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THE SAFETY OF STEAM-COOLED FAST REACTORS AS INFLUENCED BY BY THE DESIGN AND ARRANGEMENT OF THEIR COMPONENTS+)

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ABSTRACT

This paper presents the reasons why the dynamic behaviour and the safety of a steam cooled fast reactor is particularly dependent on the design and the arrangement of the cooling cycle and its components. It points out that by a suitable design and arrangement a selfcontrolled system can be obtained and the consequences of certain major accidents greatly reduced. Independent of any specific reactor design the criteria to achieve this are defined.

Contents:

1. INTRODUCTION

Steam cooled fast reactors are much more influenced regarding their dynamic behaviour and hence their safety by the cycle feedback than reactors cooled by liquid metal, the reason being the strong dependence of reactivity on the coolant density and the large density change possible. Hence, safety considerations should be based on how the design of the cooling cycle could influence the density changes and should outline the resulting possibilities.

In the following sections mainly the problem of the dynamic stability of the steam cooled fast reactor and its safety against severe accidents will be dealt with.

It will be investigated which requirements should be posed on the cooling cycle, regarding the dynamic behaviour of the reactor and how and to what extent these requirements can be met. The authors are quite aware that the inherence of dynamic stability and safety against accidents are not necessarily an overriding principle but that they constitute highly desirable characteristics. Inherently unstable reactors, as for example the gas-cooled type, can be operated quite satisfactorily by means of the control system, and the safety risks, as the rod drop accident of the boiling water reactor can also be controlled by a safety system. Furthermore other aspects have to be taken in account in the reactor plant design, e.g. the optimization with respect to the costs and cost structure, the breeding ratio and the long-term fuel supply, the exclusive use of already available material and hardware and of proven system components.

However, these last aspects will not be considered as primary factors in the following investigations, because the specific aim of this study is to show the development tendencies to be adopted in cases where the inherent safety is of primary interest. It remains than to be seen whether the resulting requirements are incompatible with the other aspects to be taken in account, or whether they even support them.

This approach seems to be justified expecially in the case of the steam cooled fast reactor, because the strong mutual interaction between reactor and cooling cycle produces new specific problems and possibilities which differ from those connected with more familiar reactor types qualitatively as well as quantitatively.

2. REACTOR AND COOLING CYCLE AS A SELF-CONTROLLED SYSTEM

2.1 The Principle of Self-control

If a small reactivity step ($\Delta k \ll 1$ ¢) is introduced into the core of a reactor operating at constant power, a new equilibrium will be established at a higher power level, $Q + \Delta Q$ provided the core is stable. In a Na-cooled reactor this concludes the event, since there is only a slight feedback of the cooling cycle. If the power is to be reduced to the initial level, this must be done with the aid of control rods. In a steam cooled reactor, by contrast, the steam temperature increase $\Delta\vartheta_D$ and the resulting density decrease $\Delta\varrho$ of the steam first effects another reactivity increase, if the steam density coefficient $\frac{dk}{d\varrho}$ is negative. Fig. 1 shows the schematic diagram of the analogue model. Now, the higher temperature of the superheated steam results in a greater steam generation in the evaporator. The steam thus additionally generated, causes the pressure in the evaporator, p_k, and hence the steam density in the core to rise, which reduces the reactivity and the power of the reactor. By suitably designing the cycle, this mechanism will reduce the power to the initial level without any movement of control rods. Thus, reactor and cycle form a self-controlled system, similar to the mutual feedback of neutron flux and void fraction in a boiling water reactor. This characteristic behaviour of a steam cooled fast breeder indicates the decisive importance of arrangement and design of its components for the dynamic stability.

2.2 Demands on the Cycle

Fig. 2a shows the pressure increase after a reactivity increase without the feedback of the pressure on reactivity. The pressure remains constant at first. It starts to rise after the hotter steam reaches the evaporator. Finally, the pressure increase will become linear, when all pipe walls and other parts contacting the superheated steam have been warmed up to the higher temperature of the superheated steam which, consequently, remains constant. This curve is called the transfer function of the cut circuit. Fig. 1 shows the location of the cut.

The reactivity feedback of the rising pressure starts the sooner the shorter the dead times and delay times of the cycle. For simplification we shall treat the dead times T_t in the same way as the delay times T_u in the following considerations.

Moreover, the feedback will be the stronger the higher the slope K of the transfer function. However, for a given total delay time T_{ges} there is a boundary value K_{krit} above which the system performs rising oscillations. Below the boundary value the oscillations die out, and this the more strongly the larger the distance from K_{krit}. If one regards the delay time as a pure dead time (Fig. 2b), the stability condition (2) reads

$$K \cdot T_{ges} < \frac{\pi}{2} \qquad (1)$$

Applying this formula one remains on the safe side, since in the real system the feedback will start earlier and more softly, which makes this system more stable than a system with a pure dead time.

The stability condition shows K_{krit} to be proportional to $\frac{1}{T_{ges}}$, i.e. the smaller T_{ges} the larger the slope K may be and the faster will be the control of disturbances. The optimum value K_{opt} is about reached, when the time integral of the deviation becomes a minimum:

$$\int x_w^2 \, dt = \text{min.} \qquad (2)$$

x_w = deviation of power, pressure, temperature or other quantities from a stationary value to which they become reduced by self-control.

Factorization of K indicates by what measures this optimum value can be reached.

The slope K of the transfer function is calculated in the case of a pressure increase from the gain K_C of the core effective for the cycle and the cypacity C of the cycle:

$$K = \frac{K_c}{C} \tag{3}$$

$$K_C = \frac{\text{power increase}}{\text{pressure decrease}} \quad \left[\frac{MW}{at}\right]$$

$$C = \frac{\text{energy requirement}}{\text{pressure increase}} \quad \left[\frac{MWs}{at}\right]$$

K_C is the larger the higher the steam density coefficient and the smaller the Doppler coefficient. The larger K_C, the more the power will increase in the case of a pressure reduction. Hence, K_C should be small but positive. As a consequence, in order to reach the optimum value of K, the capacity C of the cycle must be changed.

This results in the following requirements for the reactor cycle:

1) The gain K_C of the core should be small.
2) The total delay time T_{ges} of the cycle should be as small as possible.
3) The capacity C of the cycle should have an optimum value dependent on the two other quantities.

The possibility of these requirements being met will now be discussed.

2.2.1 Gain of the Core

The thermal power of the D-1 core at max. burnup rises by 2 % per atmosphere of pressure reduction. 60 % of the power increase go into the cycle, 40 % into the main turbine [1,3].

The gain of the core effecting self-control thus is

$$K_C = \frac{0.02 \times 0.6 \times 2500 \text{ MW}}{1 \text{ at}} = 30 \frac{\text{MW}}{\text{at}}$$

The effect of reducing K_C is very favourable on the dynamic stability, as will be shown later.

In [3] various possibilities of reducing the gain of the core are discussed.

The economically most effective way is to increase the power density of the fuel. This requires the intensification of the heat transfer from the can to the steam. This measure increases the Doppler feedback and thus improves the inherent safety.

2.2.2 Delay Time

The heat transport from the fuel to the evaporator is delayed by the following processes:

1) Heat transfer from fuel to steam
2) Transport of steam from core to evaporator
3) Heating of pipes and structures contacting the superheated steam.

A numerical consideration shows how and to what extent it is possible to reduce the total delay time.

The individual delay times are discussed in the sequence of their order of magnitude and their significance:

a) Reheater. If a reheater heated with live steam is envisaged, it will cause most of the delay time.

This time is composed of the time required for filling the volume with steam:

$$T_{R1} = \frac{V_R}{\dot{V}} \tag{4}$$

V_R = live steam volume of reheater $[m^3]$

$\dot{V}$ = steam flow $\left[\frac{m^3}{s}\right]$

and the time required for heating up the reheater pipes:

$$T_{R2} = \frac{m_R \, C_{Fe}}{\dot{m} \, C_p} \tag{5}$$

m_R mass of reheater pipes [kg]

$\dot{m}$ steam mass flow $\left[\frac{kg}{s}\right]$

C_{Fe} specific heat of steel $\left[\frac{kJ}{kg \; ^oC}\right]$

C_p specific heat of steam at constant pressure $\left[\frac{kJ}{kg \; ^oC}\right]$

For the reheater of the D-1 design [1] the sum of these periods is about

$$T_R = T_{R1} + T_{R2} = 20 \text{ sec} \tag{6}$$

This relatively long delay time prevents a fast feedback and reduces the inherent stability of the system. Since, in addition, reheating leads to disadvantages in the design, and as the economic advantage to be gained by an increase in efficiency is small due to the low fuel cycle costs of a breeder reactor, it should be investigated in each individual case whether reheating is reasonable or not. Because of the long delay time of the reheater only a reactor without one will be considered in the following.

b) The time constant of the heat transfer from the fuel to the steam depends on the diameter and conductivity of the fuel and of the can. With a fuel rod diameter of 7 mm it is

$$T_B = 2 \text{ sec.}$$

This time constant cannot be greatly reduced, since it is impossible to change the material properties of the fuel arbitrarily and since a smaller rod diameter would increase the costs of the fuel cycle. Hence, T_B is a lower limit for the whole delay time and a measure of all the other delay times. The total delay time should not be much higher than T_B.

c) The delay time caused by structures to be heated is

$$T_{Fe} = \frac{m_{Fe}\ C_{Fe}}{\dot{m}\ C_p}$$

For m_{Fe} the mass of those structures should be inserted which follows a temperature change of the superheated steam. In the case of thick-walled components the temperature disturbance will penetrate only to a depth of some 10 mm within a few seconds, cf. Fig. 3. A realistic value for the mass of iron to be heated in this way is about 20 t in a 1,000 MWe reactor.

