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1. INTRODUCTION

The present phase of nuclear reactor development is characterized by the trend to large units of thermal reactors and by the demonstrated competitiveness of nuclear power. Since the economical use of a nuclear plant in a power grid may call for a location close to industrial or population centers, siting of nuclear power plants should not be restricted by population density and other safety related site parameters.

Making a power plant independent on siting means avoiding any hazard to the environment. Therefore, the design of a containment of a site-independent nuclear reactor plant should accomplish the following points:

1. Leakage of activity during and after any accident suitably low
2. Integrity during accident
3. Leakage and integrity constant over plant life.

Several low leakage containment concepts have been proposed and have been applied to power plants of the present generation of thermal reactors /1/. Because absolute leaktightness of containment shells is technically not feasible, certain leakrates principally must be taken into account. At present a design leakage around 1 % of contained volume per day is considered reasonable with regard to safeguards requirement and economics. However, potential hazards to the environment depend both on the design leak rate of the containment and on the actual course of accident considered. In particular the time function of pressure and activity distribution in the contained volume has to be known. Hence, the containment design is dictated by the maximum conceivable accident for a particular plant, which we, therefore, call the Design Basis Accident.

In the following, we shall (1) explain the most important differences between the design basis accidents in thermal water reactors as compared to fast sodium cooled reactors, (2) describe the accident which was chosen as design basis accident in the conceptual design study of a 300 MWe prototype reactor,

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(3) draw conclusions from these considerations for the engineering layout of the containment and engineered safeguards. Finally, we will give some data on the activity release showing the effectiveness of the double containment concept chosen for that prototype reactor.

2. DESIGN BASIS ACCIDENTS IN THERMAL WATER REACTORS AND FAST SODIUM REACTORS

The accident which determines the containment design of thermal water reactors is commonly considered to be the main coolant line rupture with subsequent failure of emergency cooling /2/. Because the thermal water reactor is inherently shut down by loss of coolant, this accident leads to core meltdown only due to fission product decay heat accompanied by release of fission products and other radioactive material from the fuel to the primary system and hence into the reactor building. It is difficult to establish the release rates from the fuel for the various fission products as function of time and temperature for a given accident. However, we can state, that meltdown and subsequent fission product release in a thermal water reactor is certainly a matter of hours. This means that energy transfer from the primary system is slow and no phenomena of rapid energy transfer should be anticipated.

Furthermore, the maximum temperature possible during this design basis accident will not exceed the fuel melting temperature because of the slow power generation by fission product decay in the system. Therefore, it is concluded that in thermal reactors it is essentially thermal processes which govern the energy and activity release in a large accident and determine the requirements for the containment design.

Design basis accidents in sodium-cooled fast reactors may differ remarkably from those in thermal water reactors. This is mainly due to two reasons:

First the fuel in the core is not arranged in its most reactive configuration, i.e. inadvertent movement of fuel can lead to increase in reactivity. Secondly, the void coefficient of reactivity in sodium is positive at least over the central region of the core.

The first effect is not considered to be a major problem, because inadvertent movement of fuel into a more reactive configuration can be precluded by design. Bowing effects, which were troublesome to the Mark II-core of the EBR-I plant are now understood thoroughly.

The second effect can be of importance if a complete safety system failure is assumed accompanied by loss of flow due to pump failure and subsequent overheating in a number of coolant channels. Then, with further pessimistic assumptions the sodium void effect may lead to a nuclear excursion with subsequent fuel melting and even fuel vaporization. Although the prompt negative Doppler effect and core disassembly will terminate the reactivity excursion, an energy release of several hundred MWsec may then be envisaged /3/. Additionally, thermal processes can take place due to the fact that during the excursion fuel may be dispersed into sodium still present in the coolant channels /4/. Depending on how the fuel comes into contact with sodium, a rapid energy transfer may be encountered. This fuel-coolant interaction could add to the destructive potential of the nuclear phase of the excursion.

