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*) Work performed within the association in the field of fast reactors between the European Atomic Energy Community and Gesellschaft für Kernforschung mbH.,Karlsruhe

SAFETY FEATURES OF A 300 MWe SODIUM COOLED
FAST BREEDER REACTOR (Na 2) x)

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ABSTRACT

The conceptual design of the 300 MWe sodium cooled fast breeder reactor Na-2, presented in this paper, is the result of joint efforts of the Nuclear Research Center Karlsruhe, associated with Euratom, and the industrial group Siemens-Interatom. The main features of the design will be described. Emphasis will be laid upon the safety related design aspects. The results of the analysis of typical reactivity insertions and coolant system failures will be given. The potential consequences of severe nuclear accidents are analysed and the containment capability of the design chosen is demonstrated.

x) Work performed within the framework of the association Euratom - Gesellschaft für Kernforschung mbH. in the field of fast breeder development.

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INTRODUCTION

1.1 Following the conceptual design study of a 1000 MWe sodium cooled fast breeder reactor [1,2], the German fast breeder program has concentrated on the development work for a 300 MWe prototype reactor [3,4]. The conceptual design presented in this paper is the result of a joint effort of the Nuclear Research Center Karlsruhe and the industrial group Siemens-Interatom. The prototype reactor is scheduled for commitment in 1969 and for completion in 1973.

A rather conservative approach is taken both in the safety evaluation and engineering layout to account for incomplete knowledge in the fields of engineering, safety and physics. Nevertheless, the prototype reactor will be typical of a future 1000 MWe unit in most respects and is especially aimed to provide the technological information for the construction of a competitive large scale power plant by the end of next decade.

1.2 The following main characteristics and performance data were chosen:

Core and Reactor Vessel

Mixed oxide fuel pins of 6 mm O.D. and 95 cm active height are bundled in hexagonal wrapper tubes. The coolant flows upward in parallel through the two zone core and the blanket. The rating is approx. 1 MWth/kg of fissile material and the total breeding ratio is 1.24. The thermal power is 730 MW to produce a net electrical power of 300 MW.

Coolant System

The main coolant system consists of three identical parallel subsystems with two sodium loops in series. A loop design was chosen for the main primary cooling system. Reactor outlet temperature is 560°C yielding a steam temperature of 510°C. Two natural convection coolant systems are provided for backup core cooling at shutdown.

Safety

Because of the first-of-a-kind characteristic of the prototype reactor a high degree of conservatism is applied in the safety assessment. However, in the nuclear design no drastic modification such as pancaking, modular core arrangement or spectrum softening was employed. The additional safety margin will rather be gained by adding consequence limiting safeguards against the most extreme accident situations. Double containment was provided. The design ensures that the integrity of the containment is not impaired by accidents involving nuclear, thermal or chemical reactions. Loss of coolant from the core is eliminated even in the case of a major pipe or vessel rupture. Long term decay heat removal from the fuel is by natural convection. The reactor is inherently stable with respect to reactivity perturbations.

This paper will give only a brief description of the design and will concentrate on the safety assessment of this reactor.

DESCRIPTION OF MAIN DESIGN FEATURES

Core and Reactor Vessel

2.1 The fuel pin diameter is 6 mm O.D. with a .38 mm thick stainless steel cladding. The fuel pin consists of (starting at the lower end) a 65 cm fission gas plenum, 40 cm lower axial blanket, 95 cm active fuel column and a 40 cm upper axial blanket. The fission gas plenum is located below the core, because first, due to the lower temperature at this point there is a saving of 20 cm in overall fuel element length, and secondly, thermal stresses and bowing from radial temperature gradients across the fuel assembly are considerably reduced. The potential release of fission gas bubbles after rupture of the canning is not expected to cause severe safety problems [5]. A smeared fuel density of 80 °/o was selected to account for burnup swelling up to a maximum burnup of 85 000 MWd/t. In the reference design full pellets of 85 °/o theoretical density were chosen. However, dished pellets or cored pellets may be used if they show better performance. 169 fuel pins are positioned by honeycomb grids in hexagonal wrapper tubes of 110 mm across flats. This fuel assembly is smaller than would be desirable for a 1000 MW plant. The size was chosen as a compromise with respect to thermal stresses and forces, fueling time, improved thermal performance by more accurate adjustment of coolant flow to power profile and available test rigs. The smaller size fuel elements are also advantageous with respect to the reactivity to be handled by the fuel charge machine. The minimum distance between fuel pins is 1.9 mm. The core arrangement is shown on figure 1. The inner zone of lower enrichment fuel contains 78 fuel assemblies and 6 control elements, namely 2 regulator rods with less than 1 \$ each, 2 safety rods and 2 shim rods. The outer zone contains 72 fuel assemblies, 6 safety rods and 6 shim rods. The central position of the core is reserved for special in-core instrumentation. The core is 95 cm high and 153 cm in diameter. It is surrounded by a 50 cm thick radial blanket.

2.2 Physics calculations were performed with one- and two-dimensional codes, most of which are part of the Karlsruhe nuclear code system NUSYS. A 26 group cross section set developed at Karlsruhe was used [6]. The main physics data are listed in Table I.
Because of the uncertainties in the calculation of the sodium void reactivity a maximum value of 6 \$ was used in the safety analysis rather than the calculated value of 4 \$. The requirements on reactivity control were established to about 40 \$ to accomodate reactivity swings from cold shutdown to full power, burnup, reactivity perturbation and shutdown margin, including one stuck rod.

2.3 The main thermal-hydraulic data are listed in Table II. Depending on burnup, the radial blanket will contribute about 4 to 10 percent of the total thermal power. The end of life average temperature rise in the blanket was limited to 150°C in order to prevent excessive thermal bowing.

Various refueling intervals were analysed as to their influence on physics, thermodynamics and economy. Since it does not seem appropriate to fix a certain fuel cycle at this time, the design is such that any fuel cycle time between 1 year and 3 months can be realised according to the contemporary circumstances. At present, a 6 months refueling period is used as a reference. In the blanket residence times will be between 4 to 15 years depending on local flux level.

2.4 Special emphasis was laid on providing a precise and reproducible core geometry. In particular it was considered essential that the core subassemblies are fixed radially in an upper and lower grid plate so that bowing of individual subassemblies will not affect the overall core geometry. In the reference design (fig.2) the fuel element is supported by the lower grid plate. The hydraulic forces of the coolant are balanced by means of individual low pressure chambers below each element. The top of the fuel assembly is positioned radially in the upper grid plate which itself is supported horizontally against a rigid steel cylinder surrounding the radial blanket. Inward bowing of the fuel assemblies is prevented by wear faces attached to the outside of the wrapper tubes in order to ensure the inherent stability of the reactivity feedback.

Besides its primary function of supporting the top of the fuel assemblies the upper grid plate also serves as a positive stop against fuel element ejection from the core following failure of the hydraulic holddown. It also guides the control rod drives with respect to the core to ensure safe reactor shutdown during horizontal earthquake loading. Temperature sensors are provided above each fuel element position for monitoring the coolant outlet temperature of each fuel element individually. To prevent inadvertent lifting of fuel assemblies upon removal of the upper grid plate a strip off plate is provided as an integral part of the grid plate. Sweep arms are also provided to check the complete separation of the upper grid plate and the fuel elements before the upper grid plate will be swung away for refueling.

Surrounding the radial blanket and outside of a steel neutron shield there is a circular arrangement of spent fuel storage positions for more than one third of the core loading. The steel shield serves to limit the fission rate in the spent fuel storage annulus. Additional shielding is provided between the fuel and the vessel to reduce the fast neutron dosage to below $5 \cdot 10^{21}$ nvt during the lifetime of the plant.

Coolant entry is through the bottom of the reactor vessel. This facilitates a rather simple design of the vessel internals since it eliminates a flow divider skirt between cold and hot sodium and the safety problems related to it. Besides there are no hot nozzles in the cold part of the vessel wall and vice versa. The reactor vessel is approx. 5.2 m in diameter and 15 m high.

For removal of spent fuel from the core and replacement with fresh fuel a system of three rotating plugs was chosen. Two fuel transfer mechanisms are installed on the smallest one of the three plugs for in-vessel fuel handling. Transfer of fuel and blanket elements into and out of the reactor vessel will be by a separate machine through the fuel transfer port in the large rotating plug. With this system a high degree of flexibility and short fuel handling times can be achieved.