Thus,

$$T_{Fe} = 1 \text{ sec.}$$

This delay time could be reduced by thermal insulation

d) The time required for the outlet plenum to fill up a volume of $V_A = 30 \text{ m}^3$ is

$$T_A = \frac{V_A}{\dot{V}} = 0.5 \text{ sec} \tag{8}$$

For design and safety reasons the volume V_A and thus the time T_A cannot be reduced arbitrarily.

e) The delay time T_p in the superheated steam pipes is composed of the transport time of the steam

$$T_{p1} = \frac{L}{w}$$

L length of pipe $[m]$

w velocity of steam $\left[\frac{m}{s}\right]$

and the time required to heat a 1 cm thick layer (mass m_p) of the pipe wall

$$T_{p2} = \frac{m_p \, C_{Fe}}{\dot{m} \, C_p} \tag{9}$$

Fig. 4 shows the dependence on pipe length and diameter of the delay time. If the steam flow of a 1,000 MWe reactor is separated into four or six partial cycles and if steam velocities of 40 to 50 $\frac{m}{s}$ are selected, this results in pipe diameters of 0.4 to 0.5 m. This provides D, within close limits. The length of pipe should be below 10 m to prevent the delay time T_R from consuming too much of the total delay time.
At L = 10 m,

$$T_R = 0.7 \text{ sec.}$$

Pipes of about that length presuppose a compact design.

Under these cirumstances the total delay time is

$$T_{ges} = 4.2 \text{ sec.}$$

This is 2.1 times the value of T_B and can be reached only by compact design without reheater.

2.2.3 Capacity of the cycle

As will be shown below, this delay time of 4.2 sec is so short that now the capacity C also should be made as small as possible to approach the optimum value of K. A numerical treatment will show which are the largest capacities and in what direction the design of the components should be influenced therefore.

Realistic data of a compact cycle are used.

The capacity C $\left[\frac{MWs}{at}\right]$ is composed of four contributions:

a) capacity of the evaporators C_L

b) capacity of the steam volumes C_D

c) capacity of the pipes C_p

d) capacity of the blanket and the installations C_{BL} to the extent that their temperature is raised in the same way by a pressure increase as the saturation temperature of the steam.

a) Evaporator

1) Water Volume

The water being in thermodynamic equilibrium with the saturated steam increases its heat content in the case of a pressure increase by

$$c_w = \varrho' \frac{dh'}{dp} = 2{,}32 \left[\frac{MWs}{m^3\ at}\right] \qquad (10)$$

ϱ' density of water $\left[\frac{kg}{m^3}\right]$

h' saturation enthalpy of water $\left[\frac{kJ}{kg}\right]$

p saturation pressure of water [at]

With a volume of water $V_W = 40\ m^3$ the result is

$$C_{L1} = c_w V_w = 93 \frac{MWs}{at} \qquad (11)$$

2) Steam Volume

1 m^3 of steam volume has the specific capacity:

$$c_D = (h'' - h_{sp}) \frac{d\varrho}{dp} = 1{,}8 \frac{MWs}{m^3 \, at} \qquad (12)$$

h'' = saturation enthalpy of steam $\left[\frac{kJ}{kg}\right]$

h_{sp} feed water enthalpy $\left[\frac{kJ}{kg}\right]$

ϱ steam density $\left[\frac{kg}{m^3}\right]$

If the steam volume is $V_s = 150 \, m^3$ then the steam capacity is:

$$C_{L2} = c_D \, V_s = 270 \frac{MWs}{at} \qquad (13)$$

The total capacity of the evaporator ist then

$$C_L = C_{L1} + C_{L2} = 363 \frac{MWs}{at} \qquad (14)$$

b) Steam Chambers

A chamber volume $V_E = 50 \, m^3$ at the reactor inlet and $V_A = 30 \, m^3$ at the reactor outlet results in

$$C_D = (h'' - h_{sp}) \left(V_E \frac{d\varrho_E}{dp} + V_A \frac{d\varrho_A}{dp}\right) \qquad (15)$$

$$= 100 \frac{MWs}{at}$$

c) If there are four pipes each for saturated and superheated steam, in the dimensions L = 10 m (length) and D = 0.5 m (diameter), then the pipe capacity is

$$C_p = \frac{\pi}{4} (h'' - h_{sp}) \sum D^2 L \frac{d\varrho}{dp} = 20 \frac{MWs}{at} \tag{16}$$

d) Blanket and Structure. A blanket mass $m_{Bl} = 70$ tonnes and a structure mass in contact with saturated steam $m_{Fe} = 30$ tonnes has the capacity

$$C_{Bl} = (m_{Bl}\, c_{Bl} + m_{Fe}\, c_{Fe}) \frac{d\vartheta_s}{dp} = 18 \frac{MWs}{at} \tag{17}$$

ϑ_s [°C] saturation temperature

Accordingly, the total capacity is

$$C = C_L + C_D + C_p + C_{Bl} = 500 \frac{MWs}{at} \tag{18}$$

If this capacity must be further reduced, a reduction of the steam volume in the evaporators and steam chambers is of primary importance, a reduction of the water content in the evaporator of secondary importance.

2.2.4 Stability

The data thus calculated are inserted in (3) and (1):

$$K = 0.05 \text{ sec}^{-1}$$

$$K\, T_{ges} = 0.25$$

The distance from the thoeretical stability limit with respect to delay time and capacity in a design satisfying the above design criteria is characterized by the factor 6. This factor is relatively large; it would permit a further reduction of the capacity of the cooling cycle.

2.3 Results of Analog Computation

The theoretical predictions about the influence of the cycle design, on the dynamic behaviour were checked by means of an analog computer.
Fig. 5 shows the behaviour of a steam cooled fast breeder reactor following a 0.1 $ reactivity step.

Curve 1 applies to a reactor with reheating and small capacity of the cycle. The behaviour is unstable, characterised by oscillations (not to be seen on the figure). In curves 2 to 4 stability is achieved by various means. Curve 2 applies to a lower gain K_C of the core, which can be obtained, e.g., by a smaller steam density coefficient or a stronger Doppler feedback. The strong influence of this parameter is evident. Curve 3 applies to a reactor without reheating. The feedback of the cycle starts earlier. In curve 4 the capacity of the cycle has been increased by increasing the water content in the evaporator. The stability has been improved, but the steam temperature and thus also the can temperature reach a higher maximum.

Fig. 6 shows the transient response following the failure of one blower. The pressure at the reactor inlet decreases. Because of the decreased pressure head each of the undamaged blowers feeds more steam than before the disturbance so that the decrease in the total steam flow is partially set off. This effect is the more pronounced and the rise in steam temperature is the lesser the flatter the blower characteristic (Fig.7). In that respect a flat characteristic is favourable. Only near the stability limit a flat characteristic will entail some disadvantages, because the steam flow in this case will oscillate more strongly (Fig.8).

3. SAFETY

It has been shown so far that a design and arrangement of the cycle is possible which safeguards stable load operation, i.e. operation in which small reactivity changes are controlled automatically.

Further to the previous section, which dealt with small, finite reactivity disturbances, the severe accidents will be investigated now.

It will be examined by what design measures, with respect to the design and arrangement of the cooling cycle, the inherent safety of the steam cooled breeder reactor can be safeguarded as much as possible against those severe accidents, which are caused by a fast density change of the coolant, i.e. precisely those which are specific to this type of reactor.

Severe accidents are those in the course of which steep reactivity ramps are induced in the core which make the reactor become prompt-critical and heat the fuel to evaporation until they are terminated by partial disassembly of the core (Bethe-Tait excursion). For a given reactor design the consequences of these accidents are largely dependant upon the ramp rate (in $ /sec.).Hence, the potential of severe accidents is given and limited by the maximum possible reactivity ramps.

3.1 Severe Accidents

On the basis of the assumption that the steam cooled reactor will be flooded for loading and unloading as well as for the removal of decay heat over prolonged shutdown periods, there are three possible mechanisms that may result in the generation of steep and sufficiently long reactivity ramps:

1) Rapid reduction of the steam density in the core
2) Unflooding of the reactor from the near critical state
3) Core meltdown.

These three types of accident will now be investigated to ascertain under what boundary conditions the steepest ramps will arise.

Ad 1) - Density Reduction

The steam density in the core can be reduced by an increase in the temperature and by a reduction of pressure. In both cases a reduction of density up to very small values is possible. If one starts from the normal density of steam in the core, which is between $\rho = 0.1$ g/cm^3 (saturated steam, 150 at) and 0.07 g/cm^3 (average density at normal power), it is possible in the case of the D-1 core, see Fig. 9, to induce a reactivity in the core of $5 \div 6\ \% \mathrel{\hat{=}} 15 \div 18$ ¢ by density reduction.