Discussing more principally the safety philosophy of fast reactors we must emphasize that both effects are only virtually relevant, i.e. a number of unprobable events, particularly failure or the safety system, has to be assumed before fuel movement or sodium voiding can occur. We should always bear in mind that the safety of large nuclear plants relies on engineered safeguards anyway. Therefore, assuming the reliable performance of engineered safeguards under accident conditions is one way to develop a safe nuclear reactor design and is - as we believe - a realistic way. Assuming the failure of engineered safeguards during an accident is another way of safety philosophy which is certainly more pessimistic and conservative. As will be outlined later in detail we took the second (i.e. the conservative) way for our conceptual design study, which is not necessarily in agreement with other fast reactor development groups. For instance, the British do not implement in their design philosophy nuclear excursions which deliver major amounts of energy. Therefore, the design philosophy in Germany of taking into account extremely remote accident conditions and add costly engineered safeguards to cover hypothetical accidents should be considered as very conservative. We believe that design principles on which the present engineering layout of fast sodium cooled reactors in Germany is based can be relaxed in the near future as more safety research has built up sufficient confidence in the performance of sodium cooled breeders.

According to the present status of safety philosophy for fast sodium-cooled reactors and the foregoing considerations we can conclude that whereas in thermal water reactors essentially thermal processes govern the course of the design basis accident, in fast sodium cooled reactors nuclear and thermal processes govern energy and activity release. The result of this philosophy is the incorporation of a containment against energy release and a containment against activity release in the design of the fast sodium cooled reactor prototype.

3. DESIGN BASIS ACCIDENT OF THE NA-2-REACTOR

In the following safety assessment and design philosophy are further explained at hand of the Design Basis Accident, which has been assumed for the conceptual design study of a 300 MWe prototype fast breeder reactor. This so-called Na-2 study is currently underway as a joint effort of the Nuclear Research Center Karlsruhe and the industrial group Siemens/Interatom. A final report of this study will be published soon. A very conservative approach was taken both in the safety assessment and engineering layout. This decision was made in view of the early date of commitment, which is scheduled for 1969 and on account of today's incomplete knowledge in the fields of engineering, safety and physics.

The main characteristics and performance data of Na-2 are as follows:

Core: Mixed oxide fuel pins of 6 mm o.d. and 95 cm active height are bundled in hexagonal wrapper tubes. Power flattening is achieved by two radial zones of different enrichment. Coolant flow is upward through core and blanket. The core is positioned by a lower and an upper grid plate. The power rating is close to 1 MWth/kg of fissile Plutonium, and the total breeding ratio averages 1.3.

Primary system: A loop design was chosen for the primary cooling system consisting of three main and two auxiliary loops. Coolant entry is at the bottom of the reactor vessel. Reactor outlet temperature range is 560°C - 580°C yielding in steam temperatures between 520°C and 540°C.

Fuel handling: A system of three rotating plugs is employed for refueling. Intermediate storage positions for spent fuel are provided within the vessel.

Containment: The plant is designed to accomodate the potential consequences of a nuclear excursion releasing 1000 MWsec of destructive energy. Double containment was chosen consisting of two independent leaktight steel barriers in series and separated by a large air plenum. The inner containment is filled with nitrogen.

For the reference design of Na-2 a broad spectrum of hypothetical failures leading to uncontrolled reactivity insertions has been analysed. On the basis of these investigations it is concluded, that reactivity excursions leading to a core destruction can be discounted, unless one assumes gross failure of a main primary system component such as guillotine type pipe rupture, complete blockage of coolant flow for the entire core, loss of sodium from the core, coinciding with complete mal-function of the safety system. For the current design study, such a very unlikely simultaneous failure of a vital reactor component and the entire safety system has been assumed as the design basis accident for the plant.

A typical sequence of events culminating in the DBA is given below:

1. All primary coolant pumps fail simultaneously.
2. The reactor is not shutdown by the safety system.
3. The stagnant coolant in the core is heated above its boiling point.
4. Desuperheating is suddenly initiated, and the coolant starts being expelled from the core by the vapor pressure.
5. Due to the positive Na-void coefficient, the expulsion leads to a reactivity insertion of about 50 ¢/sec.
6. The resulting nuclear excursion is terminated by core disassembly and Doppler feedback.
7. Vaporized and molten fuel is intimately mixed with the remainder of the sodium in the core, resulting in a rapid vaporization of the sodium.
8. Mechanical work is done on the environmental structure of the core.
9. Fuel isotopes and fission products are released instantaneously.

On the basis of a careful and conservative analysis of this hypothetical accident it is concluded, that its destructive potential, i.e. the amount of energy available to do mechanical work, is less than 1000 MWsec. About one half of this energy is contributed by the nuclear excursion, and the remainder by the consequential sodium vapor expansion. However, knowing the destructive potential of the DBA is not sufficient to design engineered safeguards against its consequences. Depending mainly on the rate at which the thermal energy is transferred to the sodium, three different destructive phenomena have to be considered: shock waves, water hammer effects and internal blast pressure.

