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Institut für Angewandte Reaktorphysik
Institut für Neutronenphysik und Reaktortechnik
Institut für Reaktorentwicklung

Third FORATOM Congress, London

"INDUSTRIAL ASPECTS OF A FAST BREEDER REACTOR PROGRAMME"

German Contributions to Sessions I, II, III, IV, VI

D. Gupta, W. Häfele, K. Kummerer, D. Stegemann, D. Smidt, J. Seetzen

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KARLSRUHE

DEUTSCHES ATOMFORUM E.V.
und
KERNFORSCHUNGSZENTRUM KARLSRUHE

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*) In Association with EURATOM

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Third FORATOM Congress, London

INDUSTRIAL ASPECTS OF A FAST BREEDER REACTOR PROGRAMME

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Session I | German Report

Fast Reactor Development Objectives
General Reactor Characteristics and Fuel Requirements *)

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*) This paper is coherent with the respective papers to Session I of the Belgian and Dutch Atom Fora

**) In Association with EURATOM

1. Main Objectives

1.1 General Considerations

The development of power reactors is becomingmore and more a mature engineering enterprise. Due to the striking success of light water reactors one envisages a first generation of such power reactors. The successful development of light water reactors in Germany is the result of a large and well targeted effort of the German industry.

This effort was supported by the German government and it also evolved from the close co-operation with large US firms, so it was not a fully independent German effort. This development of light water reactors is entirely within the responsibility of the German industrial firms and therefore parallels somewhat the present situation in the United States with its similar economic structure.

With minor effort D_2O reactors and others were developed in a wholly German venture as a potential additional generation of thermal power reactors.

Besides this first generation of power reactors one envisages a second one which will be largely made up by fast breeder reactors. The development of such fast breeders will be an independent effort of Germany in connection with a number of partners, namely EURATOM, and in particular Belgium and The Netherlands. But there is of course also a rather intense exchange of information with the U.S.

This development of fast breeders started in the Kernforschungszentrum Karlsruhe in 1960. Gradually a rather large R. + D. program has been built up aiming at physics and safety, the choice of coolant and the fuel development of a large fast oxide breeder. The basic approach was to pronounce besides breeding in particular a cheap fuel cycle. The use of oxide as a fuel of a potentially high burn-up leads into a number of principle problems and decisions. Among others the question of the proper coolant was stressed. Around 1964 the work in Karlsruhe had progressed to a point where it was possible to determine more precisely the overall objectives of the whole program: In the late seventies an economically competitive fast breeder shall be available, with 1000 MWe and fuel cycle costs of ~ 1 mill/kWh or less. The capital cost should be in the neighborhood of 100 $/kW or less.

The size of a fast breeder with 1000 MWe leads into a rather delicate balance between inherent safety, breeding, and economy [1]. The status of the project has been described in greater detail more recently [2].

In the course of the Karlsruhe project quite a number of facilities has been brought up. There are the physics facilities SUAK (a pulsed subcritical fast assembly), STARK (fast thermal (Argonaut) reactor of Karlsruhe), SNEAK (fast critical assembly Karlsruhe), which is in the process of becoming critical now (Dec.66), and SEFOR which is a fast dynamical test reactor of 20 MWth as a joint venture with General Electric, the South West Atomic Energy Associates (SAEA) and the USAEC in Arkansas, USA. Furthermore facilities have been installed which allow for the development work of the closed fuel cycle such as: Pu facilities, α,ß,γ hot cells, a laboratory scale reprocessing plant and a laboratory scale refabrication line together with the necessary irradiation tools. In connection with the above mentioned choice of coolant a number of Na, He and steam loops have been built and operated.

On the basis of all these activities two larger 1000 MWe studies, one for Na and one for steam [3], [4] have been finished up to now. They serve as a basis for future action. This just described large R. + D. program has been associated with EURATOM since 1963. This association will come to an end by the end of 1967. The Karlsruhe-EURATOM association is interlinked with the fast reactor associations Belgium-EURATOM and The Netherlands-EURATOM which also expire by the end of 1967. They form together one common well concerted and coherent program.

During 1966 it became evident that the work of Karlsruhe and its Dutch and Belgian partners had come to a point where the design and construction of prototypes becomes possible and necessary. According to the industrial structure of Germany, The Netherlands, and Belgium this is clearly a task which falls under the scope of industrial activities.

Careful evaluations of the two above mentioned design studies lead to the result that both Na and steam as a coolant of a large fast oxide breeder have roughly the same potential and that Na and steam cooled fast breeders complement rather than duplicate each other. Therefore it was decided to design two prototypes by two industrial consortias under the clear responsibility and leadership of these two industrial consortias. By the same token the Karlsruhe program becomes then a base program which enhances and supports the more specific industrial activities on the level of a Kernforschungszentrum (or national laboratory). Besides that Karlsruhe will make the strongest use of its above mentioned facilities to expedite the necessary investigations for the two industrial groups.

It is felt that such an approach is a natural procedure if industry is responsible for the development, the sale and the economical success of the forthcoming 1000 MWe fast breeder power stations.

As an outcome of this approach the Siemens-AG. and Interatom formed a consortium during the course of 1966. This consortium will be responsible for the industrial development of the Na-cooled fast reactor line and in particular the Na-cooled prototype.

In parallel to that the Allgemeine Elektricitäts-Gesellschaft (AEG), the Gutehoffnungshütte (GHH) and Maschinenfabrik Augsburg-Nürnberg (MAN) joined in 1966 for bringing up the steam cooled fast reactor line and in particular the steam-cooled prototype. The two fast breeder prototypes shall have 300 MWe and their expected date of the start of construction will be 1969/1970. 300 MWe is a comparatively large size and 1969/1970 is a rather early date. Therefore it is necessary and in particular for technical reasons to have a preparational step for each line of development.

The KNK reactor of Interatom which is presently under construction at Karlsruhe is this preparational step for the sodium line. It is a Na-cooled 20 MWe reactor using zirconium hydride as a moderator and thus leading to a core which is practically as compact as that of a fast reactor. It is being contemplated to install as a second core a coupled or even all fast core. The KNK reactor will go into operation in 1968/1969.

The HDR reactor of AEG which is presently under construction at Kahl is the preparational step for the steam line. It has been considered thoroughly to install rather quickly as a second core a coupled fast thermal core leading to the STR reactor. This gives in particular the important possibility to test under true fast reactor conditions a large number of fuel pins of the steam cooled fast breeder prototype. The HDR reactor will go into operation by the middle of 1968.

Now reference is being made to the already established co-operation between the three EURATOM associations with Germany, Belgium, and The Netherlands. Starting from that it is the clearly indicated intention of both Belgium and The Netherlands to participate not only in the base program of the Kernforschungszentren (national laboratories), (that is in the case of Germany GfK, Karlsruhe, in the case of Belgium CEN, Mol with industrial partners and in the case of The Netherlands the RCN/TNO group) but also in the industrial enterprise of the Na-cooled prototype. The negotiations between Neratoom (Netherlands) and Belgonucleaire (Belgium) with the SSW/IA consortium are well advanced and are supposed to come to an end early in 1967. In that case the Na-cooled prototype will be a joint venture of Germany, The Netherlands, and Belgium. The three governments are also in process of negotiating a proper agreement in parallel to and as a consequence of the envisaged industrial agreement with the objective of bringing up together the necessary R. + D. program and constructions funds. It is expected that EURATOM will participate in a mode which is still to be defined.

It should be kept in mind that the base program of Karlsruhe, Petten and Mol is dealing with both: the sodium and the steam line therefore giving opportunity to Belgium and The Netherlands to be in close contact with the steam line too. This base program is the prolongation of the present EURATOM associations with Germany, Belgium and The Netherlands, and it is the intention to have EURATOM as a full association partner to that program beyond 1967.

The costs of the presently pursued Karlsruhe program in association with EURATOM (until 1967) amount to 280 Mio DM, about 700 people are wholly or partly engaged in this work at Karlsruhe. The costs of the preconstructural R. + D. phase of industry (1966-1970) together with the KNK and HDR/STR reactor amount to about 370 Mio DM. The two fast reactor prototypes will cost together about 860 Mio DM. The base program of Karlsruhe beyond 1967 and until 1972 will be in the order of 200 Mio DM.

As mentioned before it is the objective of the so envisaged common enterprise of Germany, Belgium, and The Netherlands in connection with EURATOM to have commercially competitive fast breeders available to the electrical grid during the late seventies. The power rating shall be 1000 MWe, the power production cost should be less than 1,6 DPfg/kWh (less than 4 mill/kWh) at 0,7 load factor, 25 years plant life, 7 % interest rate and 10 $/g Pu price. The fuel rating will be around 0,9 $MWth/kg_{fiss}$ in the case of Na-cooling and at least 0,70 $MWth/kg_{fiss}$ in the case of steam cooling. For sodium cooling the rod power will be around 500 W/cm in the case of the Na-1 study and 300 W/cm in the case of the D-1 study but this will ultimately be higher. The Pu production rate of such reactors which has to be aimed at will not be a criterion as such, it depends in a more complex fashion on the economy of the growing electric power production and the possible combinations of thermal and fast reactors [5] (s. paper, session VI). Moreover, it seems possible to adjust the breeding ratio within certain limits without major changes, e.g. by enlarging or reducing blankets.

1.2 Fuel Requirements

The choice of fuel is Plutonium. More specifically advantage is being taken from the fact that the isotopic composition of the Pu being discharged from light water reactors has a Pu-240 content of about 25 %. This roughly coincides with the Pu-240 content of equilibrium Pu of a continued refueling fast reactor cycle. But it also implies that in particular the refabrication step is being developed in such a way that a neutron activity of 300 - 400 n/sec g Pu is being cared for. The choice of fertile material is U-238, either in the form of depleted or in the form of natural uranium. There may be certain advantages for Th-232 as a fertile material in case of a steam cooled or even sodium cooled breeder. But this has not yet been analyzed completely.

The target value for the burn-up is 80 000 - 100 000 MWd/to of heavy atoms. More recent results of Karlsruhe stress the importance of fission product poisening in large fast oxide reactors. Burn-ups beyond 80 000 - 100 000 MWd/to therefore may lead into higher fuel cycle costs because fission product poisening lowers breeding. Fission products of 50 000 MWd/to (average) can lead to a decrease of breeding ratio as large as 0,1. It also lowers the through-put through reprocessing and refabricating plants. We therefore envisage the necessity of combining the reprocessing of fuel from both fast and thermal reactors, in order to lower by such an approach the specific reprocessing (and refabricating) costs.

The fuel inventory requirements are governed by the target value for the rating of the fuel, that is about 0,9 $MWth/kg_{fiss}$ in case of the Na-cooled reactor line. The rating of the steam-cooled line may be somewhat lower, but $>$ 0,7 $MWth/kg_{fiss}$ due to the properties of the canning material. It should be kept in mind that for the rating of steam-cooled breeders the point of reference is the rating of the light water reactors (o,7 - o,9 $MWth/kg_{fiss}$). The rate of installing fast reactor units of 1000 MWe has been extensively discussed in the report KFK-366 which was issued for the FORATOM-Congress 1965 in Frankfurt. Reference to that report is being made here. This report has been complemented more recently by the KFK-466 issue.

It is the intention of the German fast reactor program not to rely on the availability of Pu alone. Therefore the early start-up of fast breeders by U-235 is being studied in great detail. In particular the Belgium-EURATOM Association is dealing with this subject. The most striking result seems to be that no engineering changes of a Pu designed reactor are necessary if a U-235 start comes out to be necessary. The fuel cycle cost will be higher in that case by 0,5 - 0,7 mill/kWh roughly. After 200 000 MWd/to then the breeder operates on a Pu basis.

The minimum breeding characteristics are difficult to fix. We make reference to the formerly cited report KFK-366. A breeding ratio of 1,3 - 1,4 seems to be desirable in the long run, but the actually requested value depends to a great extent on the fraction of power production by light water reactors and the evolution of the price for uranium. In such a context also a breeder of a rather low breeding gain (for example 1,15 in case of the steam cooled breeder) has its place, it is much better than any advanced converter. Breeding and doubling time has to be judged upon in the context of capital costs, Pu price, light water reactor potential, the growth rate of electrical power, and uranium price. It is not a criterion of its own. We therefore see no real minimum breeding ratio for the next 30 years or so.

1.3 Operational Characteristics and Considerations, Choice of Refueling Schemes

The choice of refueling scheme is being characterized by the decision that there will be a common core blanket management. This leads to a rather constant isotopic composition of all stages of the Pu fuel cycle, it reduces the difficulties of reaching the state of Pu isotopic equilibrium for freshly fueled breeders, it avoids Pu hold up in the blanket, and leads to larger through-puts to reprocessing plants and therefore to smaller specific costs for reprocessing.

To sharpen the arguments of this paragraph we give a rough indication of the throughputs through reprocessing and refabricating plants of the future. It fixes more the general order of magnitude rather than the actual values. The figures are taken from a more recent study of the Karlsruhe group [6].

Table 1

	1980	1985	1990	2000
Fabrication (to/year)				
fast	117	286	596	1618
thermal	755	1128	1430	1545
total	872	1414	2026	3163
Reprocessing (to/year)				
fast	40	132	304	1059
thermal	491	819	1146	1626
total	531	951	1450	2685

The fast reactors are being developed for off-load refueling, in particular the steam-cooled breeder will use the BWR loading procedure that is: it will load and unload under water. Off load refueling will be an accepted scheme by the German utilities, their intention is to have only an option for the particular date of refueling. Furthermore it is the intention to consider three loadings during the full life time of a fuel subassembly (or core).

2. Time Scale for Development Program

The overall time schedule of the fast breeder program is shown in the table below.

1960	start of the project
1964	STARK and SUAK go into operation
1965	Pu facilities go into operation at Karlsruhe
1966	α,ß,γ hot cells and SNEAK go into operation; start of the industrial program
1967	start of the construction of the first German reprocessing plant at Karlsruhe
1968/1969	HDR, KNK and SEFOR go into operation; start of the large Na and steam component tests
1970/1971	fuel tests under true fast neutron/steam conditions are available in the STR reactor (second core of HDR)
1970/1971	completion of the first fuel test round for both fast reactor prototypes after having achieved 50 000 MWd/to in a fast flux with a large number of pins, the construction of the two prototypes starts
1973/1974	the two 300 MWe prototypes go into operation
1978/1979	the 1000 MWe demonstration plants go into operation

3. Power Reactor Characteristics

The power reactor characteristics of 1978 cannot be given today. But the present thinking can best be described if reference is being made to the two 1000 MWe studies of Karlsruhe, that is report KFK-299 for Na and KFK-392 for steam. The more detailed data are given in table 2 and 3, the data of the Na-cooled 1ooo MWe plant given there refer to a revised and more modern version of Na-1.

4. Fuel Requirements for a Program of Commercial Reactors

More detailed data referring to fuel requirements of the fast breeder development under given conditions are presented in the paper for Session VI. There one will find a broader evaluation of possible options under given constraints (cf. 1.2).

Here only few data are presented in order to answer the given questions, but one has to keep in mind that the picture is incomplete. The interdependence of a growing nuclear energy production by two or more reactor types (thermal and fast) is more complicated and cannot be described properly by the data given here.

More specifically the following answers are being given therefore:

4.1 Resources

Provided the installation of light water reactors in Germany follows the estimated scheme [7], [8], about 5-10 to Pu(fiss) will be produced by these reactors at the end of the seventies. This amount seems to be sufficient to install two large 1ooo MWe fast breeders. But as it has been mentioned before, the start-up with U-235 is being considered as an additional option.

The whole question of availability of Pu and U-235 leads into the problem of reprocessing plants and separation plants. It is the definite plan to close the fuel cycle in Germany, that is to have available the full services of the transportation of irradiated fuel elements, of reprocessing, of refabricating the Pu fuel and of waste disposal.

4.2 Initial Inventory Requirements

The initial inventory requirements are determined by the fuel ratings. As it has been mentioned before the target value for a 1000 MWe plant is about 0,9 $MWth/kg_{fiss}$ in case of Na and more than 0,70 $MWth/kg_{fiss}$ in case of steam. Due to the fact that there will be three loadings over the fuel life time, at the beginning of breeder start-up, the first reloading of the first 33 % of the core has to be available too. Therefore the effective values will be 4/3 times the values given by the fuel ratings, that is 1,5 kg_{fiss}/MWth in the case of Na and 1,9 kg_{fiss}/MWth in the case of steam.

The thermal efficiency of the plant is in both cases about 0,4. In case U-235 is being used 1,3 - 1,4 times larger amounts of U-235 are necessary than in case of the Pu with equilibrium isotopic composition. As it has been mentioned the additional hold up at the outside is roughly 1/3 of the core inventory.

4.3 Replacement Fuel Requirements

As it can be seen today, a self-sustaining fuel cycle can be obtained after the first fuel reloading where 1/3 of the core is replaced, if advantage is being taken from the mixed core - blanket reprocessing mentioned above.

