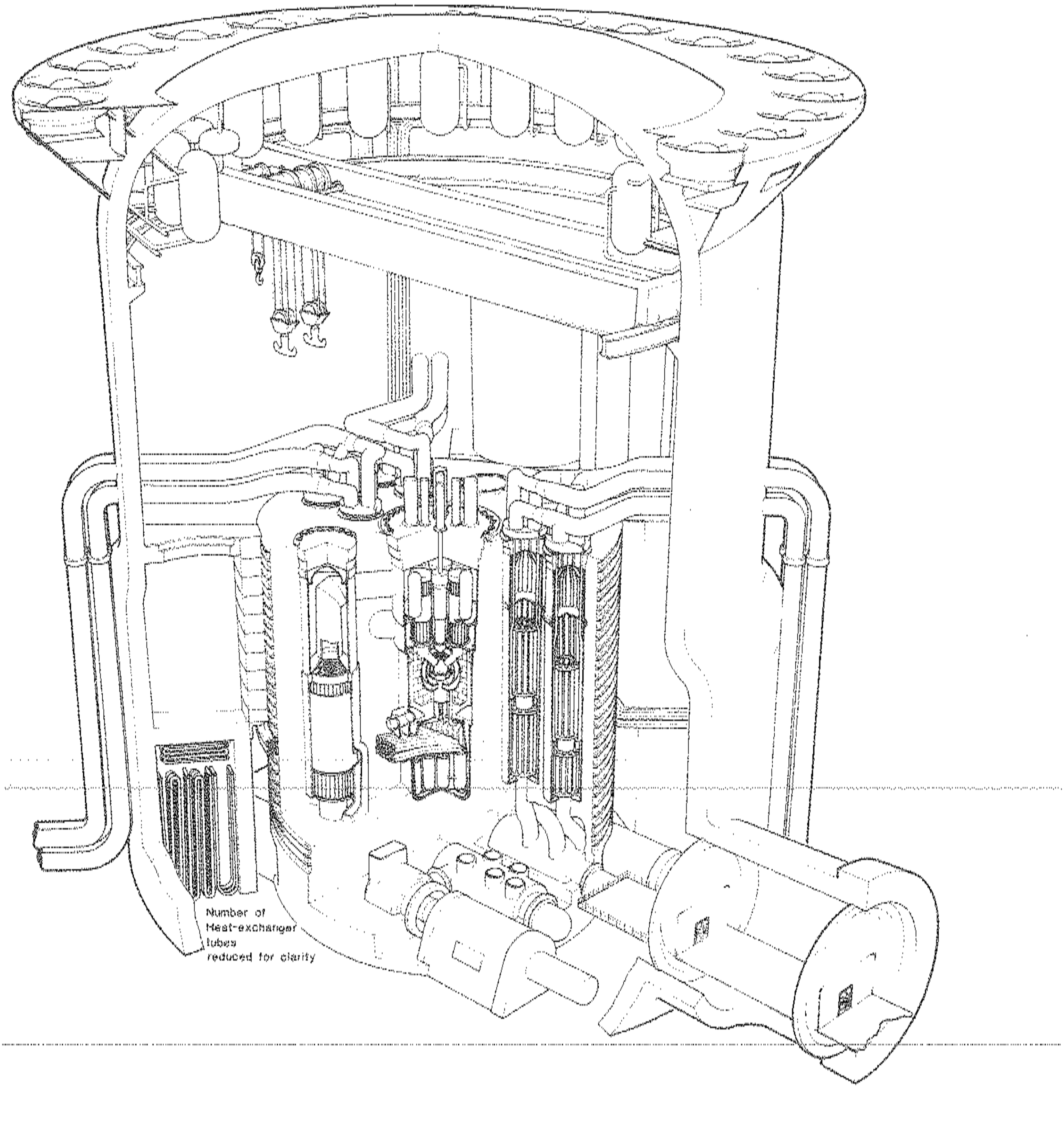
Atomic Energy Establishment, Winfrith, Dorchester, Dorset. August 1970



2500 MWe Helium Cooled MSFR with integrated gas turbine plant

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CONCEPTUAL DESIGN AND ASSESSMENT OF A HELIUM COOLED 2500 MWE MOLTEN SALT FAST REACTOR WITH INTEGRATED GAS TURBINE PLANT

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SUMMARY

A description is given of a conceptual design study of a high temperature MSFR cooled by helium and with a four-set gas turbine plant, all integrated within a prestressed concrete vessel. It demonstrates the potential for shop fabrication leading to a rapid-to-build plant, with facilities for renewal of all plant for extended life. Safety principles are discussed and sufficient design data are presented to enable a preliminary cost estimate to be prepared.

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INTRODUCTION

Previous work on Molten Salt Fast Reactors (MSFR's) as reported in AEEW-R 956[1.2] dealt with the basic physics, chemistry and corrosion aspects of molten salt fast systems together with a first stage engineering assessment. The engineering design was then concerned mainly with a lead cooled system, although other coolants were considered and a preliminary helium cooled design for steam generation was shown. Subsequent work indicated that the choice of coolant for a fast system was indeed restricted to lead or helium if good compatibility with molten salt or steam in the event of heat exchanger leakage was to be a main criterion.

The high temperature potential of molten salt, and the compatibility of helium led naturally to the investigation of a molten salt system with a gas-turbine power plant, as this appeared also to have prospects of giving an integrated plant with compact units of sizes suitable for prefabrication thus leading to shorter erection times and lower costs.

This report is presented in four parts - the first summarises the principles of the concept, the second describes the conceptual design in more detail, certain safety aspects are considered in the third part, and the final part discusses the findings, draws conclusions, and makes recommendations for future work.

PART I

PRINCIPLES OF THE CONCEPT

1.1 Choice of Secondary Coolant

A review of secondary coolants including gases, liquid metals and molten salts concluded that since some leakage in heat exchangers and steam generators and coolers was inevitable, the choice of intermediate coolant should be determined by its compatibility with both the heavy metal salts to avoid processing and precipitation problems and with water/steam to avoid corrosion difficulties. Certain of the candidates (salts) which were otherwise attractive were rejected because of concern about the enhanced corrosion local to the site of water/steam leakage which could rapidly increase the leak rate. On this basis the choice of coolant lies between lead and helium; they are both relatively inert and there is a choice of materials available which meet the corrosion requirements. Lead cooling would offer the possibility of fuel inventory at the lower end of the range when used in conjunction with salt running at high temperatures. There is however a mismatch with temperatures required for steam generating conditions which necessitates considerable recirculation of cooler lead in a loop round the steam generators to reduce thermal stresses and control is required to ensure the correct mixing with the hotter lead from the salt heat exchangers. Considerable cost would be incurred using dearer materials for the high temperature region of the lead circuit and steam generators.

It was further recognised that despite the safety attractions of a low stored energy coolant circuit, reluctance to develop a lead technology and the costs involved could prejudice the chances of introducing MSFR's.

Although therefore there are safety questions to be answered with a pressurised coolant, it was considered that helium cooling should be studied bearing in mind expectation of the development of its use as a coolant for HTR's. Furthermore the need for high salt temperatures to give low fuel inventory matches up with helium temperatures suitable for gas turbine plant, which in turn gives components of similar scale and construction to that of the reactor. The development of suitable closed circuit gas turbine plant for use with helium has now started and it seems reasonable to hope that such plant would be available before any MSFR is introduced.

1.2 Design Outline

1.2.1 The figures show the conceptual design of the reactor and integrated 4 set double shaft gas turbine plant within a prestressed concrete vessel. An outer containment building houses this and all the auxiliary plant with space for maintenance on, or replacement of, all main plant items including, in the extreme case, the reactor vessel itself.

The principal parameters are summarised in Table I, detailed parameters are given in Table II. A cycle diagram is given in Fig. 7 and a simplified overall plant flow diagram in Fig. 9.

1.2.2 A core and blanket of similar configuration to that considered for the indirect system reported in AEEW-R 956 (Design 3) was taken, with the same salt composition to give a similar nuclear performance, fuel inventory and doubling time. The thermal output has had to be increased by 10% (to 6600 MWt) to allow for the lower efficiency of the cycle compared with steam but only a marginal increase of fuel inventory is required because a higher salt outlet temperature has been used.

1.2.3 The maximum salt temperature was chosen as 1050°C. This allows some margin for transients, while still staying within desirable predicted stress limits for molybdenum at 1200 - 1300°C. The gross efficiency of the power cycle is 40.8% with helium at 61 bars nominal top pressure. The pressure and temperature were selected to be comparable to those at present proposed for HTR in order to take full advantage of the experience gained in this field, though higher working pressures could be contemplated. The high efficiency case (though requiring substantial recuperators) was chosen to keep down fuel inventory; furthermore it coincided with a cycle proposed by Maillet[1] for which details of the rotating machinery were given.

1.2.4 To overcome the objections to a pressurized coolant and to assist in giving low capital costs the whole helium circuit including reactor and gas turbine plant has been integrated within a prestressed concrete vessel, of 30 m. diameter and 35 m. in height, the reactor, heat exchangers and dump tanks occupying a central vault of 9 m. diameter as shown in Figures 1, 2 and 3. A pressure balance arrangement is used to ensure running with the maximum fuel salt pressure just below the helium gas pressure so that all leakage is normally that of gas into the active circuit and reactor vessel scantlings can be reduced. Providing it can be shown that the helium over-pressure system can be engineered with a high degree of reliability, it can be claimed that under operation with very small local failures there should be no leakage of active salt into the secondary coolant helium; and that even with a more appreciable leakage in the primary salt circuits (e.g., a weeping heat exchanger tube joint) operation could be continued until the inflow of helium reached the full capacity of the off-gas cleanup plant or active helium storage.

1.2.5 Considerable thought has been given to the dump system. A rapid dump is required to reduce the amount of salt leakage in case of the failures already discussed and also to reduce the overheating that will occur as the large volume of fuel salt in the heat exchangers and pumps, drains through the core maintaining a critical condition during this time. There is also considerable delayed neutron heating as well as fission product decay heat. A dump system has been evolved in which heat is removed using a natural circulation NaK coolant and which will, at peak, take up and transport 600 MW heat from the dump tanks and on a continuous basis dissipate 200 MW through isolating water boilers to air coolers with guaranteed supply driven fans. The difference between initial heat transport capacity and the continuous dissipation capability is taken up by increase in temperature of the NaK including that of a cool reserve.

Increased complexity has been required also to keep the fuel and the blanket salts separate to avoid the need for time consuming (and costly) chemical separation of Pu in the event of inadvertent dumping which would otherwise mix fuel and blanket salt.

1.2.6 As before, molybdenum or TZM is proposed as the material for the high temperature salt regions and for all salt heat exchanger tubing. Corrosion experiments are required to endorse the suitability of these materials at the high temperatures proposed. The working temperature of a large proportion of the outer boundary of the primary (salt) circuit has been kept to 650°C by suitable design, to allow nickel alloys, such as Hastelloy N or possibly stainless steel, to be used for the bulkier components.

1.3 Fuel Inventory

Initial fuel costs and doubling time are both dependent upon the amount of heavy metal inventory and it is a major design problem to keep this low. In previous studies it was found that even with lead cooling (giving high heat transfer coefficients) the external fuel inventory associated with the cooling circuit is about equal to or higher than that in the core. Clearly with helium the problems are likely to be greater. A large proportion of the external inventory is associated with the ductwork connecting heat exchangers and pumps. There is a certain configuration of heat exchanger, determined by helium flow conditions and a reasonable salt pumping power, which corresponds to a minimum length for connection to core and pumps. With this limitation, the only way in which inventory can be kept down is to reduce the flow cross section areas. This is done by reducing volumetric salt flow rates by employing a large salt temperature range and then using duct velocities up to the maximum believed to be consistent with vibration and pumping power limitations. The maximum fuel salt temperature of 1050°C proposed leaves some margin within the strength limit of irradiated molybdenum or its alloys, while the minimum temperature of 650°C is sufficiently above the freezing point of the salt (570°C) to prevent freezing on the heat exchanger surfaces. The layout of the heat exchangers and pumps is a compromise between minimum inventory and access for installation and removal or maintenance. The two stage semi-axial pumps are arranged to save inventory by incorporating them in the relatively long return ducts.

Although it is not of such prime importance it is also desirable to reduce the blanket inventory due to the cost of separated Cl_{37}, which is required to enhance the breeding gain; means of improving the blanket performance giving reduced salt content possibly by incorporating graphite in the outer regions or by optimisation would merit further investigation.

The high volumetric heat ratings which are needed to reduce inventory and which are attainable with fluid fuels lead to a compact reactor design which in turn enables the gas turbine power plant to be closely integrated in a small prestressed vessel and the whole housed in a relatively small outer containment. Both the latter items are approximately equivalent in size to similar units for HTR of half the power output.

PART II

MAIN PLANT DESIGN

2.1 General

Figures 1, 2 and 3 show the elevations and plan sections of the reactor, intermediate heat exchangers (IHX) and gas turbine plant.

The reactor consists of a spherical core of 3 m. nominal diameter, surrounded by a 1 m. thick blanket zone with an outer graphite and heavy metal reflector. The nuclear performance will be similar to that of the original lead cooled reference case described in AEEW-R 956, adjusted for a gross output of 6600 MWth which is 10% higher due to a lower net efficiency and allowance for a small design margin. Fuel salt, heated by passage through the core vessel to 1050°C, passes to four IHX tube bundles sited above the reactor. It returns at 650°C through four semi-axial two stage pumps to the outer core shell. The blanket salt is circulated by four small pumps to a single central IHX. The salt inlet temperature is 860°C, the outlet 650°C and, using a heat exchanger of the same dimensions as that for the fuel salt, blanket power can be up to 15% of the total output.

The helium coolant at 61 bars nominal pressure and 415°C inlet temperature passes down between the tubes of the IHX and out into the lower plenum around the IHX from which the heated gas at 850°C passes to four sets of HP turbines driving the LP and HP compressors. It then enters the separate shaft LP turbines driving the 660 MWe alternators and exhausts to the shell side of the recuperator where it heats the return flow to the IHX. The helium is precooled before entering the low pressure compressors (LPC) and intercooled before being compressed in the high pressure compressor (HPC). The gas returns to the upper IHX plenum through the tubes of the recuperator.

Four separate gas turbine plants are used which give a unit size well matched to a compact layout and provide sufficient redundancy of plant to continue substantial power production and reactor cooling in case of failure of a major component. The junction of all four helium circuits at the plenums will reduce the uneven salt temperatures due to loss of a helium compressor but it has to be recognised that it means complete depressurisation of all circuits in the event of a rupture.

The whole system is incorporated into a prestressed concrete vessel with vertical layout and access for the IHX, salt pumps, reactor, recuperators, intercoolers and precoolers. The HP turbine/compressor units and the separator LP turbine/alternators are arranged horizontally at the base of the vessel. All rotating parts and fixed blading together with bearings can be removed as units, and all plant can be handled within the containment building and replaced if necessary.

Total fuel and blanket salt pumping power is 49 MWe and there is 40 m^3 of 40/60 mol % (Pu+U) Cl_3/NaCl fuel salt in the primary circuit containing 10.5 te of Pu239 equivalent initial inventory.

2.3 Reactor

The general layout and principles of construction of the reactor are apparent from the drawings (Figs 1 and 2). Special features are mentioned below.

The reactor and whole primary circuit are situated in the central vault within the prestressed concrete vessel and the helium pressure of the cover gas in the salt header tanks will be balanced in such a way that the coolant helium pressure is slightly higher than the pump delivery pressure to ensure that any leakage is from the helium into the salt. As the greater part of the salt circuit pressure loss occurs across the IHX tubing, the main vessel containing the core and blanket and its cover plate with lower IHX tube plates are subject to an internal pressure slightly below pump delivery pressure, and will have little differential loading. The maximum (external) differential pressure occurs at the top headers of the IHX and at the pump inlet and is equivalent to 2.4 MN/m^2 (350 psi).

Thus the scantling of the primary salt circuit can be lighter and the integrity enhanced both by operating at lower pressure stresses than if the full 60 bars (6 MN/m^2) had to be taken across the tube plates and by minimising the amount of primary circuit material working under tension. In the event of a rupture in the helium coolant circuit the small volume of cover gas needed for the salt primary circuit proper can be made to follow the depressurisation closely, but because of their larger volumes special measures will be required to simultaneously depressurise the dump tanks. This method avoids having to design the reactor vessel for full helium pressure in case of severe leakage of the helium circuit which would be necessary if the cover-gas was not simultaneously depressurised.

The whole reactor cell within the PCV will be heated to 600°C, probably by preheated helium circulation, (a) to maintain it above the salt freezing point for initial filling, draining and for low power running, and (b) to allow instrumentation and small auxiliary circuit lines to run hot without the need for individual trace heating. The majority of the reactor vessel and primary fuel and blanket salt circuits can be run at about 650°C enabling nickel alloys such as Hastelloy N or stainless steel to be used for the heavier components by arranging where necessary for the walls to be cooled by return salt.

The core vessel can take the form shown in Fig. 1 where the box-shell construction of the outer core zone provides the downward flow passages carrying the incoming fuel salt , and forms a vibration resistant structure with economy in the use of metal to reduce neutron absorption. The physics calculations have been based on there being a total equivalent thickness of 1 cm of molybdenum between the core and the blanket. The pattern of ports in the lower region must be arranged to give good flow distribution and would require experimental modelling. Detailed stressing will also be required when the hydraulic loading has been established more fully.

Similar principles would be followed for the blanket vessel except that there is not so much need to economise in materials to reduce neutron absorption. It is possible that the blanket performance can be improved and inventory reduced also (important if separated Cl_{37} is used) by modifying the nuclear arrangement. Reed(2) has suggested using 50% by volume of graphite in the outer 0.5 m. thickness of the blanket to displace salt without affecting the breeding performance materially.

Around the blanket but within the main reactor vessel there is space for further graphite and shielding which could be cooled by a small flow of low temperature blanket salt returning from the IHX. Some further use of high density shielding material outside the reactor vessel is probably desirable to reduce activation and assist remote handling.

The arrangement of the core and blanket vessels minimises expansion problems by allowing the internal components to move as freely as possible both radially and axially. The reactor vessel is supported at its top cover plate, the vessel and internal components can expand freely downwards and the pump barrels with top IHX tube plates will expand upwards.

2.3 Intermediate Heat Exchangers (IHX)

The fuel salt heat exchangers consist of four tube bundles mounted above the reactor vessel, each containing 12,600 molybdenum tubes 8 mm. outer diameter and 6 mm. inner diameter and of 6 m. length. The effective heat transfer area is 1620 m^2/bundle. The tubing is spaced on a square pitch with a pitch/diameter ratio of 1.9 to reduce the helium pressure drop at the inlet and outlet regions. The tubing is roughened on the outer surface to enhance the helium heat transfer coefficient by 50%.

The blanket salt heat exchanger will be of similar form. To match the blanket power as it rises to reach the design steady state limit (up to 15% of total reactor power) the helium flow is adjusted to avoid over-cooling the salt by means of a sleeve valve at the tube bundle gas inlet.

Desirable features of the heat exchangers are: low fuel hold up (including any associated duct work carrying salt), ease of replacement or repair and high reliability with long life.

Low inventory within the IHX itself is achieved by a compact overall layout, small diameter heat exchanger tubes and high salt velocities of 8 m/s in the tubes. With the layout adopted, the ducts from the core to IHX are reasonably short, but to improve access for maintenance and to reduce the diametral space within the PCV, the return ducts are longer than if the heat exchangers had been disposed radially around the core. The effect of this has been offset to some extent by employing semi-axial pumps situated in the return ducts. High duct velocities of 10 m/s are proposed to reduce salt inventory.

The tube bundles (and pumps) may be replaced complete as the vertical units make possible access to all fixings from above. Sufficient space has been provided for locating arrangements so that remote handling tools can be used. Prior removal of the top header connections would be required. It may just be possible to plug tubes in situ if a small diameter long length remote plugging tool can be developed, e.g., explosive plugging, but this will depend on the tube weld features.

Important features in relation to reliability (in addition to corrosion/erosion and similar materials problems) are the ability to deal with steady and transient thermal stresses and avoidance of adverse vibrations. The top headers of the tube bundles are supported off the core pump barrels which are free to expand upwards. The differential movement relative to the pumps of the IHX tubing, which is possibly made from different material and running hotter, is accommodated by sine wave bends incorporated in the tubing in one or both of the zones where the helium enters or leaves the bundle.

This use of sine wave bends follows the recommendations of the EBASCO[9] investigation into intermediate heat exchangers. Locally high thermal stresses could occur at the headers and it seems possible to consider welding the tubing to a relatively thin molybdenum header, insulated by a stagnant gap and supported from a thicker header support made of less expensive material. The tube bundles are not restrained by the "shell" side of the heat exchangers which in this arrangement acts solely as a flow channel with no direct connection to the bundles.

The form of location and support of the small diameter tubing to avoid vibration induced fatigue and fretting requires considerable investigation. Offset grid systems may be a possibility.

Small leaks in the heat exchangers can be tolerated, as there should be no local corrosion or erosion to enhance the leak rate and as leakage will be into the salt thus minimising the possibility of contamination of the helium circuit. The highest acceptable leakage rate during operation will depend upon the capacity of the salt cover gas clean up plant. The size of leak will depend on the differential pressures at the point of leakage, and could clearly be also affected by the actual form of the leak. It would also be necessary to consider whether "back-contamination" of the helium side could occur.

