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Report over the

4th quarter and the year 1974

on the

Aqueous Homogeneous Suspension
Reactor Project

REACTOR DEVELOPMENT GROUP

N.V. KEMA

ARNHEM - NETHERLANDS

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INTRODUCTION

In the year past a culminating-point has been reached in the history of the KEMA Suspension Test Reactor.

After the reactor license had been issued on May 1st, the reactor became critical on May 22nd. Thereafter the reactor power has been increased gradually from about 3 kW in the first experiment up to 200 kW at the end of the year. The nuclear behaviour of the critical reactor proved to be well within the range of stability expected from subcritical experiments. The components of the reactor system all behaved very well. The containment of the reactor, the system itself, the compartments and the reactor hall proved to meet the leak tightness requirements without any difficulty. The new flexible containment sealings, which have been approved by the safety authorities, performed well. Suspension samples from the reactor showed a gradually increasing amount of erosion products, but no significant changes of the fuel particles. The distribution of fission products over the water phase, the gas phase and the inner and outer part of the fuel particles could be roughly determined.

The reactor power could be calibrated by determinating the amount of protactinium and cerium in the fuel samples.

I. KSTR OPERATION

I.1. Results of experiments with the KSTR

I.1.1. Preparation for critical operation

In the first quarter of 1974 the final preparations were made for critical operation of the reactor. The leaktightness of the containment systems was improved and the leak rates were determined both at over- and underpressure. The flow rates and pressure differences in all sections of the ventilation system were measured and adjusted. The radioactivity meters of the compartments were replaced and calibrated. The neutron flux detectors for critical operation were installed and calibrated.

Fuel rests outside the circulated part of the suspension system were recovered. Traces of chlorine in the reactor water were removed and the fuel itself was rinsed. The weighing device of the storage vessel was recalibrated and the fuel inventory was determined.

A new sampling valve was installed and tested. All electrical safety systems were tested and, if necessary, improved.

The reactor systems became ready for critical operation in March.

I.1.2. Approach to criticality

On May 1st the reactor license for critical operation was issued. During the first attempt to reach criticality the reactor was automatically shut down by an emergency stop, caused by short circuiting in the power supply of the reactor sub-systems. In the sequence of events which preceded the dump, a small part of the suspension had been transferred to the gas system. During the dump itself the remaining part of the suspension was transferred to the dump vessel. Although this involuntary test showed the reliability of the safety systems, this incident caused a delay of several weeks in reaching criticality. Two more attempts failed because of reactivity peaks at an almost critical reactor (-100 pcm). The reactor scrammed at the alarm setting of 100 kW.

I.1.3. Critical operation

On may 22nd the alarmsetting of the reactor power was enhanced to 1 MW. At a reactor temperature of 255°C 23.8 kg fuel was introduced, the flow rate of the suspension was decreased, the neutron absorbing rod was pulled partially out of the reactor and the reactor temperature was decreased to 248.2°C. The automatic temperature control was switched off and the heat input was reduced with 11.5 kW. Subsequently the reactor temperature decreased to about 246°C and at this temperature the reactor became critical with a power level of about 3 kW. The reactor power proved to be very unstable with power excursions, reaching values up to 500 kW within 1 second. This could be expected as the feed back of the negative temperature coefficient on the reactor power does not work fast enough at such a low power level.

During the rest of the year the reactor power has been stepwise increased up to 200 kW.

Table 1 gives a survey of the periods of critical operation.

Table 1.

Date	Time	Mean power level (kW)	Energy production (kWh)
74-05-22	12.08 - 13.14	2.4	2.6
74-08-27	11.24 - 11.28	8.5	0.6
74-08-28	20.06 - 20.20	16.7	3.9
74-08-29	22.09 - 23.48	28.5	40.2
74-09-17	20.19 - 21.38	41.9	52.8
74-09-19	13.28 - 14.53	41.2	51.2
74-09-19	21.16 - 22.42	33.5	47.4
74-09-20	11.25 - 16.51	40.9	202.7
74-10-10	12.25 - 17.31	37.6	191.5
74-11-05	23.32 - 00.11	-	3.5
74-11-06	11.39 - 12.48	40.0	46.2
74-11-06	16.58 - 10.30	-	4.5
74-11-06	20.32 - 00.26	41.3	162.3
74-11-07	11.19 - 15.29	30.8	128.4
74-12-10	16.57 - 22.23	100.0	503.3
74-12-11	10.58 - 15.22	150.0	411.4
74-12-12	12.13 - 15.31	214.0	476.5
			2329.0

I.1.4. Results

The reactor power has been increased up to 200 kW without any problems. Heaters have been switched off until the reactor compensated its own heat losses. Thereafter coolers have been switched on to reach the desired reactor power.

In fig. 1 a recording is given of the reactor power during the experiment of 74-12-12, at which the reactor was operated at a mean power level up to 220 kW. The fluctuations of the power level are relatively small, compared with the fluctuations at low power levels (cf. preceding quarterly report). This indicates that the nuclear stability increases with increasing power level.

This was also evident from the analysis of the power excursions, which occurred spontaneously from time to time. The ratio P/P_o (P = peak power, P_o = initial power level before the excursion occurs) decreases strongly with increasing values of the mean power level.

