STATUS OF THE DEBENE FAST BREEDER REACTOR DEVELOPMENT, MARCH 1979

U. DÄUNERT, G. KESSLER

1. INTRODUCTION

1.1 Political Situation in the Federal Republick of Germany During 1978

The nuclear debate in the Federal Republic of Germany during the second half of 1978 was dominated by the discussion about the fast breeder development in this country. Although the Bundestag (the German Parliament) had recommended late in 1977 that construction of the SNR 300 be continued, the discussion was reinforced in the context of the then necessary next partial construction license, which was ready for issue in June 1978.

Against the background of the pending law suit which was brought before the Federal Constitutional Court for a decision on whether fast breeder construction is backed by the German Atomic Energy Act or not, the discussion on the proliferation aspects of plutonium in the framework of INFCE and the more general discussion about the consequences of a plutonium economy, the responsible licensing authority of the state of North Rhine-Westphalia hesitated to issue a further license for the SNR 300. Moreover, they asked to convert the SNR 300 into a plutonium burner by changing both core and blanket of the reactor.

This controversy finally led to a debate in the German Parliament on December 14, 1978 and to a voting procedure on the continuation of the development of fast breeder technology. The Bundestag then decided:

> "In view of the steadily increasing world energy demand, the limited reserves of fossil and mineral sources of primary energy and the considerable dependence of the Federal Republic of Germany on energy imports, research and development for tapping and utilizing new and alternative sources of energy and energy technologies, and in particular renewable sources of energy as well as nuclear process heat, must receive public support. Considering the imponderabilities involved in predicting the development of our future energy demand and the limited character of world uranium supplies, the Deutscher Bundestag considers it necessary that R&D activities inthe field of the new and advanced reactor lines, such as the high-temperature reactor, the fast breeder and controlled nuclear fusion, be continued so that all conceivable options at present open concerning the long-term securing of our energy supplies can be developed, preserved and assessed.
>
> With regard to the development of the fast breeder technology, the construction of the SNR 300 prototype and the accompanying research as well as the resultant modifications, if any, are to be continued so that a final decision can be made in favour of or against this reactor line on the basis of more profound knowledge and more precise criteria. In view of the objections still remaining, the Deutscher Bundestag expects that, before the potential commissioning of the SNR 300, the Bundestag will have to take another decision following a political debate of principle. This will also apply if the prototype is to breed more fissionable material than it uses up. A decision on yet another potential fast breeder reactor (SNR 2) should not be taken until ample experience has been gained with the operation of the prototype facility. The same approach should be taken in the case of the high-temperature reactor.
>
> In preparation of these decisions, the Deutscher Bundestag will set up an inquiry commission to study these technologies as well as possibly altered or modified designs in great detail. The commission will have to set out the potential alternatives and necessities in connection with future developments in the field of the peaceful utilization of nuclear energy under ecological, economic, social and safety aspects both nationally and internationally, and to submit recommendations for the corresponding decisions."

In the meantime the Federal Constitutional Court has ruled that fast breeder construction under the German Atomic Energy Act is fully consistent with the Basic Law (Constitution) of the Federal Republic of Germany.

Consequently the next partial construction license was issued by the licensing authorities on December 20, 1978. The impact of this delay is described in the corresponding chapter of this progress report.

1.2 International Cooperation

The activities in the field of international fast breeder cooperation during 1978 were governed by the implementation of the agreements on breeder cooperation between France/Italy and the DEBENE-countries.

This cooperation is directed by a steering committee which holds biannual meetings and has installed 10 working groups. It comprises all fields of LMFBR R+D except fabrication of components and fuel elements and reprocessing.

The efforts required on the part of those concerned are illustrated by the following example: During the period from November 1977 through December 1978 roughly 100 meetings took place, tying up a remarkable amount of manpower.

For this reason among others it became necessary to rationalize the ongoing cooperation with other countries by implementing bilateral agreements on a tripartite basis between France, Germany and third parties.

The cooperation between KfK and CEA in the CABRI Project is continuing. Since the summer of 1978 the three excursion tests were performed. During 1978 NRC has joined PNC and UKAEA as junior partner to the project. NRC supplies information on ACPR in exchange. Negotiations with DOE to exchange CABRI and TREAT results are continuing.

The contact with CEA on defected fuel experiments in the SILOE reactor at Grenoble has been extended.

The cooperation between Japan and Germany was renewed on a trilateral basis between PNC-CEA-KfK/INTERATOM on June 21, 1978. It substitutes and supersades the previous contract between KfK and PNC. A third meeting on exchange of JOYO and KNK experience was held in Karlsruhe in March 1979.

The first phase of experiments in the Fission Product Loop (FPL) of Toshiba Research Centre was completed during 1978. There exists general interest to perform a second phase of experiments, but it is not yet clear how satisfactory contractual arrangements can be made.

In November 1978 agreement was reached among US DOE - CEA - BMFT to implement the bilateral umbrella agreements on a tripartite basis. A program was defined for the exchange of knowledge in the areas of safety, reactor core, systems and components and fuels and materials and is now ready for approval by the parties involved.

In a similar way a corresponding agreement between UKAEA-CEA-KfK/INTERATOM is in preparation. The cooperation with UKAEA in the frame of the BIZET project has been continued with critical experiments on large heterogeneous cores in ZEBRA.

The duration of the KNS-COVA agreement was extended by 1 1/2 years (German sodium boiling experiments versus English containment explosion code validation experiments).

Finally, the commitment of the DEBENE-utility group Schnell-Brüter-Kernkraftwerksgesellschaft (SBK) within the framework of NERSA should be mentioned. While the utilities get full access to all the experience obtained in the course of the construction of a commercial-size fast breeder reactor, the manufacturing industries in the DEBENE countries will gain experience by delivering systems and components for fuel handling and storage.

2. KNK II FAST CORE EXPERIMENTAL NUCLEAR POWER PLANT

In the period under review, the main results achieved with the KNK II compact sodium-cooled reactor facility were the issue of the permit for 40% power operation in April and for full power operation in August 1978, the occurrence of cover-gas bubbles in the primary sodium in late 1978, and attaining the 100% power output in early March 1979.

2.1 Plant Operation

Under the third partial operating permit, which allowed operation of the plant up to 40% of the rated power (58 MWth), measurements were conducted in early April 1978 initially of the post-scram behavior and for optimization of reactor control. Subsequently, the turbine was started and raised to its rated speed. The generator was first hooked up to the interconnected grid system of the local utility on April 26, 1978. KNK II generated some 4000 KWh by late July.

In the first amendment of the partial operating permit issued by the licensing authority on August 3, 1978 preconditions for full power operation of KNK II were established. However, as the power level was raised, the plant was scrammed by the "high negative reactivity" signal on August 9 and 12, 1978. The reactivity signal, which is a scram signal connected to the safety system, serves for the detection of boiling events in fuel elements which may be caused by blockages of the coolant flow.

Detailed studies always carried out in close cooperation with the licensing experts and the licensing authorities, however, indicated that the signal was certainly not due to sodium vapor bubbles caused by boiling events, but to bubbles from the argon cover-gas.

These gas bubbles never imperiled the safety of KNK II operation. It should also be noted that the relatively small core has a negative void coefficient in all regions.

The cause for the entrainment of cover-gas in the sodium was determined by specific experiments which excluded a sequence of various possibilities. For instance, it was first demonstrated by feeding nitrogen and xenon that the main coolant pumps do not take in any gas. Next, attention was focused on the reactor proper, a suspected phenomenon being the entry of a ventilation pipe from the orifice adjusting system.