Coolant System

3.1 The main coolant system consists of 3 identical subsystems which are completely independent except for the common point of the reactor vessel. A schematic of the system is shown on fig. 3. Each subsystem consists of a primary sodium loop, a secondary sodium loop and a steam loop in series. Each of the secondary sodium loops includes three steam generator-superheater units and one reheater in parallel. Steam conditions are 171 atm and 510^{o}C at the turbine stop valve. The steam generators are of the vertical, single wall, once through type with free sodium surface and tube sheets in the covergas. Rupture discs are provided for Na-H_2O reaction pressure relief.

3.2 The secondary sodium coolant pumps are located in the cold leg of the secondary loop. The sodium/sodium heat exchangers are located inside the containment building with the primary sodium on the shell side and the secondary sodium on the tube side. With this design the best protection against both damage from sodium-water-reaction pressure pulses and heat exchanger tube rupture after a severe reactor accident can be achieved. Fig. 4 shows a schematic of the primary coolant system. The primary sodium pumps are in the hot leg of the primary loops in order to provide the required net positive suction head at the pump inlet without the need of pressurizing of the reactor cover gas. Thermal expansion of the piping system is compensated by appropriate three dimensional layout. A throttling valve is provided in the cold leg of each primary loop in order to facilitate flow control after pump shutdown. Normal flow control will be by pump speed variation. A pony motor is provided for each pump with backup connection to the emergency power supply. Decay heat removal from the reactor after shutdown will normally be by the main coolant loops.

3.3 In case of loss of power to all pump motors the auxiliary coolant system, consisting of two identical subsystems, will provide shutdown cooling of the core by natural convection (fig.4). The heat is transferred to a natural convection NaK loop and finally dumped into the air. Reverse flow at reactor operation in the auxiliary sodium loops is prevented by check valves.

The auxiliary coolant system is especially designed to remain operable even after rupture of the primary coolant system including the reactor vessel and the auxiliary coolant system itself. This is achieved by minimizing the void volumes in the reactor cavity and in the component cells and by providing Na-reservoir tanks which would drain by gravity, such that the core and the entire primary part of the auxiliary coolant system will always be covered by sodium.

Containment

Fig. 4 shows schematically the layout of the reactor containment. Its main characteristics are the blast shield, a double containment barrier and full leakage control.

4.1 The blast shield which completely envelopes the reactor vessel is designed to accomodate all primary effects of the design basis accident. Shock wave energies corresponding to the detonation effect of up to 250 kg TNT can be absorbed by plastic deformation of structural material without loading the concrete walls of the reactor cavity. Water hammer effects are eliminated by avoiding a free sodium surface below the rotating plug. The rotating plug and its holddown mechanism are designed for a pressure exceeding the burst pressure of the reactor vessel. Thus, any pressure buildup in the vessel would be released into the reactor cavity prior to failure of the plug.

4.2 The concrete structure which houses the reactor and the primary coolant system forms the inner containment barrier. It is designed for 2 atm internal overpressure. The design leak rate is 50 $^{o}/o$ of contained volume per day. The heat exchangers, namely the walls between primary and secondary sodium, apparently are the most vulnerable parts of the inner containment barrier. However, we feel confident that no strong shock waves would reach the heat exchangers after having travelled all the way through the coolant system piping. Water hammer effects cannot occur because there is no free surface in the heat exchanger and, finally, the exchangers are designed for a pressure exceeding the burst pressure of the reactor vessel. Thus, if any failure would occur it would be in a place where it would not violate the integrity of the containment. The atmosphere in the inner containment will be of low oxygen and water content to eliminate the possibility of a sodium fire.

4.3 The outer containment barrier is a conventional low leakage steel building, approximately 50 m high and 35 m in diameter. Design conditions are 2 atm overpressure, 250°C temperature and a leak rate of 1 °/o of contained volume per day. These values of design pressure and temperature are based upon the analysis of a large pool type sodium fire inside the containment.

4.4 The outer containment is surrounded by a 50 cm thick concrete cylinder as a shield against direct radiation from radioactive material contained in the building following a severe accident. This concrete cylinder is covered by a leaktight steel roof. The air gap between the steel containment and this concrete cylinder is connected to a filtering and exhaust system, thus providing means to control any leakage of radioactive material out of the reactor building.

ACCIDENT ANALYSIS

In order to assess the safety of the design chosen, numerous accident situations were analysed. In the following we shall describe the plant behaviour for some typical reactivity insertion accidents and coolant system malfunctions.

Reactivity Perturbations

5.1 In the control rod runaway accident it is assumed that all control rods are withdrawn simultaneously at maximum design speed resulting in a reactivity ramp of a few cents per second. Due to the strong negative temperature coefficient of the core the sodium temperature would rise at a rate of only 6 ^{o}C/sec which can easily be controlled by the safety system. Control rod blowout by the coolant is generally eliminated because the control rod weight exceeds the hydraulic forces. Two failures would be required to cause control rod blowout: failure of the coolant inlet nozzle and inadvertent uncoupling and withdrawal of the control rod drive shaft. The reactivity ramp would be in the order of 5 $/sec which can still be controlled by the safety system without causing gross damage to the core.

Fig. 5 shows the maximum reactivity ramp which can be controlled by two types of shutdown mechanisms (free gravity fall and spring accelerated) such that no melting occurs either in the hot spot fuel element (curves 1 and 3) or in the central fuel element without hot spots (curves 2 and 4).

5.2 There was considerable concern about the potential of a refueling accident. However, with the design chosen, fuel handling can be performed only if all control rods are previously decoupled from their drives. The total control rod worth of 40 $ will provide abundant shutdown margin. Despite this fact in the analysis it was assumed that the core had become critical inadvertently and that a 31.4 $^{o}/_{o}$ enriched fuel element is dropped into central core position. The initial temperature was assumed to be 300^{o}C and the flow was set to 10 $^{o}/_{o}$ of the rated value. Fig. 6 shows that the maximum fuel temperature would rise only to about 1000^{o}C and the sodium temperature would stay well below the boiling point. However, it is anticipated that further development of subcriticality measurement techniques will provide a reliable safety device so that the restriction of maintaining at least 10 $^{o}/_{o}$ of rated flow during refueling will not be required.

Coolant System Failures

6.1 Flow blockage in single fuel channels is expected to be a relatively slow process. Since it is anticipated to provide thermocouples above the outlet of each fuel element the beginning of flow blockage is likely to be detected. Undetected flow blockage would lead to excessive sodium temperatures and to local boiling and fuel failure. Instrumentation for detection of sodium boiling and fuel failure is presently under development in various countries and the hope that at least one of the methods being developed now will become a reliable safety instrument, is well justified.

6.2 Loss of pumping power in the whole primary system would shut the reactor down either upon low flow or upon high temperature indication. The primary sodium temperatures would follow the curves shown on fig. 7. About 2 minutes after loss of power the two auxiliary natural convection loops would take over the long term heat removal.
Loss of flow even in all secondary coolant systems does not require immediate reactor shutdown.

Accidents Involving Safety System Failure

7.1 During the early phases of the design work it was decided that the reactor and containment design should be based on the requirements of a nuclear accident which might result from a severe reactor component failure coinciding with complete failure of the accident preventive safeguards, such as the safety system. In the following it is, therefore, generally assumed that the safety system is inoperable. Since the most severe accidents would result from fuel melting and vaporization in a nuclear power burst the analysis has concentrated on potential reactivity insertions and their consequences. As a first step to such an analysis a simple fault propagation tree has been set up as shown on fig. 8. The main events which might initiate severe reactivity insertions are the following:

1. Control rod ejection
2. Generation of sodium voids
3. Core meltdown.

There is an important difference between the meltdown accident and all others. Meltdown of the dry and subcritical core due to fission product heat may lead to supercritical configuration whether a safety system is operable or not. With all other incidents a well designed and properly operating safety system would detect the beginning of a dangerous situation and take corrective actions. It is, therefore, concluded that the safety system and the design provisions against loss of coolant from the core are of equally great importance in sodium cooled fast reactors, and are, therefore, to be designed to the same confidence level. We will now come to the accident initiating events individually.

7.2 The most reactive control element contains approx. 4.2 \$ reactivity. This control rod is assumed to be ejected by boiling sodium after blockage of the coolant flow through the control element and after decoupling and withdrawal of the control rod drive. At full power the heat generation in the control element is approx. 10 % of the power generated by a fuel element. The sodium would reach the boiling point within a few seconds. Assuming 300°C sodium superheat, the control rod would be ejected by a 9 atm pressure difference and might reach a velocity of 5.5 m/sec. This would result in a reactivity insertion rate of 60 \$/sec.

