The second annual meeting of the IWGFR was held in Vienna from 18 - 20 March 1969. The meeting was attended by the members of the Group from France, Federal Republic of Germany, Japan, UK, USA and USSR, observers from Czechoslovakia and Poland, and a representative from ENEA. The list of participants is attached to the Summary Report (Attachment 1).

The meeting was presided over by Mr. Vautrey (France).

The general considerations of the meeting were on national programmes on fast reactors and future activities of the IWGFR. The agenda of the meeting is found in Attachment 2.

2. APPRAISAL OF THE IWGFR'S ACTIVITIES

The scientific secretary summarised the activities of the IWGFR from the period of the first meeting. There will be three conferences on fast reactors until March 1970:


The dates for the last two conferences have been changed as compared to the dates which were discussed at the Group's last meeting. Monaco was chosen to host the Agency's Symposium as we feel that France will be able to make it possible to arrange a technical visit to Cadarache for those interested participants of
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the IAEA Symposium. The decision of the Agency with respect to the location of the Symposium was criticised by Dr. Wensch in his letter.

In accordance with the recommendations of the Group, the first meeting of specialists on sodium water reactions was held at Argonne, USA on 5 - 6 November 1968. The basic results of the meeting were reported by certain of the participants at the panel meeting of the following conference on sodium technology. All the participants unanimously agreed that it was very useful and a very wide-ranging exchange of information. However, this meeting would have produced much more beneficial results had the participants been asked to submit official reports. In this case the Agency could subsequently publish these reports, together with the discussions which took place. The scientific secretary made a suggestion that for all future meetings of specialists official reports be presented.

Agreement has been reached between the UK and the Agency with regard to holding the next meeting of specialists on the problem of values for plutonium alpha at Winfrith on 30 June - 1 July 1969.

Dr. Wensch said that, in general, recommendations of the Group came out rather well. However, after having carefully examined the details of the discussions which took place last year, one can find some changes made relating to those recommendations. He considered it important that the recommendations of the Group were followed through or, if the Agency revises the recommendations in any way, the IWGFR members should be informed of the basis for making the revision. The following cases were cited which deviated from the IWGFR recommendations.

1. It was understood last year that the fast reactor physics conference was to be held at Winfrith by the UKAEA. But the conference is being held in London by BNES.

2. Recommendations were made that the meeting at Dounreay would be beneficial if it included discussions on the behaviour of fuels and materials under fast neutron irradiation. There has not been any notice sent to the members of the Group on whether this question was discussed with the UK authorities.
3. There was an agreement that fast reactor meetings be held at sites in which fast breeder reactors were in operation, being designed, or under construction. From this point of view Monaco is not a suitable site for the Agency's Symposium.

4. A long discussion last year led to a conclusion that there should not be more than one international conference on fast reactors per year. Other meetings should be called "international topical meetings". Therefore the title of the meeting in Thurso does not quite agree with the understandings reached last year.

5. As concerns specialists' meetings, it was felt last year that they should be attended by people highly skilled in the field. Therefore preparation of official papers for such meetings is not necessary.

6. The minutes of the last IWGFR meeting were not detailed and some comments made by the participants were omitted. It was proposed to prepare more explicit minutes for future meetings of the IWGFR.

   Dr. Smith said the success of the sodium water reactions meeting at Argonne was largely due to the absence of prepared papers. Therefore he suggested to stick to that pattern for future specialists' meetings. Participants could be asked after a meeting to prepare short notes for a summary report.

   As concerns the titles of international conferences this was just not thought much about.

   The dates of the physics conference were altered because of domestic reasons. The conference is organised by BNES but this makes no difference if compared with the organisation by the UKAEA. This is a matter of administrative convenience within the UK. London was considered as a more convenient site from the point of view of available accommodation.

   At the last meeting of the IWGFR there was a conclusion that the main emphasis of the Dounreay conference was to be on the ways of getting results on fuel and materials irradiation rather than results themselves. There will be a session on irradiation
results obtained with the rigs which will be described during the conference.

Mr. Kuramoto felt it was not necessary to submit formal papers to the specialists' meetings but it was most desirable for the scientific secretary to prepare summary reports, which could be very useful.

Dr. Engelmann supported the opinion expressed by Dr. Smith and Mr. Kuramoto on specialists' meetings. He said it would under the presentation of very recent results and free discussion if official reports were requested before the meeting. On the other hand, it would be most helpful to have a report summarising the findings of the meeting.

The Chairman emphasised that the specialists' meeting at Argonne was very useful. He shared the unanimous opinion that these meetings should remain very simple in nature, and the round-table discussion character of these meetings which is very efficient should be preserved. He thought that such meetings should be convened more often, and that, at the same time, the number of conferences per year should be decreased.

As concerns the IAEA Symposium, Monaco seemed to be a very good choice since that site is not too far from Cadarache and a visit for participants to Cadarache could easily be arranged.

Dr. Schwuster offered that the Karlsruhe centre could be considered by the Agency as a site for future conferences and specialists' meetings on fast reactors.

Dr. Wensch said a problem of interpretation into other languages was encountered at the Argonne meeting, and therefore better provision must be made in future so that communications could be improved.

Extensive further discussions were carried out on the considerations of holding the IAEA Symposium in Monaco. The majority expressed their support to the Agency's choice for the location of the Symposium. Dr. Wensch expressed his disagreement with this decision.
A considerable proportion of the meeting was devoted to the presentations on national programmes made by the members of the Group. These reports, followed by discussions, are found in Attachment 3 in the order of their presentation.

In connection with Mr. Boxer's report on the studies of alternative coolants for fast reactors carried out in ENEA, a very prolonged discussion was held on whether this Group had to confine itself to sodium-cooled reactors only or could also consider information available on other coolants. The opinion of the members of the IWGFR and that of the Agency representatives, was as follows.

Dr. Engelmann pointed out that the Group has never really restricted itself to sodium. If there are some interesting results on other coolants, they should be briefly discussed at the IWGFR's meetings.

He did not see the problem at the moment since the activity in steam-cooled reactors had decreased and that on gas-cooled reactors had not increased in such a way that the Group had to worry about it.

Dr. Wensch emphasised that at the very first meeting of the IWGFR recommendations were made to the Agency that the activity of the Group should be restricted to sodium-cooled fast reactors.

It was too early to speak of widening the scope of the Group at its second official meeting.

The members of the Group were appointed by their Governments as specialists in sodium. By broadening the scope, other specialists would probably be needed, and the number of members would increase.

Dr. Smith said that the terms of reference referred to fast reactors rather than to sodium-cooled fast reactors. But since gas-cooled fast reactors form a very minor part of the national programmes, it would be appropriate for the IWGFR to devote a very small amount of attention to gas-cooled and steam-cooled
reactors. If at some later date steam-cooled or gas-cooled reactor programmes should grow, the position could be reconsidered.

Mr. Kuramoto agreed with Dr. Smith's opinion.

Dr. Spinrad mentioned that the Agency would expect to put to this Group questions concerning whether gas-cooled fast reactors would assume some importance in the Agency or in world programmes. If there are such questions, the member countries will be warned well in advance, so that they could be prepared to send appropriate experts to deal with these questions.

He commented that it was not easy for the Agency to set up another group which could deal with coolants other than sodium, and that from the point of view of administrative convenience, the IAEA would appreciate the opportunity of using this Group as a channel for questions to the States represented here on all problems of fast reactors, whatever the coolant.

He then pointed out that there was an agreement on the question between the members of the Group and the Agency. His request was that in considering possible changes of the terms of reference, the attention of the authorities should be brought to the fact that it was necessary to determine whether national fast reactor programmes on alternate coolants were to be represented here.

The Chairman agreed with Dr. Smith. The name of the Group was still valid, i.e., it included all reactor types, but it was natural to speak at IWGPB meetings almost exclusively about sodium-cooled reactors, since this was the major interest of countries represented in the Group. But this was no reason for depriving ourselves of information on studies which might be carried out on gas- and steam-cooled reactors.

Prof. Kazachkovsky had no objections to keeping the Group informed on the progress in alternative coolants for fast reactors but as concerns the planning of various meetings, he considered this Group as competent only in sodium-cooled reactors.
Dr. Wensch repeated his point of view for clarification:

(1) It was the intention of this Group to confine itself to sodium-cooled fast reactors, although this was not indicated in the terms of reference.

(2) It was the intention of this Group to limit the membership to those member countries having major fast reactor programmes.

(3) This Group was also to recommend subjects for specialists' meetings.

(4) Next year the situation could be re-evaluated to see whether the terms of reference should be broadened to include other coolants.

(5) Ad-hoc discussions on other coolants were very desirable.

4. COMMENTS ON GENERAL FINDINGS OF NATIONAL AND REGIONAL MEETINGS ON FAST REACTORS

The scientific secretary suggested to confine the discussion to comments on the principal results of the conferences held in represented countries, excluding international conferences.

Prof. Kasachkovsky considered the highlights of the Symposium on Fast Reactors held in December 1967 in Obninsk. The proceedings were already available last year. It was a conference whose participants were scientists from Eastern European countries. The Symposium was devoted primarily to fast reactors. The aspects under discussion were as follows: fast reactor physics, theoretical calculations, and experiments. Then there were papers on general technical problems of fast reactors. One could find in the proceedings a more detailed description of the EN-350 reactor and the design of the EN-600 reactor. There were various photographs of the construction site of EN-350.

A great deal of attention was paid to coolants, more specifically to sodium, and the compatibility of coolant with various construction materials. Other topics were on studies of experimental fuel elements for fast reactors.
There were a number of papers on other types of fast reactors beside those with liquid metal coolant. For example, two or three papers were on gas-cooled reactors. Mostly they dealt with safety aspects of these reactors. Then there were papers of a more exotic nature, for example, the use of molten salts for fast reactors.

Reprocessing of spent fast reactor fuel was not discussed.

To a certain extent the Symposium touched upon specific questions of economic optimisation of systems, and calculations of such important parameters as doubling time in different types of fast reactor.

Though that conference was held more than a year ago, its findings had not yet become obsolete and might be useful for those interested in the subject.

Some remarks had been made on the World Power Congress held last August in Moscow. There was a separate section for consideration of nuclear power problems. Most attention was paid to fast reactors. Unfortunately that section did not meet the expectations of fast reactor people. Everybody knows that there is a cautious attitude to fast reactors from the utility people and there was a hope that presentations at this section would help to prevent this suspicion. It so happened that many of the presentations contradicted each other and some of them even expressed wrong points of view. Therefore there was a bad impression as concerns prospects of fast reactors.

Dr. Wenocb reported on highlights of the Cincinnati Conference on Fast Reactor Systems, Materials and Components. That meeting complemented the previous meeting on fast reactors in San Francisco. The proceedings of the Cincinnati meeting had been published and he was ready to take care of assisting anybody in obtaining a copy of the proceedings.

The components for steam-cooled, gas-cooled and sodium-cooled fast breeder reactors were presented at the meeting, but from that aspect not much new was learned. Probably the most important part of that meeting was the rather extensive
information presented on the more unusual materials – the U-refractory alloys, the compatibility with sodium, advances made in some steels, and the high nickel-containing alloys. There was also considerable discussion presented on the design of large pumps for sodium systems. The initial work on the 60 000 gpm pump was presented. Some of its operating parameters were given. There was also an extensive discussion on the need for proof testing sodium components before putting them into nuclear service.

5. COMMENTS ON THE PROGRAMMES OF INTERNATIONAL MEETINGS ON FAST REACTORS TO BE HELD IN 1969 AND CONSIDERATION OF A SCHEDULE FOR MEETINGS ON FAST REACTORS IN 1970

There were no additional comments on the programmes of conferences on Fast Reactor Irradiation Testing (Thurso, UK, April 1969) and on the Physics of Fast Reactor Operation and Design (London, June 1969).

As concerns the IAEA Symposium on the Progress in Sodium Cooled Fast Reactor Engineering, the remarks were as follows.

The Chairman suggested to exclude from the list of topics Item I - design and construction of prototype fast reactors. This was agreed.

Dr. Wensch recommended to concentrate on getting more information on Items II, III and IV – primary components, steam generators and safety technology – from the point of view of systems requirements which these components must meet.

Dr. Smith said many safety problems were referred to the Item IV - safety technology – as it was obviously not possible to talk about components without talking on their individual safety linked with the particular reactor systems. But for this meeting it was suggested not to consider the overall safety aspects of reactor systems and leave them for the future meeting on safety problems of fast reactors. This was agreed.

The Chairman pointed out that in the title of the Symposium it was necessary to indicate that the Symposium related to the progress in the technology of fast reactors.
Dr. Engelmann mentioned that Item VI - in-core instrumentation for fast reactors - became more and more important and was appropriate for a specialists' meeting.

Dr. Wensch was in favour of such a specialists' meeting.

The Chairman summarised that there was unanimous agreement on the importance of the subject of in-core instrumentation for an appropriate discussion. On the one hand it was felt undesirable to confine oneself to only review papers for the IAEA Symposium; on the other hand, the problem was too vast for one specialists' meeting. Therefore it was proposed to delete this item from the list of topics for the IAEA Symposium and consider further some meetings of specialists which might cover this problem.

The further discussion was devoted to the conference on safety problems of fast reactors.

The Chairman reminded the members that at the last meeting there was a suggestion from Dr. Wensch to hold such a conference in the USA in 1969, but the Group agreed to defer it to 1970.

Dr. Smith considered it appropriate to have the conference in the Autumn of 1970 and asked Dr. Wensch whether the USA was still keen to act as the host country for such a conference.

Dr. Wensch agreed with the importance of a conference on safety problems and said there was no problem concerning holding that conference in the USA. However, he raised a question on whether it would be desirable to convene these conferences from time to time in the Member States represented in the IWGFR. The other point was that in future international conferences, the Member States of the Group should try to strengthen the participation and the presentation of papers. These two points should be considered with the other members of the Group before requesting the USA to host this meeting.

Dr. Engelmann said it was possible to hold the safety conference in Karlsruhe but at the same time he informed the Group on preliminary plans for a topical meeting organised by ANS and the European section of ANS with the title "Design and Performance
of Fast Reactor Fuel Elements" to be held, probably in Karlsruhe, in Autumn 1970.

Discussion was then devoted to whether it was necessary at all to have a safety conference in 1970, keeping in mind the IAEA Symposium and the conference on fast reactor fuel in Karlsruhe mentioned by Dr. Engelmann. The consensus was not to hold a safety conference in 1970 and return to this question at the next IWGFR meeting. As concerns the ANS conference in Karlsruhe, the members of the Group, in the absence of any details concerning the conference, did not feel ready to make any endorsement for participation in this meeting. The difficulty was recognised that in a year's time when more information on this meeting will be available, it could be too late to decide this question. Therefore this was left for a decision by correspondence.

Dr. Smith suggested to exchange opinions on preliminary ideas for holding fast reactor conferences in future.

Dr. Engelmann said they would be happy to invite a conference on safety problems to Karlsruhe in 1971.

Dr. Smith felt that agreement to hold a safety conference in 1971 had been reached and suggested to discuss the question in detail at the next IWGFR meeting. He also said that the UK would be interested in sponsoring a conference following the early power operation of the PFR. They would be interested in holding a conference on operating experience of that and other reactors which might be about in 1973.

Dr. Wensch recalled that at the previous meeting of the IWGFR, there was a general feeling that conferences on operating experience of fast reactors should be held on the occasion of the BN-350, the PFR and the Phenix coming into operation.

Prof. Kazachkovsky thought such conferences would be very interesting. He promised to do his best to examine the possibility of holding such a conference in the USSR, probably in 1971 or 1972. If there was a positive solution, he would immediately notify the scientific secretary.

The Chairman expressed everybody's satisfaction with the Specialists' Meeting on Sodium Water Reactions held at Argonne, USA in November 1968, and emphasised that representatives from different countries suggested organising a follow-up meeting on the subject at some future date. There was an opinion that the meetings on sodium water reactions should be held regularly.

The Chairman's suggestions for specialists' meetings were as follows:

1. Sodium water reactions.
2. In-core instrumentation.
3. Failure cladding detection.

Dr. Spinrad suggested a meeting on fast reactor spectrum.

Dr. Smith's suggestions were:

1. Sodium impurity measurements and control.
2. Sodium vapour control.

Dr. Wensch proposed meetings on the following to complement those proposed by the Chairman, Dr. Spinrad and Dr. Smith:

1. Latest techniques on non-destructive testing.
2. Seismic considerations of plant design.

Dr. Engelmann suggested a meeting on conceptual designs of primary cooling systems for FBR.

The meetings were written in a preferential order by all members of the Group and finally the first six meetings in the list of priorities were as follows:

1. In-core instrumentation (excluding cladding ruptures).
2. Failure cladding detection.
3. Sodium vapour control.
5. Sodium impurity measurements and control.
6. Sodium water reactions.
Dr. Spinrad said that the number of specialists' meetings which the Agency can sponsor per year was primarily a question of the wishes of the IWGFR. But he suggested that since there were in any given year three or four conferences on topical meetings on fast reactors, the number of specialists' meetings be the same from the point of view of possible combination of specialists' meetings with appropriate conferences.

Dr. Engelmann proposed to hold the Specialists' Meeting on In-Core Instrumentation in Karlsruhe in October 1969. This was unanimously agreed.

Prof. Kazachkovsky suggested to change the order of items 2 and 3 in the list of priorities and combine the Specialists' Meeting on Sodium Vapour Control with the IAEA Symposium in Monaco. The meeting on Failure Cladding Detection might be held in Karlsruhe in Autumn 1970, if the ANS conference takes place at that time.

In further discussion Dr. Wensch offered the USA as the place for the meeting on Failure Cladding Detection, realising that the IWGFR had not endorsed the ANS Karlsruhe conference.

Finally, it was decided to hold the meeting on Sodium Vapour Control in Cadarache in March 1970, combining it with the IAEA Symposium, and to postpone the decision on the place of the meeting on Failure Cladding Detection until the next annual meeting of the IWGFR. The Failure Cladding Detection meeting should be held in Autumn 1970.

The Chairman brought to the attention of the IWGFR other meetings on fast reactors organised by the Group on Sodium Boiling under the chairmanship of Dr. Grass of the Ispra Centre. The Group was set up after the conference in Aix-en-Provence and consists of representatives from Belgium, France, Federal Republic of Germany, Italy, Netherlands and the UK. Japan and the USA were also invited to the latest meetings. Actually that Group was a group of specialists who meet at regular intervals. The IWGFR had nothing to decide about that Group but just took note of its existence. Dr. Grass could be asked about an invitation to the USSR to participate in the Group.
The scientific secretary informed the Group on possibilities of publications of review papers in the IAEA Journal "Atomic Energy Review", and on the principles of co-operative research agreements under IAEA auspices.

Dr. Spinrad added that some publications could be issued in the form of monographs and elaborated further in the discussion on the purposes and shape of co-operative research agreements.

The Chairman said it was possible to organise co-operation in the field of collecting information on operating experience of fast reactors and accidents which was discussed earlier in the meeting. Taking into account remarks that such activities might overlap with those of the CREST group, it was proposed to re-consider this problem, and possibly discuss it, at the next meeting of the IWGFR.

Dr. Spinrad suggested discussing it at the next meeting of the IWGFR and then giving the Agency a formal recommendation as to what might be done on the problem of making better use of the integral basic theoretical data for calculations.

As concerns any co-operative research agreements, a negative attitude was expressed by the members of the IWGFR.

The members of the Group agreed to contact the Agency by correspondence if there were any specific suggestions as to review papers for publication by the IAEA.

Dr. Smith recollected that when the IWGFR was set up, it was understood that much of its work was to be done by correspondence in the periods between the annual meetings. In fact, during the last two years, there has been very little correspondence and there have been very few questions asked by the Agency of the Group. He suggested that the Group be made active throughout the year by correspondence, instead of waiting for an annual meeting.

This was agreed by the Group and the Agency's representatives.
TIME AND PLACE OF THE THIRD ANNUAL MEETING
OF THE IWCFR

The Chairman proposed that the third annual meeting of the International Working Group on Fast Reactors be held in March 1970 in Cadarache, France in the week preceding the IAEA Symposium in Monaco, or just after it. This was agreed. The exact dates of the meeting were to be decided between the Agency and the French authorities.
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AGENDA

1. Opening of the meeting.

2. Appraisal of the IWGFR's activity for the period from the first annual meeting of the Group.

3. Comments on national programmes on fast breeder reactors.

4. Presentation of general findings and conclusions of national and regional meetings on fast reactor problems held in represented countries and international organisations last year.

5. Comments on the programmes of international meetings on fast reactors to be held in 1969.


8. Suggestions for reviews and studies in the field of fast reactors.

9. The time and place of the third annual meeting of the IWGFR.

10. Closing of the meeting.

Introduction

A year ago I gave you a summary of the German Fast Breeder Project, its objectives and its development.

During the past year work under the basic programme an on the development of the sodium-cooled fast breeder prototype power station was carried on so that the original goal - i.e. the submission of title-one design documents for a 300 MWe prototype nuclear power plant early in 1970 - will probably be achieved.

Activities for the development of a steam-cooled fast breeder nuclear power station, which were also pursued in the Federal Republic of Germany, were discontinued in February of this year and the steam line development work was considerably cut down.

The close co-operation established between Germany, Belgium and the Netherlands for the development of the sodium-cooled fast breeder has meanwhile been further extended to include Luxembourg. In last August, the Governments of the Federal Republic of Germany and the Grand Duchy of Luxembourg exchanged a memorandum on co-operation in the fast breeder field. The Luxembourg contribution
will be made by the industrial group Luxatom and will comprise activities falling under the basic programme carried on by the nuclear research centres as well as development work on the sodium-cooled prototype nuclear power plant.

In my paper I am going to describe the progress achieved to date by industry, and Dr. Engelmann will report on the work which is being done in the nuclear research centres.

The Steam-Cooled Fast Breeder

The results available in 1966 suggested that the title-one design documents for both a steam-cooled and a sodium-cooled fast breeder prototype nuclear power station could be prepared by 1969 or 1970. Therefore, in November 1966, the industrial group consisting of the firms Allgemeine Elektricitätsgesellschaft, Gutehoffnungshütte Sterkrade AG and Maschinenfabrik Augsburg-Nürnberg AG (AEG/GHH/MAN) commenced work on the design and development of a 300 MWe steam-cooled fast breeder nuclear power plant. As work proceeded, it became obvious early in summer 1968 that the original project aim could not be achieved within the scheduled period of time.

The crucial issue was the lack of an appropriate testing facility to test the steam-breeder fuel elements under realistic reactor conditions. The original plan to test the fuel elements in the Superheat Reactor (HDR) now under construction after inserting a second fast-thermal core had to be abandoned for safety reasons;
this means that a special fuel element testing reactor would have to be built in order to prove the feasibility of the chosen fuel element design, before starting the construction of a steam-cooled fast breeder nuclear power station.

As a result, it was necessary in summer 1968 to modify the objectives of the steam breeder programme and to change the original time schedule; therefore, the Federal Government decided to reconsider the programme for the development of a steam-cooled fast breeder.

On the basis of the recommendations made by the responsible advisory bodies of the Federal Government, the Federal Minister for Scientific Research decided in February of this year:

(1) to discontinue the development of the steam-cooled fast breeder as an independent subproject of the overall fast breeder programme;

(2) to continue work on this project in the industrial sector in order to reach a stage whereby a comprehensive final report can be submitted. This should be a report which can be used as the basis for a possible resumption of activities for this project.

(3) to continue individual developments under the basic programme of the overall fast breeder project carried out by the Karlsruhe Nuclear Research Centre, in particular the development of fuel rods.
The most important reasons for this decision are as follows:

- Because of technological problems the objectives and the time schedule of the German steam breeder programme had to be revised thoroughly.

- Owing to the new international situation - no other country is going to pursue the steam line - the Federal Republic of Germany would stand alone in developing the steam-cooled fast breeder without having a chance of getting a broader basis by international partners.

- As far as can be said today, the design of the steam-cooled fast breeder does not offer an advantage over the sodium-cooled fast breeder using oxide fuel elements. In the long run it does not seem possible to improve the steam-cooled fast breeder to reach electricity generating costs to be expected of the sodium-cooled breeder once carbide fuel is introduced.

- Commercial steam-cooled nuclear power stations will not be ready for operation as early as has hitherto been expected; this would enhance the risk involved in launching this type of nuclear power plant on the market.

The industrial activities for the development of the steam-cooled fast breeder will be steadily reduced in volume and will more or less cease in the course of this year. As a result, capacities will be released to intensify efforts for the development of the sodium-cooled fast breeder.
The Sodium-Cooled Fast Breeder

On the basis of the Na-2 design study completed in 1967 by the Karlsruhe Nuclear Research Centre in co-operation with the industrial consortium Siemens/Interatom, a modified reference design was prepared by the industrial consortium for the 300 MWe prototype nuclear power station. Work on the single design for the 300 MWe sodium-cooled fast prototype reactor, SNR, has continued as planned. A fixed-price offer of the industrial consortium Siemens/Interatom, Belgonucléaire and Neratom is expected early in 1970.

The most important design data of the nuclear power plant and of the reactor core are indicated in Tables 1 and 2.

After detailed comparative studies a loop design was chosen for the primary system (see figure 3); three intermediate heat exchangers are installed in three heat transfer cells and are connected by tubes to the reactor vessel and the pumps.

The consortium feels that this design as compared with a pool arrangement has the particular advantage of involving fewer technical risks and of being more readily scaled up to fairly large plants.
The primary loops which contain activated sodium are completely jacketed to exclude a loss-of-coolant accident. The entire primary system is surrounded by a double containment. Located outside the containment are the steam generators in a separate building and some of the handling facilities which are housed in a gas-tight handling station adjacent to the containment.

The handling system comprises 2 fuel element storage areas, 2 fuel handling machines, and one observation cell. Irradiated fuel elements are transferred under sodium from the reactor core to an annular fuel storage within the reactor vessel - a temporary decay area for irradiated fuel elements - with the aid of an in-vessel transfer device permanently installed on the rotating top shield, whereas for conveying the fuel elements from the annular storage area to the fuel storage within the handling station, an argon-cooled fuel charge/discharge machine is used.

Above the rotating top shield, in a pit which comes up flush with the operating floor, both the fuel element transfer device and the control rod drives are installed in a manner permitting the fuel charge machine to proceed to any position of the annular fuel storage without interfering with the performance of the fuel element transfer device, and without any control rod drives having to be dismantled for the sake of fuel handling operations.

Essential internal components of the reactor vessel are: (see fig. 4) - below the grid plate: a separator for eliminating any gas bubbles entrained by the coolant. The separated gas is fed to the upper plenum via the breeder elements.
above the grid plate; the reactor core made up of free-standing fuel elements held down in position individually by hydraulic means. The core is surrounded by a blanket zone followed further toward the periphery by the steel-shielded decay storage positions for irradiated fuel elements. The core contains a total of 18 absorber positions. Of these, the 12 rods of the outer zone belong to the control and shim system; three rods of the inner zone belong to the first scram system; the three other rods of the inner zone belong to the completely independent second shut-down system which is of a different design.

above the core fitted to the rotating top shield; a perforated dip plate, which reduces the free sodium-surface with all the related problems like perturbations of the surface, sodium vapour production and problems of sodium hammer in case of accident. A device for positioning thermocouples and other instrumentation, as the case may be (at the same time serving as a mechanical stop to prevent lifting of fuel elements).

The support structure between the dip plate and the lower plate for positioning the instrumentation guides the control rods.

The 3 Intermediate Heat Exchangers (Fig. 5) are straight-tube units so designed as to permit the identification and repair of leaking tubes in the bundle, without dismantling and cleaning down the entire heat exchanger.
For the reference design of the steam generator, straight-tube concept was adopted, too. Each of the 3 secondary sodium loops includes 4 superheaters and 2 intermediate superheaters. The steam generators are built up from modules with a view to keeping the operating risk small in the event of damage, and to ensuring continued operation of the unit at full capacity also after shut-down of a damaged module.

Special attention was directed towards the problem of engineered safeguards. Development work has been started on in-core instrumentation (thermocouples, flow meters, and fast reading can failure detectors) for the detection of local events. A hazards analysis based on probability theory has been started at Interatom. As input data of this analysis, information on the reliability of reactor components is required. Part of this information is available; experimental investigations are planned for important components, such as control rod drives, the reliability of which is not known with sufficient accuracy.

The construction of the major test facilities has proceeded according to schedule:

The Fast Breeder Safety Test Facilities (ASB) of Interatom have been taken into operation and since mid-1968 have been available for
basic tests on the sodium-water reactions in big steam generators. The results obtained at the ASH largely proved the theory derived from previous small-scale experiments. Construction and general assembly work on the Pump Test Facility (APB) and the Fast Breeder Reactor Test Line (MBB) of Intera tok has been more or less completed.

Construction work of the 58 MWe KNK reactor at Karlsruhe is nearing completion. All major components have been installed; work is still going on to complete the electrical installation, and the check-out of components of the system has started. It is planned to fill the system with sodium in May.

Design work is also far advanced on the second KNK II fast core. A cross-section of the core is shown in fig. 6. The 7 central sub-assemblies form a hard spectrum test zone for PuO\textsubscript{2}-UO\textsubscript{2} fast reactor fuel elements. High rod ratings are obtained by use of highly enriched fuel (30\% PuO\textsubscript{2} + 70 \% UO\textsubscript{2} enriched to 80 \% in U-235). The surrounding driver zone of UO\textsubscript{2} fuel elements (35 \% enrichment) contains some ZrH to soften the neutron spectrum in order to get a larger negative Doppler coefficient. (Table 3).

During the period from September to December 1968, experimental investigations were made in SNEAK on a core mock-up of the KNK-II reactor.