In further experiments it was possible to prove the dependency of the gas entrainment on the sodium flow in this pipe, which also indicated a possibility of preventing these gas bubbles by reducing the flow (see Figure 1).

The ventilation pipe serves to ensure the maintenance of natural circulation in the emergency cooling case. Accordingly, the flow through this pipe can be throttled considerably by a valve during normal reactor operation. However, the installation of such valves constitutes an interference with the safety philosophy underlying the operating permit. For this reason, the required licensing procedure extended over a couple of months until early February 1979. In February, KNK II went on power again, and nominal power output was first reached on March 3, 1979.

2.2 Test Programm

Presently, 66 experiments are being conducted in the following areas in KNK II within the framework of the fast breeder development:

- irradiation and post-irradiation examination of fuel elements, absorber elements and structural materials
- reactor instrumentation (safety instrumentation), failed fuel element detection, sodium and cover-gas chemistry
- systems technology.

It should be mentioned that the first of a major number of so-called experimental plugs has been inserted to introduce measuring systems, irradiation rigs etc. into the reactor vessel down to the very core region.

Efforts are also being made to install several global failed fuel element detection systems in the accessible area of the tank. This work turned out to be very complicated because of the cramped space conditions; moreover, the shielded gas pipers and other facilities must be run so as to be protected against earthquakes.

3. STATUS OF CONSTRUCTION OF SNR 300

The partial permits 7/1 and 7/2 under the Atomic Energy Act granted in the years 1972 - 1977 and their amendments allowed most of the shell construction to be finished by mid-1978. Figure 2 shows the state of construction in August 1978.

Construction of the biological shield will have been completed by early 1979 (see Figure 3); hence, the only major building work still to be carried out is the construction of the redundant Diesel air intake system and the cooling tower, the latter of which has not yet been licensed. In addition, finishing work is still carried out in the buildings within the framework of machine assembly.

Assembly work in the steel plate skin, which has been stopped since mid-1976 (external sealing skin), will most probably be resumed in mid-1979. This is dependant on the completion of the test procedure going on since late 1978 with respect to temperature loads, concrete deformation, stress analyses and design. Since the pilaster strips assembled up until 1976 now longer correspond to the loads now used as a basis, partial disassembly of the steel structure is first of all necessary; it has already been initiated.

Assembly of the components of the tertiary circuit in the turbine hall is complete except for the high pressure pipelines of the feedwater/steam system. The equipment assembled is being conserved by constant exposure to dry air.

Partial permit 7/3 was issued on December 20, 1978. This partial permit for the first time covered construction work in machine and electrotechnical installations.

As a result of the considerable delay in the forthcoming of this partial permit the crew on the construction site dropped to some 300 by the end of 1978. In the long term, the further development of the project is still determined by partial permits 7/4 and 7/5 as far as the time-table is concerned. This partial permits mainly refer to the after-heat removal systems for specific lines and the primary and secondary circuits, whose assembly and commissioning is on the critical path.

Fabrication of the most important power plant components by and large follows the time schedules. Successive fabrication of many components requires increasingly more temporary storage, because the partial permits required for assembly are not available. This temporary storage is mostly organized on the construction site, partly in the special storage halls constructed for the purpose.

Fabrication of the bottom cooling system has been completed. The components can now be shipped to the construction site after completion of the shop documents (see Figure 4).

Fabrication of the reactor rotating plug system had involved more and more difficulties in the management of fabrication which no longer admitted of any continuation of the order. Accordingly the contract was cancelled in agreement with the contractor in mid-1978 and fabrication was assigned to a different companies which already built a satisfactory reactor vessel. Work performed by the new contractor has been promising so far and has confirmed the appropriateness of the measure taken (see Figure 5).

Also fabrication of the large components for the primary and secondary systems is successively being completed. In some instances fabrication has been stopped because final assembly had best be postponed until right before the installation phase.

All shells of the intermediate heat exchangers are kept in a temporary store. Fabrication, especially the welds of the last three bundles proceeds satisfactorily and according to schedule.

After repair of the compensators of the straight tube steam generators completion has been continued. The steam generators will also be put into a temporary store after completion.

The helical bundles are completed for the first helical tube steam generators. Fabrication of the central tubes, the tube plate and the shells had been interrupted inview of the in-service inspections.

The impellers of the primary sodium pumps have been cleared for mechanical working. Fabrication of the pump housing, most of which had been completely welded, has been interrupted because the results of acceptance tests must first be clarified.

In view of the requirement by the German Advisory Committee on Reactor Safety (RSK) on the power current systems for energy supplys of the safety system the reliability of the start-up behaviour of the Diesel emergency power system (electrically coupled double systems) must be proved by tests. Test programs have been elaborated for this purpose.

The tests will most probably be carreid out on site in spring 1980.

In early March 1979 a new time table for the licensing procedure will be agreed upon with the licensing authority and the experts, in which a most probable date of late 1980 will be assumed for partial permit 7/5. This plan will be the most important basis of a new time table of the whole project. In the light of these concepts the earliest commissioning date will be in late 1982.

4. RESULTS OF R+D PROGRAMS

4.1 Fuel Element Development: Irradiation Experiments

4.1.1 Experiments Connected with the SNR 300 Mk-Ia-Core

In the frame of the fuel element development for Mk-Ia the experiments RAPSODIE I and DFR-455 could be completely finished. In the case of DFR 455 the causes for clad-ruptures, which had occured during irradiation, were analysed and could be explained as being the result of fabrication failures and reactor specific influences.

For the KNK II reactor 7 fuel elements with mixed oxide fuel pins were fabricated (6 elements by ALKEM, 1 element by BN) according to Mk-Ia specification and loaded into the reactor. Fromout the fabrication of about 2000 mixed oxide fuel pins resulted a remarkable gain of industrial production experience which can be used directly for the Mk-Ia campaign.

The Mk-Ia fuel element fabrication started end of 1978 and will last for about 2 years. The irradiation of 13 original Mk-Ia mixed oxide pins in a DMSA (demountable subassembly) in the PFR is in preparation to confirm once more status of fabrication- and design-development and production experience.

4.1.2 Experiments Connected with the SNR 300 Mk-II-Core

The RAPSODIE II irradiation experiment, containing 19 mixed oxide pins with helical wire spacers has in the meantime nearly reached the foreseen goal without any problem (preceding monitor gained 1.03×10^{23} n/cm^2 ≈8.95% burnup, the subassembly reached 10^{23} n/cm ^{2}); unloading will take place end of 1979.

An important progress for the support of future SNR core design is expected by using the KNK II reactor to test fast breeder fuel element types. So the second core loading will contain fuel elements with an advanced Mk-II-design, where the mixed oxide fuel (with respect to density and solubility) and the cladding material (with respect to swelling behaviour) are improved. It is also possible to irradiate for the first time a greater number of elements with SNR design under more realistic fast breeder conditions than before.

The decision was made for the Mk-II reference concept to use spark-eroded grids with skirts as spacers and to irradiate a full-sized SNR 300 element in PHENIX in order to prove the expected high potential of this element design. Start of irradiation is foreseen for 1981.

4.1.3 Experiments with "Off-normal" Reactor Conditions

In cooperation with the French CEA a joint programme to investigate the behaviour of defective fuel pins under steady state and power cycling operation is being conducted at the SILOE reactor. Three single pin experiments were performed, the first with an unirradiated pin, the second and third with preirradiated pins. The most important results are:

- Inspite of large artificial defects at the clad hot spot location the loss of fuel, even after a power cycling period of some weeks, was small.
- Fuel loss and migration of fission products and fuel were analysed by using DND-monitor, on-line γ-spectrometry, on-line fission gas control and by the normal destructive post-irradiation examinations.