In fig. 2 the ratio P/P_o has been plotted versus P_o. The experimental points have been obtained from the analysis of the power peaks. The curve in fig. 2 has been calculated, assuming a reactivity addition of 500 pcm in 1 second and a decrease to $\rho = 0$ during the next second.

From this figure it can be seen that the experimental values are following the slope of the calculated curve very well, and that the ratio P/P_o decreases strongly with increasing initial power level. This means that the negative temperature coefficient feed back increases strongly with increasing power level, and that the nuclear stability will still improve at power levels over 200 kW.

By plotting the reactor core inlet temperature versus the critical fuel mass in the reactor system at standard power conditions (40 kW), it could be proved that the behaviour of the reactor has not changed during this year.

The temperature difference between inlet and outlet of the reactor core shows a tendency towards a too low value with respect to the reactor power level measured by the neutron flux detectors.

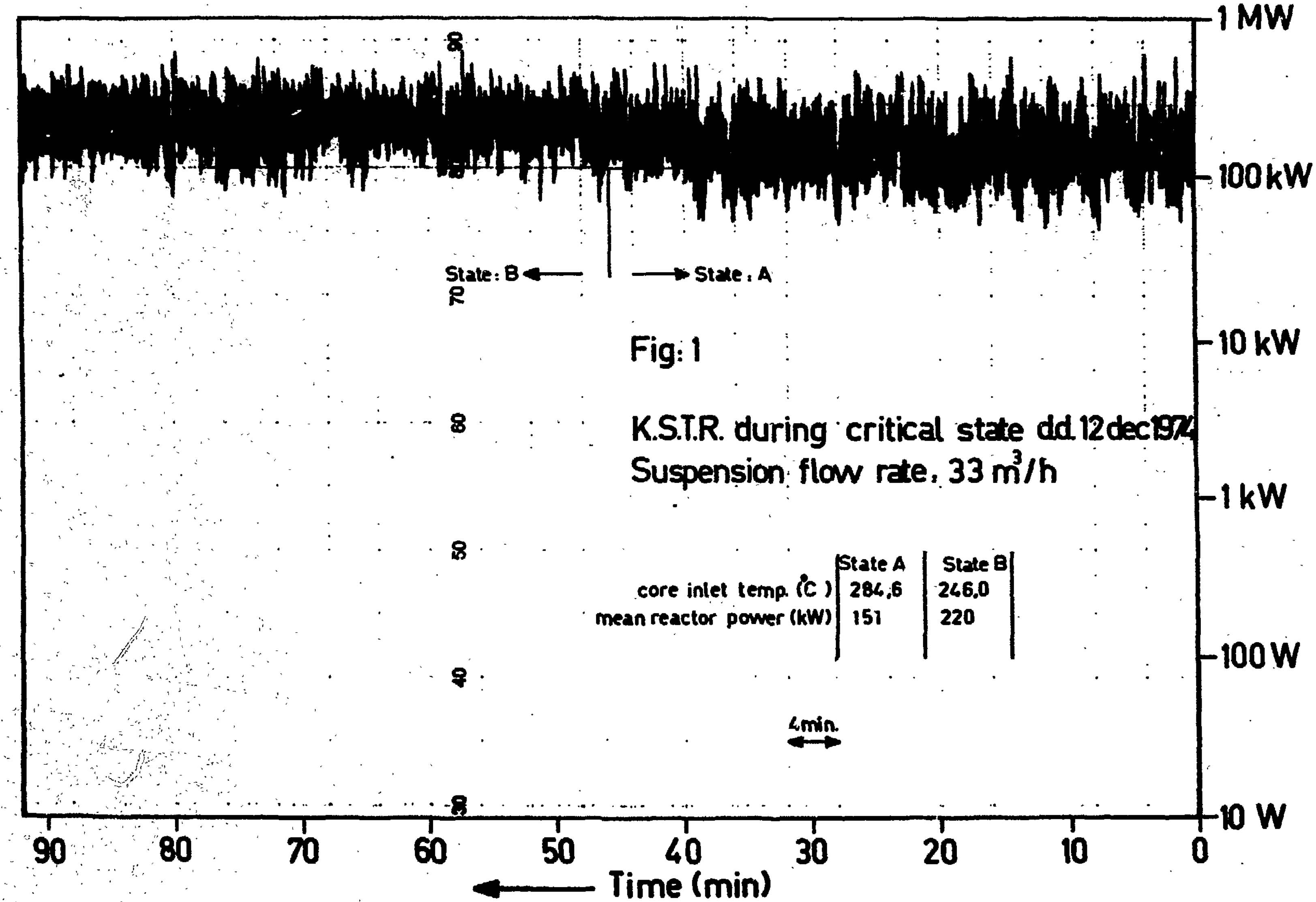
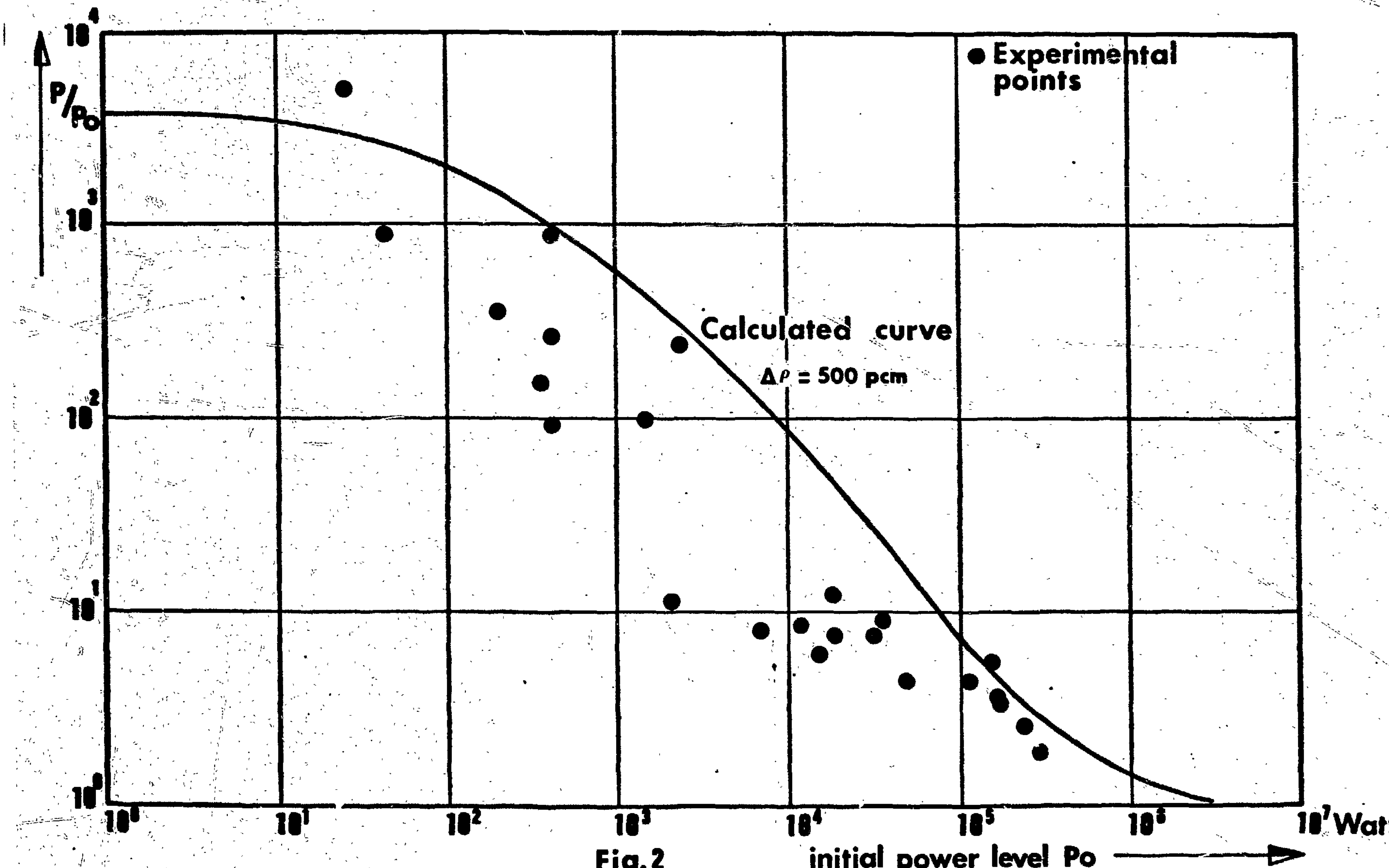