Another facility under construction is called BEVUS (see Fig. 7).
State of Development of the Activities for the Sodium Breeder
in the Karlsruhe Research Centre in Cooperation with Industry

1. Fuel Element Development

Fuel capsules and fuel pins are being tested in both thermal and fast neutron environment. UO₂ capsules have reached a burn-up of 70 000 MWD/t, UO₂PuO₂ capsules of 30 000 MWD/t in the FR-2. 7 pins of 1.08 m length with mixed oxide fuel have reached 25 000 MWD/t in the BR-2. These pins are irradiated in a sodium loop at 600 W/cm linear rod power and 600°C cladding temperature. 3 pins have been tested in the DFR up to 20 000 MWD/t, the irradiation of 77 pins in the center of the DFR has started in February.

A big program of cladding material irradiation is under way at the BR-2. Half of the irradiation capacity of this reactor will be used over the next 5 years for FBH work. A cladding material irradiation up to $10^{23}$ n/cm² will be started in the DFR this summer.

The development work on carbide fuel is being increased because of its long-time aspects as the most economic PBR fuel.

In view of the material problems, arising in high fast neutron fluxes and at high fast neutron doses, a feasibility study has been initiated on a fast high flux test reactor with a flux of 1 to $1.5 \times 10^{16}$ n/cm² sec. It is expected that first results of this study will be available by end 1969.
2. Sodium Technology

A list of the major test facilities has been given last year. This list did not mention a large number of small special purpose sodium facilities in operation or under construction at Karlsruhe.

There is a test stand for fuel subassemblies (flow tests at 560°C Na temperature, 60 m³/hr), a corrosion loop "Cerberus", where austenitic steels and vanadium alloys are being tested at 500°C, 550°C, and 600°C for periods up to 10 000 hrs, a mass transfer loop, a facility for testing reactor components in sodium etc.

A high temperature sodium facility with 3 corrosion loops of 100 kW each for testing steel and vanadium alloys at temperatures up to 500°C at flow velocities up to 12 m/sec is under construction. Connected to this facility is a heat transfer loop for electrically heated fuel element subassemblies.

Another facility under construction is called BEVUS. It will be used to study the propagation of fuel element failures from one subassembly to its neighbours in case of sodium ejection and reentry due to sudden evaporization at superheat conditions.

3. Physics and Safety

Experiments in support of the SNR design will start at SNEAK in April. The experimental program has been coordinated with the French zero power reactor program for Phenix in a way, that the same core
compositions will be studied in both MASURCA and SNEAK. This makes possible an exchange of fissile material between the two facilities and also helps to speed up the accumulation of physics information as different core arrangements can be investigated at the same time.

The integral experiments performed in SNEAK have shown some deficiencies of the nuclear data sets used at Karlsruhe. In general, criticality is underestimated, the discrepancies being larger for plutonium than for uranium systems. They increase with the softness of the neutron spectrum. A new data set therefore is in preparation which is based on both new cross section information and integral data. Major changes are a reduction of the U236 capture data in accordance with results of the PETREL nuclear explosion and measurements of Moxon and Pönitz, a reduction of Pu240 capture data according to recommendations of Pitterle et al., and the incorporation of the Pu239-alpha-data of Gwin.

The Pu239 Doppler experiments reported last year have been repeated with great care in the soft spectrum Pu-core SNEAK-3B-2. The results are consistent with resonance parameters corresponding to the Pu-alpha values of Gwin. Details of this experiment and of the $k_{\infty}=1$ experiment SNEAK-5 will be discussed at the IAEA-specialists meeting on plutonium alpha in London.

The differential alpha-measurements at the Karlsruhe van de Graaff facility will be started this spring. Last year very careful re-measurements have been made on the ratio of the fission cross sections
of Pu239 and U235 relative to U235 in the energy range 3 to 200 keV. The accuracy above 20 keV is about 1 - 2%. These remeasurements had been initiated in order to remove some discrepancies of earlier measurements which indicated even lower Pu239 fission data than measured by White. The new results will be available within the next few weeks. In 1969 a program will be started on measurements of fission product capture cross sections. This work complements the reactivity measurements with fission product isotopes to be performed at the STEK-reactor at Petten/Holland.

Research work for fast reactor safety includes the study of nuclear aerosols. A test facility TUNA has become operational last year. In this tank UO₂ pins may be evaporated by means of a large current pulse passing through the preheated pin. It is planned to investigate in addition to pure UO₂ aerosols also mixed aerosols with sodium compounds to simulate the conditions in the containment after a major accident.

4. Work on Gas Cooled Fast Breeder Reactors

The Karlsruhe laboratory has actively participated in the NSBI study on gas cooled fast breeders. This work included feasibility studies and cost estimates. The main problem areas are the high temperature fuel element, large gas turbines, and concrete pressure vessels. Industry is confident that 1000 MW gas turbines and also
The concrete pressure vessel for such a reactor could be built. The development work for a fuel element, however, is in a very early stage. Basic experiments have been started on UO₂PuO₂-chromium cermet fabricated by hot pressing. Some results obtained with vanadium alloys indicate that a cladding material could be developed on that basis.

The work on helium cooling is followed as a long-time back-up for the sodium line with much less effort.
Table 1: Main Technical Data of SNK-Plant

<p>| | |</p>
<table>
<thead>
<tr>
<th></th>
<th></th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Power</strong></td>
<td>730 MWth/300 MW el net</td>
</tr>
<tr>
<td><strong>Primary System (3 loops)</strong></td>
<td></td>
</tr>
<tr>
<td>Na temperatures</td>
<td>360° C inlet/560° C outlet (m.m.)</td>
</tr>
<tr>
<td>max. fuel element surface temp.</td>
<td>700° C</td>
</tr>
<tr>
<td>pressure drop/Na flow rate</td>
<td>6 atm/14,000 m³/h</td>
</tr>
<tr>
<td><strong>Secondary System</strong></td>
<td></td>
</tr>
<tr>
<td>Na temperatures</td>
<td>IHX: 340° C inlet/540° C outlet, steam generators: 525° C inlet</td>
</tr>
<tr>
<td>pressure drop/Na flow rate</td>
<td>5 atm/12,750 m³/h</td>
</tr>
<tr>
<td><strong>Steam System</strong></td>
<td></td>
</tr>
<tr>
<td>3 x 4 steam generators</td>
<td>52 MW and 91 t/h each</td>
</tr>
<tr>
<td>pressures and temp. inlet/outlet</td>
<td>(195 atm/285° C)/(170 atm/510° C)</td>
</tr>
<tr>
<td>3 x 2 reheaters</td>
<td>17 MW and 140 t/h each</td>
</tr>
<tr>
<td>pressures and temp. inlet/outlet</td>
<td>(44 atm/325° C)/(42 atm/500° C)</td>
</tr>
</tbody>
</table>
Table 2: Characteristics of Core and Blanket

**Fuel of Core (PuO₂ - UO₂)**

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>enrichment (2 zones)</td>
<td>17.2/25.5 % fiss.</td>
</tr>
<tr>
<td>inventory of fissile material</td>
<td>0.87 t Pu fiss.</td>
</tr>
<tr>
<td>density</td>
<td>80 % theor. dens.</td>
</tr>
<tr>
<td>Pu 239/240/241/242</td>
<td>75/22/2.5/0.5 %</td>
</tr>
</tbody>
</table>

**Performance of Core**

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>average fuel rating</td>
<td>760 MW/t fiss. mat.</td>
</tr>
<tr>
<td>max. nominal rod rating</td>
<td>420 W/cm</td>
</tr>
<tr>
<td>breeding ratio/doubling time</td>
<td>1.33/10.9 full power years</td>
</tr>
<tr>
<td>max./av. burnup</td>
<td>80,000/55,000 MW-d/t fiss. mat.</td>
</tr>
</tbody>
</table>

**Design Data of fuel elements**

<table>
<thead>
<tr>
<th>Component</th>
<th>Description</th>
</tr>
</thead>
<tbody>
<tr>
<td>core (Ø 1.5 m)</td>
<td>150 elem. each consisting of 169 rods of 6 mm diam.</td>
</tr>
<tr>
<td>blanket</td>
<td>312 each consisting of 91 rods of 9.5 mm diam.</td>
</tr>
<tr>
<td>overall/active height</td>
<td>3.17 m/0.95 m</td>
</tr>
<tr>
<td>distance across flats</td>
<td>110 mm</td>
</tr>
</tbody>
</table>
### Table 31: Kidd-II Design Data

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Test-Bore</th>
<th>Driver-Bore</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core volume, l</td>
<td>60</td>
<td>240</td>
</tr>
<tr>
<td>Core height, cm</td>
<td>60</td>
<td>60</td>
</tr>
<tr>
<td>Core composition, v/o</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Fuel</td>
<td>31.7</td>
<td>28.9</td>
</tr>
<tr>
<td>Na</td>
<td>48.7</td>
<td>46.0</td>
</tr>
<tr>
<td>Ss</td>
<td>19.6</td>
<td>19.7</td>
</tr>
<tr>
<td>ZrH&lt;sub&gt;1.7&lt;/sub&gt;</td>
<td></td>
<td>5.4</td>
</tr>
<tr>
<td>Number of fuel subassemblies</td>
<td>7</td>
<td>22</td>
</tr>
<tr>
<td>Fuel pins per subassembly</td>
<td>211</td>
<td>102</td>
</tr>
<tr>
<td>ZrH&lt;sub&gt;1.7&lt;/sub&gt; pins per subassembly</td>
<td>0</td>
<td>19</td>
</tr>
<tr>
<td>Pin outer diameter, mm</td>
<td>6.0</td>
<td>8.2</td>
</tr>
<tr>
<td>Average burnup, MWd/t</td>
<td>80,000</td>
<td>-</td>
</tr>
<tr>
<td>Full power days for 30,000 MWd/t burnup</td>
<td>450</td>
<td>-</td>
</tr>
<tr>
<td>$\bar{\text{av}}_{\text{max}}$, n/cm&lt;sup&gt;2&lt;/sup&gt;sec</td>
<td>2.3x10&lt;sup&gt;15&lt;/sup&gt;</td>
<td>-</td>
</tr>
<tr>
<td>max. linear rod power, W/cm</td>
<td>450</td>
<td>350</td>
</tr>
<tr>
<td>average linear rod power, W/cm</td>
<td>250</td>
<td>220</td>
</tr>
<tr>
<td>$^1_{\text{Na}, \text{in}}$</td>
<td></td>
<td>360°C</td>
</tr>
<tr>
<td>$^1_{\text{Na}, \text{out}}$</td>
<td></td>
<td>560°C</td>
</tr>
<tr>
<td>$\beta$</td>
<td></td>
<td>6x10&lt;sup&gt;-3&lt;/sup&gt;</td>
</tr>
<tr>
<td>Doppler constant $\frac{d\lambda}{dT}$</td>
<td></td>
<td>-4x10&lt;sup&gt;-3&lt;/sup&gt;</td>
</tr>
<tr>
<td>Na-void effect</td>
<td></td>
<td>-6x10&lt;sup&gt;-2&lt;/sup&gt;</td>
</tr>
</tbody>
</table>
FIGURE 4: REACTOR VESSEL AND ASSOCIATED EQUIPMENT
FIGURE 5: NA/NA-HEAT EXCHANGER
Figure 6: KNK-II Core Cross Section
Figure 7
Schema des BEVUS-Versuchsstandes
List of KFK-Reports Concerned with Fast Reactor Problems

(Continuation of the list distributed during the 1st Annual Meeting of the INGPR)

KFK-539/I  Y.S. Hoang
Struktur-, Ausdehnungs- und Verbiegungseffekte im schnellen Reaktor
Teil I: Theoretische Überlegungen

KFK-539/II  Y.S. Hoang
Struktur-, Ausdehnungs- und Verbiegungseffekte im schnellen Reaktor
Teil II: Programmbeschreibung

KFK-539/III  Y.S. Hoang
Struktur-, Ausdehnungs- und Verbiegungseffekte im schnellen Reaktor
Teil III: Anwendung auf Na-1 Reaktor

KFK-654  M. Cramer
Studie über einen 527 MWth natriumbeheizten Dampferzeuger und einen 98 MWth natriumbeheizten Zwischenüberhitzer

KFK-657  W. Frisch, G. Weite
Analogrechenmodell für dampfgekühlte schnelle Reaktoren mit Direktkreislauf

KFK-660  K. Gast, E.G. Schachtendahl
Schneller Natriumgekühlter Reaktor Na-2

KFK-667  M. Audoux, L. Caldarola, P. Giordano, H. Rohrbacher
Automatic Control System for Balanced Oscillator Tests

KFK-673  W. Bühr, Th. Dippel
Über die Auflösung von PuO2-haltigen Brüterbrennstoffen in Salpetersäure für die wässrige Wiederaufarbeitung nach dem Purex-Verfahren
KFK-689  M. Dalle Donne, K. Wirtz
Gas Cooling for Fast Breeders

KFK-696  R. Theisen, D. Vollath
Plutonium Distribution and Diffusion in UO\textsubscript{2}-PuO\textsubscript{2} Ceramics
Plutonium as a Reactor Fuel
Proc. of a Symp. on the Use of Plutonium as a Reactor Fuel
held by the IAEA in Brussels, 13-17 March 1967, S. 253-264

KFK-700  H. Beißwenger, H. Blank, H. van den Boorn u.a.
Die Entwicklung von Brennelementen schneller Brutreaktoren

KFK-701  L. Lindner, A. von Baeckmann
Verfahren zur chemisch-analytischen Bestimmung von Plutonium
und Uran in oxydischen Kernbrennstoffen

KFK-706  H. Hauck
Untersuchungen über das Kriechverhalten einer aushärtbaren
Nickel-Chrom-Eisen-Legierung im Temperaturbereich um 0.18 $T_\text{S}$

KFK-709  F. Baungärtner
Die wässrige Wiederaufarbeitung von Kernbrennstoffen besonders
mit hohem Plutonium-Gehalt

KFK-712  H. Böhm, H. Hauck, G. Hess
Untersuchungen über die Hochtensatempesversprödung nach
Neutronenbestrahlung von 16/13-CrNi-Stählen
Journal of Nuclear Materials 24 (1967), S. 198-209

KFK-713  H. Böhm, H. Schneider
Über das Zeitstand- und Kriechverhalten von austenitischen
Chrom-Nickel-Stählen in Gegenwart von Natrium
Journal of Nuclear Materials 24 (1967), S. 188-197

KFK-718  T.J. Connolly, F. de Kruijf
An Analysis of Twenty Four Isotopes for Use in Multiple Foil
(Sandwich Measurements of Neutron Spectra below 10 keV
J.J. Schmidt
Recommended Resolved and Statistical Resonance Parameters
for Twenty Four Isotopes
J. Kadlec, V. Pfommer  
Methode für die experimentelle Untersuchung der Eigenfrequenzen der Normal- und der Dehnungsfunctionen sowie der Dämpfung der querschwingenden Brennstäbe

R. Walze  
Aufbau und Betriebsweise des SHEAK-Reaktivitätsmeters

W. Schwetje  
Numerische Untersuchungen zu Methoden der Kritikalitäts- und Brutrateberechnung für Reaktoren mit Zylindergeometrie und teilweise eingeführten Regelstäben

W.P. Pönitz  
A "Grey" Neutron Detector for the Intermediate Energy Region

I. Langner, J.J. Schmidt, D. Well  
Tables of Evaluated Neutron Cross Sections for Fast Reactor Materials  
(KFK-750 wird nur gegen Berechnung von DM 24.-- abgegeben.)

H. Kämpf  
Einfluß der inneren Geometrie auf die radiale Temperaturverteilung von Schnellbrüter-Brennelementen

W. Frisch  
Stabilitätsprobleme bei dampfgekühlten schnellen Reaktoren

W. Baumann, V. Casal, H. Hoffmann, R. Møller, K. Rust  
Brennelemente mit wendelförmigen Abstandshaltern für Schnelle Brutreaktoren

H. Huschke  
Gruppenkonstanten für dampf- und natriumgekühlte schnelle Reaktoren in einer 26-Gruppendarstellung

J.J. Schmidt  
Abschätzung unbekannter Anregungsfunktionen für (\(\alpha, xn\))- (\(\alpha, pxn\))-, (\(d, xn\))- (\(d, pxn\))- und (\(p, xn\))-Reaktionen
KFK-774  H. Böhm, M. Schirra  

KFK-776  P. Engelmann, A.M. Raberain, D. Wintzer  
Untersuchung der Abhängigkeit der Reaktivität eines schnellen dampfgekühlten 500 l Reaktors von der Dampfdichte in SNEAK-3A

KFK-780  H.E. Häfner  
Bestrahlung von Brennstäben in instrumentierten Blei-Wismut-Kapseln  
Irradiation of fuel pins in instrument-equipped lead-bismuth capsules  
Kerntechnik, Isotopentechnik und -Chemie, 10 Jg., 1968, Heft 3 S. 136-141

KFK-786  H. Borgwaldt, H. Küsters, G. Kußmaul, H. Meister, K. Thurnay  
Reactor Dynamic Topics Recently Investigated at Karlsruhe

KFK-788  H. Borgwaldt  
Neutron Noise in a Reactor with an External Control Loop  
Nukleonik 11 (1968) S. 76-84

KFK-790  D. Smidt, P. Fette, W. Peppler, E.G. Schlechtendahl, G.F. Schultheiss  
Problems of Sodium Boiling in Fast Reactors

KFK-791  J.J. Schmidt  
Organizational and Technical Aspects in the Field of Neutron Nuclear Data Evaluation

KFK-792  H. Th. Klippel  
Control Rod Calculations for the Steam Cooled Fast Breeder Reactor D-1

Irradiation Effects on the Mechanical Properties of Vanadium-Base Alloys  
Effects of Radiation on Structural Metals Special Technical Publication No. 426, American Society for Testing and Materials
KFK-798  W. Schikarski
The Karlsruhe Research Program on Nuclear Aerosols and its
Relation to the Plutonium Hazard of Fast Sodium Reactors

KFK-808  G. Schmidt
Ein Rechenverfahren zur festigkeitsmäßigen Auslegung der
Brennstabhüllrohre bei flüssigmantalgalkühlen schnellten Reaktoren

KFK-811  R. Böhme, H. Seufert
Uranium Reaction Rate Measurements in the Steam-Cooled Fast
Reactor SNEAK, Assembly 3A-2

KFK-812  H.U. Borgstedt, I. Michael, St. Müller, G. Wittig
Untersuchung der Spannungsrißkorrosion von austenitischen
Stählen und Nickellegierungen

KFK-813  D. Braess, K. Thurney
Theoretische Behandlung hypothetischer, schwerer Unfälle
bei schnellten Leistungsreaktoren

KFK-814  H.J. Laue, H. Böhm, H. Hauck
Multi-axial In-Reactor Stress-Rupture Strength of Stainless
Steels and a Nickel-Alloy

KFK-815  K. Doetschmann
Drei FORTRAN-Programme zur Bestimmung der Heißkanaltemperaturen
in dampf- und gasgekühlten Reaktorkernen unter Berücksichtigung
der Kühlmittel-Quervermischung

KFK-817  D. Brucklacher, W. Dienst, F. Thümmler
Überlegungen zum Kriechen von UO₂ unter Neutonenbestrahlung

KFK-838  H. Böhm
Die Porenbildung in metallischen Werkstoffen durch Neutonen-
bestrahlung

KFK-840  S. Cierjacks, P. Forti, D. Kopsch, L. Kropp, J. Nebe
High Resolution Total Fast Neutron Cross Sections on Some
Non-Fissible Nuclei in the Energy Range 0.5 ≤ E_n ≤ 30 MeV
KFK-841  M. Dalle-Donne, E. Eisemann, F. Thümmel, K. Wirtz
High Temperature Gas Cooling for Fast Breeders

KFK-846  E. Kiesháber
Konsistente Behandlung des Einflusses der neuen Alpha-Werte
von Pu239 auf nukleare Kenngrößen unter Berücksichtigung der
Plutonium-Gleichgewichtsisotopenzusammensetzung (Pu₉₀)

KFK-848  K. Doetschmann
Kühlmittel-Quervermischung und deren Auswirkung auf die Heiß-
kanal-Temperaturen in dampfgekühlten Schnellen Reaktoren

KFK-852  W. Ochsenfeld
Wiederaufarbeitung der Brennelemente schneller Brutreaktoren
Atomwirtschaft, XIII (1968) Nr. 8-9, S. 422-425

KFK-1000  S. Cierjacks, P. Forti, D. Kopsch, L. Kropp, J. Nebe, H. Unseld
High Resolution Total Neutron Cross-Sections between 0.5 - 30 MeV
(KFK-1000 wird nur gegen Berechnung von DM 22.-- abgegeben)
Discussion of the Report by Dr. Schuster and Dr. Engelmann

Wensch: What is the division of effort on a gas-cooled reactor programme versus sodium in terms of dollars or marks or people involved?

Engelmann: The effort going into the gas cooling is very small as compared to the work on sodium. There are a few people working at Karlsruhe on gas cooling, but this work is at a very early stage.

If you have a look at the list of reports on fast reactors issued by the Karlsruhe centre, you will find only two reports on gas cooling: out of a total of about sixty. This reflects the division of effort in both sodium and gas cooled reactors.

Schuster: There was remarkable activity of ENEA last year in assessment of gas and steam cooled fast reactors and I think Mr. Boxer will speak on this. Our experts took part in these common efforts. But there is no doubt as yet that sodium will be a success. Considerations of gas and steam cooled reactors are regarded as a back solution in case there is a failure with sodium reactors.

Boxer: What kind of facility had you in mind when you mentioned feasibility studies on fast high flux test reactor with a flux of $1 \times 10^{16}$ n/cm$^2$ sec?

Engelmann: This is a very early stage of feasibility study. We have just started a couple of months ago considering what such a facility could look like. If we use for the driver zone helium-cooling, we are limited to $1 \times 10^{16}$ n/cm$^2$ sec in the flux. If we go to sodium cooling we could come up to $1.5 \times 10^{16}$ n/cm$^2$ sec. But nothing has been decided yet with respect to the choice of a coolant or even to whether we shall build such a facility. First results of this study will be available by the end of the year and perhaps next year we will be in a position to tell you something more.
Schuster: There is wide discussion in Europe about the need for a high power high flux test facility for the testing of carbide fuel. But this is a stage of the very early discussion. From our point of view a fast flux test facility could be a good chance for international co-operation.

Vautrey: May I ask Dr. Schuster to clarify his statement in which he compared loop design systems to integrated systems?

Schuster: The consortium feels that loop design as compared to the pool arrangement has a particular advantage of involving fewer technical risks and of being more suitable for a large plant. Two main points for choosing this design were to have a good accessibility to various components and the possibility of eliminating the intermediate sodium loop.

Engelmann: The question of which design is the best is still an open question. We can spend a long time discussing all the pros and cons of each design. This could be a subject for a specialists' meeting.

Vautrey: You mentioned that for the safety analysis you were thinking of using a probability method. Could you give us some clarification about the present stage of development in using this method at Karlsruhe?

Engelmann: Interatom is just now starting to look at this hazard analysis method with the probability approach. There is an easy way to design a maximum credible accident or design basis accident and to go along these lines. Probably for the SNR we have to follow this more classic approach. But we feel we get more insight into the safety problems if we have a kind of failure tree and assign probability factors and try to get it more qualitative. Of course one needs much more information in this case. Interatom has started a programme looking at what probability and reliability factors are available and which are not reliable. They have started some work especially in the field of control rod drives and failure of these mechanisms and others under sodium. But whether sufficient data will be available in time for the hazards analysis report of the SNR is not yet clear.
Vautrey: Could I ask you to make some comments on your system BEVUS?

Engelmann: This is a vessel containing a smaller volume with sodium. It also contains a bundle of electrically heated mock-ups of fuel element subassemblies. The pins have the same dimensions and cladding as real fuel elements. There is a pressure release system, and it is possible to simulate events happening under superheat and vaporisation of sodium. The aim of the facility is to investigate a possibility of a propagation of a fuel pin failure to the adjacent pins.

Vautrey: I wonder whether I understood correctly that you envisage having flow meters somewhere in the core.

Engelmann: There is an instrumentation plate within the reactor vessel above the core. These instruments are to be installed at this plate. Much discussion is taking place on how this could be achieved. It is not easy for these instruments to fulfil their purposes. This problem is still under discussion.

Smith: Could I ask you whether in loss of power to the pumps you consider sudden seizure of a pump or disconnection of the pump rotor from the shaft or failure of one of the ducts from the pump to the inlet of the core. This is something that represents the instantaneous stop of flow rather than a rundown of the pump due to electrical failure. Do you consider this under your probability analysis?

Engelmann: This accident has been discussed. I have no definite answer but can enquire from people at Interatom. Do you think it is necessary to include this analysis?

Smith: It is a very difficult accident to consider because you get sudden flow failure and you get back flow through that circuit. We find that an overheating of the core can be reached in a very short time with this particular accident. It can probably be satisfactorily disposed of on the probability basis. It is rather hard to dispose of it over maximum credible accident basis.
**Wensch:** On the basis of available experience with pumps, I believe this will be a very unusual situation because one obtains indications of pump functions rather early.

**Engelmann:** In our case we have three loops. The probability that all three fail by a mechanical failure is practically zero. This is the reason why this is not discussed in the report.

**Smith:** I agree that this is a most improbable accident. Nevertheless, it is hard to absolutely exclude the possibility of a seizure of pumps. These are three things which have a similar effect: one is an actual seizure of a pump, the other is the pump rotor detaching itself from the shaft in some way with the shaft breaking, the third is failure of the pipes to the inlet plenum or to the actual pump casing itself. It is not necessary for three pumps to fail simultaneously. For one pump to fail in this way in a three loop reactor will result in serious damage to the core and possible sodium boiling. This is why I said the solution lies in the probability analysis. All we can do is put a sufficiently small probability of its happening and hence convince ourselves in that way that the reactor is safe.

**Wensch:** It is obvious that safety considerations for FBR are quite sensitive as regard the way they are treated in different countries. The probability approach has been considered in the U.S. It would be a very sensitive issue because if the probability is low but the consequences of an accident are very great, the review authorities will tend to say no. They depend more almost upon the zero deflect analysis. You are acquainted with the procedures for getting a licence in the U.S. They are very stringent. I hope when your report comes out, Dr. Engelmann, that it does not cause a big problem in the U.S.

**Vautrey:** A question to Dr. Smith. Could you give us a few details about this consequence from the stopping of one pump if you have, for instance, three primary pumps.

**Smith:** The reason why I asked these questions is simply because I do not know the answer. If you do have a sudden cessation you will then have back flow through that circuit and the core flow with the three pump system may drop to about 60%
of the normal flow. The exact amount to which it drops depends on the design of pumps and the level at which cavitation starts. Whether or not one then gets sodium boiling depends on the design of the individual reactor. We find it very difficult to design pump circuits so that this does not occur.

Wensch: I do not quite understand this back flow situation. In our reactors we have check valves, which prevent flow reversal.

Smith: We also have check valves on our reactors. In any case it takes a finite time for the check valves to operate and you have to convince yourself that during the time taken for the check valve to operate, the temperature in the core does not rise too high. If the check valve is reasonably slow, and some check valves have dampers to avoid sodium hammer, it is possible to run into troubles before that check valve closes. Even if, in the normal situation, the check valve closes satisfactorily, perhaps consideration should be given to the case when it closes a little more slowly, or else fails to close at all.

Vautrey: As regards the method of analysing safety on the basis of probabilities, this method has been basically envisaged in the UK by Dr. Farmer. Could I ask whether Dr. Smith thinks that for fast reactor safety, numerical values could be available which are sufficiently valid to use this method.

Smith: This is a very good aid to design and it focuses attention on the areas where the most improvement is needed. At the present time it is very difficult to speak about safety in terms of certain figures. The probability analysis is a very good guide to design, and I personally believe that this will become the basis of safety documents, even if this is not possible today.

Wensch: We do have a programme which would, I think, be helpful to Dr. Engelmann. Anything which happens which is unplanned is reported on a standard form in which the cause of an incident is stated, saying what actions are necessary to remedy the situations. These reports go to ORNL and to the LMEC, so that
from the engineering point of view, someone can determine what are the common principles for malfunction.

Engelmann: I would like to ask Dr. Wensch whether this information is available to everyone.

Wensch: I will look into it and let you know.

Boyer: The kind of studies which Dr. Wensch mentioned are carried on within the CREST activity in ENEA. There were a number of meetings on reliability. Some people from Karlsruhe participated in this Committee and you could contact them and get a lot of information.

Smith: I believe that UK information is available through similar channels. We do collect the same information in the UKAEA. I think, however, that there is some danger in this. For example, we will have, I presume, fairly good data on the rate of failure of control rods in gas cooled reactors. Whether the same failure rate could be applied to control rods in sodium cooled reactors is, I think, anybody's guess.

Schuster: We have an institute on reactor safety in Cologne. They are collecting all this information. This institute receives information from the USA and the UK.

Vautrey: In France all incidents on Rapsodie are reported in weekly bulletins in a very detailed manner and every three months a report is issued. I would think that, taking into account what has been said about other countries, this is perhaps a subject which we might talk about a bit later on, when we come to the proposals on co-operation.
Fast Reactor Development in Japan

Masaaki KURAMOTO
Power Reactor & Nuclear Fuel Development Corporation (Japan)

1. The Power Reactor and Nuclear Fuel Development Corporation (PNC) was established on October, 1967 for the development of fast breeder reactor and advanced thermal reactor and this new organization absorbed the Atomic Fuel Corporation (AFC) which works the production, reprocessing of nuclear fuel and the exploitation, mining and refining of nuclear source materials.


3. JAEC expects the commercial uses of fast breeder power reactors around 1985. The sodium cooled type fast breeder reactors with plutonium and uranium mixed oxide fuels should be developed.

4. The prototype FBR will be a reactor of 200-300 Mwe output and it will be critical at around 1976. And the construction of experimental reactor with 100 MWT output will be started in 1969 and this reactor will be critical in 1973.

5. The experimental reactor has two big aims:
   (i) to get the construction experiences of fast reactor for the development of prototype reactor,
   (ii) to be able to use it as the irradiation facilities of fuel elements for fast reactors.

6. For this projects, Japanese Atomic Energy Commission Research Institute (JAERI), the governmental or local-governmental research institutes, private industries and
others should cooperate with PNC which is the nucleus of this development project.