In an other program fuel pins are tested under possible operational transients, i.e. severe load-follow-on operation, power ramps, local overpower and excess clad temperatures caused by minor coolant perturbation. A series of capsule and loop experiments has been initiated. 5 startup-ramp experiments and 2 middle-of-life-ramp experiments will be carried out 1979 in HFR at Petten, one loop-experiment under combined power (clad temperature transient conditions) starts late in 1979. Later on fast flux pre-irradiated fuel pins will be included. Main aims of this program are: code verification, demonstration of defection limits and consequences of continuous operation.

4.2 Development of Cladding and Core Structural Materials

For the cladding material the influence of fabrication parameters especially on the mechanical behaviour was studied; experiments to examine the influence of fission product corrosion were started. This work was necessary to assure the materials data used for the design, to understand physical processes which occur under irradiation (swelling, in-pile creep) and to evaluate models and functions for the prediction and description of the cladding material behaviour under irradiation. Most of the data came from the experiment RIPCEX I, where about 90% of the neutron dose expected for SNR 300 Mk-Ia could be reached.

With respect to a longterm improvement of the material behaviour the development programs for optimizing the steel W.Nr. 1.4970 and studies of alternative materials, including despersion strengthened ferritic steels, were continued (RIPCEX II), and after a delay of some years the irradiation experiment M-1 has started in PFR.

Planning and preparation is going on for two further experiments:
- M-2 in PFR (W.Nr. 1.4970 with various modifications, alternative- and development-alloys)

- Charlemagne (together with the CEA,the influence of fabrication parameters of the cladding material on fuel pin behaviour will be tested in a standard fuel element of PHENIX).

Concerning core structure-material W.Nr. 1.4981 experiments on the minimum creep rate at 700 °C were terminated; for the alternative wrapper material W.Nr. 1.4914 experiments were continued to improve impact strength and stress-rupture properties. First results on behaviour after irradiation are available.

4.3 Absorber Development

The R+D-work for qualification of the Mk-Ia absorber concept (closed absorber pin with B_4C-pellets) were finalized with experiments in the field of compatibility between steel and B_4C-evaluation of post-irradiation data, especially fromout the experiment DFR 510, show that no compatibility problems have to be considered for the Mk-Ia-design.

The irradiation of B_4C-pellets, produced by various industrial vendors, and of EuB_6 pellets in capsules in PFR will be continued in order to test the irradiation behaviour of absorber material foreseen for the Mk-II absorber pin.

4.4 Plant Structural Materials

4.4.1 In-air Tests

A central activity of in-air tests is the determination of the hardening and fatigue behaviour of the SNR structural material W.Nr. 1.4948. The experiments proper on monotonic hardening have been terminated. A continuation of investigation is advisable because of the wide scattering found even within one charge.

Work on cyclic hardening yielded a clear dependence of the number of load cyclings on the temperature and on the strain amplitude.

The investigations into the interaction of fatigue and creep stress showed differences in material behaviour as a function of the temperature so that the methods offered by the ASME code are applicable with restrictions only.

Long-term creep rupture with welded and unwelded steels, W.Nr. 1.4948 (≙ 304 SS) and W.Nr. 1.4919 (≙ 316 SS), at 500 °C to 600 °C are under way. Some specimens have reached a test time of 80 h. The tests shall be extended up to 200,000 h.

For verification of computer codes biaxial creep and LCF experiments were made and the influence of the multiaxial status of stress at 550 °C was studied.

4.4.2 Sodium Effects

The assumptions made with respect to the mass transport established in the primary heat transfer system, inclusive of the activated corrosion products and the Cs-137 fission product, have been confirmed by experiments. Future tasks include measures allowing to reduce the activity transport.

To verify the influence exerted by sodium on the mechanical properties of the SNR structural material, specimens made of the basic material and provided with weld seams were loaded into sodium loops set up for creep tests.

At a temperature of the cold trap of 418 K (~5 ppm O) the lifetime of both specimen types is shorter than in air. The effect can be largely suppressed by thermal treatment which destroys the sensibilization. In sodium of higher purity (cold trap temperature 393 K) the specimens reach a lifetime similar to that in air.

Results of more than one million low cycle fatigue tests in sodium are presently interpreted so that they can be taken into account in the design of high-temperature loaded components.

A test program is being prepared which relates to the investigation of crack propagation in sodium wetted defective weld seams both under creep and fatigue stress.

4.4.3 Effect of Neutron Irradiation on the Properties of SS, Type W.Nr. 1.4948

The testing program for determination of the effects of neutron irradiation on the mechanical properties of the W.Nr. 1.4948 construction steel for the SNR 300 reactor vessel and the internal components is being continued on samples originating from the real reactor tank cast ingots. The program comprises the irradiation of plate and welding specimens at 723 K and 823 K up to thermal neutron fluences of 6×10^{18} n/cm^2 and 2×10^{20} n/cm^2, achieved in core positions of the HFR Petten at thermal to fast flux density ratios of about 0.6. Post-irradiation testing comprises tensile testing at strain rates from 6×10^{-6} to 6 1/s, creep measurements up to 10,000 h rupture time and low-cycle fatigue within strain ranges from 0.6% to 2% and at a strain rate of $3x10^{-3}$ 1/s. The major effects observed in the first series of experiments are high temperature embrittlement (due to helium produced by the $^{10}B(n,\alpha)^{7}Li$ reaction in the 14 ppm boron containing steel used for the experiments), reduction of creep rupture times to 10% of original values, and a decrease of the creep strength by 60 MN/m^2 (plate material, irradiation fluence $6x10^{18}$ n/cm^2, temperature 823 K). The total creep strain of weld samples is reduced to values of 0.3% to 1.5%.

Within the program of "combined creep and fatigue stress" tests at 550 ^{o}C with holding times up to 60 minutes have been terminated for 1.5 and 1% of total strain. The number of load cyclings is markedly reduced by the introduction of a holding time due to increase in creep stress.

To determine the influence of temperature, some holding time tests were performed also at 450 ^{o}C and 500 ^{o}C. With rising temperature an increasingly negative influence of holding time on the number of cycles can be observed. The explanation is the damage by creep during holding times which increases with the test temperature. Tests with holding times up to 60 minutes performed on welded material at 550 ^{o}C have been terminated. If these results are compared with that of the basic material, a greatly similar behaviour is found for both material states. The only difference to be noticed is that the number of cycles of welding specimens is only about half the number of that of the basic material.

4.4.4 Rupture Mechanical Investigations

A new test rig was planned and erected for rupture mechanical tests to be performed on components. Commissioning of this test rig is expected in spring 1979. The target of the planned working program are investigations into the leak-preceding-rupture concept adapted to operating parameters and performed on largely true-to-scale and true-to-geometry structures. Work relating to components is supplemented by an accompanying test program concerning rupture mechanical specimens. Within the frame of this program the great variety of operating parameters such as temperature, influence of media, influence of frequency, etc., are dealt with.

4.5. Fast Reactor Physics

4.5.1 Experiments in Fast Critical Assemblies

SNEAK 11 is related to reloads of the KNK II reactor. During last year, two configurations have been investigated, each with control roads withdrawn as well as half inserted. The experiments were aiming to determine control rod worth for different rod patterns and the related power distribution. Evaluations done so far with respect to rods containing boron of varying enrichment show an overestimate of rod worth of the order of 10%. Power distributions tend to be more asymmetric then calculated.

