STATUS OF THE FAST BREEDER REACTOR DEVELOPMENT IN THE FEDERAL REPUBLIC OF GERMANY, BELGIUM AND THE NETHERLANDS

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Preface

In 1967 and 1968 the Federal Republic of Germany, the Kingdom of Belgium and the Kingdom of the Netherlands ("DeBeNe") agreed to develop, in a joint program, breeder reactors to the point of commercial maturity. The following research organizations take part in this effort:

- Kernforschungszentrum Karlsruhe (KfK)
- INTERATOM, Bergisch Gladbach
- ALKEM, Wolfgang near Hanau

- SCK/CEN, Mol
- Belgonucléaire, Brussels

- ECN, Petten
- TNO, Apeldoorn
- NERATOOM, The Hague.

The three German institutions mentioned above have been interrelated since 1977 by the Entwicklungsgemeinschaft (EG) Schneller Brüter. Between KfK, INTERATOM, and the French Commissariat à l'Energie Atomique contracts were concluded in 1977 about close cooperation in the Fast Breeder field, with association of the Belgian and Dutch partners.

The results of research and development activities carried out by the DeBeNe partners in 1981 have been compiled in this report. The report begins with a short survey of the fast reactor plants, followed by an R&D summary. The bulk of the report gives more detailed information about those plants and about results reported by the Working Groups of the R&D Program Working Committee of the Fast Breeder Project. In an additional chapter a survey is given of international cooperation.

I. EXECUTIVE SUMMARY

1. Breeder Plants in the German-Belgian-Netherlands Region ("DeBeNe")

The Fast Breeder Project, which was begun 22 years ago, has led to construction and operation of the KNK II plant and to construction work on the prototype SNR 300 by industries in Germany, Belgium and the Netherlands. For development work beyond this level, planning studies are being conducted which will also involve the French concept and the experience already accumulated within it. In addition to Italy, the German, Belgian, Netherlands and British electricity generating industries participate in the French 1200 MWe Super Phénix breeder reactor.

1.1 KNK II Experimental Nuclear Power Station

The KNK II compact sodium cooled reactor is a 20 MW experimental nuclear power plant which, after the backfitting of a reactor core for fast neutrons, has allowed operation of the first fast power reactor in the Federal Republic of Germany. The plant is being used for numerous experiments in connection with tests of measuring techniques, fuel elements and materials and for the evaluation of operating experience.

Some first experience in construction and operation of a complete sodium cooled nuclear power plant had been obtained already in KNK I, the plant working on thermal neutrons, as far back as 1974.

In August 1981, the first core of KNK II exceeded its planned in-pile time of 255 full load days. This corresponded to a peak burnup of the test zone fuel elements of 62.000 MWd/te. In the light of the positive overall plant behavior of the fuel elements, the core will now be continued in operation up to 355 full load days (and even longer, if necessary). The permit required for this extension has been granted. Peak burnup by Feb 15, 1982, was 81.000 MWd/te.

This has confirmed the Karlsruhe fuel and cladding material designs, which are used as a basis for the specifications by the manufacturers, Interatom and Alkem. Also commissioning of the SNR 300 can be expected confidently because the burnup attained in KNK II at the same time is equivalent to more than 96% of the contractual target burnup in the SNR 300.

In rods of two test zone fuel elements, which had become defective in 1979 and 1980, respectively, completion of the breeder fuel cycle was demonstrated on a kilogram scale. For reprocessing of the first three reactor cores of KNK II at Marcoule an agreement was signed with France. The fuel rods of the first reloading of KNK II have been fabricated and are now being assembled.

1.2 SNR 300 Prototype Power Plant

Almost 1000 workers are at present engaged in construction work on the site of the SNR 300 near Kalkar. The main buildings have almost been completed. The cooling tower will be added and an additional recooling system, a well cooling system, will be installed.

Some 65% of the deliveries and services have been ordered, some 50% of the machine and electrical components and systems have been completed either in the shops or on site.

Some 35% of the fuel rods required for the first core have been fabricated; assembly of the fuel elements is beginning in early 1982.

In October 1981, the North-Rhine-Westphalian State Government granted the 4th partial construction permit (7/4) for the SNR 300. This permit covers ventilation systems, the core catcher system, decay heat removal systems of specific legs, an auxiliary cooling water system, emergency core cooling system and handling facilities. A special permit for the modified steel shell is expected to come forth in the near future.

The main systems can only be assembled after the corresponding permit 7/5 has been granted. This is the factor mainly determining the date of completion of the plant. Interatom expects partial permit No. 7/5 to be granted by the middle of 1982. The time schedules are based on that assumption and provide for filling of the main systems with the sodium coolant in 1985.

Some special items in the licensing procedure should be mentioned:

Proof of the integrity of the reactor vessel in a hypothetical Bethe-Tait accident was produced in accordance with the scope defined in late 1980. This also includes retention of the fuel in the vessel.

As a consequence of changes in the postulated Bethe-Tait loads, recalculations were necessary for the pump support structures. It was proved that the higher forces can be accommodated by the pump support structure and the building. The support structure of the intermediate heat exchangers was reinforced also because of changed Bethe-Tait loads.

Work on the "multiple pipe rupture in the steam generator" accident indicated that a simultaneous rupture of several pipes in an SNR straight tube steam generator can be excluded. Also external initiating mechanisms (airplane crash, safety earthquake) do not result in such an event, as a result of the design of the building and the components in the tertiary system.

Over the past few years, the SNR project has suffered considerable time delays, some of the reasons being licensing problems related to the novel character of the plant. The cost increase in the SNR 300 resulting from those delays is a major burden on governments of the participating countries. Consequently, the German Federal Ministry for Research and Technology (BMFT) is urging the electricity utilities and the manufacturing industries in the Federal Republic to increase their financial contributions to the project. One problem in this connection was found to be the dependence on a political decision on the commissioning of the SNR 300, which the German Federal Parliament reserved for itself to take. To prepare this decision, the first Fact Finding Committee on "Future Nuclear Energy Policy" of the German Federal Parliament in 1980 recommended to draft two additional safety related studies. These studies were commissioned in 1981 by the Federal Ministry for Research and Technology with KfK and the Gesellschaft für Reaktorsicherheit, respectively. The results of the two studies are to be used as a basis by the new Fact Finding Committee set up on May 26, 1981 to elaborate, by late July 1982, a recommendation on the problem of commissioning the SNR 300.

2. Progress in Research and Development Work

The following summary is a compilation of the most important results elaborated by the "DeBeNe" partners in their research and development work for the Fast Breeder Project in 1981.

2.1 Core Elements

An in-pile experiment in the Rapsodie reactor using 19 mixed oxide fuel rods of 7.6 mm diameter with helical wire spacers was completed in late 1980 at the target burnup of approx. 9 at.%. After cooling, non-destructive post-irradiation examination was carried out at the CEA; all fuel rods were found to be in excellent condition without any traces of damage.

As had been proved already for the reference cladding material of the SNR 300, Mk. Ia temperature transients of the type relevant to operating conditions do not affect the mechanical behavior also of the Mk.II unirradiated reference material.

The susceptibility to intergranular corrosion by fission products and UO_2 of the 1.4970 type cladding tube steel was examined experimentally. Mechanical-thermal treatment was found to exercise a major influence on the extent of corrosion.

The destructive post-irradiation examinations of the pressure tubes made of 1.4981 type steel after RIPCEX-I irradiation in the Rapsodie reactor proved radiation induced creep to be dependent on volume swelling and also indicated a quantitative pattern of this dependence. Volume swelling was also found to be affected not only by the fluence and the temperature, but also by the stress applied.

Evaluation of the S-3 and RS-1 defective rod experiments performed in Siloe, like the results of earlier experiments, indicated only minor fuel dispersions at the defect points.

Trial fabrication of fuel element wrapper tubes out of welded raw tubes made of 1.4970 type steel led to positive results: fault-free weld areas, ultrasonic testing capability by immersion techniques also in the region of the weld, low variance of strength data in the undisturbed region and in the weld. The martensitic 1.4914 type material, which is regarded as an alternative choice for the wrapper tubes, does not have the susceptibility to high temperature embrittlement observed in 1.4970 type austenitic material. This was found in a comparison of ductilities after irradiation in the BR 2 and loading in a creep-rupture test.

Three-dimensional temperature field measurements in an electrically heated 19-rod subassembly subjected to a sodium flow furnished information about the effects of grid type spacers and rod bowing in a fuel element. While the maximum differential temperature around the circumference of the central rod remains low and constant, it attains five and three times the original value in the corner rod and the wall rod, respectively. The grid type spacer results in local temperature increases only in the spacer proper. Minor rod bowing only leads to insignificant local temperature increases. Only a reduction to less than 70% of the narrowest gap between the rods approaching each other in the plane of measurement provokes a major rise in the cladding temperatures.

In a model experiment, the influence on flow distribution in the rod subassembly of the geometry of grid type spacers with flow aprons, which correspond to the Mk II concept, was studied. Comparison with the velocity profiles downstream of a grid without such a flow apron indicates that the aprons cause a major reduction in coolant flow in the wall channels. In this way, excessive cooling of the wall channels is largely avoided.

2.2 Physics

Compared with measurements conducted in the single-zone uranium core, SNEAK 12A, it was possible to calculate the reactivity worths of central voids to an accuracy of 3%, while a 14% difference between calculation and experiment was found for an eccentric void. The reactivity effect of the redistribution of steel was precalculated to an accuracy of approx. 8-11%. The studies served to clarify nuclear effects of major accidents associated with material relocations.

The BZB assembly, a large homogeneous two-zone core of the BIZET program, had been used until 1980 to measure the reactivity worths and power distributions for many control rod samples. Recalculations proved sufficient accuracy, i.e., less than 10% deviation, of the calculated results from the experiment. This also applies to asymmetrical control rod models, which correspond to operation under skewed load conditons.

The nuclear data of the Am-241, Am-242m, Am-243 and Cm-244 actinides have been re-evaluated for KEDAK. For 24 actinide nuclides, also a 36-group set made up of ENDF/BV basic data was compiled. Some major differences relative to the first actinide data sets of 1979 were discovered.

2.3 Safety

Analyses of hypothetical loss-of-coolant-flow accidents for a 1300 MWe breeder reactor ("SNR 2") with a homogeneous conventional core were compared with results for the SNR 300. Simulations of the expected events show primary excursions for the two reactors to be non-energetic. Despite an overall mild development of events in the SNR 2, the power amplitudes during the primary excursion are slightly higher than in the SNR 300.

Compared to the SNR 300, the effects of axial fuel expansion are generally 50% weaker in the SNR 2, which is due to its larger flatter core. Also the Doppler effect, despite a more pronounced Doppler constant due to the lower enrichment, is slightly reduced by the higher operating temperature of the fuel rods. Given the roughly identical maximum positive void reactivity effect, the feedback effects reducing the consequences of accidents are generally slightly smaller in the SNR 2 than in the SNR 300.

In the boundary case of chains of events assumed to take the most pessimistic turns throughout, conservative models for the SNR 2 indicate an almost tenfold increase in the mechanical load potential compared with the SNR 300.

Unter transient overload conditions, the behavior of a 7-rod subassembly was studied in thermite experiments. Rod failure was initiated for all seven rods with a time coherence of 100 ms. As the rods failed, the sodium was expelled in the axial direction. Shortly afterwards, the first liquid molten materials emanating from the thermite reaction reached the simulated blanket regions. Freeze-out caused blockages growing in the axial direction and increasing in density. The blockage in the upper blanket region was very solid and impervious, while that in the lower blanket region was less solid and partly permeable.

Verification of the ARES continuum mechanics program on the basis of a second explosion test in the 1:6 scale SNR vessel model showed very good agreement between the test results and a first recalculation by ARES. In the test, water was used instead of sodium. Some remaining differences (e.g., straining of the inner vessel, time of the water hitting the lid) were studied in parameter calculations by quantifying individual effects (e.g., explicit dependence on strain rates of the materials characteristics, water level in the vessel). In this way, agreement with the experiments was further improved.

Subsequent examinations of the first two loss-of-flow experiments of the Mol 7C series showed the formation of compact steel crud at the edges of the blockage. The fuel behavior in the two experiments is clearly different. In Mol 7C/1, larger fuel particles were transported from the blockage area, building up a secondary blockage at the first spacer downstream of the blockage. In the Mol 7C/2 experiment, most of the fuel columns remained standing, although they exhibited major ballooning. There was a considerable increase in valume of the fuel in the radial direction. The different types of fuel behavior are explained by the different filling gas pressures in the Mol 7C/1 and Mol 7C/2 experiments.

The A2 transient experiment in CABRI indicated that canning failure does not occur in a fresh fuel rod even if an energy of 1.1 kJ/g is introduced into the rod during a power transient. This energy input results in an almost complete melting of the fuel at the positions of the rod most highly loaded. - Hodoscope evaluation has been advanced to a stage that it is also able now to indicate quantitatively transient movements of the fuel in the cooling channel. Such movements were observed in the A3 Experiment, in which a fresh fuel rod was made to fail.

The most important components of the SNR core catcher NaK cooling system (level probe, flowmeter, EM pump) were successfully tested under SNR conditions on a full scale basis.

To clarify the extent in which the transport of radiant heat contributes to the thermal conductivity of a fuel melt, the optical constants of liquid UO_2 were determined. In view of the high absorption constant measured in the visible and near-infrared spectral range, an increase in thermal conductivity of the melt with increasing temperature due to internal heat radiation must be excluded.

Several input data of the SOFIRE II program adopted from the United States were modified on the basis of sodium pool fire experiments. As a consequence, the program now computes the gas temperature and the gas pressure in a conservative mode.

2.4 Plant Monitoring Measurements

KNK II continued to furnish experience in employing methods of detection important for plant monitoring. They mainly include the following aspects:

For acoustic boiling detection, the absolute sound pressure encountered in sodium boiling was detected under out-of-pile conditions and, in KNK II, the background signal mainly due to flow noises was determined; the consequence derived from both noises is that boiling should be detectable by acoustic means.

Examination of the monitoring algorithms proposed for the SNR 300 to detect local loss of coolant flow on the basis of the fuel outlet temperatures measured in KNK II shows that the dynamic limit control system planned will be effective over a broad range of power levels.

