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# Kema Scientific & Technical Reports

Final report
on the aqueous homogeneous
suspension reactor project

KEMA

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# Final report on the aqueous homogeneous suspension reactor project

## Preface

During the period from 1951 to 1977 a group made up of a few highly qualified people, working under the inspiring guidance of Prof. Dr. J.J. Went, did studies and experiments aimed at evaluating the potential of a so-called aqueous homogeneous nuclear suspension reactor. It was felt that a breeding process should also be used for thermal reactors and that this process would allow a highly enhanced utilization of natural uranium and thorium. The use of an aqueous homogeneous suspension of $(U,Th)O_2$ particles as a circulating fuel at high temperature was the idea at the base of the new concept. The inherent safety of the reactor against excursions was ensured by a prompt negative temperature coefficient of the reactivity. A simple production process, developed in-house, for microspheres and the possibility of the fission recoil phenomenon occurring in the fuel, led to the expectation of an attractive fuel cycle.

The group devoting its efforts to this project found itself attracting the interest of a number of institutes and individuals. This community of interest proved to be of great value and, in certain cases, led to the actual participation of specialists in the project.

The present report was written to inform all those parties and persons whose contribution to the project has led to the successful operation of the final installation: the KEMA Suspension Test Reactor (KSTR). We now, in this preface, want to express our sincere appreciation to all those who participated in the project or gave it their support. The Dutch electricity supply companies should be the first mentioned. They carried the main financial burden of the work done at KEMA. The Stichting voor Fundamenteel Onderzoek van de Materie (FOM) and later the Reactor Centrum Nederland (RCN, now ECN) were the partners within the country for some time, while Euratom was the international partner. Additionally, profound interest was shown and personal effort was contributed to the project by the Oak Ridge National Laboratory (ORNL), the Atomic Energy Research Establishment (AERE) at Harwell, the Commissariat à l'Energie Atomique (CEA) and the Société Grenobloise d'Etudes et d'Applications Hydrauliques (SOGREAH) at Grenoble. Both national and international guiding and safety panels were set up to support the program. Additionally the potential of thermal breeders was evaluated as a combined effort with other interested parties under the auspices of the International Atomic Energy Agency in Vienna.

After being brought to critical operation in 1974 the KSTR reached its design power of 1 MWth in 1975. The operation was stopped in 1977 after a total energy production of 154 MWh. Much information has since been collected by examining a number of primary components after partial dismantling of the installation. These activities have delayed the appearance of the present report.

The project gained momentum at a time when questions concerning the resources of fissile material and safety prompted a search for new types of reactors. Improved safety and a more efficient use of natural resources of thorium and uranium were thus the objectives of the KSTR project. Although the project was successful insofar that it demonstrated a suspension reactor with a power density of 50 $kW \cdot l^{-1}$, no follow-up of this line of research could be realised. The reasons for this were that a second, albeit simplified version would have had to be built to operate at a still higher temperature. Also there was a clear absence of international support at that time and further development costs would have been too high to be met by a national effort. Although the many people involved regret that an industrial application has so far not been seriously considered, the project undoubtedly has bred valuable expertise. This expertise has become available, directly

or indirectly, not only for the Dutch electricity supply companies but also to be put to use to some extent and in specific fields of innovation elsewhere in the world.

The project was an altogether very complex technological operation, stretching technical possibilities to their limit. This research, development and construction could only be carried through because an enthusiastic group of high-quality research experts felt challenged by the idea and persisted in working to solve the many problems. The present appreciation obviously extends to the fact that this enthusiasm led to a most valuable national and international participation.

The report is a summary of the project as a whole, providing a more or less chronological survey of the main results of twenty five years of research and development. The results of so many years of diligent experimental work cannot be described in a report of limited size such as this one. It therefore highlights important aspects which are quite specific or which represent important innovative achievements. It is hoped that this report may serve as a useful source of information should there come a revival of scientific and technological interest in nuclear suspension reactors.

**M.E.A. Hermans**

# Final report on the aqueous homogeneous suspension reactor project[1]

KEMA[2]

**Abstract**

KEMA 1987 Final report on the aqueous homogeneous suspension reactor project – Kema Scientific & Technical Reports 5 (1): 1-48. ISSN 0167-8590; ISBN 90-353-0048-3.

The most important results of twenty five years of research and development work on circulating aqueous homogeneous suspensions of fissile material are presented. Experiments in the final phase were carried out with a prototype nuclear reactor (KEMA Suspension Test Reactor, KSTR), fuelled with an aqueous suspension of $UO_2/ThO_2$ particles.

Summarizing the results of the experimental and post-operational research on the KSTR, it can be stated that the suspension reactor has largely fulfilled the expectations of its designers. No operational problems worth mentioning occurred during an experimental period covering several years. From the standpoint of reactor physics the results can be termed wholly favourable, especially at a high power level. The reactor proved to be inherently safe due to the prompt action of the comparatively high negative coefficient of reactivity. The mechanical behaviour of the parts inspected has also satisfied expectations.

The problems encountered have been due exclusively to the presence of a flow stabilizer. This was installed in the period following some time of subcritical operation in order to reduce the reactivity fluctuations. A stabilizer closed on all sides might perhaps have prevented the problems mentioned and the applicability of the principle of the suspension reactor would have been demonstrated still more clearly.

## Introduction

The main argument for the development work on circulating aqueous homogeneous suspensions of fissile material concerned the degree of efficiency with which uranium can be used in thermal reactors of the existing types: graphite-moderated gas-cooled reactors and pressurized- or boiling-water reactors. Those reactor types are all strictly limited as regards the percentage of atoms present in natural uranium which are split in the fission process and which are, therefore, responsible for energy production. That percentage appears to be practically the same for each of the three types mentioned above and amounts to 0.5-1%.

It can be argued whether the considerable resources in manpower and money which are being devoted to development and application of nuclear power, should be expended for the attainment of such a limited objective. In principle that limitation need not exist. In the same way as the readily fissile isotope Pu-239 can be formed from U-238 by capture of a neutron, the fissile isotope U-233 can be obtained from Th-232 which occurs in nature in at least the same amount as uranium, but probably more than ten times as frequent.

The specific isotopes produced in the above mentioned processes are also of importance since the number of neutrons set free in the fission process is not the same for all fissile isotopes. The number of neutrons

[1] This paper is based on a large number of reports authored by KEMA employees from various divisions and departments.

[2] P.O. Box 9035, 6800 ET ARNHEM, The Netherlands.

Table 1
Number $\eta$ of free neutrons liberated when fissile isotopes are split by thermal or fast neutrons.

| Fissile isotope | thermal | fast |
|---|---|---|
| U-233 | 2.28 | 2.60 |
| U-235 | 2.08 | 2.28 |
| Pu-239 | 2.10 | 2.74 |

depends, moreover, on the kinetic energy of the neutrons which are captured by the fissile isotopes (thermal or fast neutrons). Table 1 gives the $\eta$ values which represent the mean number of free neutrons produced in the fission process of the isotopes under consideration.

It is obvious that breeding of fissile material is only possible in a reactor where $\eta$ has a value significantly superior to 2. In such a case one can obtain a total fission ratio of 25-50% instead of 0.5-1%. This has two important consequences. On the one hand, 50 times more energy can be produced from the same quantity of cheap uranium and on the other hand natural uranium or thorium may be used even if 50 times as expensive as is now the case.

If, in the case of thorium, actual breeding of fuel must be obtained, a heavy-water moderated homogeneous reactor is required since it is of great importance to remove as rapidly as possible the neutron poison Xe-135 which constitutes one of the high-yield fission products.

In such an aqueous homogeneous reactor the uranium and thorium must be evenly distributed throughout the liquid. That can be achieved either by means of a salt solution (studied as part of the homogeneous reactor project by the Oak Ridge National Laboratory) or by the utilisation of small oxide particles in suspension (suspension reactor project of KEMA). The fuel involved in both cases has all the attributes of a liquid; i.e. a simple and cheap fuel element. The cooperation with the ORNL was important because the ORNL Homogeneous Test Reactor had already been put into operation in Oak Ridge. Since the researchers at ORNL were on the point of starting up the Molten Salt Reactor, use could be made of the experience gained in that field. The transfer of data was achieved by exchange of reports and notes with photographs and by exchange visits.

The properties of the KSTR fuel changed during extended power operation. On one hand those changes had a favourable effect on reactor noise, on the other hand they resulted in loss of reactivity due to which the critical temperature decreased.

In the reducing environment of the KSTR the radiolytic products are already recombined within the reactor vessel. When helium is used as pressurizing gas, the internal recombination is much less and $H_2$ and $O_2$ are transported to the external system together with the flowing liquid. The net production is then proportional to the reactor power and is approx. 3 kg $O_2$ per MWh.

*Framework of the study*

In principle, the construction of a suspension reactor is also very simple: it must comprise a core vessel, a circulating pump and a heat exchanger. Much research and development work are needed, however, to ascertain the possibilities and limitations of suspensions under the conditions extant in a power reactor. Such research should include the following topics.

(1) Determination of the extent to which the particles can be homogeneously distributed and maintained in a circulating suspension at a sufficiently constant concentration. This problem is of particular importance since it is related to the extent to, and the manner in which the reactor can be controlled.
When the concentration of the suspension in the reactor vessel varies, the nuclear reactivity of the system also varies. The resulting fluctuations must be kept within fairly narrow limits. For this aspect of the study the first step was to build a subcritical reactor operating at low temperature (up to 343 K) which could be brought very close to the critical point ($k_{eff} = 0.99$) by varying the concentration of the particles in suspension in a continuous way. The results gained from this work show that the requirements described above could be met (see **'Subcritical experiment'** section).

(2) Investigation of the chemical and mechanical conditions that the particles will experience when suspended in water at high temperature, and development of techniques for the manufacturing of particles that can withstand such conditions (see **'Research and development'** section).

(3) Solution of construction problems related to e.g. reactor vessel, hermetically sealed and sufficiently wear-resistant pumps, valves, flanges, a gas injector, a gas separator, a heat exchanger, a fuel-storage vessel, a dumping vessel etc. (see **'Reactor design'** section).

(4) Final testing under operational reactor conditions. That purpose does not require building of a prototype power reactor. A small installation can be used, provided that all conditions such as temperature, pressure, concentration, power density etc. be identical to those required in a real power reactor. The KEMA Suspension Test Reactor (KSTR) with a power density of 50 $kW \cdot l^{-1}$ was built for the purpose (see **'Operation and experimental research'** section).

(5) Analyses of the fuel properties (see **'Changes in the physico-chemical properties of the fuel'** section).
(6) Analysis of the mechanical behaviour of the various equipment (see **'Post-operational research'** section).

**Subcritical experiment**

A subcritical reactor was built for the purpose of investigating aqueous suspensions. The principal aim was to ascertain whether a suspension could be distributed throughout the reactor system in a sufficiently homogeneous manner and whether the concentration could be maintained constant over a long period of time. It was also important to determine whether, in practice, it would be possible to regulate the concentration in a sufficiently accurate and continuous manner.

These points can be assessed with by far the highest degree of precision from the nuclear reactivity of a subcritical reactor i.e. by monitoring the neutron flux of the reactor being operated close to the critical point at a constant external neutron source: the source multiplication. The thermal and epithermal neutron flux distributions were thus determined both inside the reactor vessel and in the reflector region.

It was found during the investigation that this procedure could also yield important information concerning hydrodynamics inside and outside the reactor vessel. It was also found that the colloid-chemical properties of the suspension had a strong influence on the behaviour of the reactor.

*Fuel*

A suspension of $UO_2$ particles in light water was used as the fissile material (the $UO_2$ being enriched in U-235 up to 20% for the experimental reactor; a low enrichment would be required for a commercial breeder based on the principoles of the KSTR). The basic parameters for the particles were their size and shape and the state of their surface. The upper limit of about 15 $\mu$m for the size was set by the following two conditions.

(1) A sufficiently high yield for the 'recoil separation'. By this is meant the recoil of fission products from the $UO_2$ particles into the water. Since the size range of fission products in $UO_2$ was about 10 $\mu$m according to the available literature, an upper limit of 15 $\mu$m was assumed for particle size. Subsequent measurements by Markestein & Van der Plas (1962) showed that the size range was about 5 $\mu$m.
(2) The settling velocity of the $UO_2$ particles in the water had to be small in relation to the flow velocity of the liquid in the reactor vessel. It also had to be comparable with the settling velocity of 5 $\mu$m particles in water of 523 K.

The lower limit for particle size was fixed at 3 $\mu$m so that the concentration could be further adjusted by means of a hydroclone. In addition, the particles must have a smooth surface and a preferably spherical shape in order to prevent as far as possible mutual attrition and erosion of the walls during circulation.

Finally, in order to make the specific surface area as small as possible, the $UO_2$ particles were sintered at high temperature. The method of fabricating the particles has been published by Hermans (1958) and will be described in the **'Research and development'** section of this contribution.

The colloid-chemical behaviour of the suspension was determined by the pH of the fluid. This determines in particular the extent to which the particles are flocculated. Where pH = 7, the $UO_2$ suspension is unstable, i.e. flocculated (Kalshoven, 1962). Where pH values are higher than 9 (but lower than 12), the suspension is stable, i.e. non-flocculated. It will be seen from the results of the experiments described in this report that the question whether the suspension is or is not flocculated has considerable influence on reactor behaviour.

*Reactor construction*

Figure 1 represents a simplified diagram of the layout. The reactor vessel was made of stainless steel, placed vertically (∅ = 28 cm, volume 17 l) and had a cylindrical shape with cone-like extremities. The suspension flowed from bottom to top. There was a flow directer, the shape of which was based on scale-model tests, in the inflow opening. The purpose of the manifold was to ensure good flow distribution and a homogeneous concentration distribution in the reactor vessel.

The reactor vessel was placed in a BeO reflector which in turn was surrounded by graphite blocks. In this way, the dimensions of the vessel could be kept small and the thermal-neutron flux in the reactor vessel was more flat. In the BeO reflector – where the thermal-neutron flux was greatest – 8 aluminium tubes (∅ = 20 mm) were fixed around the reactor vessel. The tubes contained:
(1) a Ra-Be neutron source ($7.5 \times 10^5$ neutrons per second),
(2) three $B_4C$ safety rods (each containing 100 g $B_4C$),
(3) four $BF_3$ neutron detectors.
In addition, a $BF_3$ neutron detector was located in the graphite reflector. The BeO reflector also contained the target of a 500 kV accelerator tube, with which about $10^8$ neutrons per second could be generated. This neutron source was used for special investigations (see '*Re-*

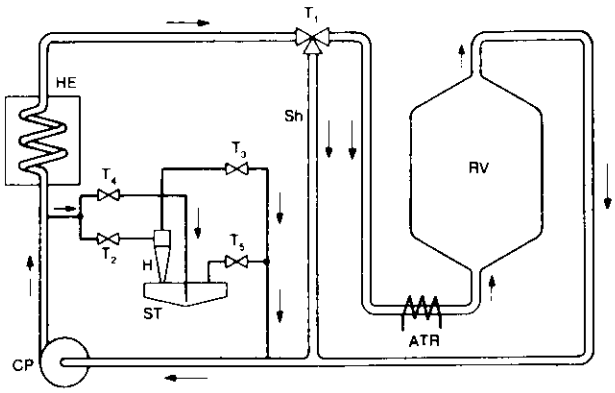



Fig. 1
Simplified diagram of the subcritical assembly. This system made it possible to adjust the source multiplication to within a few percent.
RV = reactor vessel; CP = circulating pump; HE = heat exchanger transmitting the heat through a closed circuit to a second exchanger (not shown) which is cooled with water from the main supply; ATR = automatic temperature regulator; Sh = shunt pipe across the reactor vessel; $T_1$-$T_5$ = control valves; H = hydroclone for separation of $UO_2$ particles from the water (open valves $T_2$ and $T_3$; S = particle storage tank (valves $T_2$ and $T_5$ must be opened to allow $UO_2$ particles to flow from the storage tank into the reactor circuit; $T_4$ is used for the evacuation of the last remaining particles from the storage tank, if necessary).

*coil separation efficiency of fission products*' section).

As the reactor had a high negative temperature coefficient, it was necessary to maintain a sufficiently constant temperature during measurements. The temperature was kept constant to within $\pm$ 0.05 K by electrical heating of a section of the piping with the aid of a thyratron amplifier and an appropriate transformer. Regulation was effected by means of a resistance thermometer inserted in a Wheatstone bridge.

*Experimental results*

Experiments were carried out on:
(1) the critical concentration,
(2) fuel hold-up in the reactor vessel,
(3) the temperature coefficient of reactivity,
(4) recoil separation efficiency of fission products,
(5) neutron-noise analysis,
(6) the neutron-flux distribution.
The results of these experiments are described in the following sections.

*Critical concentration* – The reactor system (total volume approx. 33 l) was filled with demineralized water (pH = 6.8); the storage tank contained 6.5 kg of $UO_2$ with a particle-size distribution as shown in Figure 2. The $BF_3$ counter located in the BeO reflector opposite to the Ra-Be source recorded 270 counts per second and the $BF_3$ counter in the graphite reflector recorded 110 counts per second.

The critical concentration was determined by extrapolation of the curve (1/M versus concentration) to 1/M = 0 (M = multiplication factor). Such an extrapolation was considered permissible since the last part of the curve must be linear. The critical concentration proved to be 198 g $UO_2$ per liter for the volume flow applied, i.e. 9.7 $m^3 \cdot h^{-1}$.

*Fuel hold-up in the reactor vessel* – Upon changing of volume the flow, it became apparent that M depended on this parameter. Since the relation between multiplication and concentration had been established, it was possible to determine the concentration (not the average concentration in a purely physical sense, but the apparent concentration as it influences the neutron physics of the reactor) in the reactor vessel as function of volume flow Q. The results are represented by the uppermost curve in Figure 3.

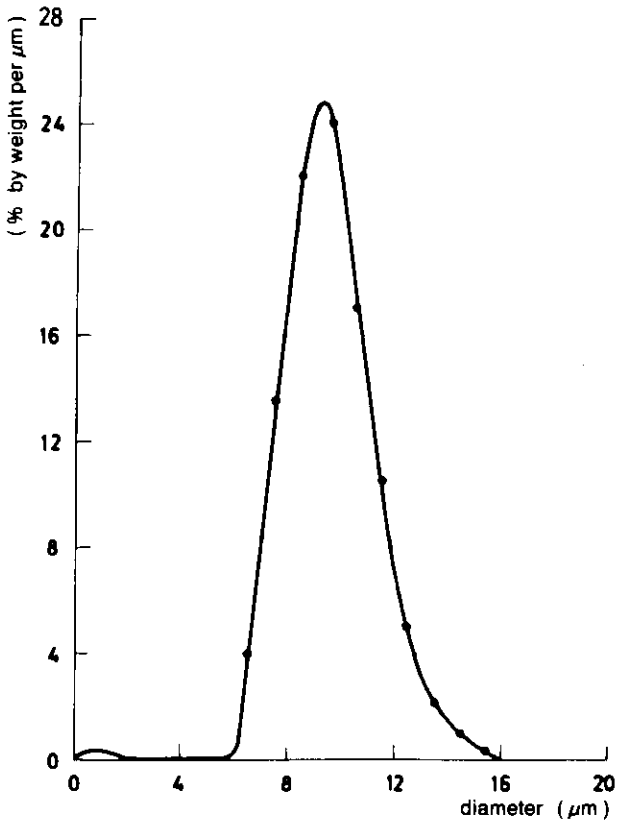



Fig. 2
Size distribution of the dispersed $UO_2$ particles prior to their use in the reactor.

The concentration at a volume flow of 24 $m^3 \cdot h^{-1}$ was arbitrarily denoted as 100. It could be deduced from the curve based on the measurements that the concentration in the reactor vessel increased by 12% when the volume flow was reduced to 1.75 $m^3 \cdot h^{-1}$. With a still smaller volume flow, settling of $UO_2$ particles occurred and the source multiplication dropped to unity within a few minutes.

The concentration increase in the reactor vessel could be explained by the fact that at a volume flow of 1.75 $m^3 \cdot h^{-1}$ the average linear flow velocity of the fluid in the vessel (8 $mm \cdot s^{-1}$) became comparable with the settling velocity of the flocculated $UO_2$ particles. That is not the case with a non-flocculated suspension and the concentration increase observed then was consequently much lower, i.e. less than 1% (Figure 3, bottom curve). With a stable suspension, the flow rate at which settling of the particles started in the reactor was found to correspond to the rate which could be calculated from the size distribution of the dispersed (non-flocculated) particles (Figure 2).

Since the reactivity – and thus the concentration – was constant for volume flows between 24 and 4 $m^3 \cdot h^{-1}$ (Fig. 3), no settling of particles occurred on the inclined bottom walls of the vessel for flows greater than 4 $m^3 \cdot h^{-1}$ either. This was also the case with a flocculated suspension for volume flows higher than 16 $m^3 \cdot h^{-1}$. Consequently, at sufficiently high volume flows, a flocculated suspension presented no difficulties as far as reactivity is concerned.

*Temperature coefficient of reactivity* – Attempts were made to measure the temperature coefficient of reactivity between 294 and 343 K. No multiplication difference was observed, however, in this temperature range with a flocculated suspension. That could be explained by the fact that the settling velocity of the particles increased with the temperature since the viscosity of the water decreases as the temperature rises. The negative temperature effect caused by the density reduction will then be compensated by the concentration increase in the reactor vessel, which in turn is a consequence of the increased settling velocity. It should be mentioned that, although a negative temperature coefficient could not be observed under these circumstances, the negative temperature coefficient of the reactor will be effective at a high temperature (475-575 K) in spite of the higher settling velocity at higher temperatures. This was to be expected, since $\delta\rho/\delta T$ increases considerably with the temperature (T) whereas $\delta v/\delta T$ is virtually independent

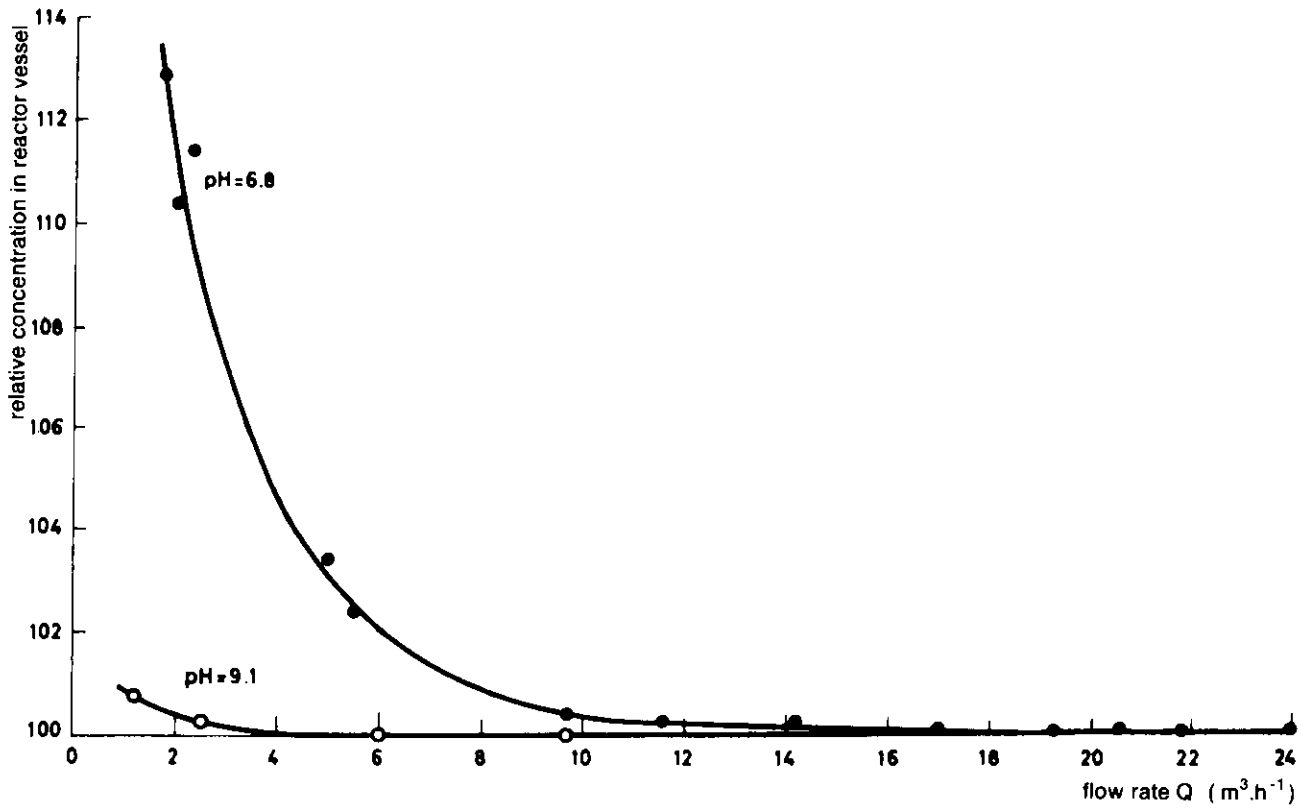



Fig. 3
The $UO_2$ concentration in the reactor vessel as a function of flow rate (Q) and pH. A concentration of 24 $m^3 \cdot h^{-1}$ is arbitrarily set at 100. The suspension was flocculated at a pH of 6.8 (upper curve), whereas it was stable at a pH of 9.1 (lower curve).

of T (e = density of water, v = particle settling velocity).

This was in agreement with the concentration as measured outside the reactor vessel, which decreased as the temperature increased. The effect of a temperature rise on the concentration in the reactor vessel was similar to the effect of a derease in flow rate. Both effects could be explained by an increase in the settling velocity relative to the flow velocity in the reactor vessel.

When a stable suspension was used the effect of the negative temperature coefficient was no longer compensated by the increase in the settling velocity of the particles when the temperature increased. With a pH of 9.1, where the particles were completely dispersed (stable suspension), a negative temperature coefficient of $-2.3 \times 10^{-4}\ \Delta k/k \cdot K^{-1}$ was measured between 294 and 343 K.