The joint DeBeNe-UKAEA-BIZET program was devoted to investigations of heterogeneous cores. Configurations showing a central breeder island as well as smaller scattered islands have been set up. Here also, measurements were performed with control rods either fully withdrawn or half inserted. Special emphasis is given to the Na-void effect and to assymmetries of power shape depending on rod pattern. During the whole year, two KfK delegates stayed at Winfrith, they were joined by a BELGONUCLEAIRE delegate during the large part of the year.

First discussions have been held with CEA, concerning the future common large critical to be performed at Cadarache. This joint venture will also be devoted to heterogeneous cores with main emphasis put on ring cores.

The decisions which type of heterogeneous assemblies to investigate within the joint programs of critical facilities were accompanied by discussions on the merits of heterogeneous cores in general involving members of the respective design teams.

4.5.2 Nuclear Data

The measurements of the capture cross section of Am-241 have been finally evaluated in the energy region between 10 keV and 250 keV.

Concerning the KEDAK data file, statistical resonance parameters of various Th, U, Pu, Am, Cm isotopes have been newly evaluated. The capture cross section of Pu-240 has been increased (20-40%) as compared to KEDAK-3 and ENDF/B IV.

ECN Petten continued their efforts to improve the fission product data file, these activities now being done in co-operation with similar CEA-CNEN activities.

Methods and data basis related to the determination of displacement cross sections have been reviewed to understood certain remaining discrepancies between DeBeNe and CEA calculations.

US-data concerning γ-heating have been adopted.

Concerning the NEACRP 1250 MWe fast breeder benchmark, two problem solutions were supplied to the specialists' meeting held at ANL. One of these was based on KEDAK 3 data, the other on the adjusted KFKINR 26 group set, which is used for design calculations.

4.5.3 Neutronics Codes

The spatial-synthesis code KASY was further tested through comparison with 2d and 3d finite difference diffusion codes. Discrepancies for high leakage assemblies were eleminated.

Diffusion codes (D3D, D3E) and burn-up codes (TRIBU, HEXABU) have reached sufficient maturity to become exchanged between DeBeNe partners.

A survey of methods to determine anisotropic diffusion coefficients has been recently completed.

The present version of the cell-heterogeneity code (KAPER-2) is being improved by taking leakage into account when calculating the collision probabilities. The implementation will first be done for rod geometry. Concerning rod followers, the work of Seki has been studied and a related code was implemented.

The KfK transport codes have been provided with a perturbation option.

The fast design code BRUST has been improved to take strategy parameters into account. This enables one, for instance, to judge a design from the view point of uranium consumption.

4.6 Fast Reactor Safety

4.6.1 Core Disruptive Accidents

In the last report to the IWGFR it has been stated that analyses performed for SNR 300 show no significant loading for the vessel and the remaining primary system. Additional analyses performed 1978 confirmed this statement. Investigations concentrated on the contribution of Na and SS vapour to the mechanical work, the distribution of core material in the vessel as a consequence of a core disruptive accident, and on the investigation of secondary criticality events.

Na and SS vapour may contribute to the conversion process of thermal energy into mechanical energy. Investigations show that the efficiency might be increased; i.e. the mechanical energy grows larger. However, even taking into account these effects the mechanical energy in the case of SNR 300 hardly exceeds 100 MWs.

Studies on core material distribution in the reactor vessel have beeen performed in order to obtain initial condition for studies on the longterm coolability of the core debris inside the reactor vessel. The studies were performed in such a way that bounding cases could be defined for various spots in the vessel. The material distribution defined in this manner were used for coolability studies.

Investigation of recriticality event were necessary because of the energetically benign nature of the primary event which does not necessarily lead to a complete disassembly. A large number of cases have been analysed. It always has been tried to define these cases in a logic and consistent manner. Results of the investigations show that also in these case of a recriticality event no major energy releases are to be expected. Mechanical energy releases again are lower than about 100 MWs.

Accident studies for large breeder reactors again could not be started in 1978 because of urgent work for SNR 300.

Material Movement Inside the Fuel Element

The technical feasibility of using thermite filled pins was demonstrated by several one pin tests under sodium. The pins have an outer diameter of 7.6 mm, the thermite section (Fe_2O_3 + 2 Al) is 500 mm long. The pins are pressurized. The exothermic energy release of the thermite reaction is similar to the energy of a core disruptive accident. Use of uranium thermite has been proved.

The one pin experiments were continued with 3 and 7 pin bundle experiments. The phenomenological aspects were tested like coherence of pin rupture, influence of different pin pressure, formation of blockages at the spacers of the bundle etc. With a high speed camera the failure behaviour and material movement within the bundle was studied. The inspection of the 7 pin test section showed that about 20% of the mass within the reaction zone was transported to the upper plenum. The detailed evaluation of the experiments is on the way. Theoretical models are used for interpretation. Several better instrumentated one pin experiments and one 19 pin-bundle experiment are planned for 1979.

Explosion Tests on Reactor Vessel

The explosion test on 1/6 scale performed early in 1977 has been fully analysed now. The agreement between calculated and measured deformation is good. As however the deformations are only about 1% (compared to 3-5% expected in reality) an additional test will be performed. This test has been prepared in 1978 and will be performed end of March 1979.

The computer code ROPLAST which has been developed by INTERATOM now is considered to be verified and is in a production stage. The code calculates the plastic deformations cause by pressure waves in the tubes of the primary circuit taking into account the coupling of structure and hydrodynamics.

Post Accident Heat Removal

In the SNR 300 there will be an external core catcher. The necessity for such a system arose because difficulties were seen in proving with sufficient reliability that the core debris can be kept inside the vessel within this framework.

R+D-activities were performed to qualify liner materials for the cooled bottom of the core catcher and insulation materials for the side walls. The results of these investigations are described in the last status report.

More recent investigations show however, a very high probability that the fuel can be cooled safely inside the vessel, therefore the meaning of the external core catcher is reduced to that of a back-up system.

This result follows on detailed analytical studies of the heat transport from the fuel to the decay heat removal systems for various representative distributions of the fuel within the reactor vessel. In addition, the local transient temperature fields within the structures supporting the dispersed fuel are determined. The investigations are based on numerous experimental and theoretical findings, which have been produced internationally in the PAHR-field in the past years.

In support a basic R+D-program at KfK is underway. These program concerns the following main subjects:

Thermo- and Fluiddynamic Experiments

1) Model-experiments with simulant materials
   A) Heat transport in volume heated liquids
   B) Meltfront propagation
      - downwards and sidewards
      - influence of soluble and unsoluble materials
2) Fuel injection experiments (Uranium-Thermite) without Na
3) Code-Development
   A) Integral Codes
   B) Differential Codes

Material Properties

1) Determination of the chemical composition of the core melting
2) Theoretical and experimental determination of the physical properties

Interaction and Compatibility of Core Materials

1) Compatibility of molten UO_2 with core retention material (high and low melting)
2) Separation and mixing behaviour of the melt components
3) Removal of molten material by boiling and evaporation events

Fission Product Distribution

1) Determination of fission product distribution in the pool with respect to the heat source distribution
2) Transport behaviour of evaporating materials from the molten pool in the sodium.

