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Institut für Reaktorentwicklung
Institut für Angewandte Reaktorphysik
Institut für Reaktorbauelemente

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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 ASPECTS OF STEAM COOLED FAST BREEDER REACTORS *)

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ABSTRACT

In the present paper some results on the dynamic and safety analysis of the first German 1000 MWe steam cooled fast breeder reactor design (D1) are presented. It is shown that all arising problems can be solved with today's technology, mainly taken over from boiling water reactors. The special results of the investigations presented here will be useful in the determination of new steam cooled fast reactor concepts.

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

1. INTRODUCTION

The German Fast Breeder Program includes the development of a steam cooled prototype besides the sodium cooled version. Naturally a comparable effort must be given to the specific safety problems of steam cooling.

The present study is based on our first detailed design of a 1000 MWe steam cooled fast breeder plant, called D1, which was published in 1966[1,2]. As it was pointed out, this D1-concept does not claim to be the optimum of all possible steam cooled fast reactors. According to our schedule a D2-study is going to incorporate now all the experience with D1 and the subsequent analysis.

The main features of the D1-design are shown on the flow scheme of fig.1. Coolant steam enters the reactor pressure vessel at 182 atm close to saturation temperature. Before entering the core, it passes a relatively large inlet plenum. It is formed by the inner volume of the pressure vessel not occupied by the core, the steam outlet ducts, the inner shielding and some other structural parts. The superheated steam leaving the core at the bottom with 540°C is divided into three different paths: some 30 °/o go to the two main turbines, some 60 °/o to the six Loeffler circuits and some 10 °/o to two subsidiary closed coolant systems. The latter is equipped with surface type heat exchangers and standard Loeffler boilers. These two subsidiary coolant systems are always in service during reactor operation providing a permanent heat-sink for decay heat removal. In addition, the cooling efficiency of these systems could be increased remarkably by moistening the coolant steam. This would be required, if the coolant flow or the pressure decrease below a certain value during a major accident. In this case, an emergency flooding process could be carried out also with the water from the two Loeffler boilers which are located at a higher level than the reactor.

During reactor operation the two Loeffler boilers together with the water volume in the six main steam generators in the Loeffler circuits serve as Ruths-Accumulators in case of power transients, equipment failure or leakage.

The core is surrounded by a heavy steel neutron- and γ-shield and is enclosed in a pressure vessel. This arrangement has the potential of absorbing large amounts of energy in case of an excursion-type-accident. In addition, the whole reactor is located in a reinforced concrete cell, filled with water in the upper part and covered by a heavy movable bridge.

All coolant pipes and the two subsidiary coolant loops are located in a dry-well formed by concrete walls. Any overpressure there is dissipated into the water pool above the reactor (pressure suppression system) and ultimately to a pressure resistent first or inner containment. This inner containment together with the six Loeffler circuits and the two main turbogenerators is enclosed in a second gastight envelope. Here a small negative pressure is maintained by a ventilating system venting the exhaust air through filters to the stack. All penetrations between the first and the second containment are equipped with two independent stop valves. In addition, the primary coolant pipes are equipped with special quick-acting check and overload stop valves, respectively, adjacent to the reactor pressure vessel.

The basic safety philosophy of steam cooled fast reactors like D1 centers around two aspects:

a) the steam density coefficient,
b) the loss-of-coolant accident.

As with sodium, large steam cooled fast reactors have a negative reactivity coefficient of coolant density or a positive void coefficient. With sodium this coefficient practically acts only with the onset of boiling, leading to a sudden, very fast reactivity increase. With steam, on the other hand, the density coefficient is always in action, since the coolant density depends very strongly on pressure and temperature. This results in a number of dynamic problems which are stressed even more by the direct cycle system. On the other hand, the reactivity changes cannot be as fast as in the case of sodium boiling over larger core regions.

The loss-of-coolant accident, which can be excluded for sodium by proper designing, must be considered very carefully for steam. Besides the scram system other engineered safeguards, such as emergency cooling or flooding systems are required. Again, the problems are stressed by the direct cycle. Emergency shut-off valves have to isolate the core from the turbine and close the primary containment. However, as will be shown, the basic requirements of these valves do not differ from those needed for boiling water reactors for the same purpose.

Finally, the direct cycle may be contaminated by plate-out of solid fission products during operation with a can failure. First answers to these problems have been reached during operation of the ESADA - Vallecitos-Experimental-Superheat Reactor [3]. We closely participated in this project. Nevertheless, some questions are still open. These will be studied in a special experimental program scheduled for the next years by using different steam loops in German reactors. Since the circuit contamination presents mainly operational problems, it is not included in this safety analysis. Therefore, this paper is divided into two main portions:

D y n a m i c A n a l y s i s and
P r o b l e m s a n d C o n s e q u e n c e s o f t h e
L o s s - o f - C o o l a n t A c c i d e n t.

2. DYNAMIC ANALYSIS

2.1 Core Stability

The stability analysis provides information on the behaviour of the core following minor reactivity disturbances. Major reactivity disturbances are taken into account as part of the examinations on coolant cycle dynamics as well as in accident models. Effects caused by the coolant cycle are not considered in the studies on core stability. It is assumed that core inlet temperature, pressure and throughput are not variable with time. The model of calculation comprises thermodynamics, neutron kinetics and reactivity coefficients. A description of the model for sodium coolant is given in [4]. In addition for steam cooling the nonlinearities and the space dependence of the steam data are considered. Fig. 2 shows the simplified block diagram. In such a system with feedback comprising several delaying elements (heat sink, delayed neutrons) two types of instability may arise: oscillatory and monotonic instability.

2.1.1 Oscillatory Instability

It occurs when the reactivity feedback is negative and very high, which means an excessive amplification of the system. It has become apparent from calculations that the core of the steam-cooled 1000 MWe reactor is extremely far from this limit. An increase in the Doppler coefficient by 4-5 decimal powers only leads into the vicinity of the stability limit. For this reason, the inherent oscillatory instability of the core was not given further consideration. This oscillatory instability of the core must not be mistaken for the oscillatory instability of the coolant circuit caused by the pressure feedback (chapter 2.2).

2.1.2 Monotonic Instability

It occurs when the feedback of the system becomes positive. This means a positive power coefficient. In this case, only feedback, i.e. thermodynamics plus reactivity coefficients, has to be considered. The following reactivity coefficients are taken into account:

α_D	$\left[\frac{1}{^oC}\right]$	=	Doppler coefficient
α_B	"	=	fuel density coefficient
α_C	"	=	can coefficient
α_S	"	=	coefficient of structural material
α_ϱ	$\left[\frac{1}{gr/cm^3}\right]$	=	coolant density coefficient

α_D and α_ϱ are the main factors in the determination of stability, α_B depends to a very large extent on the still unknown mechanical behaviour of the fuel. Therefore, an accurate evaluation is not possible. It will not amount, however, to more than 20 % of the value of the Doppler coefficient.

2.1.3 Results

The studies on stability were applied to the core of the D1-reactor(1). The Doppler coefficient α_D and the coolant density coefficient α_ϱ being the principal determinants of stability, the stability boundary (power coefficient = 0) was represented in the α_D - α_ϱ plane (fig.3). The position of the D1-core is also entered in that plane. The D1-core is situated in the stable zone. Assuming realistic errors in the calculation of the coefficients ($\pm$ 25 % for α_D and $\pm$ 40 % for α_ϱ) we obtain the shaded rectangle extending in part into the unstable zone.

In order to be sure to have a stable core we have to determine the reactivity coefficients with greater accuracy. If this is not feasible or if the more accurate values provide an unstable core, stability can be improved in different ways, which are described in the following paragraph. However, implications for other design aspects (e.g. thermal efficiency, breeding ratio) are not taken into consideration.

An inherent unstable core raises not so much problems of safety but rather problems of control. Most of the accidents considered in the accident analysis take place in the range of milliseconds, while effects of inherent instability occur in the range of seconds only. But the instable core forces the control system to intervene much more frequently; consequently permanent rod movement and considerable wear of the driving and guiding mechanism has to be expected.

2.1.4 Improvements in Stability

a) Change of Reactivity Coefficients

The increase in $|\alpha_D|$ and the reduction of $|\alpha_\varrho|$ cause an improvement in stability. The influence is nearly linear. A considerable reduction of $|\alpha_\varrho|$ can probably be reached with an annular core (5).

b) Increase in Power Density (rod power)

When the power density is increased with the temperature rise remaining unchanged, the average fuel temperature rises, whilst the average coolant temperature does not change. This enhances the effect of the Doppler coefficient. The influence is slightly smaller than linear ($\sim q^{0.8}$).

c) Shortening of Temperature Rise between Inlet and Outlet $\Delta\vartheta$

Shortening of $\Delta\vartheta$ with power remaining unchanged is reached by an increase in throughput. Assuming a constant value for the average coolant temperature $\bar{\vartheta}$, the improvement of stability is nearly linearly dependent on the increase in throughput. However, in normal circumstances, $\bar{\vartheta}$ also changes with the increase in throughput. In this case the influence of d) must also be considered.

d) Reduction in Change of Steam Density with Temperature ($d\varrho/dT$)

The dependence of steam density on temperature decreases with the increase in the average coolant temperature $\bar{\vartheta}$. If $\Delta\vartheta$ is shortened by reducing the outlet temperature ϑ_A, $\bar{\vartheta}$ will be reduced. A shortening of $\Delta\vartheta$ and a reduction in $\bar{\vartheta}$,however, have opposite effects on stability, the stability enhancing part being predominant. If $\Delta\vartheta$ is shortened by raising the inlet temperature ϑ_E (slight superheating), $\bar{\vartheta}$ is raised. In this case, both factors improve stability. Fig. 4 shows the stability boundaries $\alpha_{\varrho b} = f(\Delta\vartheta)$. The much better improvement of stability following the increase in inlet temperature ϑ_E is obvious. For illustration, the position of the D1-core has been entered.

e) Increase in Thermal Resistance

An increase in the thermal resistance can be reached by a deterioration of thermal conductivity λ. The related increase in fuel temperature results in a stronger influence of the Doppler coefficient. However, stability is improved only proportional to $(\frac{1}{\lambda})^{0.6}$. Due to the small dependence and to other technological problems, this means of improving stability does not seem to be reasonable.