2.4 Salt Pumps

Four pumps are provided for core salt circulation and an equal number for blanket salt so as to permit substantial power output even if one pump becomes inoperative. The permissible power with a failed pump will be a function of the flow redistribution which occurs in the core or blanket end of the back flow through the failed pump. Overheating of the salt (and correspondingly of the circuit materials) must be prevented. Changes in temperature patterns within the core would also affect reactivity and this is an effect which will need further study. It is of course necessary to avoid "dead" portions in the salt circuits because of the delayed neutron and fission product heating in the salt; it would be preferable therefore to avoid isolating valves, and also flow control valves if sufficient control can be achieved by pump speed variation (fluid diodes may be useful for this application). It will be recalled that the proposed method of power output variation is based on varying salt flow to keep approximately constant core mean temperatures.

The electrical drive units for the pumps are mounted outside the PCRV to improve accessibility. The pumps themselves are of the vertical semi-axial type developing 2.4 MN/m^2 (350 psi) over 2 stages. The power requirement at 70% hydraulic efficiency is 9.4 MW (1260 HP) for each fuel salt pump and 2.9 MW (390 HP) for each blanket pump (total circulation power 49 MW). Detail design of the impellers has not been carried out, but full power speeds of 1500 RPM for the fuel salt pumps, and 3000 RPM for the smaller blanket salt pumps would be typical. The hydraulic performance of pumps handling hot, dense (SG $\sim$3.3) liquids would need study, as would bearings running in molten salt (salt temperature 650°C) or in a gaseous (helium) atmosphere which might be laden with salt "sols" and volatile fission products.

In the ORNL study[10] proposals were made for a layout in which a free surface was formed at the pump suction for extraction of fission product gases and volatiles from the salt, using bubble generators to promote this. In the MSFR it seems better to carry out this "de-gassing" in a separate loop branching off the primary circuit, as there is only very limited space around the pumps and access is difficult.

2.5 Gas Turbine Cycle and Plant

The gross cycle efficiency of 41% was selected for this first study as a compromise between fuel inventory, doubling time and recuperator size. The recuperator effectiveness of 0.83 is modest by HTR direct cycle standards but might be considered unnecessarily high for a fast reactor. For example, studies of direct cycle GCFR's have shown that with low fuel costs, there is an incentive to reduce capital costs, and efficiencies down to about 33% are acceptable. However in the molten salt fast reactor there is a need to keep down initial fuel inventory both for doubling time reasons and because it accounts for over half of the fuel costs with plutonium at £5/gram. This will tend to favour higher efficiencies. There clearly is a need to optimise this interaction but for the present the view has been taken that the size of recuperator selected is on the generous side and as this could be fitted in any reduction in size can easily be accommodated.

It has the further convenience that details of the cycle (Figure 7) and sizes of rotating machinery can be matched to the Meillet proposal already referred to. Four sets are used, each comprising the following:-

(a) A 3 stage HP turbine running at 4500 rpm driving a 6+1* stage LP compressor and a 10+1* stage HP compressor (* refers to a final centrifugal stage).

(b) A separate shaft 6 stage LP turbine running at 3000 rpm drives a 663 MWe gross output alternator.

(c) A precooler removing 600 MWth with 6100 m^2 heat transfer surface and an intercooler removing 580 MWth with 5230 m^2 surface using cooling water heated from 20°C to 35°C.

Items (c) are mounted vertically in the PCV with water connections at their upper ends so that they may be installed as self-contained units at a late stage in the construction programme and can more easily be removed for repair if this is necessary. The design is adapted from a Dragon Project[3] design with banks of internally water cooled U-tubes arranged in "box" formation around the central gas outlet duct. The incoming gas flows upwards in the outer annulus across the tubes and down the inner bore. To reduce the waterside pressure drop for the high flow rates required with the greater powers for MSFR the U-tubes of the DRAGON design are replaced with double sets of straight tubes in parallel. Separate 28 cm ID pipes are connected separately to distribution manifolds outside the PCV to reduce the area of flow for helium in case of rupture. The end closures in the PCV are duplicated to eliminate the possibility of large failures which would cause rapid helium depressurisation.

(d) A recuperator consisting of a vertical tube in shell heat exchanger with 36,000 tubes 15 mm od x 12 mm id totalling 17,200 m^2 heat transfer area and exchanging 1080 MWth. High pressure gas returning to the IHX passes upwards through the tubes in counterflow to the LP turbine exhaust gases on the outside of the tubes. To avoid complications in the duct layout within the PCV and interference with the prestressing cables the flow from the low pressure turbine passes up an annular shell around the recuperator to reach the top entry. The recuperator can be installed or removed independently of the remaining plant and is also provided with double closures.

Calculations have shown that the pressure drop specified in the Maillet proposal can be met with the size of plant and ducts shown on the drawings and for the conditions given in the parameter lists.

Control studies have not been carried out on this system; the interaction between load following by the turbines and the need to adjust reactor output by a temperature perturbation followed by a return to the same mean temperature condition to maintain criticality is complex and further studies will be needed to see if the requirements can be met. A bypass valve across the LP power turbine is possible, and/or a throttle valve may be fitted at inlet to the LP turbine. Methods involving cooling of the gas leaving the HP turbine with gas from the HP compressor to protect the recuperator and reduce the helium pressure have been advocated to aid control but would need further consideration for incorporation into this system.

2.6 Prestressed Concrete Vessel (PCV)

It was decided to adopt a prestressed concrete vessel with an integral gas-turbine primarily for safety reasons, the principal ones being:-

(i) The active fuel and blanket salts are contained in the PCV acting as a secondary containment to the primary containment of the reactor vessel and primary circuit. The helium pressure can be slightly above the salt pressure to avoid contamination from minor leaks.

(ii) The pressure differentials across the primary circuit are reduced to the salt pumping pressure (see Section 2.2) at the critical points, e.g., the IHX tube plates.

(iii) The possibility of a large rupture in the helium circuit is regarded as incredible in a PCV provided that normal metallic closures are duplicated or flow restrictions are fitted. (The larger closures for the reactor, recuperator and possibly the coolers, will require special treatment so that they become an extension of the PCV concept with the loads taken in compression or by the prestressing tendons.)

Thus severe damage to the active salt circuits due to sudden depressurisation or missiles as the result of large ruptures is avoided and cooling will continue for a considerable period if smaller breaches of the helium circuit occur. (The additional protection of draining the salt to independently cooled dump tanks is described in the next section.)

The overall size of vessel required is relatively small and is not much larger than that required for a 660 MWe AGR such as Heysham and with the layout proposed, the mass of concrete is not inordinately in excess of that required for shielding and structural support. Furthermore a free standing gas turbine plant would require a larger containment building due to the space required for duct expansions, unless bellows expansion joints were employed.

An outline of the form of PCV proposed is given in the general arrangements and on Figure 6 which shows the main apertures and suggested routes of the prestressing tendons. The principles employed are similar to those of the Heysham PCV with circumferential prestressing to resist hoop stresses and part of the end loading, reinforced by longitudinal tendons around all large apertures with transverse tendons on the lower part of the vessel around the turbine apertures. Some diversion of the tendons from a straight path is required to avoid the various ducts and apertures but the amount involved seems reasonable.

No detailed stress calculations have been done except to provide sufficient area of tendons to balance the overall pressure loads, including full design pressure of 66 bars being exerted in a crack across the diameter of the PCV.

The form of liner and insulation will follow the practice being developed for AGR and HTR. Two layers of insulation with helium cooling flow between will probably be required for the hottest regions. Special provisions could be needed for areas where the decay heat from any leakage salt has to be dealt with. These are considered in Section 3.3.4.

2.7 Emergency Cooling and Dump System

2.7.1 General

Methods of removing delayed neutron and decay heat in the event of loss of main cooling but with an intact circuit can be divided into two types:-

(a) Provision of emergency cooling within the reactor vessel or primary salt circuits in some manner, or

(b) Draining of fuel and/or blanket salt to dump tanks provided with a separate reliable cooling system.

The former method avoids the necessity for dump valves and dealing with the effects of inadvertent dumping, but involves demonstration of a highly reliable circulation system and additional heat rejection circuits which raises many problems. Furthermore with a fluid fuel circuit leaks must be capable of being dealt with.

Method (b) has therefore been the one investigated and it offers the opportunity to provide a means of cooling completely independent of the reactor circuits. Inadvertent dumping is however a concomitant complication for this method. Separate dump tanks are needed for fuel and blanket salt to avoid having to reprocess the whole mixture (whereas for a major accident this might be acceptable); also dumping could occur from full power and this increases the heat removal capacity compared with accident cases. The necessary provision of separate tanks in a confined space is more complicated than if fuel and blanket salt could be drained into one large diameter vessel with a low H/D ratio.

The system proposed comprises:-

(i) Separate dump lines for fuel and blanket salt each with four quick opening valves to allow sufficient margin in case any one valve fails to open.

(ii) Separate fuel and blanket dump capacities each subdivided into 12 cylindrical tanks with the blanket tanks grouped towards the centre and the fuel tanks towards the outer regions of the space under the reactor to give a sub-critical configuration. (The layout shown is representative and more calculations will be required to give a rigorous demonstration of subcriticality especially at low temperatures; there would clearly be no problem in introducing absorbers between the tanks if this proved to be needed.) Overflow arrangements are provided between the core and the blanket dump tanks at two levels so that if a core or blanket drain valve fails to open flow will be distributed in all the tanks of the respective group (i.e., core or blanket). If the core/blanket membrane fails in addition to a valve failure flow can be distributed to all tanks by means of the highest overflow pipes. The cooling pipework and drain lines are free to move independently of the tanks, thus differential expansion stresses are minimized. These tanks are open-topped to make it easier to maintain pressure and flow balances, and are of a non-critical configuration. They are grouped together and contained in a large outer tank mounted in the lower part of the reactor cavity of the PCV (see (v) below). The individual tanks may be internally jacketed and cooled to reduce thermal shock and increase the strength.

(iii) A collecting lining forming a tundish to catch any salt escaping from the reactor vessel and primary salt circuits which will drain into the dump tanks through fusible plugs in the roof of the outer tank.

(iv) A natural circulation heat removal circuit filled with NaK which passes through U tube bundles in the drain tanks. All U-tube to manifold welds are situated above salt level. The NaK transfers the heat it removes (by radiation across an air space to prevent cross contamination) to water tubes in boilers situated in the upper part of the containment building. The water boils off at just above atmospheric pressure and passes to condenser/coolers on the outside of the containment building; these are air cooled by electrically driven fans supplied by stand-by diesel or gas turbine generators if necessary. Should failure of these generators occur, the water boiling off can be replaced through the fire main system.

(v) The outer tank surrounding the individual dump tanks is provided to contain the active gas and volatiles emanating from the open-topped tanks and to catch any leakage from the dump tank system. Cooling is provided in the lower part in case of such leakage. The gas pressure within this tank system is balanced with that in the salt primary circuit header tanks to allow gravity dumping and so is maintained at one or two bars below the helium coolant pressure. In case of loss of helium coolant pressure the dump tank/header tank helium will be vented to special dirty gas receivers. For a reasonable size of receiver, this can be done naturally to a pressure balance at around 17 bars which must be followed by pumping. A possible alternative is through a filter into the coolant circuit if it is considered this could be done without carrying through too much activity. An alternative layout which has not been explored but which could give improved access for maintenance is to locate the dump tanks in an extension to the PCV but extra fuel inventory and increased drainage times will be associated with the longer drain paths this would involve.

2.7.2 Performance of Emergency Cooling System

Estimates of the rate of depressurisation following a breach in the helium coolant/gas turbine circuit have shown that the time for pressure to fall to half working pressure is 27 seconds for a double ended 0.06 m^2 (28 cm dia.) breach with a discharge coefficient of 0.6. This corresponds to a helium precooler/intercooler CW circuit pipe or NaK dump cooling circuit pipe suffering total fracture and is taken as the limiting size for which a significant probability of failure must be catered for with a prestressed concrete vessel. It may be desirable to design both NaK dump tank or helium precooler/intercooler water circuits for full helium pressure so as to avoid arguments about breaches of these circuits. All larger apertures are doubly contained or flow restricted to below 0.06 m^2 equivalent or are extensions of the PCV with separate and redundant tensioning systems.

During depressurisation from the small apertures, cooling of the salt in the primary circuit will be maintained for some time by helium circulation; the most severe requirement for the emergency cooling system could well be that imposed by inadvertent dumping of the core whilst at power due to a control fault or mal-operation. Preliminary calculations indicate that, by making use of the heat capacity of the fuel salt and the NaK coolant, the "continuous" heat rejection duty required of the NaK-water boilers and of the air cooled condensers is about 3% of full power. Extra heat transfer surface must be provided in the dump tank coolers to allow for the higher peak rate of heat removal required in the early stages of the transient, as high levels of power will continue to be generated until the salt from the upper levels of primary circuit has drained through the core region. The reduction in power due to temperature rise of the salt in the core has been estimated in order to assess the short term heat dissipation capability required. The drain time for all the fuel salt will be about 14 seconds but it is assumed that cooling in the dump tank will commence before the end of this period (at about 10 seconds) by which time the heat generation will be equivalent to 3 full power seconds and the mean salt temperature will have risen from 920°C to 1125°C. To assist heat transport during the peak period a reserve of cool NaK is provided in header tanks so that the NaK temperature at inlet to the dump tanks does not exceed 30°C until 110 seconds after dumping.

Thorough mixing of the bulk of the fuel salt should occur during draining, thus it seems reasonable to use mean temperatures. The effect of residual cooling in the IHX, which has not been included in the calculations, will mean that cooler salt will reach the drain lines and dump tanks first, thus reducing thermal shock.

The effective temperature for NaK should be within design conditions of 300°C cool return temperature and 600°C maximum temperature at outlet from the dump tank coolers.

Removal of gamma heating from the core and blanket vessels, shielding, etc., after the salt has been rapidly drained has not yet been assessed as there is no practical evidence available on residual salt films, etc., and residual activity. It could prove to be necessary to provide an emergency cooling system for the core and blanket vessels, and should this be so, helium would be the preferred medium if the heat transfer rate can be made adequate.

2.8 Outer Containment

For the purposes of the initial assessment it has been assumed that the outer containment is required to be leak tight and will withstand a helium pressure (1.5 bars of differential) due to a breach in the PCV. There is the possibility of considering a vented containment with filters capable of trapping fission products and fuel salt in the form of smoke if leakage occurs, but a knowledge of the behaviour of salt released at high temperatures and condensation effects is required to predict performance. In either case a high standard of decontamination may be required, say about 10^{-2} or 10^{-3} overall, (including leakage effects) to deal with combined release from the primary salt and secondary coolant circuits in accident conditions if pessimistic release figures for Pu are taken.

The possibility of an internal steel lining supported by a reinforced concrete or prestressed concrete outer shell but with interspace connected to a second filtration system could be considered to give further integrity, as some form of insulation for a concrete shell will be required in any case.

An access feature of 9 m. diameter, possibly combined with a large entrance airlock is needed if the reactor vessel, outer containment for the dump tanks, and recuperators are to be removed and new ones installed complete.

2.9 Plant Layout

The layout of plant within the containment building is shown on Figures 4 and 5. Brief notes on the individual items are given below.

2.10 Core and Blanket Drain Tanks

It is desirable to provide drain tanks in addition to the dump tanks so that longer delay times may be imposed before salt processing occurs and in case the dump tanks must be emptied for servicing the tanks or reactor or other plant in the vicinity. The tanks may also be used for storage prior to initial filling or for topping up.

As the design of the drain tanks is not governed by the shape of the PCV nor the need to remove the considerable immediate decay heat, it is possible to provide a single "flat" cylindrical tank of 6.25 m. inside diameter for the fuel salt. The height has been limited to under 2 m. in order to prevent criticality occurring at the enrichments possible (this approximate estimate in limiting height will need further study, particularly for the low temperature conditions). Two similar tanks are required for the blanket salt and one or two more tanks can be fitted as spares. The salt can be stored at subatmospheric pressure to avoid contamination problems.

A total cooling capacity of 30 MW is probably adequate after a five day delay of fuel salt in the dump tanks where the majority of the heat is removed. The coolers will be connected to the same natural circulation NaK system that serves the dump tanks. They can be of similar U tube form to those in the dump tanks with parameters adjusted to suit the different geometry and rating.

Space is available for five more drain tanks above the set previously described in case it is considered desirable to have more spares available instead of removing any faulty tank from a highly active area.

2.11 Off-gas Delay and Storage System

In the fast reactor system there is not the imperative need to remove certain fission products to maintain reactivity and breeding gain that exists in the thermal molten salt system. Nevertheless there are advantages to be gained (e.g. on the safety side) by partial removal of fission products. In the present state of knowledge, there is considerable uncertainty in the species and amounts which will plate out in the system or will remain chemically combined in the salt solution. Equally there is uncertainty in the amounts of fission gases and volatiles given off, so the parameters used for this part of the plant can be only a guide at this stage.

ORNL have proposed a system consisting of bubble generators associated with the main pumps to purge the gases from the salt solution; the off-gases pass through the drain tanks to a short term delay bed to remove a majority of the decay heat. A proportion of the main flow then passes through a long term delay bed before being cleaned up to separate out the remaining fission products, water and oxygen. For MSFR it is proposed that the off-gassing is promoted by similar means but in loops external to the PCV for easier access, a short term (2 day) delay bed would be retained, but it is suggested that the bulky ORNL low pressure long term delay system could be replaced by storage of off-gases at 60 bars pressure until sufficient decay has occurred, so that mainly stable gases can be released. Cleanup can be restricted to the removal of any corrosive gases which are in contact with the fuel salt and circuit internals. The merit of this proposal will depend upon detailed assessment of the spectrum of fission products passing to the store, the amount of non-gaseous decay products remaining in the store and the heat load.

The two-day delay bed provided for consists of 114,000 m. of 5 cm bore pipe or equivalent, packed with charcoal adsorbent. The pipe is formed into parallel sets, each consisting of 6 U tubes, situated in a water pool formed in an annular segment of width 7 m. and effective length 25 m. within the containment building. The off-gas high pressure store may be conveniently formed from 15 cm bore pipe also formed into U sections immersed in the same tank. A length of about 56 m. should be sufficient for one year's off-gas storage (based on measurements of Xe and Kr after 15 days cooling), and more coils can be installed if necessary.

Heat removal from the delay bed and store is by natural convection boiling in the tank and condensation in air blast coolers outside the building or with an intermediate circuit acting as a fission product barrier if necessary.

Heat loads pro rata to MSBR are: 8 MW from the two day delay bed and about 1 MW from the off-gas store. Forced circulation of the water may be required over the U tubes of the latter item should it prove that the different fission product spectrum in MSFR compared with the thermal reactor leads to increased heat loads. A heat rejection of 40 MW to the condensers is allowed for in the parameter list.

2.12 (a) Helium Coolant Clean High Pressure Store

A volume of 480 m^3 is required to store the 11 te of helium coolant in the cold condition at 138 bars (2000 psi) for initial filling or long term storage after purification. A typical size for the above volume would be 4 storage vessels each 20 m. long and 2.8 m. ID (about 3.4 m. OD).

2.12 (b) High Pressure Dirty Gas Store (DGR)

This store is required for active or contaminated gases when it is desired to reduce pressure in the reactor vault, gas turbine plant, header and dump tanks, etc., and must accommodate complete depressurisation, in case of severe leakage, or for major maintenance. The store can also be used for helium inventory control to enable longer term control of the gas turbine plant for high efficiency at reduced power conditions. A portion of the store should be reserved for the helium in contact with the salt; extra shielding will be required for this part.

To accommodate rapid pumping down of the helium coolant circuit it is assumed that the gases will enter the store at the same temperature as the operating condition, thus 1250 m^3 storage at 138 bars is required using 10 vessels of similar size to those of the clean store. For rapid pumping down cooling may be required to remove excessive heat due to compression.

The size of transfer pumps required will depend on considerations of combined primary and secondary circuit leakage and the activity that can be tolerated within the containment building, which will dictate the time limit for transfer.

2.13 Chemical Cleanup Plant

It has been assumed for the present study that full chemical processing for complete removal of fission products from both core and blanket salts and for extraction of plutonium from the blanket salt will be carried out using an external reprocessing plant, but it is worth noting that a pyrochemical plant would be sufficiently compact to fit into the containment building, in the cells below the dirty gas receivers. Local treatment of the salt has been limited to chemical cleanup to precipitate fission products in solution as uranium compounds and to remove oxygen by precipitation of alumina as described in AEEW-R 956.

For intermittent use salt may be passed to the plant via the drain tanks where it is delayed to remove the majority of the decay heat. If continuous cleanup is required using a loop in parallel with the core/blanket circulation pumps arrangements for heat removal and adjustment of temperature for precipitation will be required.

The salt cleanup plant, together with salt preparation and helium cleanup plants will be located within the containment building in a shielded annular segment about 10.5 m. long by 6.8 m. wide and 21 m. deep above the drain tanks with easy access on two sides for hot cell techniques. These shielded rooms will probably require leaktight membranes to prevent health hazard to the operators. Salt circuits can be maintained at subatmospheric pressures to prevent active leakage.

2.14 Control and Instrumentation

As discussed in Section 2.5 and as previously stated for the lead cooled indirect system, it is hoped that no control rods will be required and that power output can be controlled by fuel salt temperature perturbations. This puts a premium on reliable salt flow control by means of pump speed variation over a wide range, but additional means may be needed at the low power end.

The control maintaining the pressure differential between the helium coolant and salt circuits must, to meet the safety and operational requirements, be of a standard similar to that achieved by control systems in solid fuelled reactors. Selection of the reference pressure point in the primary salt circuit will be important in determining the variation in differential over the power range. It has not been possible to consider the provision of instrumentation in any detail. The principal parameters available for "power" control are flow, temperature and pressure drop measurement in the individual fuel and blanket salt circuits and of the helium coolant. An important development item will be that of high temperature instrumentation capable of working for long periods in an active, radiation environment, together with reliable telemetering. In addition on-line devices will be needed for monitoring salt chemistry and plutonium concentration. So far as can be seen at this stage, apart from temperature environmental problems, nuclear instrumentation (including criticality equipment in process plant) can be of the standard type.