Sodium Fire and Aerosol Physics

The physical-chemical behaviour of sodium fire aerosols was investigated. In the case of a large sodium fire in the secondary loop system a large amount of aerosols will be generated. A sodium spray fire delivers more aerosols than a pool fire. At sodium fires with a homidity of 40% the average diameter of the primary aerosol was 0.2 µm, with the munidity of 70% 0.35 µm. The chemical behaviour of the produced aerosol is dependent on the reactions of the sodium with the O_2, H_2O and CO_2 of the surrounding air. It is of interest to know the time for the formation of sodium carbonate as end product. Chemical analysis of sodium fire aerosols staying in an air atmosphere showed that within about 60 seconds more than 90% of the aerosols were changed into carbonate. The reaction into the carbonate is faster with a higher humidity (80%) than with a lower one (40%). The reason is the higher particle radius with a larger humidity content. If sodium fire aerosols are transferred into the surroundings of the reactor plant and if there will be a wind velocity of 2 m/sec then at a distance of more than 120 m there will be no corrosive sodium hydroxide any more.

The research on aerosol leakage through concrete and through model leaks as well as on aerodynamic characterisation of uranium oxide and sodium smoke aerosols is in progress. Reactor containment grade solid concrete appeared to have relatively large leak rates, whereas steel pipes through concrete introduce negligible leakage. Low ratios of the aerosol number concentration (monosized spheres) having artificially made cracks in concrete test specimens to that entering the cracks (i.e. penetration values below 1%) were observed for microsized (2 µm) particles whereas a considerable penetration ($\geq$ 50%) was found for 0.5 µm particles.

4.6.2 Single Subassembly Event

Role of Single Subassembly Events in Licensing of SNR 300

The discussions with the licensing authorities on the importance of single subassembly events are going on. It is still open whether local core faults have to be considered as a design basis accident or not. In the first case the German licensing rules require two diverse protection systems. Actually it is considered whether the inherent, self-limiting characteristics of the propagation processes could replace the second protection system. It is recognized that a failure propagation by local melting of steel and fuel and by mechanical effects of fuel coolant interactions can be ruled out. However, the problem of a thermal subassembly to subassembly propagation by melt-through processes is still unsolved. Such processes would not immediately lead to whole core accident. But actually it is difficult to demonstrate that the damage will be limited to a few fuel subassemblies if in the licensing procedure it is presupposed that a single protection system (e.g. the DND system in the case of SNR 300) could fail for any unspecified reason. From this situation results the requirement for further research effort related to the problem of thermal subassembly propagation.

Sodium Boiling Experiments

The local boiling experiments were continued at KfK with a 21% edge blockage in a 169 pin bundle and at ECN Petten with a 35% central blockage (60% part of SNR fuel element). The first test section at KfK had a 49% central blockage respectively 70% at Petten.

The main results of the tests are:

- the maximum temperature increase behind the 21% blockage is twice of that which the 49% blockage because the mass exchange between the main stream is smaller;
- under nominal SNR 300 conditions the saturation temperature is reached with the 21% blockage;
- with the 21% blockage no strong pulsation of boiling occurred. The different phenomena during boiling between 21% and 49% blockage influenced the conditions, leading to a dry out. The safety margin between boiling and dryout is smaller with the 21% blockage than with the 49% blockage;
- argon gas was added at the entrance of the test section with the 49% blockage. In the case of 21% blockage gas was injected behind the blockage into the recirculation zone.
  Dependent of the sodium velocity and the injected amount of gas a quick dry out was noticed starting from the single phase situation. The location of adding gas is not important. The amount of gas reaching the recirculation zone is decisive. A gas rate of 15 cm^3/sec initiated the loss of cooling with the 21% blockage;
- the experiments at Petten with a 70% and 35% central blockage showed in principle the same boiling phenomena as the 49% central blockage at Karlsruhe. For the end phase of the 35% blockage gas injection is planned.
  With all blockages tested at both centres no hydrodynamic instability occurred.

4.6.3 Status of In-pile Investigations

CABRI

The CABRI reactor went into operation in 1978. Three tests have been performed: ICO, A1, A1R, The main aim of the ICO test was a check out of the test section instrumentation. Beside that the test pin was to be destroyed with a special device the test pin was not destroyed.

In A1 and A1R energy depositions of about 0.54 and 0.64 KJ/g have been brought into the test fuel starting from nominal conditions, i.e. 400 W/cm and 480 W/cm respectively. Clad failure was not expected and did not occur. Presently the PIE of the test pins is underway. Mid of March the test programm will be continued by test A2, having an energy deposition of about 1.2 KJ/g UO_2. These conditions should be very close to pin failure.

Up to now fresh fuel has been used in all the tests. Preirradiation of test pins will start in April in the PHENIX reactor. After one cycle the pins will be unloaded and will be brought to the CABRI facility. First tests with pre-irradiated pins are scheduled for December 1979.

Mol 7C

The Mol 7C program is a joint in-pile program of KfK and SCK/CEN Mol. The current program consists of three experiments with 37 pin bundles having fresh UO_2 fuel. Two experiments have been done. The post irradiation examinations for the first experiment have been finished, for the second experiment they are on the way. The third experiment which was planned for beginning of 1979 had to be postponed to the end of 1979 because the beryllium matrix of the BR2 reactor is changed during 1979. The main difference between the first two experiments is the different gas pressure of the pins:(1) experiment 7C/1: 60 and 53 bar, (2) experiment 7C/2: 12 and 10 bar). The transient was initiated by blocking the valve for the additional cooling of the blockage. The important facts for the two experiments are:

	Mol 7C/1	Mol 7C/2
Boiling temperature	after 10 sec	after 4.4 sec
Begin of dryout	after 12.6 sec	after 6.4 sec
Pin failure	after 12.6 sec	after 6.4 sec
DND signal	after 14.1 sec	after 7.7 sec
Steel melting temperature	after 15.0 sec	after 8 sec
Stabilized sodium bundle flow	after 200 sec	after 80 sec
Full reactor power	2880 sec	360 sec

Post irradiation examinations of the first experiment showed that molten cladding material has moved into the radial and axial direction and has formed a crust at the outer region of the blockage. The six fuel pins of the inner row of the bundle failed. The contact with liquid steel caused the failure of some pins of the second row. Fuel particles formed a secondary blockage at the next spacer downstream but caused no damage within this area.

The most important results for the first two experiments are:
- with a porous central blockage of 35% and an axial extension of 40 mm no sodium boiling occurs at nominal flow condition. There was local boiling at 25% of nominal flow for a SNR 300 fuel element;
- the failure of fuel pins can be detected by delayed neutron detection (DND);
- local coolant disturbance at full reactor power did not lead to a rapid failure propagation within the bundle;
- even in the event of destroyment of parts of the bundle geometry the integral cooling of the bundle stayed intact during 48 minutes reactor operation.

Experiments on Single LMFBR Fuel Pins Under Abnormal Reactor Conditions

The single fuel pin failure experiments under simulated reduced cooling (SHOT) and loss of cooling (LOC) have been continued and the program of the 19 SHOT experiments and 24 LOC experiments is almost finished. The research entails irradiations on fresh and pre-irradiated fuel pins in sodium capsules in the sodium capsules in the poolside facility of the HFR at Petten, followed by PIE in order to study the behaviour of fuel and canning and possible propagation effects at stationary high temperatures and internal pressures (SHOT) and at loss of cooling conditions, e.g. transient temp. increases (LOC).

The SHOT results on fresh fuel pins will shortly be reported in full. Highlights have been presented at the recent ENS/ANS international topical meeting at Brussels (Oct. 1978). The LOC results will be reported at the ANS/ENS international meeting in Seattle (August 1979).