2.2 Dynamic Behaviour of the Core Including the Coolant Cycle

2.2.1 Analogue Model of Calculation

The main parts of the D1-plant (1) are simulated by an extensive analogue program. Fig. 5 shows a simplified flow chart of this model. The model contains a double steam feedback, one of them represents five loops, the other one loop. By that means it is possible to simulate failures in one of the 6 loops (e.g. the failure of one blower or one steam generator).

The main characteristics of the program are:

Lumped model of the neutron kinetics, 6 groups of delayed neutrons, one fuel element representing the core, radial and axial division into several zones, representation of the heat transfer as a function of temperature and steam velocity, the steam density as a function of pressure and temperature, the specific heat of the steam as a function of temperature; pressure drop and heat capacity of the blanket are taken into account. Inlet and outlet plenums of the reactor, pipes, reheaters, steam generators, blowers, check valves, Ruths-Accumulators and turbine mass flow are represented, partly in a simplified way. The dependence of the steam characteristics on pressure and temperature is separated and represented by function generators or linearisation. Because of this simplification the accuracy of the model becomes insufficient at pressure changes of more than 25 at from the initial value.

With this analogue model the behaviour of the plant can be investigated under normal conditions (load changes) and during failures (e.g. leaks, failure of a blower or a steam generator.) The time behaviour of all important quantities, as reactor power P/P_o, max. can temperature ϑ_{Cmax}, reactivity δk, coolant temperature ϑ, outlet pressure p_A, core mass flow $\dot{m}/\dot{m}_o$ and many other quantities can be calculated.

On the one hand the results show the influence of different system parameters (e.g. reactivity coefficients, the quantity of water etc.) on the dynamic behaviour of the plant, on the other hand they are useful for the development of an optimal control system, which shall be included into the next stage of the program.

The most important parameter is the reactivity coefficient of the steam density (density coefficient α_ϱ). It depends on the burnup, and furthermore it cannot be calculated with good accuracy, but the dynamic behaviour of the plant depends strongly on this coefficient. Therefore, it is varied in a large region ($-0.46 \leq \alpha_\varrho \leq +0.18 \frac{1}{gr/cm^3}$) at all investigations. The maximum value calculated for D1 (at maximum burnup) is $\alpha_{\varrho o} = -0.37 \frac{1}{gr/cm^3}$; a mean value is $\alpha_\varrho = -0.275 \approx 0.75\ \alpha_{\varrho o}$.
The numbers at the curves in the figures 6 to 15 have the following meaning:

Curve	α_ϱ [gr/cm^3]$^{-1}$
1	$-0.090 = 0.25 \cdot \alpha_{\varrho o}$
2	$-0.185 = 0.5 \cdot$ "
3	$-0.275 = 0.75 \cdot$ "
4	$-0.370 = 1.0 \cdot$ "
5	$-0.460 = 1.25 \cdot$ "
6	$+0.185 = -0.5 \cdot$ "

2.2.2 Normal Behaviour of the Plant (Load Changes)

In order to investigate the behaviour at load changes the turbine power is suddenly raised from stationary state by 10 %. In the analogue simulation this effect is represented by a sudden rise of the turbine mass flow by the corresponding amount. The results show the behaviour of the plant after a load change without the influence of a control system or a movement of the control rods. In fig. 6 the reactor power P/P_o and the core outlet pressure p_A are represented as a function of time for different density coefficients.

The increasing turbine load (increasing mass flow) forces a decreasing pressure. When α_ϱ is negative, the decreasing pressure causes a power rise until there is a new equilibrium, so that the reactor power corresponds to the turbine load. This behaviour is good at a small negative density coefficient ($\alpha_\varrho = 0.5\ \alpha_{\varrho o}$). Increasing values of α_ϱ lead to an overshoot of the power and finally to an increasing oscillation at $\alpha_\varrho = \alpha_{\varrho o}$. At positive values of α_ϱ the reactor power doesn't follow the turbine load. Increasing turbine load forces decreasing reactor power.

This behaviour leads to the demand, that the density coefficient α_ϱ should be small and negative. It must be smaller than the boundary value of the core stability (chapt.2.1). With increasing negative values of α_ϱ the boundary of instable oscillations of the coolant cycle is reached before the stability boundary of the core. The instability of the coolant cycle is reached the sooner the greater the time delays of the cycle are; e.g. the reheater with its very long time delay makes worse the stability of the cycle (chapt.2.5).

2.2.3 Reactivity Disturbance

The investigation of reactivity disturbances is made for two reasons: First it is shown how the plant behaves at real reactivity disturbance, secondly the dynamic of reactivity disturbance is a help in designing a control system.

Fig. 7 shows the dynamic behaviour of the reactor power, the max. can temperature and the outlet pressure after a +0.2 \$ step function of reactivity for several values of α_ϱ. Because of the pressure feedback the initially rising power is finally set back to 100 %, if α_ϱ is negative.

The pressure rises until it has compensated the reactivity disturbance (in the range of some minutes, therefore not visible in fig.7). Increasing negative values of α_ϱ lead to an overshoot and to undamped oscillation. If α_ϱ is positive, the cycle becomes unstable because of the positive pressure feedback. Then the power is increasing monotonously.

2.2.4 Faults at Components of the Steam Cycle

Failure of a Blower

It is assumed that one of the 6 turboblowers fails. Both loss of drive and sudden blockage are investigated. There is only a small difference between these two kinds of failures. In both cases it is assumed, that the check valves shut, when the mass flow turns back in the disturbed cycle. The shown curves of fig. 8 refer to a sudden blockage. After the failure of one blower the other 5 blowers take over a part of the missing flow, according to the steepness of the characteristic line of the blowers. In this case a flow reduction of 10 °/o remains, which leads to increasing coolant and can temperatures. When α_ϱ is negative, the increasing coolant temperature causes an increasing power, which is reduced first by the Doppler coefficient and later on by the pressure feedback until the initial value is reached. Here also oscillations are possible if α_ϱ is too large. During the transients a too large overshoot of the can temperature is possible, so that a scram is necessary. If there are possibilities to reduce the overshoot of the can temperature, the reactor need not be scramed after a failure of one blower (chapt. 2.5).

Failure of a Steam Generator (Evaporator)

The sudden loss of feedwater in one of the 6 injectors is assumed. The steam leaves the injector nearly as hot as it enters. Because of the reduced density of the steam the blower of the disturbed cycle becomes unstable and the direction of the flow reverses. When the check valve shuts, the dynamic behaviour of the five normal cycles after a failure of the injector is the same as after the failure of a blower. The calculated curves correspond to those of the failure of a blower. But it must be noticed that the moment t = 0 is not the beginning of the failure but the time the check valve shuts.

Leaks

A leak at the outlet is similar to an increase of the turbine load. Because of the pressure reduction the power increases. At the beginning of the disturbance the can temperature decreases by reason of the larger coolant flow through the core. More dangerous are the effects of a leak near the inlet of the reactor, because the rise of the power is accompanied by a reduced coolant flow. Fig. 9 represents the course of the power and the max. can temperature caused by a leak near the inlet with an initial throughput of 500 kg/sec (15 °/o of the total mass flow). Because of the quick loss of pressure (ca. 2 at/sec) the simulation is possible only over some seconds. Power and can temperature increase more or less as a function of α_{ς} .

Break of a Main Turbine Pipe

The sudden break of one of the two main turbine pipes causes a rise of the mass flow in the disturbed pipe by more than 5 times. The results of this accident are shown in fig. 10. As the mass flow through the core increases, the power and the can temperature initially decrease. Only after 3 sec the can temperature exceeds its initial value. This time delay is a good help for closing the safety valves in due time.

2.2.5 Parameter Variations

The dynamic behaviour of the plant can be changed strongly by variation of core parameters and components of the cycle (6). Because of the very complicated relations and the multiple feedbacks an exact prediction on the influence of the different parameters is not always possible with simple means. Therefore, the analogue model was used to the parameter variations.

A very rough valuation using a simplified closed loop control system and the stability criteria of Hurwitz (7) has shown, that the t i m e d e l a y s of the circuit have to be as s m a l l a s p o s s i b l e to get a good dynamic behaviour. The second condition which must be fulfilled is a n e g a t i v e f e e d b a c k to get a stable circuit. The feedback is negative, if the coolant density coefficient is negative. A third limitation is the g a i n o f t h e l o o p. If it exceeds a

certain limit, oscillations rise. The gain of the loop depends on the reactivity coefficients and on the water contents of the steam generators. Reduced Doppler coefficient, enlarged density coefficient and diminished water contents of the steam generators lead to an enlarged gain of the loop. The gain limit is a function of the time delays. The larger the time delays are, the smaller is the gain limit.
This rough estimate shows the way to improve the stability and the dynamic behaviour of the plant. Based on these considerations the most important parameter variations were executed with the analogue model.

Reactivity Coefficients

The influence of the main reactivity coefficients has been investigated. The variation of the steam density coefficient $\alpha_{\varrho o}$ has been shown in chapter 2.2.2 - 2.2.4. The cycle is monotonously unstable if α is positive, too large negative values of α_{ϱ} cause oscillations. The stability boundary for the D1-plant is near $\alpha_{\varrho} = - 0.3 \; [gr/cm^3]^{-1}$. This boundary depends on the Doppler coefficient and on the components of the cycle. The stability becomes better with increasing Doppler coefficient and decreasing time delays of the steam cycle.