If the density is reduced only by a reduction of pressure

(e.g. in case of a pipe rupture), relatively moderate reactivity ramps will result. A detailed investigation of the D-1 reference design, for instance, showed a maximum possible ramp by density reduction of 4 ¢ / sec [3].

Only if a simultaneous fast increase in temperature is superimposed upon the pressure decrease, a considerably higher ramp rate could result. This simultaneous action occurs for instance if there is a reversal of flow in the core, i.e. superheated steam flows back into the core due to a fast pressure reduction at the reactor inlet during load operation, e.g. due to the rupture of a saturated steam pipe. If no design measures have been taken to prevent reversal of the flow, ramps up to one order of magnitude higher than the one mentioned above, may result.

Ad 2) - Unflooding Accident

Corresponding to curve 2, Fig.9, a positive reactivity may be induced by flooding as well as by unflooding. But the reactivity peak, according to present results of calculations [4], will not be higher than 5 ¢ . Hence, flooding of the cold critical reactor will not be able to induce sufficient reactivity for a Bethe-Tait excursion. In a hot reactor the blowers prevent inadvertent flooding by their delivery head. Moreover, reactivity measurements taken before flooding can ensure that the core is sufficiently subcritical not to become supercritical again during flooding, because methods of reactivity measurement are known [5], the use of which will safety detect reactivities down to - 6 ¢.

Hence, we have to consider only the unflooding accident in this context. The flooded reactor is subcritical by at least $-\Delta k_F^+$, i.e. by - 25 ¢ . Since there is a high noise level (gamma activity, spontaneous fissions) in a highly burnt up fast reactor core, all methods of measuring reactivity known at present will fail with a subcriticality of that degree. Hence, an approach to the critical state beyond $-\Delta k_F^+$ cannot be indicated directly by the control channels.

There are two possibilities of a dangerous approach to the critical condition:

(a) Control rods are withdrawn from the flooded core, by operator's error, more than necessary to make the reactor critical after unflooding.

(b) The core is reloaded by too much enriched fuel; this is credible, since the reactivity of the core is generally increased by reloading. Normally, there is an uncertainty with respect to the reactivity condition precisely when the core is to be unflooded.

In both cases the reactivity curve $\frac{\Delta k}{k} = f\ (\varrho_K)$ is shifted upwards, see curve 3 in Fig. 9. A reactivity ramp is induced by rapid, inadvertent unflooding, e.g. due to a pipe rupture or faulty valve actuation. The slope depends on the rate of unflooding, the total reactivity induced on the degree of overloading or the number of rods withdrawn, respectively.

Since there is high internal pressure and only a little water needs to leak out to unflood the core completely, very steep unflooding ramps are possible with an adequately large cross section of outflow.

Disregarding any design measures or engineered safeguards, the potential of this accident is above 100 ₵ /sec.

The probability of this accident has to be assessed in any specific case. However, since the possibility of this accident is fundamentally inherent, either the probability must be made sufficiently small, for instance by a safety system or by engineered safeguards, or the maximum possible ramp rate must be sufficiently limited by design and arrangement of the cooling cycle. This latter possibility will be investigated here.

<u>Ad (3)</u> - Core Meltdown

If the fuel ~~cans~~ melt down in the center of the core because of insufficient cooling, the core will collapse. The conseqence is a reactivity increase by core compaction, enhanced by the displacement of the coolant from the molten zone. This reactivity ramp results in the so-called second excursion.

The meltdown due to decay heat under acceleration by gravity generally does not result in such a large release of energy as to damage the pressure vessel, as is shown in [3].

However, if the meltdown is introduced by a power excursion of the types 1 or 2, the ramp is increased by 2 effects:
(a) The core will melt down at high power, i.e. the times at which all rods begin to melt will be compressed into a very short span, and

(b) The molten components are accelerated by flow forces in addition to acceleration by gravity.

This process is called " forced meltdown " and may result in a considerably higher ramp rate than the above mentioned meltdown type.

A premise of the forced meltdown is, that the preceding ramps due to density reduction or unflooding accident have not resulted yet in a disassembly and thus subcriticality of the core. This condition is readily possible in both cases. The density can be reduced so slowly that the can will melt before the fuel evaporates, and the ramp in an unflooding accident can be terminated before the core is destroyed, see curve 3, Fig. 9.

This can be summarized in the statement that the three types of accident

1) Density reduction with flow reversal
2) Unflooding accident
3) Forced meltdown

may potentially result in very steep ramps, if not design measures are provided to reduce these.

3.2 Safety Requirements

For an investigation how and to what extent design and arrangement alone can limit the consequences of fast density changes in a reactor in general and those of the severe accidents mentioned above in particular, the safety requirements of the cooling cycle should first be formulated in a general way.

The following requirements have to be fulfilled:

1) The maximum possible mass flow out of a leak, such as in the case of a pipe rupture or wrongly actuated valves, must

be limited, since in all three types of accident the maximum leak rate of water or steam determines the ramp slope.

The mass flow in a leakage is determinded essentially by the cross section of the leak. Hence, sufficient restriction of the possible leak cross sections must be ensured.

2) Hot superheated steam entering the core, caused e.g. by flow reversal due to pipe ruptures and the like, should be prevented by design.

These two requirements limit all three types of severe accidents. The specific characteristics of the steam cooled breeder reactor however, necessitate two additional requirements for safe operation without major risks:

3) Operational unflooding must permit a control of reactivity, because three reasons render unflooding a potentially hazardous operation at higher burnups:

a) Unflooding induces a positive reactivity of the order of several percent;
b) Reactivity cannot be measured accurately before unflooding;
c) Loading and shuffling of the fuel elements in the flooded core will increase reactivity.

This makes it a prerequisite that all reactivity movements be controlled during unflooding and that unflooding be interrupted and reflooding be initiated immediately if too close an approach to the critical state is indicated. This operation should be controlled along similar principles as the scram, which is characterized by the failsafe principle, redundant control channels, etc.

4) Protection of the pressure vessel must be safeguarded not only in nuclear but also in non-nuclear accidents. The pressure vessel is the only component with vital functions which is not replaceable. Hence, it merits special precautions in all accidents. If the pressure vessel should suffer cracks under tension, this will result in a final shutdown of operations, whereas the meltdown of a few rods or even the disassembly of the interior of the core will lead only to an interruption in operation, under favourable circumstances. The pressure vessel is endangered especially by fast pressure decreases due to thermal stress. A pressure reduction by blowing off is unavoidable even in simple accidents; such as the rupture of a pipe between the pressure vessel and the shutoff devices. The target should be to avoid damage to the pressure vessel by tension cracks in these cases.

At the end of these stability and safety considerations and the requirements derived from them, it should be pointed out that these considerations were not based on a specific design and consequently do not constitute a safety analysis. These requirements still have to be harmonized with the other requirements to be made of a complete design.

This approach differs fundamentally from the conventional method employed, e.g. in the Na-1 study [6,7] or the D-1 study [1,3], where a design was submitted first on the basis of which the safety investigations were then carried out. This approach, too, can result in a design that meets usual safety requirements, as shown in [3]. In addition, a specific design allows the evaluation and correlation of other aspects, which were mentioned in the introduction, in addition to individual aspects of stability and safety.

Contrary to those approaches, the emphasis on inherent stability and the safety aspects mentioned above will be followed consequently below and will give rise to the question of which design tendencies are particularly satisfactory.

4. IMPLEMENTATION OF SAFETY AND STABILITY REQUIREMENTS

4.1 Venturi Tubes

The first safety requirement, namely the far-reaching limitation of potential leak cross sections in the cooling cycle, can be met primarily by the following measures:

1) Separation of steam flows into a multitude of pipes of smaller diameters.
2) Arrangement of Venturi tubes for flow limitation in each major steam line.

These two measures are conventional and well known from the boiling water reactor.A separation of the inner cooling cycle into several (between 3 and 6) parallel loops is required already for operational safety reasons. As shown in the D-1 study [3], these measures in themselves will already contribute a sufficient amount of safety. However, if it is intended to investigate to what extent the limitation of the leak cross section may be achieved in an extreme case, the result is that the breakdown of the steam flow into very many pipes with narrow venturi tubes will result in high capital costs and large flow losses. Hence, a possibility of avoiding these disadvantages will be discussed in the following.

4.2 Integrated Design

As pointed out above, in a Loeffler cooling cycle about 60 % of the superheated steam flow leaving the reactor is returned to the evaporator by the inner cooling cycle to evaporate the feed water. Only about 40 % of the superheated steam is bled off the inner cycle and used to drive the turbines (Blower driving turbine, power turbine, and auxiliary turbine). The whole inner cooling cycle is integrated in a surrounding pressure vessel. In such a design the number of high pressure superheated steam lines penetrating the pressure vessel is reduced to less than half of the number encountered in a disintegrated design. There are no venturi tubes at all in the superheated steam lines of the inner cooling cycle since these lines are loaded no longer with pressure so that no rupture is to be anticipated. Moreover, integration avoids the possibility of flow reversal in the reactor core, because now only a pipe from the core to the turbine may rupture.

