KfK 3727
April 1984

Fusion Technology Programme

Semi-annual Report October 1983 - March 1984

Compiled by
D. Finken
Projekt Kernfusion

Kernforschungszentrum Karlsruhe

KERNFORSCHUNGSZENTRUM KARLSRUHE

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Preface

The development of nuclear energy systems has been the dominant activity of the Nuclear Research Centre of Karlsruhe (KfK). Extensive r + d has been devoted to reactor design, materials research, fuel reprocessing and safety aspects for LWR and LMBRF reactors, preceding or accompanying industrial development.

The experience gained in these domains is one essential root of the Nuclear Fusion Project, constituted by KfK in 1983. However, fusion research has many years of tradition in the organization. The development of superconducting magnets and neutral beam heating, extensive studies of reactor schemes and test facilities, and research on radiation effects of structural materials have evolved through the years and had in part been implemented into the European effort before through a link to the Institute for Plasma Physics at Garching (IPP).*

KfK has reinforced these relations to IPP by forming a joint venture ("Entwicklungsgemeinschaft Kernfusion")for nuclear fusion development, KfK's share being fusion technology. Through a direct association to Euratom, which became effective in 1983, KfK participates to the Fusion Technology Programme of the European Community.

Most of the work in progress addresses the Next European Torus (NET) and the long term technology aspects as defined in the 82/86 programme. A minor part serves to preparation of future contributions and to design studies on fusion concepts in a wider perspective.

The Fusion Technology Programme of Euratom covers mainly aspects of nuclear engineering. Plasma engineering, heating, refueling and vacuum technology are at present part of the Physics Programme. In view of NET, integration of the different aereas of work will be mandatory. KfK is therefore prepared to address technical aspects beyond the actual scope of the physics experiments. In this sense, development of high power sources at mm wavelength has been initiated in the current KfK programme. These studies on gyrotron type microwave tubes conclude this report.

The technology tasks are reported project wise under title and code of the Euratom programme. Most of the projects described here are shared with other European fusion laboratories as indicated in the table annexed to this report.

A main effort of the past half-year have been expert discussions on the NET strategy and the associated research and development to be carried out in the programme period 1985 to 1989. KfK will further take its share in NET planning by working on study contracts, by direct delegation to the design team and by scoping the r + d programme of KfK to the needs of this developing project. A research center as KfK with its wide cast experience from fission reactor development is particularly equiped and qualified for multidisciplinary research and for the development of components and systems from laboratory to technical scale. It is therefore envisaged, starting from the first round of technology projects, which have addressed only single critical aspects, to concentrate future work in important domains.

The work on blankets will emphasize on a helium cooled blanket with ceramic breeder material to be developed as a reference concept for NET. The subject of tritium extraction and the techniques of tritium processing from the exhaust gas and the breeding blanket to refuel the plasma will be another aerea of concentrated effort. In the field of structural materials, KfK will specialize on advanced materials and on an analysis of testing methods and test facilities suitable for qualification and development of such materials. Superconducting magnet development will maintain its essential role in the KfK project. In the first week of April, shortly after concluding this report, the Euratom LCT coil was tested in our laboratory and has shown the expected performance. Beyond this important step, future work will be directed towards high field conductor development and on poloidal field coils for NET.

J.E. Vetter

* Nucl. Engineering and Design 73, 2 (1982); Special Issue: Fusion Technology in the Nuclear Research Center Karlsruhe

M 1 THE LCT-PROJECT

The LCT-Project is a project of the International Energy Agency (IEA) to develop the technology for the application of superconducting toroidal field coils for future Tokamak experiments. Six toroidal field coils which dimensions and manufacturing techniques should be representative for Tokamak magnet systems will be tested in the Large Coil Test Facility (LCTF) in the Oak Ridge National Laboratory. Three coils come from outside the USA. Japan, Switzerland and Euratom participate in these experiments by the delivery of one coil each.

In early 1984 four of six coils were ready: the Japanese coil, one of the American coils (manufacturer General Dynamics Convair), the Euratom coil and Swiss coil. The Euratom coil built under the responsibility of KfK was delivered and installed in the TOSKA test facility at KfK to pretest the coil before shipping it to the USA.

The test activities in the LCTF were started by a test run with two coils, the Japanese and GDC coil. Leakage and electrical insulation problems at the GDC coil forced an interruption of the test and repairing.

The arrival of the Swiss coil in the USA lead to a reassign of test program by the participants. Three coils will be installed in the LCTF for the first tests whereby the Swiss coil will only be tested cryogenically.

In the second half of 1984 the remaining coils (General Electric, Euratom, Westinghouse) will be installed in the LCTF. The start of the six coil test is scheduled for 1985. The compatibility and a smooth coil installation in the LCTF could be reached by periodical meetings of the participants and assignment of KfK people to ORNL. A common test program was accepted by all participants.

To prevent operation of the LCTF with too many unknown objects and, therefore, to reduce the risks of untolerable delay of the six coil test it was decided to perform the above mentioned pretest of the coil in the TOSKA facility in a very similar way as the Japanese partner did in its domestic test facility. This facility has been completed end of 1982 and was operated several times in 1983 so that it was well prepared when the coil was delivered to KfK end of November 1983.

1. Completion of the LCT-coil

The Euratom coil was developed and built in collaboration of the Institute of Technical Physics (ITP) of the KfK, the Vacuumschmelze Hanau, the Siemens AG/Erlangen with subcontractor Krupp Industriebau (KIS)/Essen.

The Euratom-LCT-coil is one of the three forced flow cooled coils in the LCT-project. The winding consists of 7 double pancakes impregnated by epoxy resin to a monolythic winding block, mounted in two halves of a stainless steel case for reinforcement.

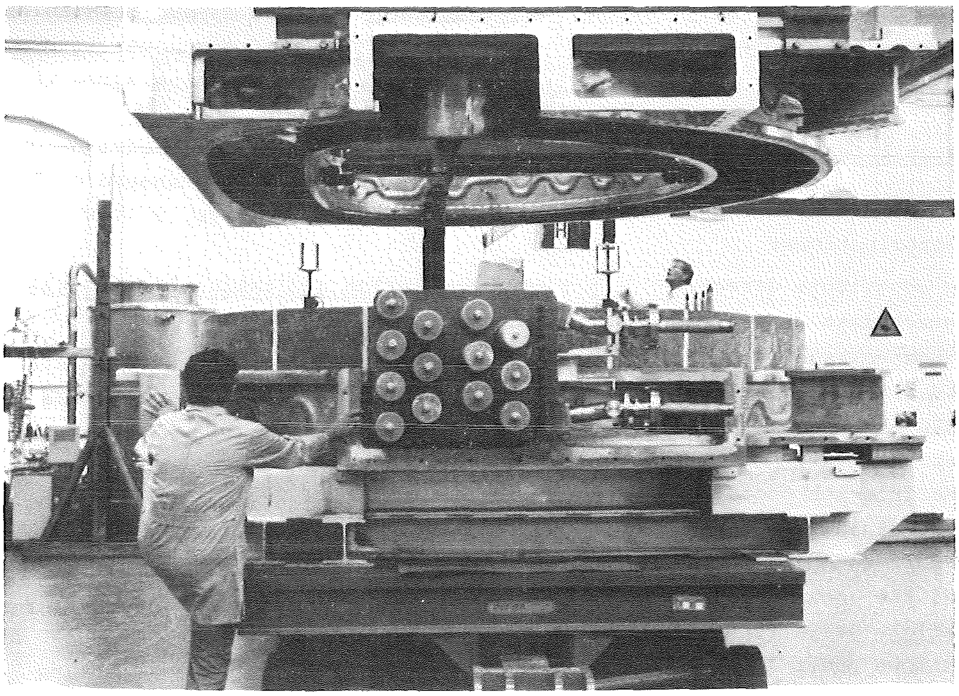

Fig. 1: Lowering of the second coil case half at the first one. In front the pancake connections with gas collectors and voltage taps

In 1983 the main components of the coil were mounted by the manufacturer (Siemens/Erlangen) (V 18944, 17650) with the following steps:

- Impregnation of the winding
- Mounting of the winding in the two halves of the coil case (Fig. 1)
- Closing of the tolerance between coil case and winding by means of stainless steel cushions mounted between winding and coil case and filling them by epoxy resin
- Bolting the two halves of the coil case by preheated bolts to get enough pretension
- Seal welding of the coil case
- Mounting of cryogenic supply pipes and electrical high current feed throughs
- Installation of sensors at the coil and wiring.

After completion of the coil the following acceptance tests were carried out:

- Geometrical dimensions
- Vacuum tightness (< 10^{-6} mbarl/s)
- High voltage test (12 kV DC,8 kV AC, $tg\delta$ measurement, dielectric strength in the Paschen-Minimum)
- Checking of the sensors operation (ca. 200).

The tests were partially repeated after the delivery of the coil to Karlsruhe and before installation in TOSKA some additional sensors were installed.

- Sensors to measure coil deformation the opening of flange groove during excitation
- Piezoelectric sensor for registration of acoustic emission.

Fig. 2 shows the coil a short time before the installation in the TOSKA facility.

2. Preparation for the coil test in TOSKA

2.1 Accompanying investigations

These are two kinds of contributions:

i) Investigations which contribute directly to the coil construction like
   - development of a high voltage (12 kV), high inside-pressure instrumentation lead feedthrough (25 bar to 10^{-5} mbar) at operation temperatures 4 K
   - Measurements of the friction coefficient for the bolted stainless steel flange of the LCT-coil to demonstrate the transmission of the shear forces
   - Measurements of fatigue properties of stainless steel bellows under load cases at 4 K like they will occur in the LCT coil piping.

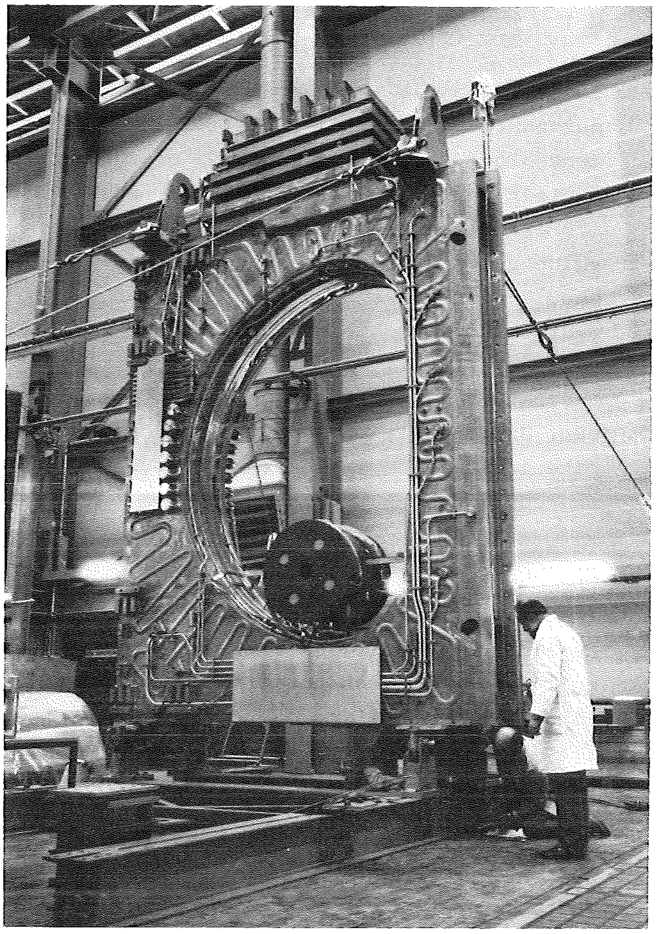

Fig. 2: The Euratom-LCT-coil (rated current 11.4 kA at 8 T, weight: 38 t)

ii) Basic investigations like
   - Investigations about conductor stability made on the LCT-coil conductor in the W7 conductor facility. Results agree with the expected values.
   - Calculations of bending moments during the winding process of D-shaped coils (17685, V 18694, V 19494, V 19493).