4.4 Breeding Characteristics and Doubling Times (cf.3)

Net Pu surplus

Na-1	302	(kg Pu_{fiss}/GWe·a)
D-1	80,9	"

Characteristic doubling times depend not only on the characteristics of a single reactor type but also on the growthrate of an expanding energy demand. Therefore doubling times are not given here. The ratio of net Pu surplus/net initial Pu requirement may serve here as a characteristic figure.

Na-1	0,113	(1/yr)
D-1	0,023	"

Table 2

Sodium-cooled Fast Breeder (Revised Na-1 Version)

General

(a) Reactor Type : Heterogeneous Fast Breeder Reactor

(b) Coolant : Sodium

(c) Fuel and Canning Material : Fuel = UO_2 - PuO_2 -Pellets
Clad = SS CrNi 16 13

(d) Reactor Thermal Power : 2410 MWth

(e) Electrical Power (Gross and Net) : 1080/1000 MWe

(f) Thermal Efficiency : 41,5 %

Fuel Cycle Data

1. Number of zones

Core : 2 ; Ax.Blanket : 1
Rad.Blanket : 1

2. Reactor Initial Fuel Loading (all isotopes)

		Core I	Core II	Ax.Bl.	Rad.Bl.	Total
U :	t	9,110	8,800	17,890	33,26	69,060
Pu :	t	1,575	1,885			3,460

3. Isotopic Composition of Plutonium Originally Loaded
(cf. Nr.4)

4. Isotopic Composition of Plutonium used in Replacement Fuel
Core : 239 : 240 : 241 : 242 = 75 : 22 : 2,5 : 0,5

5. Isotopic Composition of Plutonium in Discharged Fuel
Average Composition in a Mixed Core-Blanket-Management
239 : 240 : 241 : 242 = 75 : 22 : 2,5 : 0,5

6. In Core Fissile Inventory Pu (all isotcpes)

Core Zone I	1,220 kg/MWe	)	
Zone II	1,460 kg/MWe	)	Pu-239+241 (for a burn-up of 80 000 MWd/to heavy atoms)
Total	2,680 kg/MWe	)	

7. Mean Enrichment of Replacement Fuel Pu (all isotopes)

   Core Zone I 14,73 a/o
   Zone II 17,63 a/o } Average : 16,18 a/o

8. Mean U-235 Concentration in Fertile Material used in Replacement Fuel

   ≈0 %

9. Mean Enrichment of Discharged Fuel Pu (all isotopes)

   Average : 7,73 % (Mixing Core + Blanket)

10. Mean U-235 Concentration in Discharged Fuel (cf.Item 7)

    0 %

11. Burn-up (maximum average burn-up of a core batch as discharged)

    80 000 MWd/to ≙ 8,4 %

12. Burn-up (mean burn-up during operation of the whole Core at Equilibrium)

    after reloading a new batch : 26 600 MWd/to
    before discharging a batch : 53 300 MWd/to

13. Breeding Gain

Breeding Ratio	Core	0,94
	Ax.Blanket	0,24
	Rad.Blanket	0,19
	Total	1,37

    Breeding Gain 1,37 - 1 = 0,37

14. Inventory Outside Reactor

    33 1/3 % of Core and Ax.Blanket
    16 2/3 % of Rad.Blanket
    + 5 % of the Initial Reactor Fuel Loading as Reserves

15. Plutonium Loss due to Reprocessing

    1 %

16. Residence Time at 100 % L.F.

    Core + Ax.Blanket : 2,0 years (100 % L.F.)
    Rad.Blanket : 4,0 years

17. Refueling Scheme (on or off load)

    Batch-wise, off load

18. Refueling Time per Sub-Assembly

    1,5 hours

19. Discharge Scheme. Proportion of Fuel Discharged and Time between Discharges

33 1/3 % of Core and Ax.Blanket
16 2/3 % of Rad.Blanket

Time between discharges : 0,66 years (100 % L.F.)

20. Fuel Fabrication Time and Time Delay between Fabrication and Loading

0,22 years

21. Spent Fuel Cooling, Transport and Reprocessing Time Delays

0,50 years

Summary of Fuel Requirements

Fuel Rating : a) Core (Fissile) : 867,5 W/g Fiss.Mat.
(Fuel) : 108,8 W/g Fuel

b) Core+Ax.Blanket +1/2 Rad.Blanket: 43,1 W/g Average

Breeding Gain : 0,37

Initial Reactor Fuel Laoding :

Core	Pu*)	3,46 g/kWe
	U	17,91 g/kWe
Ax.Blanket	U	17,89 g/kWe
Rad.Blanket	U	33,26 g/kWe
Total	Pu	3,46 g/kWe
	U	69,06 g/kWe

Additional Hold-up Required outside the Reactor :

Core	Pu	1,33 g/kWe
	U	6,86 g/kWe
Ax.Blanket	U	6,86 g/kWe
Rad.Blanket	U	7,20 g/kWe
Total	Pu	1,33 g/kWe
	U	20,92 g/kWe

Gross Running Requirements at 100 % L.F. :

Core	Pu	1,75 g/kWe·a
	U	9,04 "
Ax.Blanket	U	9,03 "
Rad.Blanket	U	8,40 "
Total	Pu	1,75 g/kWe·a
	U	26,47 g/kWe·a

*) all figures of Pu (all isotopes) are not in Pu-239 equivalents

<u>Net Running Requirements (Consumptions) at 100 % L.F. (at equilibrium)</u> :

Total U : 1,83 g/kWe·a

<u>Average Plutonium Production at 100 % L.F. (all isotopes)</u> :

Total 0,39 g/kWe·a

Total Additional Fissile + Fertile Material Required to be Fed to <u>Reactor before Achievement of Self-Supporting Fuel Cycle</u>

(cf. Additional Hold-up Required outside the Reactor)

<u>Open-Cycle Doubling Time at 100 % L.F.</u> :

8,9 years (Ratio of Initial Pu-Inventory to Average Pu-Production per year)

<u>Linear Doubling Time at 100 % L.F.</u> :

14,1 years

[According to the following expression:

$$P \cdot \delta_R \left[1 - \frac{1+P\cdot Z}{2\cdot P\cdot Z}\right] + \delta_W + \delta_F + \frac{M}{\Delta Pu} \cdot m(\varrho + \frac{1}{Z}) - \frac{\delta_R}{Z}$$

whereas:

P	Ratio of Residence Time Blanket/Core
δ_R	Residence Time of Core Elements (cf.No.16)
Z	Number of Batches
δ_W	Reprocessing Time (cf.No.21)
δ_F	Fabrication Time (cf.No.20)
M	Initial Pu Inventory
ΔPu	Average Pu-Production
m	Additional Demand according to Fabrication Losses (1.01)
ϱ	Additional Demand according to Reserve Elements (1,05)]

Table 3

Steam-cooled Fast Breeder (D-1)

General

(a) Reactor Type : Heterogeneous Fast Breeder Reactor

(b) Coolant : Superheated Steam (H_2O)

(c) Fuel and Canning Material : Fuel = UO_2 - PuO_2 - Pellets
Clad = Incoloy 625

(d) Reactor Thermal Power : 2519 MWth

(e) Electrical Power (Gross and Net) : 1080/1000 MWe

(f) Thermal Efficiency : 39,7 %

Fuel Cycle Data

1. Number of zones

Core : 2 ; Ax.Blanket : 1
Rad.Blanket : 1

2. Reactor Initial Fuel Loading (all isotopes)

		Core I	Core II	Ax.Bl.	Rad.Bl.	Total
U :	t	12,525	12,055	17,650	32,240	74,470
Pu :	t	2,045	2,515			4,560

3. Isotopic Composition of Plutonium Originally Loaded

(cf.No.4)

4. Isotopic Composition of Plutonium used in Replacement Fuel

Core : 239 : 240 : 241 : 242 = 74 : 22,7 : 2,3 : 1

5. Isotopic Composition of Plutonium in Discharged Fuel

Average Composition in a Mixed Core-Blanket-Management:

239 : 240 : 241 : 242 = 74 : 22,7 : 2,3 : 1

6. In Core Fissile Inventory Pu (all Isotopes)

Core I	1,560 kg/MWe	)
Core II	1,920 kg/MWe	) Pu-239+241 (for a burn-up of 55 000 MWd/to heavy atoms)
Total	3,480 kg/MWe	)

7. Mean Enrichment of Replacement Fuel Pu (all isotopes) :

   Core Zone I 14,04 a/o ⎫
   Zone II 17,26 a/o ⎭ Average : 15,65 a/o

8. Mean U-235 Concentration in Fertile Material used in Replacement Fuel

   ≈ 0

9. Mean Enrichment of Discharged Fuel Pu (all isotopes)

   Average : 7,71 % (Mixing Core + Blanket)

10. Mean U-235 Concentration in Discharged Fuel (cf.Item 7)

    0

11. Burn-up (maximum average burn-up of a core-batch as discharged)

    55 000 MWd/to ≙ 5,8 %

12. Burn-up (mean burn-up during operation of the whole Core at Equilibrium)

    after reloading a new batch : 18 330 MWd/to
    before discharging a batch : 36 660 MWd/to

13. Breeding Gain

Breeding Ratio	Core	0,883
	Ax.Blanket	0,081
	Rad.Blanket	0,172
	Total	1,136

    Breeding Gain 1,136 - 1 = 0,136

14. Inventory outside Reactor

    33 1/3 % of Core and Ax.Blanket
    16 2/3 % of Rad.Blanket
    + 5 % of the Initial Reactor Fuel Loading as Reserves

15. Plutonium Loss due to Reprocessing

    1 %

16. Residence Time at 100 % L.F.

    Core + Ax. Blanket : 1,88 years (100 % L.F.)
    Rad.Blanket : 3,76 years

17. Refueling Scheme (on or off load)

    Batch-wise, off load

18. Refueling Time per Sub-Assembly

    44 h/batch

19. Discharge Scheme. Proportion of Fuel Discharged and Time between Discharges

33 1/3 of Core and Ax.Blanket
16 2/3 of Rad.Blanket

Time between discharges : 0,63 years (100 % L.F.)

20. Fuel Fabrication Time and Time Delay between Fabrication and Loading

0,22 years

21. Spent Fuel Cooling, Transport and Reprocessing Time Delays

0,50 years

Summary of Fuel Requirements

Fuel Rating : a) Core (Fissile) : 673,3 W/g Fiss.Mat.
(Fuel) : 80,4 W/g Fuel

b) Core+Ax.Blanket: 40,0 W/g Average
+1/2 Rad.Blanket

Breeding Gain : 0,136

Initial Reactor Fuel Loading :

Core	Pu*)	4,56 g/kWe
	U	24,58 g/kWe
Ax.Blanket	U	17,65 g/kWe
Rad.Blanket	U	32,24 g/kWe
Total	Pu	4,56 g/kWe
	U	74,47 g/kWe

Additional Hold-up Required Outside the Reactor :

Core	Pu	1,75 g/kWe
	U	9,42 g/kWe
Ax.Blanket	U	6,76 g/kWe
Rad.Blanket	U	6,98 g/kWe
Total	Pu	1,75 g/kWe
	U	23,16 g/kWe

Gross Running Requirements at 100 % L.F. :

Core	Pu	2,45 g/kWe·a
	U	13,20 "
Ax.Blanket	U	9,48 "
Rad.Blanket	U	8,66 "
Total	Pu	2,45 g/kWe·a
	U	31,34 g/kWe·a

*) all figures of Pu are not in Pu-239 equivalents

Net Running Requirements (Consumptions) at 100 % L.F.(at equilibrium):

Total U : 1,65 g U/kWe·a

Average Plutonium Production at 100 % L.F. (all isotopes) :

Total 0,106 g Pu/kWe·a

Total Additional Fissile + Fertile Material Required to be fed to Reactor before Achievement of Self-Supporting Fuel Cycle

(cf.Additional Hold-up Required outside the Reactor)

Open-Cycle Doubling Time at 100 % L.F. :

43 years (Ratio of initial Pu-Inventory to Average Pu-Production per year)

Linear Doubling Time at 100 % L.F. :

61,8 years

[According to the following expression:

$$P \cdot \delta_R \left[1 - \frac{1+P \cdot Z}{2 \cdot P \cdot Z}\right] + \delta_w + \delta_F + \frac{M}{\Delta Pu} \cdot m(\varrho + \frac{1}{Z}) - \frac{\delta_R}{Z}$$

whereas:

P	Ratio of Residence Time Blanket/Core
δ_R	Residence Time of Core Elements (cf.No.16)
Z	Number of Batches
δ_w	Reprocessing Time (cf.No.21)
δ_F	Fabrication Time (cf.No.20)
M	Initial Pu Inventory
ΔPu	Average Pu-Production
m	Additional Demand according to Fabrication Losses (1.01)
ϱ	Additional Demand according to Reserve Elements (1,05)]

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INDUSTRIAL ASPECTS OF A FAST BREEDER REACTOR PROGRAMME

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Session II

German Report

Fuel Fabrication and Reprocessing

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1. Fabrication

1.1 Fuel Elements

a) Reasons for choice of fuel material: oxide, carbide, etc.

On the basis of technical and economic analysis under German conditions the mixed oxide of plutonium and uranium appears to be the most suitable fuel for both the sodium cooled and steam cooled versions of the German fast breeder project. Therefore, it was decided that the first cores will consist of oxide fuel elements. The main reasons which influenced this decision were as follows:

The industrial background, even in Germany, is more established for oxide type ceramic. As a result less development work is necessary to ensure the availability of desired amounts of fuel in due time.

Up to now, most irradiation results available refer to oxide fuel which offers a relatively acceptable basis for a fuel element layout. Especially high burnups up to 100 000 MWd per ton of heavy metal were verified with single specimens using rod powers between 400 and 800 watts per cm.

In the reference fast reactor designs which are presently under consideration, the range of rod power attainable with mixed oxide appears to be technically feasible and economically attractive. For example a typical reactor layout with Na cooling requires a maximum rod power of 600 watts per cm and a burnup of 80 000 MWd per ton, whereas that with steam cooling is limited to a maximum rod power of 450 watts per cm and requires a minimum burnup of 50 000 MWd per ton.

According to the state of the present knowledge, the radial swelling difficulties with oxide may be less severe than with other ceramics. This may be also due to the fact that the average temperatures in oxide fuel are higher, leading more into the range of fuel plasticity. For the required burnups the resulting radial swelling can be accommodated within the internal geometry by the available void (pores, gap etc.), if the smeared fuel density does not exceed ca. 85 % of the theoretical value. Because of the low thermal conductivity of the oxide, the average fuel temperature is in reference designs above 1200°C. Therefore a large part of the fuel is already plastic and has a remarkably high fission gas release reducing the swelling rate.

Mixed oxide is characterized by its significant compatibility to cladding materials and to the cooling media. No dangerous mass transfer mechanism (as e.g. for the C in carbides) has to be taken into account. Also the overall inert behaviour with respect to air and water may be mentioned in this connection.

It is felt that these advantages outweigh under the present conditions the drawbacks of oxide fuels, namely their lower heavy metal density and their comparatively low thermal conductivity.

For the fertile component of the fuel mixture both natural and depleted uranium is possible, the actual choice being governed by the market conditions. According to recent investigations the uranium oxide is suitable for the blankets as well. The isotopic composition of the Pu in the fuel varies of course with the mode of fuel cycle operation. With Na cooling in a separate core-blanket management the composition of the Pu is around 64 % Pu 239, 30 % Pu 240, 3,5 % Pu 241, 2,5 % Pu 242. In a mixed core-blanket management the isotopic composition changes to 75 % Pu 239, 22 % Pu 240, 2,5 % Pu 241, 0,5 % Pu 242. With steam cooling the composition remains approximately in the same range.

b) Reasons for choice of canning material

Although the same fuel is chosen for both fast reactor versions the choice of canning materials has to take into account the different requirements of sodium and steam cooling.

The principal choice for sodium cooled fuel elements aims at the group of high temperature stainless steels. The present reference material is an austenitic alloy with the composition: 16 % Cr, 13 % Ni, 1,3 % Mo, 1 % Nb, 0,75 % V, 0,75 % C. It has a high creep strength and remarkable resistance against sodium corrosion including mass transfer, which are the two most important properties required. The unirradiated rupture strength for this material at 650°C and 20 000 hours has been measured to be in the range of 15 to 20 kg/mm^2.

For the steam cooled fast reactor fuel element some of the nickel base alloys are proposed. Presently, tubing out of Inconel 625, Inconel 718, Hastelloy X and René 41 (with duplex coating) are being thoroughly investigated. These alloys are characterized by their excellent mechanical properties up to 750°C, a temperature essential for steam cooling. The unirradiated rupture strength ($\sigma_{B/20000}$) at 750°C is for all the mentioned alloys in the range of 20 to 30 kg/mm^2. In this range the properties of stainless steels become inadequate ($\sigma_{B/20000}$ = 5 - 10 kg/mm^2), especially if the creep-buckling problem induced by the high steam pressure is taken fully into consideration. A further important parameter of selection was the resistance of these alloys against stress corrosion caused by high temperature steam. Although the stainless steels of the Incoloy type offer a good corrosion resistance, their mechanical properties are not sufficient. Incoloy 800 for example has a 20 000 hours rupture strength at 750°C below 5 kg/mm^2.