When the reactor was heated rapidly ($200\ K \cdot h^{-1}$) a negative temperature coefficient between 21 and 40 K was also measured ($-1.2 \times 10^{-4}\ \Delta k/k \cdot K^{-1}$) in the case of a flocculated suspension. Here the flocs did not have sufficient time for complete settling. From this it could be concluded that upon a sudden temperature increase the negative temperature coefficient of the reactor will be completely effective even with the use of a flocculated suspension.

*Recoil separation efficiency of fission products* – By using the accelerator tube as a neutron source and by operating the reactor at a multiplication of approximately 100, a power of about 0.003 W could be obtained. That was sufficiently high to produce enough fission products within a few hours to allow the chemical determination of the recoil separation effect. A detailed description of these experiments and the complete results can be found in Markestein & Van der Plas (1962). A yield of 54% was found (percentage leaving the particles) for iodine with a flocculated suspension. That yield was lower than hoped for on the basis of the size distribution given in Figure 2. It could, however, be explained by the fact that, through floc formation, the effective particle-size tended to increase.

In the case of a stable suspension, the observed yield of the recoil separation effect corresponded with the value which could be calculated from the particle-size distribution shown in Figure 2, i.e. the maximum theoretical value had been reached. This yield was 76% for iodine. These results made it possible to determine the size range of various fission products in $UO_2$ as 4-8 $\mu$m (Markestein & Van der Plas, 1962).

*Neutron-noise analysis* – Since any local fluctuations of the concentration in the reactor vessel, as well as changes with time might cause reactivity variations, the neutron noise of the reactor was investigated in detail. As the reactor had been operated at zero power, those fluctuations in reactivity i.e. in the neutron flux, could not be damped by the negative temperature coefficient. In consequence, the neutron flux was extremely sensitive to any variation in the $UO_2$ concentration inside the reactor vessel.

During the experiments, the average value of the neutron flux was found to remain constant within the measuring accuracy of $\pm$ 1% for several hours (Fig. 4). That indicated a constant concentration in the reactor vessel within a limit of 0.1% during the same period. In other words, there occurred no sticking or uncontrolled release of $UO_2$ particles during this time and within this limit (0.1% corresponds to approximately 6 g $UO_2$).

Although the average value of the neutron flux was constant, the records showed fluctuations in the flux with a maximum amplitude of $\pm$ 10% ($\Delta k/k = \pm 0.0021$) at a multiplication of 47.6 (counting rate 36,000 counts per second). A flocculated suspension was used for making these records (Fig. 5, upper curve).

It also emerged that the frequency of the fluctuations depended on the rate of flow of the suspension in the reactor vessel. Since the records showed that one or more special frequencies were likely to have prevailed in the fluctuations, the following quantitative analysis method was applied.

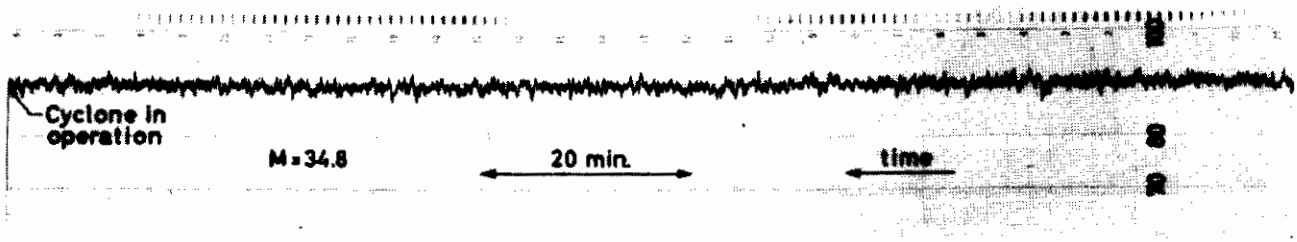



Fig. 4
Record of the neutron flux as a function of time during steady operation of the reactor for two hours. The average flux value was constant to within 1% (counting rate between 8900 and 9100 counts per second).

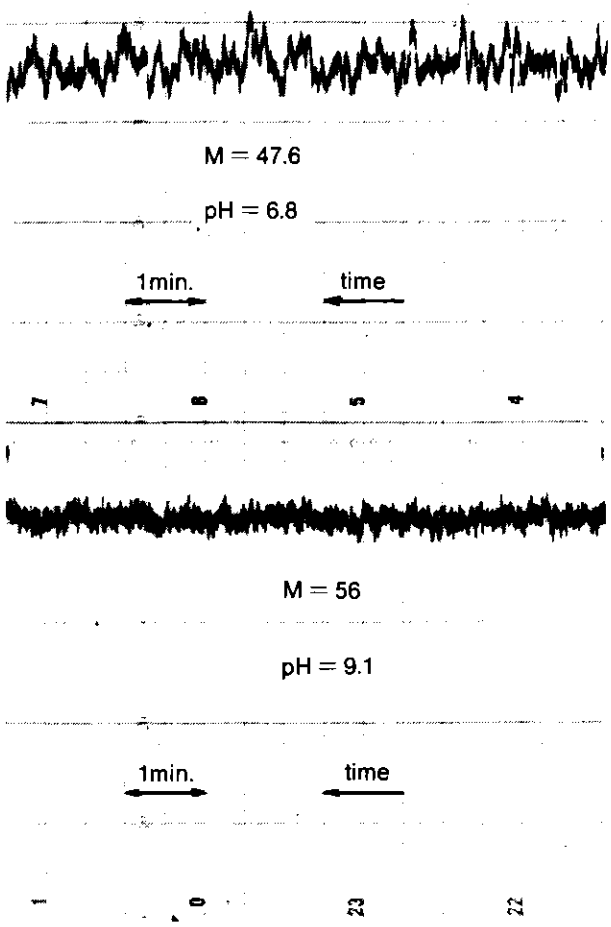



Fig. 5
Records of the neutron flux as a function of time at a low pH value in a flocculated suspension (upper curve) and at a high pH value in a stable suspension (lower curve). The two curves show remarkable differences as regards both amplitude and frequency.

If $\bar{c}$ represents the average number of counts in a given time interval (gate length),

$$\frac{\overline{c^2} - (\bar{c})^2}{\bar{c}} = 1 + Y \text{ is a measure for the fluctuations.}$$

For a Poisson distribution, Y = O.Y is thus a direct measure for the deviation of the fluctuations from a Poisson distribution. The measured values of Y can be plotted against the gate lengths used. The gate length appropriate to any peak in the curve gives the half-period of a characteristic frequency.

In order to avoid an influence of fluctuations caused by the chain character of the events in the reactor, the experiments were carried out at a relatively low multiplication (M = 35). The volume flow was 9.7 $m^3 \cdot h^{-1}$. The $BF_3$ detector impulses were amplified, transmitted to a fast discriminator and subsequently to a gate circuit. The gate length could be varied between 10 ms and 100 s. The impulses which were allowed to pass through the gate were counted by means of a scaler which was then read and reset to zero by hand. The scaler was read 400 times for each gate length used. The Ra-Be source was used for the purpose of checking whether random impulses really produced Y = O with this arrangement. After 400 readings with a gate length of one second, it was found that Y = −0.05. That was considered satisfactory. Subsequent experiments were carried out with the same counting device. The results with a flocculated suspension are given in the top curve of Figure 6. That curve shows that there were two characteristic periods, viz. 1.2 s (being 2 x 0.6 s) and 16 s (being 2 x 8 s) at the rate of flow used.

The amplitude of the fluctuations in the neutron flux was appreciably lower with a stable suspension than with the floeculated suspension (see Fig. 5). The maximum amplitude for a pH = 9.1 and a multiplication of 56 appeared to be ± 3.5% ($\Delta k/k$ = ± 0.0007).

No characteristic frequencies were observed with the method of analysis described above. The fluctuations showed a Poisson distribution. The experimental results for various gate lengths are given in Figure 6 (design Z).

*Neutron-flux distribution* – The neutron-flux distribution inside the reactor vessel as well as in the reflector region was determined for a source multiplication of about 100. This multiplication was sufficient to eliminate the influence of the external neutron source. The neutron-flux distribution in the vessel was determined first by activation of an indium wire followed by activation of a wire made of a dysprosium-aluminium alloy

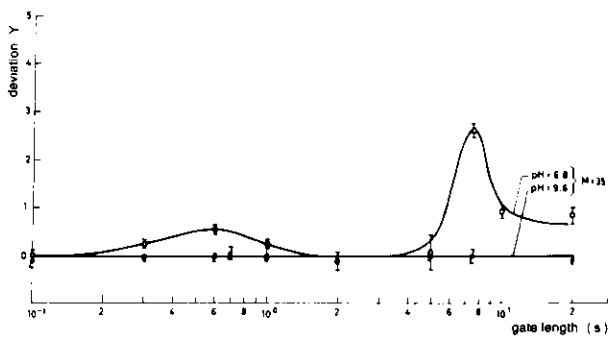



Fig. 6
Deviation (Y) from the Poisson distribution in the fluctuations of the neutron flux as a function of the counting time (gate length). The top curve (pH = 6.8; flocculated suspension) shows a definite maximum of approx. 8 s (characteristic period of 16 s). There was no deviation from the Poisson distribution at a pH value of 9.6 (stable suspension; lower curve with experimental values designated by Z).

(10% Dy). The wire (∅ 1 mm) was mounted on an aluminium fixture, brought inside the vessel through a narrow tube (∅ 18 mm) located at the top. The wire was then stretched out in a horizontal plane at the centre of the vessel and perpendicular to the radius through the external neutron source. The aluminium fixture and wire had practically no influence on the source multiplication or the neutron-flux fluctuations as a function of time.

The induced activity of the wire was measured after one hour of reactor operation at 18 sections of 15 mm length, covering 27 cm of the vessel diameter (28 cm). The neutron-flux distribution in the reflector region was determined by measuring the induced activity of small pieces (15 mm) of the same wire material which were mounted at 5 points in the same horizontal plane as mentioned above.

The experimental results are given in Figure 7. Since the measurements with the dysprosium wire showed errors smaller than those for measurements with the indium wire, the flux distribution of the thermal neutrons in the reactor vessel could be determined with greater accuracy than the flux distribution of the epithermal neutrons. Figure 7 shows that the neutron-flux distributions in the reactor vessel were symmetrical and that the flux distribution of the thermal neutrons was indeed very flat.

*Conclusions*

It can be concluded from the experimental results that the subcritical unit appears to be a most suitable and sensitive tool for investigating the essential properties of a homogeneous suspension reactor.

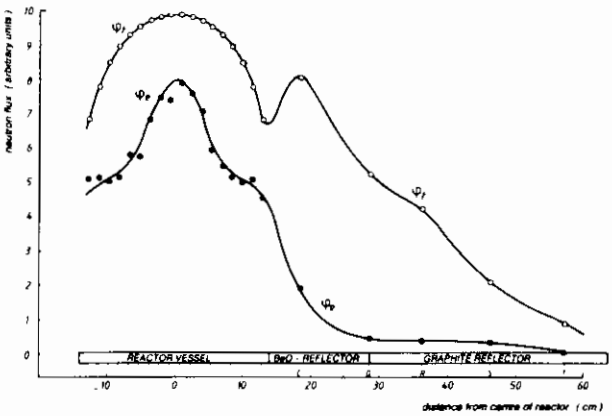



Fig.7
Thermal ($\varphi_t$) and epithermal ($\varphi_e$) neutron-flux distribution inside the reactor vessel and the reflector region.

Subcritical experiments were indispensable in the development programme for that reactor, for checking and completing basic chemical and hydrodynamic research and measurements made on models. Subcritical measurements are extremely sensitive and in this case could be carried out relatively fast since the concentration could be regulated rapidly and accurately. This was the cheapest method for ascertaining whether the colloid-chemical properties of the suspension and the hydrodynamic conditions inside and outside the reactor vessel had been correctly selected.

Because of the low radiation level during and after the experiments, there were no maintenance problems – as is indeed the case in all subcritical assemblies – and the necessary modifications could thus be carried out without difficulty.

**Research and development**

Extensive research and development work on chemistry, technology and control systems was required for both the subcritical experiment and the KSTR. Although the major part of those activities took place prior to the construction of the two installations, development work continued during operations as well. Some of the operational results from the KSTR (**'Operation and experimental research'** section) will therefore be dealt with here.

*Chemical research and development work*

The main problems concerned:
(1) fuel preparation,
(2) irradiation research,
(3) sedimentation behaviour,
(4) oxygen formation and recombination,
(5) removal of the fission products.

*Fuel preparation* – Two important processes were developed for preparation of the fuel: (a) the urea process and (b) the sol-gel process. The latter can be subdivided into (b-1) a first and (b-2) a second stage.

(a) The so-called urea process was applied for the preparation of the $UO_2$ particles for the subcritical experiment. A narrow size distribution of reasonably spherical particles can be achieved with this process.

The principle of the technique is a controlled precipitation of a stabilized amorphous $UO_3$ hydrate where germination is restricted to the start of the process, after which the particles grow by controlled flocculation and by continuing precipitation of the uranium which is still in solution. Stabilization of the amorphous form of the

precipitate is achieved by having a well selected nitrate content of the uranyl-nitrate solution and by maintaining a $CO_2$ atmosphere above the reacting liquid.

The procedure is as follows: take 1 liter of 0.1 molar uranyl nitrate, 1.32 molar nitrate content, to be partially neutralized with 50 ml of 25% $NH_4OH$ solution. Stir well using a special agitator (Hermans, 1964) and heat to 368 K, then add 500 ml of a solution with 250 g urea, also warm. After approx. 50 minutes filter off the precipitate, wash with cold water, rinse and dry with acetone, reducing in moist hydrogen at 1525 K. The resulting particles have a specific surface area of 0.1 $m^2 \cdot g^{-1}$ and the yield is higher than 99.5%.

This procedure, spread over several batches, yielded 6.4 kg of $UO_2$ with an enrichment of 21% for the subcritical experiment.

The problems in the process lay in the growth of the precipitate (shape and size of the particles) and in development of the 'pusher furnace' for reduction and the sinter oven. The preparation itself proceeded without problems.

(b-1) The fuel for the KSTR was prepared by means of the sol-gel process, developed for this purpose by KEMA (Van der Brugghen et al., 1968). Starting from $UO_2(NO_3)_2$ and $Th(NO_3)_4$, there were four steps to the sintered end-product (Fig. 8).

The sol was prepared by dispersion of a precipitate washed free of electrolyte. A Th-oxide hydrate precipitate, washed free of electrolyte, was dispersed by boiling in a $UO_2(NO_3)_2$ solution.

Van der Brugghen et al. (1968) have discussed both the necessity of starting from a $NO_3/(Th + U)$ ratio of 0.3 to 0.4 and the method used to achieve this ratio. The emulsifying of the sol took place in a mixture of carbon tetrachloride and toluene with a density slightly lower than that of the sol. Use was made of a vibrating agitator with adjustable amplitude and fixed frequency (50 Hz). The size of the sol drops, which ultimately determines the size of the fuel grains, could be set at the correct value by adjusting the agitating conditions. The gel spheres obtained could be separated from the emulsion by sedimentation.

The gel spheres were dehydrated by boiling them in a mixture of carbon tetrachloride and toluene with continuous removal of water. The dehydrated spheres were

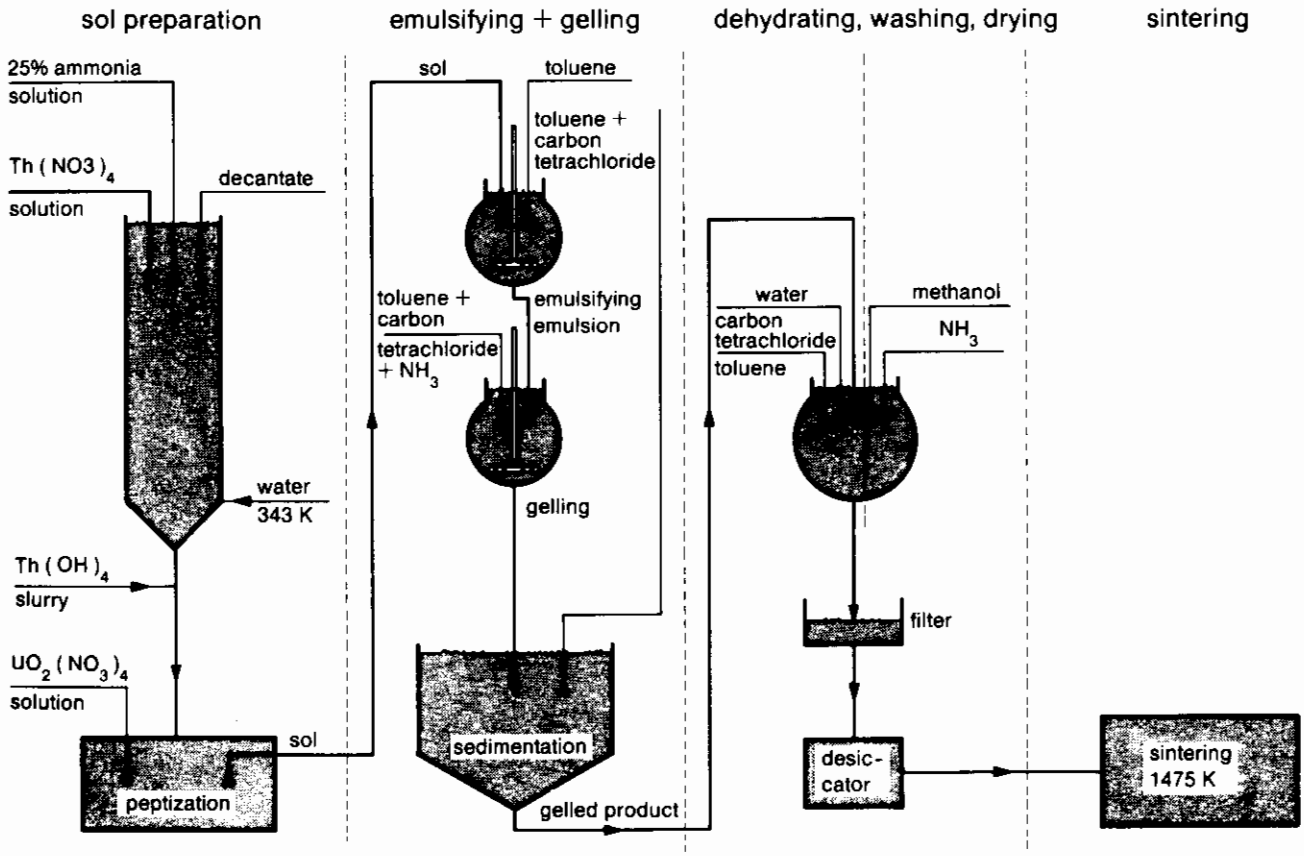



Fig. 8
The sol-gel process.

filtered and washed with methanol to remove as much as possible of the $NH_4NO_3$ present. The final step was filtration and drying at 473 K.

The fuel particles were sintered in a pusher oven in air. They remained in the hot zone of 1453-1473 K for a total of 160 minutes. Under those conditions it was possible to achieve the desired specific surface area of $0.2\ m^2 \cdot g^{-1}$. The fuel was prepared in a steel plate cabinet specially built for the purpose. Practically all the procedures required could be performed from the outside. The working room was kept under under-pressure by means of fans, to prevent leakage of matter containing U and Th.

If the quantity of uranium handled was not restricted, the processing of natural uranium yielded 1 kg of fuel per day. The processing of highly enriched uranium was restricted to 300 g per batch because of the danger of criticality.

(b-2) In addition to the initial sol-gel process, a second stage process was evolved for preparing larger particles (∅ up to 1.2 mm) with a wide range of chemical compositions such as $UO_2$, $(U, Th)O_2$ and $(U, Pu)O_2$. It proved simple to incorporate burnable (neutron) poison (such as Eu and Gd) in the fuel.

The Pu part of the process was developed by Euratom (Karlsruhe) and later by the Netherlands Energy Research Foundation (ECN, Petten). The carbide variant for (Th, U) oxide was developed partly within the Dragon Project (Winfrith, Great Britain); the (U, Pu) carbide variant was carried out by Swiss Federal Institute for Reactor Research (Eidgenossisches Institut für Reaktorforschung, EIR) (Würenlingen, Switzerland). Fuel studies were also carried out with regard to the Homogeneous Test Reactor (HTR) on controlled incorporation of porosity into the fuel.

Several sol-gel variants have been developed (see, e.g., CNEN, 1968; IAEA, 1968, 1974; Wijmer, 1968). Unfortunately, because of industrial interest in the process, little has been published in the freely accessible literature; a relatively large amount of material is accessible only through the patent literature.

It appeared that the initial process described for external gelation was no longer usable for particles larger than roughly 20 μm. Gelation progressed too slowly and as a result, at least definitely under the prevailing flow conditions, the larger particles distorted and built up osmotic imbalances manifested in subsequent process stages (washing, drying, sintering) as cracks and disintegration of the fuel microsphere.

The solution to this problem was homogeneous generation of the gelling agent ($NH_3$) within the sol drop itself. To that end hexamethylene tetramine, an agent releasing ammonia at elevated temperature under the prevailing process conditions, was added beforehand to the specially cooled sol, thus bringing about internal gelling of the sol drop. It then proved possible to prepare particles of various chemical composition with diameters up to a few mm. It should be noted here that the process mentioned is to be considered as the internal gelation of a solution of hydrolysed uranyl nitrate.

This process allowed the preparation of several kilograms of $UO_2$ in particles of various sizes, with various degrees of enrichment and different amounts of burnable poison (Eu and Gd). The material was prepared in the KEMA laboratory for radiation experiments in the high-flux reactor at Petten and the heavy-water boiling water reactor (HBWR) in Halden. As favourable results were obtained, an additional 120 kg of $UO_2$ were made for three fuel elements of the Dodewaard boiling-water plant. Vibratory compaction and assembly of the elements was carried out in all cases by the Material Testing Department of the Netherlands Energy Research Foundation.

The results of this Vibrasol project were excellent and all rods met the operational specifications (burn-up of $20\ MWd \cdot kg^{-1}$). It was also demonstrated that the interaction between fuel and cladding was favourable in comparison with an equivalent pelletized element.

*Irradiation research* – The size range of fission products in (U,Th) oxide is 4-8 μm. As a result, the surface of the 5.5 μm particles is seriously damaged by fragments escaping during the fission process. That has important consequences for fuel used without cladding, as is the case in a suspension reactor. In view of this, it is obvious that the behaviour of the fuel in this context had to be examined by neutron irradiation outside the reactor.

The fuel for the radiation experiments consisted of 15% $UO_2$ (90% $^{235}UO_2$) and 85% $ThO_2$; the average sphere diameter was 5 μm. This mixed oxide was prepared by using the sol-gel process. After exposure to radiation, the fuel was treated with dilute nitric acid. The quantities of uranium and thorium found in the etching liquid were regarded as a measure of irradiation damage. In the present report the damage will be expressed in mg uranium per gram mixed oxide.

The next step was to investigate the damaged surface by examining photographs taken with an electron microscope and by using a replica method.

Experiments on irradiation at high temperatures of mixed-oxide particles in water were carried out in the BR-2 reactor in Mol (Belgium) with a neutron-flux density of $10^{14}\ cm^{-2} \cdot s^{-1}$. Research at high temperatures (523-583 K) is important in view of applicability of the fuel for use in a power reactor. The results are shown in Figure 9.

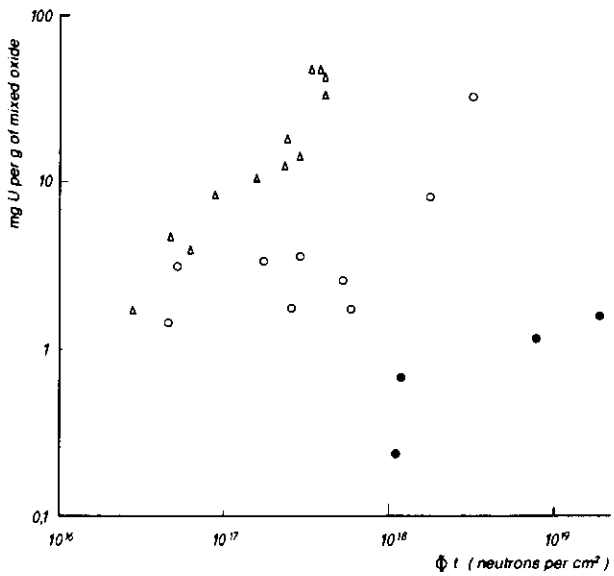



Fig. 9
Results of irradiation experiments at high temperature.
Black dots represent mixed oxide irradiated in water at 583 K; circles represent mixed oxide irradiated in water at 523 K; triangles represent mixed oxide irradiated in water at 353 K. Values expressed in mg U per g of mixed oxide.

Figures 10 and 11 give an impression of the irradiation damage at 523 and 583 K at neutron densities of $5 \times 10^{18}$ and $8 \times 10^{18}$ cm $^{2}$ respectively. Irradiation damage seemed to be less at higher temperatures. Some suspension irradiation at 583 K was also carried out at neutron densities higher than $10^{20}$ cm $^{2}$..

Electron micrographs showed that the mixed-oxide particles had completely disintegrated. At a neutron density of $1.1 \times 10^{20}$ cm $^{2}$ and at 583 K, 4.5 mg of uranium per gram of mixed oxide had apparently dissolved after irradiation due to etching; electron micrographs, however, showed completely disintegrated particles (Fig. 12). It certainly appears from this that etching methods give results differing from those obtained from electronmicrographs. The former give an indication of how inert the fuel particles still are in a chemical sense, whereas the latter give the shape and size of the particles.

There have been a number of publications (e.g. Rogers & Adam, 1962; Nilsson, 1966) concerning uranium ejecting from $UO_2$ surfaces due to fission fragments.