2.2 The completion and taking into operation of the TOSKA facility

The TOSKA facility consists of a large stainless steel vessel (5 m Ø, 9 m high with a nitrogen cold wall in it) (Fig. 3). The vessel was placed in a pit to protect surroundings against magnet stray field. A second vessel contains a He-pump with heat exchangers for the generation of forced flow circuit at He temperatures. The facility is connected by cold pipes to the existing refrigerators. An extensive cryogenic control and measuring

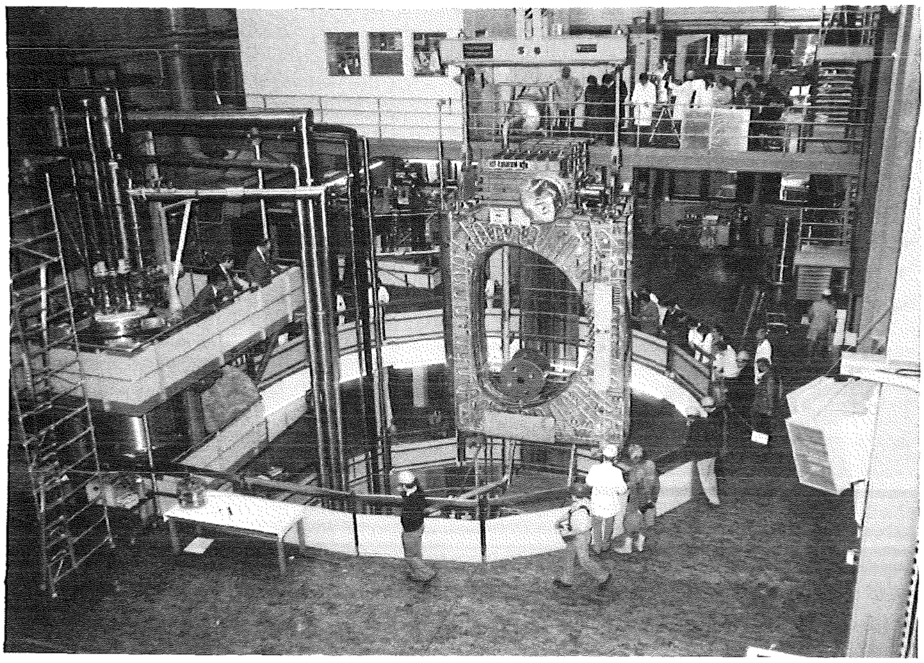

Fig. 3: The transportation procedure into vacuum vessel

system, an electrical power supply for charging and discharging coil at current of 10 kA and data acquisition system also belong to the system. The facility was built by a collaboration of different companies and many contributions from other institutes of KfK.

Test runs of the TOSKA facility serve for measurement of components. In detail the following work was done:

- Determination of the cooling power during cooldown (250-10 K) and static operation at 4 K with helium pump
- Test of a gas cooled current feedthrough with a superconducting bus (cryogenic supply, insulation properties at 8 kV, 10 kA current and contact resistance measurements ∿ some 10^{-9} Ω)
- Test of data acquisition in hardware and software parts (acquisition of data, display programs, calculation programs to convert measured signals into engineering units).

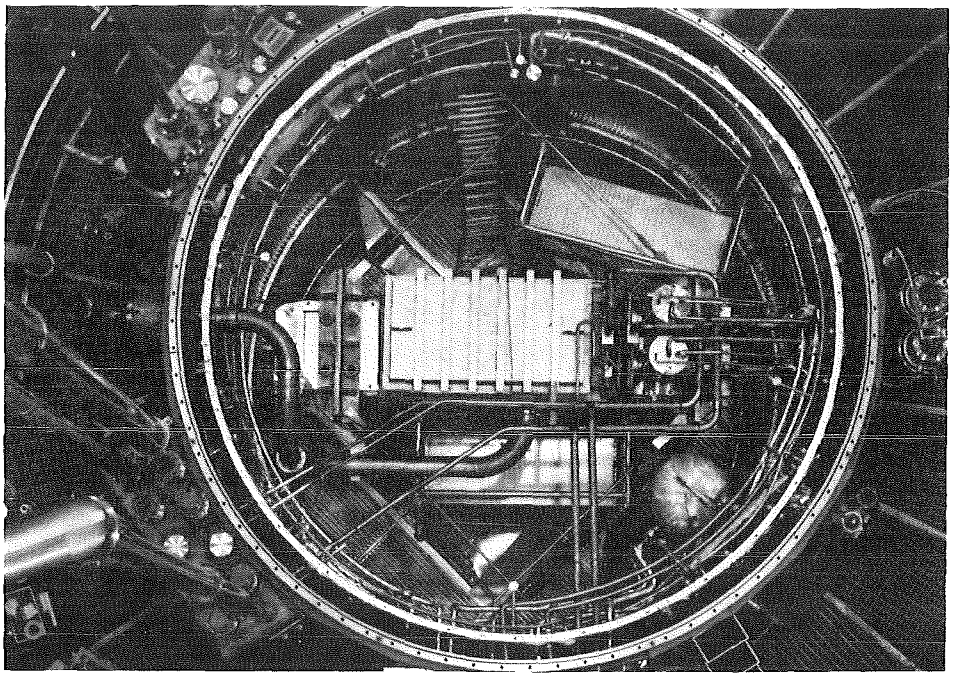

Fig. 4: The installed coil before closing the lid

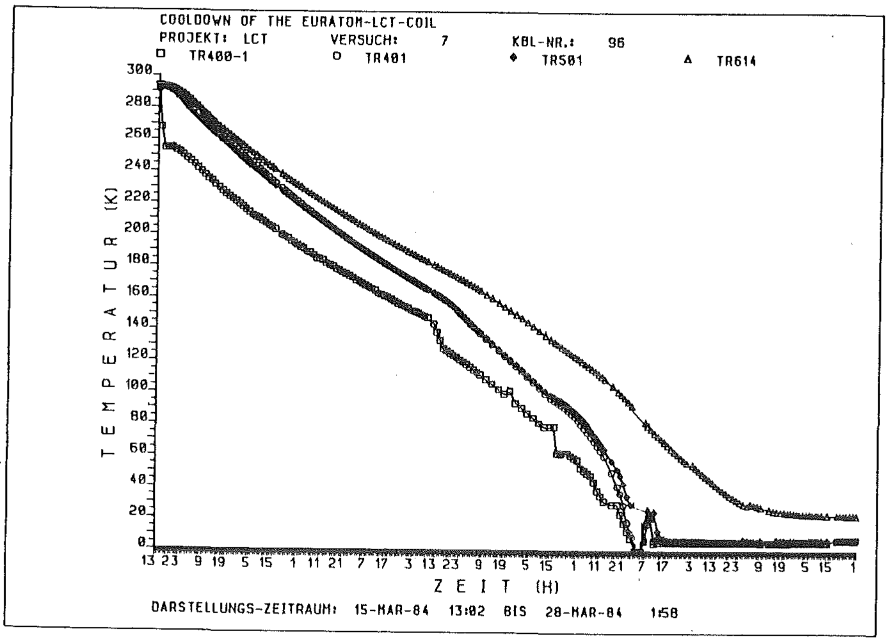

Fig. 5: Cooldown temperatures decrease at winding inlet, -outlet and different location at the coil case

2.3 Installation of Euratom-LCT-coil and cooldown

In a period of 12 weeks the Euratom-LCT-coil was installed in the TOSKA facility (Fig. 4). The main work was

- Mounting of the coil into its mechanical support
- Installation of the current leads and the superconducting bus
- Piping for the cryogenic supply, safety and measuring technique
- Wiring of instrumentation leads and operation checks
- Extended leak checks and high voltage insulation tests.

The cooldown started March 15, 1984. The coil could be cooled down successfully in 8.5 days (Fig. 5). First operation with current was performed end of March 1984. On April 1 the coil was operated with the design current for operation in TOSKA of 10 kA without any problems. Fast safety discharges (τ = 7.5 s) so far were performed up to currents of 6 kA with very small losses and small disturbance of the cryogenic systems. Discharge voltage was 1.3 kV at 6 kA.

Next steps are the extension of safety discharges up to full current of 10 kA and the performance of stability tests. After that the coil will be warmed up and will be prepared for shipping to ORNL.

THE LABORATORY TORUS COIL EXPERIMENT 'TESPE'

Construction of all six coils has been completed including mounting of the feedthroughs for measuring leads and current. Leak tests at room temperature proved tightness of all coils.

Three of the coils were installed in the test facility for a symmetrical 3-coil test. The design current of 7 kA was already reached in the first run without premature quench or any other disturbance.

Electromagnetic stability of the coils was tested when quenches were triggered artificially by heat pulses. Both quench energy and normal zone propagation were investigated in dependence of magnet current. The propagation corresponds to the calculated values with velocities between 0.05 and 2.2 m/s for currents between 3.0 and 6.5 kA, respectively. Unlimited stability was found below 2000 A. The quench energy shows the expected dependence on magnet current, its absolute values are higher than predicted by theory. This will be subject of further evaluations. Some 50 fast discharges of the stored magnet energy (3.5 MJ for 3 coils) into an external resistor have been performed. At a discharge voltage of 700 V the transfer of the energy was nearly 100 % up to 4.0 kA and still 85 % at 7.0 kA.

Deformation of the coils during charging has been measured in two axes and found to be in good agreement with results of finite element calculations. The signals of strain gages mounted on coil casing and superconducting winding were registrated and the evaluation is being performed in detail now.

During occupation of the cryogenic system by the TOSKA/LCT tests, the other three coils are under installation and the complete 6-coil tests are foreseen for summer of this year. Then the preparation for safety experiments will start.

Publications:	Staff:
17650	G. Aupelt
17685	H. Bayer
17687	M. Beckenbach
17984	DI. Drolshammer
18221	P. Duelli
18224	Ing. S. Förster
V 18694	Ing. G. Friesinger
V 18944	DP. W. Geiger
V 19493	A. Grünhagen
V 19494	DI. W. Heep
	Ing. R. Heil
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	H. Kiesel
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	Ing. A. Naschwitz
	DI. G. Nöther
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	U. Padligur
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	Prof. Dr. Yan Luguang

M 3 DEVELOPMENT OF HIGH FIELD COMPOSITE CONDUCTORS

In view of the elaboration of A15 conductor development for NET-coils, the experimental work was concentrated on the composite optimization by ternary alloying and problems of the basic conductor including stabilizer and reinforcement.

1. Optimization of Nb_3Sn composite wires by alloying

The possibility of further improving the current carrying capacity of Nb_3Sn superconducting wires by adding various elements has been investigated extensively. For this purpose, 19 core Nb_3Sn wires have been fabricated in the laboratory using the bronze technique and adding Ta, Ti to the Nb-core or Ni to the core and Zn to the Cu-Sn matrix. Measurements of J_c up to 23 T have shown that J_c of all these wires was very similar. The nature of the additives is thus of secondary importance, the main reason for the increase of J_c with respect to unalloyed Nb_3Sn wires being the perturbance of the completely ordered Nb_3Sn lattice by the introduction of 1 to 3 at% of another element, e.g. Ti, Ta or Ni. These perturbations increase the electrical resistivity of the A15 filaments and thus their upper critical magnetic field, leading to the observed increase of J_c at a given field B_o.

This result is technically relevant since it is now no longer necessary to search for other additives to Nb_3Sn, their effect being expected to be the same as for the above mentioned ones. The necessary condition for this statement, the detection of perfect ordering in Nb_3Sn, has recently been performed for the first time in our laboratory.

2. Conductor reinforcement

A further aspect of interest is the effect of the mechanical reinforcement on J_c of Nb_3Sn conductors. Different tests with stainless steel or Inconel reinforcement have shown that these materials lead to an enhanced precompression of the Nb_3Sn filaments and thus to a considerable reduction of J_c at fields above 12 T. The search for reinforcing materials having a smaller linear thermal contraction than steel or Inconel but still a high Young's modulus, has led to several possible solutions. One of them is the combination of steel and molybdenum: a Mo rod was inserted into a stainless steel tube of 20 mm Ø and drawn down to 1.3 mm Ø to lengthes of several hundreds of meters, thus proving industrial feasibility. It was found that the internal reinforcement of Nb_3Sn conductors by incorporation of a steel/Mo composite tube (ratio 1:2) yields the same critical current as for the unreinforced conductor.

This means that the degree of precompression of Nb_3Sn conductors can be controlled by an appropriate choice of the steel:Mo ratio. A precompression of $\varepsilon_m \sim 0.3$ % would be desirable, since it would provide the necessary mechanical and J_c reserve: even at full load the Nb_3Sn filaments in a fusion magnet should never be set under tensile stress or above the maximum of J_c of $\varepsilon = \varepsilon_m$. A further advantage of the presently proposed steel + Mo combination is the high Young's modulus, which was measured to E = 270 GPa compared with 220 GPa for stainless steel or 170 GPa for Incoloy 903.

The new steel + Mo combination allows to reduce the volume fraction of the reinforcing structure by $\sim$ 25 % in favor of the stabilizing material, leading to a higher safety margin for operation.

The experimental device for the measurement of the strain behavior of reinforced Nb_3Sn conductors has been extended for larger currents. A strain rig with the characteristics F = 10 kN up to I = 3000 A for the use at B = 14 T has been completed, allowing to measure conductors up to 40 mm² cross section.