2.15 Remote Handling Arrangements

The compact nature of the MSFR plant items makes it practicable to provide space for installation, removal and shielded remote maintenance areas for all plant, including the reactor vessel and dump tanks.

Thus for initial installation all plant items can be prefabricated and brought into the outer containment building after its construction and that of the PCV main structure is complete.

In order to install or replace the largest plant items, such as the reactor or outer dump tanks, removable bulkheads are fitted to the airlock (to save diameter and length) on the assumption that handling of units of this size will only be done initially or at a prolonged shut down when full outer containment is not required. Normal airlock doors will be fitted so that the smaller plant items such as alternators, gas turbines, IHX and possibly the precoolers or intercoolers can be transferred without breaching the containment.

Sufficient outline design has been done to show that the size of containment building and layout as shown on Figures 4 and 5 will allow for all plant as listed below to be handled and maintained within the building.

Removal of all active salt and decontamination of the primary circuits and dump tanks will of course be required before access is possible for maintenance involving dismantling. For this purpose the salt can be transferred into the drain tanks, after a suitable decay period in the dump tanks, and isolated. The question of salt remaining as impurities on surfaces and its removal, possibly by dilution with non-active salt, must receive attention. Experience with the ORNL MSRE[4] suggests that there was no problem due to traces of salt remaining on the walls but in that case a considerable time (7-10 months) elapsed between shut down and removal of equipment. Furthermore, there may be difference in the wetting properties of fluoride and chloride salts which could have an important effect.

The following table summarises the provisions for remote handling of the main plant items.

Plant Handling

(a) Remote handling flask: capable of handling the largest items of plant (provided that the PCV/shield plugs are removed and the flask positioned by remote operation of the polar crane). A nominal thickness of 3 in. of steel shielding has been taken giving a weight of 360 te which can be handled by the polar crane.

(b) Reactor vessel (with or without IHX and pumps))

(c) Outer dump tank containment with all dump tanks) For both these items two circular apertures each of 9.6 m. diameter give access to a space 10.5 m. x 9.8 m. x 14.5 m. deep above the delay bed area for storage or handling. Access for remote handling is available on both sides of the space and also from above.

(d) Plant up to the diameter of the recuperator including IHX, pumps, precooler, intercooler gas turbines, and drain tanks: An annular segmental space approximately 11.5 m. x 6.8 m. x 21 m. deep is available situated above the drain tanks with remote handling access on both sides and from above.

(e) Gas turbines/compressors: A separate small horizontal flask can be provided for maintaining the moving parts (which can be withdrawn separately) if shielding is necessary due to contamination of the helium circuit.

(f) IHX and main salt pumps: provision is made for in situ remote disconnection of the IHX and pumps from the top of the reactor vessel after prior removal of the top shield plug; access to the IHX tubes can be achieved by removal of the top cover.

(g) Dump valve actuators: Access is possible from a shielded gallery reached via a tunnel in the PCV.

(h) Instrumentation: Special attention will be required at the detail design stage to replacement of instruments but for measurements within the reactor vessel, particularly temperature measurement, replacement will be very difficult and development of reliable long term sensors will be required.

(i) PCV prestressing cables; Access can be easily provided for retensioning or removing the longitudinal cables. It will be necessary to remove part of the leaktight liners and shielding blockwork forming the inner sides of the handling spaces mentioned in (c) and (d) above to inspect or replace part of the circumferential stressing cables.

PART III

SAFETY

3.1 Introduction

A reactor using a liquid fuel containing plutonium salts presents novel safety problems. Whereas in a solid-fuelled reactor, it is only when circuit leakage leads to fuel overheating that release of plutonium (and fission products in quantity) may occur, any leakage with a liquid fuel presents a potential source of release. On the other hand, providing the fuel dump system can be made acceptably reliable, and by design can eliminate accidental criticality problems, the MSFR has the capability of avoiding the energy-release problems which present the sodium-cooled fast reactor with so much difficulty. Much of the safety argument must depend on demonstrating a high standard of control of leakage to limit releases to low levels under normal operation, and to avoid hazard to the public in the event of the more serious failures. The low limits for Pu inhalation have been converted by Beattie(6) into a release/frequency curve for emergency conditions by comparison with currently considered iodine release-frequency curves (the Farmer curve) as shown in Fig. 11. There is reason to believe that fractional release of Pu from the molten salt, agglomeration and plate out effects within the plant areas or containment, and condensation in the outer air which reduces the persistent airbore fraction, will give further "decontamination factors" in considering allowable plant release. Unfortunately many of these factors are as yet quite unknown for heavy metal chloride salts. Tests reported by Stewart(11) in which samples of plutonium metal were heated showed that under vaporisation conditions about half the Pu escaped in an aerosol, whereas with mechanical (and therefore partial) disruption of liquid into droplets, the fraction escaping in aerosol form could be down by as much as 2 orders of magnitude, and that melting without vaporisation reduces the aerosol fraction by yet a further 2 orders of magnitude. If the higher Pu escape fractions have to be taken as the criterion, then the permissible leakage will essentially be determined by the plutonium since inhalation of "mixed fission products" is estimated to lead to a similar level of hazard. If, however, the lower Pu escape fractions are the most likely, the situation might then revert to one similar to that in "conventional" solid-fuelled reactors where the limiting feature was the release of iodine (or similar volatile materials) and the usual Farmer criterion could be used. The very low operational limits for Pu inhalation of $3.5 \times 10^{-5} \mu g/m^3$ of soluble Pu(7) also make it important to understand the form and fractions of release to define leakage control requirements in relation to both operation and maintenance.

It should be noted however that it would be the intention to sparge out the fission product gases continuously from an MSFR as is done in the thermal system, not because of the neutronic effects which have importance for the latter, but to avoid gas blanketting and other undesirable effects which might arise. In this case, the safety problem with iodine would relate to the reliability of its storage in the off-gas system, and salt leakage limits might once again be determined by plutonium.

It is clearly a vital feature of any further work on molten salt fast reactors to carry out a programme of study of potential mechanisms and forms of release of plutonium and fission products from mixed chloride salts, which embraces a range of temperature conditions and of forms of disruption and scattering of salt drops.

The use of high pressure helium as coolant raises questions of depressurisation accidents and interdependence of primary and secondary circuit failure and these are discussed later. It does, however, confer major potential advantages if it can be satisfactorily exploited - overpressure of helium relative to the salt can be used to put almost the whole of the primary circuit into a compressive stress condition, which should minimise the chance of its failing. It can also be used to mitigate against salt leakage from the primary circuit, a feature of very substantial benefit for the small leakages which could be a problem under operational conditions, and for certain limited fault conditions.

It must be remembered that auxiliary circuits and plant, as well as the main circuits, will require careful attention to leakage and the principle of external helium pressure in excess of that of the salt must be followed for these systems as well (the analogy with the pressure balance systems of fuel reprocessing and fabrication lines for plutonium fuels is obvious here).

It will be seen that the present wide range of uncertainty does not allow analysis of any precision and in the sections which follow the discussion is intended to illustrate the form of the problem and suggest potential rather than definitive solutions.

3.2 Containment Principles

The barriers to escape of fuel, or blanket, salt are:-

(a) The primary circuit boundary (PC).

(b) Helium coolant overpressure which normally prevents salt leakage from the primary circuit and causes any leak to be of clean helium inwards.

(c) The coolant helium (secondary) circuit boundary (SC) which is formed by the liner and seals of the prestressed concrete vessel.

(d) A main or outer containment building (MC). It may be necessary to fit an inner lining combined with a sub-atmospheric interspace if the lowest possible leakage to atmosphere is found to be necessary.

It is clear that several basic principles must apply to the helium cooled MSFR if it is to achieve the required high standard of overall containment. These are:-

(i) the pressure balance system which ensures that the helium circuit pressure is always in excess of the pressure in the salt circuits must be engineered to a very high degree of reliability, comparable to that of control rod systems, to keep the helium coolant circuit normally free from contamination even with small or moderate leakage paths in the boundaries of the primary circuit.

(ii) release from the primary circuit of a quantity of salt sufficient to cause a large dependent failure of the secondary circuit is of adequately low probability.

(iii) It must be demonstrated that severe failure of the helium pressure circuit which could cause consequent severe failure of the primary circuit thus leading to a high release of fuel into the containment building is of negligible probability. This calls for the use of a prestressed concrete vessel.

(iv) lesser, though still substantial, failures of the helium coolant pressure circuit cannot cause dependent failures of the primary circuit of sufficient size to release significant quantities of salt to the secondary circuit and hence out into the containment building.

3.3 Operational Hazards

In order to gain full-time access to operational areas within the outer containment the concentration of Pu must be less than the statutory limit of $3 \times 10^{-5} \mu g/m^3$ of air. If therefore it is assumed that the target value should be $10^{-5} \mu g/m^3$, it is necessary to consider what cleanup arrangements are needed in both the helium circuit and the outer containment.

A potential argument is as follows:-

(1) The Outer Containment volume is 10^5 m^3, so the permissible total amount of Pu within it is $1 \mu gm$.

(2) With a high efficiency filter and 1000 times per day recirculation of containment air, inleaking of Pu should be reduced by a factor of at least 1000. The leakage from the helium circuit could then be 10^{-3} gms/day of Pu.

(3) DRAGON experience with pressurised helium is that leakages can be kept down to 10^{-4} of circuit volume per day. There could therefore be 10 gms of Pu present in the helium circuit.

(4) Assuming a filter system in the secondary circuit, and on the same argument as (2), the leakage from the salt circuit to the helium circuit could be 1000 times greater than this, i.e., about 1 Kg/day (or about 4 litres of salt).

When one considers that the helium circuit pressure is to be kept in excess of that of the salt, this seems a reasonable target to aim at, and perhaps it is arguable that high filtration rates need only be invoked when there are indications of a possible leakage. The problems are:-

(a) the circulation power for the outer containment filtration unit would probably be about 2.5 MW which is large but not prohibitive. The frontal area of the filters is also large (about 1000 m^2) and they would occupy a circumferential zone about 6 m high.

(b) the filtration system for the helium circuit has to take gas at pressure. If the filters were at the lowest pressure region (i.e., just before the inlet to the first stage compressor) they would operate at 15-20 ats and a temperature of 30°C. Maintaining the Δp across the filter units at the same level as under the more usual atmospheric conditions but allowing for the increased gas pressure, the frontal area would have to be about 150 m^2. Clearly development of a "stronger" filter to reduce frontal area would be advantageous.

Since the most probable source of leakage is in the salt/helium heat exchangers, it would however be desirable to develop a "trap" or filter which could operate at the top cycle temperature (850°C) to minimise any deposition which might occur round the circuit, particularly in the rotating machinery. It is, of course, uncertain what the form of the Pu-containing material may be, for example deposition may only begin below a certain temperature if at all. Once again the importance of knowing the way in which Pu may escape from the salt mixture is demonstrated.

The high temperature "trap" could assume considerable significance from the point of view of maintenance and replacement of components if it minimised deposition.

3.4 Primary Circuit Failures

In this section primary circuit ruptures with an intact secondary circuit are considered. The blanket circuit is included with the fuel salt as it must be regarded as having significant quantities of Pu present in it for a substantial part of reactor life even though the concentrations will never be as high as in the fuel salt. The individual major components and the factors affecting their design to minimise failure are discussed below. The primary circuit would be designed to take the full system internal pressure in case of total loss of secondary helium pressure, and would (if at all) be subject to this extreme condition only for short periods. Although the proposed pressure balance system may at a low probability lose control, the gas volumes in the balance system could be limited to ensure that only a nominal internal pressure excess could develop in the primary circuit, again only for a short period.

3.4.1 Reactor Vessel

Both the blanket and core salt header tanks helium are fed from the same pressure control system and are connected to the respective pump delivery pipe-work. As the salt pressure drop across both core and blanket is relatively low the pressures will remain approximately equal across the core/blanket membrane even with the different flow rate conditions necessary to cope with load variations and the variation of heat produced in the blanket during the fuel cycle. There will thus be little pressure difference to cause leakage even if cracks develop due to vibration. If blanket salt leaks into the fuel there will be some degradation of reactor performance, alternatively if fuel leaks into the blanket in large quantities it might overload the blanket heat removal system. There could be reactivity transients, which are not assessed here, but clearly an appreciable increase in blanket heat rate would call for dumping of fuel and blanket salt. By adjusting the blanket pressure to be equal to or slightly above the core pressure, the safer leakage of blanket into fuel salt could be ensured. The top "plate" of the reactor vessel will be cooled by blanket salt so no high temperature fuel salt should come into contact with it. The upper side of the plate will be in contact with the helium at outlet from the IHX with about 2 bars excess differential pressure. Insulation will be required and arching of the "plate" structure can be employed if necessary to transfer tensile loads to the outer circumference to minimise the chance of cracking.

3.4.2 Intermediate Heat Exchangers

The lower tube plates will be subject to low differential pressures similar to those across the reactor top plate but special measures may be required to protect the bulk material from the high temperature of the salt passing through the tubes at 1050°C and from the helium coolant at 850°C. Curvature could be employed to avoid central tensile loads but would complicate tube fixing. However, as an alternative a false TZM tube plate could be welded to the tubes and be supported from an arched structural plate loosely fitted over the tubing with tensile loading occurring only at the outer circumference. The upper tube plates will be subject to the basic minimum differential pressure of 2 bars plus salt pumping pressure, say about

25 bars external helium pressure, salt temperature 650°C, helium temperature 415°C. Again a separate backing arched structural member can be employed which, although subject to higher differential pressures than the lower tube plates will be running considerably cooler.

The dimensions of the IHX tubing (8 mm OD and 6 mm ID) ensure that it will withstand very high internal or external differential pressures (from measurements with TZM at 1090°C irradiated to 2.4×10^{20} n/cm^2 at 1 MeV the bursting pressure, at 1090°C, would be 15,000 psi). However, for the large total length of tubing and the large number of welds involved the probability of complete severance is about 2×10^{-1} year^{-1} based on normal pressure piping data. This figure may be substantially improved on bearing in mind the fact that the pressure is external, but the effect of vibration induced fatigue and corrosion surface effects must be examined carefully by prototype testing to ensure that common mode failure of the tubing and connections to the header plates cannot occur. It is likely that due to the small tube diameter the helium overpressure will prevent significant salt leakage if a tube does fail. (If the overpressure system failed completely while there was a faulty tube, the leakage at the normal flow rate from a double-end failure of one tube over a period of 10 seconds, assuming that it took this time to slow the salt pumps down and/or dump the salt, would be about 4 litres.)

3.4.3 Pumps and Salt Ducts

The stresses in the pump bodies as a result of the external pressure will be low and compressive, except in the top cover region, so salt leakage due to pressure failure is very improbable. Failure due to bursting impellers or shaft seizure must be guarded against in the detail design of the pumps and by providing an outer structure to ensure that the probability of major disintegration of the casing is negligible. Connections of IHX's and pumps to the reactor vessel should be in compression due to the helium overpressure; sufficient redundancy of bolting should be employed to prevent the pump delivery pressure causing leakage during the run down time if the helium overpressure fails.

3.4.4 Effect of Primary Circuit Leakage on Secondary Circuit

The foregoing discussion has aimed to illustrate the claim that with an intact secondary circuit, design can ensure that major primary circuit failures leading to substantial quantities of salt being released into the PCV can be kept to an acceptably low probability. It has in fact high-lighted the problems of leakage which are a consequence of very large heat exchange units with very large numbers of tubes and once again the helium overpressure system appears to be potentially capable of obviating most of the salt leakage conditions which could arise.

It is necessary, nevertheless, to consider what can be done to mitigate the possible consequences of a quantity of salt escaping into the PCV and reaching its lower regions. The problem is to prevent a breach of the PCV liner. If high temperature insulation (capable of taking about 2000°C) can be mounted clear of the liner so as to contain the bulk of the salt, then it has to be demonstrated that leakage of salt through this layer of insulation will not overheat the liner. The decay heat rate of the salt will be about 10 w/cc and for a liner thickness of 2 cm, with a gap between insulation and liner which is not allowed to exceed 1 cm, the "normal" liner water cooling system of tubes at 20 cm pitch should limit the liner temperature to a maximum of $1000^{\circ}C$. In the more critical areas, it appears possible therefore to provide a boosted liner cooling system which could keep temperatures well below this. The essential feature of this argument is the development of the high temperature insulant which can be installed with minimum chance of leakage through it.

3.5 Secondary Circuit Failure

It is necessary to show that failures of the secondary, i.e., coolant helium, circuit cannot cause consequent failure of the primary circuit of sufficient size to give release of fuel salt exceeding the safe limits. Major failure of the PCV forming the secondary circuit boundary can be regarded as of negligible probability providing that the PCV concept of redundant tensile members is applied to the largest closures for the reactor vault and regenerator spaces and that double sealing members or flow restrictors are applied to the remaining large closures.

With the helium overpressure system working correctly, the primary circuit should never be subjected to internal bursting forces which would lead to substantial missiles capable of breaching the PCV liner, and the rotating machinery which is confined to the pumps could have restraints fitted to prevent missiles being projected in the event of seizure or similar failure. Even so, it is clearly prudent to provide suitable protection for the liner and to screen components to minimise the chance of liner damage.

Attention must be concentrated upon the largest remaining apertures in the PCV which cannot be flow restricted; these are found in the penetrations for the NaK dump tank cooling pipework and the main cooling water pipework in the precoolers and intercoolers. The size of these pipes has been limited to 28 cm flow diameter which gives a reasonable compromise between normal operation pressure drop, the number of such apertures and the rate of depressurisation. In the case of the water piping for the precoolers and intercoolers where outward flow of helium could occur through both ends of a ruptured pipe, the rate of helium depressurisation for sonic flow through a short length would be 27 seconds to half pressure. As both the intercooler and precooler are remote from the primary circuit it is considered that a rupture of the CW piping should not cause a consequent failure of the primary circuit by any mechanical effect. Furthermore with this rate of depressurisation it should be possible to carry out a depressurisation of the primary circuit so as to avoid a consequential rupture (see the later part of this section). The NaK pipework is closer to the reactor and if a rupture occurred where the pipe penetrates the PCV lining a large flow of helium will pass through the IHX plenum space. Investigation would be required to see if precautions are required to prevent damage to the tubing of the IHX due to high gas velocities during this depressurisation. As an alternative, it can be argued that the probability of failure of the NaK pipework leading to loss of helium can be made negligible by designing the external pipework to resist the full helium pressure that would be transmitted to the NaK system if a failure in the dump tank region occurred. Yet again, the penetrations of the PCV liner could be reduced in size by doubling the numbers of pipes, or flow restrictors used to reduce the helium leakage rate to a sufficiently low value to avoid damage due to any failure in this region.

A point of concern is the possibility of normal running with, for example, a weakened IHX tube which is not detected but which was prone to failure due to extra loads imposed during a depressurisation. If the depressurisation was due to a CW pipe failure and if salt escaped from the IHX tube some of it might reach the CW system although it seems more likely to deposit on the recuperator surfaces on the way. Any salt reaching the CW piping could pass out of the outer containment without hindrance into the main source of cooling water. In this event the release can only be kept within safe limits by either (a) devising a pressure balance system which ensures that even in the case of a fairly rapid depressurisation the overpressure of secondary helium will be maintained, or (b) designing the cooling water system to be self-contained and to withstand the full helium pressure. The latter object could be achieved by using the water as an intermediate circuit passing, say, to dry cooling (forced draught air cooled) towers. The second alternative may be the more acceptable one even though the lowest temperature of the helium cycle would need to be raised slightly. Furthermore dry cooling towers as a means of heat rejection would in themselves have an attraction.

Even so, it appears practicable to depressurise the helium volume associated with the salt header tanks and dump tanks (Hv) (about 500 m^3) into the dirty gas receivers (DGRs) at a sufficiently high rate to maintain the coolant helium (H_c) overpressure in the event of the 27 second time constant depressurisation if the dump tank helium exhaust was initiated by a rapid action loss of pressure signal from the coolant helium circuit. For a DGR volume of 1250 m^3 the H_v pressure would balance at about 17 bars and pumping would be required to reduce the pressure further at a necessarily slower rate. However the fuel can be dumped (in about 14 seconds) before this stage is reached, so the chances of significant quantities of Pu reaching the CW system are remote even if a fracture of an IHX tube occurs at the same time as a secondary circuit depressurisation. Alternative (a) is therefore a possibility.

There will be a period in a depressurisation procedure of this kind (which might be invoked for other events also) when it is necessary to accept that the primary circuit contains an excess internal pressure of 17 bars. Apart from any salt leakage, care must be taken that there will be no serious leakage of gaseous and other volatile fission products held in the primary circuit and dump tanks. Again, it can be argued that this condition applies only in the event of a major secondary circuit failure and for only the short period necessary for the further pumping down to be achieved; and that to sustain 17 bars for several hours if pumping down is delayed is a very modest requirement.

The safety aspects of the dirty gas receivers when containing the active gas in the event of the rapid H_v depressurisation will require further consideration. They have been shown as free standing steel vessels; it might be safer to form them into a self-contained prestressed vessel even though the theoretical probability of a combined failure of the reactor coolant circuit and the DGRs must be negligible. The possibility of incorporating the DGR within the main PCV was considered but if it is necessary in the PCV design to protect against liner failure leading to high pressure being transmitted to any diametral crack in the PCV wall, the layout changes could involve an uneconomic and an impractical amount of prestressing.