4.3 Series Turbine

The consequences of the rupture of a superheated steam line can be reduced by using series turbines to drive the blowers. In this arrangement, as shown in Fig. 10, the whole superheated steam flow to the power turbine is first expanded in the blower driving turbines, where the major part of the head required to produce the driving power is used up in the nozzle ring. This causes high velocities in the smallest cross section of the nozzle ring, which are approaching sonic velocity depending upon the hydraulic design adopted. Since all of the turbine steam flows first through the driving turbines it is ensured that in the case of a rupture of the superheated steam line beyond these series turbines the whole mass flow coming out is limited by the velocity of sound in the nozzle ring. This is a decisive advantage, because in this way it is possible to reduce the increase in mass flow in the case of a pipe rupture to a minimum and the throttling this requires does not entail any loss in normal operation but actually serves for power generation in the driving turbines of the blowers.

The integrated design with series turbines thus is an efficient and practical protection against the consequences of the rupture of a superheated steam line. This can be very significant in the case of this line rupturing outside the containment, since the reduction of the reactivity ramps strongly diminishes the hazard of bursting cans and the activity release these will entail.

4.4 Water Reservoir

The possible rate of density changes in the case of accidents can be strongly reduced also by the arrangement of water reservoirs in the inner cooling cycle. These water reservoirs are the Loeffler evaporator and a water jacket around the core. Since the water content of the Loeffler cycle should not exceed a certain limit for dynamics reasons, it will be the water jacket around the core which has to slow down the reduction rate of the pressure level at the core inlet in the case of disturbances and accidents. It is important that the steam released by the reservoir enters the inlet plenum without any pressure loss. In addition, this water reservoir safeguards emergency cooling after accidents for a limited period of time.

The three measures

1) Integration of inner cooling cycle
2) Series turbine as blower drive
3) Water reservoir at core inlet,

are sufficient to keep even the ramps of an unflooding accident within acceptable limits, e.g. below 30 \$/sec. This ramp does not lead to a destruction of the pressure vessel in the case of a Bethe-Tait excursion, i.e. if the shutdown system should fail, as the D-1 analysis shows. The reactivity ramp induced by density reduction in the case of the rupture of a superheated steam line can be limited to about 0.1 ₰/sec.

4.5 Flooding Tank

The requirement of unflooding control can be satisfied, among other means, by having the same pressure in the reactor and in the flooding tank during unflooding and by making an unflooding pump lift the flooding water merely to the higher geodetic level of the flooding tank ('unflooding at equal pressures') so that it will flow back into the reactor automatically in case of a premature shutoff of the flooding pump. There should be no valves in the unflooding pipe system which would effect unflooding by overpressure in the case of maloperation, for this does not permit sufficient control of the unflooding procedure.

4.6 Pressure Vessel

The pressure vessel required for a 1,000 MWe reactor of the integrated design can no longer be made a steel vessel. Hence, the use of prestressed concrete is a prerequisite for the integrated design and thus an element to enhance the safety of large steam cooled reactors. If there is no reheater for reasons of dynamics and stability of the reactor cycle and, also, because of its technological complications, the dimensions of the concrete pressure vessel can be kept so small that the high pressure level of steam cooled reactors can be safely managed.

In addition, the prestressed concrete pressure vessel fulfills without any restrictions, the requirement of reliability of the pressure vessel in the case of pipe ruptures. The prestressed concrete is maintained at a temperature below 100° C at the inside by water cooling. In this way, the actual pressure-resistant concrete wall is kept free of temperature changes and the resulting thermal stresses as may be caused by rapid pressure reductions. This characteristic is a considerable advantage over a steel pressure vessel. Another advantage lies in the fact that if the burst pressure is exceeded and this causes a leakage, the inner pressure can be reduced only to the extent that the leak is closed again by the tension of the cables.

4.7 Evaporators

In order to achieve favourable dynamic behaviour, the quantity of water stored in the evaporator should be kept small. This requirement could be satisfied by using spray evaporators, because here only the water required to desuperheat the hot steam must be injected into the superheated steam flow in a finely dispersed spray, and theoretically the quantity of water stored can thus be made almost arbitrarily small.

However, along with the requirement mentioned above it must be prevented that superheated steam penetrates through the evaporator. With the spray evaporator this danger is inherent owing to the potential failure of the feed water supply, and must be prevented by engineered safeguards. Since in this connection mainly the problem of the inherent accident safety is of interest, it will be considered now whether the Loeffler evaporator inherently fulfills the two requirements, i.e.

1) Sufficiently small volume of water
2) Safe prevention of superheated steam penetration.

In the Loeffler evaporator superheated steam is blown into a stagnant volume of water. The steam is dispersed into single steam bubbles by means of nozzles, which results in a good heat exchange between the hot steam and the water to be evaporated. The geometrical shape, size, and the frequency of the steam bubbles generated decisively control the heat exchange. These factors are influenced mainly by the velocity in the nozzle. An increase in the velocity initially results in an increase in frequency at constant size of the bubbles, which makes for an improved heat exchange. However, from a certain optimum steam velocity a further increase will result in enlarged steam bubbles at constant frequency, which deteriorates the heat exchange. Consequently the optimum steam velocity provides for the smallest possible volume of water in the evaporator. Experiments carried out so far have shown that the optimum evaporation velocity referred to the free water surface is around 0.2 m/sec.

At this velocity the water level above the nozzles required to desuperheat live steam of 520° C completely is about 25 cm. Thus the necessary water content in the evaporators for a 1,000 MWe reactor is about 40 m^3. This volume is small enough to meet the requirements for dynamic stability. Another result of the experiments carried out is that most of the superheat of the steam bubbles is transferred to the water during the bubble formation and shortly after the separation from the nozzle. So the reduction of the water level below the level required for complete desuperheating will result only in a slight residual superheat of the steam generated. For instance a reduction of the water level by some 50 % results in a residual superheat of about 5°C. Hence, the penetration of superheated steam is effectively prevented in the Loeffler evaporator.

4.8 Steam Blower

For reasons of dynamic behaviour the blower should have a flat characteristic and, in addition, it should be suited for an integrated design, i.e. together with its drive it must be compact. Since a radial blower has a flatter charcteristic than a comparable axial blower, its use in the cooling cycle is particularly advantageous. Also under the aspect of hydraulics a radial impeller will show a better behaviour than an axial one, as has been shown in [8].

The steam blowers are driven by steam turbines which are designed as series turbines to the power turbine for reasons of inherent accident safety. This design results in a small turbine head and in relatively small steam volumes at the inlet and outlet sides. Compared with other possible ways of arranging the blower turbine, in the case of series turbines the absolutely smallest volumetric steam flow has to be taken out of the reactor. This also means the smallest possible diameter of the penetrations of the pressure vessel. The blower diving turbine can be designed as a single-stage impeller. The dimensions of such an impeller are about the same as those of the blower so that the whole blower-turbine unit can be designed as an enclosed system with water lubricated bearings. The use of four blower units for a 1,000 MWe reactor results in impeller diameters of some 400 mm for double suction radial impellers and dimensions of the casing of some 1.5 m diameter and 3.0 m length. Such a machine is expecially suited for an integrated arrangement of the reactor because of its small dimensions and its simple design. A smaller prototype of these blowers has been successfully tested for more than one year.

4.9 Fuel Elements

It has been shown that the improvement of stability of the reactor core is achieved in a particularly efficient way by increasing the average fuel temperature through raising the power density in the fuel. However, because of the limited permissible canning temperature a higher power density requires an improvement of heat transfer.

An increase in heat transfer was experimentally proven by the following measures:

1) By the application of turbulence promoters as surface roughness.
2) By spiral fins integral with the can.

These provisions result in an increased heat transfer by increasing the turbulence of the flow medium, by enlarging the heated surface, and, especially, by a considerable improved cross mixing of the coolant flow in the cooling channels. It was shown that the good cross mixing achieved by the spiral fins substantially reduces the hot channel factor so that for steam temperatures of 540°C maximum rod powers of some 600 W/cm at a maximum can temperature around 660° C can be realized and the pressure loss increases only slightly by these measures.

Thus, suitable design of the fuel elements can markedly increase the heat transfer and the power density and in this way improve the inherent stability of the reactor.

5. CONCLUSION

The strong dependence on the coolant density of reactivity of the steam cooled breeder raises a number of problems specific to this type of reactor referring, on the one hand, to the dynamic behaviour and inherent stability, on the other hand to the safety against accidents. An isolated treatment of these problems and of the question in what way they can be solved as far as possible by design and arrangement of the components, excluding the safety system or engineered safeguards, results in the following requirements which reactor and cooling cycle have to satisfy:

1) Total delay times of the dynamic feedback as short as possible,
2) Minimum possible integrated deviation in case of reactivity disturbances or changes of load
3) Stabilization of pressure at reactor inlet
4) Prevention of superheated steam from entering the core
5) Leak cross sections in the inner cooling cycle as small as possible
6) The pressure vessel must be insensitive to reductions of the inner pressure
7) Control of reactivity movements during unflooding.