4. ENGINEERING LAYOUT OF CONTAINMENT

4.1 Design Criteria

For the design of the containment the following requirements regarding the DBA have been established:

1. All immediate mechanical effects capable of producing damage to structural components, such as shock waves, water hammer effects, and internal blast pressure, shall not propagate beyond the reactor cavity.

2. All secondary effects, such as residual pressure, chemical reactions, decay heat of the core debris, shall not impair the integrity of the first containment barrier.
3. The first containment barrier shall be sufficiently tight to prevent a sodium fire in the outer containment and gross leakage of radioactive material.
4. The first containment shall be completely enveloped by a secondary containment as a final barrier for radioactive material.
5. Decay heat from the core debris shall be removed by natural convection of sodium.

4.2 Containment against energy release

First, the design provisions against the damage producing phenomena are briefly described. A scheme of the reactor cavity is shown in fig.1. It is shown, that design provisions against water hammer effects would have to be opposed to those tending to decrease the internal blast pressure: In order to limit the internal blast pressure to a value below the burst pressure of the reactor vessel, one would provide a rather large gas plenum below the rotating plug. On the other hand, in case of rather quick energy release from the sodium fuel reaction, the high pressure sodium vapor bubble at the core location might accelerate a column of liquid sodium towards the vessel head, thereby concentrating most of the total energy to a single and most vulnerable components of the system. Obviously there is no reasonable way to solve both problems simultaneously. Therefore, in the present design it is not intended to keep the reactor vessel intact against the internal blast pressure, because failure of the vessel would be less hazardous to the containment integrity than failure of the plug holddown. The rotating plug is divided horizontally into two parts with the cover gas plenum in between. The lower part of the plug is emerged in the sodium, so that water hammer effects are precluded. The rotating plug and its holddown mechanism are designed for a pressure exceeding the burst pressure of the vessel, which has a section of reduced wall thickness just below the plug. Thus, the internal blast pressure will be released into the reactor cavity rather than into the reactor containment building.

For the design of the explosion containment structure the effective energy from both the nuclear phase and from the sodium vapor explosion are conservatively considered in terms of an equivalent charge of 500 lb of TNT. The reactor vessel is surrounded by a closed steel structure capable of absorbing the shock wave created energy by plastic flow of material without transferring considerable forces or momentum to the cavity walls. In the analysis no credit was taken for the containment potential of the vessel, internals and the vessel itself. The peak pressure following the burst would be less than 10 at. The explosion containment structure is designed to sustain this pressure and release it to the adjacent cells of the primary system without loading the concrete walls of the cavity. Within a few seconds after the excursion the equilibrium pressure will be below 1 at.

Above the rotating plug there is a nitrogen filled cell which is separated from the operational area by a leaktight lid as part of the primary containment barrier. Its purpose is to prevent a sodium fire in the secondary containment following the DBA, should sodium be ejected through and alongside the rotating plug.

Special emphasis was paid on providing long term decay heat removal capacity from the core, respectively from the fragments of the core. Two auxiliary cooling loops are particularly designed to remain operable after the DBA, i.e. even after a severe rupture of the vessel or the pipes. Loss of coolant from the core is discounted by minimizing the free volume in the primary sodium cells and by providing sodium reservoir tanks which would drain by gravity into the vessel or into the reactor cavity. Additionally, three natural convection NaK loops are embedded in the wall of the reactor cell, where they are protected against the blast effects of the DBA, for decay heat removal and cooling of the concrete structure.

4.3 Containment against activity release

As was mentioned before, a double containment against activity release consisting of two steel shells in series was chosen for this conceptual design study. The reason for chosing a double containment design are twofold:

First, as will be shown later, uncomplete data about activity release have us forced according to our conservative safety philosophy to assume pessimistic data for the release parameter.

Secondly, the outer containment filled with air to facilitate access and maintenance shall contain any sodium fire which is credible during reactor operation, maintenance or repair work. Because the integrity of the inner containment can be guaranteed for the DBA a sodium fire together with the DBA can be ruled out.

Two alternate but similar containment layouts have been investigated from which one will be explained here. A scheme is shown in figure 2.