7.3 In the Na-2 design the maximum sodium void reactivity was calculated to be 4 \$ for a bubble volume of about 50 % of the core. However, the safety analysis is based on a value of 6 \$. Voiding the entire core would result in a reactivity increase of about 2 \$. There are principally two mechanisms which might generate a void in the sodium of the critical reactor. First, a gas bubble in the coolant, and secondly, sodium boiling.

7.4 A gas bubble in the primary coolant system might enter the core at the rated velocity of the sodium. It was assumed that the gas bubble occupies only the inner 2/3 of the core cross section where the sodium void effect is positive. The resulting maximum reactivity perturbation is 55 \$/sec.

7.5 Sodium boiling in a single element contributes only up to 14 ¢ reactivity. Therefore, in order to generate a severe reactivity perturbation sodium boiling must occur simultaneously in a great number of fuel elements. Since the sodium void effect is the most positive in the central part of the core, the analysis has concentrated on those situations, where boiling would start in the center of the core and would propagate radially at a high rate. The most important parameter which determines the axial growth of the bubble within the fuel elements is the sodium superheat. Because there is a great lack of experimental data in this field the maximum sodium superheat has been deliberately set at 300°C [7], because above this temperature the fuel cladding is expected to fail and to initiate sodium boiling by fission gas bubbles injected into the sodium. The method which was used to compute the sodium ejection process is described in another paper of this conference (7). As a worst case it was assumed that boiling starts in the core midplane after instantaneous loss of flow at full power operation. Fig. 9 shows the combined axial and radial bubble growth. It was assumed that sodium boiling propagates radially according to the power profile and starts at the same temperature of 1200°C in all fuel elements. In reality due to the differences in hot spot factors and burnup sodium boiling would start sooner in some elements and later in others. Thus the total reactivity input would be smeared out in time compared to the above model. Even for the conservative assumptions made here the maximum reactivity rate was calculated not to exceed 60 $/sec.

7.6 Emphasized by the Enrico Fermi incident considerable effort was made to analyse the potential consequences of coolant flow blockage to individual fuel elements. Again, since the reactivity effect of voiding a single fuel element is rather small, only rapid propagation of this incident would result in severe reactivity perturbations. It was found that after sudden complete blockage of a fuel element it takes approx. 2 to 3 seconds before the sodium would reach the boiling point. If all fuel elements surrounding the failed fuel element were blocked completely by the pressure pulse from sodium boiling, this process would propagate at the rate of about 1 annulus every two seconds. We are confident that the hexagonal wrapper tube of the fuel element can be designed against the maximum pressure during boiling. However, severe deformation of these wrapper tubes by the recondensation pressure pulses cannot yet be discounted (7). It is expected that the ejection and recondensation processes in several fuel elements would cause considerable reactor noise which can be detected by both nuclear and acoustic instrumentation and that the reactor can be shut down before greater damage would occur. Simultaneous ejection of sodium out of 30 fuel elements, which would correspond to a 45 $/sec reactivity ramp, is considered to be a conservative estimate of the maximum reactivity perturbation which might result from propagation of fuel element blockage.

7.7 Contrary to the metal fueled fast reactors of the first generation core meltdown is a very unlikely event for the present fast reactor design employing ceramic fuel. As long as there is sodium in the core meltdown can be discounted on the basis of the great difference between sodium boiling point and fuel melting point. Any serious situation would rather lead to sodium boiling than to gross fuel meltdown. Core meltdown, therefore, is of concern only after loss of sodium. In the Na-2 reactor loss of sodium is eliminated by the geometrical design of the reactor cavity and the heat exchanger rooms. Nevertheless it is worthwhile to analyse what might occur if the design were different. Because removal of sodium in itself would have caused a reactivity excursion meltdown at power is not possible. Meltdown due to decay heat, however, can be envisaged after shutdown. Due to the heat which is still stored in the fuel and which is still being generated, clad melting would start very soon and would propagate within a matter of seconds over the whole reactor. Fuel melting, however, would take several minutes. If we assume that melting fuel falls under gravity and fills up all available void volume in the core, the effective core height would be reduced by about 40 cm. The corresponding reactivity rate would be in the order of 5 \$/sec. However, if we assume that the fuel would fall out of the canning as soon as the canning melts, then a maximum reactivity rate of approx. 60 \$/sec would result. Although the experimental evidence of irradiation experiments indicates that the fuel column would not break down immediately after clad melting a reactivity ramp of 60 \$/sec was used in the analysis of the dry meltdown accident.

7.8 The energy release of the nuclear excursion was calculated according to the Bethe-Tait theory (8). A one-dimensional spherical model of the Na-2 reactor core was used. In all cases where there is still sodium in the reactor a modification was applied to the one-dimensional model to account for the fact, that the fuel cladding and the hexagonal wrapper tubes of the fuel elements can prevent radial core disassembly. It was assumed that this disassembly takes place only in the axial direction. Consequently a reactivity ramp of 60 \$/sec would produce about the same energy in the wet core as it does in the dry core, although the Doppler coefficient of the wet core is about twice as good as in the dry core. About 500 MWsec of destructive energy might be released in both cases.

7.9 In most of the nuclear accidents investigated there is still sodium in the core or in the environment of the core and the potential destructive energy release of the excursion may be increased due to subsequent vaporization of sodium[9]. An upper limit for the potential mechanical energy of a fuel-sodium interaction was determined on the basis of instantaneous heat transfer from the vaporized and molten fuel to the liquid sodium and isentropic expansion of the fuel and sodium vapor mixture. Fig. 10 shows the maximum mechanical work which can be done by the sodium vapor bubble as a function of the initial fuel temperature and the final pressure to which the bubble will expand. For an average fuel temperature of 4000°K which is typical for a severe nuclear excursion of the Na-2 reactor, the potential mechanical work of an expansion down to 20 atm is about 200 MWsec per tonne fuel or 1000 MWsec total. In this we have assumed that the sodium vapor bubble does destructive work only above 20 atm, because this is the design pressure of the reactor cavity.

7.10 In the nuclear power burst most of the fuel is melted and a significant portion is vaporized resulting in the release of radioactive fission products and fuel isotopes. Iodine and Plutonium are of special significance in this context [10] because they represent the most important hazard potential to the environment.

Assumptions made to calculate the leakage of the radioactive material through the containment barriers were very conservative. Following the nuclear excursion and the fuel sodium interaction a great part of the primary sodium was assumed to be spread into the primary containment heating up its inert atmosphere to the sodium temperature. For the resulting overpressure of less than 3 ata the time dependence of gas leakage from the inner containment into the outer containment and from there into the atmosphere was calculated assuming pressure decay due to leakage only. No credit was taken from the fact that the containment atmosphere will cool down during the accident.

The release of radioactive material from the fuel was conservatively estimated by assuming 100 °/o for the noble gases, 10 °/o for the halogens, 50 °/o for the volatile solids and 1 °/o for the solids. These numbers are rather pessimistic in view of the fact that the fission products and the fuel isotopes released are in close contact with sodium before being dispersed in the inner containment atmosphere. Due to the good trapping capability of sodium the release fractions should be lower than assumed in these calculations. Because the fuel will be destroyed very rapidly by a fast nuclear excursion the released fractions of radioactive material were assumed to become airborne in the inner containment immediately after the power burst. Decrease of the activity concentration in the containment atmosphere was accounted for according to an exponential law of the plate out with half times of 1 h for the halogens and volatile solids and 10 h for the solid material. The weather conditions during the accident were considered to be constant. In the calculation a ground level release accompanied with inversion conditions were assumed. No credit was taken from the decontaminating effect of the exhaust and filtering system which controls the air gap around the steel containment. The accident doses given below were calculated for the design leak rates of 50 Vol. °/o / day for the primary and of 0.5 Vol. °/o / day for the secondary containment barrier.

Design Basis Accident Doses: (500 m, downwind, inversion)

Time of Exposure (h)	2	8
Bone Dose (rem)	0.98	22
Thyroid Dose (rem)	0.17	0.55

The numbers show that the bone dose is the governing radiation burden. This is due to the high inventory of Plutonium in the reactor which together with the Strontium isotopes adds considerably to the bone dose. The interrelation between the containment system and the various isotopes released during a large accident is discussed in more detail in another paper (10) of this conference.