To fulfill these requirements the following design criteria can be imposed on the cycle and its components:

1) Simple, compact cooling cycle without reheater
2) Integration of reactor with inner cooling cycle in one common pressure vessel, preferably a concrete pressure vessel
3) Increase of the power density in the fuel by the application of integral spiral fins and turbulence promoters.
4) Use of high performance Loeffler evaporators with small water and steam volumes
5) Sufficiently large water reservoir near boiling temperature at core inlet
6) Arrangement of blower driving turbines as series turbines with high Mach number in the nozzle ring
7) Unflooding into an overhead flooding tank at equal pressures

These design criteria should be regarded as trends which have to be harmonized with the other design aspects in the case of a specific project. However, it is apparent that far-reaching implementation of the requirements outlined above is possible by a suitable design of the cooling cycle and its components.

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Figures:

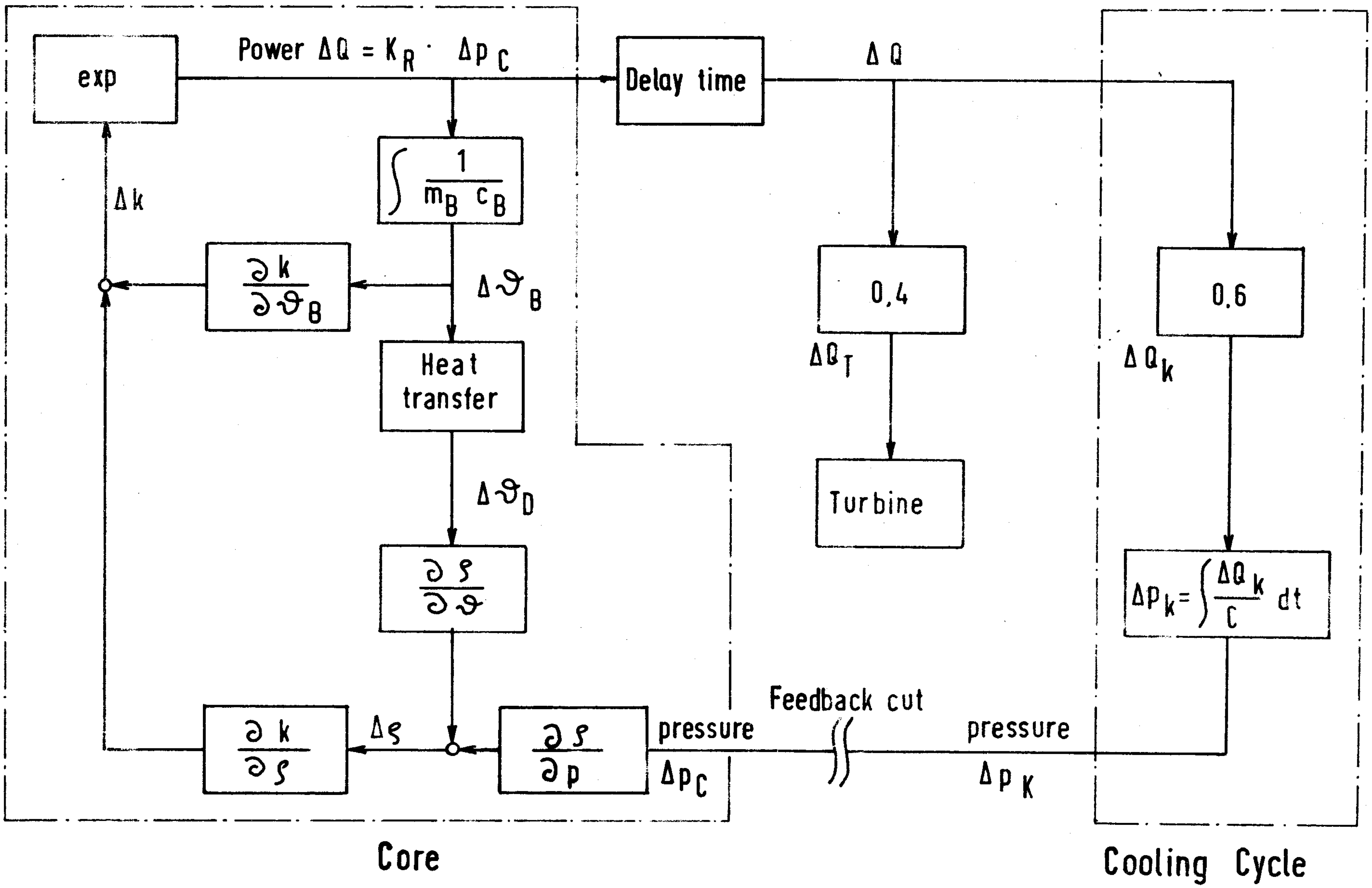
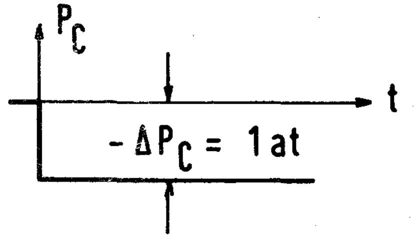