7. The scientific and modern project management techniques and systems are applying and the project is managed rationally with grasping the status of progress and reviewing and evaluating the results of the development works.

8. The exchange of informations and personnel and other co-operations with foreign countries developing fast breeder reactors shall be actively performed in order to make a development effectively and economically within a shorter period.

9. JAEC expects that the at least half amount of money for the construction of prototype reactor shall be contributed by private enterprises.

10. The Hitachi Ltd. and JAERI made fundamental researches of sodium coolant technology, fast reactor physics, etc. during these ten years. In 1963, the fast critical facility (FCA) was authorized to build in Tokai Laboratory of JAERI and R & D on fast reactor was started.

11. The initial design of fast experimental reactor was done by JAERI in 1964-65. And the first conceptual design of the experimental reactor was also done by JAERI. JAERI and AFC (the former body of PNC) co-operated to develop plutonium and uranium mixed oxide fuel.

12. The second conceptual design by JAERI was finished on June 1968 and this design was transferred to PNC. PNC asked to French CEA the check of this design and comments on it. The third design will be completed by the end of March 1969. The construction of this experimental reactor will be started at the end of this year.
13. PNC will construct the fast prototype reactor from 1972 and make it critical in 1976. The initial design studies are under going.

14. The schedule of the development of fast reactor in Japan is shown in Fig.1.

14. The fast experimental reactor is sodium cooling type, with 100 MWt output and PuO₂ - UO₂ mixed fuel. This experimental reactor will construct for three purposes:

(i) to acquire the experiences to design, build and operate fast reactor,
(ii) to make the first step to develop commercial fast reactor,
(iii) to be used it as the irradiation bed of fuel elements for the development of future reactors.

15. The design is undergoing by the five japanese nuclear industrial groups as follows:

Fuji Denki K.K.  ° Participating to nuclear and thermal design,
° charge and discharge machine of fuel
° waste disposal equipment
° inert gas system
° sodium and NaK supply system
° various electric equipments

The Hitachi Ltd. ° participating to nuclear and thermal design
° reactor core: fuel assemblies
° reactor vessel: primary shield tank
° primary coolant system
Mitsubishi Atomic Industries K.K,
¢ participating to nuclear and
thermal design
¢ reactor upper mechanism:
rotary plug
¢ secondary coolant system

Sumitomo Atomic Industries K.K.
¢ participating to nuclear and
thermal design
¢ reactor core: fuel assemblies

Tokyo Shibaura Electric K.K.
¢ coordinating company of five
groups
¢ nuclear and thermal design
¢ anti-seismic design
¢ shielding design
¢ heat balance
¢ flow sheet
¢ reactor core (including reflector,\n  blanket fuel elements)
¢ driving mechanism of control and
  safety rods
¢ reactor container: reactor building
¢ instrumentation and control
  mechanism

18. The design of experimental reactor is principally conservative.\nThe coolant is metallic sodium in high temperature. This\ncoolant is very chemically active and having high heat\nconductivity compared with water or carbondioxide in conventional\ntype reactors as light water moderated reactor or gas cooled\ntype reactor. Then it is quite necessary to be secured in\nsafety construction.
17. The R & D on main components and nuclear thermal transfer will be mainly concerning to the confirmation tests for this experimental reactor. The design of the experimental reactor is very conservative, but for the confirmation of its reliability and safety, the full size mock-up tests are expecting to be done. Even in the case of no problem in design, the full-size or certain size mock-up tests should be tried.

18. The main components of fast experimental reactor would be expected to be developed by Japanese technology. But small equipments or components, which the development takes long period, might be imported or introduced foreign technology.

19. The irradiation tests of fuelmatertials are very important and PNC wishes to use foreign facilities as Rapsodie, Dourneray, (France) (UK), Enrico Fermi Reactor, (USA)
DEVELOPMENT PROGRAM OF FAST BREEDER REACTOR

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<td>48</td>
<td>49</td>
<td>50</td>
<td>51</td>
</tr>
</tbody>
</table>

**Experimental Reactor**
- Design No. 2
- Design No. 3
- Design No. 4
- Construction
- Critical
- Low power
- Operation
- Irradiation of fuel

**Components**
- Design
- R & D
- Tests
- mfg.
- R & D

**Na technology**
- Constr. of loop
- R & D
- R & D

**Reactor physics**
- Research
- Mock up
- Pu-mock-up
- R & D

**Safety**
- R & D
- R & D
- R & D

**Fuel**
- R & D
- R & D
- R & D

**Prototype Reactor**
- Design No. 1
- Design No. 2
- Design No. 3
- Construction
- Critical
Experimental Fast Reactor of PNC

1. The design of the experimental fast reactor of PNC is now under the final stage. The purpose of the reactor is to obtain the experience with a fast reactor and to provide fast neutron flux for the development of fast reactor fuel, in order to facilitate the design and construction of the prototype fast reactor being planned and larger fast reactors to be constructed in the future.

The construction of this reactor is expected to be started late this year and the criticality is anticipated to be achieved late 1973 or early 1974.

2. The second conceptual design of the reactor using mixed oxide fuel of plutonium and uranium was completed by the JAERI in the last spring.

Based on the conceptual design, further design work was started by the PNC in collaboration with the five major Japanese industrial groups and the JAERI. The design work is nearly completed.

In parallel with the endeavors, review of the second conceptual design was made by the French CEA under special contract. Suggestions and advices from the reviewing facilitated the current design work.

The hazard report of the reactor is being prepared and the safety committee of the AEC will be opened in near future.

The reactor will be constructed in the Ohara site of the PNC and excavation for the reactor is expected to begin late this year.

3. Main characteristics of the reactor are in the following table including the nuclear properties of core, the thermal hydraulic characteristics, the reactor cooling system parameters, the mechanical dimensions of main reactor components and the weights.
Detailed examinations of the fuel design and core characteristics show that the reactor power is to be about 90 MWt at the initial state of the core in the present design. It is estimated that the power is to be able to increase up to 100 MWt adding the fuel subassemblies to make up burn-up reactivity losses. This is mainly because of the improvement in the peaking factor of the power distribution in the core. At the equilibrium state of the burn-up, the 100 MWt output is to be attained.

The 110% over power factor for the reactor may not necessarily be appropriate in view of the safety margin of the operation. Main limitations come from the design criteria of the temperatures limits on the clad temperature. These criteria are subject to change according to the development of fast reactor technology.


Main Characteristics of the Reactor

early core

1st loading initial condition 2nd loading initial condition

1. Reactor thermal power
   Total output 92 MWt 95 MWt
   Core output 82 MWt 85 MWt
   Axial blanket output 2 MWt 2 MWt
   Radial blanket output 8 MWt 8 MWt
2. Na Coolant System

Number of Cooling Circuit
(Primary/Secondary) 2

Number of Pumps in Primary Circuit 1/Circuit

Number of Intermediate Heat Exchanger 1/Circuit

Number of Pumps in Secondary Circuit 1/Circuit

Number of Main Air Cooler 1/Circuit

3. Core, Blanket and Reflector

Dimensions

Core Height 600 mm
Equivalent Core Diameter 710 mm
Core Volume 240 &
Axial Blanket Height 400 mm
Radial Blanket Height 1,650 mm
Effective Radial Blanket Thickness 300 mm
Effective Reflector Thickness 140 mm

Number of Assemblies

Core Fuel (with Axial Blanket) 61
Inner Radial Blanket 54
Outer Radial Blanket 137
Movable Reflector 42
Fixed Reflector 72
Coarse Control Rod 6
Fine Control Rod 2
Safety Rod 4
Neutron Source Rod 1
### Compositions (Volume Ratios)

**Core**
- **Fuel** ($\text{PuO}_2$-$\text{UO}_2$) 39%
- **Coolant** (Na) 38%
- **Structure Steel** (AISI-316) 21%

**Radial Blanket**
- **Fuel** ($\text{UO}_2$) 49%
- **Coolant** (Na) 30%
- **Structure Steel** (AISI) 18%

**Axial Blanket**
- **Fuel** ($\text{UO}_2$) 39%
- **Coolant** (Na) 38%
- **Structure Steel** (AISI) 21%

**Radial Reflector**
- **Reflector** (AISI-316) 95%
- **Coolant** (Na) 5%

**Axial Reflector**
- **Reflector Steel** (AISI-316) 41%
- **Coolant** (Na) 38%

#### Early Core

<table>
<thead>
<tr>
<th>(1st loading)</th>
<th>(2nd loading)</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core Fuel Material</td>
<td>$\text{PuO}_2$-$\text{UO}_2$</td>
</tr>
<tr>
<td>Blanket Fuel Material</td>
<td>$\text{UO}_2$</td>
</tr>
<tr>
<td>Core Fuel Enrichment</td>
<td>16 w/o</td>
</tr>
<tr>
<td>$\text{PuO}_2/(\text{PuO}_2+\text{UO}_2)$</td>
<td>20 a/o</td>
</tr>
<tr>
<td>$\nu^{235}/\nu$</td>
<td>0.7 a/o</td>
</tr>
<tr>
<td>Blanket Fuel Enrichment</td>
<td></td>
</tr>
<tr>
<td>$\nu^{235}/\nu$</td>
<td></td>
</tr>
</tbody>
</table>
## Isotopic Composition

### Core Fuel
- **Pu 239**: 70 a/o
- **Pu 240**: 25 a/o
- **Pu 241**: 5 a/o
- **U235**: 20 a/o
- **U238**: 80 a/o

### Blanket Fuel
- **U235**: 0.7 a/o
- **U238**: 99.3 a/o

## Effective Densities

<table>
<thead>
<tr>
<th></th>
<th>PuO₂-UO₂ (20°C)</th>
<th>UO₂ (Blanket) (20°C)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>PuO₂</strong></td>
<td>0.94 T.D.</td>
<td>0.94 T.D.</td>
</tr>
<tr>
<td></td>
<td>(=10.39g/cm³)</td>
<td>(=10.30g/cm³)</td>
</tr>
<tr>
<td><strong>UO₂</strong></td>
<td>0.84 T.D.</td>
<td>0.84 T.D.</td>
</tr>
<tr>
<td></td>
<td>(=9.30g/cm³)</td>
<td>(=10.30g/cm³)</td>
</tr>
</tbody>
</table>

## O/K Ratio
- **PuOₓ x**: 1.98
- **UOᵧ y**: 1.98

## Inventory in the Core
- **PuO₂**: 680 kg
- **UO₂**: 150 kg

## Nuclear Characteristics

### Critical Mass
- **Pu 239**: 92 kg
- **Pu 240**: 33 kg
- **Pu 241**: 7 kg
- **U 235**: 120 kg
- **U 238**: 460 kg
<table>
<thead>
<tr>
<th>Property</th>
<th>Inner</th>
<th>Outer</th>
</tr>
</thead>
<tbody>
<tr>
<td>Pu239+Pu240*Pu241</td>
<td>132 kg</td>
<td>151 kg</td>
</tr>
<tr>
<td>U235+U238</td>
<td>600 kg</td>
<td>502 kg</td>
</tr>
<tr>
<td>Total Breeding Ratio</td>
<td>1.07</td>
<td>1.06</td>
</tr>
<tr>
<td>Inner Breeding Ratio</td>
<td>0.24</td>
<td>0.22</td>
</tr>
<tr>
<td>Outer Breeding Ratio</td>
<td>0.83</td>
<td>0.86</td>
</tr>
<tr>
<td>Core Average Neutron Density</td>
<td>$2.3 \times 10^{15}$ n/cm$^2$sec</td>
<td>$2.4 \times 10^{15}$ n/cm$^2$sec</td>
</tr>
<tr>
<td>Peak to Average Power Production Ratio</td>
<td>1.75</td>
<td>1.70</td>
</tr>
<tr>
<td>Core Average Neutron Energy</td>
<td>300 keV</td>
<td>310 keV</td>
</tr>
</tbody>
</table>

Isothermal Temperature Coefficient (at 350°C)

**Doppler**

- Core: $-2.6 \times 10^{-6}$ Δ $k/k/°C$
- Radial Blanket: $-1.5 \times 10^{-6}$ Δ $k/k/°C$
- Axial Blanket: $-0.9 \times 10^{-6}$ Δ $k/k/°C$

**Fuel Axial Expansion**

- Core: $-2.02 \times 10^{-6}$ Δ $k/k/°C$
- Radial Blanket: $-0.19 \times 10^{-6}$ Δ $k/k/°C$
- Axial Blanket: $-0.04 \times 10^{-6}$ Δ $k/k/°C$

**Sodium Expansion**

- Core: $-9.34 \times 10^{-6}$ Δ $k/k/°C$
- Radial Blanket: $-1.78 \times 10^{-6}$ Δ $k/k/°C$
- Axial Blanket: $-1.78 \times 10^{-6}$ Δ $k/k/°C$

**Structural Expansion**

- Core: $-13.05 \times 10^{-6}$ Δ $k/k/°C$
- Radial Blanket: $-1.50 \times 10^{-6}$ Δ $k/k/°C$
- Axial Blanket: $-1.69 \times 10^{-6}$ Δ $k/k/°C$

**Power Coefficient**

- $-73.3 \times 10^{-6}$ Δ $k/k/°C$
### Reactivity Change due to Burn Up

<table>
<thead>
<tr>
<th></th>
<th>$-1.66 \times 10^{-4} \Delta k/k/day$</th>
<th>$-1.64 \times 10^{-4} \Delta k/k/day$</th>
</tr>
</thead>
<tbody>
<tr>
<td>Prompt Neutron life Time</td>
<td>$2.0 \times 10^{-7}$ sec</td>
<td>$2.0 \times 10^{-7}$ sec</td>
</tr>
<tr>
<td>Effective Delayed Neutron Fraction</td>
<td>0.0051</td>
<td>0.0047</td>
</tr>
<tr>
<td>Maximum Design Burn Up</td>
<td>2.5 a/o</td>
<td>5 a/o</td>
</tr>
</tbody>
</table>

### Thermal Hydraulic Characteristics

1. **Primary Coolant System**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Reactor Inlet Temperature</td>
<td>370°C</td>
<td>370°C</td>
</tr>
<tr>
<td>Reactor Outlet Temperature</td>
<td>500°C</td>
<td>500°C</td>
</tr>
</tbody>
</table>

2. **Coolant Flow**

<table>
<thead>
<tr>
<th>Component</th>
<th>Flow Rate</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total</td>
<td>554.8 kg/sec</td>
</tr>
<tr>
<td>Core</td>
<td>426.0 kg/sec</td>
</tr>
<tr>
<td>Radial Blanket</td>
<td>84.4 kg/sec</td>
</tr>
<tr>
<td>Leakage</td>
<td>44.4 kg/sec</td>
</tr>
<tr>
<td>Total Pressure Drop</td>
<td>4.4 kg/cm²</td>
</tr>
</tbody>
</table>

3. **Core**

<table>
<thead>
<tr>
<th>Temperature Parameter</th>
<th>Nominal</th>
<th>Hot Spot</th>
</tr>
</thead>
<tbody>
<tr>
<td>Maximum Coolant Temperature</td>
<td>587 °C</td>
<td>626 °C</td>
</tr>
<tr>
<td>Maximum Clad Temperature</td>
<td>596 °C</td>
<td>642 °C</td>
</tr>
<tr>
<td>Maximum Fuel Temperature</td>
<td>2091 °C</td>
<td>2449 °C</td>
</tr>
</tbody>
</table>

4. **Maximum Flow Rate per Assembly**

| Rate                | 9.1 kg/sec | 9.3 kg/sec |

5. **Maximum Flow Velocity at Element Gap**

<p>| Velocity            | 5.0 m/sec  | 5.1 m/sec  |</p>
<table>
<thead>
<tr>
<th>Parameter</th>
<th>Average</th>
<th>Maximum</th>
</tr>
</thead>
<tbody>
<tr>
<td>Assembly Total Pressure Drop&lt;sup&gt;1)&lt;/sup&gt;</td>
<td>3.0 kg/cm^2</td>
<td>3.1 kg/cm^2</td>
</tr>
<tr>
<td>Fuel Linear Heat Rate</td>
<td>247 W/cm</td>
<td>255 W/cm</td>
</tr>
<tr>
<td>Power Output per Assembly</td>
<td>1.34 MW</td>
<td>1.38 MW</td>
</tr>
<tr>
<td>Power Density&lt;sup&gt;2)&lt;/sup&gt;</td>
<td>401 kW/ℓ</td>
<td>413 kW/ℓ</td>
</tr>
<tr>
<td>Radial Blanket</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maximum Coolant Temperature</td>
<td>591 °C</td>
<td>591 °C</td>
</tr>
<tr>
<td>Nominal Maximum Hot Spot</td>
<td>591 °C</td>
<td>591 °C</td>
</tr>
<tr>
<td>Max Clad Temperature</td>
<td>637 °C</td>
<td>637 °C</td>
</tr>
<tr>
<td>Nominal Maximum Fuel Temperature</td>
<td>958 °C</td>
<td>958 °C</td>
</tr>
<tr>
<td>Max Fuel Temperature</td>
<td>1074 °C</td>
<td>1074 °C</td>
</tr>
<tr>
<td>Maximum Flow Rate per Assembly</td>
<td>0.9 kg/sec</td>
<td>1.0 kg/sec</td>
</tr>
<tr>
<td>Volume</td>
<td></td>
<td></td>
</tr>
<tr>
<td>Maximum Flow Rate at Element Gap</td>
<td>0.69 m/sec</td>
<td>0.74 m/sec</td>
</tr>
<tr>
<td>Assembly Total Pressure Drop&lt;sup&gt;3)&lt;/sup&gt;</td>
<td>0.11 kg/cm^2</td>
<td>0.13 kg/cm^2</td>
</tr>
<tr>
<td>Maximum Linear Heat Rate</td>
<td>141 W/cm</td>
<td>141 W/cm</td>
</tr>
</tbody>
</table>

1. Including Orifice Pressure Drop
2. Core Power Output/Core Volume (Fuel Assembly)
3. Including Orifice Pressure Drop
Secondary Coolant System

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
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<tbody>
<tr>
<td>Intermediate Heat Exchanger Primary Inlet Temp.</td>
<td>500 °C</td>
</tr>
<tr>
<td>&quot; Primary Outlet Temp.</td>
<td>370 °C</td>
</tr>
<tr>
<td>&quot; Secondary Inlet Temp.</td>
<td>340 °C</td>
</tr>
<tr>
<td>&quot; Secondary Outlet Temp.</td>
<td>470 °C</td>
</tr>
</tbody>
</table>

Coolant Flow

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Primary Flow Rate</td>
<td>$2 \times 1.075 \times 10^6$ kg/hr</td>
</tr>
<tr>
<td>Secondary Flow Rate</td>
<td>$2 \times 1.075 \times 10^6$ kg/hr</td>
</tr>
</tbody>
</table>

Air Cooler

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Sodium Inlet Temp.</td>
<td>470 °C</td>
</tr>
<tr>
<td>&quot; Sodium Outlet Temp.</td>
<td>340 °C</td>
</tr>
<tr>
<td>&quot; Air Inlet Temp.</td>
<td>30 °C</td>
</tr>
<tr>
<td>&quot; Air Outlet Temp.</td>
<td>333 °C</td>
</tr>
</tbody>
</table>

7. Fuel Element (Pin)

Core Fuel Element

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Type</td>
<td>Integral and Sealed</td>
</tr>
<tr>
<td>Pellet Diameter</td>
<td>5.5 mm</td>
</tr>
<tr>
<td>Pellet Height</td>
<td>10 mm</td>
</tr>
<tr>
<td>Clad Material</td>
<td>AISI-316 Stainless Steel</td>
</tr>
<tr>
<td>Clad Outer Diameter</td>
<td>6.3 mm</td>
</tr>
<tr>
<td>Clad Thickness</td>
<td>0.35 mm</td>
</tr>
<tr>
<td>Total Pin Length</td>
<td>1,830 mm</td>
</tr>
<tr>
<td>Gas Plenum Length</td>
<td>200 mm</td>
</tr>
<tr>
<td>Fuel Pin Pitch</td>
<td>7.69 mm</td>
</tr>
<tr>
<td>Spacer Type</td>
<td>Grid and Spiral wire</td>
</tr>
<tr>
<td>Number of Pins/Assembly</td>
<td>91</td>
</tr>
<tr>
<td>Total Fuel Element Weight</td>
<td>0.42 kg</td>
</tr>
</tbody>
</table>

Radial Blanket Element

<table>
<thead>
<tr>
<th>Description</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Type</td>
<td>Integral and Sealed</td>
</tr>
<tr>
<td>Description</td>
<td>Value</td>
</tr>
<tr>
<td>-------------------------------------------------</td>
<td>-----------</td>
</tr>
<tr>
<td>Pellet Diameter</td>
<td>13.5 mm</td>
</tr>
<tr>
<td>Pellet Height</td>
<td>15 mm</td>
</tr>
<tr>
<td>Clad Material</td>
<td>SUS-32</td>
</tr>
<tr>
<td>Clad Outer Diameter</td>
<td>15 mm</td>
</tr>
<tr>
<td>Clad Thickness</td>
<td>0.6 mm</td>
</tr>
<tr>
<td>Total Pin Length</td>
<td>1,030 mm</td>
</tr>
<tr>
<td>Gas Plenum Length</td>
<td>80 mm</td>
</tr>
<tr>
<td>Pin Pitch</td>
<td>16.47 mm</td>
</tr>
<tr>
<td>Spacer Type</td>
<td>Spiral wire</td>
</tr>
<tr>
<td>Number of Pins/Assembly</td>
<td>19</td>
</tr>
<tr>
<td>Total Element Weight</td>
<td>2.9 kg</td>
</tr>
</tbody>
</table>

8. Fuel Assembly

Core Fuel Assembly

- Distance Across Flats inside Wrapper Tube: 75 mm
- Thickness of Hexagonal Wrapper Tube: 1.8 mm
- Total Length: 2,800 mm
- Total Weight: 64 kg

Radial Blanket Assembly

- Distance Across Flats inside Wrapper Tube: 75 mm
- Wrapper Tube Thickness: 1.8 mm
- Total Weight: 67 kg

9. Control Rods

- Number of Rods
  - Fine Control Rods: 2
  - Coarse Control Rods: 6
  - Safety Rods: 4
- Number of Absorber Pins per Rods: 7
| Reactivity Worth/Fine Control Rods | 0.5 S |
| Reactivity Worth/Coarse Control Rods | 0.67 S |
| Reactivity Worth/Safety Rods | 2.0 S |
| Absorber Material | B₄C |
| Clad Material | AISI-316 |

### Dimensions of Fine and Coarse Control Rods

| Absorber Pin Length | 300 mm |
| Absorber Pin Diameter | 16 mm |

| Absorber Length | 700 mm |
| Absorber Diameter | 16 mm |

### Driving Speed

<table>
<thead>
<tr>
<th>Fine Control Rods</th>
<th>Coarse Control Rods</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>Stationary</strong></td>
<td>500 mm/min</td>
</tr>
<tr>
<td><strong>Scram</strong></td>
<td>40,160 mm/min</td>
</tr>
<tr>
<td><strong>Safety</strong></td>
<td></td>
</tr>
<tr>
<td><strong>Stationary</strong></td>
<td>40,160 mm/min</td>
</tr>
<tr>
<td><strong>Scram</strong></td>
<td>Less than 500 msec</td>
</tr>
</tbody>
</table>

(Up to the Center of Core)

(Up to Full Length Insertion)

### Reflector

<table>
<thead>
<tr>
<th>Material</th>
<th>AISI-304 or AISI-51</th>
</tr>
</thead>
<tbody>
<tr>
<td>Distance Across Flats</td>
<td>78.6 mm</td>
</tr>
<tr>
<td>Total Length</td>
<td></td>
</tr>
<tr>
<td>Movable Reflector</td>
<td>2,800 mm</td>
</tr>
<tr>
<td>Fixed Reflector (5-1)</td>
<td>2,800 mm</td>
</tr>
<tr>
<td>(5-2)</td>
<td>2,190 mm</td>
</tr>
</tbody>
</table>
Fixed Reflector (5-3) (5-4) 2,290 mm 2,190 mm

Weight
Movable Reflector 89 kg
Fixed Reflector 83 kg

II. Reactor
Vessel
Type Double Wall
Inner Diameter 4,000 mm
Height 11,525 mm
Wall Thickness 25 mm (Inner)/15 mm (Outer)
Material AISI-316/AISI-304
Weight (vessel + Jacket) 45 ton

Core Support Plates
Type Double Plates, High Pressure Plenum Fixed Orificing Device
Material AISI-316

In-Vessel Structure
Material AISI-316

Reactor Upper Structure

Upper Shield Plug
Type Double Rotating Plug
Shield Material Stainless Steel, Graphite, Steel
Coolant \( N_2 \) gas
Total Weight 103 ton (Larger Outer)/ 53 ton (Small Inner)
12. Fuel Handling System

- Fuel Exchange Type: Linear Telescope on Plug
- Type of Load Unload Machine: Rotating Twin Coffins
- Type of In-Vessel Fuel Storage: Rotating Rack Around Core
- Maximum In-Vessel Storage Capacity: 32 Assemblies
- Type of Fuel Transfer Rotor: Suspended Rotor
- Maximum Transfer Rotor Storage Capacity: 40 Assemblies

Accessories
- Fuel Transfer Pot
- Periscope
- Sweeper

13. Container

- Type: Double Container
- Height
  - Above Ground Level: 33.0 m
  - Below Ground Level: 21.5 m
- Inner Diameter: 26 m
Discussion of the Report by Mr. Kuramoto

Wensch: I am interested in hearing your comments on a heat exchanger for 100 MW reactor. You are going to remove the heat using air. Do you consider this component developmental or do you consider it to be engineered and readily available?

Kuramoto: At this time the industrial people do not think this will be a difficult problem.

Wensch: Really one must look at an air heat exchanger as a system, because it is quite complex. Otherwise one can encounter temperature oscillations in the reactor which make the reactor very difficult to control. We had experience in air heat exchangers on our reactors and we had a very difficult time maintaining stable operation. I should broaden my question asking: do you look upon your heat exchanger as a good way to remove heat for a very sizeable reactor such as 100 MW(th)? How do you expect to engineer this system to make the heat removal constant?

Kuramoto: I think there are some difficulties in constructing such a system. We shall make some experiments in this respect. At this moment we believe that these problems can be solved.

Vautrey: I would like to add a short comment on this question to say that I am rather surprised by Dr. Wensch's question, because if I think of Rapsodie, which as you know, is cooled solely by air exchangers, we do not have any problem of the kind you mentioned. These terminal air heat exchangers were provided with an automatic temperature control. In fact, they are so stable and influence of air temperature is so slow and so low on the temperature in the reactor that this automatic control was eliminated and now these air exchangers are controlled solely using a few simple manual movements which do not take place very often.

Wensch: The experience I was referring to was one we had with the Sodium Reactor Experiment. Its air heat dump capacity was 20 MW(th) and the air flow was controlled. We finally had to
modify the circuit and use a steam-generator to get more uniform temperature distribution within the reactor. We are investigating this problem much more thoroughly today because FFTF are likely to use air heat exchangers to remove 400 MW of heat energy. There is very good behaviour of this system in Rapsodie and our systems people would be very much interested in looking at these systems.

Schuster: I did not get the figures about the financial means. I noticed that $20 million were allotted for 1969. And for 1968?

Kuramoto: For 1969 $8 million has been allotted.

Schuster: What is an idea of the total cost of the fast experimental reactor in Japan?

Kuramoto: The fast experimental reactor will cost $47 million. This is in addition to the $20 million for research and development this year.

Schuster: What did you mean when you said that PNC will construct the experimental fast reactor?

Kuramoto: PNC is a semi-governmental body, more than 90% owned by the Government, and they have a responsibility to construct an experimental and prototype fast reactor. PNC has contacts with industrial groups for making design studies which are undertaken by a special group of people from PNC and industry. The Toshiva is the main contractor which makes the actual design.

Schuster: There was French supervision of your fast reactor design. Have you some interesting remarks from CEA?

Kuramoto: These considerations have not been finished yet. We are expecting a final report shortly.
March, 1969

REVIEW OF FAST REACTOR PROGRESS IN THE U.K.

1968 - 1969

by R.D. Smith

INTRODUCTION

During the last year construction of the PFR has made good progress. The reactor tank is in place and all the buildings are essentially complete, though there have unfortunately been some welding difficulties in the construction of the reactor roof which will lead to some delay in the completion of the reactor.

Design studies for subsequent fast reactors have been extended to include 600 MW(E) reactors as well as the larger 1300 MW(E) units studied previously.

The Dounreay Fast Reactor has been brought back into operation after successful repair of the leak in one of the primary coolant pipes and our fuel irradiation programme is now going ahead rapidly.

PHYSICS

A series of plutonium fuelled test zones has been built into Zebra. The composition of each zone was adjusted so that the k-inf was near to unity. The P.C.T.R. technique in which removal of central lattice cells is required to give near zero reactivity change, has been used to test that the required condition for k-inf was met. In these test zones, particular attention has been given to keeping the number of regions in a lattice cell small so that the heterogeneity can be calculated by rigorous methods. The important capture and fission events have been measured in the appropriate plates using foil techniques, and particular attention has been given to reducing systematic errors wherever possible and to checking measurements with alternative techniques. For example, solid state track recorders have been used to check on fission rates normally measured from fission product activity in foils. The foils are calibrated in a separate experiment in a dummy fission chamber. Great importance is attached to accurate measurement of the U.238 (n, X) reaction. An absolute technique is now being used to measure U.238 capture events using Am-243 to calibrate the coincidence counter; this is an alternative to the previous Zebra technique in which a thermal neutron calibration is used. Good agreement between the two methods has been achieved.

In the assemblies studied the neutron spectrum has been measured using time of flight techniques, proportional counter measurements and Li-6 solid state spectrometers. The proportional counter measurements have been extended down to energies of 3 KeV using pulse rise time discrimination. Most data has however come from measurements using the 14 MeV pulsed electron linear accelerator. This accelerator has a beam of 200 ma and gives a yield of 5 x 10^14 neutrons/sec. in the pulse from a water cooled uranium-molybdenum target. There is a 200 metre flight tube and Li.6 and Boron scintillator detectors are used. Results obtained so far have been encouraging. The main difficulty has been in the accurate determination of the variation of sensitivity with energy of the detectors used.

