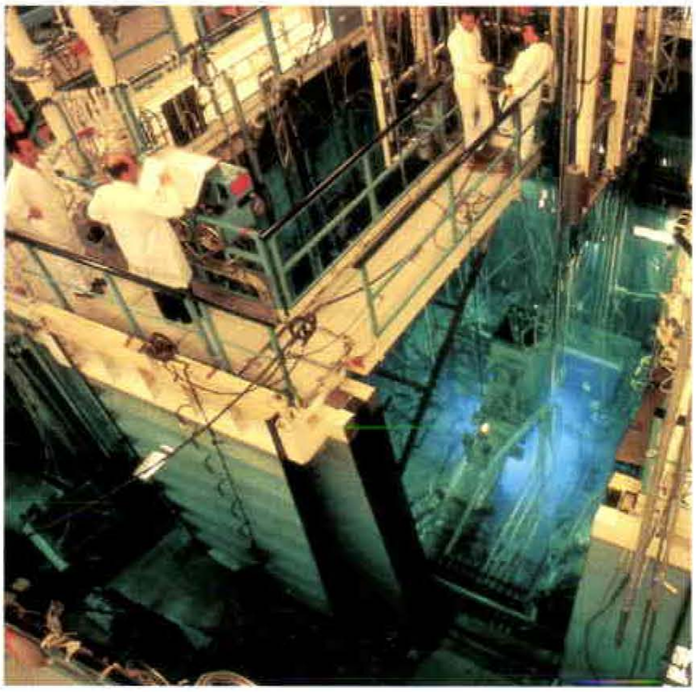
IGORR 2



2ND MEETING OF THE INTERNATIONAL GROUP ON RESEARCH REACTORS

18, 19 MAY 1992

SACLAY - FRANCE

COMMISSARIAT A L'ENERGIE ATOMIQUE

technicatome

International Group on Research Reactors
IGORR

Charter

The International group on Research Reactors was formed to facilitate the sharing of knowledge and experience among those institutions and individuals who are actively working to design, build, and promote new research reactors or to make significant upgrades to existing facilities.

IGORR Organizing Committee

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IGORR II

2nd Meeting of the International Group on Research Reactors
May 18-19, 1992
INSTN/SACLAY, France

* AGENDA *

MONDAY, MAY 18, 1992		SPEAKER	TIME
.	Registration (final)		8:45
.	Welcome	B. FARNOUX	9:15
.	Opening	J. BOUCHARD	9:25
.	Agenda of the meeting	C. D. WEST	9:45
.	Organization	C. DESANDRE	10:00
		Coffee break	10:20

SESSION I

Research Reactor Reports - Chairman A. AXMAN

1.	ILL	J. M. ASTRUC	10:50
2.	FRM-II	K. BONING	11:10
3.	HIFAR	J. EDWARDS	11:30
4.	PIK	Y. PETROV	11:50
5.	JAERI	S. MATSUURA	12:10
	Discussion		12:30
		Lunch	12:50

SESSION II		SPEAKER	TIME

Research Reactors (cont'd) - Chairman J. AHLF

6.	MAPLE	A. LEE	14:10
7.	ANS	C. D. WEST	14:30
8.	NIST	H. PRASK	14:50
9.	MURR	J. C. McKIBBEN	15:10
10.	TRIGA	J. GANLEY	15:30
11.	BR2	E. KOONEN	15:50
12.	SIRIUS 2	P. ROUSSELLE	16:10
	Discussion		16:30
	Adjourn		16:50
		Bus departure	17:00

CONFERENCE BANQUET

Bus departure from IBIS hôtel	19:00
River Boat Cruise	20:00

TUESDAY, MAY 19, 1992

SESSION III		SPEAKER	TIME
Other Neutron Sources - Chairman B. FARNOUX			
1.	SINQ	G. BAUER	9:00
2.	New European neutron source	A. TAYLOR	9:30
	Discussion		10:00
	Coffee break		10:20

SESSION IV

Workshop I - R&D Results and Needs - Chairman K. BONING
Sec. K. ROSENBALM

- Reports on progress in needed R&D areas identified at IGORR 1

1.	"Pulse Irradiation Test on Low) Enriched Silicide fuel Plates) in JAERI")	S. MATSUURA	10:40
2.	"The CNS Facility and Neutron) Guide) Tubes in JRR - 3M")		
3.	Silicide fuel tests & fabrication development, aluminium corrosion, thermal-hydraulic & fuel plate stability test results	C. D. WEST	11:10
4.	Fuel plate Development for FRM-II	Y. FANJAS	11:40
5.	Irradiation effects on aluminium and beryllium	M. BIETH	12:00
6.	Reactor Controls Research at MIT	KWAN KWOK	12:20
	Discussion		12:40
	Lunch		13:00

SESSION V

TIME

Workshop II - User & R&D Needs - Chairman J. HAYTER
Sec. K. ROSENBALM

.	Reactors and Physics Education	J. HAYTER	14:15
.	New R&D needs identified since IGORR-I		
.	New facility needs identified by users		
	Coffee break		15:45

Closing session 16:15

.	Adjourn	17:45
	Bus departure	18:15

WEDNESDAY, MAY 20, 1992

.	Visit ORPHEE and OSIRIS (passeport or identity card compulsory)	9:20
	Bus departure	11:30

OPENING

BY

JACQUES BOUCHARD
DIRECTOR
NUCLEAR REACTOR DIVISION

Ladies and Gentlemen,

It is my pleasure to welcome at SACLAY the second meeting of the International Group On Research Reactors. The organizers asked me to give a brief introduction and I would like to take this opportunity to underline some of the key points of the present status and future prospects for research reactors.
As you know, we are in CEA very much involved in this business, with our three reactors, OSIRIS and ORPHEE at SACLAY, SILOE at GRENOBLE, and in our position of Associate Member of the ILL which operates the high flux reactor at GRENOBLE. We are also cooperating with TECHNICATOME, an industrial subsidiary of the CEA which has, among its activities, the design and construction of research reactors.

Producing neutrons, for basic research or applied technologies, is the main purpose of the so-called research reactors. They are also used, quite often but on a lower scale, for commercial productions, such as radioisotopes used in medical or industrial applications, or doped silicon for electronics purpose.

There are a quite large number, near forty, of such reactors located in both the Western and Eastern parts of Europe. They are good experimental tools, which play an important role for increasing our scientific knowledge as well as for improving nuclear technology.

It does not mean that we have no difficulty. Let me say a few words about three problems we encounter with most of our facilities.

First, the safety. More than ever we must be very cautious in operating the research reactors. In a general way, they are quite simple and efficient as compared to the large reactors used for electricity generation. But, the risk is never zero and the consequences of anykind of large incident, not speaking of a real accident, could be far more important for the future of all our activities and of the nuclear industry than the particular contribution of a given facility.

Safe operation of nuclear facilities, not only the reactors, is an international concern. As for electricity generation plants, international cooperation must be set up in order to give to all countries sufficient garantees on a safe operation of all existing research reactors. If some countries, in the Eastern part of Europe, or elsewhere, meet particular difficulties with their own facilities, we must be ready to help them either to assess the present safety level or to improve it.

I would not give the impression of being worried of catastrophic situations. As most of the operators, I am quite confident in the good safety features of research reactors, but I cannot let aside some realities: Most of these reactors are quite old, many of them are in the vicinity of populated areas, some of them have difficulties to be maintained or updated.

In our country, not speaking of ORPHEE, which is one of the youngest research reactor, we have spent and we are still spending a lot of money to improve the safety of OSIRIS and SILOE. Faced to strong requirements from our Safety Authority, such as very low releases in case of a BORAX type accident, we proceed with continuous renewal or improvement of the safety grade components.

The fuel cycle is a second difficulty which concerns most of the existing research reactors. Large amounts of spent fuels are accumulated in pools, waiting for a satisfactory solution of the end of cycle problem.

In Western countries, at least, the previous solution is no more available. Interruption of the US DOE services has let the operators of research reactors with a temporary dead end. No restart is expected in the near future and, even if it could occur, the practical conditions would certainly be very different of the previous ones, more expensive and less simple.

Alternative solutions are being studied by europeans countries, some services are already offered and I am sure that other proposals will be made soon. But, we must be realistic: the cost of the fuel cycle will be higher and this applies for high enriched fuels as well as for the new low enriched ones.

My third concern is with the financing of research reactors. It is not directly related to the two previous points, even if they contribute to increasing the difficulties.
In a more general way, it will be useful to clarify two important questions, in order to prepare the future of research reactors:

What will be the needs and the requirements in the next ten or twenty years?

How do we calculate the cost of a given service?

Obviously, these questions are more difficult for applied technologies and commercial productions than for basic research. They are of particular concern for multipurpose reactors.

I am sure you don't expect me to bring the answers now, but I can say that we are devoting a great effort in CEA to try to solve these problems and we are ready to participate in any kind of international thinking on the subject.

Anyway, I am confident that we will find solutions to all these problems and looking to the future, I would like just to mention some recent results concerning the research reactors in our country :

First, the decision which has been taken to refurbish the high flux reactor of the ILL. It is a big operation, involving a complete rebuilding of the reactor tank; it will take more than two years and it will be necessary to have a new licensing; but we hope it will offer the possibility to operate the reactor for at least fifteen more years, thus contributing to satisfy the important needs of the scientific community.

The good results which have been obtained in testing a new cold source at ORPHEE let appear the possibility of extending the experimental capacity around this reactor; a project has been set up and it could also increase the available means for basic research.

A new experimental device , the OPERA loops, will be soon implemented in OSIRIS, and it will allow us to offer wider services for irradiations in LWR simulated conditions. We have an important programme for improving the performances of the UO2 and MOX fuels used in our electricity generation plants and such a device will play an important role for the validation of new designs.

The design of new research reactors is also an important objective, and we are working with Technicatome on it. The project SIRIUS II, based on the best French knowledge in the field, is now offered on the market.

The aim of IGORR meetings is to exchange experience and to work together on common problems of research reactors. Let me wish you fruitful discussions and I hope you will enjoy the two days meeting in this place.

RESEARCH REACTORS

INSTITUT LAUE LANGEVIN
GRENOBLE

Refurbishment Programme of the reactor and progress of work

J.M. ASTRUC
DRe-ILL

1. PRESENTATION OF THE RHF (High flux Reactor)

The ILL was founded in Grenoble in 1967. The reactor started in normal operation, at full power (57 MW nominal value) at the beginning of 1972.

Up to last year, It has been operated for more than 4100 Equivalent Days at Full power.

It can be useful to present rapidly a figure of the reactor : fig.1.

During these 20 years of operation, we had to face to some problems. For instance, between 1982 and 1985, we had to face to the problem of the ruptured heavy water collector, in the upper part of the reflector tank. About at the same period, we had to replace all the beam tubes, due to the evolution of the mechanical caracteristics of the aluminium alloy under irradiation.

2. BACKGROUND

On 30.03.91 the reactor was shut down for a normal maintenance period after an operating cycle with no problems.

Some days after, an inspection of the internal elements of the Reactor was carried out in the course of the regular inspection programme. Then , we have found cracks on the grids which ensure the regular flow of cooling water.

The details of the grids are shown in the figure 2

We started immediatly a program to found the reasons of these cracks, which included calculations and measurements in the reactor.
The investigations showed that the cracks are due to a design fault, aggravated by the effects of mechanical fatigue on highly irradiated material. figure 3

We studied, first, a solution in order to temporarily restart the reactor as soon as possible, but these efforts were rapidly stopped, for different reasons.

It is not possible to repair the cracked grid, and this must replaced. This involves, at least, the dismantling of the internals parts of the reactor tank.

Our Associates asked for an intervention programme which would result in an overall state of the reactor compatible with a long term expectation of operation for at least 10-15 years.

An analysis carried out jointly by ILL and a group of industrial experts, from the three member countries, concluded by defining several solutions :

One solution was to cut the grid and to replace it by a new grid made of 8 or 12 pieces which could be introduced in the reactor tank by the existing openings;

Another solution was to open the upper part of the reactor tank, and to replace the grid in one piece.

Another consisted in replacing the reactor block and the ancillary elements.

In October 91, the ILL was requested to go in greater depth into this last solution, in association with an international consortium. The aim set was to provide figures on costs and timescale on the basis of industrial tenders.

3. PRESENTATION OF THE PROGRAM

3.1. Technical content of the refurbishment

The reactor refurbishment programme provides for the replacement of the reactor block, the coupling sleeves, the antiturbulence grids and the diffuser, and of the ancillary elements.

The main items of equipment to be replaced are as follows: see figure

- the reactor block consisting of the reactor vessel and its cover, known as the "upper structure",
- the heavy water collectors,
- connecting sleeves between the reactor block and the flanges of the various beam tubes.

These three items constitute the primary circuit in the swimming pool.

It is also planned to replace some internal parts of the reactor tank, such as the beam-tubes, the grid and diffuser and the chimney.

Some parts of the present reactor, which are not at the end of their life, would be reused, for instance the two cold sources, the safety rods, and some other pieces.

The parts replaced would be cut up and packaged in accordance with current standards and disposed of.

After removal of the reactor block, thorough cleaning and decontamination of the pool will be made. Furthermore, we have already verified that the activation of the stainless steel liner of the pool is very low. This will allow to carry out the work of reconstruction and re-installation of the central parts of the reactor in a non-active environment. This concerns in particular the sensitive work to ensure leaktightness on all metallic seals.

The emptying of the reactor swimming pool will permit a complete inspection of the state of the lining, the replacement of the bellows which provide the link between the swimming pool lining and the flanges of the different beam tubes, and also a metrology necessary to determine the actual geometry of the areas for the intervention work.

3.2 Modifications planned

All items are in principle to be replaced by identical equipment. This concerns in particular performance, mechanical characteristics and the choice of materials.

However the experience gained over 20 years' operation indicates certain minor modifications and some simplifications. This concerns , for instance, the omission of the heavy water collector in the upper part of the reactor vessel, or some modifications of the present grid in order to prevent another failure.

A study is started to examine the possibility of replacing the existing grid, which cannot be dismantled, by an grid capable of being dismantled. *see figure*

Calculations have already been made, showing the possibility of such a new design. This design will be tested in a mock-up, at a reduced scale, in order to obtain all the qualification necessary for the authorisation to use this new grid. If that fails, it will always be possible to use the present grid.

3.3. Regulatory aspects

The calculation of the new reactor-block, following ASME section III, division 1, class 1, is in progress. Up to now, only minor changes in the design of the present reactor are introduced by this calculation.

A study is in progress to verify the behaviour of the central parts of the reactor in case of earthquake, with a level 7-8 MSK. This study concerns both the engineering structures of the swimming pool and the storage pools, and the reactor block proper.

It appears that a new decree has to be obtained, due to the fact that the normal operation is stopped for more than two years, and perhaps this will need a new public enquiry, but this point is not entirely clear up to now. The total delay to obtain a new authorisation is estimated at about two years.

4. PROGRESS OF WORK

The replacement of the reactor block necessitates a complete dismantling of the equipment in the reactor block, and of the structures in the reactor swimming pool.

4.1. Dismantling work

All the instruments in the areas where work is planned have been dismantled and these areas are now completely clear.

Almost all the reactor equipment proper has been dismantled, and the active parts not scheduled for subsequent re-installation have been cut up, packaged and sent for disposal. See the figure for the reactor-bloc before and after the dismantling of the equipments .

The intervention work in an active environment involving handling active parts has been more than half completed. The collective dose for all the operators concerned, from the beginning up to now, is 0.103 man.Sievert. The preliminary estimate 5 time more than that. So, this work was carried out in good radiological conditions.

The cutting of the reactor-bloc itself will be made under water, by a specially developed milling machine. see the figure. Some other machines have to be bought for cutting the tubes in the upper part of the reactor-bloc.

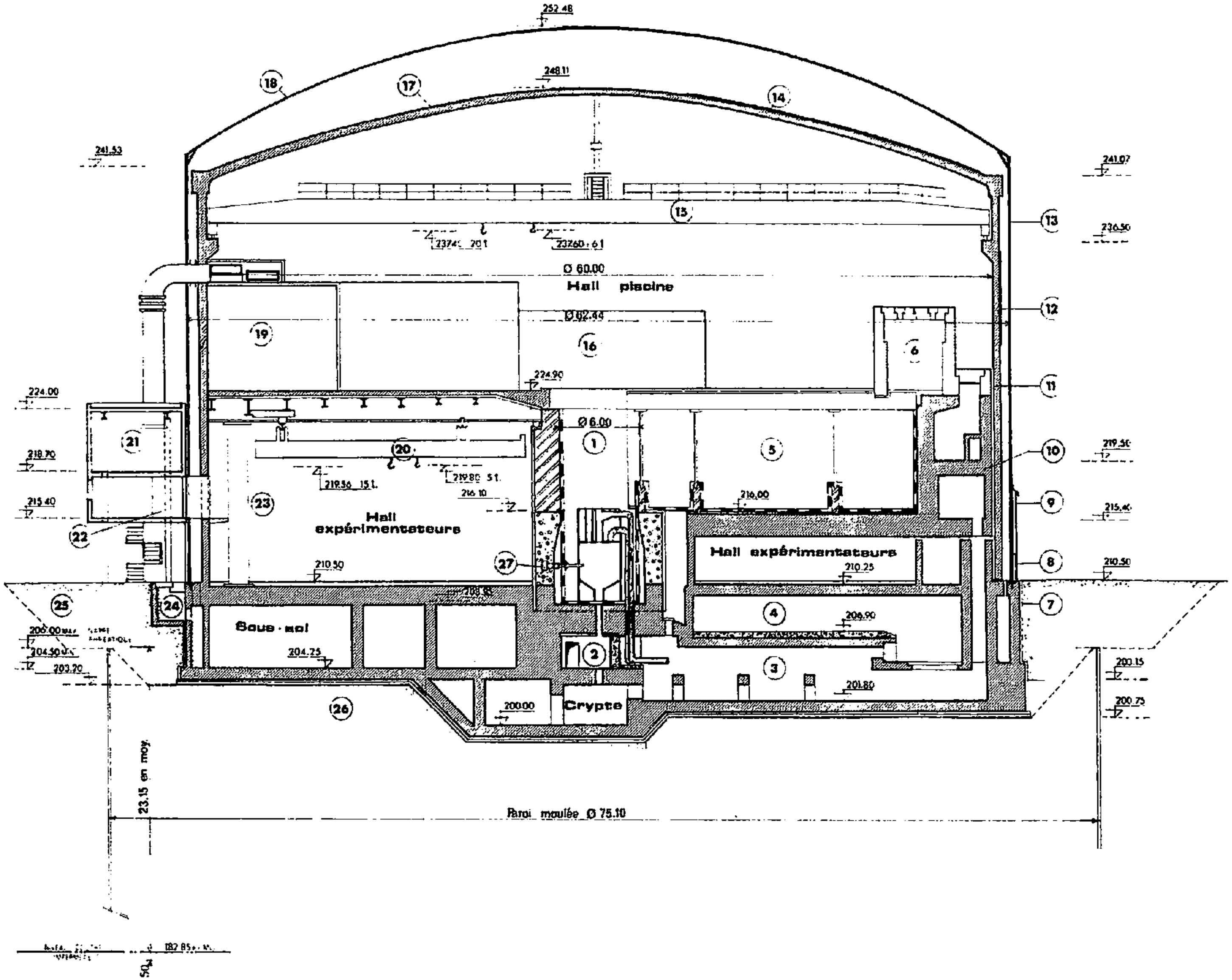
4.2. Invitations to tender

The invitations to tender for the principal equipment and services were issued jointly by ILL and an industrial architect. Detailed discussions with the different suppliers, have confirmed the expectations as regards prices and delivery dates, and made it possible to draw up the order documents within a short time.

4.3 Delays

The expected delays are presented on the following table.

If a decision is taken, as expected, at the end of this month, we expect to be able to restart the reactor in the middle of 1994.



LEGENDE

- Béton lourd : densité 5.5
- Béton lourd densité 3.5
- Béton 1ère phase
- Béton 2ème phase

1. Piscine reacteur
2. Salle des mécanismes
3. Galerie des collecteurs
4. Galerie des vannes
5. Canal de transfert
6. Cellule chaude
7. Etancheite sous-sol
8. Calorifuge
9. Bardage
10. Bloc arriere
11. Jupe béton
12. Isolation thermique sur toute la surface
13. Jupe métallique
14. Isolation thermique sur toute la surface
15. Pont roulant
16. Locaux périphériques
17. Coupole béton
18. Coupole métallique
19. Centrale pulsion
20. Pont roulant
21. Prise d'air frais
22. Sas de secours
23. Passerelle
24. Accès à l'espace annulaire
25. Remblai
26. Etanchéité et couche de glissement
27. Canal expérimental

COUPE GENERALE DU BATIMENT REACTEUR

FIG. 1

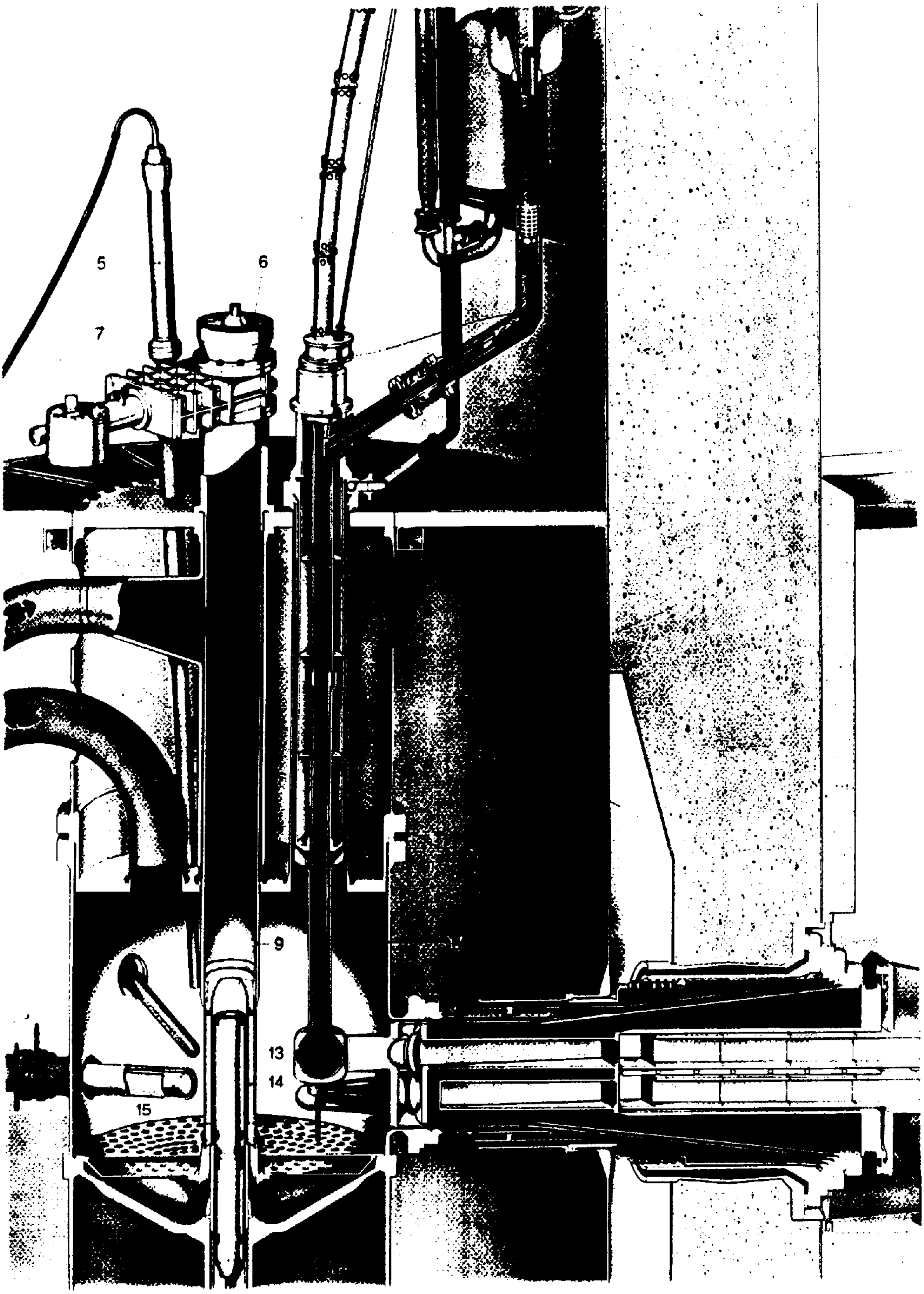


FIG. 2

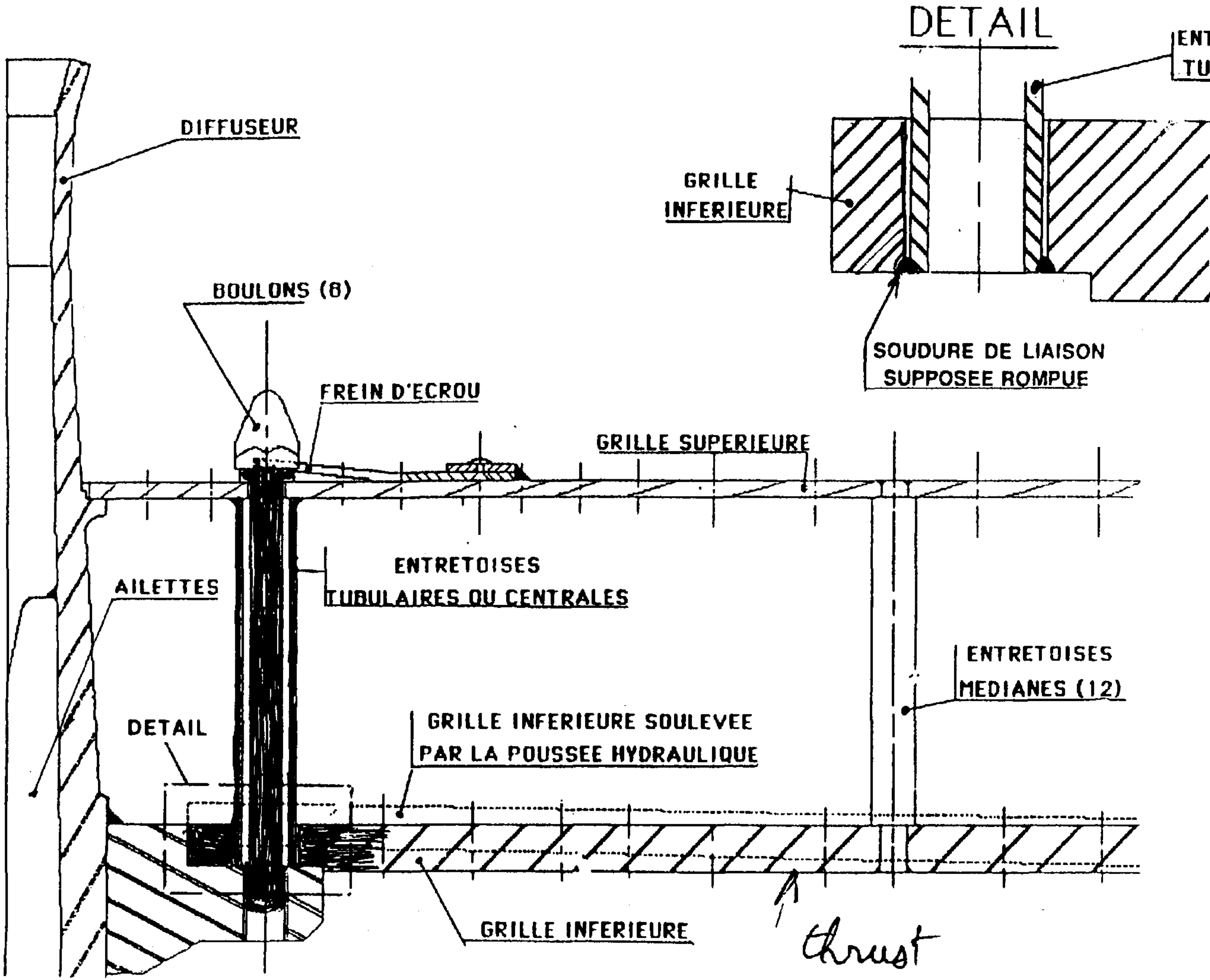


FIG.3 DETAIL DES GRILLES -DETAILS OF THE GRIDS

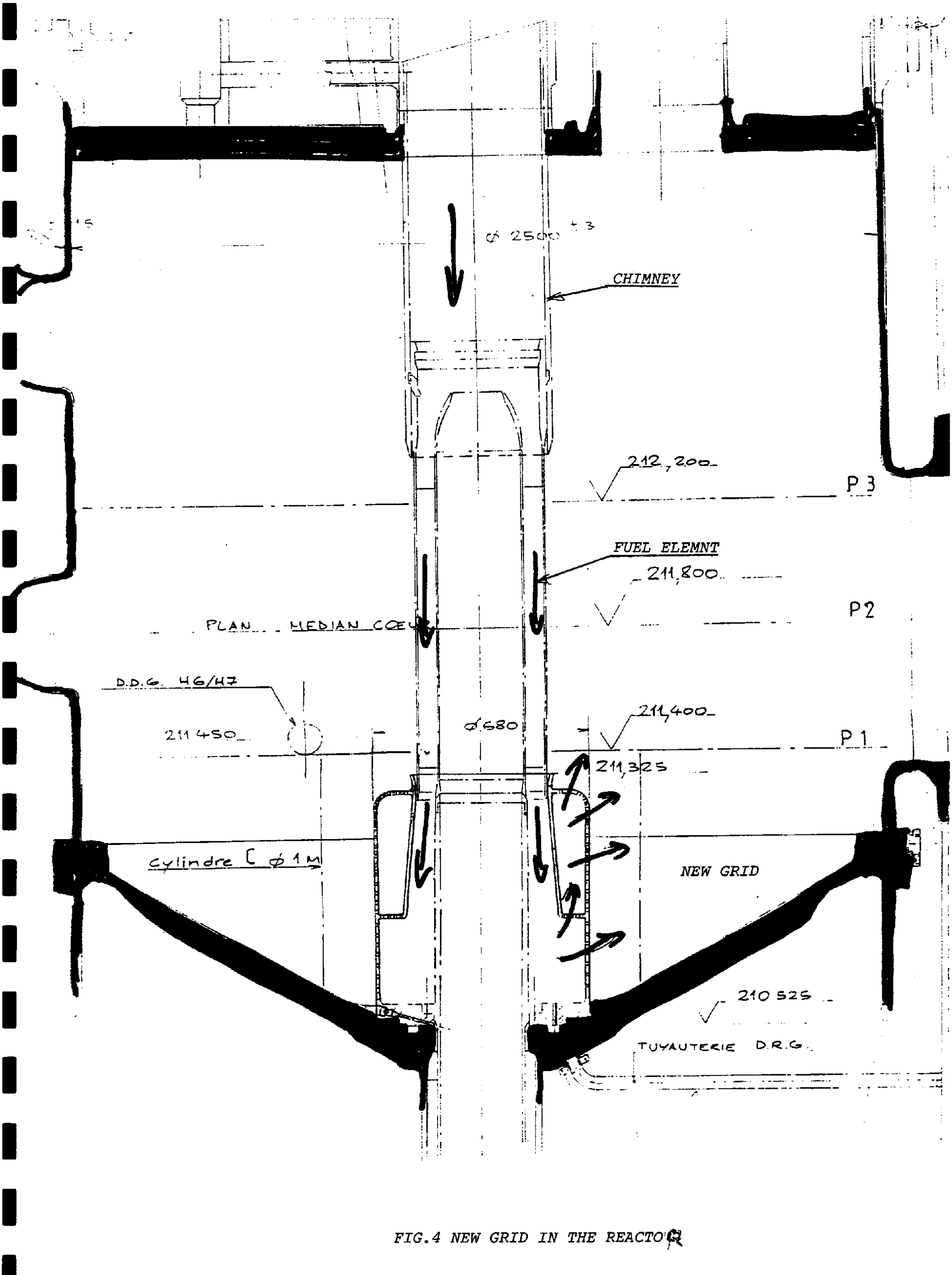


FIG.4 NEW GRID IN THE REACTOR

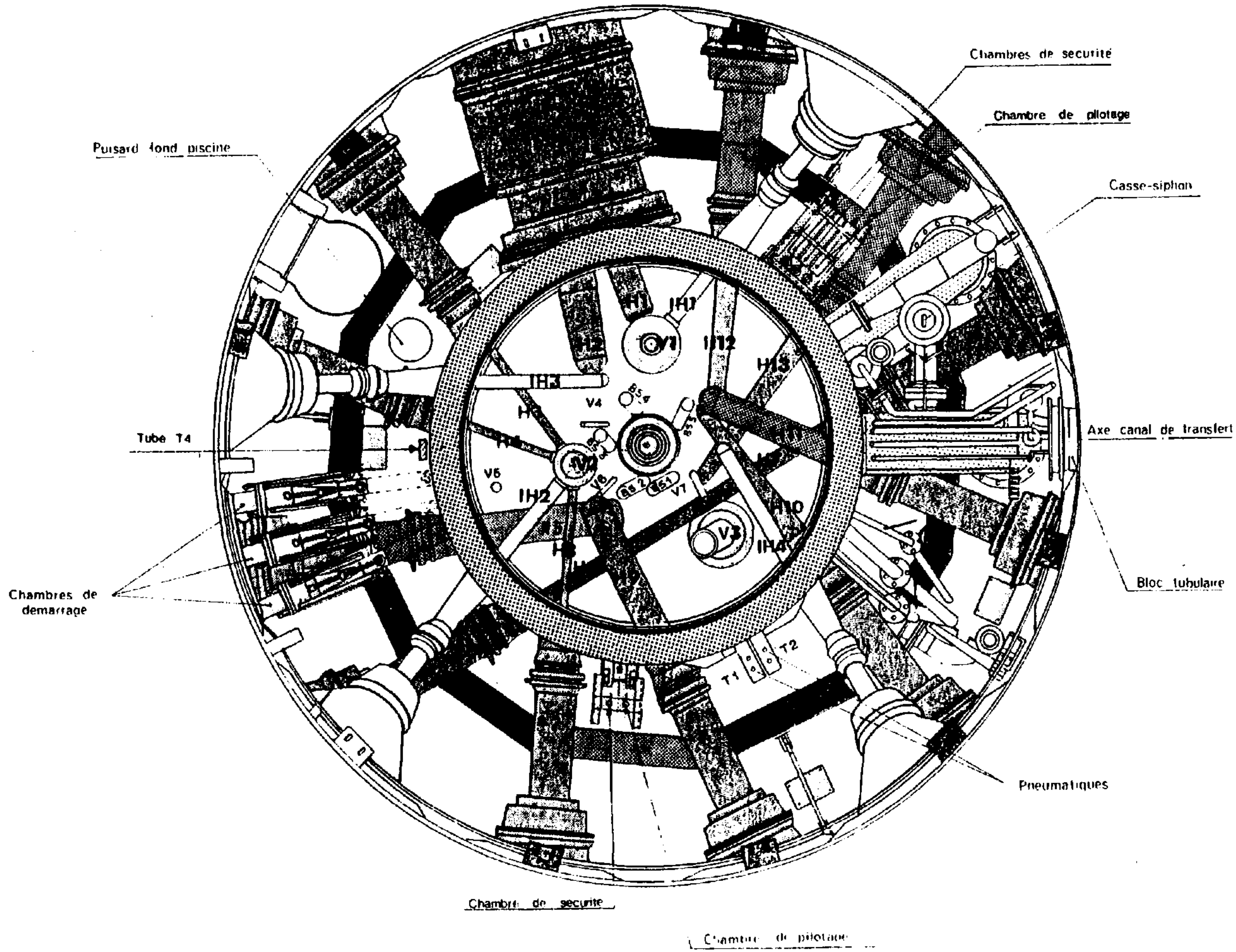


FIG.5 HORIZONTAL VIEW OF THE POOL

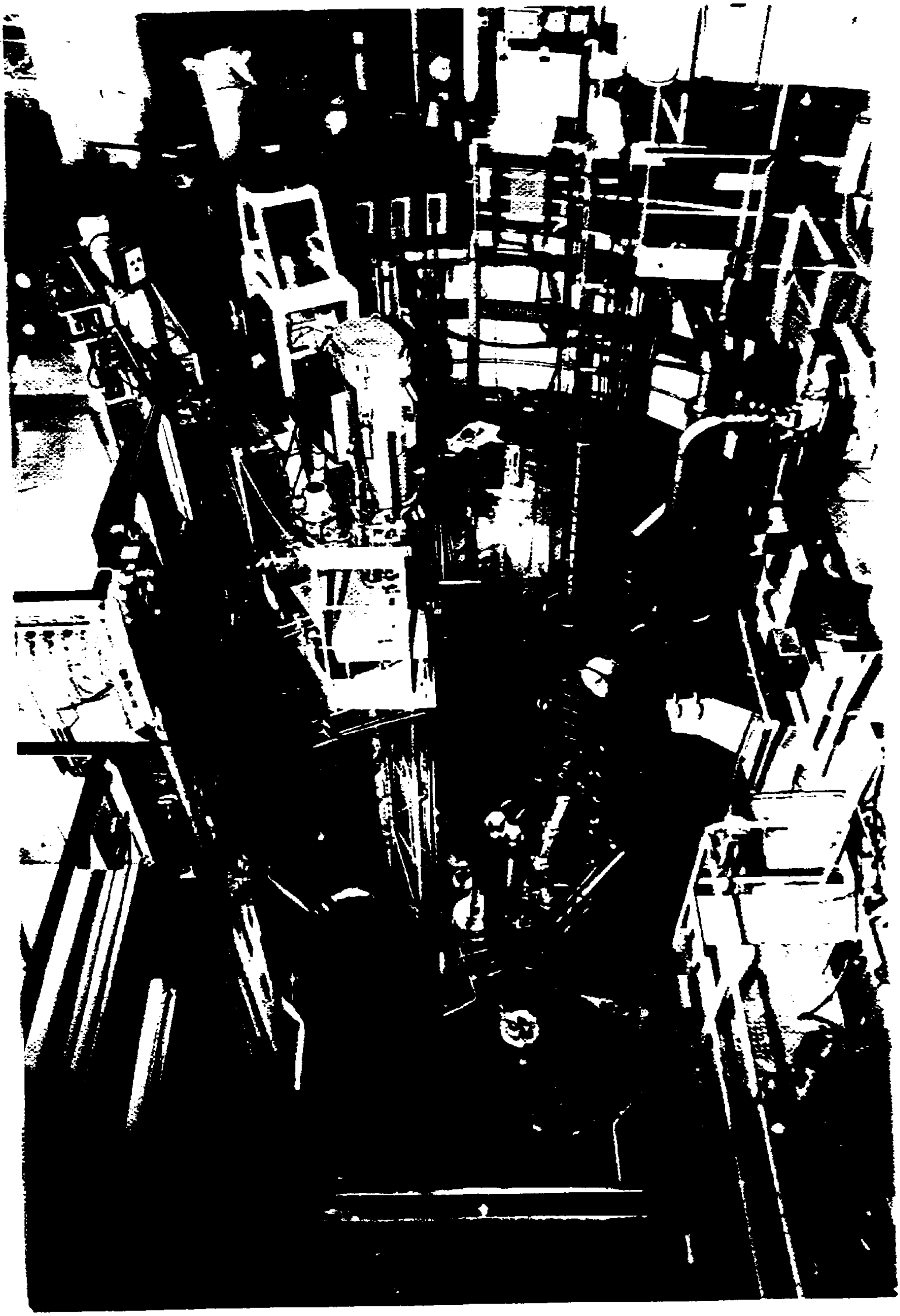


FIG. 6 ONE YEAR AGO

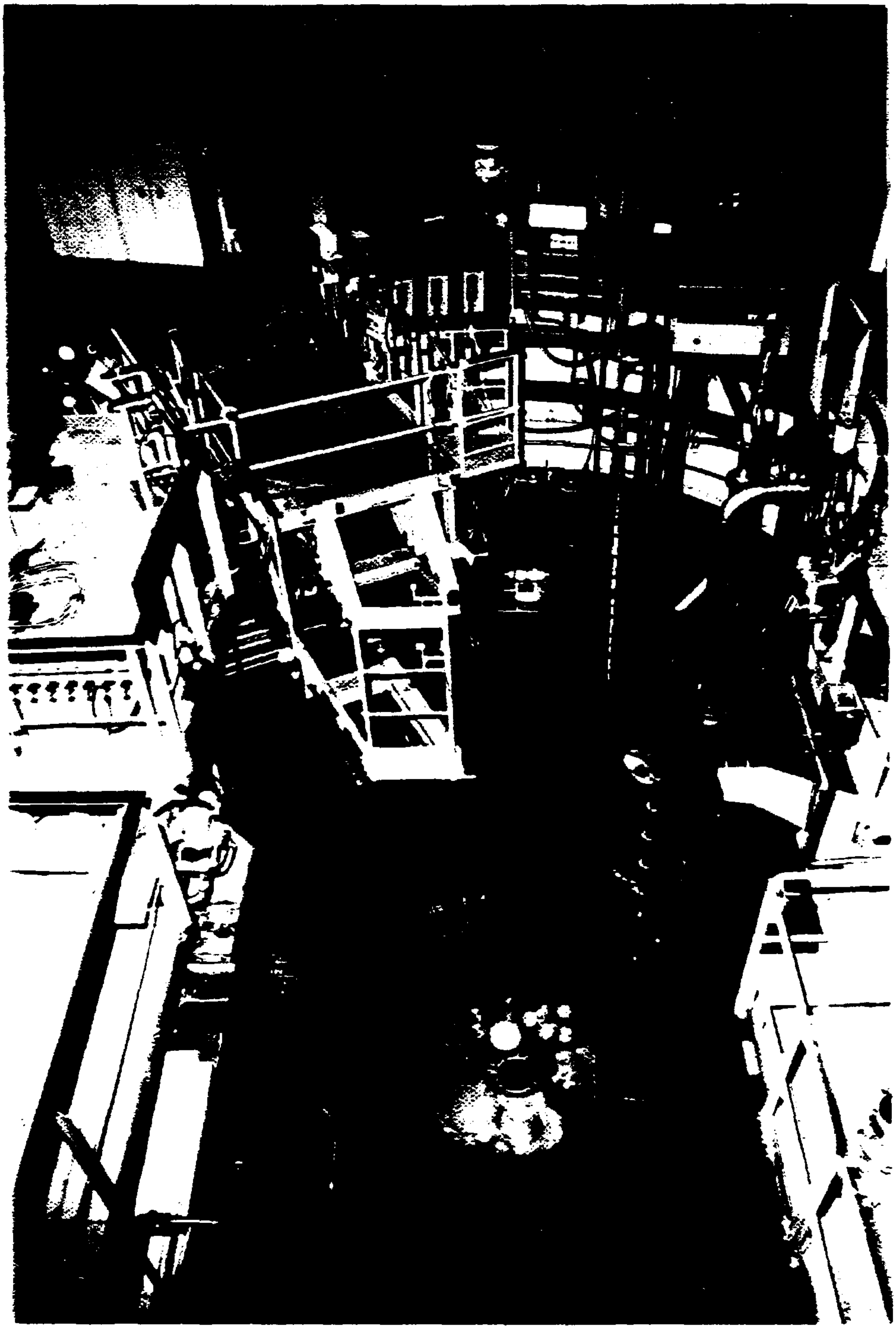


FIG.7 PRESENT STATE (MAY 92)

Status Report on the FRM-II Project

K. Böning

Fakultät für Physik, Technische Universität München,
D-8046 Garching, Germany

ABSTRACT

The new German neutron source FRM-II is predicted to yield a thermal neutron flux of about $8 \cdot 10^{14}$ $cm^{-2}s^{-1}$ outside of the reactor core at 20 MW power. Its main design features are a very compact core cooled by light water and placed in the center of a large heavy water moderator tank. - The paper reports on progress the project has made since the first IGORR meeting in 1990. These achievements include a new evaluation of the total costs, a new time schedule of the project and some modifications of the facility design. An updated version of the safety report has been practically completed.

INTRODUCTION

The planned new neutron source FRM-II will be a national high flux research reactor, which has been optimized primarily with respect to beam tube applications but which will also provide very attractive possibilities for other fields of utilization. It will be operated by the Technical University of Munich in replacement of its existing research reactor FRM.

Reports on the design of this facility have been given, e.g., in references /1,2/ and also on the first IGORR meeting /3/. In what follows we will briefly recall some principal design features of the FRM-II, report on the progress which has been made since the first IGORR meeting and finally mention some examples of modifications which have been incorporated in the design since then.

FRM-II DESIGN FEATURES

The FRM-II concept provides for a very small "compact core" which will be cooled by light water and placed in the center of a large heavy water moderator tank. The cylindrical single fuel element has an active volume of 17.6 liter containing about 8 kg of high enriched uranium in 113 fuel plates of involute shape. The reactor is controlled by a central hafnium absorber rod with a beryllium follower underneath. Two independent fast shutdown systems will be realized, - firstly - by the central control rod just mentioned and - secondly - by five hafnium shutdown rods in the moderator tank which are fully withdrawn during normal reactor operation. The reactor power will be 20 MW and the cycle length about 50 days. An unperturbed thermal neutron flux maximum of $8 \cdot 10^{14}$ $cm^{-2}s^{-1}$ will be obtained in the moderator tank - the corresponding flux-to-power ratio being higher than in any other reactor /1/. The large volume and high values of thermal neutron flux in the moderator tank will be made use of by 10 horizontal tangential beam tubes and 9 vertical irradiation channels.

The reactor building has the cross section of a 40 x 40 m^2 square at the ground floor level ("experimental hall") and of an octagon on the 12 m level ("reactor hall"). Additional space, also for offices and laboratories, will be provided for in an adjacent "neutron guide hall" offering an experimental area of 60 x 25 m^2.