Fig. 1 Feedback of core and cooling cycle



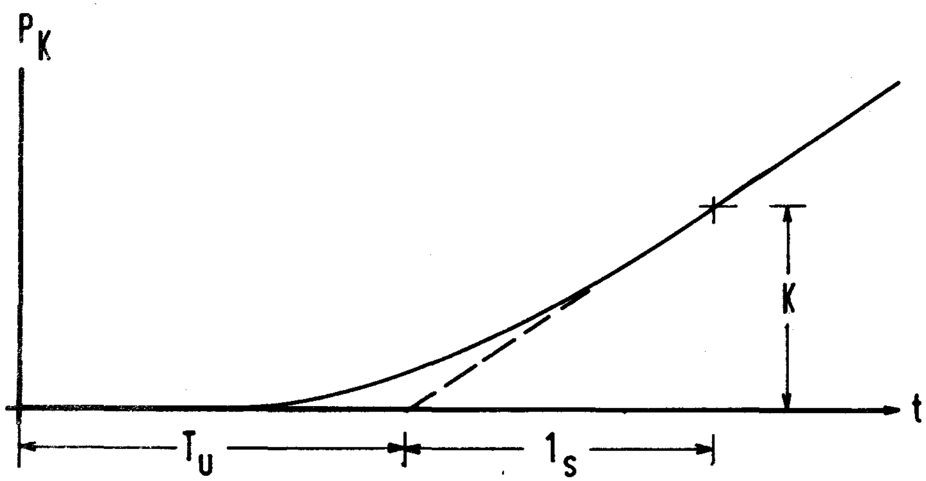


Fig. 2a Real system

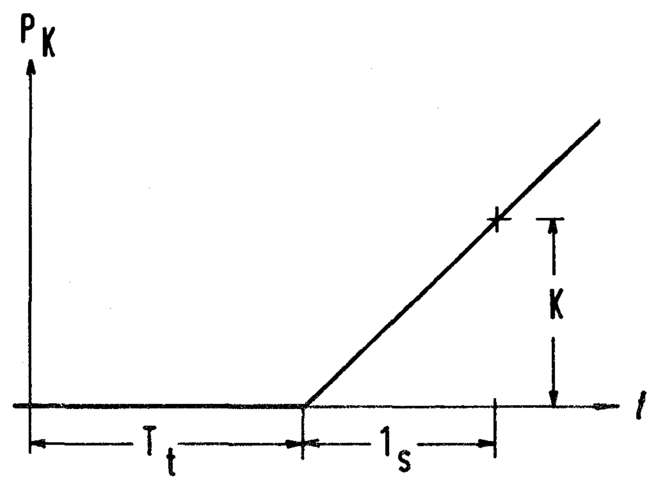


Fig. 2b Idealised system

Fig. 2 Transient response of the cut circuit

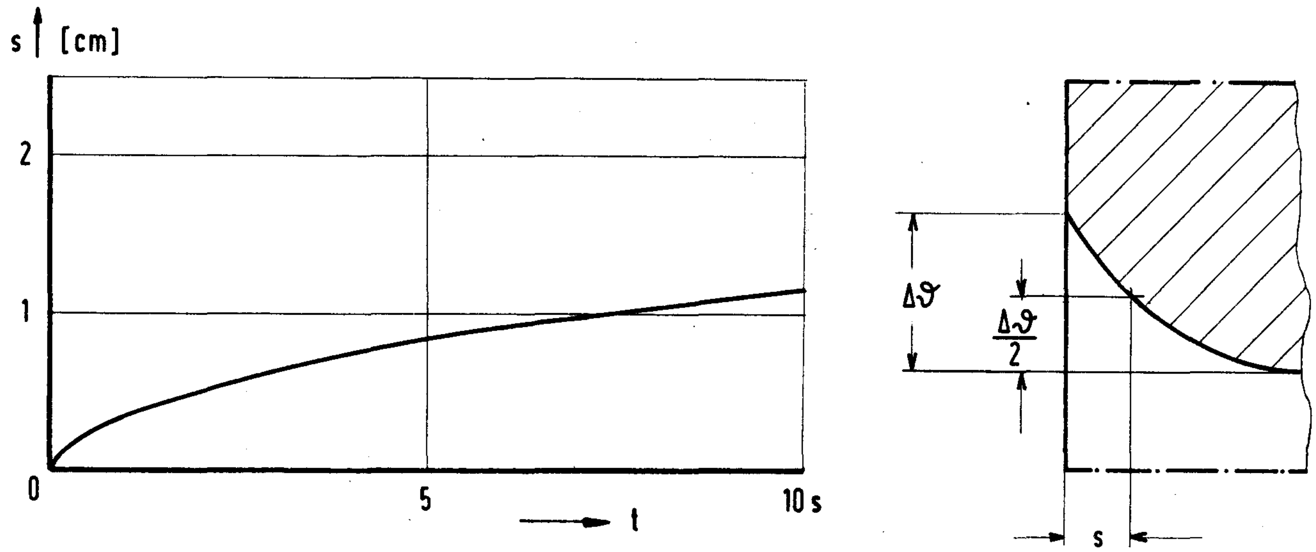


Fig. 3 Penetration depth of a temperature disturbation in steel

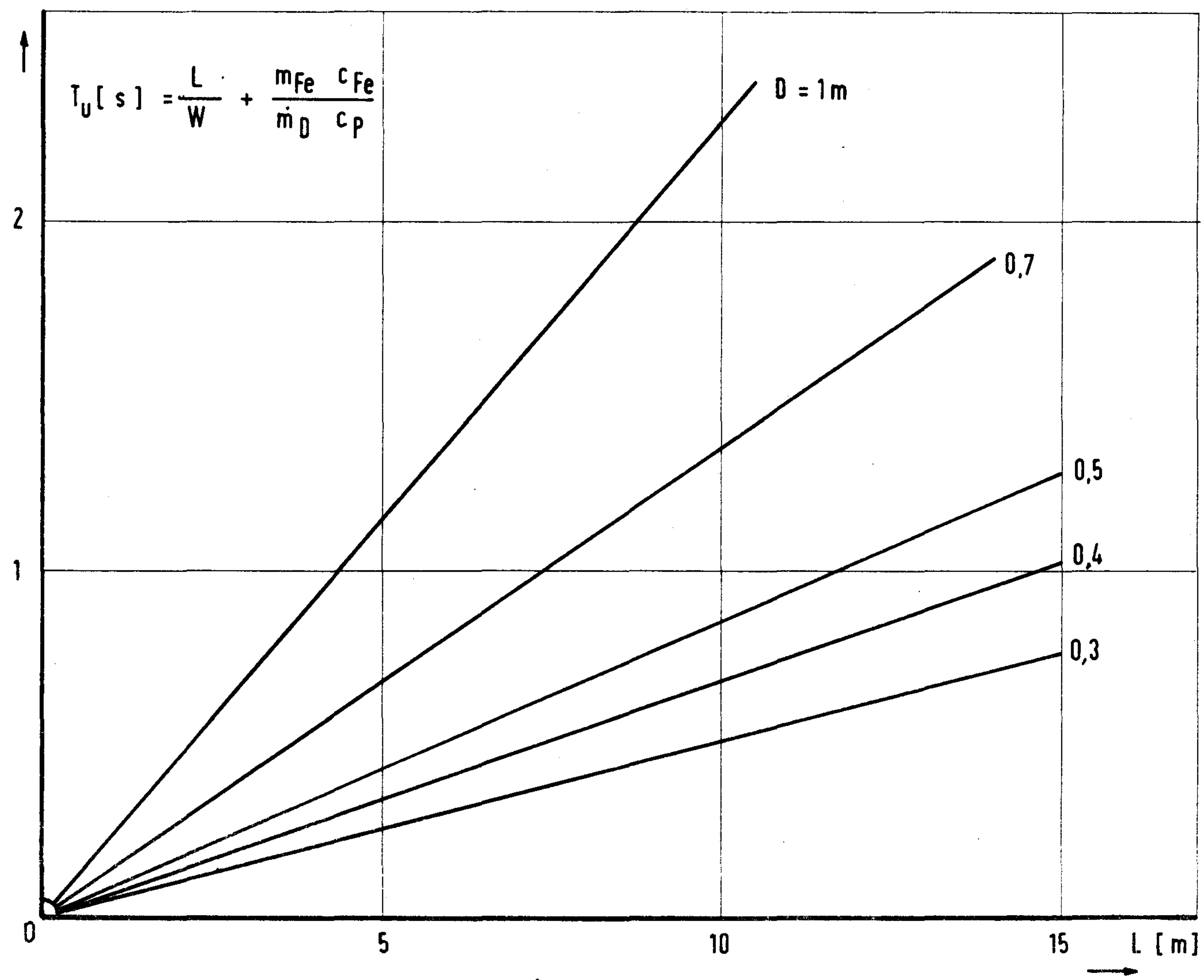


Fig. 4 Delay time in steam pipes

Curve Nr.	1	2	3	4
K_C [MW/at]	30	17	30	30
T_{ges} [s]	21	21	6	21
C [MWs / at]	320	320	320	860
$\frac{K_C \quad T_{ges}}{C}$ [1]	1,97	1,11	0,56	0,73