All these results are being used to test current differential nuclear data, and, coupled with existing integral data, are being used to adjust the data, within the errors, to provide the best fit to all the measurements.
The F.D.4. data set has been brought into general use during the year and comparisons between calculated and experimental values have been made for a large number of assemblies. In general the higher values of plutonium alpha reported last year have been confirmed, but a review of the present position will be made at the specialist meeting in June this year.

Detailed analysis of the Zebra PFR mock-up experiment has continued and resulted in revised predictions of physics parameters for the P.F.R., together with better estimates of the uncertainties in these predictions.

**FUEL**

The fuel irradiation programme so far completed has successfully endorsed the reference design mixed PuO$_2$ - UO$_2$ fuel element up to a maximum burn-up of 7.5%. Based on this reference design of fuel element a new fabrication plant was developed and the construction of this plant is now well advanced. Sections of the plant will be handed over for trials during this year and manufacture of the first charge for D.F.R. will commence in 1970.

The continuing irradiation programme in D.F.R. includes a number of important variations on the reference fuel. These include the use of differing cladding materials, can thicknesses and fuel densities, but owing to the interruption of the programme by the leak it will be some time before these experiments have completed the full cycle irradiation cooling and examination.

At the present time the reactor is on run 62 which started on February 6th and is due to be completed on April 2nd. The reactor contains a total of 59 experiments in the core and 41 in the blanket. Of these 9 are for overseas countries including one central subassembly. These irradiations cover topics of interest both to fast reactors and to thermal reactors, for which the high damage rate obtainable in D.F.R. is of major importance.

**MATERIALS IN SODIUM CIRCUITS**

Owing to the prolonged shutdown of D.F.R. little progress has been made on the effect of fast neutron irradiation on the physical and mechanical properties of fuel cladding materials. Further work has been performed on the phenomenon of void formation using a Heavy Ion Accelerator. Voids have been produced in type 316 stainless steel which had previously been irradiated at 20°C with He$^+$ ions of various energies so as to produce a reasonably uniform distribution of implanted helium atoms to a depth of some 3000 Å at an average concentration of between $10^6$ and $10^7$. Subsequent irradiation in the temperature range 450-600°C with 100 keV protons, carbon ions or iron ions in the Harwell 150 keV Heavy Ion Accelerator produced voids in the helium implanted steel. After all cases of bombardment with carbon or iron ions and in most cases after proton bombardment, cavitation was only observed when the steel had previously been implanted with helium. The work indicates that the presence of helium or hydrogen is necessary before void formation will occur in type 316 steel. It is thought that the voids result from the growth of helium bubble nuclei under conditions of vacancy supersaturation in an irradiation environment.

Further information has been obtained on the effect of fast neutron irradiation on the creep-rupture properties of type 316 stainless steel tubes. D.F.R. irradiation produced a loss of rupture ductility, but in general no significant change in creep strength to biaxial tests of solution-treated material at 650°C. Cold worked tubes suffered loss of creep strength at 550°C. Irradiation of solution-treated tubes at 650°C in D.M.T.K. caused a considerable loss of high temperature ductility and a significant reduction in creep rate both during and after irradiation.
It is believed that an irradiation-induced carbide precipitation process is responsible for the increase in creep strength.

Further work on the compatibility of circuit materials in sodium has confirmed previously published data. It is clear that the mass transfer of stainless steel in oxygen-containing sodium is very complex and involves a number of processes, some of which may interact upon each other. Mass transfer rates in isothermal loops do not differ widely from those observed at the highest temperature in more complex non-isothermal loops. Therefore, clearly, the most important factor is the oxygen level in the circuit. Preliminary work on vanadium alloys shows that low oxygen contents are required in the sodium in order to avoid high corrosion rates. The work indicates that alloys with high Ti, Zr and Al contents are unlikely to be satisfactory in sodium unless the sodium has been hot trapped to remove oxygen.

DONREAY FAST REACTOR

At this time last year the leak in the primary vessel of the D.F.R. had been located and preparations for its repair started. A two metres long section of 10 cm diameter pipe within the neutron shield was replaced and welded in place using a specially developed internal welding machine. The welding parameters had previously been determined on a full scale mock up where a large number of welds were done under conditions similar to those in the vault. All the remaining welds were made using standard orbital welding equipment. During the whole operation the argon atmosphere in the reactor vessel was maintained, so that many operations had to be done in glove boxes. Expandable plugs were used to seal the repair section wherever possible. Some trouble was experienced when joining some of the hot trap pipes due to NaK contamination of the welds. This was overcome by using tightly fitting plugs of solid sodium that were easily dispersed after the weld was complete by warming the outside of the tube. In addition to replacing the faulty section of pipe in the affected primary circuit the small pipes from the other seven hot trap circuits were blanked off. A total of 148 welds was completed within the vault of which 22 were on the primary circuit itself and the rest on the leak jacket.

After pressure and leak tests the reactor was recommissioned on June 19th 1968. Initially the oxide content of the core remained high at about twice the normal operating level of 15 ppm. After a period of power operation, purification of the NaK and tests of corrosion of the driver charge cladding, the liquid metal condition was satisfactory for loading experimental rigs.

The reactor returned to normal operation in September and operated satisfactorily. Shortly after full power was reached analysis of the argon blanket gas showed unambiguous presence of radon. The only source of radon in the reactor was from the radium tracer inserts in some of the experimental pins in the reactor. The suspect experiments were removed and failures were located in two of them. Each of the failures was at the upper end of a pin (coolant inlet end) and was typical of failure due to local gas bubble blanketing of the heat transfer surface probably due to gas entrainment. A check of selected highly rated experiments indicated that no other failures had occurred.

Acoustic measurements of the 24 individual primary coolant circuits showed that two circuits transported gas bubbles into the reactor when the primary electro-magnetic pumps were being brought to full flow. It was found that in these two circuits, which are different to the remainder in that they do not contain expansion tanks, gas entrainment occurs when the NaK rises past an outlet pipe in the purification traps as flow is increased. An operating technique to flood these outlet pipes continuously while the primary flow was being raised was evolved. Acoustic measurements taken throughout a full operating cycle showed that gas entrainment from the two circuits had been cured. This has been confirmed by the successful irradiation of a full range of experiments including a subassembly.
At the beginning of December when the reactor was shut down to recharge fuel and change experiments, the top rotating shields stuck due to NaK entering the mercury gas seals. The resultant alloy was dissolved by prolonged purging of the seals with clean mercury. The next run commenced in February and the reactor has operated normally since that date.

**Prototype Fast Reactor**

Work on the major buildings has proceeded to plan so that the Reactor and Decontamination Hall, Steam Generator Building, the Turbine Hall and Electrical Annex are now weather tight. The Sea Water Pump House is complete and the installation of equipment has commenced. The N.S.H.E.E. have commenced the preparation of the transformer and switchgear site and erection of the first ten miles of 275 kv transmission line is completed.

Whilst the civil engineering work was being carried out on site, contracts were placed with British industry for all the major engineering sections of the reactor and steam generation plant. The concrete reactor vault, 12 m. diameter 15 m deep, is complete and the steel leak jacket, fabricated on site, has been inserted into the vault. The primary tank which will contain the whole of the primary circuit of the reactor is complete except for a small amount of final welding.

The components of the core support structure are now being assembled at D.E.R.E. into a single unit ready for insertion into the primary tank. Fabrication of the reactor jacket which separates the hot and cold sodium pools is proceeding to programme.

Difficulties arose in the fabrication of the biological shield roof which is a complex fabricated steel structure forming the top closure of the reactor from which are suspended the heat exchangers, pumps and reactor core. The examination of completed welds by ultrasonic techniques indicated 'pull out' in the plate material underlying the weld. The principal cause of the 'pull out' is the presence of laminar weaknesses in the plate material, which escaped detection during comprehensive non-destructive testing methods employed in the inspection of the plate since at that stage they had not opened up. The defective welds are being treated by chipping out the metal in the faulty areas and refilling with weld metal.

Tests on the components of the fuel handling machine have now been completed. The various bearings, universal joint and ball, nut and screw mechanisms have all proved satisfactory and a minimum maintenance-free life of five years is predicted for the in-pile operation of the fuel-handling machine. The manufacture of the actual fuel handling system for the reactor is almost completed and this is being subjected to trials at the manufacturers prior to delivery to site. In the Reactor Engineering Laboratory at Risley a saddle coil flow meter has been calibrated against a British Standard Venturi using the mechanical pump rig over a range of sodium flows from 1,200 to 6,500 gallons per minute. The flow was measured by the Venturi to an accuracy of better than ± 2% and the flow as calculated from the saddle coil measurements was within the experimental error of the Venturi over the complete flow range. Mechanical performance tests on a prototype control rod have commenced. A complete control rod and actuating mechanism have been fitted into a 30 ft. high x 30 in. dia. vessel filled with sodium. The test programme includes a comprehensive series of mechanical tests at sodium temperatures up to 600°C.

The core support grid is on final machining and will soon be set up at the manufacturers works for tests with a complete dummy core. Satisfactory progress is being maintained on the manufacture of the rotating shield for the reactor and manufacture of the sodium pumps is in progress.
On the steam generation plant considerable development work has been carried out on tube to tube plate welding for both intermediate heat exchangers and boiler units to meet the very high standard of integrity which has been set for these welds. All the main components have now been manufactured for the heat exchangers and by using the tube to tube plate welding techniques developed, final assembly should proceed satisfactorily. The manufacture of the boiler circulating pumps, which are among the largest glandless pumps manufactured to date in the U.K., is on schedule, the first pump being under test at present.

The problems encountered in the manufacture of the roof have resulted in the completion of construction data for the reactor being delayed. The construction programme is at present being reassessed and a new completion date for the reactor will be announced shortly.

COMMERCIAL FAST REACTORS

Work has continued on the designs for future civil fast reactor power stations with reactors of about 625 MW(E) as well as the 1250 MW(E) reactors previously studied. Particular attention has been paid to methods of handling short cooled fuel and to safety problems especially containment systems.

Economic studies of the development of the U.K. generating system have confirmed the advantages of introducing fast reactors in the U.K. at the earliest possible date, even though fuel costs for the early fast reactors will be higher than the later stations due to the small throughput of the fuel fabrication and reprocessing plants. In consequence the choice of a date for the building of the first commercial station depends on a complex interaction of technical and economic considerations. As noted last year it would be possible for a large commercial fast reactor to be on power in 1976.

R.D.S.
March 1969.
Discussion of the Presentation by Dr. Smith

Engelmann: Is there any large report on the FFR mock-up experiments available yet?

Smith: The answer is no. These results are in several small reports. Some of the results will be reported at the London conference.

Engelmann: You mentioned you have done spectrum measurements. I wonder whether there are new results at Zebra with a good agreement of measurements made by different techniques.

Smith: I really try to avoid answering this question because we have not finished sorting this out yet. There are discrepancies between the techniques and between the techniques and calculation. Certainly some results would appear at the London conference. It seems that the time of flight results in particular forecast a larger number of low energy neutrons than we had suspected. I would rather not say it as a definite conclusion, because this depends on an energy calibration of the detectors and this is a rather difficult thing to do accurately.

Wensch: I wonder whether components for PFR will be fabricated and tested to meet the codes.

Smith: Certainly such components as piping and vessels will meet the normal codes. I do not know whether the codes embrace every aspect, for example steam generators, but I believe the majority meet standard codes.

Schuster: Is there any work under way on carbide fuels?

Smith: We are irradiating carbide fuels in the DFR. This fuel will not be used for the first charge of the PFR. We regard this as an advanced fuel. Early tests tended to show that it will not readily go to higher burn ups as an oxide fuel.

Schuster: You said nothing about materials behaviour in sodium about the displacement, the distortion and some other events in the high fast neutron flux. Are there any experiments in this respect?
Smith: During the last year most of our attention was to these swelling phenomena. We consider this effect as probably the most important.

Schuster: Can you give us some estimate about the delay in construction of PFR?

Smith: Because of troubles with the roof, the reactors cannot now be completed before the end of 1971. If we can manage to keep to this date, it would mean that the reactor is on power by the end of 1972, which means of delay of twelve months.

Schuster: You said that a possible date for a commercial fast reactor to be in operation in your country was 1976. You cannot start the construction of such a reactor without first having some experience with the PFR.

Smith: If we have to wait for experience of the PFR, it turns out that CFR should be postponed to about 1980. The 1976 date is based on the assumption that we would be prepared to start construction of a CFR before we had experience with the PFR. But this depends on many decisions - Government decisions, the view of C.E.G.B., and so on.

Wensch: Do you have some cost estimates for the construction of the PFR?

Smith: I don't have such information with me. There are cost estimates in existence, but I do not know whether these figures were released.

Boxer: You mentioned that design studies for subsequent fast reactors have been extended to include 600 and 1300 MW units. I wonder if you are able to comment upon the impact and of re-optimisation on these designs when the new plutonium alpha date are introduced.

Smith: We said at the Karlsruhe conference that the effect of new data compared with the old is to reduce the breeding gain by about 0.1. The effect on an optimisation of a reactor design is not perhaps quite as great as one would have thought, since in any event one tended to design a reactor so that it had the
highest breeding gain that one could attain. The highest breeding gain you can get is a little lower than you would have thought. The things you do in order to get the higher breeding gain — high fuel fraction, good design, low parasitic capture — are the same with new data as they are with old. The only thing that this may do in your analysis is to shift your optimum slightly towards things which give you high breeding gain.

Engelmann: We have calculated the loss in breeding gain due to the higher plutonium-239 alpha values. The higher Pu alpha values seemed in the beginning to reduce the breeding ratio by 0.06 to 0.1 for soft spectrum fast reactors and in the order of 0.06 to 0.08 for harder sodium-cooled reactors; the recalculation of the new infinite plutonium composition in fact reactors changes this reduction and it tends to reduce the loss in breeding gain. So our calculations show that taking into account only the plutonium alpha change, the reduction in breeding gain seems to be for sodium-cooled fast reactors in the order of 0.04 rather than in the order of 0.1.

Spinrad: We have done fairly popular exercises at the Agency in trying to guess what the price and value of plutonium might be in future decades. The estimate at the moment shows a rather steadily and constantly rising cost of plutonium alpha value. I think that this feature will do a lot to negate the plutonium alpha business because the higher value of plutonium inevitably forces you to squeeze a little more out of breeding gain and to go to higher power densities in order to have a relatively inexpensive fuel cycle. The net result I suspect is going to be a spectrohardening of most fast reactor designs to where rather lower energy troubles with plutonium alpha are no longer so important. We are trying to predict things that are likely to be in the 1980's.

Smith: I do not think we are in great disagreement with Dr. Engelmann. I did say the total 0.1 took into account the other cross section changes which have tended to occur over the last year, and in particular the 238 capture cross sections come down and the steel cross sections seem to show signs of going
up. These three together give the result which obviously gives us a hardness of the spectrum, but it is of the order of 0.08 to 0.1. You get some of this back again if you perhaps have a plutonium of degraded composition with a lot of plutonium-240. By and large I think we still find that the breeding gains are about 0.1 lower than they were 1.5 years ago.

Simmons: A question concerning a break in the pipe at Dounreay. Did you notice any effects of surface cracking or other types of indication of thermal stresses?

Smith: No, the whole of the inside pipe-work was almost like new. There was very little evidence of a surface attack of any sort.

Simmons: If I remember correctly the design of your pool system, you have an open duct from the reactor proper to the heat exchanger. This permits the underside of the roof of the reactor vessel to see the high temperature of the sodium. Having now done a design of that system and going to another system, would you reconsider that design arrangement?

Smith: Yes, we are looking at arrangements which do keep the surface of the pool at a more cool and more constant temperature. I do not think we feel that this is a problem which would cause us too much worry. In the CFR studies some of the designs do not have these hot surfaces.
I. INTRODUCTION

The use of nuclear energy for the generation of electric power continues to attract world-wide attention. The locations and capacities of nuclear power plants in the United States are shown in Figures 1 and 2. Utility orders in 1968 for nuclear steam-supply systems continued at a high level but less than the rate in 1967 (Figure 3). A comparison of annual additions to non-nuclear and nuclear utility generating capacities in the U. S. is shown in Figure 4.

The continuing high commitment of the utility industry to nuclear energy places increasing pressure on the need for a continuing and increasing supply of reasonably priced fuel. In response to these pressures, industry has increased its exploratory efforts for uranium, and exploratory drillings in 1968 were about twice that in 1967. The prospect of availability of increasing supplies of plutonium recovered from light water reactors emphasizes the need for its economical use, including accelerated work on plutonium recycle and the development of the breeder reactor.

During the past year the AEC's breeder development work was directed chiefly to the liquid-metal fast breeder reactor (LMFBR). The LMFBR Program has as a goal the development of technology that will, in cooperation with the nuclear industry and the utilities, result in a commercially competitive, central-station power plant. An LMFBR Program Plan has been developed which outlines the requirements and courses of action to accomplish the objective. This plan includes the construction of a fast flux test facility and at least three demonstration plants. These demonstration
FIGURE 1

NUCLEAR POWER PLANTS IN THE UNITED STATES

The nuclear power plants included in this map are ones whose power is being transmitted or is scheduled to be transmitted over utility electric power grids and for which reactor suppliers have been selected.

NUCLEAR PLANT CAPACITY
(KILOWATTS)

| OPERABLE | 2,782,200 |
| BEING BUILT | 28,387,100 |
| PLANNED REACTORS ORDERED | 31,284,900 |
| REACTORS NOT ORDERED | 9,950,000 |
| TOTAL | 72,404,200 |

LEGEND

| OPERABLE | □ (14) |
| BEING BUILT | ▲ (39) |
| PLANNED (Reactors Ordered) | ◇ (36) |

*11 more plants have been announced for which reactors have not yet been ordered.

U.S. Atomic Energy Commission
September 20, 1960
<table>
<thead>
<tr>
<th>SITE</th>
<th>CAPACITY (Kilowatts)</th>
<th>UTILITY</th>
<th>STARTUP</th>
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</thead>
<tbody>
<tr>
<td>ALABAMA</td>
<td>1,054,500</td>
<td>Tennessee Valley Authority</td>
<td>1970</td>
</tr>
<tr>
<td>Atmosphere</td>
<td>1,054,500</td>
<td>Tennessee Valley Authority</td>
<td>1971</td>
</tr>
<tr>
<td>Arkansas</td>
<td>1,054,500</td>
<td>Tennessee Valley Authority</td>
<td>1972</td>
</tr>
<tr>
<td>ARKANSAS</td>
<td>820,000</td>
<td>Arkansas Power &amp; Light Co.</td>
<td>1972</td>
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<tr>
<td>CALIFORNIA</td>
<td>68,500</td>
<td>Pacific Gas &amp; Electric Co.</td>
<td>1963</td>
</tr>
<tr>
<td></td>
<td>430,000</td>
<td>Southern Calif. Edison and San Diego Gas &amp; Electric Co.</td>
<td>1967</td>
</tr>
<tr>
<td></td>
<td>462,000</td>
<td>L.A. Dept. of Water &amp; Power</td>
<td>1973</td>
</tr>
<tr>
<td></td>
<td>1,060,000</td>
<td>Pacific Gas &amp; Electric Co.</td>
<td>1974</td>
</tr>
<tr>
<td></td>
<td>800,000</td>
<td>Sacramento Municipal District</td>
<td>1976</td>
</tr>
<tr>
<td></td>
<td>330,000</td>
<td>Public Service Co. of California</td>
<td>1976</td>
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<tr>
<td></td>
<td>462,000</td>
<td>Conn. Yankee Atomic Power Co.</td>
<td>1967</td>
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<tr>
<td></td>
<td>653,000</td>
<td>Northeast Utilities</td>
<td>1969</td>
</tr>
<tr>
<td></td>
<td>828,000</td>
<td>Northeast Utilities</td>
<td>1974</td>
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<tr>
<td></td>
<td>661,500</td>
<td>Florida Power &amp; Light Co.</td>
<td>1970</td>
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<tr>
<td></td>
<td>825,000</td>
<td>Florida Power Corp.</td>
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<td></td>
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<td>Florida Power and Light Co.</td>
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<td></td>
<td>786,000</td>
<td>Georgia Power Co.</td>
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<td></td>
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<td>Commonwealth Edison Co.</td>
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<tr>
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<td>715,000</td>
<td>Commonwealth Edison Co.</td>
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<tr>
<td></td>
<td>715,000</td>
<td>Commonwealth Edison Co.</td>
<td>1969</td>
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<tr>
<td></td>
<td>1,050,000</td>
<td>Commonwealth Edison Co.</td>
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</tr>
<tr>
<td></td>
<td>1,050,000</td>
<td>Commonwealth Edison Co.</td>
<td>1973</td>
</tr>
<tr>
<td></td>
<td>515,000</td>
<td>Northern Indiana Public Service Co.</td>
<td>1970</td>
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<tr>
<td></td>
<td>537,600</td>
<td>Iowa Electric Light and Power Co.</td>
<td>1973</td>
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<tr>
<td></td>
<td>790,000</td>
<td>Maine Yankee Atomic Power Co.</td>
<td>1972</td>
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<tr>
<td></td>
<td>800,000</td>
<td>Baltimore Gas and Electric Co.</td>
<td>1973</td>
</tr>
<tr>
<td></td>
<td>800,000</td>
<td>Baltimore Gas and Electric Co.</td>
<td>1974</td>
</tr>
<tr>
<td></td>
<td>175,000</td>
<td>Yankee Atomic Power Co.</td>
<td>1970</td>
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<tr>
<td></td>
<td>635,000</td>
<td>Boston Edison Co.</td>
<td>1971</td>
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<tr>
<td></td>
<td>70,300</td>
<td>Consumers Power Co.</td>
<td>1962</td>
</tr>
<tr>
<td></td>
<td>700,000</td>
<td>Consumers Power Co.</td>
<td>1969</td>
</tr>
<tr>
<td></td>
<td>65,900</td>
<td>Detroit Edison Co.</td>
<td>1963</td>
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<tr>
<td></td>
<td>1,100,000</td>
<td>Detroit Edison Co.</td>
<td>1974</td>
</tr>
<tr>
<td></td>
<td>1,060,000</td>
<td>Indiana &amp; Michigan Electric Co.</td>
<td>1972</td>
</tr>
<tr>
<td></td>
<td>1,054,000</td>
<td>Indiana &amp; Michigan Electric Co.</td>
<td>1973</td>
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<tr>
<td></td>
<td>685,000</td>
<td>Consumers Power Co.</td>
<td>1974</td>
</tr>
<tr>
<td></td>
<td>685,000</td>
<td>Consumers Power Co.</td>
<td>1975</td>
</tr>
<tr>
<td></td>
<td>22,000</td>
<td>Rural Cooperative Power Assoc.</td>
<td>1962</td>
</tr>
<tr>
<td></td>
<td>471,700</td>
<td>Northern States Power Co.</td>
<td>1970</td>
</tr>
<tr>
<td></td>
<td>530,000</td>
<td>Northern States Power Co.</td>
<td>1972</td>
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<tr>
<td></td>
<td>570,000</td>
<td>Northern States Power Co.</td>
<td>1974</td>
</tr>
<tr>
<td></td>
<td>457,000</td>
<td>Omaha Public Power District</td>
<td>1971</td>
</tr>
<tr>
<td></td>
<td>778,000</td>
<td>Consumers Public Power District and Iowa Power and Light Co.</td>
<td>1972</td>
</tr>
</tbody>
</table>

**FIGURE 2**

Names and Capacities of Plants

<table>
<thead>
<tr>
<th>SITE</th>
<th>CAPACITY (Kilowatts)</th>
<th>UTILITY</th>
<th>STARTUP</th>
</tr>
</thead>
<tbody>
<tr>
<td>NEW HAMPSHIRE</td>
<td>800,000</td>
<td>Public Service Co. of N.H.</td>
<td>1974</td>
</tr>
<tr>
<td>NEW JERSEY</td>
<td>515,000</td>
<td>Jersey Central Power &amp; Light Co.</td>
<td>1969</td>
</tr>
<tr>
<td></td>
<td>910,000</td>
<td>Jersey Central Power &amp; Light Co.</td>
<td>1973</td>
</tr>
<tr>
<td></td>
<td>1,050,000</td>
<td>Public Service Gas and Electric Co. of New Jersey</td>
<td>1971</td>
</tr>
<tr>
<td>NEW YORK</td>
<td>265,000</td>
<td>Consolidated Edison Co.</td>
<td>1962</td>
</tr>
<tr>
<td></td>
<td>820,000</td>
<td>Consolidated Edison Co.</td>
<td>1969</td>
</tr>
<tr>
<td></td>
<td>965,300</td>
<td>Consolidated Edison Co.</td>
<td>1971</td>
</tr>
<tr>
<td></td>
<td>500,000</td>
<td>Niagara Mohawk Power Co.</td>
<td>1968</td>
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<tr>
<td></td>
<td>420,000</td>
<td>Rochester Gas &amp; Electric Co.</td>
<td>1969</td>
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<tr>
<td></td>
<td>523,000</td>
<td>Long Island Lighting Co.</td>
<td>1973</td>
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<tr>
<td></td>
<td>829,000</td>
<td>New York State Electric &amp; Gas Co.</td>
<td>1973</td>
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<td></td>
<td>1,115,000</td>
<td>Consolidated Edison Co.—Orange and Rockland Utilities, Inc.</td>
<td>1973</td>
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<td>NORTH CAROLINA</td>
<td>821,000</td>
<td>Carolina Power and Light Co.</td>
<td>1973</td>
</tr>
<tr>
<td></td>
<td>821,000</td>
<td>Carolina Power and Light Co.</td>
<td>1974</td>
</tr>
<tr>
<td></td>
<td>821,000</td>
<td>Carolina Power and Light Co.</td>
<td>1975</td>
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<tr>
<td>PENNSYLVANIA</td>
<td>40,000</td>
<td>Philadelphia Electric Co.</td>
<td>1966</td>
</tr>
<tr>
<td></td>
<td>1,065,000</td>
<td>Philadelphia Electric Co.</td>
<td>1971</td>
</tr>
<tr>
<td></td>
<td>1,065,000</td>
<td>Philadelphia Electric Co.</td>
<td>1975</td>
</tr>
<tr>
<td></td>
<td>1,065,000</td>
<td>Philadelphia Electric Co.</td>
<td>1977</td>
</tr>
<tr>
<td></td>
<td>90,000</td>
<td>Duquesne Light Co.</td>
<td>1967</td>
</tr>
<tr>
<td></td>
<td>783,000</td>
<td>Duquesne Light Co.—Ohio Edison Co.</td>
<td>1973</td>
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<tr>
<td></td>
<td>381,000</td>
<td>Metropolitan Edison Co.</td>
<td>1971</td>
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<td>1,052,000</td>
<td>Pennsylvania Power and Light Co.</td>
<td>1975</td>
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<tr>
<td></td>
<td>1,052,000</td>
<td>Pennsylvania Power and Light Co.</td>
<td>1977</td>
</tr>
<tr>
<td>SOUTH CAROLINA</td>
<td>663,000</td>
<td>Carolina Power &amp; Light Co.</td>
<td>1970</td>
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<td></td>
<td>841,100</td>
<td>Duke Power Co.</td>
<td>1971</td>
</tr>
<tr>
<td></td>
<td>841,100</td>
<td>Duke Power Co.</td>
<td>1972</td>
</tr>
<tr>
<td></td>
<td>841,100</td>
<td>Duke Power Co.</td>
<td>1973</td>
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<td></td>
<td>58,500</td>
<td>Northern States Power Co.</td>
<td>1964</td>
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<td>TENNESSEE</td>
<td>1,125,000</td>
<td>Tennessee Valley Authority</td>
<td>1973</td>
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<tr>
<td></td>
<td>1,125,000</td>
<td>Tennessee Valley Authority</td>
<td>1974</td>
</tr>
<tr>
<td>VERMONT</td>
<td>513,900</td>
<td>Vermont Yankee Nuclear Power Corp.—Green Mt. Power Corp.</td>
<td>1970</td>
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<tr>
<td>VIRGINIA</td>
<td>783,000</td>
<td>Virginia Electric &amp; Power Co.</td>
<td>1971</td>
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<td>783,000</td>
<td>Virginia Electric &amp; Power Co.</td>
<td>1972</td>
</tr>
<tr>
<td></td>
<td>800,000</td>
<td>Virginia Electric &amp; Power Co.</td>
<td>1974</td>
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<tr>
<td>WASHINGTON</td>
<td>790,000</td>
<td>Washington Public Power Supply System</td>
<td>1966</td>
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<td>WISCONSIN</td>
<td>50,000</td>
<td>Dairyland Power Cooperative</td>
<td>1967</td>
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<td>454,600</td>
<td>Wisconsin Michigan Power Co.</td>
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<td></td>
<td>454,600</td>
<td>Wisconsin Michigan Power Co.</td>
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<tr>
<td></td>
<td>507,000</td>
<td>Wisconsin Public Service Co.</td>
<td>1972</td>
</tr>
</tbody>
</table>

*Site not selected.*
plants will be spaced at approximately two-year intervals, with commitment of the first as early as 1970. Each of these plants is expected to have a capacity of 300 - 500 Mwe. The AEC is presently seeking Congressional authorization to initiate this year an effort leading to the first demonstration plant.
II. PLANNING

The year 1968 produced significant accomplishments toward the realization of the LMFBR as a commercial power source in the U.S. Since our last meeting a year ago, the LMFBR Program Plans have been completed and implementation of the Plans has begun.

The LMFBR Program Plans are national in scope. The plans address the key problem areas and provide courses of actions and possible alternatives for their resolution. The plans are based on the belief that a systematic, engineering approach will be needed to achieve the successful introduction of the liquid-metal fast breeder reactor into the utility environment in the U.S. It is recognized that the first edition of the plans is only one of a series of steps that will be needed.