Besides the properties which are specific to the coolant type the canning materials for both versions need to be discussed with respect to their mechanical behaviour under fast neutron irradiation, with respect to their neutron absorption cross section and to their economics. The high temperature embrittlement seems to be acceptable, however there is a remarkable lack of experimental data, particularly for neutron doses above 10^{22} nvt. While the components of stainless steel have no major detrimental influence to the neutron economy of a sodium cooled system due to their fast neutron absorption cross sections, the nickel base alloys under consideration for steam cooling affect the breeding ratio adversely. Also additions to the alloys as Mo, Nb and W influence the neutron economy. As a general rule, components in the canning materials with absorption cross sections above 20 mbarns have to be investigated carefully with respect to the improvements which they may bring in mechanical strength and economics. The canning material selection was also made in view of their availability and cost potential under expected proper future European conditions.

c) Items of future development work

Concerning future development lines for fast reactor fuel, both the carbide and nitride compounds of Pu and U have certainly some potential mainly because of their **high** heavy element density and their higher thermal conductivity. The higher density would lead to an increased breeding ratio (above 1.5), whereas an improved thermal conductivity would allow a rod power up to 2000 watt/cm or/and larger fuel diameters as with oxide. Of course the mixed carbides could be used only in sodium cooled systems due to their chemical affinity to steam. Nitrides on the other hand, appear to have some potential for use in both versions, although there might exist similar problems concerning their affinity to steam. The present fabrication technology of mixed oxides may be improved by an extensive use of vibrocompaction techniques and by new preparational methods like resistance melting. Furthermore, substoichiometric oxides may offer some improvements in their thermal conductivity.

In the field of canning materials, future developments will refer to vanadium base alloys and to improved ferritic steels. Since a few years the development of V-base alloys suitable to Na cooling is in progress. Such advanced canning material promises better mechanical strength than austenitic steels at high temperatures and less sensitivity to high temperature embrittlement under fast neutron irradiation. Concerning the corrosion of these alloys, however, there are some open problems. Because of their basic economic potential and their better behaviour under high temperature irradiation the ferritic steels may become attractive provided the mechanical properties at high temperatures can be improved satisfactorily, e.g. by dispersion hardening with ceramic substances. The potential of fuel venting was investigated in some detail. There is possibly an economic future for such a design, however the technical difficulties prevent a quick realization of this concept.

d) Possibilities of standardisation

A possibility for standardized specifications becomes more and more relevant in the field of tubings. Especially the permissible tolerances for outside and inside diameter and for the ovality of the tubing have to be adjusted to the requirements of nuclear industry. Also test procedures for different stages of fuel element fabrication should be mentioned in this context.

1.2 Fuel Manufacture

a) Glove box facilities

All the presently operated glove box units for handling of fast reactor fuels are situated at the Karlsruhe Research Center. In the course of the last years very large facilities were here established and came into full operation. These facilities include special units, where diversified basic and applied research work in the field of plutonium technology can be carried out. The Euratom owned Transuranium Institute offers the most versatile setup both on laboratory and on pilot scale. The total glove box volume amounts to 250 m^3. This institute is a part of the common research establishments of Euratom and performs research and development work also for the Euratom associated fast breeder project. Further glove box lines are established in smaller laboratories of the Gesellschaft für Kernforschung, dealing with test sample fabrication and also with rather extensive analytical measurements. There is also a production unit available, owned by the firm ALKEM, with a capacity of about 2 tons of mixed oxide per year. Here, during recent time valuable practical experience was gained in the field of precision sintering (e.g. up to tolerances of a few microns) in course of the large scale production of Pu-containing oxide fuel elements for the fast reactor SNEAK.

b) Radiation and toxicity control

All the standard procedures in this connection are applied in the Karlsruhe facilities. These include air monitoring, criticality control by geometry and administration, as also a well experienced health physics control of personal by dosimeter and medical examination. In this connection the extensively used whole-body-counter and the thoroughly examined bio-assay (urine analysis for alpha-emitters) are to be mentioned.

c) Future development of automatic plant

With the present and foreseeable future level of fast reactor fuel element production - especially for the supply of fast prototype reactors - an automatization of fabrication does not yet appear to be essential. However, with increasing throughputs expected further, there might be some stronger technical and economic incentives in this direction. Also the future large scale handling of dirty plutonium will lead to automatization of those steps in the production line where an uneconomic increase in the number of working personal due to dose rates must be avoided. A recent very detailed analysis carried out at Karlsruhe investigated the local irradiation doses at all the steps in a plant for refabrication of reprocessed dirty Pu fuel. Taking into account the neutron emission rate of about 300 neutrons per sec and per gram Pu and also the gamma radiation of U 237 and Pu as well as of residual fission products (at a decontamination factor of 10^7) it was found, that all but one refabrication steps can be modified and adjusted to "dirty Pu-conditions", e.g. by applying a semi-remote technique or/and a simple shielding, so that the personal dose rates do not limit economic operation. However, in the last step, that is the assembling of the fuel pins, rather high dose rates of up to 30 mrem per hr are not avoidable by simple shielding and semi-remote handling techniques. Therefore this step will gain the maximum advantage by automatization. It could be also envisaged that a future large scale routine production would need a nearly totally automatized plant, in which the elements of human errors are reduced to a minimum.

1.3 Fuel Accounting and Measuring of Isotopic Constitution

The distribution, movement and intermediate control is carried out strictly according to administrative procedures established by the government in collaboration with the Euratom Supply Agency. The normal fuel accounting methods are applied in all the laboratories handling fast reactor fuels. The analysis for Pu content uses well advanced methods as potentiostatic coulometry and roentgen fluorescence absorption. The determination of the isotopic constitution of Pu is carried out by mass spectrometer methods. The Karlsruhe Transuranium Institute uses for this purpose a thermal ionization mass spectrometer. In the Institut für Kernverfahrenstechnik a similar device is in operation routinely. The independent results of these two institutions are crosschecked from time to time to avoid systematic error.

2. Reprocessing

2.1 Present Plant Capacity and Plans for Future Expansion

Presently in Germany there is no facility for the reprocessing of fast reactor fuel elements in operation. The future requirements are expected to be in the range of 50 tons per year in 1980 increasing to about 700 tons per year in 1990. The necessary plant capacity for the next decade is at present under thorough examination. Especially combined reprocessing of fast and thermal reactor fuel elements is also envisaged in order to reduce costs on account of reduction of fissile material concentration and on account of larger plant size.

2.2 Reprocessing Costs and Reprocessing Times

On the basis of prevalent reprocessing costs of about DM 80.- per kg of irradiated heavy metal content for thermal reactor oxide type fuel, a further analysis under German conditions indicated that reprocessing costs for fast reactor fuel would be DM 100.- to 150.- per kg of mixed core and blanket heavy metal, using a modified Purex process.
For a single fast reactor system the reprocessing time is determined by the fuel cycle requirements. The total out-of-pile time is governed by the chosen batch loading schedule. In an increasing population of reactors, however, this point of view becomes less important because the reprocessing management may be so organized as to reduce the out-of-pile inventory to an economic minimum. Under such future conditions the reprocessing time alone may vary between some days and a few months. For the reference designs a total reprocessing time (including cooling and transport time) of about 0.5 years was considered to be realistic.

2.3 New Reprocessing Methods

a) Decontamination factor versus cost

For fast reactor purposes the well established Purex process with minor modifications appears to be quite adequate both from the technical and economic points of view. The major difference of the fast breeder fuel elements from thermal converter elements is the higher fissile material concentration (10 to 15 %) and the higher fission product content (50 to 80 grams per kg fuel) because of higher burnup. A lower Pu concentration of 4 to 6 % and fission product content of 20 to 30 grams per kg can be obtained by simply following a mixed core-blanket management. If at all necessary, the fissile material concentration may be further reduced by mixing the fuel elements with those from converter type reactors before reprocessing. The recovery and purification of fissile materials can then be carried out in conventional reprocessing plants based on solvent extraction, without any major modification. Only the head end and the tail equipment have to be properly dimensioned. The isotopic mixing of Pu from fast and converter type reactors would not cause a major shift in composition, as Pu 240 concentrations from both these sources are in the same range (22 % in coreblanket mixtures of fast breeders against 25 % in light water reactors). Such a mixing might be slightly disadvantageous for U 235 if the discharge concentration from light water reactors exceeds the natural concentration. For other thermal converters this problem does not arise.

Some detailed internal analyses carried out at Karlsruhe indicate that a decontamination factor (DF) of about 10^7 would be required both for U and Pu, from the point of view of refabrication. This corresponds, approximately, to the background activity of natural uranium. For plutonium, a higher DF does not bring in much, as the neutron emission of the dirty Pu and the gamma activity of U 237 and Am 241 (both produced from Pu 241) predominantly determine the dose rates in the fabrication steps.

A DF of 10^6 could be attained in two extraction cycles, whereas a third cycle is required to obtain decontamination factors of 10^7 to 10^8. Since a DF of 10^7 is essential for U and Pu refabrication and the range of 10^7 to 10^8 can be obtained in that single additional cycle, no significant differences in the reprocessing costs are expected for this range of decontamination factors.

b) Dry-way reprocessing, etc.

Amongst the known dry reprocessing ways, the fluoride volatility method may be an advanced alternative to aqueous reprocessing. Therefore, a combined venture has been initiated by Belgium and Germany in connection with the fast breeder project. In this context reference is made to the report of the Belgian Atomic Forum. The experimental development work is being carried out at the C.E.N. laboratories at Mol with active participation of the German side.

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INDUSTRIAL ASPECTS OF A FAST BREEDER REACTOR PROGRAMME

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Session III

German Report

European Fast Reactor Experimental Programmes and Further Development

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Description of Laboratories and Facilities, and Current Programmes

This section lists major experimental facilities devoted to fast reactor research and development work. For each item a description is given of the experimental setup and a summary of the current and future programme indicating the major types of work.

Research and developmental work within the scope of the fast breeder program in Germany can be described in two sections, the first of which is the general basic program at the Karlsruhe Nuclear Research Center and the second is the R.+D. work for the two fast prototype reactors which is carried out under full responsibility of the industrial firms as detailed in the paper to Session I. In addition to that, the R.+D. programs of The Netherlands and Belgium, as described in the paper of these countries to Session III of this congress, contribute extensively to the common fast breeder program on both levels, the basic and the industrial one.

1. Basic Program

1.1 Physical Experiments

1.1.1 The Fast Zero Power Assembly SNEAK (Schnelle Null-Energie-Anordnung Karlsruhe)

SNEAK is a fixed vertical assembly with fuel elements suspended from a grid. The reactor has a maximum diameter of 3.2 m and a maximum height of 2.7 m. It is assembled out of square elements which are suspended from a top grid plate, and, after completion of loading operations, are fixed in their positions by clamping devices at the bottom. There are two loading machines: a lower loading machine which transports elements between the reactor and the fuel element transit lock; an upper loading machine which hoists the elements into the reactor.

The fuel elements in the core section are square stainless steel tubes of 51 x 51 mm inner dimensions, the elements in the blanket section are square tubes of 153 x 153 mm inner dimensions. The lattice pitch is 54.4 mm in the core and 163.2 mm in the blanket. At present, the core section contains a 24 x 24 array of elements, it can later be extended by replacing blanket elements with bundles of 9 fuel elements. The core elements have a specially shaped cross section so that cooling channels are created between adjacent elements.

The complete facility and its surrounding buildings have been designed for use of plutonium as fuel.

The experimental program of SNEAK is starting with measurements on a well-known fast uranium core, a repeat of the ZPR-III assembly 41. This assembly has been chosen because all materials necessary are available. The purpose of "SNEAK 1" is to test the SNEAK facility and its experimental equipment. The data obtained will be compared with those of ZPR-III.

In support of the fast reactor development work at Karlsruhe, the next two assemblies are designed to investigate special features of sodium- and steam-cooled fast plutonium oxide reactors. Preparations for the sodium assembly "SNEAK 2" began in 1964, and for the steam assembly "SNEAK 3" in 1965. Since information on steam-cooled fast reactor systems with plutonium fuel is still scarce, it has been decided that "SNEAK 3" will be built before "SNEAK 2". It is planned that mock-ups of sodium- and steam-cooled fast proto-type reactors will follow these initial assemblies.

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KFK-472

1.1.2 The Fast-Thermal Argonaut-Reactor STARK (Schnell-Thermischer Argonaut-Reaktor Karlsruhe)

The zero-power facility STARK is a flexible fast-thermal reactor consisting of a subcritical fast core and a surrounding Argonaut type thermal driver zone. The basic concept of the system dates back to 1962 when it was recognized that a flexible source reactor would be needed in the earlier stages of the Karlsruhe fast reactor program, and therefore, it was decided to convert the Argonaut-Reactor at Karlsruhe into a coupled fast-thermal assembly.

The thermal core consists of normal Argonaut type fuel plates and light water as a moderator. The thermal zone is operated at 80°C in order to guarantee a high degree of inherent shutdown capability of the reactor. The fast core (37.2 cm average diameter) is formed by an array of 37 vertical stainless-steel matrix tubes which are fixed by a bottom grid plate. The tubes can be filled with platelets (2 inches by 2 inches) of various core materials. In order to prevent strong peaks in the U-235 fission rate at the edge of the fast zone, the core structure is enclosed in a 5 cm thick natural uranium casing which absorbs slow neutrons incident from the driver zone. Thus, coupling between the zones is maintained mainly by the exchange of fast neutrons.

The experimental program takes aim at two main points:

(a) development of experimental techniques for investigating fast zero power systems, especially fast reactor mock-ups on the critical assembly SNEAK,

(b) investigation of the reactor physics properties of the fast-thermal coupled system STARK and comparison of experimental data with multigroup calculations.

In part (a) of the program new techniques have been developed and existing ones improved for fast reactor application. Among them are neutron spectrometry, pile-oscillator for heated samples and heated foil furnaces for studies of the Doppler effect. Further, investigation on pulsed source and noise analysis techniques, especially in order to determine the neutron lifetime on one side and reactivity during shut-down and approach to criticality on the other side.

In part (b) of the program the analysis of the experimental data has shown that most of the fundamental properties of the coupled system STARK are described fairly well by multigroup diffusion calculations. Satisfactory agreement was found, in particular, for the critical mass and the partition of reactor power between the various zones.

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KFK-482

1.1.3 The Fast Subcritical Facility SUAK (Schnelle Unterkritische Anordnung Karlsruhe)

The assembly is essentially composed of platelets 2" x 2" x $\frac{1}{8}$" of 20% enriched uranium, which are held in aluminium square tubes of 1 mm wall thickness. 25 or 36 tubes can be closely packed together on an aluminium base plate of 3 cm thickness to form a parallel epiped with side lengths of 27.0 or 32.3 cm, respectively. The height of the assembly can be varied by changing the filling height of the tubes. By admixture of platelets of other materials the composition can be varied. The tubes are closed on top with a plug of 3 cm aluminium and at the bottom with 3.5 cm stainless steel. Two central tubes act as safety rods and drop out of the assembly if a given multiplication is exceeded.

To minimize the influence of backscattered neutrons the assembly is located 5 m above the ground in the centre of a thin-walled building of dimensions 10 m x 15 m x 10 m. A 200 keV Cockcroft-Walton accelerator and auxiliary equipment is placed on a grid 3.5 m above the ground.

SUAK is primarily used for pulsed neutron decay measurements and for spectrum determinations by the time-of-flight technique. For easier interpretation of measurements only bare assemblies are constructed which also contain as few materials as possible. So far three assemblies have been investigated. The first assembly consisted of pure uranium (20% enriched) only. In the other assemblies different amounts of hydrogen-containing material was added to shift the neutron spectrum to lower energies. The experimental results were compared to calculations in which different approximations and cross sections were used.

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1.1.4 The Southwest Experimental Fast Oxide Reactor (SEFOR)

The SEFOR is located near Fayetteville Arkansas, USA. It is being built by the General Electric Company (GE) and owned by the Southwest Atomic Energy Associates (SAEA). It is commonly funded by SAEA, GfK, USAEC, and GE.

The 20 MW sodium-cooled reactor is wellknown and need not be described here once more. It will provide a series of key-informations with respect to safety problems of sodium-cooled fast breeders. In the experimental program GE and GfK cooperate. This program is devoted to

(a) measurement of Doppler effect during normal operation conditions

(b) measurement of Doppler shut-down effect under transient conditions.

SEFOR is scheduled to start its operation in early 1968.

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1.1.5 Van-de-Graaff Accelerator (Institute of Applied Nuclear Physics, Karlsruhe, Nuclear Research Center)

A 3 MeV Van-de-Graaff accelerator with a bunching facility is used for cross section measurements for the fast breeder development programme.