Fig: 1

K.S.T.R. during critical state dd.12dec1974
Suspension flow rate. 33 m³/h



Experimental points
Calculated curve
Δρ = 500 pcm
P/Po
10^4
10^3
10^2
10^1
10^0
10^0
10^1
10^2
10^3
10^4
10^5
10^6
10^7 Watt
initial power level Po

Fig. 2

Also the heat exchange between main system and cooling system indicates a lower value. To solve this problem a fuel sample has been taken from the reactor system after a well-known power production to check the power calibration of the neutron detectors by means of activation analysis.

The influence of the reactor power fluctuations on the behaviour of the process parameters is very small, except for the variations of the liquid level in the gas separator. During large power excursions the alarm setting at 280 cc above the mean liquid level is reached and a continuous surveillance of this parameter is necessary.

No radiolytic gas could be detected in the gas system, even under special measuring conditions.

To minimize the influence of fluctuations in reactor power and heat exchange in the main system, two control systems have been tested. The first system consists of a neutron absorbing rod, the movement of which is controlled by the neutron flux in order to minimize the power fluctuations. This automatic control system showed to be appropriate to minimize the power fluctuations as well as the temperature fluctuations in the main system. As a consequence the liquid level variations decreased too.

The second system consists of a controlled heat exchange in the main heat exchanger. This system is successfully used to decrease measuring errors in the heat balance of the reactor system.

I.2. Operation of the reactor

I.2.1. Operational performance

Operation of the reactor in the year past can be divided into three periods:

1. Preparations for critical operation of the reactor;
2. First attempts to make the reactor critical;
3. Critical-state operation with stepwise increased reactor power.