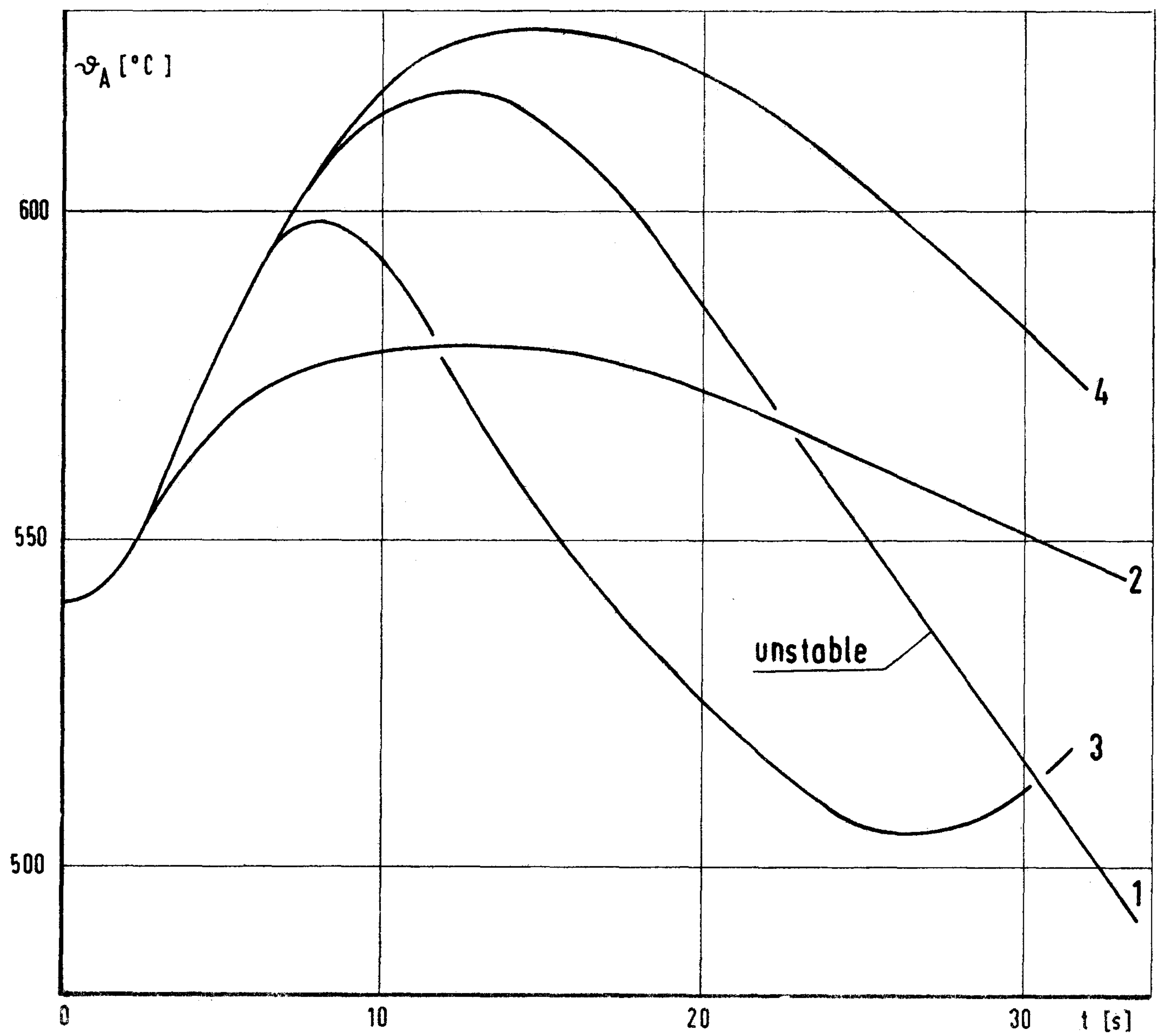


Fig. 5 Exit steam temperature transient following a 0.2 $ reactivity step

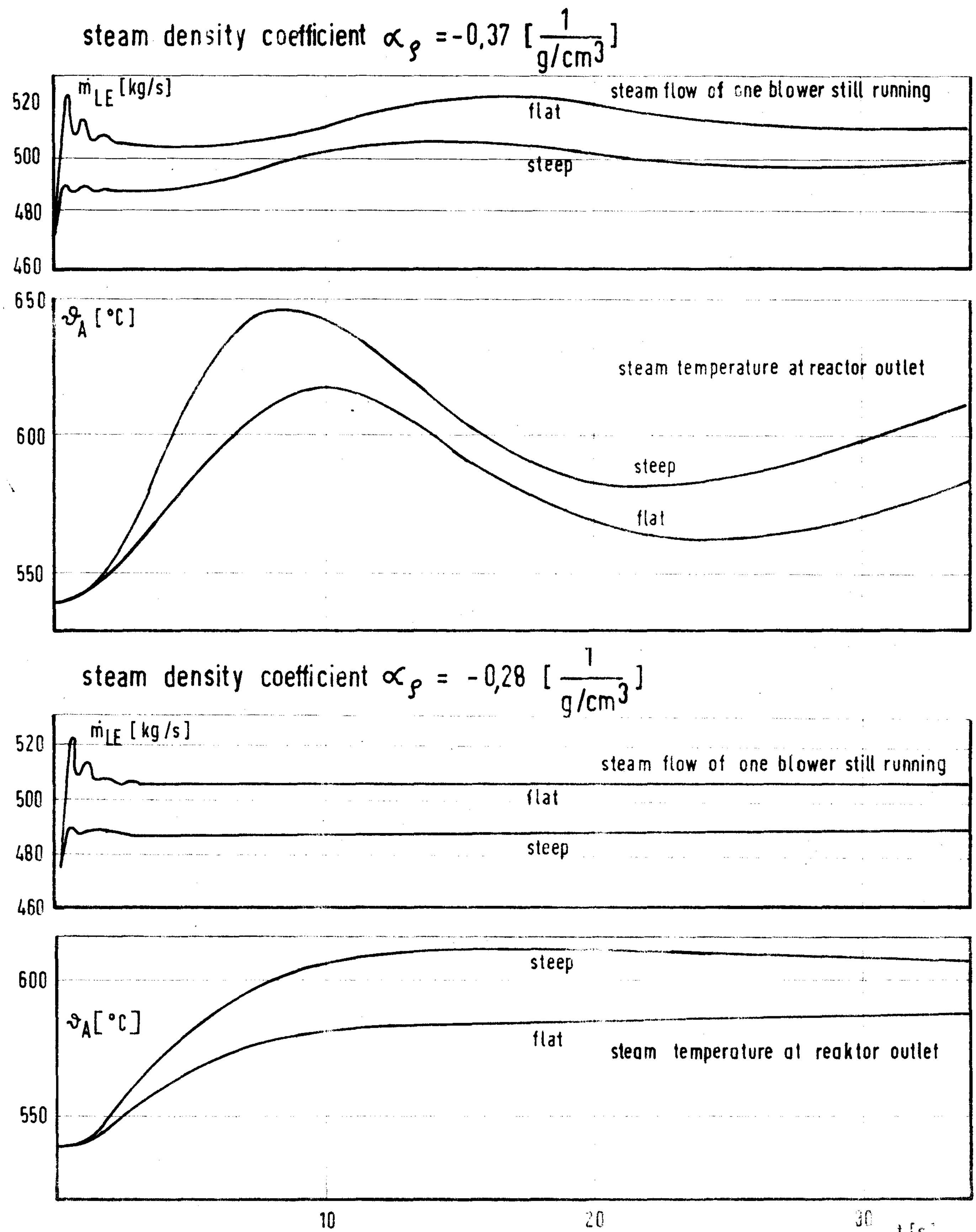


Fig. 6 Steam mass flow and exit temperature transients following the failure of 1 out of 6 blowers