The experimental programme is concerned with the measurement of capture cross sections of structural material and fission products in the energy range of 10 - 200 keV. Also fission cross section of the plutonium isotopes 239, 240, and 241 are measured. Further, effort is under way for absolute measurements of cross sections and neutron fluxes, as well as for self-shielding factors and resonances in the keV-energy region.

It may be mentioned in this connection that cross sections for shielding materials are studied in the MeV-range by use of the Karlsruhe Cyclotron.

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KFK-454

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KFK-455

1.2 Facilities for Closed Fuel Cycle Investigations

1.2.1 FR2 Irradiation Loop (Karlsruhe Nuclear Research Center)

This irradiation loop is a closed circuit with helium, in which small cylindrical samples of fuel material can be irradiated up to 100 MWd/kg. Inside the core an extensive gas purification and gaseous fission product removal system are attached to this loop, as canning damage under these high burn-up conditions cannot be ruled out. The circuit consists of a fuel sample introduction system into the central channel of the FR2 core, helium blowers with gas bearings, gas purification system, and the necessary accessories. The important operational data are:

operating pressure of helium		30 atm
helium throughput:	at the blower max.	440 kg/h
	at the test line max.	350 kg/h
total heat generation in the samples		30 kW
rod power of the specimen		300-1000 W/cm
gas entrance temperature to test line		50°C

The experimental program includes the irradiation of UO_2-PuO_2 samples with different enrichment, burn-up conditions and testing of the fission gas pressure build-up. The system has been in operation since early 1966.

1.2.2 Irradiation Test Facilities

Besides the FR2 at Karlsruhe which is intensively used for irradiation purposes for fuel specimen as well as for cladding material, the following reactors are also used or intended to be used for irradiation tests:

a) BR-2, Mol, Belgium
b) EFFBR, Detroit, USA
c) DFR, Dounreay, Great Britain
d) VKL, test loop in the Kahl BWR, Germany.

The bulk of cladding material specimen are irradiated in BR-2. Several test groups will be carried out within the fast breeder fuel element development program.

Fuel irradiation tests are mainly carried out in the FR2, while integral tests of whole fuel pins in a fast neutron flux at present can only be carried out in the DFR or, if possible, in the EFFBR. Later on integral tests of fast breeder fuel pins with mainly fast neutrons are planned in

a) BR-2, Mol, Belgium
b) STR, Kahl, Germany
c) KNK, Karlsruhe, Germany.

The STR will be a modified reconstruction of the HDR-core with a central fast core-zone. This reactor will then provide for enough irradiation space for the steam-cooled fast breeder version.

1.2.3 Hot Cells (Karlsruhe Nuclear Research Center)

The five hot cells erected at the Karlsruhe Center provide excellent possibility for carrying out research work on irradiated materials. Each of the three larger cells can handle a maximum α,γ activity of 10^7 Curie (at 1 MeV).

According to the flow of materials each of the cells has a specific function:

cell 1: disassembling
cell 2: mechanical operation
cell 3: chemical analysis
cell 4: technological tests
cell 5: non-destructive measurements.

In the field of the fast breeder project all the testing of irradiation samples with fuel elements and canning materials are carried out in these cells. They have been in full operation since the beginning of 1966. Another hot cell facility will start operation with AEG in 1968.

1.2.4 Reprocessing (Hot Chemistry Institute, Nuclear Research Center Karlsruhe)

In the field of reprocessing a laboratory scale plant has been designed to test the wet chemical method of reprocessing for the recovery and purification of U and Pu from high burn-up fast reactor fuels. For example, the different steps in the plant have been so designed as to reprocess fuel elements with a burn-up of about 60-80 MWd/kg fuel and Pu- or U-235-concentrations of about 10 - 20%. All the required data on extraction of fissile material in presence of large amounts of fission products (60-80 gr/kg fuel) and the associated high radiation field, the required decontamination factors and the losses of fissile material as well as the accounting and instrumentation can be obtained and checked.

The plant will have a capacity of about 1 kg per day. The whole of this facility is located in the institute of hot chemistry. In this institute R.+D.work on the chemistry and technology of transuranium elements like Np, Cm, Am is also carried out in the frame-work of the fast reactor project. In a significant joint venture between Belgium (Nuclear Research Center at Mol) and Germany (Nuclear Research Center at Karlsruhe) the fluoride volatility process is also being investigated (see paper to Session III of the Belgian Atom Fora).

1.2.5 Laboratory for Fuel Fabrication (Karlsruhe Nuclear Research Center)

The Laboratory for Fuel Fabrication was established at the Karlsruhe Research Center for investigating the various problems on fabrication of fast reactor fuel elements containing plutonium. It consists mainly of a glove-box room, a fissile material storage, and necessary accessories. All the necessary equipments and instruments for fabrication (mixing, grinding, sintering, pressing, vibration-compaction, etc.) and for general chemical and metallurgical investigations are inside the boxes. The transfer of active material is either carried out directly or through conveyor belts.

The total amount of active material is restricted to 1 kg Pu + U-235 in the working facility, and to 30 kg Pu + U-235 in the storage.

The programme of the laboratory comprises the production of fuel rod specimen for a fast breeder prototype and special fuel samples containing vibration-compacted fuel for irradiation purposes, as well as fundamental research work on chemical and metallurgical problems.

1.2.6 Fabrication and Refabrication

NUKEM, Wolfgang/Hanau

The firm NUKEM is, besides their regular activities, responsible for the development of different fabrication processes of fast breeder fuel elements. The main emphasis, however, is put on the technological development of different operations under inactive conditions with full-scale mock-up assemblies.

ALKEM, Leopoldshafen

The firm ALKEM is a subsidiary of NUKEM. Besides other fuel elements, it manufactures Pu-containing fast breeder elements based on processes developed by NUKEM. They are carrying out developmental work under active conditions in their well equipped glove-box set-up. The production facilities at ALKEM have a capacity of about 2 tons per year of mixed oxide containing Pu. It may be mentioned at this point that about 140 kg of fissile Pu have been fabricated to date - this means a very large amount handled by an industry in the private sector for peaceful purposes. All the production units are laid out for extreme precision work e.g. the square pellets produced by this firm for the SNEAK were made with tolerances of $\pm$ 40 μ with no PuO_2-particles larger than 200 μ. Because of the highly developed sintering techniques these tolerances could be achieved directly by sintering without any further polishing or grinding.

1.2.7 European Institute of Transuranium Elements (EURATOM, Common Research Center located at Karlsruhe Nuclear Research Center)

This EURATOM institute which is located at Karlsruhe center is a part of the common research center of EURATOM. The institute is working in processes for utilizing Pu as nuclear fuel in the framework of the common fast breeder program. In order to reach this aim, fundamental as well as applied research is conducted, above all in the fields of metallurgy, ceramics, physics, chemistry of Pu alloys and compounds. Besides the possibility of fabricating a fairly large number of fast reactor fuel elements, the institute is also in a position to prepare special fuel samples with pellets for the irradiation research. As special facilities there are a large number of glove-boxes for analytical purposes as well as for fabrication and 14 well equipped hot cells up to

50 000 Curie activity for technological metallography and chemistry. R.+D.work of this institute for the fuel element development of the fast breeder project is described in more detail in the German paper to Session II of this Congress.

1.2.8 Fuel Element Development and Manufacturing (AEG and KRT, Großwelzheim)

AEG has in collaboration with GfK the responsibility for the development of the fuel for the steam cooled breeder prototype. It has two installations for fuel fabrication: KRT a subsidiary of AEG is equipped for large scale commercial pellet fabrication and fuel element manufacturing. The development work for fuel elements is done in the nuclear laboratories of AEG in Großwelzheim. The facilities - a new section is under construction - allow the manufacturing of uranium fuel pins in laboratory scale and will give the possibility of Pu handling in 1968.

1.3 Test Facilities for Coolant Technology

1.3.1 Sodium Cooling

1.3.1.1 Sodium Test Loops (Institute of Reactor Components, Karlsruhe Nuclear Research Center)

The following loops have been built for studying special fundamental and applied problems of sodium cooling.

I. Fuel Element Test Rig. This rig has the following characteristics and consists of a test tank for a fuel element assembly, and all necessary accessories:

throughput	30 m^3/h
pressure drop	4,5 atm
electrical heating	280 kW
circuit temperature	400°C
temperature of test section	600°C .

This loop will start operation in 1967.

II. Sodium Corrosion Loops. In these relatively small facilities corrosion and mass transport characteristics of canning materials with different composition can be investigated. The characteristic data are:

a) Corrosion Loop

number of test sections	3
throughput	2,7 m^3/h
temperature in test section max.	650°C
circuit temperature max.	400°C
electrical heating per test line	12 kW

b) Mass Transport Loops

number of loops	3
throughput per loop	0,5 m^3/h
temperature in test section max.	600°C
temperature in loop max.	400°C
electrical heating per loop	3 kW .

Loop (a) has been in operation since 1966, loop (b) will go into operation in early 1967.

1.3.1.2 Sodium Testing Facilities (Institute of Reactor Development Karlsruhe Nuclear Research Center)

A. Testing of Moving Parts in Reactor Vessel

This set-up consists mainly of tanks and accessories and permits the testing of moving parts like manipulator arms, control-rod drives in an atmosphere of hot sodium. The highest attainable temperature is about 700°C. The set-up has a sodium purification system attached to it.

B. Test Facility for Investigating Boiling Characteristics of Sodium

In this closed circuit set-up provisions have been made for extensive investigations on the boiling, two-phase-flow, and superheating characteristics of sodium. The set-up has been laid out for a max. rod power of 500 W/cm^2 and a max. temperature of 900°C.

1.3.1.3 Hydraulic Loop (Institute for Neutron Physics and Reactor Technology, Karlsruhe Nuclear Research Center)

This test rig is used for investigations connected to steam and sodium cooling. The hydraulic loop is run in a closed circle. The maximum water flow is 360 $m^3\ h^{-1}$ using a pressure of 12 kp cm^{-2}. The pumping power is 125 kW and is cooled away in a heat exchanger using fresh water supply. The water flow is controlled by a bypass.

The experimental program is concerned with measurements of spacer resistance of spacer grids in quadratic and hexagonal arrays of rod bundles. Further, investigations are performed on pressure drop in homogeneous rod bundles with spiral spacers as a function of wire wrap pitch, ratio of centre distance to rod diameter (P/D-ratio) and number of rods in the bundle.

1.3.2 Steam Cooling

1.3.2.1 Experimental Test Facilities for Steam Cooling (Institute of Reactor Components, Karlsruhe Nuclear Research Center)

If steam is used as a coolant, the evaporation system of a fast breeder has to be laid out outside the reactor core. This system has been defined as the "Löffler circuit".

In the Institute for Reactor Components at the Karlsruhe Nuclear Research Center a number of test facilities are available for testing the different aspects of steam cooling of fast reactor systems.

A. Low pressure "Löffler circuit"

This circuit is mainly used for collecting data on the dynamic parameters of a closed Löffler-system. The circuit has the following characteristics:

Operating pressure	35 atm
saturated steam temperature	241^{o}C
superheat temperature	540^{o}C
electrical heating power	3 MW
saturated steam amount	17 to/h .

The two important components of this circuit are the Löffler-evaporator and the steam-compressor-turbine-unit. These two components can be tested within the closed cycle either separately or integrated in a common vessel. The dynamic studies carried out with this facility are checked and investigated by similating the operation in an analog computer.

B. A normal Benson boiler is installed to supply steam for separated test rigs. The steam may have up to 520^{o}C and 150 atm. The production rate is 7.5 to/h.

There are three separate test rigs for testing (a) Löffler-evaporation, b) spray evaporation, c) steam compressor turbine units with water-lubricated bearings, d) fuel element mock-ups for heat transfer, pressure drop, corrosion and vibration measurements.

Additional test set-ups allow for preliminary measurements with compressed air.

1.3.2.2 Corrosion Loop with Superheated Steam (Karlsruhe Nuclear Research Center)

A corrosion loop with superheated steam is being planned by the Institute for Reactor Components and will be operated by the Institute for Meallurgy and Solid State Physics at the Karlsruhe Nuclear Research Center. This loop will serve to test the corrosion of canning materials under conditions of a steam cooled fast breeder without irradiation but with heat transfer. It has been found that the corrosion is higher under heat transfer than under isothermal conditions.

The loop consists of a closed steam circuit with six parallel test sections each with an electrically heated tube (7 mm Ø, 1000 mm length). The characteristics of the loop are:

steam pressure	300 atm
steam throughput	2000 kg/h
steam exit temperature	540°C
max. rod-power	200 W/cm^2
max. clad-temperature	800°C
steam velocity	70 m/sec .

The corrosion loop will be in operation in 1968.

1.3.2.3 FR2 Contamination Loop (Karlsruhe Nuclear Research Center)

The problems of system contamination on account of canning material damage in a steam cooled direct circuit fast breeder can be investigated in this loop. It will consist of a fuel element rod with a controlled canning material damage.

The system pressure will be 180 atm. with a steam entrance temperature of 540°C. The system is being designed now.

1.3.2.4 MZFR Steam Loop (Karlsruhe Nuclear Research Center)

The possibility of creep collapsing of canning materials under the high pressure of the cooling medium in a steam cooled fast reactor system has to be investigated thoroughly. For this purpose an in-pile steam loop has been planned for the MZFR Karlsruhe. The loop will have the following operating data:

power in the test section	500-600 kW
max. operating pressure	180 atm
steam temperature	360-540°C
steam throughput per test section	4000 kg/h
number of irradiated rods	8-10 per test section.

The creep collapsing is influenced by the strength of material, temperature, outside pressure, fission gas pressure build-up, swelling of the fuel, and the in-pile time. All these parameters will be tested in this loop, in addition to corrosion studies.

The loop is under design.

1.3.3 Helium Cooling

1.3.3.1 Test Rig for Gas Technology (Institute for Neutron Physics and Reactor Technology, Karlsruhe Nuclear Research Center)

A test rig has been installed for helium technology. The main data of the test rig are: pressure 50 atm, maximum temperature 525°C, heat exchanger 700 kW, mass flow of blower 1 kp sec^{-1} (power 180 kW), blower inlet temperature 300°C, maximum power at test section 600 kW, test section length 2.8 m, and diameter 120 mm.

The experimental program is concerned with the measurement of heat transfer coefficients and pressure drop coefficients by use of rod bundles in a square array.

2. Test Facilities for the Industrial Program for Fast Reactor Prototype Development

2.1 Sodium Cooled Fast Breeder Prototype Development

2.1.1 5 MW Sodium Test Facility (INTERATOM, Bensberg)

The 5 MW Sodium Test Facility has been installed within the R.+D. program of the KNK at Bensberg and has been operated successfully since early 1965.

All the heat transfer systems in this facility represent those of a sodium-cooled reactor. The heat source is an oil-fired boiler. The flow-rate of the loop is about 120 t/h sodium. The maximum sodium temperature is 580°C. The primary sodium loop transfers the heat through an intermediate heat-exchanger to a secondary sodium loop. This loop is cooled by a sodium heated steam generator which produces 7,5 t/h steam at normal large boiler conditions (510°C, 80-100 atm).

The test facility can be used in a most flexible way as all main components can be by-passed in order to maintain or exchange them, without having to shut down the system.

The test program for this facility in the scope of the fast breeder program will comprise tests of control devices and of component models, especially steam generators, cold, hot, and vapor trap plugging meters, fuel-element handling and pump bearings. Moreover, shock-tests can be carried out. This year the water steam system is to be modified to match steam pressure up to 200 atm and temperature up to 540°C.

2.1.2 Large Sodium Pump Test Rig (INTERATOM, Bensberg)

A very large test facility for sodium pumps will be installed at Firma INTERATOM, Bensberg, in order to have the opportunity of testing pumps up to 15 000 m^3/h pumping capacity. Such large pumps will be necessary for fast breeder power plants of 1000 MWe. The operation characteristics of such pumps can only be obtained in full scale experiments over an extended period of time. However, this set-up will be used first to test the prototype reactor pumps in full scale, i.e. about 5000 m^3/h of sodium.

The rig will have a cooling device for 5 MW, in order to remove the heat produced by friction in the loop. Provision will be made for shock tests up to 5°C/sec over a range of 200°C to be carried out in the test rig. The piping is expected to have about 900 mm diameter.

Start-up of operation is planned for 1968/69.

2.1.3 Experimental Facility for Safety Technology (Sodium Water Reactions) (INTERATOM, Bensberg)

Of great importance for sodium-heated steam-generators are the safety devices for release of the products of a sodium-water reaction to the atmosphere without danger. A test facility has been installed at Bensberg within the framework of the KNK project in order to study the basic problems of Sodium water reactions in pipes and to obtain experimental data for the design work for steam-generators. This test facility will be used for the fast reactor project as well. But on the basis of the large amount of experience obtained with this facility, a larger test set-up for the investigation of sodium water reactions is planned to be in operation in 1968. This large test facility will consist of a complete sodium loop, separate test sections, and storage tanks for sodium and high pressure (280 atm, 370°C) steam vessels. The maximum filling capacity of the test section will be 500 kg Na. The electrical power consumption will be around 600 kW. The hot sodium will be pumped to the test section by a standard sodium pumps. The reaction products which will be obtained in large quantities have to be separated in special devices.