To detect loss of coolant flow by measurements of temperature fluctuations, the out-of-pile results were confirmed in the central element of KNK II according to which the (undistrubed) background signal in the power mode is independent of reactor power.

The compensation method of precision measurements of the fuel element temperature rise was used for determining the distribution of gas bubbles in the KNK II reactor core, which is important for safety reasons. The method is at present being studied by CEN Cadarache with respect to its applicability in the Phênix reactor.

Progress was made towards proving the long-term functioning capability of permanent magnetic flowmeter probes. Since March 1980, double-magnet transit time probes have been installed in KNK II above two fuel elements. So far, they have worked well continuously. This positive result can mainly be traced to artificial pre-aging of the magnets prior to use. This pre-aging is done by heating to a temperature clearly above the maximum operating temperature.

Neutron flux noise analyses in the power operation of KNK II showed pronounced resonances in the 3 Hz range. Correlation with a multitude of other signals indicates coherence with measured fuel element outlet temperatures. Consequently, it is concluded that fuel element vibrations occurred. One interesting feature to note is the dependence on power, i.e., on flow, of the resonance frequency.

2.5 Components and Structural Materials

The influence of radiation on the creep behavior of the 1.4948 type steel used for the reactor vessel was examined. The variance of the data for the irradiated and the unirradiated conditions is comparable. The ultimate elongation and the creep-rupture strength of all batches examined was reduced considerably by irradiation.

The creep-rupture strength of welds is influenced by radiation less severely than that of the base metal. The creep strain of welds reaches clearly lower levels than that of the base metal. The filler metal is hardly affected by irradiation.

Time-to-rupture experiments on the combined stresses of creep and fatigue with hold times up to 24 h were continued on irradiated 1.4948 type material. The numbers of load cycles to rupture showed only minor changes beyond hold times of approx. 10 h. However, this tendency still needs to be confirmed by other experiments.

With respect to the SNR 2, two steam generators with a power of 435 MW each are being planned per system. Both a straight tube and a helical tube steam generator were designed to this power level. It was found that steam generators for pool type plants differ from those for loop type plants only in certain details. A primary sodium pump in the cold leg will reduce the design pressure in the steam generator.

The most suitable steam generator material for large breeder power plants with a required life of 300.000 h and a main steam temperature of 455 °C, in the light of studies conducted so far, seems to be the ferritic 12 Cr steel, material No. 1.4922.

II. REACTOR PROJECTS

1. Operation of KNK II

1.1 Fuel Elements of the First Reactor Core

The first core of KNK II exceeded its planned in-pile time of 255 full load days in August 1981. This corresponded to a peak burnup of the test zone fuel elements of 62.000 MWd/te. As a result of the positive overall operating behavior of the fuel elements, operation of the core is now being continued to 355 full load days (if possible, also beyond this limit). The permit required for this purpose has been received. By Feb 15, 1982, the peak burnup amounted to 81.000 MWd/te. This implies that more than 96% of the target burnup contractually agreed upon for the SNR 300 has been reached.

Two fuel elements turned defective in 1979 and 1980, respectively, have meanwhile been disassembled in the Hot Cells of KfK, and some of them have already been subjected to post-irradiation examinations. While the first element was kept in the cooling pond of KNK II, a disassembly system was designed and built. This system, which is approx. 5 m long and made all of stainless steel, has proved to be quite effective in dismantling the fuel elements. The experience collected at that stage has shown that fuel elements with grid type spacers can be disassembled without major difficulties even at higher burnups. Previous sodium removal from the elements is not necessary for this purpose.

Reprocessing of the fuel rods from the first fuel element in the Milli facility has been completed; that of the rods of the second element is in progress. In the dissolution of the low density fuel in 7-molar nitric acid, 10% remained undissolved. The three extraction cycles (separation of U,Pu from the fission products; separation of U from Pu; uranium decontamination) were then carried out without any difficulties due to the fuel. The reprocessed uranium and plutonium of the first fuel element were used by the Alkem company to fabricate fuel rods for the first KNK II reloading. This means that also in the DeBeNe region the breeder fuel cycle has been closed on a kg-scale.

100 other rods, 60 of them from the first defective subassembly with a peak burnup of 17.500 MWd/te and 40 from the second one with 48.000 MWd/te, were shipped to SCK/CEN, where they are to be used for two follow-on experiments of the Mol 7C LOF test series.

1.2 KNK II/2 Follow-on Core

While the first core of the KNK II fuel elements is of the Mk.Ia type of the SNR 300 (6 mm fuel rod diameter, fuel of lower density), fuel elements of the Mk.II type (7.6 mm fuel rod diameter, fuel of higher density) are being fabricated for the second core. Fabrication of the fuel rods has largely been completed. For reloading of the KNK II test zone, Alkem fabricated a total of 1146 fuel rods designed for two central elements and seven outer elements. Some temporary difficulties in the fabrication of the fuel element wrapper tubes (welding defects in the pads) have been overcome. Assembly of the fuel elements began in early 1982.

In addition, Belgonucléaire is fabricating a test zone fuel element for KNK II, which is to be equipped with helical wire spacers.

1.3 Waste Management

The reference concept for waste management in KNK II has been defined in all points by concluding a reprocessing agreement with the French CEA. The agreement has meanwhile been approved by the governments of France and Germany and entered into force in June 1981.

The KNK II fuel elements of the first three reactor cores are to be reprocessed in the TOR/SAP facility of Marcoule beginning 1986. The fuel rods are to be delivered individually, i.e., following disassembly of the subassemblies. The TOR facility is to begin hot operation in 1984 and will be added upstream of the existing SAP extraction plant.

The KNK II waste management reference concept comprises these stages:

- Cooling storage of fuel elements in the Na storage facility of KNK II for up to one year,
- canning of the fuel elements in KNK II,
- fitting of a shipping cask,
- temporary storage of the canned fuel elements in the MZFR water pool,
- disassembly of the fuel elements and canning of the individual rods in the Hot Cells,
- temporary storage of the canned rods in the MZFR water pool,
- reprocessing at TOR/SAP of Marcoule,
- return of uranium and plutonium to Germany.

1.4 Preventing Gas Entrainment into the Primary Sodium

It had been possible already in 1979 to reduce to a minimum the influence of Ar-gas bubbles in the coolant by relatively minor modifications of the equipment. When the second core will be loaded in 1982, some additional measures will be taken to avoid the passage of gas bubbles. They include a modified coolant pathway upstream of the entrance to the fuel elements and the inclusion of cyclones for separating gases. To optimize these new systems, Interatom has carried out extensive full scale model tests with water, inter alia of the whole lower plenum.

Additional measurements will be performed in KNK II in order to clarify the causes of the introduction of gas and the accumulation of gas in "voids". A factor decisively influencing the measurements was a permit granted in spring 1981 for the shutdown of reactivity meters for experimental purposes. Only in this way is it possible to trace the whole pathway of a larger gas bubble passing through the core without any interruption by a scram operation.

2. Construction of the Kalkar Nuclear Power Station (SNR 300)

2.1 General Overview

The main buildings have been completed on the Kalkar construction site. East of those buildings, the cooling tower will be added; north of them, there will be an additional recooling system, the well cooling system.

Some 65% of all deliveries and services have been ordered, some 50% of the mechanical and electrical components and systems have been completed in the shops and on site.

Some 35% of the fuel rods and 80% of the blanket rods required for the first core have been fabricated, assembly of the core elements is in progress.

For the SNR 300, the North-Rhine-Westphalian State Government granted the fourth partial construction permit in mid-October 1981. This permit covers ventilation systems, the core catcher system, specific decay heat removal systems, ancillary cooling water system, the emergency core cooling system, and handling facilities. A separate permit for the modified steel containment is expected to come forth in the near future.

The main systems can be assembled only after the appropriate permit has been granted, which will be the fifth partial construction permit. This mainly sets the date of completion of the plant.

2.2 State of Construction Work

Almost 1000 workers are at present doing construction work on the site.

Reactor Building:

The support ring for the reactor vessel has been assembled and aligned. Installation work in the primary cell is progressing expeditiously, so that the vessel can be mounted as soon as the fifth permit has been granted.

After initial delays, the assembly of the sodium ancillary systems is now making good progress. Pressure tests have been carried out on some of these systems.

Switching Systems Building:

In October 1981, the plant was connected to the power grid for its own load requirements. The step was preceded by

- the construction and installation of the process computer,
- cable running and connection,
- assembly and commissioning of the DC switching systems for the actuating voltages of controllers, transducer consoles and signals to be supplied to the process computer.

Turbine Hall:

The electric auxiliary boilers were assembled and have been put into service.

External Systems:

Machine assembly of the cooling water cleanup systems has been completed, the electrical systems are at present being assembled.

2.3 Workshop Activities

Reactor Vessel with Internals:

After several years of storage, the reactor vessel (and also the support ring for the shielding and the bottom collecting tank) showed intergranular corrosion. The three cases concerned refer to parts made of 1.4948 type steel, which had been "sensitized" by the fabrication process. Only the outer surface of the reactor vessel is affected. No indications have been found on the inside, where nitrogen inertization was and is available. The corrosion cracks on the average are up to 0.7 mm, in maximum cases up to 2 mm. Although any

crack propagation or new development of cracks can be excluded with absolute certainty if the atmosphere is kept properly dry, it was decided in agreement with the Advisory Committee on Reactor Safeguards (RSK) to remove the cracks by abrasion of the surface. Estimates have indicated that a wall thickness reduced by 2 mm is permissible both for operational and accident type loads. Abrasion of the vessel will have been completed by May 1982, at which date the vessel will be ready for installation in the reactor building.
The ring support has been refabricated because of intensive corrosive attack on all sides.
Extensive investigations of all components made of 1.4948 type material turned out additional damage only on some of the valves, they have recently been repaired. To protect components against intergranular corrosion attack, existing rules and regulations, particularly for storage and assembly on site, were cast into more precise terms, especially by a more stringent definition of permissible environmental conditions (humidity, temperature).

Core Element Fabrication:

Fabrication of the fuel rods for the SNR 300 Mk.Ia core is progressing. By the end of 1981, about 12000 fuel rods (35% of the total), 5200 blanket rods (80%), 1550 grid spacers (54%) and 450 wrapper tubes (81%) had been fabricated. The balance will have been fabricated by late 1983.

The fabrication of core element structural components is proceeding according to schedule.

2.4 <u>Work within the Licensing Procedure</u>

Reactor Cell:

Demonstrations of the vessel integrity under conditions of a hypothetical Bethe-Tait accident have been produced in accordance with the scope defined in late 1980. This also includes the retention of fuel in the vessel.

The experts require a system for the detection of loose parts in the reactor vessel. After an extensive literature search, in which especially facilities for acoustic detection of loose parts in light water reactors were taken into account, an acoustic detection system for loose parts in the reactor vessel of the SNR 300 has been designed. The experiments necessary to back the concept will be carried out until late 1984.

In-service Inspections:

For in-service inspections of the outer wall of the reactor vessel, the German Advisory Committee on Reactor Safeguards (RSK) now requires no ultrasonic techniques, but feels that visual inspections are sufficient. Consequently, the focus of the development work was put on optical inspection of the outer wall of the reactor vessel by means of TV cameras. Special importance is attached to the description and definition of the sequences of movement of the manipulator moving the TV camera under remote control in the annulus between the reactor vessel and the guard vessel, and elaboration of the necessary camera techniques.
Also development work on the in-service inspection of welds of the primary pipes no longer serves the previous concept of ultrasonic inspection, but visual inspection with leak detection. This is also due to a recommendation by the German Advisory Committee on Reactor Safeguards. For the regions in the pipes subjected to the highest stresses, remotely operated endoscopes are planned for use in inspecting the welds, which are to be moved around the pipes in rails. This system is being tested at present.

Heat Transfer System:

Materials purchasing for the main sodium pipe tangential bends to be fabricated anew is according to schedule.
It is being assumed that the longitudinally welded pipes for the main sodium carrying systems will be used without having been ground. Intensive development work by the pipe manufacturers has resulted in a major improvement of the quality of circumferential welds. Special preparation and calibration of the pipe ends and the use of internal restraining systems during welding allows circumferential welds to be made with a minimum misalignment of edges and good surface quality.

HEDL was contacted for the development of suitable pipe clamps; in this way, experience might be applied.
Due to changes in the postulated Bethe-Tait loads, recalculations had been necessary for the pump support structure. It was proved that the higher forces can be accommodated by the pump support structure and the building.
Also because of modified Bethe-Tait loads, the support system of the intermediate heat exchangers has been reinforced. The material required for modifications in the bundle suspension and the supports of the helical tube steam generators has been ordered. Fabrication for the IHX and steam generators is in progress.
Work on the "multiple pipe rupture in a steam generator" accident has been completed in the light of the design made, its implementation and the quality assurance of the components. The result has been that simultaneous rupture of several pipes in a straight tube SNR steam generator can be excluded. Also external initiating mechanisms (airplane crash, safety earthquake) will not initiate such an event, because of the design of the building and of the components of the tertiary system. Yet, the safety potential of the intermediate heat exchanger under conditions of a simultaneous rupture of all pipes in the straight tube steam generator has been demonstrated for the postulated "multiple pipe rupture in a steam generator" case. The loads occurring at the interfaces between the primary and the secondary sodium are far below the load limits permissible for this postulated accident.

Ancillary Sodium Systems and Auxiliary Systems:

Sodium aerosol tests with coolers in the inertization systems revealed damage to the aluminum fan impellers due to caustic soda. Therefore the impellers will now be made of steel, and a corresponding test program has been drawn up on these impellers.

As a result of various planning changes and accident analyses heat releases from the systems in the reactor building are now assumed to be much higher than in previous analyses. Thus it is now necessary to cool the recirculated air to lower temperature levels with the containment remaining closed, in order to keep electrical facilities in the reactor building operable. Instead of the component cooling system used so far, this requires a special intermediate cooling system which, in turn, is cooled by a well system. The corresponding licensing procedure under the Atomic Energy Act is under way, the clarification necessary for a permit under the laws pertaining to the use of water has been arrived at, the site of the well has been defined in the light of the requirements to be met by the emergency cooling well.