The plans will be aggressively prosecuted in a disciplined manner. Emphasis will be placed upon the development and application of adequate quality-assurance procedures throughout all phases of the program and insistence upon adherence to the highest possible standards of engineering and workmanship.

The LMFBR Program Plans consist of an overall plan and nine other technical elements as shown below:

1. WASH-1101, Overall Plan
2. WASH-1102, Plant Design
3. WASH-1103, Components
4. WASH-1104, Instrumentation & Control
5. WASH-1105, Sodium Technology
6. WASH-1106, Core Design
7. WASH-1107, Fuels & Materials
8. WASH-1108, Fuel Recycle
9. WASH-1109, Physics
10. WASH-1110, Safety

The overall plan summarizes the other nine elements, treats some problems common to all elements, and discusses overall program
goals* and the approach to achieving a competitive and self-sustaining industry. The other elements describe, for a specific area of technology, the technical objectives, scope, and course of action for the research and development programs that are needed to develop the LMFBR.

The plans were developed by the LMFBR Program Office at the Argonne National Laboratory with the assistance and advice of industry, national laboratories, and other specialized groups. The Program Office is staffed by a group of about 45 senior personnel who have had extensive experience in the various disciplines in the LMFBR field.

The plans were completed in August 1968 and widely disseminated to all likely participants in the program. In addition to the AEC, related government agencies and the national laboratories, the plans were distributed to numerous industrial and research organizations, universities, and electric utilities. Many sets were made available through U.S. representatives to interested organizations overseas. To encourage industrial involvement in the program, a presentation seminar was sponsored by the Program Office at Argonne National Laboratory on October 23-24, 1968.

* The Liquid Metal Fast Breeder Reactor (LMFBR) Program has been assigned the highest priority in the Atomic Energy Commission's broader program for the development of civilian nuclear power. The primary objective of the civilian power reactor development program in the United States is widespread use of nuclear energy for the production of heat and electricity with full exploitation of the energy available in our resources of uranium and thorium. The AEC's objective also includes fostering the development of a self-sufficient and competitive nuclear industry. The need for a power reactor that can fully and economically exploit the energy reserves contained in uranium and thorium was recognized in the "Civilian Nuclear Power--A Report to the President--1962" which stated:

"The overall objective of the Commission's nuclear power program should be to foster and support the growing use of nuclear energy, and importantly to guide the program in such directions as to make possible the exploitation of the vast energy resources latent in the fertile materials uranium-238 and thorium."
The seminar attracted more than 260 representatives from about 75 organizations. Each element of the plan was described, key problem areas were discussed, and courses of action to resolve these difficulties were presented.

The Program Office will continue to improve and periodically update the plan and to assist the AEC in the broad area of program planning.
The LMFBR Program is currently focused upon the development of satisfactory fuels and materials, the construction of a fuels and materials irradiation facility (FFTF), the development of components including adequate test facilities, and the preparation for near-term demonstration plants. The specific accomplishments and events in the R&D program during the past year are described in the following paragraphs using the format of the Program Plan.

A. Plant Design

Since our last meeting our plan for plant development has been further developed. The 1000-Mwe LMFBR Follow-On Study program essentially has been completed and the highlights of the work released. Activities such as this will provide much of the understanding of the needs of the LMFBR. The industry-financed program to study demonstration plants has quickened, particularly in light of the AEC's issuance of the plans and their recent decision to implement in FY-70 the first phase of the LMFBR demonstration plant program. The conceptual design of the Fast Flux Test Facility (FFTF) was selected and the preliminary design has been initiated. The EBR-II operated as a fuels test facility with an improved plant factor. The repairs to the Enrico Fermi plant have proceeded well and its return to operational status is expected in 1969. Four major meetings devoted to or featuring LMFBRs were held.

The plan for the development of plant-design technology and the construction of demonstration plants is described in WASH-1102, LMFBR Program Plan -- Plant Design. The approach to be used consists of:
(1) Development of a thorough understanding of LMFBR target plant* design technology, engineering considerations, and related R&D requirements — through AEC-funded studies

(2) Investigation of demonstration plant** concepts and identification of related R&D requirements — through industry-funded activities

(3) Identification of other R&D needed to support the Program — through industry- or AEC-funded activities, as appropriate

(4) Construction of demonstration plants — as jointly funded ventures by AEC and industry

Figure 5 illustrates the approach summarized above.

The LMFBR Program will be guided by design studies of target plants with capacities of 1000-Mwe or greater. These studies will identify features, problems, and design alternatives of 1000-Mwe units that appear to be potentially attractive for further development. The studies are financed by the AEC and performed by five major reactor manufacturers: Atomics International, Babcock & Wilcox, Combustion Engineering, General Electric, and Westinghouse Electric. Each has developed a 1000-Mwe reference plant concept and performed extensive parametric studies and trade-off evaluations. The reference plant concepts derived from this program are considered the target plants toward which our current program is directed.

* A target plant is a commercially competitive LMFBR central-station power plant for use in the utility environment. A 1000-Mwe capacity was established as a ground rule for the 1962-1964 AEC-sponsored Large Fast Reactor Design Study. At the time, the LWRs being committed were chiefly about 500 Mwe, and plants of 700 to 800 Mwe were being considered seriously.

** An LMFBR demonstration plant is a sodium-cooled, fast breeder reactor facility, with an electric capacity of 300 - 500 Mwe, and designed, constructed, and operated as a central-station plant in a utility system to demonstrate technical and economic aspects of a future nuclear facility of the same general type.
BASIC APPROACH OF THE PLANT DESIGN PROGRAM

AEC-Funded

Target Plant Studies → R&D → Evaluation → Demonstration Plants Construction and R&D → Target Plant

Industry-Funded

Demonstration Plant Studies → Proposals → R&D
Three of the five reactor manufacturers, in conjunction with utility
groups, are currently funding the study of conceptual designs of demonstration
plants that are intended for early commitment. The makeup of these demonstra-
tion plant studies is shown in Table 1. Only fragmentary details of these plants
have been published. However, because this work is being done simultaneously
and in parallel with the AEC's 1000-Mwe studies, the designs of the demonstra-
tion plants should bear a close resemblance in many respects to the reference
designs of the large, 1000-Mwe target plants. This is borne out by some of the
early results that were presented at the National Topical Meeting on Fast
Reactors in San Francisco in April 1967. Some pertinent design parameters
taken from these papers are shown in Table 2.

The LMFBR Program encourages the cross-fertilization of technology
between groups developing basic technology with AEC funds and those develop-
ing applied technology with industrial funds.

The major results of the 1000-Mwe Follow-On program were des-
cribed at the International Conference on Sodium Technology and Large Fast
Reactor Design at Argonne National Laboratory on November 7 - 9, 1968.*
Table 3 summarizes many of the significant plant parameters and features.
Two manufacturers selected the pool-type primary system; one using a hot-cell
with refueling through an open reactor vessel, the other an under-the-plug
refueling system similar to the EBR-II. Three manufacturers selected the loop-
type primary system; one using a hot-cell for refueling, the other two using
the through-the-plug technique. Steam conditions are 2400 psig and 900°F or
950°F, usually with reheat. Net plant efficiency varied only slightly from

* The proceedings of this conference are to be issued as report ANL-7520, with
publication scheduled for about June 1969.
Table 1
Current Industrial Programs for Demonstration Plants

<table>
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<tr>
<th>Reactor Designer</th>
<th>General Electric</th>
<th>General Electric</th>
<th>Westinghouse</th>
<th>Atomics International</th>
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<tr>
<td>Program Scope</td>
<td>Prepare a preliminary design of a 500-Mw prototype reactor system including: a. System design b. Equipment specifications c. Safety criteria d. Capital and fuel-cycle costs. (This effort complements the GE-ESADA effort)</td>
<td>Prepare and perform an R&amp;D program in the areas of: a. Mechanical design of fuel assemblies b. Fuel-loading and -unloading systems c. Instrumentation to monitor reactor cooling conditions d. Standby control systems e. Steam generator (This effort complements the GE-utility design study)</td>
<td>Development of a demonstration plant concept and the preparation of a proposal for consideration by the participants. Performance of a test program for a demonstration plant and preparation of a concept proposal.</td>
<td></td>
</tr>
<tr>
<td>Program Goal</td>
<td>Development of a thorough understanding of the demonstration plant and its many facets. A proposal is not contractually involved. Resolution of key problem areas is anticipated. The participant has the right of first refusal on construction of a demonstration plant. Development of a thorough understanding by each participant of the demonstration plant and its problems. A definitive plant concept will be developed. Each of the participants has the option to accept or reject for construction. Development and performance of a test program to resolve outstanding questions concerning a breeder system. Plant designed in study will be for construction by participants if it is satisfactory and will fit into AEC program.</td>
<td></td>
<td></td>
<td></td>
</tr>
<tr>
<td>Schedule</td>
<td>Program began in March 1967 and is scheduled to be completed by about March 1969.</td>
<td>Program began in August 1967 and is scheduled to be completed by August 1970. In 1970 participant will have the option to construct the breeder plant prepared during the study.</td>
<td>Program began in mid-1967 and is scheduled to be completed by July 1970. At that time, the participants are expected to exercise the right of acceptance or rejection of the plant.</td>
<td>Program began in October 1967. Program is scheduled to provide sufficient information to permit a design in 1970 on construction of a 350- to 500-Mw breeder system.</td>
</tr>
</tbody>
</table>
### Table 2

Selected Parameters of Suggested U.S. Demonstration Field Concepts

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Alhambra International</th>
<th>General Electric</th>
<th>Westinghouse</th>
<th>Pacific &amp; Others</th>
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*Where two values are shown, the first represents the initial core operation.
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<th>Babcock &amp; Wilcox</th>
<th>Atomics International</th>
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Table 3 (Cont)

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**Steam Pressure (psia)**

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**Core Factors**

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<td>Specific power, fissile, avg (Kw/Kg)</td>
<td>1304</td>
<td>--</td>
<td>880</td>
<td>--</td>
<td>--</td>
</tr>
<tr>
<td>Doppler coefficient ((T \frac{\Delta k}{\Delta t}))</td>
<td>-0.0037</td>
<td>-0.00243</td>
<td>-0.005</td>
<td>--</td>
<td>-0.0042</td>
</tr>
<tr>
<td>Na Void, core plus axial blanket ($)</td>
<td>2.24</td>
<td>0.51</td>
<td>7.5 core only</td>
<td>--</td>
<td>3.90</td>
</tr>
</tbody>
</table>

**Core Parameters**

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Diameter (in.)</td>
<td>97.4</td>
</tr>
<tr>
<td>Height (in.)</td>
<td>30</td>
</tr>
</tbody>
</table>

* Total in-pile and out-of-pile.
<table>
<thead>
<tr>
<th>Parameter or Feature</th>
<th>General Electric</th>
<th>Westinghouse Electric</th>
<th>Babcock &amp; Wilcox</th>
<th>Atomics International</th>
<th>Combustion Engineering</th>
</tr>
</thead>
<tbody>
<tr>
<td>L/D ratio</td>
<td>0.308</td>
<td>0.84</td>
<td>0.29</td>
<td>--</td>
<td>0.226</td>
</tr>
<tr>
<td>Volume (liters)</td>
<td>3660</td>
<td>4415</td>
<td>6516</td>
<td>--</td>
<td>3350</td>
</tr>
<tr>
<td>Fuel type</td>
<td>(U, Pu)O$_2$</td>
<td>Cr$_{23}$C$_6$ mod (U, Pu)C</td>
<td>UO$_2$-PuO$_2$</td>
<td>UO$_2$-PuO$_2$</td>
<td>PuC – UC</td>
</tr>
<tr>
<td>No. of fuel assemblies</td>
<td>--</td>
<td>244</td>
<td>288</td>
<td>274</td>
<td>219</td>
</tr>
<tr>
<td>No. of elements/assembly</td>
<td>--</td>
<td>168</td>
<td>331</td>
<td>217</td>
<td>163</td>
</tr>
<tr>
<td>Total no. of elements</td>
<td>--</td>
<td>40,992</td>
<td>95,328</td>
<td>59,458</td>
<td>35,697</td>
</tr>
<tr>
<td>Fuel pellet OD (in.)</td>
<td>--</td>
<td>0.246</td>
<td>--</td>
<td>--</td>
<td>0.350</td>
</tr>
<tr>
<td>Fuel element OD (in.)</td>
<td>0.25</td>
<td>0.302</td>
<td>0.280</td>
<td>0.30</td>
<td>0.40</td>
</tr>
<tr>
<td>Pitch/diameter ratio</td>
<td>1.20</td>
<td>1.24</td>
<td>1.20</td>
<td>--</td>
<td>1.128</td>
</tr>
<tr>
<td>Cladding thickness (in.)</td>
<td>0.010</td>
<td>0.012</td>
<td>0.010</td>
<td>--</td>
<td>0.011</td>
</tr>
<tr>
<td>Cladding material</td>
<td>--</td>
<td>316 SS</td>
<td>304 SS</td>
<td>304 or 316</td>
<td>316 SS</td>
</tr>
<tr>
<td>Fuel channel across flats (in.)</td>
<td>5.67 HEX</td>
<td>5.29 SQ</td>
<td>6.45 HEX</td>
<td>5.323 HEX</td>
<td>6.527 Dia + Flutes</td>
</tr>
<tr>
<td>Fuel element type</td>
<td>Gas bonded, vented</td>
<td>Na bonded, vented</td>
<td>Gas bonded, vented</td>
<td>Gas bonded, non-vented</td>
<td>Sodium bonded, vented</td>
</tr>
</tbody>
</table>
41 percent. Core designs varied considerably. Two designs employed spoiled geometries for safety reasons — one consisted of 4 modules with an L/D of 0.84, the other used a high-leakage, pancake geometry with an L/D of 0.23. The remaining designs had L/Ds ranging between 0.29 to 0.405. In general, 1300°F was considered as an upper limit for the peak clad temperature with mixed-mean coolant temperatures from the reactor ranging from 1000 to 1150°F. The fuel is either mixed oxides or mixed carbides, with peak fuel ratings of about 16 Kw/ft for oxides and 40 Kw/ft for carbides. Burnup goals are, in most instances, 100,000 Mwd/T average. Cooling of the reactors is accomplished with multiple primary and secondary loops.

The Program Plan calls for these studies to be augmented with certain of the concepts to be investigated in greater depth. The new studies will concentrate on the reactor plant, placing special emphasis on the primary system, the refueling system, and transient performance and safety analyses. Some work has been authorized and contractor selection is now in progress.

Last April the Edison Electric Institute (EEI) published a Fast Breeder Reactor Report which gave the results of a comprehensive review and assessment of the status and development needs of fast reactors. The report found a quickening of the pace of fast breeder reactor development and a strong incentive for the electric utility industry to participate in this development. The report encourages the utility industry to support these findings. It recommends that at least two 300- to 500-Mwe LMFBR demonstration plants be constructed during the early 1970s, with firm commitment of at least one plant by about 1970.

The planned schedule for the commitment of demonstration plants is shown in Figure 6. As shown, the plan considers three demonstration plants for commitment at intervals starting as early as 1970.
FIGURE 6

PLANT DESIGN PROGRAM MAJOR MILESTONES

|------|------|------|------|------|------|------|------|------|------|------|------|------|------|------|------|

- 1000 MWe FOLLOW-ON PROGRAM
- FUEL HANDLING
- POOL AND LOOP PRIMARY SYSTEMS
- CONTAINMENT AND SHIELDING
- SYSTEM ANALYSIS
- PLANT SAFETY ANALYSIS

IN-DEPTH TARGET PLANT STUDIES

STARTUP FFTF

1st DEMONSTRATION PLANT

2nd DEMONSTRATION PLANT

3rd DEMONSTRATION PLANT

DEMONSTRATION PLANT STUDIES (INDUSTRY FINANCED)

GENERAL PLANT STUDIES OF LONGER RANGE INTEREST

ASSESSMENT OF PROGRAM

COMPLETE USER'S ASSESSMENT

COMPLETE PLANT ECONOMICS, SPECIFICATIONS AND STANDARDS
The AEC- and industry-financed programs are expected to intensify in light of the recent AEC decision to seek authorization to implement this year a $4 million first phase of the demonstration plant program.

Currently, the AEC's principal plant project is the Fast Flux Test Facility. The results from the FFTF project will contribute to all aspects of the LMFBR Program. Many FFTF requirements are as demanding as those for LMFBR central-station power plants. The project will help develop and consolidate technical knowledge and industrial capability. It will make substantial contributions to the development of LMFBR specifications, codes and standards, and quality-assurance procedures. Rigorous procedures have been instituted to assure that systematic and disciplined engineering practices are followed. High standards are being set for all phases of development, design, construction, inspection, test, and operation and the best technical competence and the strongest possible management have been sought.

Operating and construction experience with fast reactors has been acquired with EBR-II, Enrico Fermi, and SEFOR. EBR-II has operated satisfactorily at 50 Mw; up 5 Mw from its previous operating power level. The gross thermal power generated during 1968 was 171,899,000 Kwh. A plant-capacity factor of about 56% is reported for the last quarter. Some minor operating difficulties occurred and were successfully resolved. Some fission gas leakage from experimental and one driver fuel assemblies occurred with only minor inconvenience to reactor operation. The anomalous changes in the reactor-power coefficient of reactivity reported last year have been investigated and are believed due to the replacement of the original inner blanket assemblies with reflector assemblies made completely of stainless steel. Removal of the stainless steel units and return to the original core configuration
has reestablished approximately the original values and characteristics of the
power coefficient.

If all goes well, the Enrico Fermi plant should return to operational
status this year. The recovery operation has proceeded exceptionally well and
the objects causing flow blockage have been removed from the coolant inlet
plenum. All three steam generator units have been repaired and are considered
operational.

The construction of SEFGR is complete. The reactor is in a preoper-
ational checkout status. Permission to operate to a level of 1 Mwtt has been
announced.

The ZPPR is complete and should be operational at or about the time
of this meeting.

Four technical meetings, national or international in scope, were
held during the past year. The National Topical Meeting on Fast Reactor Sys-
tems, Materials and Components was held in Cincinnati, Ohio, April 2 - 4,
1968. This meeting and the International Conference on Sodium Technology
and Large Fast Reactor Design, held in November at Argonne National Laboratory,
were exclusively directed at the LMFBR. Fast reactor sessions were featured
in the Institute of Electronic and Electric Engineers meeting in Montreal, Canada
in June and the International Conference sponsored by the American Nuclear
Society and the Atomic Industrial Forum in Washington, D. C. in November,
1968.

B. Components

The development of components is receiving increasing emphasis.
During the past year, contracts have been negotiated for the development of
steam generators, intermediate heat exchangers, piping, and valves. Repairs
after a suitable period of time. Three identical in-vessel systems are used —
each servicing one-third of the core.

A new fuel-transport facility has been designed and fabricated to
replace the original cask car at Fermi. The new equipment separates into
modular subsystems the important functions of fuel transfer that interact for
the transfer of fuel between the Reactor Building and Fuel and Repair Building.
Acceptance testing of key equipment items has been completed at Fermi.

Completion of the SEFOR fuel-handling tests has established the
feasibility of the hot-cell approach to reactor refueling. The success of the
test program has attracted the interest of other plant designers in the possible
use of this approach to future LMFBR power plants.

3. Piping. A guide for the design of LMFBR piping has been started.
The guide will cover piping analysis, design, specifications, fabrication,
quality assurance, and inspection.

4. Pumps, Seals, and Bearings. About two years ago the AEC
contracted with Byron Jackson and Westinghouse Electric to develop large
sodium pumps for use in the LMFBR. The first phase comprised design analy-
ses, parametric studies, problem identification, the conceptual design of
60,000 gpm pumps for a 350 ft. head, and the definition of associated devel-
opment needs. This phase is complete. The second phase, which has been
initiated, consists of the definitive design and fabrication of pumps, leading
to their test in the Sodium Pump Test Facility (SPTF). Two pumps will be
included -- a low capacity, high head pump similar to those needed for the
FPTF, and a larger capacity pump of 60,000 gpm.

Special emphasis is being given to the development of shaft seals
and sodium bearings. The construction of a facility for testing shaft seals was
completed this past year. The facility consists of two rigs: one for testing oil-type seals, and the other for testing gas-type seals. Seals for the FFTF pumps will be tested first, followed by seals for 60,000 gpm pumps.

5. **Vessels.** The principal reactor vessel work underway is the design of the vessel for FFTF. A variety of vessel concepts has been studied and a firm concept is anticipated soon.

Considerable interest is developing in non-destructive techniques for the in-service inspection of vessels and other equipment. Acoustic-emission methods are being developed.

6. **Sodium-Water Reactions.** The investigation of a small leak of water into sodium is continuing. Based upon the metallurgical examination of 26 tubes of 2-1/4 Croloy, 8 tubes of Incoloy 800, and 4 tubes of stainless steel whose wastage patterns and rates were considered representative, it was concluded that: (1) metal removal was by abrasive, erosive attack; and (2) both Incoloy 800 and the stainless steels (Types 304 and 321) showed greater resistance to wastage than the Croloy. It is planned to conduct several tests this year using water-injection rates larger than 0.01 lb/sec, with some approaching 1 lb/sec. A literature survey and evaluation was made of small Na–H₂O reactions. The survey showed that no significant wastage has ever been reported for tests in the U.S. involving large leakage rates. In contrast, significant wastage was observed in the UKAEA's NOAH tests. The evaluation concluded that, in every U.S. case, water or steam injection continued after most of the sodium was flushed out, while, in the NOAH tests, water injection was terminated earlier. A topical report on the survey will be issued soon.
C. Instrumentation and Control

The LMFBR Program Plan has given greater visibility to the needs and programs for the development of instrumentation and control. Application to LMFBRs will encounter new or more difficult environmental and functional problems. For example, fast reactor flux ranges from startup to full power are greater than are required for LWRs; coolant temperatures are higher, ranging to 1200°F (1400°F for FFTF closed loops); and the coolant is non-transparent to the human eye. Each of these conditions imposes problems that require development work before the LMFBR can be commercialized. New techniques or devices may be required to resolve them.

1. Facilities. A number of facilities are needed for the development of instrumentation and controls. Facilities are being built at the LMEC for the test, calibration, and standardization of instruments. These facilities will accommodate instruments for measuring coolant conditions of temperature, flowrate, pressure, level, and purity, and for measuring the strain of structural materials -- all at conditions of interest to the LMFBR Program. Other facilities, particularly in the national laboratories, are being made available to develop instruments for measuring fuel temperature, fuel-element fission-gas pressure, fuel-assembly coolant conditions, and to develop in-reactor electrical connectors and neutron sensors and associated circuitry. An instrument thimble in the EBR-II primary tank will be converted for testing neutron sensors to 1200°F. The design is currently underway. A computer center will be established at PNL for work on control systems with special emphasis upon FFTF for the immediate future. These facilities will be available to all participants in the program and are complemented by facilities owned by industrial participants in this work.
2. **Sensors.** The development of sensors is receiving considerable emphasis. Fission counters for 1200°F service are needed. Fission counters rated at 1100°F from three manufacturers have been tested out-of-reactor to 1200°F with promising results. Commercial chambers have also been tested in an EBR-II thimble at ~120, 300, 500, and 700°F with the reactor shut down. Test results were satisfactory. After ~1 week operation at 700°F and during approach to full power (50 Mwt) following an unexpected shutdown, a malfunction occurred. The cause is being investigated. Fission counters and compensated ion chambers using boron coatings rated at 1200°F with a 1400°F survival requirement will be tested as soon as thimble modifications are complete.

Permanent-magnet and eddy-current flowmeters for measuring coolant flow through FFTF fuel assemblies are being developed. Permanent-magnet flowmeters have been developed for use in instrumented fuel assemblies in SEFOR and EBR-II.

LMFBRs will require the measurement of sodium flow rates in piping greatly exceeding 18-inches. No experience exists for sodium flowmeters in piping of these diameters. Accordingly, flow-measurement techniques which are not unduly affected by pipe size are being investigated. Proof-of-principle has been shown for ultrasonic techniques. One technique measures the time difference of an ultrasonic sound pulse traveling an equal distance upstream and downstream through the coolant. This time difference consists of flow measurement by analysis of the transit time of thermal eddies in the coolant. This method is based on the fact that the mean-coolant temperature from a heat exchanger varies with time in a random manner. A random but distinctive thermal pattern is created which persists for some distance down the pipe. Flow rate can be related to the time required for this thermal pattern to pass between two
points. This requires proper temperature-measuring devices at two locations in the pipe and suitable electronic circuitry. The circuitry must correlate the random temperature patterns in the coolant and measure the time difference for identical points in the pattern to pass between measurement stations. The method has been demonstrated in aqueous systems with accuracies of ± 1/2%.

3. Failed-Element Detection and Location. The development of satisfactory failed-element detection and location systems is receiving a high priority. The cover-gas monitor and the delayed-neutron monitor systems for detecting a failed-fuel event in the EBR-II have successfully operated. Xenon-tagging methods are being investigated to assist in the location of a failed element. This technique consists of adding carefully controlled mixtures of xenon isotopes to fuel elements during fabrication. A distinct ratio of isotopes is used in each assembly to be tagged. The feasibility of this method has been experimentally demonstrated and is now required for all unencapsulated fuel-irradiation experiments in EBR-II.

4. Miscellaneous. An ultrasonic technique for viewing under sodium is being developed. Proof-of-principle has been demonstrated. The specific application of this technique to fuel handling and reactor maintenance will be undertaken.

Work on signature analysis techniques is receiving increasing interest. This work is spurred by the need for better in-service inspection techniques and improved nondestructive testing methods.

Electronic circuitry for measuring neutron flux over a range of 10 decades is under development. A 10-decade neutron monitor using low-temperature fission-counters is being tested in the EBR-II. Initial results have been satisfactory. These activities will be continued and expanded to include use of the monitor with higher temperature fission-counters as they become available.
D. Sodium Technology

The extensive AEC program in sodium technology continued to produce information of direct application to the LMFBR Program during 1968. The results are described in numerous periodical and topical reports. In addition, two large meetings pertinent to this topic were held during this past year:

1. 1968 AEC Corrosion Symposium -- Battelle Memorial Institute, Columbus, Ohio, May 6 - 8

2. International Conference on Sodium Technology and Large Fast Reactor Design -- Argonne National Laboratory, Argonne, Illinois, November 7 - 9

1. Corrosion and Mass Transfer. A substantial part of the U.S. effort has been concerned with problems of the compatibility of materials in a high-temperature sodium environment. Work in this area on iron-base alloys, which has been carried out by General Electric for nearly 10 years, continues to yield useful data. A summary of the results obtained in the earlier work was given at the IAEA Symposium on Alkali Metal Coolants held in Vienna, November 28 - December 2, 1966.* In this program corrosion and mass transfer have been studied in a series of nine pumped loops made of Types 304 and 316 stainless steel, Croloy (2-1/4 Cr - 1 Mo) and a 5% Cr - 0.5% Mo - 0.5% Ti ferritic steel, alone and in combination, for periods as long as 30,000 hours. The results were subjected to detailed statistical analysis, and were expressed in a corrosion-rate equation which included the effects of temperature, oxygen content, flow velocity, geometric configuration, and other factors. The continuation of this work in 1968 has focused on a number of.

interesting phenomena. The earlier work on the effect of flow velocity indi-
ted a $v^{0.8}$ dependence. It has now been found that at higher flow rates, the
corrosion rate is independent of velocity, as was suggested by British work.
The effect of heat flux on mass transfer is being investigated; work to date
has revealed no systematic differences at heat transfer rates of up to
500,000 Btu/hr.-ft$^2$, and work is continuing with much higher heat fluxes,
up to $10^6$ Btu/hr.-ft$^2$. The program has been expanded to include some
nickel-base alloys (Incoloy 800, for example) with possible application in
secondary heat exchange systems, where resistance to corrosion by water
as well as sodium is essential. Early results indicate a significant increase
in corrosion rate with increasing nickel content.

2. Effects on Mechanical Properties. Extensive work on the
alteration of mechanical properties by sodium has been carried out by the
MSA Corporation. In these studies, on both austenitic and ferritic alloys,
the emphasis has been focused sharply on the effects of carbon transfer,
which seems to be the principal factor in the observed changes in proper-
ties. Initial test results were summarized in the 1967 IAEA Conference in
Vienna.* Detailed studies have also been made on carbon transfer in ferritic
alloys, and on the compositional and property changes accompanying this
process for conventional alloys, as well as some special materials suggested
for their resistance to decarburization.

The General Electric program has also included work (microprobe
studies) on changes in composition and mechanical properties of various types
of steel accompanying the process of corrosion by sodium.

Because of the possibility that the stainless steels presently favored for sodium service may be limited by temperature or radiation-damage considerations, the behavior of refractory metal alloys in sodium is being investigated: namely, tantalum, niobium, vanadium, and cobalt-base alloys in sodium-filled, stainless steel systems.

3. Impurity Monitors. The LMFBR Program has a strong commitment to develop continuous-reading, on-line, impurity-monitoring instrumentation for the control of contaminants in sodium systems. The solid-state electrolyte (Y$_2$O$_3$-doped ThO$_2$) meter, originally suggested in the U.K., has developed into a very useful instrument. Improvements to it are being sought. This instrument has also assisted in the study of the nature of oxygen-containing species in sodium.

An important accomplishment has been the development by UNC and application of a carbon meter, which depends on the permeation of carbon from sodium through a thin iron membrane. Similar work is in progress on the development of meters for hydrogen in sodium using the permeation principle, some in combination with electrochemical cell concepts.

Components for sodium-cooled plants have naturally drawn a great deal of attention. Concepts for improving plugging-meter and cold-trap designs have developed from observations on precipitation processes in sodium, made at Los Alamos, in both experimental and analytical studies.