FRM-II PROJECT STATUS

In the year of 1991 the evaluation of the site parameters was completed and the conceptual design and the basic safety concept of the FRM-II facility was established /2,3/. As a consequence, the first draft of the safety report of the facility was worked out. (Note: According to German regulations, the safety report is mainly for the benefit of the public so that individuals can consider whether or not they might be affected by the project. The licensing authority requires much additional and more detailed information).

This first draft of the safety report was then discussed with experts from other organizations. At the same time, the conceptual design of the facility was reviewed in order to identify design simplifications with the potential to reduce costs (examples of such modifications will be given in the next section). As a result of these efforts all the essential design features of the facility can be considered now as firmly established. According to this situation the Siemens AG Company - into which the previous Interatom GmbH Company has been incorporated - found itself in a position to submit, in December 1991, a fixed price offer for the detailed design and construction of the research reactor facility.

This offer, then, represented the basis for the evaluation of an updated cost estimate of the whole project, yielding a result of slightly less than 500 million DM (in December 1991 money). This value includes 60 million DM for the basic experimental equipment, disregarding the beam tubes themselves which are considered as an inherent part of the research reactor facility.

At present, all design modifications resulting from the initiatives as mentioned above are being worked into the safety report to obtain an updated version. In this way all necessary documents should be ready to submit - immediately after a new political decision which is expected for spring or early summer of 1992 - the official application for the nuclear licensing of the FRM-II. In parallel to this another official request ("Raumordnungsverfahren"), which includes an environmental impact examination, can be started.

Fig. 1 shows the FRM-II time schedule which begins with the date when the two offical requests will be submitted, presumably in about June 1992. The choice between the two versions of this schedule will be one of the subjects of the political decision just mentioned. The "basic schedule" is consistent with the fixed price offer of the Siemens AG Company whereas the "alternative schedule" proceeds at a somewhat reduced speed. Not shown in Fig. 1 is the project phase 1a (conceptual design and safety report) which has already been completed. If we focus our attention on the "alternative schedule" this begins with the "basic design" (for licensing) in phase 1b which means a further improvement of the conceptual design up to a degree that a preliminary positive assessment could be granted by the nuclear licensing authority. This first licensing step, which could be worked out by the authority during phase 1c, would also involve the authorization to begin with the construction of the new buildings. After a final political decision (phase 1d) the detailed design (phase 1e) could be made and, finally, the construction of the whole facility could be performed (phase 2). The completion of the facility and the begin of nuclear operation requires two more licensing steps. - The "basic schedule" differs from the alternative schedule only in the phases 1b and 1e and in the phases 1c and 1d being carried out at the same time.

RECENT DESIGN MODIFICATIONS

In what follows we will give some examples of significant modifications of the facility design which have been made recently.

In 1990 it was already decided to separate the project of the new FRM-II from the decommission of the existing FRM. Shortly before the new FRM-II goes into operation the old FRM will be permanently shut down, but the formal procedure of decommissioning the FRM will be subsequently performed so that any possible uncertainties in its time schedule would be without effect for the FRM-II project. Nevertheless, the option is maintained that at some later date the neutron guide hall of the new FRM-II could be extended into the FRM reactor building to obtain additional space for scattering experiments on the neutron guides (see also Fig. 3 of Ref. /3/).

The scheme of the preliminary cooling concept of the FRM-II has been shown in Ref. /2,3/. In the meantime the number of primary pumps has been increased from two to four because of safety reasons, i.e. to two pumps in each "primary cell". The N16 decay tank in the primary cooling circuit turned out to be unnecessary and has been cancelled. The number of two independent cooling systems for the reactor and decay pool has been reduced to one - and this single system has no longer to be safety-graded since the thermal capacity of the pool water is so large, considering the low value of reactor power, that the decay afterheat of the fuel element could be removed also by a tolerable increase of the pool water temperature and by pool water evaporation etc..

A vertical cut through the reactor block is shown in Fig. 2. One distinguishes the reactor pool with the D_2O moderator tank and the fuel element in its center, and part of the storage pool for the storage of spent fuel elements etc.. The tubes of the primary cooling circuit lead from the fuel element through the tubing channel to two primary cells (only one of them is shown). A hot cell placed on the top is accessible (not shown) from the storage pool. Some of the beam tube penetrations of the D_2O tank can be seen in Fig. 2, the beam tube on the left containing five neutron guides which lead through the neutron guide tunnel to the adjacent neutron guide hall. Finally, the five inclined shutdown rod drives can be seen on the top of the D_2O tank, the driving concept of which has also been changed recently from spring to gas pressure loaded.

Significant modifications have also been performed in the FRM-II buidings. As can be seen from the vertical cut of Fig. 3, the reactor building has now only one basement - instead of previously two - and in compensation for this, additional space has been created - more economically - in a new basement under the neutron guide hall. As motivated in the same way, additional offices have been created in the side wings of the neutron guide hall rendering a separate office building not necessary any more.

ACKNOWLEDGEMENTS

This paper is a summarizing report on a project which many colleagues and coworkers from various institutions have contributed to. These include numerous members of the Faculty of Physics of our Technical University and of the Siemens AG Company (previously Interatom GmbH).

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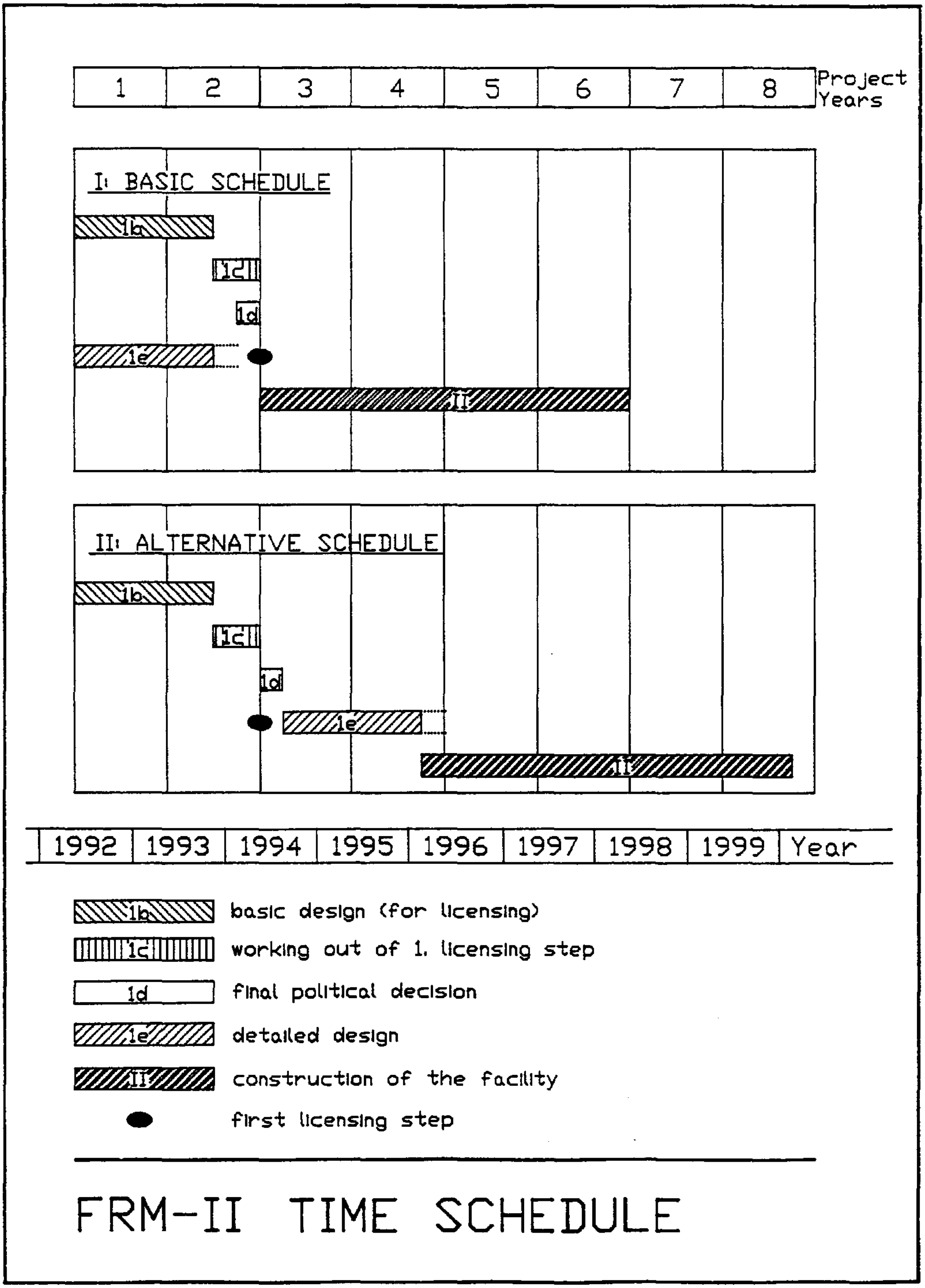

Fig. 1: FRM-II Time Schedule. Shown are two versions which both begin on the date of the official application for nuclear licensing which is expected for about June 1992.

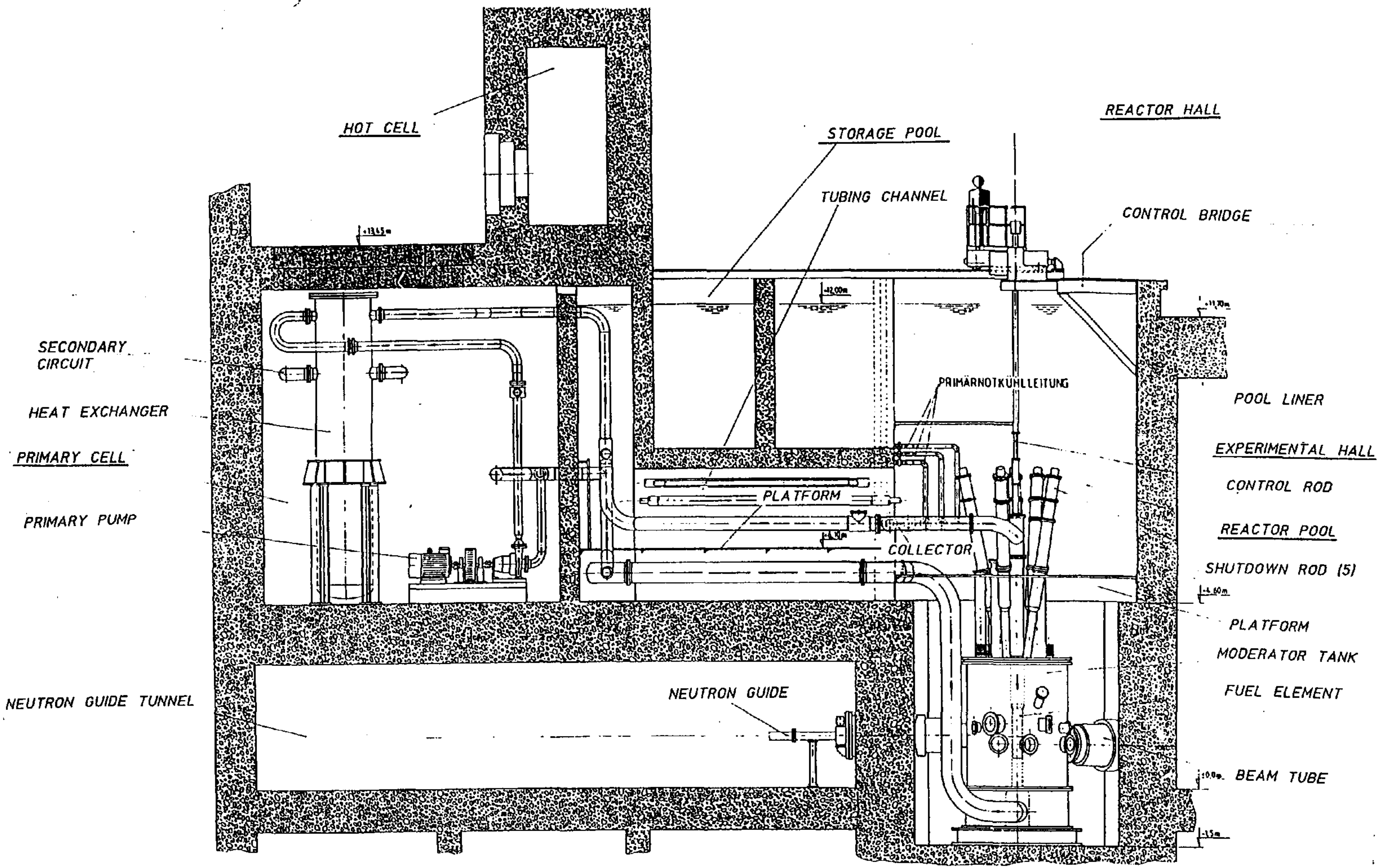

Fig. 2: Vertical cut through the reactor block showing the reactor pool, the storage pool, the tubing channel to the primary cells and the neutron guide tunnel.

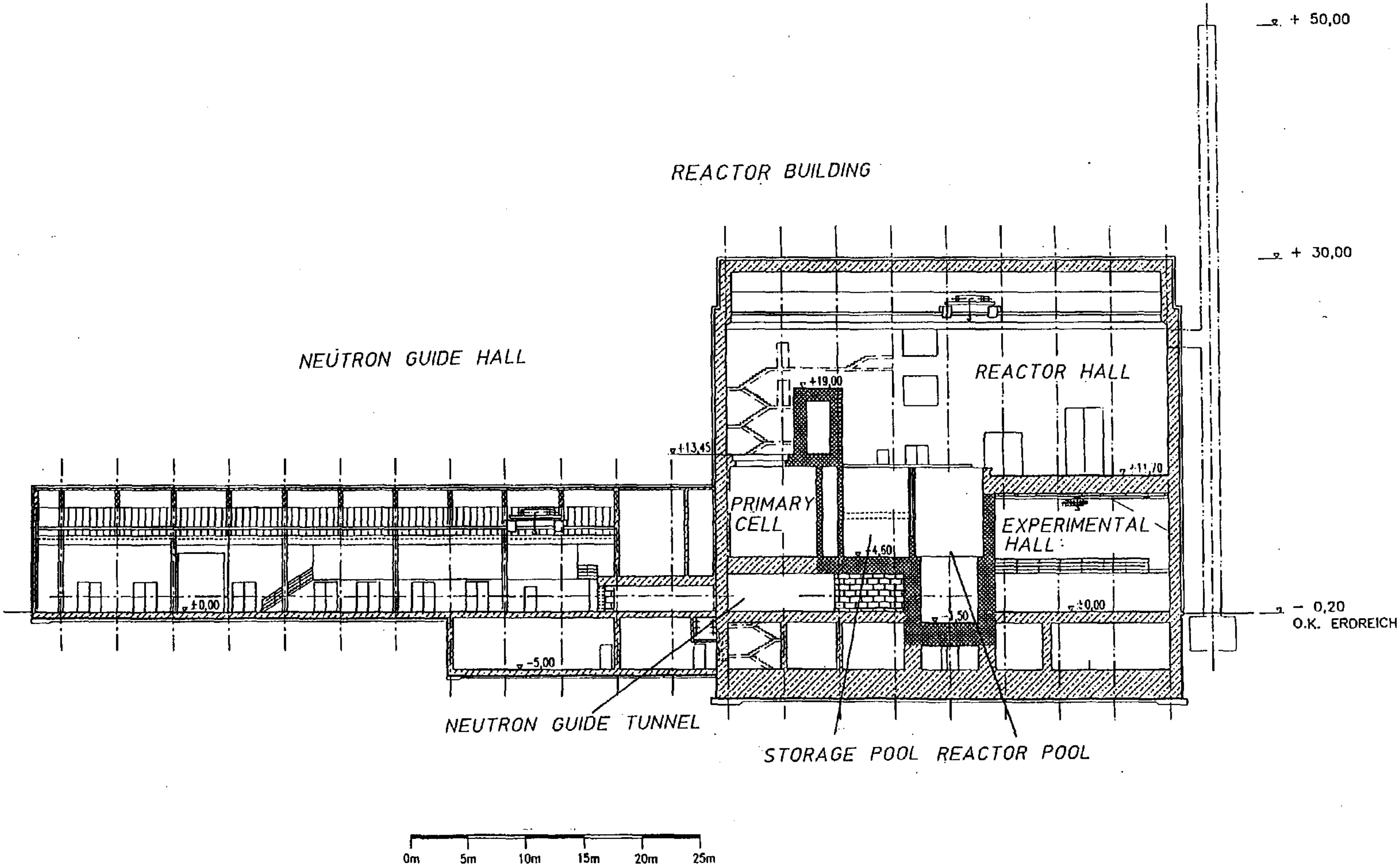

Fig. 3: Vertical cut through the FRM-II reactor building (right) and neutron guide hall (left). The egg-shaped building of the existing FRM (not shown) would be on the far left.

International Group on Research Reactors

Saclay Meeting

May 1992

New Research Reactor for Australia

By R. Miller

Australian Nuclear Science and Technology Organisation

New Research Reactor for Australia

HIFAR, Australia's major research reactor was commissioned in 1958 to test materials for an envisaged indigenous nuclear power industry. HIFAR is a Dido type reactor which is operated at 10 MW. With the decision in the early 1970's not to proceed to nuclear power, HIFAR was adapted to other uses and has served Australia well as a base for national nuclear competence; as a national facility for neutron scattering/beam research; as a source of radioisotopes for medical diagnosis and treatment; and as a source of export revenue from the neutron transmutation doping of silicon for the semiconductor industry.

However, all of HIFAR's capabilities are becoming less than optimum by world and regional standards. Neutron beam facilities have been overtaken on the world scene by research reactors with increased neutron fluxes, cold sources, and improved beams and neutron guides. Radioisotope production capabilities, while adequate to meet Australia's needs, cannot be easily expanded to tap the growing world market in radiopharmaceuticals. Similarly, neutron transmutation doped silicon production, and export income from it, is limited at a time when the world market for this material is expanding.

Ansto has therefore embarked on a program to replace HIFAR with a new multi-purpose national facility for nuclear research and technology in the form of a reactor

- for neutron beam research
  - with a peak thermal flux of the order of three times higher than that from HIFAR,
  - with a cold neutron source, guides and beam hall,
- that has radioisotope production facilities that are as good as, or better than, those in HIFAR,
- that maximizes the potential for commercial irradiations to offset facility operating costs,
- that maximizes flexibility to accommodate variations in user requirements during the life of the facility.

Ansto's case for the new research reactor received significant support earlier this month with the tabling in Parliament of a report by the Australian Science and Technology Council on recommended priorities for government expenditure on major national research facilities over the next ten years. A new research reactor was one of seven proposals recommended by the Council for priority during that period.

As basis for Ansto's normal activities is nuclear science and technology rather than reactor development, it will be necessary to purchase much of the nuclear specific technology and hardware with the emphasis being on modern but proven technology.

In January 1992 Ansto commenced a two year preliminary engineering and financial study that will define the user requirements, assess the availability of reactor designs compatible with those requirements, complete preliminary design and provide a detailed costing and schedule for the provision of the facility. The report of this study will form the basis of a submission to Government for funding for detailed design and construction. Initial operation of the reactor is scheduled for 2003. Figure 1 shows the overall project schedule.

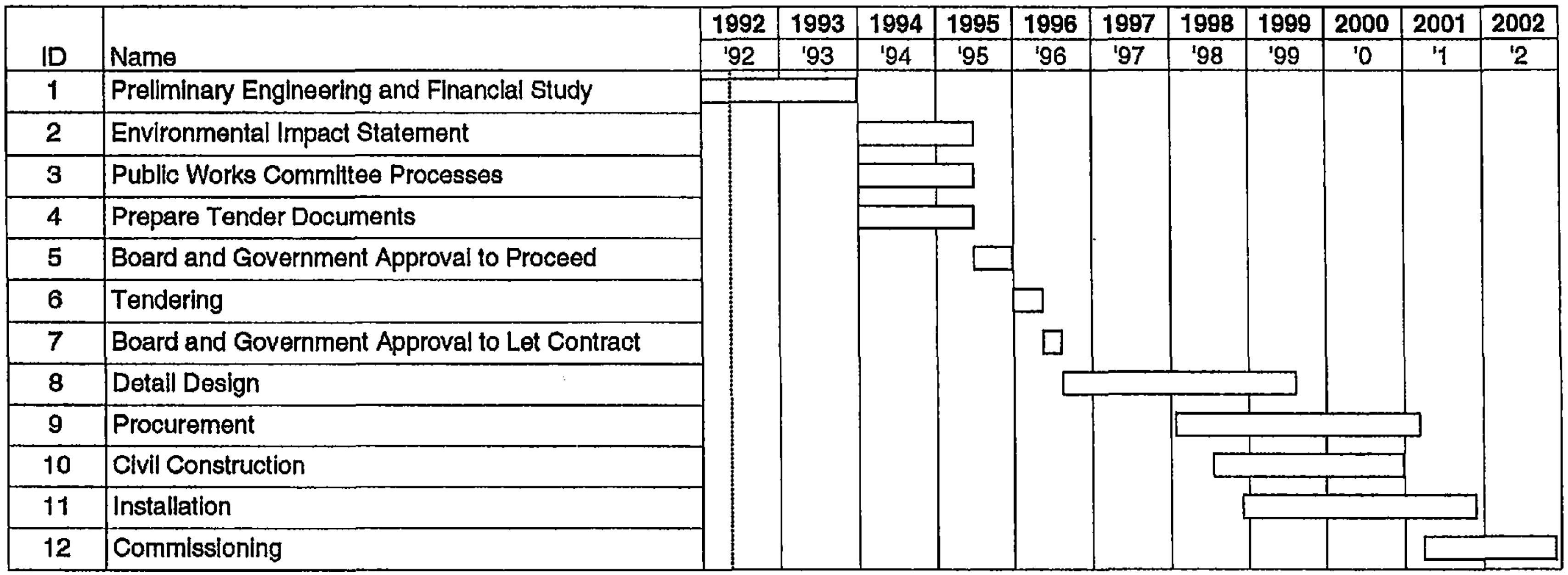

Figure 1 : Schedule for Provision of a New Research Reactor Facility.

5.2 CORE DESIGN

The reference 19-site core (Figure 1) currently consists of six cylindrical 18-element fuel assemblies in the control and shutdown sites, two experimental assemblies for materials testing and eleven hexagonal 36-element fuel assemblies. Cylinders of hafnium surround the circular flow tubes holding the 18-element fuel assemblies. Three of these hafnium cylinders provide reactivity control, the other three are maintained as a poised safety bank. All six cylinders, acting as one shutdown system, are inserted into the core upon detection of a trip signal.

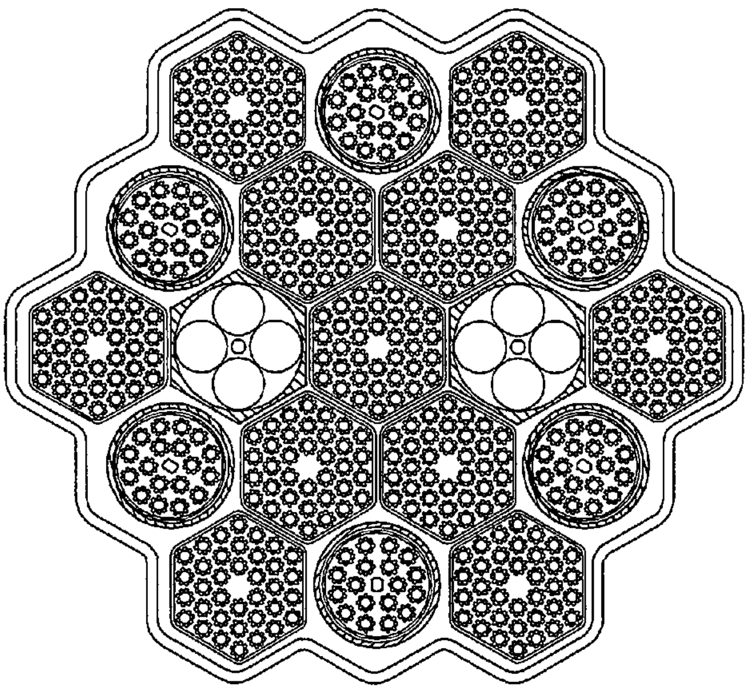

Figure 1: Schematic Representation Of The MAPLE-MTR Core

With a core volume of 63 L, the MAPLE-MTR core is considerably smaller than that other materials test reactors, such as HFR-Petten at 210 L, OSIRIS at 250 L and SILOE at 113 L, and has fewer sites for irradiation experiments. This small core volume represents a compromise between in-core space for high fast-neutron flux irradiations and thermal-neutron fluxes in the D_2O reflector. However, since the power densities are similar (i.e., 250 kW/L for MAPLE-MTR, 210 kW/L for HFR-Petten and 280 kW/L for OSIRIS), it is expected the MAPLE-MTR will have comparable fast-neutron fluxes in the range of 2 to 3 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$. Preliminary physics calculations indicate that the MAPLE-MTR has perturbed fast-neutron fluxes of ~1.4 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$ in the samples. These fluxes are at the lower end of the high fast-neutron fluxes requirements for the accelerated ageing studies. The core power will be increased if higher fluxes are demanded.

The thermal-neutron flux in the D_2O reflector will determine the value of the MAPLE-MTR for the fuel development program and for beam-tube applications. The basic requirements for high thermal-neutron fluxes in the D_2O reflector are a high power level and a small external surface area to the core. With a core volume of 63 L, the MAPLE-MTR core is about 10% larger than the ORPHEE core (56 L) and has about 10% greater external surface area. Since the power densities are similar, it is expected that the thermal-neutron fluxes in the D_2O reflector will be similar. Preliminary calculations indicate the peak thermal-neutron fluxes are $\sim 3 \times 10^{18}$ $n \cdot m^{-2} \cdot s^{-1}$.

5.3 FN-ELEMENT DESIGN

With the small core volume, the in-core space for materials testing is limited to two or three core sites. Additional fast-neutron irradiation space is provided by introducing fast-neutron rods (FN-rods) in the D_2O reflector. These FN-rods consist of annular fuel assemblies placed in the D_2O reflector to locally convert thermal-neutron fluxes into medium fast-neutron fluxes. Such FN-rods have been used in the NRU and DIDO reactors for many years to provide fast-neutron irradiation facilities with fluxes of $\sim 0.6 \times 10^{18}$ $n \cdot m^{-2} \cdot s^{-1}$.

The FN-rod concept for the MAPLE-MTR assumes use of the same fuel element as the driver fuel. The development of the FN-rod concept must account for the compromises between the volume of irradiation space in the centre of the assembly and the magnitude of the fast-neutron fluxes. As well the physical location of the FN-rods with respect to the core will determine how much power and fast-neutron flux can be produced. Figure 2 shows a schematic representation of four FN-rods and the core. Preliminary calculations indicate that with an inner diameter of 74 mm for the FN-rod to hold the experiment assembly, the average power in each FN-rod is about 1.2 MW and the peak fast-neutron fluxes in the experiment assembly is $\sim 0.6 \times 10^{18}$ $n \cdot m^{-2} \cdot s^{-1}$. These medium fast-neutron fluxes will satisfy the irradiation requirements for corrosion studies. Although initial studies have used four FN-rods, additional FN-rods will be considered if more irradiation space is required.

The presence of the FN-rods in the D_2O reflector has several effects on the overall performance of the reactor. The available excess reactivity of the reactor can be increased by as much as 100 mk, depending on the amount of ^{235}U in the FN-rods. The FN-rods decrease the local thermal-neutron fluxes and increase the local fast-neutron fluxes. The fast-neutrons escaping from FN-rods are thermalized elsewhere in the D_2O reflector and contribute to higher thermal-neutron fluxes in other regions of the D_2O reflector.

5.4 FUEL TEST LOOP CONCEPTS

Several fuel test loops will be included in the D_2O tank for the MAPLE-MTR. The irradiation requirements from the fuel development and reactor safety research programs have identified fuel test loops to hold small multi-element (up to 7) assemblies and loops to hold full-diameter CANDU bundles.

To achieve peak linear power ratings of 70 kW/m requires thermal-neutron fluxes of 3-4 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$. Preliminary calculations indicate that the regions between each pair of FN-rods have the requisite thermal-neutron fluxes. However, it is expected that the performance of the small-diameter test loops will differ, depending on the number of fuel elements. Preliminary scoping calculations have been performed to determine the fuel performance as a function of distance from the core. In these calculations, a four-element assembly with natural uranium UO_2 fuel was modelled. For a test loop centred about 100 mm outside of the core, the peak linear ratings in each of the fuel elements was ~67 kW/m, which is in the upper range of CANDU fuel performance. When the same test loop was centred about 270 mm outside the core, the peak linear ratings in each fuel element was ~47 kW/m, which represents the normal operating range. The peak linear ratings for the fuel test loop located 270 mm outside the core can be increased by enriching the uranium, thereby enabling the locations nearer the core to be occupied by the full-diameter test loops.

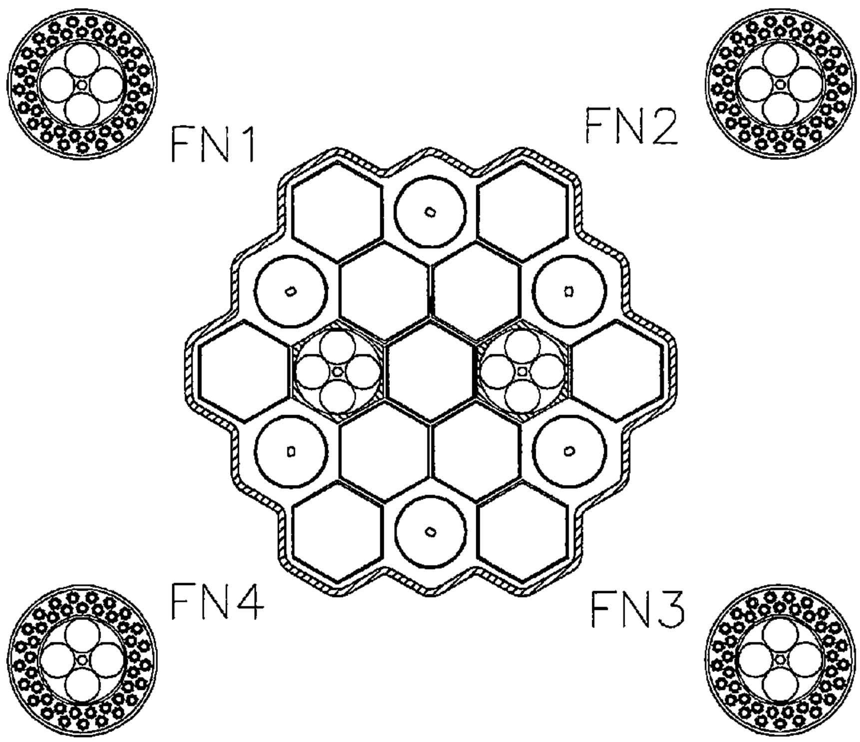

Figure 2: Schematic Representation Of FN-rods Around The MAPLE-MTR Core

As CANDU fuel bundles represent much stronger thermal-neutron absorbers, some enrichment will be needed to achieve the desired linear power ratings. Scoping calculations to determine the effect of enrichment on the power distributions through the bundles are in progress. To meet the needs of the fuel development program at least four test loop positions will be needed. These test loops will be located between pairs of FN-rods to take advantage of the prime thermal fluxes. In addition to providing the facilities for fuel irradiations, the local fast fluxes produced by the test fuel will provide the appropriate conditions for proof testing pressure tubes.

5.5 D_2O TANK

The D_2O tank for the MAPLE-MTR is adapted from the design of the MAPLE-X10 tank. At present, the reference diameter is 1.6 m, but the demand for many fuel test loop positions, four or more FN-rods and the beam tubes may require the diameter to be increased. The reference height for the MAPLE-MTR D_2O tank is 1.5 m, which is taller than the MAPLE-X10 tank. The tank height has been increased to increase the axial thermal flux region for the fuel test loops and to accommodate the fast D_2O dump system, which provides a second diverse shutdown system. Although placement of the beam tubes awaits finalizing the positions for the fuel test loops, it is expected that 6-8 beam tubes and a cold neutron source position can be accommodated.

The second shutdown system, which provides rapid removal of the D_2O is based on the shutdown system for the WR-1 reactor. The D_2O tank consists of two parts, the upper D_2O reflector region and the lower dump region, separated by a weir. When the upper region is filled with D_2O, the lower region is pressurized with helium to hold up the D_2O. In response to a trip signal, the pressure in the upper and lower regions is equalized and the D_2O spills over the weir.

6. SUMMARY

The research programs that require reactor-based irradiation facilities have been identified and include fuel channel materials research, fuel development, reactor safety research, condensed matter science and radio-isotope production. Table 1 summarizes the neutron flux requirements associated with these programs.

To provide the requisite neutron fluxes, the MAPLE-MTR would have a compact (63 L) H_2O-cooled, H_2O-moderated core with several high fast-neutron flux irradiation sites and a relatively high core power density (i.e., at least 250 kW/L). The in-core irradiation sites would provide the high fast-neutron fluxes needed for accelerated ageing studies on fuel channel materials. To provide medium fast-neutron flux irradiation sites, it is proposed to add at least four FN-rods in the D_2O reflector. To achieve the desired fast-neutron flux performance, each FN-rod will need to contribute about 1.25 MW to the reactor. Hence the total fission power of the reactor, excluding the power contributed by the fuel test loops will be at least 20 MW. To provide irradiation facilities for fuel development, reactor safety research and fuel channel proof testing, at least four fuel test loop positions for full-diameter CANDU bundles will be positioned in the D_2O tank. To complete the complement of fuel test facilities, several small-diameter multi-element test loops will be included. Sites will be chosen for these test loops once locations for the full-diameter test loops have been selected. Beam tubes will be placed in the remaining space in the D_2O tank. This forms the starting point for developing the MAPLE-MTR concept. Studies in physics, thermalhydraulics and engineering are in progress to develop a feasible concept.

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PLANNING A NEW RESEARCH REACTOR FOR AECL: THE MAPLE-MTR CONCEPT

by

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BACKGROUND

- NRU is 35 years old
  - components old, difficult to replace
  - large costs for refurbishing
  - uncertain regulatory requirements
- New and different needs for irradiations
  - interest in end-of-life phenomena
  - interest in life extension
  - support for advanced CANDU design
- AECL examining options
  - refurbish and upgrade NRU
  - design new reactor: MAPLE-MTR concept

AECL

AECL RESEARCH PROGRAMS

- CANDU Reactor Support
  - Fuel channel materials research
  - Fuel development
  - Reactor safety research

- Neutron Beam Applications
  - Condensed matter science
  - Neutron radiography
  - Applied neutron diffraction for industry

- Radioisotope Production
  - Facility to back-up MAPLE-X10

AECL

IRRADIATION REQUIREMENTS

CANDU Reactor Support

- CANDU fuel channels
  - CANDU design has array of fuel channels
  - co-axial pressure and calandria tubes
  - form main pressure retaining boundary
  - encloses fuel bundles
- Accelerated ageing research
  - define end of life for materials
  - develop new fuel channel materials
  - $2 - 3 \times 10^{18}$ $n \bullet m^{-2} \bullet s^{-1}$
  - $\sim 3 \times 10^{25}$ $n \bullet m^{-2}$ annual fluence

AECL

IRRADIATION REQUIREMENTS

CANDU Reactor Support

- Corrosion research
  - investigate hydriding phenomena
  - high temp. (300 C) & press. (10 MPa)
  - high radiation fields
  - D_2O water chemistry
  - $0.6 - 0.7 \times 10^{18}\ n{\bullet}m^{-2}{\bullet}s^{-1}$

- Pressure tube testing
  - proof-test new pressure tube designs
  - re-irradiate old pressure tubes
  - examine end of life phenomena
  - high temp. & press.
  - 0.2 to $0.3 \times 10^{18}\ n{\bullet}m^{-2}{\bullet}s^{-1}$

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IRRADIATION REQUIREMENTS
CANDU Reactor Support

- CANDU fuel bundle
  - short lengths of fuel elements
  - thin end plates
- Fuel element irradiations
  - single and multi-element tests
  - new fuel element designs
  - CANDU conditions, 300 C, 10 MPa
  - peak linear element ratings up to 70 kW/m
  - normal and off-normal tests
- Fuel bundle irradiations
  - fuel design proof-testing
  - bundle junction interactions
  - fuel burnup behaviour
  - peak linear element ratings up to 70 kW/m

AECL

IRRADIATION REQUIREMENTS

Other Applications

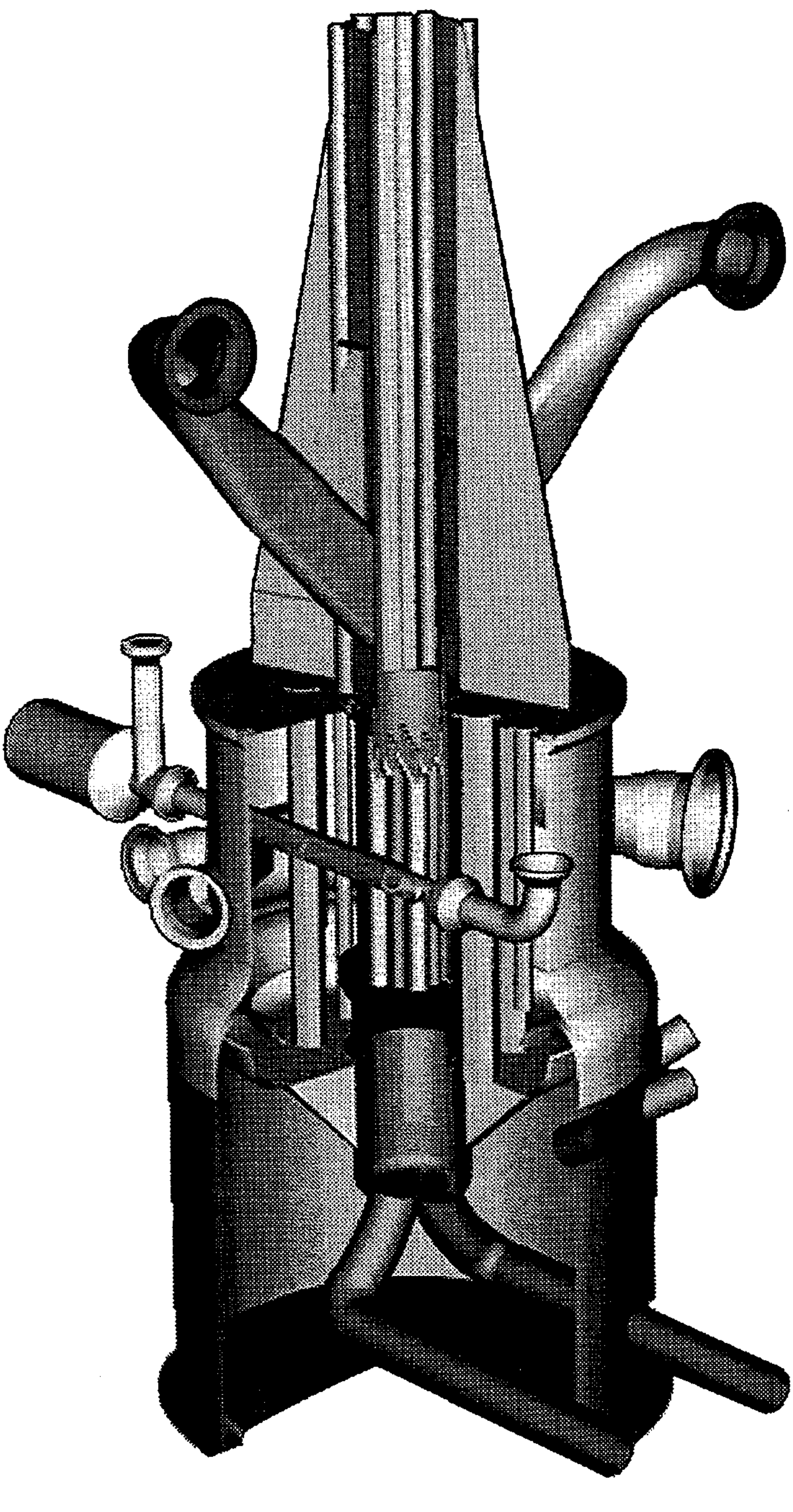
- Neutron beam applications
  - condensed matter science
  - neutron radiography
  - well thermalized flux
  - at least $2 \times 10^{18}\ n \bullet m^{-2} \bullet s^{-1}$
- Back-up radioisotope production
  - prudent to have back-up for MAPLE-X10
  - key short-lived isotopes: Mo-99 & I-125
  - compatibility with MAPLE-X10 targets

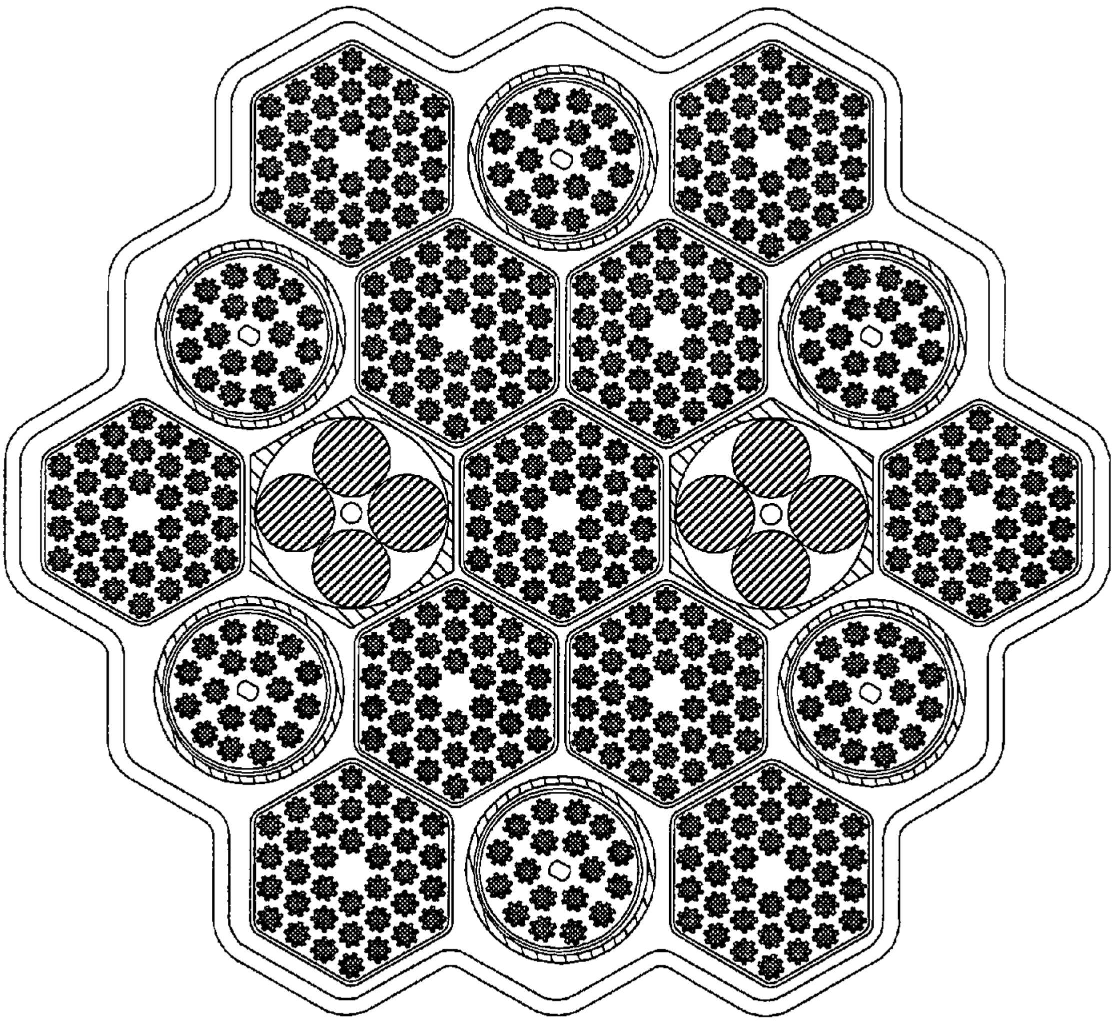
AECL

MAPLE-MTR CONCEPT

- Starts with MAPLE-X10 design
  - 63 L, 19-site core
  - H_2O-cooled & moderated
  - D_2O radial reflector
  - LEU rod fuel assemblies
  - cylindrical hafnium absorbers
- Adapted to multipurpose research reactor
  - reference 15 MW power, 250 kW/L power density
  - several in-core materials test sites
  - perturbed fast flux of about $1.4 \times 10^{18}\ n \bullet m^{-2} \bullet s^{-1}$
  - FN-rods for medium fast flux irradiations
  - full-diameter CANDU test loop sites
  - small diameter CANDU test loop sites
  - beam tubes