Emergency Diesel Power Systems:

New contracts have been awarded for the emergency power diesel systems as monoblocks. Emergency diesel power systems type tested and already used in several power plants have been selected. The purchase of new emergency diesel power systems has no consequences on the targets in the time schedule. The planning work resulting from the new systems (modification of the licensing documents, changes in building installations and cooling water supply) has been started.

III. REPORTS BY THE "DEBENE" WORKING GROUPS

1. DEVELOPMENT OF CORE MATERIALS AND CORE ELEMENTS

1.1 Fuel, Blanket and Absorber Materials

For the first (Mark Ia) reactor core of the SNR 300, fuel rods will be used with 6 mm diameter and with fuel of low density; for the Mark II follow-on core, fuel rods with 7,6 mm diameter and fuel of medium density are to be used.

1.1.1 Behavior of Fuels and Blanket Materials

Since 1980, studies of fuel behavior have been concentrated on the high density and highly soluble types of fuel newly developed by the industrial manufacturers (cf. Section III.8). In-pile experiments devoted to studies of structural change processes and the solubility behavior within the framework of the FR2 Experimental Group 7 have been completed; post-irradiation examination has been begun.

1.1.2 Studies of Absorber Material

To test vented absorbers, a test rod was loaded in the Phênix reactor as a common action of DeBeNe/CEA. The in-pile time will be six months. At the same time, the IAMCOS absorber code has been completely revised. Studies have been performed on rods for large reactors.

Irradiations of B_4C and EuB_6 pellets in the British PFR have been continued, but have been delayed because of reduced reactor availability.

1.2 Technological and Mechanical Behavior of Core Materials

1.2.1 Cladding Material

As proved already for the reference cladding material of the Mk.Ia SNR 300, temperature transients relevant to operating conditions do not influence the mechanical behavior also of the unirradiated Mk.II, 1.4970, 20% c.w. reference material.

It was also found that wall defects (axial V-notches) down to depths of 100 μm do not cause the tube samples to fail as a result of one temperature ramp to 950 ^oC and σ_t = 56 to 110 N/mm^2.

The susceptibility of the 1.4970 type steel for the cladding tubes to intergranular corrosion caused by fission products and UO_2 has been studied experimentally. According to these studies, mechanical and thermal treatment has a major influence on the extent of corrosion.

For the Charlemagne in-pile experiment conducted jointly with the CEA (for comparison between the German 1.4970 and the AISI 316 cladding material used in Phênix) and for the subassembly experiment with spark eroded spacers, 100 percent of the cladding tubes to be used for rod fabrication were retested non-destructively.

1.2.2 Fuel Element Wrapper Tubes

A trial fabrication batch of 1.4970 wrapper tubes out of welded raw tubes led to positive results: complete absence of flaws in the areas of the welds, ultrasonic testing capability by immersion techniques also in the region of the welds, little variance of the strength data in the undisturbed region and at the weld.

A large batch of the ferritic alternative wrapper tube material, 1.4914, whose chemical composition and heat treatment had been optimized in a preliminary program, has been delivered. A characterization program, in-pile experiments and technological studies up to fabrication of a fuel element wrapper tube were conducted on this material.

1.3 Systematics of the In-pile Behavior of Core Materials

The destructive post-irradiation examinations of the pressure tubes made of 1.4981 steel of the RIPCEX-I irradiation in the Rapsodie reactor have proved that radiation induced creep depends on volume swelling and have also furnished a quantitative description of this dependence. Moreover, it was found that volume swelling is not only influenced by the fluence and the temperature, but also by the stress applied.

In the VEC III simulation irradiation with nickel ions, the influence of different batches (within the given specifications) of 1.4970 material was clarified. Moreover, alternative commercial materials with different Cr/Ni ratios were studied with respect to their swelling behavior and phase stability. The martensitic 1.4914 material did not exhibit the susceptibility to high temperature embrittlement observed in the austenitic 1.4970 material. This was found in a comparison of ductilities following irradiation in the BR 2 and loading in a creep-rupture test.

For the trilateral DOE/CEA/BMFT in-pile program to be conducted in Phénix (Mathusalem Experiment) and in FFTF (Monix Experiment) the test matrix was agreed upon among the parties involved. Specimens of 1.4970 are available.

Pressure capsules are to be used to study creep and swelling of the cladding material; in the Monix experiment, high temperatures will be used.

Also in cooperation with the CEA, preparation has been started of an irradiation of two Phênix fuel elements with cladding tubes made of dispersion strengthened ferritic material (IDEFIX). As a precursor of this subassembly irradiation, single rods with ferritic cladding tubes are prepared for the CARAFE experiment in the BR 2 and the COUCOU experiment in Phênix, respectively.

1.4 Core Element Irradiations

The in-pile experiment in Rapsodie conducted on 19 mixed oxide fuel rods of 7.6 mm diameter and with helical wire spacers was completed in late 1980, when the target burnup of approx. 9 at.% had been reached. After cooling, non-destructive post-irradiation examination was conducted at the CEA. All fuel rods were in an excellent condition and exhibited no indications of any fault or damage.

The next step in the advanced development of breeder elements is testing the Mk.II type with improved fuel variants, an optimized spacer concept and larger rod diameters. For the first KNK reloading, Alkem, following a redefined design and specification, produced the rods for nine fuel elements with high-density, highly soluble fuel, spark eroded grid type spacers, and 7.6 mm fuel rod diameter.

Belgonucléaire fabricate 127 rods of the same diameter for a fuel element with helical wire spacers, which is also planned for KNK reloading.

In cooperation with the CEA, irradiation of 2 Mk.II fuel elements in the Phênix reactor is being prepared. For this purpose, all structural parts of the fuel element have been ordered and fabricated on the basis of the fundamental data defined in late 1980, the detailed design has been completed, final negotiations about the specifications for fuel rod fabrication have been held with the manufacturer, and the corresponding flow measurements in water and the long-term tests in sodium have been completed by CEA. For the detailed design phase, the IAKAST computer program for fuel element wrapper tubes has first been used. Moreover, computation methods have been developed for spark eroded spacers. Also the irradiation of two fuel elements in Phênix is being prepared, in which helical wire will be used as a spacer concept. 1983 has been envisaged as the starting date of irradiation of the four fuel elements.

For use in KNK II, another materials test element ist being prepared for the irradiation of cladding and structural materials. This element can be reloaded and will be used in a reflector position. Encapsulation of the samples allows in-pile temperatures up to 600 °C.

1.5 Work on Core Design

The program on the transient loading of fuel rods has been continued by experiments in the HFR (startup and operating ramp experiments in the DUELL and KAKADU capsules) and in the BR 2 (VIC-1 loop experiment), all of which serve to study the impacts upon fuel rod performance of potential operating transients, such as overload, startup ramps and minor increases in cladding temperature. Post-irradiation examination of the first DUELL series was begun in 1981.

Cooperation between KfK and the IRT Darmstadt in the field of fuel rod model theory has been expanded to cover also the transient behavior of fuel rods. Especially the results of the Mol 7C blockage experiments are described excellently by the URANUS code.

The studies of the behavior of defective fuel rods conducted jointly with the CEA were continued with the RS-5 fuel rod loaded in the SILOE reactor (Grenoble). Important results were furnished by the studies completed at KfK on the S-3 and RS-1 rods. Operation of both fuel rods (S-3 had been pre-irradiated in RAPSODIE to 91,000 MWd/te, RS-1 to 110.000 MWd/te) was continued with defective claddings for another 40 days, partly with power cycling. In this mode of operation, the amount of fuel escaping from S-3 was 1500 mg with a defect of 30 x 1 mm size. In the case of RS-1, whose defect of 10 x 1 mm had been generated only at 90 % power in SILOE, only 60 mg fuel were released.

1.6 Development Work on Core Element Disassembly

Development work on the disassembly of fast breeder elements serves to design a concept allowing the safe removal of fuel rods from the bundle with a minimum of time and fixtures required. For the Mk.Ia and Mk.II fuel elements, disassembly concepts have been drafted, for the verification of which individual tests were conducted, such as load experiments of the spacer attachments and measurements of the maximum load-carrying capacity of spacers during disassembly. In an experiment conducted on a complete Mk.Ia fuel subassembly, which had both overlapping of rod and spacer cell circle diameters and element bowing, element torsion and rod bowing for simulation of in-pile deformations, the disassembly concept was confirmed.

2. MEASUREMENTS, INSTRUMENTATION AND SODIUM CHEMISTRY

2.1 Radiochemical and Chemo-technical Studies

2.1.1 KNK Primary System Contamination

In the repair phase of KNK II, direct measurements of pipes and components of the primary system were performed for the first time, the systems in one case being filled with sodium, in the other case being empty. The measurements were carried out with a portable measuring station. The spectra were later evaluated in the laboratory.

In the light of gamma spectrometric measurements, the ratio of activities of $^{137}Cs/^{134}Cs$ in a rod of the first defective KNK fuel element, which is 35, corresponds to the ratio of the two Cs isotopes in the sodium condensate samples taken from the upper rotating shield gap.

Rough estimates indicated a release of approx. 7% of the ^{137}Cs inventory generated in the defective rod for the second KNK fuel element defect.

For monitoring the tritium activity of the primary system, probes operating by the permeation principle were developed, whose sensitivity of detection in sodium is 1 nCi/g of sodium and, in the cover gas, 1 nCi/liter of gas. A test rig for testing the probes has been completed.

2.1.2 Radionuclides in the KNK II Cover Gas

The new sampling station for the primary KNK cover gas has been completed; gas samples are drawn through evacuated steel tanks and analyzed in the laboratory.

Each of the samples (580 ml) was measured repeatedly. As in the case of KNK I, the expected linear relationship between the reactor power and the activity concentration again was found neither for the Ar-41 activation product, nor for the "fission product gases" over the whole power range. All gaseous radionuclides exhibited a kind of "saturation" above 80% reactor power, i.e., lower values than expected. It is at present being debated whether this is in any way related to the gas bubble reactivity decreases, which had occurred mainly below 80% rated power.

In the development of radionuclide traps it was found that cesium can be removed best by means of graphite traps added to existing sodium purification systems or by means of distillation systems for treating the aerosol condensate discharged from sodium vapor traps. A cesium trap has been designed, which is either connected upstream of the cold traps or is to be integrated in them.

2.2 Systems and Components Monitoring

Further background measurements have been conducted on the planned acoustic boiling experiments in KNK II. For this purpose, a total of three plug rigs with high temperature sonic sensors have been used, whose applicability had been studied in previous out-of-pile experiments under sodium and at temperatures up to 600 °C. The acoustic background behavior, which is to furnish reference values for the in-pile boiling experiment planned for 1982, is influenced mainly by flow noises of the coolant and narrow-band electric noise frequencies and their harmonics in the range up to 20 kHz. The noise spectra of the sodium pumps recorded as dominant in KNK I years ago do not play a role in this case.

2.3 Core Monitoring

2.3.1 Global Core Monitoring by Boiling Detection

Development of high temperature resistant sonic sensors with $LiNbO_3$ crystals has been continued. Fabrication of a miniature sonic sensor with an outer diameter of 7.6 mm has been started for use in the Cabri reactor. It is expected that the use of microphones with dimensions of the rod geometry results in improvements in acoustic signal coupling and in higher resolution of the time signals.

2.3.2 Global Core Monitoring by Nuclear Measurements

In KNK II, the on-line gamma spectrometer installed at a bypass of the cover gas pipe has been commissioned. A collimator with various inlet openings is now being moved by computer control as a function of the fission product activity in the cover gas pipe, which produces optimum counting rates in the GeLi system, depending on the activity concentration. For failed fuel detection by means of gas chromatographs, precipitators and xenon adsorption in the activated carbon filter, a test facility for measurements of active cover gas (without leakage traces) has been commissioned. Further measurements are still being conducted using free residual fuel areas of various sizes. A compact DND system is also being developed for failed fuel detection.

2.3.3 Individuel Core Monitoring by Nuclear Measurements

The sensitivity of the measurement of the isotopic ratios of stable noble gases (especially xenon) by means of a combination of gas chromatograph and mass spectrometer has been further improved. As was demonstrated by fuel element leaks in KNK, it is possible to determine the isotopic ratios at concentrations of 100-200 vpb with relative standard deviations of 1-2%. For concentrations of approx. 20 vpb, the respective standard deviation is ± 5%. When applying the measuring technique to problems of fuel element tagging (Concentrations of 0.5 vpb), the noble gases must be pre-enriched on activated carbon. In experiments using Xe-133 as a carrier, the working conditions for adsorption and desorption and the failure volume were determined. The results constitute the basis of an automated enrichment and measuring system.

A so-called beta detector for direct detection of fission products in the coolant outlets of fuel elements has been designed, and its basic functioning has been confirmed in the laboratory. The system measures the beta particles during decay of the fission products. The main problem encountered is the high gamma background of the coolant. A prototype of this novel detector is at present under construction with Toshiba; it is to be tested in the forthcoming Mol 7C experiments.

Finally, on the basis of results obtained in the SILOE experiments, a method of detection using gas sampling lines during reactor operation is being proposed for direct localization of failed fuel elements.

2.3.4 Individual Core Monitoring by Flow Measurements

Since March 1980, one double-magnet transit time probe each has been installed above two fuel elements in KNK II. They serve to monitor the functioning of permanent magnetic flowmeter probes over prolonged periods of operation. So far, the probes have worked consistently well, especially due to artificial pre-aging of the magnets before installation. This pre-aging is achieved by heating to a temperature clearly above the maximum operating temperature. 650 °C was selected for a period of one hour. The magnetic field strengths suffered a permanent reduction of approx. 6% as a result of this procedure. A flow probe of 2 mm outside diameter has been developed successfully for use in test loops. It allows, e.g., to measure the velocity profile across the subassembly outlet.