4. Sodium Chemistry. With respect to more fundamental studies of sodium, some important discoveries should be mentioned. At GE, at least five different forms of carbon have been found to exist in sodium, explaining in part the difficulty in identifying the carbon-bearing species responsible for carbon transfer. Using isotopic tracer techniques, ANL has suggested that
sodium acetylide, $\text{Na}_2\text{C}_2$, is the most important carbon carrier. Argonne has also found that NaCN has a surprisingly high solubility and stability in high-temperature sodium. During 1968, measurements of the solubility of nitrogen gas in sodium were completed; nitrogen dissolves in the diatomic form, and is not dissociated. The use of alkaline earth metals as soluble getters for oxygen in sodium has been subjected to new analysis and evaluation. Serious doubts have been raised about the use of this technique, which will be resolved only by continuing work in this field.

E. Core Design

The AEC's LMFBR core design program for the past year has consisted of the conceptual design phase of the FFTF project and the 1000-Mwe Follow-On Studies. In addition, three reactor manufacturers have been preparing conceptual designs for demonstration plants in the 300 - 500 Mwe size range.

The core design effort on fast reactor concepts other than the LMFBR is small by comparison. All government- and privately-financed work on the steam-cooled fast reactor has been terminated.

Although none of the three major areas of LMFBR core design activity has reached a conclusive stage, certain design trends and problems are of interest.

1. **FFTF Core Design.** The principal core design parameters of the FFTF conceptual design are given in Table 4. Of the many design features not yet firmly chosen, the following three are particularly important to the overall LMFBR Program:

   (1) Whether axial-motion restrictors are required in the mixed-oxide fuel elements
### TABLE 4

SUMMARY OF PRELIMINARY FFTF CHARACTERISTICS AND DATA

**General Plant Data**

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Total power, initial (Mwt)</td>
<td>400</td>
</tr>
<tr>
<td>Peak flux (n/cm²/sec)</td>
<td>$7.3 \times 10^{15}$</td>
</tr>
<tr>
<td>Core volume (liters)</td>
<td>1025</td>
</tr>
<tr>
<td>Reactor coolant flow rate (lb/hr)</td>
<td>$1.5 \times 10^{7}$</td>
</tr>
<tr>
<td>Reactor pressure drop, Design maximum (psi)</td>
<td>100</td>
</tr>
<tr>
<td>Reactor bulk inlet/outlet temperature</td>
<td></td>
</tr>
<tr>
<td>Initial core (°F)</td>
<td>600/900</td>
</tr>
<tr>
<td>Design maximum (°F)</td>
<td>900/1200</td>
</tr>
<tr>
<td>Core temperature rise average,</td>
<td></td>
</tr>
<tr>
<td>Initial/design maximum (°F)</td>
<td>300/400</td>
</tr>
</tbody>
</table>

**Reactor Vessel**

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Diameter (ft)</td>
<td>~18</td>
</tr>
<tr>
<td>Height (ft)</td>
<td>~51</td>
</tr>
<tr>
<td>Wall thickness (in.)</td>
<td>2</td>
</tr>
<tr>
<td>Wall fluence, total (nvt)</td>
<td>$10^{21}$</td>
</tr>
<tr>
<td>Material</td>
<td>304 SS</td>
</tr>
</tbody>
</table>

**Core Design**

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Core lattice positions</td>
<td>91</td>
</tr>
<tr>
<td>Driver fuel subassemblies</td>
<td>73</td>
</tr>
<tr>
<td>Closed loops</td>
<td>6</td>
</tr>
<tr>
<td>In-core open test positions</td>
<td>3</td>
</tr>
<tr>
<td>Peripheral control rods</td>
<td>15</td>
</tr>
<tr>
<td>Equivalent core diameter (in.)</td>
<td>~47</td>
</tr>
<tr>
<td>Active core height (in.)</td>
<td>36</td>
</tr>
<tr>
<td>Fuel pin heat transfer area (sq. ft.)</td>
<td>2,800</td>
</tr>
<tr>
<td>Core coolant velocity, maximum (ft/sec)</td>
<td>30</td>
</tr>
</tbody>
</table>

**Driver Fuel**

<table>
<thead>
<tr>
<th>Characteristic</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Fuel composition</td>
<td>20-30 vol % PuO₂</td>
</tr>
<tr>
<td>Cladding material</td>
<td>316 SS</td>
</tr>
<tr>
<td></td>
<td>70-80 vol % UO₂</td>
</tr>
<tr>
<td>Parameter</td>
<td>Value</td>
</tr>
<tr>
<td>---------------------------------------------------------------------------</td>
<td>---------------------</td>
</tr>
<tr>
<td>Linear heat-generation rate, average (Kw/ft)</td>
<td>7.85</td>
</tr>
<tr>
<td>Overpower factor</td>
<td>1.25</td>
</tr>
<tr>
<td>Engineering hot channel factor</td>
<td>1.13 to 1.25</td>
</tr>
<tr>
<td>Peak linear heat generation at overpower (Kw/ft)</td>
<td>18.0</td>
</tr>
<tr>
<td>Target burnup, average (Mwd/T, Metal)</td>
<td>45,000</td>
</tr>
<tr>
<td>Maximum fuel temperature (°F)</td>
<td>4,200</td>
</tr>
<tr>
<td>Fuel assembly length (ft)</td>
<td>14</td>
</tr>
<tr>
<td>Pin diameter (in.)</td>
<td>0.230</td>
</tr>
<tr>
<td>Spacer-wire diameter (in.)</td>
<td>0.056</td>
</tr>
<tr>
<td>Number of pins per assembly</td>
<td>217</td>
</tr>
<tr>
<td>Subassembly cross-section outside dimension (across flats, in.)</td>
<td>4.615</td>
</tr>
<tr>
<td>Lattice spacing (in.)</td>
<td>4.715</td>
</tr>
<tr>
<td>Duct-wall thickness (in.)</td>
<td>0.140</td>
</tr>
</tbody>
</table>

**Physics Data**

- Delayed neutron fraction: 0.003
- Neutron lifetime (sec.): $3.5 \times 10^{-7}$
- Doppler, $\left(\frac{dk}{dt}\right)$: -0.004

**Power density (Mw/liter)**: 0.4

**Power distribution (peak/average)**
- Radial: 1.35
- Axial: 1.24
- Total: 1.68

**Test Facilities**

**Closed loops**

- Power handling capability (Mw): 4
- Test flow rate (gal/min): 30 - 300
- Test-section outlet temperature (°F): 1400 (bypass flow permitted)
- Test-section length (in.): 36
- Test-section diameter (in.): 2.5
- Pump head, primary (lb/in.$^2$): 250
- Test-section pressure drop, max. (lb/in.$^2$): 90
- Material (in-core tube): 316 SS
In-core open test positions

Power

Coolant-flow rate

Test-assembly length

Test-assembly cross section

Coolant

**Heat Transport System**

Primary loops

<table>
<thead>
<tr>
<th>Number</th>
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</tr>
</thead>
<tbody>
<tr>
<td>Primary loop material</td>
<td>304 SS</td>
</tr>
<tr>
<td>Primary loop flow rate (per loop) (lb/hr)</td>
<td>$5.0 \times 10^6$</td>
</tr>
</tbody>
</table>

Primary pumps

<table>
<thead>
<tr>
<th>Number</th>
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</tr>
</thead>
<tbody>
<tr>
<td>Design pump head (ft)</td>
<td>500</td>
</tr>
<tr>
<td>Net positive suction head (ft)</td>
<td>43</td>
</tr>
<tr>
<td>Design temperature (°F)</td>
<td>1050 (a)</td>
</tr>
<tr>
<td>Speed control</td>
<td>wound rotor motor with liquid rheostat</td>
</tr>
<tr>
<td>Motor power, brake/rated (HP)</td>
<td>1655/3000</td>
</tr>
</tbody>
</table>

Intermediate heat exchangers

<table>
<thead>
<tr>
<th>Number</th>
<th>3</th>
</tr>
</thead>
<tbody>
<tr>
<td>Type</td>
<td>vertical shell and tube</td>
</tr>
<tr>
<td>LMTD, initial/ultimate (°F)</td>
<td>75/100</td>
</tr>
<tr>
<td>Capacity, initial/ultimate (Mw)</td>
<td>133/177</td>
</tr>
</tbody>
</table>

(a) Pump tank will be designed for 1200°F
(2) Whether the fuel elements will be spaced and supported by a helical wire wrap or by intermittent grids

(3) How to design the system to assure core radial restraint

The first involves an issue of safety and operational reliability (reproducibility of the power coefficient). Experimental and analytical investigations are now in progress on the vulnerability of oxide to dry slumping mechanisms, the need to make mechanical provisions to limit molten slumping, and fuel-cladding mechanical interactions.

The other two items have assumed a greater importance in the past two years due to the discovery of irradiation-induced growth of stainless steel. Correlations of empirical data, and theoretical models to predict the magnitude of this effect, are becoming available in large numbers. Figure 7 is one such correlation for the 450 - 550°C temperature range with combined 304 and 316 stainless steel data. The data are weighted in accordance with the reported or estimated accuracy of the measurements, which varies appreciably.

For the maximum design burnup of FFTF fuel elements (80,000 Mwd/T corresponding to a fluence of ~1.5 x 10^{23} of E > 0.1 Mev), the designers believe that the fuel-element support system must be designed to accommodate about 3% ∆D/D. Perhaps half of this could be permitted by original clearances in a wire-wrapped fuel-element bundle. Out-of-pile tests have shown that the other half can be accommodated by deflection of the elements, which occurs preferentially to denting of the clad by the wire. A grid design, which could permit the entire 3% diametral growth without introducing bending stresses, is being developed as an alternative approach.

Accommodating volumetric growth of stainless steel in the design of the core radial support is seen as an even more challenging problem. Several concepts are being studied.
Figure 7

Available Clad Swelling Data
450–550°C

\[
\frac{\Delta V}{V} = 0.191(\phi)^{1.143}
\]

Weighting Factors

LEAST SQUARES LINE
75% CONFIDENCE BANDS
95%
2. **Large Plant Core Designs.** The 1000-Mwe Follow-On design program is still in progress but some of the interim reference parameters are summarized in Table 3. Of the five industrial contractors participating, three have chosen oxide, and two have chosen carbide. All have studied the alternative fuel in limited depth. Four of the five use vented fuel elements, though even the fifth acknowledges the attractiveness of fission-gas venting as a long-range objective.

In contrast to the 1963 - 65 round of 1000-Mwe design studies where three of the five participants used moderator materials in or near the core to enhance Doppler feedback, only one employs moderator in the present studies. The developments on the Pu$^{239}$ a value are responsible for a loss of interest in spectrum-softening features. Also in contrast to previous studies, where all contractors adopted a spoiled geometry to avoid a positive sodium-void coefficient, only two contractors have taken a deliberate economic penalty to avoid this problem in the Follow-On studies.

Four contractors arrange fuel elements on a triangular pitch in a fuel assembly with solid walls; the fifth employs a square pitch with an open assembly.

An average fuel burnup of 100,000 Mwd/T is assumed by most participants. Implicit in achieving this, is the assumption that fuel elements will operate passively under failed conditions for an extended period of time (residence times of up to three years are involved). This has yet to be clearly demonstrated for oxide or any other suitable LMFBR fuel, and it is being investigated in two high-priority experimental programs.
F. Fuels and Materials

1. Cladding and Structural Alloys. During the past year, additional data on the swelling of stainless steels for fuel element claddings and other core components were generated. Although the fluences are still below $10^{23}$ nvt, these new data are useful in narrowing the uncertainty as to the extent of irradiation-induced swelling. Figure 7 is an accumulation of data available as of approximately January 1, 1969, for the 450 - 550°C temperature range. Swelling is recognized as a problem that must be accommodated through design, at least for the near term. Most investigators are convinced that swelling is the result of vacancy clustering. The mechanisms and kinetics of the phenomenon are being actively investigated in hopes of developing a basis for reducing its magnitude in long-term LMFBR applications.

The promise of improved performance of stainless steel for LMFBR cladding and structural applications has been enhanced by experimental results suggesting that alloy additions and/or thermomechanical treatment may be effective in reducing the degree of irradiation induced embrittlement.

2. Recent Irradiation Experience on LMFBR Fuels and Fuel-Element Concepts. In the U.S. Program, fast-flux irradiations have been performed exclusively in the EBR-II. The approximate allocation of space, by material, is shown in Figure 8. Owing to EBR-II space limitations, thermal reactor space is also used for LMFBR-oriented irradiation testing, especially for the investigation of phenomena not thought to be highly dependent on neutron energy. Some recent developments in the irradiation testing of fuels are:

(1) Mixed-Oxide Fuel: Conclusions, based on interim and destructive examinations of nearly 40 fuel elements after fast-flux irradiation up to 7 a/o burnup, indicate that the cladding provides an effective restraint for retarding fuel
Utilization of EBR-II Core Space by LMFBR Materials

- Core portions resulting in reactivity loss:
  - Metal
  - Clad
  - Carbide
  - Oxide

Percent

expansion. It is also estimated that half of the measured increase in diameter (1.5% at maximum burnup) can be attributed to the swelling of stainless steel cladding.

(2) Mixed-Carbide Fuel: Vibratory-compacted mixed-carbide fuel was irradiated successfully in a fast flux at about 25 Kw/ft to a maximum burn-up of 7 a/o. Helium-bonded, pellet-type, mixed-carbide fuel elements, clad with Type 316 SS and Incoloy 800, were also irradiated successfully in EBR-II to 4 a/o burnup at a peak linear power of 28 Kw/ft.

(3) Mixed-Nitride Fuel: Thermal irradiations of both sodium- and helium-bonded mixed-nitride fuel at 20 to 30 Kw/ft to 6 a/o burnup showed swelling between 0.4 and 1.5% ΔV/V per a/o burnup. These rates appear to be less than the swelling rates of mixed carbides.

In out-of-pile studies of the mixed-nitride fuel system, excellent compatibility with stainless steel was found under typical LMFBR temperature conditions. Also, analysis of existing data indicates that neither volatility of mixed-nitride fuels nor nitride-phase instability are expected to be a problem under envisioned operating conditions.

(4) Metal-Alloy Fuel: To date 40,000 EBR-II driver-fuel elements have sustained 1.2 a/o burnup with no failures. Five EBR-II driver fuel elements (Mark IA type) have been successfully irradiated to 2.5 a/o burnup.

3. Fuel Development in Support of FFTF. The major thrust of the FFTF fuel-development program is summarized below:

(1) Steady state irradiations in EBR-II, and transient irradiations in TREAT, will be made to test the necessity and effectiveness of axial-motion restrictors. Encapsulated fuel elements will be involved. Nineteen encapsulated fuel specimens, clad in 316 stainless steel, will be irradiated to 50,000 Mwd/T to determine the effect of void deployment on swelling and growth.
(2) Another test assembly in EBR-II, with encapsulated elements, will be used to verify performance of mixed-oxide fuel, clad in 316 stainless steel, to 100,000 Mwd/T with interim inspections and selective removal of individual elements for destructive examination.

(3) Nine 37- or 61-pin prototypic assemblies will be used to provide statistical confidence in the adequacy and reliability of the FFTF fuel-element design. These tests will incorporate varying degrees of cladding cold work, and linear power ratings corresponding to the low, mean, and high FFTF fuel-element linear power ratings. These are unencapsulated tests in EBR-II.

(4) Additional 37- or 61-unencapsulated-pin assemblies will be used to gain statistical data on the effects of fuel form, smeared density, void deployment, and burnup. A recently-developed instrumented assembly will be utilized in a test of 37 prototypic unencapsulated FFTF fuel elements. Accurate coolant and cladding temperature data will be obtainable in this vehicle. Most of the data needed for the development of mixed-oxide fuel for early cores of demonstration plants will be obtained as part of the FFTF program. Many parameters of interest are similar, as shown in Figure 9.

G. Fuel Recycle

Fuel recycle includes recovery and purification of uranium and plutonium in discharged fuel assemblies, and fabrication of the recovered elements into new fuel assemblies for return to the reactor. For LMFBRs, fuel-recycle operations are in an early stage of development. Near-term efforts are aimed generally at providing essential capabilities; long-term efforts are aimed at improving economics at least sufficiently to make an LMFBR a competitive source of power.
### FIGURE 9

LMFBR PROGRAM  
**FUELS & MATERIALS**

DIFFERENCES IN MIXED OXIDE FUELS FOR LMFBR AND FFTF FAST REACTORS

<table>
<thead>
<tr>
<th></th>
<th>LMFBR</th>
<th>FFTF (PRELIM.)</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>ACTIVE FUEL LENGTH</strong></td>
<td>36&quot;</td>
<td>36&quot;</td>
</tr>
<tr>
<td><strong>IN-CORE LIFE</strong></td>
<td>2 YEARS</td>
<td>1 YEAR</td>
</tr>
<tr>
<td><strong>AVERAGE POWER</strong></td>
<td>9.6 KW/FT</td>
<td>7.8 KW/FT</td>
</tr>
<tr>
<td><strong>MAXIMUM POWER</strong></td>
<td>18 KW/FT</td>
<td>14.4 KW/FT</td>
</tr>
<tr>
<td><strong>CORE POWER</strong></td>
<td>165 KW/LITER</td>
<td>400 KW/LITER</td>
</tr>
<tr>
<td><strong>PuO₂ MAXIMUM</strong></td>
<td>15 W/O</td>
<td>21 W/O</td>
</tr>
<tr>
<td><strong>AVERAGE BURNUP</strong></td>
<td>112,000 MWD/T</td>
<td>45,000 MWD/T</td>
</tr>
<tr>
<td><strong>MAXIMUM BURNUP</strong></td>
<td>150,000 MWD/T</td>
<td>80,000 MWD/T</td>
</tr>
<tr>
<td><strong>MAXIMUM FUEL TEMPERATURE</strong></td>
<td>5,200° F.</td>
<td>4,200° F.</td>
</tr>
</tbody>
</table>
The major near-term effort is on fabrication rather than reprocessing technology because initial cores for reactor startup are required several years before there is a need to recover spent fuel. Both the fabrication and reprocessing technologies benefit greatly from experience with LWR fuels. Therefore, the listing of major accomplishments includes those which, although directly applicable to LWRs, ultimately will benefit LMFBRs.

Major accomplishments in 1968, and the major programmatic thrusts are described below:

1. Fuel Fabrication. Fabrication of the mixed-oxide fuel loading for SEFOR was completed. A pilot facility for developing and demonstrating fabrication of LMFBR-FFTF fuel was set up and put into operation at Pacific Northwest Laboratory. Industry will fabricate FFTF fuel. Specifications and criteria for fabricating FFTF fuel and cladding were established preparatory to qualifying commercial vendors.

2. Fuel Shipping. The first shipping cask to be shielded with laminated uranium was built. The immediate application is for shipping LWR fuels, but future application to LMFBR fuels is probable. A guide for design and fabrication of lead-shielded casks was developed at Oak Ridge National Laboratory. In addition, regulations covering shipment of radioactive materials were published by the Department of Transportation and subsequently adopted. These items are evidence of a maturing technology that will benefit LMFBRs.

3. Fuel Processing. Recovery and recycle of EBR-II fuel by means of the EBR-II Fuel Cycle Facility (FCF) is being discontinued. In the FCF, discharged fuel cooled only 15 days, was routinely handled, processed, and fabricated into new fuel elements — all by remote operations. Although the fuel was metallic and much easier to handle than mixed oxides, successful
demonstration of the process for a prolonged period of time, some four years, is of significance to the LMFBR Program. This is because some form of remote operations may be required for the production of large quantities of fuels using high exposure. The FCF is now being devoted exclusively to servicing the irradiation program. During the four years of operation as a production facility, some 40,000 fuel elements were manufactured in the FCF, none of which failed in the reactor. The process equipment was operating or operable greater than 90% of the time and all maintenance that was required was carried out remotely.

4. Current Programmatic Thrust. Major remaining fuel-recycle developments include:

1. Developing a capability for handling short-cooled fuel having high decay-heat rates in such operations as: transfer, sodium cleaning, storage, and disassembly, chopping, and other steps to prepare fuel for inspection, shipping, or dissolution.

2. Establishing the feasibility of shipping short-cooled LMFBR fuels and, if feasible, developing the needed technology.

3. Extending the aqueous processing technology to accommodate, first, early LMFBR fuels discharged from FFTF and demonstration plants requiring only a modest extension of technology, and second, fuels discharged from commercial plants requiring a substantial extension of the aqueous technology. The feasibility of recovering commercial (high-performance, short-cooled) LMFBR fuels by aqueous methods has not been established. Fluoride-volatility and pyrochemical processes are also being investigated.

4. Developing the necessary technology to economically fabricate LMFBR fuels to the exacting specifications and high standards needed to ensure the high performance sought for these fuels. This will require that considerable
attention be given to quality assurance and, hence, to the development of quality-assurance methods and nondestructive testing and inspection procedures. An important matter to resolve in the near future is whether recycle LMFBR fuels must be fabricated remotely as a result of a high intrinsic radiation level of the plutonium isotopes present. Remote fabrication would increase greatly the effort needed to develop suitable fabrication methods and equipment.

H. Physics

The major objective of the physics program is the development of reliable methods and data for nuclear design. The development work is directed toward three broad technical areas:

(1) Determination of basic nuclear data including cross sections and other nuclear properties of materials. This encompasses measurement, evaluation, compilation, and theoretical understanding.

(2) Obtaining integral data

(3) Developing and improving calculational and analytical capability

Selected major accomplishments in these three areas of development are described below.

1. Basic Data. Among the rather extensive results obtained in the program of cross section measurements carried out under LMFBR auspices, the one measurement most publicized recently is that of $\alpha$ of Pu$^{239}$. This was a joint effort by the Oak Ridge National Laboratory, who provided detection equipment and analysis of data, and Rensselaer Polytechnic Institute, whose linear accelerator was used as the neutron source. The measurements covered the neutron energy range up to about 25 kev. While the results are still being scrutinized, no major changes to the values originally published by Gwin, et al, as plotted in Figure 10, have been identified.
FIGURE 10
Values of $\alpha$ for Pu$^{239}$

- GWIN ET AL.
- SCHOMBERG ET AL.
- SCHOMBERG ET AL.
- DE SAUSSURE ET AL.
- HOPKINS & DIVEN
- KAPL

$\alpha(E)$ vs. Neutron Energy (eV)
In general, the results confirm that $\alpha$ is higher between 200 ev and 20 kev than that which has previously been used (see ENDF/B curve in Figure 10), but not as high as had been indicated by the preliminary results of other recent measurements such as Schomberg, et al, (IAEA Karlsruhe meeting in October 1967). Of further interest is the resolution of specific structure up to 1 kev. It is noted that the Gwin and the Schomberg measurements are in concert here. Work is now underway unravelling the capture and fission cross sections.

A significant increase in the quality and quantity of data on cross sections for Pu-239 and other fissile materials is expected in the near future, as the linear accelerator at Oak Ridge (ORELA) and the fast neutron generator at Argonne (AFNG) are brought into operation. It is planned to initiate experiments on these this spring and summer. The Fast Neutron Generator will be used to extend measurements of scattering and other reactions at high energies. ORELA is expected to produce a higher neutron intensity with lower background than has previously been available and will be used in the range up to a few hundred kev. Effort on the RPI Linac will be concentrated on the non-fissile materials.

Another accomplishment, closely associated with measurement of cross sections, was the distribution of the first complete version of the evaluated nuclear data file, ENDF/B. This cross section set has now been used by a variety of organizations throughout the U.S. and in some instances outside the U.S. Significant and meaningful feedback from users has indicated a variety of areas in which improvements are required, and a coordinated effort to provide these improvements is underway.
2. **Integral Data.** A broad effort is underway at a number of centers toward obtaining a capability for routine measurement of neutron spectra in bulk media. Results have recently been distributed of experiments in which linear accelerators have been used to pulse subcritical configurations to measure the neutron spectra using the time-of-flight technique. Simultaneously, extensive measurements are underway using proton recoil counters, with supporting efforts on other methods. Time-of-flight capability is being added to the ZPR-VI facility at ANL.

As of the date of this writing, only one U.S. facility was licensed to perform large scale plutonium fueled critical experiments. Most of the available time on this facility (ZPR-III) during the past year has been used for experiments related to the design of the FFTF. A basic core configuration has now been selected for FFTF, and the critical experiments are concentrating on a generalized representation of that design, to be followed by a detailed engineering mockup about a year from now.

Another important accomplishment has been the completion of the Zero Power Plutonium Reactor (ZPPR). This facility has been completed and checked out, and is now awaiting receipt of a license to operate. Also significant has been the conversion of ZPR-VI and -IX for plutonium use. This work is almost complete, and it is expected that these facilities will be available for use with plutonium within the next few months. Meanwhile, plutonium fuel has been fabricated for use in all three facilities. A total of approximately 3000 kg is available for use interchangeably among the four ZPR facilities.

Initial operation of these facilities will allow the rapid accumulation of urgently needed integral data. Initial configurations will concentrate on large scale benchmark plutonium assemblies, progressing later to configurations which are of more specialized interest.
3. **Computational Methods.** The development of computational methods has recently been characterized by a number of small advances rather than any major ones. For example, a variety of computer codes have been made available recently for various types of two-dimensional theory calculations. Some of these codes will improve calculational efficiency and data handling for specific machine configurations. There has also been a considerable increase in the capability for generating group cross sections for reactor calculations. Much of this capability has been obtained by coupling older computer codes with the ENDF/B library, thus providing a standard source of microscopic cross sections. At the same time, a format has been established that should contribute significantly to the future sharing of evaluated nuclear data.

Several computer codes for calculating the effects of burnup and recycling of fuel are nearing completion. In addition to further developments of the calculational capabilities just mentioned, work on the use of synthesis techniques will be intensified in the near future. A number of methods already proposed are being investigated. The major concern is to determine which methods are dependable for fast reactor calculations and to develop computer codes for their application.

A major effort is currently being mounted to attempt to improve the degree of coordination and compatibility of coding efforts in the U.S. A working group is being collected to seek specific areas of standardization, such as code interfaces.

I. Safety

The LMFBR safety program is a continuing effort designed to improve the technological base needed to provide realism and confidence in the
understanding and analysis of accident situations, to develop and evaluate safety systems for the prevention of accidents and mitigate their consequences, and to develop standards and codes for the safe design, siting, construction, and operation of both USAEC test facilities and commercial LMFBRs.

A major source of guidance for the safety program is the 1000-Mwe Follow-On Study program, which is now nearing completion. Five industrial design groups have completed reference 1000-Mwe LMFBR designs and performed limited safety analyses to support the designs. These analyses have demonstrated, to some extent, the approach to safety through specification of preventive and consequence-limiting safety systems. Negotiations are underway for more extensive safety analyses on at least two of the reference plant concepts. These more comprehensive safety analyses will provide a mechanism to promote meaningful discussions on LMFBR safety between all interested parties, including industry, AEC laboratories, regulatory groups, the AEC, and the LMFBR Program Office. They will also guide further development of the safety program and provide an evaluation of how the AEC "General Design Criteria for Nuclear Power Plant Construction Permits" applies to the conceptual LMFBR plant designs being considered.

1. Analytical Models and Codes. The major thrust of the current safety program is toward the development of accident analysis methods and the research and testing required to substantiate these methods. The development of codes for analysis of initiating accidents is proceeding at ANL, GE, and PNL. ANL is developing a code system, SAS-IA (Safety Analysis System-IA). SAS-IA is a package of accident-analysis modules linked together to form an integrated code. One of the modules describes the voiding and two-phase flow associated with flow-coastdown or blockage incidents; another treats the
axial motion of fuel up to the time of cladding rupture. GE and PNL efforts are focused on the development of code packages which combine simplified voiding and fuel-melting models to provide more realistic estimates of reactivity-insertion rates associated with hypothetical accident situations. The ANL and GE codes will be described at an ANS meeting in April 1969.*

The development of codes for the computation of the energy release in large nuclear excursions was highlighted by the publication by PNL of the MAX code, which includes a hydrodynamics model consisting of many movable and compressible mesh points, this being a significant improvement over older codes. The ANL SAS-IA also includes, as a module, an improved version of the MARS code, originally developed by APDA, for analysis of core disassembly. A two-dimensional hydrodynamic code to describe energy absorption by primary systems and containments has been developed at ANL.

2. Coolant Behavior. Studies are in progress which are directed toward the understanding of sodium boiling and two-phase flow required to describe void formation, coolant motion, and pressure generation during reactor accidents. Incipient boiling superheat has been under study at ANL where pool-boiling experiments with sodium have been completed. Comparison of superheat data from a sandblasted surface with that from a polished surface showed little difference. A theory developed by Singer and Holtz of ANL, based on the pressure-temperature history of the surface, describes the observed effects. A more refined theory, which includes the effects of dissolved gases, is expected to further improve the understanding of liquid superheat. Some superheat data resulting from tests with flowing sodium have been published by AI. These data

indicate a minimal dependence of the pressure-temperature history on incipient boiling superheat. A report, AI-AEC-12767, of AI's forced convection boiling studies will be published in the summer of 1969. Studies at ANL of the flashing of liquid sodium in a cylindrical, divergent test section have shown that sodium exhibits strong non-equilibrium effects. Propagation velocities of pressure disturbances in two-phase mixtures have been measured.

Out-of-pile simulations of sodium behavior in multirod geometry are planned at several sites. The development of resistance-heater rods for the simulation experiments is underway at AI and at PNL. At ANL, electron-bombardment heater rods are under development. The multipin simulations will facilitate observation of the combined effects of superheat, void growth, void collapse, and two-phase flow in realistic configurations.

A major objective of the sodium-boiling studies is the development and testing of mathematical models of channel voiding. Existing models have been reviewed and intercompared.

Another aspect of sodium boiling which is under vigorous investigation is the potential for buildup of large pressures when molten fuel contacts sodium. Experiments to define the rate of heat transfer from heated spheres to water and to sodium have been completed. Heat-transfer rates in excess of $10^7$ Btu/hr sq. ft. have been found for heat transfer to sodium. Laboratory experiments and TREAT capsule tests, which are designed to combine the effects of particle heat transfer, fuel fragmentation, and pressure-pulse generation under conditions of realistic geometry, are proceeding at ANL.