7.11 It can be concluded that the double containment system applied to the Na-2 design provides a suitable measure to reduce the radiation hazard to a tolerable level if we use the 25 rem accident dose as a yardstick. It should be noted that the assumptions made in these calculations are rather pessimistic. We believe that with more and better information about the important parameters of activity release in fast reactor accidents the accident doses reported will be lowered considerably.

Furthermore, if plant sites near cities or areas of high population densities should be selected, additional reduction of accident doses can be achieved by accounting for the concrete cylinder around the outer containment shell as an additional barrier. Venting the air gap between the outer containment shell and the concrete cylinder to an exhaust stack would lead to better activity dispersion in the atmosphere. Adding filters to the exhaust system would even more reduce the radiation burden to the environment in case of an accident. From these studies it can be concluded that the Na-2 plant has the capacity of following the economical site requirements without hazard to the public.

CONCLUSIONS

From the safety analysis of the Na-2 design a number of conclusions can be drawn, which may serve as a guideline for future work. Concentrated effort is required in the fields of accident analysis and development of engineered safeguards.

8.1 Greatest emphasis is to be placed on development and demonstration of reliability of all critical reactor components and accident preventive safeguards, such as to prevent the occurrence of any situation which might lead to release of active material. Realization of the principles of diversity and redundancy in the entire safety system and in the emergency cooling equipment appears to be the appropriate method. Success in this area would eliminate the problems of the two following paragraphs.

8.2 The possibility of a severe nuclear accident is extremely remote, but the present state of technology and analysis leaves some doubt as to whether they can be ruled out entirely. Until sufficient confidence has been established in the safety of sodium cooled breeder reactors, it appears appropriate to account for such severe accidents in the design as was done in the Na-2 concept.

The containment of the Na-2 reactor would be capable to accomodate the potential consequences of severe nuclear accidents without hazard to the public.

8.3 Because of the pessimistic estimates made in every single step of the safety analysis, the design of the consequence limiting safeguards incorporates a high degree of conservatism. Further theoretical and experimental investigation of fast sodium cooled reactor safety will likely lead to more appropriate and more economic solutions.

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Table I

Physics Data

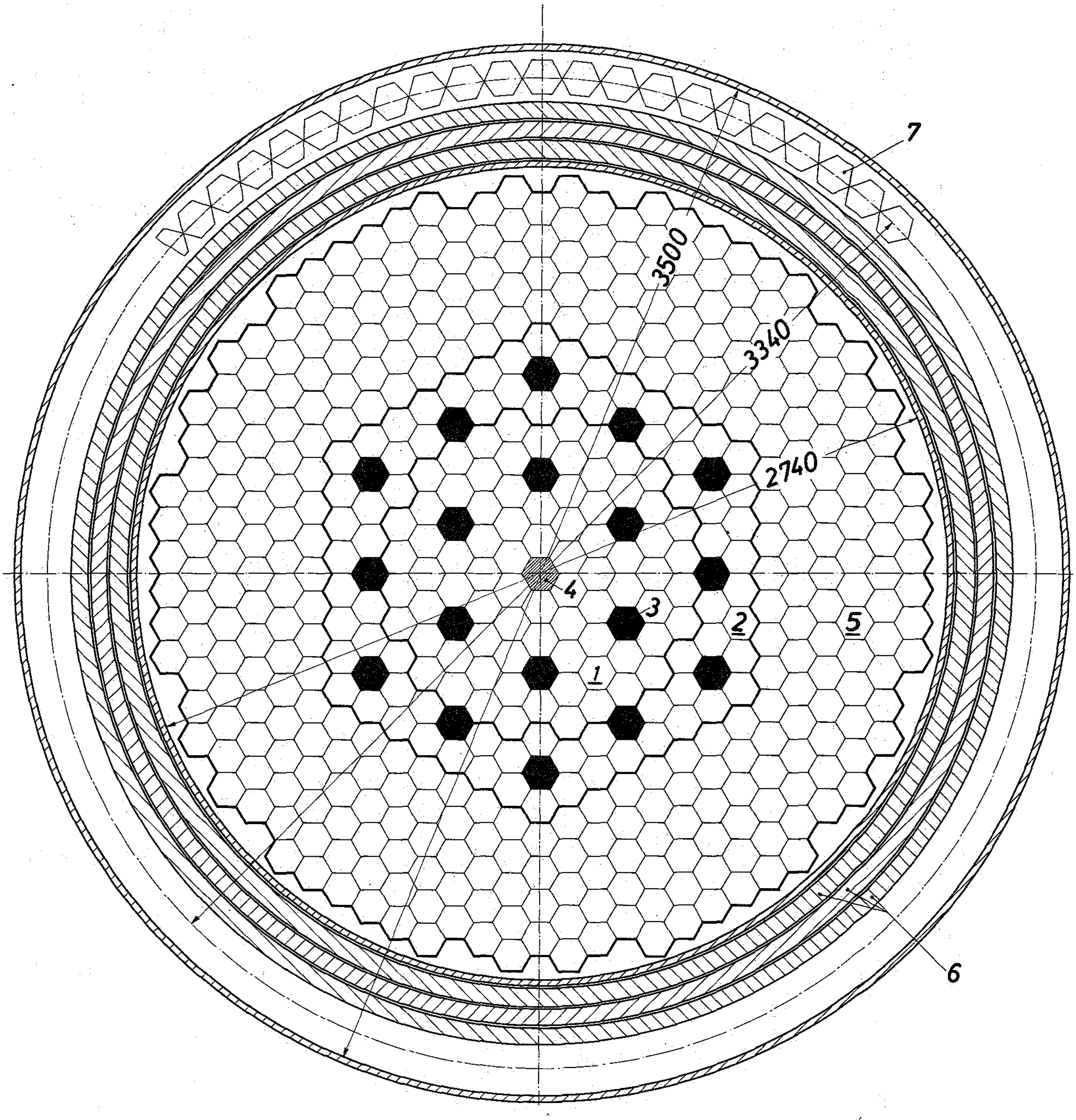
Core height	95 cm
Core diameter	153 cm
Core composition	
fuel	29.8 °/o
sodium	49.6 °/o
steel	20.6 °/o
Critical mass	
fissile	773 kg
fertile (core)	3867 kg
fertile (blanket)	26800 kg
Pu-enrichment	
inner zone	21.2 °/o
outer zone	31.4 °/o
Pu-composition (as loaded)	
Pu 239	75 °/o
Pu 240	22 °/o
Pu 241	2.5 °/o
Pu 242	0.5 °/o
Internal conversion ratio	0.55
Total breeding ratio	1.24
Prompt neutron lifetime	$3.9 \cdot 10^{-7}$ sec
Effective delayed neutron fraction (1 β)	$3.0 \cdot 10^{-3}$
Reactivity coefficients	
fuel (expansion only)	$-9.45 \cdot 10^{-6}$ grd^{-1}
clad	$+0.37 \cdot 10^{-6}$ grd^{-1}
structure	$-31.24 \cdot 10^{-6}$ grd^{-1}
sodium	$+0.37 \cdot 10^{-6}$ grd^{-1}
Doppler constant ($T \frac{dk}{dT}$)	
with sodium	$-3.76 \cdot 10^{-3}$
without sodium	$-1.76 \cdot 10^{-3}$

Reactivity effects	
cold shutdown - full power	8.5 $
burnup swing (16000 MWd/to)	9 $
sodium void (maximum)	4 $
One outer zone fuel element loaded into the center of inner zone	2.5 $
One control rod (maximum)	4.2 $
All shim rods	20 $
All safety rods	20 $

Table II

Thermal Hydraulics Data of the Core

Power	
total	730 MW
core and axial blanket, high/low burnup	660/700 MW
radial blanket, high/low burnup	70/30 MW
Peaking factor	
radial	1.20
axial	1.24
Rod power	
core average	270 W/cm
maximum hot spot	540 W/cm
Temperature	
reactor inlet	380 °C
reactor outlet	560 °C
core outlet, high/low burnup	564/576 °C
radial blanket,outlet, high/low burnup	530/440 °C
cladding hot spot	690 °C
fuel hot spot	2700 °C
Coolant velocity maximum	5 m/sec
Core pressure drop	2.1 at



1 Subassembly Inner Zone
2 Subassembly Outer Zone
3 Control Rod
4 Instrumention Position
5 Breeder Element
6 Neutron Shield
7 Spent Fuel Storage