3.6 Fault and Accident Analysis

3.6.1 Release Levels and Containment Effectiveness

The curve of Fig. 11 can be used to derive a permissible frequency for an incident when an estimate has been made of the potential Pu release. This curve, while not using the more pessimistic of Beattie's[6] values, is conservative when compared with a recent MRC publication[13] which would for a given frequency yield permissible release values about six times higher.

It is useful at this stage to consider how effective a barrier the secondary circuit envelope can be if it remains intact. As has been said previously, DRAGON experience has shown that leakages from a well-designed tight helium circuit can be kept to below 1 Kg per day. This rate of leakage constitutes $\sim$0.01%/day of the secondary circuit gas inventory, and is equivalent to a DF of $\sim 10^4$ even if (a) the helium circuit is kept at pressure for 24 hours after an incident and (b) any cleanup which a secondary circuit filtration system achieves (see Section 3.3) is neglected. The design is such that the helium can be discharged to dirty gas receivers in about 2½ hours, and the system pressure can then be maintained near to or just below atmospheric so as to reduce the leakage rate substantially. This ability to depressurise the helium system will, on a time-basis alone, afford a further order of magnitude improvement in the effective DF of the intact helium system, increasing it to 10^5.

Although there is no direct evidence to guide, a decontamination factor (DF) for the main containment of 100 for Pu aerosols seems a reasonable and cautious target value to take, bearing in mind plate-out effects and contributions from the containment main cleanup plant (see Section 3.3), or an emergency cleanup unit.

In the discussion below, the failures are grouped under the main headings of (a) primary circuit failures, (b) secondary circuit failures, (c) fuel dump tank failures. The arguments are developed in the text and begin almost entirely with the initiating event; at the end of this Part III of the report (pages 33 & 34) cases are presented in diagrammatic form to supplement the text and for ease of reference.

3.6.2 Primary Circuit Failures

(a) IHX failures

It has already been noted in Section 3.3 that the large number of tubes and welds in the IHXs may mean individual tube failure occurring at a frequency of 10^{-1} yrs^{-1}, and further that a significant leak would only take place if there were a failed tube when the helium pressure balance system also failed. With care in design the balance system should be capable of high reliability, but taking what is believed to be a pessimistic value of 10^{-2} failures/year means that the frequency of the leakage from this group of events should be no greater than 10^{-3} yrs^{-1}, as a single tube failure can hardly be considered to be severe enough in itself to precipitate pressure balance system failure.

Dumping of the fuel will lower the salt level below the IHXs in about 10 secs and if normal full flow of salt persisted for this time about 4 litres of salt containing about 1 Kg of Pu would escape into the helium circuit.

With an intact secondary circuit, the combined DF of this low leakage circuit and the containment is at least 10^6 (see Section 3.6.1) so the release to atmosphere would be about 10^{-3}gm (or about 10^{-6}gm if the cleanup plant were effective) - both negligible amounts at the assessed frequency of 10^{-3} events per year. Alternatively it can be argued that from the point of heat exchanger tube failures, a secondary circuit leakage rate much higher than that achieved by DRAGON would be acceptable.

In the extreme, even if there were at the same time a substantial leak from the secondary circuit, the containment would reduce the Pu escape to atmosphere to less than 10 gms. Reference to Fig. 11 shows that a release of 10 gms is acceptable at a frequency of 10^{-3} yrs^{-1}, which corresponds to the assessed permissible frequency for the fault. It seems reasonable to conclude that there are substantial margins in hand against this kind of accident.

Unless there is some common mode effect, the frequency of failure of greater numbers of IHX tubes will be inversely proportional to the number of tubes considered, so the effects are self-compensating. It is clearly crucial that the IHX development programme demonstrates elimination of common mode failures, and additionally, though this seems unlikely, that failure of one tube does not cascade into failure of others.

The previous paragraphs have assumed that fuel dumping has been effective in restricting the amount of salt released from a failed IHX. Retaining the assumption of pressure balance system failure, and assuming now that dumping does not occur for, say, 1000 seconds, at which time there is manual intervention to effect it, the salt discharged into the helium circuit would be about 400 litres, which contains 100 Kg Pu. Following the same type of argument as in the early part of this section, an intact secondary circuit combined with the containment retention would reduce the release to 10^{-4} g - again negligible. Similarly if all the Pu escaped into the containment, the release from the latter with its DF of 100 would be 1 Kg for which the permissible frequency is 10^{-5} yrs^{-1}. Provided therefore the dump system failure rate were not greater than 10^{-2} yrs^{-1}, the requirement for this extreme assumption is met and to achieve this reliability in the dump system does not seem an unduly onerous requirement.

The ultimate in this argument is that of total failure to dump fuel in the event of substantial IHX failure. The end result is similar to that of gross failure of the primary circuit which is discussed in (b).

(b) Large failures of the primary circuit

A major failure of the primary circuit could release a large fraction of the salt to the secondary helium circuit, even if the dump system operated successfully. If the pessimistic assumption is made that the Pu from the salt gets widely dispersed in the helium, then between 10^3 and 10^4 Kg (the total inventory) is involved. A "tight" helium circuit (see (a) above) with cleanup circuits operating would reduce the amount escaping to the containment to less than 10 g of Pu (or alternatively with no cleanup benefit but exhausting helium to dirty gas receivers about 100 g Pu).

The release to atmosphere would then be in the range 1 - 10 g, permissible at a frequency of 10^{-2} to 10^{-3} yrs^{-1} - a reliability demand which the primary circuit bearing in mind its operation under external pressure of helium should easily be able to meet. Conversely, if this primary circuit failure rate frequency were considered to be in the 10^{-4} yrs^{-1} range, the leakage from the secondary circuit would be acceptable at a 10^2 higher level than discussed in part (a) of this section.

The two main questions about this most severe accident are (i) can it be ensured that the lining of the PCRV will retain the spilled salt to the requisite degree of reliability? It has already been argued that it should be possible to avoid missile damage to the lining, and so the principal design problem is to achieve a layout in which it can be guaranteed that only very small fractions of leaking salt escape being directed by the tundish or catchment system to the dump tanks so coming into direct contact with the lining. (ii) will the escaping salt cause pressure surges in the helium system which would breach it, or less seriously, significantly increase the leak rate? Since there does not appear to be any reaction mechanism which could cause rapid pressure rises, and any caused in slower time by heating could be dealt with by suitable venting to the gas storage, this does not seem to constitute a problem which could not be handled satisfactorily.

3.6.3 Secondary Circuit Failures

These may be categorised as:-

(i) Small failures which will neither cause a primary circuit failure nor prevent operation of the overpressure system. Failure of the precooler or intercooler cooling water small bore tubing should fall into this category as even continued sonic flow through two ends of a 1 cm bore tube would only cause the secondary circuit pressure to halve in 6 hours.

(ii) Intermediate size failures up to the maximum "credible" size of 28 cm (intercooler water pipe or emergency NaK cooling system pipe penetrations) giving a depressurisation time constant of 27 seconds to half pressure through both ends of the severed pipe. These pipes are sufficiently remote from the primary circuit to believe they should not cause it severe damage but could conceivably cause certain failures if incipient weakness were present.

(iii) Very large secondary circuit failures which clearly could damage the primary circuit.

Considering the release-probabilities for the three categories:-

(a) The length of small bore (about 1 cm) tubing in the precooler and intercooler is very high (1150 Km), and the probability of failure at 10^{-7} failures per annum per foot run (based on conventional piping) is about 4×10^{-1} yrs^{-1} or one in $2\frac{1}{2}$ years. This raises the question of reliability and down time for repairs, which is a feature likely to be common to all large reactors, rather more acutely than that of release of activity as no failure of the primary circuit or of the helium overpressure system should occur.

(b) the amount of larger bore piping and connections to the subheaders in the coolers is substantial, and it may be difficult to claim a failure rate for complete severance of better than 10^{-3} yrs^{-1}. If we assume pessimistically that a depressurisation (arising partly because the helium pressure balance fails to cope) causes incipient weaknesses in the IHX tubes as being the most vulnerable primary circuit component to result in some failures, then by the arguments of Section 3.6.2 (a), about 1 Kg of Pu could enter the helium circuit before fuel dumping lowers the salt level below the IHXs. How much of this could enter and subsequently escape from the cooler water circuits is hard to estimate. Certainly there seems to be cause (which has not been fully pursued in the present studies) for making the cooler water circuits of high integrity, possibly to take full helium circuit pressure. Since a release from the cooler circuit would be direct to atmosphere, the full 1 Kg of Pu could only be allowed to escape at a frequency of 10^{-5} yrs^{-1}. This means that either a further factor of cooler circuit failure to atmosphere of 10^{-2} yrs^{-1} is needed to supplement the initial assumed failure rate between helium and water circuits of 10^{-3} (see above) - which seems a plausible target - or only about 1/100 of the Pu will escape - which also seems plausible. Failure of automatic fuel dump (again see Section 3.6.2 (a)) could increase the potential Pu release by a factor 100 but if the fuel dump failure rate is 10^{-2} for each demand of this type, the acceptability of the relationship between frequency and release is maintained.

The case of failure of the NaK cooling system in the dump tanks leads to similar arguments; again the best solution may be to design it for full pressure conditions.

(c) if there is a large failure of the PCRV, it must be expected that a severe failure of the primary circuit could result. The Pu escape from the primary circuit might by analogy with the discussion in Section 3.6.2 be in the range 10^3 - 10^4 Kg, and if it is assumed that all of this is uniformly dispersed in the helium, clearly it is highly important that the containment remains effective. The volume of the containment is 100 times that of the helium circuit, so making no correction for temperature, the balance pressure would be about 1.6 ats - not an unduly high pressure to expect to be able to contain to the extent of avoiding any significant breaches. The problem may rather therefore be that of being able to pump to dirty storage in a reasonable time so as to reduce outward leakage from the containment to as low a value as possible. Since there are no chemical reactions that can be foreseen which would be a cuase of pressure pulses or increases over a longer time, and it is an implicit assumption that the design must avoid critical accumulations, the remaining problem may then be that of decay heat disposal from the dispersed fuel. Water sprays for drenching may be the only design solution.

Returning to the release question, a DF of 100 by the containment would mean an escape of 10 to 100 Kg of Pu, for which Fig. 11 demands an occurrence rate not greater than 10^{-7} yrs^{-1}. The catastrophic failure rate of the PCRV would then in the absence of plate out or other mitigating factors require to be 10^{-7} yrs^{-1} also; since this type of major depressurisation failure, in a PCRV with double closures and similar constraint devices is usually considered to be incredible, there seem to be possibilities that further examination of this case will lead to an acceptable result.

3.6.4 Dump Tank Failures

The relability requirements of the system which dumps fuel from the primary circuit have been discussed in previous sections. Since the dump tank system in a molten salt reactor is claimed to be one of the most important features of the safety concept in that it provides a means of emergency cooling under controlled conditions, it is necessary to demonstrate adequate reliability of the components of the dump tanks.

3.7 Commentary

The discussion in Section 3.5.1 draws attention to the great importance of knowing to what extent and in what form Pu (and fission products) would be released from the mixed chloride salts of an MSFR in the event of leakage, in any subsequent overheating of leaked salt, and in certain fault conditions. The balance between fission products and Pu as the greater hazard could be very much influenced by their relative release fractions and there clearly could be several orders of magnitude change in Pu release if persistent aerosols did not form. The influence of these uncertainties on design containment and operational philosophies is large. For example, in considering whether it is possible to undertake operational access inside the outer containment, either on a routine basis, or during a prolonged shutdown for maintenance, the permissible levels of Pu in the containment air mean that on the more pessimistic figures only micrograms of Pu could be allowed to leak from the salt circuits if they became dispersed through the helium coolant circuit in the PCV. Even with the more dramatic improvements arising from orders of magnitude reductions in the Pu fractional releases, very small spills could still be a problem.

Unless the activity is predominantly retained in the salt, it is clear that the helium overpressure and its maintenance to a rigorous and high standard forms an integral and important part of the safety argument. Furthermore, it must be appreciated that the auxiliary circuits may well have to be operated with an excess of external gas pressure.

It is debatable whether one should consider regular entry to the containment during operation (though instrumentation may be the factor determining the need for this). The problem of access for maintenance has however still to be met and here again there is no real body of evidence helping to indicate to what extent the salt, its contained Pu, and the fission products will disengage from circuit components on drainage. All that is known on a plant scale is the encouraging picture that ORNL experienced no difficulty in dismantling MSRE but more will need to be known about the "drainage" properties of chloride salts compared with fluorides before this question can be assessed.

Considering first the reliability of the dump system, heat removal system, circuits, etc.:-

(a) Circuits to each tank are duplicated and there are 24 separate natural circulation circuits. A guaranteed electrical supply for the air blowers totalling 1300 kW is needed but this could be sub-divided and the ultimate if this fails would be to supply water to the boilers and let them boil off. Initially valve operation is called for to allow a reserve of cool NaK to enter the circuits but failure of these valves would not be intolerable in the extreme. One point to consider during the development of the design is ensuring that there is a negligible risk of failure of both cooling circuits in one tank.

(b) The construction of the tanks is simplified to reduce the risk of failure particularly in the region below the salt levels. Thermal stresses upon dumping can be reduced to acceptable levels by the use of

TZM or molybdenum inner insulating linings. If a tank did fail cooling is provided by the outer catchpot, and if a failure of the NaK circuit reduced the capacity of any tank, overflow arrangements are provided and in the limit the salt would be contained in the outer catchpot. It is therefore considered that the risk of release due to tank failure should be lower than that of primary circuit failures.

The studies made during this investigation have supported the feeling that heat removal under these controlled conditions and the retention capability of the dump and catchpot system can be engineered to the requisite degree of reliability as the designer has a number of techniques which can be exploited. It is an area which because of its relative novelty would benefit from investigations which explore various alternatives.

In the consideration of accidents, it is not claimed that the case coverage has been exhaustive; the aim has rather been to seek out the principal ones which illustrated some basic problem or requirement. The performance of the outer containment is an important factor, as in a large proportion of the cases considered most difficult, a failure in the secondary circuit is the initiating event; it has then to be shown that any concomitant release from the primary circuit can be adequately contained. The cases where the pessimistic limits are barely or not quite held are:-

(a) if an inherent weakness developed in the IHX's which made them prone to "galloping" failure, a helium overpressure system error or a secondary-circuit CW pipe failure could lead to big salt releases. The IHX's could be the Achilles' heel of the system if they were not fully developed and stringently tested and fabricated. A means of checking their continued quality would also have to be devised.

(b) the consequences of a secondary circuit failure due to a break in the NaK cooling system for the dump tanks, or the CW circuit for the gas-turbine heat exchangers suggest it may prove desirable to design these circuits for full coolant pressures, and in the CW case to have a closed circuit.

(c) major failures of the primary circuit can perhaps be dealt with by suitable development of schemes for protecting the PCV liner and designing for good drainage of the PCV, but there would be a very great gain if further work could confirm that a primary circuit designed carefully to take advantage of a normally external pressure condition could have a very high standard of potential integrity (failure rate 10^{-6} to 10^{-7} per year).

Having enumerated all these queries and problems, it nevertheless can be said that the combination of the pressurised gas coolant and a molten salt fuel can be used to considerable advantage in dealing with safety questions - provided the basically high integrity of a pre-stressed concrete pressure vessel against major failure is accepted. It appears possible to design the system so that no major missiles will develop which would seriously damage the PCV and equally it appears possible to avoid energy-releasing accidents which might breach the containment.

It is of interest, in fact, to attempt a comparison with the lead-cooled MSFR concept reported in AEEW-R 956[12].

Considerable pumping pressures are required in both the salt and lead circuits of a MSFR (in the region of 2 and 1 MN/m^2 or about 300 and 150 psi respectively), thus the pressure problem is not eliminated. The possibility of serious failure of the primary circuit must still be shown to be low, although intuitively it is felt that the salt would be better cooled if it entered the lead circuit than in helium. If small leakages into the coolant circuit were to be avoided an overpressure of lead would be needed, so lead pressures in the region of 300 psi would be required. The probability of failure of the large lead ducts under liquid pressure would be higher than that for large failures of the PCV of the helium case and the consequences of a duct rupture with the inertia forces involved could lead to serious primary circuit failure.

This short examination suggests that the lead-cooled system is unlikely to offer dramatic safety advantages - again provided the high-integrity PCV philosophy is accepted.

PRIMARY CIRCUIT FAILURES

SMALL FAILURES* (10^{-1} yrs^{-1})

- SC overpressure prevents salt leakage
  - No release
- SC overpressure fails also (failure rate should be better than 10^{-2} yr^{-1})
  - 1 Kg Pu released to SC
    - Filter in SC DF 10^3
      - SC leakage effective DF 10^3
        - Containment DF 10^2
          - Release about 10^{-6} gms at frequency 10^{-3} yrs^{-1} – large margin in hand
    - Assume SC filter and DF effect negligible
      - Containment DF 10^2
        - Release 10 gms at frequency 10^{-3} yrs^{-1} – acceptable

LARGER FAILURES†
(in all cases, SC overpressure assumed not effective)

- Fuel dump effective
  - Typical Pu release to SC about 10 Kg
    - SC intact (DF with filters 10^7)
      - Release to containment 10^{-3} gm
        - Containment DF 10^2
          - Release about 10^{-5} gms – negligible
    - SC leaky
      - Assume full release to containment of 10^4 gms
        - Containment DF 10^2
          - Release about 10^2 gms – acceptable at 10^{-4} frequency. Requires combined frequency of initial failure and reliability of SC to give factor 10^{-4} which is a modest demand
- Delayed dump (failure rate 10^{-2}/demand)
  - Pu release to SC about 100 Kg
    - filters in SC + SC minimum leakage + containment DF total 10^9
      - Release 10^{-4} gm – negligible
    - Leaky SC all Pu to containment
      - Containment DF 10^2
        - Release 1 Kg at permissible frequency $\sim 10^{-5}$ requires initial PC failure probability together with SC failure on this demand to be $\sim 10^{-3}$ which seems quite plausible
- Total failure to dump (failure better than 10^{-3}/demand)
  - Pu release to SC 10^3–10^4 Kg (full inventory)
    - SC filters ineffective but SC remains intact with DF 10^4
      - Containment DF 10^2
        - Release 1–10 g at frequency significantly less than 10^{-3} per demand which leaves a substantial margin
    - Gross SC leakage
      - Containment 10^2
        - Release 10^4–10^5 g permissible at frequency $10^{-6}/10^{-7}$ yrs^{-1} – only fails to meet criterion if whole Pu inventory escapes into containment, and then by factor 10 only

Note: If automatic dump of fuel fails (rate say 10^{-2} per demand), the release increase is estimated as a factor 10^2, so the situation is still acceptable.

*e.g., 1 IHX tube failure.

†i.e., ranging from multiple (progressive) IHX tube failure (say) at lower end of scale to gross circuit failure in worst case.

Code – PC refers to Primary (Salt) Circuit.
SC " " Secondary (Helium) Circuit.
Containment or
MC refers to Main containment building.
DF " " Decontamination factor.

SECONDARY CIRCUIT FAILURES

FAILURES LEADING TO RELEASES TO MAIN CONTAINMENT

- **SMALL FAILURES** not leading to consequential PC failure, eg from IHX tube
  - assume small existing PC leak
  - Assume Pu passes to containment, failure upper limit likely is 1 Kg Pu
  - Containment DF 10^2
  - Release 10 gm ~ if frequency of SC small failure less than 10^{-3} yr, acceptable even assuming no effective DF in secondary circuit

- **LARGER FAILURES ASSUMED TO RESULT IN PRIMARY CIRCUIT FAILURES** (very low basic failure rate for a prestressed vessel)
  - MODERATE PC failures
    - Fuel Dump Operates
      - Typical release from PC 1-10 Kg
      - Assume all Pu passes to containment
      - Containment DF 10^2
      - Release 10-100 gm acceptable at 10^{-3}/10^{-4} yr^{-1} ~ should be an easily attainable target as SC failure rate should achieve 10^{-3}/10^{-4} yr^{-1}
    - Delayed Fuel Dump
      - Typical release from PC 1-10 Kg
      - Assume all Pu passes to containment
      - Containment DF 10^2
      - Release 1 Kg Pu acceptable at 10^{-5} yrs^{-1} frequency ~ for dump delay reliability 10^{-2}/demand case is acceptable at SC failure rate as high as 10^{-3} yr^{-1}
    - Total Failure to Dump Fuel
      - Typical release 10^3/10^4 Kg (full inventory)
      - Assume all Pu passes to containment
      - Containment DF 10^2
      - Release 10-100 Kg Pu acceptable at 10^{-6} to 10^{-7} yr^{-1} ~ attainable for SC failure rate 10^{-3}/10^{-4} yr^{-1} and for total failure to dump $\sim 10^{-3}$ per demand
  - LARGE PC failure
    - Failure too large for fuel dump to have much effect
    - Typical release 10^3/10^4 Kg (full inventory)
    - Assume all Pu passes to containment
      - Containment DF 10^2
        - Release 10-100 Kg acceptable at 10^{-6} to 10^{-7} yr^{-1} ~ attainable if SC gross failure is considered to be at this low level
      - Containment fails
        - If containment total failure rate $\approx 10^{-2}$ per serious event of this kind, the SC reliability level demand remains as in previous case

FAILURES WHICH MIGHT BY-PASS MAIN CONTAINMENT

(ie SC leaks to NaK dump tank cooling or CW systems)

- NaK or CW circuit remains intact
  - If effective DF of intact circuit is not less than 10^2 cases are the same as or better than releases to the main containment ~ it seems arguable that this requirement could be met
- NaK or CW circuit fails
  - If failure rate of these circuits is $\sim 10^{-2}$ per demand under these circumstances, these cases are also similar in release/probability to corresponding releases to containment and therefore acceptable

PART IV

SUMMARY AND CONCLUSIONS

Many of the comments on salt chemistry, materials, nuclear performance and fuel cycles which were made in AEEW-R956 (An Assessment of a 2500 MWe Molten Chloride Salt Fast Reactor) are appropriate to this further study of an MSFR with a direct-coupled helium gas-turbine power plant, so for reference the conclusions of that report are reproduced in Appendix I of this report, and the remarks below are intended to summarise the salient points of the new features.