The test program comprises

- effects of pin-holes
- measurements of the pressure and temperature tansients during the reactions
- reliability of the pressure release system and separation devices.

2.1.4 General Test Facilities for Sodium Technology (INTERATOM, Bensberg)

A number of test facilities for sodium technology are installed or will be installed at the Bensberg site of INTERATOM. A general description of these set-ups is given below. The largest of these test facilities will be a reactor mock-up of the sodium-cooled prototype to test in full scale all refueling and control components and processes over extended periods of time, especially the rotating top shield. In other smaller facilities a great number of special tests will be carried out (and has already been carried out for the KNK project) in order to check and improve components and instrumentation.

Some special problems regarding electrical drives of components (e.g. valves) or the condensation mechanism of sodium vapor or sodium oxide, will also be investigated.

The test program for these detailed investigations is at the moment part of the R.+D.work for the sodium-cooled prototype, but will be extended as much as necessary for the demonstration reactor R.+D.work.

2.1.5 KNK (Kompakte Natrium-gekühlte Kernenergieanlage)

This sodium-cooled thermal test reactor is being constructed at KFK by the firm INTERATOM, Bensberg, and is located at the Nuclear Research Center Karlsruhe.

The characteristics are:

thermal power		58 MW
electrical power		20 MW
fuel UO_2		(6% enriched)
canning material		SS 1613
moderator		zirconium hydride
coolant temperature	primary	$550^{\circ}C/360^{\circ}C$
	secondary	$540^{\circ}C/310^{\circ}C$
steam conditions:	pressure	80 atm
	temperature	$505^{\circ}C$.

The functions of this reactor in the framework of the fast breeder project are:

a) to operate as a first step for providing experience on sodium technology under reactor operating conditions,

b) to serve as a complete set-up for fuel irradiation experiments also under reactor operating conditions.

KNK is scheduled to start operation at the end of 1969.

Literature

Atomwirtschaft, Jg. XII, No.7, 3/9 (1966).

2.1.6 Test Set-up for Fuel Elements and Control Rods (Siemens AG., Erlangen)

A special facility for the out of pile testing of fuel elements and control rods will be installed at Siemens research center at Erlangen. Mechanical, hydraulic, and thermodynamic investigations will be performed with an hexagonal arrangement of up to 7 elements. The maximum sodium temperature will be $600^{\circ}C$, the flow rate 500 m^3/h.

2.2 Steam Cooled Fast Breeder Prototype Development

2.2.1 Test Facilities for Testing Components of the Steam Cooled Fast Breeder Prototype

For the development of a steam cooled fast breeder it is essential to test the more important reactor components and accessories under actual operating conditions. Besides the fuel element development they require the largest amount of R.+D.work. Since AEG is responsible for the development of the steam cooled breeder prototype this company is installing a highly versatile test set-up. The main tests during the various development stages of the blower and the steam generator respectively will be performed in the modern test facilities of GHH which will build the prototype of the blower-turbine combination and in the especially arranged set-ups for the steam generator tests with MAN which is developing in the steam generator. Final tests and other experiments will take place at the AEG site in Großwelzheim.

1. Facilities at GHH, Sterkrade

Within the scope of circulator development and testing carried out at Gutehoffnungshütte AG. (GHH), optimum and functional tests are being prepared on the following test stands:

a) Optimum tests on circulator impeller and diffusor

Flow tests with air at atmospheric pressure with allowance for the Reynold-analogy are conducted on available test stands of 500 and 1000 kWe, respectively. These tests serve to obtain the most favourable impeller diffusor combination coupled with the highest efficiency and partial load behaviour available.

b) Functional tests on water lubricated bearings

These tests are conducted as preliminary tests partially at an already existing bearing test stand. Various axial thrust forces are pneumatically simulated. Further experiences on bearings will be performed in conjunction with point c) and d).

c) Joint tests of circulator and turbine at reduced steam pressure. The tests are based on steam conditions available at GHH test stands, (35 atü, 350^{o}C). The complete aggregate can, therefore, be tested at a reduced pressure and similar flow conditions. The water pressure of the bearings will at the same time be adjusted in accordance with this pressure level.

d) Testing of steam circulator and water lubricated bearings at full pressure and maximum temperature

At a further test stand the turbine will be replaced by an available 5 MWe motor drive which renders the possibility of testing the circulator at 145 ata and saturation temperature. The circulator will at the same time be operating in a closed circuit in which the driving power will partially be converted to compression power. The desired circuit pressure will be attained by water injection and steam discharge.

2. Facilities at MAN, Nürnberg

For the steam generator tests Maschinenfabrik Augsburg-Nürnberg (MAN) has facilities at its Nürnberg site for the following purposes:

a) Screening test at low pressure with water steam mixtures in various geometries of steam generator internals.

b) In spring 1967 a test stand will go into operation for more detailed tests at 64 atmospheres and 500°C steam temperature. The available steam amount is 15 t/h.

c) 1968 a facility for 160 atm pressure and 500°C at a steam flow of 10 t/h will be available.

These loops will be used for the testing of three different types of steam generators: an advanced Löffler boiler, a desuperheater type in which water is sprayed into the superheated steam, and a cyclon water-steam mixer.

Further facilities allow the measurements of heat transfer quantities from electrically heated pins to steam up to highest power densities with a capacity of 800 kW electric power.

3. Facilities at AEG, Großwelzheim

There is a steam generator under design with the following variable capacity:

pressure	50-170 ata
temperature	260-520°C
steam flow	2.5 - 25 t/h superheated 4 - 40 t/h saturated.

A cooling loop for 20 MW will be used to dissipate the energy of the steam and of the 5 MW dc power supply.

With this steam and electrical supply several test stands will be run. After the development stage mentioned under 2. the steam generator will be tested in larger scale, especially under transient conditions. Another set-up will be used for a final testing of the blower-turbine combination up to 70% to full turning speed with full temperature and pressure. An electrically heated fuel bundle mock-up will allow extended measurements of overall heat transfer coefficients, pressure drop and mechanical behaviour of the multirod arrangement with various spacer designs.

2.2.2 Turbine Model for Checking the Deposition of Fission- and Corrosion-Products (Firma AEG, Frankfurt)

An important safety and maintenance problem in steam cooled fast reactor systems with open cycle lies in the fact that fission products may get deposited on turbine components or other parts of the system in case of a fuel element leakage and because of the presence of activated corrosion products. In order to assess the magnitude of these problems a small turbine in a special loop will be attached to the already existing superheat loop of the Kahl BWR.

The turbine is expected to be in operation in 1968.

2.2.3 VKL (Versuchskreislauf)

Since 1963 a loop for testing thermal superheat elements has been in operation in the Kahl boiling water reactor. The loop which has a cooling capacity of 18 MW allows 4 different test elements to be installed in the core and controlled separately.

For tests of steam cooled fuel pins of fast breeder dimensions and linear rod powers up to 500 W/cm in thermal neutron spectrum two fuel pin arrangements are under construction.

Literature

Kerntechnik 6, p.303-311 (1964).

2.2.4 HDR (Heißdampf-Reaktor)

The HDR is a thermal combined boiling water-superheat reactor with two coolant circuits in series. The reactor is being built at Kahl near Frankfurt by the Allgemeine Elektricitäts-Gesellschaft under contract of Gesellschaft für Kernforschung Karlsruhe. Its important characteristics are:

thermal power	100 MW
electrical power	25 MWe/35 MWe
fuel UO_2	2.7 % enriched
steam conditions (primary circuit):	pressure 60 atm temperature 500°C .

The function of this reactor may be itemized as follows:

a) To obtain complete information and experience on the construction and operation as a first step to master full scale superheated steam cooling of reactors.

b) Redesigning a part of the core as a fast zone in order to obtain extended experience on a fast steam cooled system.

c) To irradiate fuel element under proper fast reactor conditions.

It is possible to obtain valuable information on fuel irradiation in fast steam cooled systems which is of cardinal importance for the overall development of this type of reactors.

Literature

Atomwirtschaft, Jg. XI, No.6 (1966).

Third FORATOM Congress, London

INDUSTRIAL ASPECTS OF A FAST BREEDER REACTOR PROGRAMME

24 - 26 April, 1967

Session IV German Report

Prototype Designs

D.Smidt

1. Sodium Cooled Prototype Reactor

   Gesellschaft für Kernforschung, Karlsruhe (GfK) *)
   Interatom, Bensberg (IA)
   Siemens AG, Erlangen

2. Steam Cooled Prototype Reactor

   Gesellschaft für Kernforschung, Karlsruhe (GfK) *)
   Allgemeine Elektricitäts-Gesellschaft, Frankfurt (AEG)
   Gutehoffnungshütte, Sterkrade (GHH)
   Maschinenfabrik Augsburg-Nürnberg (MAN)

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*) In Association with EURATOM

Introduction

According to the German Fast Breeder Program [1] both the sodium and the steam cooled fast breeder will be developed in parallel.

This paper describes the main features of the preliminary designs as of October 1966. Work is still in progress and some of the design features may be subject to changes before completion of the current studies. Alternate designs which are being considered will not be discussed in this paper.

Both conceptual design studies will be finished in 1967, start-up of the prototype reactors in 1973 is anticipated. The whole work will be a joint effort of the Nuclear Research Center Karlsruhe and the industrial groups Siemens/Interatom (sodium cooled version) and AEG/GHH/MAN (steam cooled version) in liaison with Belgian and Dutch firms as already mentioned in the paper to Session I.

[1] "Fast Reactor Development Objectives"
Paper to Session I of this Congress.

1. The Sodium Cooled Prototype Reactor

1.1 Main Design Objects

The prototype design is based upon the principle that the prototype reactor should provide - at reasonable cost - the information required for installation and operation of a commercial fast reactor power plant of about 1000 MWe in the late 1970's or early 1980's. After considering the uncertainties in the extrapolation of today's technology to a large scale power plant mainly in the development of high heat flux and high burnup fuel elements, large sodium components and the engineering problems of the reactor a power output of 300 MWe was chosen as a reasonable intermediate step. Preliminary analysis indicates that a 300 MWe prototype reactor can be built at reasonable capital costs and with fuel cycle costs which are well competitive with other power plants.

A considerable degree of conservatism has been applied to the 300 MWe conceptual design because of the uncertainties which still exist in engineering, safety and physics. In particular a very conservative approach was taken in the hazards evaluation leading to the inclusion of some safety design features which are not likely to be required in future plants.

Hence, the prototype reactor will not necessarily be typical of the 1000 MWe plant in all respects. However, it will incorporate most of the basic design features of the large reactor established in the previous 1000 MWe design studies /¯1,2_7, such as:

Ceramic fuel
Fuel pin bundles in hexagonal wrapper tubes
A core of 95 cm height
Two core zones of different enrichment
Upward flow in parallel through core and blanket
A core pressure drop around 3 at.
A multiloop two stage sodium coolant system
Sodium temperature at reactor outlet 560°C to 580°C
Steam conditions in the range of 170 at and 520°C to 540°C.

1.2 Description of the Design

1.2.1 Core and Reactor Vessel

The core is divided into two radial zones of different enrichment (fig.1). The inner zone consists of 79 fuel elements and 6 control elements, the outer zone consists of 72 fuel elements and 12 control elements. The core is surrounded by a 50 cm thick radial blanket and shielding annulus. Each fuel element contains 169 fuel pins in a hexagonal array. The spacing of the pins is achieved by means of honeycomb grids. The axial distance of the grids was established such that coolant flow induced vibrations and unstable thermal bowing will be prevented. The pin diameter of 6.0 mm was selected after a detailed study of fuel cycle economics and design feasibility. This value turned out to be the optimum choice with respect to a whole generation of sodium cooled fast breeders rather than the optimum with respect to the single reactor [3]. The fuel pins contain a 65 cm long fission gas reservoir at their lower end, a 95 cm long fuel section and two blanket sections of 40 cm length each. The fuel clad thickness is 0.38 mm.

The fuel elements are supported by a lower grid plate. Hydraulic forces are balanced by means of an individual low pressure chamber below the foot of each element. The upper end of the fuel elements is guided in an upper guide plate which itself is positioned horizontally with respect to a rigid support shroud surrounding the radial shielding. The guide plate also provides a positive stop against fuel elements being expelled from the core after failure of the hydraulic holddown. Besides, this plate serves as a control rod guide and contains thermocouples above each fuel element position. Wear faces attached to the hexagonal wrapper tubes of the fuel elements are arranged such that the core will deflect into a less reactive configuration under the influence of radial temperature gradients.

The size of the prototype fuel element is below the optimum value for a large scale reactor. The smaller size was chosen as a compromise with respect to thermal stresses and forces, fuelling time, improved thermal performance due to coolant flow adjustment according to the power profile and available out-of-pile test facilities.

The core support shroud is surrounded by an annular arrangement of spent fuel elements which have been removed from the core and are stored for afterheat decay before they are removed from the vessel. The fuel clad and the whole internal structure of the core is made of X8 Cr Ni Mo V Nb 16 13 steel.[*]

The coolant flow enters the bottom of the reactor vessel, flows upward through core and blanket to an upper plenum and leaves the reactor vessel through three nozzles (fig.2). The reactor vessel is approx. 15 m long and 5 m in diameter and is made of X8 Cr Ni Nb 16 13[**] steel, since the German codes do not allow the use of 304 type steel so far. Besides this, the use of Nb-stabilized steel 10 Cr Mo Ni Nb 9 10[***] is under consideration.

1.2.2 Primary Coolant System

Several concepts have been analysed for the primary coolant system. A loop type design was chosen for Na-2 (fig.3), however a pool design will be further investigated as an alternate concept. Detailed stress analyses indicate that thermal expansion of the system can be compensated by two- or threedimensional loops between the reactor vessel, the pumps, and the heat exchangers.

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*) X8 Cr Ni Mo V Nb 16 13 contains:

< 0,1 % C	16 % Cr
13,0 % Ni	1,35 % Mo
1,25 % Mn	0,75 % V
≤ 1,2 % Nb	0,45 % Si

**) X8 Cr Ni Nb 16 13 contains:

< 0,1 % C
13 % Ni
≤ 1,2 % Nb
16 % Cr
1,25 % Mn
0,45 % Si

***) 10 Cr Mo Ni Nb 9 10 contains:

< 0,1 % C	0,55 % Ni
2,25 % Cr	0,6 % Mn
1,0 % Mo	≥ 1 % Nb

Three main coolant systems and two auxiliary coolant systems were chosen for the design. This arrangement provides adequate heat removal capability in all normal operating and shutdown conditions. Failure of one out of three pumps is not expected to necessitate immediate shutdown of the reactor. After failure of one loop the reactor may continue to operate at reduced power.

The hot sodium leaves the reactor vessel from the upper plenum and flows into the shell side of the intermediate heat exchanger. The pumps and throttling valves are located in the cold legs of the loop systems.

The reactor cover gas is maintained at a slightly elevated pressure in order to provide the necessary net positive suction head to the pumps.

Throttling valves are located between the pumps and the reactor in order to assist flow control by pump speed variation in each loop individually during normal operation and after pump failure as well as during forced or natural convection decay heat removal.

All primary circuit components are made of X8 Cr Ni Nb 16 13 steel.

1.2.3. Fueling

Various fuel handling principles have been analysed. A final choice had to be made between a system consisting of three manipulators attached to a stationary vessel lid and a rotating plug system. Although the manipulators apparently have the greater potential for development of onload refueling and increased plant availability, the choice was made in favor of a rotating plug system because of the better known technology and because of the possibility to position a repair access hole above any position of the reactor assembly.

A system of three rotating plugs seems to be the most economic with respect to fuel handling time and reactor vessel size.

Handling of spent fuel elements will be performed under sodium. After removal from the core position the elements are lowered into the decay storage annulus surrounding the blanket from where they will be removed through the vessel head after the next reactor cycle. The decay heat is removed by natural convection. The blanket elements are removed from the vessel without intermediate storage.

1.3 Safety Assessment

Reactivity excursions are controlled by limiting the maximum rate of reactivity insertion to a value that can be controlled by the period, flux and temperature trips of the safety system without exceeding the design limitations of the core and primary system.

Undetected criticality during fuel handling is discounted because of the excessive shutdown capability of the control rods.

Coolant flow failure in one of the three loops does not cause an unacceptable temperature transient and is not expected to cause a safety trip. Coolant flow failure in either two or all loops will trip the reactor.

In this case the inertia of the coolant system and emergency cooling by the two auxiliary loops will limit the temperature transient to below the design value.

Coolant flow blockage in single fuel elements cannot be discounted so far. However, sodium boiling or gross fuel slumping would not cause an unacceptable reactivity excursion provided that the failure does not propagate rapidly throughout the core. There is reasonable confidence that failure propagation can be precluded. Experiments which are to confirm this confidence are under way.

Loss of sodium from the core is discounted because the flexible design of the coolant system will produce rather low combined stresses. In the reference design there are no hot nozzles in a cold vessel-wall or vice versa. The design of the reactor cavity and the primary system cells is such that even gross leakage of sodium from the system will not uncover the core nor the auxiliary coolant systems from sodium.