2.3.5 Novel Measuring and Analytical Techniques

A method has been tested to determine the performance of individual fuel elements also without individual flow measurements. For this purpose, the transit time of the coolant and the sodium flow, respectively, through the corresponding fuel elements is determined from a correlation of the measured fuel element outlet temperature and the measured neutron flux, a model being used for the heat transfer from the fuel to the coolant. Some preliminary evaluations of KNK II data have indicated the potential of this method, which still needs to be developed further. One of the applications of this technique may be in cases where no other possibility exists to determine the power distribution, e.g., in the SNR 300.

Neutron flux noise analyses in the power operation of KNK II showed pronounced resonances in the 3 Hz region. Correlation with a multitude of other signals indicates a coherence with measured fuel element outlet temperatures. Consequently, it is concluded that vibrations occurred in fuel elements. It is interesting to note that the resonance frequency is a function of the power, i.e., the flow.

To detect losses of coolant flow by measurements of temperature fluctuations, the temperature signals of the measurement probe equipped with three-wire thermocouples were analyzed at the outlet of the central fuel element of KNK II. The k-values determined in this way (RMS values of the temperature variations as a function of the temperature rise of the respective fuel elements) remained constant throughout the period of use, irrespective of the reactor power and the coolant flow. However, gas entrainment through the KNK core resulted in considerably higher k-values in all fuel elements monitored. This for the first time demonstrates in a reactor the suitability of this technique for the sensitive detection of minor losses of coolant flow.

3. NEUTRON PHYSICS AND NUCLEAR CORE DESIGN

3.1 Planning and Evaluation of Nuclear Experiments

3.1.1 Reactivity Effects of Material Relocations (SNEAK 12)

The SNEAK 12 project mainly serves to study the reactivity effects of material relocations of the kind which may occur in connection with accidents. It is carried out at Karlsruhe as a common action of DeBeNe and CEA/CNEN.

In the first assembly, SNEAK 12A, a single-zone uranium platelet core of 78 cm diameter surrounded by a blanket 30 cm thick of depleted uranium has been set up. So far, the reactivity effects of voids, streaming channels of various geometries and of steel relocations have been determined. In addition, some control rod experiments have been performed, which were intended to establish a connection between SNEAK and BIZET experiments. At the present time, experiments are carried out on fuel movements.

The reactivity worths of central voids were calculated easily (within 3%), as was to be expected, while a difference of 14% between calculations (S_4 transport calculation) and experiments was found for an eccentric void.

Streaming channels of various geometries, but identical void volumes, exhibited only very small reactivity differences. The reactivity effect of the relocation of steel without any changes in the total quantity was calculated in advance to a precision of at least 8% when doubling the local concentration, while a discrepancy of 11% was found for four times the local concentration.

3.1.2 Critical Experiments for Large Breeders

RACINE:

The RACINE project is being carried out in the French MASURCA zero power reactor in Cadarache as a joint experiment of DeBeNe and CEA/CNEN. It serves for studies of heterogeneous annular cores.

In the course of the measurements performed in 1980 on the first assembly, RACINE 1A, flux asymmetries had been encountered, which were traced to a slight asymmetry of loading. For correction, the experimental setup was modified. By June 1981, those measurements had been repeated which had suffered particularly from asymmetries, especially those of power distributions. Subsequently, the position of the central annular blanket was shifted 5 cm to the inside (RACINE 1B) and to the outside (RACINE 1C), respectively. The development of the fission ratio of Pu-239 in both cases showed the market influence of the position of the annular blanket.

In the further course of the program, the distribution of gamma power was studied in the 1A assembly. In this connection, 400 Li-7 thermoluminescence detectors were irradiated with a Co-60 source at CEN Mol for calibration purposes.
Evaluation of earlier gamma measurements performed in RACINE 1A indicated that the gamma dose rate in the central blanket island was 40% of that in the first fission ring.

Finally, the extensive control rod program was started in November 1981, which is going to extend into late 1982.

Throughout the year, staff members of KfK, IA and BN had been delegated to Cadarache for participation in the RACINE project.

BIZET:

The BIZET project was carried out at the British ZEBRA zero power assembly of Winfrith as a joint experiment of DeBeNe and UKAEA in the period between 1976 and 1980. It served for studies of large homogeneous and heterogeneous cores. Evaluation work on the DeBeNe side in 1981 was concentrated on control rod experiments. The reactivity worths and power distributions had been measured for many control rod patterns in the BZB assembly, a large homogeneous two-zone core. Recalculation indicated that, also for the asymmetrical control rod patterns corresponding to skewed load operation, the reactivity worths could be determined with sufficient accuracy, i.e., less than 10% difference between the calculation and the experiment.

Also for the BZC/1 island core, subcritical control rod experiments had been performed. For symmetrical control rod patterns, the deviation between calculation and experiment was around 5%, which roughly corresponds to the results obtained in homogeneous cores. When comparing "natural" and enriched B_4C, a discrepancy of a few percent was found between the calculation and the experiment, as previously in the homogeneous BZA and BZB assemblies.

Comprehensive calculations had to be performed for all control rod evaluations so as to take into account deviations from the point reactor model.

3.2 Nuclear Data Sets

3.2.1 Measurement and Evaluation of Nuclear Data

Work has been performed for some time already on a more recent 26-group set, KfKINR 2, whose weighting spectrum is tailored to the cores of large breeder reactors (roughly 1 GWe). The source data have been made available. Fitting, testing and documentation still need to be carried out.

The RCN-3 data library for fission products has been increased to 36 materials, the most recent additions referring to the stable isotopes of Pd and to Pd-107. The whole ENDF/BV fission product library has been converted into 26-group data, from which pseudo-fission product data were generated for application in the computation of fast reactors.

The cross section library for corrosion products in the primary systems of fast breeder reactors has been expanded by the Cr-50, Fe-54, and Zn-64 nuclides.

Much of the interest in nuclear data is devoted to the actinides, whose knowledge is required also specifically in connection with problems of the fuel cycle. Measurements have been begun on the Van-de-Graaff generator to determine the capture cross section of Am-243. The new evaluation of the actinides according to the present state of the art has been completed, for the time being, for Am-241, Am-242m, Am-243 and Cm-244 for KEDAK; one-group cross sections and resonance integrals have been set up. At present, comparisons are being conducted with other evaluations and integral measurements. In this connection, and for project application, a data set has been created from ENDF/BV basic data for 24 actinide nuclides in 36 energy groups. Major differences were encountered in some respects relative to the first actinide data file created in 1979.

3.2.2 Adoption and Supplementation of Nuclear Data

Within the framework of an international benchmark comparison for a heterogeneous fast breeder core of the SNR-2 power category, major differences in the fission and capture cross sections of Pu-239 and U-238 had been found between Studsvik and Interatom, although both sides had started on ENDF/BIV nuclear data for group data generation. In a number of comparisons with identical group structures and weighting spectra, differences in the resolved resonance region partly in excess of ±10% were found, and others up to 4% occurred in the corresponding one-group cross sections. Comparison of the Interatom processing code with the corresponding US codes mainly resulted in agreement better than 1%. Systematic differences up to +4% merely occur in the unresolved resonance range for Pu-239 (n,f), the cause for which has not yet been fully elucidated.

3.3 Diffusion and Transport Programs for Nuclear Core Design

The BRUST design program has been further improved by

- matching enrichments to the power peaks as a function of burnup,
- taking into account various isotope vectors,
- considering the development of power as a function of time,
- taking into account reprocessing costs as a function of zones.

One- to three-dimensional diffusion codes and one- and two-dimensional coarse mesh transport codes have been adopted from the University of Illinois, which operate by the method of nodal Green's functions. However, these codes are still in the early stages of development. While the examples furnished along with the codes resulted in impressive reductions of computation times (in one case, e.g., by a factor of 20 compared with the three-dimensional D3D diffusion code), application to a realistic sodium breeder problem still resulted in convergence difficulties.

The KASBA and D3BIER programs for three-dimensional calculations of reactivity changes due to core element movements have now been completed. As a consequence, this important contribution to the dynamic behavior of large fast breeder cores can now be determined with greater accuracy and reliability than before. Minor shortcomings and faults appearing only in application will still have to be removed.

3.4 Basic Core Design Work

The influence of rod diameter upon relevant core characteristics has been studied over a wide range (6-12 mm outside diameter). The maximum positive void effect is some 40% smaller in the thicker rods than in the thinner ones. The number of absorber positions required for an SNR 2 has been reduced to 37 from originally 55 by making use of various effects and by careful configuration. This seems to solve the space problem for in-core instrumentation and the absorber drives in the rotating shield.

4. SAFETY

4.1 Studies of Hypothetical Accidents

4.4.1 Initiation and Disassembly Phases

Comparisons of the LOF accident in homogeneous SNR 2 Cores with that in the SNR 300:

The SAS3D analyses of hypothetical loss-of-flow (LOF) accidents for a large 1300 MWe breeder reactor (SNR 2) with a homogeneous conventional core were compared with corresponding results for the SNR 300. Simulation of the expected events showed non-energetic developments of the primary excursion in both reactors. Early recriticality following the primary excursion is more probable in the SNR 300 than in the SNR 2.

In the SNR 2, which was studied with partly and fully burnt-up elements at the end of the equilibrium cycle, the power amplitudes produced during the primary excursion were slightly higher than in the SNR 300, despite a milder overall development.

For reasons of geometry, the effects of axial fuel expansion are generally some 50% weaker in the larger flatter core of the SNR 2 than in the SNR 300. Also the Doppler effect, despite a more pronounced Doppler constant due to the lower enrichment, is slightly reduced because of the higher operating temperature of the fuel elements. At approximately the same maximum positive void reactivity effect of 4.6 $, the effects mitigating the accident in the SNR 2 are therefore generally slightly lower than in the SNR 300.

In the assumed boundary case of the most pessimistic chains of events, conservative models show an increase in the isentropic fuel work potential from 64 MJ in the SNR 300 to 622 MJ in the SNR 2, i.e., an almost tenfold increase in the mechanical load potential compared with the SNR 300.

Analyses Performed with the SIMMER Code:

The SIMMER code, which had so far been used by KfK mainly for the expansion and voiding phases, has now also been applied to describe phenomena in the transition phase. In a first step, some model cases for energetic recriticalities were recalculated in good agreement with earlier results.

For subsequent analysis of the transition phase, the basic data of the SAS3D analyses on the LOF accident in the SNR 300 were used. That accident is characterized by a mild sequence of events, but the dispersive forces are not sufficient for final shutdown.

The transition from the SAS calculation to SIMMER is taken at a point in time at which most of the fuel rods are destroyed, but the wrapper tubes are still intact. The ensuing redistribution of the freely movable masses of fuel and steel was modeled by means of SIMMER. It was found that there will be prompt excursions, but no high energy releases.

Risk Studies for the SNR 300:

In a joint effort by KfK and Interatom together with the American SAI consultant firm, a study of the risk entailed by a LOF was initiated. For this purpose, two initiating mechanisms were postulated which could result in a LOF in cases of the shutdown systems failing.

4.1.2 Fuel-Sodium Interaction and Material Movement

Events of rod failure and material movements up to the potential formation of blockages under conditions of a major accident in a breeder reactor are being studied by means of the "SIMBATH" simulation technique. The nuclear heat generation is replaced by an exothermal thermite reaction within cladding tubes of dimensions typical of fuel rods. Material movements in the simulation fuel rod and in the cooling channel are recorded by X-ray cinematography.

Rod failure was studied under a number of different experimental conditions in ten single-rod experiments. The parameters varied related to the internal rod pressure, the sodium temperature and the coolant flow rate. Here are the main findings made in these tests:

- Some 60-80 ms after the occurrence of the first rod defect, the sodium has been expelled from the cooling channel by the expansion of the multiphase-multicomponent mixture.

- The fluid materials mainly enter a channel from which the sodium has been voided.

- The thermal interactions with sodium residues do give rise to high peak pressures (up to 160 bar, periods of 0.3 ms), but are not very energetic.

The behavior of a 7-rod subassembly was investigated under transient overpower (TOP) conditions. The subchannel geometry and the annular channel additionally arranged around the hexagonal test section tube corresponded to the SNR 300 Mk.Ia core. Rod failure was initiated for all seven rods with a time coherence of 100 ms. As the rods failed, the sodium was expelled in an axial direction. Shortly afterwards, the first fluid materials generated in the thermite reaction reached the simulated blanket regions. Freezing caused blockages expanding axially and increasing in density. The blockage in the upper blanket region was very solid and impervious, whereas that in the lower blanket region was less solid and partly permeable.

For direct transmission to reactor conditions of the individual phenomena observed in the SIMBATH experiments, the CALIPSO computer program is being developed, which computes the thermohydraulics following failure of a single fuel rod with the corresponding cooling channel.

4.1.3 Vessel Explosion Tests

Verification of the ARES continuum mechanics program on the basis of a second explosion test in the 1:6 SNR vessel scale model showed a high degree of agreement between the test results and a first ARES recalculation. Some remaining discrepancies (e.g., internal vessel expansion, time of water impact on the lid) were studied in parameter calculations by quantifying individual effects (e.g., explicit dependence on strain rates of the materials data, water level in the vessel). This clearly improved agreement with the experiment.

4.2 Studies of Fuel Element Failures

4.2.1 Loss of Coolant Flow with Integral Boiling

Some first experiments on the LOF accident (pump coastdown) and decay heat removal were performed on a 37-rod subassembly. The pump coastdown curve and the pressure loss coefficients were determined. Some preliminary experiments in the single-phase region preceding the actual LOF studies served to detect the influence upon the temperature distribution of the flow velocity and the temperature rise (flow velocity, 1-1.4 m/s; mean rod power, 60-216 W/cm). Several single-phase experiments under LOF conditions were stopped shortly before the boiling temperature was reached. Also the influences on temperature distribution of the pump coastdown curve, skewed load and partial load were studied.

In connection with decay heat removal, studies on the same 37-rod bundle are conducted under natural convection conditions. Single-phase experiments served to determine the conditions initiating boiling. Boiling experiments under skewed load conditions followed (average rod power, 8 W/cm; skewed load, ±8%).

The experimental results on LOF and decay heat removal are used, among other purposes, to verify the corresponding computer codes. The BACCHUS computer program adopted from CEN Grenoble and modified by KfK (two-dimensional, single-phase, transient) was used to recalculate three 7-rod experiments (LOF experiments up to the onset of sodium boiling). Calculations and experimental values agreed well. A three-dimensional version was developed and tested. It is used for single-phase transient experiments on the 37-rod subassembly.