Influence of the Reheater

The reheater is simulated by a second order time delay of the coolant temperature. This representation of the transport time and the heat capacity of the reheater is simplified but good enough to show the influence of the reheater on the dynamic behaviour of the cycle. The time delay of the reheater is the largest of the plant and the dynamic behaviour is much better if there were no reheating (or an extern reheating). Fig. 11 shows the comparison of the D1 cycle and a cycle without reheater. Without reheater curve 3 ($0.75\ \alpha_{\varrho o}$) shows a very good dynamic behaviour with a small overshoot, whilst the same curve 3 shows an oscillation, if a reheater exists. Curve 4 shows an unstable behaviour with reheater and a stable one if there is none. During a failure of a blower too, the omission of the reheater is advantageous. The reason for this good behaviour is the more quickly acting pressure feedback, which stops the power rise earlier. If α_{ϱ} is small enough ($\alpha_{\varrho} < \alpha_{\varrho o}$), it is not necessary to scram the reactor after a failure of one blower, because of the sudden increase and overshoot of the can temperature. The reactor can work on at a reduced power level.

Variation of the Water Contents in the Steam Generator (Evaporator)

The most important parameter of the evaporator is the water volume. The 6 injectors of the D1-design have a water volume of 45 t alltogether. In the variation this volume has been enlarged to the fourfold value (180 t), which is the water volume of a corresponding Loeffler boiler. The enlargement of the water volume has a stabilizing effect, because the pressure changes are smaller. This is advantageous at load changes (fig.12), but brings a small disadvantage after the failure of a blower because of the weaker pressure feedback, which reduces the increasing power (fig.13).

Ruths-Accumulator at the Inlet Plenum

The Loeffler boilers of the two emergency coolant circuits (fig.1) have the effect of a Ruth accumulator with regard to the main coolant cycles. It works against a pressure decrease by delivering steam into the inlet plenum. The opposite effect is very small, so that increasing pressure which is often wanted to reduce the power, is not influenced by the Ruth accumulator. Because of this behaviour and because of the favourable position in the D1-design near the inlet plenum the effect of the Ruth accumulator is very good. The difference between a cycle with and without Ruth accumulator can be seen in fig. 14 for a leak near the inlet plenum and in fig. 15 after the failure of a blower.

3. PROBLEMS AND CONSEQUENCES OF THE LOSS-OF-COOLANT ACCIDENT

3.1 Types and Interconnections of Accidents

The probability of creating a major leak in a well designed and fabricated high pressure system is very low as well as a complete malfunction of an important engineered safeguard. Consequently, the probability of the simultaneous occurrence of two or more of such independent events is extremely small. The following considerations are not concerned with these probabilities in any respect. As a first step they show the principal logical interconnections of different events. As will be seen, the situation is similar to that of boiling water reactors.

As for these reactors, our assumptions are being restricted to a major pipe rupture of a single main coolant pipe. This leak may occur anywhere, directly at the reactor vessel or outside the inner containment. In principle we distinguish according to fig. 1:

Leak A in the region of the coolant inlet pressure inside of inner containment, where the position directly by the reactor vessel is the most dangerous one.

Leak B in the region of the outlet pressure, also inside the containment.

Leak C outside the inner containment in the turbine building.

Taking the dependence of reactivity versus steam density for the max.burnup according to Abb. 7.2-6 of [(1)], the critical velocities of the leak flow and the effect of the steam and water storage capacities (Ruths-accumulators and main steam generators) the following ramps have been calculated:

Leak A:	4	$/sec
Leak B:	.6	$/sec
Leak C:	.3	$/sec

In all three cases critical mass flow is reached. Up to the leak opening the saturated steam depressurizes to a water-steam-mixture. The critical mass flow is calculated according to Moody's theory (max.possible critical mass flow).

The steam density in the coolant channels is calculated for different axial sections; the average value gives the ramp-rate.
The mass flow delivered by the blowers remains constant as long as no pipe rupture occurs in the corresponding coolant circuit. The pressure drops across the core, between outlet plenum and steam generator, and between Ruths accumulator and inlet plenum were taken into account. The pressure drop between pressure vessel (inlet or outlet plenum) and leak, however, is neglected for leaks A and B, but taken into account for leak C.

The results are shown in fig.16 for leak A and fig.17 vor leak B. The sharp decrease of the steam density at the beginning for leak B is caused by the quick pressure reduction in the outlet plenum. After a certain delay time this tendency is turned by decrease of the outlet temperature (similar

behaviour as is shown in fig.10, break of the main turbine pipe). Only after that period, a stable density gradient is generated leading to the reactivity ramp of .6 \$/sec mentioned before.
The first transient does not give additional safety problems, because the maximal positive reactivity increase amounts only to some .3 \$.

The reactivity ramp at rupture of an inlet pipe (leak A) can be remarkably reduced by increasing the cross section of the pipe between the Ruths-accumulator and the inlet plenum, i.e. by reducing the pressure drop between those two. For a diameter corresponding to the main steam pipes a ramp of only 2 \$/sec was calculated. The results obtained with this digital calculation are in good agreement with those of the analogue program described in 2.2. Therefore the behaviour in the case of leak C can be seen in fig.10.

Let us now follow the chain of events as given in fig.18.
Leaks A and B are combined in one diagram, where A is the more dangerous one, because the flow rate in the core decreases remarkably as was shown in fig.16. The engineered safeguards protecting the public against radiation hazards act in four major steps:

a) Safety or scram system
b) Emergency cooling system
c) Inner containment including the pressure suppression system.
d) Isolation valves which shut off all coolant pipes penetrating the inner containment.

In addition, there is an outer building enclosing the whole inner containment, the turbine plant and the Loeffler circuits being sufficiently leak-tight to guarantee activity release only via the stack and not through building leakage.

Ad a)

The scram system has to deal with the reactivity ramps given by the coolant density change. The first problem is to get a signal fast enough for the scram from the loss of coolant. This may be generated by a pressure decrease, by coolant flow change or by neutron flux increase. Using the signal of 125 °/o nominal flux and assuming a 100 ms delay for the flux detectors, the electronic equipment and the scram-rod-drives, it would be able to shut off reactivity ramps even up to 16 $/sec without fuel or can melting if the cooling behaviour is not affected. This provides a sufficient margin for cases B and C, but further studies are necessary on A with respect to the decrease of coolant flow.

Ad b)

Emergency cooling as described in [1,2] relies on the two subsidiary coolant loops, water storage in the Ruths-accumulators and main steam generators and water injection into the coolant flow. Finally the core has to be brought into the flooded condition. Some questions in connection with this flooding procedure will be solved by an experimental program under way now, especially with respect to the Leidenfrost-effect.

Ad c)

The inner containment with the reactor cell and the pressure suppression system is engineered to wellknown rules and can withstand the energy release of a Bethe-Tait-excursion following core-meltdown.

Ad d)

Like in boiling water reactors the coolant pipes must be sealed by emergency isolation valves. All pipes have two valves in series reducing the probability of malfunction. Additionally flow reversal valves are applied at the reactor vessel entry and overflow valves at the reactor exit.

As shown in fig.18 we may arrive at harmless incidents (HI), severe accidents (SA) if some malfunctions lead to major fission product release, but where containment and isolation valves work, or very severe accidents, if the containment integrity is disturbed (probabilities not considered).

3.2 Radiological Calculations

The upper limits for the radiation burden to the environment were calculated by means of a digital program (MUNDO) described elsewhere [8]. The assumptions concerning the course of the accident and the magnitude of the most important parameters will be briefly mentioned here.

3.2.1 Severe Accidents (SA)

Melting and vaporization of large parts of the core lead to instantaneous release of certain fractions (see below) of fission products and fuel into the inner containment; uniform distribution and an exponential plate-out in the containment is assumed.

Fission products and fuel	Release fractions	plate-out half-times
Inert gases	100 °/o	(∞)
Halogens	50 °/o	1 h
Volatile solids	50 °/o	1 h
Solids	1 °/o	10 h

A leak rate of 1 vol.°/o per day of the inner containment, a venting rate of 100 °/o per day of the outer containment, vent filtering with an efficiency of 99 °/o for vapors and aerosols and succeeding release through a 75 m stack lead to a radiation dose below 25 rem even for unfavourable (unstable) meteorologic conditions and exposure times up to 24 hours. Doses due to incorporation of radioactive materials as well as Gamma-radiation of the cloud have been considered.

3.2.2 Very Severe Accidents (VSA)

Two types of VSA have been considered:

a) Release of 100 °/o of the inert gases due to can melting only (isolation valves may already be closed at the time when core-vaporization occurs). If there is no leak of type C a slug flow of all the gas through the main turbine, the off-gas hold up piping and the stack into the atmosphere will lead to cloud Gamma doses above permissible levels in the vicinity of the reactor.

b) Release of the same fractions of fissionp product as in case 3.2.1 while leak C has occurred and isolation valves stay open. Uniform distribution of the released material in the second containment is assumed, plate-out, vent rate and filtering efficiencies, stack height and meteorologic conditions are the same as in case 3.2.1. The resulting radiation dose in the environment due to incorporation of aerosols, especially

Pu, will now be higher than the cloud radiation dose of the inert gases and again far exceed permissible levels.

Therefore, the consequences of any very severe accident must be limited by engineered safeguards. In case a) the liberation to the atmosphere can be restricted with the help of the off-gas hold up piping. Similar to boiling water reactors, the relatively long hold up time between the turbine and the stack (some 30 min) will allow to trap the fission product gases by closure of the off-gas isolation valve. The leak rate of this system is neglegible and the leak rate at the turbine gland seals is reduced by feeding in nonradioactive steam. Therefore, the fission gases trapped in the turbine, the condensor and the hold up piping would be held until radioactive decay or meteorological conditions allowed dispersal without significant environmental effects.
In case b) (malfunction of the scram-system or the emergency cooling system) it must be made sure, that at least one of the independent isolation valves in the turbine steam line has been closed 3 to 4 sec after leak C has occurred at the latest.