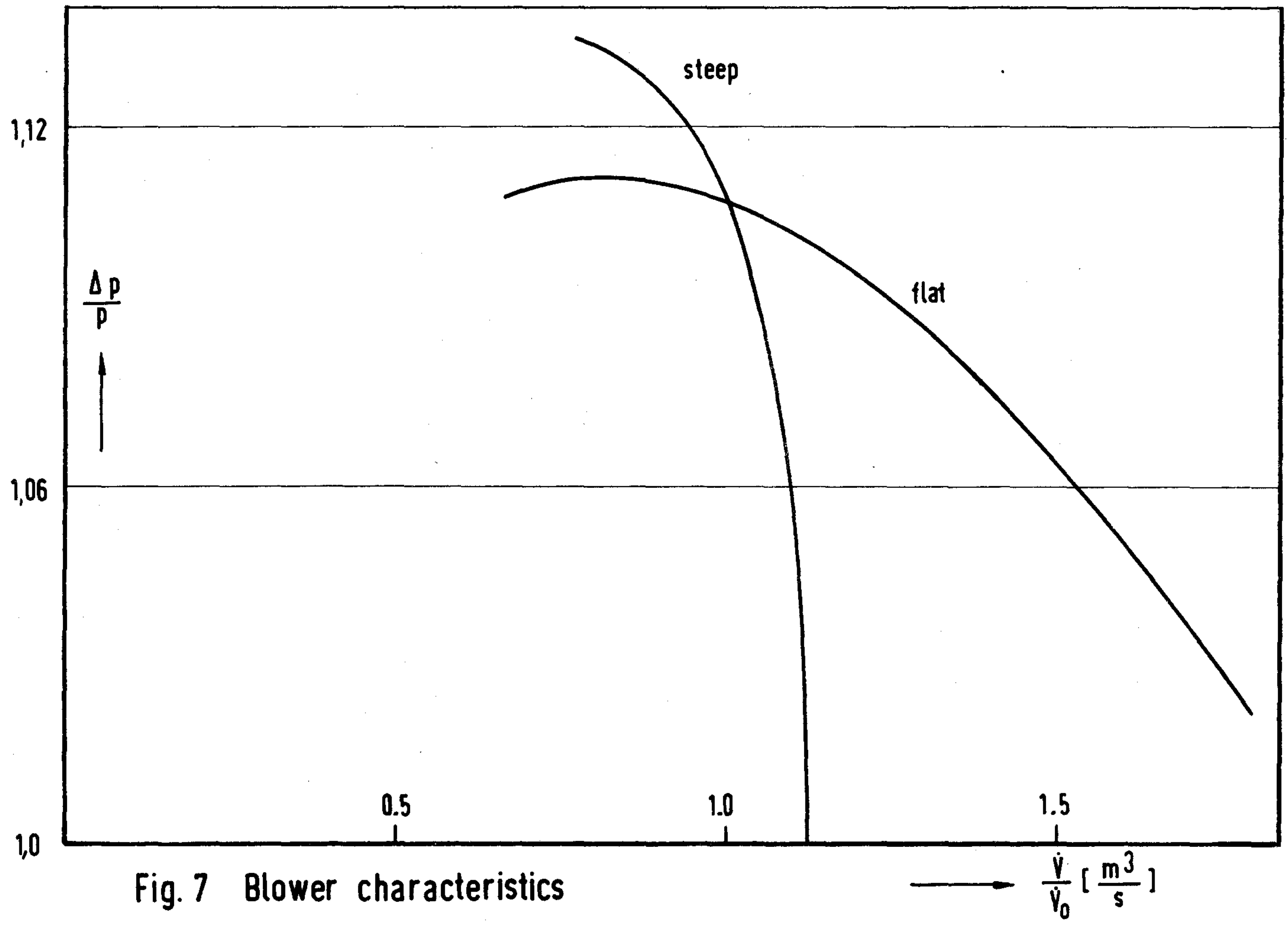


Fig. 7 Blower characteristics

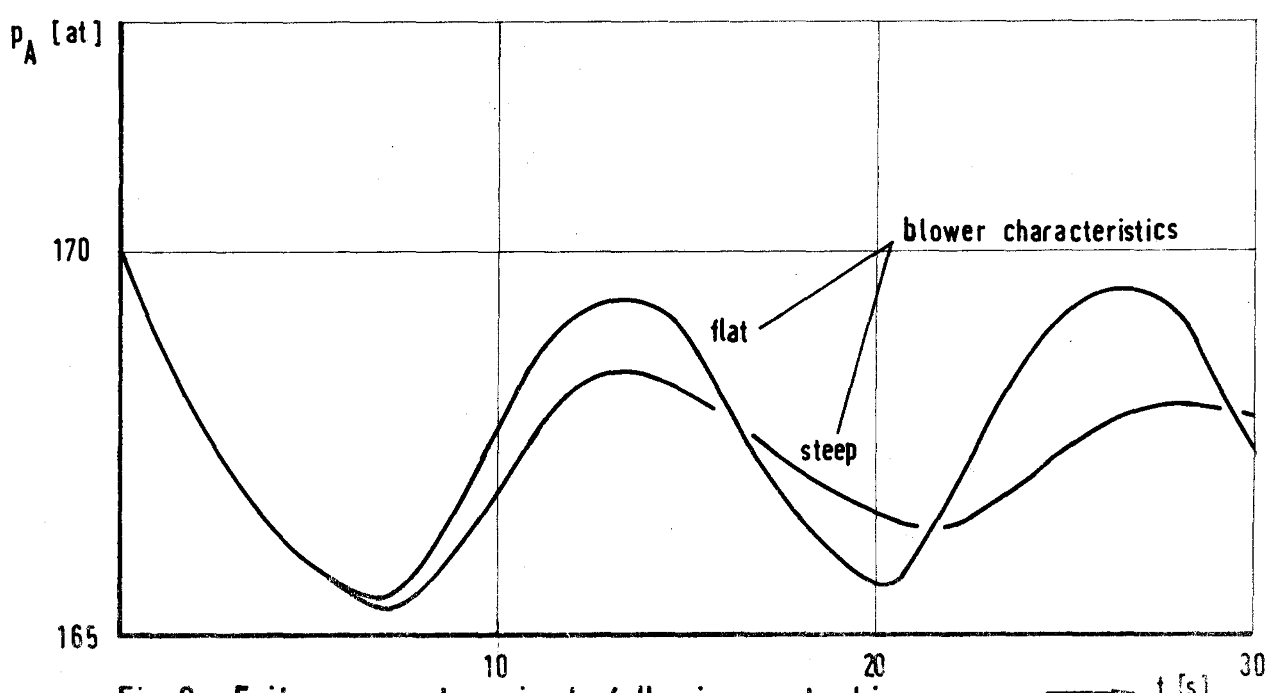


Fig. 8 Exit pressuretransient following a turbine load increase (90 → 100 % output)

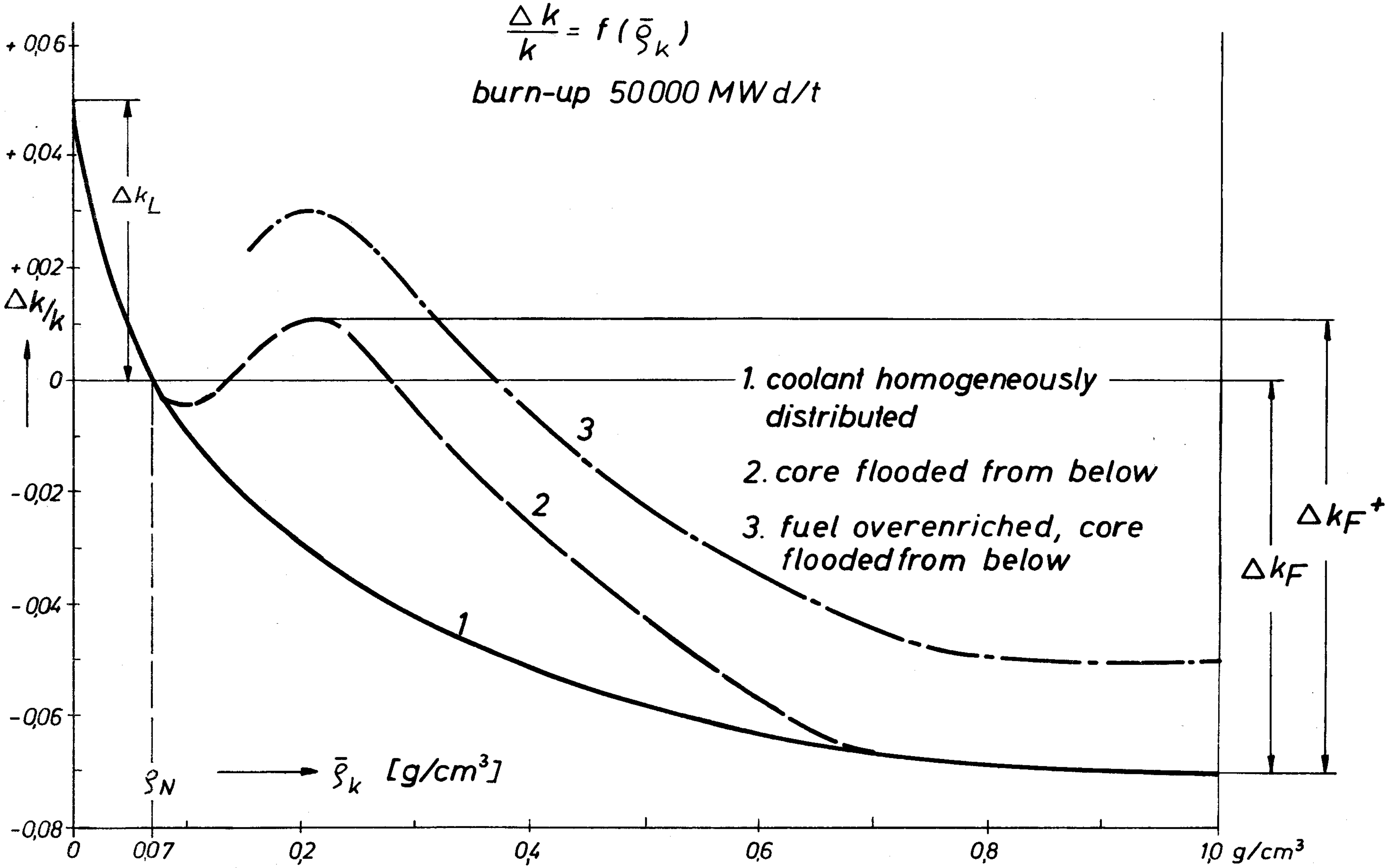


Fig. 9

Reactivity as a function of the average coolant density $\bar{\rho}_k$

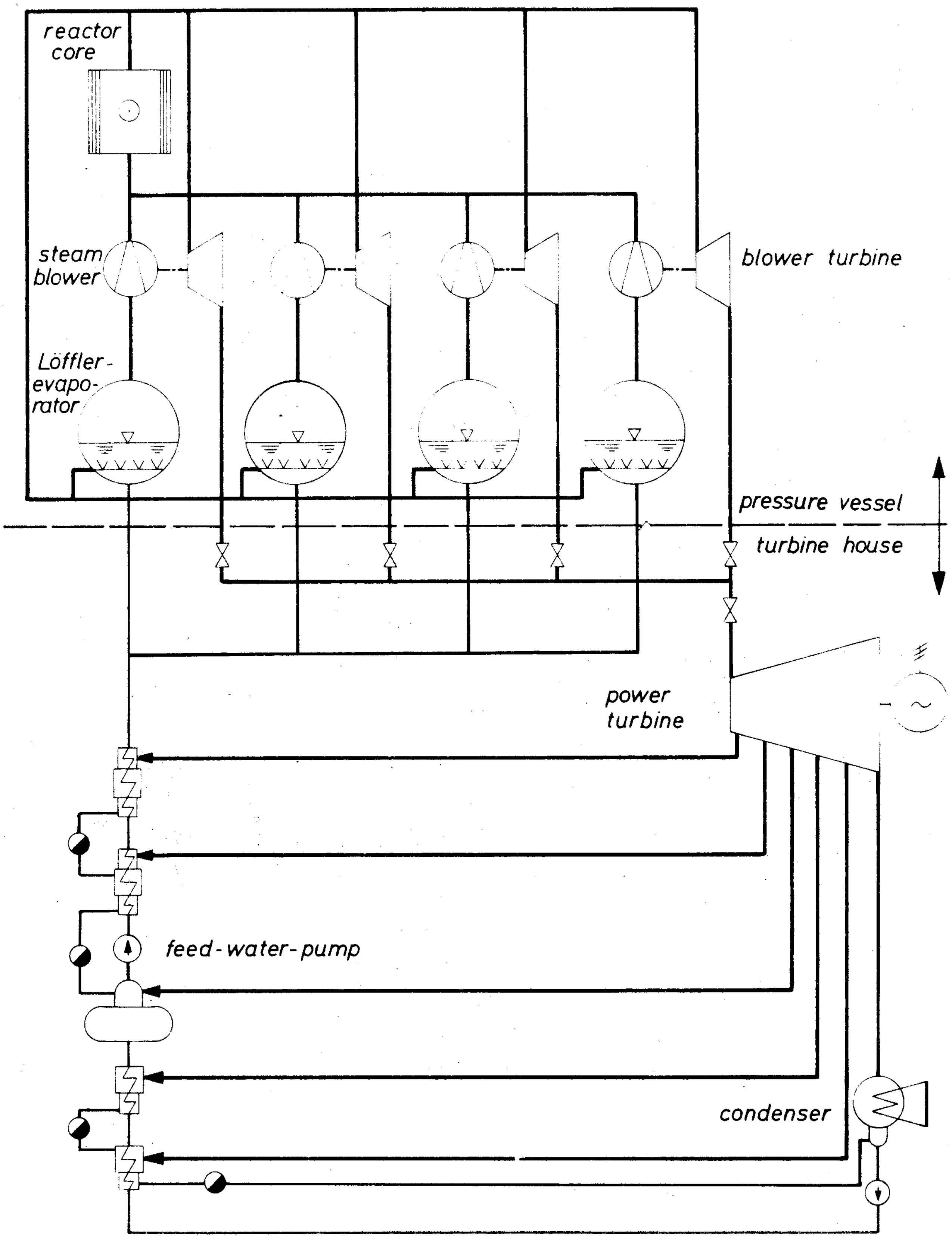


Fig.10 Heat flow diagram of a steam cooled fast reactor