Ad. 1. These preparations mainly consist of:

a. Overall check, cleaning and leak testing of nuclear systems and compartments.
b. Determination of the total fuel inventory.
c. Leak test of the reactor hall.
d. Testing of the safety functions of the reactor.
e. Testing of the procedures to reach criticality and the critical state operation of the reactor.

Ad. 2. The first attempt to reach criticality failed, because the core vessel pressure safety systems blew off (cf. quarterly report, 2nd quarter 1974). The second attempt was more successful and the reactor went critical on the 22nd of May for a period of about 1.5 hour with a mean power of 3 kW.

During the period June-August some components of the gas purification system had to be opened in order to recollect the fuel, which had been transferred to the GPS during the blowing off of the core vessel safety systems.

Ad. 3. During the period August 27th - December 12th the reactor has been made critical several times. The power level has been increased stepwise to 40-100-150 and 200 kW, respectively.

The last step of the procedure to reach criticality is to switch off 40 kW of heat power. As a result the temperature of the main system decreases slowly and the reactor becomes critical at a power level of 40 kW.

This item of the procedure was altered and now this last step is to switch on control heat exchanger HE-11, so that 40 kW will be extracted from the main system. The temperature decreases now at a rate of 1°C/min and the reactor becomes critical in less than 5 minutes.

To increase reactor power from 40 kW up to 160 kW, heat power has to be switched off and to increase reactor power from 160 kW up to 1000 kW, the base power heat exchangers HE-9A/B have to be switched on.

The only drawback in the operation of the reactor was caused by the level indicator of the liquid level in the gas separator. The instrument indicates the required overpressure of a gas flow bubbling through the gas separator. This gas system repeatedly became plugged and the level control had to be switched over to a less accurate level indicator, which needed continuous attention.

I.2.2. Instrumental performance

I.2.2.1. Modifications of electric instruments

To prevent a new blow-off of the gas system of the reactor, the electrical fuses of this system have been regrouped in such a way that the failure of one fuse cannot cause a reactor blow-off.

A radiation monitor in the off gas system gave several false alarms because of direct radiation of the reactor. The monitor has been shielded.

Another monitor which measures the iodine activity in the ventilation air to the stack has been provided with a thermostat in order to obtain the required accuracy. During power operation the battlement-shaped changes in the neutron flux recording also became visible on the recordings of the temperature differences between several thermocouples on the reactor core wall. These temperature differences seemed to be proportional to the power level of the reactor. To enable the study of this phenomenon the measuring facilities for these thermocouples have been improved.

I.2.2.2. Additional reactor instruments

To improve the security of the reactor site all reactor entrances have been provided with door signals which are visible on a display in the control room. Some other measures have been taken to meet the new Dutch security requests.

An electric barometer has been installed, which was needed for the interpretation of reactor hall leaks.

Safety in the hot lab has been improved by installing a better alarming system and by mounting a pressure switch. ' th this switch the air pressure in the hot lab can be increased if the underpressure is too high to open the door.

The signal of the meter which measures the pressure difference in the transport tube of the sampling system has been transduced to the control room. Here the pressure difference can be adjusted.

I.2.2.3. Maintenance and repairs

Besides the routine maintenance which was carried out in accordance with the computer programme, hardly any maintenance work was necessary. Several meters have been recalibrated, f.i. some space monitors and the power indicator of the control heater in the main system.

The flux control by the neutron absorbing rod could be connected to a linear flux measuring channel instead of the logarithmic channel. This improvement was possible because of the better feed back of the negative temperature coefficient of the reactor at higher power levels.

I.2.3. Mechanical performance

Apart from the routine maintenance and some minor repairs the mechanical group has been busy with the recovering of the lost fuel. The fuel which had been transferred to the gas system during the gas blow off in the middle of the year has been regained. For this purpose the compartments and part of the primary system had to be opened. After closing the leaktightness had to be checked and the results proved to meet the specifications.

During critical operation it appeared that there was some leakage of γ-rays through a part of the east wall of the reactor building. Therefore this place has been covered with a layer of earth. Also a γ- and neutron leakage was detected through the instrument thimble. To thermalize the neutrons, the thimble has been filled up with paraffin, and a shield of boron and lead has been installed outside the thimble.

The blower of the recombination system of the compartment air does not meet the leak requirements. This system will be improved.

II. REACTOR PHYSICS

II.1. Theoretical work

II.1.1. Uranium consumption of a 250 MWe suspension reactor

Calculations have been made on the total uranium consumption of a 250 MWe circulating suspension reactor in 30 years. The ratio of core volume to outer volume of the reactor and the fuel concentration have been varied. The results indicate a lower uranium consumption at a high fuel concentration and a high ratio of outer volume to core volume. These results can be explained by a better conversion ratio under these circumstances.

II.1.2. Fuel cycle costs

The fuel cycle costs of a 250 MWe circulating suspension reactor have been calculated. The ratio of core volume to outer volume, the fuel concentration and the fuel cycle duration have been varied.

The results indicate lower fuel cycle costs at lower concentrations and a longer cycle duration. Th and U prices have hardly any influence. Very important are the costs of particle fabrication, reprocessing and separative work.