All primary sodium and auxiliary equipment cells in the lower part of the building are filled with nitrogen and are interconnected to take advantage of maximum available volume for pressure relief after the DBA. They are completely enveloped by a steel liner, whose primary function is to prevent a sodium fire and consequential pressure and temperature buildup in the air filled room above the operating floor. Since for this purpose it has to be rather leaktight, it is considered as the primary containment barrier also with regard to fission products and aerosols. This containment will be designed so that operating personnel may enter through locks for repair or maintenance of the auxiliary systems during reactor operation without interrupting containment integrity.

The outer containment is a conventional all welded low leakage steel building, approximately 54 m in overall height with a diameter of about 38 m. A flat bottom is used for maximum space utilization. A particular feature of this concept is that both containment barriers are completely separated by a continuous airfilled gap. At the bottom, this is achieved by a grid type support of the entire inner containment. Whether or not the principle of complete structural independence of the two containments will have to be verified at the bottom also, has not yet been decided. The large air volume in between the two containment barriers will act as a high capacity low pressure plenum effectively reducing leakage of radioactive material in the DBA.

The activity release and subsequent radiation burden due to inhalation of radioactive material was calculated according to the conditions of the DBA for the cases of single and double containment concept. Because the site of this reactor has not been selected, the meteorological diffusion conditions of another

German reactor installation were used. Data about fission product release from fast sodium cooled reactors are not yet available as they are for thermal water reactors. Therefore we took the classical values from TID-14844 /5/ (release fraction 0.5 for halogens, 0.01 for solid fission products) assuming an additional retention factor for the halogens and volatile fission products because of the good trapping capability of sodium. Plate-out for halogens and volatile fission products was assumed to be exponential in both containments with a half time of 1 hour. For the solid fission products including Plutonium we assumed a plate-out half time of 10 hours. The leak rates at maximum pressure were 1 Vol %/day for the single containment concept and 10 Vol %/day for the primary and 1 Vol %/day for the secondary shell in the double containment concept. The resulting numbers are given in table 1.

It can be concluded that a double containment without filter system or other

TABLE 1: Design Basis Accident Doses due to Incorporation

(800 m, downwind, inversion)

	Single Containment			Double Containment		
Time of Exposure (h)	2	3	24	2	8	24
Bone Dose (rem)	850	2900	5200	0,07	3,5	30
Thyroid Dose (rem)	230	300	305	0,01	0,07	0,08

the containment atmosphere decontaminating equipment is capable of minimizing the radiation burden of the environment to the point of site independence for the plant if we use the 25 rem accident dose as a yard-stick. It may be emphasized that this is accomplished without any engineered safeguards which must be put into operation after the DBA has occured. The simple double containment concept proves suitable for large fast sodium-cooled reactors of the size of 300 MWe avoiding any hazards to the environment.

It should be noted that larger reactor plant sizes up to 1000 MWe are the ultimate goal of the present fast reactor development phase. Depending on the DBA of that size reactors it may come out, that even the very conservative and pessimistic assumptions made at present in the activity release may lead to a higher radiation burden than reported in table 2. This would make necessary further improvement of the containment concept described. However, two possibilities are seen to imply further reduction of the environmental radiation burden.

First the concrete cylinder surrounding the outer containment to attenuate the direct γ-radiation of fission products through the containment shell may be covered by a leaktight steel roof. Connecting the air gap in between those shells to an exhaust- and filtering system the leakage could be reduced further by a few orders of magnitude. However, this system has to be put into operation immediately after the DBA has occured which might be questionable in respect to reliability.

Secondly, the release models currently in use may be improved taking into account the real physical and chemical processes and transport phenomena occuring in the DBA. For instance, the release of 1 % solids from the core assumed in the calculations reported above would mean for a 1000 MWe fast sodium-cooled breeder that about 100 kg core material can stay airborne as aerosol in the containment during the accident. This seems to be a rather unrealistic assumption. In the Karlsruhe Nuclear Research Center, therefore, a program was initiated to study activity release and transport characteristics of fission product and Plutonium aerosols from fast reactor cores as function of large fast reactor accident conditions. We hope to provide soon sufficient data which may allow even reduction of the number of containment shells for large fast sodium cooled reactors.

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SUMMARY

For the conceptual design of a 300 MWe sodium cooled fast breeder prototype reactor a severe nuclear excursion is assumed as the Design Basis Accident.

Based on a very conservative analysis of this accident a double-containment concept is chosen for that reactor design.

It is shown that this concept is capable of containing the Design Basis Accident without any hazard to the environment.

Improved knowledge in accident analysis, activity release, and engineered safeguards are expected to facilitate single containment designs for large fast breeder reactors sodium cooled.