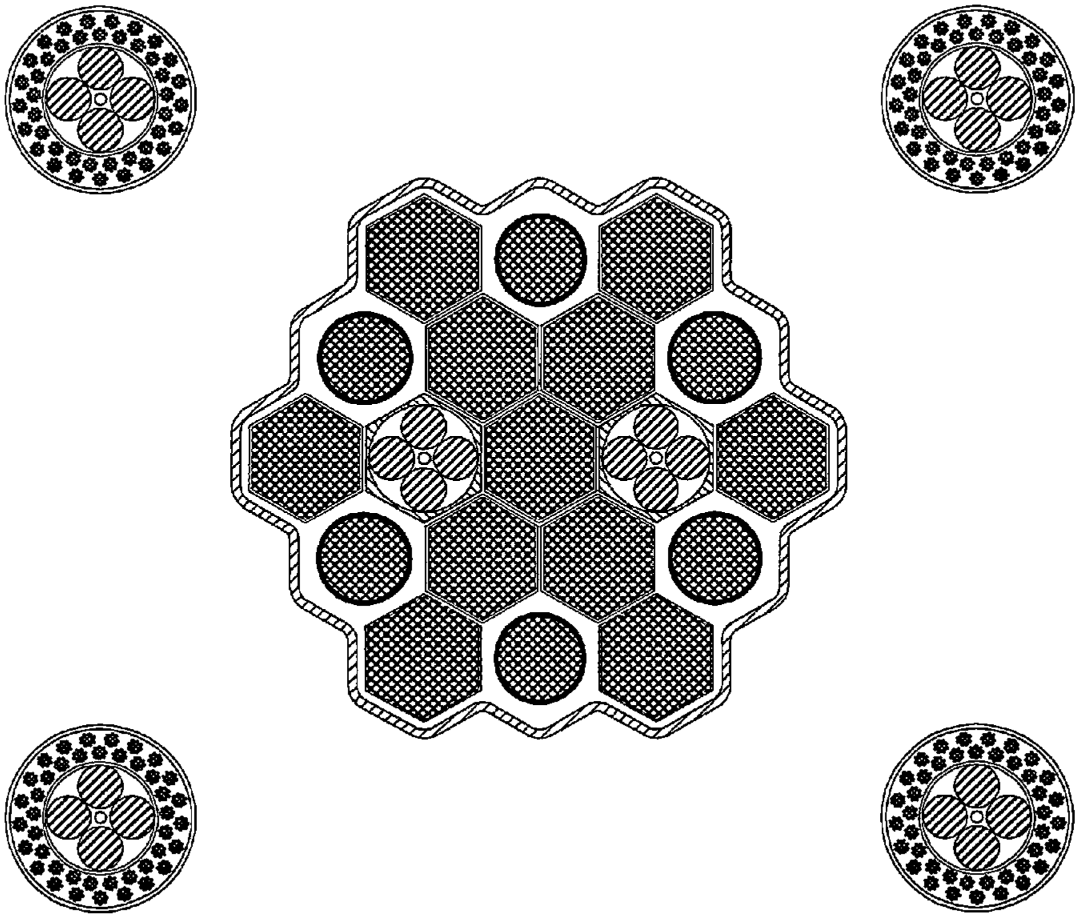
AECL





MAPLE-MTR CORE

IGORR-MAY18/92



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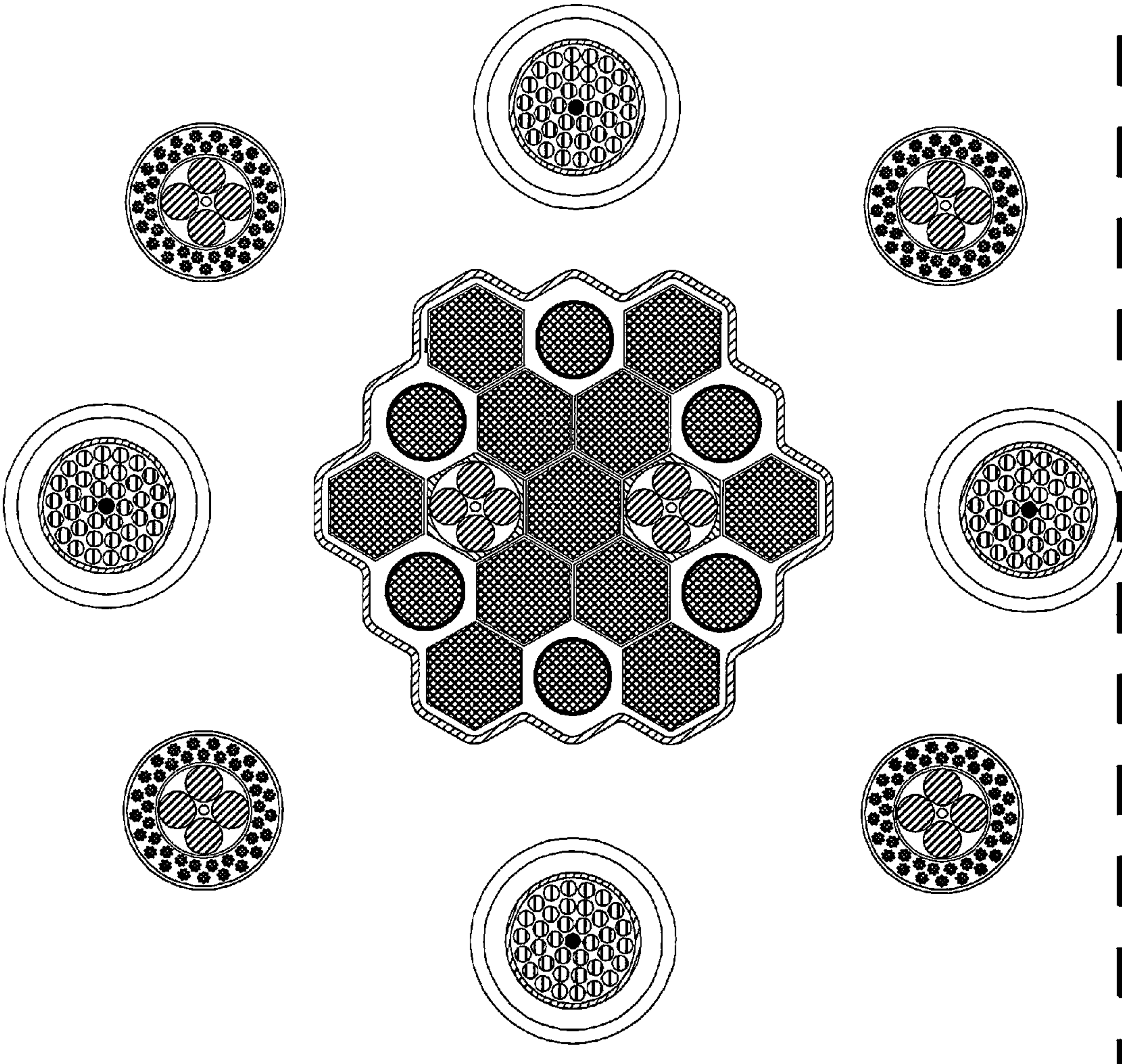
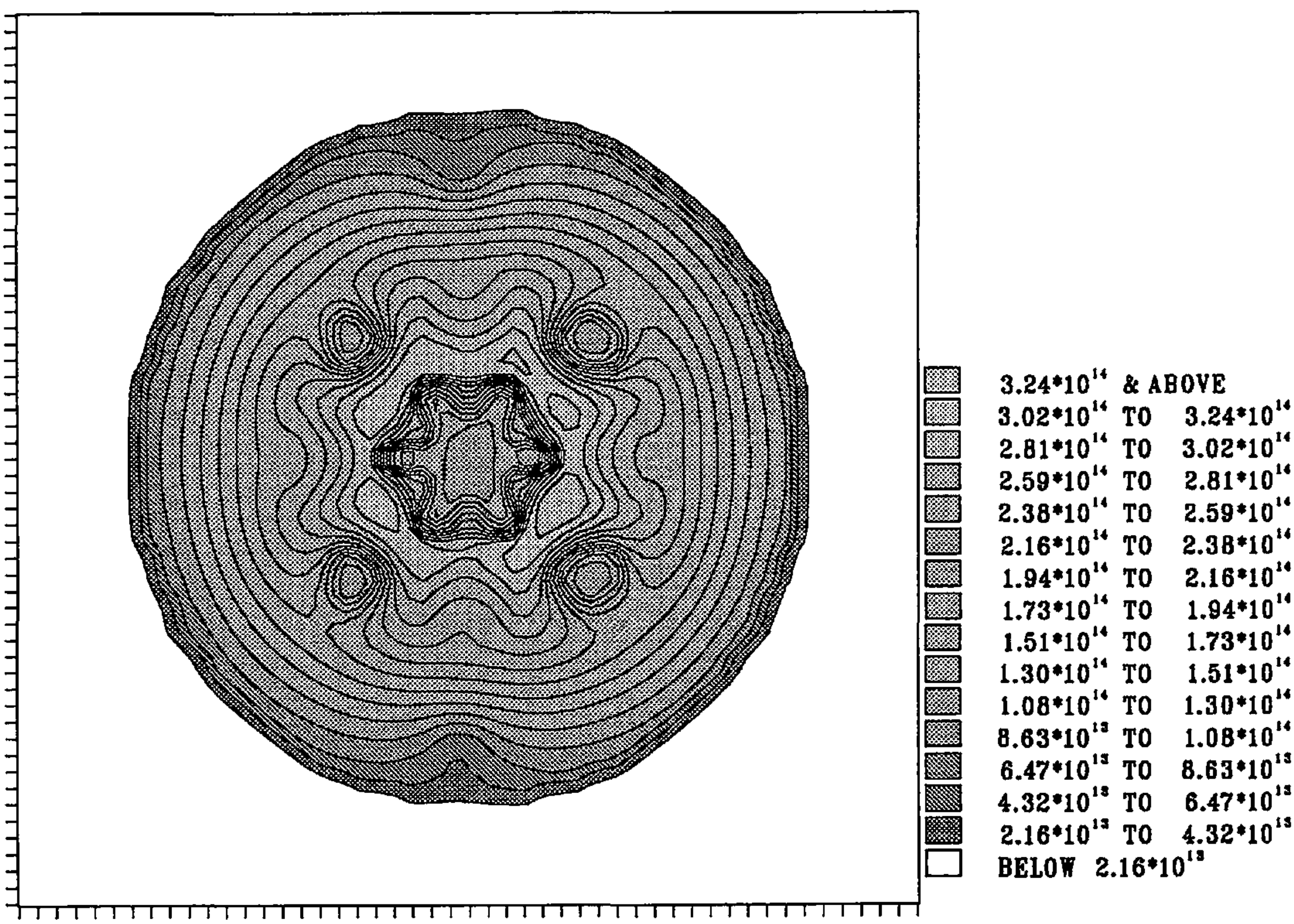
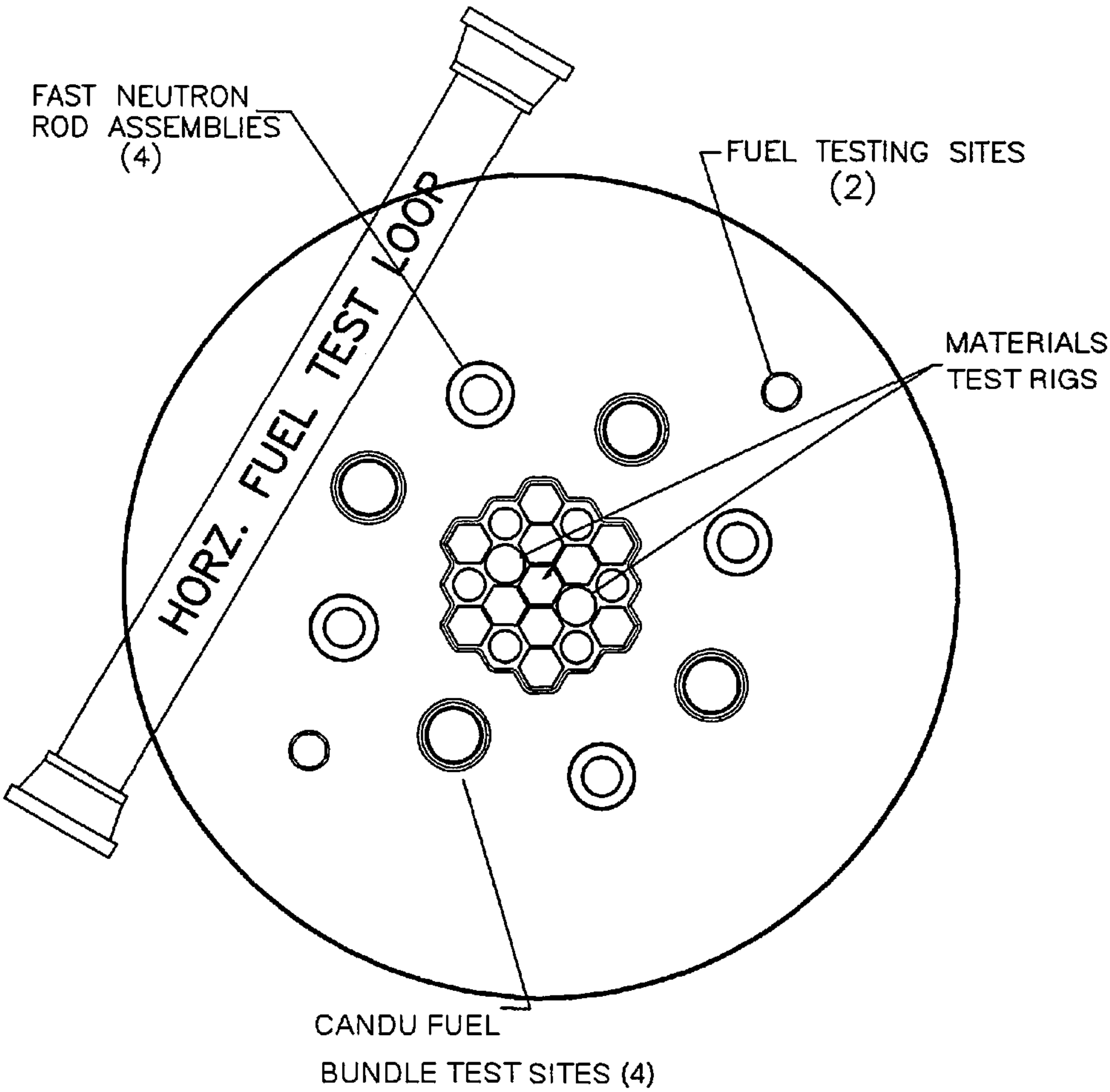


MMTR, EQUM CORE, FOUR CANDU FTLS, FOUR FN SITES
X−Y Flux, Z= 80.20 Group 5





FAST NEUTRON ROD ASSEMBLIES (4)
FUEL TESTING SITES (2)
HORZ. FUEL TEST LOOP
MATERIALS TEST RIGS
CANDU FUEL BUNDLE TEST SITES (4)

THE ADVANCED NEUTRON SOURCE DESIGN - A STATUS REPORT*

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Abstract

The Advanced Neutron Source (ANS) facility is being designed as a user laboratory for all types of neutron-based research, centered around a nuclear fission reactor (D_2O cooled, moderated, and reflected), operating at approximately 300 MW_{th}. Safety, and especially passive safety features, have been emphasized throughout the design process.

The design also provides experimental facilities for neutron scattering and nuclear and fundamental physics research, transuranic and other isotope production, radiation effects research, and materials analysis.

Design Basis

The basic reactor design concept is derived from the technical objectives (Table 1) and the project's philosophy of minimizing technical risks and safety issues by relying only on known technology to meet the minimum design criteria.

The main scientific justification for the project, expressed by the National Academy Committee on major materials facilities[1] is the U.S. need for a world-class neutron scattering facility. The essential requirements are for a very high flux of thermal neutrons in a region that is accessible to beam tubes and with space for one or more cryogenic moderators large enough to remoderate the thermal neutrons to much lower energies, producing so-called cold neutrons.

Design Concept

It is clear, to meet those requirements that the reactor must produce a large number of fission neutrons, i.e., it must have enough power. In addition, the requirements dictate certain other design features (Table 2) which are very different from power reactors and have implications (many of them positive) for the safety analysis of the reactor (Table 3).

The annular, involute geometry of the fuel plates is copied from the High Flux Isotope Reactor (HFIR) and the reactor at the Institut Laue-Langevin (ILL) at Grenoble. The aluminum clad mixture of U_3Si_2 fuel particles and aluminum powder has been developed and extensively tested

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by the Reduced Enrichment Research and Test Reactors (RERTR) program although more tests at higher temperatures and burnup rates are underway or planned by the ANS Project. The short heated length of the HFIR core design and the long neutronic length of the ILL core have been combined (Fig. 1). Nominal specifications of the reference conceptual design are given in Figure 2 and in Table 4, which also includes the major thermal-hydraulic parameters of the core assembly.

The core cooling system design concept, which evolved through iteration among the design, safety, and R&D groups, incorporates many passive safety features (Fig. 3), as does the reactor system coolant components (Fig. 4).

Core Pressure Boundary Tube

The primary coolant pressure boundary in the region of the core is called the Core Pressure Boundary Tube (CPBT). It fits fairly closely around the upper fuel element (Fig. 5), and is made from aluminum 6061, for which ASME Section 3 Code Approval is being sought. Aluminum 6061 is chosen because of its high thermal conductivity and relatively low neutron absorption, and because there is extensive experience with it as a structural material in U.S. research reactors (for example, in the HFIR and the HFBR).

The fracture mechanics properties of aluminum 6061 are such that one cannot, as with steel, take any credit for leak-before-break detection nor, unless unacceptably thick sections are used, can one completely rule out flaw growth in components that are subject to tensile stress during operation. Thanks to the primary coolant circuit safety features illustrated in Figs. 3 and 4, we expect the core to survive without damage a large break in the CPBT downstream of the fuel elements, but not upstream. Therefore, a pressure vessel with an integral guard pipe concept has been adopted. In this design (Fig. 6), a continuous outer tube is the pressure boundary in normal operation. An inner guard tube is separated from it by a narrow, annular cooling channel. Holes or slots at the bottom of the lower guard tube would restrict the coolant loss rate to an acceptable value following failure of the outer tube. The flow rate in the cooling annulus is limited by a restrictor placed at the end of the upper fuel element: the flow rate must (7 m/s alongside the upper element sideplate, 4.2 m/s elsewhere) be high enough to cool the tubes, but low enough to provide a Bernoulli pressure rise that keeps the lower guard tube in compression (i.e., with no tendency for flaw growth) during normal operation.

Reactivity Control

The reactivity control system includes three hafnium rods in the central hole region, driven together from below by a mechanical system based on the successful HFIR design. These three rods can also be scrammed, with high acceleration, by individual springs that are individually released by magnetically held, fail safe mechanical latches, a design also based on the HFIR (Fig. 7). This system alone is capable of meeting the reactor shutdown criteria even if one rod fails to scram, and of shutting down the reactor (although by a smaller margin) even if two rods fail.

A second independent and diverse shutdown system is provided by eight rods in the reflector tank outside the CPBT. This set of rods is driven from above (so that, for example, a single missile will not damage both drive units). These outer rods are reset and latched hydraulically, and driven in for a scram by a combination of hydraulic and spring forces (Fig. 8).

In addition, burnable poison (boron) in the core controls the excess reactivity in fresh cores, reducing the negative reactivity that must be provided by the moveable control rods.

Table 5 outlines the reactivity balance of the core and other major contributors.

Reactor Internals

Among the components inside the core pressure boundary tube are the core support structure; the inner control rod guides, supports, linkages, stops, position switch actuators and drive springs; the flow screens at the entrance to the fuel elements; the irradiation facilities for materials testing and transuranium production; the instrument lead assemblies for the materials testing capsules; and the central hole flow restrictor.

Reflector Tank Internals

There are major items of equipment in the reflector tank which can influence the core reactivity and the thermal neutron flux. Table 6 is a preliminary listing of items included in the conceptual design.

Building Design

The overall budget design (Fig. 9) is also an important contributor to the safety, security, and accessibility of the scientific and reactor facilities. It provides physical barriers, formed from the massive containment and shielding structures, between sensitive or higher risk areas on the one hand, and the laboratories, offices, and experimental space of the users on the other.

IN-CORE MATERIALS IRRADIATION FACILITIES

<table>
<tr><th rowspan="2">Parameter</th><th colspan="2">Goals</th><th rowspan="2">ANS</th></tr>
<tr><th>Small specimens</th><th>Larger specimens</th></tr>
<tr><td>Capsule dia, mm</td><td>16</td><td>46</td><td>48</td></tr>
<tr><td>Capsule length,mm</td><td>500</td><td>500</td><td>500</td></tr>
<tr><td>Total no. of positions</td><td>10</td><td>10</td><td>10</td></tr>
<tr><td>No. of instrumented positions</td><td colspan="2">8</td><td>5</td></tr>
<tr><td>Fast flux, 10^{19} $m^{-2}.s^{-1}$</td><td>≥1.4</td><td>≥0.5</td><td>2.9</td></tr>
<tr><td>Fast/thermal ratio</td><td>≥0.5</td><td>≥0.3</td><td>1.1</td></tr>
<tr><td>Axial flux gradient over 200 mm, %</td><td colspan="2">≤30</td><td>14</td></tr>
<tr><td>Damage ratio (dpa/y in stainless steel)</td><td>≥30</td><td>≥8</td><td></td></tr>
<tr><td>Nuclear heating rate (w/g in stainless steel)</td><td>≤54</td><td>≤15</td><td>78(max)/72(av.)</td></tr>
</table>

Table 1. Advanced Neutron Source Project technical objectives

- To design and construct the world's highest flux research reactor for neutron scattering
  - -5 to 10 times the flux of the best existing facilities
- To provide isotope production facilities that are as good as, or better than, the High Flux Isotope Reactor (HFIR)
- To provide materials irradiation facilities that are as good as, or better than, the HFIR.

Table 2. Design features and scientific requirements

Feature	Relationship to Scientific Requirements
Heavy Water Coolant	Reduces (compared with light water) the moderation of fission neutrons within the fuelled region, thus increasing the number of neutrons thermalized outside the core where they are potentially available to the beam tubes.
Small core volume	Reduces the surface area through which the reflected, thermalized neutrons must pass, thereby increasing the flux. Also reduces the moderation of fission neutrons within the fuelled region.
Heavy Water Reflector Region	Low absorption of thermalized neutrons, thus increasing the number potentially available for extraction in beam tubes or guides. Provides a large volume of high thermal neutron flux in which the cold moderators can be accommodated outside the core. Can easily accommodate complex shapes of experimental equipment, and is not subject to radiation damage.
Low temperature reflector	Thermalize neutrons at lowest practical temperature coolant (for thermal neutron beams and as source of neutrons for the cryogenic moderators).
High power (for a research reactor, much lower than power reactors)	Produces a large number of fission neutrons for subsequent moderation to thermal energies.
Large containment building	Provides floor space for experimental stations on thermal and hot-neutron beam tubes.

Table 3. Some features of the ANS research reactor with significant safety implications compared with typical tower reactors

Feature	Comment
Low thermal power level	~ 300 MW compared with ~ 3000 MW means less stored and circulating energy
Low fission product inventory at end of cycle (smaller core contains much less fuel)	~ 6 kg compared with ~ 100 kg means much lower source term
Small core	~ 100 kg total mass compared with ~ 100,000 kg means less chemical energy available for release
Lower core and reflector coolant temperature	<100°C compared with ~ 325°C, so that coolant water would not flash into steam during a pressure loss
Lower primary coolant pressure	~ 3 MPa compared with ~ 15 MPa means much less stored energy
High degree of containment	Containment bigger than a typical power reactor, to provide space for neutron beam experiments, but 10 times lower power level
Heavy water moderator	Longer neutron lifetime means slower reactivity transients
High power density (from high power and small core)	~ 5 MW/Litre compared with ~ 5 kW/Litre means more rapid heat up and dryout possible.
Occupied containment	Thermal neutron beam experiments must be located and operated at positions close to the reactor
High coolant velocity	High coolant velocity to accommodate high power density leads to very large flow forces on fuel plates and reactor internals
Short core life	The high power density leads to a short core life, with more frequent opportunities for refueling accidents.

Table 4. ANS reactor core nominal specifications

Quantity & Unit/Item	Reference Value/Material
Heat deposited in fuel, MW	303
Fission power, EOC, MW(f)	330
Core life, d	17
Core active volume, L	67.6
Core dimensions	See figure 2
Fuel form	U_3Si_2
Fuel enrichment, %	93
Fuel matrix	Al
Vol. % of fuel in fuel meat	11.2
No. of plates in upper element	432
No. of plates in lower element	252
Mass of ^{10}B, gm (BOC)	13
Fuel plate thickness, mm	1.27
Aluminum clad thickness, mm	0.254
Coolant channel gap, mm	1.27
Coolant	D_2O
Heated length, mm	507
Coolant velocity in core, m/s	25
Inlet pressure (in plenum), MPa	3.2
Core Inlet temp, °C	45
Annular gap in CPBT, mm	5
Coolant velocity in Annular gap, m/s	7
Equivalent break diameter of inner CPBT holes, mm	76 (lower) 51 (upper)

CDW 5/8/92

Table 5. Conceptual reactivity balance

Potential Reactivity of the core at 20°C (BOC)	31,070 pcm
Temperature effect at full power	- 459 pcm
Core pressure boundary tube assembly	-5,150 pcm
Irradiation facilities	-2,000 pcm
Beam tubes	-3,820 pcm
Cold sources	- 470 pcm
Hot source	TBD
Other experimental facilities in reflector tank	-2,670 pcm
Central control rods (3)	-20,390 pcm[1]
Outer shutdown rods (8)	-15,330 pcm[2]

---

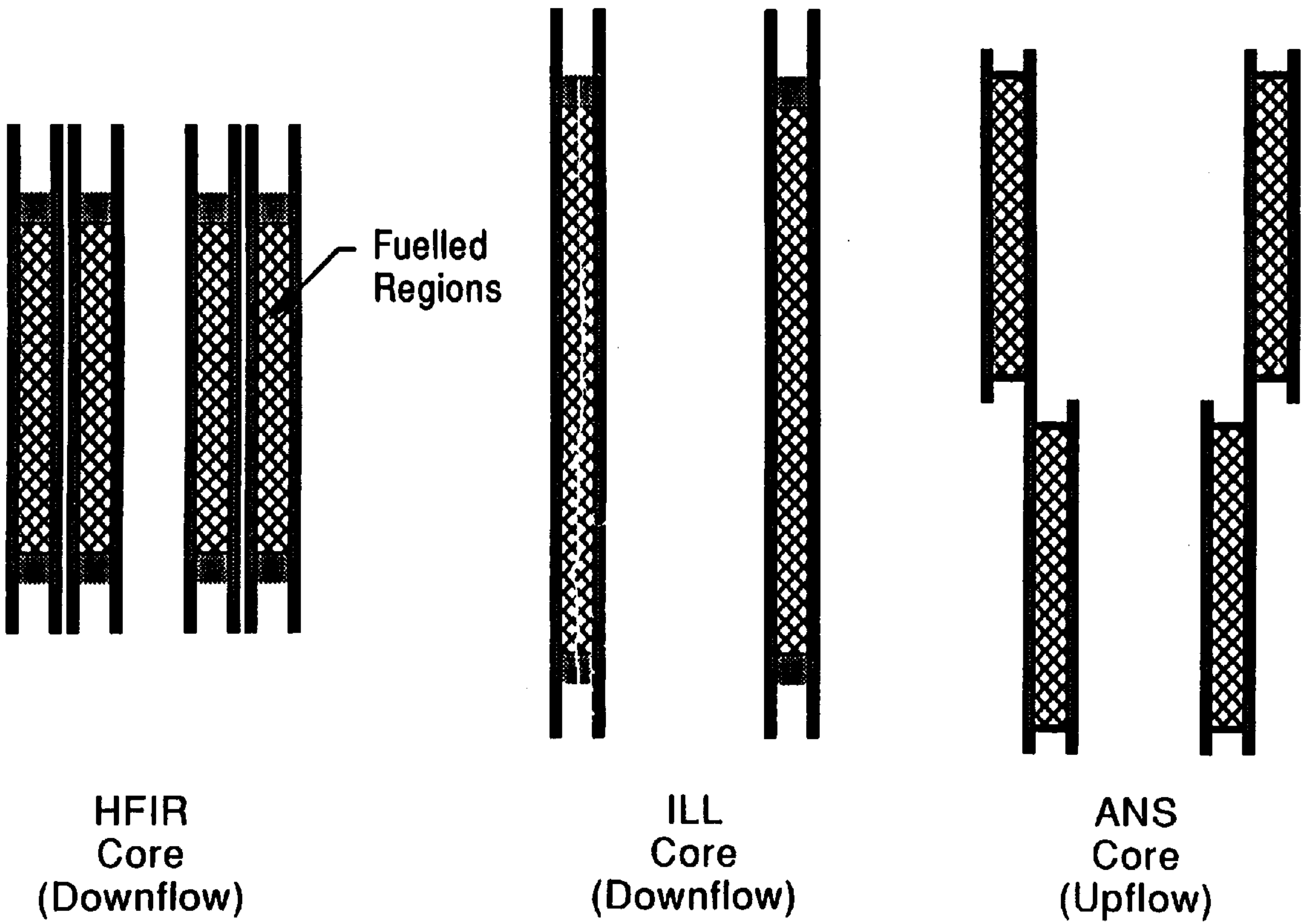
[1]) With outer rods fully withdrawn

[2]) With inner rods fully withdrawn

Table 6. Reflector tank internals

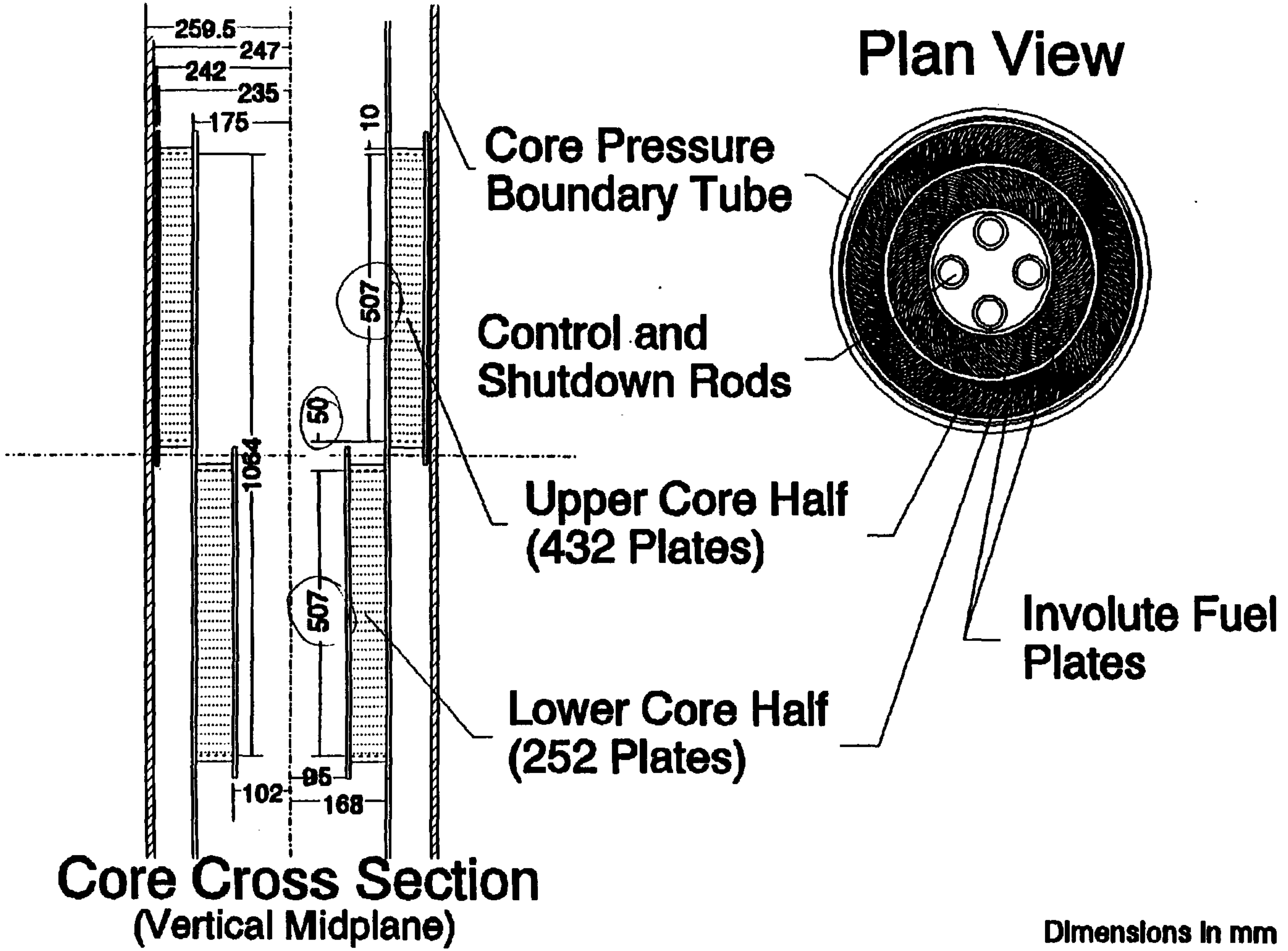
Outer shutdown rod assembly
Hydraulic lines to outer shutdown rods
Tangential thermal beam tubes (7)
Thermal through beam tube
Slant thermal beam tube
Hydraulic rabbit tubes for light isotope production (3)
Hydraulic rabbit tube for transuranium production
Pneumatic rabbit tubes for analytical chemistry (5)
Isotope production vertical holes (4)
Slant irradiation tubes (2)
Cold source thimbles (2)
Hot source thimble
CPBT fasteners and seals

The ANS Reactor Core Design Combines The Best Features Of Earlier Designs



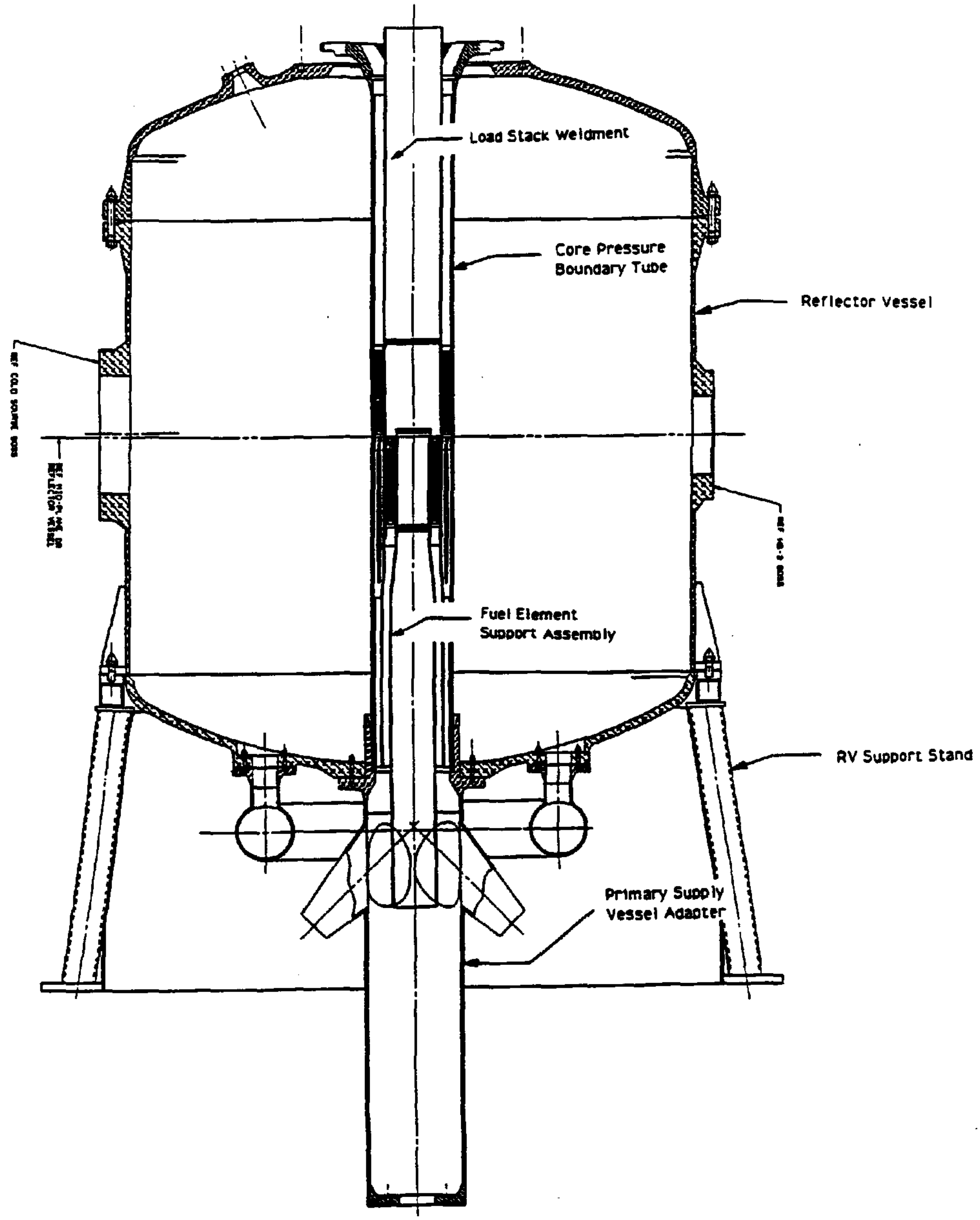
Side by Side Comparison of ILL, HFIR, ANS Cores