As an extension of the foregoing research and complementary to the fast power transients in the CABRI facility at Cadarache, France, an irradiation project is started to investigate the effects of relatively moderate power transients on single SNR type fuel pins (TOP). The irradiations will again be performed in the poolside facility of the HFR making use of the experience gained from the SHOT and LOC experiments. However, whereas the SHOT and LOC capsules had stagnant sodium, for the TOP experiments forced convection sodium rigs will be used. The first part of the TOP programme consists of a series of 12 experiments. Power steps up to 1800 W/cm in 0.5 sec are being considered. Other parameters are: stationary power 250 and 420 W/cm; burn-up 0.05, 2 and 5% FIMA; energy deposition up to about 1.5 KJ/gr UO_2; variable gap width between fuel and canning. Pre-irradiation in a fast reactor is being investigated. The design of the forced convection rig, the development of the in-pile instrumentation and data handling system is in an advanced stage. Various components of the test rig are being tested in an out-of-pile sodium test rig. The first irradiations are planned to start in 1980.

4.7 Measuring Methods Development

4.7.1 General Remarks

The development and application of special measuring methods has two objectives, namely
- to provide the means for core and plant protection (safety)
- to get high plant availability by the very early detection of failures (early warning system).

Some radiochemistry works serve also for the last aspect, as will be shown later.

4.7.2 Failed Fuel Detection for SNR 300 by DND

The investigations on local coolant disturbances in fuel elements have shown, that a fuel defect and a slow failure propagation cannot be excluded absolutely, but that this will be detectable with a suitable instrumentation.

For this the DND-method - that is the measurement of delayed neutrons within the coolant, but outside the reactor core - seems appropriate. Delayed neutron emitting fission products are carried from the fuel into the coolant if a open defect arises. The efficiency to detect local fuel defects and thus to interrupt a defect propagation was demonstrated in the experiments of the "SILOE-failed-fuel program" and with the blockage experiments of the Mol 7C-series. The measurements with artificially defected pins in SILOE showed a slowly increasing DND-signal, which is on the other hand strongly dependend on the reactor or pin power.

For a better understanding of the fission product release mechanism a first model was set up, which considers diffusion and recoil. The high dependency of the DND-signal on temperature dependency of fission product diffusion in the failure region. The following diffusion coefficients were determined experimentally:

380 ^{o}C at the failure location: $6x10^{-9}$ cm^2/s
470 ^{o}C at the failure location: $1.0x10^{-8}$ cm/s
560 ^{o}C at the failure location: $7.5x10^{-7}$ cm/s

More experience on early failure detection was also gained by the monitoring of fission gases in the covergas.

Future work will be concentrated on the determination of reactor shut down criteria for operation with failed fuel. These studies include to get more knowledge on the development of failures and the correlation and interpretation of different signal, as DND, covergas-spectroscopy and others.

Concluding it can be stated that DND signals are suited to protect FBR's against incidents which have their causes in defect fuel elements.

The design phase for the delayed neutron detection system for SNR 300 is finished. The measuring sodium volume is about 20 ltr and He-3 counters will be used.

The calculated sensitivity of DND is better then 150 cm^3 fuel free surface in the driver zone. This is the limit at which the safety instrumentation has to be initiated for a scram. These investigations are confirmed by studies on SNR 300 mock-up.

4.7.3 Other Work on Core Surveillance

For the early detection of coolant disturbances before boiling the temperature-fluctuation at the fuel element onlet seems to be an appropriate method. It is known that a local blockage needs at least 30...50% of the fuel element cross section in order to be detected by normal thermometry at the fuel element outlet, whereas the fluctuating component of this signal is much more sensitive.

Measurements on the 60^{0}-blockage bundle in Petten and with the KfK-full bundle (169 pins) indicated, that the detection limit for a local blockage by temperature noise analysis is in the range of $\geq$10% blockage area. A blockage detection before Na-boiling seems therefore possible in principle. Further investigations are under preparation.

Further progress has been made in acoustic measuring methods. The detection systems, using Li-Niobate piezo-electric transducers and also magnetostrictive devices, are well developed and applied in out-pile and in-pile experiments. The application for SNR will be:

- boiling detection
- steam generator leak detection.

For the acoustic boiling detection in SNR a series of measurements was made in the out-pile facility NABEA in order to determine the transmission characteristics of sound in sodium. These measurements gave good results. As a next step measurements under real reactor conditions are prepared: an electrically heated bundle will be installed in KNK II, boiling will be initiated and the signal to noise ratio be determined. The reactor is in a shut down state during these experiments.

The acoustic leak detection in steam generators is considered as a fast method, compared withthe more sensitive, but slow responding hydrogen-monitoring devices. Model-experiments with different leakage rates were performed. They indicated that the acoustic frequency spectrum from a steam/Na-leakage is remarkably higher than the normal background frequency, which allows a clear differentiation between both sources. The extrapolation of the model-experiments to reactor dimensions was shown by a series of acoustic background measurements on a Hengelo-steam generator. The results until now give a good basis for the final development of the method.

For the general development of sensitive diagnosis methods the "System Identification" by noise analysis is a major task. The signal-processing-system NOASIS was completed which allows the simultaneous and on-line analysis of power-spectral-densities, distribution-functions etc. of up to 16 input-signals. First models and programs for the KNK reactor transfer functions were developed. It was possible to determine the fuel-gap conductivity, which is not directly measurable.

The development of methods for in-service inspection is in progress. After completion of basic research on the transmission behaviour of sound in Na (KfK), a prototype for an under sodium viewing system was constructed and first tested under water (INTERATOM). It was possible to range parts and holes of 1 mm Ø, also contours and shapes of submerged test-specimen. For the further under-sodium-application, the necessary high-temperature sensors are under development.

For monitoring the steam generator leakage of SNR's two hydrogen detection systems are tested in KNK II for application in SNR 300. Each of these devices are in operation for 7000 hours. The operation was free of troubles. With these devices, during the power operation of KNK II, the rise in concentration through hydrogen diffusion has been measured. The measurements resulted in a high reproducibility so that the purification circuit depending on hydrogen indication was switched over.

The development of oxygen probes represents a further important step in monitoring the steam generator. Here better mechanical strengths had been achieved specially with the modification in electrolyts construction. The main object was to increase the resistance in temperature change. The electrods assembly could also be improved.

For flow measurement in main pipes, which are of importance in the power determination of SNR 300, a development of a permanent magnetic flowmeter had been concluded for a pipe diameter up to 600 mm. To reach this point, testings are carried out beforehand in a plant with the measurements at original dimensions. That could be yielded on electrods assembly which enables a definite linear measurement over temperature range with sufficient accuracy.

A study for Na-flow measurements for main ducts of advanced SNR's with even larger diameters has been made, following a general recommendation of the IAEA-meeting on instrumentation and control, April 1978.

Eleven different methods were studied in respect to the requirements of the SNR-2. Selection criteria had been among others: integral flow measurement, measuring accuracy, installation inside or outside the pipes, exchangeability, redundancy and diversity.

Summarizing the results it can be stated, that no method will exist being able to meet the requirements in full. As there are at least two diverse measuring devices necessary, the development should not concentrate on one method only. The suggestion is to promote the development on

- permanent-magnet flowmeters
- ultrasonic flowmeters
- flux distortion flowmeters
- transit time measuring techniques.

4.7.4 Radio-chemistry Work

The deposition of released fission products and of activated corrosion products on the system surfaces can cause maintenance problems even after the sodium has been drained. The identification and the transport and deposition behavior of these nuclides has been studied in KNK, indicating that Zn 65 and Mn 54 are the main active corrosion products which are deposited in the cold part of the system (pumps). Investigations for suitable materials for radio-nuclide traps are finished, a suitable material was found for operation up to 510 °C. The layout for a prototype absorber has started. The main long lived fission products in sodium from open fuel surfaces are J and Cs, as found in the experiments SILOE (failed fuel program) and FPL. The deposition is reversible with the temperature. As J can be concentrated in cold traps, the future work is concentrated on methods for Cs-trapping. Basic studies are in progress on the mock-up loop of INTERATOM.