Despite the excessive safety provisions designed into the reactor it was decided that the reactor should be placed into a double containment designed against the potential consequences of an assumed destructive nuclear excursion. This decision was made uniquely for the prototype reactor because of the lack of experience with large fast reactors which will still exist in the early 70's, especially with respect to the reliability of engineered safeguards and the possibility of fuel failure propagation.

On the basis of a conservative analysis of hypothetical failures - such as guillotine pipe rupture, complete flow blockage of the entire core, loss of sodium from the core and gross overpower coinciding with complete failure of the safety system - it was concluded that a nuclear excursion followed by rapid vaporization of sodium in intimate contact with vaporized and molten fuel would not release more destructive energy than the "design basis accident", namely 1000 MWsec. The reactor cavity and the vessel head are conservatively designed to withstand the destructive potential of waterhammer effects, shock waves and internal blast pressure of the design basis accident.

The double containment is realized by means of a steel clad concrete structure containing the reactor cavity and the entire primary system completely enclosed in a conventional steel containment building. The inner (concrete) containment is filled with an inert atmosphere to prevent sodium fire.

The reactor cavity is covered by a leak tight lid during reactor operation which is removed after shutdown for access to the rotating plug system for fuel handling and maintanance. The integrity of the containment is not affected by the design basis accident. The possibility of a significant sodium fire inside the containment is so remote that the containment has not to be especially designed for sodium fire conditions. However, the design criteria imposed upon the containment by the design basis accident will ensure that the containment can withstand large sodium fires.

Special provisions have been made to maintain longterm decay heat removal even after destruction of the core and the reactor vessel. Both auxiliary coolant loops are designed for decay heat removal by natural circulation. In addition three natural convection NaK loops are embedded in the wall of the reactor cell, protected against the blast effects of the design basis accidents to take care of the decay heat if the auxiliary systems should fail. The NaK loops transfer the heat to a secondary NaK loop which is led out of the containment.

Preliminary estimates indicate that the cost of the design requirements of the design basis accident will not excessively increase the total capital costs of the plant.

References

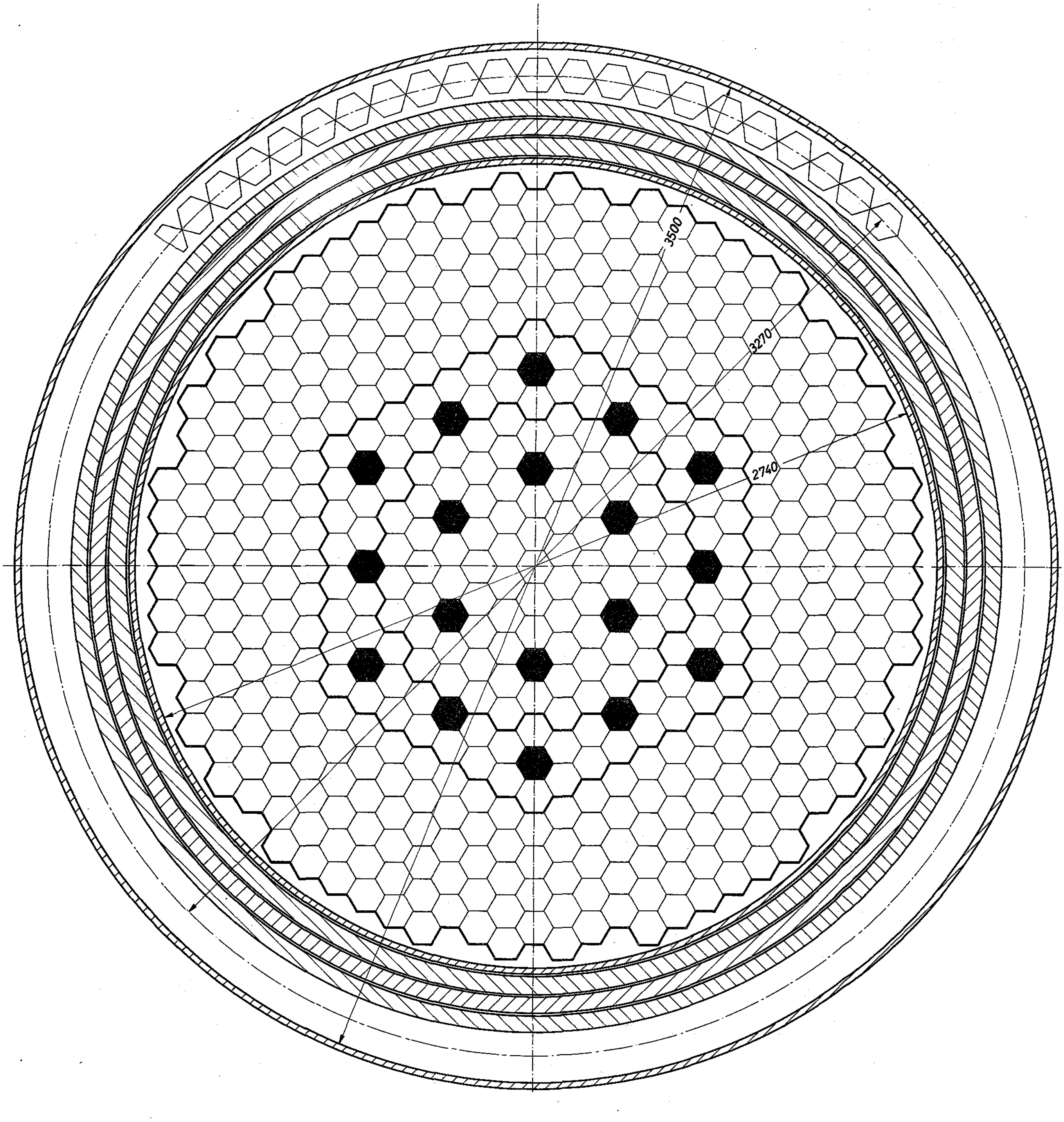
[1] D. Smidt, A. Müller et al.: Referenzstudie für den 1000 MWe natriumgekühlten Schnellen Brutreaktor (Na 1). KFK 299, Dec. 1964

[2] D. Smidt et al.: Safety and Cost Analysis of a 1000 MWe Sodium Cooled Fast Power Reactor. PSB 152, Paper Conf. on Safety, Fuels and Core Design in Large Fast Power Reactors, Argonne, Oct. 1965

[3] H. Grümm, D. Gupta, W. Häfele, P. Jansen, E. Schmidt, J. Seetzen: Kernbrennstoffbedarf und Kosten verschiedener Reaktortypen in Deutschland. KFK 366, Sept. 1965

List of Main Parameters

Net electrical power	300 MW
Thermal output	750 MW
Primary sodium inlet temperature	380°C
Primary sodium outlet temperature	580°C
Primary coolant flow	3 to/sec
Steam temperature	520°C - 540°C
Steam pressure	170 at
Core diameter	153 cm
Core height	95 cm
H/D-ratio	0,62
Number of core zones	2
Core volume	1,75 m^3
Core composition:	
coolant	47 %
structural material	20,6 %
fuel	29,8 %
helium	2,6 %
Fuel pin outside diameter	6,0 mm
Fuel clad thickness	0,38 mm
Number of pins per fuel element	169
Distance across flats of fuel element	110 mm
Number of fuel elements	151
Number of control elements	18
Maximum linear power	467 W/cm
Average power rating	1 MW/kg fissile
Average enrichment	18,4 % Pu 239 + 241
Weight of Plutonium in core	982 kg
Weight of fissile material in core	760 kg
Weight of mixed oxide in core	4774 kg
Height of axial blanket	40 cm
Thickness of radial blanket	50 cm
Internal breeding ratio	0,64
Total breeding ratio	1,3
Doppler constant (= -Tdk/dT)	$3,86 \cdot 10^{-3}$



Na 2 Cross Section
Reactor Core Arrangement

Fig. 1

Na-2

Vessel and Internals (Schematic)

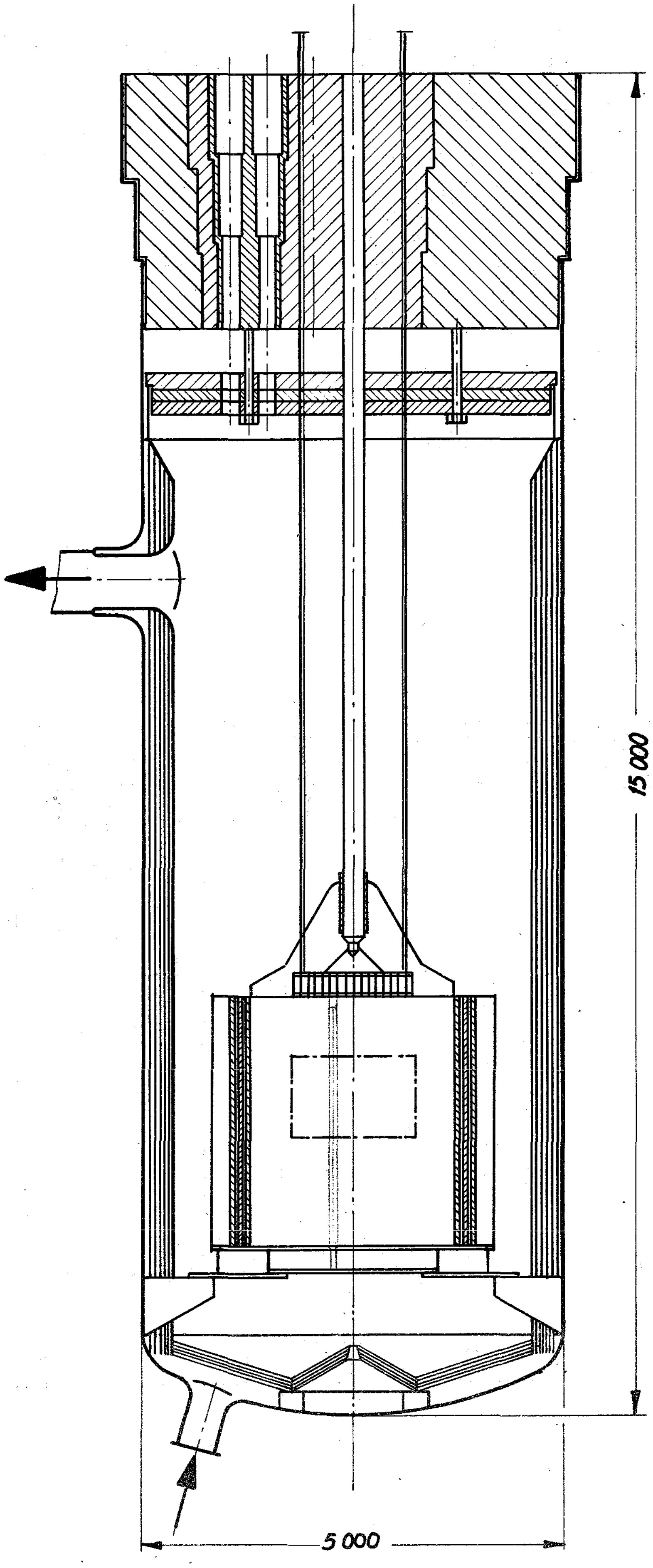


Fig. 2

Primary System and Containment (schematic)

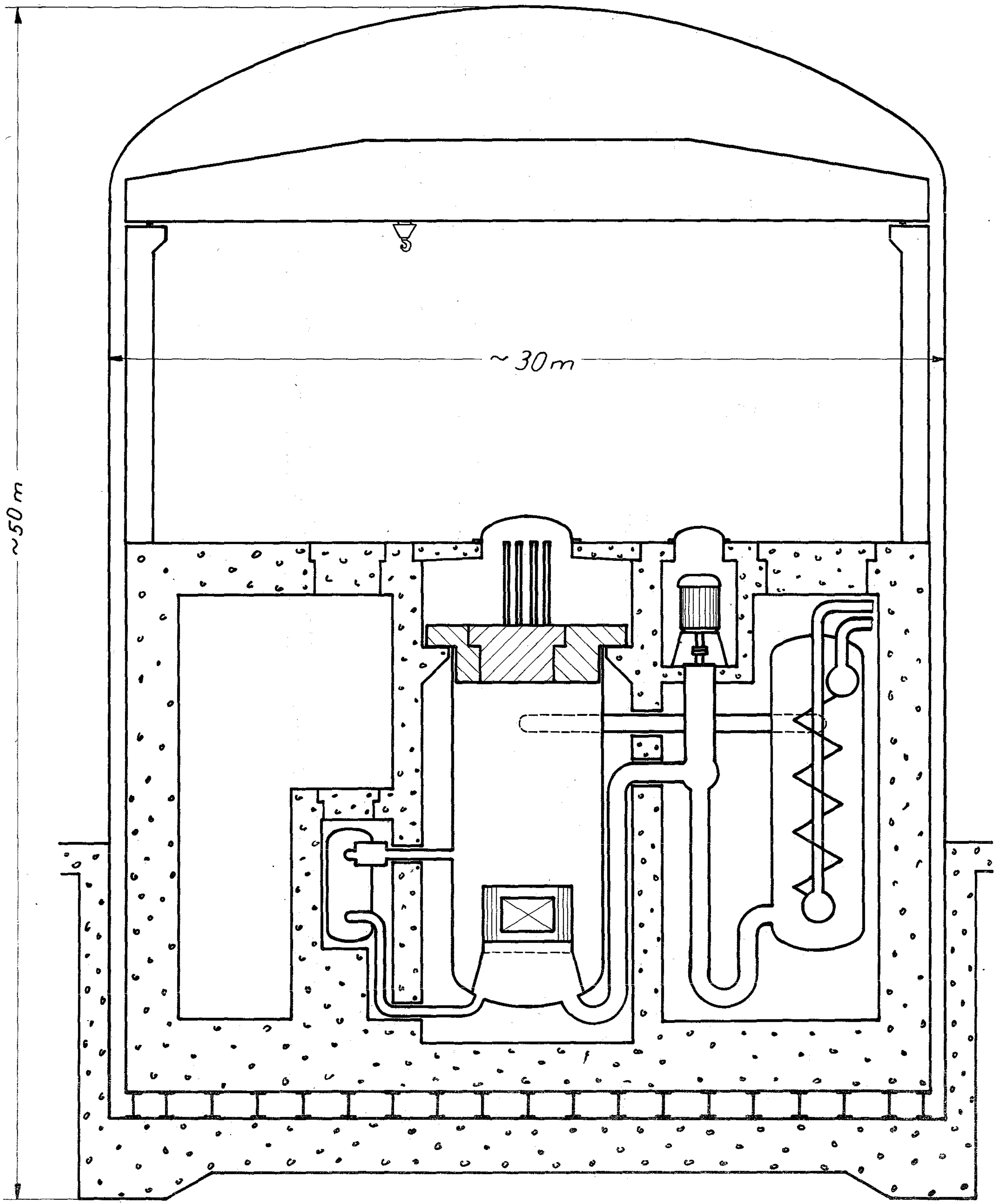


Fig. 3

2. The Steam Cooled Prototype Reactor

2.1 Overall Design Criteria

The main general objective for the prototype design is to provide all information necessary for the construction of large fast steam cooled reactors. [1,2]

Furthermore it is the aim of the prototype to demonstrate the integrity, the operability, the dynamical behaviour and the breeding potential of the concept, to gain experience with large components and to prove the predicted physical behaviour.

To meet these requirements the size of the prototype should not be too much lower than 300 MWe. Of course, the higher the power level, the easier it would be to extrapolate physical data, as for instance burn-up and critical enrichment. Otherwise, the first step in this line should not be too big. Therefore, a design size of 300 MWe was chosen. The system components will be built in rather large size to have a sound basis for extrapolation to larger stations. This means, the number of parallel circuits will be only four in comparison to 6 for a 1000 MWe plant.

As fuel a UO_2/PuO_2 mixture will be used. The fuel elements, containing a set of numerous fuel pins installed in a hexagonal wrapper tube, are similar to those planned for large size reactors.

For the core a cylindrical shape with a H/D-ratio of 0,66 was chosen. Consequently the total loss of coolant will result in a relatively high positive reactivity. We don't feel this will give any unusual problems if we make sure, that total loss of coolant will need some seconds under all circumstances. Then the produced reactivity ramp will not be excessive.

Light water steam at a pressure of approx. 160 at and a suitable outlet temperature of 490 degree celsius is used as coolant. The plant will operate in direct cycle with reheat of the expanding steam. With this cycle an overall efficiency of approx. 39 percent is achieved. With this system we expect to be able to overcome the system contaminations problems. The indirect cycle has been studied and was discarded on account of capital cost considerations.

A unique feature with operational fluctuations is the steam density coefficient of reactivity. Owing to the effect of fission products this coefficient becomes more and more negative during burn-up. Nevertheless this swing is tolerable if only a part of the whole core loading is exchanged each time. The change in steam density will influence the reactivity as well as the coolant conditions. Preliminary calculations for such incidents indicate that no unusual problems are to be expected from this behaviour. Nevertheless, its role in dynamic behaviour is given intense attention.