Fig. 1



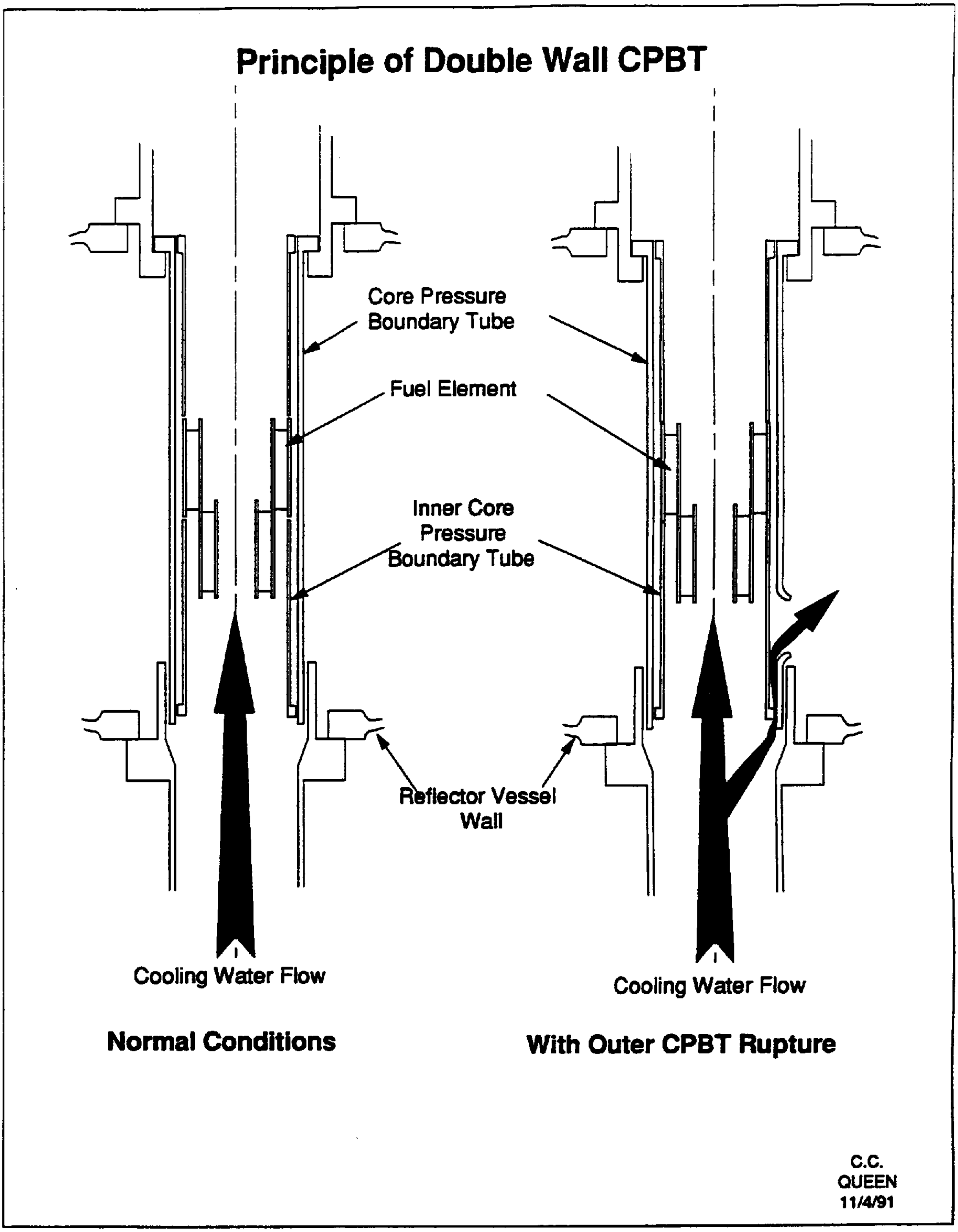
Dimensioned Core Drawing

Fig. 2



Core Pressure Boundary Tube

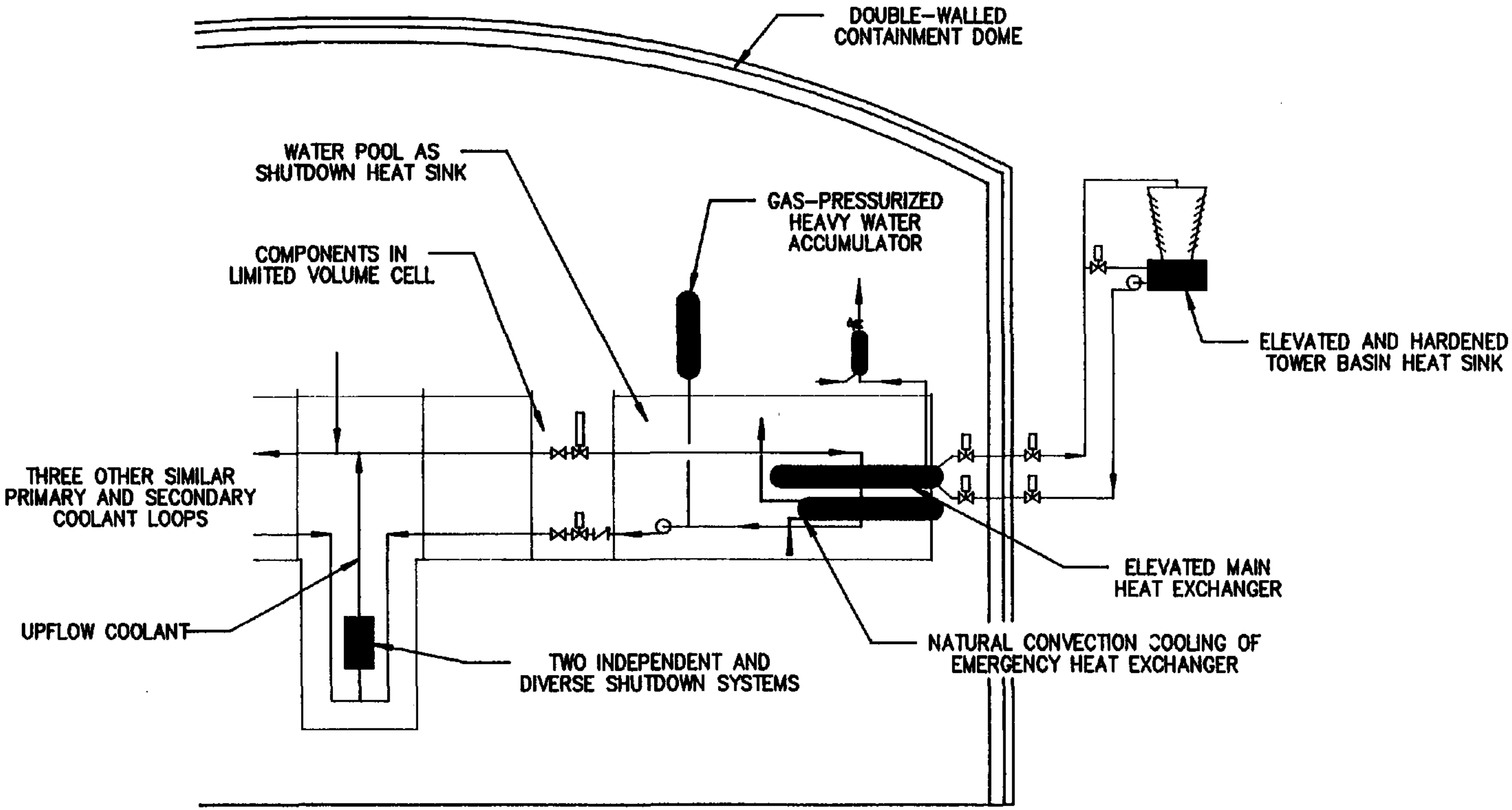
Fig. 5



Principle of Double-Wall CPBT

Fig. 6

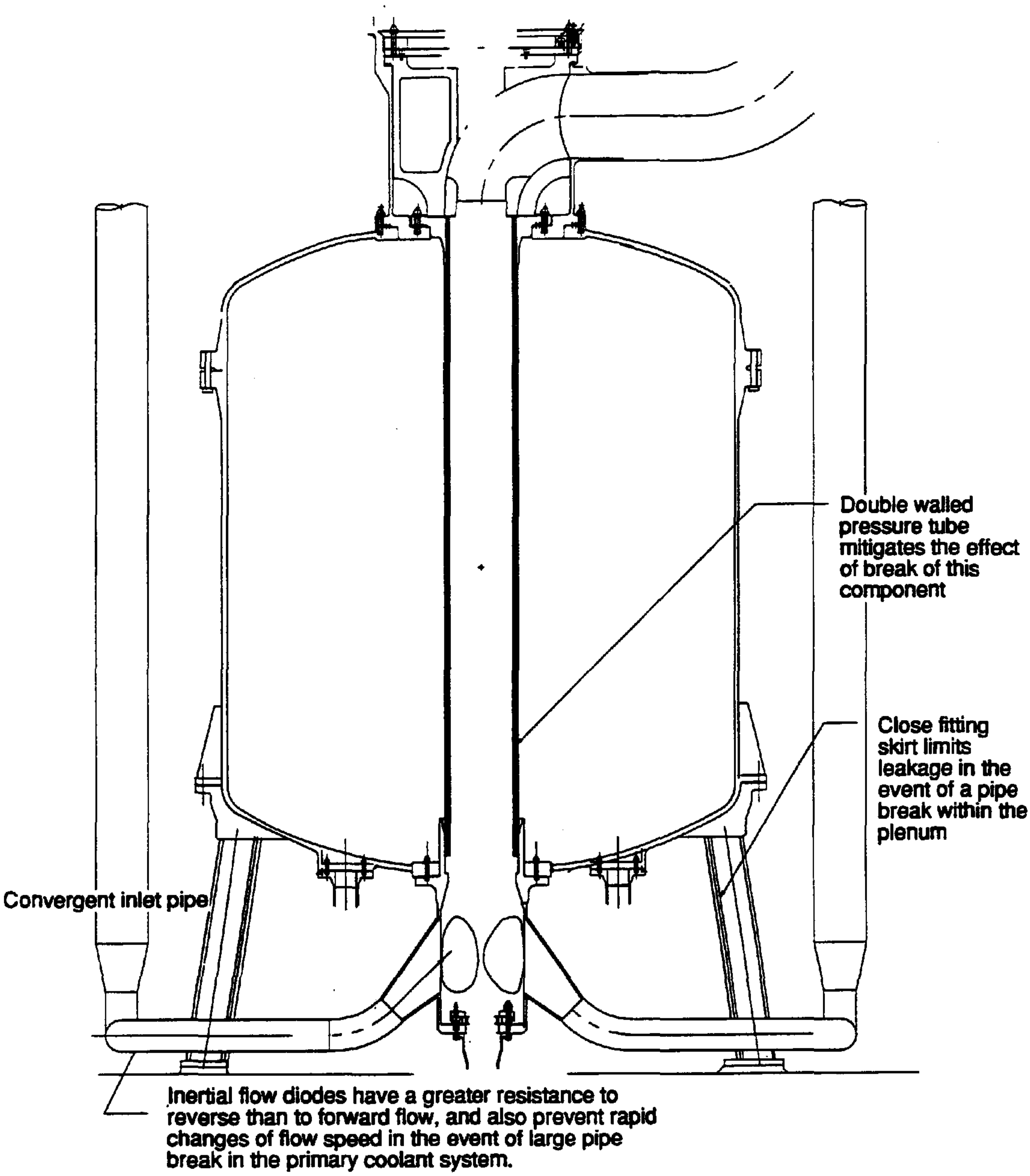
SOME SAFETY FEATURES OF THE ADVANCED NEUTRON SOURCE REACTOR DESIGN



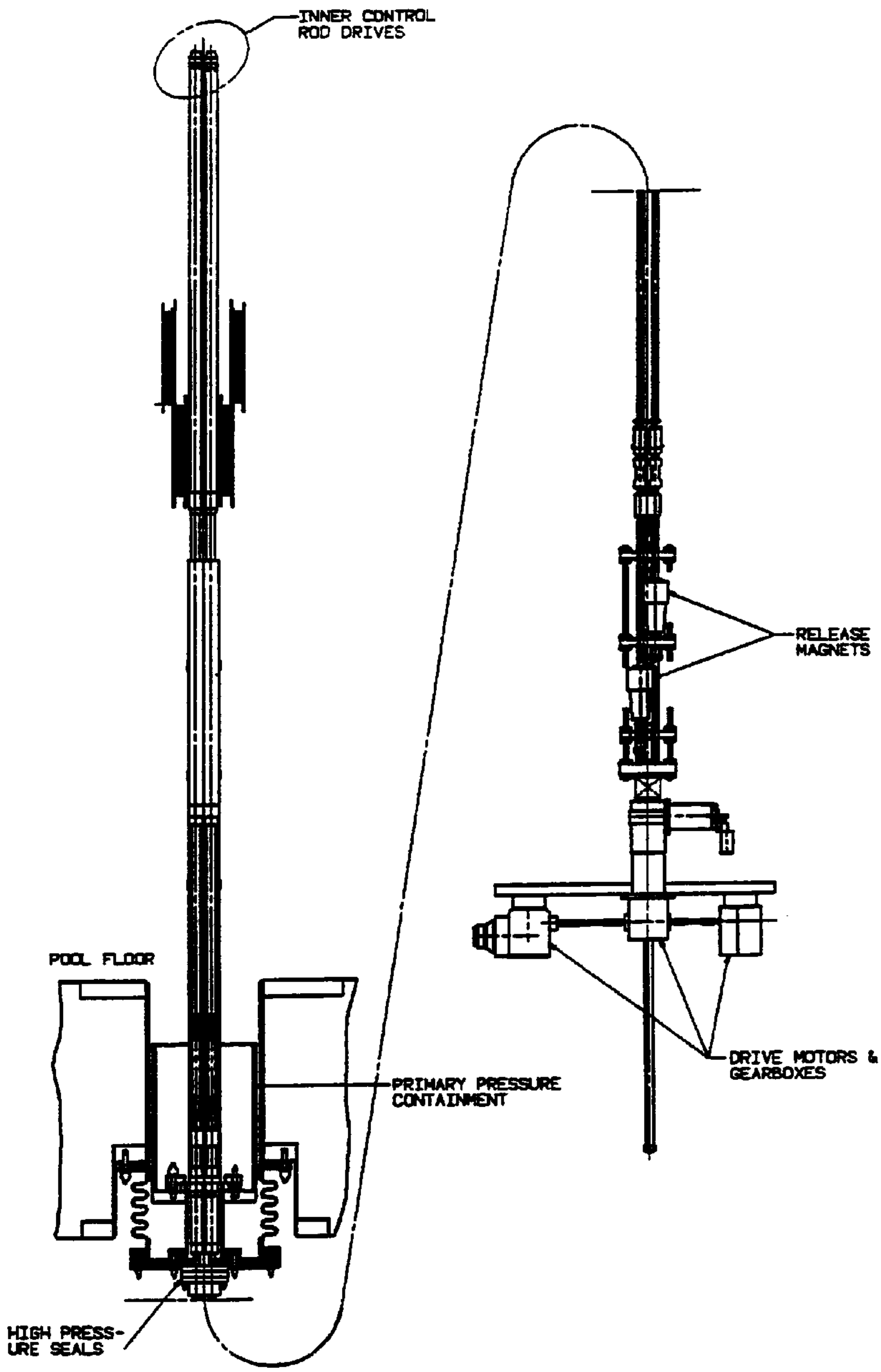
Some Safety Features/Reactivity

Fig. 3

Fig. 4

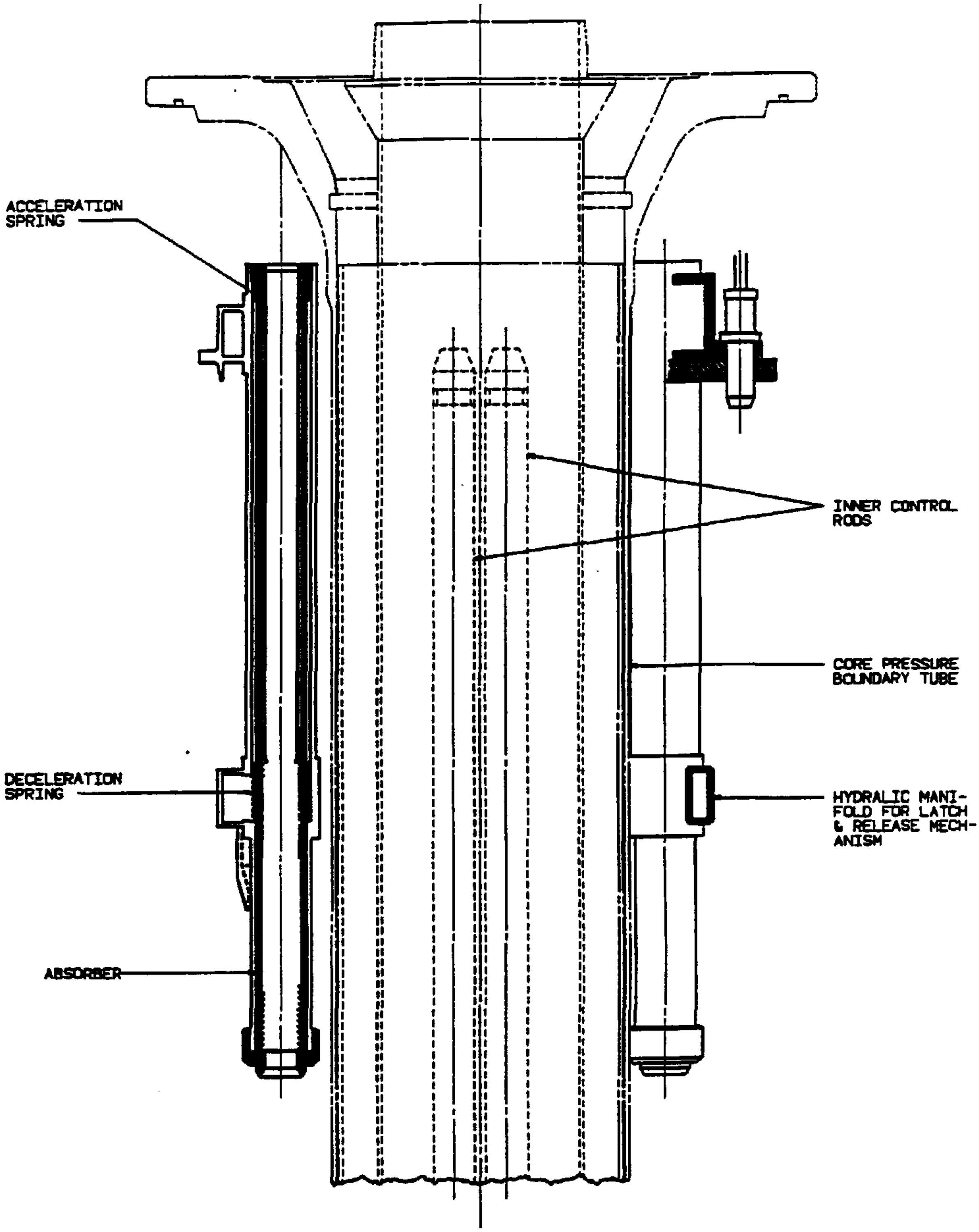


Some Safety Features of the Reactor System Coolant Components



Inner Control Rod Drives

Fig. 7



Outer Shutdown System

Fig. 8

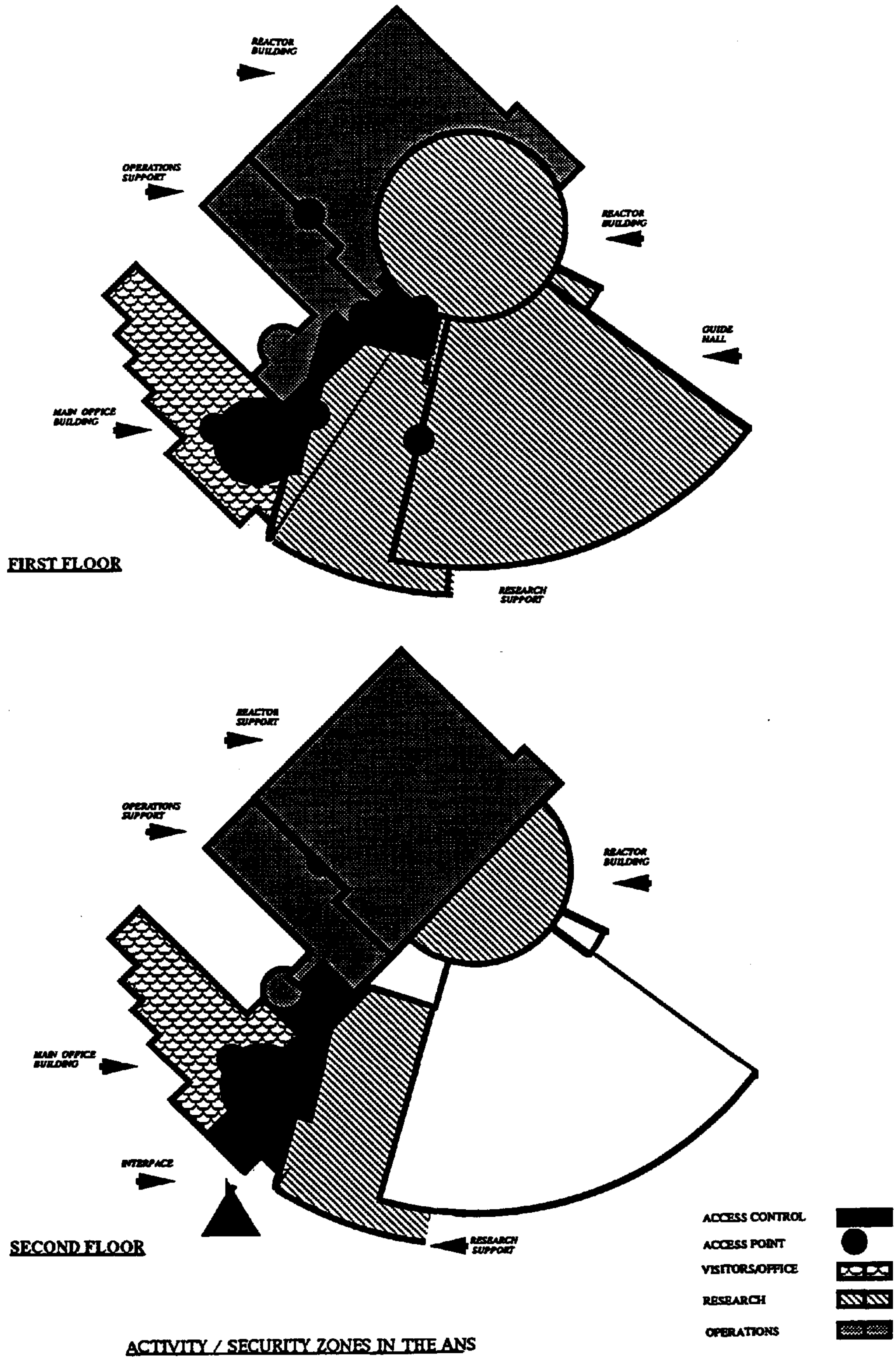


Fig. 9 Building Design

THE ADVANCED NEUTRON SOURCE DESIGN - A STATUS REPORT

C. D. West
Oak Ridge National Laboratory

presented to the
2nd Meeting of the
International Group on Research Reactors
(IGORR-II)

Saclay, France

May 18, 1992

PROJECT TECHNICAL OBJECTIVES

- To design and construct the world's highest flux research reactor for neutron scattering

  — 5-to-10 times the flux of the best existing facilities

- To provide isotope production facilities that are as good as, or better than, the High Flux Isotope Reactor (HFIR)

- To provide materials irradiation facilities that are as good as, or better than, the HFIR

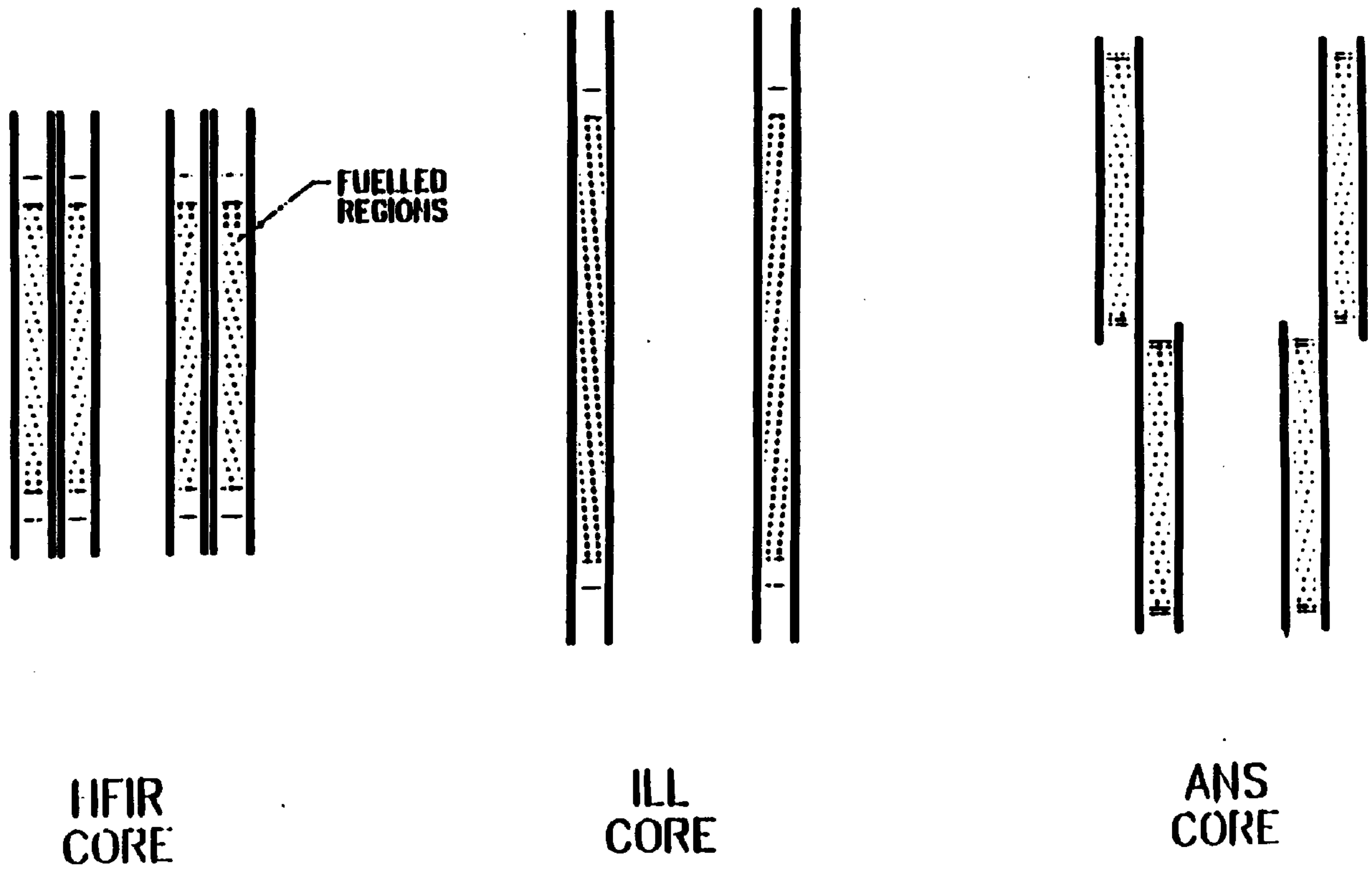
LIST OF PARTICIPANTS ON THE ANS PROJECT

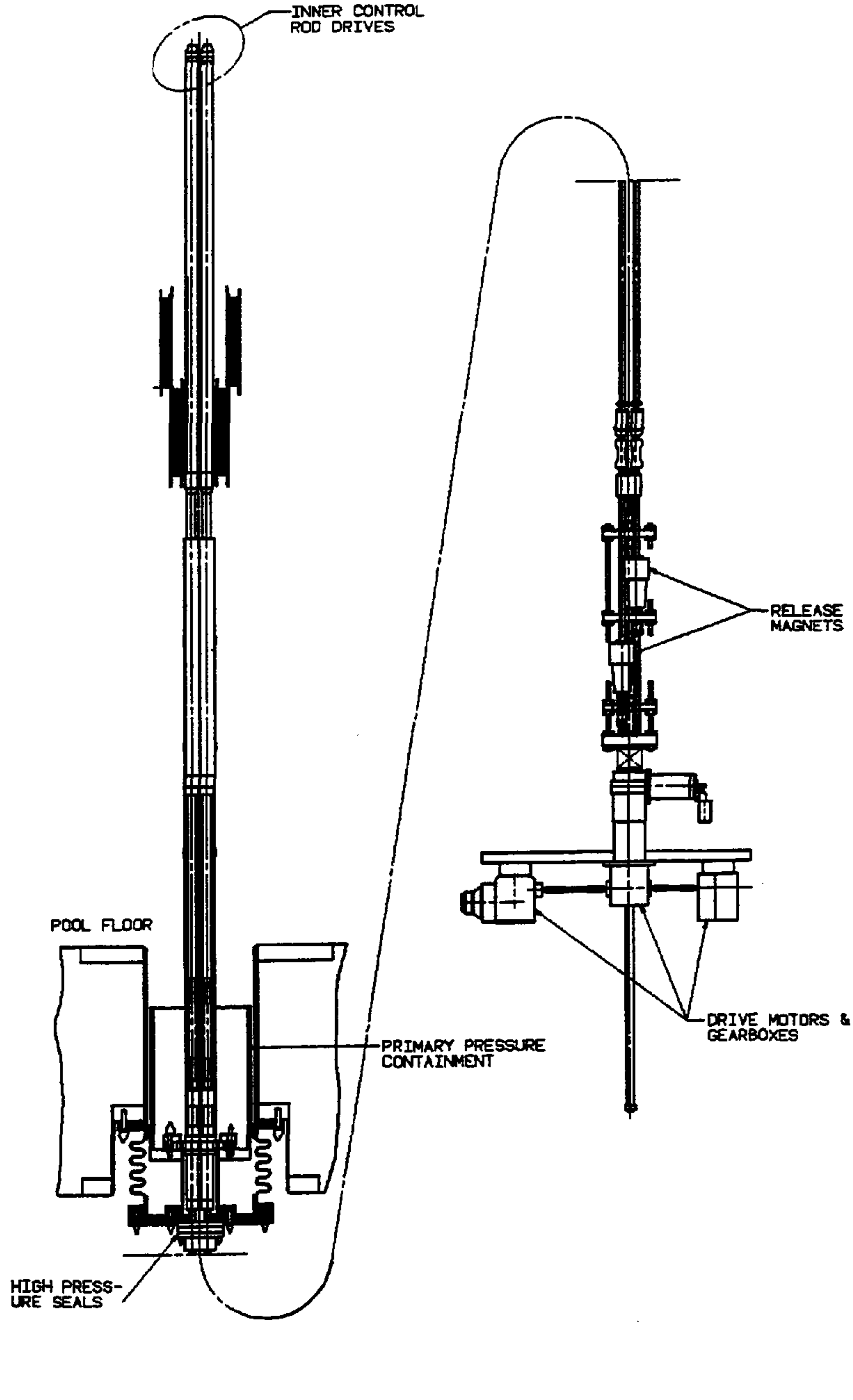
Industry

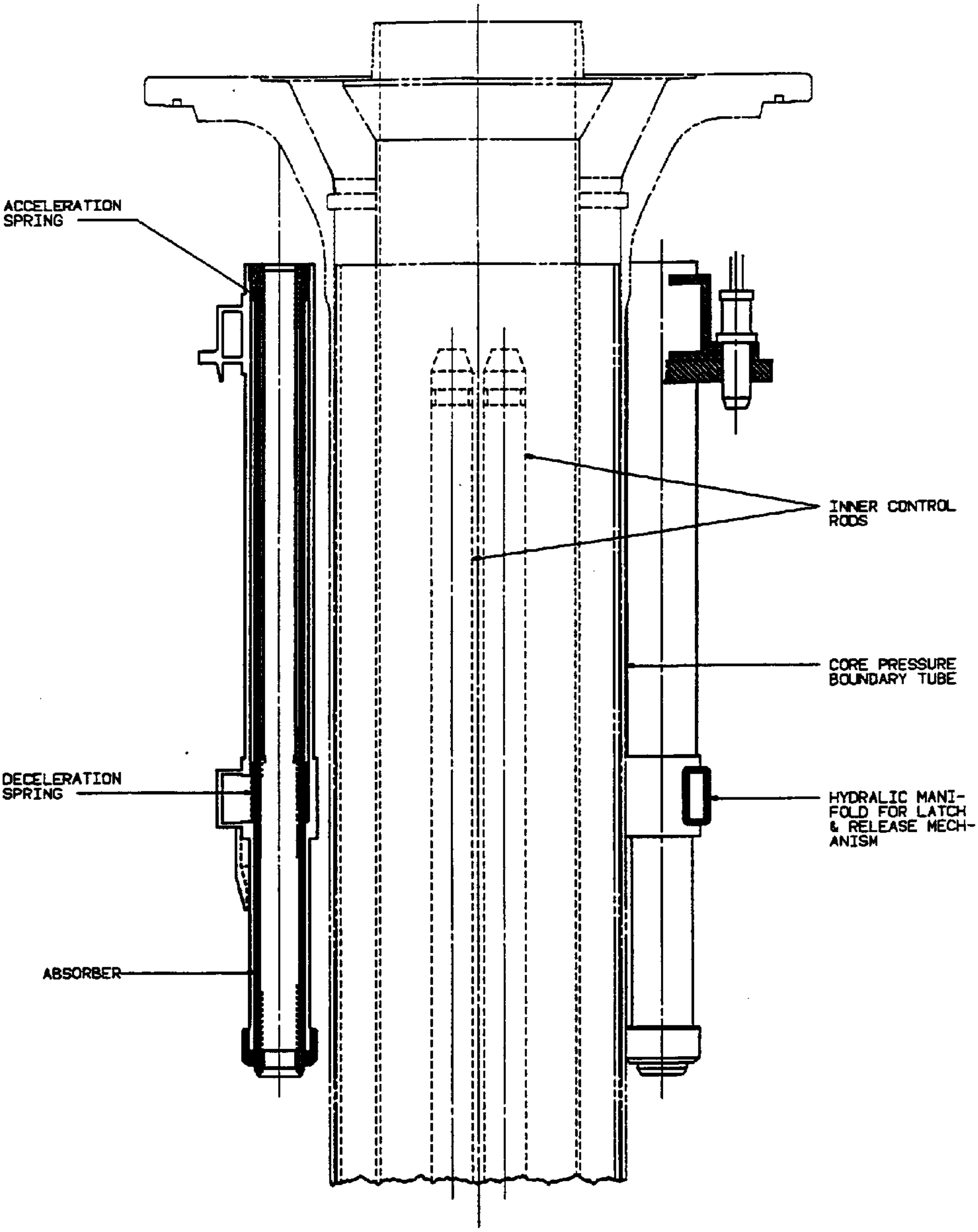
Air Products
Argonne National Laboratory
Atomic Energy of Canada, Ltd.
Babcock & Wilcox
Brookhaven National Laboratory
DRS
Gilbert/Commonwealth, Inc.
Idaho National Engineering Laboratory
Interatom GmbH
Law Engineering
National Institute of Standards and Technology
NUS
PLG
SAIC
Sandia National Laboratory
Australian Nuclear Science & Technology Organization
Japan Atomic Energy Research Institute

Universities

Kinki University
Kyoto University
Kyushu University
McGill University
Nagoya University
Osaka University
Tohoku University
University of Tennessee
University of Virginia
University of Wisconsin
Technical University of Munich
Massachusetts Institute of Technology
Georgia Tech
University of Southern California


FUELLED REGIONS
HFIR CORE
ILL CORE
ANS CORE


INNER CONTROL ROD DRIVES
RELEASE MAGNETS
POOL FLOOR
DRIVE MOTORS & GEARBOXES
PRIMARY PRESSURE CONTAINMENT
HIGH PRESS-URE SEALS


ACCELERATION SPRING
INNER CONTROL RODS
CORE PRESSURE BOUNDARY TUBE
DECELERATION SPRING
HYDRALIC MANI-FOLD FOR LATCH & RELEASE MECH-ANISM
ABSORBER

Core Reactivity Balance at BOC

Potential reactivity of core at 20°C	31,070 pcm
Temperature effect at full power	-460 pcm
Boron-10 burnable poison (13g)	-8,470 pcm
CPBT assembly	-5,150 pcm
Irradiation facilities (in-core)	-2,000 pcm
Beam tubes	-3,820 pcm
Cold sources	-470 pcm
Hot sources	TBD
Other experimental facilities in reflector tank	-2,670 pcm
Total balance without control	+8,030 pcm
Three central control rods	
inserted to ~130mm above the core midplane	-8,030 pcm for a balance of 0 pcm
fully inserted	-20,390 pcm for a balance of -12,360 pcm
Eight reflector shutdown rods inserted	
with central rods removed from the model	-15,300 pcm for a balance of -7,300 pcm
with central rods at failed to scram position	TBD
with central rods fully inserted	TBD

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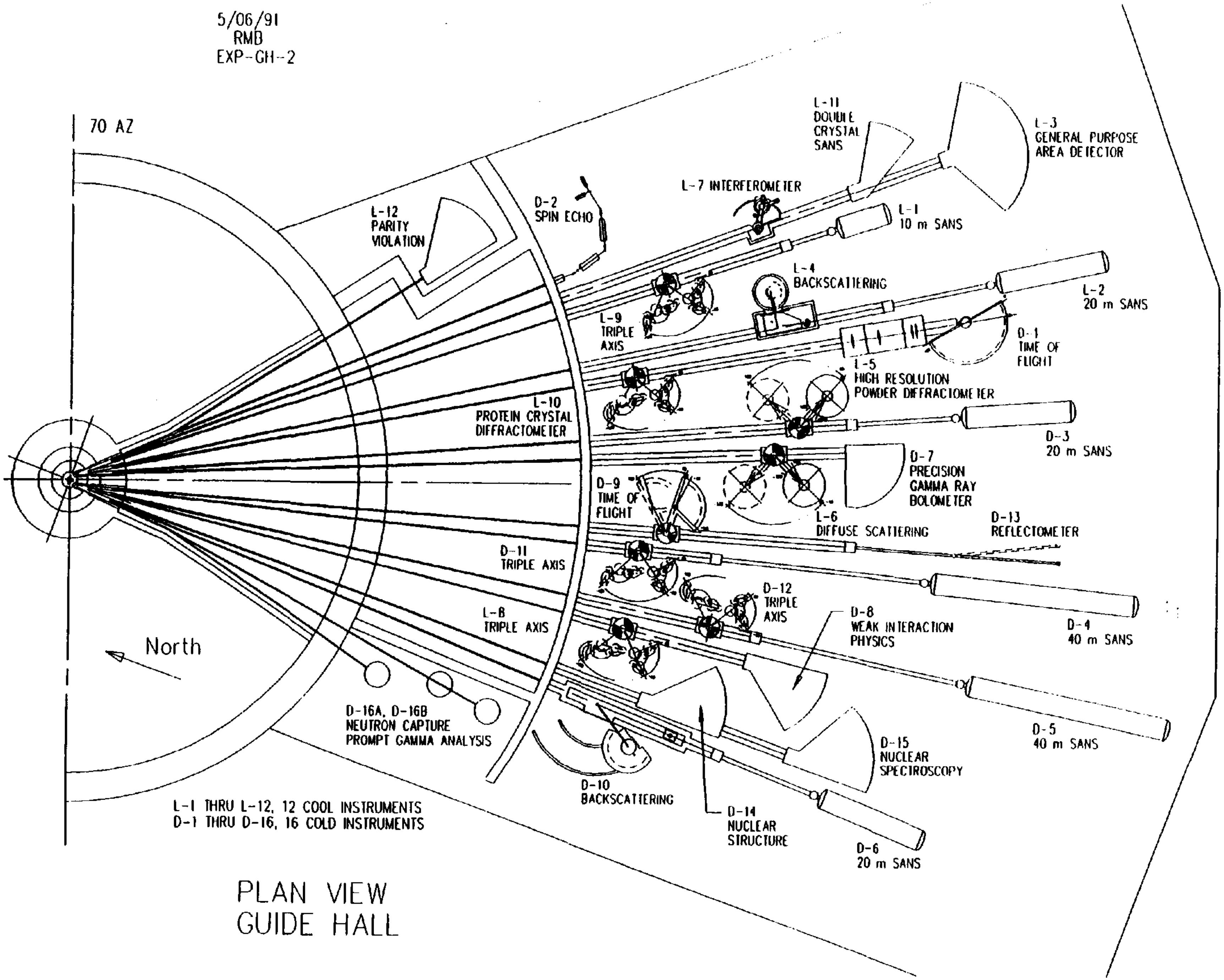
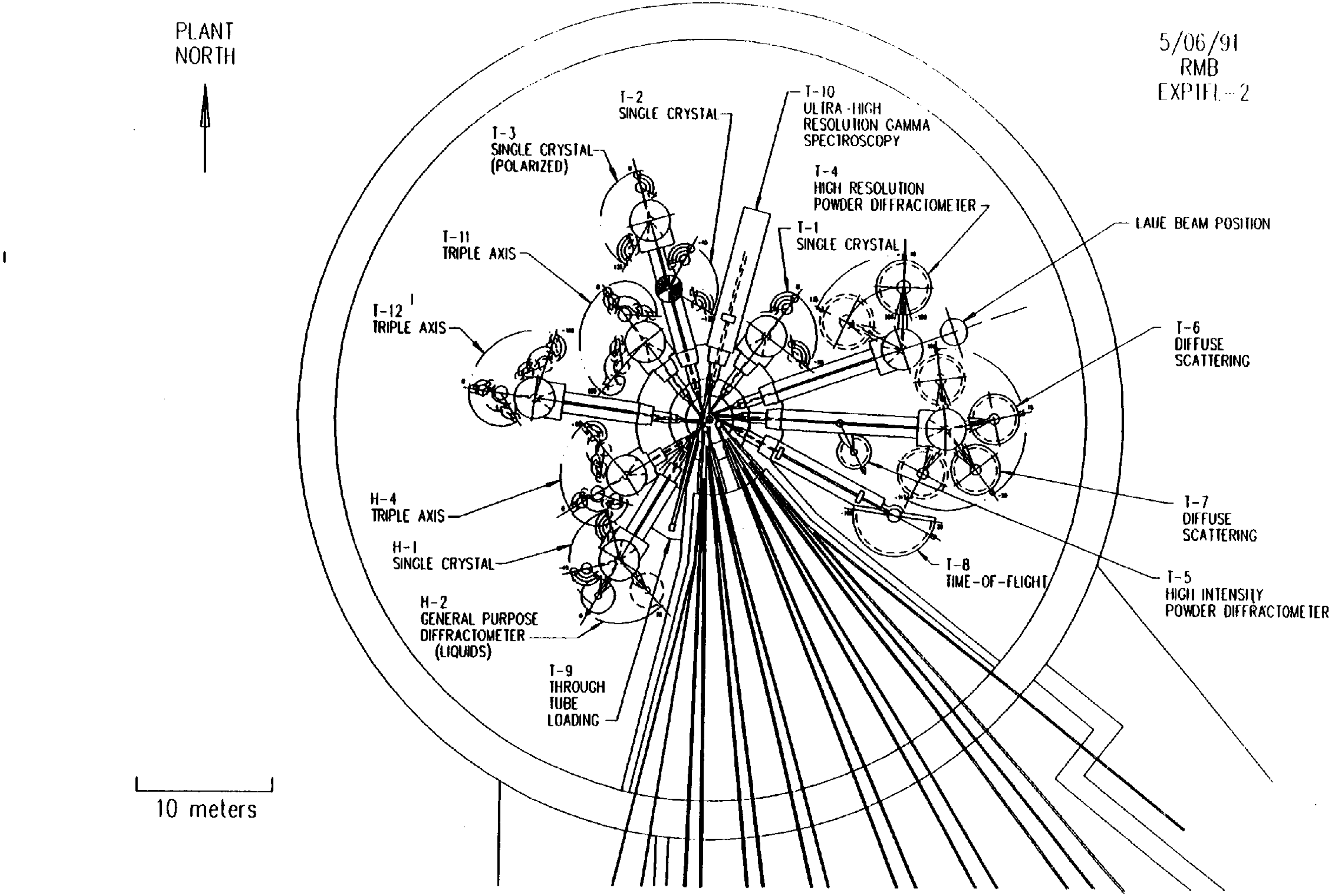
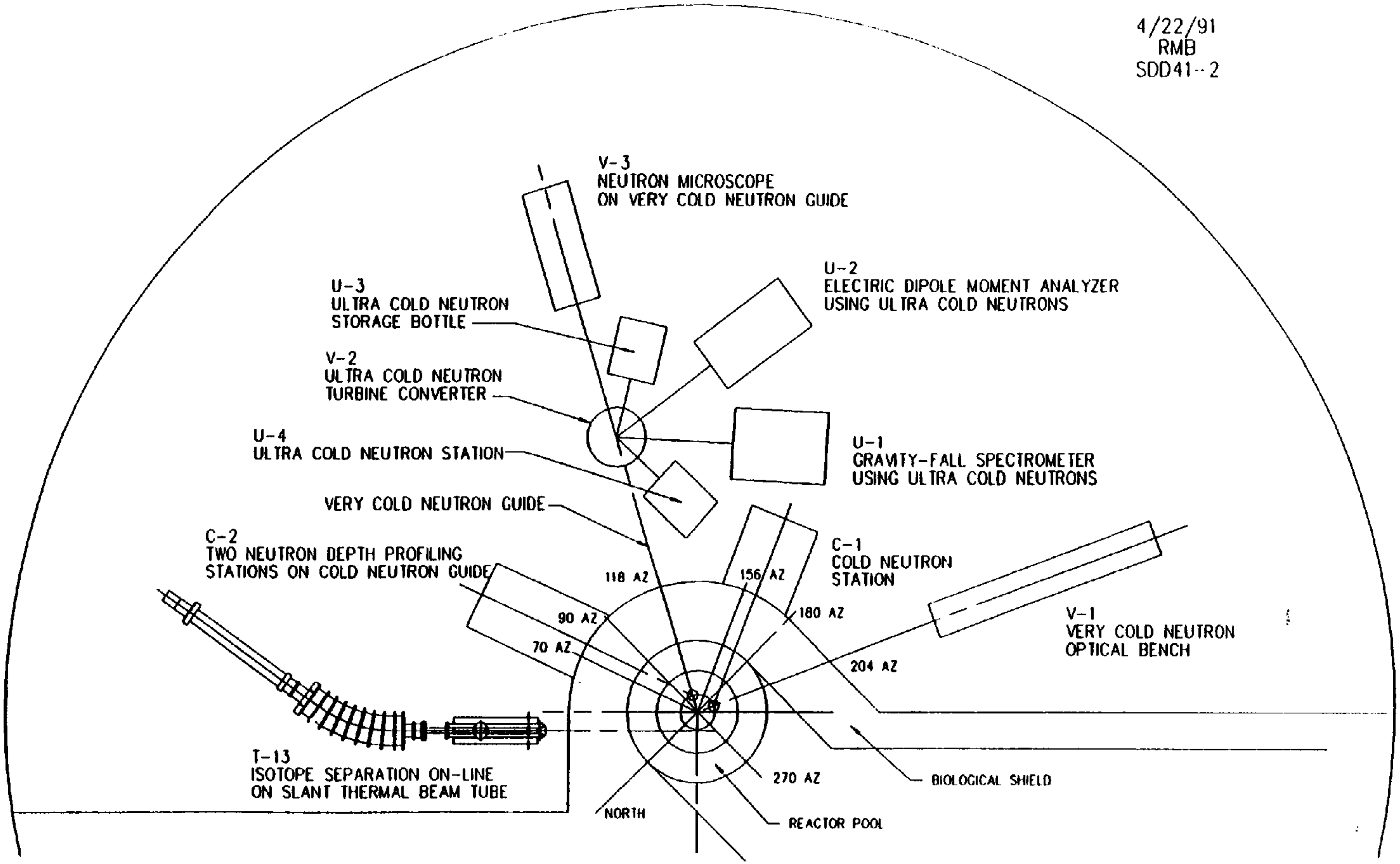


PLAN VIEW
GUIDE HALL



PLAN VIEW
GROUND FLOOR BEAM ROOM



PLAN VIEW
SECOND FLOOR BEAM ROOM

Figure 2

IN-CORE IRRADIATION FACILITIES

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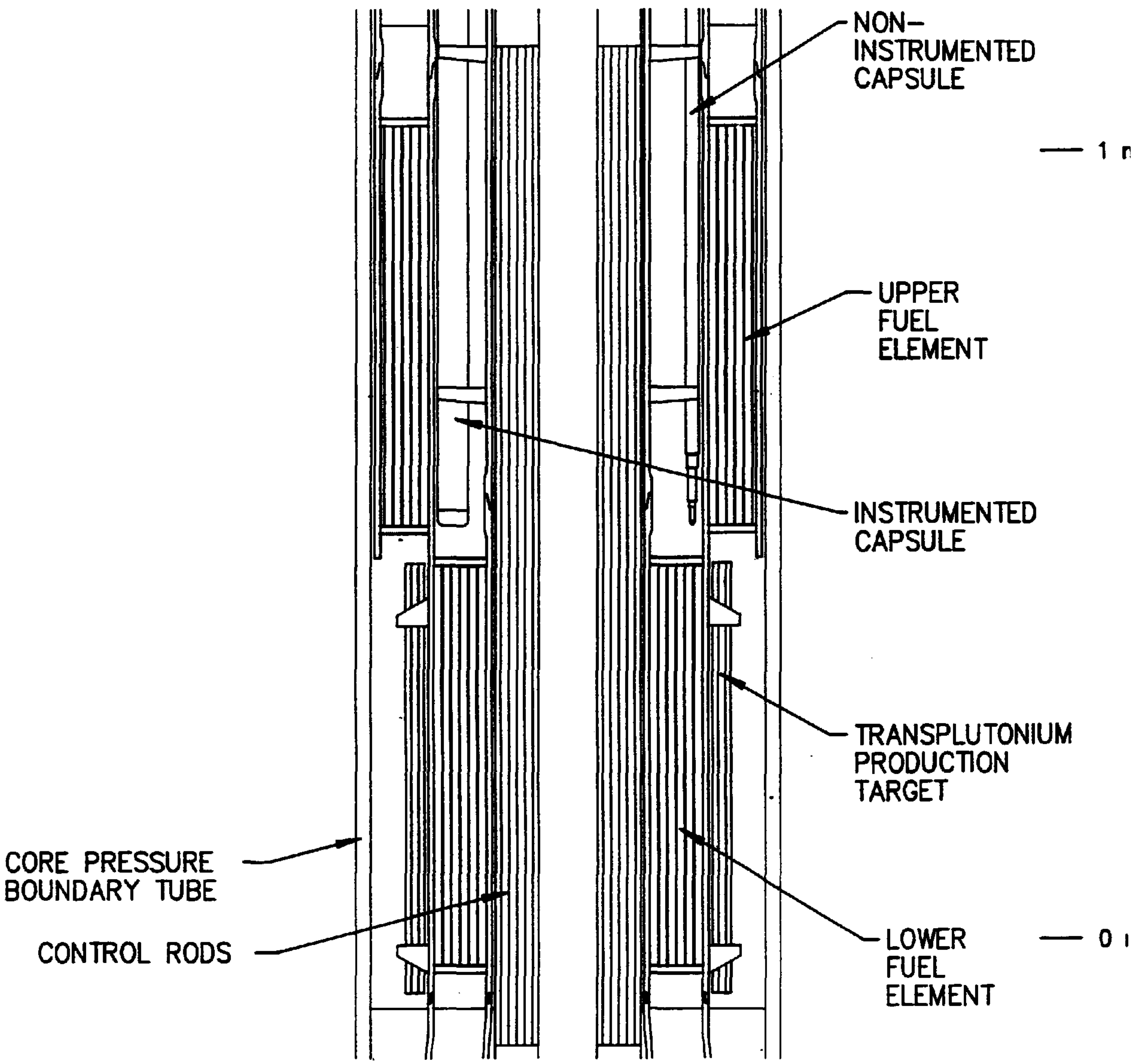

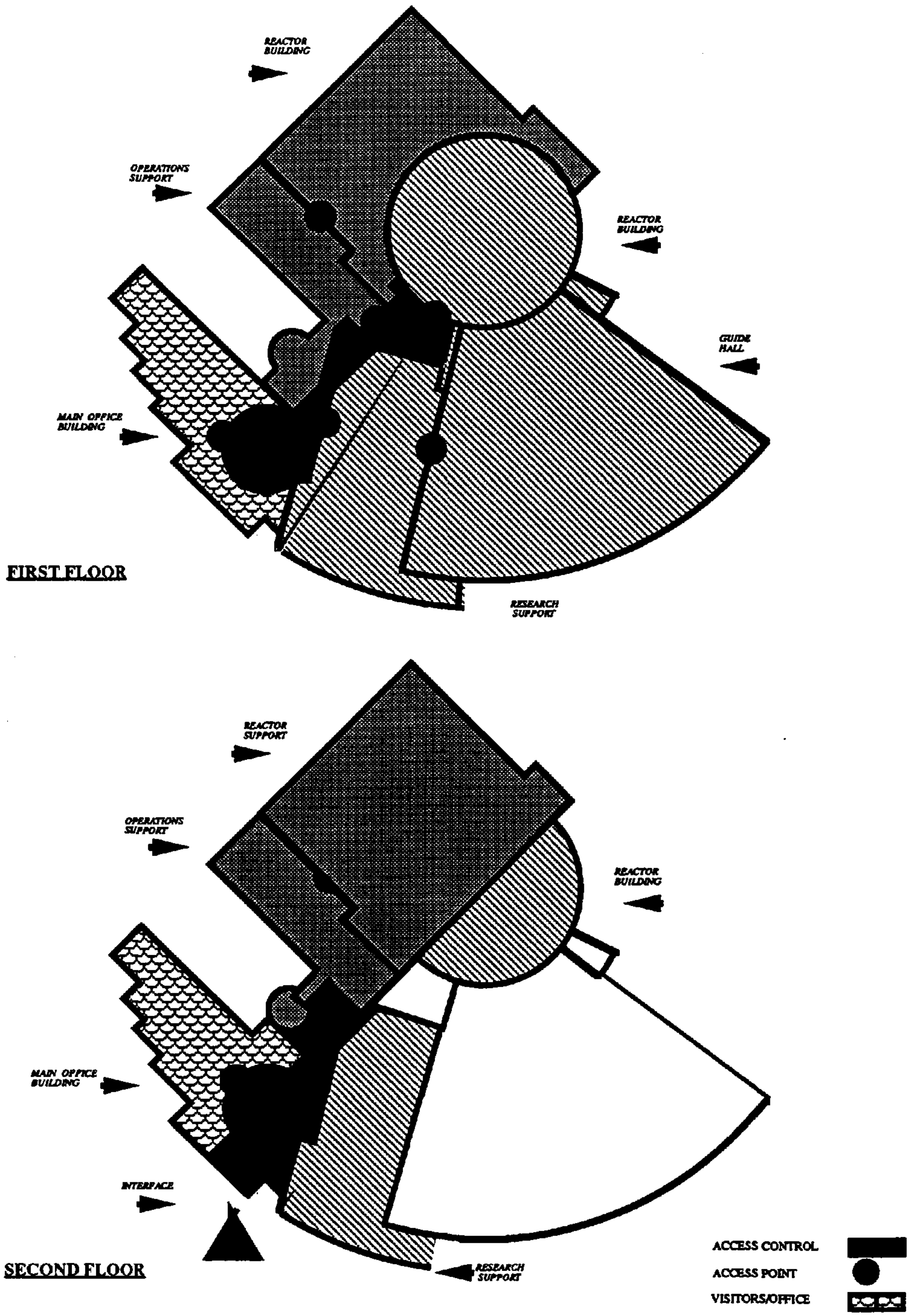

ACTIVITY / SECURITY ZONES IN THE ANS

The Reactor and Cold Neutron Research Facility at NIST

H.J. Prask and J.M. Rowe

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National Institute of Standards and Technology

Gaithersburg, MD 20899

ABSTRACT

The NIST Reactor (NBSR) is a 20 MW research reactor located at the Gaithersburg, MD site, and has been in operation since 1969. It services 26 thermal neutron facilities which are used for materials science, chemical analysis, nondestructive evaluation, neutron standards work, and irradiations. In 1987 the Department of Commerce and NIST began development of the CNRF--a $30M National Facility for cold neutron research--which will provide fifteen new experimental stations with capabilities currently unavailable in this country. As of May 1992, four of the planned seven guides and a cold port were installed, eight cold neutron experimental stations were operational, and the Call for Proposals for the second cycle of formally-reviewed guest-researcher experiments had been sent out. Some details of the performance of instrumentation are described, along with the proposed design of the new hydrogen cold source which will replace the present D_2O/H_2O ice cold source.