Fig. 1 Core Arrangement

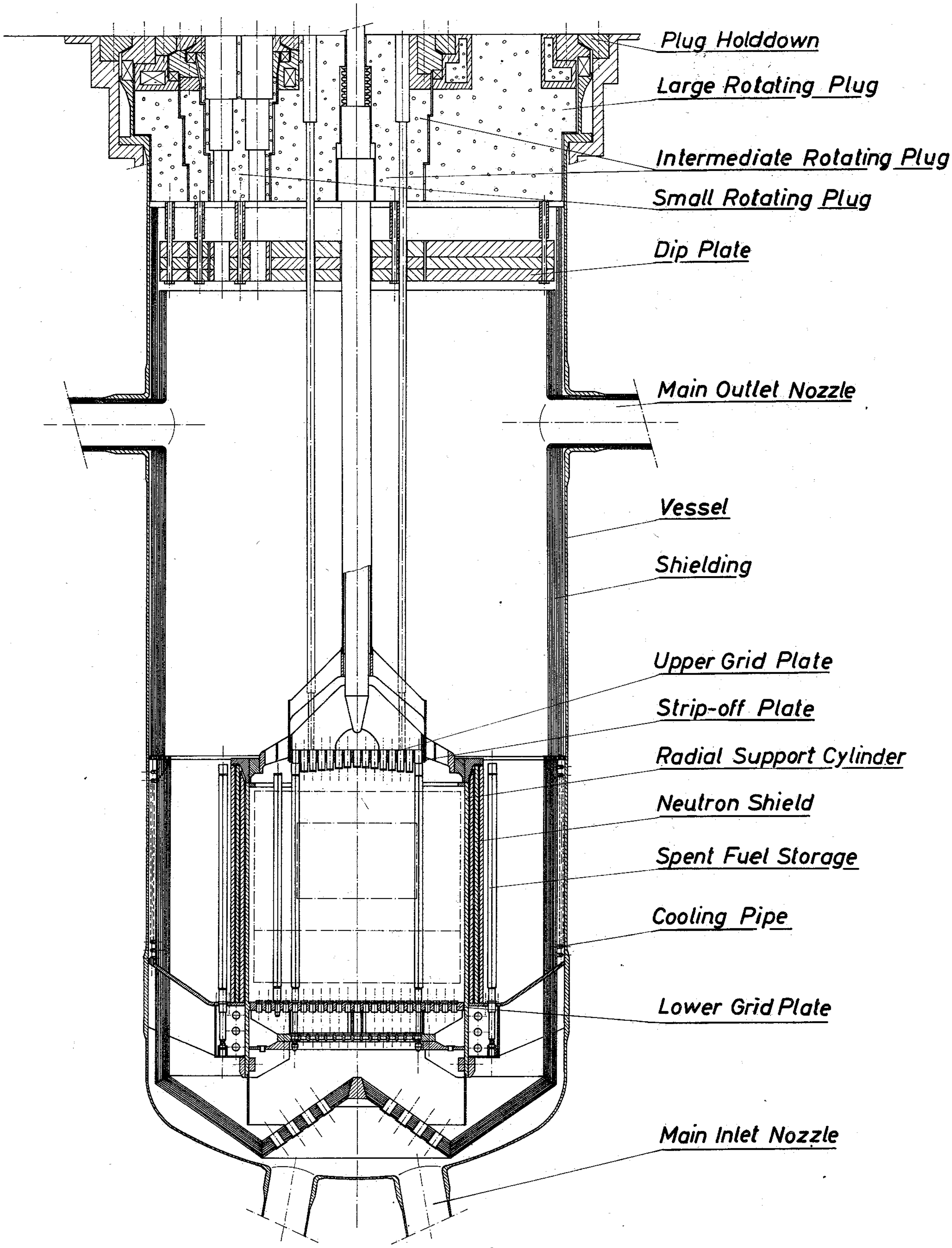


Fig. 2 Core Assembly and Reactor Vessel

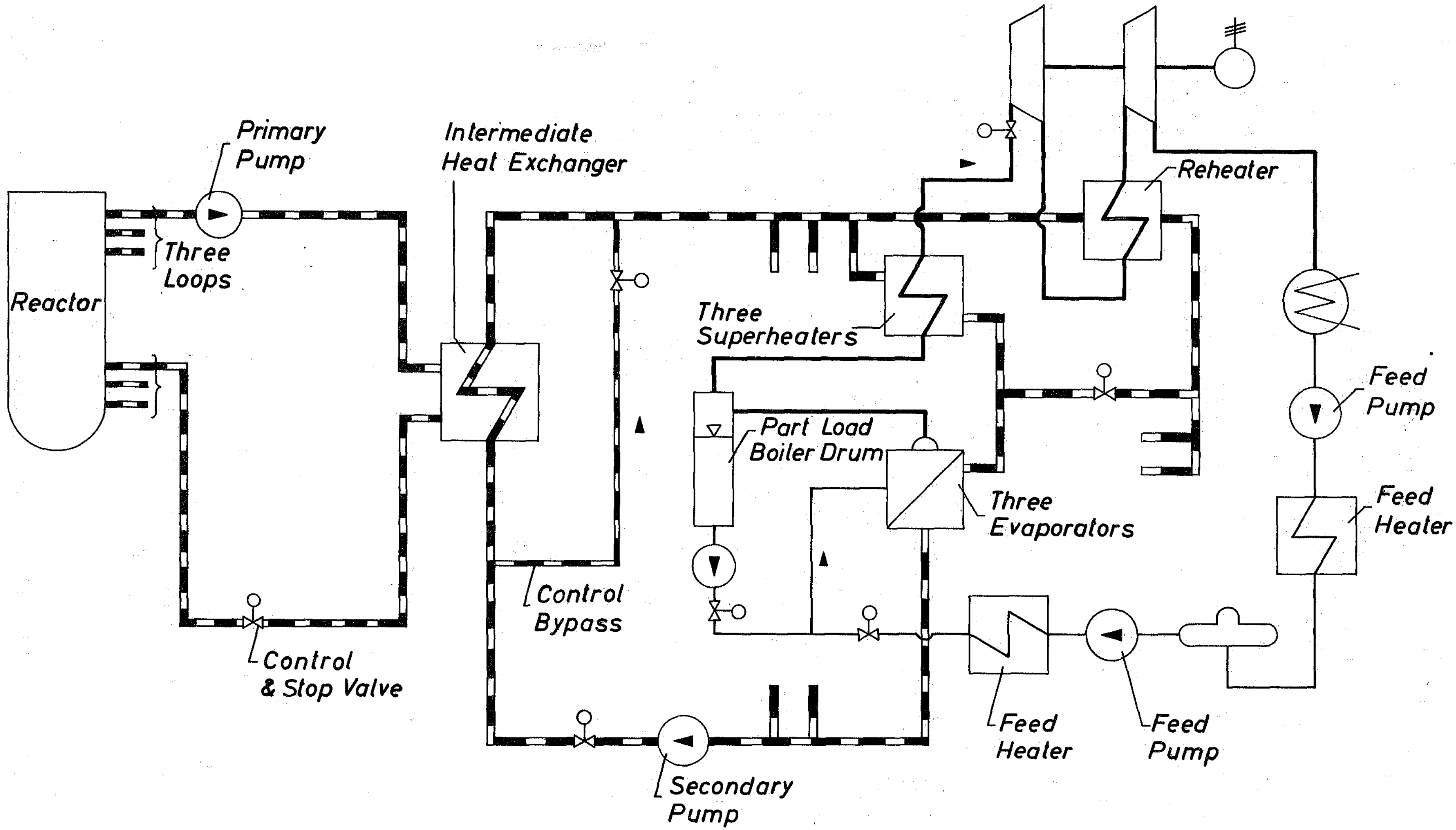
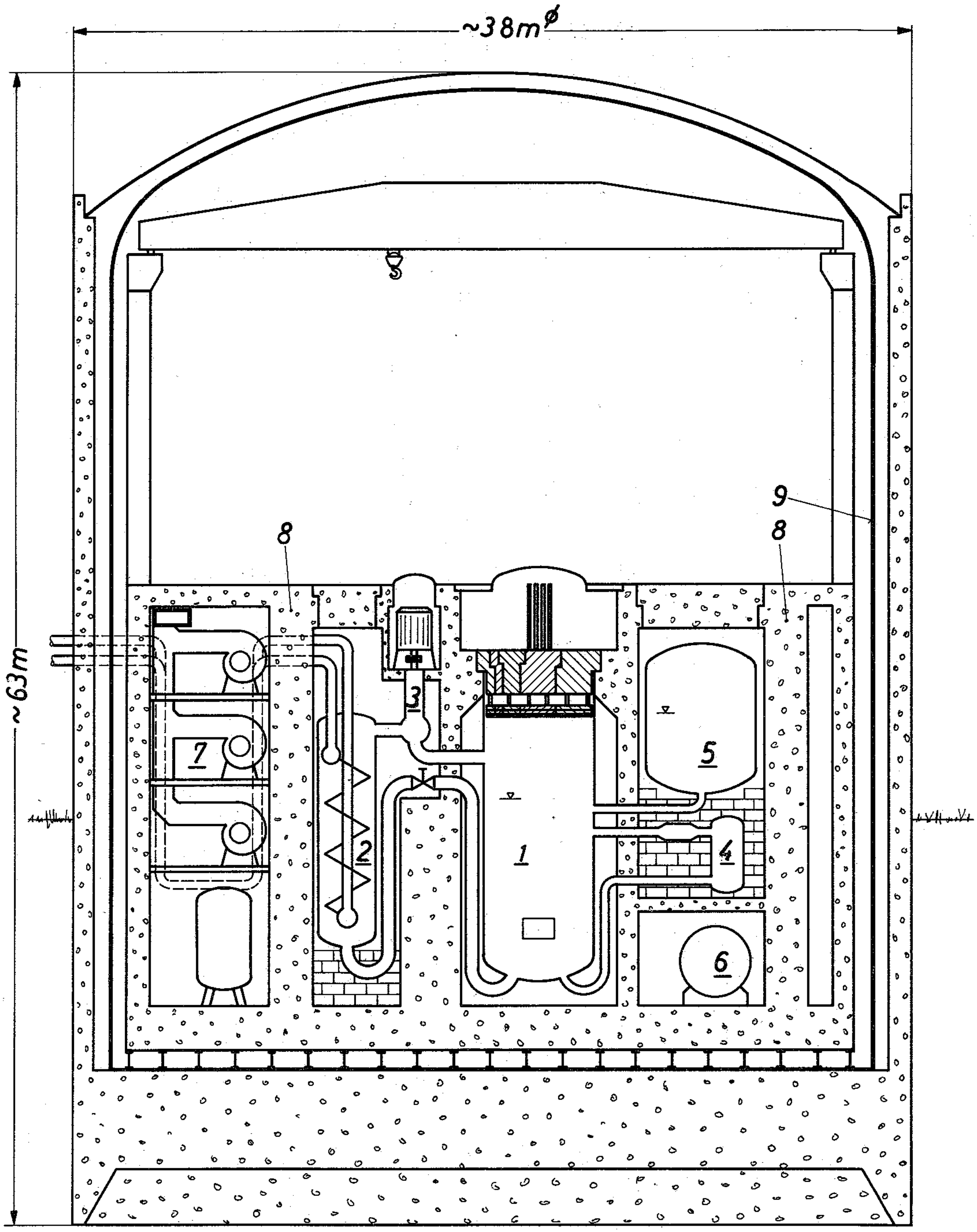