4. CONCLUSIONS

a) The results have shown that without a control system the reactor has no ideal selfcontrolling behaviour; but the power follows the demand of the turbine if the negative density coefficient is not too large. Though a control system shall be necessary, a plant shall have a good dynamic behaviour (no oscillations) without a control system because in this case the control system has to act little (small and few motions of the control rods) and there will not arise a dangerous situation if the control system fails. The plant of the D1-design can be governed by a normal control system, as the transients are not very quick.

b) The behaviour of the plant during load changes of the turbine depends strongly on the density coefficient α_ρ. Oscillations may arise, if α_ρ is negative and too large; monotonous instability arises, if α_ρ is positive.

c) The dynamic behaviour of the plant could be improved very much, if the reheaters of the main coolant cycles could be removed from this position.

d) The Ruths-accumulator at the inlet plenum has an advantageous effect on the dynamics, especially in the case of leaks and failure of a blower. or steam generator.

e) Under certain conditions the reactor must not be scramed after the failure of one blower or steam generator. Working on is possible with reduced power.

f) No disturbance or accident in the coolant loop is dangerous, if the safety system operates. Even the worst coolant accident, the total break of a main pipe at the inlet plenum, induces a reactivity ramp of at most 4 $/sec, which can be governed by the safety system of the D1-design.

g) Compared to a fast sodium cooled reactor, where the loss-of-coolant can be excluded by design, in a steam cooled reactor the loss-of-coolant accident is of significance. This requires additionally a reliable emergency coolant system <u>and</u> isolating valves in case of the direct cycle.

h) The isolating valves should have the same degree of reliability as the inner containment. At a rupture of a turbine pipe they should be closed before any can melting has occurred. Dynamic studies show that for this action 3 - 4 sec are available, the signal generation and processing and valve shut off time included. Fortunately, these times are within the range of BWR-experience. Therefore, the isolation valve problems of steam cooled fast reactors do not differ from the problems already encountered in the BWR development.

i) The problems of emergency cooling and the flooding procedure require further experimental support.

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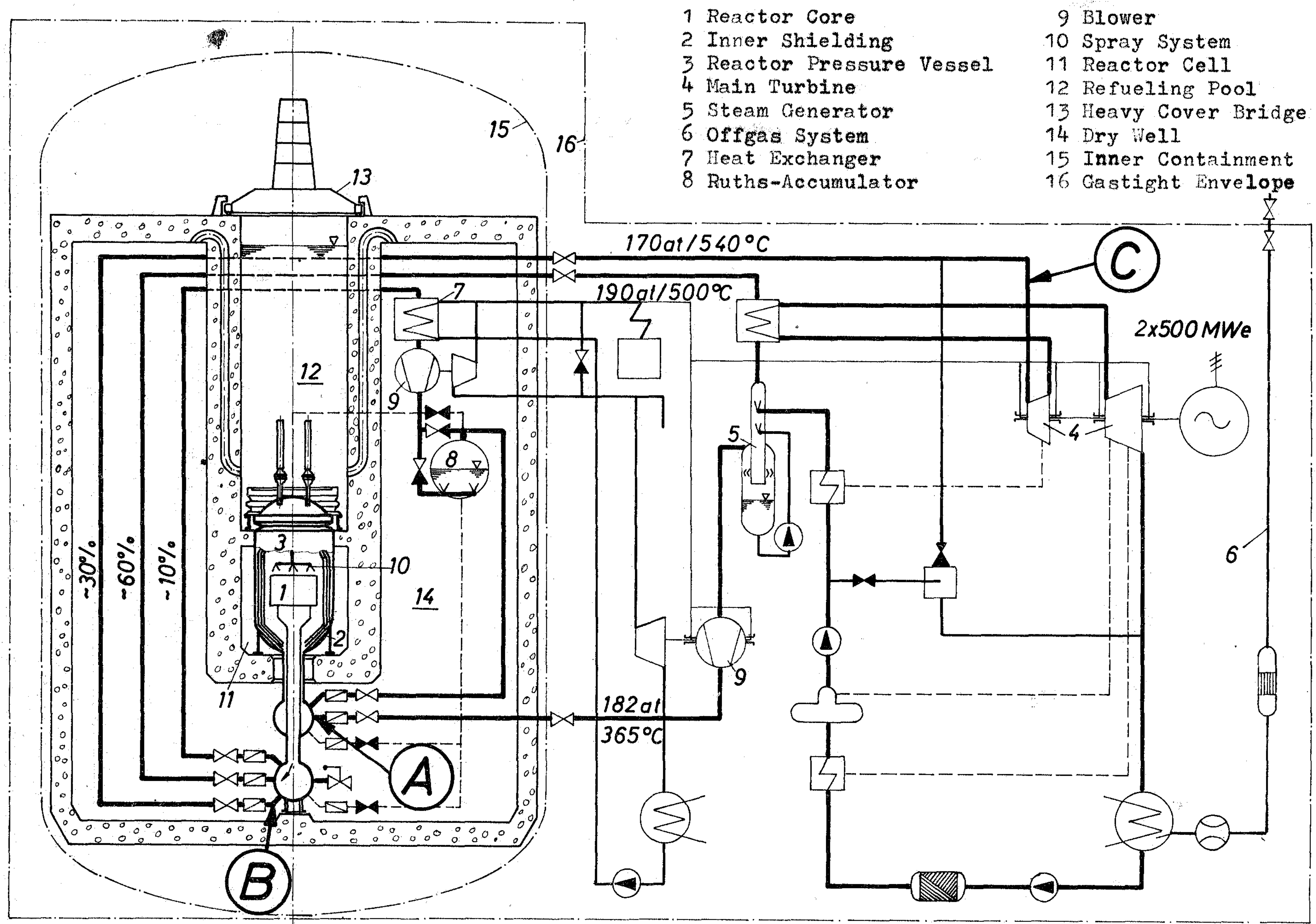


Fig. 1 Simplified Flow Diagram and Containment System

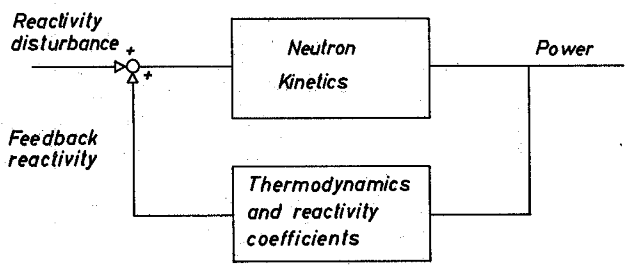


Fig. 2 Block diagram of the core

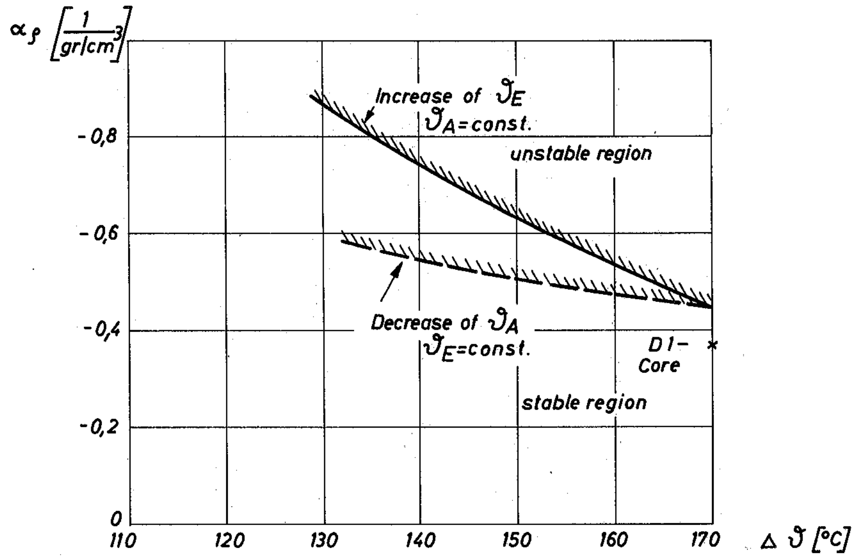


Fig. 4 Stability boundary as a function of the temperature rise $\Delta\vartheta$

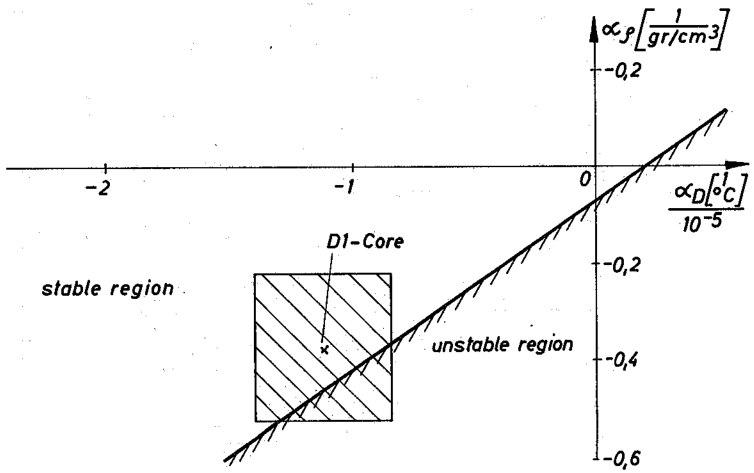
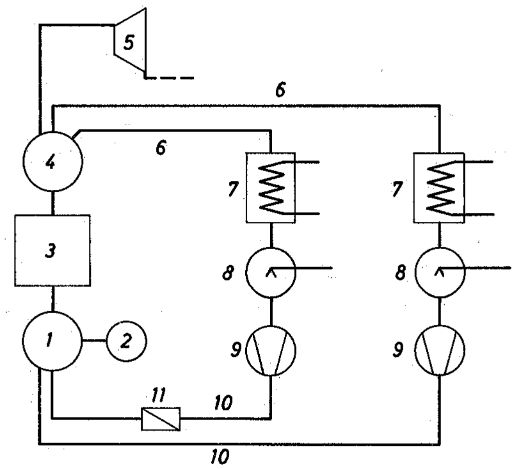