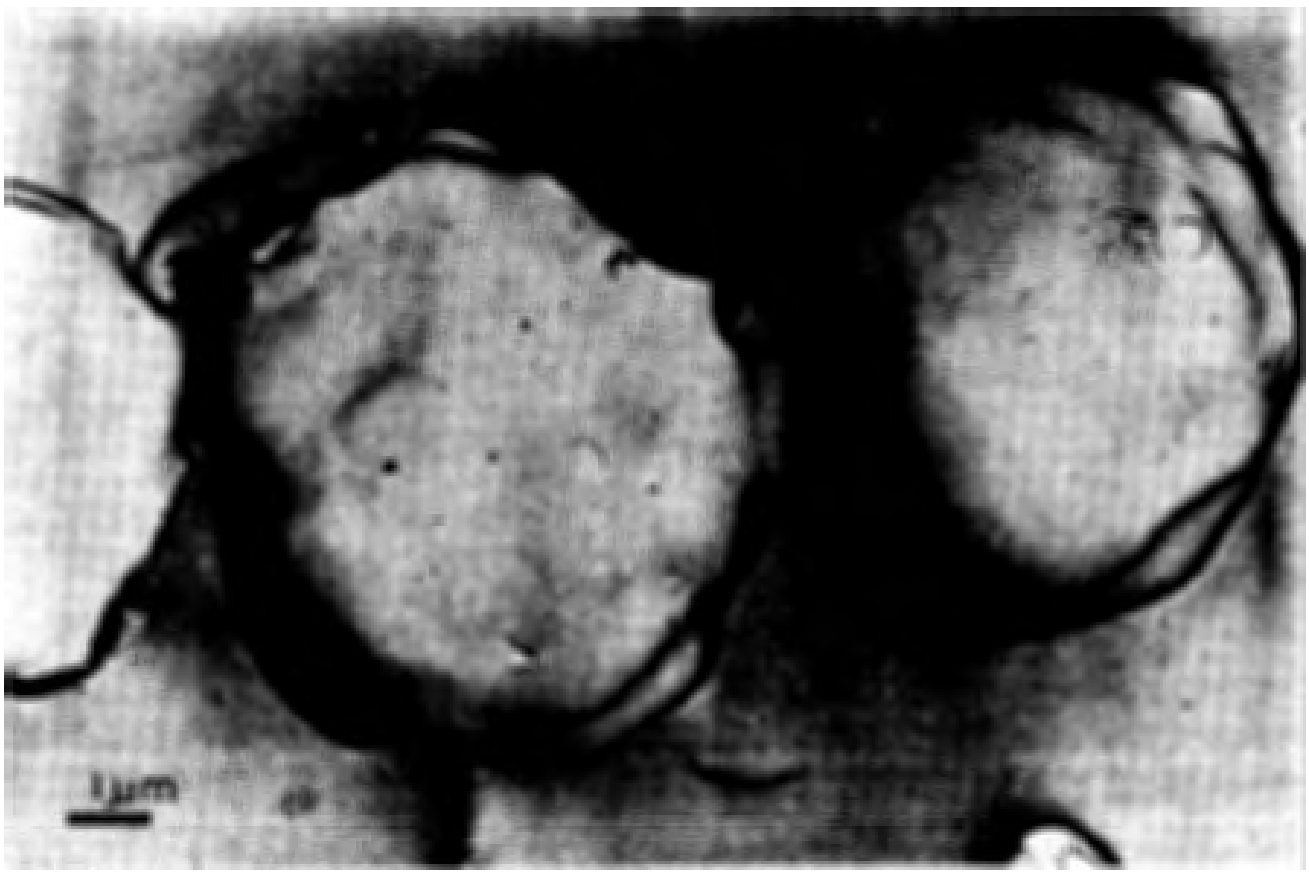



Fig. 10
Fuel particles irradiated at 523 K ($5 \times 10^{18}$ neutrons per cm$^{2}$).


Fig. 11
Fuel particles irradiated at 583 K ($8 \times 10^{18}$ neutrons per $cm^2$).

Fig. 12
Fuel particles irradiated at 583 K ($1.1 \times 10^{20}$ neutrons per $cm^2$).

KEMA also investigated this during the irradiation experiments. The uranium ejected during the KEMA project was gathered on a collector placed opposite a uranium oxide tablet. Uranium transport from the collector back to the uranium oxide surface was also investigated. Both ejection and transport took place during dry irradiation experiments. Because there is a dynamic equilibrium between both effects, the damage caused by these irradiations was approximately constant.

Such an equilibrium is not obtained in the case of irradiation in water. It was discovered that there was a large quantity of amorphous material around the irradiated fuel placed in water prior to etching. This amorphous material was dissolved during treatment with dilute nitric acid. The proportion of thorium and uranium thus dissolved is approximately equal to the proportion in the particles before irradiation. The matter ejected during irradiation forms a separate phase and the surface of the particles is not protected by this layer. For that reason additional matter is continuously ejected during irradiation. The results of irradiation in butanol and in 1.5. molar potassium iodide indicate that decomposition products of water ($H_2O_2$, H, OH and $HO_2$) also play an important part.

As already mentioned, little uranium is dissolved after irradiation at elevated temperatures. There are two possible explanations for this: (1) hydrogen peroxide is very stable at this temperature and (2) recrystallization may occur. A similar effect has also been discovered in ion implantation for germanium and silicon (Mayer et al., 1970). However, the fuel had entirely disintegrated during irradiation at 583 K, although it hardly dissolved afterwards when treated with nitric acid. Specific attack of the particle surfaces also probably played a part.

*Sedimentation behaviour* – In a nuclear reactor where fission takes place in suspended fuel particles, the settling rate of those particles is one of the most important factors. Due to the large difference in density between particles ($\rho = 10$) and liquid ($\rho = 0.8$), the sedimentation rate (settling rate) is relatively high.

In the case of the KSTR, sedimentation, or rather the movement of suspended solid fuel particles in relation to the water, could influence the following.

(1) The critical concentration: due to the relatively small upward flow in the reactor vessel (average approx. 12 cm.s$^{-1}$ in the centre) in relation to the settling rate (max. 3 cm.s$^{-1}$), the average fuel concentration in the reactor vessel is higher than in the rest of the circuit. That phenomenon is termed the 'hold-up'. It will be obvious that the hold-up becomes less at an increased circulation rate and at a lower settling rate. The latter can be influenced slightly by changing the temperature. Variations in hold-up due to sudden changes in the flow pattern were the cause of stepwise flux or temperature changes in the case of the KSTR.
(2) The disappearance of fuel: fuel may become concentrated in system parts with low flow rates due to sedimentation. In the case of the KSTR this normally occurred in crevices such as those in flanges.
(3) The representativeness of the samples: due to movement of particles in relation to the liquid in the bends, the concentration was usually higher at the outside than at the inside radius. This led to homogenizers being installed upstream of the sampling points.
(4) Erosion: due to the centrifugal effect described above, erosion was greater in bent and branched than in straight pipes.

The colloid-chemical properties of the suspension which determine the magnitude of the interacting attractive and repulsive forces between the particles are of specially great importance: if the attractive forces predominate, flocculation of the particles occurs and a flocculated suspension has a much higher settling rate than a non-flocculated suspension. The sizes of flocs are determined by the attractive forces between the particles and by the shear forces resulting from settling and/or flow. The shear forces are greater for large than for small particles during sedimentation. The difference between the settling rates of homogenized flocculated and non-flocculated non-flowing suspensions is consequently relatively smaller for large than for small particles. For example, the colloid-chemical properties are irrelevant for the sedimentation rate in the case of suspensions of coarse sand, but are relevant for clay suspensions.

The Van der Waals attractive forces between the particles themselves are always present; the repulsive forces originate at the surface of the particles from electric charges of equal sign. The charges may develop by ionizing or by adsorption of ions such as $H^+$, $OH^-$, $P_2O_7^{2-}$, $Fe^{3+}$, etc. The repulsive forces decrease when salts are dissolved in the water. The magnitude of the recoil effect in the KSTR is largely determined by whether or not a suspension in the KSTR is flocculated.

The sediments of flocculated and non-flocculated suspensions may differ rather considerably. The former are often of looser construction and therefore can often be more easily resuspended.

Measuring instruments had to be developed for analysing sedimentation behaviour. Sedimentation phenom-

ena of suspensions are best examined in vertical cylindrical tubes with samples that are well homogenized beforehand by shaking or by pumped circulation. The gradient of the particle concentration in a thin layer parallel to the surface is then recorded in the stoppered tube.

A sedimentometer which can give the particle distribution directly on a linear scale of quantities as well as on a linear scale of particle sizes has been developed at KEMA and is now patented. The latter reading was achieved by changing the depth of measuring during measurement. The sedimentometer is shown in Figure 13. The concentration is measured by means of X-rays. The amount of radiation transmitted by the suspension is compared continuously with the radiation transmitted by a wedge shaped body. If there is a difference, the wedge is moved by the recorder motor in such a way that the transmissions become equal. The position of the wedge, which is linked to the pen of the recorder, is thus a measure of the concentration. In later versions both the wedge and the suspension were placed in one single beam. Differences in sensitivity of the receivers thus no longer had any influence. Another version has been built with a beam interruptor, in which one receiver suffices. In all cases the X-ray tube was an AC device. The differential amplifier also amplified only AC pulses. This had the advantage that radiation originating from radioactive suspensions did not cause interference. Consequently, a sedimentometer could be installed in the hot cell as well. Particle-size distribution could also be determined in very fine suspensions by measuring the concentration as a function of the elevation in a homogeneous suspension centrifuged for a given period.

The addition of sodium pyrophosphate, hydrochloric acid or alkali was usually sufficient to counteract flocculation of suspensions for particle analysis. The particles were occasionally magnetized or water repellent, in which case deflocculation was virtually impossible.

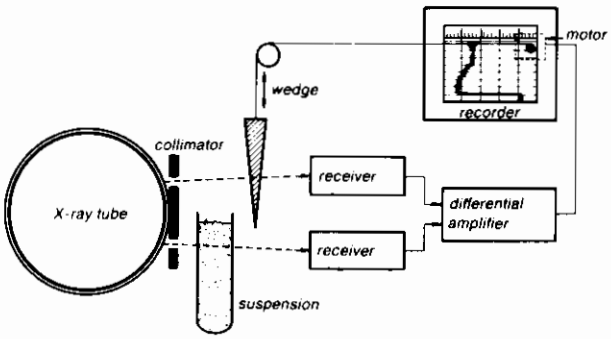



Fig. 13
Schematic presentation of the sedimentometer.

A major argument in favour of using X-rays for concentration measurements was that they permit measurement in metallic tubes as well. As a result the sedimentation rates of suspension samples from test loops and from the KSTR could be measured at the operating temperatures of those installations. Tubular autoclaves of some 60 cm length were built for the purpose. They were shaken by turning in a special device until immediately prior to measurement. The extended length was necessary because the sedimentation rate at the operating temperature (up to approx. 583 K) could be high ($3\ cm \cdot s^{-1}$). Ideal homogenizing could, however, no longer be achieved in such cases.

Measurements of hold-up and recoil effect were carried out in test arrangements and in the KSTR, but the data obtained on the magnitude of slip are difficult to interpret. It could nevertheless be inferred from the data that the suspensions were virtually always flocculated.

Particle-size analysis was an important aid in indicating whether the correct particle size for fuel preparation was being achieved. KSTR samples showed the following results:

(1) original material: 84% < 8.4 μm; 50% < 5.6 μm; 16% < 3.7 μm;
(2) after 134 operating hours: 84% < 7.0 μm; 50% < 5.1 μm; 16% < 3.6 μm.

The reduction in particle size was mainly due to breaking of particles which had been sintered together during preparation. At the start of the critical operation so many erosion products had deposited on the particles that these first had to be cleaned with HCl to exclude flocculation of the particles altogether. Following this the result was:

(3) after start of critical operation: 84% < 7.0 μm; 50% < 5.0 μm; 16% < 3.3 μm.

This treatment was no longer necessary at the end of the critical operation, perhaps because the magnetized matter had been converted. The result was then:

(4) at the end of critical operation: 84% < 6.2 μm; 50% < 4.2 μm; 6% < 2.7 μm.

*Oxygen formation and recombination* – The principle of a suspension reactor is that the fission fragments, which are rich in energy, transfer their energy to the water by direct contact. A small portion of the water is decomposed during this energy transfer, forming hydrogen and oxygen (the so-called radiolysis). The KSTR is operated with a gaseous atmosphere of hydrogen; oxygen/hydrogen mixtures are explosive over a wide range of mixing ratios. Much work has been done over the

years to recognize and solve those problems that might affect the safe operation of the KSTR.

When oxygen gas develops, the oxygen/hydrogen ratio in the gas system will increase to form a mixture that might cause explosions. As such gas mixtures – not entirely correctly called detonating gases – should not be allowed to form, the oxygen must be removed. This can be done by recombination with hydrogen which is available in excess. A recombiner was thus developed. It had a dual purpose: recombination of the oxygen and determination of the quantity of oxygen generated based on the heat released in the recombination reaction.

In view of the problems the Oak Ridge National Laboratory (ORNL) had experienced with the recombiners developed there, it was decided to develop a completely new recombiner at KEMA. A system was designed for the purpose, allowing oxygen to be added in discrete doses to a flow of hydrogen. Such a system should make it possible to test various recombination catalysts, to develop measuring equipment for oxyhydrogen production and monitoring of the explosion limit, and finally to work out the mechanical design of the reactor recombiner. Special measuring cells were developed to measure oxygen concentrations ranging from less than 1 ppm up to a few per cent of $O_2$ in $H_2$ (Kuypers & Ruiter, 1968).

Platinum metals were selected for the catalyst to be used in the recombiner mainly because of the hydrogen atmosphere and the low temperature (approx. 355 K). After various experiments with both Pt and Pd with and without carrier, Pd on an $Al_2O_3$ carrier was selected. The advantage of Pd over Pt is that at low temperatures it performs better than Pt; one disadvantage is the lower so-called Tamman temperature, at which the specific active area decreases substantially. Temperature alarms were therefore installed in the recombiner. Thermocouples were also placed in the catalyst bed to determine where in the bed recombination occurred. The action of the catalyst can be nullified due to poisoning by, for example, iodine which occurs as fission gas. For that reason among others the catalyst bed was well oversized.

During experiments the recombiner showed starting problems in the presence of moisture. An electric heater was therefore installed ahead of the recombiner so that the gas would enter the recombiner unsaturated by aqueous vapour. Following this alteration the recombiner converted oxygen completely even during long-term tests and under varying circumstances. It appeared that, by measuring the temperature rise through the recombiner, high (a few per cent) to very low (a few ppm) $O_2$ concentrations could be determined accurately. It also appeared that both the measurement of gas temperatures and the thermal insulation had to satisfy stringent requirements. It was necessary to develop a flow meter for measuring $H_2O$ production. The product of flow and oxygen concentration gives the converted amount of oxygen and is thus a measure of the production of $H_2O$. Flow was measured by heating the gas electrically by a few degrees centigrade. The electric energy input was measured accurately. The energy released in the recombiner could be calculated from the temperature rise through the recombiner, as could therefore also the amount of oxygen converted. The method proved to be perfect in the test arrangement as well as in the reactor itself.

The measurement of the temperature difference through the recombiner was used to monitor the explosion limits. The maximum allowable oxygen concentration was set at 2%. Were it to appear that radiolysis in the reactor was higher than the equivalent of this value, the power of the reactor would have to be decreased. A second recombiner was installed in series with the first for additional safety. Both recombiners could be tested and calibrated under operational reactor conditions with oxygen supplied externally.

*Removal of fission products from the suspension reactor* – Removal of fission products may be realized by (1) utilization of the recoil effect, (2) adsorption of fission products, (3) the use of a second adsorbent, and (4) the use of active carbon.

(1) If the diameter of the suspended fuel particles is less than the range of the fission products in the material of the particle, all fission products will leave the particle and come to rest in the surrounding water or within a fuel particle in their proximity (Hermans & Van der Plas, 1957). There is a high probability that a fission product will come to rest in an adjacent particle. If the suspended particles were (and remain) homogeneously distributed, virtually all fission products would come to rest in the water, at least with suspensions at the concentrations used in actual practice. In a flocculated suspension the particles form aggregates of much larger diameter than a single particle. The particles within those aggregates are close together.

Determinations (Markestein & Van der Plas, 1962) during the subcritical experiment (see section above 'Subcritical experiment') showed that approximately 40% of the fission products were found within the particles. The formation of aggregates apparently predominated. The homogeneously distributed state could be adjusted approximately by changing the pH of the suspension; the above percentage then decreased to 20%.

The fraction of fission products finally present in the water could – in principle – be removed without deterio-

ration of the solid particles. Fission products, just as corrosion products, contribute substantially to the parasitic absorption of neutrons that occurs progressively in a nuclear reactor. It is thus important to remove these products if the best possible neutron economy is to be attained.

(2) The removal of fission gases is relatively easy as was discussed by Kreijger (1962). Most of tbe remaining fission products are isotopes of elements which will occur as oxides or hydroxides in the aqueous phase at approx. 525 K, the environment considered here. It has been found that those materials attach themselves to the surface of the fuel particles in this environment. The portion present in the liquid phase is small. Only strong chemical agents seem able to dissolve the attached materials.

This phenomenon prevents the separation of the fuel and the fission products from being achieved by simple means ('simple means' is to be understood here as a mechanical separation method, for example with the aid of hydroclones which largely separate the liquid from the solids).

(3) The following has been suggested as a way of reaching the intended goal (Hermans & Van der Plas, 1957; USP, 1962). As the adsorption mentioned above is not specific, attachment to the fuel can be prevented by adding another solid with such particle size and at such a concentration that the materials attach primarily to that solid. It is then essential that the material can be separated from the fuel by simple means, e.g. by passing through a hydroclone. The action of the added material, called a second adsorbent (the fuel itself is considered the first adsorbent), can best be imagined as a homogeneous distribution of both solid materials. The second absorbent can be regarded as a screen preventing the fission products from reaching the fuel particles.

When the fuel particles are aggregated, the only active portion of the second adsorbent may be that present within the aggregates. The requirements for the second adsorbent and proposals for materials to be used for the purpose have been discussed by Van der Plas (1968). Active carbon and fine thorium oxide, possibly even the finely distributed fuel itself seemed to offer the best chances of success. 'Fine' implies a particle size small in relation to that of the fuel itself, hence small in relation to 5 $\mu$m.

(4) The possibility of using active carbon as a second adsorbent raises two problems: the nature of a suitable carbon and the stability of that carbon in an operating reactor. Both problems have been discussed at length (Van der Plas, 1968); this author made the following points:

(a) as far as the nature is concerned: a carbon which allows good dispersion in water has a surface mainly covered with groups containing oxygen; those groups can be introduced by using several types of oxidizing processes, including careful oxidizing with air; the number of those groups decreases in water at 525 K but the carbon remains dispersable;
(b) the stability of carbon under irradiation conditions was examined by irradiating carbon suspensions in a reactor without fuel; the overall result was that carbon was oxidized to carbon dioxide by the oxidizing products of water radiolysis while hydrogen was released.
(c) if it is assumed, as an estimate, that the rate of this oxidation is proportional to the absorption of energy in the water, then it appears that in an operating reactor this rate is too high; the carbon is 'burned' too fast. Active carbon thus does not appear to be suitable as a 'second adsorbent'.

*Technological research and development work*

The special nature of the medium (fuel in the form of a suspension of $ThO_2/UO_2$ particles in water) and the question of safe and reliable operation under radioactive conditions made technological study and development work necessary. The objectives were:
(1) to provide the required data for the design and manufacturing specifications;
(2) to indicate the technical possibilities and limits for meeting the requirements set by the design.

*Methods* – In many cases development work had to be subdivided into a number of phases. For problems involving the flow aspects of the suspension the following phases were judged preferable:
(1) an L.T.P. (low temperature and pressure) phase covering visual evaluation or testing of certain measuring methods in actual use;
(2) an H.T.P. (high temperature and pressure) phase under conditions comparable to those in the reactor, mostly involving testing and evaluation of prototypes;
(3) functional testing of the final components and constructions in the reactor systems.

Components for which the flow aspects had no relevance, were developed in the course of endurance or functional tests of the prototypes. The final assessment was made during the functional test of the complete reactor systems.

The medium was modelled in the H.T.P. tests by particles composed of mixed oxide (85% $ThO_2$ and 15% $UO_2$) of the same size (5 $\mu$m) as was to be used in the

reactor, but not enriched. Some tests were done with model materials for which the settling rate was selected as the critical parameter. One difficulty in this case was that other factors having an effect with aqueous suspensions would not be entirely excluded (flocculation phenomena, interactive influences).

*Erosion* – The target of the research on erosion was to obtain data for the design and manufacture of suspension handling systems such as process margins, shapes and sizes, that would sufficiently guarantee that the integrity of the primary system (the first containment of the suspension) would be maintained during the entire period of the reactor's operation. That operating period was arbitrarily set at 10,000 hours of operation with the suspension.

The research concerning erosion partly had an experimental character and partly was directed at the erosion processes that occurred during the operational period of the KSTR.

Experimental testing of components had to be done under supervision of KEMA. That was necessary since, due to the special nature of the medium (5 $\mu$m, $ThO_2/UO_2$ particles in water) and the conditions (523 K, $6 \times 10^6$ Pa, concentration 400 $g.l^{-1}$), neither literature nor the manufacturer could provide the required experimental data.

The tests were carried out in the test circuits for suspension just mentioned under conditions analogous to those to be expected in the reactor, apart from radiation. The experiments carried out could be classified roughly into three groups that will be dealt with in the following paragraphs:

(1) erosion by 'undisturbed' flow, assumed to occur in straight pipes and bends;
(2) erosion by a 'disturbed' flow pattern such as in the centrifugal circulation pump;
(3) erosion in shut-off devices (suspension valves).

(1) It is assumed that with an 'undisturbed' flow pattern the laminar sub-layer of the flow offers some protection of the wall against particles moving in its direction. The thickness of the boundary layer depends on e.g. temperature and flow rate. At a specific, critical value, some of the particles will no longer be sufficiently decelerated in the boundary layer and will therefore reach the wall with a certain impulse and angle of incidence. That may result in damage. The wall material then becomes the determining factor for the degree of erosion. No measurable erosion occurs at rates below the critical value.

It could be concluded from the test results that the critical rate was approx. 8 $m \cdot s^{-1}$ for 5 $\mu$m $ThO_2/UO_2$ at 273 K and 400 $g \cdot l^{-1}$. Based on those data, 5 $m \cdot s^{-1}$ was maintained in the design for the flow rate in pipes and bends.

The many test-circuit operating hours have shown that no measurable wear occurred in straight pipes and similar flow arrangements.

(2) Basically the same mechanism as mentioned in the above paragraph applies to the 'disturbed' flow pattern. The flow rates occurring and the influence of turbulence upon the walls are so great that the boundary layer offers insufficient protection and the wall is damaged.

To investigate the degree of erosion in such a case and to reduce it if possible, four centrifugal, canned rotor pumps of different makes and shapes were compared in the test circuits in endurance tests. The degree of erosion could be established qualitatively by visual evaluation of the pump casing and the impeller after specific numbers of operating hours. Quantitative data were obtained from the increase in the amount of erosion products in the 'fuel' samples taken from the flowing suspension. It appeared that all pumps tested initially showed an unacceptably high degree of erosion, already after 120 hours of operation, but that the erosion could be reduced by a factor of 10-25 by a better adaptation of the shape of the pump casing and the impeller to the flow.

Specifications were arrived at for the design of the final version of the pump. An operating lifetime of 5,000 hours was aimed at. The pump might then have to be replaced by an identical spare only once during the 10,000 hours of reactor operation planned.

Figure 14, showing the erosion in a pump, illustrates the improvement in the operating lifetime.

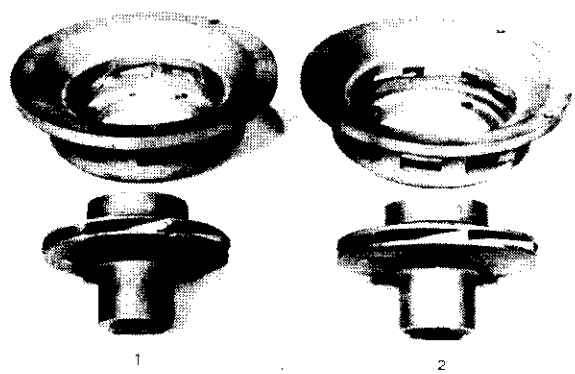



Fig. 14
Erosion damage to test pump.
1: before modification after 120 hours; 2: after modification and after 1000 hours.

(3) Although the use of shut-off devices in suspension systems was avoided as far as possible, their use was unavoidable in the concentration control system (10 mm bore) and in the dump system (25 mm bore).

Two effects arise in shut-off devices for erosive suspensions, i.e. (a) the erosion of valve body and seat by the suspension flowing past them in the open position, and (b) damage to valve body and seat by suspension particles trapped on closing of the valve. Both effects will greatly influence the decrease in tightness during use. As no further improvement could be achieved by adapting the conical shape of the valve body and seat to the flow, it was necessary to find which combination of materials for valve body and seat would be the most suitable for satisfying the long-term tightness requirement, namely 1 $cm^3 \cdot h^{-1}$ at a differential pressure of 500 kPa. The two effects mentioned above could be separated by using a suitable testing method, and thus be examined for each combination of materials.

The result of testing four combinations of materials was that, after an initial selection at 293 K with clean water and with suspensions, only $Al_2O_3/Al_2O_3$ and WC/WC (sintered tungsten carbide) were found suitable. A second selection at 523 K showed that the $Al_2O_3/Al_2O_3$ combination was not acceptable any more due to leaching of the binder.

The WC/WC combination remained acceptable even at 523 K. The quantitative results were a leakage of 3 $cm^3 \cdot h^{-1}$ at 5 $kg.cm^{-2}$ differential pressure after 750 hours in the test series, 0.5 $cm^3 \cdot h^{-1}$ at 5 $kg \cdot cm^{-2}$ after 300 hours and 1800 closing cycles in a functional test with the final design. In addition it appeared that the damage caused by trapped particles during closing had more influence on tightness than had damage due to suspension flowing by. Alignment and centering of the composing parts of the valves are essential conditions for good tightness as became obvious during research on the initial prototypes in which that condition was not fulfilled by the usual manufacturing tolerances.

*Erosion during operation* of the KSTR was evaluated on the basis of both the suspension samples and the plant itself.

Although the components for and the layout of the suspension systems had been optimized as far as erosion was concerned on the basis of the experience with the test circuits, it was necessary to monitor the degree of erosion in the KSTR itself. That was done by regular measurement of the increase in the amount of erosion-corrosion products, i.e. Fe, Cr and Ni, in the samples taken from the main system during operation. It was thus assessed whether continuation of the operation was justified (Fig. 15). The erosion could be localized more precisely because three specific materials each characterized three areas of the main system, i.e. Cr for the pump casing and impeller, Ni for the heat exchanger, and Fe for pipes and other components.