4.8 Component Tests

4.8.1 Sodium Aerosol Deposition Tests with the Reactor Plug System

The concept for the protection against sodium aerosol deposition in the vertical gaps between rotating plugs and vessel support structure was tested under different conditions. During reactor operation each of the three plugs sits on a convection barrier. Nevertheless there will exist a small horizontal gap and so it is in principle possible that cover gas enriched with sodium aerosols can enter into the vertical gaps. An auxiliary gas loop is installed to prevent this. Argon gas is pumped against the gas coming from the reactor vessel through the horizontal gap at the convection barrier. Two main cases were tested. The handling operation phase at a sodium temperature of 200 °C and the reactor operation phase at 560 °C. During the handling phase the plugs are lifted. This means that a larger gap exist at the convection barrier than during reactor operation. Then the gas loop is out of operation. During these conditions and a test time of 60 weeks under different test conditions no depositions occurred within the vertical gap. Part of the test time sodium temperature was rised above normal operating temperature to generate a higher aerosol concentration in the reactor cover gas. During reactor operating temperature of 560 °C no depositions were noticed. The conclusion of the time consuming tests is that the geometrical convection barrier and the foreseen auxiliary gas loop system protect the vertical gaps of the rotating shield system against sodium aerosol depositions.

4.8.2 SNR 300 Cold Trap

The SNR-300 cold trap (same type for primary and secondary system) was tested for 500 h. The trap was at the same time part of a large pump facility. The result of the tests was an improved construction. Objectives of the tests included

- effectiveness of the trap per time unit,
- operation time in the system,
- heat exchange behaviour (radial and axial temperature profile),
- testing of the cooling system,
- operation behaviour under normal and fault conditions.

With the final version of the cold trap an effectiveness of 16 g oxygen per hour was gained.

The trap for SNR 300 is designed for a 2 years operation time. With the test results it was possible to reduce the number of traps to 4 for the primary and to 3 for the secondary system of SNR 300.

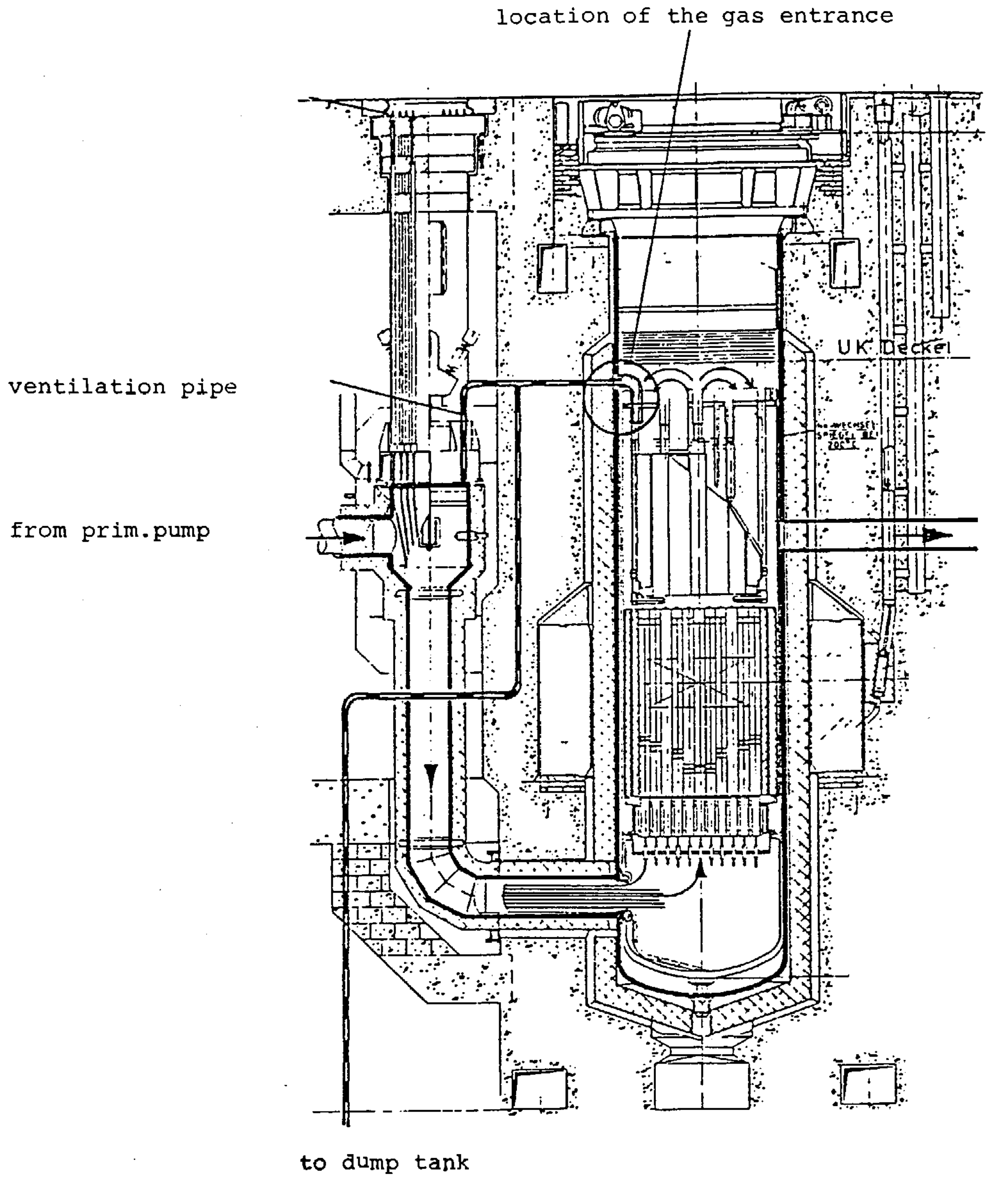

Fig. 1: Vertical Cross Section of KNK II with Special Emphasis on the Ventilation Pipe