Fig. 3 Region of core stability



1. Inlet plenum
2. Ruths accumulator
3. Core
4. Outlet plenum
5. Turbine
6. Pipe
7. Reheater
8. Injector
9. Blower
10. Pipe
11. Check valve

Fig. 5 Flow chart of the analogue model

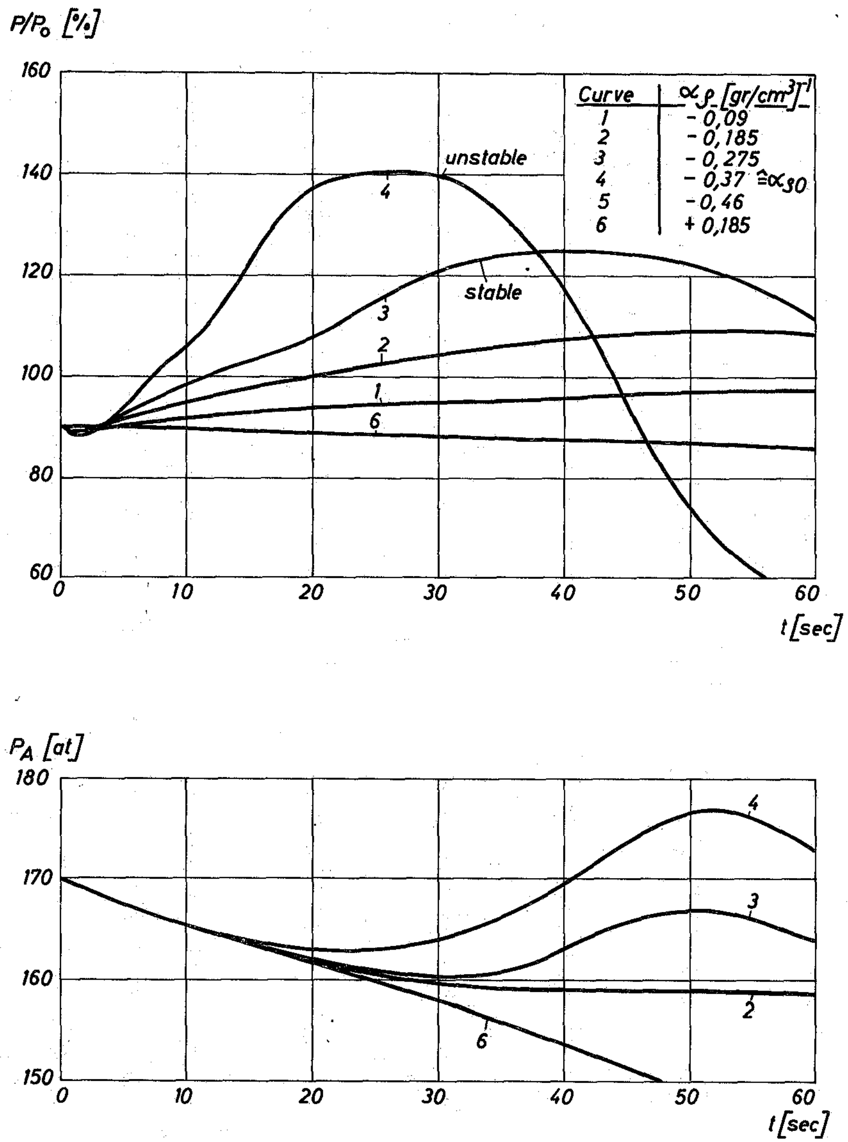


Fig. 6 Load change from 90 % to 100 %.
Power P/P_0 and outlet pressure p_A

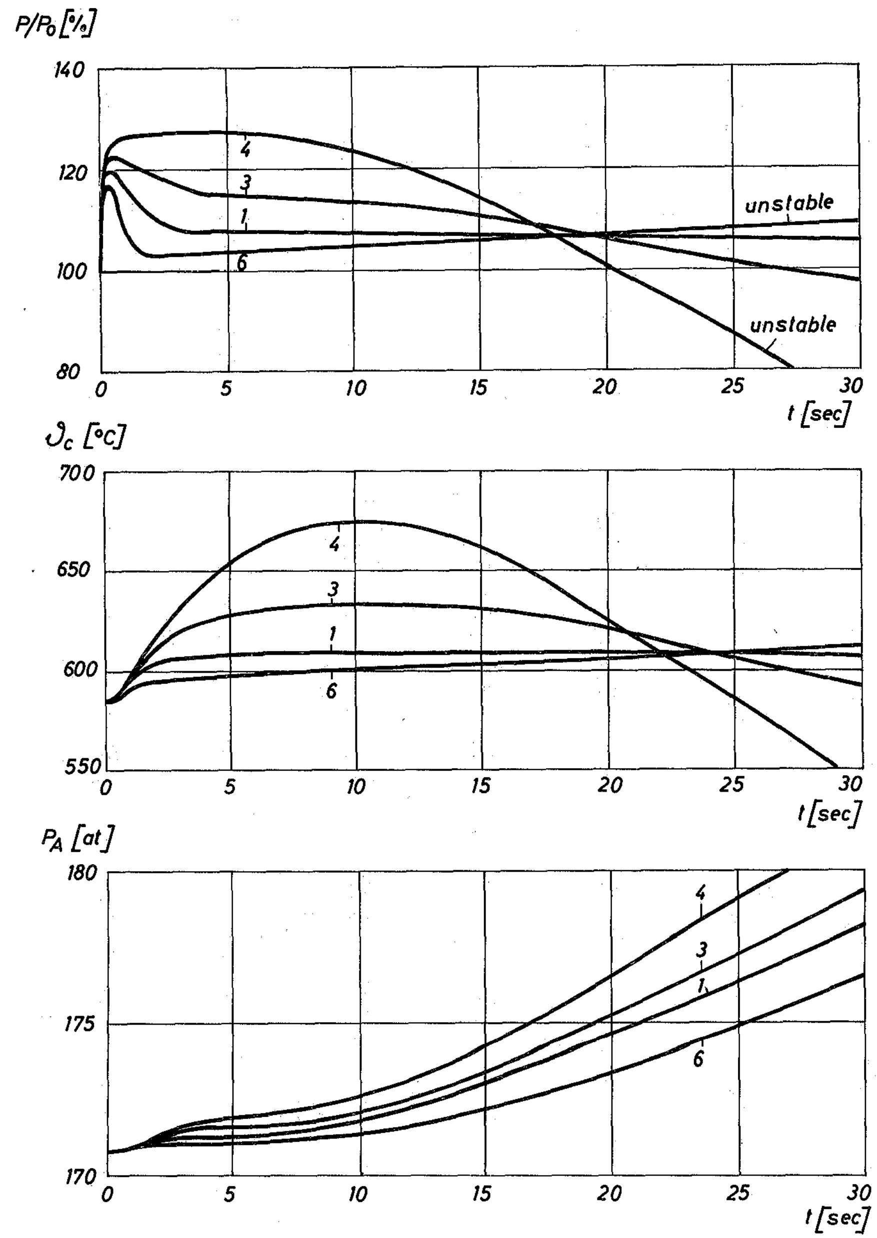


Fig. 7 Reaktivity step disturbance of + 0,2 \$.
Power, max. can temp. ϑ_c and outlet pressure.