(i) Plant performance - although the gas-turbine cycle selected leads to a lower efficiency than that of the steam-plant version of AEEW-R956, the bigger salt temperature range accruing from the increase in salt top temperature assumed to accommodate the gas-turbine cycle offsets this in such a way that the fuel inventories of the two cases will be effectively the same and fuel costs will therefore be indistinguishable from those estimated in AEEW-R956. Doubling times would also be the same (see paras (v) and (vi) of Appendix I of this report). There is scope for further optimisation of the balance between capital and fuel costs for the gas-turbine case.

(ii) Plant layout - the concept of a direct-cycle gas-turbine unit with pressurised helium as salt coolant, when used in conjunction with a prestressed concrete pressure vessel lends itself to an extremely compact layout. The complete MSFR plant for a 2500 MWe unit (excluding CW pumping units, workshops, laboratories, control room and administrative accommodation, but including all generating plant) can be accommodated in a containment building of 50 m. dia, 77 m. high (see Fig. 10 for a size comparison with other reactors and Figs 4 and 5 for a general layout).

(iii) Plant components and plant costs - the individual items of plant can all be kept down to sizes permitting prefabrication, thus leading to quicker construction times; it is also possible within the plant layout devised to make provision for maintenance or replacement of <u>all</u> items using the shielded flasks and storage/maintenance positions provided, within the outer containment building. It is therefore possible to envisage a site usage being extended to the life of the PCV (for which the liner may be the limiting feature). The potential for reducing erection times with prefabrication indicates on a first assessment several areas where significant cost reductions relative to an LMFBR might be seen, and there are several factors suggesting that plant costs could be noticeably lower than for the lead-cooled version reported on in AEEW-R956 (para. (v) of Appendix I of this report). It has been necessary, as in that report, to assume that molybdenum (or its alloys) will be developed so that it can be used as the principal material for the salt circuits.

(iv) Safety aspects - the studies have illustrated very clearly the importance of gaining further information on the way in which Pu and fission products can escape from salt under various temperature conditions and as a result of mechanical breakup of drops and jets of salt under fault conditions. The chief problem is the fractional release which is in the form of a persistent aerosol. The present range of uncertainty makes it difficult to do other than attempt to assess what principles should be followed, and what these might achieve. They have also shown that:-

(a) provided the high basic integrity of a PCV against major failure is accepted, a high pressure helium circuit can confer a number of advantages. If the helium is maintained at a suitably chosen pressure in excess of the salt maximum pressure, minor leaks of salt from the primary circuit should be prevented and the fact that the primary circuit is virtually all under compression in the operating condition should confer an added degree of reliability against gross failure. There is little previous experience to guide this approach and it is an avenue worth considerable further exploration. There is of course the implication that the helium overpressure control system must have a high degree of reliability.

(b) in case substantial leaks do occur from the primary circuit into the PCV it is important to devise a high temperature protection for the PCV liner if major breaches of it are to be avoided, as this would otherwise negate the argument of no large failures of a PCV.

(c) a dump system can be devised with potential for the required reliability, but care must be taken to ensure that its heat rejection system has the requisite integrity against mechanical failure if activity release possibilities are to be kept to a minimum.

(d) the CW system for heat rejection from the helium circuit could be a weak point for activity release and may need a special closed circuit.

(e) the large intermediate heat exchangers with their great lengths of tubing and numerous welds will need much development, proving and life-testing if they are not to be (1) a nuisance in normal operation because of leakage although small amounts of helium inleakage can be tolerated and (2) a problem in fault conditions if multiple tube failures cascade from initial moderate faults in the heat exchangers themselves or in other circuit components.

(f) unless the helium overpressure is capable of eliminating leakage from the primary circuit in normal operation, maintenance of general plant and access to the outer containment could be very difficult to achieve.

Although clearly there are many questions still unresolved about MSFR's, and indeed many queries still to be formulated, it is believed that this study has shown distinct possibilities that a concept with the requisite accident safety characteristics could be evolved, with the pressurised helium coolant system as an important contributing feature. The increasing concern about the problems and cost of fabrication of solid fuels containing plutonium, together with the quantity of highly-active waste that solid-fuel assemblies lead to, is a material incentive to continued investigation of MSFR's.

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APPENDIX I

The conclusions which can be derived from the studies summarised in Report AEEW-R956 are:-

(i) of the alternative forms of MSFR examined, the indirect-cooled versions (see Figures 1b or 2) appear to offer most potential for further investigation. There are however uncertainties in a number of physical properties (notably thermal conductivity) which reflect noticeably on choice of design conditions, and these, coupled with some uncertainties on nuclear data, make it difficult to achieve any substantial degree of optimisation in design at this stage. Choice of secondary coolant is an important feature, and of the liquids, lead appears to be the most suitable taking into account compatibility with the fuel salt. Since, however, helium technology will be developed for HTRs and it should pose less materials problems, study of the possible use of helium in an MSFR, which will require careful consideration of the safety issues of a pressurised coolant, should be undertaken.

(ii) the chemistry of the chloride salts proposed for fuel and blanket shows, on the limited work to date, encouraging signs of being controllable to the required degree; but both in this area, where the effects of potential precipitating species (e.g., due to oxygen and nitrogen from adventitious air ingress, sulphur from Cl-35, fission products) and the possible deposition of noble metal fission products need study. In the materials and corrosion fields, much more work is clearly needed. The extent of the materials work will be affected by the choice of secondary coolant, but it is evident that considerable reliance will have to be placed on the demonstration of good corrosion performance and development in the technique of fabrication of molybdenum or its alloys for the key components (e.g., core vessel and heat exchangers) of the core and blanket circuits. Suitable design should enable more conventional materials to be used for the main vessels and other components.

(iii) there appear to be good prospects of being able to operate an MSFR without conventional nuclear control systems. This together with the avoidance of refuelling equipment, offers the possibility of a compact system with more the character of a chemical plant. It is from these possibilities that the potential for capital cost savings compared with an LMFBR can be discerned, but there are the important areas of containment, emergency dumping and ancillary systems requiring further investigation to assess any off-setting effects.

(iv) the reference lead-cooled indirect design of this report is inferior to a 1990 oxide-fuelled LMFBR in fuel inventory but current evidence suggests that an increase in primary salt top temperature could be contemplated to a level where the inventories are comparable. The doubling times with natural chlorine in the salt would then also be comparable, and if separated Cl-37 were used doubling times in the range 20-25 years seem possible.

(v) the preliminary costing (which is all it has been possible to carry out) indicates that the reference MSFR has potential for savings relative to a 1990 LMFBR both in the capital cost (about £12-20/kWe) and fuel cost areas (£8/kW). The increased top temperature scheme referred to in (iv) shows further gains of £5/kW for capital and £5/kW for fuel costs, the latter reflecting the reduced inventory charges with plutonium taken at £5/gm. These costs are for natural chlorine salt; Cl-37 although giving the gains in doubling time already mentioned, leads to a slightly higher fuel cost because of increased inventory charges, so its use must be judged (apart from any chemistry gains in reducing the amount of sulphur formed from Cl-35) on its value in an overall generating system where increased installation rates can have discounted worth.

(vi) the fuel costings have taken aqueous solvent extraction as the process for heavy metal handling and show favourable results. It is not therefore necessary to invoke a close-coupled pyrochemical method, and indeed the preliminary investigations made, while indicating technical feasibility, raised doubts about the costs of this method with the small scale batching technique considered; this is an area requiring further examination.

(vii) the investigations reported here have indicated the value of the intrinsic self-regulatory characteristics of the MSFR with its favourable temperature coefficient, and the potential for fuel dumping in the event of faults but have also brought out the importance of further careful study of fault conditions, including cross-leakage with a fluid fuel, and of containment requirements.

<u>Parameters for Indirect and Direct Cooled Versions of a Molten Salt Fast Reactor - Summary</u>

Reactor power (total)	MW(th)	6000
Gross electrical output	MW(e)	2 x 1350
Nett electrical output	MW(e)	2 x 1250
Steam conditions - TSV	MN/m^2	16.3
	psig	2350
	^oC	565
Overall plant thermal efficiency	%	41.8
Fuel salt	$NaCl:UCl_3:PuCl_3$	60/37/3 mol %
Blanket salt	$NaCl:UCl_3$	60/40 mol %
Melting point		577^oC (850 K)

TABLE I

HELIUM COOLED MSFR GAS TURBINE PLANT PARAMETER SUMMARY

Nett station output		2500 MWe
Gross alternator output	4 x	654 MWe
Gross LP turbine power	4 x	663 MWe
Cycle Efficiency $\frac{\text{(LPT Power)}}{\text{(Nett heat input)}}$		40.8%
Nett efficiency		38.4%
Nett heat from reactor		6500 MWth
Maximum fuel salt temperature		1050°C
Fuel salt temperature at core inlet		650°C
Maximum helium pressure		62.5 bars
Helium temperature at IHX outlet		850°C
Helium temperature at IHX inlet		415°C
Cycle temperature ratio		3.72
Cycle pressure ratio		3.16
Helium flow	4 x	732 Kg/s
Fuel salt flow	4 x	8250 Kg/s
CW inlet/outlet temperature		20/35°C
High pressure turbine (HPT) power	4 x	760 MW
HPT speed		4500 rpm
HPT number of stages		3
High pressure compressor (HPC) power	4 x	382 MW
HPC number of stages, axial + centrifugal		10 + 1
Low pressure compressor (LPC) power	4 x	378 MW
LPC number of stages, axial + centrifugal		6 + 1
LP Turbine speed		3000 rpm
LPT number of stages		6
Recuperator effectiveness		0.83
Recuperator heat transfer	4 x	1080 MW
Precooler heat transfer	4 x	600 MW
Intercooler heat transfer	4 x	380 MW
Number of tubes, size and heat transfer area:-		
IHX - Fuel 12,600 8mm OD x 6mmID x 5.8m long	4 x	1620 m^2
Blanket 12,600 8mm OD x 6mm ID x 5.8m long	1 x	1620 m^2
Recuperator 36,200 15mm OD x 12mm ID x 11.2m long	4 x	17200 m^2
Precooler 12,150 12.5mm OD x 10mm ID 2x6.4m long	4 x	6100 m^2
Intercooler 12,150 12.5mm OD x 10mm ID 2x5.5m long	4 x	5230 m^2

TABLE II

REACTOR AND PLANT PARAMETERS

(Plant List No. Ref. designated PLN..)

REACTOR

Thermal Power:-

	Total	6600 MWth
	Maximum core (at start of fuel cycle)	6600 MWth
	Maximum blanket (at end of fuel cycle)	990 MWth
	Nominal core diameter	3.0 m
PLN1	Core volume (effective)	16.5 m^3
	External core salt volume in primary circuits	24.0 m^3
	Total core salt volume	40.5 m^3
	Composition of core salt: (Pu ≙ 10% of heavy metal)	40/60 mol% $(Pu+U)Cl_x/NaCl$
	Melting point of core salt	577°C (850 K)
	Pu239 (equivalent) inventory, core and external primary circuit	≙ 10 te
PLN3	Volume of blanket surrounding core	50 m^3
	External blanket salt volume in primary circuit	6 m^3
	Initial composition of blanket salt	40/60 mol% $UCl_3/NaCl$
	Specific core power based on nett core volume	400 MW/m^3
	Approximate neutron fluxes:-	
	(a) At core centre	2×10^{16} n/cm^2s
	(b) At core blanket membrane	5×10^{15} n/cm^2s
	(c) At outer vessel	3×10^{14} n/cm^2s

CORE/BLANKET MEMBRANE PLN2

Formed from double walled spherical shell, outer wall forms membrane separating core and blanket salt, inner wall separates downward cool fuel salt flow from central upward flow region, box construction for stiffness.

Outer diameter (core boundary)	3.0 m
Inner diameter at flow separation	2.4 m
Thickness of material	4-5 mm
Material of construction -	Molybdenum,TZM
Outlet flow duct diameter 4 off	600 mm
Inlet flow duct diameter 4 off	600 mm
Core drain central duct diameter	600 mm

OUTER REACTOR VESSEL PLN4

Nominal dimensions

Height	6.5 m
Outer diameter	7.0 m
Inner diameter	6.8 m
Top flange diameter (including support rim)	8.6 m
Accident design differential pressure	5 bars
Working differential pressure (nominal)	2 bars
Working temperature (cooled by blanket salt internally)	≃ * 650°C
Material of construction	Hastalloy N or, possibly, stainless steel
Shell thickness, nominal	60 mm

(Top plate fabricated and webbed)

*Top flange insulated and cooled with incoming blanket salt where necessary.

REFLECTOR PLN5

Material - not fully specified	Graphite blocks + dense metal
Nominal thickness allowed	1 m
Cooling	Cool blanket salt
Working temperature (not calculated)	≃ 700°C

SHIELDING AND CATCHMENT FUNNEL PL6

Details not specified

SALT TO HELIUM HEAT EXCHANGERS (IHX)

	PLN10 Fuel Salt	PLN20 Blanket Salt	
Number of tube bundles	4	1	
Heat transferred, each	1650	990*	MWth
Salt temperature inlet/outlet	1050/650	860/650	°C
Salt mass flow rate (each)	8250	9400	Kg/s
Salt velocity ducts/tubing	10/8	10/8.3	m/s
Helium temperature inlet/outlet	415/850	415/800	°C
Helium mass flow rate (each)	732	495*	Kg/s
LMTD	215	122	°C
Tubing ID/OD	6/8	6/8	mm
No. of tubes per bundle	12,600	12,600	
Tube spacing, square pitch	15.1	13.0	mm
Length bare tubes (minimum)	5.9	5.9	m
(Heat transfer coefficient on helium side enhanced by 50% with roughening)			
Mean heat flux	102	61	w/cm^2
Heat transfer area	1620	1620	m^2

*At end fuel cycle, core parameters reduced accordingly.

FU~~ AND BLANKET SALT PRIMARY PUMPS

	PLN11 Fuel (Core) Salt	PLN21 Blanket Salt	
Number of pumps	4	4	
Inlet salt temperature	650	650	°C
Inlet salt density	3380	3380	Kg/m^3
Flow rate (each pump)	2.85	0.73	m^3/s
Pressure rise	2270	2400	KN/m^2
Number of stages	2	2	
Pressure rise/stage	1200 (say)	1200	KN/m^2
Velocity increase across each stage required	27	27	m/s
Delivery pressure (nominal)	6000	6000	KN/m^2
Typical dimensions for shaft speed of	2000	3000	rpm
Shaft diameter	150	100	mm
Outer diameter of flow passage at inlet for 10 m/s	620	350	mm
Impellor throat diameter	400	250	mm
Impellor outer diameter	580	360	mm
Outer diameter of removable flow passage and impellor impellor components	640	384	mm
Delivery pipe diameter for 8 m/s	670	370	mm
Outer diameter of pump barrel casing	900	500	mm
Pumping power at 70% overall efficiency (each pump)	9.4	2.5	MWe
Total pumping power	4 x 11.9 = 47.6		MWe

(see Figure 8 for salt pressure distribution and balance with helium pressure)

FUEL AND BLANKET SALT HEADER TANKS

	PLN12 Fuel Salt	PLN22 Blanket Salt	
Salt volume in header tank (at melting point)	1	1	m^3
Expansion volume	8	7	m^3
Total internal volume (including gas space)	12	10	m^3
Four tanks each:-			
Height	4	3.5	m
Internal diameter	0.98	0.98	m
Wall thickness	60	60	mm
Design pressure	5	5	bars

DUMP SYSTEM

PLN43	Core dump valves	4 off effective bore	0.4 m
	Minimum flow area, 3 out of 4 valves		0.39 m^2
	Core salt discharge time taking C_D as 0.5 (including inventory in heat exchangers, pumps and ducts)		14 seconds
	Venturi effects will be used to reduce actual diameters of valves and central core drain duct to reduce inventory and heat generated.		
PLN44	Blanket dump valves - as core PLN43		
PLN45	Fusible discs in case of reactor vessel or primary circuit gross failure	4 off nominal diameter	0.4 m
PLN42	Outer dump tank containment diameter		8 m
	height		9 m
	thickness (in SS)		70 mm
	Nominal accident design differential pressure		5 bars
	Material TZM lining, SS outer with NaK cooling in between		

Dump Tanks

	Core Salt	Blanket Salt	
PLN	40	41	
Number of dump tanks	4x3	4x3	
Internal diameter	1.16	1.30	m
Overall height	4.8	5.8	m
Effective salt height	3.75	4.4	m
Volume of salt in each tank	3.33	5.05	m^3
No. of NaK U tubes in each tank	54	22	
Size of U tubing, outer diameter	44	44	mm
inner diameter	40	40	mm
Heat transfer area/tank	55.5	28	m^2
Material, tubing and tank (lining?) - TZM			

Peak core salt heat removal conditions at t = 11 seconds after loss of normal cooling:-

Maximum salt temperature (with thorough mixing)	1124	°C
NaK coolant inlet temperature	30	°C
NaK coolant outlet temperature	$\simeq$ 600	°C
Overall heat transfer coefficient for salt natural convection with salt to tubes Δt 400°C	0.13	w/cm^2°C
Heat flux (mean) across U tubes	105	w/cm^2
Total heat transferred (12 tanks)	700	MWth
Heat transport is maintain up to t = 115* seconds with 170 m^3 (total) reserve of cool NaK equivalent to	700	MWth
Natural circulation heat transport from t = 116 to t = 315 seconds	300	MWth

*Flow restriction may be employed to spread this heat capacity over a longer period.

...f sustaining Core Salt Heat Removal Conditions

NaK inlet temperature	300	°C
NaK outlet temperature	600	°C
Maximum salt temperature (with thorough mixing)	≏ 800	°C
Heat transfer capacity (total)	234	MWth
Heat transport with NaK natural circulation	300	MWth
Heat transfer to steam in NaK-H_2O boilers	200	MWth

These conditions will apply for t >315 seconds. Up to this time the sum of heat removal at boilers plus enthalpy rise of fuel salt and NaK will equal the delayed neutron and fission product heating.
NaK U tube subheaders in dump tanks are duplicated, but total number of U tubes is not.
NaK pipework is duplicated in area as well as in number of pipes, i.e., 24 flow and return pipes each 280 mm ID/310 mm OD.
Boilers and fan coolers are duplicated in numbers but heat transfer area is not duplicated. Cool NaK reserve is not duplicated.
Thus in the event of a failure of part of the NaK or boiler circuits heat will still be removed from a dump tank but as only half the required heat transfer area is available in the dump tank and boiler some overheating will occur but can be tolerated as an accident condition (≏1350° maximum salt temperature for 200 MW heat removal).

NaK-H_2O Boilers PLN49 - 24 off

Heat transfer/boiler	8.4 MWth
Water/steam temperature (atmospheric pressure)	100 °C
Radiation mean heat flux (across insulating air gap)	1 w/cm^2
Surface area/boiler	840 m^2
NaK tubing ID/OD	23/25 mm
Water tubing ID/OD	28/32 mm
Number of tubes at 4 m long/boiler each 0.315 m^2 surface area	2680
Internal boiler csa for s/d = 1.5, square pitch	6.1 m^2
ID/OD boiler	2.8/2.85 m

Air Cooled Condensers PLN50 24 off + 3 off similar PLN107

Heat transferred to air, each	8.4 MWth
Surface area, each	2600 m^2
Water/steam flow (velocity = 2 m/s natural circulation), each	3.7 Kg/s
Air flow for 50°C rise in temperature, each	150 m^3/s
Frontal area 5.5 m x 5.5 m each (or subdivided into 3x2 standard)	30 m^2
Fan power, each of 24/total for all dump circuits (72 HP/1720 HP)	53/1300 kW

RECUPERATOR PLN 165

Parameters quoted for one of four recuperators

Heat transferred	1080	MWth
Helium flow (same for shell and tube side)	732	Kg/s
No. of tubes	36,200	
Tube size: inner diameter	12	mm
outer diameter	15	mm
Tube pitch (Square pitch s/d = 1.32)	19.8	mm
Tubes flattened at inlet/outlet of shell side to 4.5 mm outer radius edges with 9.4 mm flats		
Tube length	11.2	m
Tube bundle diameter	4.22	m

	Shell side	Tube side	
Helium pressure (nominal)	18	62	bars
" pressure drop	.45	.61	bars
" inlet/outlet temperature	473/188	130/415	^oC
" velocity	63	32	m/s
" heat transfer coefficient*	0.165	0.331	$W/cm^2 {}^oC$

Overall heat transfer coefficient	0.11	$W/cm^2 {}^oC$
Temperature difference	58	^oC
Heat Flux	6.3	W/cm^2
Effectiveness	.83	

*Heat transfer v. pressure drop has not been optimised, relatively poor heat transfer is used in this case for low pressure drop conditions to obtain conservative size. Higher pumping pressures could be optimised against saving on heat transfer area.