3. Fuel-Element Failure Effects. Experiments, both out-of-pile and in-pile, are being conducted by ANL to study failure propagation for oxide fuel elements. The current out-of-pile experiments are to study the effects of gas release from a clad failure in a 7-pin bundle in a water loop. A flowing
sodium loop is being developed. The in-pile experiments are being planned and will be performed in a thermal test reactor. These will make use of a 19-pin bundle in a flowing sodium loop.

In-pile experiments to study defected fuel element behavior are underway at GE. Capsule-type experiments are performed in the GETR in which fuel is melted in sodium flowing by forced convection; single pins are being used in the first experiments with three pin tests expected shortly. These latter tests should provide useful information on fuel element failure propagation. Experiments are being performed in the TREAT reactor to study transient mobility of molten fuel. Fuel has been preirradiated to 20,000 Mwd/T for these experiments; further, transient tests on higher burnup material are underway.

The Mark II loop is a flowing-sodium package loop, developed at ANL, which will be installed soon in the TREAT reactor. This is the first dynamic loop designed to meet the high containment requirements for complete meltdown of oxide fuel in sodium. In this 7-pin loop, experiments are planned to study such problems as: fuel-failure thresholds; movement of material prior to failure; mechanism of failure; fuel and coolant movement accompanying failure; and pressure generation.

4. Miscellaneous. Experiments are being performed at BNL to measure the thermodynamic properties of fission products in sodium solutions. Some results of these experiments show that the vaporization of cesium from liquid sodium into an inert-gas stream can be predicted from the limiting condition of equilibrium vaporization. Other elements being studied are rubidium, barium, tellurium, antimony, and iodine.

Studies of PuO$_2$-Na and UO$_2$-Na aerosols at Brookhaven and AI have shown that initially the aerosols are composed of individual fuel and sodium
particles. After several hours of aging, the fuel and sodium become incorporated into composite particles. The results of coagulation and settling behavior of aerosols are in good agreement with theory over the first one and one-half orders of magnitude decrease in aerosol mass concentration.

Experiments on sodium-pool fires are carried out at AI in the Large Fires Apparatus. A computer code, SOFIRE, has been developed from the results of these experiments to predict the sodium burning rates and resultant pressures. A similar experimental system is being developed to study sodium spray fires.

A study is underway at ANL to evaluate, in-depth, the type and number of in-pile experiments needed in the LMFBR safety program, as part of a determination of test-facility requirements. This work will then determine the most effective utilization of existing or planned facilities as well as the need for modified or new facilities.
IV. PROJECTS AND FACILITIES

The current status, past year's accomplishments, and future plans for selected projects and facilities are summarized below:

A. EBR-II

The EBR-II, which is the only facility in the United States for fast flux steady-state irradiation testing of LMFBR fuels and structural materials, operated considerably better during this past year than it did the year before. Last year's summary paper, you may recall, reported a plant factor of only 20%, a serious anomaly in the power coefficient, a sodium fire in a building containing secondary system piping, and several other difficulties. The rectification of these problems, and other recent accomplishments and near-term plans, are briefly summarized below.

1. Plant Factor. The plant-factor problem with EBR-II has had considerable attention in the past year. An improvement program was undertaken and the results are impressive. The plant-capacity factor has been raised in each of the last three operating quarters. The last 3-quarter period averaged just under 50%, as compared with a previous average (1965 to 1968) of less than 33%. The last quarter averaged 55.8% and, while increases are expected to be smaller in the future, steady improvement will be a goal.

2. Power Increase. In August 1968, power was increased from 45 to 50 Mwt. Except for scheduled operations at lower power (for certain experiments and to locate failed fuel), this level has been held since then. Further increases to 62.5 Mwt, in several steps, are contemplated for the future.

3. Reactor-Physics Anomaly. The reactor-physics anomaly
discovered during Run 25 (April 1967) was reported last year. Since then, the cause of the flattening of the power/reactivity curve has been determined and corrected. Flattening was caused by substituting a stainless steel reflector for the inner blanket (uranium) assemblies. This reflector had greater structural rigidity than the blanket, and the assemblies experienced a much different radial thermal gradient during power operation. This resulted in mechanical restraint to the core and an interrupting of the normal free thermal bowing of fuel assemblies in the mid-power range. The effect was corrected by replacing the reflector with blanket assemblies. Figure 11 illustrates the power/reactivity curve with the reflector intact (Run 29A), with partial replacement (Run 29B) and full replacement (Run 29C). Redesign of the reflector, avoiding the previous problems, is contemplated for the future.

4. Unencapsulated Fuel Experiments. Five new types of fuel experiment are now in EBR-II. These are mixed-oxide assemblies, each containing 37 unencapsulated fuel elements in flowing sodium. With fully enriched elements, these 37-pin assemblies substantially match the reactivity of the replaced metal-driver assembly. This essentially eliminates the reactivity limitations previously encountered in the use of encapsulated elements, of which only 19 could be placed in a test assembly. Future plans also include unencapsulated elements of smaller diameter with 61 elements per assembly.

The projected large-scale use of the facility for oxide irradiations raises questions of the kinetic behavior of the reactor. The approach adopted is to perform a series of experiments at increasing oxide/metal fuel ratios. These are just beginning.

5. Instrumented Fuel Assembly and Other Facilities. An
FiguRE 11
EBR-II Power Coefficient Data (Run 28)

- RUN 28A, 1/29/68, REFERENCE CORE
- RUN 29B, 7/10/68, S.S. REMOVED FROM ROW 7
+ RUN 29C, 7/17/68, ALL S.S. REMOVED
instrumented fuel assembly has been developed for EBR-II. This assembly utilizes former control-rod positions and hence is smaller in cross-section than a regular fuel assembly. Up to 23 separate sensors can be accommodated; not all, however, in any one assembly. These sensors will measure coolant flow rate, coolant and fuel temperatures, fuel compartment pressure, and flux level (monitor wires). The first dummy and UO₂ units are being assembled and checked out at present with the first mixed-oxide experiment being scheduled for the next convenient shutdown period.

An out-of-core instrument test facility and a radioactive sodium chemistry loop will be fabricated. Other increased experimental capabilities for EBR-II are under study. These include an in-core instrument test facility.

6. HFEF. The Hot Fuel Examination Facility (HFEF) is a hot-cell complex being designed specifically to support the LMFBR program and will be located adjacent to the EBR-II. Fuels and materials irradiated in EBR-II, TREAT, ETR, PBF, and elsewhere will receive interim and terminal examinations in this facility. The assembly of components (fuel assemblies and loops) containing pre-irradiated fuels also will be performed in HFEF prior to insertion in an irradiation facility.

The heart of HFEF is a shielded rectangular cell, 70-ft long by 30-ft wide by 27-ft high (Figure 12). Fifteen work stations with leaded-glass windows, mechanical master-slave manipulators, and utility services will be located in this cell. In-cell materials will be transported by overhead systems. The cell will have an inert-gas atmosphere.

Two smaller, shielded support cells will be located beside the large cell. These cells will contain an air atmosphere and will be used for
supporting activities, such as decontamination of fuel and equipment as well as remote maintenance of equipment. Eight work stations will be in these cells.

Operation of the HFEF is planned for mid-1972.

7. FCF Conversion. The Fuel Cycle Facility (FCF) at EBR-II is being phased out of active fuel-fabrication operations owing to the introduction of a commercial source of fuel. This facility has always had a role in driver-fuel-surveillance examination. This role has been expanded and the FCF is now being equipped to provide interim examinations and to prepare discharged experiments for shipment. This role will continue until HFEF is available.

B. Enrico Fermi Reactor

The industry-owned Enrico Fermi Atomic Power Plant, with a rated capacity of 430 Mwtt (155 Mwe), is the largest fast reactor in the U.S. It was inoperative during the past year as a consequence of the incident on October 5, 1966 involving extensive melting in two fuel assemblies.

As you may recall, last year's summary paper reported that the object causing flow blackage of the assembly nozzles had been seen and photographed through a borescope. The object, though badly bent and mutilated, was identified as one of the six, thin zirconium segments that covered the conical flow guide in the inlet plenum. Ironically, these pieces were not a part of the original design of the vessel. They were installed as a last-minute "safety feature" in the event of a major core meltdown. Their purpose was to prevent molten fuel from reacting with the stainless steel flow guide and to disperse the fuel into a shallow, zirconium pan near the bottom of the vessel, where it would assume a subcritical configuration.
1. **Removal of Objects from the Reactor Plenum.** Last March, work began on the task of penetrating a primary elbow in one of the three, 14-inch, sodium inlet lines to provide access for inserting a specially designed retrieval device into the inlet plenum. The plan was to operate a spine-type manipulator through the core-support plates to hand the object to the retrieval device. The device, gripping the object, was then to be snaked through the 35 feet of 14-inch pipe to the site of the penetration. Figure 13 depicts the complex operation. After two unsuccessful retrieval attempts and the correction of numerous, intervening equipment problems, the object was removed from the reactor on March 22, 1968 — 17 days after the patch was removed from the elbow of primary pipe. Figure 14 is an approximately full-scale photograph of the object after cleaning. The torn material around the screw holes is clearly visible.

The apparently inadequate method of attaching the zirconium segments to the flow guide made it necessary to remove the remaining five pieces. This was an even more difficult assignment because the plates were still attached. Two alternative methods were considered for detachment: (1) burning off the screw heads by a remote arc-melting operation, and (2) cutting off the screw heads with a chisel-like tool. After considerable development of both alternatives, including trial runs with a full-size mockup facility, the arc-melting operation was adopted. Figure 15 shows one of the zirconium segments removed from the mockup during the procedure-development phase, and also illustrates the as-fabricated shape of the segments. Thanks to the extensive development work, the remaining five segments were removed without incident in December. During that operation, it was discovered that one of these five had also detached from the flow guide and was loose in the plenum.
Fig. 13. Removing Object from Enrico Fermi Reactor Core Inlet Plenum

Courtesy of APDA
Fig. 14. Object Responsible for Enrico Fermi Reactor's Fuel Melting Incident of October 5, 1966 (After Cleaning)
Fig. 15. Zirconium Segment from Enrico Fermi Reactor Mockup after Removal of Screwheads by Arc-melt Process

Courtesy of APDA
Current status is: the lower plenum has been vacuum cleaned (about 1/2 cupful of weld spatter was removed), and the opening in the 14-inch inlet pipe has been sealed in preparation for refilling the system with sodium.

2. Future Plans. The present schedule for the Fermi Project, assuming no unforeseen delays in relicensing, is:

- Fuel reloading -- May 1969
- 200 Mwt demonstration run -- September 1969
- Commence 110 Mwt steady-state operation with partial core loading of fuel and cladding test assemblies -- January 1970.

About 11 core-lattice positions, comprising about 10% of the core, are envisioned for the test loading. Potential sponsors of Fermi irradiation experiments are: The USAEC, Euratom, Japan, and the Edison Electric Institute. Counting driver fuel undamaged by the October 1966 incident and that in dry storage, about 250 days of operation at 110 Mwt are possible. This is adequate for fully enriched mixed-oxide fuel elements to reach 50-70,000 Mwd/T, depending on core position.

The next core loading for the Enrico Fermi Reactor will be oxide, although it has not yet been decided whether it will be UO$_2$ or (U,Pu)O$_2$. Atomic Power Development Associates expects to complete a fuel specification this month, collect bids from industry this summer, and place an order in September 1969. The final choice between a UO$_2$ and a mixed-oxide loading will be made after further assessment of the relative licensing complexities and relative fabrication costs.

C. SEFOR

There has been a great deal of progress on the SEFOR reactor in the past year. Construction is complete; title for the plant was conveyed
to the customer, SAEA, on November 1, 1968.

Non-nuclear pre-operational testing has been completed. The present schedule is to initiate fuel loading by mid-March, as soon as the 1 Mwt operating license is officially conferred. The license for full power (20 Mwt operation) is anticipated later this year.

It is, of course, anticipated that minor problems will arise during the startup and initial operation. The startup schedule allows a normal amount of time for these. In early January, for example, after a period of primary-sodium circulation, a filter that had been installed to remove particulate matter which may have been left from plant construction, was found to have been partially torn from its moorings. Most of the pieces of the ruptured filter were trapped by the backup filter and removed, and the small amount of filter material remaining in the system is judged not to present an operational problem. These operations caused a delay of a few weeks in the startup schedule. Unless something quite unusual happens, however, results of the first oscillator measurements are expected within the next year. A rather extensive series of such measurements is planned. In addition, power oscillations at essentially constant coolant temperature eventually will be included so that the reactivity coefficient of fuel can be isolated.

The schedule provides for putting the fast reactivity excursion device (FRED) into operation within the year. This device will produce abrupt reactivity increases to test the consequent behavior of the system. The tests will be carried to or beyond prompt criticality, depending on the size of the resulting transients.

In all, a three-year program of physics tests is planned. The further use of the facility is under consideration and the Program
Plan calls for a technical evaluation of potential uses of SEFOR beyond its current program.

D. Fast Flux Test Facility (FFTF)

During 1969 the design of the FFTF was firmed. The conceptual design of the overall facility was selected and the preliminary design was started. Functional and specific design requirements were identified and guidelines for concept development were established. An architect-engineer and a reactor plant designer were selected and began work. Conceptual design work on some major components is still in progress. Development programs have been initiated and a pilot line for fuel fabrication is 95 percent complete and partially operative. A site was selected.

The FFTF is being managed by the Pacific Northwest Laboratory (PNL) which is operated by the Battelle Memorial Institute for the AEC. The FFTF will provide an adequately controlled and instrumented fast-neutron environment in which fuels and materials in support of the LMFBR Program will be tested and examined. The facility will have a capability to test fuel up to and including failure (short of planned meltdown) in dynamic sodium.

The reactor concept is a vertical, trisected core design with six sodium-cooled closed loops and two contact and one proximity instrumented open test positions. Fuel handling is accomplished through three ports in the reactor vessel head -- each port serving one-third of the core. The reactor is located below the operating floor level within a containment structure, as shown in Figures 16 and 17. An air-filled machinery dome is located over the reactor to provide additional protection from the consequences of an accident.
FIGURE 16

Reactor Arrangement - FFTF Vertical Core Reference Concept

* Failed Element Detection and Location
FIGURE 17
Plant Layout - FFTF Vertical Core Reference Concept

Containment Vessel

Fuel Handling Machine

Location for Nuclear Proof Test Facility

Fuel Handling and Examination

Reactor

Machinery Dome

Closed Loop Cells

Halo Transport System
The reactor core is comprised of 91 assemblies. The assemblies are about 14 ft. long and are hexagonal in cross-section. The driver fuel assemblies contain a 217-pin array of fuel elements. Each element contains PuO$_2$-UO$_2$ fuel clad in a 0.230-in. diameter, Type 316 stainless steel tube. A 36-in. long gas plenum can be accommodated. Three scram-assisted control rods are located within the core. Fifteen gravity-drop control rods are located in the radial reflector. All control rods use B$_4$C as the neutron absorber and all are driven by top-mounted drive mechanisms.

The closed loop and open test positions are vertically oriented and located along three radii of the core in a Y-shaped array. The arms of the Y are located at ~120° intervals. Two closed loops are located at essentially the core center, one closed loop and two open test positions are located at approximately the mid-radius of the core, and three closed loops are positioned at the core periphery.

All driver-fuel positions are instrumented for outlet-coolant temperature, coolant flow rate, and fuel-failure detection. The instrument sensors are located in instrument probes attached to an instrument-plate assembly located directly above the core and supported by the refueling plug. Three such refueling plugs and instrument plates are provided -- each serving a one-third sector of the core. The instrument plate provides an instrument probe for each fuel assembly in the sector, and provides back-up holddown for the fuel assemblies (primary holddown is provided by hydraulic balance).

Each closed loop is instrumented for inlet- and outlet-coolant temperature, coolant flow rate, failed-fuel detection, differential pressure across the test section, and flux monitoring and mapping. Additional capability is provided which permits the measuring of fuel
Refueling of the core is accomplished by a combination of in-vessel and ex-vessel mechanisms. The in-vessel device is a fixed-length grapple assembly positioned by means of a plug located in the refueling port. This machine transfers fuel assemblies between locations in the core and finned pots outside the core. The refueling ports have rotatable plugs for positioning the instrument probes and the in-vessel fuel-handling machine. Fuel assemblies are brought into and out of the reactor vessel in the finned pots by a shielded; ex-vessel fuel-handling machine. A forced-convection argon-coolant system is used to remove decay heat.

The reactor coolant system consists of three primary and three secondary loops, each having an ultimate capacity of 177 Mw. Heat generated in the reactor is dissipated to the environment by sodium-to-air heat exchangers. The coolant system is designed to accommodate a 1200°F mixed-mean outlet temperature from the reactor with an average core-temperature difference of 400°F. Initially, the reactor will be operated at reduced levels of power and system conditions. At an initial power level of 400 Mw (133 Mw per loop), the expected operating conditions are 900°F mixed-mean outlet from the reactor and a 300°F rise across the core.

Table 4 is a summary list of characteristics and data.

During the past year, Westinghouse Electric was selected to be the designer of the reactor plant, with Atomics International serving as the principal subcontractor in the area of fuel handling and sodium-coolant-system technology. The Bechtel Corporation was selected to be the architect-engineer for design of the more conventional areas of
the facility. All three organizations have participated in the development of the conceptual design and are currently beginning or preparing to begin the preliminary design phase, preparatory to start of construction.

Some additional accomplishments during the past year are:

- Integrated FFTF Reactor Refueling Concept selected
- Basic three-loop heat transport systems selected
- Preliminary fuel cladding, and fuel element process and product specifications established and transmitted to vendors
- Preliminary design initiated
- Full size S/A flow test completed in water and initiated in sodium
- Resistance heated prototypic geometry pins for thermal testing achieved heat fluxes of 500,000 Btu/hr-ft²-°F in 800°F sodium
- Life test of 7-pin bundle in 1060°F Na completed (8500 hrs)
- FTR critical experiments in progress at ANL operated facilities, first phases complete
- Initial hydraulic tests on mockup of core holddown and reactor vessel features completed
- Fast irradiation complete and post irradiation examination of initial test fuel pins initiated
- Fast and thermal reactor irradiations of prototypic fuel pins in progress

E. Liquid Metal Engineering Center (LMEC)

The LMEC will serve as the principal AEC center for the testing of components for sodium systems. Currently, it has a large variety of test facilities, ranging from a sodium chemistry laboratory to a 35-Mw test loop. The addition of facilities for testing large, sodium pumps; shaft seals and bearings; instrumentation; and other components is underway.

A facility for testing large sodium pumps, 60,000 to 120,000 gpm, is being designed for installation at the LMEC. Construction of the SPTF will provide the first U.S. experience in the design, fabrication,
and operation of a large, high-temperature, sodium-piping system that is typical of future LMFBRs. The SPTF will also provide capability to test instruments (e.g., large flowmeters), large valves, and other components of a large sodium-piping system. Valuable experience will be gained from its operation and maintenance. The conceptual design has been completed and the preliminary (Title I) design has been started. The facility is scheduled for completion in 1971.

The LMEC also serves the AEC as technical managers of many of the component development programs.

Highlights of accomplishments of the center during 1968 include:

- Initiated training of industrial personnel in sodium technology
- Completed optimization studies and started the definitive design of the Sodium Pump Test Facility
- Repaired the steam system of the Sodium Component Test Installation, made some minor modifications, and prepared for return to service
- Completed two designs of pump-seal rigs, and nearly completed their fabrication
- Completed a draft of the Liquid Metal Engineering Handbook; publication is expected about mid-1970.

In addition to these accomplishments, two other activities of note were undertaken.

A maintenance and malfunction reporting and analysis system has been established for LMFBRs. It is planned to collect operation, maintenance, and failure data from all LMFBR facilities. The objective of this undertaking is to upgrade operational reliability and availability of these facilities, and to help develop adequate specifications and standards. A Miracode storage and retrieval system became operational.
A Failure Data Handbook is being prepared.

The LMEC, in conjunction with Oak Ridge National Laboratory, is developing specifications and standards for components, materials, processes, and operations for application within the LMFBR Program.

Draft standards for six components were completed.
V. CONCLUDING STATEMENT

The American representatives to the International Working Group on Fast Reactors have presented to you the important events which have occurred since our last meeting here in March 1968.

In the two previous meetings of the International Working Group on Fast Reactors, the U.S. representatives also presented to you information of the same character. The presentation of substantive detailed information, including problem areas and their means of resolution has been done because of our sincere intention for close cooperation with the International Atomic Energy Agency.
ACKNOWLEDGEMENT

The authors are grateful for the cooperation of the following organizations contacted during the preparation of this paper: Argonne National Laboratory, Atomic Power Development Associates, General Electric, Liquid Metal Engineering Center, and Pacific Northwest Laboratory. The contributions and assistance of numerous personnel in the AEC Division of Reactor Development and Technology and the LMFBR Program Office at ANL are acknowledged.
Discussion of the Paper by Dr. Wensch

Schuster: What are the programmes in your country relating to fuel development for fast reactors and, in particular, your plans on the development of carbide fuels?

Wensch: For 1000 MW(e) studies, Combustion Engineering and Westinghouse did select the mixed carbide. Since that time Westinghouse has, and is, re-evaluating 1000 MW(e) plant design to use mixed oxide. We believe that the latter is better for the first cores. All the demonstration plant designs use mixed oxide.

Smith: I think when you mentioned major milestones of the U.S. programme, it differed from the official programme.

Wensch: What is different today from the A.E.C. plans is that we have not requested Congress, or the Bureau of the Budget, for the money for the second and third demonstration plants. I think it is quite likely that we will seek to proceed with the three plants all together, but officially we have plans to proceed with only one; so we think of three plants but invest only for one.

Schuster: What is the way to decide in your country which manufacturer will be chosen to construct the first demonstration plant?

Wensch: It will be by competitive means, evaluation of proposals, the evaluation of the capability of the organisations, and the cost for the Government.

Smith: I notice from your report that the fuel refabrication facility for EBR-II is now ceasing to make fuel. I wonder whether you feel this has now demonstrated the feasibility of this type of reprocessing or whether you are losing interest in rapid pyrometallurgical reprocessing because you are interested in oxide fuel or are you still looking to that sort of reprocessing for oxide fuel?
Wenschi: The fuel cycle facility is not being used for reprocessing of fuel right now because it is mainly used as a hot core facility for the examination of the irradiations coming out of the EBR-II.

The pirometallurgical method of reprocessing of metallic fuel has been successful. The quality of reprocessing has been of a fairly uniform standard.

Work at ANL is under way to fully develop the process using the fluoride volatility technique. GE is also giving some consideration to the fluoride volatility refining of hot fuels in the construction of one of their fuel element plants.

Kazachkovsky: I remember that when the first experiments on pirometallurgical methods were started at ANL, there were large losses of fuel in this process. Have you managed to decrease these losses and, if so, to what extent?

Wenschi: I do not know the details of these losses. The yields in reprocessing have been of the order of 90%.

Kazachkovsky: As concerns the chlorine corrosion, I know that American purity standards for sodium provide a certain maximum permissible content of chlorine in sodium. Are there any experimental proofs of this permissible amount? Has it been proven that if the chlorine content in sodium is more than that provided by the standards, a certain danger for steel may appear?

Wenschi: It was not chlorine in the sodium which caused the problems. We believe that it was chlorine ions in water because it was the water side in some experiments that failed.

Kazachkovsky: How do you establish the standards for the chlorine content in sodium? Is this just a content you get in processing sodium or are you concerned that a larger amount of chlorine in sodium may be dangerous for steel?

Simmons: Principally calcium limitation is a main concern. We usually limit the chlorine, as a precautionary measure, to be sure that we do not get excessive amounts. There is no evidence of stress corrosion, or cracking from chlorine in sodium.
Kazachkovsky: Have you considered the possibility of using bellows to compensate thermal deformation in main circuits?

Wensch: We do use bellows to compensate where we can the deflections of piping due to thermal expansion, particularly in control rod mechanisms, in valves. For large piping we would prefer not to use bellows. Thermal expansions can be accommodated in this case by using some expansion loops.

Kazachkovsky: Dr. Wensch's opinion was that from the point of view of the influence of irradiation upon the construction materials it is more preferable to use molybdenum steel, such as 316. Did you mean the employment of steel for fuel element cladding, or for reactor vessel, or for some structures inside the vessel?

Wensch: I addressed myself really to the FFTF when I said 316, and this is for cladding of fuel elements. There is no large difference between 316 and 304. The 316 apparently has a little better fit strength at high temperature. 304 is a more common steel.

Kazachkovsky: You mentioned that you use an intermediate layer of sodium or helium in fuel pins for improving heat transfer. Do you consider it possible to dispense with such an intermediate layer at all? Probably you could start with air inside your fuel pins.

Wensch: For oxide fuel, helium is quite appropriate. With sodium there are some reactions, depending on the oxygen to metal ratio. With air, the linear power rating would tend to go down.

Spinrad: Air inside fuel elements is strongly to be suppressed. It would make the uranium hypostoichiometric, which is not good.

Kazachkovsky: There have been experiments at ANL on metallic fuel: U-Pu-Ti or U-Pu-Zr in stainless steel cladding. We used to believe that such fuel elements could withstand temperatures on the cladding of up to 600°C. Are there any new results in this direction?
Spinrad: These experiments were carried out in order to see whether it was reasonable to try to develop such things as vanadium-alloy as a cladding material; in fact, there are some good irradiation results which one can see in progress reports. I believe the vanadium-tantalum alloy clad irradiating metal samples achieved 2.5% burn up with very good condition of the tubes. Of course the metal inside looks terrible. I recall that the temperature in experiments did not exceed 600 - 650°C.

Wensch: I can add that there is a metallic alloy programme under way at ANL with the aim of developing and improving fuel for the EBR-II because the limitations in burn ups have been limited so far to 1.2%. The design of fuel elements is also changed to allow more room for expansion. They expect to be able to achieve somewhat greater burn up.

Vautrey: In one of your slides there was a figure of 150000 MWd/T for the burn up. What are your hopes to achieve such a burn up?

Wensch: This is just the maximum rather than the average figure. The average burn up for that study was something like 86000 MWd/T.

Vautrey: I agree that this is a maximum figure. But it means that there are some pins in the reactor which can reach the value of 150000 MWd/T. Did you put in this figure just for studies and calculations or do you have good reason to believe that this figure has a real significance?

Wensch: That figure is an objective only. That is not a real figure.

Vautrey: You mentioned codes and in particular for pumps, for the IHX as having been edited at the end of 1968. I would like to ask you whether these codes are available.

Wensch: There is a difference between specifications and codes. In the USAEC we are establishing standards for vessels, for purity of sodium and things like that. These standards, if accepted by the USA or by the professional societies, will then have the force of law and will then become codes. Most of our codes are available from the various societies and the A.E.C.
Vautrey: You said that in the FFTF the flux of $10^{16}$ n/cm$^2$ sec could not be achieved because of the small core. Then you indicated that in this facility the neutron flux would double that of any existing fast reactor. What exactly is the flux that you are expecting to be achieved in the FFTF.

Wensch: The objective was to achieve $10^{16}$ n/cm$^2$ sec for a fairly large length. But really the flux will be around $7.3 \times 10^{15}$ n/cm$^2$ sec.

Vautrey: Did I understand it correctly that FFTF will be equipped with steam generators rather than air heat exchangers?

Wensch: The present plan is to use air heat exchangers with variable speed blowers. Some consideration has been given to steam generators but it seems unlikely that we shall use them.

Schuster: Could you give us the figure for the total estimated costs for the US sodium breeder programme, including FFTF, INE and three demonstration plants? What is the estimated amount for the construction of the first demonstration plant? And what means modest A.E.C. assistance in 1000 MW(e) plants?

Wensch: The estimates for demonstration plants have been conducted by the private groups but they are not detailed, very preliminary ones.

The A.E.C. wants to get for this programme $18$ million for research and development, for assistance in fuel use charges which are quite high for a private company, in some components and in architect engineer work.

Kuramoto: I wonder whether your safety criteria will be applied to the first demonstration plant?

Wensch: Complete criteria for FBR does not exist. Applicable codes must be used or even procedures followed which are more stringent than the applicable codes.

There are some uncertainties which could cause in problems in obtaining licensing.
SURVEY OF FAST REACTOR WORK IN THE SOVIET UNION (1968-69)

O.D. Kazachkovsky

I intend to summarize the results achieved from the work that has been done on fast-neutron reactors during 1968.

First let me make a few general remarks concerning the fast reactor philosophy which has gained general acceptance in my country.

Last year we continued working on the main problems presented by fast reactors, in particular sodium-cooled reactors using solid—principally ceramic—fuel. At the same time a small amount of work is in progress on fast gas-cooled reactors, though this work is still in the exploratory stage, devoted to laboratory calculations.

The main tasks we set ourselves in 1968 were the same as in previous years. Basically they comprise two problems:

(1) To demonstrate the technical feasibility of constructing and operating commercial fast reactors of size comparable with existing power reactors using thermal neutrons;

(2) To determine the optimum technical parameters of fast reactors in relation to subsequent large-scale plants. By "optimum" I mean "optimum from the economic point of view". This refers primarily to the working temperature of the coolant, the degree of burn-up and the thermal stress on the fuel elements.

The first problem is being solved by construction of the BN-350 reactor. Our work along these lines will be expanded in future with the construction of the BN-600 reactor. Fast-reactor experts are already convinced that such systems offer sufficient reliability and excellent operating properties. We are now faced with the difficult task of securing acceptance of this fact by those who will be responsible for running nuclear power stations with fast-neutron reactors, i.e., the power engineers.
Another related objective is to obtain working experience with large-scale commercial fast-neutron systems, in particular experience in the operation of sodium circuits. It is of prime importance to make a sufficiently comprehensive study of the operating conditions of equipment working in radioactive sodium. It is also important to determine which of the two possible types of fast reactor designs should be regarded as preferable, the integrated or the non-integrated. In the Soviet Union both types have their supporters and their opponents. The BN-350 reactor is, of course, based on the non-integrated type of design whereas the BN-600 will be integrated.