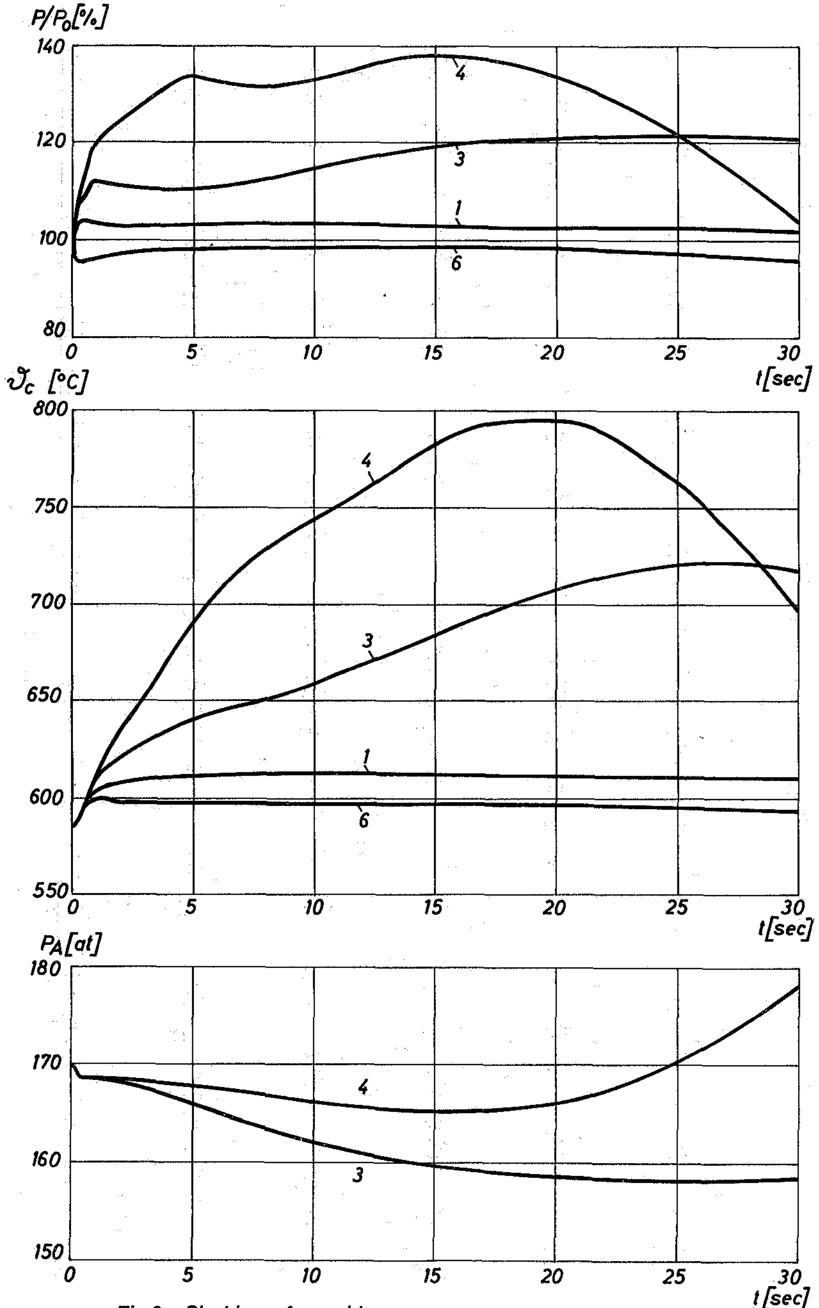


Fig.8 Blocking of one blower.
Power, max. can temp. and outlet pressure

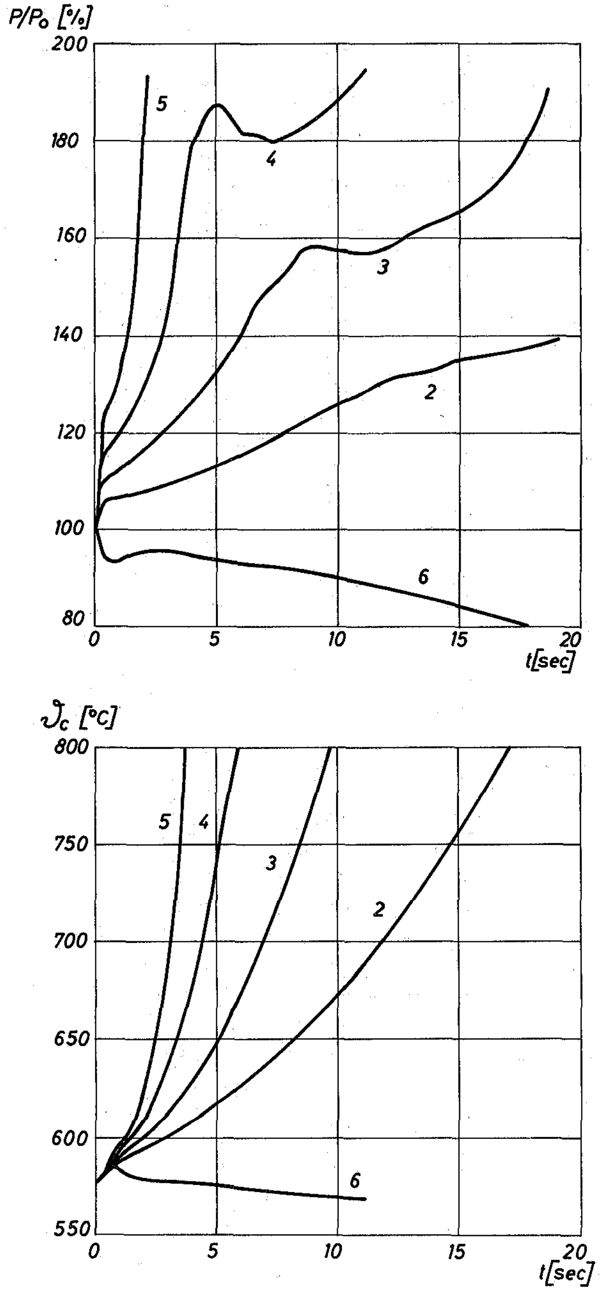


Fig. 9 Leak near the inlet plenum.
Power and max. can temperature

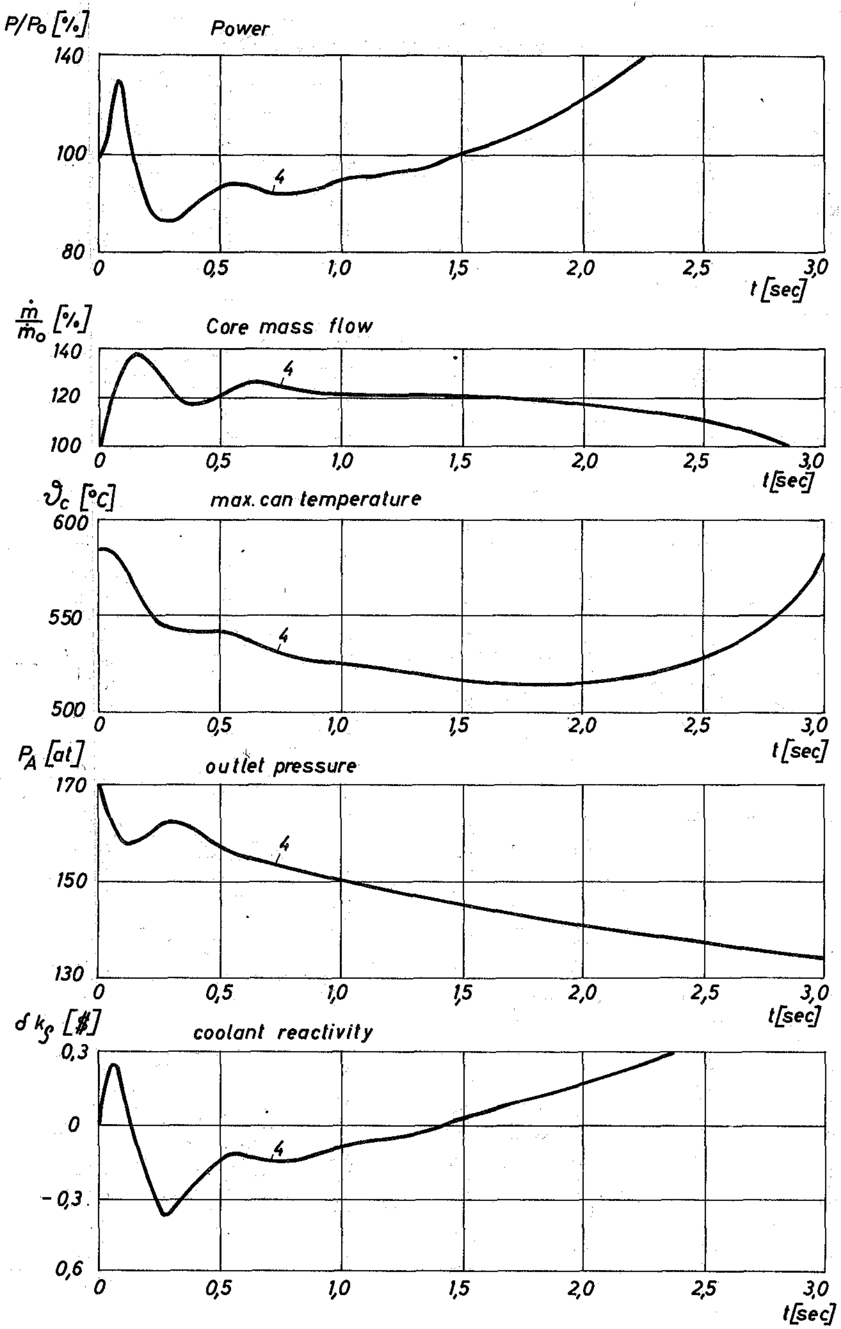


Fig. 10 Break of the main turbine pipe

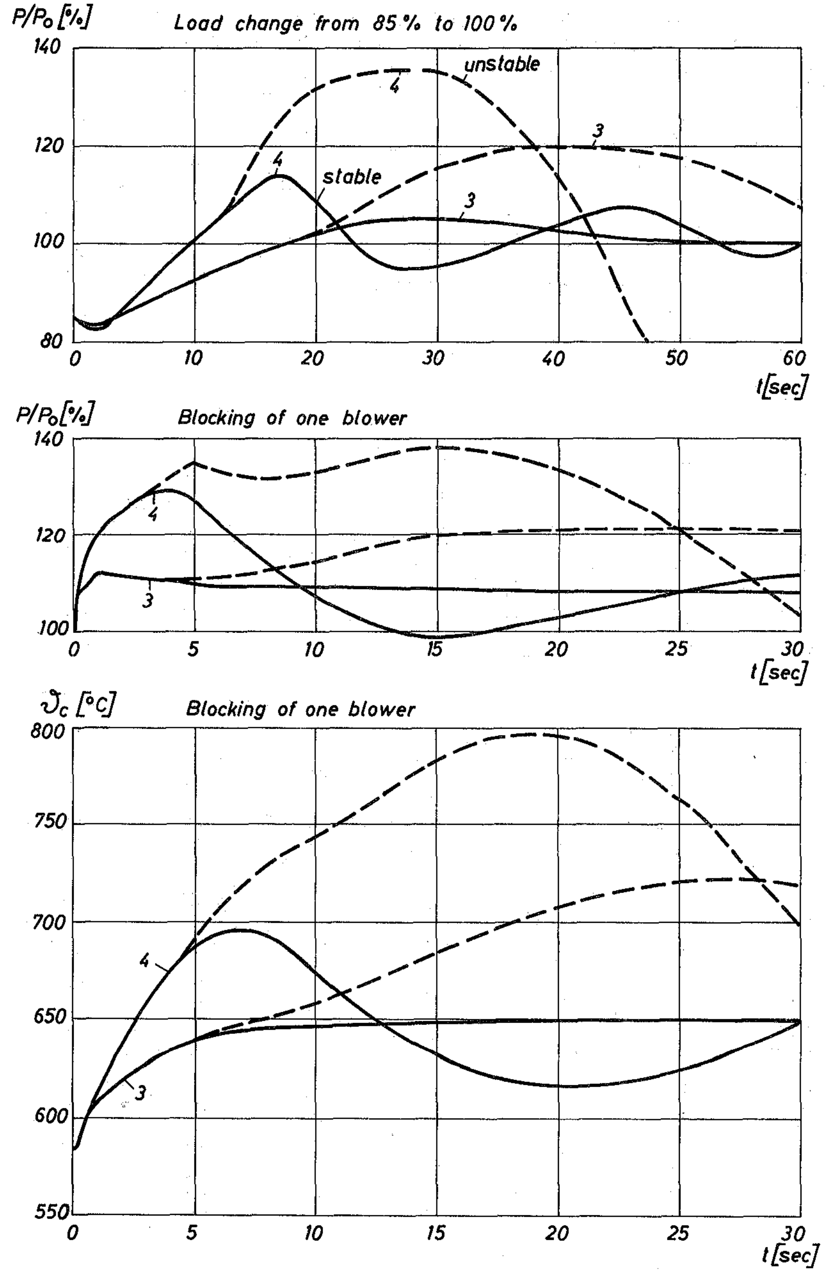


Fig. 11 Comparison: With (---) and without (—) reheater

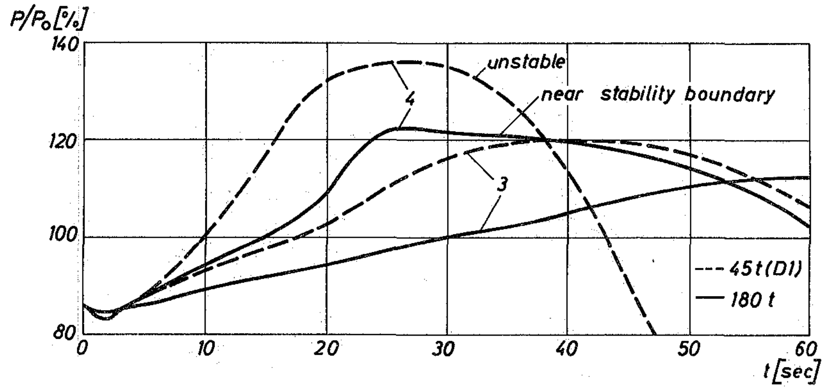


Fig. 12 Different water contents in the steam generators