3. Conductor stabilization

For a 15 T insert winding to the HOMER test magnet a Nb_3Sn composite conductor is planned with an aluminium stabilizer to make use of the low magneto-resistance of aluminium. The composite conductor will be soldered together from a prereacted Nb_3Sn flat cable and a copper coated aluminium tape. The aluminium was coated with copper via a drawing process to get a better bond after soldering. At first conductor lenghts of 10 m on 170 mm dia. spools were heat treated and the mechanical handling studied, i.e. straigthening, bending and soldering. The conductor did not suffer damaging in the range of permitted strain of smaller than 0.6 %. Tests in one layer coils demonstrated a sufficient cryogenic stability in the field and current regime strived for, due to the copper coated aluminium. However, in low fields at high current densities the conductor was not fully cryogenically stable and mechanical stresses of more than 160 N/mm^2 at the Nb_3Sn bronze cross section led to premature normal transitions. A phenomenon which needs further clarification. The manufacture of this composite conductor in a sufficient length (800 m) is the next technical goal, under development with VAC.

4. HOMER tests

So far the HOMER test facility is running at a field of 10 T with an operation temperature of 1.8 K. The Nb_3Sn conductor development work described above will soon lead to composite conductors for 12 T and 15 T insert coils.

Investigations at 4.2 K and 1.8 K of test windings with the internally steel strengthened conductor have shown that the gain in current density with decreasing temperature does not correspond to the values expected from literature. From measurements of Nb_3Sn without any reinforcement a gain of 35 % at 10 T could be expected, but in reality only a gain of 23 % was found. The reasons for this discrepancy have not been clarified yet.

Publications:

17079
17080
17359
17782
18221
18795

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Ing. W. Barth
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R. Berggötz
Ing. N. Brünner
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M 4 SUPERCONDUCTING POLOIDAL FIELD COILS

Activities on the development of superconducting Poloidal Field (PF) coils were initiated beginning of 1983. They are conducted within a joint programme of KfK and CEA/France. The aim of the programme is the demonstration of the PF coil technology suited for NET and for future fusion reactors. Intermediate goals are the construction and test of prototype coil(s) in the environment of an intermediate size tokamak (TORE SUPRA, ASDEX-UPGRADE). First steps initiated in this period were design and test of the conductor and other components. A second step being presently prepared is the design, construction and test of a model coil in laboratory scale.

To define the requirements an assessment of the relevant parameters of near and medium term tokamaks was made (TORE SUPRA, ASDEX-UPGRADE, INTOR/NET, ALCATOR-DCT, TFCX). On this basis preliminary conductor and coil design studies were carried out and first verification tests were performed /V 19478/.

A PF coil superconductor has to withstand different severe thermal loads. Most stringent are fast plasma current rise and disruption leading to high transient loads and, on the other hand, plasma position control leading to relatively small but quasi-stationary heat loads. With properly designed cooling channel geometry bath cooled conductors would perhaps be most suitable with respect to this aspect. But forced flow internally cooled conductors give the advantages of better electrical and mechanical integrity. Therefore, the potential of internally cooled conductors was investigated more deeply and designs with a double cryogenic system with two kinds of helium were developed. Stagnant subcooled helium is used for stabilization (high transient heat transfer) and forced flow helium for cooling (heat removal from the coil), respectively. First experiments were performed to verify the feasibility of this concept.

To reduce ac losses a PF conductor consists of a cable with a few 100 strands, depending on operational current. The strands are NbTi/Cu/CuNi mixed matrix conductors. Different strand geometries were developed on the basis of ac loss, stability and manufacturing aspects together with industry. Sample conductors are under fabrication. Different cable geometries and insulation designs are still under discussion.

Due to the pulsed nature of PF coils the question of ac losses also in the structural material and fatigue properties become of importance. Mainly three materials were regarded as candidate materials: austenitic stainless steel (SS), fibreglass reinforced plastics (GRP), carbon fibre reinforced plastics (CRP). According to the measurements performed GRP in most cases has better characteristics than CRP and, in addition, is compatible with stainless steel and the superconductor, because the thermal expansion coefficients can be matched and GRP can serve simultaneously as the insulating material. SS, on the other hand, has the best structural properties but must be subdivided to reduce ac losses so that finally a combination of SS and GRP should be used in a PF coil mechanical structure. Preliminary designs are presently being made for both, a bath cooled and forced flow cooled version of an Equilibrium Field (EF) coil to identify problem areas and to get a better comparison between these two options.

All activities mentioned above will be continued with special emphasis on conductor sample fabrication and testing (ac losses, stability). Additional experiments are being prepared for the verification test of the stabilization and cooling idea of the internally cooled conductor. Design considerations of EF coils and a respective model coil will be continued for the two different cooling modes, with the goal to finally select one design solution.

Publications:
V 19478

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B 1 BLANKET DESIGN STUDIES

The blanket studies include feasibility evaluations of blanket design concepts, basic feasibility studies and as a result of these more general investigations a proposal for a NET reference blanket. KfK concentrates on helium cooled ceramic blankets. The studies are coordinated with the CEA efforts by a common working group in periodic meetings.

First a breeding ball blanket concept was investigated which exhibits the following features:

- The blanket consists of a mixture of Li-ceramic breeder spheres with multiplier spheres, both of about 30 mm diameter.
- The balls are filled in poloidally arranged tubes of about 300 mm diameter and are cooled by boiling water at 100 atm pressure (310°C)
- Tritium extraction is made by continuous or batchwise removal of balls and ball heating up to 700°C, a temperature at which tritium permeates through the clad.
- Niobium getter material is added to the ceramic in the balls to prevent oxidization of the tritium.

Main advantages of this concept are the totally canned breeder units with external tritium extraction and the flexibility in changing the blanket composition. The principle feasibility and the possibility of its implementation in the INTOR geometry were shown. In spite of this the study was discontinued mainly since the He-pressure build-up in the balls makes the application of the concept for power reactors questionable. /V 19567/

Then the studies have been concentrated on helium cooled ceramic blankets. The geometrical constraints, the maximum temperatures, heat fluxes and neutron fluences are those given by the NET team. The objective is to design a blanket as reliable as possible with a minimum of development work and a tritium breeding ratio as high as possible.

The first calculations were for a blanket with the following characteristics /18959/.

1. Separate helium cooling system for first wall (FW) and lead multiplier at very low temperatures (helium pressure 50 bar, helium maximum temperature 90°C).
2. Use of Li_2SiO_3 as ceramic breeding material in form of 2 mm diameter pebbles contained in perforated stainless steel tubes. High temperature helium (helium pressure 50 bar, inlet temperature = 400°C, maximum outlet temperature = 450°C) is used to cool the blanket and to carry away the tritium. An oxidizing atmosphere in helium ensures that tritium is available in form of oxide and cannot permeate through the steel walls. The helium pressure tubes are arranged in toroidal direction. The presence of a moderator ($ZrH_{1.7}$) in the back of the blanket improves the tritium breeding ratio (TBR) and the power distribution in radial direction.

The resulting local TBR is quite high (1.34 for natural lithium, 1.44 for 30% ^{6}Li enrichment). However the one-dimensional calculations did not take account of the not complete coverage of the torus surface and of the fact that the inboard section of the torus must be considerably thinner.

The use of the cooling helium in the function of tritium purge flow simplifies the blanket design. However subsequent calculations have shown that a separate helium purge flow is required. Indeed, even if the oxidizing atmosphere avoids tritium losses, the helium losses from the primary circuit are probably of the order of 0.025 - 0.05%/d. which means, for a helium slipstream flow in the purification plant of 0.1%, a tritium loss 25 to 50 times higher of the maximum allowable (10 curie/d). The helium atmosphere in the purge system must be oxidizing to avoid tritium losses in the main helium circuit. The resulting equilibrium partial pressure of T_2O (19 Pascal) probably prevents the use of Li_2O as breeder material, due to large increases of tritium inventory in the ceramic because of T_2O solubility and surface adsobtion. However the behaviour of litium silicates and especially of the aluminates should be considerably better in this respect.

Recently a new design has been investigated. The lead multiplier is still integral with the first wall, however the helium cooling tubes and the pressure tubes containing the ceramic breeder are running in poloidal direction. Four rows of tubes are used in the outboard section and two in the inboard. Due to the considerable length of the tubes (6 meters), the helium pressure has to be increased to 80 bars to limit the helium

pumping power for the blanket to 3% of the heat output. The helium enters the FW and lead cooling system at 80^{o}C, then enters the ceramic cooling system at 180^{o}C coming out with a temperature of 380^{o}C. The condition that the lead should be kept below the melting point to avoid liquid metal embrittlement in the steel structural material has as a consequence that one third of the ceramic material is at temperatures below 320^{o}C. To avoid excessive tritium inventory in the blanket material, helium has to be circulated in the breeder pressure tubes alternatively in two opposite directions.
To avoid the temperature limitations due to the lead, we are now studying the use of beryllium as multiplier. In front of the blanket the beryllium is placed together with the breeding material: this is contained in form of sphere-pac in a calandria structure. The beryllium is canned and placed in annular regions between the helium cooling tubes and the sphere-pac material, and in steel clad-rods. The volume ratio of beryllium to ceramic being 80:20. In the back of the blanket beryllium is partly replaced with $ZrH_{1.7}$. For this arrangement radial, poloidal and toroidal solutions are being investigated. The tritium breeding ratio is slightly lower than that for the lead multiplier version and depends strongly on the allowable beryllium density.

Design work for the blanket studies is done at KfK and by industry by means of a KfK-contract. A contribution concerning the He-header system is performed by KFA. Within the rest months a feasibility study of the main cooling system and the tritium purge system will give complementary information for the blanket design.

Publications:	Staff:
18959	Prof. M. Dalle Donne
V19567	Dr. S. Dorner
	DP. M. Küchle
	Dr. S. Taczanowski

B2 DEVELOPMENT OF COMPUTATIONAL TOOLS FOR NEUTRONICS

During the HIBALL study /18239/ the $^{7}Li(n,n't)$ cross section was evaluated at KfK on the basis of then available "new" experiments. The result of this evaluation is displayed in Fig. 6. In the 6 MeV region our evaluation is a bit higher than the new data but well within the experimental uncertainty. In the meantime the Harwell data have been revised and fit better to our evaluation. The accuracy of this evaluation is about 7 %. This may be regarded as sufficient especially because in most of the recent blanket designs the $^{7}Li(n,n't)$ process contributes only a few percent to the total breeding ratio. Only in the case of a Li or Li_2O blanket is a higher accuracy required.

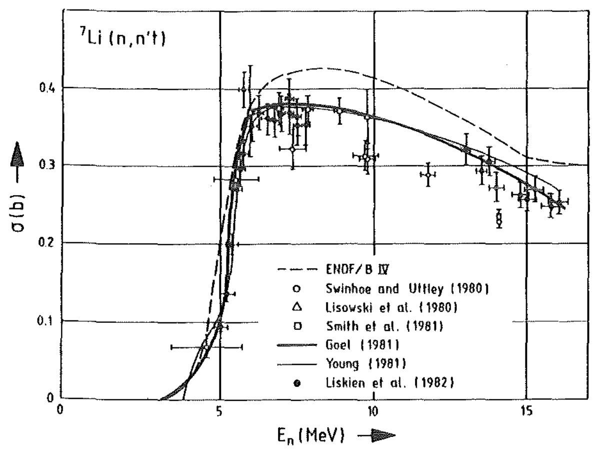

Fig. 6: $^{7}Li(n,n't)$ cross section

In some of the blanket designs the use of ZrH_x is proposed /18959/. KERMA-factors for both these isotopes are missing in the used data base. To make up for this deficiency we have implemented the MACK-IV code at Karlsruhe and will use it to generate KERMA-factors for the required isotopes.

At the last B2 meeting (June 1983) a participation in the precompound benchmark exercise was encouraged. For this purpose we have modified our version of the HAUSER*4 code /18137/ to enable the simultaneous calculation of (n,n') and (n,2n) neutron emission spectra. The precompound contribution is calculated using the geometry dependent hybrid model. As shown in Fig. 7, experimental data for ^{93}Nb are well reproduced with this code system without resorting to a fitting procedure.

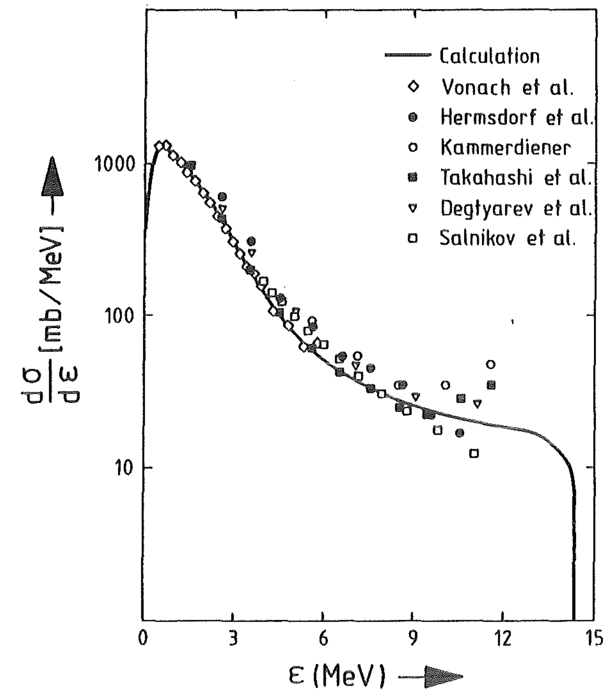

Fig. 7: Neutron emission spectrum for ^{93}Nb

In the neutron multiplication experiment on Pb by Takahashi et al. more neutrons were observed than predicted by ENDF/B-IV data. Another experiment by Aleksandrov et al. leads to the same conclusion. These observations are in conflict with the (n,2n) cross section measurement by Frehaut et al. A critical analysis of the Takahashi experiment reveals that in this experiment 13 % of the Pb(n,2n) neutrons should remain above the (n,2n) threshold. This is kinematically highly improbable. Thus the experiment should be reinvestigated.