REFERENCES

/1/ COTTRELL, W.B., DAVIS, W.K., Containment and engineered safety of nuclear power plants, Proc. 3rd UN Int. Conf. PUAE 1964, P/276

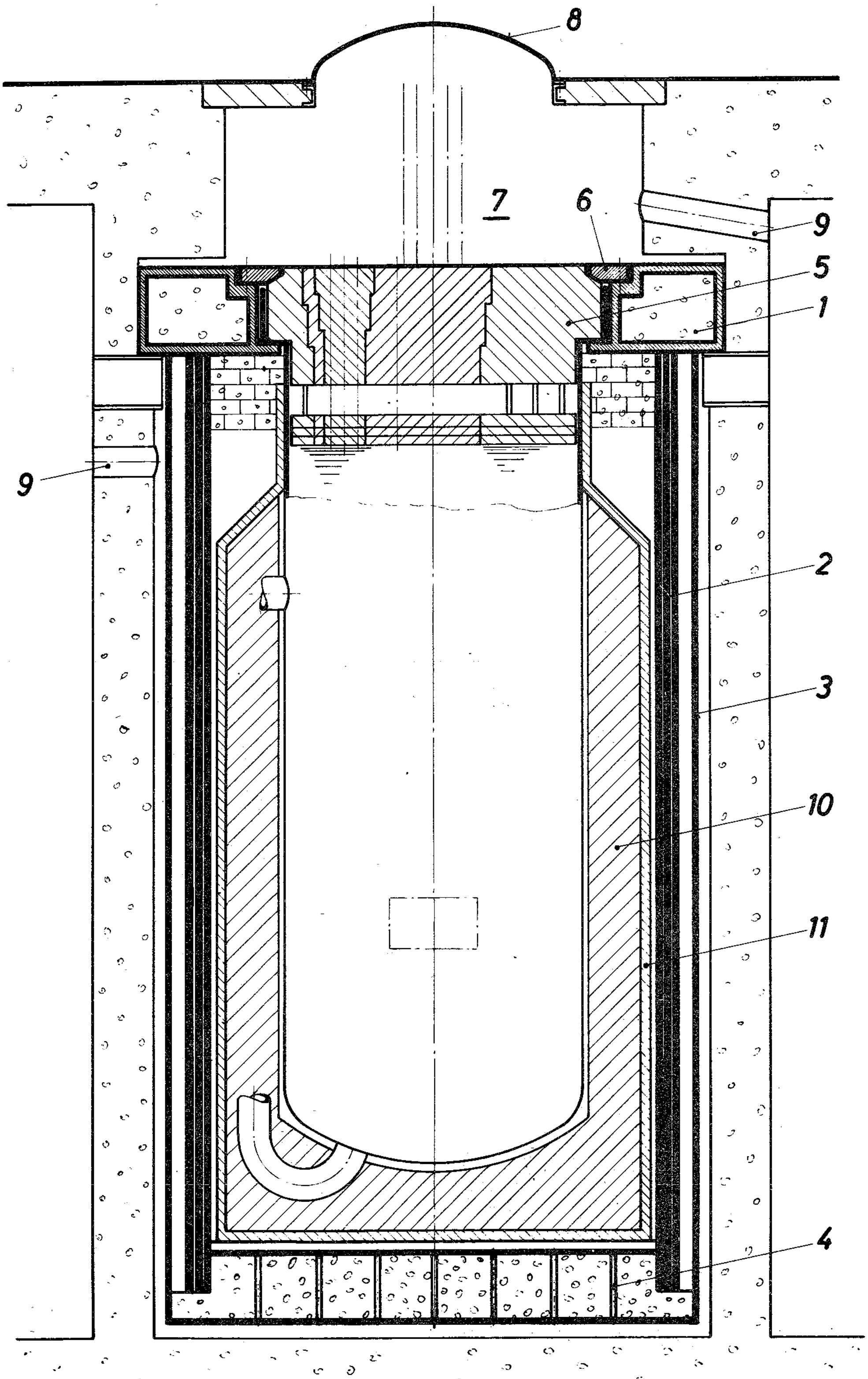
/2/ BECK, C.K., Power reactor accidents in perspective, CONF-650407 (1965) p. 908

/3/ HICKS, E.P., MENZIES, D.C., Theoretical Studies on the fast reactor maximum accident, ANL-7120 (1965) p. 654