II.2.1. Transient behaviour of the KSTR

When the reactor is shut down at a high power level by inserting the safety rods (reactivity worth 1200 pcm), the main system of the reactor will be cooled down very rapidly, because of the low temperature in the primary cooling system. The decrease in concentration by the hydrocyclone is too slow to compensate the effect of the negative temperature coefficient of reactivity. Therefore, after a stop the reactor may become critical again at a lower temperature. In order to be able to predict the behaviour of the reactor after a fast regular stop, calculations have been made on an analog computer.

The reactor power, the temperature in the core, the mean temperature in the main loop and the temperature in the primary cooling system have been calculated as a function of time after a fast regular stop. The results are in good agreement with the observed phenomena in the reactor. At higher power levels the procedure for a fast regular stop has to be changed.

II.2.2. Heat balance of the KSTR

Several attempts have been made to improve the heat balance of the reactor during subcritical and critical operation. The power level as indicated by the neutron flux detectors has been compared with the concentration of some isotopes in samples and with the heat balance. It is expected that a better agreement between these methods will be achieved at higher power levels.

III. CHEMISTRY

III.1. Fuel irradiation

III.1.1. Irradiation in water at high temperature

Introduction:

The previous annual report indicates already the reason why we irradiated two kinds of fuel. A model fuel (UO_2-ThO_2 15%-85%), which has more or less the composition of the KSTR fuel (25-75%), obtains a certain burn-up in a shorter time than the other fuel (UO_2-ThO_2 1.5%-98.5%). Consequently, the activation of the capsules will be less, which facilitates post irradiation work. The second fuel is called power reactor fuel since it has the composition which is foreseen for a power reactor, based on the principle of a suspension reactor.

a. Model fuel ($(U_{0.15}Th_{0.85})O_2$)

a.1. The LISA-1 experiment.

The first part of the post irradiation examination has already been described in the previous annual report. In this year the investigation was terminated.

Some electron- and X-ray pictures were taken from two samples, containing irradiated fuel, corrosion products and platinum black. The irradiated solid material consists partly of 5 μm spheres and irregularly formed pieces with a size of 5-10 μm. From the pictures it appeared, that a part of these irregularly formed pieces contained uranium and thorium and another part iron, nickel and chromium. Also some platinum was found.

a.2. The LISA-2 experiment

The purpose of this experiment is to examine the irradiation damage of the 5 μm fuel particles in the temperature interval of 250°C and 310°C, while the burn-up varies between 3000-8000 MWd/t mixed oxide. During the irradiation the thermocouple of one capsule failed, while the thermocouple of another capsule was positioned on the boundary level of water and gas space.

No difficulties were experienced with the other capsules. Two irradiated capsules are already transported to Arnhem. Post irradiation examination will probably start in January 1975.

When the investigation of LISA-1 and -3 is terminated, we hope to be able to have a picture of the behaviour of the model fuel between 250°C and 310°C and between 3000 and 8000 MWd/ton mixed oxide. By then we also have a confirmation whether our present idea is sound that the model fuel indeed allows us to obtain reliable information for the power reactor fuel.

b. Power reactor fuel $((U_{0.015}Th_{0.985})O_2)$

The LISA-2 experiment.

The post irradiation examination of the capsules has nearly been finished. Only the examination of the solid material by means of a micro probe still has to be done.

From the examination of the irradiated suspension it was found, that:

- on the platinum wires of all the capsules, except one, a knob was present. The composition of these knobs is about 97% thorium, 1.5% uranium and 1.5% platinum.
- the amount of uranium and thorium dissolved during etching, varied from 1-10% (with respect to UO_2 and ThO_2).
- electron microscope pictures showed both spheres and pieces (size 5-10 µm. Probably a part of the pieces contains uranium and thorium, while the other part consists of iron, nickel and chromium (see LISA-1 experiment).

A micro probe examination of the irradiated solid material must yet be done.

c. The LISA-4 experiment

The purpose of this experiment is to test the irradiation damage of some other fuels which have been discussed for different purposes:

- 5 µm UO_2 particles containing 0, 5 or 10% ThO_2, which could be used in a Pu-producing suspension reactor.
- 800 µm UO_2 particles which offer possibilities for a suspension reactor with a fuel-free out-of-core system.

The burn-up will be from 2000-8000 MWd/ton oxide. Probably the irradiation will start at April, 1975.

III.1.2. Irradiation at low temperatures

Irradiation experiments have been carried out in vacuo with a low neutron dose ($\simeq 10^{15} n/cm^2$). The Nu-values scatter much, but we may conclude that at very low doses ($\leq 10^{14} n/cm^2$) these disintegration rates will be constant. The specific surface of the fuel is hardly changed.

We remind that the amounts of uranium, dissolved during the etching procedure, is expressed as the number of uranium atoms per passage of a fission fragment through a particle surface. This number will be called the Nu-value.

III.2. Development of an adsorbent for fission fragments

Results of thorium oxide fines, coated with a layer of thorium phosphate are disappointing. The larger part of these fines were adsorbed on the surface of the 5 µm particles during autoclaving at 250°C.

Then some experiments were carried out to examine the adsorbing properties of the fines, compared with those of the 5 µm particles. This was done by means of an autoclave experiment at 250°C with a very diluted radioactive cerium solution. We found that a small part of the cerium has been adsorbed onto the fines, but the larger part on the wall of the autoclave. However, quantitative conclusions cannot be drawn.