Load change from 85 % to 100 %

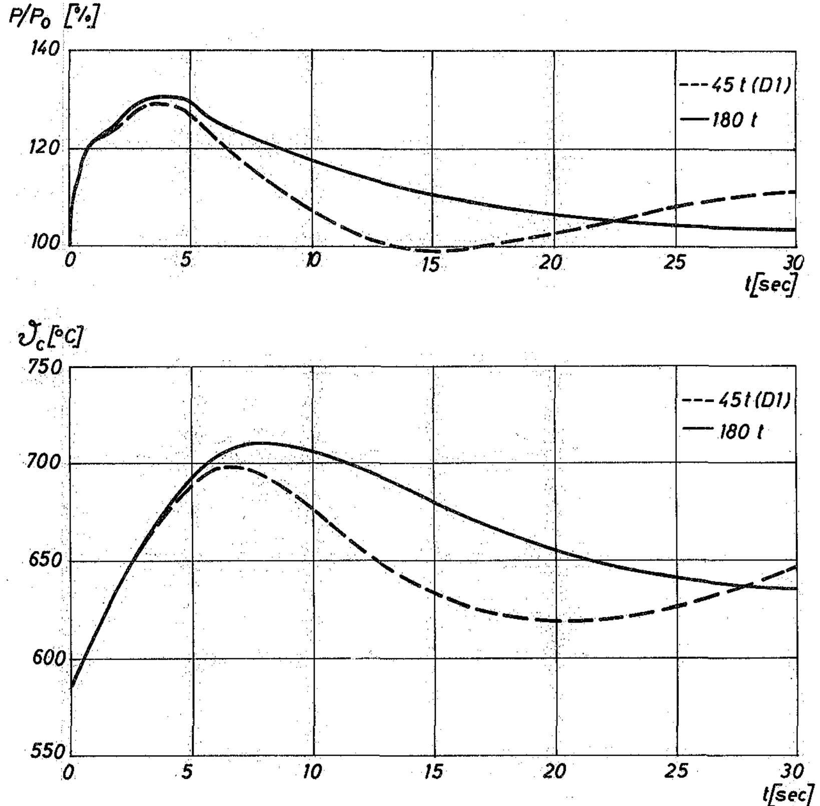


Fig. 13 Different water contents in the steam generators
Power, max. can temp. after the blocking of a blower

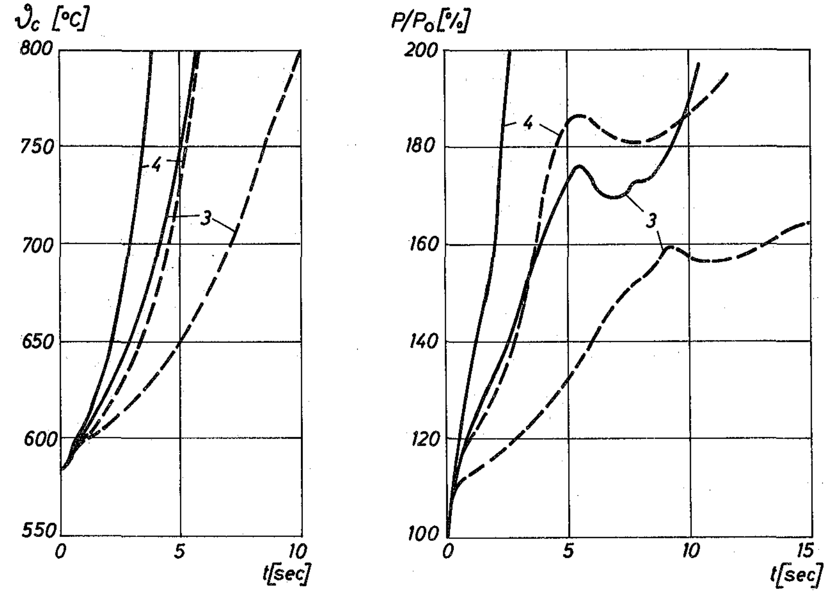


Fig. 14 Comparison: with (---) and without (—) Ruths accumulator
Leak near the inlet plenum

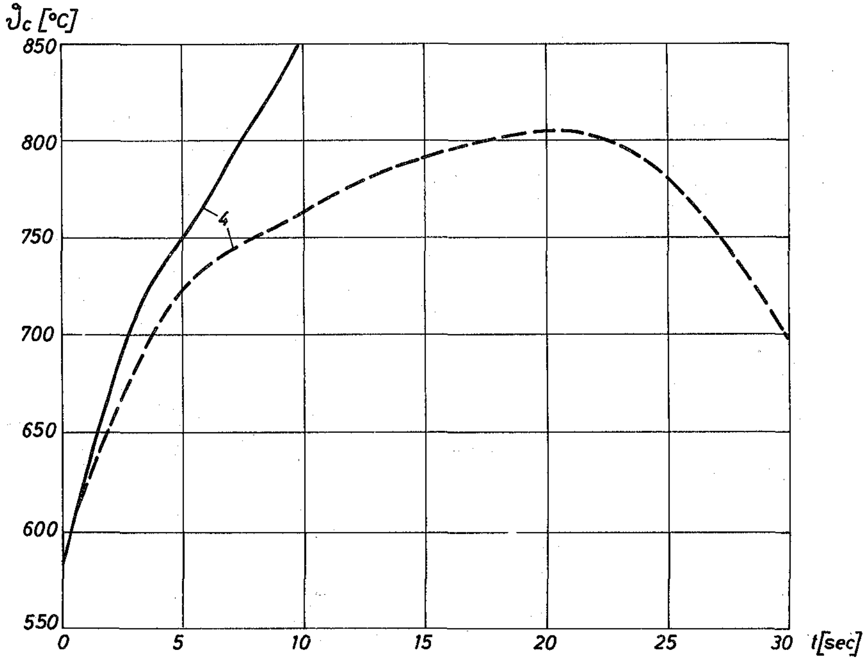


Fig. 15 Comparison: with (---) and without (—) Ruths accumulator
max. can temp. after the blocking of a blower