4.2.2 In-pile Studies of Fuel Element Failure

In-pile experiments on the behavior of fuel elements under local loss of coolant flow (Mol 7C):

Post-irradiation examinations in the hot cells have been completed for the first two experiments, Mol 7C/1 and Mol 7C/2. For the Mol 7C/3 experiment they will be started in early 1982. In all three experiments there has been no propagation of the fault from the destroyed blockage region to the whole subassembly.

The preliminary results of the signals evaluated to date are these: Despite the higher operating pressure of the filling gas in the case of Mol 7C/1 (68 bar), the time required for the blockage to be decomposed is roughly the same as in Mol 7C/2 (14 bar). The corresponding period of time for Mol 7C/3 (14 bar) is shorter than in the case of Mol 7C/2, despite the higher power density in the blockage, due to the use of UO_2 spheres instead of steel spheres in Mol 7C/1 and Mol 7C/2. In Mol 7C/1 and 2, compact steel crud was formed at the peripheries of the blockages. However, fuel behavior showed clear differences in the two experiments. In Mol 7C/1, larger fuel particles were removed from the blockage region, building up a secondary blockage at the first spacer downstream of the blockage. In Mol 7C/2, most of the fuel columns remained standing, even though they exhibited major swelling. The fuel volume increased greatly in the radial direction. The different types of fuel behavior are explained by the different filling gas pressures in Mol 7C/1 and Mol 7C/2. Post-irradiation examinations of Mol 7C/3 are still under way.

Two other experiments with active blockages and rod configurations are being prepared. Fuel rods from the KNK II reactor with different burnups will be used, namely 17,500 and 48,000 MWd/te, respectively. The overall length is 2475 mm with 950 mm for the fissile core region; the UO_2-PuO_2 fuel largely corresponds to that of the SNR 300.

TOP Program at HFR Petten:

The TOP program at ECN provides for 18 mild to moderately strong transient experiments on fuel pre-irradiated both for short and longer periods of time. The test parameters are the burnup, energy input, rate of energy input, rod power and, in a number of cases, also the width of the cladding/fuel gap. The experiments are to furnish quantitative information on the behavior of advanced fuel rod models in the phase immediately preceding failure. For 1982, three TOP experiments have been planned with fuel pre-irradiated for short periods of time.

CABRI Experiments:

The stainless steel loop so far used in the CABRI reactor is being replaced by a Zircaloy loop. This allows higher energy levels to be released in the test rod. Recommissioning of the reactor was delayed by unexpected difficulties in a Zircaloy-steel weld; consequently, a LOF experiment without a power transient was begun only in December 1981. The test rod used is a rod pre-irradiated to 8000 MWd/te in the Phénix reactor.

Evaluation of previous experiments has been continued. Agreement between the experiments and advance calculations was found to be good. Experiment A2 showed that rod failure in a fresh fuel rod will not occur even if an energy of 1.1 kJ/g is added to the rod during a power transient. This energy input results in almost complete meltdown of the fuel at the most highly loaded parts of the rod. The experiments performed so far have shown with respect to thermal axial fuel expansion that free movement must be expected to occur as long as there are no major contact pressures between the fuel and the cladding tube.

Hodoscope evaluation has been developed so that now also transient movements of the fuel in the cooling channel can be indicated quantitatively. Such movements were observed in the A3 experiment, in which a fresh fuel rod was made to fail.

Transient Experiments in ACRR:

In the ACRR test reactor of Sandia National Labs., U.S.A., transient experiments are being conducted in cooperation with KfK, in which open problems of fuel rod failure are being clarified.

In the fuel disruption series of the total of 10 FD2 and FD4 experiments, specimens are raised to typical steady-state conditions and carried to, or just short of, failure by means of a single pulse. Increasing the power of ACRR allows the radial flux depression to be drastically cut from 4 to 1.5 and a preheating time of roughly 3 s to be achieved. In this way, temperature gradients typical of reactor conditions can be established and temperature inversions during transients avoided. Boundary effects are greatly reduced by extending the length of the fuel rod sections from 1 to 5 cm. It is possible to start with an internal gas pressure up to 3 bar. A device for extracting the fission product gases at the grain boundaries for analysis in a mass spectrometer has been built at Karlsruhe, a corresponding system at Sandia for analysis of intragranular gases. The test series was started in late November 1981 with a few tentative experiments conducted with unirradiated fuel.

In the Equation-of-State (EOS) series in ACRR, especially the fuel vapor pressure at extremely high temperatures corresponding to major accidents is measured. For those experiments, crucibles and specimen capsules were fabricated at KfK and delivered to Sandia. Two delegates from KfK work with Sandia. Additional experiments on the same subject are being conducted in the SILENE reactor in France and the VIPER reactor in Britain.

4.3 Safety Studies of Specific Component Groups

4.3.1 Containment Studies

To differentiate between the effects of thermal and radiological burdens on the thermal insulating materials of the containment, a pre-irradiation test of a purely thermal kind was carried out in BR 2. Under startup conditions of the planned in-pile experiment, the releases of humidity from the different materials were determined as a function of time, temperature, and flow velocity of the nitrogen carrier gas.

4.3.2 Studies of the Core Catcher System

NaK System in the Core Catcher System for the SNR 300:

The most important components of the NaK cooling system (level probe, flowmeter, EM pump) of the SNR 300 core catcher were tested on a full scale basis under SNR conditions. If the necessary safety precautions are observed, handling the liquid metal at room temperature involves no problems. Also cleaning components wetted with NaK created no difficulties. Functioning of the components of the NaK cooling system is ensured under the planned operating conditions.

Experiments on the Coolability of Debris Beds:

Phenomena studied were the dependence on particle diameter of the dryout heat flux, the influence of surface tension, the power history, and the dependence on the height of the bed. A first experiment was performed with a stratified bed. Only the model by Lipinski halfway satisfactorily reflects the dependence found experimentally, but also this model overestimates by a factor of two the dryout heat fluxes for small particles. The stratified bed studied in this case had been built up of four partial layers 2.5 cm thick with different particle diameters, the large particles being in the bottom layer. In the stratified bed, a formation to be expected whenever fuel particles settle in a coolant, the dryout heat fluxes are more than a factor of two lower than in the mixed bed. Similar findings were observed in the in-pile experiments on UO_2 particles in sodium performed at SANDIA.

Materials Data for Accident Analyses:

The question to what extent the transport of radiative heat contributes to the thermal conductivity of the molten fuel can be clarified only if the optical absorption constant of the fuel is known. For this purpose, optical measurements were conducted on liquid UO_2 in order to determine the optical constants of the liquid fuel as a function of wavelength. Knowing the absorption spectrum, it is possible to determine the optical transmission of liquid fuel for heat radiation. It was found that liquid oxide fuel is opaque and does not become transparent for heat radiation even at higher temperatures. An increase in the thermal conductivity of a molten core with rising temperature due to internal heat radiation can thus be excluded in the light of the high absorption constants found in the visible and near-infrared spectral range.

4.3.3 Studies of the Heat Transfer System

All modules of the program for calculating dynamic long-time transients in the steam generator system were integrated in the overall program. It now describes the following components:

- The steam generator with the evaporator, superheater and interconnecting pipes,
- the water separation system with a sediment bowl, a normal and a fast drain, and the corresponding controls,

- the feedwater supply system with the main and the emergency feedwater pumps, piping system, actuators, and control system,
- the main steam system with the main steam pipes, turbine, bypass station, actuators, and control systems,
- the feedwater preheating system with the emergency feedwater preheater, drain heat exchanger, decay heat condenser, emergency feedwater tank, pipes, and the control system.

To check the program, the post-scram behavior of the system was studied with the different possibilities of decay heat removal.

4.4 Environmental Impacts

4.4.1 Aerosol Problems

A model for calculating the behavior of aerosols penetrating fissured leakage pathways in containment walls has been developed by ECN and fitted to experimental results. This aerosol behavior can now be described in a consistent way. For particles of a size above 1 μm, inertia effects were found to be the dominant plate-out process. In the case of smaller particles, there may also be diffusion. At low flow velocities, sedimentation is important. Difficulties in interpreting the results are due to inaccurate knowledge of the true shapes of the leakage pathways and the sizes of aerosol particles.

4.4.2 Sodium Fires

Extensive studies of sodium pool fires were conducted by KfK on burning areas of 2-12 m^2 and sodium quantities of 150-500 kg.

Intercomparison calculations with the SOFIRE-II code developed in the U.S.A. and adopted by KfK initially showed the code to underestimate the values determined experimentally for the sodium temperature, burning rate and gas temperature. As a consequence, the burning rate equation used in the code was modified by the values found in several experiments for the gas transport coefficient and for the stoichiometric burning rate. With this modification, the gas temperature and the gas pressure were now calculated in a conservative way. The NABRAND code developed by Interatom was used for recalculations besides SOFIRE II.

The PARDISEKO code developed by KfK was employed to calculate airborne mass concentrations of the burning aerosols. Agreement between the PARDISEKO calculation and the experiment is relatively good for the time after the fire, but less so while the fire is burning. For this reason, it seems to be necessary to modify the code for this case.

5. THERMOHYDRAULIC AND TECHNOLOGICAL STUDIES OF CORE COMPONENTS

5.1 Theoretical Studies of Core Elements

5.1.1 Thermohydraulics of Fuel Elements and Blanket Elements

After completion of the VITESSE version for fully developed flows in subassemblies, the VITESSE program was further developed at ECN for two-dimensional calculation of turbulent events.

At Interatom, the COBRA III C program was expanded (axially variable rod geometry, iterative calculation of the heat flux density at the rod surface), so as to allow non-steady state temperature distributions to be calculated in absorber elements.

Fuel element outlet temperatures measured in KNK II agree relatively well with the calculated data.

Recalculations of core structure experiments at Cadarache (19 SPx-1 elements) were continued and analyzed. It was seen that the three-dimensional DDT program, in the version including friction, interprets the experiments satisfactorily, uncertainties probably being due primarily to the locally dependent friction coefficients.

5.2 Studies of Fuel Elements and Blanket Elements

5.2.1 Flow and Temperature Distributions in the Subassembly Region

Thermohydraulic Experiments in Nominal Geometries:

In a model experiment, the influence of the geometry of grid type spacers upon flow distribution in the rod subassembly was studied. The test section used was a replica of a section of a fuel element subassembly on a 6.5:1 scale with 16 full and nine partial rod dummies. Two Mk.II spacers with flow aprons completed the subassembly section.

Velocities in the subchannels of the subassembly were measured by means of the laser Doppler anemometer. At six different levels downstream of the spacers, the velocity profiles were measured in the corner channel and the wall channel and, as far as possible, in the adjacent central channel.

The deformations of the velocity profile caused by the grid are still clearly recognizable at L/D = 2.5. Disturbances in the velocity profile gradually decrease with increasing distance downstream of the grid, a relatively smooth profile being developed after 36 D.

Comparison with the velocity profiles downstream of a grid without a flow apron indicates that the aprons cause a major reduction in coolant flows in the wall channels. This largely avoids supercooling of the wall channels.

Thermohydraulics in Non-nominal Geometries:

Studies in air on flow and turbulence distributions in subchannels of rod subassemblies have been continued. The results confirmed the high anisotropy of the turbulent momentum exchange in the gaps between the rods and between the rods and the channel wall, which had been found earlier for the same conditions, also for asymmetrical positions of the rod subassembly within the rectangular channel. Intercomparison calculations with the VELASCO computer program showed good agreement for the wide wall channel, but differences up to 25% in the narrow wall channel for the distribution of wall shear stresses over the rod.

Measurements of local velocity distribution in a 19-rod subassembly in water demonstrated the influence of eccentric subassembly positions in the fuel element wrapper tube. At an eccentricity of 1.5 mm relative to the concentric position, which was studied in this case, the velocity profile changed along the edge zone of the subassembly corresponding to the change in flow cross section. Along a transverse line across the subassembly, the eccentricity causes sizable changes in velocity practically only up to the first central channel row. In the innermost central channels, the velocity fields measured practically show no differences between concentric and eccentric subassembly arrangements.

The three-dimensional temperature field measurements in an electrically heated 19-rod subassembly with grid type spacers in a sodium flow have shown the formation of azimuthal temperature profiles to take place in very different ways with increasing heated length, depending on the position of the rod in the subassembly cross section. While the maximum differential temperature on the periphery of the central rod remains low and constant, it reaches five times and three times, respectively, that value in the corner rod and the wall rod in the axial position of $L/D_h = 97.2$.

- The influence of the grid type spacer results in local temperature increases only within the grid type spacer proper. The maximum differential temperatures around the rod periphery in the center of the grid are up to 180% (central rod) above those right outside the grid type spacers.

- Slight rod bowing only results in minor local temperature increases. Only reduction to less than 70% of the narrowest gaps between the rods approaching each other in the plane of measurement causes a major increase in cladding wall temperatures.

The CIA, VELASCO, VITESSE computer programs were tested to convert into design programs the data obtained experimentally. For the 19-rod subassembly without a spacer, the subchannel velocities in the central and wall channels calculated by these programs and referred to the average subassembly velocity show deviations between the calculation and the experiment, which are slightly higher than the experimental error band of ±2%. For the corner channel, the deviations between the velocities determined experimentally and those calculated are up to 10% in the lower range of Re numbers. These deviations necessitate improvements in the model setups.

BN, inter alia, determined the velocity distribution of a water flow in an SNR Mk.II subassembly by spark eroded grid type spacers with flow aprons. In six triangular and two rectangular subchannels, the individual mass flows were measured by an isokinetic sampling technique at eight positions each downstream of the grid. In most of the triangular subchannels, the flow velocity decreases downstream of the grid, before the next apron causes the flow to be deflected inward. In rectangular subchannels, the velocity initially increases sharply downstream of the grid and then decreases when reaching the apron.

The temperature distribution in an electrically heated 19-rod subassembly in sodium with a spark eroded grid type spacer with a flow apron was also determined by BN. A special device made it possible to adjust the subassembly eccentric relative to the wrapper tube. In eccentric positions, a major increase in temperature of the wrapper tube was found on the side where it was contacted by spacers.