PRECOOLER AND INTERCOOLER PLN 166 and 167

Parameters quoted for one of four units	Pre-Cooler	Inter Cooler	
Heat transferred	600	380	MWth
Helium flow	732	732	Kg/s
No. of tubes (water flow inside tube)	12,144	12,144	
Tube size inner diameter	10	10	mm
outer diameter	12.5	12.5	mm
Tube pitch ($\triangle$ s/d = 1.33)	16.6	16.6	mm
Tubes arranged 12 sets of 22 x 46 in annular array.			
Total length, divided into two sections to give water flow in parallel to reduce water Δ p	2 x 6.4	2 x 5.5	m
Inner/outer diameter of tubing "annulus"	1.5/2.84	1.5/2.84	m
Helium pressure	18	33.3	bars
Helium pressure drop	0.5	0.5	bars
Helium inlet/outlet temp.	188/30	129/30	^{o}C
Helium nominal velocity	24	16	m/s
Helium heat transfer coefficient	0.215	0.231	
Cooling water (tube side):			
Flow	9.6	6.0	m^3/s
Temperature inlet/outlet	20/35	20/35	^{o}C
Tube pressure drop	2.83	<Precooler	bars
Water heat transfer coefficient	2.5	2.5	$W/cm^{2o}C$
Overall heat transfer coefficient	0.19	0.19	$W/cm^{2o}C$
LMTD	52	38	^{o}C
Mean Heat Flux	9.8	7.25	W/cm^2
Heat transfer surface area	6,100	5230	m^2
Maximum core size of cooling water pipes within pressurised area and at penetrations:	280	280	mm
Design pressure (external)	66	66	bars
Material - carbon steel			

COOLING WATER SYSTEM FOR PRECOOLER & INTERCOOLERS PLN 169

Inlet water temperature	20°C
Outlet water temperature	35°C
Total heat removed from all precoolers and intercoolers	3920 MW
PCV cooling	12 MW
Total c.w. flow rate	62.4 te/s
Four sets of piping each consisting of two supply pipes and pumps and two return pipes (see Flow diag. Fig. 10))	
Outside building pipe bore	2.16 m
Inside building pipe bore	1.5 m
Distributor pipework and headers max. size = 1.2 m bore.	
Water velocity in pipework outside building	2.1 m/s
" " " " inside building	4.2 m/s
" " " " in precooler and intercooler tubes	5.0 m/s
Overall CW pressure drop (for pipework shown on figures and precooler).	450 KN/m² 65 psi
Overall CW pumping power for above conditions at 85% η	33 MW

PLANT POWER BALANCE

Net station output	2500 MWe
Core and blanket salt pumping power	50 MWe
Remaining station load including C.W. pumping and auxiliary circuit heating to prevent salt freezing	$\simeq$ 50 MWe
	2600 MWe
Transformer loss 0.5%	13 MWe
Gross alternator output 4 x 654 MWe	2613 MWe
Mechanical losses in T/A set and alternator losses 1.5%	39 MW
Total low pressure turbine gross power 4 x 663 MW	2652 MW
Cycle efficiency = LPT Gross power	40.8 %
~~Nett heat input required to gas turbine cycle~~	~~6500 MWth~~
Heat loss to PCV liner	12 MWth
Allowance for gas by-passing	50 MWth
Total heat transferred at IHX 4 x 1640 MW	6562 MWth
Nuclear power allowing for 45 MW pumping losses)	6517 MWth
Net station efficiency = $\frac{\text{Net Station Output}}{\text{Total Nuclear Heat}}$	38.4 %

HELIUM PRESSURES

See Fig. 7 cycle diagram for pressure distribution in helium circuits.

Helium flow	4 x 732	Kg/s
High pressure compressor (HPC) delivery pressure	62.5	bars
Recuperator (tube side) Δp	0.61	bars
IHX ("shell" side) Δp	1.32	bars
Duct losses (H.P. side)	0.37	bars
H.P. turbine inlet pressure	60.2	bars
H.P. turbine outlet pressure	19.2	bars
Recuperator (shell side) Δp	0.41	bars
Precooler Δp	0.52	bars
Duct losses. LP side	0.27	bars
LPC inlet pressure	18.0	bars
LPC outlet pressure	33.0	bars
Duct losses. LPC - intercooler	0.12	bars
Intercooler Δp and duct loss to HPC	0.19	bars
HPC inlet pressure	32.7	bars

DRAIN TANKS

CORE DRAIN TANK PLN 61 - 1 off

Net storage volume	44 m^3
Internal diameter	6.25 m
Net height for salt (allowing 80% of csa)	1.75 m
Overall height	2.5 m
Heat removal capacity, through NaK natural circulation system in parallel with dump tank cooling allowing for 5 days decay in dump tanks if all core salt is drained.	30 MW
Working pressure (in case of mal operation allowing helium over pressure)	60 bars
Working temperature (bulk salt)	≏ 800°C
" " vessel walls cooled by NaK	650°C

Materials of construction

Vessel	Hastelloy N
Cooling U tubes	Molybdenum or TZM

BLANKET DRAIN TANK PLN 62 - 2 off as PLN 61

Total net storage volume for 2 tanks	88 m^3
Less heat removal capacity is required say	5 MW

SPARE DRAIN TANK(S) PLN 63 1 or 2 off as PLN 61

There is room for two spare drain tanks if required on the same level and a further 5 replacement tanks on the floor above.

FUEL/BLANKET SALT PREPARATION AND CLEAN-UP PLNos. 70, 71, 73, 80

Not detailed

HIGH PRESSURE CLEAN HELIUM STORE PLN 110

Mass of helium to fill reactor vault and gas turbine plant at working conditions of temperature and pressure = 11 te (approx.)

Storage volume at 20°C and 138 bars (2000 psi) = 480 m^3 say 500 m^3.

Say 4 vessels. 20 m effective length x 2.82 m ID each 125 m^3 capacity (Room for 6)

Thickness 28 mm outer diameter 3.38 m say 3.4 m O.D.

HIGH PRESSURE DIRTY GAS STORE PLN 130

To receive active or contaminated gas from reactor vault, gas turbine plant, header and dump tanks in case of large leakage requiring immediate pumping down to atmospheric pressure.

Assuming rapid pumping out and loss of enthalpy equalling pumping energy, i.e. gas enters store at the same temperature as in working condition but compressed to 138 bars pressure, required volume = 1250m^3 nominal value (subject to revision on auxiliary plant requiring any urgent pumping down).

10 vessels of size as for the H.P. Clean Active Store allowed for.

Mean temperature = 480°C max.

Material for PLN 110 and PLN 130: Low alloy steel

OFF GAS SYSTEM

OFF GAS RECIRCULATION LOOPS PLN 100

Inc. Bubble generators across main core/blanket pumps - not detailed.

Delivery of off gases, via the dump tanks for preliminary cooling and deposition of particulates, to short term delay bed.

2-DAY SHORT TERM GASEOUS FISSION PRODUCT DELAY BED PLN 103

Total length of bed formed from 50 mm bore piping	= 114,000 m
No units in parallel at 75 mm pitch	= 666
No U tubes in series/unit	6
Depth of immersion of U tubes	14 m
∴ Length of each unit	170 m
Wt. of charcoal adsorbent	110 te

LONG TERM GASEOUS FISSION PRODUCT STORE PLN 105

Storage required $1 m^3$/year at 60 bars

Allow double storage space for alternate operation

i.e. 2 x 56 m of 150 mm bore pipe

(Space is available for whole reactor life storage if required)

COOLING TANK FOR 2 DAY DELAY BED AND LONG TERM STORE PLN 104

Effective length	25 m
Width for short term delay bed	5 m
Additional width available to accommodate long term store	1.6 m
Heat removal capacity equivalent to $\gamma + \beta$ heating at 1 day after discharge (initial heat will be removed in dump tanks)	40 MW

PLN 106 Condenser situated above tank has 40 MW capacity

PLN 107 3 Fan coolers provided each allowing operation

For limited time on two units as PLN 50. Removal 16.6 MW

N.B. The above data for the off gas system are subject to revision when fission product spectrum from Pu in molten salt is fully investigated, some reduction may be possible as not all fission products will be gaseous or volatile.

PRESTRESSED CONCRETE VESSEL PLN140

See Fig. 6 for details.

Height	35.4	m
Overall diameter	30	m
Cavity design pressure	66	bars
Circumferential loading for full design pressure in cracks	10,000	te/m
Circumferential prestressing - number of bands	30	
load/band	10,000	te/m
- channel dimensions: width	600	mm
depth	300	mm
Longitudinal prestressing:-		
Reactor cavity - No. of tendons	48	
Load/tendon	1000	te
Tendon	36x18	mm dia.
He Recuperator cavity - No. of tendons	22	
Load/tendon	780	te
Tendon	28x18	mm dia.
Precooler/Intercooler cavities - No. of tendons (effective)	11	
Load/tendon	780	te
Tendon	28x18	mm dia.

Turbine machinery cavities. Local horizontal and hoop tendons will be provided.

N.B. Figs 1 to 4 show circumferential stressing channels incorrectly.

CONTAINMENT BUILDING PLN150

For full helium release from coolant circuit including gas turbine plant, IHX volume, space around reactor and dump tanks at respective working pressures and temperatures. Total pressure of air and helium in containment assuming no energy loss = 1.24 bars above atmospheric pressure. Temperature ≏ 140°C.

Height of containment	77	m
Internal diameter	50	m
Design differential pressure	1.5	bars

Construction: reinforced or prestressed concrete possibly with inner steel lining and interspace for insulation and dealing with leakage.

REMOTE HANDLING FACILITIES

Handling Flask to take all plant. PLN180

Height (internal)		20	m
Internal clear diameter		9.2	m
Wall thickness (3" steel)		75	mm
Overall max. diameter		9.8	m
Weight	approx.	360	te

Storage/Remote Maintenance Areas

For heat exchanger, pumps, intercoolers or precoolers, drain tanks. PLN182.

Access hole	dia.	4	m
Plan size (with PLN183)	approx.	11x6.8	m
Depth		21.5	
Accessibility: One side and above			

For recuperators or items above PLN183

Access hole	dia.	5.5	m
Depth		21.5	m
Accessibility: One side and above			

For reactor with IHX and pumps if necessary. PLN184

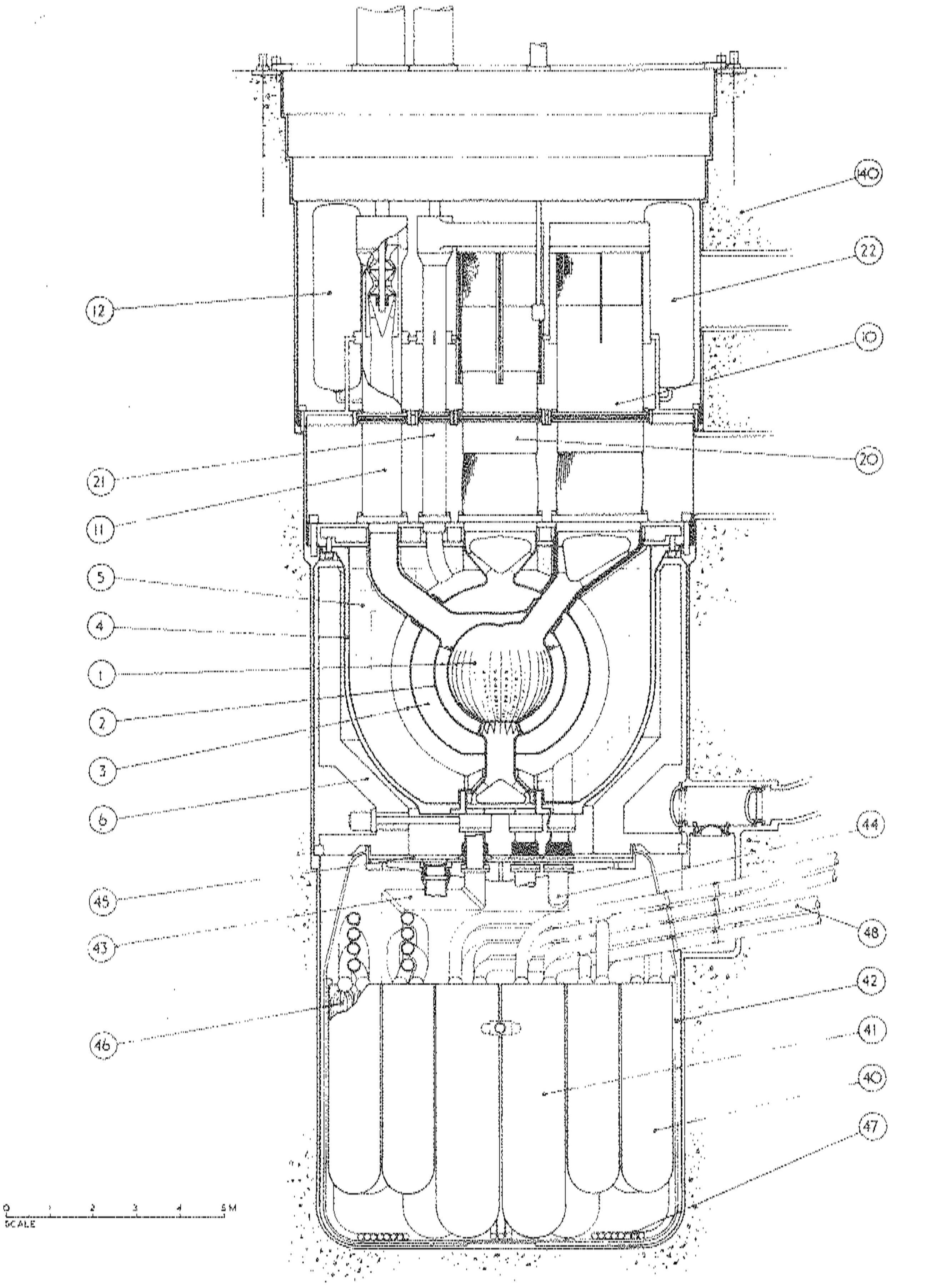
Access hole	dia.	9.2	m
Depth		14.5	m
Accessibility: Two sides and above			

For outer dump tank and contents PLN185 - identical to PLN184.

Remote handling for fuel salt preparation and cleanup, salt processing cells and helium cleanup plant, i.e., for PLN72, 73, 80, 109, 135.

Cell space - plan area	10x6.8	m
depth	21.5	m

Accessibility from both sides.



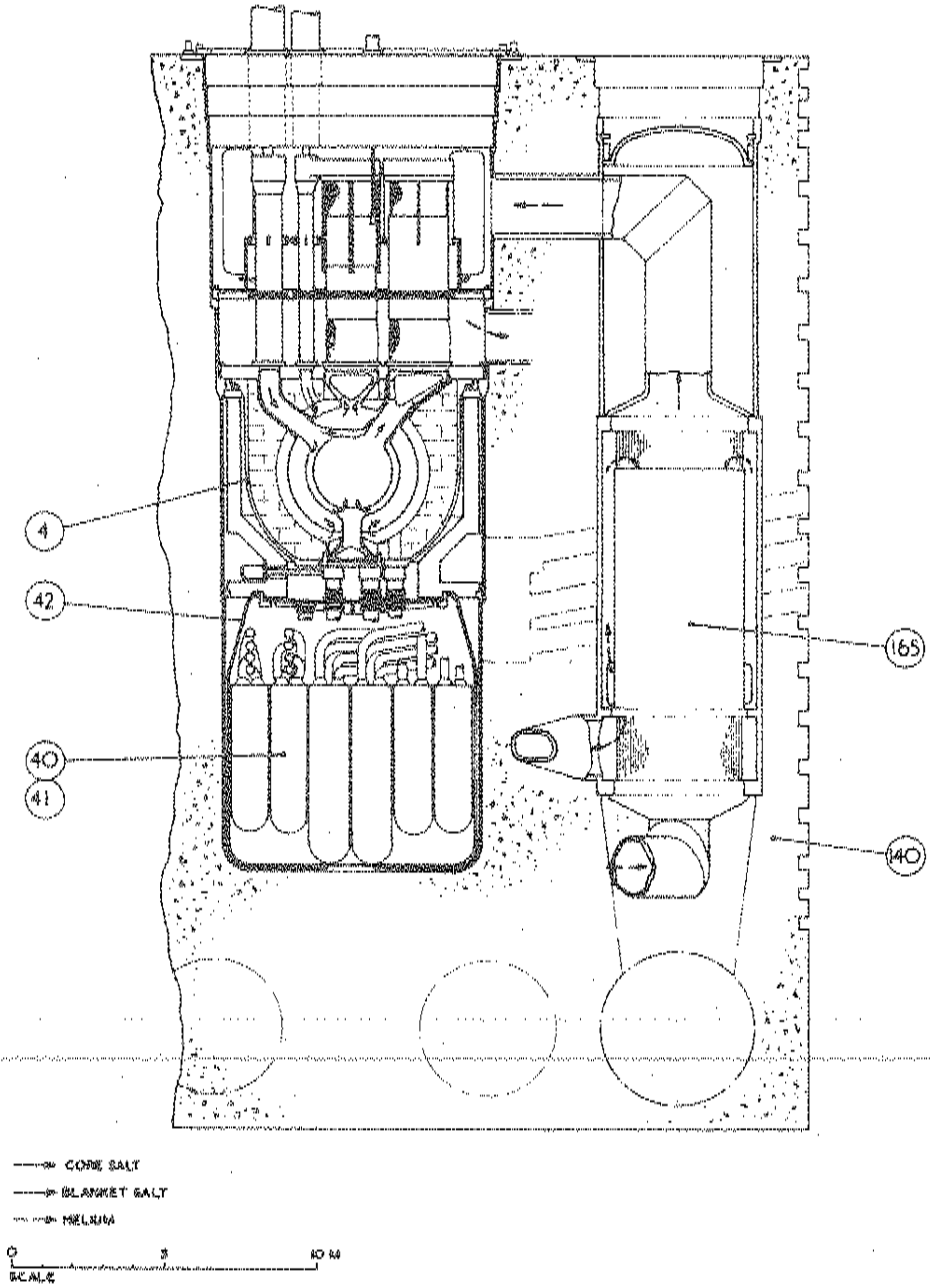
2500 MWe HELIUM COOLED MSFR

REACTOR, INTERMEDIATE HEAT EXCHANGERS & DUMP SYSTEM

FIG. 1

ITEM NUMBERS REFER TO PULL-OUT ITEM LIST AFTER LAST FIGURE

NOT-FOR-PUBLICATION (Commercial)