Through analysis of the local conditions twenty suspect locations which could possibly be affected by erosion were selected and arranged in their sequence of risk. An ultrasonic sensor was installed at the most suspect location and was used for making five measurements after specific periods of operation to detect any reduction in wall thickness. Up to the start of high power operation (approx. 2,300 hours with suspension) no measurable reduction of the wall thickness was found at that location. Other locations were checked regularly and as long as possible (up to the start of high-power operation) by means of X-ray photographs of the walls.

*Development of components* – This section will only touch on those components to be developed for which specific requirements were applicable in the context of suspension being used in an environment with ionizing radiation. These components are therefore primarily those in the main system, particularly in the M cell.

Specifications of the requirements and the result of the development work are described below. Most components were developed with the aid of the low- and high-temperature test circuits already mentioned.

The components required for the KSTR suspension operation were as follows:

(a) MR-1 (reactor vessel),
(b) MP-1 (main circulation pump),
(c) ME-1 (main heat exchanger),
(d) MV-10 (storage vessel),
(e) MV-3 (dump vessel and agitator),

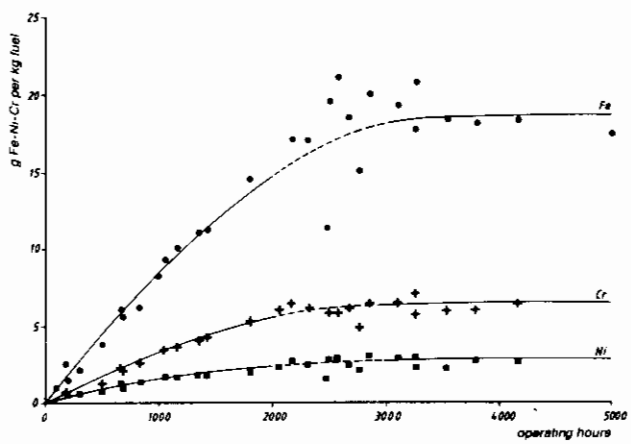



Fig. 15
Progress of erosion-corrosion products as a function of the number of operating hours with suspension.

(f) connecting elements,
(g) sampling equipment,
(h) MJ-2 gas/liquid contactor and MJ-1 gas/liquid separator, and
(i) cell closure.

(a) The process engineering development work for the MR-1 reactor vessel mainly involved the control of the flow pattern of the flowing suspension. The major requirements for the flow pattern were: (1) adequately stable flux distribution and nuclear reactivity, (2) the absence of large differences of the particles' time of residence in the various zones of the vessel, (3) avoiding deposition of particles on the walls by means of sufficient local flow rates, and (4) avoiding local boiling.

A first, provisional selection was made of the shape of the vessel and the flow distributor based on tests at 293 K on models with various inlet pieces and with sand used as a model material. The choice fell on an oval shaped vessel with an axial flow from bottom to top on which a slight rotation was imposed.

The flow pattern (i.e. axially symmetrical with vertical position of the vortex core) appeared to be very sensitive to the position of the flow distributor (concentric and perpendicular) and to the correct position of the straight section of the supply pipe (in the plane of the centre line of the vessel and with a length of more than ten times the pipe diameter). Those effects were verified step by step during the manufacturing phases of the final vessel and corrections were made as required.

After completion of the pilot study, precise measurements were carried out on the model selected in order to determine more accurately the speed distribution (e.g. by using microreels) and residence times (through the response of injected salt solutions). Those measurements were carried out by SOGREAH in Grenoble. The shape of the model selected initially had not to be changed drastically as a result of this study.

Subsequently a measurement was carried out of the concentration distribution of a flowing $ThO_2/UO_2$ suspension in the vessel at a temperature of 523 K by means of gamma-absorption methods. It became obvious that the flow pattern was sensitive to the shape of the vessel itself (very divergent inlet section) and to any minor deviations in the positioning of the inlet piece. Finally, after initial subcritical measurements in the KSTR, research was carried out on the short-time changes in flow pattern. It appeared that changes in flux occurred with excessive amplitudes (2% $\Delta k/k$). Installing a stabilizer tube in the centre of the reactor vessel made it possible to reduce the amplitude to 0.5% $\Delta k/k$.

In addition to doing flow tests for finding the correct shape of the stabilizer tube, special measures were required for guiding in and fixing the stabilizer tube in the vessel so that it could be mounted remotely controlled. The final reactor vessel is shown in Figure 16. The manufacturer made two prototypes of the reactor vessel according to current manufacturing methods. Various tests were carried out on those vessels, some at an internal pressure of 5.5 MPa.

(b) Apart from studies on erosion control carried out on the main circulation pump MP-1, it was necessary to check whether the motor space of the vertical centrifugal pump MP-1 could be kept free of particles. Due to the tightness requirement, a canned type pump was

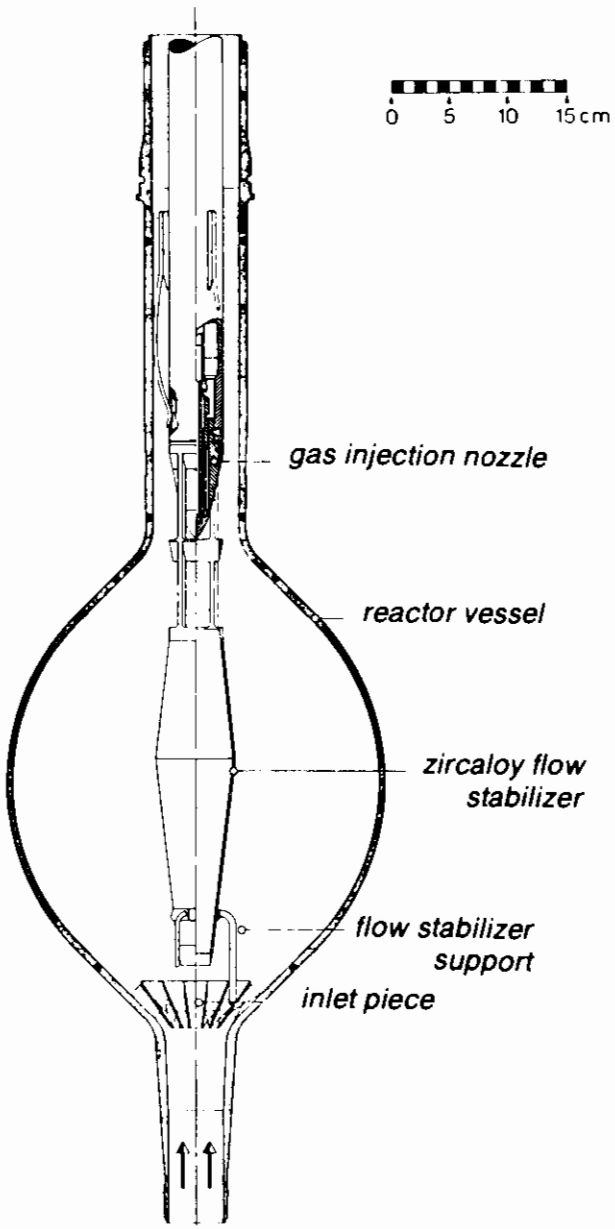



Fig. 16
Reactor vessel with inlet piece and flow stabilizer.

chosen in which the rotor bearings moved in the medium to be pumped. It was a condition for correct operation, however, that the bearings be lubricated and cooled in particle-free water and that any gas collected could be removed.

Examination of, and experience gained with the four test pumps showed that keeping free of particles could be achieved by allowing enough flushing water into the motor space (approx. $10\ l \cdot h^{-1}$) and flushing this partly through the connecting channel to the pump space ($9\ l \cdot h^{-1}$). The balance, carrying any gas with it, was discharged at the top.

(c) The main heat exchanger ME-1 was compact and comprised three concentric tubes with only the intermediate annular space carrying suspension and being without branches.

Apart from applying the endurance tests mentioned before for possible erosion of inlet and outlet pieces, an empirical check was made on a prototype to find the operating conditions permissible with the design selected. The check included: a thermal test using steam at 411 K and 280 thermal cycles, a leak test to measure any leak between the three spaces, and a destructive test using pressurization of each of the three spaces.

(d) The entire stock of fuel (approx. 30 kg) could be stored (trapped) in the MV-10 storage vessel; with the aid of a hydroclone the fuel could be re-flushed into the main circulation system. This arrangement also permitted the concentration in the reactor vessel to be set at a specific value. The requirement for these activities was a reasonably fast and reproducible manoeuvering of the fuel, allowing even small quantities to be flushed in.

Weighing equipment had to be developed for recording the fuel balance. This equipment was required to allow reproducible weighing of the amount of particles in the vessel at both high and low temperature with reasonable accuracy (0.3%). It was to be permanently linked to the vessel in the system. A special shape of supply and discharge pipes (square layout) and the use of strain gauges provided a satisfactory solution.

(e) The purpose of the MV-3 dump vessel was to permit fast draining of all suspension from the reactor vessel and from the main system to a nuclear safe and thus, under all circumstances, subcritical vessel, especially when emergency situations would necessitate it. The following equipment was developed for this purpose.

(1) A cooling and heating system separated from the vessel contents by a double barrier. The need for fast switching from heating (operating position) to cooling (emergency position) was met by a cast aluminium cooling (heating) coil around the dump vessel.
(2) Suitable geometry and selection of the neutron-absorption medium. The flow pattern required to prevent particles settling on the bottom imposed restrictions in geometry that resulted in a cylindrical absorption unit being built with radially arranged baffles. A silver-indium-cadmium alloy was selected as absorption medium, the prime consideration being that the solid state of the material be suitable for providing internal support to the walls of the absorption unit against the prevailing high external pressure (max. 6.5 MPa). The nuclear value of the geometry selected was verified by both calculations and measurements of the source multiplication on the actual construction. Measurements showed a maximum k-factor of 0.7, which was considered sufficiently safe.
(3) An agitator intended to prevent locally high wall temperatures due to settled particles sticking to the bottom with insufficient cooling. The drive of the agitator was, if possible, not to pass through the wall of the vessel. An agitator was installed which moved up and down driven by a solenoid around a top extension of the vessel. Due to a suitable combination of the shapes of the bottom and of the agitator (conical), it proved possible, with an acceptable stroke (5 cm) and frequency (60 strokes per minute), to keep the suspension off the bottom for as long as was necessary to prevent overheating.
(4) A weighing device for the fuel balance. Its casing was similar to that of the device for the concentration control vessel, although the weight of the vessel itself was greater and the supply pipes had a thicker wall. An accuracy of 4% was ultimately achieved.

(f) Although welded connections were used as much as possible when installing nuclear systems, flanges were still needed at some locations for connecting replaceable components because of maintenance and/or inspection. The leak tightness requirement per connection in the nuclear systems (primary containment) was less than $10^{-7}$ N $cm^3$ of helium per second at a pressure difference of 6 MPa. In addition, detachable connections had to be provided with leak-prevention and leak-detection facilities.

Various types of flanged connections were considered. The so-called oval type ring joint connection, which also gave good results at ORNL, was finally selected. Both flanges and bolts are highly stressed in that type of connection. Thus, to achieve acceptable pressures for the required leak tightness, the rings had to be silver-plated and the bolts conditioned by stressing them for some time prior to installation.

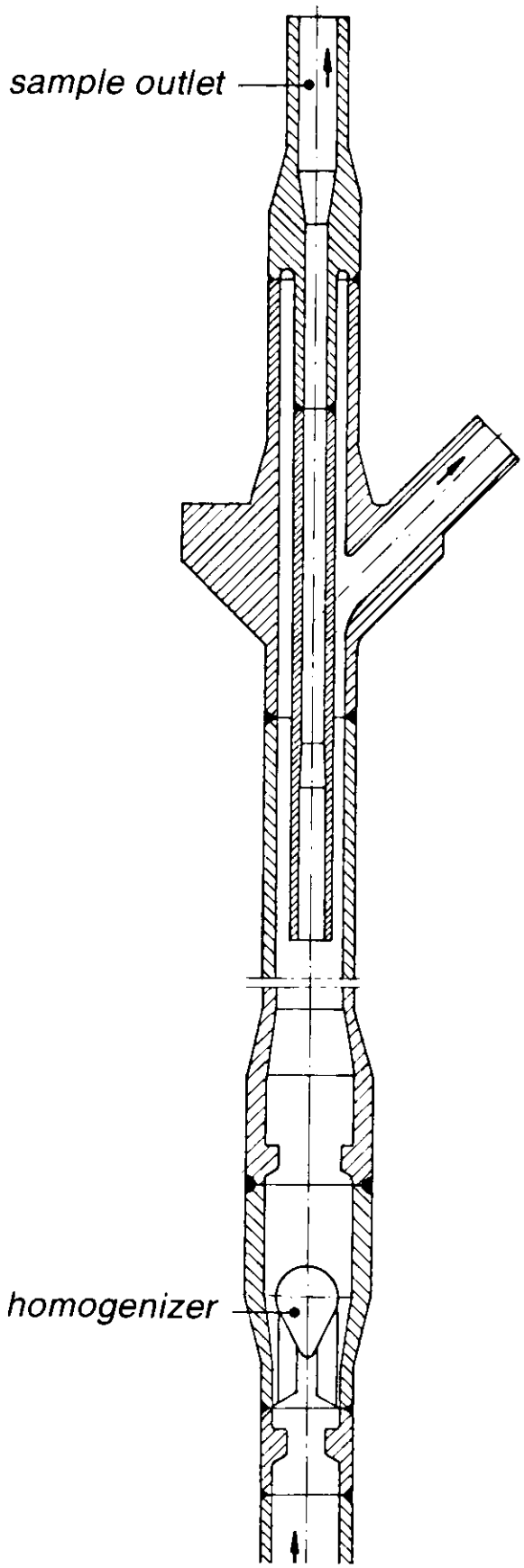



Fig. 17
Sampling point with homogenizer.

(g) The prevailing principle for the actual sampling equipment has always been isolation and enclosure of part of the flowing suspension in a bypass from the main circuit in an isolating chamber, letting a small portion of the suspension flow from there into a 15 $cm^3$ sampling vessel which is under vacuum. Experience with prototypes in the test circuits led to the development of a device which still had a variation of 20% in solids concentration. Although not perfect, the model was acceptable in practice for operational and experimental purposes as it allowed interpretation of the data from samples. It was, however, very important that the prescribed procedure for sampling be strictly adhered to.

For sampling with this model, the suspension was taken out through two Pitot tubes installed in series in the centre of the pipe. The first was to branch off from the main flow, the second to branch off to the sampler. For the sample to be representative, the flows in both Pitot tubes must be isokinetic. It proved necessary to install a homogenizer upstream of each Pitot tube to homogenize the suspension as uniformly as possible over the section of the tube (Fig. 17). The shape of the homogenizer had to be such that the turbulence generated for correct mixing would not lead to unacceptable erosion of the tube walls.

(h) An important group of components for correct operation during the experiments was the combination of the MJ-2 gas/liquid contactor and the MJ-1 gas/liquid separator. Their function was to inject hydrogen into the suspension flow directly above the exit of the reactor vessel and to again separate the gaseous products from the liquid after a certain period of contact. The aim of this part of the procedure was to remove radiolytic gas and gaseous fission products, and produce clean, particle-free water from the suspension by condensation of the vapour carried along.

The most important requirements for the MJ-2 gas/liquid contactor were a maximum bubble size of 3 mm for the gas dispersed in the suspension and a minimum period of one second for contact with the liquid. Examination of the possibilities resulted in a nozzle design with an adjustable inlet gap through which the gas discharged radially from the nozzle. The amount of gas was controlled by changing the gap width using thermal displacement. The required time of contact was achieved by having a contact length of approx. 2 m.

The most important requirements for the MJ-1 gas/liquid separator were that the discharging liquid had to be free of gas bubbles (less than 0.5%) in view of possible disturbances in the reactor's nuclear behaviour due to excessive gas in the reactor vessel, and that the discharged gas had to be free (less than 10 mg solids

per hour) from suspended solid particles. After extensive development work, a satisfactory solution was found in the use of a cyclone type separator (see Fig. 18) with tangential gas/liquid mixture inlet and axial discharge of liquid at the bottom and of gas at the top. Special facilities were installed to straighten the rotating liquid flow in the discharge by using vanes and to prevent droplets from being carried along in the gas discharge. The separating effect was greater than 99.9%. There was some carry-over detectable in the prototype but it remained within the requirement set. Due to the presence of an open level, the consistency and adjustability of the level was an additional important aspect in view of the requirement that the main circuit contents be as constant as possible. The design had to be further developed to permit measurement of the open level, shaped like a paraboloid by its rotation, in the high-pressure vessel.

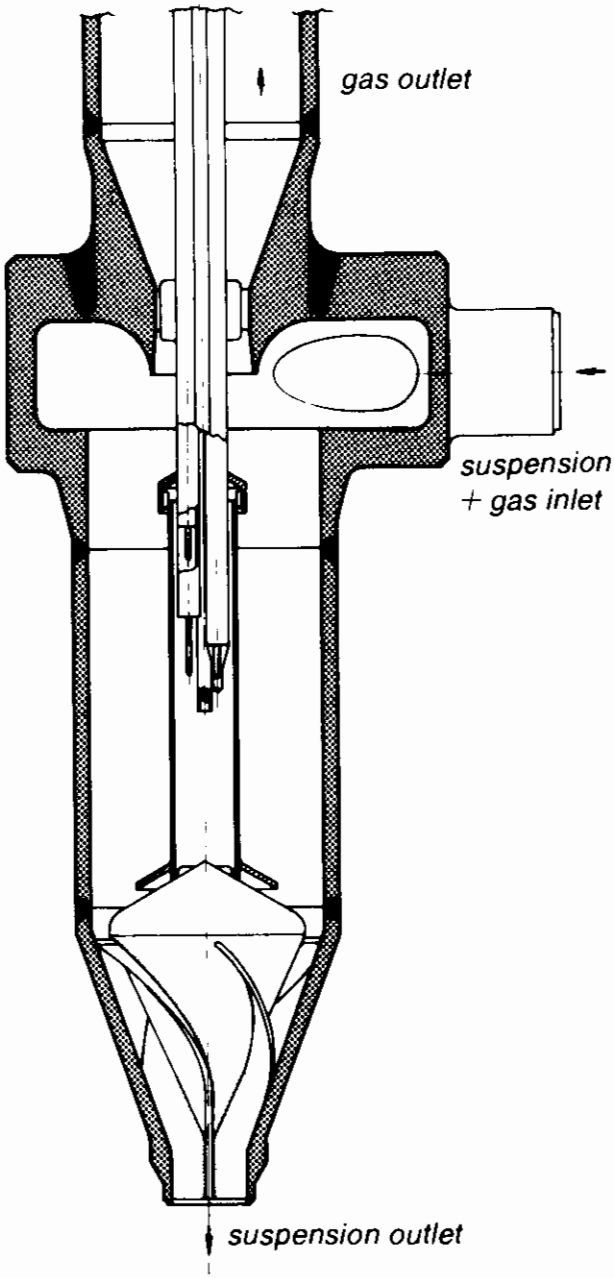



Fig. 18
Gas separator MJ-1.

(i) Gas-tight closure of cells M, G, R and Y (into which the fuel can spread in case of leakage or bursting of the primary containment), although not directly involved with suspension operations, was of major importance for safety. Welded plate constructions (so-called sealing pans) which would be welded in prior to the start of critical operation were first considered. That plan was changed at a relatively late stage, i.e. after the initial subcritical tests, and it was decided to install a more easily removable closure without welded connections but with flexible seals. The cells could be opened more often with that type of closure (e.g. for activity measurements or repairs) and the quality of the seal would be better maintained. There was no certainty that the leakage requirements for critical operation could be achieved by this method. The requirement was a maximum leak of 6000 N $cm^3$ of nitrogen per hour at an effective pressure of 20 kPa in the cells. This leak could occur in the case of a so-called design basis accident, so development of the seal required detailed research into the properties of the components to be used (such as compression and ageing of the rubber to be applied).

Research resulted in a design consisting of a circumferential square rubber strip (50 x 51 mm) and a rubber sheet (thickness 5.2 mm) spread over the cell, both pressed down by the weight of the top beams by means of cleats that were held down and locked by tension bolts (Fig. 19). The rubber seal had to keep sufficient remaining compression over the entire length of the relatively rough sealing edge on the cells and at maximum actual pressures to provide adequate sealing pressure.

Cell R was used as the test object for ultimate application. The test took the form of a pressure-drop test at various cell pressures. Leakage measurements and leakage discontinuities showed where any local adjustments to the seal were required.

It appeared that the requirements set for critical operation could be achieved in stages. Reproducible values were obtained, provided that the installation procedure for the closure was strictly adhered to. The measuring procedure was always repeated before and after each period of operation, after each single opening and closing of the cell. It was found that the quality of the seal, characterized by the gradient of the leakage rate vs. cell pressure, remained good (Fig. 20).

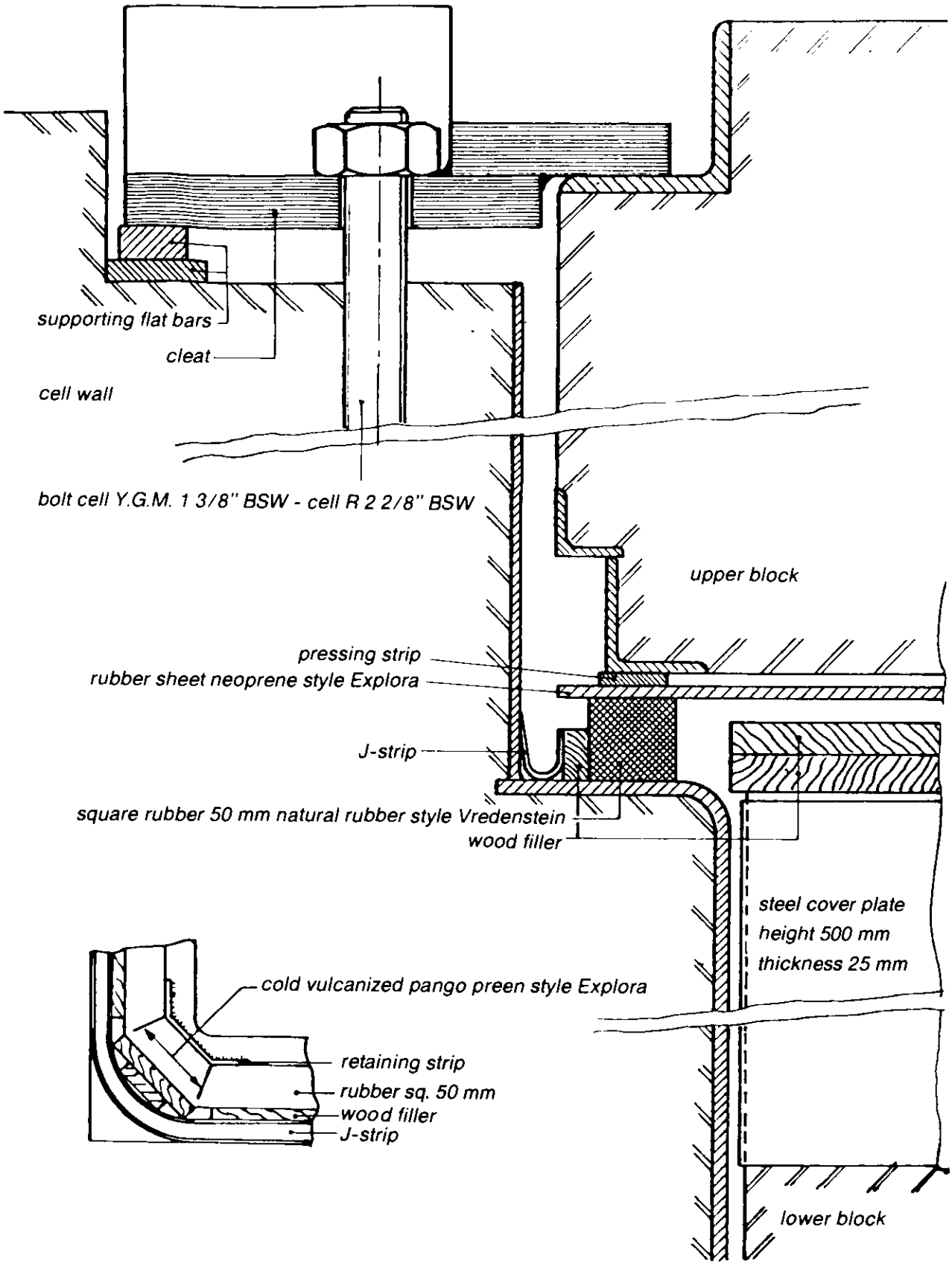



Fig. 19
Cell-closure construction.

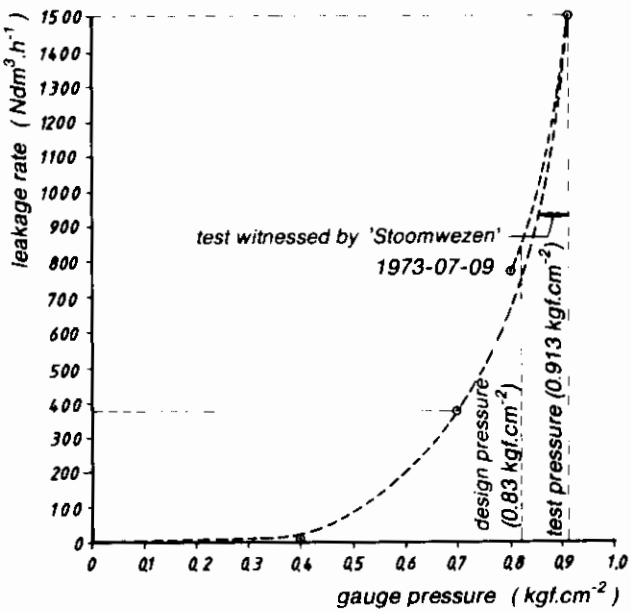



Fig. 20
Leakage rate of the M-cell closure. M-cell medium: air temperature 293 K; method: pressure-drop test with reference vessel.