Introduction

Since the last IGORR meeting, progress on several fronts has continued at the NIST Reactor. This includes the installation of new instruments, near completion of other instruments, the initial use of the Cold Neutron Research Facility (CNRF) as a National Facility, and some positive and negative facilities developments. These, with some performance measurements, are described in this paper. Complete details for each instrument are given in Ref. [1], and in a set of review articles soon to be published [2].

Facilities Developments

Cold Ports

Guide Performance. The neutron guides are straight tubes of rectangular cross-section (either 120 by 50 or 150 by 60 mm^2), made from boron containing glass, polished on the inside to an rms surface roughness of 2 nm and a flatness equivalent to 10^{-4} radians, coated with approximately 80 nm of ^{58}Ni, and evacuated most of their length. The guides are manufactured commercially by Cilas-Alcatel[1]. In order to maintain the integrity of the reactor building containment, a "shutter"--which closes automatically by gravity whenever the reactor is shut down, providing a complete seal of the building--is inserted at the wall for each guide. These shutters also serve to interrupt the neutron

[1] Name brands or company names are given for purposes of clarity and do not imply an endorsement by NIST.

beams when desired, so that work can be performed in the experimental hall while the reactor is operating. The first four guides and a cold "port" (in the confinement building) have now been installed as shown in Figure 1. Overall, the performance of the guides has met expectations--typically 15% intensity loss at 40 m distance from the source. Measured fluence rates at certain instruments are provided later in the text.

For several years, coatings with larger critical angles have been sought, and in fact so-called super-mirrors have been developed which consist of many layers of materials with different neutron properties, arranged in a particular sequence. Researchers at NIST, in collaboration with Oak Ridge and Brookhaven National Laboratories, and with two small industrial firms, have recently demonstrated that these devices can be made with the requisite reflectivity for use in guides, at least on the laboratory scale [3]. Use of these coatings is planned for at least the tops and bottoms of the three remaining guides.

Guide Failure. It has been known for some time, as a result of observations at the Institut Laue Langevin, that the borosilicate glass used in normal neutron guides suffers radiation damage in a thermal or cold neutron field, primarily as a result of neutron capture in boron, followed by emission of an alpha particle. Accordingly, the first section of guides in the CNRF installation, which is almost wholly within the reactor biological shield, is made of non-borated glass, and is not evacuated, but rather is both filled with and surrounded by helium gas at one atmosphere (the helium filled section is indicated by dashed cross-hatching in Figure 2). However, the downstream guides are made of borosilicate glass, and were evacuated since their installation in 1990.

This design was chosen after discussions with the vendor, based upon experience at the Saclay reactor, which has fluence rates comparable to the NIST reactor. The installation at Saclay had successfully operated for more than six years when the NIST design was first begun. In approximately September of 1991, it was learned that the Saclay guide installation had suffered two successive guide implosions, presumably as a result of radiation damage, after more than 8 years of continuous operation. Upon receipt of this information, a plan was made to add an additional aluminum window in the guides at 5 m further out ("removable sections" of Fig. 2) in February, 1992, so as to allow filling of this section with helium, thereby removing the stress from them. However, before this could be done, the first element in NG-6 failed on December 1, 1991, in the area shown in the insert in Figure 2.

It appears from the reconstruction of the occurrence, using fragments recovered from the floor beneath the failed section, that one of the side plates failed over a 25 cm length. Based upon all previous experience, and the known total neutron fluence on this section, this failure occurred at a fluence five to ten times less than should have been expected. Preliminary analysis indicates that there may have been a defect in the glass, perhaps as a result of faulty annealing.

When the side plate failed, air entered the guide at high velocity, and carried about 700 g of glass consisting of particles ranging in size from dust to 1 cm radius, down the 80 m length of the guide into the guide hall. There was no release of radioactivity, no effect on the reactor, no personnel were endangered, and no compromise of reactor safety resulted from this occurrence. The NBSR resumed operation in April after modification of NG-5, 6, and 7 for He filling in the 5 m sections indicated.

Cold Sources

A schematic diagram of the existing cold neutron source is shown in Figure 2, with the neutron guides installed (for scaling, the O.D. of the D_2O Ice Moderator is 34 cm). The main components are the lead/bismuth shield lining the beam port (required to reduce γ-ray heating in the moderator), a cryostat containing the D_2O ice (which is surrounded by an insulating vacuum and a helium blanket), and the cooling tubes (which carry helium gas at 30 - 40 K to cool the source). The other main component (which is not shown) is a helium gas refrigerator which supplies 1.0 KW of cooling at a helium mass flow rate of 28 g/s. The cryostat is fabricated of a magnesium alloy (AZ31B), in order to reduce the heat load on the system and increase the neutron efficiency. With the reactor operating at 20 MW, the helium gas enters the cryostat at 32 K and exits at 40K, giving an average temperature in the ice (as a result of thermal conductivity) of approximately 45 K.

The gain which this moderator provides for cold neutrons is of the order of 10, measured as the ratio of the number of cold neutrons produced with the moderator filled with D_2O ice at low temperature to the number produced with the moderator filled with D_2O at 300 K. By direct experiment (as well as calculation), we have determined that this ratio is maximized by the addition of 7% H_2O, and the moderator is operated this way. This moderator has been in service since 1987, and system reliability is now quite good (> 98% availability). As a result of radiation damage in the solid ice, the moderator is warmed up to approximately 80-100 K every two days to allow recombination of the constituents produced by radiolysis.

With this source in operation, we have turned our attention to a second generation source which will utilize liquid hydrogen. A schematic drawing of the proposed hydrogen source is shown in Figure 3, from which several features should be noted. First, as a result of the lower density of hydrogen, the lead/bismuth shield can be left out, with a resultant gain in intensity and simplicity. Second, the source itself is much thinner, as a result of the high scattering power of hydrogen. The moderator chamber is a 2 cm thick, spherical shell with an outside diameter of 32 cm, containing about 5 liters of liquid hydrogen. A 20 cm diameter reentrant hole fully illuminates the guides with cold neutrons from the flux-trap in the interior of the sphere. A large buffer volume is open to the moderator chamber so that the entire liquid hydrogen inventory can vaporize and expand into the buffer without over-pressurizing the system. The cooling mechanism is a gravity fed flow of liquid into the moderator chamber, where evaporation removes the heat produced by the radiation. The hydrogen is in a cooled loop, reliquified outside the beam port, completing the naturally circulating thermosyphon requiring no pumps or moving parts.

The entire hydrogen system is surrounded by an insulating vacuum, which is in turn surrounded by gaseous helium (not shown in the figure) in order to prevent the entry of oxygen into any volume containing hydrogen. This is a central part of the safety philosophy--namely, to limit the oxygen available for combination with hydrogen. The liquefaction is done in the hydrogen condenser, with cooling provided by a 3.5 KW helium gas refrigerator. The first engineering tests of the proposed new source will take place this summer, and a full safety analysis is being prepared for submission to the Nuclear Regulatory Commission this year. Calculations of the performance of this source (performed both by analytical and Monte Carlo methods) indicate a further gain of cold neutrons available for experiments of at least a factor of two over the existing source, with somewhat larger gains at the lower energies.

Instrument Development

Operational

8-m SANS Spectrometer [1,4]. This SANS spectrometer is installed on NG-5. The instrument utilizes a mechanical velocity selector and pinhole collimation to provide a continuous incident beam whose wavelength is variable from 0.5 to 2.0 nm. The neutron detector is a 65 x 65 cm^2 position-sensitive proportional counter which pivots about the sample position to extend the angular range of the spectrometer. The low-Q limit of the instrument (0.025 nm^{-1}) is achieved with a multiple-beam converging pinhole collimation system. A 50-cm diameter sample table accepts standard cryostats, furnaces and electromagnets as well as a sample chamber which houses a temperature-controlled multiple sample changer. The instrument has a dedicated microcomputer which is networked to an interactive color graphics terminal with specialized software to provide rapid imaging and analysis of data from the two-dimensional detector. The instrument is suitable for the study of structural and magnetic inhomogeneities in materials in the 1 to 100-nm range. Intensity characteristics are listed in Table 1.

Table 1. Intensity Characteristics of the 8-m SANS Instrument.

Neutrons on Sample vs Q_{min}	Q_{min} (nm^{-1})	I_s[a] (n/sec)
	0.04	6×10^3
	0.10	3×10^5

[a] measured on 1.5 cm diam. sample, present cold source, with $\Delta\lambda/\lambda \approx 25\%$.

The NIST/Exxon/University of Minnesota 30-m SANS [1,4]. This instrument, developed jointly by NIST, the Exxon Research and Engineering Co., and the University of Minnesota, extends the measurement range and sensitivity of the NIST 8-m SANS instrument in both the small and intermediate angle regions, and, in addition, provides exceptional flexibility in optimizing beam intensity and resolution to meet the requirements of a particular measurement. The Q-range of the instrument extends from 0.01 nm^{-1} to nearly 10 nm^{-1} which enables structural features in materials ranging from roughly 0.5 to nearly 500 nm to be studied. This wide Q-range is achieved through the use of two large 2D position-sensitive detectors (see Fig. 4): a primary detector that moves axially within the cylindrical portion of the evacuated post-sample flight to vary the sample-to-detector distance continuously from 3.5 to 15 m, and a second detector (not yet installed) that will move along a circular arc (over a 40° range) at a fixed distance of 2 m from the sample. Flexibility in beam collimation is achieved by incorporating eight neutron guide sections in the 15-m long pre-sample flight path that can be easily shifted in or out of the beam to change the effective source-to-sample distance in 1.5 m increments from 3 to 15 m. A mechanical rotating velocity selector with variable speed and pitch provides the capability to vary both the wavelength and wavelength resolution of the beam over a wide range.

This instrument began operation using pinhole collimation and the primary detector in the spring of 1991. Development work continues on two novel components for this instrument, which if successful will significantly expand its measurement capabilities. These are a Fe/Si supermirror array for polarizing the incident beam, and a doubly curved, grazing incidence mirror (to replace the pinhole collimation and increase the flux on the sample) to focus the beam onto the detector. Performance is indicated in Table 2.

Table 2. NIST/Exxon/U. of Minn. 30-m SANS Spectrometer

Neutrons on Sample vs Q_{min}	Q_{min} (nm^{-1})	I_s [a] (n/sec)
	0.01	2.5×10^3
	0.02	3×10^4
	0.04	3×10^5
	0.10	1×10^6

[a] measured on 1.5 cm diam. sample, present cold source, with $\Delta\lambda/\lambda \approx 25\%$.

Medium Resolution Time-of-Flight Spectrometer [1,5]. Two time-of-flight spectrometers are planned for the guide hall of the CNRF. The first of these instruments, which is primarily designed for medium resolution applications, is located on guide NG-6. The second instrument, to be located on guide NG-1, uses a number of disk choppers, and is intended for high resolution measurements but may also be operated with relaxed resolution when required. Installation of the latter is expected in 1993.

In the medium-resolution spectrometer, the incident beam wavelength (0.23-0.61 nm) is determined using a double monochromator. The principle of this device is similar to that of a single monochromator, but an important advantage is that the selected neutron wavelength can be changed without having to move the sample (and everything downstream of the sample). This significantly simplifies the design of the instrument. The monochromators are made of individually aligned pyrolytic graphite (PG(002)) crystals. A 60′ Soller collimator is located between the monochromators. The first monochromator is flat, whereas the second monochromator is vertically curved in order to focus intensity at the sample position. Each monochromator is made of two layers of crystals, each layer possessing a 25′ mosaic, but staggered horizontally with 25′ angular offset. This effectively yields a more desirable anisotropic mosaic distribution, 25′ vertically and 50′ horizontally.

The neutron beam leaving the second monochromator is filtered, using pyrolytic graphite or liquid-nitrogen-cooled beryllium, in order to remove higher order contamination, as well as epithermal neutrons. It is then pulsed using a "Fermi chopper". The curvature of the slots and the speed of the chopper determine the optimum transmitted wavelength of the chopper. Two slot packages are available, corresponding to optimum transmitted wavelengths of 0.4 and 0.15 nm, respectively, at close to the maximum chopper speed.

The sample chamber can accommodate a wide variety of cryostats and furnaces. An oscillating radial collimator between the sample and the detectors blocks most of the scattering from material components which surround the sample (e.g. heat shields of cryostats). The evacuated sample-to-detector flight path contains an array of ^3He neutron detectors covering a scattering range of 5-140°. Some instrument parameters are given in Table 3.

Table 3. Characteristics of the Medium-resolution TOF Spectrometer

Incident Wavelength Range:	$0.23 < \lambda < 0.61$ nm (÷2 for PG(004))
Incident Energy Range:	$15.5 > E > 2.2$ meV (x4 for PG(004))
Elastic Momentum Transfer Range:	$1 < Q < 50$ nm^{-1} (x2 for PG(004))
Flux on Sample[a]:	~1 x 10^4 n/cm^2/s at λ = 0.24 nm
	~5 x 10^3 n/cm^2/s at λ = 0.40 nm
Elastic Energy Resolution[a]:	40 μeV at λ = 0.60 nm
	150 μeV at λ = 0.40 nm
	600 μeV at λ = 0.24 nm
Q Resolution[a] :	$\Delta Q/Q < 3\%$

[a] Expected performance.

Cold Neutron Depth-Profiling (NDP) Facility [1,6]. NDP determines distribution of isotopic concentrations vs. depth within the first few micrometers of a surface. This is achieved by energy analysis of the prompt charged particles emitted following neutron capture--for which energy loss vs. path length is known. The instrument is operational on NG-0 in the reactor hall with a beam filtered by 13.5 cm of single-crystal sapphire. It includes a new 61 cm diam stainless-steel sample chamber, vacuum system, detectors, electronics, and data acquisition system. The new depth-profiling chamber is capable of operating at

UHV pressures and incorporates many design features that enhance the capabilities that currently exist at the thermal depth-profiling instrument in the reactor hall. The chamber has an X-Y positioning system that will allow computer-controlled scanning of up to 15 x 15 cm^2 samples. The computer is also capable of rotating both the sample and the detectors to adjust the sensitivity for depth-profiling. The ability to use time-of-flight depth profiling techniques has also been incorporated into the chamber design. An electron gun for in-situ surface etching of samples will also be available. Pancake type uranium fission chambers are used for monitoring the neutron fluence. A comparison with the BT-3 facility is given in Table 4.

Table 4. Neutron Depth-Profiling Facilities at the NBSR.

	BT-3	Cold Beam
Fluence rate[a]	4×10^8	1.2×10^9 $n\text{-}cm^{-2}\text{-}s^{-1}$
Peak energy	22.5 meV	~8 meV
^{10}B sigma	3404 barns	~5700 barns
Rel. sens.	1	3
Gamma dose	400 mR/h	~400 mR/h

[a] Thermal neutron equivalence.

Prompt-Gamma Activation Analysis (PGAA) Spectrometer [1,7]. PGAA gives simultaneous nondestructive determination of many major, minor, and trace elements in a variety of bulk matrices. The sample is irradiated with a beam of neutrons. The constituent elements of the sample absorb some of these neutrons and emit prompt gamma rays which are measured with a high-resolution gamma-ray spectrometer. The energies of these gamma rays give qualitative identification of the neutron-capturing elements, and the intensity is proportional to their concentrations.

Design of the facility has focused on three related topics. First, the high quality of the neutron beam and the low level of environmental background in the guide hall allow closer sample-detector spacing, resulting in higher counting efficiency and better sensitivity. Second, the high count rates possible with this high efficiency (> 50k counts per sec) can be measured without loss of quality with recent advances in instrumentation. Finally, the improved efficiency makes attractive the use of γ-γ and γ-conversion electron coincidence counting in analytical measurements, with considerably improved specificity in some cases. Also, the counting system is designed to handle multiple gamma detectors with Compton suppression. Future modifications will include multiparameter coincidence counting.

Structural and shielding materials for this and neighboring instruments have been chosen as far as practicable to avoid generating a background of capture and decay gamma rays. Hydrogenous absorbers are avoided. The section of the beam tube adjacent to the sample position is made of boron-free glass. ^{6}Li is used wherever possible for collimators and absorbers, and antimony-free lead is used for gamma shielding. As a result, the sensitivity for most elements is expected to be at least tenfold better than any with any thermal beam in existence. The detection limit for hydrogen is calculated to be about 1 μg. Characteristics are summarized in Table 5.

Table 5. PGAA Characteristics	
Measured reaction rate per atom at sample position:	$1.4 \times 10^8\ \sigma_0$, where σ_0 is the thermal neutron cross section.
Maximum sample size:	5x5 cm^2
Measured gamma rate:	< 1 mR/hr

Center for High Resolution Neutron Scattering (CHRNS) [1,4]. Two of the new CNRF instruments that will begin operation in 1992, a 30-m, high resolution small angle neutron scattering (SANS) instrument and a cold neutron polarized beam triple-axis spectrometer (see below), have been constructed with support from the National Science Foundation. The 30-m CHRNS SANS instrument and its dedicated guide, NG-3, are nearly complete. Up to 85% of the time on the CHRNS 30-m instrument will be allocated by the Program Advisory Committee (see section on Researcher Program, below) compared to 25% for the PRT-developed NIST/Exxon/U. Minnesota SANS instrument.

The performance characteristics of the single-detector CHRNS SANS instrument are very similar to the NIST/Exxon/U. of Minn. 30-m SANS spectrometer. It will, however, be the first in the United States to utilize a 2-d detector of the ILL-type, capable of counting rates in excess of 20,000/sec. To take full advantage of this high count rate capability, the instrument's post-sample vacuum flight path has been designed to allow the detector to approach within 1.2 m of the sample. As a result the instrument will not only be able to perform high resolution (low-Q) measurements, but will also cover a wide Q-range (0.01 to 6 nm^{-1}) and significantly improve current capabilities for time-resolved SANS measurements.

The Neutron Lifetime Experiment [8]. The end position on the NG-6 beam line at the CNRF is devoted to experiments in the field of fundamental neutron physics. In contrast to the CNRF neutron interferometry facility, there is no permanently installed instrumentation; the properties of the beam must be matched to the requirements of each experiment individually. Aside from the beam shutter and a beam stop, the only other permanent equipment on the beamline is a filter cryostat to remove fast neutrons and gamma rays. A variety of collimators, polarizers, spin flippers, and transport tubes can be made available for individual experiments. The capture flux at the end of the 15 x 6 cm^2 NG-6 guide has been measured to be about $4x10^8$ $n/cm^2/s$, which is about a factor of nine lower than the 3 x 5 cm^2 SN-7 guide at the ILL. This flux will increase significantly after the installation of the hydrogen cold source.

The first experiment at the end of NG-6, a measurement of the lifetime of the neutron, is currently underway. This experiment is a collaboration between NIST, the University of Sussex and the Scottish Universities Research and Reactor Center, with related work on advanced methods of neutron flux measurements performed by NIST, the Central Bureau for Nuclear Measurements, the Scottish Universities Research and Reactor Center, Harvard University, and Los Alamos National Laboratory. A more detailed description than that given here can be found in Ref.[9].

The strategy of the experiment is to measure the neutron decay rate N_{decay} and the mean number of neutrons N_n within a well-defined volume traversed by a cold neutron beam. The decay rate is related to these quantities by the differential form of the exponential decay law $N_{decay}=N_n\tau_n^{-1}$. Figure 5 shows a schematic outline of the method. N_{decay} is measured in the experiment by trapping the protons produced in the decay with a Penning trap and counting the trapped protons. N_n is measured by requiring the neutrons leaving the decay volume to pass through an accurately calibrated neutron monitor.

The first experiment using this apparatus, performed at the ILL, measured a neutron lifetime of 893.6 seconds with a 1σ error of 5.3 seconds. This result is in excellent agreement with other recent direct measurements of the neutron lifetime using completely different techniques. In the next experiment, to be conducted at the NIST CNRF, we hope to reduce the 1σ error to 1.5 seconds. Most of the improvement in the anticipated error comes from better counting statistics and improvements in the calibration of the neutron monitor detector efficiency. If the uncertainty in ϵ_o, the efficiency at a single neutron velocity, can be reduced by calibration with the black detectors, the neutron lifetime measurement at the NIST CNRF may turn out to be the most accurate measurement of the neutron lifetime.

Also on NG-6, test runs for an experiment to search for time reversal invariance violation in neutron beta decay are underway. Other experiments in the near future may include searches for parity violation in neutron-nucleus scattering and studies of polarized neutron interactions with polarized nuclei.

Near-Operational Instruments

The CHRNS Spin-Polarized Inelastic Scattering Spectrometer ("SPINS") [1,10]. The installation of the SPINS spectrometer has begun on an NG-5. Assembly and testing of this instrument will continue through the summer with the first experiments expected to commence by year's end. It will initially be configured as a cold neutron triple-axis spectrometer with focusing pyrolytic graphite monochromator and analyzer crystals. Polarizing the incident beam, and analyzing its polarization after scattering, will be provided by iron-silicon supermirrors used in transmission geometry. The development of these polarizers has now reached the stage where the first set of wide-beam polarizers is about to go into production. Also under development for this instrument is a Drabkin-type, energy-dependent spin flipper. Used in conjunction with the polarizers, this device partially decouples the momentum and energy resolution of the instrument making it possible to achieve an energy resolution of 10 μeV under conditions of moderately relaxed momentum resolution.

Neutron Interferometer [8]. With the installation of the primary base slab for a sophisticated vibration control system in early November 1991, construction of the neutron interferometry facility on NG-7 has passed a major threshold. This experimental station will be devoted to a wide variety of investigations concerned with advanced neutron optics, matter wave interferometry, sensitive tests of quantum mechanics and other fundamental theories as well as measurements of the basic parameters describing the interaction between neutrons and matter.

When operational, neutrons extracted by a pyrolytic graphite monochromator mounted inside NG-7 will be directed into an environmentally controlled room containing a perfect crystal interferometer. The monochromator and its control system have recently been installed and tested, and work very well. Due to the extreme sensitivity of such interferometers, very stringent precautions against seismic, acoustic and thermal "noise" are being taken. Internal vibrational modes or bulk displacements at sub-Angstrom levels are sufficient to destroy the interference signal or to render the interferometer insensitive. For this reason, the neutron interferometry station at the CNRF will be equipped with a sophisticated vibration isolation system to reduce the effect of seismic and acoustic perturbations. This system will include both passive and active vibration isolation technology to achieve sufficiently low levels of disturbance so that interferometers of comparatively large dimension (tens of centimeters) may be employed. The prospect of such "long baseline" interferometry offers the possibility of a very significant increase in the sensitivity of this technique.

The initial scientific program planned for this facility includes a the detailed study of a variety of quantum mechanical effects. Recent measurements carried out at the University of Missouri concern a very novel electromagnetic interaction between the neutron magnetic moment and a static electric charge distribution. This so-called Aharanov-Casher effect represents a geometrical phase, which while analogous to effects seen with charged particles has not previously been observed. Detailed studies of this effect are planned at the CNRF. Neutron interferometry is sufficiently sensitive to detect the quantum mechanical phase shift which arises from the difference in the gravitational potential corresponding to a height difference of less than a millimeter. The current best measurements of such quantum mechanical gravitational phase shifts are inconsistent with theoretical expectations. Several measurements with improved sensitivity and increased immunity to possible systematic effects are planned. A wide variety of other investigations are also anticipated.

Construction of this facility is being carried out by NIST staff from the Physics Laboratory and the Materials Sciences and Engineering Laboratory. This project is supported by NIST and the National Science Foundation through a grant to the University of Missouri (Columbia). Included among the institutions which are anticipated to participate are the University of Innsbruck (Austria), Atom Institute (Austria), Hampshire College, University of Melbourne (Australia).

High-Resolution 32-Detector Powder Diffractometer [11]. The new high-resolution powder diffractometer is ready for installation at BT-1 in the reactor hall. A schematic of the instrument is shown in Figure 6. The provision for the selection of one of three monochromators--arranged in series, at take-off angles to produce 0.154 nm neutrons--will allow an instrumental resolution to be selected in which the d-spacing range of maximum interest can be examined at optimum conditions. For 7′-40′-7′ collimation, a minimum FWHM of ~0.18° is calculated for d's of~ 0.133, ~0.113, and ~0.091 nm for the three monochromators. Additionally, according to the angle range they cover, different vertical divergences (up to 9°) for different detectors are provided for. The detector array subtends an angular range of 155° and can reach a maximum scattering angle of 167°.

Other New Capabilities

In addition to the instruments described in the previous sections, several other instruments are in detailed design or fabrication stages:

NIST/IBM/U. Minn. cold neutron reflectometer. This instrument is planned for installation on NG-7 in the guide hall in mid-1992. It will have horizontal sample geometry, counters for both small-angle and large-angle scattering--for specular reflection and grazing-angle diffraction measurements, respectively--each with polarized-beam capabilities. Reflectivities less than 3×10^{-7} are expected to be measurable, with a Q-resolution, $\Delta Q/Q$, of 0.02 to 0.1—depending on the choice of collimating slits.

Residual stress, texture, and single crystal diffractometer. This instrument is in the final design stage with installation on BT-8 in the reactor hall expected by early 1993. It will feature a monochromator take-off angle of up to 120°, a vertically-focusing monochromator, a large full-circle goniometer, and a PSD or multi-detector.

Researcher Program

The first guest researcher experiments scheduled through the formal review system of the CNRF were performed in November. In response to the limited first call for proposals, the CNRF received 40 proposals, of which 23 were accepted and allotted beam time after a thorough review process. For the 30-m NIST/Exxon/U. of Minn. SANS instrument, one of three stations available through the guest researcher program for the first cycle, the beam time requested (160 days) exceeded that available by a factor of three. In the next proposal cycle, it will be possible to accommodate many more guest researcher experiments when a second 30-m SANS machine and other instruments become operational.

The review of the proposals submitted in response to the first call was accomplished through a system whose basic features were proposed by the Program Advisory Committee (PAC)--and described in the April 1991 NEUTRON STANDARD. The proposals were first peer reviewed by mail, resulting in several written reviews for each proposal. The proposals with their reviews were then scrutinized and rank-ordered for scheduling by the PAC. Actual scheduling is taking place on an ongoing basis through June of 1992.

For the future, it is expected that the number of proposals and the number of guest researcher experiments at the CNRF will increase dramatically over the next few years, and will involve hundreds of scientists from all over the world.

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4. B. Hammouda, S. Krueger and C.J. Glinka, in ref. 2.

5. J.R.D. Copley and T.J. Udovic, in ref. 2.

6. R.G. Downing, G.P. Lamaze, J.K. Langland and S.T. Hwang, in ref. 2.

7. R.M. Lindstrom, in ref. 2.

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9. Reviewed in Nuclr. Instr. Meth., A284, No. 1, 1989.

10. S.F. Trevino, in ref. 2.

11. E. Prince and A. Santoro, in NBS Reactor: Summary of Activities July 1985 through June 1986, NBS Tech. Note 1231 (1986), p.82.

Figure Captions

1. NBSR reactor and guide hall with completed guides shown.

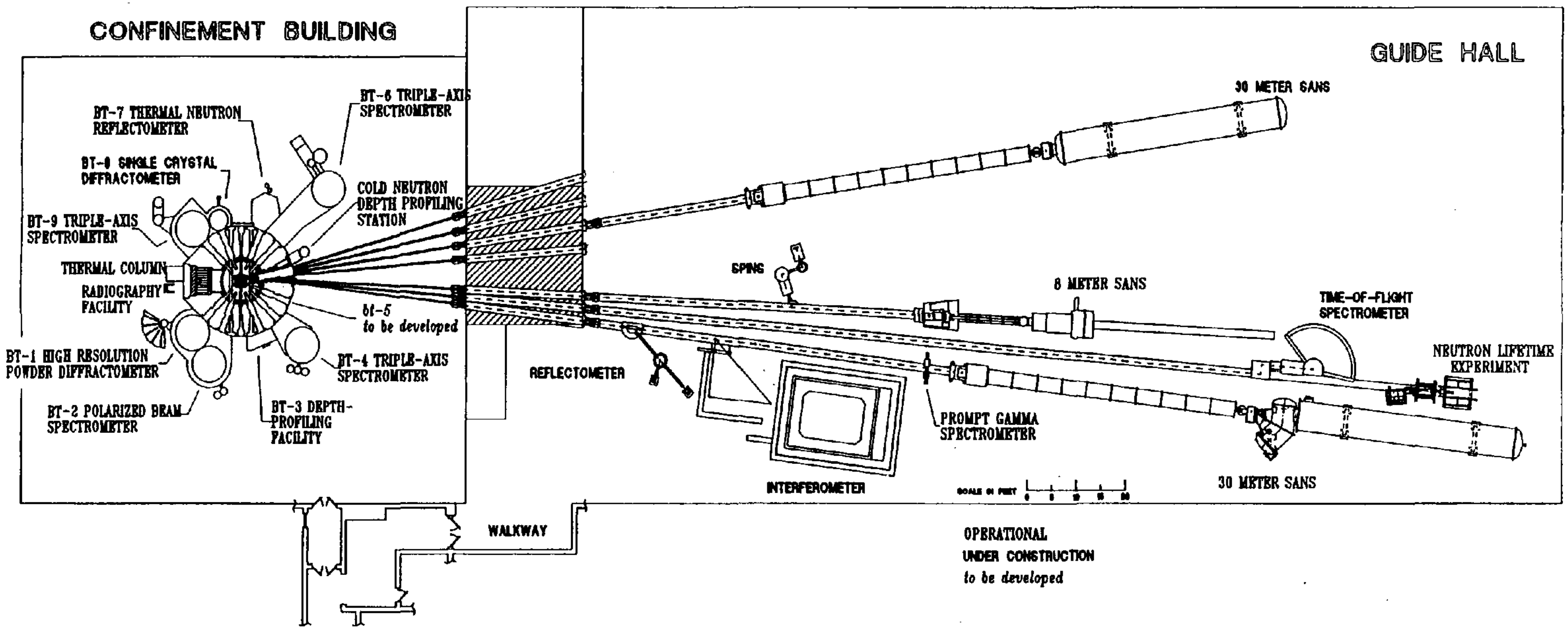
2. Present cold source and guide failure detail.

3. New cold source.

4. NIST/Exxon/U. of Minn. 30-m SANS spectrometer schematic.

5. Schematic of neutron lifetime experiment.

6. High-resolution 32 detector powder diffractometer.


CONFINEMENT BUILDING
GUIDE HALL
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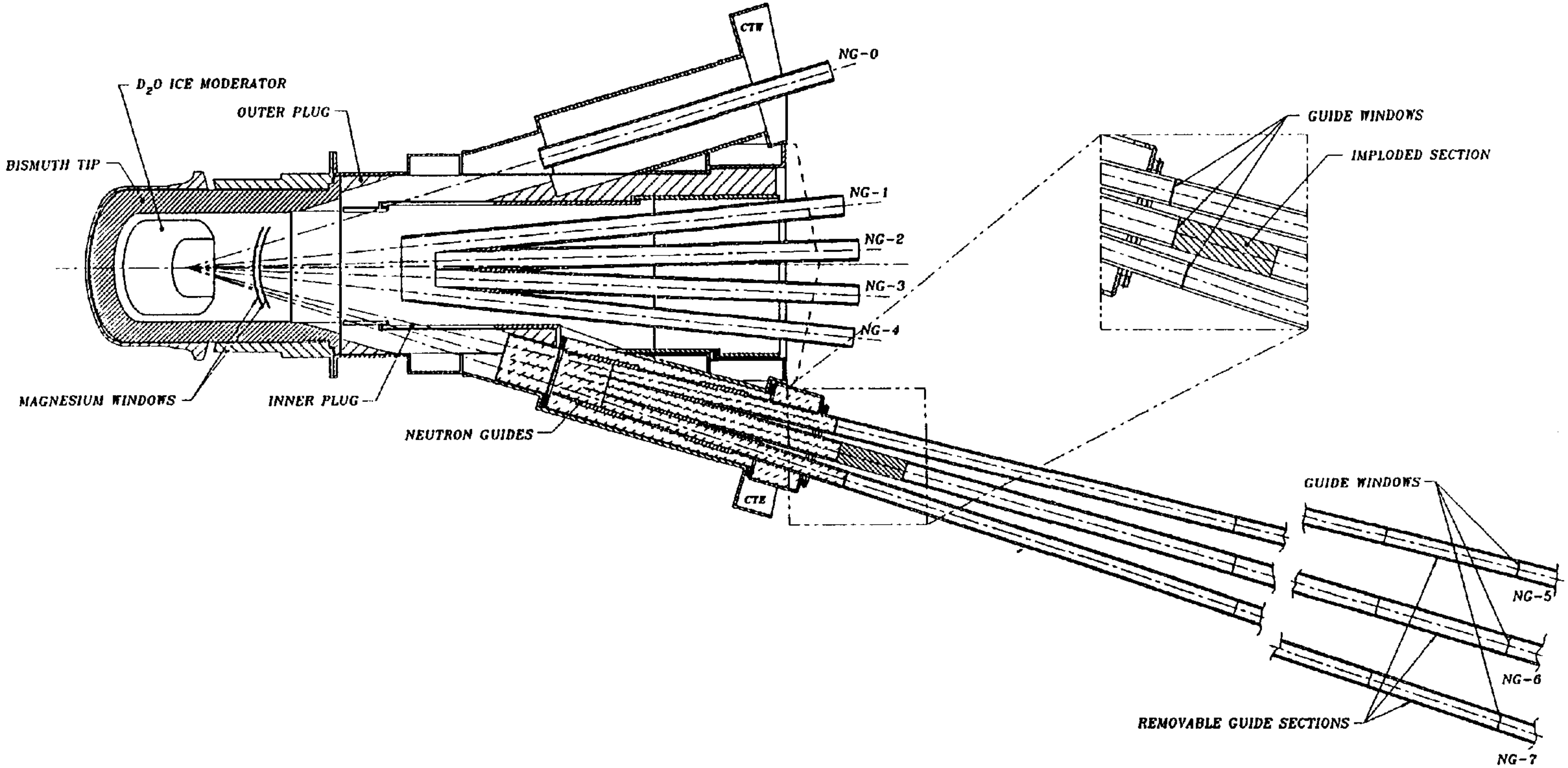
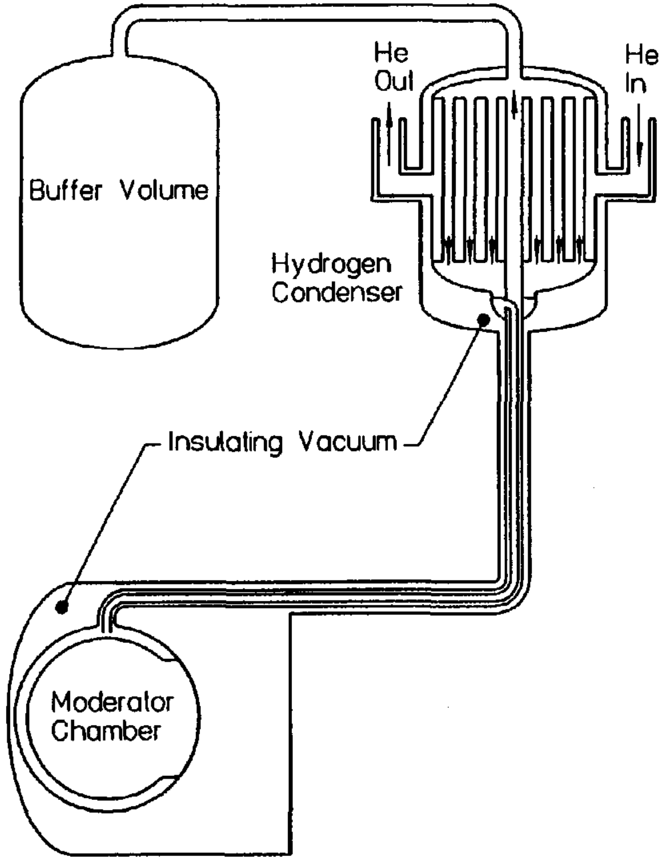
BT-6 TRIPLE-AXIS SPECTROMETER
BT-8 SINGLE CRYSTAL DIFFRACTOMETER
COLD NEUTRON DEPTH PROFILING STATION
BT-9 TRIPLE-AXIS SPECTROMETER
THERMAL COLUMN
RADIOGRAPHY FACILITY
bt-5 to be developed
BT-1 HIGH RESOLUTION POWDER DIFFRACTOMETER
BT-4 TRIPLE-AXIS SPECTROMETER
BT-2 POLARIZED BEAM SPECTROMETER
BT-3 DEPTH-PROFILING FACILITY
30 METER SANS
SPINS
8 METER SANS
TIME-OF-FLIGHT SPECTROMETER
NEUTRON LIFETIME EXPERIMENT
REFLECTOMETER
PROMPT GAMMA SPECTROMETER
INTERFEROMETER
30 METER SANS
WALKWAY
OPERATIONAL
UNDER CONSTRUCTION
to be developed

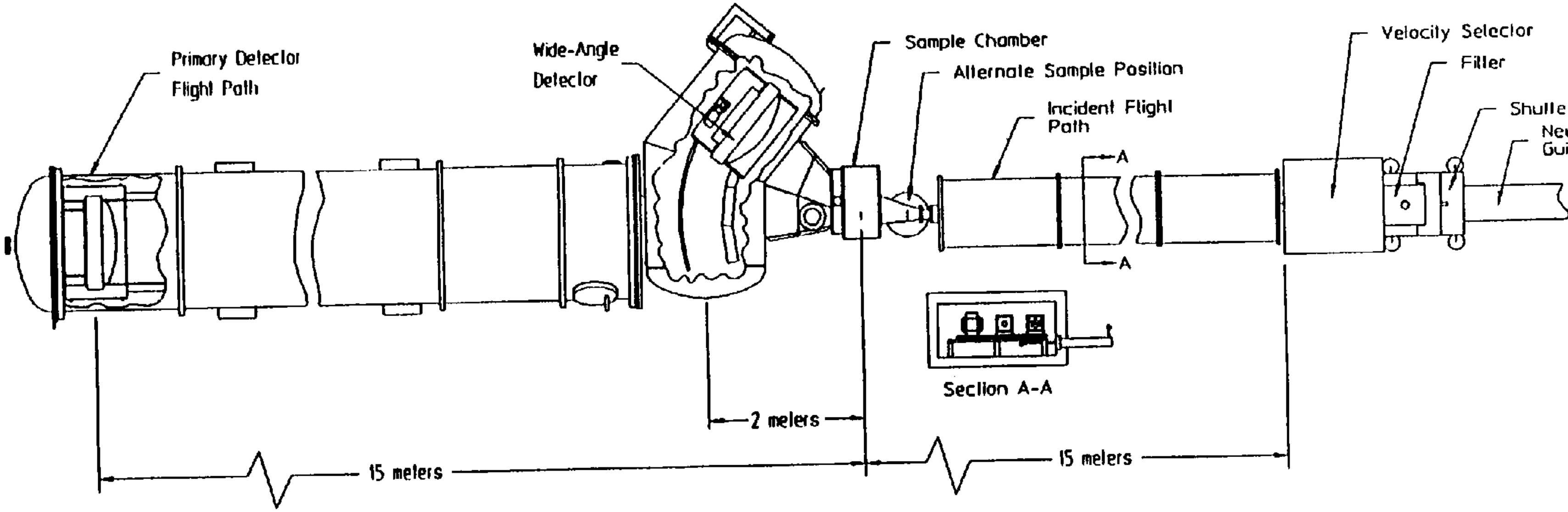


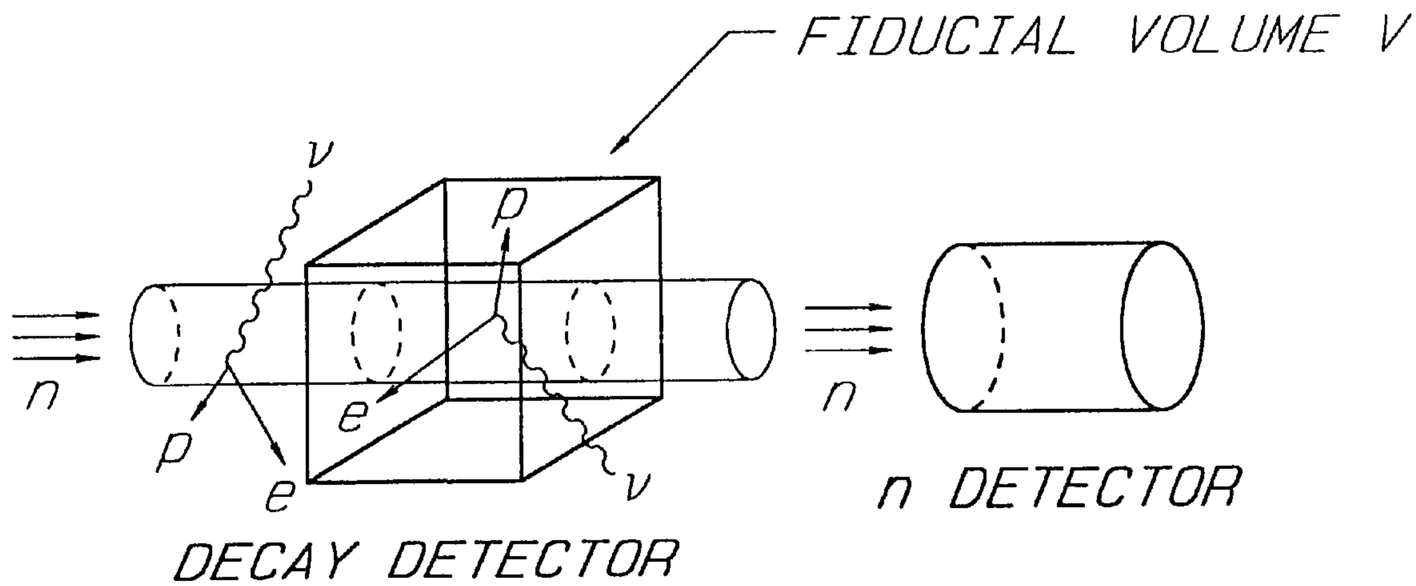
PLAN VIEW – NBSR COLD SOURCE & GUIDES

HYDROGEN COLD SOURCE THERMOSYPHON



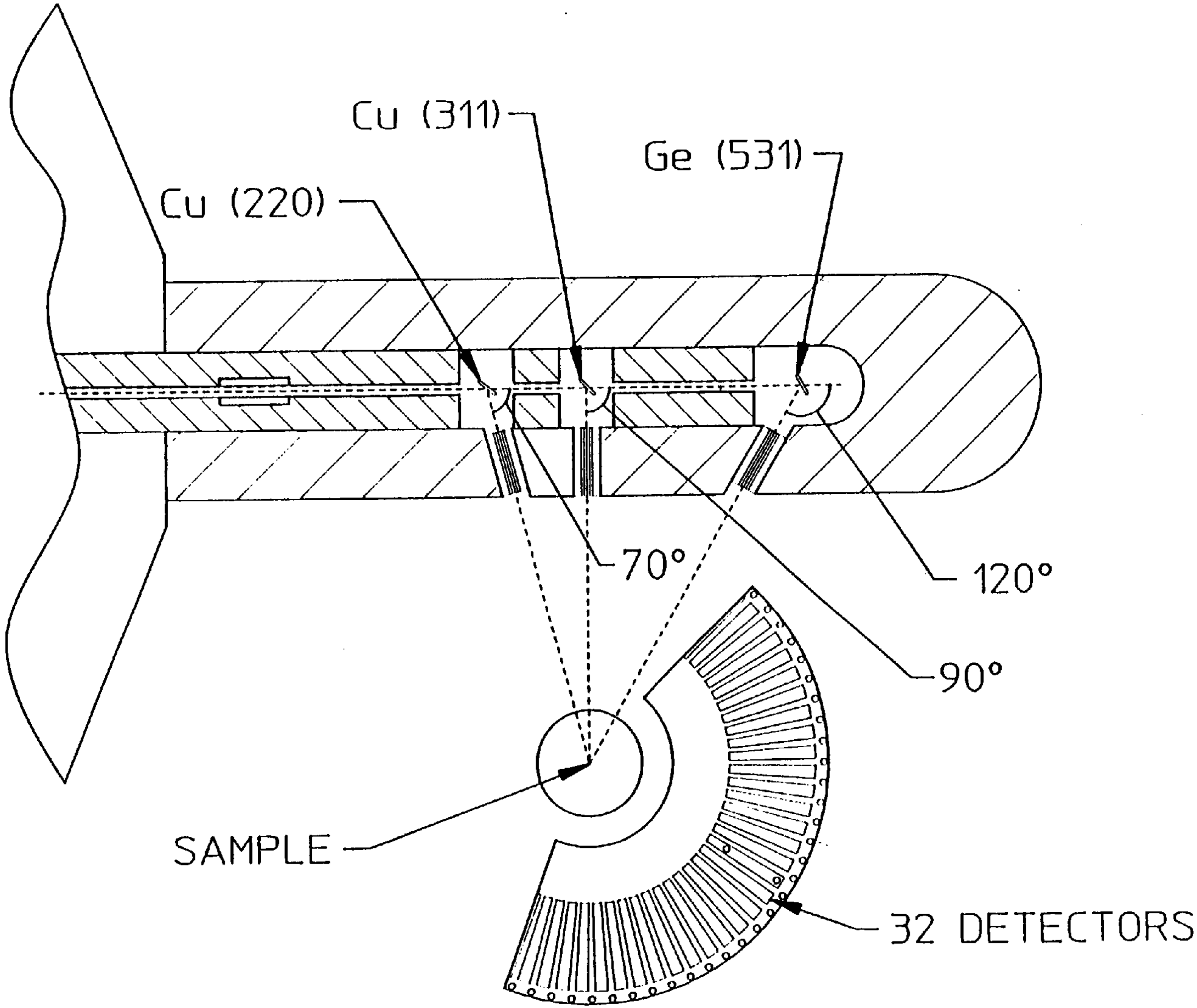
NIST / EXXON / UNIVERSITY OF MINNESOTA
SANS SPECTROMETER





$$\dot{N}_{decay} = \rho_n V \tau^{-1}$$

$$\dot{N}_{decay} = \frac{\dot{N}}{\bar{v}} V \tau^{-1}$$


Cu (311)
Ge (531)
Cu (220)
70°
120°
90°
SAMPLE
32 DETECTORS

NBSR STATUS REPORT: NAT'L. INST. OF STANDARDS & TECHNOLOGY GAITHERSBURG, MD

H. Prask, Reactor Radiation Division

HISTORY

Dec. 1967	Criticality achieved
Feb. 1969	10 MW
Feb. 1985	20 MW
Oct. 1986	Cold Neutron Research Facility approved
Jul. 1987	Cold source installed
Nov. 1987	Guide hall "Ground Breaking"
Jan. 1989	Guide hall dedicated
Jun. 1989	Shutdown for guide penetrations
Mar. 1990	Startup
Jun. 1991	Call for CNRF proposals
Nov. 1991	Begin National User Facility operation