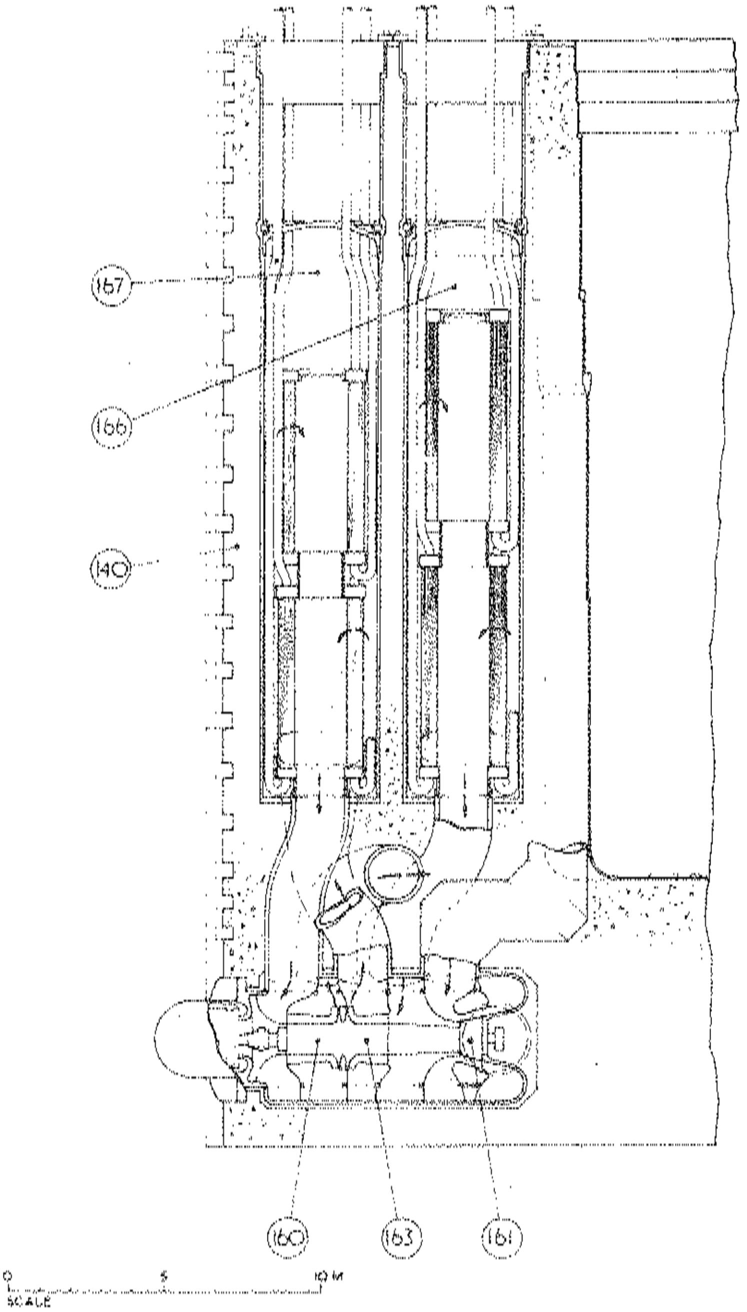


2500 MWe HELIUM COOLED MSFR

REACTOR & INTEGRATED GAS TURBINE PLANT WITHIN PCV

SECTION A - A FIG. 2a

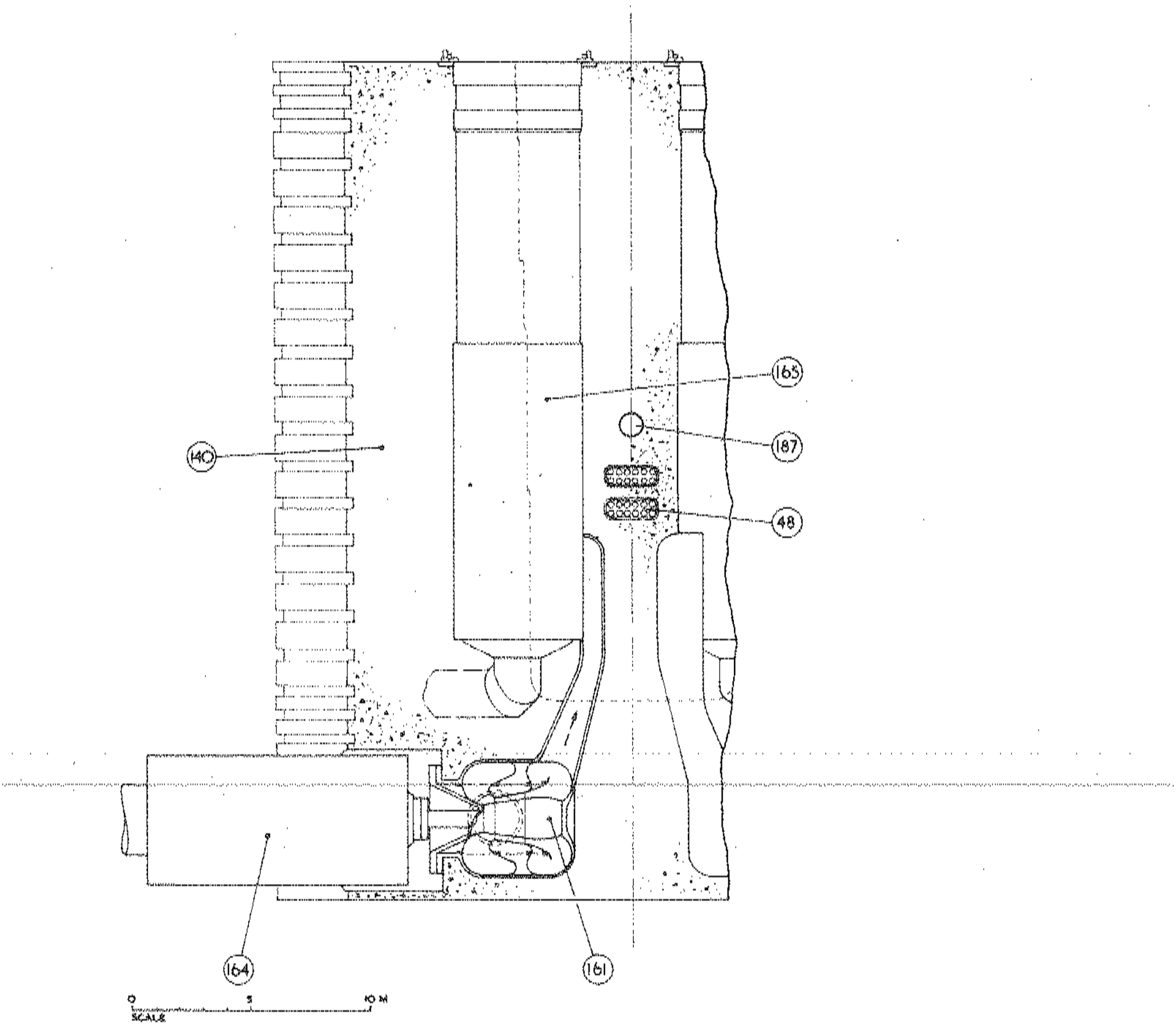
ITEM NUMBERS REFER TO PULL-OUT ITEM LIST AFTER LAST FIGURE



SECTION B-B

FIG. 2b

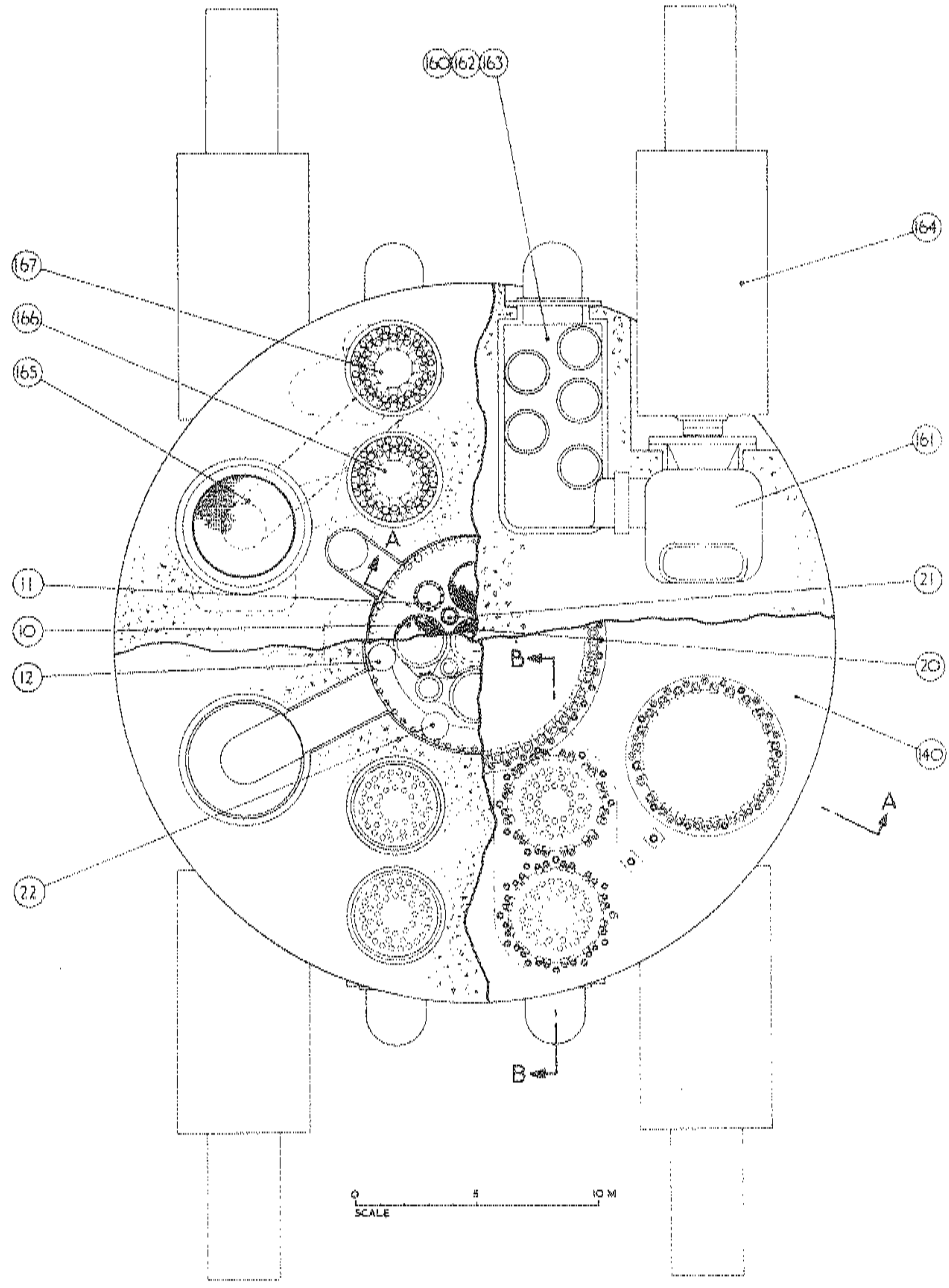
ITEM NUMBERS REFER TO PULL-OUT ITEM LIST AFTER LAST FIGURE



SECTION C - C

FIG. 2c

ITEM NUMBERS REFER TO PULL-OUT ITEM LIST AFTER LAST FIGURE



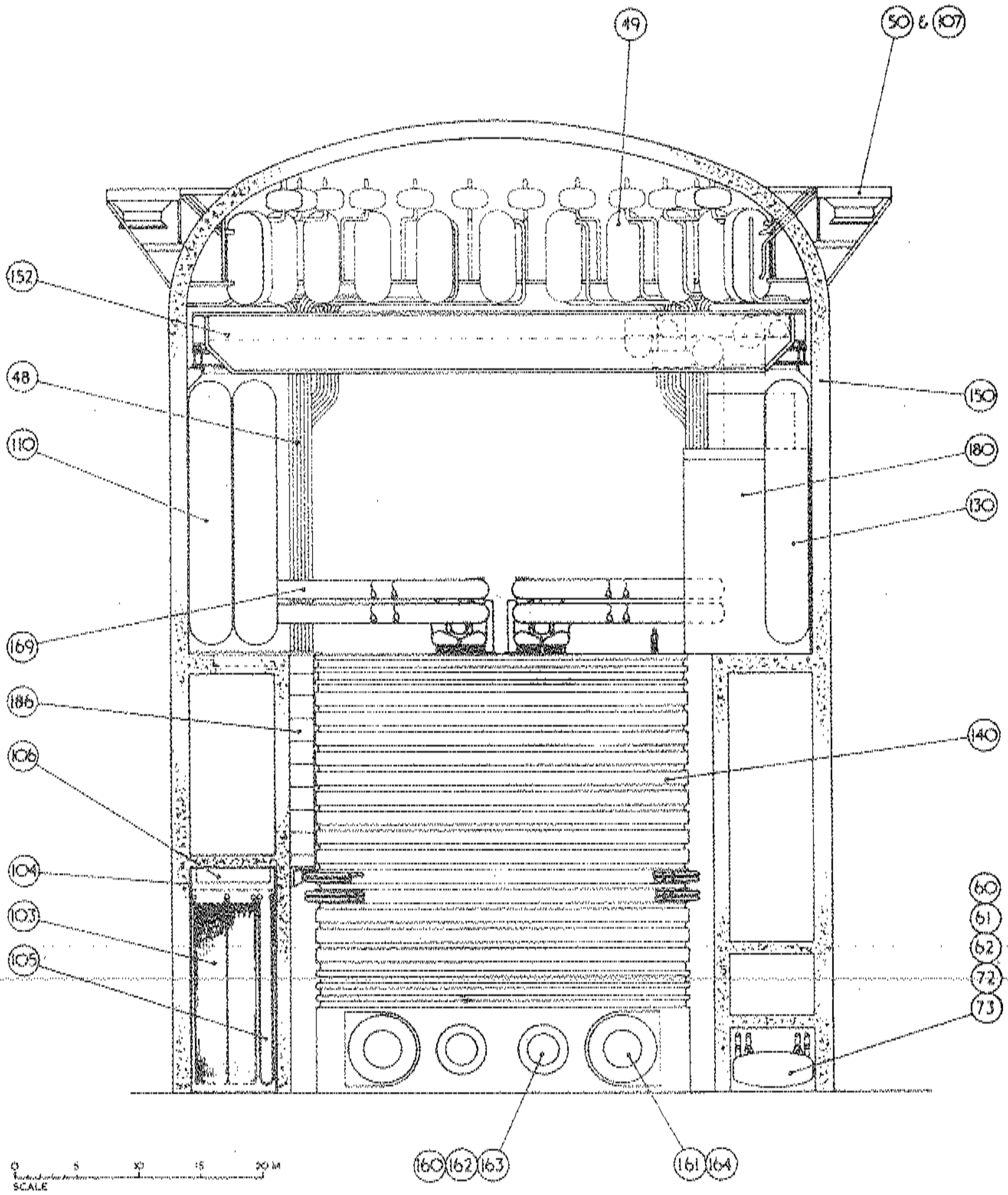
2500 MWe HELIUM COOLED MSFR

REACTOR & INTEGRATED GAS TURBINE PLANT WITHIN PCV

- PLAN SECTIONS -

FIG. 3

ITEM NUMBERS REFER TO PULL-OUT ITEM LIST AFTER LAST FIGURE

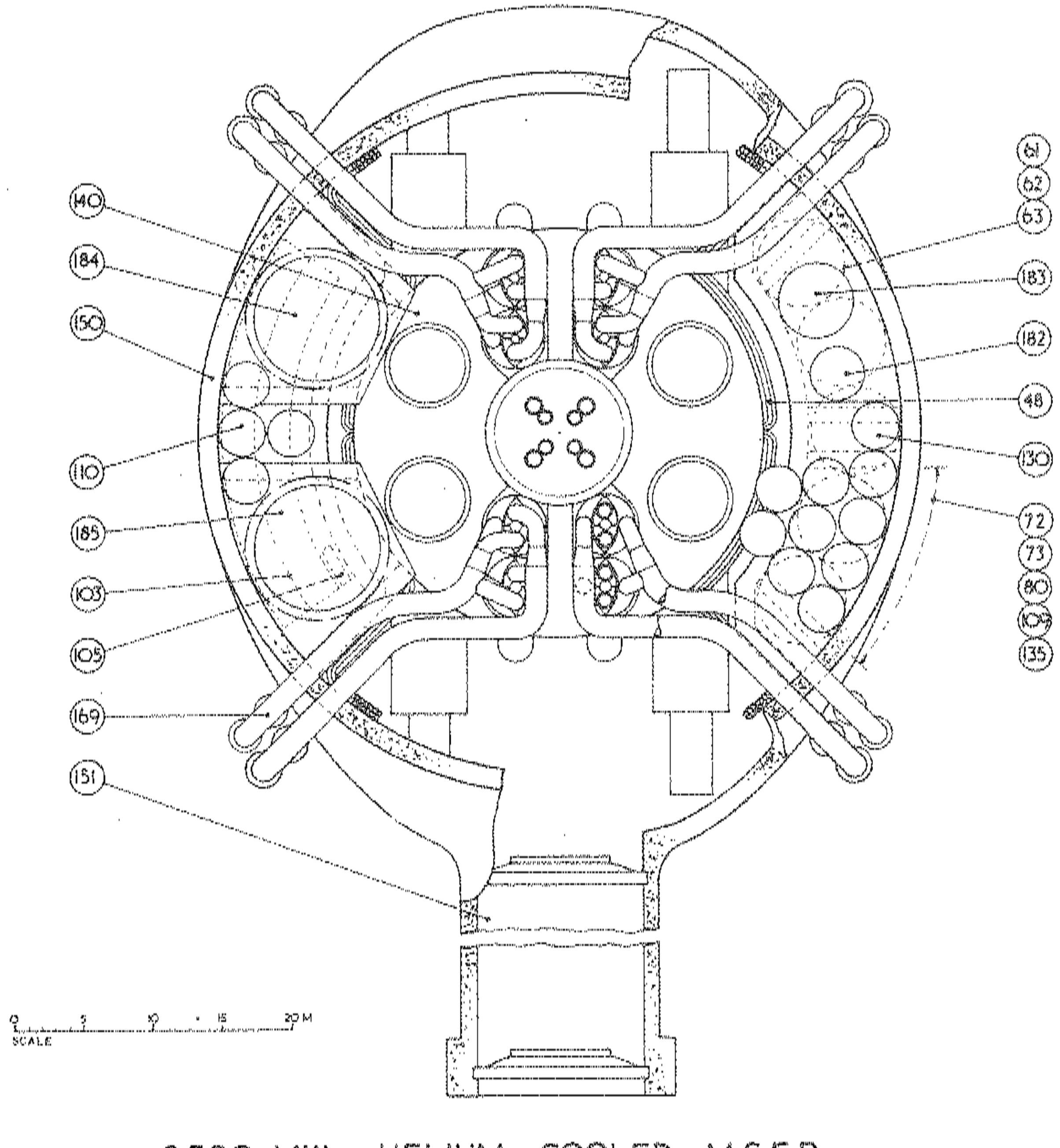


2500 MWe HELIUM COOLED M.S.F.R

BUILDING & AUXILIARY PLANT LAYOUT — ELEVATION

FIG.4

ITEM NUMBERS REFER TO PULL-OUT ITEM LIST AFTER LAST FIGURE

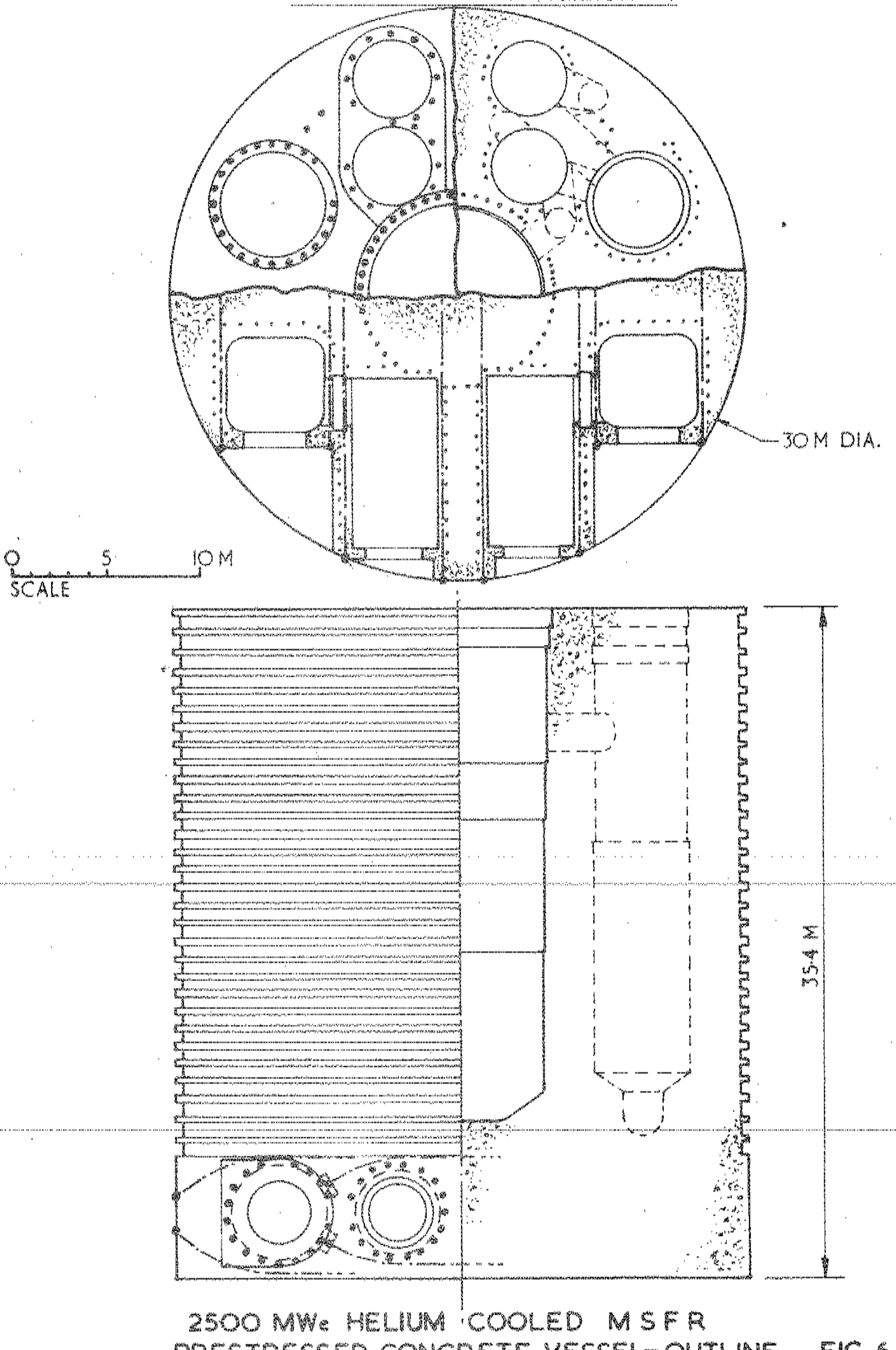


2500 MWe HELIUM COOLED MSFR

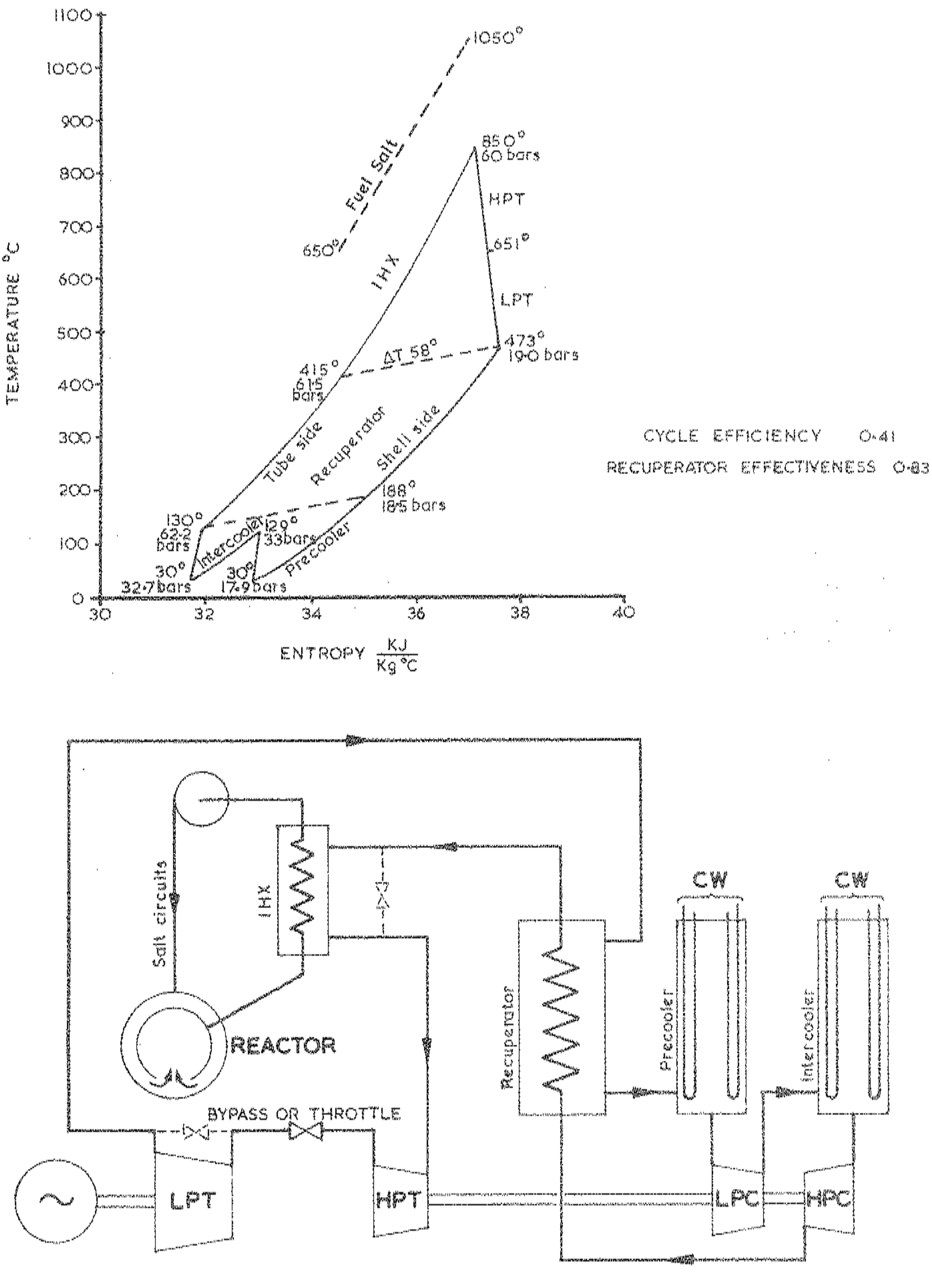
BUILDING & AUXILIARY PLANT LAYOUT — PLAN