*Control-systems research and development work*

The special nature of the suspension reactor being developed determined to a large extent the nature of the development work on the control systems. Apart from the special aspects of a nuclear plant in general, the KSTR would feature measurements and controls of extensive radioactive systems which would be located in cells with difficult access. Sensors and control devices thus often had to be radiation proof. The devices had to satisfy stringent leak-tightness requirements and had to be reliable and require little maintenance. Should defects occur, replacement had to be possible by using remotely operated tools. Development work consisted not only of developing separate measuring and control circuits but involved the entire instrumentation of the reactor in minute detail.

*Development of control systems* – Stringent requirements were set for (a) the temperature control of the suspension, (b) the level control in the gas separator which had an effect on the concentration, and (c) the gas-injection control. Those requirements were necessary in order to diminish the influence of those parameters as much as possible. The effects of changes in these parameters could be examined during special tests by applying sinusoidal and/or stepwise variations in these parameters.

(a) Setting average values of the neutron flux for subcritical tests should be done by adjusting the concentration and temperature of the suspension. Slow ajustment and stabilizing of the average power should be possible, besides the built-in fast internal control, in the case of power operation. The special point about this *power control* is that the flux-measuring equipment is in no way involved in the control. Hence defects in the flux-measuring equipment could never be the cause of nuclear excursions. As it was ensured that the power control had a built-in natural inertia, defects in the control system could never be transmitted quickly to the temperature in the reactor vessel and so they could easily be coped with by the reactor physical control through the immediate negative temperature coefficient. With such an arrangement operation could be considered as inherently safe as far as concerns nuclear excursions.

The purpose of the power control was to permit adjustment and stabilizing of the withdrawal of heat from the main system via the main heat exchanger (ME-1). The control must have a range of 0 to approx. 940 kW to permit proper operation. Considering the heat losses of the main system as 60 kW (mainly due to evaporation), the corresponding reactor power would be 60 and 1000 kW respectively.

The disturbances to be met by the control system would originate on the one hand from the main system where a fluctuating critical temperature would affect heat transfer in the main cooler, and on the other hand from the outside, where changing weather conditions (showers) might alter heat transfer in the air cooler.

The disturbances in the main system were analyzed with the transfer function of the main cooler and its lag times being included in the analysis. It could be concluded that at low reactor powers the disturbances would be in the order of $\pm$ 50 kW if the variation of the suspension temperature was $\pm$ 2 K. Furthermore, periods of minimally 20 s had to be considered. At higher power levels the relative effect of the suspension temperature on heat transfer would be considerably less due to the much larger temperature difference through the main cooler.

Disturbances from outside could be dealt with by a separate control at the air cooler and by the buffering action of the intercooling system (secondary cooling system). A power control of this type was designed, the principle of which is shown in Figure 21.

Power was actually measured by finding the product of the water flow and the temperature difference through the cold branch of the main cooler. Deviations falling between that value and the desired value were to be converted by a control device into a control signal to a mixing valve (323-325), permitting more or less cold water to be supplied as required to the injection point (P) upstream of the main cooler.

The capacity of the mixing control would be reduced in the case of higher power levels, since the temperature difference between the primary and the secondary cooling system would then become less. Although not absolutely necessary in view of the decreased relative effect on the disturbances at higher power levels, the control capacity of the mixed cooling could then be raised again by increasing the total flow using valve 301. Rough adjustment of the power level could be hand-controlled via the large coolers HE-9A and HE-9B.

The control system eventually developed was a stable and well adjustable instrument for setting reactor power. However, the high inertia of the system, particularly for disturbances higher than approx. 50 kW, also had a distressing consequence which attracted too little attention at the development stage. The problem arose later, during operation of the reactor, when an emergency stop was carried out at a reactor power level higher than approx. 200 kW without the mixed oxide being dumped. The reactivity due to dropping the control rods decreased by approx. 1.2 $\Delta k/k$ in the emergency stop. However, due to continued cooling by the inert control cooling system the temperature in the main system could drop so far that the reactor again became critical and started to supply power.

Model studies therefore were again carried out on the interaction between the main system and the control cooling system during actual power operation. An analogue computer was used to permit forecasting of the progress of the power after an emergency stop. It was found, among other things, that even if dumping could have been carried out, it could not have been done fast enough to prevent the repeated criticality. These facts had great bearing on reactor safety, implying that the approach for designing the fast stop and the emergency stop had to be revised.

(b) One of the important parameters of the reactor was the suspension content of the main system together with the concentration when a given amount of mixed oxide was flushed in. Disregarding the fraction of gas bubbles in the suspension (approx. 1%), the suspension content of the main system was set by the level of the free liquid surface in the gas separator. *Level control in the degaser*, resulting in keeping the level accurately constant would consequently fix the content of the main system and with it the concentration.

In view of the subcritical measurements – e.g. determination of the magnitude of reactivity fluctuations – the requirement was set initially that the content of the main system had to remain constant within $\pm$ 70 cm$^3$, corresponding to a desired concentration of $\pm$ 0.1%. A change of $\pm$ 70 cm$^3$ in the system content meant a variation of $\pm$ 1.75 cm in the liquid level in the degaser. Level disturbances resulting from temperature variations in the main system during subcritical operation could be neglected due to the high quality to be expected of the temperature control. Variations in gas injection thus remained the major source of disturbance.

At normal reactor temperature and pressure and with normal gas injection, the volume of gas bubbles in the gas/liquid contactor would be approx. 1 litre. Hence a variation of $\pm$ 5% in the gas injection would be accompanied by a change of approx. $\pm$ 50 cm$^3$ in the suspension volume. It was to be expected that gas injection could easily be kept constant within 5%.

It was not expected that many problems would arise during critical operation. Surges in reactor power could indeed be a reason for expansion of suspension and gas bubble volume. The volumetric variations caused by them would, however, be restricted since it was expected that such surges could never last long, due to the high negative feedback action of the negative temperature

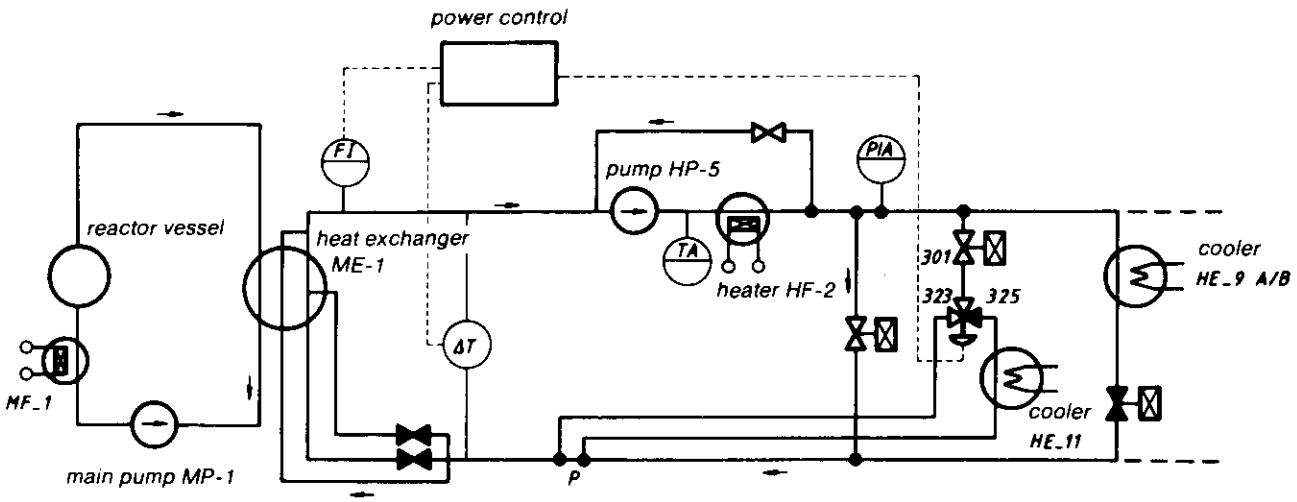



Fig. 21
Primary high-pressure cooling-water system.

coefficient of the reactivity. The order of magnitude of those effects is given by the calculation that a power surge with an energy of 100 kW in 70 litres of suspension (the content of the main system) gives a temperature rise of 0.34 K. The accompanying expansion of the suspension is 50 $cm^3$ and the increase in the volume of gas bubbles approx. 15 $cm^3$. It thus seemed that installing a well operating level control in the degaser would not present too many problems.

Reliability of the control obviously had to be high because a failure could result either in too low a level in the gas separator (with the chance of gas breaking through and harmful consequences for the pump) or in too high a level with a chance that suspension could be pushed across to the gas-purification system.

Despite the favourable prospects, the level control has been a source of trouble almost throughout the entire time of reactor operation. Frequent extensions and modifications have had to be made during operation. The result was a rather complicated system the major details of which are illustrated in Figure 22.

(c) It will have become obvious that constant *gas injection* was important (concentration; evaporation). Stability within ± 5% of the nominal value would, however, be sufficient. This was not a stringent requirement considering the speed at which the flow meter could operate and the disturbances to be expected. In view of the relatively high gas speeds, time lags would not play a part. Also the time constant determined by the gas flow and the volume between the sensor and the control device (the injection gap) was small enough to permit stable adjustment of the gas flow.

To prevent pressure fluctuations in the reactor vessel from pushing the suspension up into the gas injection pipe during critical operation, it was chosen to have a rather high pressure drop through the gap (approx. 0.3 MPa), so that there would be little pressure available for a control valve installed in series. Another reason for this choice was that the recombiners would already give a substantial pressure drop. It was therefore decided to use a bypass control arrangement in which the gas control valve would always have a pressure at least as high as the pressure drop through the gap.

The control arrangement itself has performed well during reactor operation, but blocking of the injection gap has often made good gas injection impossible.

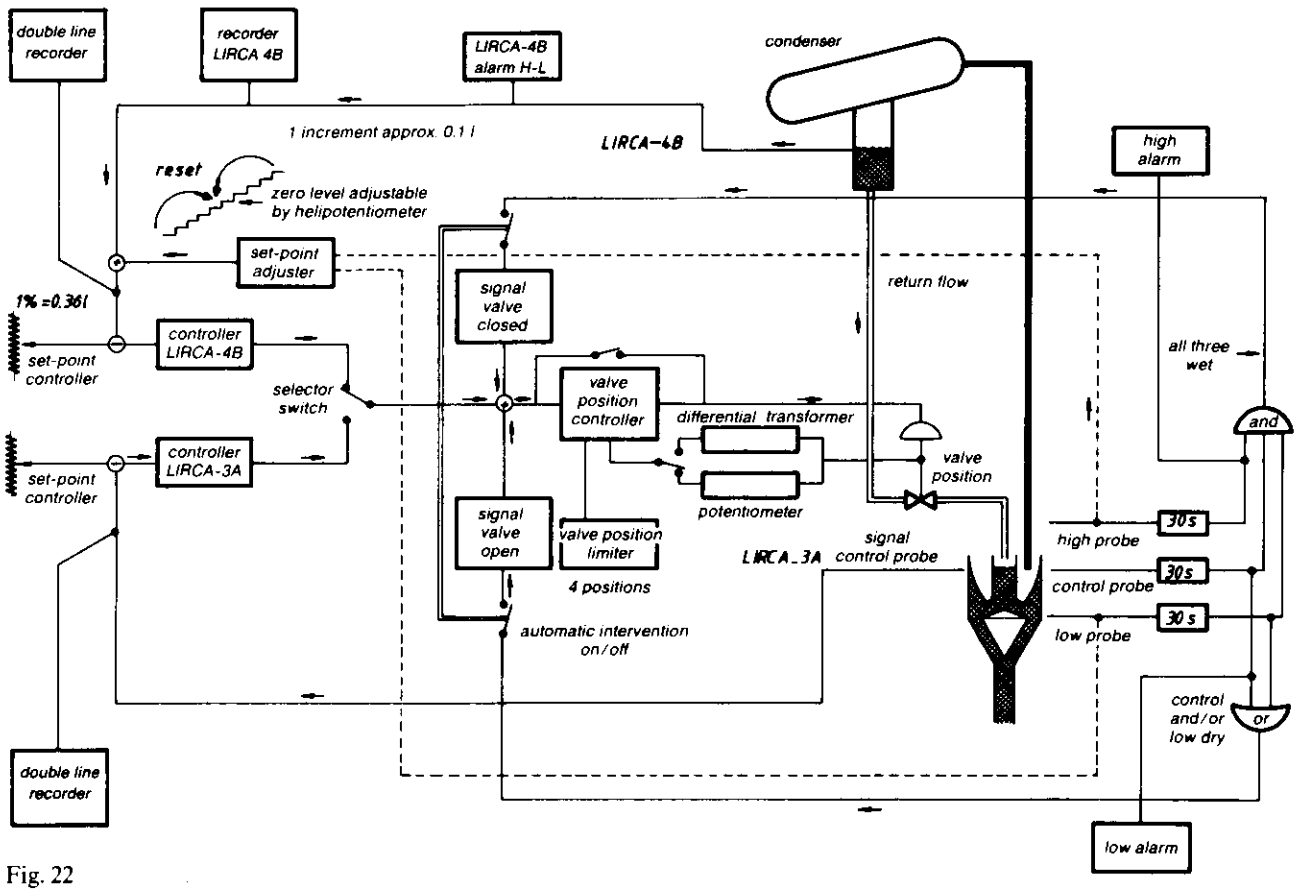



Fig. 22
Level control of the gas separator.

*Flux-measuring channels* – Contrary to the control function of flux-measuring equipment in conventional reactors, the KSTR flux-measuring channels were required mainly for measurements of secondary functions, for checking and monitoring. During the subcritical phase of operation, flux detectors would be installed in the pressure vessel in the neutron reflector; during power operation they would be installed in a so-called thimble located in the main cell at some distance from the pressure vessel. The thimble was accessible from the hall, so that maintenance could be carried out easily even after the systems had become radioactive. A location close to the reactor vessel was necessary during subcritical operation, since otherwise a rather strong neutron source would be required for measurements up to a source multiplication of M = 1000, in view of the sensitivity of the detectors.

The flux-measuring equipment consisted of two count-rate channels and a spare count-rate channel for the low-flux measurements, as well as five current-measuring channels for power operation. Two of the current-measuring channels were of logarithmic design for protection over the entire flux-measuring range. The maximum of the measuring range was equal to approximately 10 MW reactor power, and the mininum which could be measured by the count-rate channels was 1 mW.

The sensitivity to disturbances of the main power supply could be suppressed to the extent that disturbance contributions were reduced to 10% of the minimum detectable. To achieve that it was necessary to have the flux-measuring equipment entirely insulated, to permit grounding at a point to three grounding spikes specially driven for that purpose. In addition a number of grounding points in the test amplifiers had to be modified.

Table 2
Basic data for the design of the reactor.

| | |
|---|---|
| fuel | 22.5% $^{235}UO_2$ plus 75% $ThO_2$ |
| fuel concentration | 0-400 $g \cdot l^{-1}$ $H_2O$ |
| particle size | 5 $\mu m$ |
| moderator | $H_2O$ |
| reflector | BeO + graphite |
| nominal thermal power | 1,000 kW |
| specific power | 50 $kW \cdot l^{-1}$ |
| max. operating temperature | 528 K |
| max. operating pressure | 6 MPa |
| pressurizing gas | $H_2$ |
| reactor-vessel configuration | spherical with conical inlet and outlet sections |
| volume of the reactor vessel | 18.3 l |
| max. reactor-vessel diameter | 310 mm |
| mass flow | 15 $kg \cdot s^{-1}$ |
| flow speed in pipes | 5 $m.s^{-1}$ |

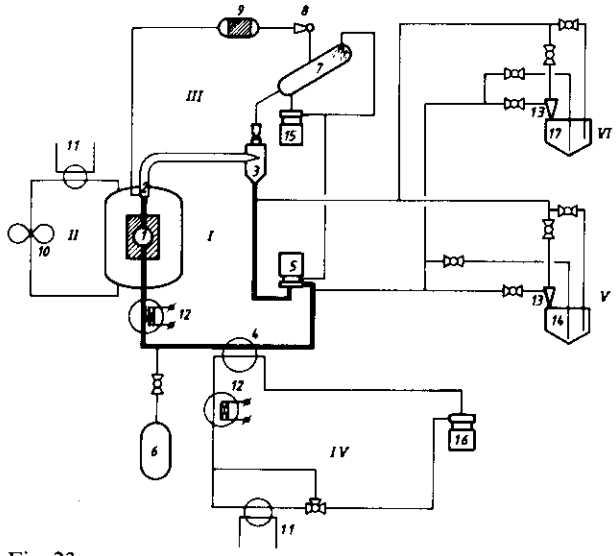



Fig. 23
Schematic diagram of the KSTR.
I = main suspension circuit (MS); II = reflector-cooling system (RCS); III = gas-purification system (GPS); IV = primary cooling-water system (PCSA); V = concentration-control system; VI = fuel supply and discharge system.
1 = reactor vessel (MR-1); 2 = gas/liquid contactor (MJ-2); 3 = gas separator (MJ-1); 4 = main heat exchanger (ME-1); 5 = suspension pump (MP-1); 6 = dump vessel (MV-1); 7 = spray condenser; 8 = ejector; 9 = recombiner; 10 = gas pump; 11 = heat exchanger; 12 = electrical heating; 13 = hydroclone; 14 = storage vessel (MV-10); 15 = auxiliary pump; 16 = cooling-water pump; 17 = storage vessel (MV-60).

**Reactor design**

The entire process of using an aqueous suspension in a homogeneous suspension reactor depends greatly on the behaviour of the suspension. In addition to physical transport phenomena, the physico-chemical properties such as colloid-chemical stability and interaction of the particles play an important role.

Some of the properties of $ThO_2/UO_2$ particles which affect the process and plant design are:

(1) specific mass: 10 $g \cdot cm^{-3}$; this implies a heavy material, possibly resulting in settling, and a minimum transport velocity of 1 $m \cdot s^{-1}$;
(2) size: $d_{50}$ = 5 $\mu m$; this small size implies a chance of blocking and of particles adhering to each other and to the walls;
(3) hardness: Mohs' scale 8 to 9; this implies that the particles are hard. Erosion of the casing walls should be prevented by a maximum allowable speed of 5 $m \cdot s^{-1}$ in pipes, bends and branches.

The basic data are summarized in Table 2. Detailed process data can be found in the various design specifications for systems and components.

*Process diagram*

The (ultimate) process diagram of the nuclear system and the cooling and gas-discharge systems connected with it, is shown in Figure 23; digits between brackets in subsequent sections refer to the identical digits in this figure. The process diagram includes the following systems that will be dealt with in the same order:

(1) main systems (MS), subdivided into:
the suspension circuit,
the concentration-control system,
the fuel-supply and fuel-discharge facility,
the dumping system, and
the sampling system;
(2) gas-purification system (GPS);
(3) reflector-cooling system (RCS);
(4) primary cooling system A (PCSA); and
(5) outlet-gas system (OGS) (partly).

Those systems were considered to be nuclear systems, i.e. systems in which fuel or radioactivity was present or might be present.

*Suspension circuit* – In the main circuit (I) the suspension, consisting of fuel particles (5 $\mu$m $ThO_2$-$UO_2$) suspended in demineralized water, is circulated by a centrifugal pump (5). The suspension flows from bottom to top through a virtually spherical reactor vessel (1) (diameter 310 mm, volume 18.3 l, stainless steel wall thickness 5 mm) with a slightly rotating motion for flow stabilization and with a high speed along the wall to prevent the particles from settling.

A stabilizer tube was installed in the centre of the vessel during the period of subcritical operation for additional flow stabilization in a lateral direction, in order to reduce fluctuations of the neutron flux. The tube was made of zircaloy and was fixed at the sides at the top and bottom allowing differences in expansion between vessel and tube to be taken up.

The heat generated by fissions in the reactor vessel was transported by the liquid to a heat exchanger (4) for transfer to the primary cooling system (IV).

At the discharge of the reactor vessel, hydrogen was injected into the suspension in the so-called gas/liquid contactor (2) and separated from it again in the gas separator (3). That was necessary since the fission process in the reactor vessel could lead to partial decomposition of the water to form hydrogen and oxygen. Injection of excess hydrogen increased the hydrogen/oxygen ratio, thus preventing the creation of an explosive mixture. In addition, the gaseous fission products were stripped and partly separated from the liquid. Furthermore, water vapour was released and was condensed elsewhere for the preparation of particle-free water. After the gases had been separated from the liquid flow they were conveyed to the gas-purification system (III).

An effective pressure of 2 MPa was applied in the main system to prevent boiling. Hydrogen was selected as pressurizing gas as it could also be used in the process to shift to a more favourable hydrogen/oxygen ratio should decomposition of water occur (radiolysis). Oxygen was not used for this purpose since it has many disadvantages, and other gases (e.g. argon) lack the advantages of hydrogen.

At 523 K the overall pressure in the main system was approx. 6 MPa.

*Concentration-control system (V)* – To permit adjustment of the concentration of fuel particles in the reactor vessel, a bypass including a storage vessel (14) was connected with the main circuit (Fig. 23). The particles were separated from the main flow by a small hydroclone (∅ 25 mm) on top of the storage vessel. Inversely the particles could be 'resuspended' into the main flow by having water flow through the vessel, in which case the particles were carried along due to flushing.

The contents of the vessel could be weighed when either cold or hot to allow the amount of fuel flushed in to be determined.

*Fuel-supply and fuel-discharge facility (VI)* – Parallel to the storage vessel (14) an identical vessel (17) was connected into the main circuit. This vessel permitted fuel from the outside to be supplied to and discharged from the system for storage and processing elsewhere.

*Dumping system* – A dumping system was connected directly to the main circuit, providing safe and fast removal of the contents of the reactor vessel in emergencies. The system included a dumping vessel (6) whose operating pressure during normal operation was lower than that of the main system. The entire contents of the main circuit and of some of the systems connected with it could be dumped into this vessel in such a way that the reactor vessel was emptied within 15 seconds. After dumping, the released decay heat was removed by an entirely independent cooling system (dumping cooling system) which could also operate on natural circulation.

The nuclear safety of the vessel was ensured by screens of neutron-absorbing materials, i.e. a silver-indium-cadmium (AgInCd) alloy.

In addition to the function just mentioned, the dumping system had also a function in some of the fuel-handling procedures. The fuel balance could be determined by weighing the dumping vessel before and after a dump. The vessel was provided with a weighing device and a level gauge for this purpose.

The dumping vessel was provided with an electro-

magnetically driven, vertically moving agitator to keep the dumped suspension in motion. This prevented the particles from settling on the bottom, specially during a short period after dumping. Possible adherence of particles caused by subsequent heating was thus prevented.

The suspension could be fed back to the main circulation system by pushing it out with hydrogen or by circulation.

*Sampling system* – Samples could be taken from the main circuit by means of a bypass flow which could be connected with a 15 $cm^3$ sample vessel. That vessel could be transported by remote control from the system to the hot cell for further examination.

*Gas-purification system* – The gaseous fission products withdrawn from the main circuit were conveyed to the gas-purification system (III) together with radiolytic gas and water vapour. Hydrogen was circulated in this system and injected again into the main circuit.

The presence of the gas in the GPS reduced the content of Xe-135 and other gaseous fission products and as a result Xe poisoning of the primary system. The system included a device (9) for recombination of the radiolytic gas (oxygen and hydrogen) and a condensor to condense the water vapour to particle-free water which was then supplied to the main system for flushing the circulation pump and the valves.

*Reflector and reflector-cooling system* – The reactor vessel was encased by a reflector consisting of stacked beryllium oxide bricks topped by stacked graphite bricks.

The dimensions selected for the reactor vessel and the reflector were such that a chain reaction of fissions could be maintained in the reactor vessel while manageable fuel concentrations were used in the main system. To obtain a neutron-flux distribution which was as flat as possible, the wall thickness of the reactor vessel was restricted to a maximum of 5.5 mm. To restrict the stresses in the vessel wall at the prevailing pressure in the main system (6 MPa), the pressure was partly balanced by a counterpressure of 4 MPa in the reflector. This was achieved by encasing the reactor vessel and the reflector in a pressure vessel (MV-1).

Heat was to be generated in the reflector during power production by the reactor. That heat could be removed by the RCS (II) which could be used to recirculate $CO_2$ gas and to convey it into the reflector through cooling channels.

The lower portion of the pressure vessel was so designed that all fuel could be collected in it in a nuclear safe fashion in case of leakage of the reactor vessel. To this end the reactor vessel was provided with a jacket with a pipe which discharged in the lowest section of the pressure vessel.

*Primary cooling system A* – The heat generated in the reactor vessel was withdrawn during passage in the main heat exchanger (4) by the PCSA (IV). That system also provided heat when the reactor was operated subcritically and so had to be kept at a constant temperature. This was done by installing electric heating blocks (154 kW) on the pipes. The heat was withdrawn from the PCSA by the secondary cooling system (SCS).

The primary cooling system B extracted the heat from all other components of the nuclear systems which were essential for operation at low temperature.

*Off-gas system* – Connections with the off-gas systems were provided at the main system and the gas-purification system. Pressure in the main system or in the gas-purification system was reduced in the recombiner by combustion of the hydrogen, after which the remaining radioactive gas was vented to a storage vessel. It could be dispersed from there after decay through the reactor disposal line and the ventilation system.

The (vacuum) pressure in the cells was kept at a constant difference with that in the reactor hall via the normal disposal line.

*Containment and hall*

As already mentioned, the nuclear systems were enclosed in completely sealed and gas-tight steel cells (containment cells) separated from each other by 1 m thick concrete walls. The entire assembly was enclosed in a gas-tight hall (reactor hall). The screen to the hall was formed by two layers of concrete beams. A nitrogen atmosphere of 80 kPa absolute pressure in relation to that in the hall was maintained in the cells. The hall itself was kept at a pressure of 500 Pa lower than the ambient air.

A containment structure was thus obtained that provided a triple barrier preventing the escape of radioactive material to the atmosphere, even in cases of bursting or leaking of the primary systems. The primary systems were interconnected through the cell walls by leak-tight crossing arrangements which joined at the steel walls. The cells were connected with each other to deal with a possible build-up of pressure in the cells beyond a specific level.