Fig. 3 Coolant System Schematic



1 Reactor Vessel
2 Intermediate Heat Exchanger
3 Primary Pump
4 Auxiliary Cooling System
5 Na-Reservoir Tank
6 Drain Tank
7 N_2-System
8 1. Containment
9 2. Containment

Fig. 4 Primary Coolant System and Containment Schematic

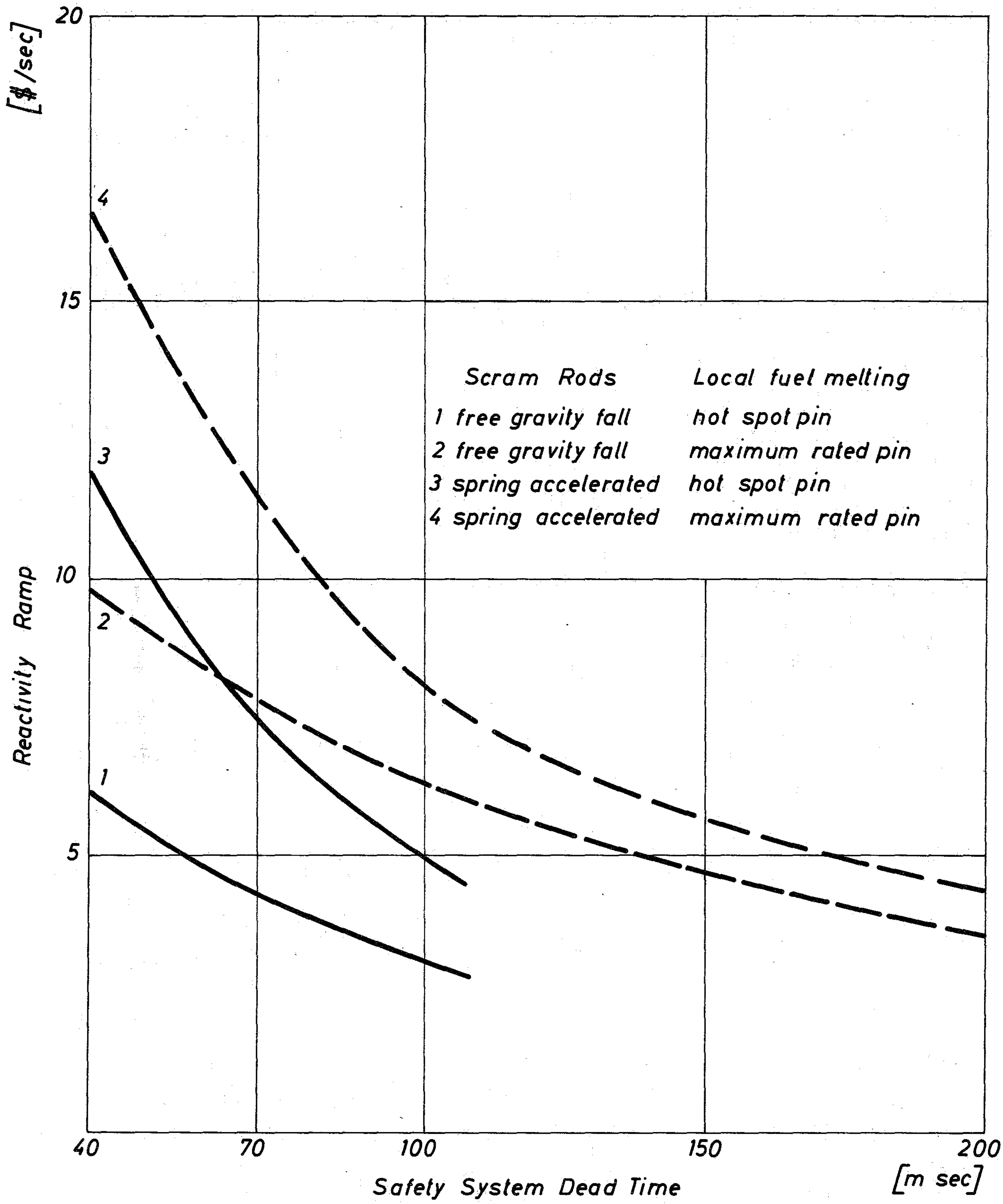