6. REACTOR VESSEL, HANDLING AND SODIUM TECHNOLOGY

6.1 Studies of Temperature Fluctuation across a 7-Element Assembly in Sodium with a Representative Cross Section

The temperature fluctuations encountered across an SNR 300 core element assembly consisting of two fuel elements and five blanket elements were studied at a sodium temperature of 720 K in the fuel elements. The difference in temperature relative to the colder blanket elements was between 45 K and 100 K at the element outlet, depending on the total flow.

Stochastic temperature fluctuations were detected in the frequency range up to 20 Hz. Depending on location, the amplitudes of the fluctuations were up to 80% of the differential temperature of the sodium jets ejected and were attenuated in the boundary layer by factors between 0.8 and 0.4, again as a function of frequency. The maximum amplitudes were encountered above the instrumentation plate at a level of approx. 200 mm above the outlet plane and decreased subsequently with increasing height. The measurements will be evaluated systematically in 1982 and will be compared with the results of a full scale model experiment to be performed with the same geometry, but on water as a flow medium. This method is to furnish criteria for extrapolability of model experiments.

6.2 Laminar Sodium Flows in Pipes

In horizontal pipes of large nominal widths, steady state and non-steady state temperature profiles may result in the formation of laminar flows and temperature streaks at low flow rates. A loop has been built to study these phenomena. It contains a straight pipe section 14 m long of 800 mm nominal width, at the top of which a sodium flow can be fed into the main flow from the top. The difference in temperatures between the two sodium flows may be up to 250 K in absolute terms. Downstream of the feed nozzle, 20 thermocouples each have been installed at five measuring planes over a vertical diameter and around the periphery of the pipe on the inside. The throughput of the main pipe can be controlled between 1800 m^3/h and almost zero; quantitatively, the feed flow can amount up to 800 m^3/h.

In the range of these parameters and with the additional possibility to feed hot or cold relative to the sodium temperature of the main flow, measurements will be performed in order to detect the boundary conditions for the formation of laminar flows and create an experimental basis for theoretical calculations.

The loop was started up in early October 1981. Some first test results obtained at flow rates around 0.2 m/s in the test section showed a major dependency of the temperature profile generated on the flow ratio of the two sodium flows. At specific test parameters, a stable temperature profile was established across a vertical diameter, which showed differences in temperature up to 70% of the differential temperature of the two sodium flows.

6.3 In-service Inspections

6.3.1 Visual Inspection of the Cover Gas Space Inside the Vessel

The experts require that the reactor vessel and its internals be inspected during in-service inspections. Above the operating sodium level, optical means can be used. Studies have shown that such optical inspections can be performed by means of TV cameras.

The inspection system has been tested in the cold condition. Some first pictures from the reactor vessel model of Interatom are available. Some hot tests have been carried out without a camera to test the cooling system. The cooling system is being optimized at present. The lighting system for the camera must be improved considerably. The inspection system has been tested for use with sodium.

6.3.2 Establishing a Concept of Handling Facilities for In-service Inspections of Primary Pipe Systems

The prototype of a system for guiding flexible endoscopes for optical inspections of primary pipe seams is under fabrication. Some preliminary tests were conducted in November 1981.

7. HEAT TRANSFER SYSTEMS AND COMPONENTS

7.1 Sodium Pumps for Large Breeders

7.1.1 Design of Centrifugal Pumps

Theoretical and Experimental Studies of the Hydraulic Designs of Large Sodium Pumps:

The development of an upstream impeller ("inducer") has been further pursued at Neratoom. The combination of inducer and impeller proved to work satisfactorily; the speed can be increased, for the same systems conditions, without causing dangerous cavitation. The dimensions and the weight of a sodium pump can be reduced compared with those of the previous design.

Hydraulic Design of the Primary Pump:

The first design draft of a pump for the loop type concept is now available. Application of an inducer allows the diameter to be limited to 2.35 m for the systems boundary conditions outlined below; the speed for this design is 800 rpm. Without an inducer, the speed would be 650 rpm and the diameter 2.75 m, which would mean a 37% increase in weight.

7.2 Intermediate Heat Exchangers

7.2.1 Design of an Intermediate Heat Exchanger (IHX) for the SNR 2

IHX for the Loop Type Concept:

The feasibility of an SNR 2 primary system of the loop type with a primary pump on the cold leg is being studied. This concept requires an IHX with a unit size of 864 MW and a low pressure drop on the primary side. The possibilities for such a component have been studied.

Strength Analyses of IHX:

The vibration behavior of the tubes of an IHX bundle for the pool type is being studied; in particular, strength calculations have been carried out on this design. It has been proved that, under Super-Phénix conditions, the load accumulated after a certain number of cycles remains within the limits of the ASME Code. The calculations are being pursued further, taking into account creep and fatigue phenomena.

Construction of an IHX Test Model:

A scaled-up version is now being studied with respect to the design of an IHX model to be tested in the 50 MW test facility of Hengelo. This will serve to minimize the radial differential temperatures on the secondary and primary sides.

7.2.2 Na-Na Heat Exchangers for Decay Heat Removal

In order to study the behavior of a heat exchanger of the bayonet type for decay heat removal under natural circulation conditions, a model with four rows of tubes has been designed (4 to 5 MW). A decay heat removal cooler is to be tested at the 50 MW test facility of Hengelo in the near future. A two-dimensional computer program for steady-state conditions has been established to describe natural circulation in a heat exchanger bundle of the bayonet type.

7.3 Steam Generators (SG)

7.3.1 Conceptual SG Design

Development of a Forced-Convection SG and Alternative Concepts:

Within the development of the SNR 2 steam generators it has been agreed to plan for two SG per system, i.e., for a power of 435 MW per SG. Both a helical and a straight tube SG have been designed to these power levels. One separate detailed study referred to the bellows compensating expansion in the 435 MW straight tube SG.

It was also found that SG for a pool type reactor, such as SPx-1, differ from SG for loop type reactors, such as the SNR 300, only in certain details; a primary sodium pump on the cold leg will reduce the design pressure in the SG.

7.3.2 SG Design

An improved computer model for major leakages is being developed. A program has been made available to describe a non-steady state flow of liquid sodium (inter alia for calculating pressure waves initiated by a sodium-water interaction) in pipes with elastic and plastic straining of the material, respectively. The laws of conservation of mass and momentum are solved by the characteristics method. The program is able to describe pipe branchings, pumps, rupture disks, vessel at constant pressure, pipe blank covers, etc. Fitting to characteristic materials data has been attempted. Several experiments have been run and some supplements have been planned.

For pressure waves in three-dimensional structures (such as SG parts), a method has been devised to solve the three-dimensional wave equation for potential flows. The solution is based on Kirchhoff's law for retarded potentials. Integration only relates to the boundary of the volume.

7.3.3 Development of SG Components

Development of Large Compensators:

The De Schelde company of Vlissingen develops bellows compensators for large straight tube steam generators with diameters between 1000 and 2000 mm with an expansion range up to 80 mm.

Very high accuracies have been achieved in fabrication.

7.4 Process Development for Large Components

7.4.1 Development of Welding Techniques for Heat Exchangers

Development of Alternative Tube-Tube Plate Joints:

For high temperature brazing of tube-tube plate joints a brazing cycle has been developed for No. 1.4922 type material, which produces sufficient strength characteristics of the brazed joints. In the tensile test, the tube-tube plate joints failed in the tube, not in the joint.

Electron beam welds of tube-tube plate joints are at present being studied. The first results are very positive.

7.4.2 Studies of the Fabrication Problems of Alternative SG Materials

In studies of the choice of materials for the steam generators of large fast breeder power plants, ferritic 12 Cr steel (material No. 1.4922) was found to be the most suitable material with respect to the required design life of 300,000 h and main steam temperature of 455 ^{o}C.

This material is frequently used in superheaters. Tests have been planned and, in part, already performed to examine the behavior of this material for application in steam generators, especially with regard to corrosion behavior under "departure from nucleate boiling" (onset of film boiling) conditions. A definitive statement is to be made only after completion of further studies also relating to stress corrosion.

7.4.3 Studies of In-service Inspections of Large Components

In 1981, the first mass produced unit of a wall thickness measuring system has been completed. Qualification tests for more precise definition of the measuring range and areas of application of that unit will be performed in 1982.

Also the first definitive unit of the mechanically rotating probe for straight tube bundles, which is driven by a water turbine, has been completed. Some modifications have been introduced following experience with the prototype, which make the probe more universally applicable and allow it to be run in a more stable mode. Extensive qualification tests with this probe have also been planned for 1982.

For the helical tube steam generators of the SNR 300, a prototype of a mechanically rotating probe has been made. This prototype has been tested for manipulation capability in the difficult tube coils of helical tube steam generators. In the course of those experiments it was proved that even in the tubes with narrow bends (R = 80 mm) the measuring principle of the wall thickness measuring unit is feasible. This probe will be further developed and is to be qualified in 1982.

Also the accessories for manipulation in measurements of wall thicknesses in straight tube bundles have been fabricated as prototypes and the experience accumulated in their use will be integrated in the final designs.

7.5 Pipes and Valves

7.5.1 Development of Pipe System Concepts

The studies conducted at elevated sodium flow velocities in pipes have been completed and are at present under evaluation. It was found that it is not so much the vibrations but rather the pressure fluctuations, which require attention. In the light of the findings, maximum velocity has been made a design base parameter with respect to the wall thicknesses of pipes.

7.5.2 Studies of Pipe Systems

Experiments run on compensators in multi-layer pipe bends are about to be completed. The first evaluations of the measurements seem to indicate encouraging results. Experiments with axial compensators are being prepared and will be started shortly.

In studies of thin walled austenitic pipes in the areas of pipe bends of the SNR 300 it was found, in agreement with criteria of the American FFTF reactor, that additional factors giving rise to peak stresses must be taken into account. This clearly exceeded the permissible stress levels in the pipe bends so far designed in the main sodium carrying systems in certain regions. It resulted in the need to use pipe bends with straight ends, so-called tangential bends, and reject the normal pipe bends manufacured several years ago. The use of tangential bends makes for a more uniform, lower stress level within pipings.

8. FABRICATION OF FUEL ELEMENTS AND BLANKET ELEMENTS; FUEL CYCLE

8.1 Developments in Fuel Pellet Fabrication

8.1.1 Fabrication, Characterization and Specification of Pellets

Fabrication of $(U,Pu)O_2$ Microspheres and Pellets:

The COGEPEL program conducted in Belgium serves for the development of a method of fabrication based on co-precipitated $(U,Pu)O_2$ spheres. The purpose is the fabrication of dust-free spheres, which can be processed into high-density homogeneous pellets. Plutonium bearing solutions are used to study two types of feed for external gelation. The working parameters already selected for the fabrication of UO_2 spheres can also be applied to mixed oxide spheres, if these are dried by azeotropic distillation. This drying method has been tested in several batches of 100 g each and resulted in sinterable products capable of pelletization. Although both the SNAM and EGU processes exhibit comparably good results in terms of sintering density, only the EGU method will be pursued further.

Characterization of Mixed Oxide Fuels:

A sputtering installation and a special transfer lock allow fabrication and testing to be performed at SCK/CEN of Pu powder or pellet samples by means of a microprobe or a scanning electron microscope outside gloveboxes. The X-ray facility has been adapted to operating conditions within the glovebox and is available for measurements of the Pu/U+Pu ratio in primary solutions and feed solutions by the COGEPEL process. In this way, the time required for material analysis can be shortened. The applicability of two methods of assaying for the free acid content in a (U,Pu) solution has been tested. The methods are based on complex formation and are now routinely applied in mixed solutions with high metal/acid ratios.

A new gas supply system has been designed at Alkem for the thermogravimetric O/M assay. Compared with the old system, the new facility produces gas of a higher purity and allows the sequence of working steps to be rationalized by programmed gas switching. In the next stage of implementation, furnace control will be equipped with a programmable temperature control system. For analyzing gas residues from pellets and filling gases from fuel rods, two gloveboxes will be designed in an arrangement resembling fabrication conditions. They are to be equipped with facilities for analyses of filling gas and residual gas, quick assays for H_2/N_2, and O/M assays. Methods of assaying for plutonium in filtrates and washing waters have been tested under active conditions. Because of its higher analytical precision, the gamma spectrometric technique was found to be suitable.

Non-radioactive commissioning and testing of the scanning electron microscope has been completed. Work on quantitative ceramography was concentrated on
- improved preparation to meet the more stringent requirements of quantitative structural analysis,
- development of an improved etching technique for mechanically mixed MOX ceramics (especially SNR Mk.Ia),
- elaborating analytical and evaluation techniques for microstructures and porosities.

The existing data bases were used to prove that gamma autoradiography can be employed to check fuel rods from SNR production for mix-ups in enrichment.

For subsequent adjustment of the stoichiometries of SNR pellets, parameters for thermal treatment in a sintering furnace have been elaborated and then qualified as a procedure.

The pressure gaging system installed on the powder press for hydraulic pressure equalization has been transfered into routine operation in SNR production. This facility can be used to monitor the stability of the pressing process.

8.1.2 Methods of Scrap Recycling

To study the influence of increased quantities of recycled scrap on the properties of SNR pellets, scrap from sintering was added to the powder mixes in quantities between 15 and 30%. This produced

- no negative impact on the ceramic structure,
- no undue change in post-sintering behavior,
- no undue change in O/M stability and resultant H_2 values.

8.2 Developments in Fuel Rod and Blanket Rod Fabrication

Belgonucléaire have fabricated more than one third of their fuel rods for the first core of the SNR 300 without any major problems to date. A certain slow-down in the rate of fabrication is due to renewed qualification requirements to be met by welds due to a change in the supplier of the cladding tubes and also to more stringent requirements imposed by the licensing authorities. As a first stage of further development of the future fuel fabrication process, a sub-assembly is at present being fabricated for the first reload of KNK II. This fuel has a higher density and slightly improved solubility. Moreover, fabrication of 50% of four bundles for Phénix has been planned, which is to be based on the MIGRA granulation technique.