**Operation and experimental research**

The objective of KSTR operation was to examine the behaviour of a suspension reactor under the process conditions applicable to a power reactor of that type.

The research was carried out in four phases.

The first phase involved tests of the various KSTR systems using particle-free water. A functional test of the main system was carried out during the second phase using a suspension of natural $UO_2/ThO_2$ particles. This phase was primarily devoted to the study of the concentration control. The particle concentration was measured both directly (absorption method) and indirectly (weighing method). The emergency shutdown procedure with suspension (dumping) was also tested during this phase. Subcritical experiments were carried out during the third phase to obtain the data necessary for start-up and operation of the reactor, as well as to evaluate the safety of the systems. The last phase consisted of a three-year period of critical operation during which the behaviour of the reactor parameters was studied as a function of the power, together with the effect of increasing radiation load on the physico-chemical properties of the fuel.

*Subcritical experiments*

In view of experience acquired with the low-temperature subcritical set-up (see 'Subcritical experiment' section) it was obvious that the KSTR research should also be started under subcritical conditions. Even at this stage, however, the fluctuations of the neutron flux appeared substantially larger at low source multiplication. The fuel concentration in the reactor system was therefore restricted for the time being to such an extent that the value of $k_{eff}$ remained $< 0.95$ (source multiplication $M < 20$). The relation measured between $1/M$ and the quantity of fuel flushed in (determined from the weight reduction of the storage vessel) could be used to make an initial estimate of the critical mass of fuel flushed in by extrapolating to $1/M = 0$. This value was approx. 21 kg at 498 K and approx. 23 kg at 528 K. Those measurements were also used for determining the temperature coefficient of the reactivity for the relevant temperature range. The value of $-0.058$ $\Delta k/k$ per Kelvin, found during experiments, appeared to be well in line with the calculated value, i.e. $-0.060$ $\Delta k/k$ per Kelvin at 523 K.

The effect of the circulation rate was demonstrated by an increase in source multiplication when the circulation rate was decreased. This effect was due to the settling of fuel particles in water. The settling rate of 5 $\mu$m $(U,Th)O_2$ particles was only a few mm per second at room temperature and thus was negligible in comparison with the flow rate in the reactor vessel (15 $cm.s^{-1}$). At 523 K, however, the settling rate was a few centimetres per second, as a result of which – and depending on the circulation rate – the concentration in the reactor vessel increased ('hold-up': this phenomenon did not occur in the external system since the flow rate there was 30-fold higher than in the reactor vessel). The rise in concentration proved to be 0.090 $\Delta k/k$ at a decrease in pump capacity of 1 $m^3 \cdot h^{-1}$ (normal capacity 33 $m^3 \cdot h^{-1}$).

The fluctuations of the neutron flux were the main topic of research during the subcritical phase. A representative recording of the neutron flux is given in Figure 24. The average fluctuation of the reactivity was $\pm$ 1 $\Delta k/k$, with peaks to 1.8 $\Delta k/k$.

In view of the favourable effect of the colloid-chemical stabilization of the suspension in the low-temperature subcritical set-up, an attempt was made to stabilize the suspension by increasing the pH value. Autoclave tests had, however, shown that it was difficult to stabilize the suspension above approx. 500 K. The effect of the pH increase was still noticeable at 498 K (decrease in fluctuation amplitude by 20-30%), but had disappeared at 528 K.

Flow tests in a plexiglass model of the reactor system had shown clearly that hydrodynamic instabilities of the flow pattern in the reactor vessel could be the cause of local changes in fuel concentration, which in turn would result in the fluctuations in reactivity observed. Analysis of possibilities for stabilizing the flow pattern resulted in the design of an approx. 30 cm long stabilizer (cf. 'Research and development' section) installed vertically in the centre of the reactor vessel (Fig. 16). The effect of the flow stabilizer on the flux pattern satisfied all expectations. A representative recording of the neutron flux after installation of the flow stabilizer is given in Figure 25. The remaining reactivity fluctuations were only $\pm$ 0.3 $\Delta k/k$, whereas peak reactivity was approx. 0.5 $\Delta k/k$. Approval was obtained to make the reactor critical once calculations had proved that such reactivity fluctuations could be compensated by the immediately acting negative temperature coefficient without the occurrence of extreme rises in temperature and pressure.

Experience with the homogeneous reactor in Oak Ridge, where tests with deliberate reactivity additions of up to 0.8 $\Delta k/k$ in one second had run well was of undoubtedly great value at this stage of the research.

*Critical operation and experimental research*

The two main aspects of this phase were (1) the reactor start-up and (2) critical operation to full power (1000 kW).

*Reactor start-up* – The KSTR system offered a number of possibilities for varying the reactivity of the reactor to make it critical and to bring it to power level. First, reactivity could be influenced by changing the fuel concentration in the reactor vessel. This could be achieved

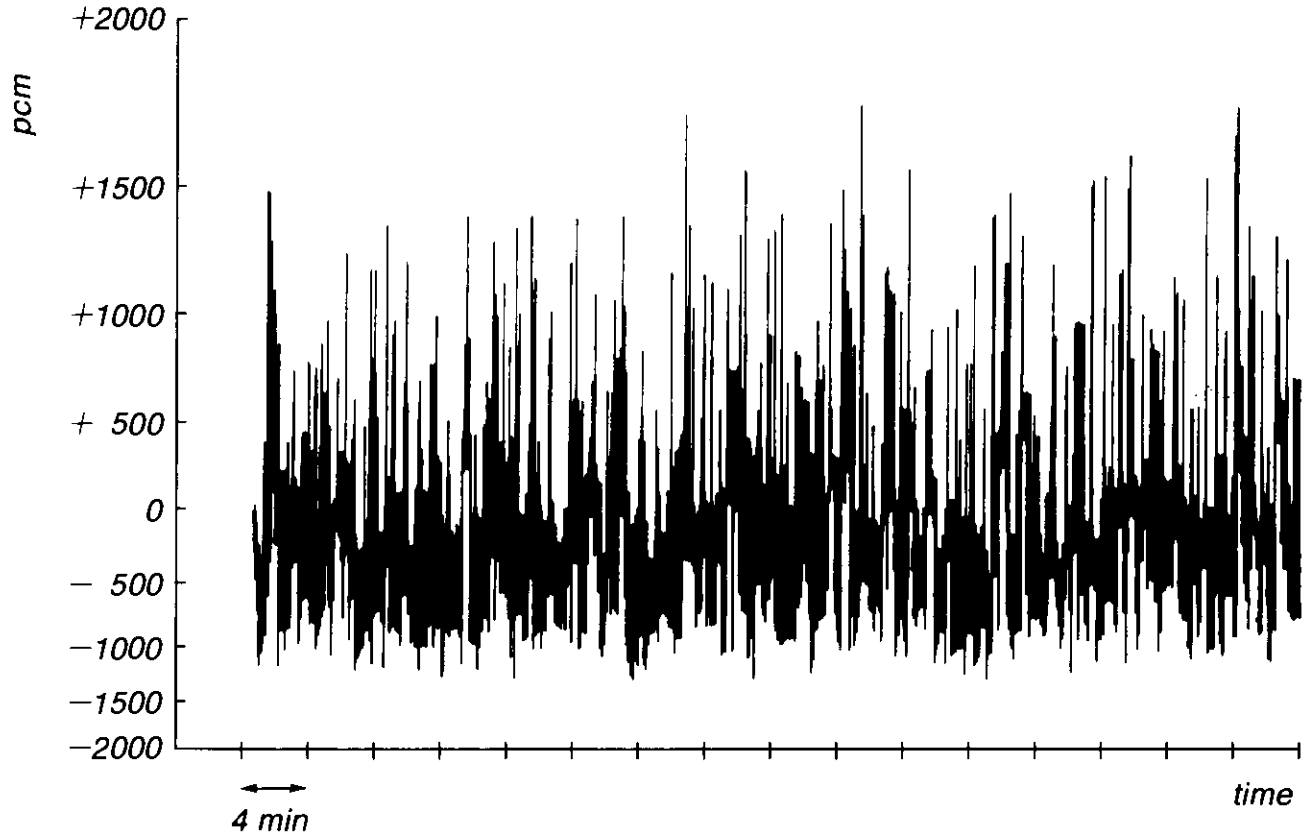



Fig. 24
Recording of the neutron flux without stabilizer. System temperature 528 K; flow rate 33 $m^3 \cdot h^{-1}$; R-C time of the measuring channel 1 s.

both by flushing in or catching out fuel, and by changing the fuel hold-up in the reactor vessel by altering the circulation rate through regulation of the suspension pump speed. Reactivity could also be changed by altering the position of the control rod. Finally, reactivity could be affected by changing the moderation by varying the reactor temperature i.e. reactivity could be increased either by switching off the heating in the main system or by switching on the cooling.

The first time the KSTR was made critical a procedure was followed which allowed a slow and accurate approach to the critical situation so that the process of becoming critical could be evaluated. The starting point was a reactor temperature of 528 K and increased pump delivery of 36 $m^3 \cdot h^{-1}$. Fuel was flushed in up to a source multiplication of 125 ($\rho = -0.8\ \Delta k/k$) with the control rod still in the reactor. Pump delivery was subsequently decreased to 33 $m^3 \cdot h^{-1}$, causing the reactivity to rise to

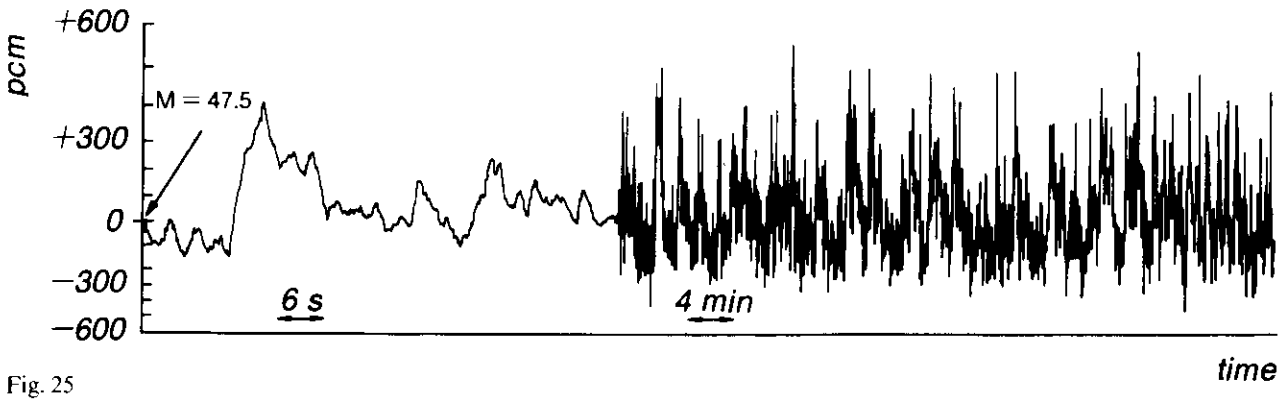



Fig. 25
Recording of the neutron flux with stabilizer. System temperature 528 K; flow rate 33 $m^3 \cdot h^{-1}$; R-C time of the measuring channel 1 s.

-0.5 $\Delta k/k$. Reactivity was then raised further by partly withdrawing the control rod (total reactivity value approx. 0.25 $\Delta k/k$) from the reactor and letting the temperature fall by adjusting the set-point of the automatic temperature control. The reactor became critical at a temperature of 518.3 K and a power of 3 kW with a weight of 23.8 kg of fuel in the reactor system. In addition to the slow fall of the average temperature, fast temperature rises occurred due to power excursions in the reactor.

The reactivity increases ($\rho$ excess) were calculated from the ratio between peak power (P) and initial power ($P_o$), giving 0.33-0.48 $\Delta k/k$, values that were well in line with reactivity fluctuations measured during subcritical KSTR operation. The feedback effectiveness of the negative temperature coefficient of the reactivity is based on the relation between initial power $P_o$ and the ratio $P/P_o$. These initial tests proved clearly that feedback rises sharply with rising initial power.

*Critical operation to full power (1000 kW)* – The procedure for 'becoming critical' was adapted on the basis of experience gained when the KSTR was first made critical. First, slow reduction of the heating power was replaced by fast opening of the control cooler in the primary cooling circuit so that approx. 40 kW power was removed. As a result, the reactor temperature fell quickly to the critical temperature, after which reactor power increased to approx. 40 kW. Power changes could then be achieved by altering the set-point in the control cooler. That procedure had the advantage that the area of very low reactor power – where the negative temperature coefficient is still insufficiently effective – was crossed rapidly. This procedure was used during the entire test period for regular operational start-up of the reactor.

When the reactor was repeatedly made critical, it appeared that the critical mass (quantity of fuel flushed in) matching a specific temperature was reproducible within the errors of measurements (Fig. 26). The quantity of fuel flushed in was determined by differential weighing of the storage vessel during operation. Weighing could be carried out with an accuracy of $\pm$ 100 gram. The critical temperature also fluctuated due to spontaneous reactivity changes, resulting in an error of $\pm$ 1 K in the temperature measurements. The fluctuations in the instantaneous power of the reactor decreased relatively sharply when the average power was increased. That was a favourable result of increasing the feedback effectiveness of the negative temperature coefficient of the reactivity. At a sufficiently high initial power level, each reactivity increase immediately resulted in a temperature rise, due to which the original excess reactivity was rapidly brought back to zero. The

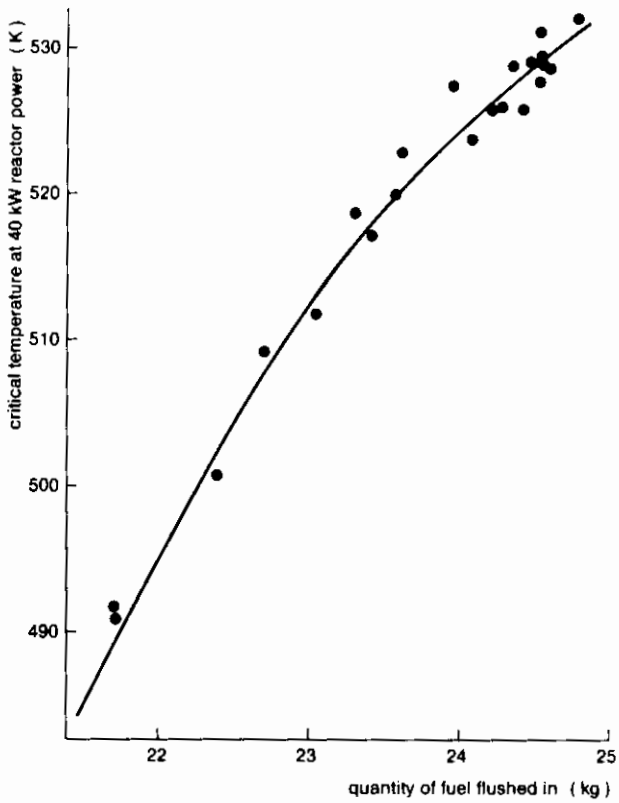



Fig. 26
Critical temperature as a function of the quantity of fuel circulating.

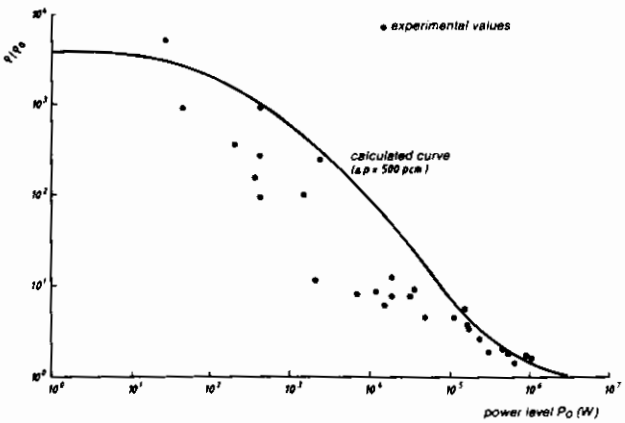



Fig. 27
Ratio between peak power (P) and initial power ($P_0$) as a function of $P_0$.

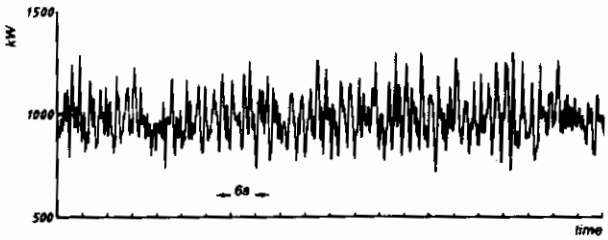



Fig. 28
Recording diagram of the neutron flux at maximum power (1000 kW).

ratio $P/P_o$ ($P$ = peak power, $P_o$ = the initial power before a power peak occurred) expressed this clearly. The ratio is plotted in Figure 27 as a function of $P_o$: the curve gives the calculated relation, the starting point being a reactivity increase ($\rho$) of 0.5 $\Delta k/k$ in one second, followed by a decrease to $\theta = 0$ in the next second. It seems that the experimental values determined from the highest power peaks followed the calculated relation. It was clear that feedback increased sharply with rising power level and that the reactor could as a result be designated as inherently safe. Figure 28 shows a recording of power at 1000 kW, from which it appears that the fluctuations were comparatively small at full power.

The temperature difference through the reactor vessel increased linearly with the power and was 301 K at 1000 kW. The flow pattern in the reactor vessel was thus not essentially affected by the increased heat generation, which may be explained by the fact that heat was not being generated uniformly in the vessel. With a homogeneous distribution of fuel concentration the neutron flux in the centre of the vessel was approximately twice as high as that at the wall and hence the heat generation at the wall was also higher than in the centre. Consequently, temperature gradients could develop which might disturb the original flow pattern. That apparently did not occur in the KSTR. Start-up, operation and shut-down of the reactor always progressed without marked difficulties. The power level was stepped up by switching gradually from heating to cooling via the heat exchanger which connected the main system with a circulation system (PCSA) comprising both a heater and a cooler.

When the reactor was to be shut down those steps were carried out in reverse sequence and at a slow rate. That rate was, however, often accelerated when a so-called automatic fast shut-down occurred due to a specific alarm setting being exceeded. Three such alarms were installed: one for the power (neutron flux), one for the integrated power (MWs) and one for the outlet temperature of the reactor vessel. The 1 MWs alarm was once exceeded during one of the experiments at an average power of 400 kW and a temperature of 526 K, leading to an automatic fast shut-down. The shut-down rods (reactivity value approx. 1.2 $\Delta k/k$) then dropped due to which the reactor became subcritical. Since the reactor no longer produced heat and since switching the

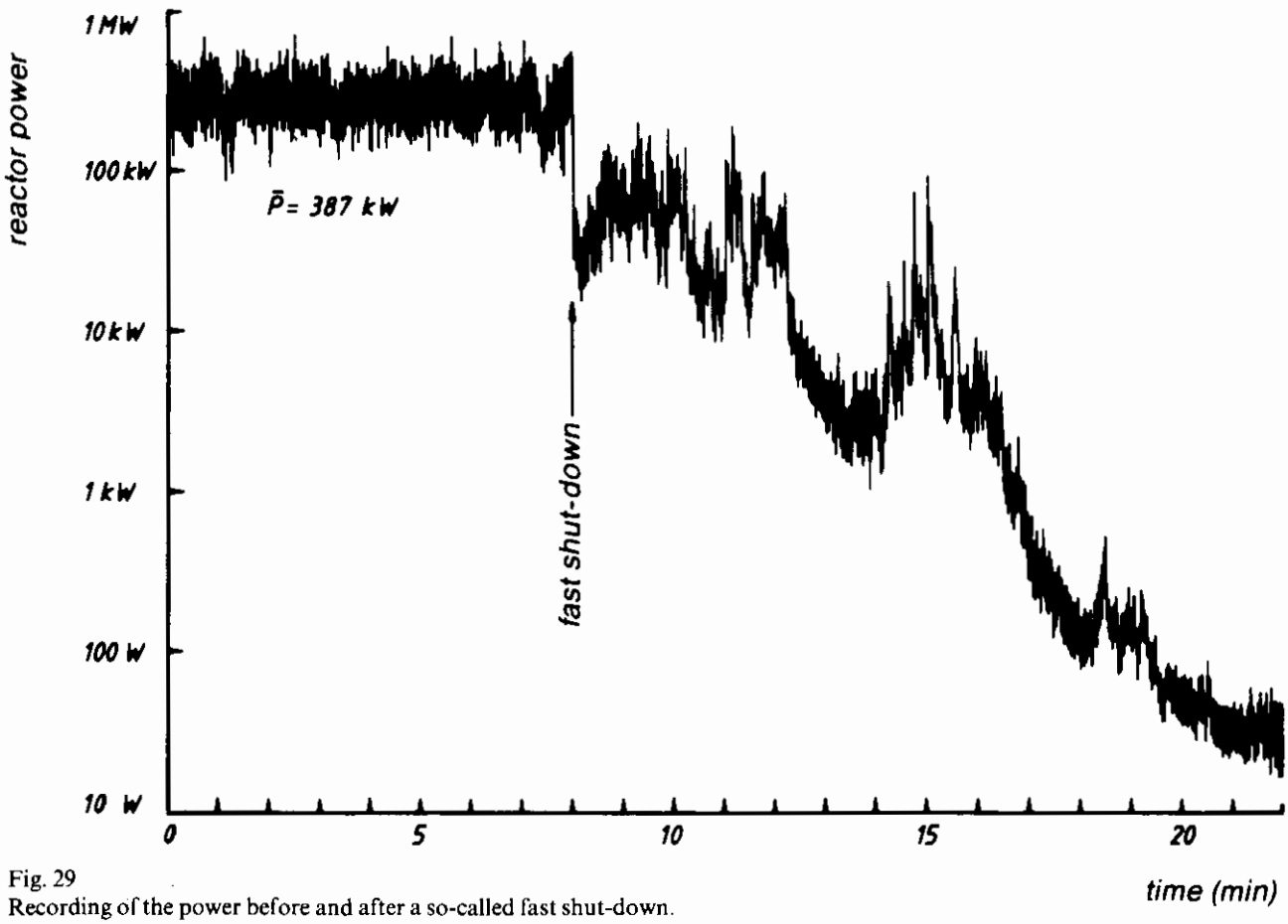



Fig. 29
Recording of the power before and after a so-called fast shut-down.

PCSA system from cooling to heating could not be effected rapidly, the temperature of the main system fell, due to which the reactor again became critical (at 498 K). Figure 29 provides a recording of the power during that fast shut-down. The figure shows that, after the start of the fast shut-down, the reactor was still critical up to a power level of approx. 200 kW before the PCSA system heated up sufficiently to be able to raise the temperature of the main system and, by so doing, keep the reactor in a subcritical condition.

If an alarm setting was again exceeded after there had been a fast shut-down, there followed a so-called emergency shut-down. This involved dumping of the entire content of the main system. No criticality could occur because the dumping vessel designed for the purpose was provided with a substantial quantity of neutron-absorbing material. The relevant calculations were verified experimentally in advance.

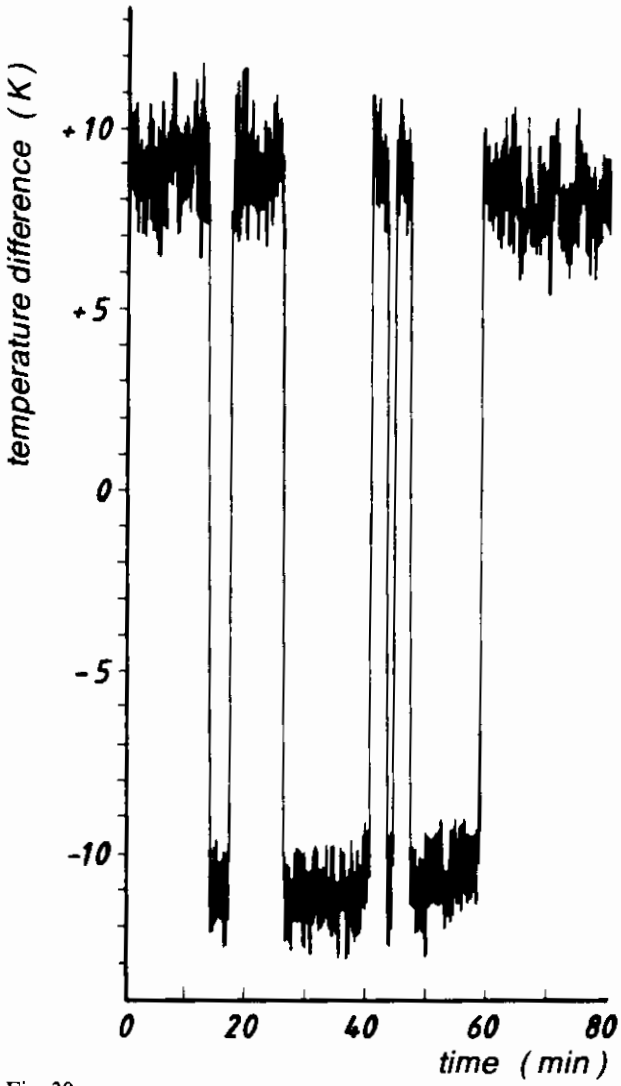



Fig. 30
Recording of the temperature difference between two points located in a horizontal plane at a mutual distance of approx. 20 cm, in the lower half of the reactor vessel. Reactor power 1000 kW.

Although critical operation generally satisfied the expectations some problems arose. They concerned mainly (a) local temperature differences, (b) block of the sampling loop, (c) loss of reactivity, (d) apparent loss of fuel and (e) noise.

(a) A still unexplained phenomenon occurred during the power operation of the reactor when the vessel wall *temperature* was being measured. To facilitate early detection of possible hot spots, 18 constantan wires were spot-welded to the external wall of the stainless steel reactor vessel. The 18 welding points were grouped in three horizontal rings, each with 6 points equally distributed over the circumference. In the event of temperature differences between the welding points, a thermoelectric voltage was created between the relevant constantan wires. It was thus possible to detect higher than normal temperature differences occurring at some locations and which increased with the power level of the reactor. Moreover, those differences changed signs spontaneously from time to time at some points. The largest temperature difference found at 1000 kW, being between +9 and −11 K, indicated a temperature jump of 20 K (Fig. 30). That difference was measured between two points approx. 20 cm apart and located in a horizontal plane in the lower half of the reactor vessel. Since the temperature difference through the reactor vessel was 28 K at 1000 kW, differences in temperature were to be expected between points at different elevations. This was unexpected, however, for points in a horizontal plane and the spontaneous change of sign was a further problem. Several hypotheses have been put forward to explain this phenomenon, all based on a spontaneous change in flow pattern. Experimental verification is, however, extremely difficult and might have been possible only by means of time-consuming model testing outside the KSTR.