Still another experiment has been performed to determine the adsorbing properties of the thorium oxide fines: some KSTR water, containing sufficient fission fragments, was added to a suspension of 5 µm mixed oxide particles and thorium oxide fines. Then this has been autoclaved at 250°C.

After this heat treatment the mixed oxide and the fines were separated and by means of a γ-spectrometer the quantities of fission fragments, adsorbed on the mixed oxide, and the thorium oxide fines, resp., were measured. Results cannot yet be given. However, as preliminary conclusions we can say:

- we can make thorium oxide fines which only for a small fraction adsorbe on the 5 µm particles during autoclaving for two weeks at 250°C;
- the adsorption of fission products on these fines needs further investigation;
- we have no information available concerning the dynamic properties of the fines.

III.3. Hot laboratory activities

The post irradiation examination of the LISA-2 capsules and of 15 KSTR samples has been done. An improvement of the apparatus makes it possible to perform grain-size determination on irradiated fuel in the hot cell now.

Of several KSTR samples we determined the amount of erosion/corrosion products on the particles after 2186 operation hours (reached in December 1974). This was:

17.0 mg Fe per g fuel
2.6 mg Ni per g fuel
6.4 mg Cr per g fuel.

The rate of the increase per operation hour of the Fe, Cr and Ni content on the particles is practically constant from 200 operation hours. Calculations of Spruyt with our results (B-A/Int.R-2/74) show that even when the erosion takes place at some well-described "dangerous areas" erosion does not yet lead to an unsafe situation for the KSTR.

Electron microscope pictures of the KSTR fuel clearly demonstrate that its surface is altered. Initially we could observe very well small crystals. Now, after 2186 operation hours, the surface is an even, amorphous layer. After etching with the usual procedure this layer is locally removed. At these places the crystalline structure reappears.

III.4. Sol-gel work

The main effort of the group presently is devoted to other activities than the KSTR and is, consequently, reported elsewhere. Part of their time, however, was used for experiments on the recycling of the organic liquid wastes, generated in previous production runs and experimental work.

One of the problems is the lack of an adequate detection method for the radioactivity of the organic liquid which is the criterion to decide whether the disposal as inactive waste is possible. The work continues as part of the working program of Interfuel B.V.

III.5. Miscellaneous activities

III.5.1. Analytical chemistry

As far as the analytical group (S.B.A.) did work for the KSTR and related projects, its activities can be described with the following number of determinations.

	1974, quarter				total
	I	II	III	IV	1974
a. operation of KSTR	45	30	39	64	178
b. fuel preparation	7	2	-	-	9
c. fuel research	96	112	36	50	294
d. sol-gel research	48	50	131	151	380
e. Interfuel support	19	8	38	61	126
f. supporting studies for KSTR	-	-	3	40	43
total	215	202	247	366	1030

III.5.2. Electron microscopy

Approximately 10 samples of KSTR fuel were prepared for electron microscopic investigation. As already mentioned in III.3. it is clear that gradually a layer of erosion products is covering the surface of the mixed oxide spheres.

Far more samples than of the KSTR fuel were prepared from the LISA experiments, described earlier in this report. Initially we had difficulties with the handling (embedding, stripping and polishing) of these highly radioactive preparations. With increasing experience of the operators, however, the problems decreased. To-day the handling of these samples is done without difficulties, provided a careful working method is followed.

The problem of the desintegration of the irradiated LISA fuels was also tackled by this group. Much effort was spent in trying to determine quantitatively the residual number of particles. However, we did not succeed.

The debris which we always see together with the residual particles has been a puzzle since long time. After many experiments with non-radioactive material we finally succeeded in preparing a sample for electron microprobe research which was done at S.C.K.(Mol).

Another technique which was new for us is remote light microscopic work. A television camera is used to observe the radioactive sample (a Co wire in this case) under the microscope.

The Health Physics Group (G.B.D.) was assisted with the initial work on the Central Dose Registration System T.N.O. is starting with.

Finally, although not related to KSTR work, the electron microscope was used for studies of fracture surfaces of metals.

III.5.3. Texture

The available instruments were used to do several kinds of analyses as: specific surface area, pore volume, adsorption/desorption isotherms as well as thermographic analysis (T.G.A.), differential thermal analysis (D.T.A.) and pore distribution measurements with mercury.

We started with an investigation of the KSTR fuel. Earlier experiments demonstrated that, when it is sintered in protecting gas 80% N_2 - 20% H_2, the U/O ratio of 2.6 at room temperature of this fuel decreased to 2.08 at 450°C and even to 2.02 at 600°C.

We started a program to investigate the influence of this O/U ratio on the crystallite size, lattice-constant, grain size distribution and pore volume and specific surface. Only limited time was available for this work and important results cannot yet be given.

IV TECHNOLOGY

IV.1. Suspension behaviour

IV.1.1. The flow pattern in the KSTR core vessel

Several experiments have been carried out in the perspex model of the KSTR core vessel. A fluorescent ZnS suspension is used to imitate the fuel particles under KSTR conditions. Phototransistors were arranged along a meridiane of the core vessel. The whole core vessel was exposed to U.V. light. The response of the phototransistors showed the presence of two distinct levels of light intensity, completely in agreement with the alternating flux levels in the KSTR. The results proved to be reproducible. The visual examination of the associated flow pattern was facilitated by placing the whole loop in a dark room. By changing the arrangement of the flow stabilizer and the inlet piece of the core a reproducible situation could be created with one definite light intensity. In this situation the visual examination revealed one relatively quiet flow pattern.