The work on generating a data base for the investigation of a suitable group structure for fusion studies and an improved handling of scattering in neutron transport calculations has been started.

Neutronic calculations for a helium-cooled, ceramic and liquid breeder material are being performed.

Publications:	Staff:
18239	I. Broeders
18137	U. Fischer
	B. Goel
	H. Jahn
	B. Krieg
	H. Küsters
	M. Segev (Guest Scientist)
	E. Stein
	E. Wiegner

B 6 CORROSION OF STRUCTURAL MATERIALS IN FLOWING $Li_{17}Pb_{83}$

In order to study the corrosion behaviour and mass transfer effects of vanadium alloys (V-3Ti-1Si) and austenitic or ferritic stainless steels in the flowing lithium - lead eutectic alloy a small loop with electromagnetic pump and flow meter was designed. The loop, which is shown in Fig. 8, is integrated to a dry argon glove box. The components of the loop, the test section with air cooler and specimen bearer, the magnetic trap, the pump and flow meter and the heater, are already constructed. They will be assembled by welding as soon as the glove box will be available. The piping is made of the stainless steel X10 CrNiMoTi 18 10. The

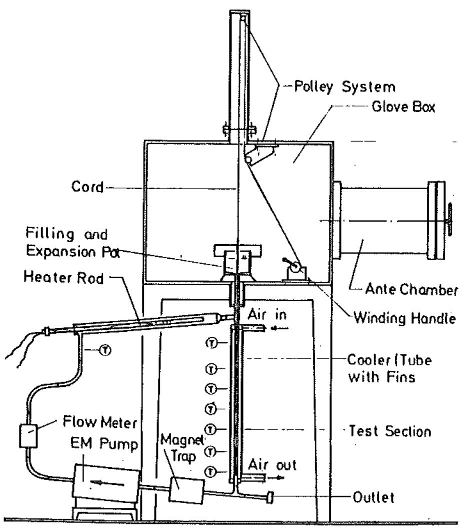

Fig. 8: Dry argon glove box with the loop below the bottom of it

glove box will be provided with the filling and expansion pot of the loop. At this position, the loop can be loaded or unloaded with the corrosion specimens. Also samples of the alloy $Li_{17}Pb_{83}$ can be taken for chemical analyses. The pot serves also for filling the loop with fresh portions of the molten alloy, while for draining there is an outlet at the bottom of the piping. The temperature will reach a maximum value of 450 ^{o}C, the temperature of the cold leg will be at 400 ^{o}C. The capacity of the pump is 1.4 m^3/h lead at 2 bars pressure drop.

Some samples of the alloy $Li_{17}Pb_{83}$ have been already molten starting from the pure components. For this procedure the box atmosphere can be trapped for nitrogen by an additional nitrogen purifier. The alloy composition was controlled by the measurements of the melting point as well as by metallography and by chemical analysis.

The Metallgesellschaft AG, Frankfurt, Germany, is asked for supply of the lithium-lead alloy. They are starting tests of the fabrication of the alloy in the 10 kg order. Since they are producing lithium and lead, they are a potential source for delivery of it.

The first runs in the $Li_{17}Pb_{83}$ loop will be used for studies of the liquid metal chemistry. Mainly the Li / Pb ratio and the contents of non-metallic impurities will be analyzed. If the results of these chemistry tests will be satisfying, corrosion tests of the V-3Ti-1Si alloy, the reference steel AISI 316 and the martensitic steel 1.4914 will be started by September 1984. They should be extended to several thousand hours.

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B 9 TRITIUM EXTRACTION BASED ON THE USE OF SOLID GETTERS

Several methods were proposed to extract tritium from the liquid $Li_{17}Pb_{83}$ getter material. Task B9 will study the use of solid getters. The advantage of this method is its simplicity and a low tritium inventory in the blanket, while most of the breeded tritium is fixed in the getter.

The work for task B9 has started during the reported period. So far only preparatory work has been done. A number of facilities were modified or improved and some techniques developed.

$Li_{17}Pb_{83}$ has to be handled in an inert atmosphere without oxygen and nitrogen. A glove-box system, designed for sodium handling, is now equipped with a system for the removal of nitrogen. The box atmosphere contains now less than 10 vpm oxygen and nitrogen.

A technique was developed for the controlled loading of materials with tritium and hydrogen. For the measurement of tritium we have a Beckmann LS9000 liquid scintillation counter. Before the measurement tritium has to be converted to tritium-water. It is now possible to convert hydrogen (and tritium) concentrations as low as 10^{-5} vpm from an inertgas stream and to collect the formed water. The tritium analysis was improved, less than 1 pCi can now be measured in a sample.

The allowed total amount of tritium handled in our laboratory is limited. Therefore tritium will have to be diluted with hydrogen or deuterium in most of the experiments. This requires the H-D-T isotopic analysis. To enable this a gaschromatograph was equipped with a low temperature chamber for the separation column.

One major goal of task B9 will be the study of the solubilities of getter materials in liquid $Li_{17}Pb_{83}$. A Perkin Elmer ICP-6000 analyzer will be used to determine the concentration of getter materials in the alloy. Most of the interesting elements can be determined in concentrations down to 1 ppm.

During the next reporting period, the mentioned methods will be further improved and used for the investigations. The solubilities of getter materials and its compatibilities will be studied in batch-type experiments. These experiments will be performed in molybdenum beakers, because this metal is very stable against the liquid alloy. However the influence of impurities in the metals and the liquid alloy has to be considered. In parallel to these experiments methods for the extraction of tritium and hydrogen from $Li_{17}Pb_{83}$ and getter materials have to be developed. The study for the gettering of tritium from $Li_{17}Pb_{83}$ will start.

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B 11/B 12 FABRICATION AND CHARACTERIZATION OF CERAMIC BREEDER MATERIALS

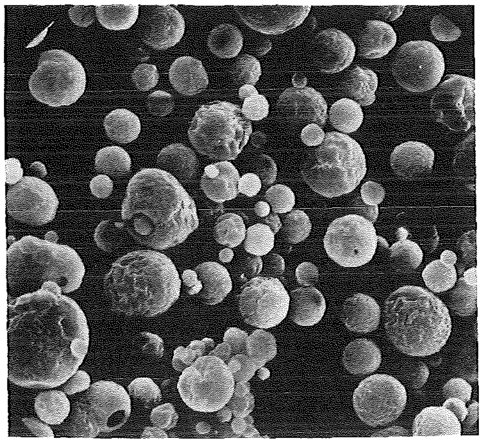
The preparation and fabrication are discussed of lithium containing ceramic materials, such as γ-$LiAlO_2$ and Li_2SiO_3, to be used as breeder materials for fusion reactors, with ceramic Li_2SiO_3 to be developed above all at KfK, whereas CEA concentrates on aluminates. The development started with the preparation of aluminate and silicate powders and pellets to obtain basic information about physico-chemical and mechanical properties, and to gain first experience in the behaviour of Li_2SiO_3 under neutron irradiation.

The methods of preparation have been developed under the aspect of a possible technological fabrication within the kg-range, i.e. production of Li_2SiO_3 powders by spray-drying methods and densification of those powders by granulation and sintering.

Preparation of Li_2SiO_3 Powders and Pellets

A suspension of silicon dioxide in an aqueous lithium hydroxide solution, which is suited for spray-drying, has been obtained by using amorphous silica (AEROSIL) produced on an industrial scale by DEGUSSA as a starting material. The stoichiometric mixture already reacts in the slurry stirred for about one hour by formation of the meta-silicate, which is partly dissolved in the aqueous phase or precipitated as a very fine powder in the form of $Li_2SiO_3 \cdot H_2O$. The powder obtained by spray-drying of the suspensions are spherical in shape (3-5 µm); most of the spheres, at least the bigger ones, are hollow (Fig.9-10). Chemical analyses of the powders shows that about 6 % Li_2CO_3 is present as an impurity, which reacts or is decomposed during the sintering process.

The spray-dried powders can be densified to about 92 % th.d. by calcination of the powders at about 600 °C and sintering of the pressed powders at a temperature of 1100 °C. Lower densities of sintered Li_2SiO_3 pellets have been obtained after granulation of the powders (250-400 µm in size) and calcination of the granules at higher temperatures; a densification of 85-65 % th.d. has been obtained at calcination temperatures of 750-900 °C. X-ray determination of the sintered pellets show a mono-phase Li_2SiO_3 structure, which is in agreement with the ASTM reference data.



(600 x)

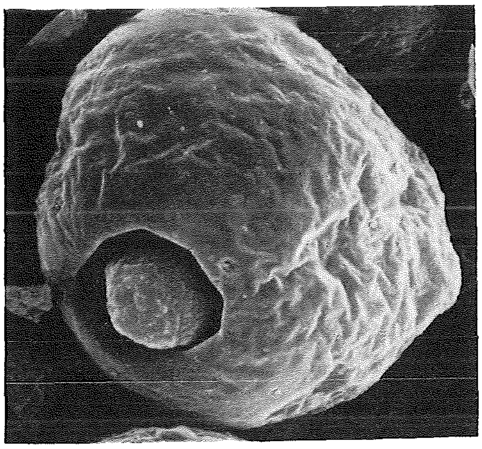


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Fig. 9-10 Electron micrograph of spray-dried Li_2SiO_3 powder

The metallic impurity contents of the powders are very low (Na = 60, K < 50, Cs = 20, Fe < 50, Cr < 50, Ti < 10, Al < 40, Pb < 500 ppm). Some slightly higher amounts, especially of iron and chromium, have been found after sieving and pressing of the powders.

Structure of the Li_2SiO_3 Pellets

Up to now, no typical specification has been given for the structure of the lithium metasilicate to be used as breeder material in forthcoming fusion reactors. Special specifications for crystallite sizes, thermal stability, tolerable water content, or chemical impurities are being discussed. But necessarily, the lithium ceramic pellets used as breeder material should have an open structure, i.e. a high open porosity, which favors the release of tritium produced by neutron irradiation, and a high thermal stability of the structure.

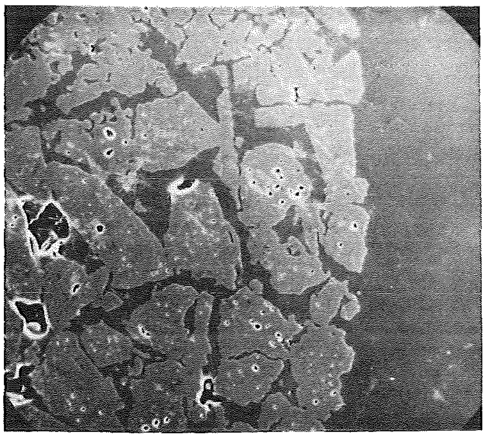


Fig. 11 Micrograph of a sintered Li_2SiO_3 sample showing the open structure (as polished, 60 x)

The open porosity of the metasilicate pellets prepared is in the range of 10-30 μm (Fig. 11 . The pellets show a high thermal stability; due to the sintering conditions exaggerated grain growth has been observed as shown in Fig.13-14. Use of binder materials in the preparation process or change of the sintering conditions leads to nearly the same density of the pellets, but the microstructure is quite different, and shows sometimes the structure of the powder.

Preparation of γ-$LiAlO_2$ Powders and Pellets

Metallic aluminium and lithium hydroxide have been used for the preparation of pure lithium aluminate powders by dissolving stoichiometric amounts of aluminium in an aqueous lithium hydroxide solution. The precipitating lithium dialuminate, $Li[Al_2(OH)_7]\cdot n\ H_2O$, remains in the hydroxide solution as a very finely dispersed powder . By spray-drying a stoichiometric mixture of lithium hydroxide and lithium dialuminate has been obtained from this slurry as a fine and homogeneously mixed powder (Fig.12). Calcination of these powders leads to the formation of a mixture of β- and α-$LiAlO_2$ at temperatures up to 400-600 ^{o}C, the transformation to γ-$LiAlO_2$ occurs at 900 ^{o}C within four hours without any sintering of the finely dispersed powder. Such powders can be densified to 80-85 % th.d. by pressing into pellets and sintering in air at temperatures up to 1250 ^{o}C. Higher densities in the range of 90 % th.d. have been obtained by sintering at 1500 ^{o}C. The products are mono-phase γ-$LiAlO_2$ as determined by x-ray diffractometry. The chemical purity depends mostly on the purity of the metallic aluminium, which is used in the process.