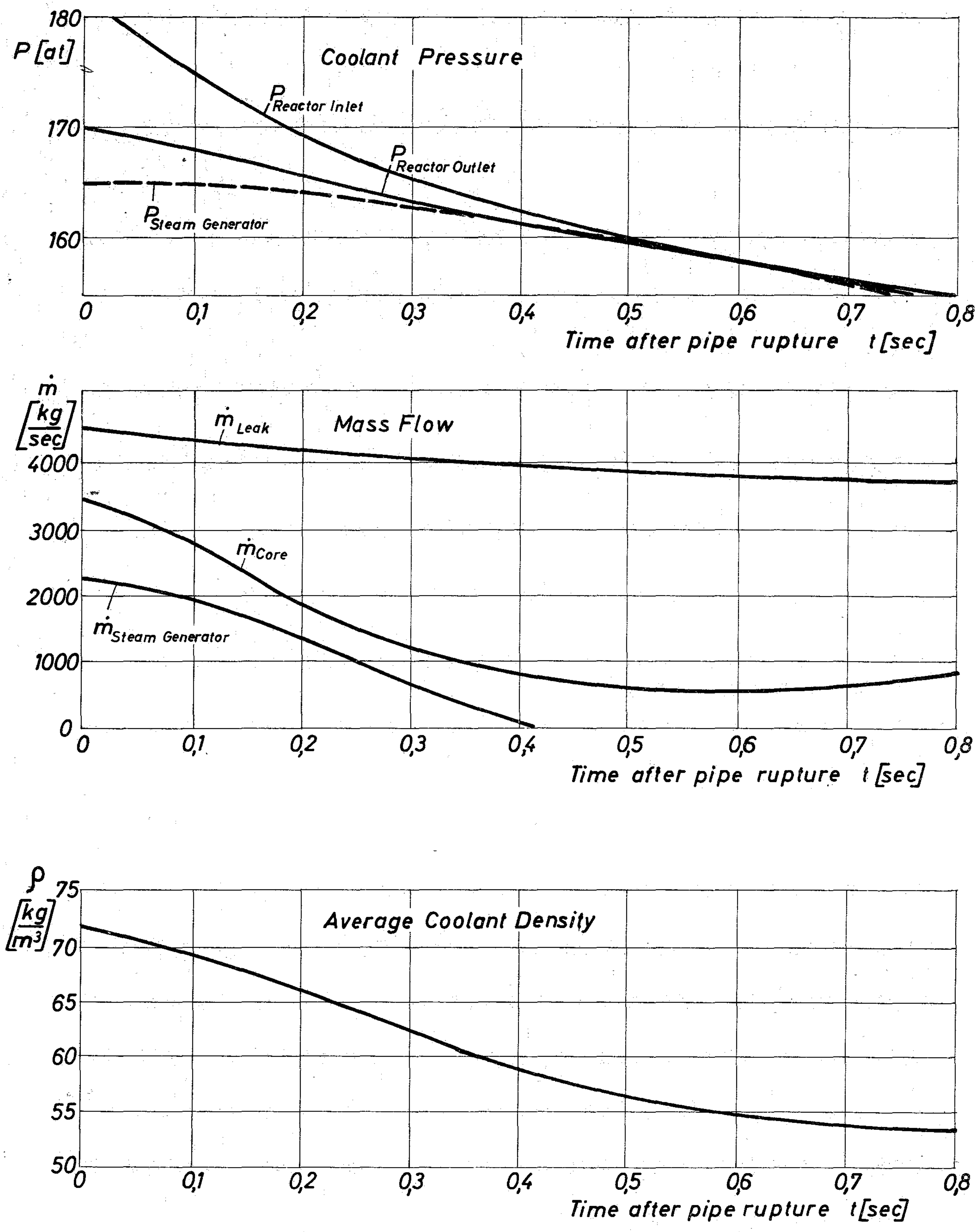


Fig. 16 Leak A, Steam inlet pipe

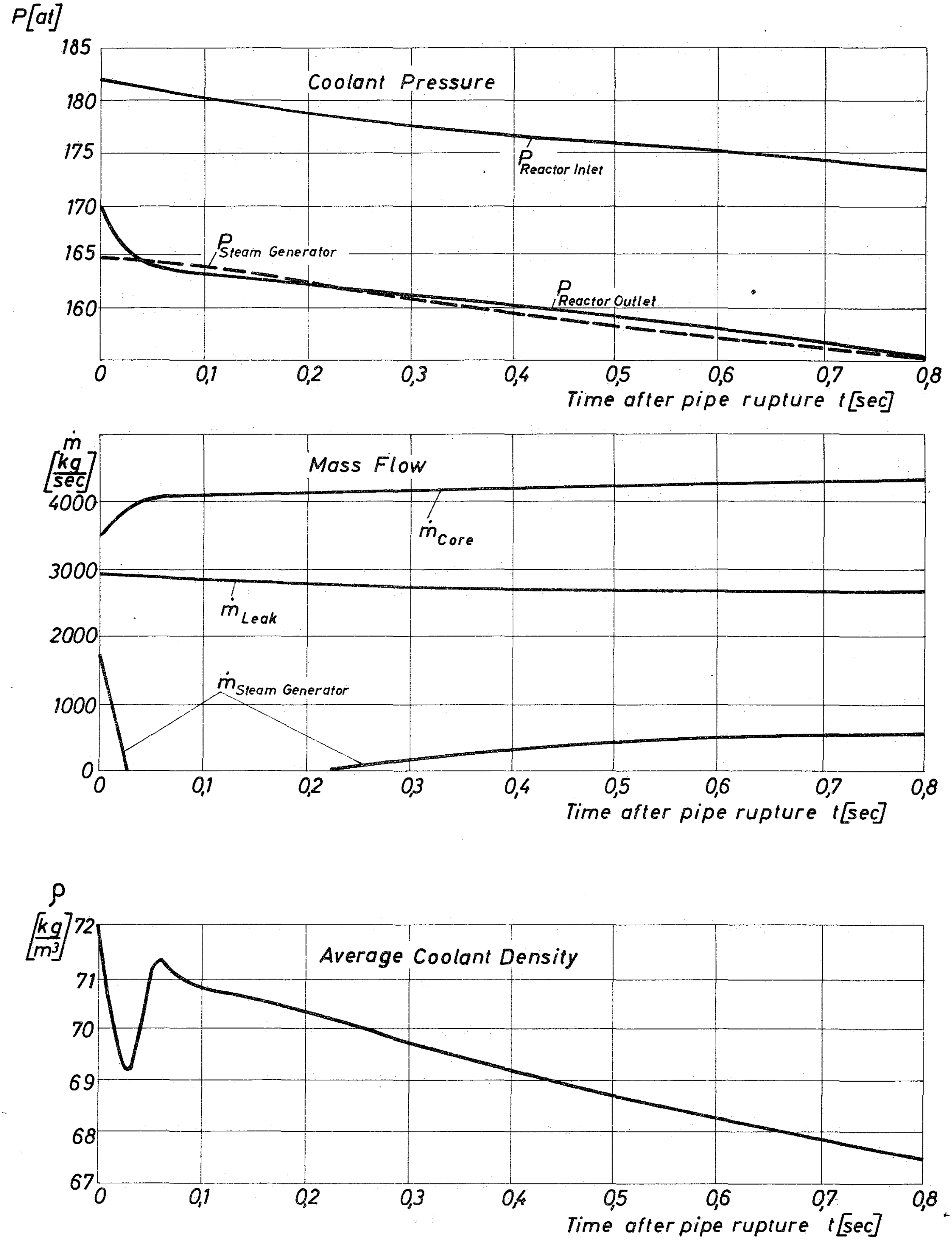
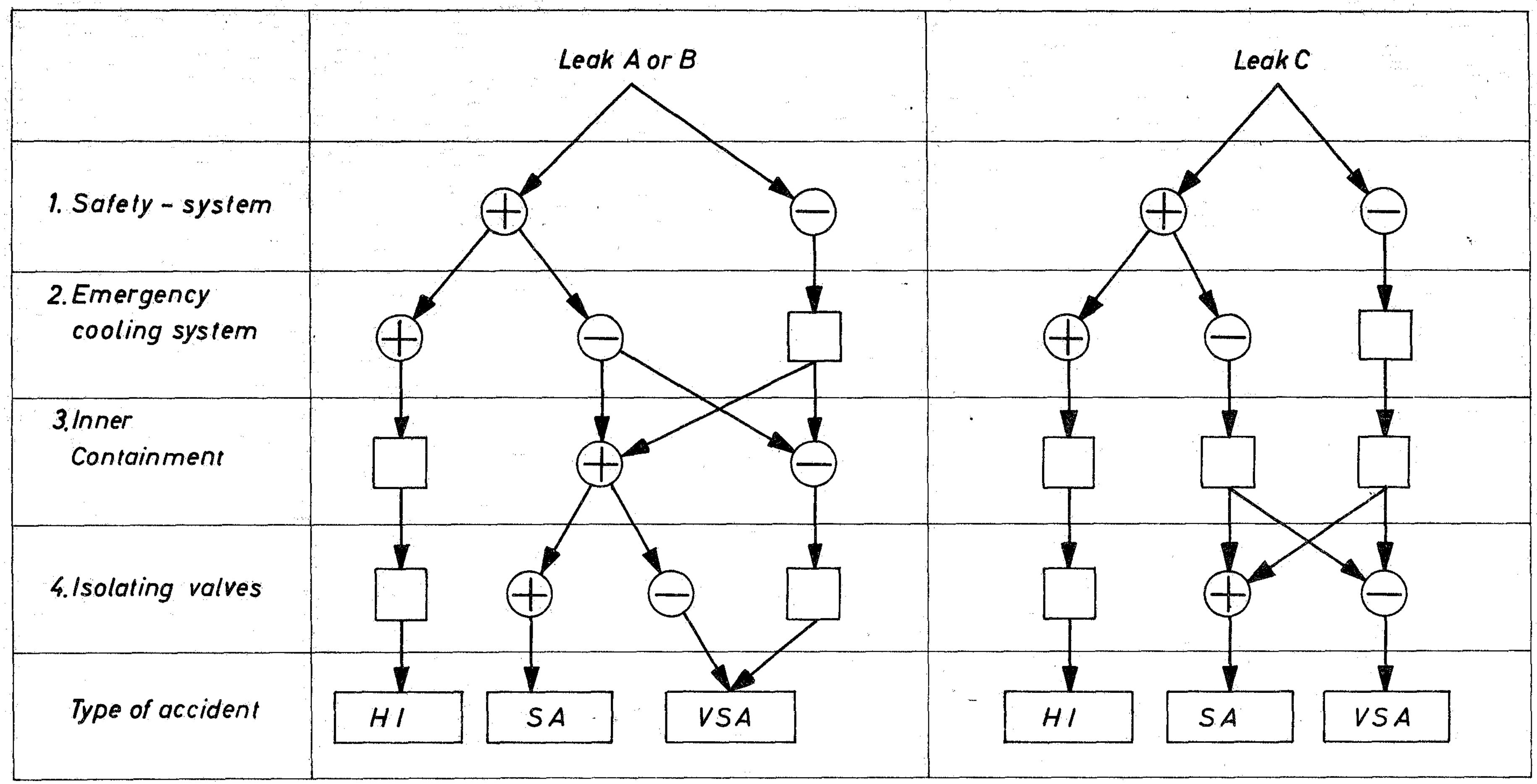


Fig.17 Leak B, Steam outlet pipe



function — HI harmless incident
malfunction — SA severe accident
without sense — VSA very severe accident

Fig. 18 Chain of events after a major leak