IV.1.2. Erosion behaviour

To examine the influence of the fuel concentration on the erosion, four experiments have been carried out with conical test pieces in a loop. At 10 bar and 150°C test runs of 100 hours have been made with different concentrations ranging from 0 - 500 g heavy metal per liter. The erosion rate was measured by weighing the different sections of the conical test piece. No erosion was found below flow rates of 5 m/s. At higher flow rates an increase of the erosion rate was observed.

An influence of the concentration on the erosion could not yet be found, this result is rather surprising. There was a steady decrease of the erosion rate with the subsequent experiments, independent of the concentration.

The experiment with a concentration of 510 g heavy metal per liter will be repeated to test the reproducibility.

IV.2. Suspension power reactor

IV.2.1. Cyclone-shaped core vessel

An alternative cyclone-shaped core vessel has been designed. In this vessel the fuel particles are kept in the core vessel by an adequate use of the properties of the flow pattern. A model of this vessel has been constructed. The first experiments were carried out with 3 mm P.V.C. spheres and the results were very promising. The same results have been observed with Pb spheres of 100–120 μm, and glass spheres of 200–220 μm. The spheres are moving in a relatively small toroidal space. The height at which the spheres are moving can be varied by changing the support flow. The vessel has been equipped with a dump vessel. The dumping operation still gives some problems. A study has started to investigate whether this type of reactor vessel can be mounted in the existing KSTR installation.

IV.2.2. High temperature suspension test loop

The high temperature loop for the study of the behaviour of fuel suspensions at higher temperatures than 250°C has been tested with water. Material and welding problems of the concentration measuring sections have been solved and the concentration measuring device is ready. The loop is available for experiments.

V. SAFETY

V.1. Reactor Safety Commission

As in October and November again two runs with the KSTR were made at a mean power level of 40 kW and in December one run up to 200 kW, several proposals for experimental programmes had to be checked by the R.V.C. (Reactorveiligheidscommissie KSTR).

For the transition of the power level restriction at 100 kW, a considerable delay was caused by the fact, that the license (of May 1st 1974) required a formal consent from the safety authorities (Begeleidingscommissie). In this case also the advice of some members of the I.A.E.A.-Panel (of 1964 and 1972) was required. For this purpose several reports with information on the recent experimental results were prepared.

With a representative of the UKAEA Safety Branch a discussion on KSTR safety aspects was held on November 21st. The formal confirmation, that power operation was allowed up to 500 kW, came on December 3rd. This, however, meant that for the time being, the mean reactor power level should not exceed a limit of about 200 kW. This limit was reached on December 12th, during run 13.

No serious trouble was encountered during the experiments in this quarter; all instruments and control functions operated satisfactorily, apart from a recurring irregularity in the measurement of the Main System level in the gas-liquid separator. Because this level can be controlled also with a signal from another type of measurement, the trouble with the first method - a gas injection back-pressure type - could be tolerated up to now, though it is considered necessary to find and eliminate the cause.

In expectation of the permit for the operation at levels above 100 kW, more attention was paid to the safety aspects, connected with the next phase, in which significant amounts of power and fission products would be generated. These aspects concerned (a.o.):

- discharge of heat instead of supply and consequently a different way of operation of the Primary Cooling System (PCS-A).

- the consequences of a "fast regular stop" by the dropping of the three safety rods.
- the effects of decay heat after the separation of the fuel from the suspension and storage in the collecting vessel.

As soon as the power produced by the reactor exceeds the thermal losses of the system (inclusive the evaporation after the injection of hydrogen), heat has to be discharged from the Main System through heat exchanger ME-1 and transferred to the PCS-A and to the SCS (Secondary Cooling System). For this purpose additional coolers had to be switched in between the PCS-A and the SCS. The transient situation might give rise to thermal stresses in the material and to control problems. These aspects were checked.

Former experiments had indicated the magnitude of the change in reactor temperature after a given reduction of the reactivity. After the insertion of the three safety rods (-1200 pcm), as part of the "fast regular stop", a temperature drop of $\sim 25^{\circ}$C was to be expected, unless the supply of heat from the PCS-A to the Main System was resumed in this time. But at high power level before the stop, -i.e. with a low PCS-A temperature ($\sim 150^{\circ}$C at maximal power)- this would involve again a transient situation, a fast temperature rise of the PCS-A water, thermal stresses and control actions.

Calculations indicated, that after such a stop at power levels higher than 200 kW, it would be possible that the reactor would reach criticality again - at a lower temperature level and during the short time to heat up the PCS-A. This situation had to be evaluated from a safety view point and it became clear, that the procedure for a fast regular stop (speedstop) had to include additional safety actions. It also was considered, that further reduction of activity, e.g. by removal of fuel without sufficient supply of heat, might lead to a further drop of temperature to a level too low for the production of sufficient water to control the level in the main system.