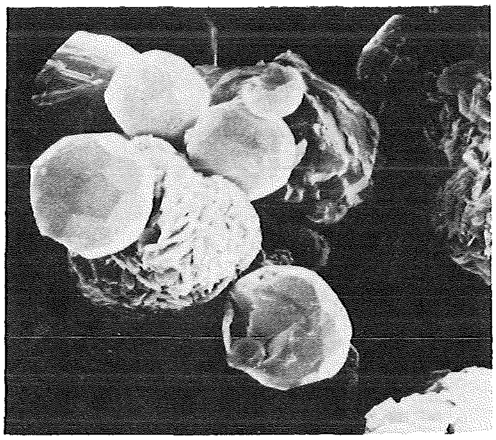
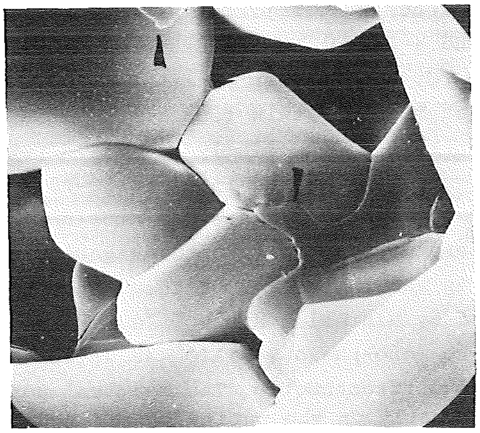


Fig. 12 Electron micrograph of spray-dried $LiOH$ / $Li[Al_2(OH)_7]\cdot n\ H_2O$ powder (3000 x)

5. First Irradiation Test of Li_2SiO_3 Pellets

In a first test lithium metasilicate pellets have been successfully irradiated for about one month in the FRJ-1 at Jülich at a low specific power and a temperature of approximately 200 ^{o}C to a burnup of about 0.8 %. The visual inspection of the irradiated pellets showed a change of colour from white to greyish-brown but no change in the geometry or break-up of the structure. From this we can conclude that the lithium metasilicate pellets are suited for further irradiation tests (Task B 15).



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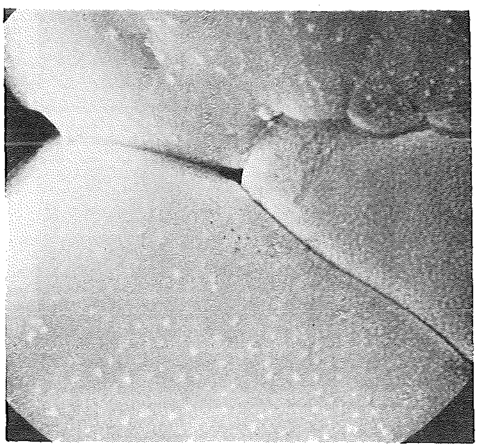

(15000 x)

Fig.13-14 Electron micrograph of a sintered Li_2SiO_3 sample showing exaggerated grain growth

Summary

Ceramic lithium metasilicate powders can be prepared easily from suspensions of amorphous silica (AEROSIL) in aqueous lithium hydroxide solution using spray-drying methods. This procedure is especially suited for technological fabrication, as no special efforts are necessary for further development. The densification of the powder leads to pelletized or granulated materials which are suited as breeder materials for forthcoming fusion reactors.

This process can also be developed for the preparation of lithium orthosilicate or other similar lithium containing compounds which are of interest for fusion technology.

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B 13 MEASUREMENT OF PHYSICAL, MECHANICAL AND CHEMICAL PROPERTIES OF CERAMIC BREEDER MATERIALS

A revised and extended version of an internal report was completed, which gives a literature review concerning physical and chemical properties of oxide breeder materials. The main importance is attached to the arguments for the operating temperature interval to be observed, to the knowledge on the chemical compatibility with cladding materials, and to considerations on the irradiation behaviour.

Constitution, thermodynamics and chemical analysis

Several breeder compounds (Li_2SiO_3, Li_4SiO_4 $LiAlO_2$ and Li_5AlO_4), each of them fabricated by different methods (a) solid state reaction b) precipitation with subsequent sintering), have been characterized by X-ray diffraction and thermal analysis. Their cell constants and transformation temperatures were compared mutually and with literature data. The melting temperature of $LiAlO_2$, measured by differential thermal analysis (DTA) as well as by direct observation, is 1760°C in inert gas (1 bar/Ar). DTA measurements of Li_4SiO_4 confirm the findings of other investigators, showing three distinct heat effects between 600 and 770°C.

Phase stability studies in the ternary system Li_2O-Al_2O_3-SiO_2 were started with the aim to improve the thermal stability of some phases by admixing minor amounts of other chemical constituents. The compounds Li_2SiO_3-$LiAlO_2$ and Li_4SiO_4-$LiAlO_2$ are stable, whereas the compound Li_4SiO_4-Li_5AlO_4 turned out to be unstable since new unknown phases have been formed. The cell constants of Li_4SiO_4 were indicated to change by admixture of Li_5AlO_4. The $\alpha \rightarrow \gamma$ transformation temperature of $LiAlO_2$ seemed to increase when Li_4SiO_4 was added.

Differential thermal analyses and thermal gravimetry (TG) measurements were started with Li_2SiO_3, Li_2O, γ-$LiAlO_2$ and Li_2ZrO_3 powder samples in order to investigate the absorption and desorption of water at temperatures up to 1000°C.

Sample materials were characterized by chemical analysis of the main impurities, particularly of the water or LiOH content.

Physical and mechanical properties

The thermal conductivity of γ-$LiAlO_2$ was measured using the laser-flash-diffusivity method. The results indicate a possible important role of the preparation process for the thermophysical properties. Fig. 15 shows values of the thermal conductivity of samples not prepared via Li_2CO_3 in comparison to data of samples with this preparation step.

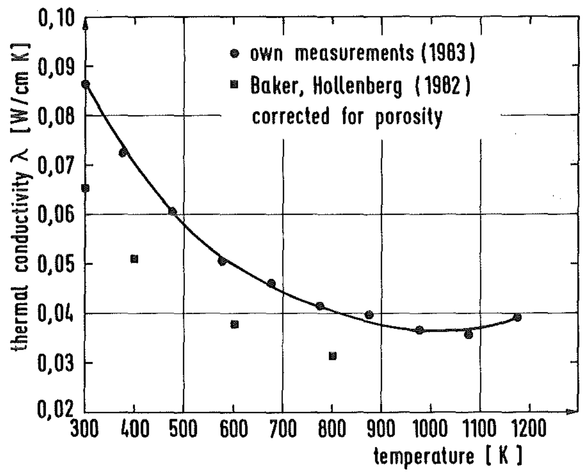

Fig. 15: Thermal conductivity of $LiAlO_2$ (without Li_2CO_3 ●, with Li_2CO_3 ■ as intermediate product)

Concerning the measurement of mechanical properties, IMF (the Institute for Materials Research) at KfK has been preparing to measure Young's modulus, thermoshock resistance and compressive creep on cylindrical pellets. Compressive strength measurements are also possible, if wanted. Young's modulus is determined by sound velocity measurement, and thermal shock resistance by dipping the pellets into a liquid metal bath of variable temperature. In this case, the property value to be measured is the critical temperature difference for crack formation.

Up to now, the main problem has been the insufficient supply with test samples. Thus, it was not yet possible to start thermal shock tests, which require a rather large number of pellets. Young's modulus could only be measured on a few Li_2SiO_3 samples of 95% TD and amounted to about $8x10^4$ MPa at room temperature. Compressive creep measurements were restricted to the starting series of some $LiAlO_2$ samples of 92% TD, because there were troubles with the equipment already used for various other investigations before. Under a compression stress of 20 MPa and at temperatures from 800 to 1100^{o}C the $LiAlO_2$ samples showed a temperature-dependent incubation period followed by rather fast deformation which perhaps is due to a microcrack growth process. The rate of sample length reduction was about $2x10^{-5}$/h at 800^{o}C, and its temperature dependence corresponded to an activation energy of about 210 kJ/mole.

In the next reporting period, a new compressive creep test machine will be installed. We expect a sufficient supply with Li_2SiO_3 samples manufactured at KfK. Thus, the main test series on Li_2SiO_3 can be started.

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B 14 COMPATIBILITY OF CERAMIC BREEDER MATERIALS WITH CLADDING MATERIALS

The results of a literature review concerning the chemical compatibility of oxide breeder materials with cladding of CrNi stainless steels are summarized in Fig. 16 .

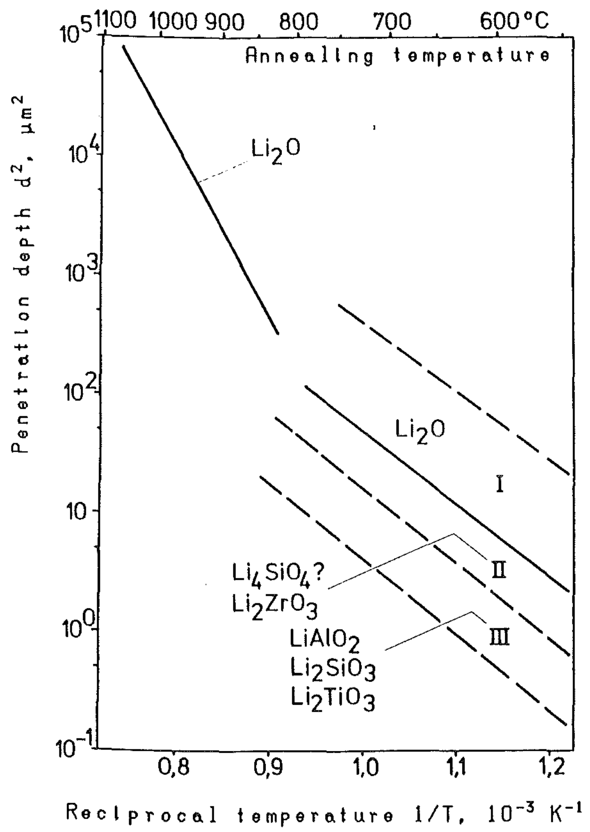

Fig. 16: Chemical reaction of oxide breeder materials with stainless steels in 100 h.

The penetration depth of the cladding attack is technically relevant only above about 10^2 μm^2. In this respect, the examination of Li_2O (with its scattering range "I" in Fig.16) is of major importance.

At IMF (Institute for Materials Research) of KfK first annealing experiments were made on CrNi stainless steel capsules filled with pressed powders of Li_2O, LiOH and Li_2CO_3. LiOH and Li_2CO_3 were used to investigate the influence of major impurities. The annealing tests are conducted at 500 to 700°C for 100 to 1000 h. Chemical reactions are demonstrated by metallographic examination of capsule cross-sections.

The penetration depth with commercial Li_2O was found about the upper bound of the scattering range "I" in Fig.16. The cladding attack was weaker with Li_2CO_3, and much deeper with LiOH, at least above 600°C. It was concluded, that the water impurity content of Li_2O is probably responsible for the cladding attack, and that LiOH is the reactive agent.

At present, compatibility tests are being conducted on Li_2O, $LiAlO_2$, Li_2SiO_3, and Li_2ZrO_3 with controlled water or LiOH additions. Preliminary results indicate that (a) carefully dried samples (2 h at 900°C) did not cause any cladding attack, (b) the water impurity was of less influence for $LiAlO_2$, Li_2SiO_3, and Li_2ZrO_3 than for Li_2O.

In the next reporting period these tests will be evaluated and maybe supplemented. The further procedure depends on the supply of the reference cladding material ss 316 L.

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B 15 IRRADIATION TESTING OF CERAMIC BREEDER MATERIALS

In the frame of the European Technology Program the investigation of fusion ceramic breeder materials was initiated. The activities in this field are co-ordinated with CEA, France. According to an agreement, KfK concentrates on the investigation of lithium silicate and CEA on that of lithium aluminate. The main goal is the study of irradiation effects in the material (dimensional stability, changes of pore structure, irradiation damage etc.) and of the tritium behaviour. Since mid of 1983 the co-operation with CEA has been established by four meetings. The irradiation tests will be carried out in the OSIRIS reactor at Saclay. A contract comprehends irradiation tests of lithium silicate in modified COLIBRI test rigs. The first of these rigs was designed. It will contain 33 pellet columns of 45 mm height within a stainless steel cladding, and six pellet columns of 90 mm height without cladding. The nominal diameter of the pellets is 4.94 mm. Further irradiation parameters are:

- specific power
  20 ... 50 W/cm³
- temperature levels
  500 and 700 °C
- tritium production rate
  2.5 ... 6.5×10^{13} atoms/cm³s
- sample density
  1.63 g/cm³ ≙ 65 % th.d. and
  2.13 g/cm³ ≙ 85 % th.d.
- moisture content of ceramic
  < 5 ppm and > 1000 ppm

The fabrication of the samples has been started. A first irradiation test of lithium silicate pellets was carried out in the MERLIN reactor at Juelich. The first irradiation in the OSIRIS reactor is planned for September 1984. During this irradiation, sample temperatures will be monitored. They can be adjusted by a controllable gas mixture in a gas gap. The neutron dose will be measured. Gas purging is not provided with this experiment. The post-irradiation examination of the breeding ceramic covers analysis of the irradiation behaviour and the tritium release kinetics. Experimental equipment for tritium measurement was designed. It will be installed in the hot cells at KfK. Other equipments, i. e. mercury porosimeter and others, are available.