The technological equipment for the BN-350 is at present being assembled on the construction site at Shevchenko, whereas construction of the BN-600 is still at the stage of earth-removal operations.

The second problem, that of determining optimum parameters, is being solved by means of test-bench and research assemblies. An important stage in this respect was when "dry" criticality was attained in the BR-60 reactor. Construction of this reactor was begun in Melekess in April 1966. The reactor is fuelled with enriched uranium oxide. "Dry" physical start-up of the reactor was successfully accomplished by the close of 1968, and work is now in progress on preparing the circuits for filling with sodium. The maximum coolant temperature for which the BR-60 reactor is designed is 650°C and the arrangements for compensating thermal stress were determined on the basis of this temperature, though that does not mean that the reactor will operate at such a high temperature.

In view of the possibility of high-temperature operation, close attention was also paid to the risk of the steel cracking in the area of welded joints. The steel selected was chrome-nickel steel without stabilization, since a series of experiments had shown that stabilization with titanium, which is normally used in my country for these purposes, causes an appreciable deterioration, by almost two orders of magnitude, in the strength of the area surrounding the joint at high temperatures. Steps were taken to ensure that all welded joints which will be subjected to high temperatures were austenitized after assembly.

The BR-5 reactor, situated at Obninsk, last year continued operating successfully with its second loading, using uranium monocarbide. A burn-up of about 3.5% was achieved.
A sodium loop was also put in service in the MIR reactor at Melekess. This is a reactor for testing experimental fuel elements of various types with beryllium moderator and aqueous coolant. The maximum power level of the sodium loop is 1 MW. Naturally there were dangers involved in introducing a sodium loop into a water reactor, and the start-up and loading operations were therefore carried out slowly and carefully. The loop is now in the initial stage of operation, though there are of course difficulties still to be overcome in regard to reloading a sodium loop surrounded by water.

Hot-cell research on experimental fuel elements for fast reactors was continued. The fuel elements were irradiated in the BR-5 reactor and the water loops of the SM-2 reactor. In practice the work was devoted almost entirely to oxide fuels. The results obtained bear out the possibility of obtaining in ceramic oxide fuels assured burn-ups of the order of 10%, i.e., 100,000 MWd/t. In our view, no fine-precision accuracy is required in fabricating these elements. In particular, if pellets of sintered material are used, relatively large interstices between fuel and cladding can be tolerated.

Work was done on the principles governing the development of structure in fuels formed from vibro-impacted powder made of uranium oxide. The optimum schedule was determined for raising a reactor using such fuels to rated power.

Close attention was paid to a study of the effect of neutron irradiation on structural materials. It is confirmed that there is some danger of a high-temperature embrittlement effect. Fig. 1 shows the results of research carried out on steel from the can of a bundle taken from the BR-5 reactor. The integral neutron flux was $2.5 \times 10^{22}$ neutrons/cm². The left-hand graph shows the temperatures at which the steel samples cut from the can of the bundle were tested. The following three graphs are parameters relating to the unirradiated steel. On the right are the same parameters after irradiation. The three graphs show successively the elasticity limit, the strength and the relative elongation. If the relative elongation before and after irradiation are compared, the marked difference between them can be seen. Admittedly our metallurgists believe that this difference can be explained, at all events in part, not by the irradiation, but simply by the conditions in which the steel finds itself at high temperature and under
some stress. In their view, these conditions may of themselves be expected to cause deterioration in the steel's properties. It must be said that the radiation effect increases as the nickel content of the steel increases and decreases if the steel is alloyed. In this sense a good component for alloying is molybdenum.

Fig. 2 gives the results of tests carried out on steel with a high nickel content. The temperature at which tests were carried out is plotted along the axis of abscissa, the relative elongation along the axis of ordinates. The upper curve relates to unirradiated steel, the lower curve to steel exposed to a flux of the order of $2 \times 10^{20}$ fast neutrons per cm$^2$. As can be seen the two curves first diverge in the region of 500°C.

Fig. 3 gives the results of tests carried out on a similar steel with an alloy composed mainly of molybdenum. The broken curve relates to the irradiated steel. As can be seen, in this case the two curves first diverge at a temperature above 600°C.

1968 saw an improvement in the conditions for carrying out experimental work on reactor physics in the Soviet Union. In particular, an accelerator was put in service in conjunction with the BFS reactor at Obninsk in order that neutron spectra might be investigated on this critical assembly, using the time-of-flight method. Installation of a larger critical assembly, which will make it possible to carry out research in connection with large commercial fast reactors, is at present nearing completion.

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Fig. 1 Mechanical properties of material comprising the can of the bundle at various temperatures.
Fig. 2

Fig. 3.
Vautrey: Could you give some clarification on the sodium loop at the reactor MIR in Melksee and on its utilisation?

Kazachkovsky: The maximum power of this loop is 1 MW. The maximum temperature which is calculated for this loop is something like 600 - 620°C.

We are doing work on testing fuel elements for fast reactors, and for this we use, or shall use, three reactors in our country: SM-2, MIR and BR-60. On SM-2 we are carrying out tests of small samples of fuel elements in capsules. That is without sodium flow. Once we have determined on SM-2 which fuel elements are essential, and which fuel elements it is necessary to further test, then we will go on to the next stage, that is, the testing of the full-scale fuel elements. This will be carried out on the reactor MIR. Here we are able to test fuel elements which are one metre long, which corresponds to the length of fuel elements in large fast reactors. It is possible also to test subassemblies of fuel elements. Certainly there are difficulties with respect to the depression of neutron flux. Nonetheless, these experiments might well give very useful results. We are thinking of testing fuel elements here in groups under conditions which are very close to the conditions which will exist in big fast reactors. After this we will be able to test shorter fuel elements in fast neutron flux in the reactor BR-60.

Engelmann: Do you use boron or cadmium filters around your loop in the MIR reactor in order to harden the neutron spectrum and to reduce the flux depression inside the subassemblies?

Kazachkovsky: This is true. We are forced to use absorbers in order to make the neutron flux even around the channel because this channel is not in the centre of the reactor. The gradient of the neutron flux is already by itself capable of leading to some rather unfortunate consequences which are connected with the bending of fuel elements. We are decreasing the thermal
neutron flux using an absorber. It is possible to use blocked cadmium or boron which should, of course, be changed during the campaign.

Smith: Could you say what sort of fuel element you are thinking of for a gas cooled fast reactor at your early stage of consideration?

Kazachkovsky: In Melekes we are not going to have any work on gas cooled fast reactors. This work is being carried out in Kurchatov Atomic Energy Institute by the group under Dr. Feinberg. This work has not reached the stage where it is necessary to choose fuel elements for this reactor.

Schuster: What is the present stage of development of the reactor BR-60?

Kazachkovsky: The reactor BR-60 has achieved dry criticality. This was at the end of last year. The situation at present is as follows. The core has been disassembled. The heating system is being mounted in the primary loop. On the secondary loop all the mounting and welding has actually been completed, including air heat exchangers. Work is being carried out on the auxiliary systems. Perhaps next month we will be able to begin the filling up with sodium in certain parts of the first and second loops.

As regards the fuel we shall in our first load use enriched dioxide of uranium for the time being, even without plutonium.

Schuster: What is the real size of steam generators for BN-350 and when do you expect this reactor to go into operation?

Kazachkovsky: On the BN-350 we have six separate loops. One of these is a spare loop, so only five will operate simultaneously. The thermal power of the reactor is going to be something like 1000 MW. I do not remember the data on steam generators, but I think these parameters have been published in the Proceedings of the Obninsk Conference held at the end of 1967.

As regards the time when the reactor will be put into operation, the construction is somewhat delayed. Our task now and the task facing our industry is to finish principal works in 1970.
Schuster: Are there any assessments of the electricity cost from the large power fast reactors in your country?

Kazachkovsky: The results of calculations are very preliminary with respect to the cost of the electrical power on fast reactors. I think that for reactors of 1000 MW(e), it is something like 0.5 kopeks per kWh but I am not sure of this.

There is reliable evidence that the fuel elements with uranium carbide in the BR-5 do leak. There is quite noticeable activity of delayed neutrons here. Changes of this activity in recent times have not been great, but I am rather hesitant to speak with great conviction. This load is not to be changed this time.

Wensch: What is your philosophy for operating BN-350 or BR-60 with failed fuel elements?

Kazachkovsky: The philosophy is as follows. We are not preparing to pull out elements if we find evidence that fuel elements are leaking. We do not have a special system which would help us to identify in what spot this change is taking place. Naturally we are aware that there will be failures of fuel elements and we will continue to work with them.

In BN-350 we are going to continue the operation if some fuel elements fail.

In BR-60 we are determined to carry out research with vented fuel.

The reactor BR-60 has research loops where we are going to study migration of radioactivity along the circuit, the possibility of retention of this radioactivity with cold traps, the possibility of removing the radioactivity from the walls of the circuits by means of actual sodium, and cleaning of argone in the gas pillow from radioactive fragments, and so forth.

Smith: Do you intend to use oxide or carbide in your vented fuel?
Kazachkovsky: First of all we are thinking of working in the direction of oxide fuels.

Wensch: I would imagine that the sodium void effect in the reactor BN-600 is going to be positive. Is it so?

Kazachkovsky: I cannot quite recall what is the sodium coefficient here. But I would like to draw your attention to the fact that on BN-600 a ratio of diameter to the height of the core will be a factor of 3. And therefore the sodium coefficient might be slightly negative on or around zero.

Smith: Where is BN-600 being built and when do you expect this reactor to be on power?

Kazachkovsky: It will be built in the Urals. The date, I think, will depend on our successes or failures with the reactor BN-350.

Wensch: What has been the worst problem encountered in the building of BN-350?

Kazachkovsky: As far as I can recall there was a lot of discussion about the pumps for this reactor. In BR-60 and BR-600 the pumps will have low hydrostatic bearing and the upper bearing will be of the usual sort. In BN-350 we are using console type pumps. This means that in this reactor there is no bearing which will work in sodium. Both bearings will work above the working wheel. In this case the possibility of change of the sodium level within the body of the pump should be very low. On the reactor BR-5 we had problems that concerned the transient regimes. There have been very quick changes of the sodium level in the pump.

We were not able to envisage hydrostatically bearing pumps for BN-350 when it was designed. Therefore we had to have more complex circuits.

There is also the very difficult question of the steam generator. I think we cannot boast that we have already solved this problem. We are still working on studying the interaction of sodium with water and steam. Last time I mentioned that steam which goes into the sodium causes a more unpleasant con-
sequence than water.

We foresee the possibility of testing different steam generators on BR-60.

There were other problems such as choosing the fuel. We chose for the first load uranium dioxide although formerly speaking we have the least amount of experience on that type of fuel. But we do not expect any surprises here. We think this is the simplest and best tried out fuel.
In this short paper I shall try to describe for you some of the results obtained in France between March 1968 and March 1969.

I should like to mention straight away two important points about which I shall go into more detail later on; these are:
- the excellent operation of the Hapsodie reactor; and
- the beginning of construction on the Phénix prototype.

For Hapsodie the past twelve months have been very profitable. Let me first quote a few figures on availability and load factors. To avoid any misunderstanding I shall recall the definitions which we give to these two terms. The availability factor is the time during which the reactor is actually capable of operating, i.e. total possible operating time free of incidents and defects, divided by total time minus fuel handling time:

\[
\text{Annual availability factor} = \frac{\text{Possible operating time}}{365 \text{ days minus fuel handling time}}
\]

The load factor is actual operating time (expressed as the number of days at full power) divided by total time:

\[
\text{Annual load factor} = \frac{\text{Equivalent days at full power}}{365 \text{ days}}
\]

Given these definitions, we have obtained the following results:

For 1968
- Availability factor 90.1%
- Load factor

Since 1 January 1969
- Availability factor 100% (except for one week at 65% and another at 94%)
- Load factor (up to 2 March) 76%
blies exposed to equivalent or even greater irradiation showed no abnormality. After considering various possible explanations, we thought that there must have been a local blockage in some particular channels of this assembly (between pins) which led to localized overheating and detachment of the cladding. This detachment must itself have accentuated the effect and spread it somewhat in the downstream direction as well as upstream. We assume, although this is only a guess, that this momentary blockage might have been caused by undissolved oxide fragments, originating perhaps in the deposits which accumulate at the top of the upper argon-filled spaces — on the shielding plugs of the reactor of the pumps or of the heat exchangers. It is encouraging to find that all the pins remained intact, since they all contained fission gases at a pressure of 80 – 100 bars (at operating temperature).

With regard to the fuel assemblies as a whole, we decided last January to limit their time in the reactor, as a general rule, to a burn-up not exceeding 50 000 MWD/tonne; here let me recall again that the burn-up originally scheduled was 30 000 MWD/tonne.

At the beginning of February, about a day before the normal reactor shut-down for unloading of the first assembly that was coming up to 50 000 MWD/tonne, the delayed neutron detection system in the sodium gave a signal — rather a weak signal — for the first time. To give you an idea of what this amounted to, the normal reading due to background had been about 30 counts/second; the signal rose slowly to 40 counts/second and then more rapidly to 55 counts/second. As this was not an alarming reading, normal operation was continued until the burn-up of 500 MWD/tonne had been obtained. Since the reactor was started up again in the middle of February the burst slug detection system in the sodium has given a stable signal of about 50 counts/second, which seems to indicate that this phenomenon is not developing any further. We hope to identify the faulty assembly during the next shut-down, which is to come in the very near future. I might add that sodium samples have revealed some iodine and tellurium, though in very
small quantities (e.g., a few times $10^{-3} \mu$Ci/g for $^{131}$I). No trace of plutonium has been found. This incident has therefore done nothing to hamper the running of the reactor. Soon we hope to have the opportunity of examining a failed can, a study which we are eagerly awaiting because we expect it to be highly instructive.

As regards incidents which have occurred in the reactor since March 1968, the figures quoted for availability suffice to show that they have been rare and not particularly serious. On this point I shall therefore be very brief.

There have been only two incidents significant enough to be mentioned; from 5 to 9 March 1968 the reactor had to be shut down because of a blockage in the argon piping caused by sodium deposits. Partial blockage had shown up earlier but had been eliminated simply by heating the pipes. This time, however, in order to clean the system more thoroughly, we resorted to the method envisaged at the design stage: the level of the sodium was raised so that it flowed into the argon pipes and washed them out. We succeeded in removing the blockage in this way, but not completely. Accordingly, during the annual shut-down which took place between 25 September and 25 November, we modified the argon circuit with a view to making this cleaning process simpler and more efficient.

The second incident which forced a reactor stoppage took place in July. On this occasion Rapsodie was shut down for about two weeks because of an incident involving the control rod mechanism: the gripper device which grasps the top of the control rod had become deformed. The mechanism was accordingly replaced by a spare one.
The power coefficient of the reactor is measured regularly so that we can follow its evolution. As anticipated, the coefficient decreases slowly; its present value is about $6 \times 10^{-5}/$MW. The overall evolution of the power coefficient is fairly well understood, although we cannot claim to be able to explain all the occasional fluctuations.

The principal role of Rapsodie is still that of an installation for testing fuel; and, more generally, it can be described as an irradiation reactor. The performance of the reactor's fuel, of which I have already spoken, is still a point of primary interest because the fuel used (UO$_2$-PuO$_2$ mixed oxide) is very similar to that likely to be used in future reactors.

However, it should be borne in mind that Rapsodie is at present being used for 42 irradiation experiments. Among the tests now under way, the most interesting are undoubtedly those involving 12 experimental fuel assemblies designed as prototypes of the assemblies to be used in the Phénix reactor. Two of these had reached 56,700 MWd/tonne by 14 March 1969.

Another matter to which we are giving our attention at the moment is the detection and location of cladding failures in the assemblies. This is, of course, an important problem, and here too we hope to get from Rapsodie results which will be of value for the Phénix reactor. Experimental work on this problem in Rapsodie is something quite recent, needless to say, because the first true cladding failure happened only a month ago. We are accordingly pleased that this incident occurred, because it gives us an opportunity to improve our understanding and our experience of the problem.

To close my account of Rapsodie, I should like to say a few words about the future. If all continues to go well we can hope to have accumulated, by the end of 1969, statistical material which will be valid for burn-ups between 50,000 and 100,000 MWd/tonne. On the other hand, a point about which there will
continue to be doubts affecting the fuel pins of future reactors is the performance of the cladding under high integrated fluxes (exceeding $10^{23}$ n/cm$^2$). The flux now available in Rapsodie is too low to permit testing of the cladding under integrated fluxes of this order. For this reason we are planning to modify the characteristics of the reactor in order to increase the maximum flux from $2 \times 10^{15}$ to about $3 \times 10^{15}$ n/cm$^2$ sec. For this purpose the power will be raised from 24 to about 40 MW, and the core assemblies, while retaining the same external geometry, will contain 61 pins of 4.1 mm diameter instead of 37 pins of 5.6 mm diameter. Modifications to the pumps and heat exchangers will be comparatively minor. This new version of Rapsodie, to be called Rapsodie Fortissimo, may be in operation in about a year's time.

As far as Phénix is concerned, studies on what we have called the preliminary plan were finished a year ago. At that time all the essential choices, many of them involving detailed specifications, had already been made. Since then work on the designs and plans has gone forward in close co-operation with the industrial firms concerned, with a view to completing the final construction drawings.

This work has been undertaken by what we call an "integrated team", consisting of members of the Commissariat à l'Énergie Atomique (CEA), Electricité de France (EDF) and the industrial architect appointed for the project, which is the GAAA Company. This team, led by a CEA engineer, has the task of co-ordinating all the construction studies and administering the supply contracts, up to and including the final tests on the plant as a whole.

Parallel to these studies and to the construction work, a very considerable experimental programme is getting under way.

For example, a number of tests have been undertaken in connection with the strength of the top and base of the main reactor
vessel, sodium hydraulics, the efficiency of the baffles and heat insulation, and the likely behaviour of the main vessel in the event of an accident involving an explosion. These studies and tests are to be continued in 1969, but by the beginning of the year they had already given us enough information to quote specifications for the long-term orders, in particular for the primary vessel, the main vessel and the double envelope. The constructor has already been selected and the requisite supplies are being obtained.

As far as fuel handling is concerned, the main effort has gone into designing and constructing the prototypes of the loading arm and ramp. These two units have been set up and tested in air. In 1969 they are to be tested in hot argon and afterwards in sodium, under actual reactor conditions.

Planning and design work on the piping for the loops has advanced to the point where consultations can take place during the first half of 1969.

The prototype of the steam generator has been run successfully for more than 4,000 hours at our test installation at Grand Quevilly and very soon we are going to begin construction on the 36 moduli of the Phénix steam generators. Let me add, by the way, that the 50 MW steam generator testing installation designed by EDF in collaboration with CEA is now under construction and should go into operation in about a year.

The building site for the Phénix reactor was opened at Marcoule in the latter half of 1968. The principal contracts have already been awarded to industry, for the main civil engineering work, the essential parts of the reactor block, the primary and secondary sodium pumps and the arm and ramp for fuel handling. In 1969 all the long-term contracts will be placed, and almost all commitments for capital expenditure will have been made by the

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* The arm in question is the device which transfers fuel from the core to the internal storage. The ramp is a chute by which fuel assemblies are loaded or unloaded in the reactor vessel.
end of 1970. The goal at present is to put sodium in the Phénix reactor in a little over three years' time.

Let me in closing say a few words about fuel reprocessing. Our AT-1 pilot installation, situated at the La Hague Centre, was designed to handle irradiated fuel from Rapsodie; 220 fuel pins, i.e. a little more than 6 assemblies, were recently processed there (this is wet reprocessing). The operation went off without a hitch and the decontamination factor obtained was higher than $6 \times 10^7$.

I should add, finally, that our work on dry reprocessing has given very encouraging results at the laboratory level. A decontamination factor of $10^7$ has been obtained.
Discussion of the Presentation by Mr. Vautrey

Kazachkovsky: You have got interesting results with fuel elements which had no inside gas pressure, so these were actually vented fuel elements. Have you found some traces of sodium inside these elements?

Vautrey: In no case did we find any traces of sodium inside element cladding.

Kazachkovsky: A question on the fuel element which differed from the others. You supposed that plugging had taken place as a result of some elements which had not been sufficiently cooled. Is it possible to derive a metallographic picture from the temperature of cladding? I mean, if the temperature was above 550 - 600°C, the grain sizes increases due to the depositing of carbides. Have you noticed any changes on the surface of the steel cladding?

Vautrey: It seems evident that there was very localised plugging inside the subassembly. There was no plugging at the entrance of the bundle. We have not yet obtained indications for the temperature of the cladding, according to metallographic analysis. In examining the fuel pins we have found that in the centre of the fuel the temperature was about 2500°C, which means a cladding temperature about 200°C above the normal temperature.

Kazachkovsky: Do you have any indications regarding the plutonium behaviour in dry reprocessing of fuel; I mean a fluoride processing. This method was established to be good enough for uranium but with plutonium some difficulties might arise due to decomposition of Pu F₆ and due to the fact that plutonium fluoride is far less well separated from fission products.

Vautrey: I do not have sufficient information for this question. This was the first test we made with this fluorine reprocessing method, and this first test was made with uranium fuel.
Smith: A very important factor in the amount of plutonium one actually produces is the percentage which is lost in the reprocessing of spent fuel. For all our studies in the UK we take 2% loss per cycle in reprocessing. We are not sure quite how realistic this is. On one hand reprocessing people consider this figure a very small loss but on the other in the big fast reactor programme 2% is a very large amount of plutonium. I would like to ask whether you think that 2% is a reasonable figure to choose. I would also like to ask the other members of the Group what their expectations are of plutonium loss during reprocessing of fast reactor fuel.

Vautrey: Unfortunately I do not think that I can say anything new. As I said, the small experiment on reprocessing is quite a new one. The first reprocessing of spent Rapsodie fuel was carried out by aqueous means. The losses were less than 1%. In our economical studies we have estimated that losses in reprocessing would be of the same order of magnitude, i.e. about 1%.

Kazachkovsky: The maximum permissible losses in reprocessing which we fix are 2%. But we have no experimental confirmation with which we can guarantee this figure.

Wenschi: We assume 2% losses in aqueous reprocessing in the U.S. This is based pretty much on actual experience. However, dry reprocessing methods should have a potential to decrease these losses.

Can you tell us about your experience in steam generators and what materials do you use in each of the units?

Vautrey: I have nothing new to say about steam generators. A steam generator envisaged for "Phénix" was described in various publications. We employ a module system so that an accident would not lead to large consequences for the whole plant. A module for Phénix has a power of about 15 MW.

We are going to use in evaporators a ferritic-chromium-molybdenum steel in superheaters — austenitic steel 321.
The first prototype steam generator, including evaporator, which has been tested was made of steel 321. This prototype was shut down after 7000 hours of operation, because it was necessary to start testing a Phénix prototype. No traces were found of corrosion, but water was carefully treated.
I should like to give you a brief account of a recent ENEA initiative in the field of alternative coolants to sodium for fast reactors, which is being undertaken in co-operation with Foratom, and to explain to you some of the thinking behind it.

More than a year ago, considerable concern was being expressed in a number of our Member countries that, while a great amount of European effort had been concentrated on sodium cooling, the potential of alternative coolants had received only preliminary assessment. The argument put forward was that sodium-cooled fast reactors are still some way short of being commercially available, and a substantial amount of proving would be needed before sodium could be confirmed as the preferred solution to the fast reactor coolant problem. This would take time, and meanwhile it was desirable that the potential of steam and gas as alternative coolants should be assessed. Commercial interests in the United States had reached similar conclusions, and were already investigating alternative fast reactor coolants with the same objectives.

The scale of expenditure required for such assessments would discourage countries in Western Europe from embarking on national programmes, especially if these were to compete for resources with work already under way on sodium cooling. Development of alternative coolants for fast reactors therefore appeared a particularly suitable matter for international co-operation, involving governmental as well as industrial interests.
As a first step, two specialist teams were set up to evaluate the merits of steam and gas cooling relative to sodium. The technical material upon which their studies were based was contributed in the form of national studies from Belgium, Germany, Sweden, Switzerland (including GGA Europe) and the United Kingdom. In addition, the draft findings of a similar exercise by the USAEC in collaboration with American industry were made available.

Information for sodium-cooled fast reactors which was used as the basis for an attempted comparison with GCFR and SCFR designs was derived from data published at the Third Foratom Congress on "Industrial Aspects of a Fast Breeder Reactor Programme", London, 24th-26th April 1967.

However, no attempt was made to carry out a detailed appraisal or survey of sodium-cooled fast reactor systems, and opinions were divided upon the figures which should be quoted to reflect the current status of technological achievement.

Furthermore, it was felt that any range of figures which could be agreed would still not provide a true basis of comparison with the corresponding findings of the two specialist teams for gas and steam cooled fast reactors, firstly because the recent changes in the value ascribed to alpha for plutonium have required a certain degree of re-optimisation of all fast reactor designs. These changes were taken into account in the re-optimised steam-cooled cases, while it was only possible to re-calculate the effects of the changes in alpha data for typical examples of the various gas-cooled alternatives. Since the results of corresponding studies on sodium systems were not made available, it was not possible to include an assessment of the effect of these changes in the technical evaluation.
Secondly, at the present time, there are large differences in the relative state of development of the three concepts. Current sodium-cooled performance figures are based upon achievements anticipated for the year 1980, whereas in the post-1980 period when the gas and steam cooled reactors would be offering potential competition, it can be expected that further experience and improvements (e.g., development of vented fuels and carbides) would result in reductions.

Classical ground rules were adopted for the economic evaluation, which as may be expected, turned out to be entirely inconclusive due to the conjunction of all the uncertainties involved.

The technical evaluation can be briefly summed up as follows:

It was confirmed that for gas and sodium cooling, lower doubling times are achievable than for steam cooling, and that the inventory of sodium-cooled systems is smaller than for steam and gas cooled systems with metal cladding and oxide fuel. A certain decrease of inventory is possible for the gas-cooled systems with coated particle fuel, with some penalty on doubling time on the basis of the new Pu data due to the softer neutron spectrum. Both sodium and gas systems would have reduced inventories from the use of carbide fuel, which of course, is not applicable to steam.

Fuel was identified as being the central development priority for gas and steam cooled fast reactors. A number of options were studied, which included alternative nickel-based alloys for the SCR, prepressurised to withstand the force of the outer coolant pressure, also strong clad and weak clad (vented)
fuels for the GCR as well as coated particle fuel designs.

One of the chief problems at this stage is to narrow the choice amongst these alternatives in order to minimise costly and time-consuming multiple development work. On balance, gas-cooled fast reactors offer a larger development potential than steam-cooled systems either by employing the direct cycle plant or carbide fuel.

Although the GCiFR metal-clad oxide-fuelled design has a fuel inventory comparable with that of the SCFR (i.e. about 4 g/kWe) the coated-particle design has an inventory similar to that of the 1980 sodium case (about 3.5 g/kWe), while the carbide fuel concept may achieve system inventories in the region of 3.0 g/kWe. The GCFR can in this sense have the potential to compete with developed sodium designs and so has a wider timescale of application than the SCFR.

Agreement was unanimous upon the necessity of a large fuel test facility which might also be conceived as a prototype power station.

As regards Safety, a good deal of faith is placed in the integrity of prestressed concrete pressure vessels for gas-cooled designs, and in the case of the steam-cooled reactor, upon the incredibility of a major rupture in a steel vessel, already accepted for thermal neutron water-cooled reactors. It is recognised that the principal bogey, the loss-of-coolant accident can only be effectively remedied by engineered safeguards with a high non-failure probability. Much more work is required also in this field.
In order to get something started, the first priority will be to eliminate some of the options, so as to streamline the development programme in the right direction and make it more manageable. Discontinuation of the work on steam-cooling for fast reactors, firstly in the U.S.A. and more recently in Germany means that further development of this variant is unlikely to be followed up within the framework of ENEA. Nevertheless, in order to establish a sound basis for future decisions on gas-cooling for fast reactors, the experts estimate that a good deal of experimental results are needed in respect of three types of fuel, strong clad, vented, and coated particle, and on two types of coolant, CO₂ and helium.

Efforts are now in hand to attempt a voluntary distribution of this work among the industrial and governmental establishments of interested countries within the framework of an ENEA Development Syndicate.

Obviously successful commercial development of the HTGCR will be a significant factor in deciding the longer-term emphasis which will be placed upon gas-cooling for fast reactors. This combined with the degree of success which can be achieved in the modest co-operative programme which we hope to start in the summer, means that the next two years are likely to be crucial in deciding the future of the GCFR in Western Europe.
Wensch: There has already been much experience gained from sodium-cooled fast reactors and one can say that all the accidents in these reactors have not been due to the sodium but simply due to some mechanical failure. I wonder if you could really focus the central problem with the sodium-cooled fast reactors as you see it?

Boxer: I can only reflect the views of the experts who participated in our working party. There were two schools of thought and the study group was fairly evenly divided among these schools. There were people who believed that it was desirable to have a back-up solution in the event of failure with sodium without necessarily specifying whether such failures might come. There were others, and this reflected more the industrial point of view, who felt that irrespective of the merits of sodium, there was plenty of justification for pursuing an alternative coolant on the basis that the future market is likely to be large enough to be able to accommodate more than one fast reactor type in the same way as it now does in the thermal reactor market.

Some figures were produced which pre-supposed that if, in a decade after 1980, about 300000 MW of nuclear capacity are likely to be built in Western Europe, it is reasonable to assume that about one third of this would be allocated to fast reactors, which implies a construction programme of something like about ten fast reactors a year.

The question was raised whether at present it would be wise to allow a potential market of this magnitude to rely upon a single reactor concept, the engineering of which is still in a transition stage.

Secondly, if an alternative coolant fast reactor absorbs only 20% of this postulated market, it would still be required to furnish a capacity of the order of 20000 MW which the protagonists of this school thought was ample justification for spending developmental money to bring this reactor type to a full commercial stage.
Smith: Did I understand you correctly that there are some new developments in this field in connection with Foratom?

Boxer: This was the first time that we had a joint intergovernmental, interindustrial study group on a particular point. So it was new in this respect.