NBSR Specifications

Power	20 MW
Peak Thermal Neutron Flux	4×10^{14} n/cm^2-s
Peak Fast Neutron Flux	10^{14} n/cm^2-s
Neutron Ports	
Beam tubes:	11
Cold source ports:	8
Thermal Column	$137 \times 132 \times 94$ cm^3 graphite
Vertical Thimbles:	17
Rabbit Tubes	4
Operating Cycle	30 days on, 7 days off

RESEARCH PROGRAMS AT THE NBSR

Neutron Scattering
- molecular solids
- magnetic materials
- H in metals
- bonding to catalysts and surfaces

Neutron Diffraction
- powder and crystal technique development
- structure determinations of
  - -ceramics, ionic conductors, alloys,
  - -zeolites, minerals, molecular solids,
  - -magnetic materials, biological materials
- phase transition mechanisms

SANS and Reflectometry
- polymer and bio-molecule structures
- suspensions under shear
- void and microstructure evolution in metals
- internal structure of thin films

Neutron NDE
- radiography and autoradiography
- residual stress & texture

Chemical Analysis
- trace analysis & depth profiling

Neutron Standards Development

Fundamental Neutron Physics
- interferometry
- lifetime of neutron

Irradiations and Isotope Production

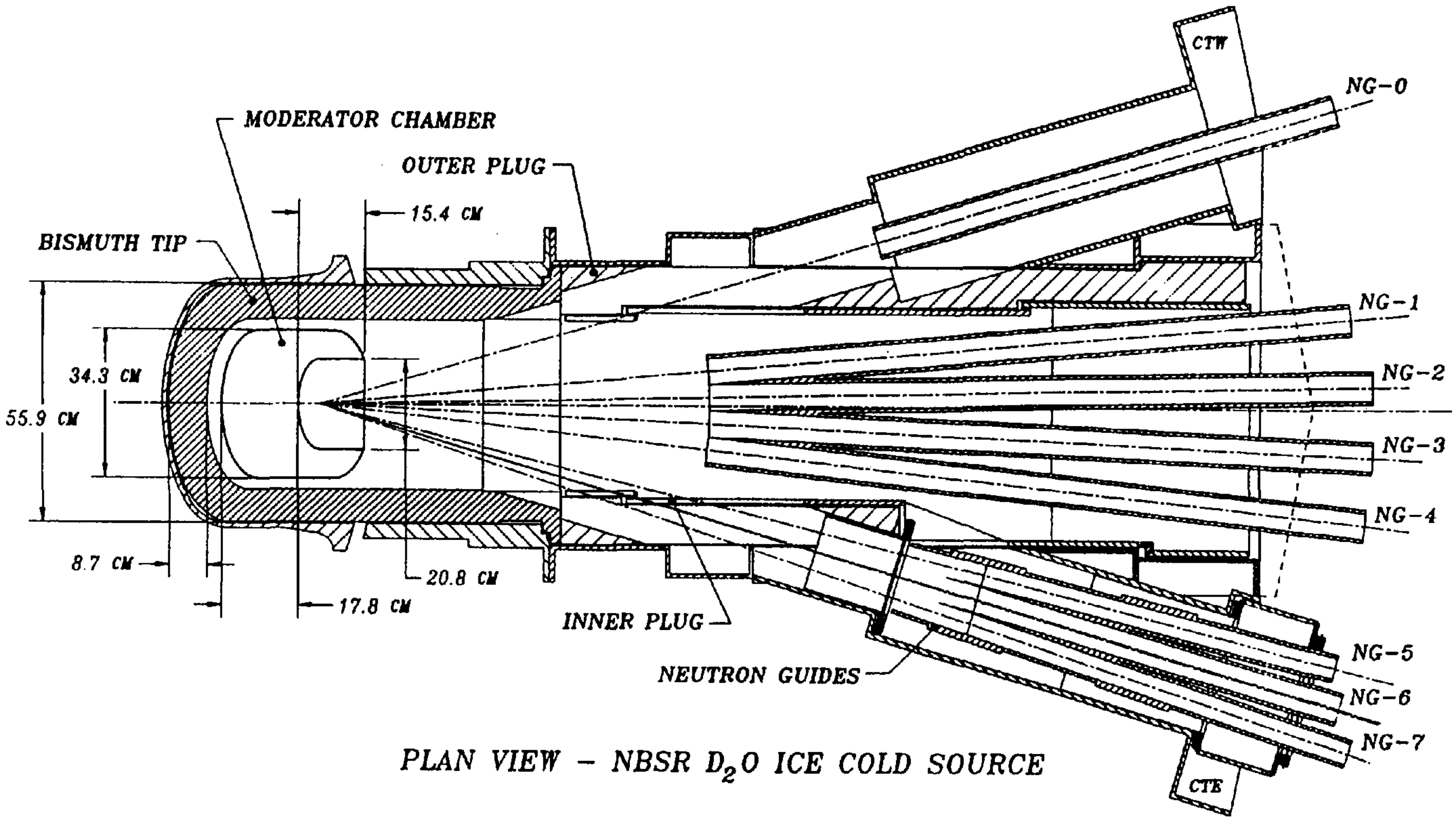
Cold Neutron Research Facility (CNRF)

- Construction continuing of $30M Facility
  - To be completed in 1993
  - 14 to 15 new experimental stations
- National User Facility
  - 1/3 of instruments funded and operated by non-NIST organizations (PRTs)
  - Program Advisory Committee allocates time on non-PRT instruments, following mail review of proposals.

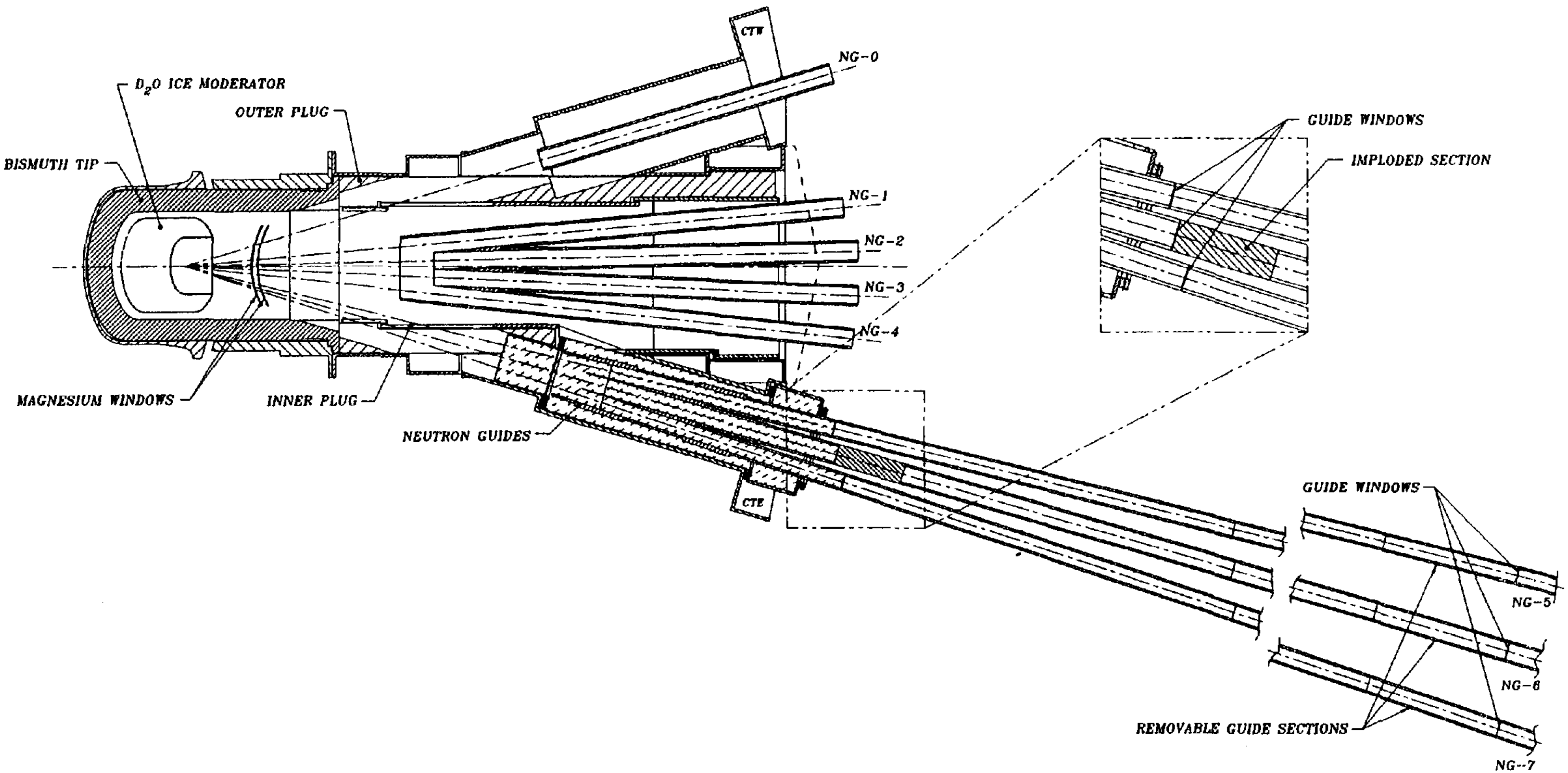
CNRF Instrumentation[1]

MATERIALS STRUCTURE			
•**NG7**	**30m-SANS**	NIST/EXXON/U.MINN.	**oper.**
•**NG3**	**30m-SANS**	NSF	1992
•**NG5**	**8m-SANS**	NIST (POLYMERS)	**oper.**
•**NG7**	**REFLECTOMETER**	IBM/U. MINN.	1992
•NG2[2]	Grazing-angle diff.		1993
MATERIALS DYNAMICS			
•**NG5**	**3-AXIS SPECTROMETER**	JOHNS HOPKINS	1992
•**NG4**[2]	**SPINS**	NSF/NIST	1993
•**NG6**	**MEDIUM-RESL. TOF**		**oper.**
•**NG1**[2]	**HIGH-RESL. TOF**	SANDIA N.L.	1993
•NG2[2]	Backscatter spect.	SANDIA N.L.	1993
•NG4[2]	Spin-echo spect.		1993
CHEMICAL ANALYSIS	KODAK/IBM/INTEL		
•**NG0**	**NEUTRON DEPTH PROFILING**		**oper.**
•**NG7**	**PROMPT-γ ACT. ANALYSIS**		**oper.**
FUNDAMENTAL NEUTRON PHYSICS			
•**NG7**	**NEUTRON INTERFEROMETER**		1992
•**NG6**	**FUNDAMENTAL NEUTRON PHYSICS**		**oper.**

[1]Hydrogen cold source to replace D_2O/H_2O source 1992.
[2]NiTi supermirror guides [$\gamma_c = 3\gamma_c$(Ni)] planned.

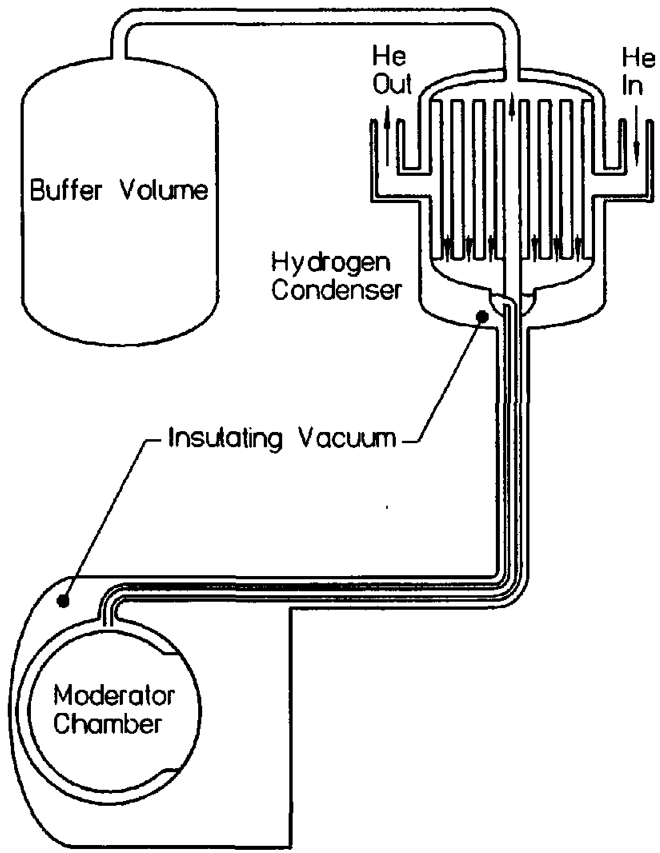


PLAN VIEW - NBSR D_2O ICE COLD SOURCE



PLAN VIEW – NBSR COLD SOURCE & GUIDES

HYDROGEN COLD SOURCE THERMOSYPHON



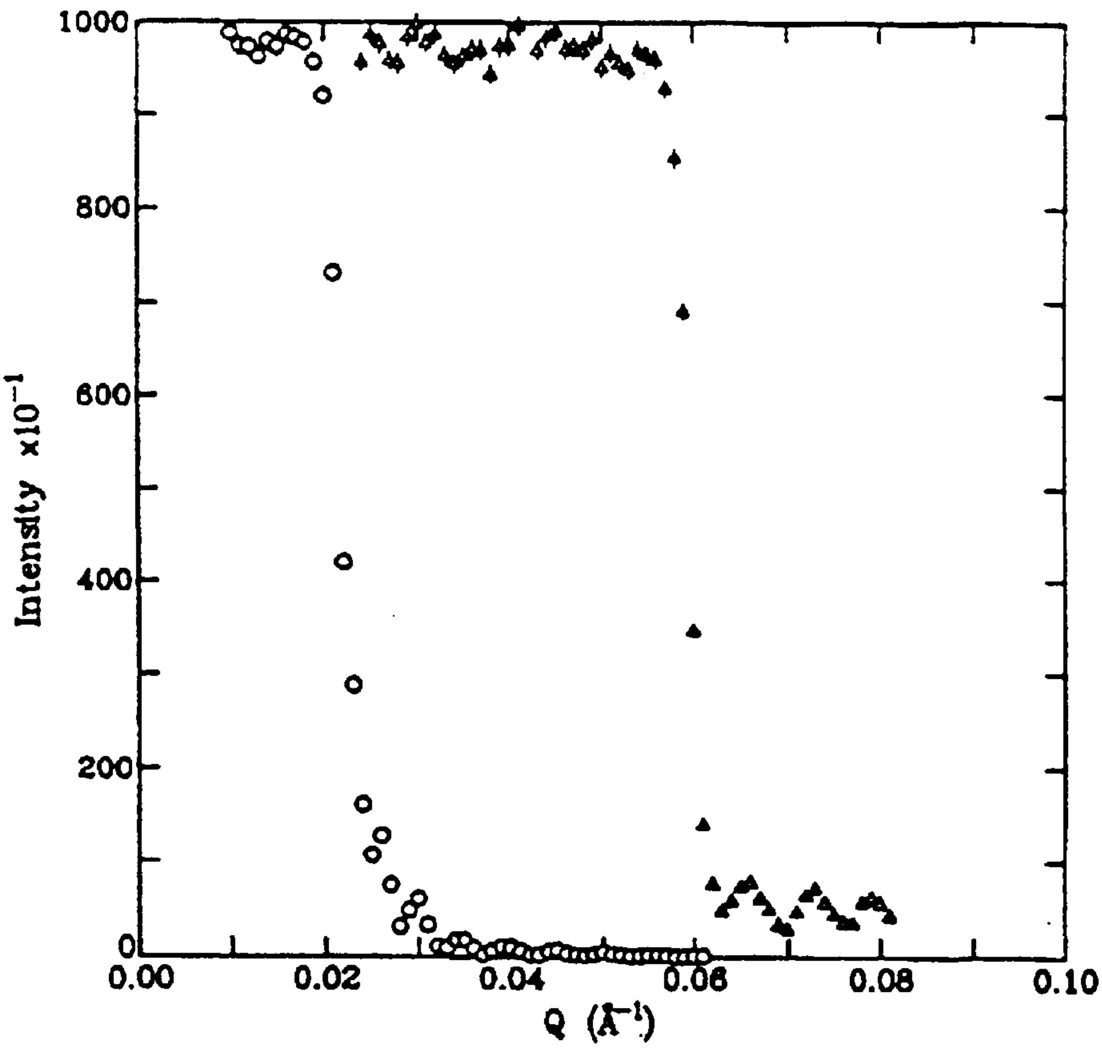
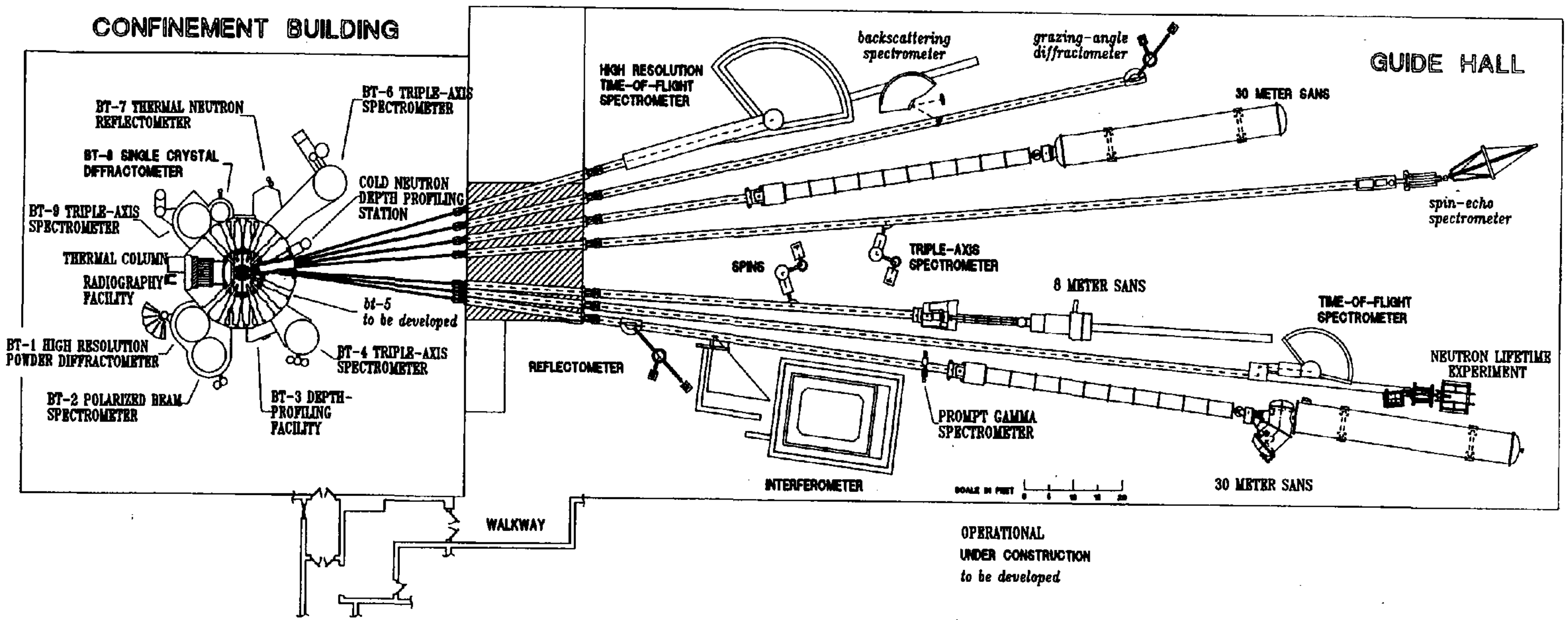


Figure 5. Neutron reflectivity as a function of wavevector transfer for a Ni-C-Ti supermirror superimposed on the reflectivity profile of a 1000-A thick Ni film. This supermirror has an effective critical angle about three times that of the Ni film and a nearly uniform reflectivity of better than 95%. This supermirror is composed of discrete sets of bilayers of different thicknesses as described in the text.

SUPERMIRROR GUIDE COATINGS

- Ovonics Corp., Optoline Corp. are developers
- NIST, ORNL, and MURR have participated
- Ovonics: carbon buffer layer between between alternating polycrystalline Ni and Ti layers <u>or</u> alternating Ni/C alloy -- Ti layers
- Optoline: amorphous Ni(x)C(1-x)-Ti(y)Mn(1-y)
- RFP out



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HIGH RESOLUTION TIME-OF-FLIGHT SPECTROMETER
backscattering spectrometer
grazing-angle diffractometer
30 METER SANS
spin-echo spectrometer
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TRIPLE-AXIS SPECTROMETER
8 METER SANS
TIME-OF-FLIGHT SPECTROMETER
NEUTRON LIFETIME EXPERIMENT
REFLECTOMETER
PROMPT GAMMA SPECTROMETER
INTERFEROMETER
30 METER SANS
WALKWAY
OPERATIONAL
UNDER CONSTRUCTION
to be developed

NBSR - Thermal Beams

BT1	5-det. Powder diff. ---> **32 det.powder diff.**
BT2	pol. beam 3-axis
BT3	Neutron depth-profiling
BT4	3-axis <u>or</u> inv. filter spect.
BT5	(to be developed)
BT6	3-axis (tex./res. stress)--->3-axis
BT7	**pol. Reflectometer**
BT8	-----------> **1-crystal diff./texture/res. stress**
BT9	3-axis

Update on the University of Missouri-Columbia
Research Reactor Upgrade

J. Charles McKibben, James J. Rhyne (University of Missouri-Columbia)

I. INTRODUCTION

The University of Missouri-Columbia (MU) is in the process of upgrading the research and operational capabilities of the MU Research Reactor (MURR) and associated facilities. The plans include an expanded research building that will double the laboratory space, the addition of new research programs, instrumentation and equipment, a cold neutron source, and improved reactor systems. These enhancements, which are in various stages of completion, will greatly expand the present active multidisciplinary research programs at MURR.

II. DISCUSSION

Current Facility

Built in 1966, MURR is a 10 MW, pressurized loop, open pool-type reactor that has operated at full power more than 90 percent of the available time over the past fifteen years. It is the highest power and most versatile research reactor located on a university campus in the world. It provides a breadth of research facilities and opportunities in the fields of nuclear-related science and engineering unequalled at any other educational institution. A sampling of programs includes nuclear medicine (radioisotope applications); archaeometry, epidemiology, and human and animal nutrition (neutron activation analysis); actinide chemistry; materials science (neutron and gamma-ray scattering and radiation effects); health physics; and nuclear engineering.[1]

Need for Upgrade

This unique facility presents to MU the opportunity and the obligation to become a leading university in nuclear related fields. The development and advancement of industries and laboratories in these areas of science and engineering as well as national projects such as the Advanced Neutron Source depend upon the availability of highly qualified and well trained personnel.

Over the past 24 years, MURR has progressed through a series of upgrades that have greatly increased its versatility for research. During 1985-89, a major, three-part plan[2,3] was developed to include:

- an increase in the reactor power level
- an addition to the research laboratory building
- a cold neutron source for enhanced beam research opportunities

With the evolution of reactor programs and MURR's recent administrative transfer to MU from the University of Missouri system, the need for an expanded research laboratory building clearly emerged as the highest priority. The reactor (as a neutron source) is not the limiting entity; it is fully capable of providing greatly expanded isotope production, sample irradiations and beam research opportunities whenever the necessary laboratory facilities become available.

Research Laboratories

As an interim solution nine new laboratories were established during 1990. An alpha laboratory was constructed for a new actinide chemistry program. With a facility ventilation exhaust system upgrade that doubled the flow rate,[4] several offices were relocated to a contiguous temporary building and four laboratories were recovered for:

- the archaeometry program
- isotope production sample preparation
- neutron activation analysis sample preparation
- cell culture work with radionuclides

Additionally four new laboratories were created in rented space to support collaborative research among MURR, MU Radiology and MU Chemistry that focuses on nuclear medicine and therapeutic application of radioisotopes.

During 1992, a third contiguous temporary building is being installed to relocate offices to free up another radioisotope applications laboratory and the space on the beamport floor for the new horizontal SANS. By early 1993, a preengineered building addition will be installed and attached to the back of the current building. This will provide 930 square meters of additional office, laboratory (dry), and shop space. This will free up more radioactive wet laboratory space that is currently used as offices and dry laboratories.

For the long term solution, MU and Sverdrup Corporation are completing a schematic design of a 8,000 square meter two-part building addition. One part will contain a neutron guide hall and low background research laboratories. The second will house research laboratories centered around new hot cells and shielded glove boxes designed for work with higher levels of radioisotopes. Together they will add around 40 new laboratories and the necessary associated offices/support spaces.[5]

Neutron Scattering Facilities

Upgrades also are being made to the neutron beam facilities and instrumentation. A new high resolution powder diffractometer (PSD-II) and a second neutron interferometer have been installed. Three neutron beam facilities are being built and two additional instruments are in the design or development stages:

- TRIAX, a joint MU-Ames Laboratory triple axis spectrometer
- a new 15 meter horizontal small angle neutron scattering instrument (SANS)
- a neutron reflectometer
- conversion of a long wavelength, five-element diffractometer to a dedicated residual stress instrument
- a double perfect crystal sans diffractometer

Cold Neutron Source

To meet the rapidly increasing demand for long wavelength neutrons, the MURR upgrade includes plans for adding a cold neutron source (CNS) facility. The preliminary CNS designs from three suppliers of this equipment (Technicatome, Interatom, and AECL) were evaluated to determine feasibility and cost.[2] The schedule for the CNS is dependent on funding.

Reactor Systems

In August 1990, culminating a four year review process, the NRC approved the new Extended Life Aluminide Fuel (ELAF) element for the MURR.[6] The first elements are tentatively scheduled to be received in 1993. This new fuel design will cut the fuel cycle cost by 50 percent for the current 10 MW power level by at least doubling the allowable burnup per fuel element. The variation in uranium loading density between plates provides a much more uniform power density in the core. Feasibility studies based on the ELAF core show that power can be increased to between 20 to 30 MW.[7,8,9] The detailed safety analysis for the power increase will be submitted to the NRC in 1994.

The reactor instrumentation and control (I&C) and electrical systems are being upgraded with the following new systems installed or in process:

- 275 KW emergency generator–1989
- uninterruptable power supply–1989
- area radiation monitoring system–1990
- stack monitor–1991
- nuclear instrumentation systems–being installed in 1992
- control rod drive mechanisms–funding requested for 1993

III. CONCLUSION

MU is investing its resources for a significant expansion of the research capabilities and utilization of MURR to provide it the opportunity to deliver on its obligation to become the nation's premier educational institution in nuclear related fields which can provide scientific personnel and a state-of-the-art research test bed to support the Advanced Neutron Source.

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TRIGA HEXAGONAL FUEL ELEMENT DEVELOPMENT

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ABSTRACT

General Atomics has continued development of a fuel element using a hexagonal array of fuel pins. This element builds on existing, TRIGA fuel technology by using the well-proven uranium-zirconium hydride fuel material in 13.8 mm OD fuel pins. The use of a hexagonal fuel pin array instead of a square array permits the core design to have a higher power density and higher neutron flux per watt. For example, a 10 MW reactor can provide a peak in-core thermal neutron flux of 3 x 10^{14}n/cm^2-s and a thermal flux at the beam port entrance greater than 10^{14}n/cm^2-s.

INTRODUCTION

Improvements in the capabilities of research reactors invariably involve attempts to increase the neutron flux. Higher fluxes offer reduced irradiation times and increased ability to examine effects not seen at lower flux levels. The principal requirements for a research reactor are neutron fluxes over certain energy ranges (usually peak values of thermal or fast neutrons at certain locations, e.g., beam port entrances or in-core) and the number and size of in-core irradiation locations. These specifications are typically derived from a survey of the expected usage of the reactor. The reactor power level, although often specified first, is actually a result of the design studies which determine the acceptable maximum power density and the number and arrangement of fuel elements. For reactor designs that use TRIGA LEU fuels, the limiting power density will be set by the maximum allowable operating fuel temperature and not by reactor power.

The uranium-zirconium hydride ($UZrH_x$) fuel material is the fundamental feature of TRIGA reactors that accounts for its well recognized safety, good performance, and economical operation. For higher power reactors that are expected to have high duty cycles erbium is also included in the alloy as a burnable poison material. This allows very high uranium loadings and very long core lifetime. The erbium loading is typically varied between 0.5 wt-% and 1.5 wt-%.

Since the erbium is also contained in the fuel material together with most of the moderating hydrogen (the rest is, of course, in the coolant water), it enhances the prompt negative temperature coefficient of the TRIGA fuel, allowing the safe insertions of much larger positive reactivity than would otherwise be tolerated. This feature protects the TRIGA reactor against unplanned reactivity excursions. Figure 1 shows the absorption cross-section for Er-167 and the TRIGA neutron flux at two fuel temperatures.

HEXAGONAL ELEMENT DESIGN

In our effort to design a fuel element that would yield higher neutron fluxes it was decided to base the design, insofar as possible, on well proven technology. This led to the choice of the TRIGA LEU half-inch (actually 13.8 mm) diameter fuel pin that had been qualified* at Oak Ridge under the RERTR program. The same fuel rod design with HEU accumulated more than ten years of successful operation in the 14 MW TRIGA reactor at Pitesti, Romania. The LEU pins were tested in the ORR over the range of fuel loadings (20, 30, 45 wt-%) to burnup values reaching 66% averaged over the entire fuel pin. Peak values of up to 83% burnup were attained in the Romanian reactor without pin/clad interaction or other fuel performance problems.

Figure 2 is a drawing of this fuel pin. Using uranium with an enrichment of 19.75% (nominal value) these fuel pins have been fabricated with loadings of 20, 30, and 45 wt-%. A

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*TRIGA LEU fuel was the first LEU fuel qualified under the RERTR program and approved by the USNRC single 45 wt-% pin, for example, contains 54.3 g of U-235. A very important characteristic of this fuel is that the most heavily loaded fuel (45 wt-%) has a uranium volume fraction of only about 20%. This is well below manufacturing limits and offers the possibility of further increases in loading and power density.

It is well known that a hexagonal array of fuel pins offers the core designer neutronic and thermal-hydraulic advantages compared to a square array but at the price of increased complexity of fabrication. The hexagonal in-core irradiation locations also more closely approximate a cylinder, the typical geometry for capsules and targets. After several design iterations we selected a hexagonal array that used the same minimum spacing (0.101 inches, or 0.26 mm) between rods in the existing, well-proven square array. This simplified some of the thermal hydraulic calculations and, again, allowed us to base the design on existing, well-proven technology. The number of fuel pins per element was selected at 19 although 7 or 37 could also have been chosen. This yields an element that has a conveniently small size to allow "building" of a core but is large enough to reduce the number of fuel elements to be handled.

A plan view of the standard element is given in Figure 3 and an assembly drawing of the entire element is given in Figure 4. Figure 5 shows a control element that could be used with this design although it should be noted that control rod design studies and thermal-hydraulic calculations are still in progress. Table 1 summarizes the main parameters for the hexagonal element.

Some of the important design aspects of the hexagonal fuel element concern the grid spacer design, the shroud construction, and the bottom end fitting. Each spacer grid consists of a simplified parallel beam configuration, Fig. 3A, with four beams spacer strips either welded or brazed to an outer hexagonal band. Therefore, at each grid elevation the fuel rods are spaced in one direction. To provide spacing in the other two directions the adjacent grids are rotated 60° as shown in Figs. 3B and 3C. Total fuel rod spacing within the bundle is maintained by a grouping of three individual spacer grids (see Fig. 4).

The major advantage of this spacer grid system is simplicity which facilitates fabrication. Standard brazing or welding methods can be used without complicated and expensive tooling which are normally associated with other spacer grid designs. In addition, because of the low cost and excellent thermal-hydraulic characteristics, a relatively large number of grids can be included in a bundle design without severely limiting the bundle flow characteristics. The actual number and axial location of the grids will be experimentally determined by flow tests.

Since hexagonal channel is not commercially available in aluminum alloy of the size required for the ideal fuel pitch, the shroud will be manufactured using two aluminum alloy plates 1 mm (.039 inches) thick, bent to the configuration shown in figure 4 and seam welded down the length of the assembly on the two sides as shown. The Inconel 718 grids (7 per fuel bundle, see Figure 4) will be positioned one at a time from the bottom up and then riveted in placer using the predrilled holes in the positions as shown. The top nozzle is integral with the top of the shroud incorporating the handling holes for handling the assemblies both in and out of the core. The mechanical design of these hexagonal fuel elements is such that fuel pins may be removed and replaced without any other disassembly being required.

The bottom end fitting provides positioning into the lower grid plate and also flow control. The latter is particularly important for control rod fuel elements to avoid major perturbations to the core flow distribution and amount of bypass flow when the control rods are moved. It is envisioned that the control rods will have non-fueled followers to control the by-pass flow and limit undesirable flux peaking. The bottom end fittings, therefore, will have to accommodate the followers whose design will be dictated primarily by neutronic considerations.

HEXAGONAL CORE DESIGN

The hexagonal fuel element offers a high degree of flexibility in the design of a research reactor core. Needless to say, there are a large number of possible designs when requirements for irradiation locations, beam tube interfaces, and control rod locations are taken into account. One arrangement for a 10 MW core is shown in Figure 6. This particular design has 17 standard fuel elements, 3 in-core irradiation locations and 5 fuel elements with control rods. The Be reflector on the west side serves as the interface with the beam tubes. Additional beam ports, both tangential and radial, can be accommodated around the core. The Be reflector on the east side serves as a coupling to a D_20 tank. This coupling allows the design of the D_20 tank that encloses the cold source to be simplified. It probably can be designed as a nearly rectangular box which should facilitate fabrication and inspection.

The estimated peak thermal flux (in a central H_20 hole) in this core operating at a power of 10 MW is about $3.5 \times 10^{14} n/cm^2$-s and the thermal flux in the reflector (at a typical beam port entrance) is approximately $1.7 \times 10^{14} n/cm^2$-s.

SUMMARY AND CONCLUSIONS

General Atomics' TRIGA Group is continuing development of a hexagonal array high performance fuel element for future research and test reactors. The mechanical design of the standard element, based on the existing ½ inch LEU fuel pin is complete and the design for the control rod element is nearing completion. Remaining studies include an examination of various control rod designs, completion of core flow calculations, and confirmatory full scale flow tests.

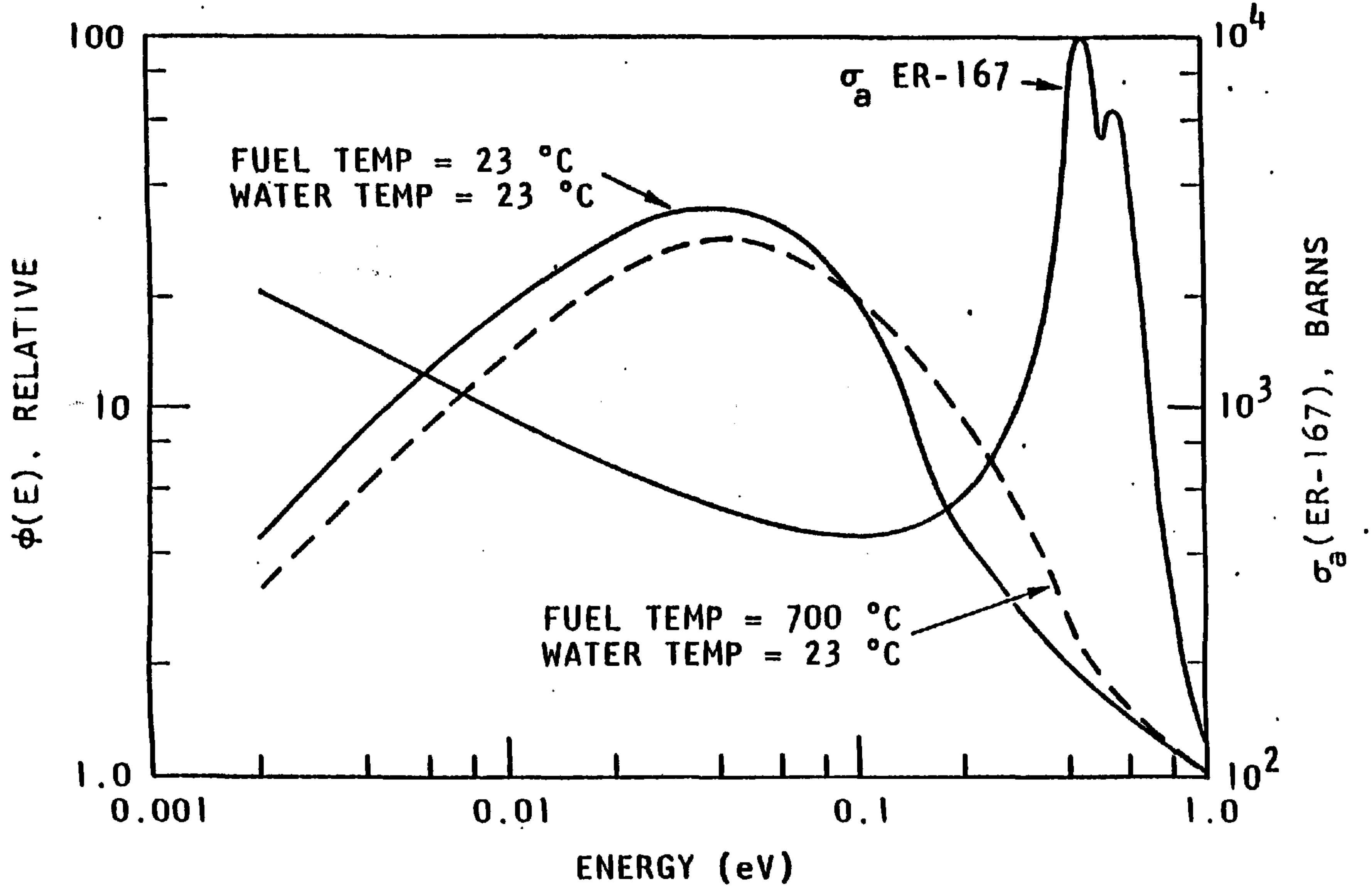

Figure 1. TRIGA Thermal Neutron Spectra vs. Fuel Temperature relative to σ_a vs. Energy for Er-167

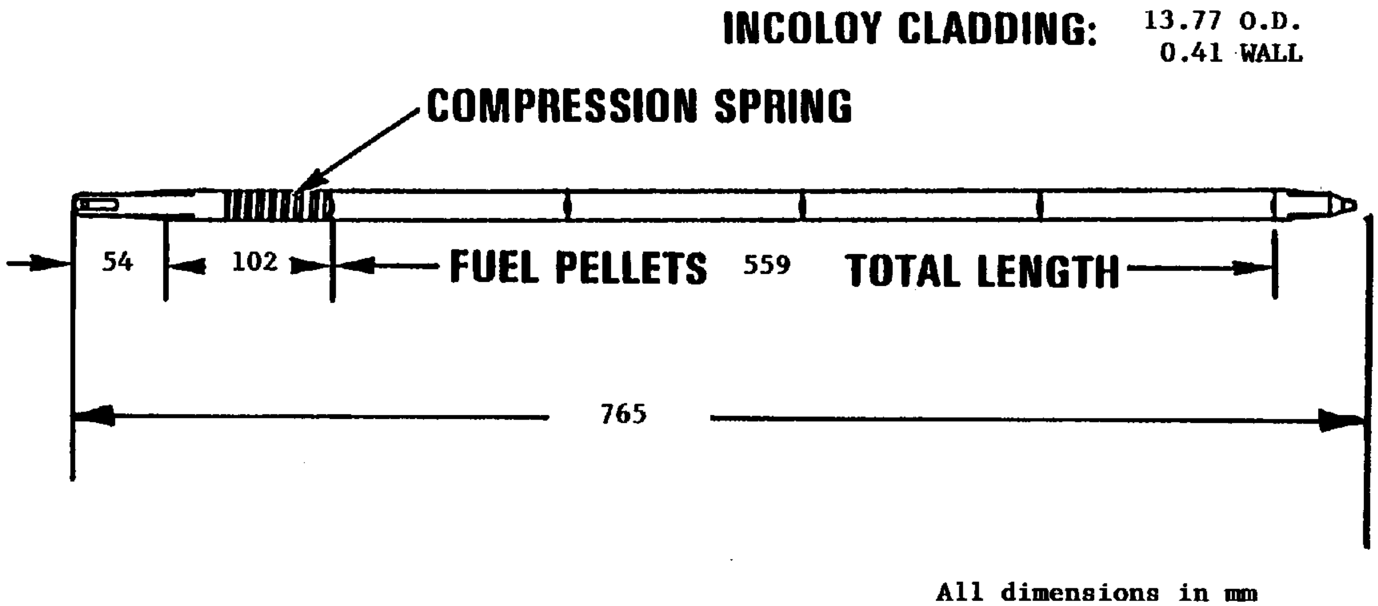

Figure 2. TRIGA 1/2 INCH FUEL ROD

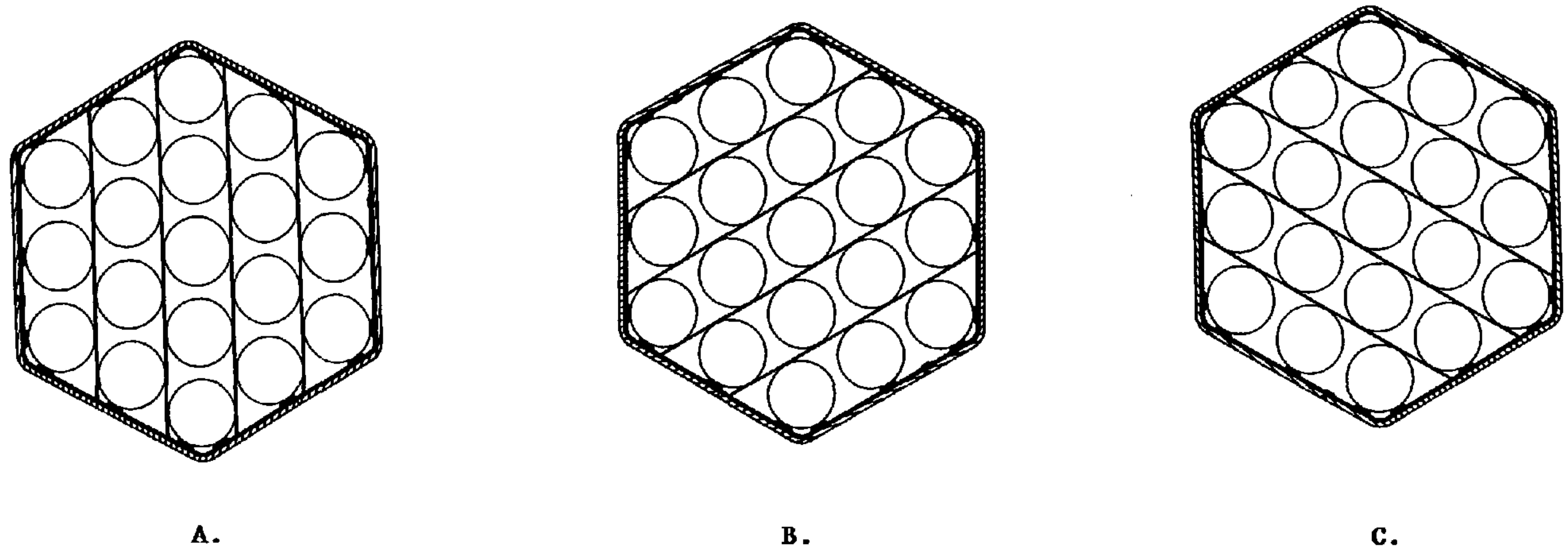

Figure 3. Arrangement of Standard Hexagonal Fuel Element Spacer Bars at 3 Axial Locations