(b) Another problem arose from *block of the sampling loop*. A sampling device was installed in a bypass of the main system to facilitate sampling of the suspension at specific times. A 'homogenizer' was placed in the bypass to provide a uniform concentration distribution at the sampling site thus making the sample representative. The homogenizer consisted of a pear-shaped metal body with a maximum diameter of 15 mm, installed in the sampling tube (∅ 20 mm). An annular gap of 2.5 mm was thus created locally.

Quite soon after the KSTR reached maximum power, the suspension flow in the sampling loop was found to decrease gradually and the critical tempera-

ture of the reactor to have fallen simultaneously by some 20 K. Fuel was apparently withdrawn from the main system, probably due to fuel particles settling in the sampling loop, in which the flow rate had dropped below the required minimum value for reasons as yet unknown. After shutting down the reactor and catching out the fuel in the storage vessel, it indeed appeared from weighing the storage vessel that more than 1 kg of fuel had remained behind. Most of it could be reclaimed by flushing. X-ray pictures of the sampling loop showed that the gap at the homogenizer was almost entirely blocked by a number of more or less globular shaped objects with a diameter 2.5 to 3 mm (Fig. 31). It was decided to remove that part of the loop to examine the 'lumps'. These proved to consist of approx. 95% fuel with the rest made up of erosion products (Fe, Ni and Co).

Similar lumps were replicated in autoclave tests. They developed at locations with a steep temperature gradient and only when the uranium-thorium oxide was mixed with one of the erosion products just mentioned. How and where the lumps developed in the reactor will be discussed in the 'Post-operational research' section.

(c) After a new homogenizer had been installed, critical operation of the reactor was resumed with a closed sampling loop. The critical temperature again appeared to fall, albeit at a lower rate than previously but with *loss of reactivity* as the consequence. The temperature fall occurred at power levels above 40 kW and the rate appeared proportional to the power. At 150 kW the rate was 2.5 $K \cdot h^{-1}$ (Fig. 32), one order of magnitude higher than could be explained by xenon poisoning. There was a slow, exponentially progressing recovery of the critical temperature when the power was reduced to below 40 kW. The threshold value of 40 kW was gradually decreased to 10 kW.

An unequivocal explanation of the phenomena described here has not yet been found. There is certainly a relation with the changing properties of the suspension under the influence of energy production: irradiation tests on capsules containing fuel suspension from the HFR in Petten and from the BR-2 in Mol showed that the fuel particles (∅ 5 mm) disintegrated as a result of discharging fission products. The disintegration increased with the number of fissions occurring, hence with the power production of the reactor in the case of the KSTR. The average fuel-particle diameter thus decreased with time and so-called fines developed, i.e. particles with a diameter of less than 1 $\mu$m. Consequently the settling rate, keeping the concentration in the reactor vessel higher than that in the external circuit, reduced the hold-up. That happened where the upward component of the flow velocity was comparatively small. If the settling rate decreased, hold-up was reduced and with it the critical temperature of the reactor at a specific quantity of fuel circulating in the main system. The plot of the critical temperature (at low power level) versus the quantity of fuel flushed in (Fig. 33) shows that the hold-up decreased with time. The results

Fig. 31
Picture of the lumps from the sampling loop.

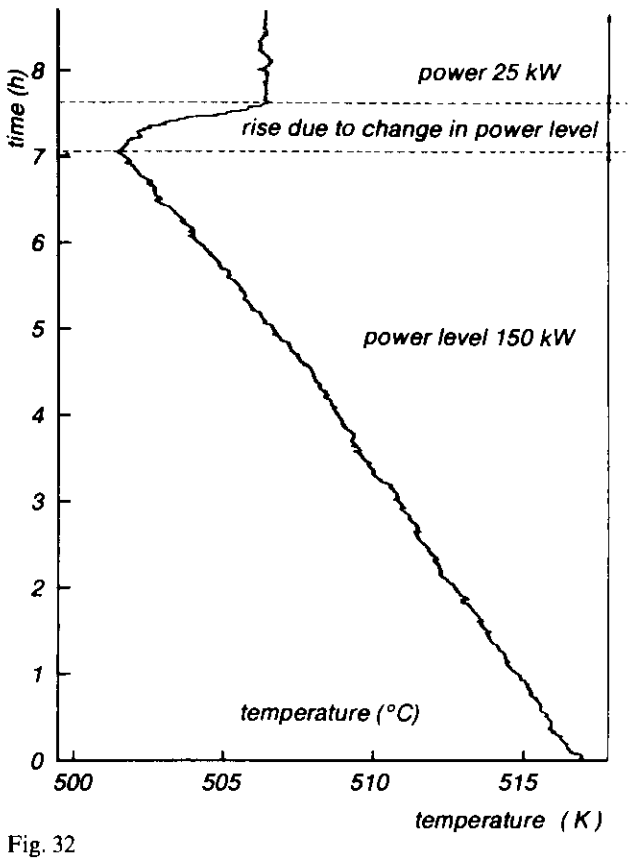



Fig. 32
Recording of the reactor outlet temperature. Fall at 150 kW (loss of fuel); constant at 25 kW (no loss of fuel).

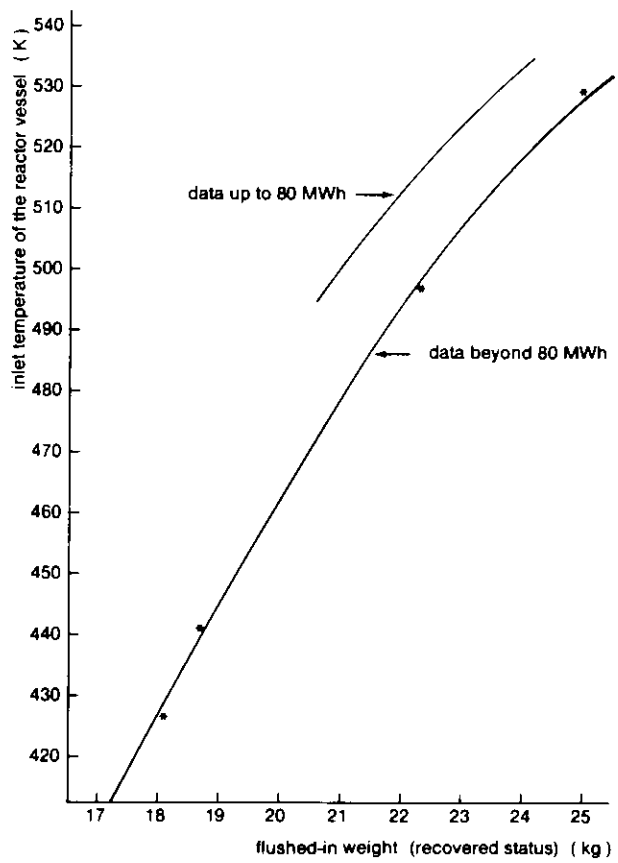



Fig. 33
Critical temperature as a function of the quantity of fuel circulating.

of direct hold-up measurements point to a reduction by a factor of 2, i.e. from 23% to 11%, at 528 K.

This, however, does not yet explain why the critical temperature fell by more than 20 K during such a short period of operation (Fig. 32). Apparently the colloid-chemical properties of the suspension changed rapidly under the influence of, and in proportion to, energy production in the reactor vessel. Slow recovery subsequently took place in the external circuit, but also at a low power level. Radiolysis in the reactor vessel most probably played an important role. The development of radiolysis products due to decomposition of reactor water was proportional to the power level, but the reverse reaction also took place simultaneously. Net production, as $H_2$ and $O_2$, could be measured using the temperature rise of a calibrated hydrogen/oxygen recombiner installed in the gas system of the reactor. The gas system was normally charged with hydrogen gas at 20 $kg \cdot cm^{-2}$; a temperature rise of the recombiner was only rarely observed under those reducing conditions and this only at high power levels (> 350 kW) (Fig. 34). Apparently the oxygen produced by radiolysis was usually already recombined with hydrogen in the reactor vessel.

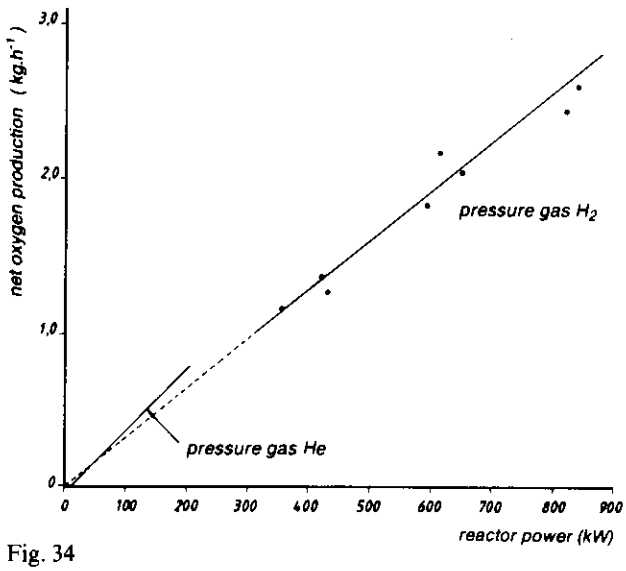



Fig. 34
Net oxygen production versus reactor power.

Hydrogen gas was replaced by helium to influence radiolysis and with it perhaps the colloid-chemical properties of the suspension. The effect of this on the net radiolysis was very clear and could be measured by using the temperature rise of the recombiner, starting from the lowest power level, as shown by Figure 34 (lowermost curve). Even after the replacement by helium the critical temperature fell to the same extent as before so that the power could not be stepped up any higher than 150 kW, because an excessive fall in temperature would have created operational problems.

(d) After each period of reactor operation, the fuel was gathered in the storage vessel which could be weighed in situ. *Fuel weight* could be computed from the weight of the storage vessel filled with settled fuel particles and water, if the specific weight of the $UO_2/ThO_2$ particles was known. This specific weight decreased with time from 10.3 to 9.1 $g \cdot cm^{-3}$, due (among others) to oxidation of the fuel and the accompanying change of the lattice constants. When this density reduction was taken into account, the quantity of fuel calculated from the weight of the storage vessel appeared to have been reduced from 29.1 to 26.4 kg during an operational period of three years (Fig. 35). This could imply that 2.7 kg fuel remained behind in the reactor vessel; a possible explanation for this will be presented in the section on **'Post-operational research'**.

(e) The parameters of the main system fluctuated greatly during the stationary operating of the KSTR. In acoustics terminology such fluctuations are called noise. *Noise measurements* have received much attention.

The nature of the noise in parameter x(t) can be given by the function $W_{xx}(f)$, i.e. the spectral noise density. That quantity describes the intensity of the frequency components of the noise as a function of the frequency.

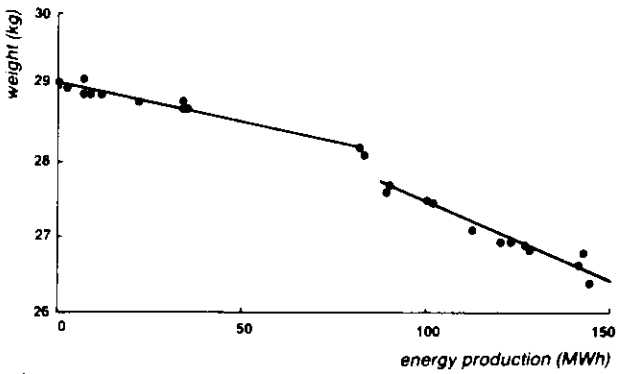



Fig. 35
Weight of the fuel in the storage vessel (recovered situation).

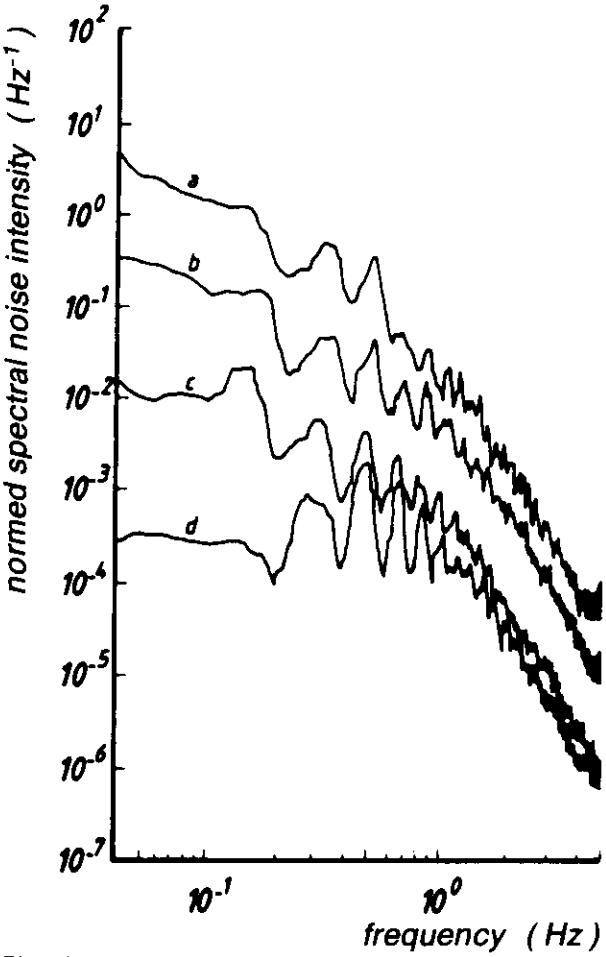



Fig. 36
Spectral noise intensity of the reactor power (in kW) as a function of that power and the burn-up (in MW days per ton).
Curve a: reactor power 20 kW; burn-up 20 $MW.d.t^{-1}$;
Curve b: reactor power 150 kW; burn-up 6 $MW.d.t^{-1}$;
Curve c: reactor power 150 kW; burn-up 260 $MW.d.t^{-1}$;
Curve d: reactor power 1000 kW; burn-up 36 $MW.d.t^{-1}$.

In mathematical terms, $W_{xx}(f)$ reflects the square of the Fourier components of the noise. In the specific case of reactor power, the spectral noise intensity is normalized according to $W_{xx}(f)/P^2$, in which P is the average reactor power. The correlation between the noise and the parameters x(t) and y(t) is characterized by the spectral cross noise intensity $W_{xy}(f)$. As a rule $W_{xy}(f)$ is considered in relation to the noise intensities of x(t) and y(t). One then refers to the coherence function $\delta$ (f), which is defined by the formula

$$\gamma^2_{xy}(f) = \frac{|W_{xy}(f)|^2}{W_{xx}(f) \cdot W_{yy}(f)}$$

The normalized spectral noise intensity of the reactor power is shown in Figure 36 as a function of this power. The strongly fluctuating nature of the reactor power at low power level (20 kW) was expressed by a high value of the noise intensity at low frequencies. Those fluc-

tuations were a result of reactivity fluctuations caused by substantial changes in fuel distribution in the reactor vessel. They developed due to combination of two effects.

Due to the colloid-chemical instability of the suspension, fuel flakes formed with a comparatively high settling rate. As a result there developed in the reactor vessel local increases in fuel concentration, inversely proportional to the local flow rate. Instabilities of the flow pattern led to variations in fuel distribution. The spectral noise intensity of the reactor power then reflected in principle the behaviour of the flow-pattern instabilities as a function of the frequency. The normalized spectral noise intensity of the reactor power decreased substantially with rising reactor power. That was a result of the reactivity feedback caused by the negative temperature coefficient, which became effective at power levels higher than a few tens of kilowatts.

Up to a reactor power level of 150 kW the influence of the negative temperature coefficient was most noticeable in the low-frequency range (Fig. 36, compare curve a with curve b). Increasing and extending it to higher frequencies showed that at a high power level there was, together with the high speed of feedback, a now substantial reduction of noise intensity in the high-frequency range (Fig. 36, curves a, b and d).

When the radiation load was increased the average diameter and thus the settling rate of the fuel particles decreased. As a consequence the local differences in concentration in the reactor vessel decreased and with them the power fluctuations due to flow instabilities in the reactor vessel. The decrease in spectral noise intensity of the reactor power with an increasing degree of fission (Fig. 36, b and c) thus provided an indication of the irradiation damage to the fuel particles. Independent of the reactor power and the status of the fuel, the circulation time of the inhomogeneities in the fuel concentration could be obtained from the frequency. That time was approx. 20% shorter than the circulation time calculated from the flow rate measured in the external circuit and the volume of the main system. This result confirms the presence of a fast flowing layer along the reactor vessel wall, caused by a flow distributor at the inlet to the reactor vessel and installed specifically to prevent fuel particles from settling on the vessel wall. The harmonic frequencies were a logical result of circulation of the concentration fluctuations with their phase differences being a function of the frequency and of the mixing in the reactor vessel.

Changes in reactor temperature were mainly caused by fluctuations in reactor power. The major temperature fluctuations occurred on power changes of some tens of kilowatts and, at a low power level, these changes only arose in the low-frequency range. At low power level there was thus coherence between reactor power and reactor temperature only to approx. 0.2 Hz (Fig. 37-A). At high power level the coherence spreaded to higher frequencies. The harmonic components with a basic frequency of 0.16 Hz mentioned above then also became visible (Fig. 37-B).

The circulating inhomogeneities of the fuel concentration caused load fluctuations in the suspension pump. The fluctuating nature of the absorbed electrical power of the pump was hence a measure of the changes in fuel concentration. This was evidenced by the coherence between reactor power and absorbed pump power. The coherence function clearly shows the circulation frequency of 0.16 Hz together with its harmonics (Fig. 38). The lack of any coherence for very low frequencies is a remarkable feature. It means that slow concentration changes were present in the reactor vessel only and did not appear in the external circuit.

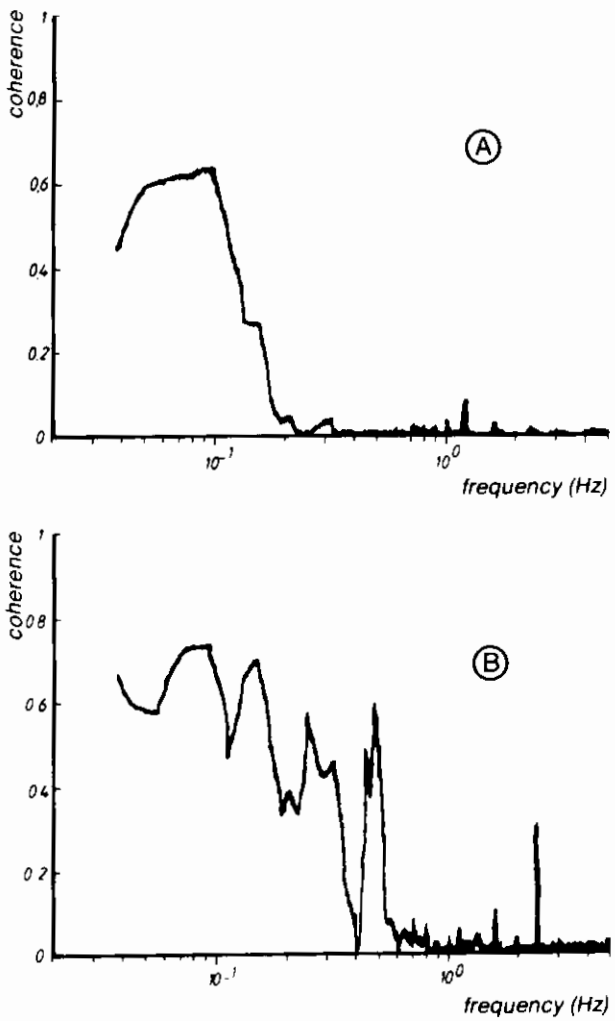



Fig. 37
Coherence between reactor temperature and reactor power.
A: low reactor power (20 kW).
B: full reactor power (1000 kW).

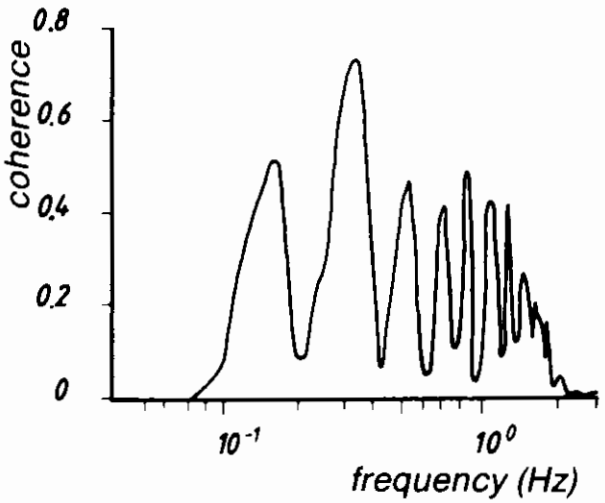



Fig. 38
Coherence between pump load (phase angle $\Phi$) and reactor power.

*Conclusions*

The following conclusions can be drawn from the results of the KSTR experiments.

(1) It was possible to solve the original problems of reactivity changes which were considered too high by installing a flow stabilizer in the reactor vessel.

(2) Even when the reactor was first made critical at low power level the fast action of the feedback due to the comparatively high negative temperature coefficient of the reactivity was obvious. The effectiveness of the feedback rose steeply together with reactor power. Even increases in reactivity above the prompt critical value were compensated completely and rapidly. The reactor was inherently safe as a result of this.

(3) The properties of the fuel changed during extended power operation. On the one hand those changes had a favourable effect on the noise of the reactor but on the other they resulted in loss of reactivity due to which the critical temperature fell in an unacceptable way.

(4) In a reducing environment the radiolysis products were already recombined within the reactor vessel and seldom came out in the form of hydrogen or oxygen. When helium was used as pressurizing gas, internal recombination was much less and $H_2$ and $O_2$ were transported to the external system with the liquid flow. Net production was in proportion to reactor power and amounted to approx. 3 kg $O_2$ per MWh.

**Changes in the physico-chemical properties of the fuel**

The physico-chemical properties of the fuel underwent various changes during operation of the KSTR; the properties affected were fuel density, specific surface area, grain-size distribution, and quantity of erosion products.

*Determination of the fuel density*

The fuel density was determined in an attempt to account for 'lost' fuel. Because of the strong radiation, a method was developed for filling the pycnometer and weighing the material in a manner safe for the operator. The filling device from the mercury porosimeter and type B6 Mettler scales were used for the purpose. Work was carried out with 'remotely controlled' tongs and with lead blocks as shielding. Table 3 provides a comparison of the results obtained with this method and those obtained with the standard laboratory method using natural mixed oxide. The densities of a series of samples from the CH-1 test loop were also determined with these methods. The results are given in Table 4 and Figure 39.

The density of various 'active' samples was measured according to the method described above. The results (also see Fig. 40) are shown in Table 5.

Sample L 77 was divided into two parts to check the influence of sample homogeneity on the density. It appeared that the two parts had slightly different densities, 9.72 and 9.58 $g \cdot cm^{-3}$, respectively. The density of sample L 120 was also measured to examine the influence of any gas adsorbed by the fuel. After determination of its volume the sample was weighed, showing a density of 8.84 $g \cdot cm^{-3}$, whereas sample L 119 (same number of hours of operation as for L 120) had a density of 9.00 g·cm-3. Sample L 119 was measured by the regular method (degasing by vacuum extraction of the pycnometer before topping up), whereas L 120 was not degased. In view of the reproducibility (sample L 77), it cannot be concluded with certainty that adsorbed or dissolved gas leads to an apparently different density.

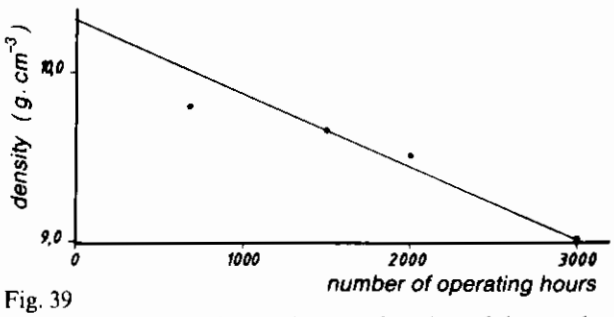



Fig. 39
Density of natural mixed oxide as a function of the number of operating hours in a high-temperature test circuit (C.H.-1).

Table 3
Comparison of fuel-density measurements carried out in the 'hotlab' and in the current laboratory.

| sample | 'hotlab' equipment | normal laboratory equipment |
|---|---|---|
| KSTR-18 | 10.32 $g \cdot cm^{-3}$ | 10.24 $g \cdot cm^{-3}$ |
| GV-60 | 10.20 $g \cdot cm^{-3}$ | 10.12 $g \cdot cm^{-3}$ |

Table 4
Fuel density of samples from the CH-1 test loop.

| sample | operation time | density |
|---|---|---|
| 60 | 688 h | 9.80 $g \cdot cm^{-3}$ |
| 108 | 1500 h | 9.64 $g \cdot cm^{-3}$ |
| 124 | 2000 h | 9.50 $g \cdot cm^{-3}$ |
| 176 | 3000 h | 9.00 $g \cdot cm^{-3}$ |

Table 5
Fuel-density of 'active' samples.

| sample | operation time | density |
|---|---|---|
| L 77 | 1550 h | 9.65 $g \cdot cm^{-3}$ |
| L 89/90 | 1878 h | 9.54 $g \cdot cm^{-3}$ |
| L 110 | 2515 h | 9.27 $g \cdot cm^{-3}$ |
| L 119 | 3104 h | 9.00 $g \cdot cm^{-3}$ |
| L 122 | 3258 h | 9.00 $g \cdot cm^{-3}$ |

Table 6
Determination of the specific surface area of KSTR fuel samples.

| number of operating hours | specific surface area ($m^2 \cdot g^{-1}$) | |
|---|---|---|
| | before etching | after etching |
| 1878 | 0.5[x] | – |
| 2066 | 1.1 | – |
| 2186 | 1.0 | 0.8 |
| 2209 | 1.1 | 0.9 |
| 2259 | 0.95 | 1.7 |
| 2316 | 1.2 | 1.2 |
| 2396 | 0.9 | 0.45 |
| 2464 | 6.2 | 0.7 |
| 2688 | 2.5 | 1.0 |
| 2774 | 11.9 | 0.7 |
| 2866 | 3.5 | 2.3 |
| 3127 | 5.1 | 0.7 |
| 3541 | 6.1 | 3.4 |
| 2788 | 5.6 | 1.5 |
| 4150 | 4.4 | 2.1 |
| 4986 | 8.7 | 3.1 |

[x] This sample was taken immediately prior to the start of critical reactor operation.