Fig. 5 Maximum Controllable Reactivity Ramp

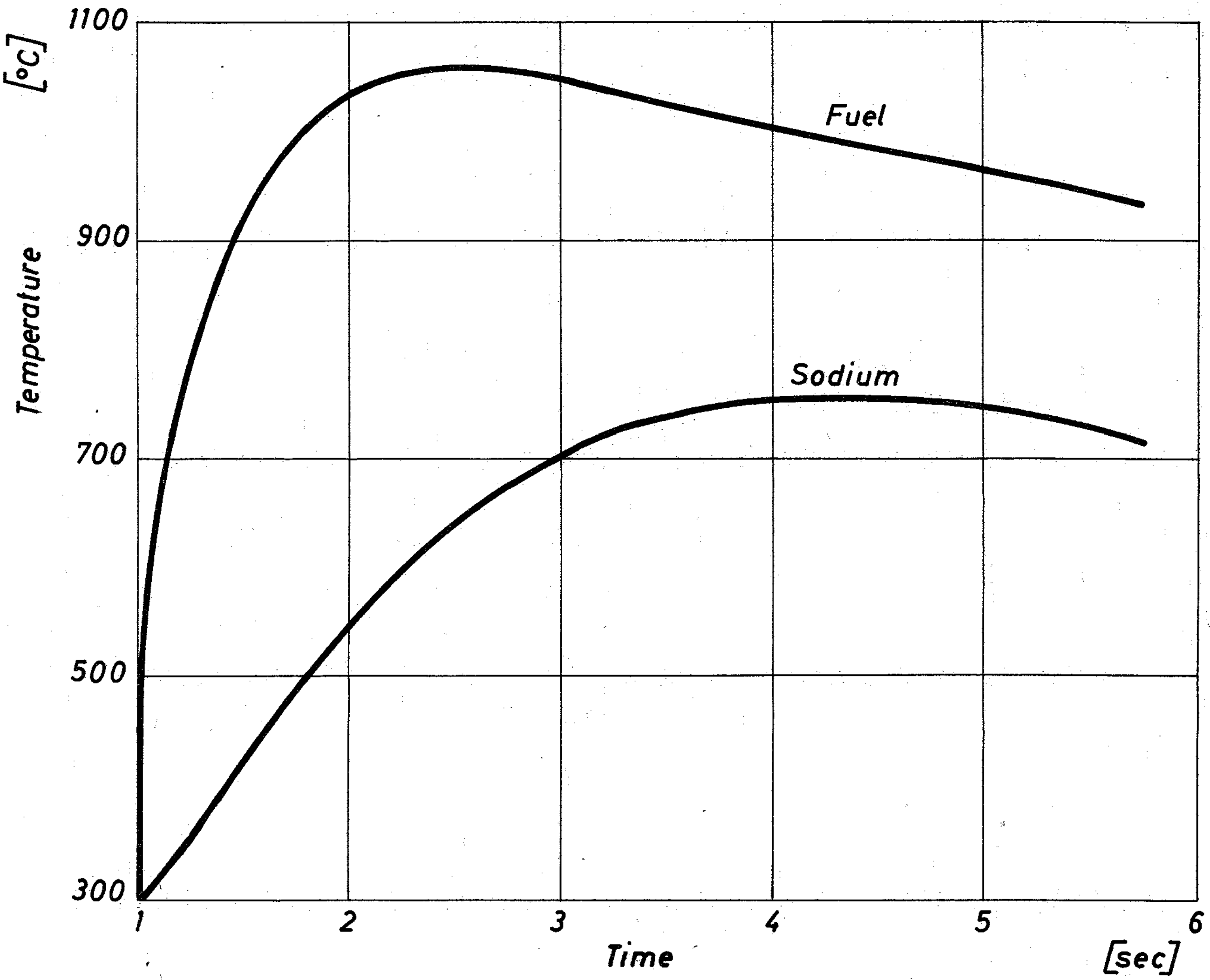


Fig.6 : Refueling Accident : Maximum Fuel and Sodium Temperatures

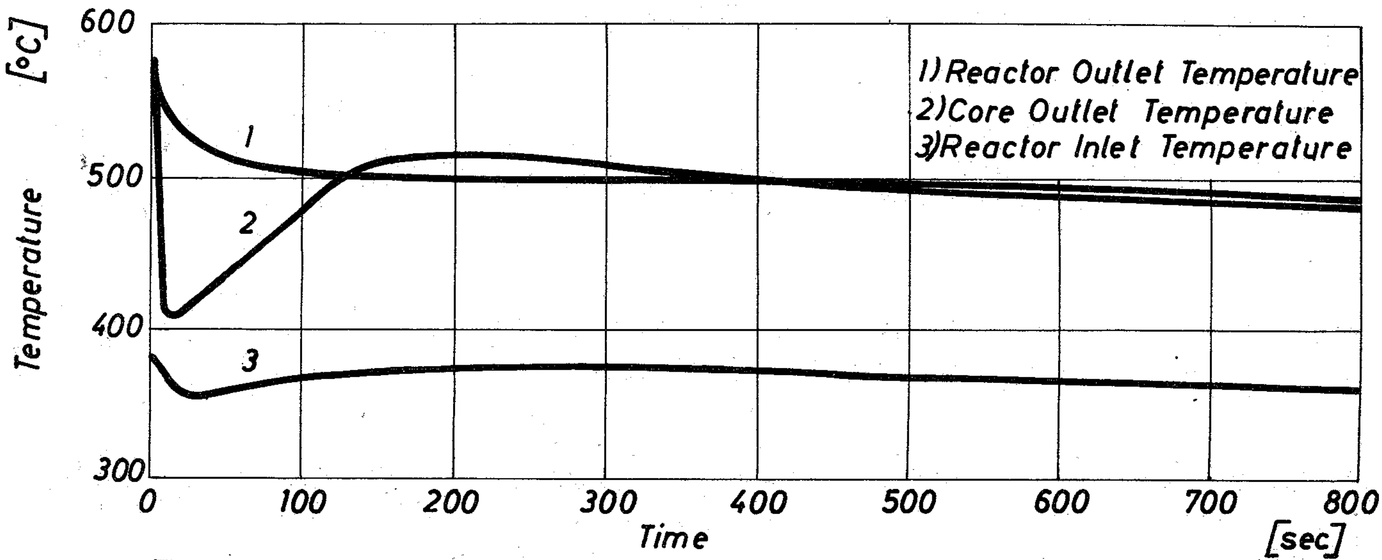


Fig. 7: Loss of Pumping Power in Primary Coolant System

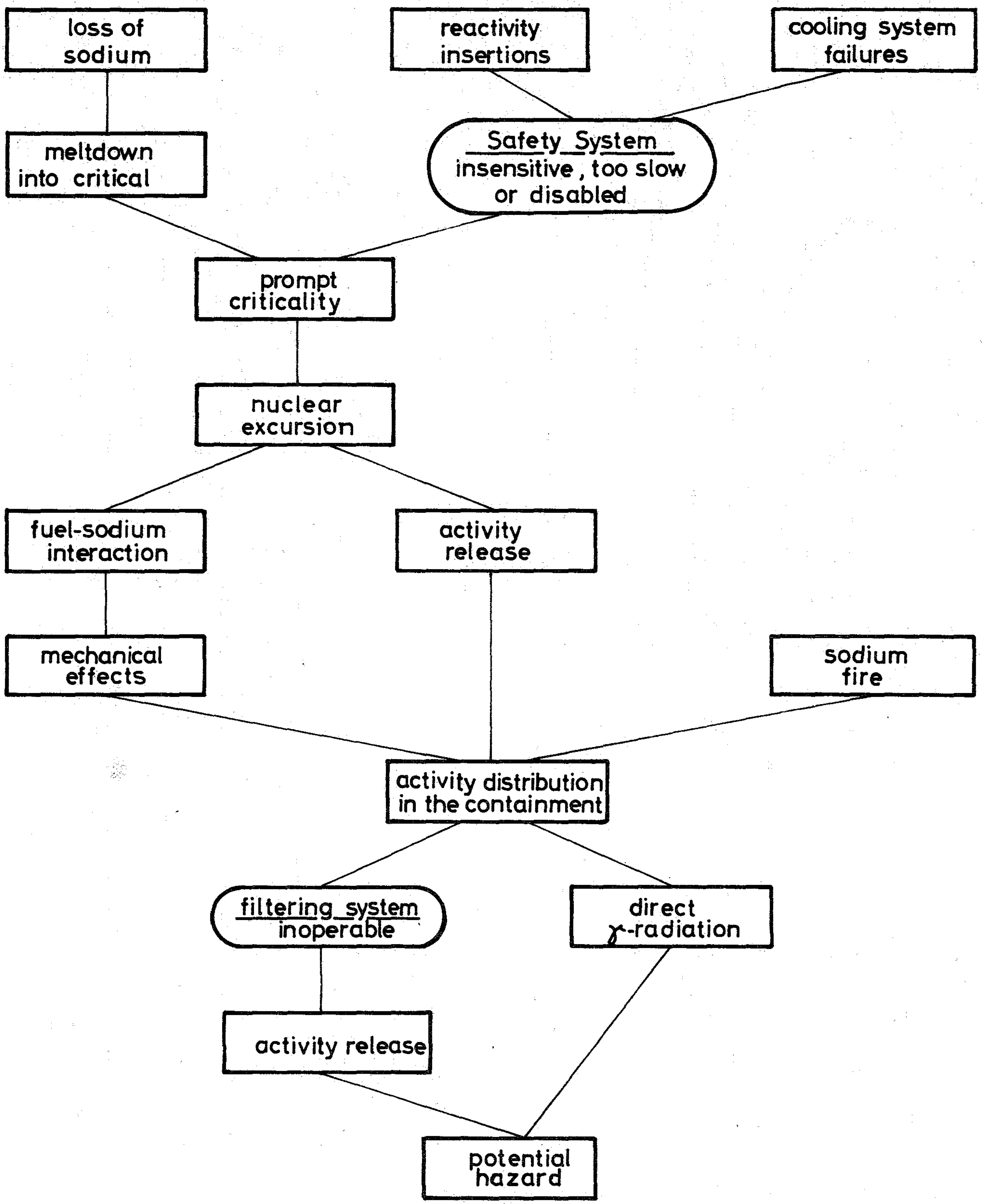