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B 16 TRITIUM RECOVERY FROM CERAMIC BREEDER MATERIALS

In a similar way as for B15 the activities under B16 are co-ordinated with CEA, France. The main aim of this activity is the investigation of the tritium release from fusion breeder materials. For this purpose ceramic samples of lithium silicates will be irradiated in the CHOUCA rig in the SILOE reactor at Grenoble. This facility allows continuous monitoring of tritium release at various temperature levels.

The irradiation of the first test rig is planned for the end of 1984. It will contain six samples of 8 mm diameter and 80 mm height in quartz tubes. The materials to be tested are lithium metasilicate and lithium orthosilicate in the form of pellets and spheres of about 85 % density. The sample temperatures will be in the range between 400 and 700 °C. The actual values can be adjusted by a gas gap and controlled by electrical heating. The specific power will be around 5 W/cm³ leading to a tritium production rate of about 7×10^{12} atoms/cm³s. The flow of the purge gas helium will amount to 0.25 to 1.3 l/h.

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MAT 1 POST IRRADIATION TESTING OF STAINLESS STEEL

During the past half year two main lines were followed at KfK in Fusion Technology Program MAT-1:

1. Development of a Test Specimen and
2. Development of Creep Equations for Materials Subjected to Variable Loads and/or Temperatures.

ad 1) The choice of appropriate specimens for MAT-1 fatigue experiments requires careful considerations. Because fatigue experiments are to be conducted also under irradiation the size of the specimen is restricted by the transmission power of the particle beam. On the other side fatigue deformation of thin specimens offers the problem of their mechanical stability (buckling). Hollow cylindrical specimens seem most appropriate to meet both the requirements: transparency and stability. In the present version hollow specimens were machined (see Fig. 17) from so called GRIM specimens (made from SS AISI 304), which in the past were used for LCF tests in different laboratories. Preserving the mechanical stability, the wall thickness in the center part could be reduced to 0.4 mm.

In Table 1 results of LCF test performed on GRIM-Hollow (GRIM/H) - specimens are summarized. Though high total strain amplitudes were used the reproducibility of the number of cycles to failure was as good as that known from the usual (solid) GRIM/S specimens. At present, 750^{o} C is the highest test temperature at which successful LCF test with GRIM/H specimens were performed. Because the outer dimensions of the GRIM/H specimens are the same as that of GRIM/S, the former specimens can be also handled in the hot cells by remote operating. This specimens (internally pressurized) will be also used for biaxial fatigue tests. The wall thickness of 0.4 mm is transparent for dual beam irradiation to be conducted at the KfK.

The comparison of data from measurements conducted on samples with different geometry is one of the basic problems of fatigue. In this connection calculations of the stress distribution in different tests specimens are necessary. These calculations for GRIM/H specimens were recently started. In order to investigate the effect of sample geometry upon the lifetime a standard test programme was proposed at the Workshop on the Use of Nonstandard Miniaturized Specimen for Different Mechanical Tests, held at KfK in February 1984, the programme is on the way.

In next future experiments will start the aim of which is to investigate first the influence of non uniform azimuthal temperature distribution in the GRIM/H specimen caused by a particle beam upon the mechanical stability of these specimens during fatigue loading.

ad 2) Two main problems arise during the design of structures at high temperatures, when they are subjected to varying loads and temperatures. First, it is important to know what is the lifetime of the material creeping under non-stationary loading conditions. Second, the problem of describing inelastic deformation of the structures must also be considered. Compared with lifetime predictions, much less attention has been paid to the description of thermal creep under variable tensile loading conditions. The prediction of material behavior precludes the knowledge of the actual loading conditions as forces, temperature, environment, radiation "loads" etc. In practice, each of these "loads" can change in time and space leading to inhomogeneous distribution of such "loads". This inhomogeneity increases the complexity of interdependencies of different kinds of "loads". Moreover, material properties, which in many cases at stationary loading conditions can be considered as constant may change under complex loading. Hence with increasing complexity of loading the description of the loading conditions by means of a standard service cycle becomes dubious.

In a first step our investigations are concerned with thermal creep which constitutes only one part of the inelastic behavior of structural materials of fusion reactors. In the absence of radiation, tensile loads and/or temperature may lead to plastic strain.

A description of the thermal creep behavior and a method of lifetime prediction was elaborated for materials subjected to non-stationary tensile loading conditions. The calculations are based on HART's tensile test equation and on a phenomenological cavitation damage model. From this model the life fraction rule (LFR) is derived. Analytical expressions for the lifetimes are derived, which contain only stationary stress rupture data. The creep behavior of non-cavitating and ideally plastic materials is derived from the solution of the tensile test equation for the particular loading conditions considered. Cavitation damage is known to influence the creep behavior by reducing the load bearing capability. A constitutive equation containing the loading conditions as well as the damage function is also derived. From this equation the creep curves for cavitating materials are then constructed. The following loading conditions were considered:

i) creep at constant load F and temperature T; ii) creep at linearly increasing load and T=const.; iii) creep at constant load amplitude cycling and T=const; iv) creep at constant load and linearly increasing T; v) creep at constant load and temperature cycling and vi) creep at superimposed load and temperature cycling.

When thermal creep is the dominant source of strains, the results of these calculations, which are summarized in the Tables 2 and 3, can be directly used by designers. The symbols and notations are explained in Table 4. As a next step, the influence of hydrostatic pressure upon the lifetime and creep of materials subjected to different loading conditions will be examined by calculations.

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Fig. 17: Low Cycle Fatigue Test Specimen With 0.4 mm Wall Thickness

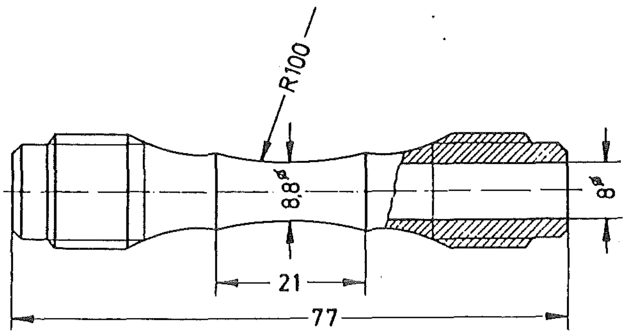

Table 1: Low Cycle Fatigue Results of Tube Specimens with 0.4 mm Wall Thickness

HOLD-TIME [min]	STRAIN RANGE [%]	STRAIN RATE [1/s]	TEMPERATURE [°C]	CYCLES TO FRACTURE
0	1.0	$3 \cdot 10^{-3}$	550	220
0	"	"	"	200
0	"	"	"	198
0	"	"	"	192
0	"	"	"	192
0	1.0	$3 \cdot 10^{-3}$	750	45
10	1.0	$3 \cdot 10^{-3}$	550	104
60	"	"	"	66
0	0.3	$3 \cdot 10^{-3}$	550	12948
0	0.4	"	"	4518
0	1.5	"	"	49

Table 2: The Lifetimes

Loading Procedure	Lifetime
$\langle F,T\rangle$	$t_f\langle F,T\rangle$
$\langle \dot F,T\rangle$	$t_f\langle \dot F,T\rangle = F_o/\dot F\{[(n+1)\dot F t_f\langle F_o,T\rangle/F_o+1]^{\frac{1}{n+1}}-1\}$
$\langle cF,T\rangle$	$t_f\langle cF,T\rangle = t_f\langle F_m,T\rangle \dfrac{(a-1)(n+1)}{a^{n+1}-1}$
$\langle F,\dot T\rangle$	$P\dot T t_f\langle F,T_o\rangle+1 = Y_f^{\,2}\exp(P(1-Y_f^{\,-1}))$
	$Y_f = 1 + \dot T t_f\langle F,\dot T\rangle/T_o$
$\langle F,cT\rangle$	$t_f\langle F,cT\rangle = \dfrac{t_f\langle F,T_m\rangle(b-1)}{b^2\exp(P(1-1/b))-1}$
$\langle cF,cT\rangle$	$\bar t_f\langle cF,cT\rangle = \dfrac{t_f\langle F,cT\rangle t_f\langle cF,T\rangle}{t_f\langle F,cT\rangle+t_f\langle cF,T\rangle}$

$$a = F_M/F_m$$

$$b = T_M/T_m$$

$$P = Q_f/RT_o$$

Table 3: Equations for creep curves of cavitating specimens

Loading Procedure	Equation
$\langle F,T\rangle$	$\Pi\langle F,T\rangle = [1-A_o\exp(ne)]^{-n}\exp(ne)$
$\langle \dot F,T\rangle$	$\Pi\langle \dot F,T\rangle = (1+[(n+1)/\beta][1-\exp(-ne)])^{n/(n+1)} \cdot \exp(ne)[1-A_o\exp(ne)]^{-n} = (1+x/\beta)^n\Pi\langle F,T\rangle$
$\langle cF,T\rangle$	$\Pi\langle cF,T\rangle = [1-A_o\exp(ne)]^{-n}H^n\exp(ne) = H^n\Pi\langle F,T\rangle$
$\langle F,\dot T\rangle$	$\Pi\langle F,\dot T\rangle = z^{\alpha}\exp(ne)[1-A_o\exp(nze)]^{-n}$
$\langle F,cT\rangle$	$\Pi\langle F,cT\rangle = u^{\alpha}\exp(ne)[1-A_o\exp(nue)]^{-n}$
$\langle cF,cT\rangle$	$\Pi\langle cF,cT\rangle = H^n u^{\alpha}\exp(ne)[1-A_o\exp(nue)]^{-n} = H^n\Pi\langle F,cT\rangle$

$$\Pi\langle p,q\rangle = \dot e_\phi\langle p,q\rangle/\dot e_o$$

$$\beta = F_o/\dot F\, t_f\langle s_o,T\rangle$$

$$H = \left(\frac{a}{(a-1)(n+1)}\,[1-a^{-(n+1)}]\right)^{1/n}$$

$$z = t_f\langle F,T_o\rangle/t_f\langle F,\dot T\rangle = f(\dot T)$$

$$u = t_f\langle F,T_m\rangle/t_f\langle F,cT\rangle = f(b)$$

$$\alpha = Q_c/Q_f$$

$$a = F_M/F_m\ ; \quad b = T_M/T_m$$

$$\dot e_o = B_o\sigma_o^{\,n}\exp(-Q_c/RT_o)$$

Table 4: Symbols and Notations

$a=\sigma_M/\sigma_m$	engineering stress amplitude ratio
A_o	initial damage value
$b=T_M/T_m$	temperature amplitude ratio
B_o	model constant in the strain rate equation
$\langle cF,cT\rangle$	CLAC, T-cycling-loading procedure
$\langle cF,T\rangle$	CLAC, T=cqnst.,-loading procedure
CLAC	constant load amplitude cycling
e	true strain
$\dot e_o=\dot e_o=B_o\sigma_o^{\,n}\exp(-Q_c/RT_o)$	strain rate at t=0 (minimum creep rate)
$\dot e_\phi\langle p,q\rangle$	true strain rate for cavitating specimens and given loading conditions
$F, F_o=F(t=0)$	load
F_M	maximum load amplitude
F_m	minimum load amplitude
$\langle F,cT\rangle$	F=const., T-cycling-loading procedure
$\langle \dot F,T\rangle$	$\dot F$,T=const. -loading procedure
$\langle F,\dot T\rangle$	F,$\dot T$=const., -loading procedure
$\dot F=dF/dt$	load rate
$\langle F,T\rangle$	F,T=const. -loading procedure
$H(a) = \left(\dfrac{a}{(a-1)(n+1)}\,[1-a^{-(n+1)}]\right)^{1/n}$	
$-1/n$	slope of the stress rupture line in the $\log\sigma/\log t_f$-diagram. n is equal to the reverse of the strain rate sensitivity m of the true stress s
$P=Q_f/RT_o$	
$\langle p,q\rangle$	general notation for stress and temperature loading procedure
Q_c	apparent activation energy for secondary creep
Q_f	apparent activation energy for failure
R	universal gas constant
$t_f\langle p,q\rangle$	lifetime value for given loading procedure
$T, T_o=T(t=0)$	temperature in K
T_M	maximum temperature amplitude
T_m	minimum temperature amplitude
$\dot T=dT/dt$	temperature rate

MAT 6 IRRADIATION BEHAVIOUR OF CERAMICS AND GRAPHITES

MAT 13 ELECTRICAL AND MECHANICAL PROPERTIES OF INSULATORS: SIMULATION OF RADIATION DAMAGE-SPECIAL CERAMICS

Ceramics may be used for various applications in fusion reactors such as electrical insulators, windows for RF heating, protection or structural materials for the first wall and canning materials for breeders. The aim of the MAT-6 task is to investigate the materials properties after neutron irradiation, whereas under the MAT-13 task accelerator irradiation with charged particles will be performed.