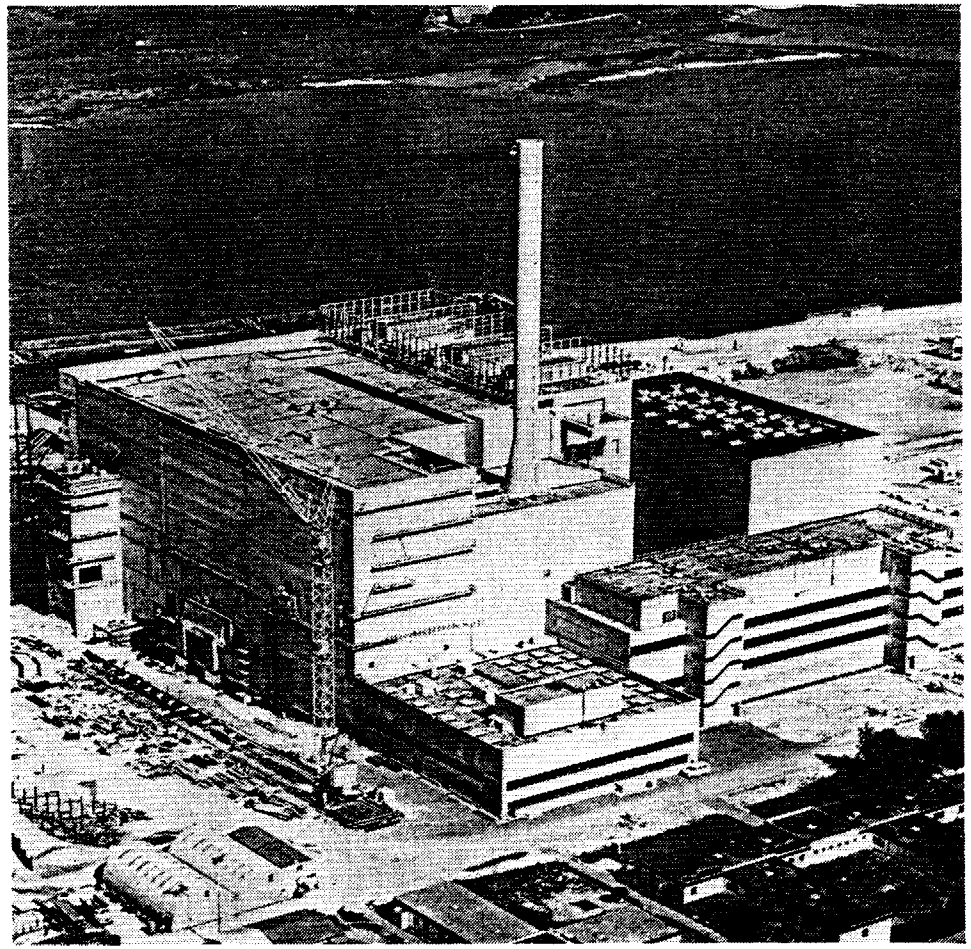

Fig. 2: Aerial View of SNR 300 Plant Site (August 1978)

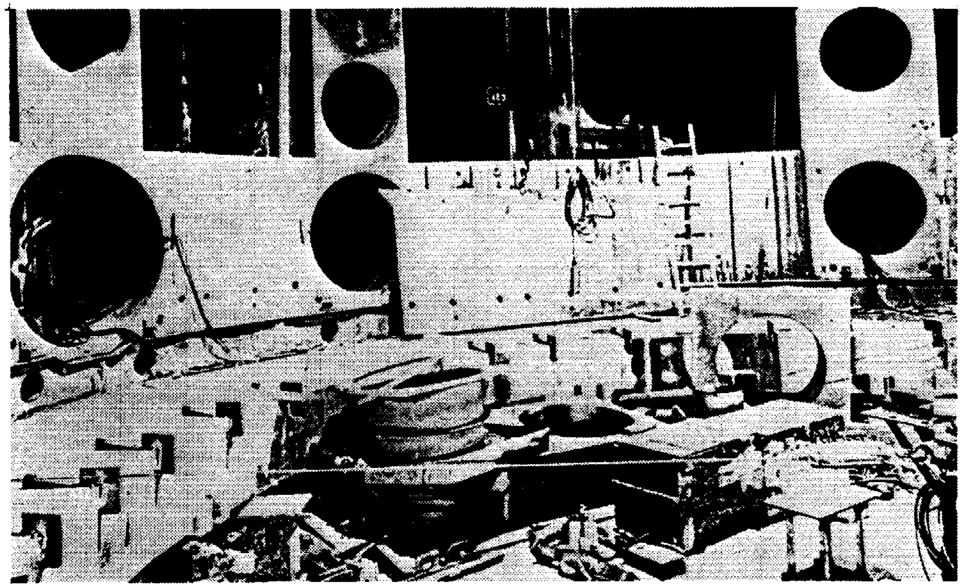

Fig. 3: SNR 300 - Biological Shield

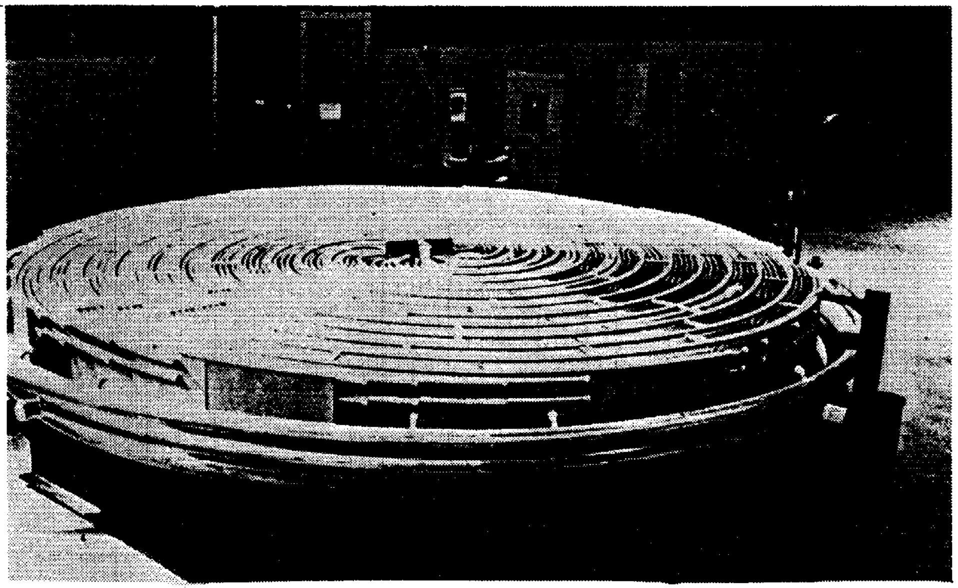

Fig. 4: SNR 300 - Bottom Cooling System

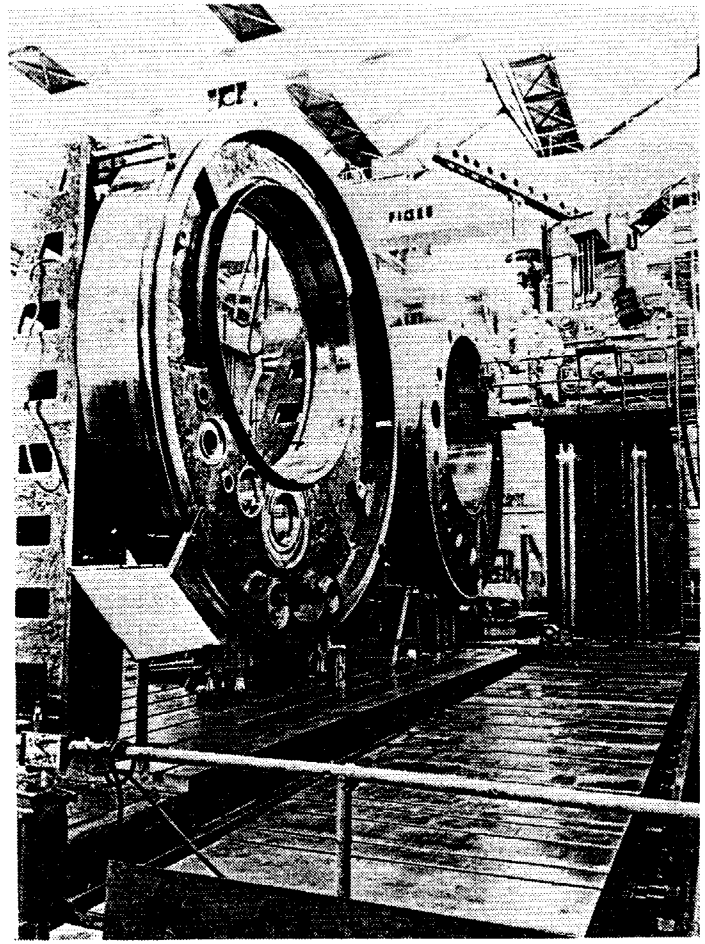

Fig. 5: SNR 300 - Rotating Plug System