*Measurement of the specific surface area of the fuel samples*

The specific surface area of the fuel particles was determined using the BET method with krypton as adsorbing gas. It was assumed that the surface area of a krypton atom is 18.5 $A^2$. The determination was carried out for each KSTR sample both before and after treatment with nitric acid (and hydrogen peroxide). The values obtained are given in Table 6.

The specific surface area of the fuel after sintering was measured for each production batch (1 kg) and varied between 0.12 and 0.2 $m^2 \cdot kg^{-3}$. After 1878 operating hours, i.e. immediately before critical KSTR operation, the specific surface area had increased to 0.5 $m^2 \cdot kg^{-3}$. The increase was probably caused by the adsorption of erosion products on the fuel particles. Electron micrographs established that the fuel particles were covered by an amorphous layer consisting mainly of erosion products. It is remarkable that the specific surface area became approximately twice as large on changing from subcritical to critical operation (after 90 hours of critical operation). No explanation could be found for this. Irradiation damage to the fuel was still minimal at that moment. The specific surface area remained more or less constant for up to 2396 operating hours before etching occurred. The electron micrographs made at this stage also showed globules with still minor irradiation damage.

The specific surface area subsequently fluctuated quite substantially, but despite rather large individual differences it was obvious that the specific surface area had increased. The electron micrographs showed that irradiation damage increased considerably during the critical period. The globules became more jagged and the number of small particles was much increased. One reason for these fluctuations might have been that the sample was not representative of the total amount of fuel as the sampling loop was more or less blocked during this period.

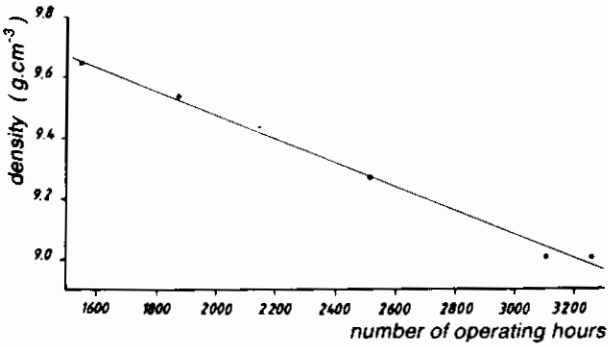



Fig. 40
Density of KSTR fuel as a function of the number of operating hours with the reactor.

*Grain-size distribution in the fuel samples*

The grain-size distribution of the samples in a lead cell could be determined by using a sedimentometer (which has been described in the 'Sedimentation behaviour' section). The distribution can be expressed as $d_{16}$, $d_{50}$ and $d_{84}$ values which indicate the grain sizes below which 16%, 50% and 84%, respectively, of the particles are included. The values are presented in Table 7.

Table 7
Grain-size distribution of KSTR fuel samples after an increasing number of operating hours.

| Number of operating hours | $d_{50}$ | $d_{16}$ | $d_{84}$ |
|---|---|---|---|
| 2066 | 4.92 | 3.19 | 6.69 |
| 2186 | 4.77 | 3.08 | 6.34 |
| 2209 | 4.61 | 2.94 | 6.18 |
| 2259 | 4.68 | 3.00 | 6.13 |
| 2316 | 4.60 | 2.88 | 6.26 |
| 2396 | 4.62 | 2.88 | 6.16 |
| 2464 | 4.56 | 2.55 | 6.06 |
| 2688 | 4.82 | 3.13 | 6.51 |
| 2774 | 4.40 | 2.62 | 6.03 |
| 2866 | 4.50 | 2.65 | 6.21 |
| 3104 | 4.37 | 2.51 | 6.03 |
| 3260 | 4.26 | 2.17 | 5.84 |
| 3788 | 4.56 | 2.71 | 5.97 |
| 4160 | 4.60 | 2.00 | 6.12 |
| 4986 | 4.19 | 2.74 | 6.24 |

Despite the fluctuations, it could be concluded that the value of $d_{50}$ decreased with the number of operating hours and that $d_{16}$ became smaller at the same time. Thus the number of smaller particles increased slightly. This conclusion is similar to that reached on the basis of the results obtained from the electron micrographs and from measurements of the specific surface area (see previous section).

*Erosion products*

The quantities of iron, nickel and chromium were determined in various suspension samples. It was assumed that all erosion/corrosion products formed were either adsorbed to the surface of fuel particles or circulated as solids along with the fuel. The correctness of the assumption was confirmed by determination of the iron, chromium and nickel content of the water. The quantities of dissolved erosion/corrosion products in the water proved to be negligible compared to the quantities on the particles.

The quantities of corrosion/erosion products adsorbed to the fuel were plotted (Fig. 41) versus the number of hours of operation with the suspension. The KSTR system was first tested with natural mixed oxide. Critical operation started with enriched mixed oxide, after 1878 hours of subcritical operation. The figure shows that the quantities of iron, nickel and chromium increased more or less in proportion with time during

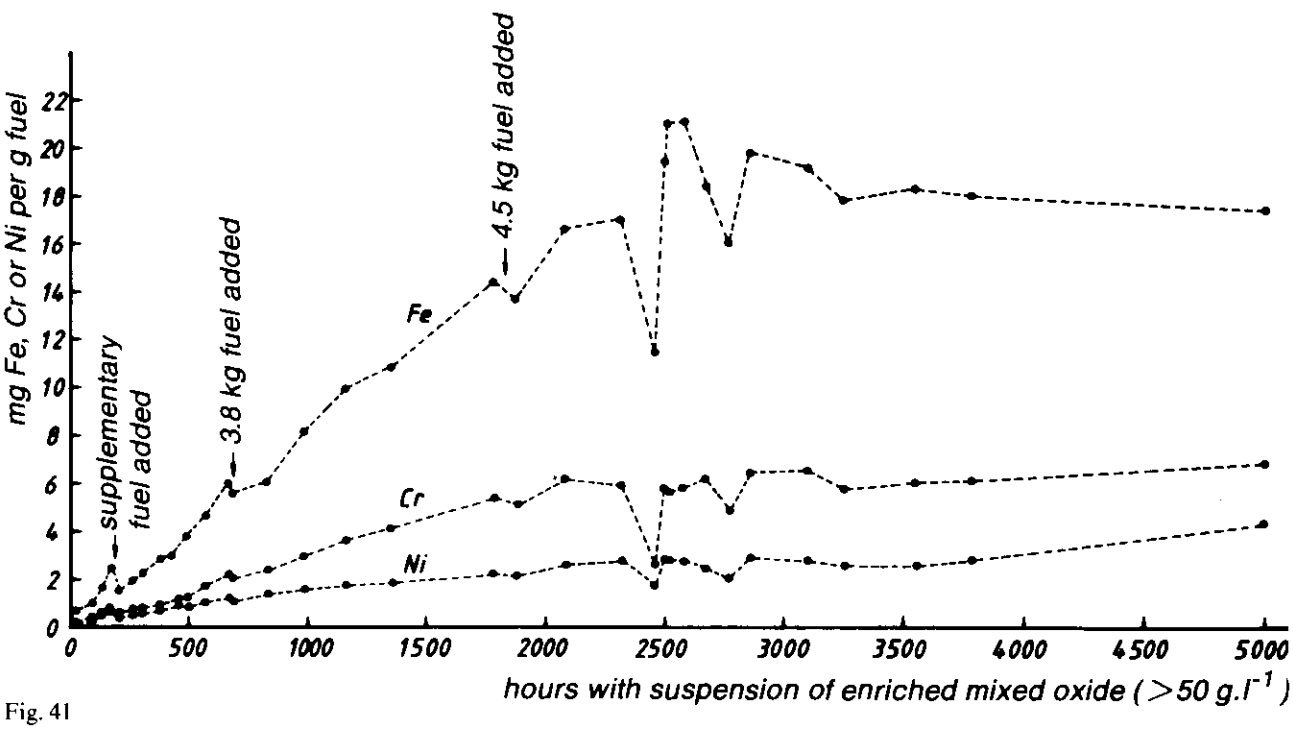



Fig. 41
Corrosion-erosion products adsorbed to the fuel as a function of the number of operating hours with suspension.

subcritical operation, with the exception of the periods when supplementary fuel was added.

The distribution of erosion products changed during critical operation. The two minima after 2464 and 2774 operating hours are noteworthy. The following is a tentative explanation. Initially the part of the natural mixed oxide left in the system at the time of changing over to enriched mixed oxide would not have continued to circulate. The natural mixed oxide would, however, have somehow become detached and would have started to circulate. Its smaller content of erosion/corrosion products compared to that of the enriched mixed oxide would explain the lower quantities of iron, nickel and chromium. Mass spectrometric determinations showed, however, that the ratio U-238/U-235 was equal to that in the two batches of enriched material when they were received (0.1055 and 0.1084). The ratio was still the same at the end of KSTR operation, so a different explanation was required. It should be noted that the two samples showing the minima had been taken immediately after the occurrence of net radiolysis. In such a case, recombination of the radiolysis products within the reactor vessel progressed at a lower rate than their generation and oxygen thus escaped. The net yield of oxygen could be determined by having a calibrated recombiner in the gas system.

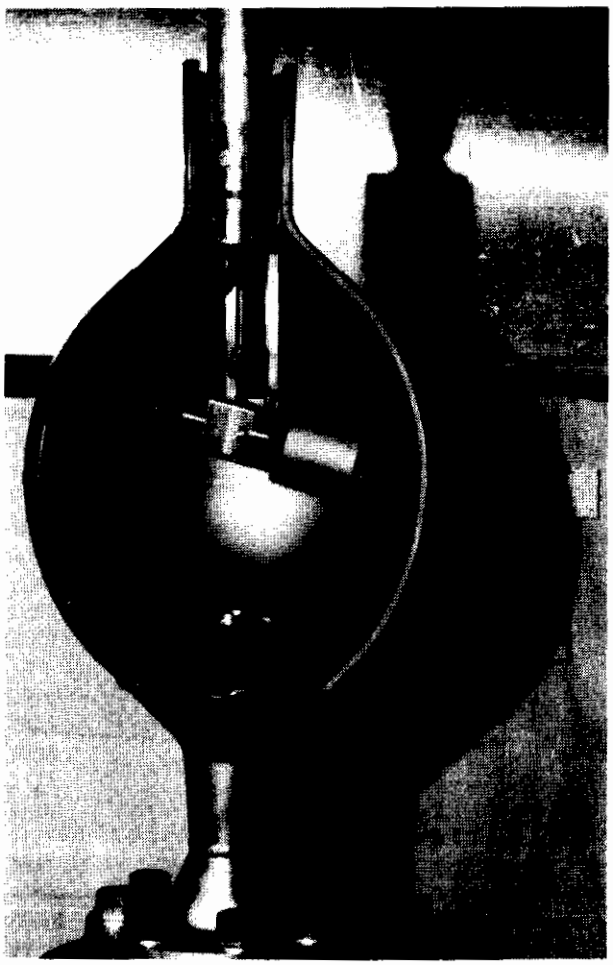


Fig. 42
Manipulator with mini TV camera in cutaway model of the reactor vessel.

The erosion products absorbed to the fuel particles may have become detached under the influence of the excess oxygen then present but the difference in erosion/corrosion products between the two relevant samples and the other samples could not be accounted for in the water, so deposition probably occurred somewhere in the system. The erosion/corrosion products must have again ended up on the fuel particles because the usual adsorbed quantities were found later.

Figure 41 shows that erosion decreased during critical operation, probably due to softening of the fuel particles as a result of irradiation damage.

**Post-operational research**

Post-operational research on the KSTR was primarily aimed at evaluating the mechanical behaviour of the nuclear part of the plant and locating the 2.7 kg fuel which, according to the fuel balance, had not been collected in the storage vessel after shut-down of the reactor and hence had remained in the reactor system. This research required the nuclear systems to be partly opened and inspected visually at a distance.

*Analysis of mechanical behaviour*

The first parts whose mechanical behaviour was to be evaluated were: the reactor vessel, the flow stabilizer, the main circulation pump and a suspension valve.

*Reactor vessel* – The gas-tight main compartment was opened for internal inspection of the reactor vessel by removing some of the concrete shielding blocks and the gasket sheet. The combination of gas-injection head and integral flow stabilizer was subsequently removed from the reactor vessel, by using remotely operated tools. A mini TV camera was introduced with a manipulator specially designed for the purpose. The images of the internal wall of the reactor vessel were recorded on videotape. Figure 42 shows the manipulator with the mini TV camera in a cutaway model of the reactor vessel. The results of the inspection may be summarized as follows:

(1) no deformations of the vessel occurred;
(2) no fuel had been deposited on the vessel wall;
(3) there was no erosion of wall or inlet piece;
(4) a slight scaling of the inlet piece was observed and there were two small pieces of fuel (each approx. 2 $cm^3$) on the inlet piece.

It could be concluded on the basis of these findings that the reactor vessel satisfied both the expectations and the requirements that had been set.

*Flow stabilizer* – Pictures of the flow stabilizer were taken after it had been removed from the reactor vessel. Fuel had deposited on the outside of the stabilizer at two locations diametrically opposite each other (Fig. 43). The remainder of the outside was clean and undamaged, but much more fuel was deposited on the inside of the stabilizer. X-ray pictures showed that the deposit extended over 75% of the length and formed such a thick layer that there remained only a channel with a diameter of 1-2 cm open in the centre of the stabilizer. The stabilizer was sent to the Belgian Study Centre for Nuclear Energy at Mol for further assessment. This assessment involved making a cut in a longitudinal direction through the stabilizer (Fig. 44). The results of the work at Mol may be summarized as follows:

(1) the amount of fuel (including erosion products as well) in and on the stabilizer was 2046 g;
(2) the fuel appeared to be very porous in the central part of the layer. The pores were more or less radially oriented. The fuel no longer consisted of the original 5 $\mu$m particles; re-sintering or re-crystallizing had apparently taken place;
(3) the outside surfaces of the thick fuel layers (on the stabilizer wall and on the sides of the channel which remained open) were much less porous and actually showed only cracks. 'Loose' particles could still be distinguished in those outer layers.

*Main circulating pump* – Research on circulation pumps in high-temperature/high-pressure test circuits with suspensions of natural uranium-thorium oxide, done prior to the KSTR period, had shown that the ordinary commercially available pumps showed unacceptable erosion damage after only a very limited number of operating hours. The Technological Development Group of the KEMA Nuclear Reactor Laboratory therefore devoted much time and effort to modifying such a pump so as to make it suitable as far as construction shape and material were concerned. The result was a hydraulically adapted pump with a 30% Cr-steel impeller and a 347 steel pump casing with a 30% Cr-steel insert. This pump was expected to survive 5000 operating hours with suspension and would then have to be replaced by a spare unit.

At the end of the KSTR experimental program there had been 5131 operating hours with suspension and at no time had there been any indication of faulty functioning of the main circulation pump or of excessive wear of the impeller or pump casing. That could be established from the Cr content in the suspension samples regularly taken from the main circuit. After KSTR operation, the pump unit, together with the cooler and measuring facilities, was disconnected from the reactor system, lifted as an entity from the cell and transferred to the inspection room for internal inspection. The seal weld between pump casing and motor compartment was then chipped out by means of pneumatic tools. The motor compartment was next lifted sufficiently to allow inspection of pump casing and impeller. A picture of the pump casing is shown in Figure 45. Visual inspec-

Fig. 43
Fuel deposits on the outside of the flow stabilizer after shutdown of the KSTR operation.

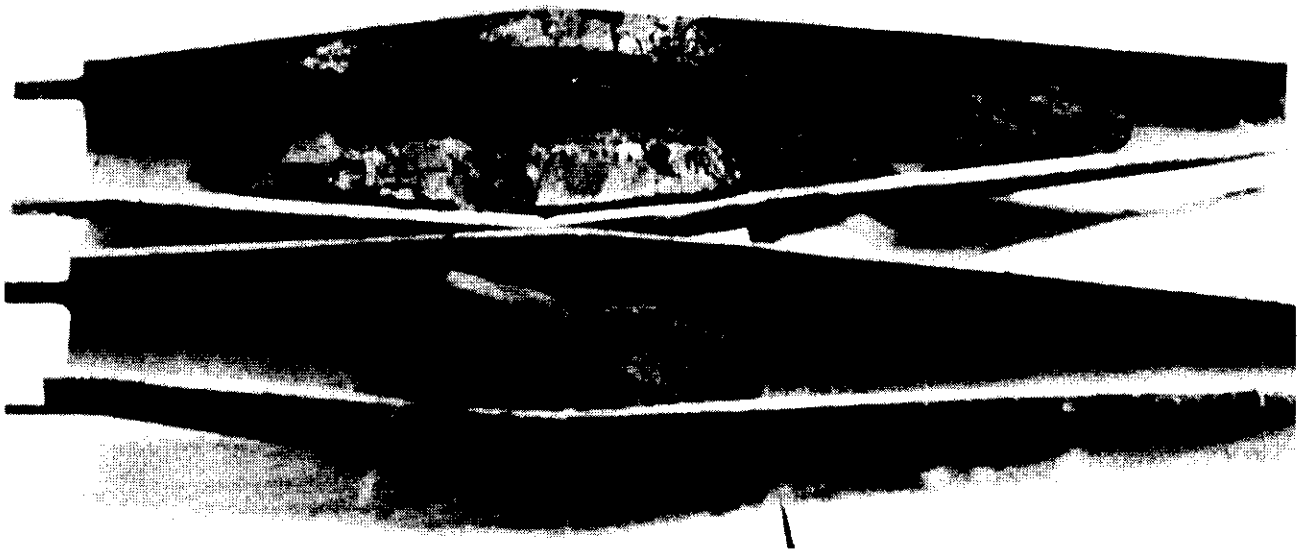
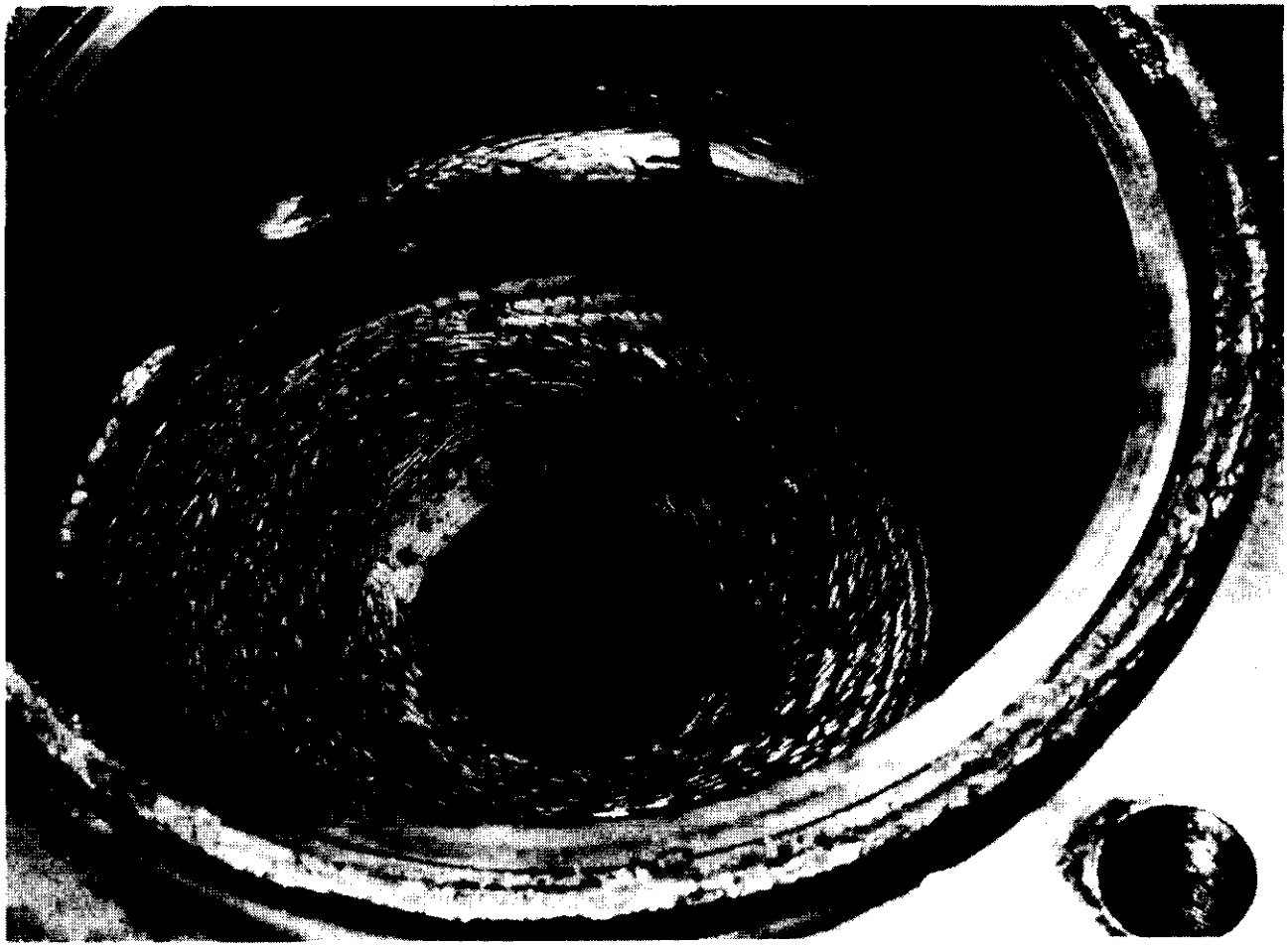


Fig. 44
Picture of both halves of the cut flow stabilizer. The fuel deposits had partly disappeared during cutting.

Fig. 45
Pump casing of the main circulation pump after more than 5000 operating hours with suspension.



tion and the pictures taken demonstrated the following:

(1) the impeller showed almost no erosive attack and only minor erosion at the vane tips. No fuel deposits were visible on the impeller;
(2) the pump casing showed erosion in the form of spirally shaped grooves with a maximum depth of 3 mm in the horizontal part. The vertical walls were attacked only in the immediate vicinity of the outlet opening (the impeller rotated around a vertical axis);
(3) based on the data from the last suspension sample taken (6.9 mg Cr per gram fuel) and assuming that all chromium originated from the grooved part of the pump casing (276 $cm^2$), an average reduction of the wall thickness by 2.3 mm must be assumed. Since the local wall thickness was 70 mm, that implied a 3.3% reduction;
(4) the performance of the pump was maintained because the impeller was not attacked;
(5) after 5000 operating hours the behaviour with regard to erosion was no worse than could have been expected on the basis of experience with test circuits using the adapted shape of the pump.

*Suspension valve* – The most intensively used valve, i.e. the cyclone inlet valve, was selected for internal inspection of a suspension valve (number of hours with flow 200, number of open/close switchings 500). No leakage had occurred through the valve during operation at a differential pressure of 2 MPa. The valve was disconnected by remote manipulation, was lifted from the cell and then partly dismantled by chipping off the seal weld. Inspection showed the following:

(1) the valve body and seat (both of tungsten carbide) were neither eroded nor damaged;
(2) a thin scaling was present on the valve body and sealing below it.

*Decrease of critical temperature and of weight of the fuel collected*

An attempt was made above (*'Critical operation and experimental research'* section) to explain some of the less favourable phenomena, i.e. (1) loss of reactivity (decrease of the critical temperature) at power levels above 40 kW, (2) gradual weight reduction of the fuel quantity collected after operations up to a final figure of 2.7 kg, and (3) blockage of the sampling loop by more or less globular fuel lumps with a diameter of 2.5-3 mm.

Since 2046 gram fuel had been traced on and in the flow stabilizer during the post-operational research, the weight reduction of the fuel quantity collected was largely explained. The balance may perhaps be found during further decommissioning of the reactor as the above mentioned lumps seen at the three lowest locations in the suspension circuit. In view of Figure 43, it is likely that those lumps originated from the primary deposits on the stabilizer and were dislodged by the flow. After circulating for a short time the comparatively heavy lumps would have been deposited at locations with a low vertical flow rate. The large lumps could not cross a local narrow gap in the sampling loop.

The fact that primary deposition of fuel occurred exclusively in and on the stabilizer is easily explained as the conditions necessary for fixing to the wall only existed there: i.e. high temperature, temperature gradients and a low flow rate. Flow measurements on a reactor-vessel test model revealed that the flow rates within the stabilizer were comparatively low over the entire length, whereas the flow rates on the outside were low only locally and temporarily. Because of this deposits within the stabilizer occurred over the entire length, but at only two locations on the outside. The fact that deposits only occurred above a certain power level during critical operation can be explained by the fact that only then did temperature gradients develop due to local differences in heat generation (flux distribution) and due to heat removal (determined by the flow rate). Moreover, an important part was played by the presence of erosion products as well as by the reversible conversion of ferro- into ferri-ions under the influence of radiolytically developed oxygen. Changes in the colloid-chemical properties of the suspension due to irradiation damage of the fuel particles may also have had an effect. All those effects depended on the reactor power and partly on the energy supplied during operation. The reversibility of some of the processes explains why reactivity recovered somewhat at low power level or during subcritical operation.

**Conclusion**

Based on the results obtained during the post-operational research, it can be stated that the expectations for the reactor were for the greater part fulfilled.

It appeared that the reactor could be operated reliably, and – furthermore – the reactor appeared to be inherently safe, due to the prompt feedback of the high negative temperature coefficient of reactivity. Even reactivity increases above the prompt critical value proved to be compensated completely and rapidly. Also the relative power fluctuations ($P/P_o$) decreased strongly with increasing power density.

The mechanical behaviour of the parts examined post-operationally also satisfied expectations.

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