The investigations within the MAT-13 task are coordinated with CEA-Saclay and UKAEA-Harwell. For the MAT-6 task a joint proposal of CEA-Saclay, CEN/SCK-Mol and KfK is under preparation; the reactor best suited for this purpose has to be evaluated (Osiris, HFR, BR2). Several coordination and information meetings took place at Paris, Petten and Karlsruhe.

For different application areas a list of candidate ceramics has been established:

Insulators: alumina, spinels (e.g. $MgAl_2O_4$), yttrium-alumina garnet ($Y_3Al_5O_{12}$) magnesia, silicon nitride

Windows for RF heating: alumina, beryllium oxide, silicon nitride

Structural materials: silicon carbide and silicon carbide-silicon carbide composites, reaction bonded silicon carbide on graphite.

All materials to be tested are purchased from different manufacturers; no laboratory grade materials will be included. After pre-irradiation, characterization and testing the materials for reactor-or accelerator-irradiation will be selected. Several types of specimens which cover the different tests have been defined.

The change of a number of physical and mechanical properties due to irradiation damage will be investigated: dimensional stability, thermal diffusivity, thermal expansion, electrical resistivity, dielectric changes, Young's modulus, bending strength. The work at KfK is mostly related to silicon based ceramics; present indications show the critical role these high performance ceramics can play in future energy systems. The characterization of a dense high-purity hot-isostatic pressed silicon carbide, to be included in the irradiation programme, has started.

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T 1 FUEL CLEAN-UP SYSTEM

Impurity removal from the D-T fuel stream of a fusion reactor by metallic getter material is investigated in laboratory scale at KFA Jülich. Using the KFA-results, KfK Karlsruhe will develop a technical scale fuel clean-up unit and test it under D-T conditions.

KfK and KFA agreed on the beginning of the design work for the technical scale test facility in the second half-year 1984. KfK will limit its work to technical experiments.

A first data-set concerning the ZrAl-getter material St 101, SAES Milano, is transfered to KfK.

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T5 DEVELOPMENT OF TRITIUM DECONTAMINATION SYSTEMS

The experiments to fix hydrogen at C=C double bonds of unsaturated fatty acids have been terminated. It was shown that this hydrogenation process works in a satisfactory manner only in an inert gas atmosphere. In the presence of oxygen oxydation of the acids competed with hydrogenation. Since both reactions have approximately the same reaction rate this method cannot be employed to bind tritium of unsaturated acids in the presence of oxygen.

Therefore, the next step is to search for organic getters and efficient catalysts which are not affected by the presence of other gases. At first, a new test stand will be installed which includes a mass spectrometer for the analysis of hydrogen during the experiments. This construction will be finished in the middle of 1984.

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T 6 INDUSTRIAL DEVELOPMENT OF LARGE COMPONENTS FOR VACUUM SYSTEMS

The objective of this task consists in the development of large components for the vacuum systems of the NET reactor and in testing these components with tritium containing gas. The following activities are being performed in detail: specification of the components, inquiries to the industrial firms, elaboration of a feasibility study, acquisition of the prototype components, planing and construction of a testing facility for prototype testing, performance of the tests with tritium containing gas. The work is being performed in cooperation with the CEA. A joint working group was founded. The work is coordinated with the NET team in addition.

Pending the availability of NET specifications, the INTOR study serves as a basis for component design.

A list of requirements was compiled for component development as regards the torus and neutral particle injection vacuum systems. For the UHV range isolating devices of up to 2000 mm NW in size and fast shutter valves as well as high vacuum pumps with suction capacities between 5×10^4 l/s for helium and 4×10^6 l/s for deuterium are required. For the initial vacuum zone roughing pumps are necessary with suction capacities up to 1000 l/s. After consultation of the NET team the individual components were grouped by priority classes.

As tritium is used, no oils and lubricants are acceptable for the components. On account of the high radiation dose ($> 10^{10}$ rad) no organic substances can be admitted.

Regarding the all-metal isolating valves the maximum permissible leak rate in the seat of 10^{-8} mbar l/s at 1 bar helium differential pressure was prescribed. Such valves are at present on the market up to 400 mm NW sizes only. Consequently, compared with the status of the art, an increase by the factor 5 would have to be achieved. The number of approximately 50,000 cycles for the isolating valves which is required for cryo-pump regeneration can be attained today with much smaller components only, NW 50. The 400 NW fittings now attain 5000 cycles at the maximum. This means that an upgrading by the factor 10 would be necessary here.

A technical specification was compiled for the all-metal isolation valves. First contacts have been established with the manufactures of components.

Unlike in the INTOR concept, turbo-molecular pumps will be provided on the UHV side for evacuation of the plasma chamber. Suction lines per unit of up to 50,000 l/s are required for helium transport. This is the status of the art: maximum suction capacity for oil lubricated turbo-molecular pumps 5000 l/s and for the oil-free pumps (magnetic bearing) 500 l/s. Consequently, upscaling by the factor 100 will be required for the NET reactor.

A technical specification was also prepared for the turbo-molecular pumps. First contacts have been established with the manufacturing firms.

Within the technical specification the ambient conditions were defined for the place of installation of the components in the reactor building. The magnetic fluxes and the radiation doses were determined from the INTOR operating data. One problem consisted in the determination of the radiation dose because the fluctuation of the material released from the first wall due to sputtering has not yet been confirmed in quantitative terms. More attention must be paid to this problem in the future and this not because of carry-over of the radiation sources but mainly on account of the danger of quantitative material plate-out on the sealing surfaces of the fittings and the mobile pump elements.

By end of 1984 the technical specification will be completed for the fast shutter valves and for the roughing pumps. Planning will start of the testing facilities for prototypes running with tritium.

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S+E 1 RADIOACTIVE EFFLUENTS: BEHAVIOUR OF GASEOUS TRITIUM IN THE AIR, PLANT, SOIL SYSTEM

It is intended to study the diffusion of elemental hydrogen into the soil in order to determine the depth of penetration as well as reactions taking place in the soil. After delivery of a climatic chamber at the end of 1984 and its commissioning first tests will be conducted on the uptake of gaseous tritium and the incorporation of tritium in plants. Inactive preliminary tests will start in the mid of 1984.

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S+E 2 ACCIDENT ANALYSIS

In the framework of this task KfK/IRE is performing functional and qualitative analysis of specific subsystems and stress analysis for structural elements. During the period under review investigations were started on the following subjects:

- Functional and qualitative analysis of the protection system for a neutral beam injector (NBI),
- investigation of the thermal stress of the neutral beam injection shine-through area during normal operation and in the case where the absorption by the plasma breaks down, and
- analysis of buckling phenomena of the vacuum container.

KfK contributions to accident analysis is coordinated with the NET team. KfK has also participated in the NET expert meetings of the safety group and took part in the "IAEA Technical Committee on Environmental and Safety Aspects of Fusion".

Availability and reliability analysis of sub-systems

The complex construction of a fusion device gives rise to ask for availability and reliability analysis in an early state of the design.

Since the middle of the seventies in the Institut für Reaktorentwicklung (IRE) there were methods under development for reliability and safety analysis of large and complex technical systems. In the meantime this methods were successfully applied in the field of fission reactors. The methods, developed by Caldarola, are based on the Boolean algebra with restricted variables. The advantage of the new method is the handling of multistate failure modes of components in the fault trees. This possibilities and further improvements in the display and the compaction of the results make the methods very promising also for the application in the fusion field.
As a first task analyses have been started related to the protection system for the neutral beam injector (NBI) of JET. This work should be considered as an example to demonstrate the ability of the applied method.

Beginning with a reference accident, e.g. plasma disruption in the main plasma chamber, all components are identified which influence that accident. A simplified flow diagram for the protection system is shown in Fig.18. The safety action for the NBI is

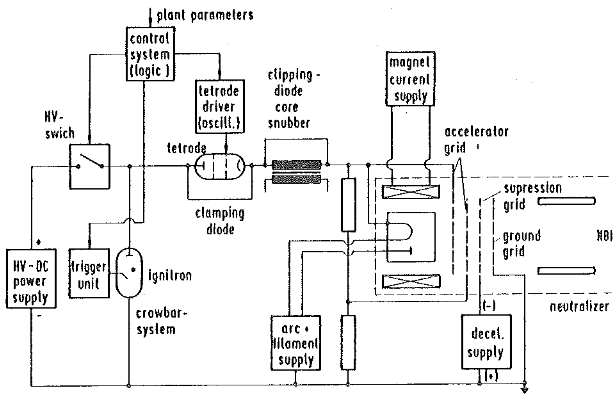

Fig.18: Flow diagram for the NBI protection system

current interruption in the acceleration grid. The most critical event is "no current interruption on demand" which is also defined as the TOP-event in the fault tree shown in Fig.19. In the analysis only those failures are of interest, which influence the system in the direction of the defined accident. In the protection system plant

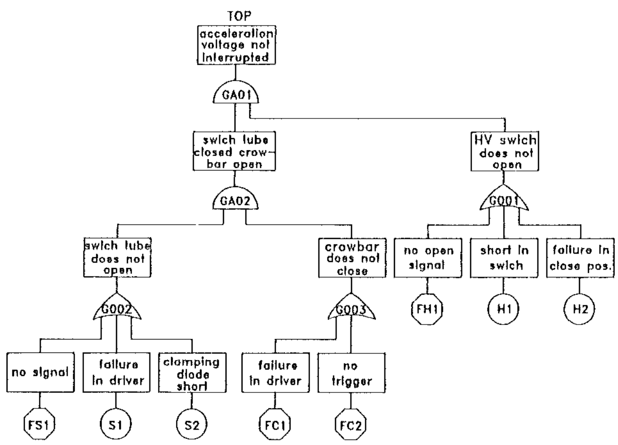

Fig.19: Fault tree for the NBI protection system

parameters, e.g. plasma density and wall temperature are monitored and controlled on maximum values. If these values are exceeded, the system swiches the current down

in the NBI. To improve the reliability of the swich down, a staggered swich system is used. The fastest swich component is the swich-tube. If the tube failes to open, the crowbar shorts the current. If the crowbar also fails, there are a few mechanical High Voltage (HV) swiches, which then short the current.
Reliability analyses of the NBI-protection system have been identified to be quite difficult, because as in all new designs there are hardly reliability data of the respective components available. In the next step therefore only parametric analysis are possible to be performed, to identify possible weak points of the currently foreseen system. This then might help to develope proposals for modifications if found to be necessary.

Stress analysis of the first wall

The first wall of a fusion reactor will be subject to high thermal and radiation loads, no matter which design concept is followed. Some regions of the first wall experience especially high power densities at normal operational conditions and even more so in the case of an accidental event. The resulting stresses, strains, swelling rates, and creep rates have to be analysed. For this aim, the computer code TSTRESS was developed at the University of Wisconsin - Madison (UWM). It has been taken over by KfK/IRE in 1983. The modelling of material properties (e.g., elastical behaviour, thermal and irradiation induced creep, swelling) has been modified. The data implemented at present in the code describe the properties of ANSI 20% CW 316 SS. If other materials are to be considered, these models have to be and can be exchanged.

The stress analysis may only yield relevant results if the initial conditions of the transient have been chosen realistically, especially in the case of investigations into accidental events. Such accidents may eventually arise after a long operational period with 10^3 to 10^6 operational cycles. It is both impossible and unnecessary to trace the pre-accidental history in detail. In order to provide realistic initial conditions, the algorithms of TSTRESS have been enhanced by KfK, yielding the long term version TSTRELT, which allows to cover long periods of cyclic operation without resolving each cycle. TSTRELT (which includes the capacities of TSTRESS, too) has been implemented on the CYBER 205 vector computer. The results of a long-term calculation serve as initial conditions for subsequent short-term analyses of accidental events.