/4/ MEYER, R.A., WOLFE, B., FRIEDMAN, N.F., A parameter study of large fast reactor meltdown accidents, ANL-7120 (1965) p. 671

/5/ DINUNNO, J.J., et al., Calculation of distance factors for power and test reactor sites, TID-14844 (1962)

Reactor - Cavity
(Schematic)



1 Support Ring
2 Steel Cylinders
3 Support Shroud
4 Concrete Block
5 Rotating Plug
6 Plug Hold Down
7 Nitrogen Cell
8 Leaktight Lid
9 Pressure Releave Duct
10 Thermal Shield
11 Isolation

Fig. 1

Primary System and Containment (schematic)

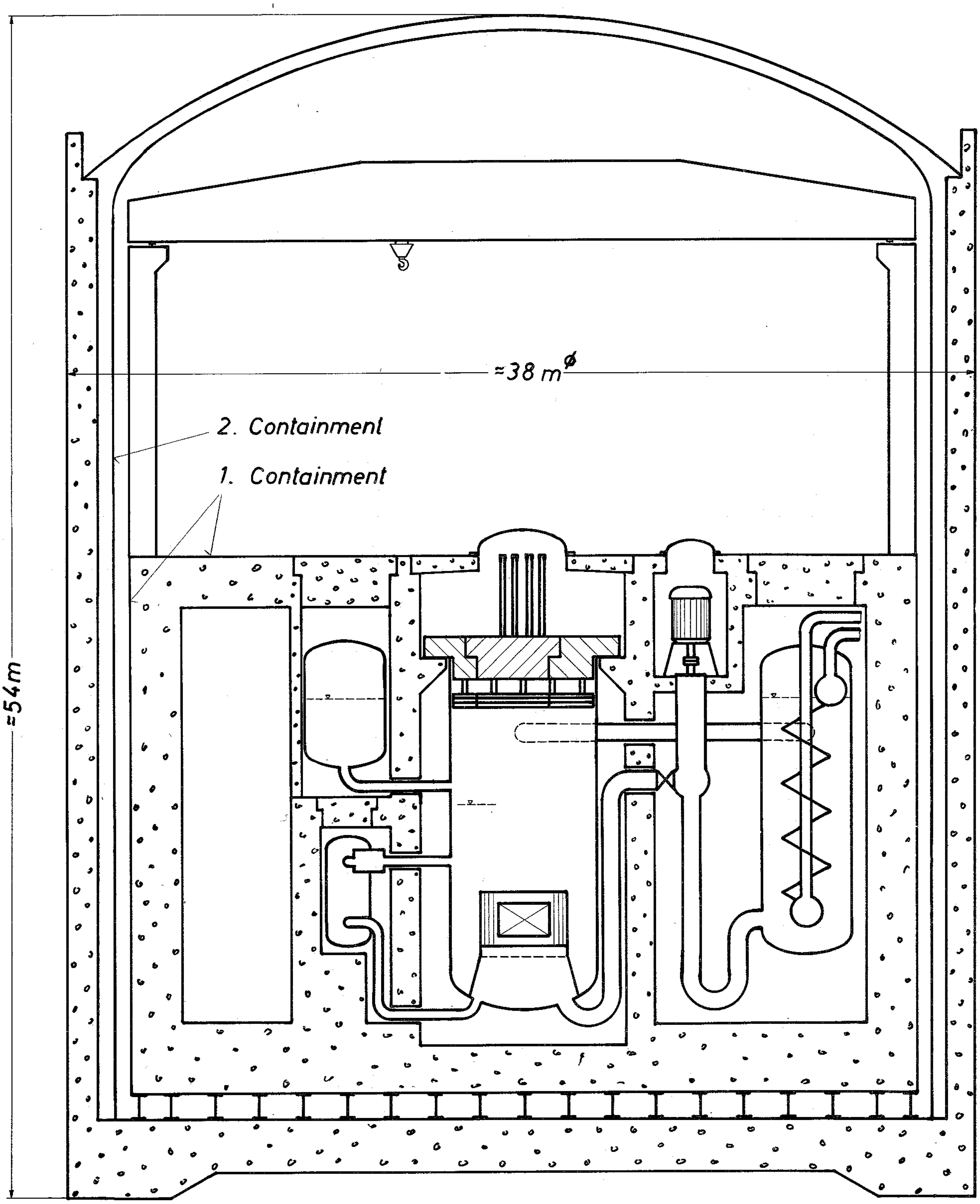


Fig. 2