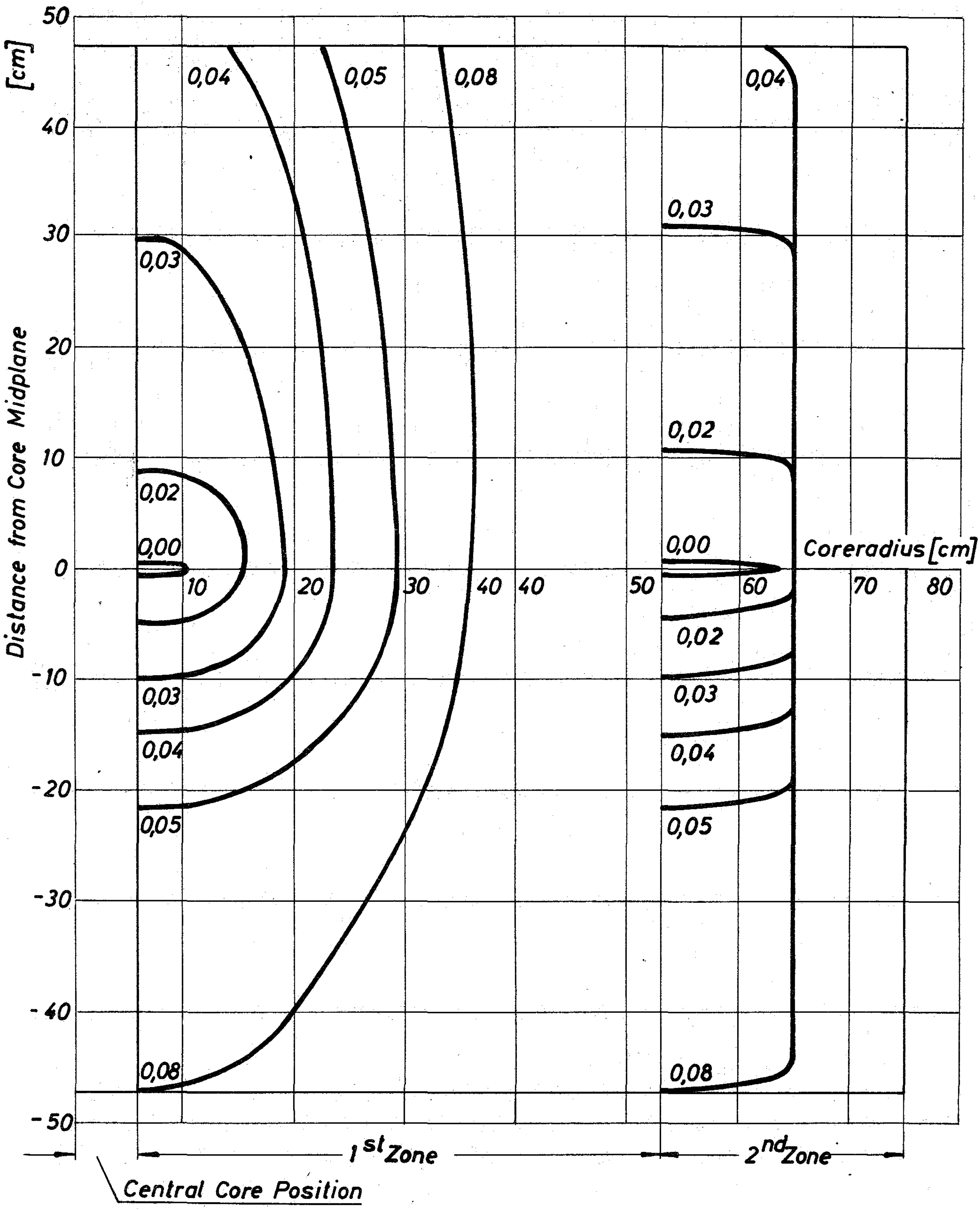


Fig. 8: Severe Accidents, Fault Propagation Tree



Curve Parameter : Time after first Bubble Nucleation

Fig.9 : Sodium Void Growth after Instantaneous Loss of Flow at Full Power

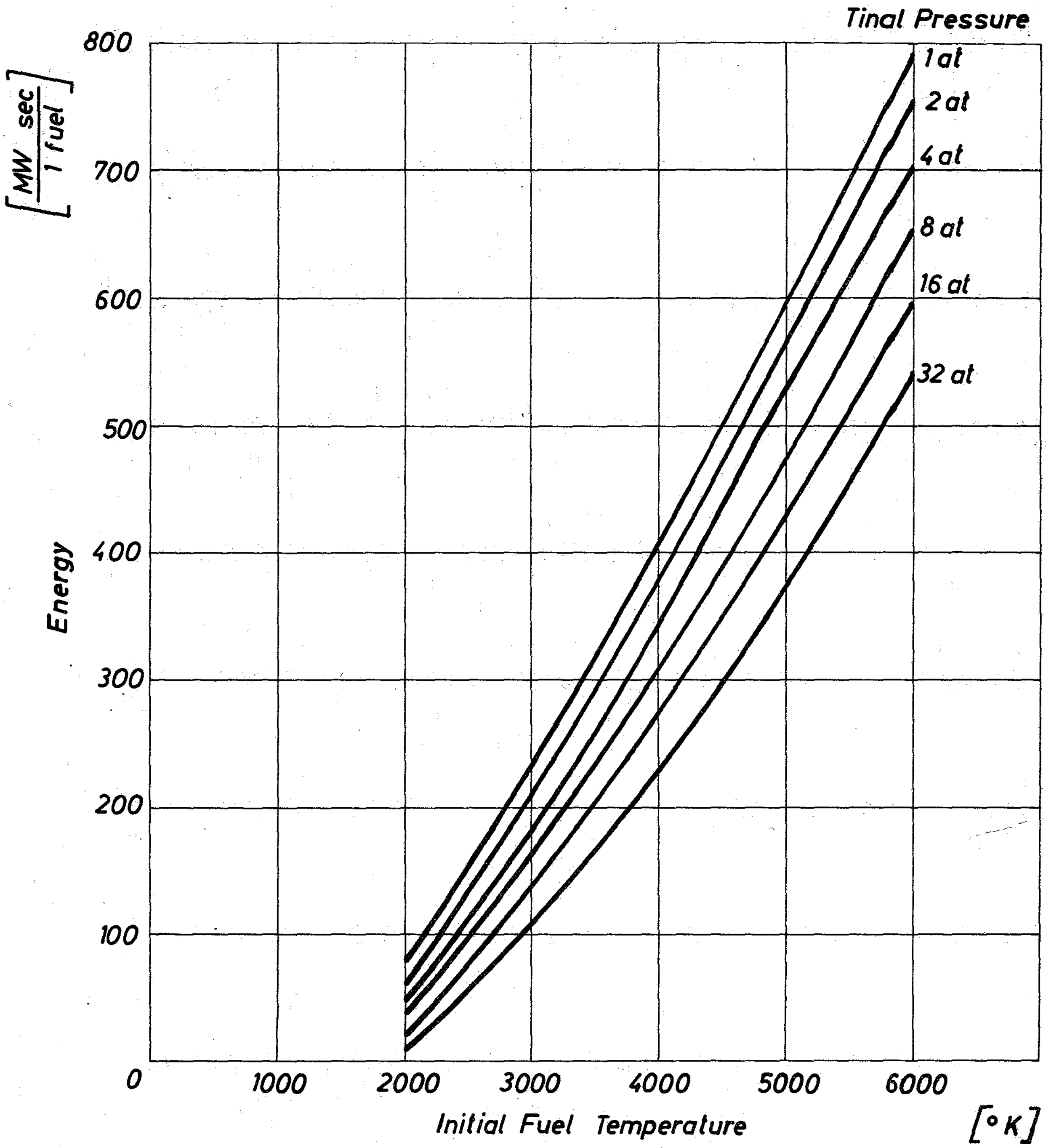


Fig.10 : Potential Energy Release from Fuel-Sodium Interaction