As in other fast reactor concepts, safety rests on the negative Doppler coefficient as a fast acting reactivity feedback.

Until now the exact mechanism of the design basis accident has not yet been defined. It is expected that not too much additional expenditures are required for blast shields or other heavy equipment to withstand the consequence of a nuclear explosion.

2.2 Core and Reactor Vessel

The thermohydraulic and physical parametric studies and the whole design work are still in progress and no final decisions have been made yet. So only preliminary results can be given.

The cylindrical core and blanket arrangement consists of 82 fuel subassemblies, 9 control subassemblies and a suitable number of blanket subassemblies of identical hexagonal cross section. The central core region contains the fuel in two enrichment zones, the outer region the fertile material. The elements rest on a bottom structure with a top grid plate holding them in proper spacing.

The dimensions of the core are: 1,12 m height; 1,7 m diameter. The fuel pins have an outside diameter of 7 mm, cladding material is Inconel 625. The maximum linear rod power with hot channel factors amounts to 490 W/cm, the maximum cladding temperature to 650°C.

The core is installed in a steel pressure vessel with an overall height and diameter of 9,8 by 3,5 m and a wall thickness of 170 mm. As steel "20 Ni Mo Cr 36" will be used. The vessel is equipped with concentric coolant pipes penetrating the outer concrete shielding.

Saturated steam enters the pressure vessel through the annular channels of the concentric studs and is flowing first through the radial blanket and through the inner radiation shield. Here any residual moisture in the coolant will be evaporated. Then flowing through the core the steam is heated up to its final temperature. It leaves the reactor pressure vessel through the inner channels of the concentric studs. With this design the heavy loaded vessel wall is in contact only with the entering saturated steam. Therefore, the pressure vessel is kept on low and fairly constant temperature. Around the blanket and core array there is an internal thermal shield and possibly a closed water-filled annulus vented to the main coolant system.

Some additional equipment is provided on the reactor to carry out the flooding and draining process.

2.3 Coolant System

The reactor coolant system uses the Loeffler cycle. No decision has been made whether the main steam generator will be of the injection-type or of the Loeffler-type. Together with the injection type steam generator one or more additional smaller Loeffler boilers could be used to act as buffer and storage in case of power transients, equipment failure, or arising leakages. For the flooding process after reactor shut down a ruths storage tank will provide the required water volume at high temperature and pressure.

The steam circulators will be of the radial compressor type with a speed of approx. 10.000 rpm, driven by a centripetal type steam turbine housed compactly together.

2.4 Fueling

For fueling the reactor pressure vessel will be flooded with water and be depressurized. The pressure vessel cover is taken off and some internal parts are removed. This gives access and direct view to the fuel and blanket elements. The transport of the elements is carried out under water, so that no additional provision for decay heat removal is required. The spent fuel elements are discharged into a fuel element storage bay outside the containment.

References:

/1/ A. Müller et al.: Referenzstudie für den 1000 MWe dampfgekühlten schnellen Brutreaktor (D 1). KFK 392, August 1966

/2/ A. Müller, F. Hofmann, E. Kiefhaber, D. Smidt:
Design and Evaluation of a steam cooled fast breeder reactor of 1000 MWe.
London Conf. on Fast Breeder Reactors, May 1966, Ref. 1/6.

List of Main Parameters

Net electrical power	300 MW
Thermal output	765 MW
Reactor inlet pressure	160 ata
Reactor inlet temperature	350°C
Reactor outlet pressure	148 ata
Reactor outlet temperature	490°C
Reactor coolant flow	1,2 t/sec
Core diameter	170 cm
Core height	112 cm
H/D-ratio	0,66
Core volume	2,5 m^3
Core composition:	
coolant	36 %
structural material	22 %
fuel	37 %
control-rods	5 %
Number of core zones	2
Fuel pin outside diameter	7,0 mm
Fuel clad thickness	0,4 mm
Number of pins per fuel element	331
Number of fuel elements	82
Number of control elements	9
max. linear power	490 W/cm
max. cladding temperature	650°C
max. power rating	197 KW/kg (U + Pu)
Average enrichment	14 % Pu 239 + 241
Isotopic composition of Pu	63 % Pu 239
	30 % Pu 240
	5 % Pu 241
	2 % Pu 242
Blanket thickness	25 cm
Core conversion ratio	0,80
Total breeding ratio	1,12
Doppler constant (= -Tdk/dT)	$1,3 \cdot 10^{-2}$

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INDUSTRIAL ASPECTS OF A FAST BREEDER REACTOR PROGRAMME

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Session VI German Report

European Fast Reactor Programmes and Economics

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1. Cost of Fast Reactor Power

1.1 Capital Cost Estimates (Direct Costs Incurred by the Contractors)

Detailed cost estimates have been carried out for 1000 MWe fast breeder reference design studies. The cost analyses for the fast breeder prototypes which are now being planned are in progress. It may, however, be noted that the energy cost for prototypes would not represent those for large scale fast breeders as the prototypes would not be operating under optimum economic conditions.

The cost data for the reference design studies as published [1,2] which are summarized below (Table I) represent consistent estimates on the basis of careful designs (although not necessarily complete in all details). The cost data are from the suppliers of components although no formal bids were obtained from them. For this purpose unit prices for individual items were obtained and the total costs were calculated on that basis. These unit prices naturally include besides the raw material and operation costs all the overheads and contingencies of the suppliers. These cost items, however, do not include the expenses for the fast breeder development as the major part of it is borne by the authorities.

The comparison of the two sets of direct investment costs as given in Table I indicates a certain increase in the cost of the conventional parts for the D-1 type over those for the Na-1 type. This is on account of the detailed knowledge gained during the preparation of the second study. A careful analysis of these two sets of cost figures shows that a saving of about 30-40 Mio DM in the direct costs for D-1 would be probable. On the basis of calculating methods developed in [4] in which a part of the contingencies added to the direct costs in Table I, are included under indirect cost, the total direct costs for the two breeder types come out to be:

Direct Cost Estimates K_D for 1000 MWe

Sodium Cooled Fast Breeder [4]	416 Mio DM
Steam Cooled Fast Breeder	386 Mio DM

Table I

Comparison of Direct Cost Estimates of two Reference Studies of a 1000 MWe Sodium Cooled Fast Breeder (Na-1) and a 1000 MWe Steam Cooled Fast Breeder (D-1)

	Na-1 (KFK 299)		D-1 (KFK 392)	
	(Mio DM)			
Land + site improvement	3,00		6,12	
+ contingencies				0,63
Reactor (excl.first core) incl.vessel, control system, fueling system, cooling circuits, emergency cooling, accessories	143,00		121,82	
+ contingencies		57,00		48,38
Reactor-buildings	54,40		44,52	
+ contingencies		7,60		6,78
Power-plant	157,60		166,00	
+ contingencies		15,00		23,30
Power-plant-buildings	22,00		30,80	
+ contingencies		3,00		5,40
T o t a l Direct plant cost	380,00		369,26	
+ contingencies		82,60		84,49
T o t a l (Mio DM)	462,60		453,75	

1.2 Fuel Cycle Cost Estimates

Fuel cycle costs are estimated according to the following formula which has been derived in full detail in [3,4]:

$$k_{Br} = \frac{R \cdot \delta_R}{E_i(1-q^{-L})} \left[\varrho \cdot \left\{ k_o^x - k_1^x \cdot q^{-L} \right\} + \right.$$

$$\left. + \frac{k_o^x - k_1^x}{Z} \cdot \frac{q^{-\frac{\delta_R}{Z}} - q^{-L}}{1-q^{-\frac{\delta_R}{Z}}} \right] \quad [\text{DPf/kWh}] \qquad (1)$$

with: $k_o^x = k_o \cdot q^{\delta_F} \cdot (1 + \frac{S}{100}(\delta_F + \frac{1}{2}\delta_R))$ $[\text{DM/kg}]$

that means present worth of fuel costs k_o including material, fabrication and transport at the time of fuelling

$$k_1^x = k_1 \cdot q^{-\delta_w} \cdot (1 - \frac{S}{100}(\delta_w + \frac{1}{2}\delta_R)) \quad [\text{DM/kg}]$$

that means present worth of gain for used fuel at time of unloading including material minus reprocessing and transport

$q \;=\; 1 + \frac{R}{100}$ [1]

R : interest rate [%/a]

S : tax rate [%/a]

δ_F : refabricating time [a]

δ_w : reprocessing time [a]

δ_R : inpile time [a]

$E_i = 24 \cdot \eta \cdot a$ [kWh/kg]

η : net thermal efficiency [1]

a : burn-up [MWd/kg heavy atoms]

ϱ : factor allowing for spare fuel elements

Z : number of partial loadings

L : plant-lifetime

The first term of formula (1) in the brackets represents the costs for the first core, the second term those for the running fuel cycle expenses.

The following parameters have been used for calculating fuel cycle cost:

	Sodium-Cooling	Steam-Cooling
1. Fabrication cost k_F (Mixed Core+Blanket) Fabrication time δ_F = 0,22 [a]	384	266 [DM/kg]
2. Reprocessing + Transport cost k_R (Mixed Core+Blanket) Reprocessing time δ_W = 0,5 [a]	257	202 [DM/kg]

3. Pu price 10 [\$/g Pu_{spalt}]

4. U_3O_8 price 8 [\$/lb]

5. Interest rate R = 7 [%/a]

6. Tax rate S = 2,7 [%/a]

7. Load Factor $\varkappa$ = 0,7

8. Plant-Lifetime L = 25 [a]

The fabrication and reprocessing costs have been arrived at after detailed analyses [9,10] carried out at the Karlsruhe Research Center taking into consideration the expected nuclear energy growth rate and probable plant sizes /5,6/. The fabrication costs are a function of the plant size and the fuel pin diameter whereas those for reprocessing depend on plant size, fissile material concentration in the irradiated fuel and batch size of the fuel elements to be reprocessed.

TABLE II

Comparison of Fuel Cycle Costs of a 1000 MWe Sodium Cooled Fast Breeder and a 1000 MWe Steam Cooled Fast Breeder (as given in the German paper to Session I)

	Sodium-Cooled Version	Steam-Cooled Version
1. Critical Mass /t Pu_{fiss}/	2,68	3,48
2. Breeding Ratio /1/	1,37	1,14
3. Burn-up /a/		
max.average in Core	80	55
average in Core+Ax.Blanket + 1/2 rad.Blanket /MWd/kg heavy atoms/	31,5	27,5
4. Specific Power r /MWth/kg heavy atoms/	0,043	0,040
5. Inpile time δ_R /a at 0.7 LF/	2,87	2,69
6. First Core Cost /DPf/kWh/	0,14	0,19
7. Running Fuel Cycle Cost /DPf/kWh/	0,20	0,35
8. Total Fuel Cycle Cost k_{Br} /DPf/kWh/	0,34	0,54

1.3 Other Running Cost

The running costs for plant operation are calculated according to the following formula (cf.4):

$$k_B = \frac{100}{8760\,\varkappa \cdot P_e} \,[K_P + K_V] \qquad [\mathrm{DPf/kWh}] \tag{2}$$

with:
$\varkappa\,[1]$ load factor
$P_e\,[\mathrm{kW}_e]$ electrical plant capacity
$K_P\,[\mathrm{DM/a}]$ personnel cost
$K_V\,[\mathrm{DM/a}]$ running cost for maintenance and materials

K_P is taken as 2.5 Mio DM/a and K_V as 4.8 Mio DM/a. With a load factor of 0.7 this gives for a 1000 MWe plant

$$\underline{\underline{k_B = 0{,}12}} \quad [\mathrm{DPf/kWh}]$$

1.4 Ground Rules for Calculation of Generating Cost

Power generating costs are composed of the three parts 1) Capital Costs, 2) Operation Costs, 3) Fuel Cycle Costs.

The capital costs are calculated as follows:

$$k_I = \frac{1}{8760 \cdot \varkappa} \cdot \left[K_A \left(\frac{R+S}{1 - q^{-L}}\right) + V\right] \quad [\mathrm{DPf/kWh}] \tag{3}$$

with:
$\varkappa\,[1]$ load factor
$K_A\,[\mathrm{DM/kWe}]$ specific investment
$R\,[\%/a]$ interest rate
$S\,[\%/a]$ tax rate
$L\,[a]$ plant-lifetime
$V\,[\%/a]$ insurance rate

$$q = 1 + \frac{R+S}{100}$$

The specific investment K_A comprises of direct plus indirect capital costs and interest during construction and start-up period.

The indirect capital costs are taken generally as 30 % of the direct costs in order to allow for about 10 % contingencies, 15 % engineering expenses of the customer and 5 % for start-up costs. In addition to these direct and indirect costs, that means 1.30 x K_D, a certain percentage for interest during construction and start-up period is added according to time of construction and interest rate. This is described in detail in [4]. If planning-, construction-, and start-up-period lasts for 5,5 a and the interest rate amounts to 7 %/a the interest during this time sums up to 11 % of the direct and indirect costs.

Thus

$$K_A = \frac{1.11 \times 1.30 \times K_D}{P_e} \quad [DM/kWe] \tag{4}$$

with: K_D [DM] direct capital cost

P_e [kWe] electrical plant capacity

and

$\varkappa = 0,7$

$R = 7,0\ \%$

$S = 2,7\ \%$

$L = 25\ a$

$V = 1\ \%$

one gets the following capital cost for the sodium cooled and steam cooled 1000 MWe power station

	Sodium cooled	Steam cooled
k_I [DPf/kWh]	1,15	1,07 (0,97)

Adding operating cost plus fuel cycle cost one arrives at the following power generating cost:

k_B [DPf/kWh]	0,12	0,12
k_{BR} [DPf/kWh]	0,34	0,54(0,50)
k [DPf/kWh]	1,61	1,73(1,59)

This result shows that there is an economic potential for both coolant versions sodium and steam of fast breeder power plants compared to that for presently known converters, especially if one takes into consideration that these cost figures for fast breeders are thorough but first and in capital costs unoptimized estimates. For the steam cooled fast breeder type the figures given in parenthesis may be achieved, as later estimates show.

1.5 Cost Trends as Fast Reactor Development Process

There is no doubt that fast breeder power plants which are considered at present have a considerable potential for further reduction of power production costs.

First of all capital costs may be reduced in the course of future technical development and on account of more sophisticated optimization. This will be true especially in the case of the steam cooled fast breeders which can be developed along the lines of the well established light water reactor technology arriving at capital costs equal to that of light water reactors. As 36 Mio DM direct investment costs or 52 Mio DM direct + indirect investment costs are equivalent to 0,1 DPf/kWh according to the calculational scheme given above one may not, however, expect much more reduction in capital cost than 0,1 - 0,2 [DPf/kWh] in the first period of fast breeder operation. With respect to power cost degression with plant size it may be stated that sodium cooled fast breeders have obviously a larger degression than steam cooled breeders on account of the pressureless cooling system. But this question has not yet been studied quantitatively.

On the other hand fuel cycle costs, though already extremely low compared with other reactor systems, have a remarkable potential for further reduction. In the course of time fabrication and reprocessing costs may be lowered. Quantitative analyses show a cost reduction of about 0,10 to 0,12 DPf/kWh for the figures given under 1.2 if fabrication and reprocessing costs are halved. Some detailed investigations [5] carried out more recently at the Karlsruhe Research Center show, however, that there is no great economic incentive for sodium cooled oxide fueled fast reactors to vary such parameters as pin diameter, rod power, blanket thickness or blanket management so long as the capital costs are assumed to be constant. The parameters on which the given figures are based seem to be rather optimal. These parameters are: diameter of fuel pin (without clad) 6,5 mm, rod power 460 W/cm, blanket thickness 30 cm and management of blanket such that the blanket is kept twice as long in the reactor as the core. But changing the fuel type of sodium cooled fast breeders from oxide to carbide would mean another considerable advantage with respect to fuel cycle costs. The relevant data are given below in Table III and may be compared to those figures given in Table II as the data have been produced in such a way that all remaining parameters of the power plant are kept constant.

Table III

Core and Fuel Cycle Data for a 1000 MWe Sodium Cooled Fast Breeder Power Plant with Carbide Fuel

1.	Rod diameter (fuel)	6,5 /mm/
2.	Rod power	920 /W/cm/
3.	Core volume	3,7 /m^3/
4.	Critical mass	1,75 /t Pu_{fiss}/
5.	Breeding ratio	1,47
6.	Burn-up	/MWd/kg heavy atoms/
	average of Core	80
	average of Core + Blanket	30,8
7.	Specific power	0,0610 /MW/kg heavy atoms/
8.	Inpile time	1,97 /a at 0,7 LF/
9.	First core cost	0,11 /DPf/kWh/
10.	Running fuel cycle costs	0,14 /DPf/kWh/
11.	Total Fuel Cycle Costs	0,25 /DPf/kWh/

The promising figures of Table III show that there is a strong economic incentive for the development of fast breeder fuel of the carbide type for sodium cooled systems.