GENERAL ATOMICS

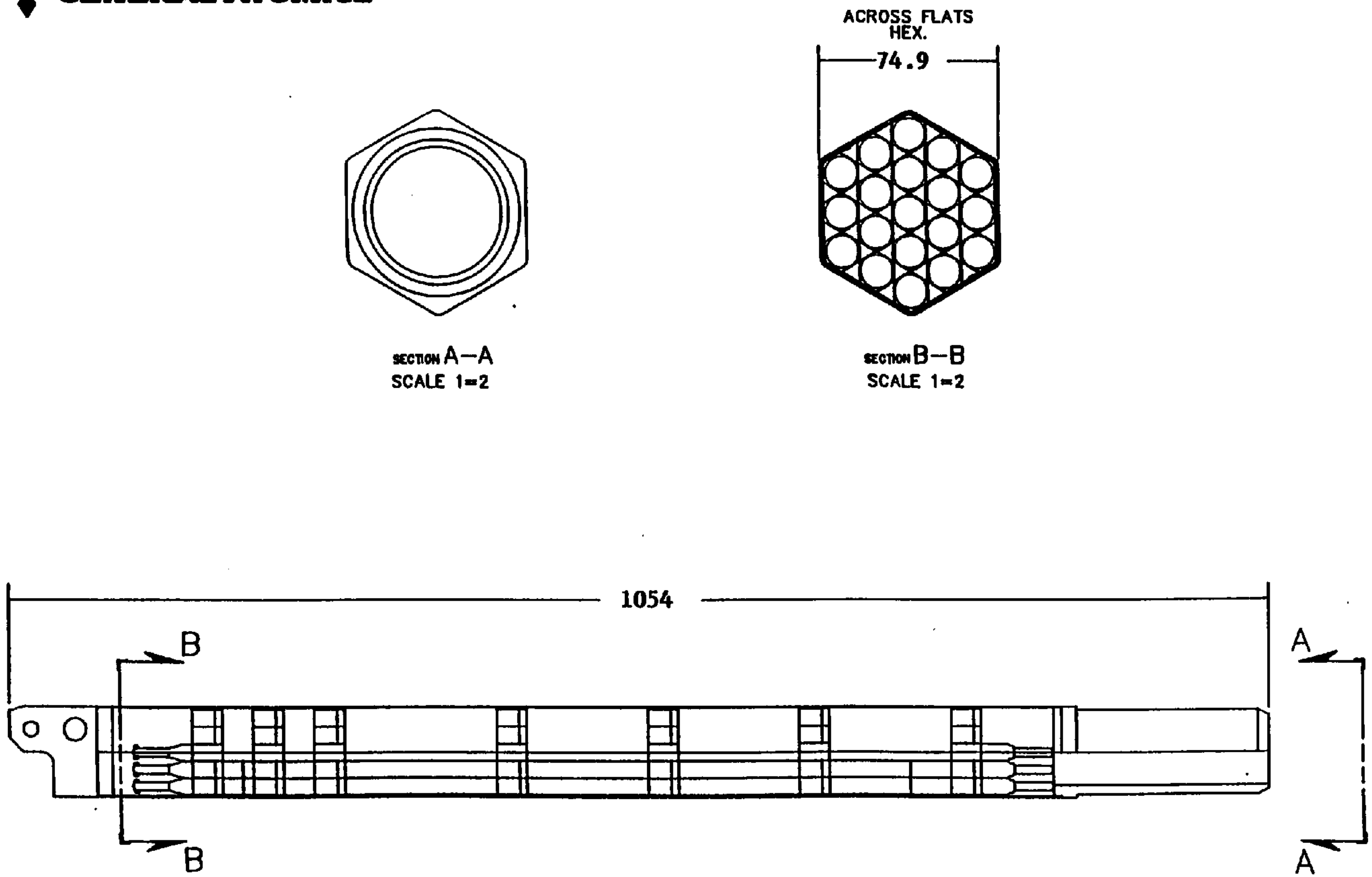

Figure 4. Hexagonal Fuel Element Assembly

GENERAL ATOMICS

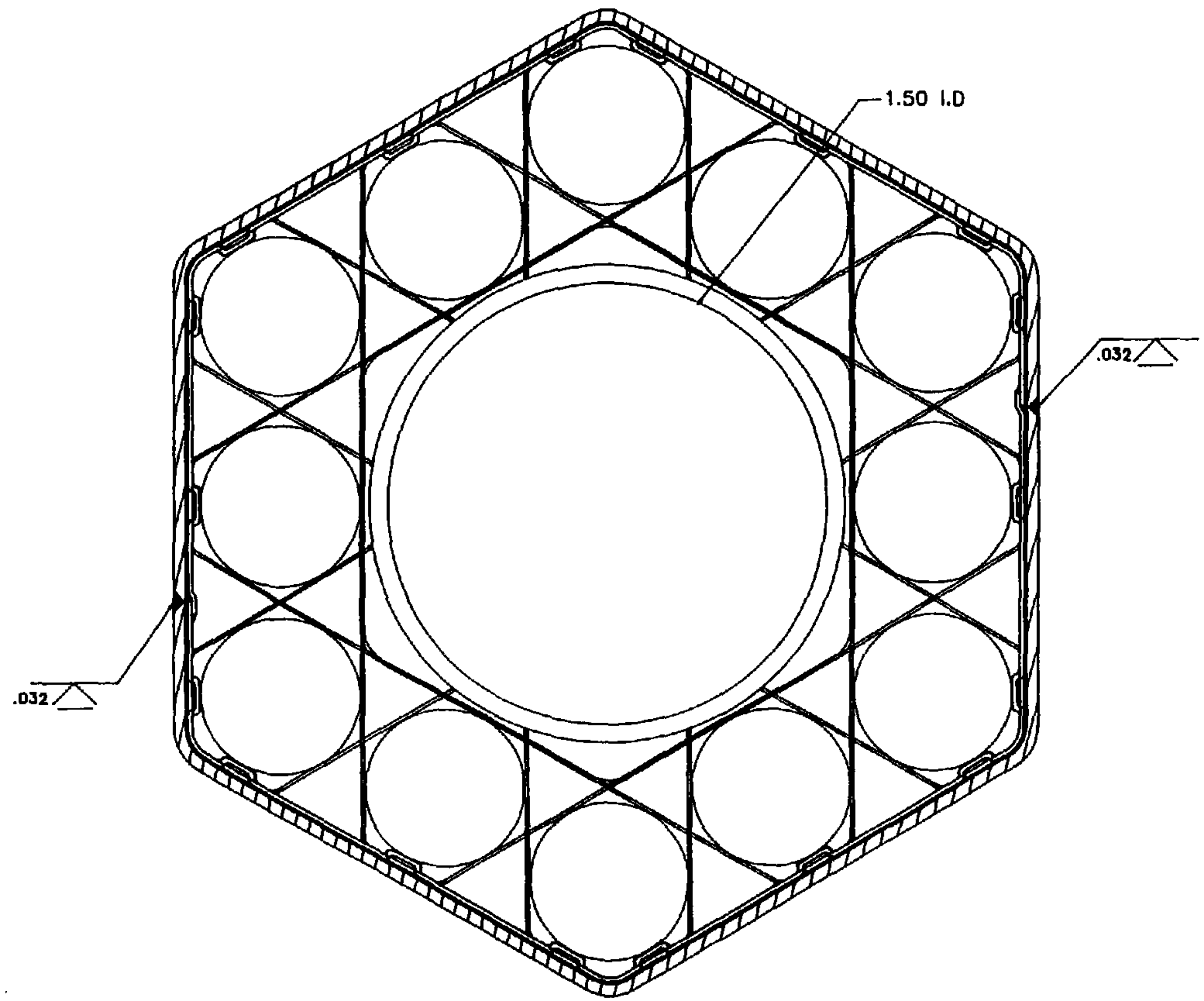

Figure 5. Hexagonal Control Rod Fuel Element

Table 1

Hexagonal Fuel Element Parameters

Fuel pin OD (mm)	13.8
Fuel composition	U-$ZrH_{1.6}$ Er
Weight of U-235 (g) - std. FE	1032
Uranium loading (wt-%)	45
Uranium enrichment (%)	19.75
Active length (mm)	559
No. pins / standard element	19
No. pins / control element	12
Pin-Pin pitch (mm)	16.3
Minimum rod spacing (mm)	2.56
Distance across flats (mm)	74.9
UZrH/H_20 ratio	1.41
Cladding material	Incoloy 800
Cladding thickness (mm)	0.41

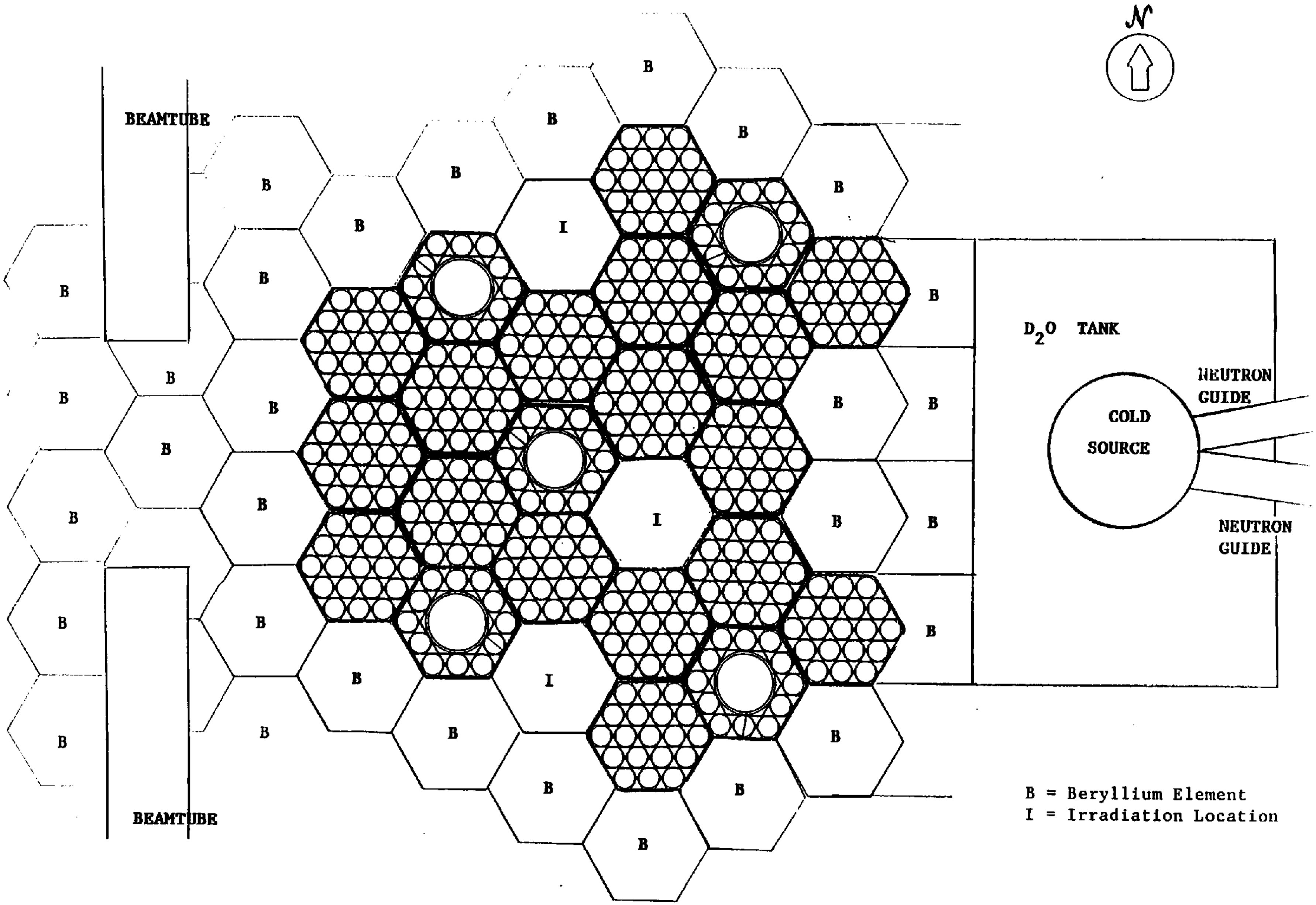

FIGURE 6. 10 MW CORE ARRANGEMENT USING HEXAGONAL FUEL ELEMENTS

Centre d'Etude de l'Energie Nucléaire
Studiecentrum voor Kernenergie
BR2 Department

IGORR-II Meeting
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Saclay, FRANCE

REFURBISHMENT PROGRAMME

FOR THE BR2-REACTOR

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I. Introduction

BR2 is a high flux engineering test reactor, which differs from comparable material testing reactors by its specific core array (fig. 1).

It is a heterogeneous, thermal, tank-in-pool type reactor, moderated by beryllium and light water, which serves also as coolant. The fuel elements consist of cylindrical assemblies loaded in channels materialized by hexagonal beryllium prisms. The central 200 mm channel is vertical, while all others are inclined and form a hyperbolical arrangement around the central one. This feature combines a very compact core with the requirement of sufficient space for individual access to all channels through penetrations in the top cover of the aluminium pressure vessel. Each channel may hold a fuel element, a control rod, an experiment, an irradiation device or a beryllium plug.

II. The refurbishment programme

According to the present programme of C.E.N./S.C.K., BR2 will be in operation until 1996. At that time, the beryllium matrix will reach its foreseen end-of-life. In order to continue operation beyond this point, a thorough refurbishment of the reactor is foreseen, in addition to the unavoidable replacement of the matrix, to ensure quality of the installation and compliance with modern standards.

Some fundamental options have been taken as a starting point: BR2 will continue to be used as a classical MTR, i.e. fuel and material irradiations and safety experiments with some additional service-activities. The present configuration is optimized for that use and there is no specific experimental requirement to change the basic concepts and performance characteristics. From the customers viewpoint, it is desirable to go ahead with the well-known features of BR2, to maintain a high degree of availability and reliability and to minimize the duration of the long shutdown. It is also important to limit the amount of nuclear liabilities.

So the objective of the refurbishment programme is the life extension of BR2 for about 15 years, corresponding to the expected life of a new beryllium matrix with the present operation mode.

The underlying assumption is that the extent of refurbishment-work should be consistently minimized with maintaining a reliably operating reactor and without, in any way, prejudicing safety standards.

III. The phase 1 study

A phased approach has been adapted as outlined in fig. 2. The objectives of phase 1 were:

- to give an overview of the refurbishment needs, i.e., compile a comprehensive list of plant items, identify the critical items and carry out a preliminary assessment of them against some agreed on criteria of aging, safety and upgrading,

- to provide confidence that by making commitments towards operating beyond 1996, the investment will not be jeopardized by the need for supplementary work or major failures,

- to define a schedule of work for all plant items to be assessed and inspected in phase 2, and to provide an estimate of costs and timescale for the overall refurbishment programme.

The approach chosen was to comment on all systems, but concentrate on items or sub-systems, that are most fundamental.

Fundamental critical items are the safety-critical ones, and those whose failure, or if refurbishment requirements dictate replacement or major intervention, could lead to significant cost or prolonged shutdown of the reactor.

According to these criteria, some items could be excluded from consideration in this context. At the first stage, the depth of consideration in subsidiary systems is limited. Actually those items and systems are adressed by the preventive maintenance programme.

The requirements on the critical items were determined by aging considerations, possible needs for safety improvements and opportunities to upgrade particular features.

Aging criteria are essentially the same as the ones used in the prevention maintenance in use since 1988. Maintenance staff are in the best position to judge the reliability of the equipment, yet they may lack best judgement on long-term fitness. For that reason, some additional inspections by external specialists will be foreseen in phase 2.

Safety criteria concern comparisons against "modern thinking" in the broad sense. Besides the official requirements written down in the operating license, a review on international criteria (e.g. IAEA), requirements for other MTR´s and power reactors has been done and some supplementary requirements have been adopted in accordance with the licensing authority. A general theme seems to be the link between PSA and modern criteria.

Concerning upgrading activities, the possibility will be considered and a cost/benefit balance will be made. Although the assumption is made that there are no requirements to change the basic concepts, there may be other reasons for upgrading, like availability of modern technology, reduction of operating costs, ergonomic and "presentational" improvements ...

The Phase 1 study has been undertaken with the assistance of AEA Technology Engineering Business and took about 4 months. The licensing authority has been closely involved from the start.

IV. Phase 1 conclusions:

The conclusion of the phase 1 study was that BR2 is in good physical condition, but some significant items of work have been identified and will be required for operation after 1996. A list of the phase 2 work packages is given in fig. 3. Some urgent actions which will determine options and refurbishment/ replacement work are presently being undertaken. They concern major components of the reactor:

- the question whether or not to replace the pressure vessel will have to be answered definitely. Although there is some concern about embrittlement under irradiation, some arguments have been put forward which tend to indicate that the vessel is in a good state of conservation and will not have to be replaced.

- whether a new matrix is needed or the matrix of the zero power nuclear model BR02 can be used, is another question which will have to be answered as soon as possible.

A probabilistic safety assessment (PSA) will be carried out to determine if there are any system weaknesses and because such an assessment is considered an emerging requirement for all types of reactors. Possible design changes will be deferred until the preliminary conclusions of the PSA are known.

Other areas, like control-room arrangements and fire mitigation, will also be considered as to their compliance with "accepted" modern standards.

Finally, some specific areas, like spent fuel storage, will have to be adressed by long-term solutions in order to secure the future.

V. Future programme

The phase 2 work items, i.e. assessments and inspections listed in fig. 3, are foreseen to be accomplished in about 1.5 years on a project management basis.

The outcome of phase 2 will be:

- a clear definition of the actual refurbishment needs,
- a definite decision on issues still left open at the end of phase 1,
- a precise estimation of the overall costs,
- a planning and organization scheme for phases 3 and 4.

It is important that phase 2 is executed without delay as it is not only desirable to remove the existing uncertainties as soon as possible, but also to leave sufficient time in phase 3 to procure any major items of hardware that may be required.

In phase 3, design, procurement and some installation work compatible with normal shutdowns will take place.

Phase 4 work covers everything to be done during the long refurbishment shutdown, including recommissioning.

VI. Conclusions

The refurbishment needs for the BR2 reactor, if it is to operate beyond the end of life of its current beryllium matrix, have been investigated.

A phased refurbishment programme is envisaged. The first phase of this programme has been executed and produced quite encouraging results.

Phase 2 inspections and assessments are presently being undertaken in order to define more precisely the extent of actual refurbishment work needed. A definite decision to operate BR2 beyond 1996 is expected on completion of phase 2, after the key assessments have been completed and when overall costing can be given with a higher degree of confidence.

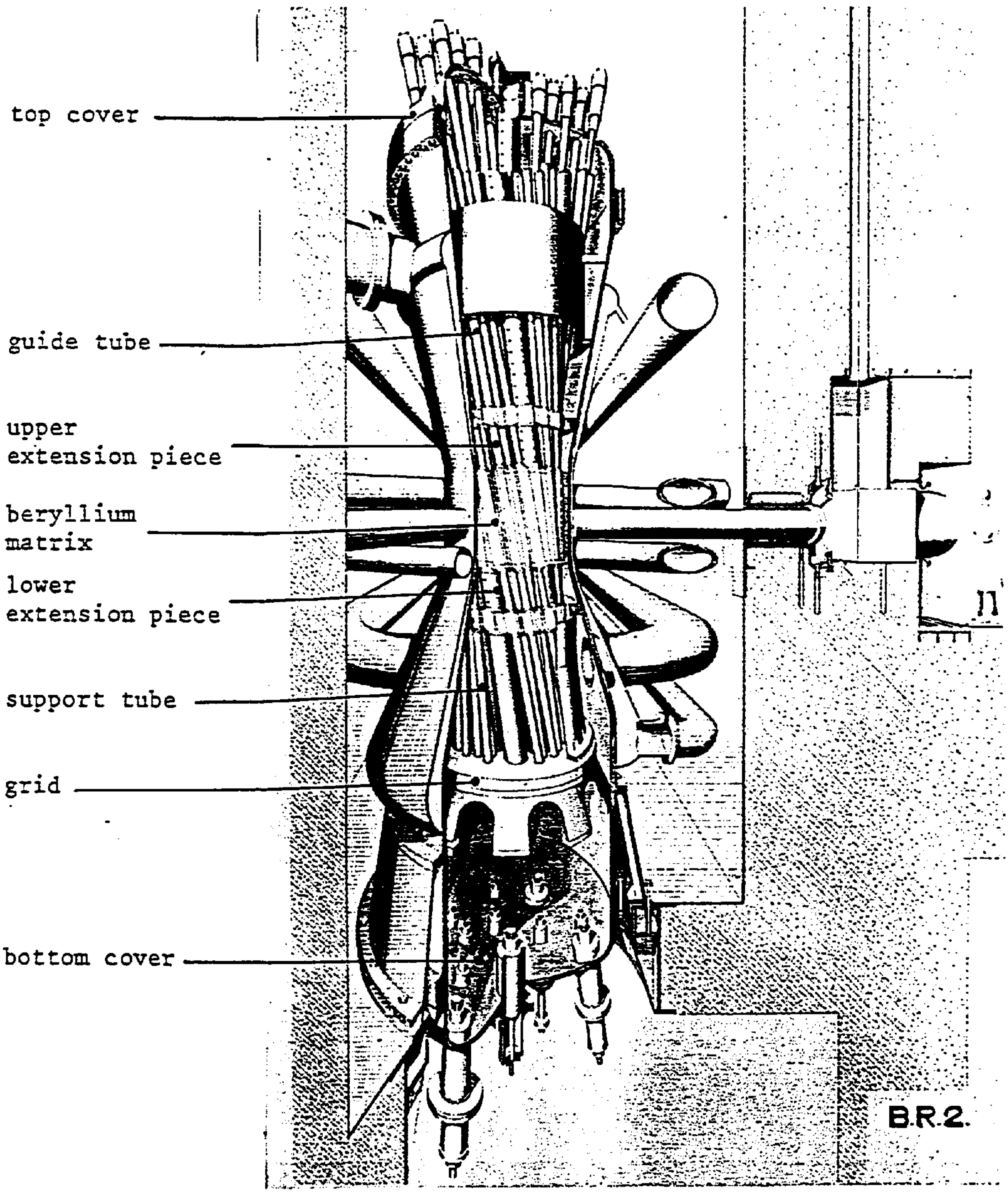

FIG.1: General view of the BR2 reactor

Fig. 2: BR2 REFURBISHMENT STUDY

Overall Objective

´to carry out the studies and refurbishment necessary to keep BR2 operational beyond 1996 and to approximately 2010´.

PHASE 1

list areas of further work, preliminary programme and costing

PHASE 2

detailded inspections, assessments, planning and costing

PHASE 3

design, procurement and some installation

PHASE 4

long shutdown-replacement, installation and recommissioning

Fig. 3: PHASE 2 Work Packages

1.	Probalistic Safety Assessment	Assessment
2.	Structural Inspection of Buildings	Inspection
3.	Fatigue Analysis of Vessel	Assessment
4.	Early Inspection of Vessel	Inspection
5.	Assessment of BR02 Matrix	Dismantle, inspect & assessment
6.	Inspection of Reactor Pool	Inspection
7.	Inspection of Pipework	Inspection
8.	Inspection of Storage Pool	Inspection
9.	New Spent Fuel Store Study	Study
10.	Control Room Study	Study
11.	Electrical Inspection	Inspection
12.	Fire Protection Study	Study
13.	Seismic Assessment	Assessment

SIRIUS 2

A VERSATILE MEDIUM POWER RESEARCH REACTOR

by PASCAL ROUSSELLE
TECHNICATOME

1. FOREWORD

Most of the Research Reactors in the world have been critical in the Sixties and operated for twenty to thirty years. Some of them have been completely shut down, modified, or simply refurbished; the total number of RR in operation has decreased but there is still an important need for medium power research reactors in order :

- to sustain a power program with fuel and material testing for NPP or fusion reactors;

- to produce radioisotopes for industrial or medical purposes, doped silicon, NAA or neutron radiography;

- to investigate further the condensed matter, with cold neutrons routed through neutron guides to improved equipment;

- to develop new technologies and applications such as medical alphatherapy.

Hence, taking advantage of nearly hundred reactor x years operation and backed up by the CEA experience, TECHNICATOME assisted by FRAMATOME has designed a new versatile multipurpose Research Reactor (20-30 Mw) SIRIUS 2 taking into account

- More stringent safety rules;

- the lifetime;

- the flexibility enabling a wide range of experiments and,

- the future dismantling of the facility according to the ALARA criteria.

II. KEYWORDS FOR SIRIUS 2 DESIGN

SIMPLICITY -------> * a reasonable cost for :

. construction
. operation.
. dismantling

VERSATILITY/FLEXIBILITY

------> * It can evolves with scientific and/or technologic research programs. (Possibility of adding new options during reactor lifetime with the minimum of work)

QUALITY ASSURANCE

-----> * Use of qualified team for design
* Use of qualified technology for equipment (qualification mainly based upon CEA's experience)

SAFETY

(last but not least) -----> * application of up-to-date safety rules/ recommandations for R.R.
* ALARA principle

LEADING IDEAS FOR THE PROJECT

* **EXPERIENCE** (based on main CEA's R.R.)

SILOE
1962/1987 >>

- Plate type Fuel Elements
- Core configuration, accessibility and flexibility
- Core block structure
- open pool
- downward core cooling
- experimental devices

OSIRIS
1966 >>

- experimental loops (IRENE, ISABELLE)
- hot cells

ORPHEE
1980 >>

- pool liner (Borax type accident proof)
- nuclear ventilation
- containment penetrations
- NGB (option)

* IMPROVEMENTS (EXAMPLES)

NON PROLIFERATION	- LEU fuel
SAFETY	- Up-to-date recommandations and rules - Maximal confinement of radioactive products inside B.R. - Physical segregation of the reactor protection systems - Emergency control panel - Reinforced protection against external aggressions for control room & protection systems (civil work)
QUALITY ASSURANCE	- QA plans for design, construction and erection of the facility
LARGE POSSIBILITIES OF USES	(according to options) - Neutron beams (up to 4/5) - In-core and out of core irradiation positions - Casemates for pressurized loops - NGB - BNCT building - Heavy water reflector - Cold source ...
NEW TECHNOLOGY	- When it leads to progress (digital technology for some I&C control system)
SIMPLIFICATIONS	For operation/modification/ Dismantling : - General lay-out (accessibility) - 2 primary coolant loops - 1 decay tank ...

III. TECHNICAL OPTIONS

	BASIC PROJECT	MAIN OPTIONS	OTHER POSSIBLE OPTIONS
Thermal power	25 MWth		30 MWth : * adaptation of coolant circuits (civil work and biological shielding designed for 30 MW)
Reflector	* Be elements (without cold source)	* 4 Be elements with diam.32 mm	* D_2O tank
Core configuration	* Standard : SILOE type . 6 irradiation elements . 5 control elements . 22 standard elements		* Boronated core
Neutron beams	* 2 tangential * 2 radial	* <u>Cold source</u> located in a Be block	* <u>Cold source</u> in D_2O tank * hot source * BNCT * neutron guide Bdg * neutronradiography
Permanent auxiliaries	* 1 hot cell : . dismantling experimental devices . visual examination . samples unloading [max.A : 100 kCi (Co60)] * 2 pneumatics	* 1 small cell for radio-elements unloading	
Removable experimental devices	* 6 to 12 in core positions (diam.36mm) * 0 to 4 reentrant positions * 17 peripheral positions (1st raw) * 2 (maximal) pressurized water loops	* CHOUCA * CYRANO * GRIFFON * ISABELLE loop * IRENE loop * THERMOPUMP * GAMMAMETRY * NEUTRON radiography * ...	

IV. MAIN FEATURES OF SIRIUS 2 PROJECT

IV.1. CIVIL WORK/GENERAL LAY-OUT

REACTOR BUILDING

- Cylindrical shape (reinforced concrete)
  * protection from external missiles : (Learjet 23 - CESSNA 210)
  * cheaper than a square building and safer (Borax over pressure proof)

- Controlled leakoff containment
  * all penetrations into the Reactor Building are grouped together in one room in Auxiliary building with specific ventilation.

- Basement (-4.50)
  * it allows areas for :
    - primary water storage tank
    - hot workshop for experimentators
    - liquid and solid waste storage

REACTOR AUXILIARY BUILDING

- a "connecting room" between Reactor Building and Auxiliary Building allows the location of <u>all</u> mechanical and electrical penetrations RB/AB in a single room with its specific ventilation in such a way to lower outward leakages.

- important equipment (control room - protection system - radiation monitoring syst.) are located in the central part of the building, being less vulnerable to external aggressions.

- It houses : I&C, Electrical, compressed air, demineralized water, hot and chilled water production and distribution systems.

OTHER BUILDINGS (options)

- neutron guide building.

- BNCT building with specific access for medical assistance, patients' reception area and therapy monitoring.

Civil work : Main figures

REACTOR BUILDING :	
ϕ internal	: 28 m
H (above the raft)	: ~ 35 m
Concrete volume	: ~ 5.700 m^3
Steel	: 740 t
Wall thickness	: 55 cm

Building and internal structures have been checked for following conditions :

- seismic horizontal acceleration : up to 0.5 g (RCCG)
- Borax type accident : . internal overpressure : 150 mbar after 25 mins
  . water hammer on the roof : 5 t/m^2
- aircraft crash : Learjet or CESSNA

IV.2. REACTOR BUILDING - INTERNAL ARRANGEMENT

Internal arrangement has been designed as follows :

- the East side (close to Auxiliary bdg) is reserved for the water block and includes :
  . auxiliary pool (access to Hot cell and gammametry device) and Fuel storage pool;
  . three casemates for primary cooling circuit (2 for main loops and one for auxiliary circuits);
  . decay tank (-90m^3)
  . primary water storage tank (basement).

  Water block is designed to assure at least one meter of water above the core under any designed accidental conditions.

- The West side is available for experimental areas
  . beam tubes ports at ground level, (with handling facility).
  . shielded cells/casemates for pressurized loops.

- Accesses :
  . a hatch for trucks at ground level (south side)
  . personnal airlock at level +10.00 (close to control room)
  . a specific access for α therapy area at ground level (eventual)

The control room is situated in the auxiliary building, outside the containment. However operators may supervise the working area at pools water level from the control room. A TV network allows operators to watch over important areas which cannot be seen directly from the control room.

Visitors may have a direct view inside the containment (lev.+10.00) through a window above the main control room, without disturbing the operators.

IV.3. MAIN CORE CHARACTERISTICS (reference configuration)

Core thermal power	25 MW
Core Flow rate	Approx. 2 400 m^3/h
Average heat flux	45 W/cm^2
Core inlet temperature	40°C
Core outlet temperature	49°C
Standard fuel element number (23 plates)	22
Control fuel element number (17 plates)	5
Irradiation element number (12 plates)	6
Beryllium reflector element number	10
Fuel cycle duration	21 days
Maximum fast neutron flux (E > 0.1 MeV)	$3.8 \times 10^{14} n/cm^2.s$
Maximum thermal neutron flux (E < 0.625 eV)	$3 \times 10^{14} n/cm^2.s$

FUEL ELEMENTS : MTR - Plate SILOE type . LEU - 19,75%

	Plate number	U5(g)	Burn-up(%)
Standard	23	413	55
Control	17	305	40
Irradiation	12	215	30

CORE :

	U5(g)	U8(g)	Pu (g)	BU(%)
BOC	8.646	46.400	300	23.8
EOC	8.024	46.300	400	29.3

IV.4. POOLS : MAIN CHARACTERISTICS

There are three pools which can be isolated by two cofferdams

	REACTOR POOL	AUXILIARY POOL	FUEL STORAGE POOL
Main functions	- reactor block housing - core cooling inlet and outlet - 2 natural convection valves - fuel storage (one core) - neutron beams sleeves	- access to hot cells (underwater) - experimental devices storage - position for fuel transportation cask	- spent fuel storage racks : ~ 140 fuel elements
Dimensions	- upper part (square): 4.80 x 4.80 - lower part (cylinder): Ø 4.30 m - H : 11.20 m	L = 7.00 m l = 2.40 m H = 6.50 m	L = 6.20 m l = 2.40 m H = 6.50m
Specific point	- annular cavity at core level : Borax - penetrations - S.S. inner liner - C.S. outer liner	S.S. liner	S.S. liner

IV.5. NUCLEAR AUXILIARIES

FUEL STORAGE CAPACITY

- Fresh fuel : the storage capacity is designed for 750 EFPD
- Spent fuel : - réactor pool : 1 core (34 elements)
  - fuel storage pool : for 600 EFPD

HOT CELLS

- 1 non destructive examination hot cell (basic project)
  [5.20 m x 4.90 m width x 5.00 m height] for :

  . removal of irradiated samples from experimental devices;
  . loading of samples (irradiated or not) into experimental devices;
  . repairing or dismantling the auxiliary elements of experimental devices;
  . cutting of worn mechanical of rig parts.

  It is designed to allow, inside biological shielding, up to 100 kCi (37.10^5 GBq) of Co 60 for $2.5 . 10^{-6}$ $Sv.h^{-1}$ at the cell wall contact.

- 1 smaller hot cell (option) can get out baskets in which tight containers for irradiated samples are situated.

  [1.70 m x 1.95 m x 2.6 m height]

NEUTRON BEAMS

- 2 tangential : north and south-west sides

  the north one can be devoted to neutrontherapy (possibility of a separate area for patients)

  . thermal neutron flux at beam port : ~ 1.8 . 10^9 $n.cm^{-2}.s^{-1}$

  . $\phi th/\phi f$: 26

- 2 radial : west side

  . thermal neutron flux at beam port : ~ 3.10^9 $n.cm^{-2}.s^{-1}$

  . $\phi th/\phi f$: 1.7

  They can be aiming at a cold source (option) located either in a Be block or in D_2O tank (option).

CONVENTIONAL PNEUMATICS (basic project)

2 pneumatic tube systems.
Irradiation position : behind Beryllium blocks

$\phi th > 10^{13}$ $n.cm^{-2}.s^{-1}$
with ϕth / ϕf ~20

Maximum speed of shuttle : $5 m.s^{-1}$
(internal diameter : 14 mm - total length : 124 mm)

Samples are loaded and unloaded from a specific room located close to the west side of Reactor pool at level + 6.50.

NUCLEAR VENTILATION

- normal nuclear ventilation includes :

  . an air supply set to compensate air exhausts and to control the ambient temperatures;

  . one "active" exhaust network connected to the hazardous rooms equipped with absolute filters;

  . one circuit for air recycling from the non active rooms.

  Normal nuclear ventilation is stopped under abnormal conditions inside the containment area; it has electrical back-up.

- emergency ventilation (exhaust only)

  This exhaust network, equipped with absolute filters and iodine traps is designed to :

  . perform air filtration on iodine trap in case of excessive contamination inside the containment;

  . maintain a minor negative pressure inside the Reactor building when normal ventilation network is stopped;

  . allow the temporary over pressure to be filtered out through active carbon filters if a Borax type accident occurs.

- hot cell specific exhaust network equipped with absolute filters and iodine trap.

- gazeous wastes exhaust.

NUCLEAR VENTILATION : MAIN FIGURES

* Negative pressure inside RB	Zone IV (hot cell)	:	> 2.2 mbar
	Zone III	:	1.2 to 1.4 mbar
	Zone II	:	0.8 to 1 mbar
* Air supply flow rate		:	38.700 Nm³/h
* Air exhaust flow rate		:	29.300 Nm³/h
* Recycling max.flow rate		:	23.600 Nm³/h
* Emergency ventilation air exhaust		:	800 Nm³/h
* Hot cell air exhaust		:	2.000 Nm³/h

Existing Reactor

- Hifar
- Tank (Dido) type reactor
- 10 MW
- In operation since 1958
- Modernisation Programs
  - Safety Upgrading
  - Facility Improvement

Mission

To provide a multi-purpose national facility for nuclear research and technology in the form of a reactor

- for neutron beam research with a peak thermal flux of the order of 3 times higher than for HIFAR, and cold neutron source facilities
- with isotope production facilities as good as or better than those in HIFAR
- that maximises commercial potential
- that maximizes flexibility to accommodate variations in user requirements throughout its life

Facility Priority

- The Australian Science and Technology Council has recommended that priority be given to the development of a replacement reactor as a major national research facility
- Other priority items are
  - Australia Telescope
  - Marine Geoscience Research Vessel
  - Mining Materials Research Facility
  - Synchrotron Research Facility
  - Tropical Marine Research Network
  - Very High Speed Data Network

ansto engineering

Schedule

ID	Name	1992 '92	1993 '93	1994 '94	1995 '95	1996 '96	1997 '97	1998 '98	1999 '99	2000 '0	2001 '1	2002 '2
1	Preliminary Engineering and Financial Study											
2	Environmental Impact Statement											
3	Public Works Committee Processes											
4	Prepare Tender Documents											
5	Board and Government Approval to Proceed											
6	Tendering											
7	Board and Government Approval to Let Contract											
8	Detail Design											
9	Procurement											
10	Civil Construction											
11	Installation											
12	Commissioning											

THE STATUS OF THE PIK REACTOR

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Abstract

This report describes the 100 MW research reactor PIK which is now under construction. The thermal neutron flux in the heavy water reflector exceeds $10^{15}\,cm^{-2}s^{-1}$; in the light water trap, it is about $4 \cdot 10^{15} cm^{-2}s^{-1}$. The replaceable core vessel allows to vary the parameters of the core over a wide range. The reactor provides sources of hot, cold and ultracold neutrons for 10 horizontal, 6 inclined neutron beams, and 8 neutron guides. At the ends of the beam tubes, the neutron flux is $10^{10} - 10^{11}\,cm^{-2}s^{-1}$. The flux of the long wave neutrons exceeds $10^9\,cm^{-2}s^{-1}$. To ensure precise measurements, the experimental hall is protected against vibrations. The project meets all modern safety requirements. The calculated parameters of the reactor were verified using a full-scale mock-up. Seventy percent of the reactor construction and installation were completed in the beginning of 1992.

Version of the preprint: A.N.Erykalov, O.A.Kolesnichenko, K.A.Konoplev, V.A.Nazarenko, Yu.V.Petrov, S.L.Smolsky: PIK Reactor, PNPI-1784, St. Petersburg 1992

1 The aims of the reactor

45 km south from St. Petersburg and 4 km from the town of Gatchina, the high flux research reactor PIK [1] – [4] is being constructed. Fig.1 shows a drawing of the reactor building, and Fig.2 shows the building site in autumn 1991. The reactor is designed for a broad range of research in nuclear and solid state physics, for studying the basic properties of matter, e.g., of newly developed materials, including high-temperature superconductors, for radiobiological research and also for solving many applied technical problems.

The envisaged high flux of thermal, cold and hot neutrons permits to plan the investigation of basic neutron characteristics such as the electric and magnetic dipole moments, charge, life time and to study the fundamental neutrons interactions, e.g., strong interactions in neutron collisions, and weak interactions after neutron capture.

The reactor allows to develop independent sources of neutrinos and antineutrinos ($\sim 10^{17}\ /s$) with known spectra: this can contribute to the development of neutrino physics. The high flux of thermal neutrons and the low background of fast neutrons and γ-quanta permit to continue the traditional research in nuclear physics, including β and γ spectroscopy and various experiments with polarized neutrons and targets.

Solid state physics will be represented by studies of the kinetics of non-equilibrium systems, by neutronographic research on high-temperature superconductors, ceramics and new materials, and also by structural research on magnetic materials etc. The research in biology will include neutron and structural analysis of biological objects, studies of membranes etc.

A more detailed description of the experimental program is given in ref. [5]. The experimental program will be carried out in co-operation with the leading scientific research centers in Russia and abroad.

The project of the reactor supervised by B.P.Konstantinov, Petersburg Nuclear Physics Institute of the Academy of Sciences of Russia, is being realized by the Scientific Research Institute for Power Reactor Design and by other organizations of the State Committee for Atomic Energy.

2 Reactor design and neutron parameters

The design and parameters of the PIK reactor were chosen so as to provide the maximum number and minimum cost of events in the experimental detectors [6,7]. The actual technological and heat transfer limitations were observed.

The light water core with a volume of about $50\, l$ is placed in a heavy water reflector and serves as an intense source of fast neutrons with a power of $100\, MW$ (see Figs. 3 - 5). The heavy water reflector in which the fast neutrons are slowed down gives the best ratio of thermal neutron flux to power as compared to other moderators [1]. Due to the large diffusion length in $D_2O\,(L_{D_2O} = 1\,m$ at 0.2% of $H_2O)$ and to the considerable dimensions of the heavy water tank (diameter $2.5m$; height from 2.5 to $2.0m$), the thermal flux is rather high at large distances from the core where the background of fast neutrons as well as that of γ quanta is small (Fig.6). A reflector of this type makes it possible to displace and to replace experimental channels either before or after the reactor starting-up. This reflector is safe against radiation damages. Accumulated tritium and hydrogen are removed by a special isotopic purification circuit, and therefore the tritium activity in the D_2O does not exceed $0.1\,TBq/l$ [8]. The reflector has its own $5\,MW$ heavy water cooling circuit which allows to maintain its temperature within the range from 50 to $60^{\circ}C$.

Light water is used as a cheap coolant in the core of the reactor. Light water as in-core moderator provides a small neutron migration length which permits to design a compact core.

The core with high pressure (up to $5\ MPa$) and high energy release (about $2MW/l$ on the average) is separated by a double core vessel from the reflector where the thermal neutron flux is formed and where the pressure is low ($0.3\,MPa$). Precautions are taken in case of damage of experimental channels. Thin membranes through which the neutrons can easily pass are installed at the output of the heavy water tank to prevent the penetration of radioactivity into the experimental hall. A water pool of $12\ m$ depth protects of the staff from possible damages of the circuits. To prevent the contamination of the hall, ventilation is provided above the water surface. The vertical cylindrical core vessel serves as internal wall of the reflector tank and is connected to the water supply tubes that it can be replaced without affecting the tank itself. Owing to these measures, the PIK will be a versatile unit permitting to change the arrangement and dimensions of the core even after the reactor is put in operation. Every two years when the core vessel is replaced because of the radiation damages, it is possible (if necessary) to change the type of the fuel elements or even to install special experimental facilities inside the core. In the beginning of the PIK exploitation, the core vessel will be made of austenitic steel which has sufficient viscosity to prevent the vessel from cracking. Later on, this vessel will be replaced by an aluminum or zirconium vessel.

In order to ensure the high flux of neutrons in the reflector and the light water trap, the fuel elements of the reactor should provide a high neutron multiplication factor k_∞. The fuel elements should also ensure a high specific power to obtain a high absolute neutron flux. These requirements are met owing to the high density of fuel (90% enriched ^{235}U with average density 540 g/l) and owing to the increase of the specific heat transfer surface (6.5 cm^2/cm^3). The fuel elements of the PIK reactor are twisted rods with crosslike section and an external diameter of 5.15 mm (see Fig. 5a). The twisting around the axis with a thread-spacing of 300 mm ensures a fixed distance between the fuel elements inside the bundle. The steel cladding of the fuel element is 0.15 mm, the fuel loading is 7.14 $g^{235}U$, and the meat density is $\sim 2.2\,g^{235}U/cm^3$. The fuel elements are placed into a triangular lattice with a spacing of 5.23 mm inside the fuel assembly. The core is formed by 12 hexagonal and 6 square fuel cassettes (Fig.5). The hexagonal fuel cassette contains 241 fuel elements, the square cassette contains 161 elements. The fuel elements were tested at 7.5 MW/l core power density (see also [9]). The neutron parameters of the reactor with the above fuel elements are shown in Fig.6 and are tabulated in Table 1. For longer terms it is planned to use aluminum fuel elements providing a higher neutron flux [10].

The control and safety system consists of a central control unit and 8 rods in the reflector. The central unit (so-called "shutter") is made of two europium oxide rings embracing the central light water trap and absorbing neutrons. In order to avoid asymmetrical distortion of the neutron flux, the rings are simultaneously moved apart from the central reactor plane. This system will compensate the burn-up and provide automatic control and emergency protection. Both halves of the central control unit normally have a clearance permitting quick emergency input of not less than $0.5\,\beta_{eff}$ 0.35 s. The absorber rods are realized as rectangular cassettes located in the heavy water reflector and containing europium oxide. Some of the rods serve as safety rods, the others are used as starting absorbents.

The lateral shielding of the reactor is divided into biological and "experimental" shielding. The biological shielding consists of iron, water ($0.5\,m$), and heavy concrete ($0.9\,m$, $\gamma = 3.6\,g/cm^3$). It diminishes the radiation to a level permitting attendance of the equipment when the reactor is stopped. The experimental shielding ($1.0\,m$ thick) is part of the physical instruments and consists of movable units. The shielding reduces the reactor radiation to 14 $\mu Sv/h$, i.e., half of the limit permitted by the existing standards.