A fast extraction of fuel also is limited by the heat removal capacity of the fuel storage vessel, as soon as substantial production of decay heat is present. That capacity had been calculated with the assumption, that only a limited amount of the fission products should adhere to the solid particles. Recent measurements have indicated, that this is the case for nearly 100%.

All these considerations have safety aspects, that are being clarified both by calculation and experiment and will eventually result in changes in the present state of safety procedures and automatic safety actions.

VI. CONCEPTUAL DESIGN

VI.1. General

The work on the conceptual design for a 250 MWe suspension reactor, which started in April 1973, made further progress.

The main purpose of this conceptual design is to provide information on the technical feasibility of a power suspension reactor and to make clear on which points further development work might be necessary. As a model for this study a 250 MWe unit is chosen as part of a 1000 MWe power station.

For this work, which actually is carried out by the mechanical project office, support was given by members of the technological groups with experience from KSTR design and operation. In order to intensify the co-operation with other groups supplying information, a steering group was formed and effectuated.

In December most of the work was done, and a provisional version of a report containing all information on the conceptual design will be available in January 1975. Below some special points will be mentioned.

VI.2. System analysis

Studies and considerations on the design and operation of the main suspension circulating loops led to the choice of a system in which the suspension circulates through the four loops (connected to one core vessel) without reduction of the concentration (200 g/1). The suspension passes the pumps and the heat exchangers in the four loops (cf. quarterly report 1973-IV). Although in this way well-known unit operations have been used, two main disadvantages are inherent to this system for a power unit with sufficient lifetime and safety of the components.

These are:

a. Erosion on the long term in certain components despite optimization of design and fabrication.

b. A high fuel inventory in the outer loops relative to the core inventory (volume ratio 5 : 1).

An alternative system in which the suspension for the greater part would be retained inside the core vessel, has been designed. This system makes use of special properties of the flow pattern in a cyclone type core vessel. The flow pattern and the movement of the particles can be influenced by particle-free water flows. Preliminary tests on this system have been carried out by the technological group and gave encouraging results.

Further indications on nuclear feasibility and heat transfer are necessary.

VI.3. Component design

Studies and proposals were made for the principle design of

a. a contact column for the removal of fission products;

b. the four main circulating pumps comprising special adaptions of the pumphousing for minimum erosion attack;

c. main heat exchangers, which also comprise adaptions for minimum erosion attack;

d. steam generators.

VI.4. Fuel handling

Procedures were proposed for the loading, refuelling, reprocessing and purification of the fuel during different operational steps.

VI.5. Control

As a result of a study of several possibilities for power control by regulation of the neutron absorption in the reflector, a separate boric-acid system was chosen as the most promising method.

LIST OF REPORTS AND PUBLICATIONS

B 101/74	C.A.Rietman Spoedstop met schoonwater bij 255°C.
Ch 157/74	P.J.C.Bloem KSTR-splijtstof afkomstig uit het GPS.
G 57/74	G.v.d.Lugt, H.Wijker Metingen met behulp van gammaspectrografie ter bepaling van de aanwezigheid van splijtstof in de reactorcompartimenten van de KSTR.
G 60/74	L.C.Scholten Meting van het rendement van de koolfilters van het ventilatiesysteem KSTR.
G 62/74	G.v.d.Lugt Enkele voorstellen betreffende een uniforme regeling voor het meten en rapporteren van radioaktieve stoffen in de geloosde ventilatielucht van toekomstige kerncentrales.
RCG 68/74	J.A.H.Kersten, J.H.C.v.d.Veer Experiments with the KSTR.
RCG 69/74	H.J.C.Boekschoten Over de invloed van de vervalwarmte op de temperatuur van de voorraadvaten van de KSTR.
RCG 70/75	H.J.C.Boekschoten, G.de Weerdt Een derde onderzoek naar de verblijfplaatsen van splijtingsprodukten.
T 198/74	D.Peters Lekmeting met N_2-overdruk in de compartimenten van de KSTR (meetserie 4).
T 202/74	J.Matteman Wanddiktebepaling ME-1.
W 190/74	A.Spruyt Leak rates of containment and reactor hall.
W 191/74	A.Spruyt et al The closure of the KSTR compartments with resilient elements.
W 193/74	A.Spruyt Kruisstuk in PCSA, thermische spanningen t.g.v. QΔT-regeling.
W 194/74	A.Spruyt Suspensiereactor kringloop met deeltjesvrij buitensysteem.

Ch-E/R-37/74	O.Votocek Fluor-Gel II. Chemische eigenschappen en interne gelering van uranylfluoride.
III 7894/74	L.C.Scholten In-situ test koolfilterinstallatie TL-76 van de kerncentrale Borssele.
III 8346/74	G.v.d.Lugt, P.IJkelenstam Resultaten van de radioaktiviteitsmetingen in de omgeving van de kerncentrale Borssele.
III 9647/74	G.v.d.Lugt, P.IJkelenstam Resultaten van de radioaktiviteitsmetingen in de omgeving van de GKN-centrale te Dodewaard over het 1e halfjaar 1974.
III 9940/74	L.C.Scholten Periodieke rendementsmetingen en in-situ test koolfilters kerncentrale Borssele.