Up to now, a sample calculation has been performed using TSTRELT, with the input data chosen close to the specifications of INTOR, and the temperature distributions inside the wall determined using the finite element code ADINAT. Fig.20 shows, as one of the results from these calculations, the stress distribution across the wall for one operational cycle which starts after about 13 months of cyclic operation.
There are significant residual stresses

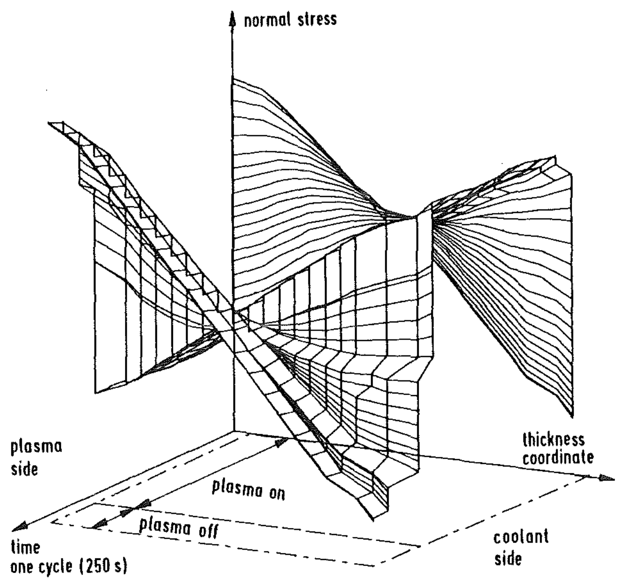

Fig.20: Stress profile across the first wall during one cycle, after 13 months of cyclic operation

inside the wall (recognizable at the beginning and at the end of the cycle) which have been developed during the operation, mainly due to irradiationally induced creep. The tensile stresses on the hot side of the wall are diminished when the heating is switched on, but it does not reach compressive stress levels as it did during the most early operational cycles, which started from flat stress distributions. The curvature of the residual stress distribution curve is caused by swelling effects which are becoming perceptible about six or seven months after the start of operation.

At present, an analysis is being prepared, where the loadings on the first wall of the JET vacuum chamber due to the neutral beam injector shine through will be investigated. The accidental scenario underlying this investigation assumes the plasma shut down, but the injectors continuing to operate.

Analysis of buckling phenomena

Another difficult problem in fusion reactor safety is to guarantee the structural integrity of the vacuum container (bell jar) which surrounds the reactor torus and the superconducting coils. It is known that even for large wall thicknesses with many reinforcing ribs relatively small container imperfections or accident forces may reduce the buckling load considerably. However, a reliable analysis of container buckling is rather difficult and requires tremendous computational effort. For instance, the finite element description in Fig.21,which is related to the INTOR-design,consists of more than 2200 elements with about 13 000 degrees of freedom. For conventional buckling analysis this is far beyond the computer capacities available. Therefore an approximative method is under development which allows for reduction of the computational effort without essential loss of accuracy. With this method buckling is not described as an eigenvalue problem, but as a simpler boundary value problem. For verification, the method will be applied to geometries, where closed form solutions or experimental results are available. For buckling of a rod, the critical load obtained by the approximative method was in good agreement with the exact value. Furthermore investigations for other geometries of the fusion reactor vacuum container will be carried out, in order to find out an optimal geometric configuration.

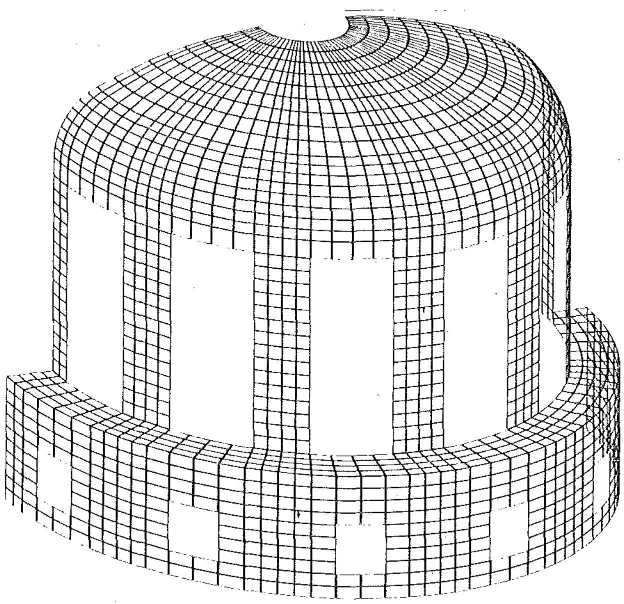

Fig.21: Finite element discretization of the INTOR vacuum container, diameter about 20 m

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NET/INTOR Studies

Within the frame of an INTOR contract, the design of the INTOR magnet system has been completed.

The INTOR TF magnet has been designed as a two zone winding in a casing with thin side walls at the central region. Two integrated wedging cylinders are formed by the circumferential ribs at the inner leg. In the low field zone up to 7.6 T the LCT-NbTi-Euratom conductor has been taken. For the high field part of the winding up to 11 T a Nb_3Sn upgrade version of the Euratom-LCT conductor has been designed with 20 kA conductor current. Supercritical forced flow cooling has been foreseen for both conductors. Comparative conductor designs have been carried out by the SULTAN group (ECN, ENEA, SIN) and an alternative TF coil structure by SIN.

The PF magnet design was based on a 50 kA bath cooled conductor proposed by JAERI (which reflects the situation that KfK/ITP had not yet started a PF conductor development). All PF coils are located outside the TF coils. The coil cross sections were evaluated and at least the outer 24 m diameter EF coil needs a subdivision. For all PF coils the NbTi technology appeared to be applicable as the induction at the conductors did not exceed 8.4 T.

Recently a NET-study contract has been started concerning the assessment of the status of A15 type conductor development and of the design implications in the case of the NET TF-coils. Within this frame, several new conductor designs are under consideration. Furtheron, an evaluation of radiation damage limits and the implication of nuclear heat on the conductor and winding design is under preparation for this study.

Further NET-study contracts are under discussion with the NET-team to be started rather soon. They might mainly concern material questions.

Publications:

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17646

19533

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DEVELOPMENT OF ECRH POWER SOURCES AT 150 GHz

This programme is devoted to the research on high power gyrotrons for electron cyclotron resonance heating (ECRH) at 150 GHz. A principle drawing of the experimental gyrotron is given in fig. 22. This gyrotron is a modular tube construction designed for operation in the TE_{031} mode. Design work for the first tube version still continues. The following results are achieved:

The generation of the hollow electron beam has been calculated using the Hermannsfeldt-code. A good beam quality with low velocity spread ($\Delta v_{\perp}/\bar{v}_{\perp} \approx \pm 1,8$ % at $\alpha_{\perp} = 1,4$) was achieved. The final optimization of the beam was started in connection with the design of the superconducting magnet system. The behaviour of the cathode materials under the specific operating conditions of the experimental gyrotron was tested and a mechanical design of the gun section was completed by the industry.

The codes for the beam-microwave interaction have been improved. Design calculations for the TE_{031}-mode are completed (for example, see fig. 23). The cavity consists of three slightly conical sections with different taper angles. The implementation of a new code was started, which includes further modes of interest (whispering gallery modes e.g.). As a microwave transition from the cavity to the large collector, a nonlinear taper was calculated with low reflection and low mode conversion (- 20 dB).

The window was studied by the industry; a preliminary study at mechanical stresses and materials was carried out.

The HV-power supply, the supplementary facilities together with the set-up of the experiment was planned in cooperation with industry. The preparation of the test facility was started.

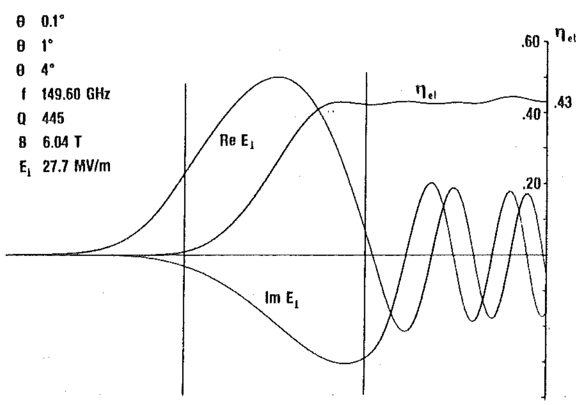

Fig. 23 Field patterns and efficiency for a TE_{031} cavity

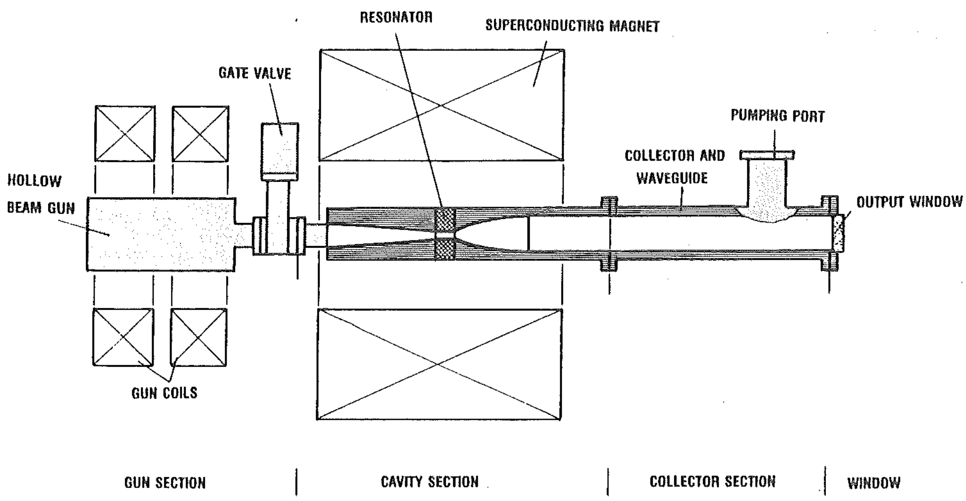

Fig. 22 Experimental 150 GHz-gyrotron

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Appendix

Taks Code No.	Title	KfK Departments	Partners
M 1	Large Coil Project	Institute for Technical Physics (ITP) Institute for Data Processing in Technology (IDT)	IEA agreement: USA, Japan, Switzerland, EC; Industry
M 3	Development of High Field Composite Conductors	Institute for Technical Physics (ITP)	CEA; ENEA; FOM/CEN; SIN/EIR
M 4	Poloidal Field Coils	Institute for Technical Physics (ITP)	CEA
B 1	Blanket Design Studies	Institute for Neutron Physics and Reactor Engineering (INR) Institute for Reactor Components (IRB)	CEA; ENEA; JRC Ispra; KFA; UKAEA; Industry
B 2	Development of Computational Tools for Neutronics	Institute for Neutron Physics and Reactor Engineering (INR)	CEA; ENEA; FOM/ECN
B 6	Corrosion of Structural Materials in Flowing $Li_{17}Pb_{83}$	Institute for Materials and Solid State Research (IMF)	CEA; SCK/Mol
B 9	Tritium Extraction Based on the Use of Solid Getters	Central Engineering Department (IT)	JRC Ispra; SCK/Mol
B 11–B 16	Ceramic Breeder Materials		CEA; ENEA; FOM/ECN; KFA; SCK/Mol; UKAEA
B 11	Fabrication of Ceramic Breeder Materials	Institute for Materials and Solid State Research (IMF)	
B 12	Characterization of Ceramic Breeder Materials	Institute for Materials and Solid State Research (IMF)	
B 13	Measurement of Physical, Mechanical and Chemical Properties	Institute for Materials and Solid State Research (IMF)	
B 14	Compatibility with Cladding Materials	Institute for Materials and Solid State Research (IMF)	
B 15	Irradiation Testing	Institute for Materials and Solid State Research (IMF) Institute for Neutron Physics and Reactor Engineering (INR)	
B 16	Tritium Recovery	Institute for Materials and Solid State Research (IMF) Institute for Neutron Physics and Reactor Engineering (INR)	
MAT 1	Post Irradiation Testing of SS	Institute for Materials and Solid State Research (IMF)	CEA; FOM/ECN; SCK/Mol; Studsvik
MAT 6	Irradiation Behaviour of Ceramics and Graphites	Institute for Materials and Solid State Research (IMF)	CEA; SCK/Mol
MAT 13	Electrical and Mechanical Properties: Simulation of Radiation Damage - Special Ceramics	Institute for Materials and Solid State Research (IMF)	CEA; UKAEA
T 1	Fuel Clean-up System	Central Engineering Department (IT)	CEA; KFA
T 5	Decontamination System	Institute for Radio Chemistry (IRCh)	CEA
T 6	Industrial Development of Large Components	Central Engineering Department (IT)	CEA
S+E 1	Radioactive Effluents: Behaviour of Gaseous Tritium in the Air, Plant, Soil System	Central Safety and Security Department (HS)	CEA; Studsvik
S+E 2	Accident Analysis	Institute for Reactor Development (IRE)	CEA; FOM/ECN; JRC Ispra; Risø
Gyrotron Studies		Institute for Nuclear Physics (IK) Institute for Data Processing in Technology (IDT)	Industry; Universities