At Alkem, 4000 fuel rods with C1 enrichment had been fabricated for the SNR 300 by May 1981. Fabrication was continued until September 1981. After revamping the production facility for fabricating the C2 enrichment (15,000 fuel rods), production was started with the qualification of the ceramic part. For the first time, mixtures are prepared in this step which will have a plutonium enrichment of up to 38 wt.%. At present, 5000 fuel rods are available, which have been tested by the Alkem quality control department and have been found to be up to specifications. The designer and the Technical Inspectorate (TÜV) have accepted some 3300 fuel rods, including mix-up checks. Alkem has to fabricate 5500 fuel rods of C1 enrichment.

8.3 Developments in Fuel and Blanket Element Fabrication

The methods of fabrication and inspection for subassembly completion have been frozen, industrial installations examined. Also fabrication of a dummy part of a fuel element served the same purpose. This allowed the necessary pretesting documents to be written.

The fixtures for assembling components, which are used to pre-assemble fuel elements, have been fabricated and delivered, their operational status tested, and the plants have been modified in accordance with test results. Minor ancillary fixtures, such as units for assembling top and bottom fittings components, have been delivered. Qualification parts and part lots of mass fabricated quantities are available of all fuel element parts. Consequently, mass fabrication of the elements can begin, starting with qualification of component assembly for fuel elements and blanket elements.

8.4 Fuel Cycle Activities

8.4.1 Offgas Cleaning

A test facility for the removal of nitrogen oxides and oxygen by catalytic reduction from dissolver offgases has been commissioned at KfK. The catalyst is a noble metal, the reducing agent is hydrogen.

A Belgian system for the removal of airborne oxygen (down to a few ppm) and an electrolyzer for hydrogen generation have been installed in the GASTON gas cleaning loop at SCK/CEN and are at present undergoing individual tests. The system for electrolytic separation of iodine from a solution has been fully tested on a laboratory scale; a small pilot plant is under construction.

8.4.2 HERMES Facility

The lead shielded cells for recirculating washing solutions and storing the solutions have almost been completed. Installation of the mechanical cell has advanced far, that of the dissolver cell is being tested on a full scale model.

Another dissolution test of 0.665 kg of DFR-455 fuel confirmed the results of the first one. 85.3% of the total amount of plutonium was dissolved in the bulk solution, 8.2% in the second solution (both 10 M HNO_3). 5% of the plutonium remained in the solution as a residue, and 1.5% remained attached to the cladding tubes. After ultrasonic cleaning, only 0.17% of the plutonium remained on the claddings.

Following Belgian-German discussions, the liquid-liquid extraction flowsheets were improved and a design report was prepared.

8.4.3 MILLI Facility

The reprocessing of KNK II fuel rods (from fuel elements turned defective) is covered in Chapter II.

8.4.4 Concept of a MILLI II Pilot Plant

A study will be drafted at KfK by the spring of 1982 aimed at elaborating the concept of a pilot plant for reprocessing breeder fuel elements. This "MILLI II" should be regarded as a precursor to a commercial plant. The minimum cooling time of the fuel elements should be 180 days, the throughput 50 kg of heavy metal/day.

Studies on the Temporary Storage of KNK II Fuel Elements:

Conceptual design studies have been elaborated on behalf of KfK for the temporary storage of spent fuel elements of KNK II, namely

- for dry temporary storage in shipping/storage casks by Transnuklear,
- for dry temporary storage in a block type storage facility by Nukem,
- for temporary storage in water by Kraftwerk Union.

9. PLANT STRUCTURAL MATERIALS

9.1 Materials Qualification for Structural Materials

9.1.1 Basic Aspects of Materials Qualification in Terms of High Temperature Performance

The tests conducted to study the influences of temperature, batch and welding joints on the fatigue behavior of 1.4948 type steel have been completed. Tests conducted at different average stresses so far have only resulted in a minor reduction in fatigue strength with increasing average stress.

The experiments conducted on the thermal stability of 10 CrMoNiNb 9 10 type steel have been completed; in the corresponding low cycle fatigue (LCF) hold time tests, some final experiments for long hold times still need to be conducted.

Work on qualification of the Ni base alloy, Inconel 718, has been continued with studies of thermal stability, creep behavior and weldability.

9.1.2 Qualification of Filler Metal

The creep test program conducted on a 40 mm submerged-arc welded joint has been completed. After a creep prestress load, the mechanical short-term characteristics were determined in a tensile test at 550 °C. Evaluation of the test results exhibited a drop in tensile strength and ultimate elongation over the non-prestressed condition, while the $R_{p\,0.2}$ yield point is slightly higher.

Within the framework of qualifying the type 16-8-2 welding filler metal for long-term high temperature use, another series of stress-rupture tests was started at 550, 600 and 700 °C on a 20 mm 1.4919 plate weld. The test material was tested in two different heat treated conditions.

A first creep test of a large composite specimen (280 mm^2 test cross section) was started.

9.2 Mechanics of Structural Materials

9.2.1 Studies of Inelastic Materials Behavior

On the basis of results obtained from stress-rupture tests and tensile tests of 1.4948 type steel, permissible yield points for long-term and short-term loading conditions were fixed for design purposes.

The stress-rupture tests of a 1.4919 heat (20 mm plate) at 550, 600 and 650 °C were completed at test times of ≤11,900 h.
Another series of tests at 700 °C has been planned to improve the basis for extrapolation to long times.

The response function for the creep behavior of 1.4948 was matched to the 1% yield points as a function of time at 100,000 h. The influence of temperature on the physical properties of differential linear thermal expansion coefficients and the Poisson ratio were represented by third and second degree polynomials, respectively, in the range between 20 and 750 °C.

The relaxation tests conducted at TNO have been completed. The development of stress as a function of time during relaxation loads was measured on four batches of 1.4948 structural material as a function of temperature, initial strain and initial strain rate as parameters.

LCF tests conducted on 1.4948 material were extended so as to cover particularly short strain amplitudes (0.3% and 0.4%). Unlike the behavior at higher strain amplitudes, a second solidification phase was found towards the end of the time to rupture, which was expressed in a more pronounced stress increase. The influence of the strain rate, $\dot{\varepsilon}$, upon the LCF behavior was clarified by tests at $\dot{\varepsilon} = 3 \times 10^{-4}$ and 3×10^{-5}/s (the standard value being 3×10^{-3}/s). As was to be expected, the reduction in the strain rate decreased the number of cycles to failure because the material, due to the longer cycle times, was kept in the range of maximum stresses for a longer period of time.

9.2.2 Mechanical Properties of Irradiated Structural Materials

The irradiation coefficients for long-time stresses and fatigue relevant to design purposes have been diminished following a request by the experts. Influence of irradiation is now being taken into account already starting at 10^{17} n/cm^2 (total).

At KfK, hold time tests were conducted on irradiated 1.4948 type material for combined creep and fatigue loads with hold times up to 24 hours. The numbers of fracture load cycles changed only very slightly beyond hold times of approx. 10 hours; however, this tendency must still be confirmed by further experiments.

At ECN, three other batches of 1.4948 steel were studied within the framework of variations of heats to detect influences of radiation on the creep behavior. The scattering of data is comparable both for the irradiated and the unirradiated condition. Ultimate elongation and stress-rupture strength of all heats investigated are reduced considerably by irradiation. The creep properties of the forged specimens used depend decisively on the final size of the forging.

The stress-rupture strength of welded joints is influenced by irradiation less than be base metal. The creep strain of welded joints attains clearly lower values than the base metal. The filler metal is hardly affected by irradiation.

9.2.3 Fracture Mechanical Studies of Materials

The fracture mechanical work conducted on 1.4948 type steel by Interatom has been continued according to schedule, and additional fatigue and creep induced crack growth experiments were conducted. J-integral measurements were performed at room temperature, 200 °C and 550 °C. Creep induced crack growth tests were evaluated by a number of concepts.

In a number of tests conducted at ECN on the crack growth behavior of 1.4948 type steel it was proved that

- crack growth increases strongly with decreasing frequency from 10 to 0.01 Hz,
- irradiation considerably enhances crack growth at low frequency (around 0.01 Hz),
- variation of the R-value has only little influence on crack propagation in the irradiated and unirradiated conditions.

The measurements will be continued to cover also additional parameters.

9.3 Special Materials Problems with Respect to Sodium Technology

KfK determines the influence of flowing sodium and its impurities upon the fatigue behavior and crack propagation in welds for 1.4948 type steel.

To prepare the LCF experiments under sodium in the FARINA loop, experiments were conducted in air. The fracture load cycles measured by the newly commissioned machine and the variances of readings agree with the values determined earlier in a number of laboratories in the DeBeNe region. Moreover, it was found that specimens age hardened prior to the LCF experiment attained clearly higher load cycles, which must be taken into account when determining the sodium effect.

Some first results on the influence of sodium upon the growth of faults in welded joints under tensile stresses have been determined on round tensile specimens with artificial defects in the CREVONA loop. Symptoms of a sodium effect on crack propagation were detected, because the range of tertiary creep was shortened and creep strain was decreased.

Studies on the influence of the coolant were continued by Interatom on specimens of the 1.4948 type base metal steel additional test results were produced on creep behavior, creep induced crack propagation, and high cycle fatigue characterization.

10. TEST PROGRAMS IN KNK II

10.1 In-pile Experiments

10.1.1 Post-irradiation Examination of Defective KNK II Fuel Elements

The two test zone fuel elements, which had become defective in 1979 and 1980, respectively, were brought into the Hot Cells of KfK after a cooling time of roughly one year. They had not previously been cleared of sodium in KNK II according to the wet gas cleaning technique, because changes in the defects caused by sodium-water interactions were to be avoided. Disassembly showed that the adhering sodium did not disturb the work in any way.

For disassembly of the fuel elements, a prototype disassembly system was used successfully, which is planned for waste management in KNK II.

The very minute fuel damage in the first defective fuel element was found by means of a dry sipping test at around 500 °C. Other localization techniques, which had been employed for test purposes, had failed. In the second element, which had been left in the reactor and operated in the power mode with the defect for more than three weeks, the defective rod was identified by direct observation. The causes of both defects have not yet been clarified unequivocally.

Wear phenomena in the cladding tubes and spacers at their point of contact are probably due to vibrations of the fuel rods/fuel elements. Their extent seems to be influenced greatly by the spacer design.

10.1.2 In-pile Experiments in KNK II/2

For the second core, KNK II/2, the in-pile experiments
- irradiation of structural material rods in operating fuel elements,
- determination of the integral actinide cross sections,

- use of a second materials test element,
- depressurized absorber rods,

were prepared along with the fabrication of the second core. The irradiation facilities will be available in time.

The two ring shaped fuel elements originally planned for the third reactor core will now be advanced in time and will be loaded in the core shortly after the commissioning of KNK II/2.

10.2 Experiments on Reactor Instrumentation

10.2.1 Failed Fuel Detection Systems

The methods of detecting failed fuel elements, which proved to work satisfactorily also in KNK II, are based on the detection of fission product gases and emitters of delayed neutrons. However, because of the high radiation background, interpretation of the signals needs much experience.
Consequently, all known methods of improving the signal-to-noise ratio are tested in KNK II. They include the separation of fission product gases by gas chromatography and their enrichment by adsorption on to activated carbon, and precipitation in an electric field.

Three types of precipitators are tested, which were developed by KfK, in France and in the United Kingdom. The Ar-41 background can also be discriminated by means of a Ge(Li) detector system. Evaluation by process computer control of the changes in time of the gamma spectra obtained in this way furnishes information as to the type of fuel rod failure and the expected development of the damage. The new measuring systems have been combined in KNK II in an extensive test rig which, after completion of the fundamental measurements, will be used for long-time trials and for some preliminary tests with artificial fuel rod failures of different sizes.

10.2.2 In-core Instrumentation

In long-time tests, the fuel element flowmeters, temperature measuring probes and piezo-electric and magnetostrictive sound transducers, respectively, have worked satisfactorily.

To detect losses of coolant flow by measurements of temperature fluctuations, the temperature signals of the measuring probe equipped with three-wire thermocouples were analyzed at the outlet of the central fuel element in KNK II. The RMS values of the temperature variations determined from these data relative to the temperature rise of the respective fuel elements remain constant throughout the period of use, irrespective of the reactor power and the coolant flow. However, gas entrainment through the KNK core gave rise to considerably higher RMS values in all fuel elements monitored. The suitability of the technique for the sensitive detection of minor losses of coolant flow has thus been proved in a reactor for the first time.

10.2.3 Additional Experience in the Application of Measuring Techniques for Plant Monitoring Purposes

For acoustic boiling detection, the absolute sound pressure produced in sodium boiling was determined out-of-pile, and the background signal mainly caused by flow noises was detected in KNK II; from both measurements the finding is derived that boiling should be detectable acoustically.

Studies of the monitoring algorithms proposed for the SNR 300 for the detection of local losses of coolant flow by means of the fuel element outlet temperatures as measured in KNK II have shown the planned dynamic limit control system to be effective over a broad range of power levels.

The compensation method of precision measurements of the fuel element temperature rise has been used for safety related assessment of the distribution of gas bubbles in the KNK II reactor core. The method is at present being studied by CEN Cadarache with respect to its applicability in the Phênix reactor.

For the prototype computer protection system to be installed in KNK II, a structure consisting of four computer groups was selected in which a data exchange is planned by meshing, which is expected to make individual failures tolerable.

10.2.4 Reactor Chemistry

In continuation of the work begun in the KNK I experimental program, the methods of sodium sampling and sodium analysis planned for the SNR 300 are being tested in the primary system of KNK II. A sampling system without any valves has been installed and tested and is now available for sodium analysis.

A method used to detect minor leakages in the steam generator is based on the pronounced rise of hydrogen concentration in sodium due to the sodium-water interaction. Already during the operation of KNK I, hydrogen detectors were installed in the two steam generators for test purposes; their development has now been completed after 20,000 h of operation.

The changes as a function of time in impurity concentrations in the sodium/steam system brought about by changes in the mode of operation (temperature, pressure, flow) affect the corrosion behavior and the formation of protective coatings, respectively, in the tertiary system. Since the impurities enrich in particular in the stable boiling zone, numerous data are monitored in various positions of the tertiary system of KNK II by means of continuously working analytical units.