Referring to steam cooled fast breeders fuel cycle costs may be reduced to some extent e.g. by using a higher specific power and by an improved breeding ratio with another canning material. But this question has to be investigated in more detail.

Finally recent investigations at the Karlsruhe Research Center [6] have shown that there is no considerable economic potential in an increased burn-up. This is true for the sodium cooled as well as for the steam cooled version. Fission product poisoning leads to increased fuel cycle costs above a certain optimal burn-up. This is on account of an increase of the critical mass and a decrease of breeding ratio which outweigh the cost reduction by higher power production per kg fuel. The exact optimum has still to be evaluated but should lie in the given region.

2. Assessment of Rate of Introduction of Fast Reactors

The extent to which fast breeders will be introduced for power production depends mainly on three conditions:

1. technical feasibility and safe operation
2. better economy than the best competitive thermal reactor then available
3. availability of fissile material that means ultimately plutonium

The first point has to be demonstrated in large prototypes and as the principle of fast breeders is too important for the future energy production, it seems to be better not to rely only on one type of technical realization that means to follow all promising technological variants of this second reactor generation until they prove their realistic advantages or disadvantages.

Provided the second condition, a better economy than proven thermal reactors, is given and can be demonstrated in the course of the development, then the introduction rate of fast breeders is primarily dependent on the availability of fissile material. As long as only plutonium shall be fed to newly installed fast breeders the rate of installation is wholly dependent on the plutonium production of already installed thermal reactors. Such a procedure has been studied in all detail in KFK-366 [7] which was prepared for the FORATOM-Congress September 1965. In the meantime extended studies have been carried out on the possibility of U-235 start-up of fast oxide breeders [8]. One of the major results of these studies has been the realization of the fact that the use of U-235 as fissile material in a fast reactor does not necessitate any alteration in the engineering design of a Pu fueled fast breeder. That means no change in control or fueling system has to be carried out if a plant is operated with one of the two fissile materials and breeder reactors may be started with U-235, and afterwards operated with self-produced Pu. Furthermore one may state that U-235 start-up of fast breeders will not lead to an alteration of reprocessing plants if such plants are centralized for several types of reactors, which seems to be the most economic way. It may, however, be mentioned that the fuel cycle costs for such reactors would be higher than those for the Pu-fueled ones.

Actually the possible start-up of fast breeders with U-235 results in a considerable freedom in choice of the optimal strategy for the introduction of a fast breeder generation. The data given below in Table IV show that there will be no drastic economic penalty for the use of U-235 in a sodium cooled fast breeder and show that the Pu production per GWa exceedes by far that from best "advanced converters".

Table IV

Fuel Cycle Data for a 1000 MWe Sodium Cooled Fast Breeder Power Plant Operated with U-235 (Remaining parameters as in Table II)

1.	Critical Mass [t U-235]	3,62
2.	Conversion Ratio [Pu_{fiss}/U-235]	0,94
3.	Pu-Production [t Pu_{fiss}/GWe·a]	0,930
4.	Total Fuel Cycle Cost [DPf/kWh]	0,66

In order to demonstrate the influence of the use of fast converters in a growing nuclear industry three characteristic examples shall be given (cf.Table V and Fig.1-3). The nuclear energy demand is taken according to KFK-366 and KFK-466 [4,7] starting in 1970 and arriving at 132 GWe installed in 2000. The installation rate of fast breeders is calculated in such a way that all excess Pu of converters and breeders is used for the Pu inventory of new fast breeders.

The rate of introduction of fast breeders is demonstrated in this paper only by examples of combined sodium cooled fast reactors. The economic aspects of the introduction of steam cooled fast reactors will be discussed in more detail in [8]. Steam cooled fast converters and breeders have the potential of replacing light water reactors in a growing nuclear power industry due to their improved economy (cf.chapter 1). In combination with sodium cooled fast breeders there are complementary features of steam cooled fast reactors with respect to the cost structure, i.e. higher fuel costs but lower capital costs (cf.chapter 1).

<u>Combination I</u> is a modern version of a boiling water reactor (LWR) with improved economy but slightly reduced Pu-production with a sodium cooled fast breeder with oxide fuel and Pu as fissile material as given in Table II.

<u>Combination II</u> is the sodium cooled fast reactor with oxide fuel and U-235 as fissile material as fast converter with the fast breeder of combination I.

<u>Combination III</u> is the sodium cooled fast converter of combination II with a sodium cooled fast breeder with carbide fuel and Pu as fissile material as given in Table III.

It may be noted from the Table V that the maximum number of converters to be installed decreases as one goes from combination I to combination II and finally to III. This means that the number of the more expensive converters can be decreased remarkably. Although the fast sodium cooled converters may be more expensive than light water reactors, because of the extremely low numbers of the former types required to meet the given nuclear energy growth, the total costs of energy and the total amount of natural and especially enriched uranium for the period 1970-2000 are reduced significantly. This will also be true for steam cooled fast converter and sodium cooled fast breeder combinations.

Table V

Characteristic Figures of Two Type Strategies for Introduction of Fast Breeders

Strategy Converter	Strategy Breeder	Number of Converters in 2000	Max.Number of Converters in the year cited	Consumption of U_{nat} x 1000t 2000	Consumption of U_{nat} x 1000t 2040
I LWR -	$NaBR(PuO_2)$	80	140(2015)	170	1100
II $NaBR(enr.UO_2)$	$NaBR(PuO_2)$	25	25(1995)	95	155
III $NaBR(enr.UO_2)$	NaBR(PuC)	1	10(1985)	35	40

SHARES OF INSTALLED NUCLEAR CAPACITY

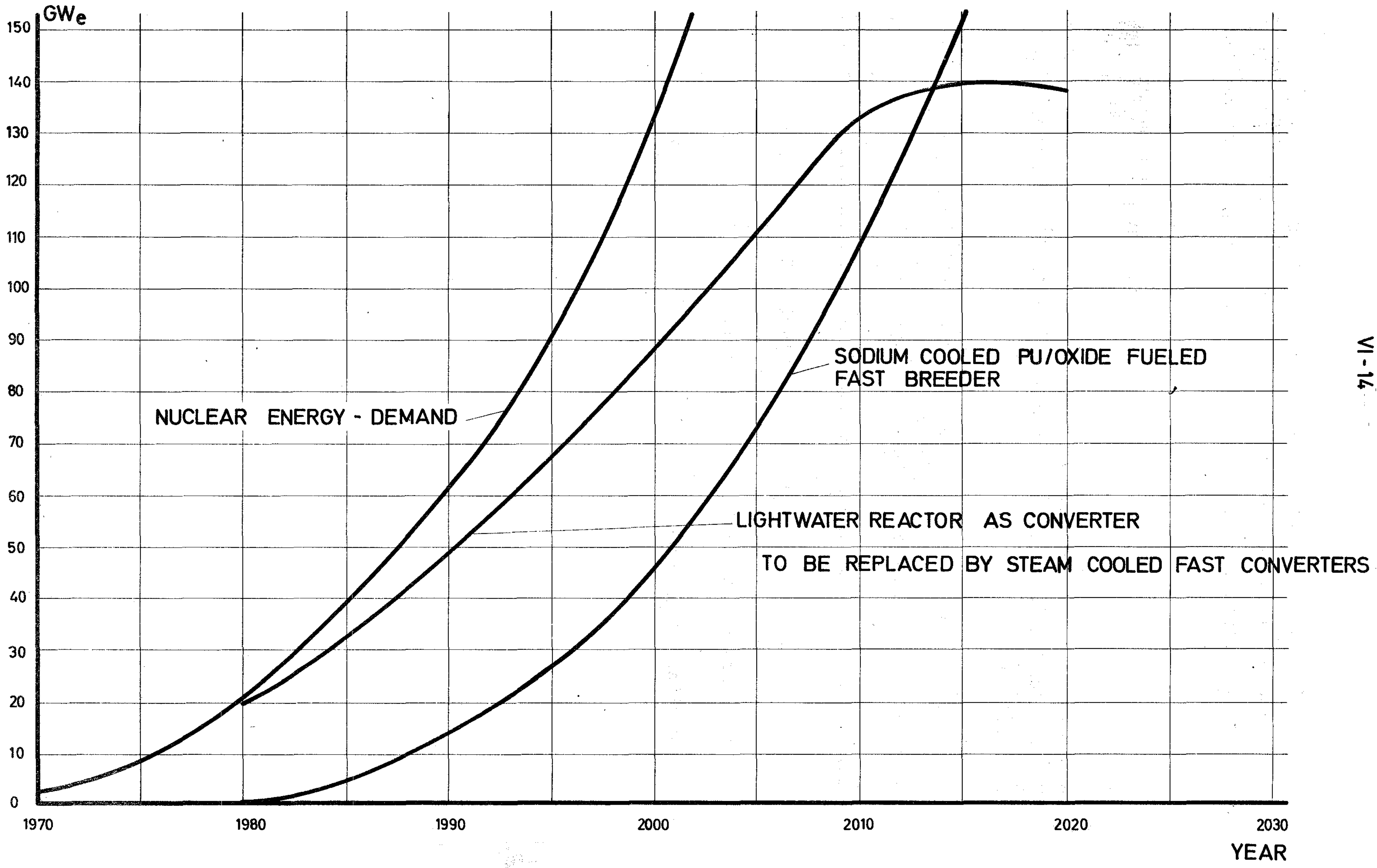

FIG. 1

SHARES OF INSTALLED NUCLEAR CAPACITY

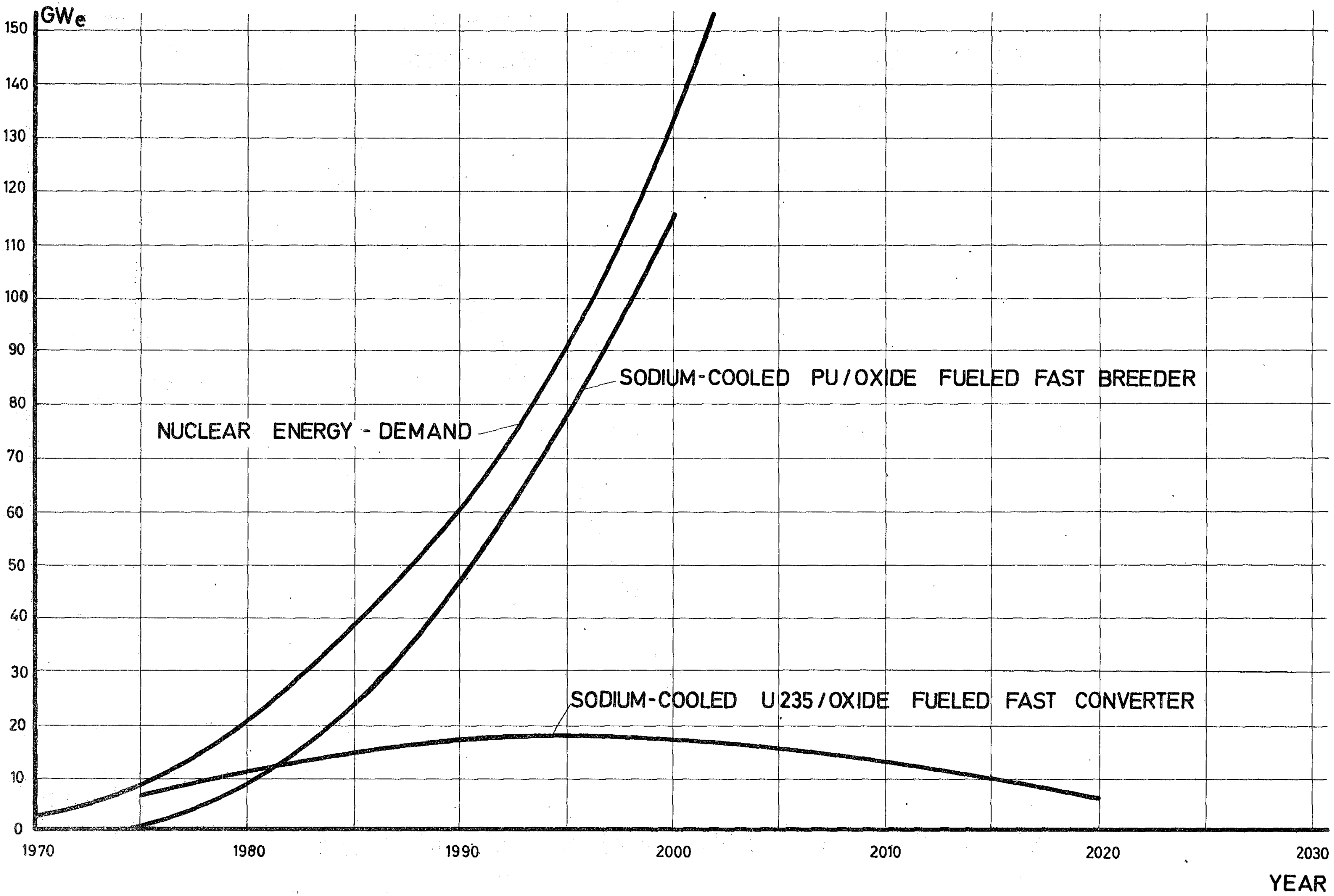

FIG. 2

SHARES OF INSTALLED NUCLEAR CAPACITY

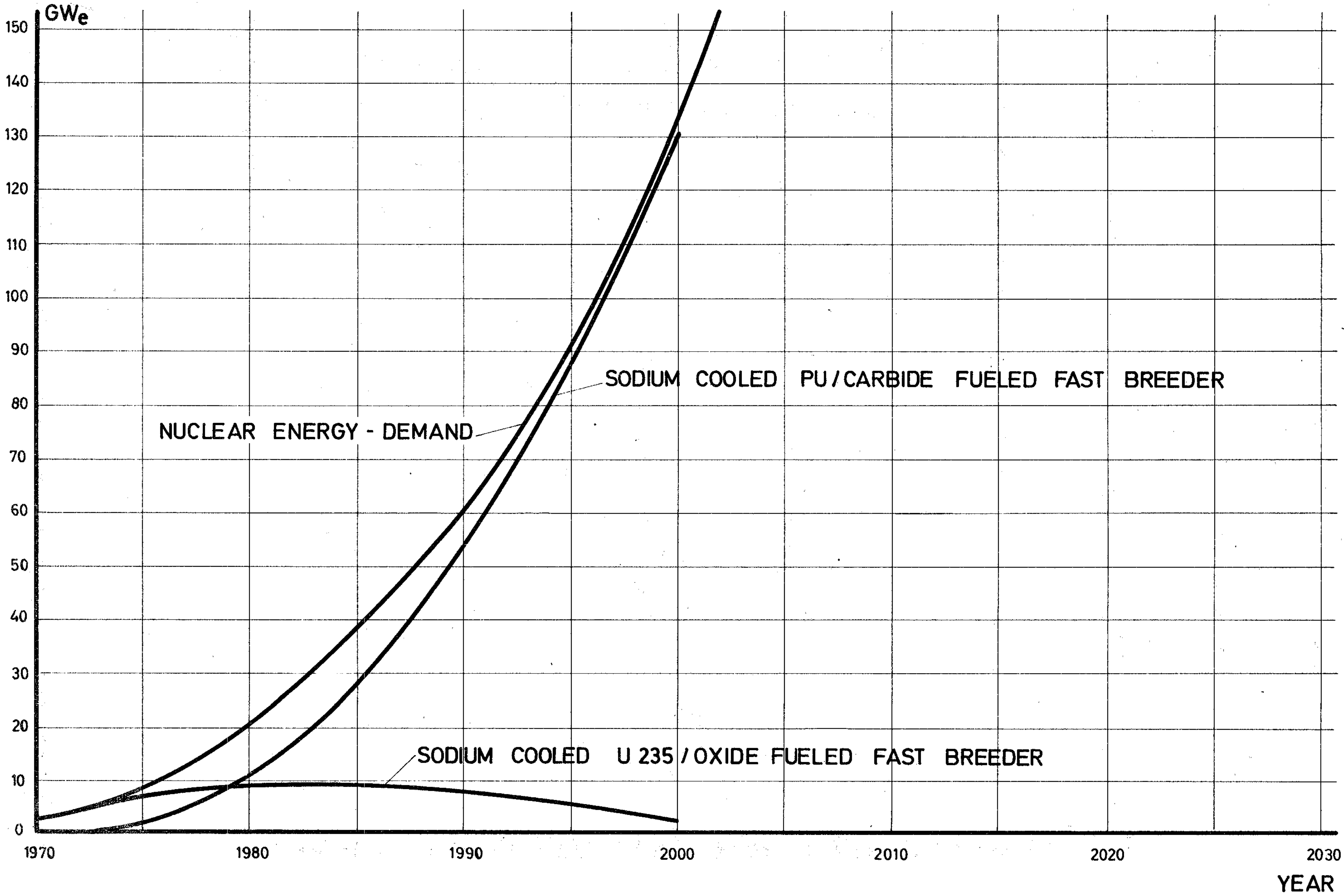

FIG. 3

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