FIG. 5

ITEM NUMBERS REFER TO PULL-OUT ITEM LIST AFTER LAST FIGURE



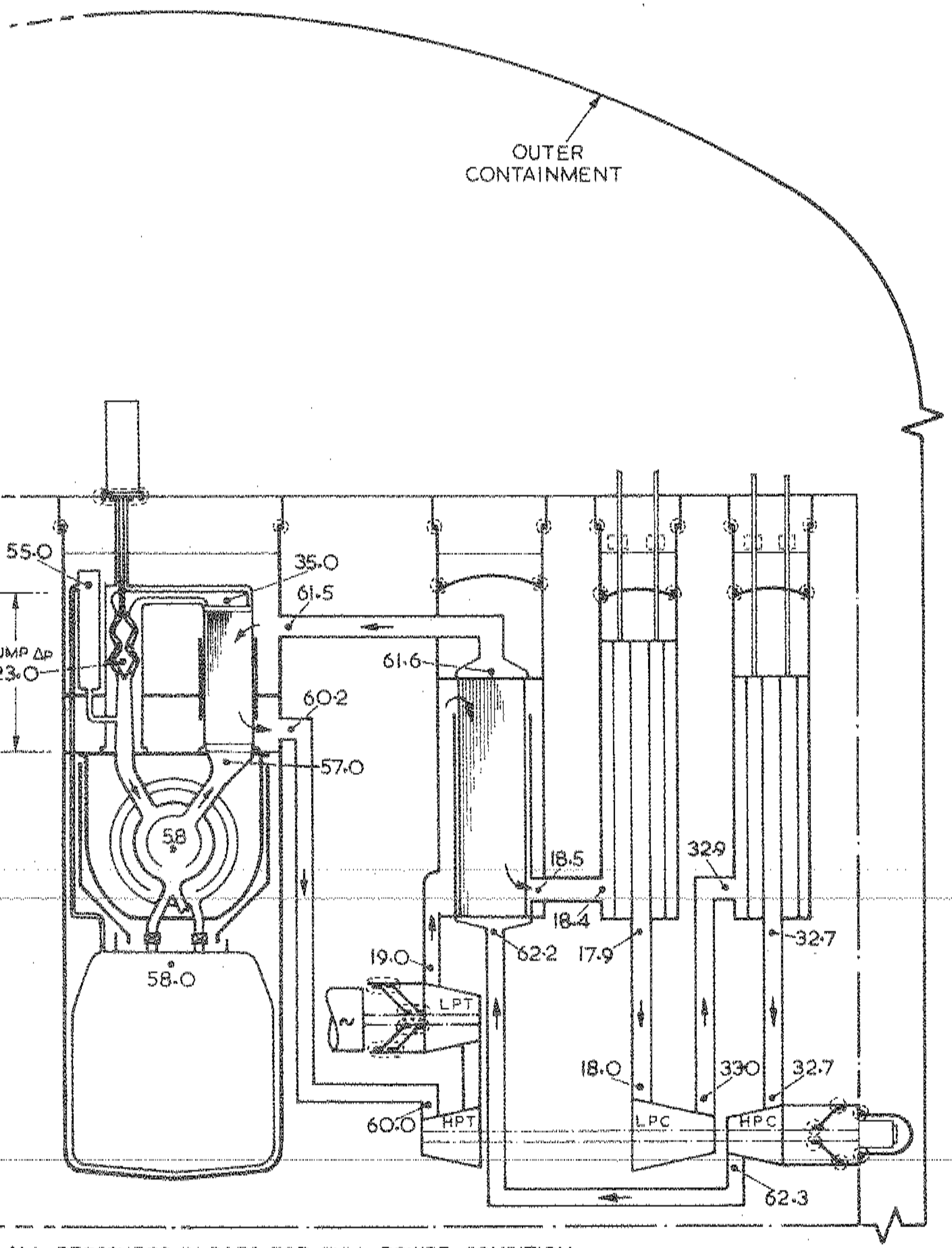
2500 MWe HELIUM COOLED MSFR
PRESTRESSED CONCRETE VESSEL-OUTLINE FIG.6



2500 MWe HELIUM COOLED MSFR

GAS TURBINE CYCLE FIG.7

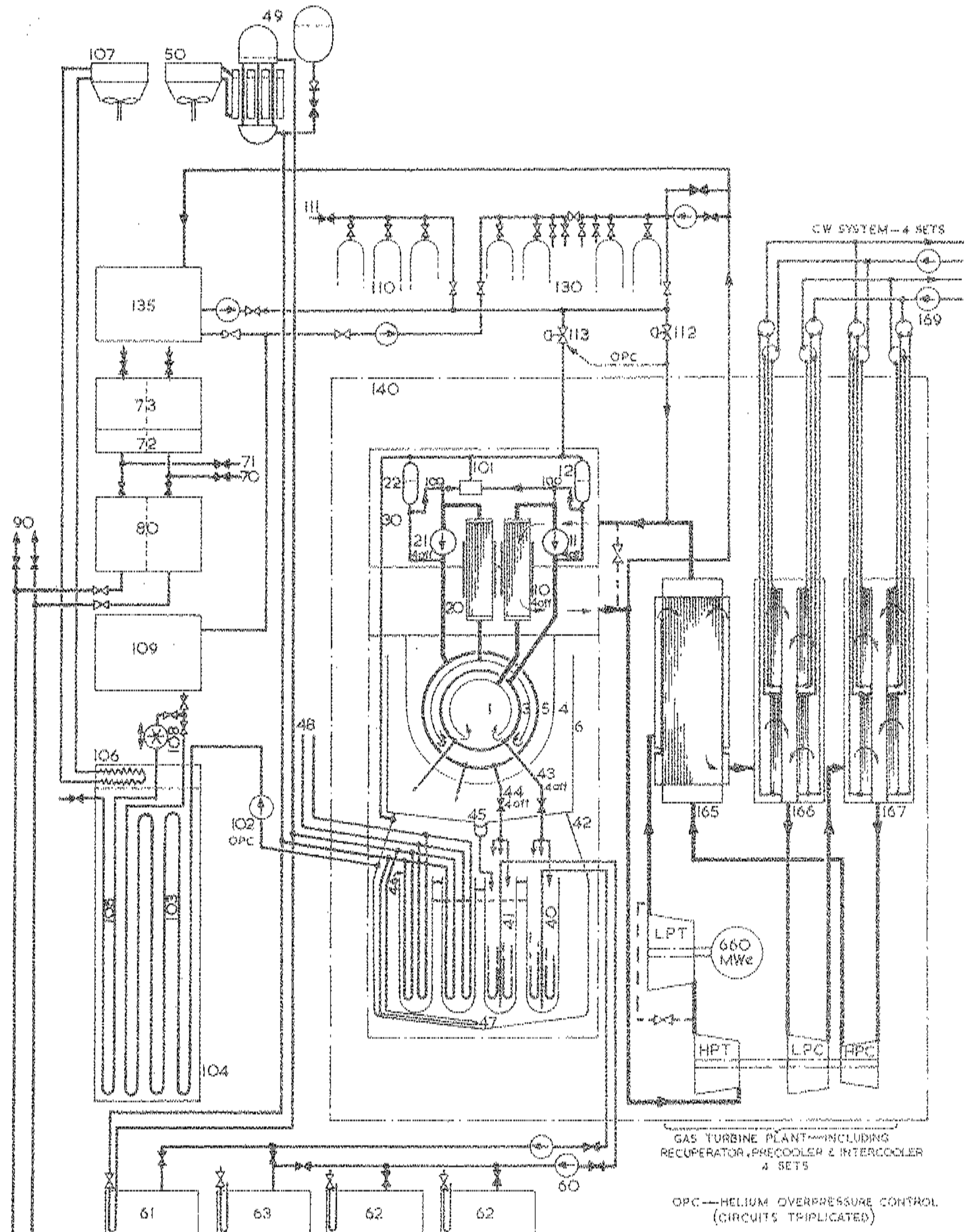
NOT-FOR-PUBLICATION (Commercial)



ALL PRESSURES IN BARS FOR FULL POWER CONDITION.
⊛ SEALING He TO ATMOSPHERE. ⬚ FLOW RESTRICTION

2500 MWe HELIUM COOLED MSFR

PRESSURE DISTRIBUTION & CONTAINMENT ENVELOPES FIG. 8

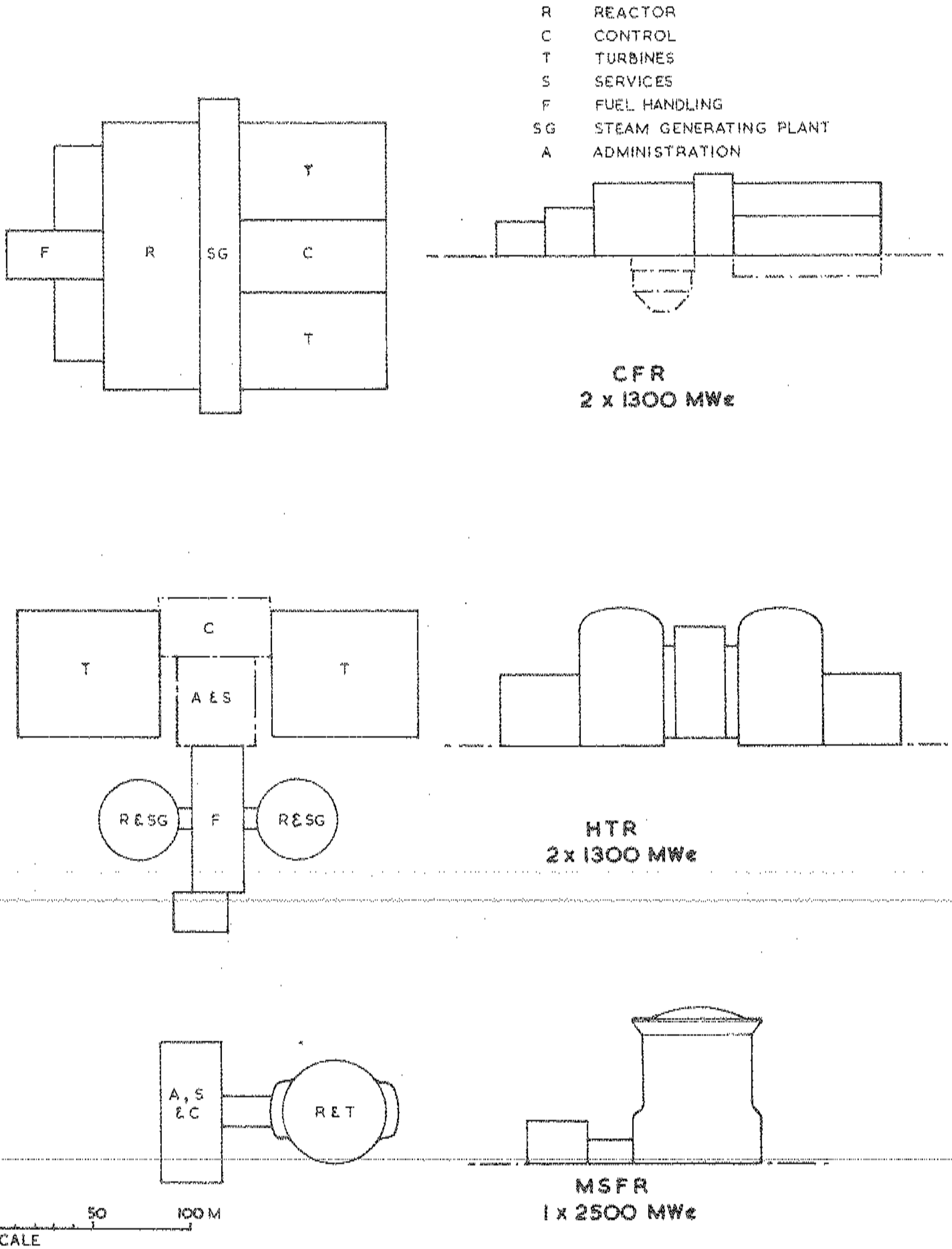


2500 MWe HELIUM COOLED MSFR

FLOW DIAGRAM

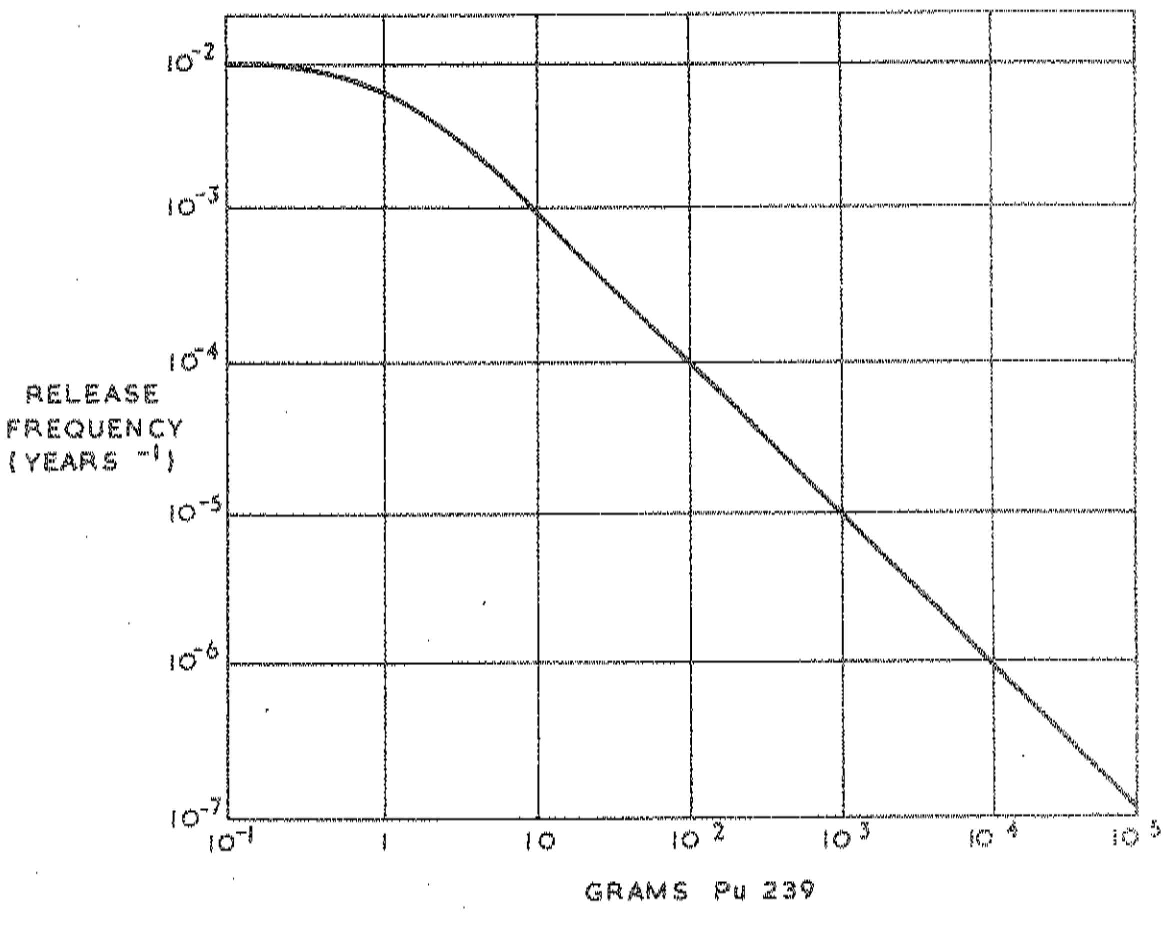
FIG. 9

NOT-FOR-PUBLICATION (Commercial)



2500 MWe HELIUM COOLED MSFR

SIZE COMPARISON OF MSFR WITH CFR & HTR FIG. 10



LIMITING Pu 239 RELEASE AGAINST PROBABILITY OF FAILURE

FIG. 11

ITEM LIST FOR HELIUM COOLED 2500 MWe MSBR WITH GAS TURBINE PLANT

Plant list Numbers refer to all figures, flow diagram and parameter list

1 Core
2 Core/blanket membrane
3 Blanket
4 Reactor vessel
5 Reflector (cooling not shown)
6 Reactor shielding and catchment funnel.
10 Core salt/helium heat exchangers (IHX) - 4 off.
11 Core salt pumps - 4 off
12 Core salt header tank
20 Blanket salt/helium IHX - one off with helium flow control.
21 Blanket salt pumps - 4 off
22 Blanket salt header tank
30 Pressure balance lines - header tanks to dump tank gas space.
40 Core salt dump tanks - 4 x 3 with interconnections
41 Blanket salt dump tanks 4 x 3 with interconnections
42 Outer dump tank containment
43 Core salt dump lines and valves- 4 sets
44 Blanket salt dump lines and valves - 4 sets.
45 Catchment area and fusible discs (4) in case of primary salt circuit leakage or failure
46 NaK U-tubes for cooling dump tanks with duplicated headers
47 Outer dump tank cooling NaK U tubes.
48 NaK cooling system pipework (duplicated) 24 flow, 24 return lines jointly serving core and blanket dump tank U tubes
49 NaK - water boilers with isolation air gap - 24 off
50 Steam condenser coolers with air blast fans - 24 off
60 Dump tank drain lines and pumps
61 Core salt drain tank - 1 off
62 Blanket salt drain tank - 2 off
63 Spare drain tank - 1 or 2 off
64 Drain tank salt removal pumps
70 Fuel salt supply to core and primary circuit
71 Blanket salt supply
72 New core and blanket salt holding and metering tanks with helium overpressure for pumping
73 New fuel and blanket salt preparation.
80 Clean up plant for fuel and blanket salts
90 Fuel and blanket salt major processing (for Pu separation and non-volatile fission product separation on or off site)
100 Off gassing loops
101 Separated off gases to dump tank space for cooling.
102 Off gas line and pump to fission product delay system.
103 2 day delay beds
104 Cooling tank for delay beds and long term store.
105 Gaseous fission product long term store.
106 Delay system heat removal isolating steam condenser.
107 Delay system heat removal fan driven CW cooler - 3 off
108 Off gas pumping from delay beds to long term store or return to salt head tanks or clean helium store via clean up plant.
109 Off gas clean up plant
110 Clean helium store
111 Helium filling point
112 Helium coolant pressure control
113 Fuel and blanket helium cover gas pressure control
120 Dump tank pressure relief to helium coolant circuit.
130 Dirty gas receiver
131 Dump tank cover gas blowdown to DGR
135 Coolant helium clean up plant
140 Prestressed concrete vessel
150 Outer containment building
151 Air lock for plant access (removable doors for very large plant)
152 Polar building crane

Gas Turbine Plant (4 sets)

160 High pressure turbine, HPT
161 Low pressure turbine, LPT
162 Low pressure compressor, LPC
163 High pressure compressor, HPC
164 660 MWe alternator
165 Recuperator
166 Precooler
167 Intercooler
168 Bypass control system (tentative)
169 Cooling water system and pumps (ext.)
180 Shielded handling flask for all plant

Storage positions with remote maintenance facilities for:-

182 Heat exchangers (IHX), pumps, intercoolers and precoolers, 21m deep
183 Recuperators, intercoolers, precoolers or smaller items
184 Reactor vessel and IXH/pumps complete if necessary, 14.5m deep
185 Dump tank complete assembly, -do-
186 Removable liner & blocks for access to prestressing cables
187 Access passage for maintenance of dump valves and disconnection of dump lines and NaK coolant pipes.

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