10.3 Experiments Accompanying Operation

10.3.1 Input Data for Reliability Analyses

Assessing the reliability of technical systems requires empirical plant data. Such data exist in part for conventional power plant components, but are largely absent for the components of sodium systems. For this reason, from the construction of KNK I on, all data about the behavior of some 300 plant components of KNK, such as sodium valves, steam generators, intermediate heat exchangers etc., have been assessed and evaluated. In the meantime, this has produced reliability data with sufficient statistical backing for most sodium components. Special measurements, e.g., have shown the failure behavior of sodium valves to be almost independent of the frequency of actuation.

Above and beyond reliability analyses, particular attention is devoted to observing the operating behavior of the valves in the primary system and the shutdown systems. Major problems associated with the valves were the thermal design of the freeze sections and lubrication of the valve heads. The grease selected after extensive tests is resistant against the influences of radiation and temperature in the primary cell. Regular comparison of specific data of the shutdown systems (torque during raising and lowering, tightness, surface quality of the lifting rods) with the respective values measured during commissioning furnishes early data with respect to changes in systems performance.

10.3.2 Maintenance Concepts

Radioactive substances are deposited in the components of the primary system - pipes, coolant pumps, intermediate heat exchangers, valves etc. -, which make maintenance of these components difficult. To develop a maintenance concept, empirical data are therefore collected about the rise of radioactivity in the primary cell as a function of the length of operation of the reactor.

The dose rate in the region of the primary cell accessible to personnel can be determined at any time during reactor operation by means of a special system. One of the findings has been that the dose rate drops to 10^{-2} R/h within two weeks after reactor shutdown.

This residual radiation determines the exposure of the maintenance personnel during work in the primary cell. For measurements of residual radioactivity, which have been performed at regular intervals since 1973, measuring stations equipped with lead collimators have been installed in various places of the primary cell. Calibrations in dummy systems resembling KNK installations allow quantitative statements to be made about the specific radioactivity deposited.

In late 1981, also a control system in the post-scram mode of operation has been tested using the core outlet temperature.

IV. IMPLEMENTATION OF INTERNATIONAL AGREEMENTS

1. German-French Agreements

Cooperation in research and development and joint commercialization of fast breeder nuclear power plants was agreed upon in 1977 between the German side and CEA France. All knowledge in the breeder field generated in the past and in the next twenty years will be made available to the partners. Items not included are reprocessing and fabrication know-how and any know-how of subcontractors. The agreements expressly provide for both sides undertaking comparably intensive efforts in breeder development. Belgium and the Netherlands are associated to the German contracting parties, Italy is associated to the French partner.

The most important joint R&D projects have reached the following status:

- So far, nine transient experiments have been performed in the CABRI test reactor. Evaluation of the results is still going on. The CABRI experiments are to help clarifying fuel rod behavior under accident conditions. The UK, USA and Japan participate in the experiments as junior partners.

- The RACINE program for studies of heterogeneous breeder reactor cores in the MASURCA zero power facility has been continued in 1981 with nuclear measurements of three ring shaped assemblies.

- The joint SNEAK 12 project mainly serves for studies of the reactivity effects of material relocations in accidents. So far, voids, streaming channels and steel relocations have been studied in a uranium core; experiments on fuel relocations are being carried out at present.

- Studies in the FAUNA test facility on the development of sodium fires are evaluated jointly.

- 19 mixed oxide fuel rods of 7.6 diameter equipped with helical wire spacers, which had been irradiated in the Rapsodie reactor, were subjected to post-irradiation examinations.

- Rod fabrication for the CHARLEMAGNE in-pile experiment was begun at Cadarache in 1981. Within the framework of this experiment, the performance of German and French fuel rod cladding materials is to be compared in Phénix in 1982.

- In order to make available vented absorbers, the PRECURSAB-B2 test rod was put into the Phénix reactor in mid-August 1981 within the framework of a joint DeBeNe/CEA action.

- The first German-French defective rod experiment within the VOLGA program in the Siloe reactor has been completed and evaluated. This program serves to study the behavior of defective fuel rods under operating conditions.

- Irradiation in Phénix of two SNR Mk.II type fuel elements with grid type spacers has been prepared by hydraulic preliminary experiments.

- Future plans also provide for in-pile experiments to clarify the mechanical interaction of fuel with the cladding, study the wrapper tube material planned to be used for Mk.II subassemblies and two fuel elements with helical wire spacers. Transient experiments in the SILENE reactor are being prepared to measure materials data at high temperatures.

Since the breeder agreements were signed in mid-1977, 42 common actions have been agreed upon, one of which has been completed in the meantime.

2. Agreements with the USA

The LMFBR Umbrella Agreement concluded between the Federal Republic and the U.S.A. in 1976 was joined by France in 1978. The agreement provides for a balanced exchange of R&D know-how in the sectors of fuel elements and materials, physics, components and safety. In some areas of the latter three sectors, such exchange has been implemented in the meantime. Special mention should be made of the further development of American programs on calculating major hypothetical accidents. The SAS3D code, which originally came from ANL, has already been used to study a loss-of-coolant accident for the licensing procedure of the SNR 300. The SIMMER code, which originated at LASL, is being verified experimentally by KfK in cooperation with LANL (LASL) and SRI.

In cooperation with SAI, work is being performed on a risk oriented analysis of the SNR 300 and has already produced some first results. In the ACRR experimental reactor of Sandia National Labs., transient experiment are being conducted to clarify fuel rod failure mechanisms, especially to measure the fuel vapor pressure at extreme temperatures. KfK delegates cooperate both at SAI and Sandia.

As a result of trilateral specialist meetings on breeder safety (May 20-22, 1981, Grenoble, and June 15-17, 1981, Washington), moreover, plans provide for

- cooperation on breeder decay heat removal systems with natural circulation,
- mutual information about local faults in breeders. This would apply to cooling channel blockage experiments in the Engineering Test Reactor of Idaho, which are a valuable addition to the Mol 7C series, and possibly to test results in EBR II and FFTF. The U.S.A., in turn, is interested in results of Mol 7C and Scarabee, defective rod experiments in the Siloe reactor, and out-of-pile sodium boiling experiments of KfK, CEN and CEA.

Within the framework of a German-American-French agreement, cladding material irradiation experiments in the Phénix reactor and the Fast Flux Test Facility (FFTF) are being prepared.

3. Agreement with Japan

The exchange of experience with Japan is covered by a trilateral agreement concluded by Germany, France and Japan in June 1978. It covers the R&D exchange in some twenty areas of the breeder field, which list can be adapted to the respective needs. The Japanese PNC together with the UK and USA participate as "junior partners" in the German-French CABRI Project. The present exchange of reports with PNC inter alia covers out-of-pile coolant blockage experiments in sodium boiling loops.

On January 19-26, a trilateral expert conference on radiological impacts of postulated breeder accidents was organized in Tokyo. In the light of the results of the discussions, continuation of studies seems to be indicated especially in the following areas:

- the radiological source term for a Bethe-Tait accident (as yet known only very imprecisely),
- programs to compute larger sodium fires,
- sodium-concrete interactions (on which so far only very few experimental and theoretical data are available).

The continuation of experiments on fission product plate-out and detection in sodium loops is under discussion. These are experiments in the Japanese fission product loop and in the KNK II and BR 2 reactors.

4. Contracts with the United Kingdom

The CAPT agreement signed 1980 by KfK, CEA and UKAEA covers the exchange of results of the German-French CABRI safety experiments against the part accessible to the UKAEA of the safety test results elaborated in the American TREAT test reactor. This part refers to unirradiated and irradiated test rods made available by Britain. A total of 28 experiments are planned with single UKAEA fuel rods and rod bundles in capsules and in the sodium loops of TREAT.

The final report on the COVA project (code validation of the ARES computer program by comparison with experiments) has been presented. The ARES computer program developed by Interatom is being used to analyze the mechanical effects of hypothetical power excursions in the SNR 300. The British experiments comprise explosion tests of tanks filled with water with and without internals.

German-British cooperation also includes fuel rod and materials irradiation experiments in the PFR (partly with French participation) and in the VEC, evaluation of the joint BIZET measurements completed in 1980 on six large breeder reactor cores, experiments on loss of coolant flow, fuel rod defects and ways of detecting them. Possible expansion of the German-British-French cooperation was discussed at a meeting in Risley on January 28-29, 1982.

List of Abbreviations Used

ACRR	Annular Core Research Reactor, Sandia Labs., Albuquerque, N.M., USA
ALKEM	Alpha-Chemie u. -Metallurgie GmbH, Wolfgang near Hanau
APB	Anlage zur Erprobung von Pumpen für Brüter (Breeder pump test facility), Interatom, Berg. Gladbach
BMFT	Bundesministerium für Forschung und Technologie, Bonn (German Federal Ministry for Research and Technology)
BN	Belgonucléaire S.A., Brussels
BR 2	Belgian Reactor 2, Mol
CABRI	Test reactor at Cadarache
CAPT	British-German contract on information exchange concerning CABRI-PFR/TREAT safety tests
CEA	Commissariat à l'Energie Atomique, Paris
CEN	Centre d'Etudes de l'Energie Nucléaire, Mol, and Grenoble, and Cadarache, respectively
COVA	Code validation by British tank explosion tests
DeBeNe	German-Belgian-Netherlands breeder project
DMSA	Demountable Subassembly for fuel subassembly irradiation in the British PFR
DND	Delayed Neutron Detector
DOE	Department of Energy, Washington
dpa	displacements per atom (due to irradiation)
ECN	Energieonderzoek Centrum Nederland, Petten
FAUNA	Forschungsanlage zur Untersuchung nuklearer Aerosole (Nuclear aerosol research facility of KfK)
FR 2	Forschungsreaktor 2, karlsruhe (research reactor)
FPL	Fission Product Loop of Toshiba, Japan
HCDA	Hypothetical Core Disruptive Accident ("Bethe-Tait" accident)
HFR	High flux reactor, Petten
IAEA	International Atomic Energy Agency, Vienna
IHX	Intermediate heat exchanger
INB	Internationale Natrium-Brutreaktor-Baugesellschaft mbH, Berg. Gladbach
INTERATOM	Internationale Atomreaktorbau GmbH, Berg. Gladbach
IWGFR	International Working Group on Fast Reactors of IAEA, Vienna
KBG	Kernkraftwerk-Betriebsgesellschaft, KfK
KfK	Kernforschungszentrum Karlsruhe GmbH
KNK	Kompakte Natriumgekühlte Kernreaktoranlage, Karlsruhe
LOF	Loss-of-flow
MASURCA	Maquette Surrégénérateur Cadarache (zero power assembly)
Mark Ia, Mark II:	first and second reactor cores of the SNR 300
MILLI	High level reprocessing test facility at KfK
NALA	Hot sodium pool aerosol studies at KfK
Neratoom	Netherlands industrial consortium for nuclear technology, The Hague
Novatome	French breeder manufacturing company
NRC	U.S. Nuclear Regulatory Commission, Washington
OECD	Organization for Economic Cooperation and Development, Paris
PAHR	Post-Accident Heat Removal
PFR	Prototype Fast Reactor, Dounreay, U.K.
Phénix	French prototype breeder, Marcoule
PNC	Power Reactor and Nuclear Fuel Development Corp., Tokyo, Japan
PSB	Projekt Schneller Brüter (Fast Breeder Project)
RAPSODIE	Experimental fast reactor at Cadarache
RIPCEX	Cladding materials irradiation in RAPSODIE to determine swelling and creep
SAI	Science Applications Inc., Palo Alto, Cal., USA
SBK	Schnell-Brüter-Kernkraftwerksgesellschaft mbH, Essen
SCK/CEN	Studiecentrum voor Kernenergie, Mol
SERENA	French-German breeder systems company
SG	Steam generator
Siloe	Test reactor at Grenoble
SNEAK	Schnelle Nullenergie-Anlage Karlsruhe (fast zero power assembly)
SNR 300	Schneller Natriumgekühlter Reaktor, Kalkar Nuclear Power Station

SRI Stanford Research Institute, Stanford, Cal., USA

Super Phénix: French demonstration breeder power plant at Creys-Malville

TNO Netherlands industrial research organization, Apeldoorn

TOP Transient Overpower

TREAT Transient Reactor Test Facility, Idaho, USA

UKAEA United Kingdom Atomic Energy Authority, London

ULK Versuchsanlage für Umluftkühlung (Reventing air cooling test facility), Karlsruhe

UNC United Nuclear Corp., USA

VEC Variable Energy Cyclotron, Harwell

VIC "Variable Irradiation Conditions" loop in BR 2

ZEBRA Zero energy facility at Winfrith, U. K.

REVIEW OF FAST REACTOR ACTIVITIES IN INDIA

S.R. PARANJPE

1.0 Introduction

It may be recalled that the 1st stage of India's nuclear energy programme is based on natural uranium fuelled heavy water reactors. Fast breeders have been considered only as a second stage to utilise the plutonium produced by the heavy water reactors. Attitudes and public reactions to nuclear energy are therefore governed by the developments in areas related to heavy water reactors rather than the developments in the field of fast breeder reactors themselves. Although some measure of success has been achieved by the country in increasing the gross national product and reducing the rate of inflation, prices of various commodities have recorded an increase and the same has resulted in an increase in the capital cost of heavy water reactors. But inspite of these increases, these reactors have maintained their competitiveness vis-a-vis coal-fired power stations in areas away from coal fields. On the other hand, there are increased stresses in the transportation system due to increased industrial activity and this is making transportation of increasing quantities of coal to remote parts of the country more and more difficult. Hence, need for nuclear energy is being felt more and more acutely. In this sense, climate for nuclear energy is quite good. While the work on existing projects for generation of electricity has recorded progress, a twin reactor station at Kakrapar in Gujarath State on the west coast of India has been announced and the work has been initiated. Moreover, the current five-year plan provides for start of work on four more reactors of 235 MWe capacity in the next 3 years. A working group has also been formed to evolve a project report complete with cost estimates for a 500 MWe proto-type fast breeder reactor to be made