The reflector tank is located 9 m deep in the pool. The pool communicates with the operating hall situated on the top floor of the reactor building. The experimental hall, into which the neutron beams are guided, is separated from the operating hall (Fig.7).

3 The safety

The PIK reactor meets the strictest requirements for modern atomic power stations. After the Chernobyl catastrophe, the project was fully revised [11] to satisfy the new safety rules valid in Russia.

The core of the PIK reactor belongs to the undermoderated systems, hence, the reactivity has a negative temperature coefficient due to a decrease of the water density. The analysis shows that the fuel elements would not melt even under the hypothetical condition that about 1% positive reactivity is put in extremely quickly (during $0.1\,s$) [11,12].

The PIK reactor has a 3-circuit cooling system. Due to the intermediate circuit (where the pressure is lower than that of the first and of the second circuit), the contamination of the third circuit or the environment is fully prevented in the case when the heat exchangers are damaged. If the cooling system fails the reactor stops automatically. During the first minutes, the cooling is provided by means of hydroaccumulating units, after that it is provided by a pump ($300\,m^3/h$) until the core is fully cooled; then, it can be unloaded. The emergency coolant mixture is duplicated and provides the reactor cooling even in the case of a coolant-loss accident when a circuit is broken.

The emergency power supply is provided by three independent, parallel 600 kW channels. Each channel incorporates diesel and accumulators connected to one of the major circulating pumps. The work of one of the channels is sufficient to cool the reactor.

The analysis shows that even in the case when the major part of the core is melt in an extremely short time (this situation is practically impossible), this would cause an energy release thousands of times less than that of the Chernobyl catastrophe [11] because the mass of the fuel elements is considerably smaller ($0.16\,t.$ as compared to $200\,t.$), and also because of the lower melting point of the PIK elements. This energy release is not sufficient to destroy the reactor building; hence, all the radioactivity would stay inside the building. The radioactivity accumulated in the PIK reactor will be a small fraction of that of the RBMK-1000 reactor: the radioactivity of short-living isotopes (iodine and others) is 32 times less than in the RBMK-1000, and the radioactivity of long-living isotopes (cesium and others) is even hundreds of times less. In order to prevent the contamination outside the building, an emergency sealing system is provided. This system permits to localize even the most improbable effluent radioactivity release and later, it allows to clean the building. In addition, a strengthened safety vessel was constructed around the reactor. This vessel is capable to withstand a shock wave up to $4\,kPa$ (Fig.7).

The described measures permit to exclude the possibility of environmental contamination.

4 Experimental facilities of the reactor

The PIK reactor is supplied with modern equipment for carrying out physical experiments. There are facilities like a light water neutron trap, cold and hot neutron sources, neutron guides, horizontal and inclined beam tubes for radiation etc.; all facilities are described in [5].

The PIK reactor and the neutron laboratory form of a single complex system. Some facilities are used by both of them, e.g., by the hot laboratory.

4.1 Neutron trap

The cylindrical light water trap with optimum diameter of about $10\,cm$ is arranged in the center of the core. In the central experimental channel, the unperturbed flux of thermal neutrons amounts to approximately $4 \cdot 10^{15}\,cm^{-2}s^{-1}$, and the flux of fast neutrons with energies $E \geq 1.2\,MeV$ is

$5 \cdot 10^{14}\ cm^{-2}s^{-1}$. Only two reactors (specially designed trap reactors) have a neutron flux similar to that of PIK: the Russian material testing reactor SM-2 [9], and the U.S. reactor for the production of transuranium elements HFIR [13]. The PIK combines the advantages of "trap" and "beam" reactors. Although the installation of a "trap" leads to some loss of reactivity, this is an advantage, because the neutron flux density in the center is three times higher than at any point of the reflector. The PIK is particularly efficient for beam measurements of the neutron characteristics of short-living isotopes obtained in the trap. In this case, the number of events in the experiment is proportional to the product of neutron fluxes in both systems, i.e. to the square of the flux of the reactor. The central channel in the trap with an internal diameter of $6\,cm$ is cooled by an independent water circuit with a capacity of $400\,kW$ and a pressure of 0.1 to $5\,MPa$, depending on the heat release in the irradiated samples. It is convenient to use this channel for irradiating targets for neutrino sources with high specific activity. The channels for the irradiation of standard samples (neutron activation analysis etc.) are located at the bounds of the core and of the trap. Some samples with vessel materials are located in the fuel cassettes and exposed to a fast neutron flux ($E > 1.2\ MeV$) two times higher than in the vessel structure itself; thereby, the changement of the vessel durability with the fluence can be predicted [14].

4.2 Cold neutron source

Experiments in solid state physics and biology require intense beams of cold neutrons with a wavelength $\lambda \geq 4\ A\,(E \leq 0.05\,eV)$. The intensity of the long-wave neutrons can be raised by orders of magnitude if the temperature of the neutron spectrum is lowered sufficiently. The reactor equipment includes two cold neutron sources. One of them is a $19\ cm$ radius sphere containing about $25\ l$ of liquid deuterium at a pressure of $0.15\ MPa$. A $11\ kW$ cryogenic helium circuit with communication through the vertical channel decreases the temperature of deuterium to about $25\,K$. The center of the source is located $78cm$ distant from the core center. The unperturbed flux of thermal neutrons at this point is on the average $3.5 \cdot 10^{14}\ cm^{-2}s^{-1}$ (Figs.4,6). The cold source is connected by the horizontal beam tube HEK 3 with the neutron guide system. An inclined channel starting from the cold source transports the beam of cold polarized neutrons in the experimental inclined channels hall.

Another source of cold and ultracold neutrons with a liquid hydrogen moderator is proposed to be installed in the center of the tangential through tube $HEK4 - 4'$. The unperturbed flux of thermal neutrons is about $1 \cdot 10^{15}\,cm^{-2}s^{-1}$ at this point. The power release into the moderator is $5.3\,kW$. The cold neutron beam goes to one side of $HEK4 - 4'$, while the ejected cold and ultracold neutrons go to the other side of the $HEK4 - 4'$ (there is large experience of using such sources in the WWR-M reactor). The expected source strength of ultracold neutrons is $5 \cdot 10^{6}\; n/s$ [15].

4.3 Hot neutron source

For research in the fields of biological, crystalline and magnetic structures, and also for the study of high energy excitations in solids and liquids, short wavelength ($\lambda > 1\; A$) neutrons are needed. Therefore, the PIK reactor is to be equipped with a hot neutron source. It is a cylindrical graphite unit with 20 cm diameter and 30 cm height, and separated by a double zirconium shell from the reflector. It is isolated from the internal shell by a graphite felt filled up with helium. The graphite is heated by the gamma radiation of the core to approximately $2000^{o}C$. The center of the hot source is located 65 cm from the center of the core and 40 cm above the central plane. An unperturbed flux of thermal neutrons at this point is on average about $3 \cdot 10^{14}\; cm^{-2}s^{-1}$. Technical links from the hot source pass through the vertical channel into the technology hall. The hot neutrons go through a horizontal beam tube HEK 8 into the experimental hall.

4.4 Beam tubes

In several horizontal planes, there are three through tubes, one radial beam tube, one V-shaped and three tangential beam tubes (Fig.4). In addition, one horizontal beam tube goes to the cold source and the other to the hot source. Since the pressure in the reflector tank is low, the wall thickness of the reflector beam tubes is rather small (less than $6\,mm\,Al$). The horizontal beam tubes can be of inner diameter up to 25 cm, since all of them, apart from the V-shaped channel, are replaceable. The only fixed parameters are the position and diameter of the holes in the tank and shielding. For example, each through channel can be replaced by two tangential thimbles.

Additionally, six inclined beam tubes of inner diameters varying from 8 to 14 cm start from the reflector. These tubes direct the neutron beams to the inclined channels rooms situated above the main experimental hall. Besides, it is possible to install 5 vertical channels 4.1 cm diameter and one channel of 15.5 diameter in the reflector tank to irradiate sample containers. Since a large number of the experimental channels could reduce significantly the neutron flux in the reflector, only those channels are installed which are to be operated in the near future.

Because of the intense neutron flux, the induced Ar activity in the air, if it filled a beam tube, could be up to 1.5 TBq. To avoid such an activity (and its release into the atmosphere), the horizontal and inclined channels are either hermetically sealed or evacuated, or CO_2 or helium is pumped continuously.

4.5 Neutron guides

To reduce the fast neutrons and gamma background, it is reasonable to lead the slow neutrons away (out of the direct visibility of the radiating beam tube) without significant losses. For this purpose, the PIK reactor is equipped with a system of mirrored neutron guides [16]. The isotope ^{58}Ni covers the internal surfaces of the neutron guides. It has a cut-off wavelength λ_c of about $500\AA$ ($V = 8.2\, m/c$). Neutron guides with a radius of curvature ρ lock the neutrons within a small angle $\vartheta^* = (2a/\rho)^{1/2}$ (where "a" is the width of the neutron guide with rectangular cross section) and transport them along an arc of length L.

Neutron guides are installed in two beam tubes: the tangential tube $HEK2$ (2.4 m long) and the cold source beam tube $HEK3$ (1.7 m long). Both beam tubes are rectangular in shape, narrowing in the core direction of the beam tube. All the auxiliary equipment of the neutron guide system, mirrored collimators and biological shielding, is designed for a neutron guide system with 10 neutron guides (5 thermal neutron guides of length L = 42 - 66 m and 5 cold neutron guides of length L = 27 - 46 m). These characteristics are described in detail in [5]. Since the geometrical characteristics of the neutron guides of the PIK reactor are similar to those of the ILL neutron guide system at Grenoble [17], both systems should produce nearly the same output fluxes. At the neutron guide outputs at considerable distances from the source, these fluxes for cold and thermal neutrons exceed

$10^9\ cm^{-2}s^{-1}$ at a particularly low background.

4.6 Protection against vibrations

The excellent parameters of the PIK reactor require the most up-to-date equipment of the neutron laboratory. The intense neutron beams provide a high rate of accumulation of statistical events required in precise measurements. Therefore, systematic errors resulting, e.g., from the neutron noise [18] or from the vibration of the building, appear to be essential. In designing the PIK reactor, particular attention was paid to reducing the vibration amplitude in the experimental hall. The main source of vibrations - the pumps of the primary cooling circuit - has been installed in a separate building (block $100B$, Fig.2). The technical equipment of the reactor, installed near the physics hall, is mounted on antivibration supports. The whole circular experimental hall rests on a special vibration-absorbing "pillow". As a result, the vibration amplitude for the various parts of the spectrum is reduced by one or two orders of magnitude.

5 Reactor mock-up: calculations and experiments

In order to define more precisely the reactor parameters, a number of theoretical and experimental studies has been carried out. The purpose was the development of an adequate calculation model, the measurement of neutron and physical parameters and the achievement of the necessary experience in the operation of the new technical systems.

To accomplish all these purposes, the mock-up of the PIK reactor was constructed [19] and put in operation in 1983 in the Petersburg Nuclear Physics Institute (Fig. 8). The core, the vessel, the reflector tank, experimental channels and mechanisms of the control systems are similar to those of the real PIK reactor. The power of the model is 100 W. The small induced activity permits to measure the core parameters quickly and effectively. The experiments on the mock-up provided significant experience in handling the heavy water. The technical systems ensure a permanent level of 99 the heavy water. The model permits to put in dosed amounts of the absorbents: boric acid into the core moderator and gadolinium nitrate into the clearance between the case and the vessel. The critical assembly of the mock-up is placed into a concrete block with $1.5\,m$ thick walls which ensure the safety of the staff.

To calculate the neutron fluxes, energy release and multiplication factor k_{eff}, a 4-group diffusion reactor model was used. Previous research showed that the slowing-down process in water-metal mixtures (metal fraction $w = 0.4$) can be described in good approximation by two diffusion groups [20]. As the thermal neutron spectrum in the core and the reflector (Fig.9) are completely different, the two "overlapping" groups of neutrons were put into the thermal region [21]. The effective age (which determines the slowing-down cross section) was verified experimentally. To test the accuracy of the theoretical model, special measurements were performed at the critical facility PIK-04. The experiments were carried out using a critical ring assembly of PIK fuel elements without cassettes and a light water reflector. The accuracy of the calculation of the critical mass turned out to be 0.6% (or $\Delta K/K = \pm 0.13\%$) [22].

The mean deviation of the calculated k_{eff} from the experimental values for 5 critical masses with different dimensions of the inner flux trap was $0.16 \pm 0.13\%$ (Fig.10). The experimental results confirmed that the calculations of thermal neutron distribution were correct as well as those of the hot spot (Fig.10). The calculated reactivity of the central control shutter and the other reactivity effects are in agreement with the experiments [23]. The values obtained experimentally can be used as benchmarks when testing other calculation methods, e.g., the multi-group Monte-Carlo method. In order to determine the accuracy of the 4-group model when applied to the real core of the reactor (including the replacement of the light water reflector by heavy water, a special series of experiments was performed on the mock-up. In these experiments, various concentrations of boric acid (up to $24\,g/l$) were added to the water, and the excess reactivity was compensated by the central control unit. The deviation of the calculated reactivity for 27 experiments differing in the boron concentrations did not exceed 0.3%.

With the help of the mock-up, the excess reactivity and irradiation characteristics were measured with various charges of the core and with various arrangements of the channels inside the reflector. Flux density and spectra of thermal and fast neutrons, fission rate, number of displacements per atom and energy release in materials were measured and calculated for the core and the beam tubes [24]. A special technique was developed to measure the power of the critical facilities; this technique has an error of about 3%. According to this technique the power is measured by the activity of the fission products in the fuel elements [25]. Tabulated in Table 2 are the unperturbed fluxes and the real fluxes taking into account the large-diameter beam tubes [24]. It is evident that such channels reduce significantly the neutron flux. Fig.11 shows the calculated distribution of the thermal neutron flux in the horizontal middle plane of the reactor. As mentioned above, the neutron flux can be doubled if the core is replaced by aluminum fuel elements and an aluminum vessel; therewith the thermal reactor power will not change [10].

The PIK reactor mock-up is permanently in operation. It serves to check the various propositions and design ideas to improve the structure of the reactor and to enhance its neutron and technological performance. Thus, compensation methods for the excess reactivity by homogeneous (boric acid) and heterogeneous (gadolinium, cadmium) absorbents are studied in order to increase the operating period of the reactor. The reactivity effects caused by replacing steel structure elements of the reactor by zirconium or aluminum structures were determined.

The results of the mock-up measurements will be taken into account for the planning of experiments with the reactor itself and for the positioning components affecting the neutron fluxes and reactivity.

6 Progress of construction

The construction of the PIK reactor began in 1976. After the Chernobyl catastrophe, the reactor project was fully revised and redesigned in 1990 to meet the new stricter safety requirements. By now, 70% of the construction are completed. The main part of the reactor equipment is arranged, but some of the items are not being manufactured; their delivery deadlines are in 1992 and 1993. The following structures are fully completed and in operation now: sanitary zone ($100P$), intermediate-circuit pumping station (100Γ), electricity station ($100D$), compressor station, storehouse, and chemical water cleaning station (see Fig.1). The construction works on the reactor building including laboratories ($100A$), primary circuit pumping station with operating rooms ($100B$) , ventilation center (101), laboratory building (105) and entrance hall are completed. Conclusive works on these buildings are to be made as well as the reinforcement of the spans of the buildings $100A$ and $100B$ to arrange the protection cover. To complete the project, the reactor has to be supplied with softened water, isotopic cleaning facility, emergency diesel station, liquid nitrogen stations, waste tanks for radioactive liquids, and the electric power substation has to be completed. Fig.12 illustrates the equipment which is already installed in the water circulation station (100Γ). The date of completion of the reactor construction works depends on the financing.

The neutron beam characteristics of the PIK are close to the most powerful research reactor, the HFR in Grenoble [17] the operation experience of which was used in the design of some experimental equipments. However, due to the trap with a neutron flux three times greater than in the reflector, and the possibility of changing core and experimental channels, the PIK reactor will offer a wider range of experimental possibilities. The high neutron flux, the versatility of the reactor together with the special measures ensuring precise measurement foster the hope that unique experimental information can be obtained during a considerable period.

Acknowledgement
The author wishes to express his sincere thanks to all who have contributed to the design of the PIK reactor with their creative power. Furthermore, he is greatly indebted to Chr.Desandre (Technicatome) and the CEA for the generous invitation to Saclay. The revision of the english manuscript by F.M.Wagner (Reaktorstation Garching) is gratefully acknowledged.

Table 1. PIK reactor parameters.

Power	100 MW
Flux of thermal neutrons in the trap	$4\ 10^{15}$ n/cm^{2}s
Flux of thermal neutrons in the reflector	$1{,}3\ 10^{15}$ n/cm^{2}s
Moderator and coolant	H_2O
Reflector	D_2O
Diameter	2,4 m
Height	2,5÷2,0 m
Core	
Inner diameter of the vessel	0,39 m
Height	0,5 m
Volume occupied by fuel assembly	51 liters
Fraction of water in the fuel assembly	0,59 liters
Load ^{235}U (90% enrichment)	27,5 kg
Type of the fuel elements	cross-shaped twisted pins
Specific heat-transfer surface of the fuel elements	6,5 m^2/cm^3
Spacing of triangle lattice	5,23 mm
Primary circuit	
Core input pressure	5 MPA
Core pressure drop	1 MPa
Water consumption	to 3000 m^3/h
Shielding	
D_2O	1 m
Heterogeneous iron-water shielding	0,55 m
Heavy concrete (γ=3,6 g/cm^3)	0,9 m
Experimental shielding	1 m

Table 2. Thermal neutron flux density and energy output in experimental channels at 100 MW reactor capacity. Experimental error is 15%.

Designation (see fig.4)	Channels	Coordinates of the channel bottom		Channel diameter	Calculated unperturbed flux	Measured perturbed flux	Φ_f for	
		R	Z		$\Phi_{th,c}$	Φ_{th}	E>1,2 MeV	q
		cm	cm	cm	$10^{14} n/cm^2 s$	$10^{14} n/cm^2 s$	$10^{14} n/cm^2 s$	W/g
CEK	central	0	0		41	45	5	45 ÷ 53
HEK 1	radial	30	21	9	7,6	5,7	0,21	3,5÷4,7
HEK 2	tangential	52	1,5	25,2	7,7	4,6	$2,2\ 10^{-2}$	
HEK 3	tangential	97	1,5	35,0		0,4		
HEK 4	going-through	48	1,5	8,2	9,1	9	$2,6\ 10^{-2}$	2,6÷3,2
HEK 5	going-through	53	-59		2	1,8		
HEK 6	going-through	53	-38	8,3	4	3,6	$2,6\ 10^{-3}$	
HEK 7	V-shaped	40	-80	19,4	1	0,73		
HEK 8	tangential	63	44,5		2,6	2,2	$9\ 10^{-3}$	1÷0,68
HEK 9	tangential	39	40	8,2	4,4	3,6		
HEK 10	tangential	49	7	25	8	4,8		
IHEK 1÷6	inclined	37÷79	-10÷27	4÷9	~ 6	~5	$\sim 5\ 10^{-3}$	
VEK 1÷6	vertical	74÷103		6,1		0,6÷1,4	$\sim 10^{-2}$	

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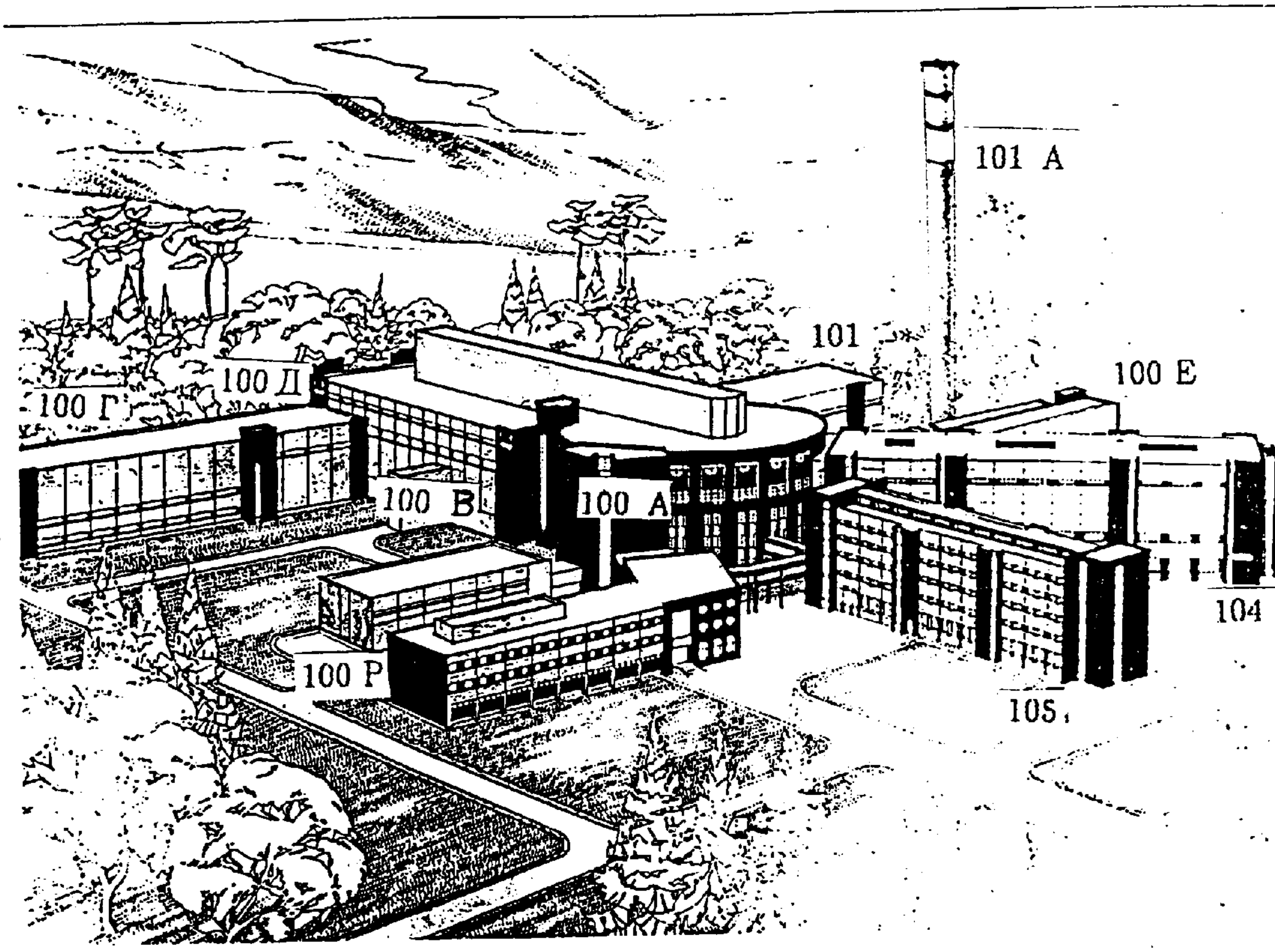


Fig.1: The main buildings of the PIK complex.
100 A - reactor; 100 B - hot cells, pumping station of the primary circuit and other equipment; 100 P,P' - administrative and sanitary buildings; 100 Γ - pumping station of the intermediate circuit; 100 D - electro substation (6 kV) for the blocks *B* and Γ; 100 E - hot neutron facility and isotopic D_2O purification unit; 101 - ventilation center; 101 A - ventilation chimney; 104 - neutron guide hall; 105 - physics laboratories.

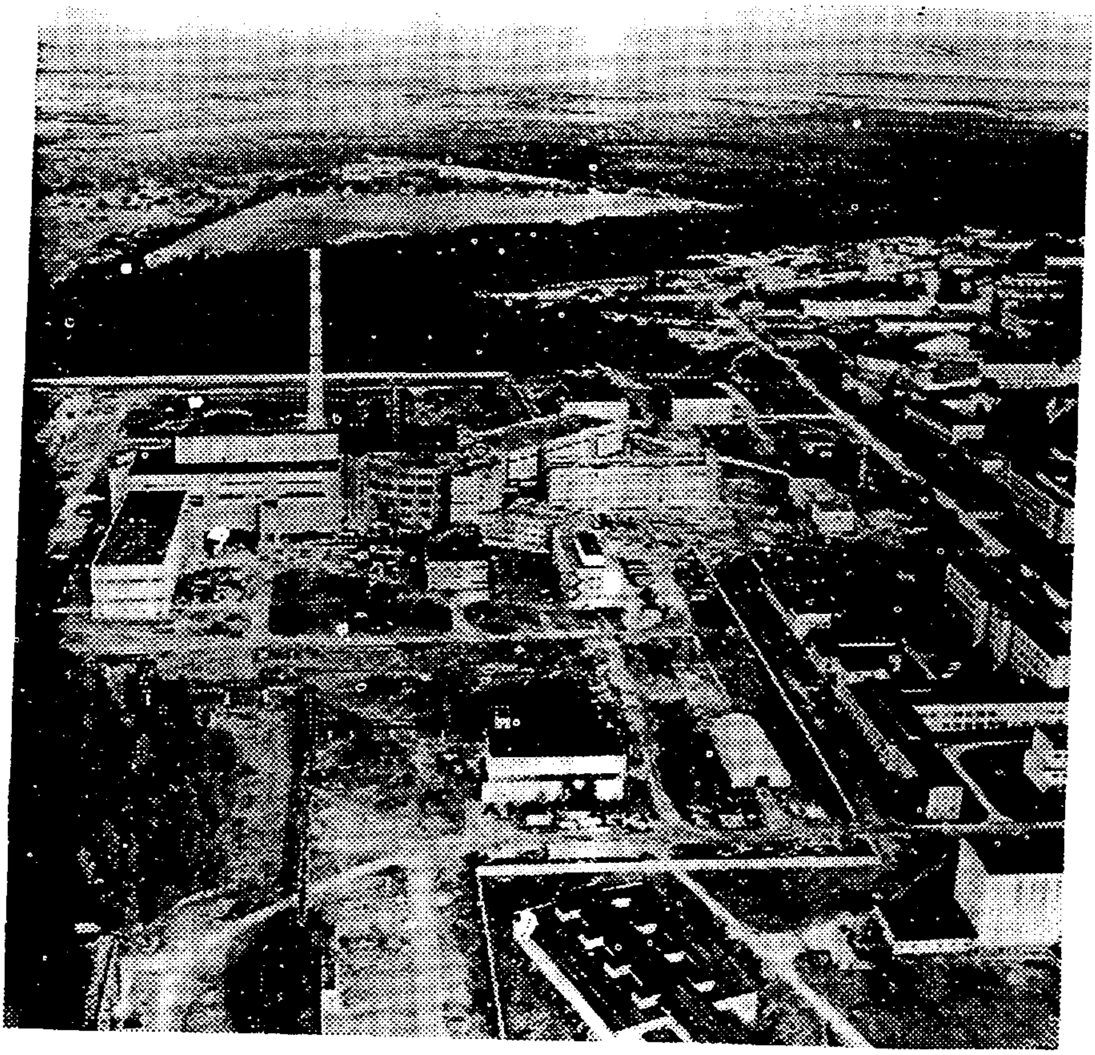


Fig.2: View of the PIK reactor complex (1991).

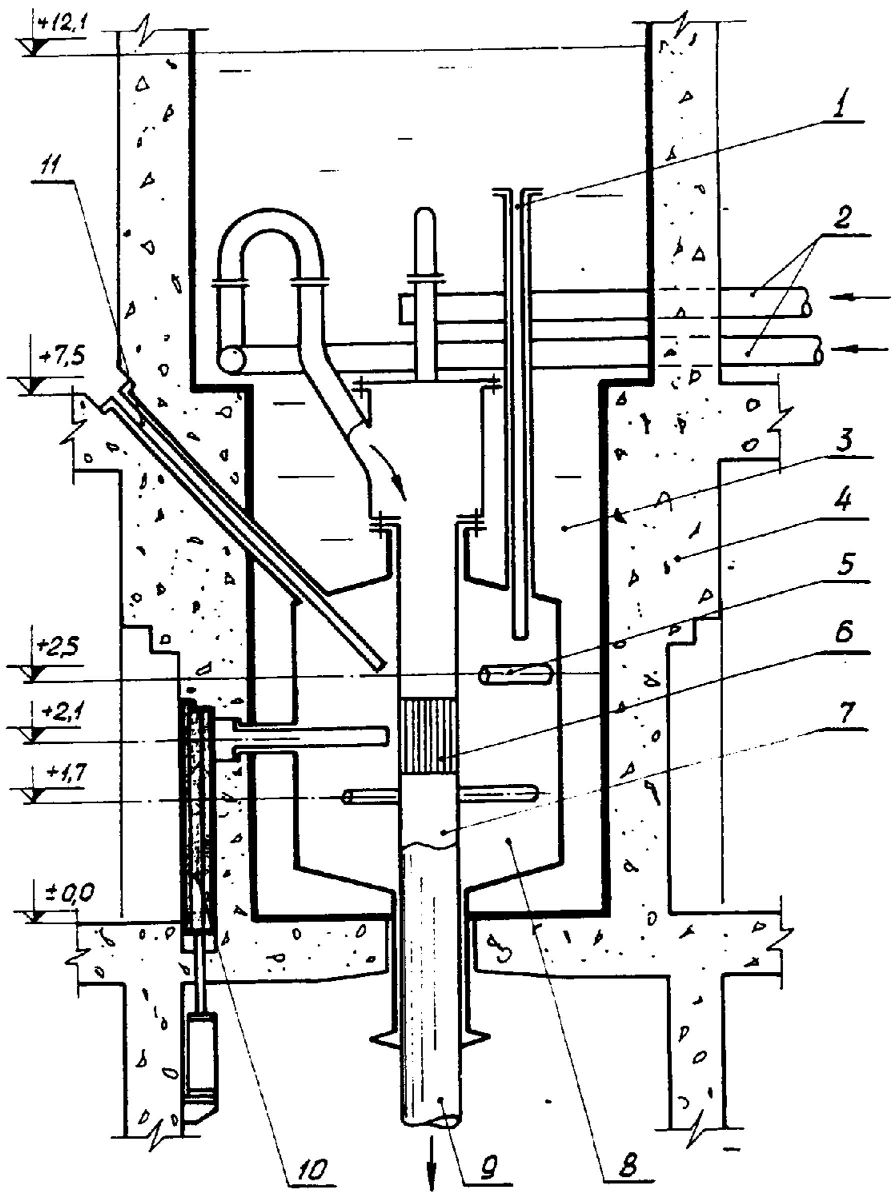


Fig.3: Schematic vertical section of the PIK reactor.
1 - vertical channel; 2 - coolant inlet; 3 - water pool; 4 - biological shielding; 5 - horizontal experimental beam tube; 6 - core; 7 - replaceable vessel; 8 - heavy water reflector; 9 - coolant outlet; 10 - plug; 11 - inclined experimental beam tube.
On the left are the distances from the floor of the experimental hall in meters.

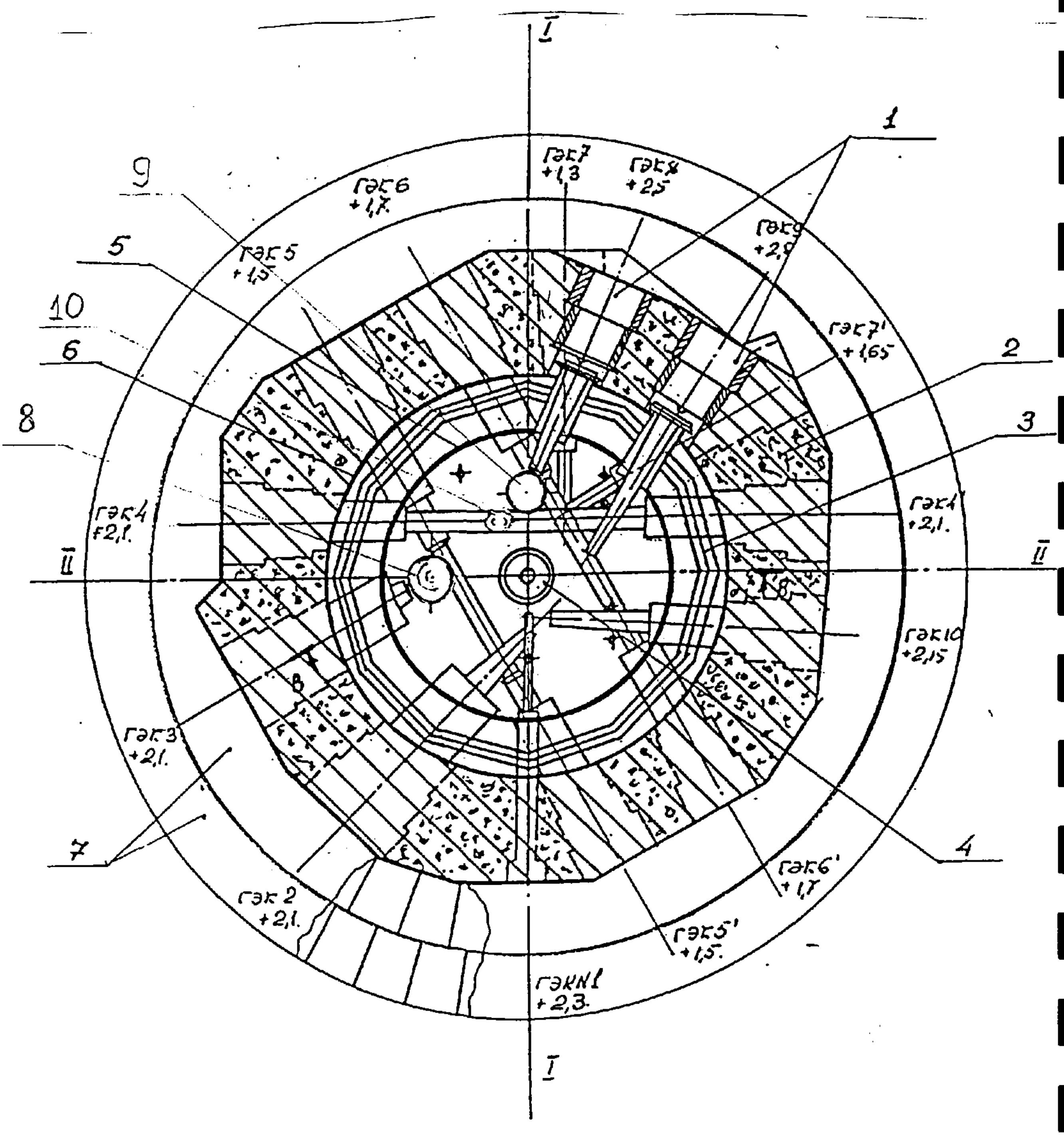
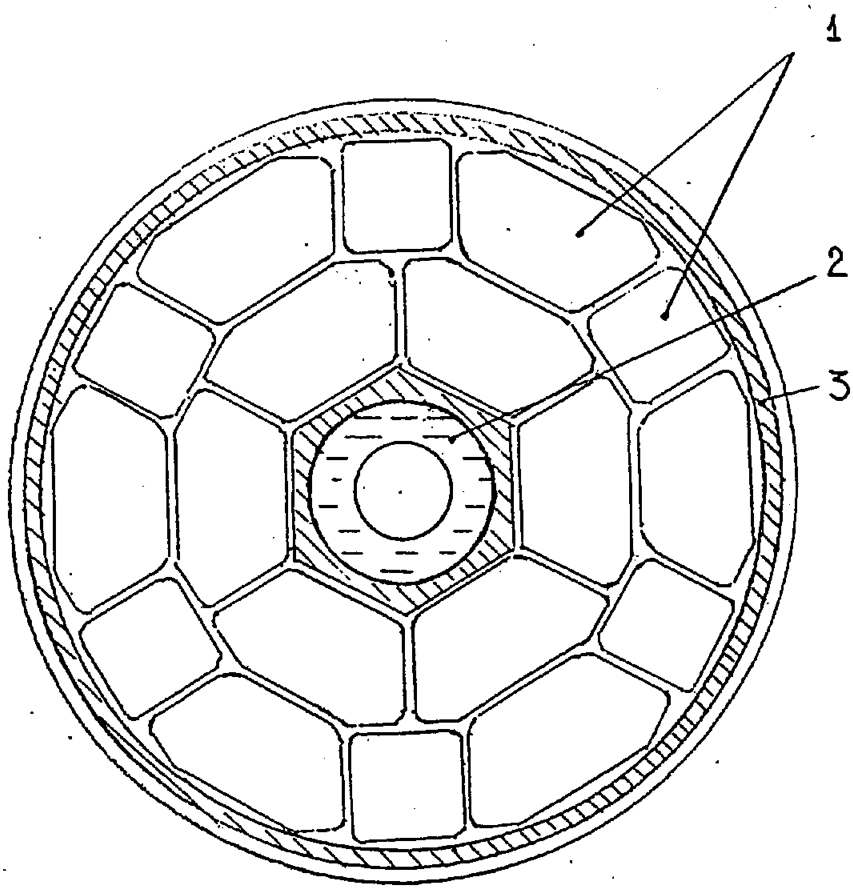


Fig.4: Layout of the experimental beams in the reflector of the PIK reactor.
1 - HEK plugs; 2 - biological shielding; 3 - iron-water- shielding; 4 - core; 5 - heavy water reflector tank (D_2O); 6 - lineer of the reactor pool; 7 - experimental shielding; 8 - cold neutron source; 9 - hot neutron source; 10 - cold and ultracold neutron sources



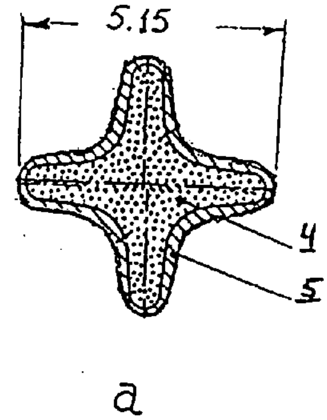


Fig.5a: Section of a fuel element
4 - fuel meat; 5 - cladding.

Fig.5: Section of the core of the PIK reactor.
1 - fuel assembly; 2 - water trap with central vertical experimental channel;
3 - vessel.

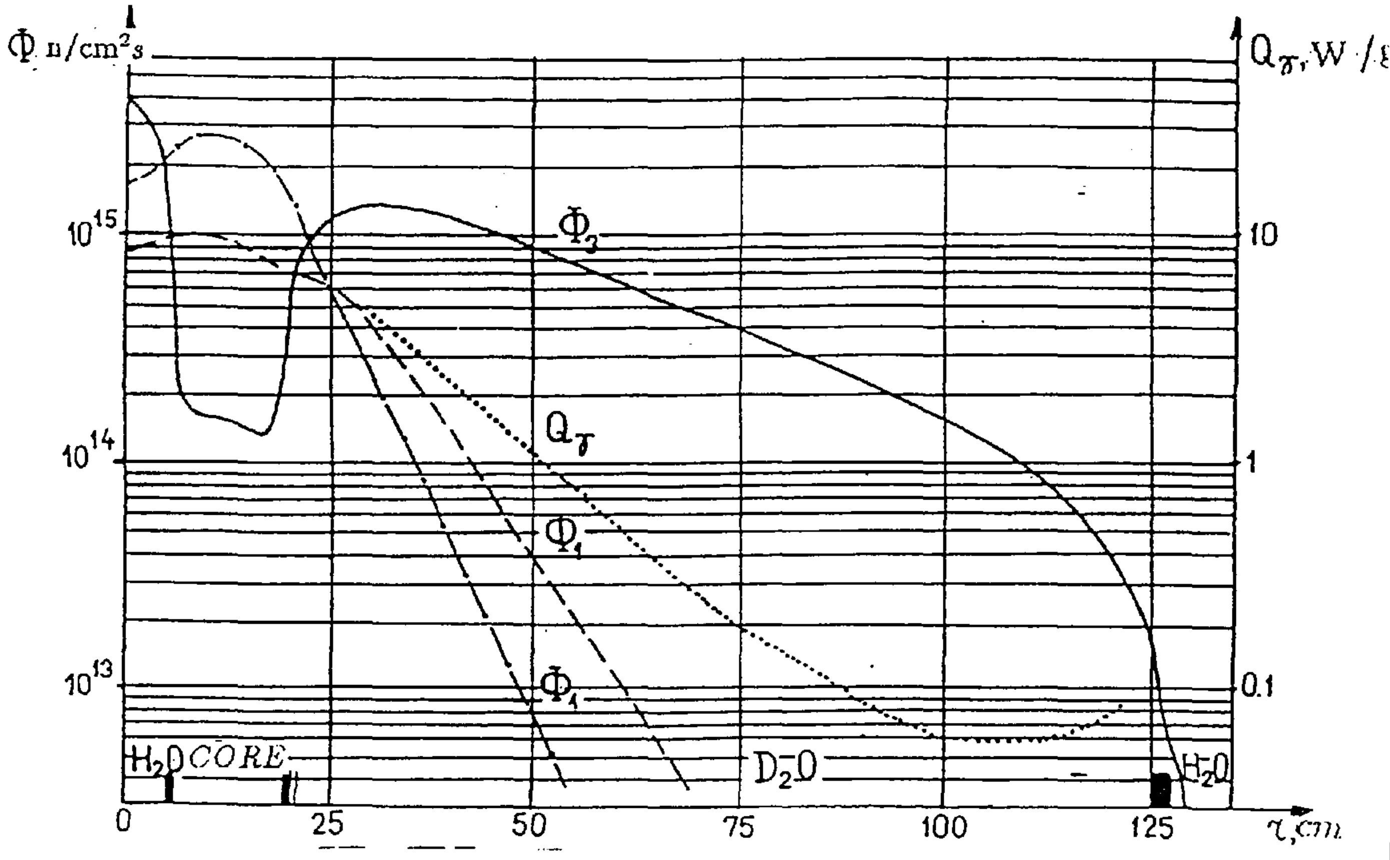


Fig.6: Distribution of unperturbed neutron flux Φ and energy output Q_γ for 100 MW reactor power.
Φ_1 - fast neutron flux $E > 5\ keV$; Φ_2 - epithermal neutron flux $5\ keV > E > 0.6\ eV$; Φ_3 - thermal neutron flux $E < 0.6\ eV$

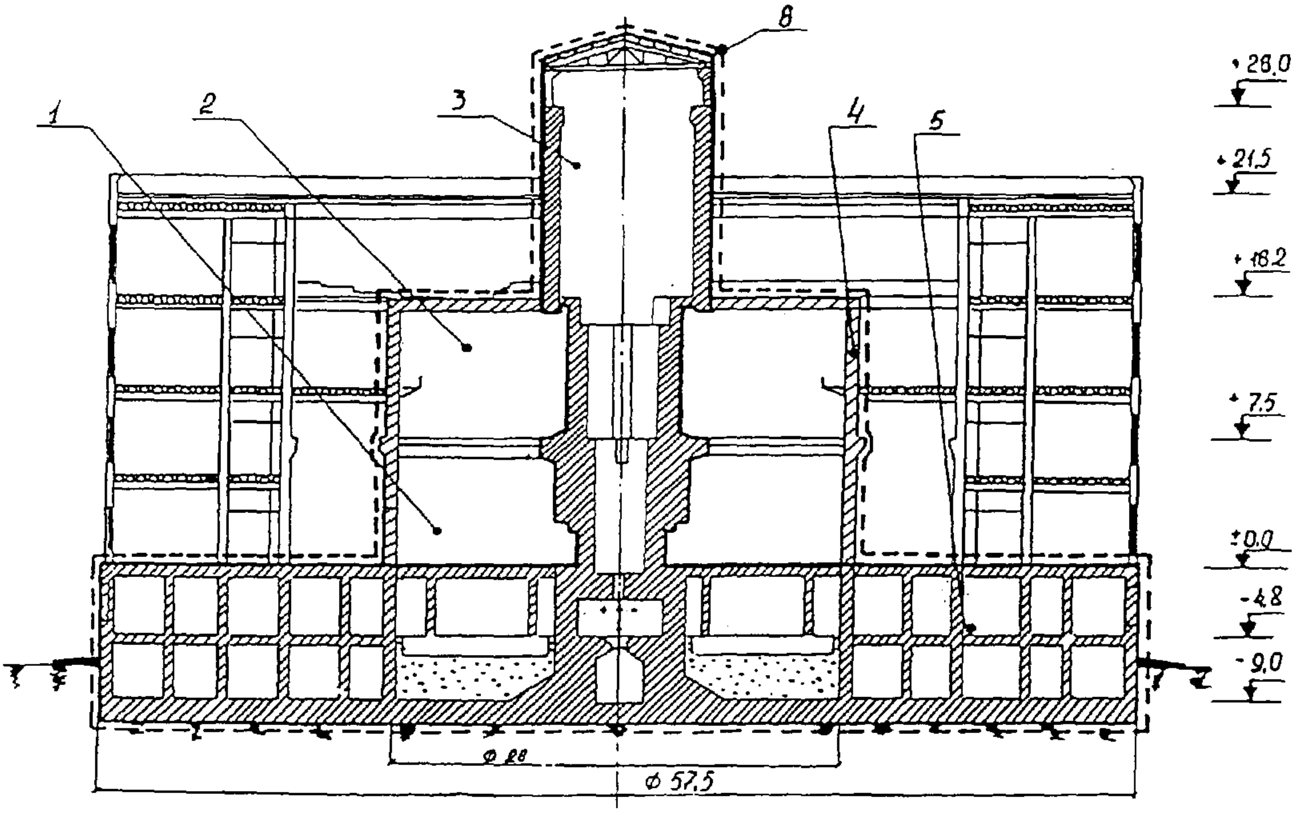


Fig.7: Vertical section of the PIK reactor.
1 - horizontal beam experimental hall; 2 - inclined beam experimental hall; 3 - technical hall; 4 - reinforced concrete cylinder; 5 - containment. Dimensions in meters from the floor of the experimental hall.

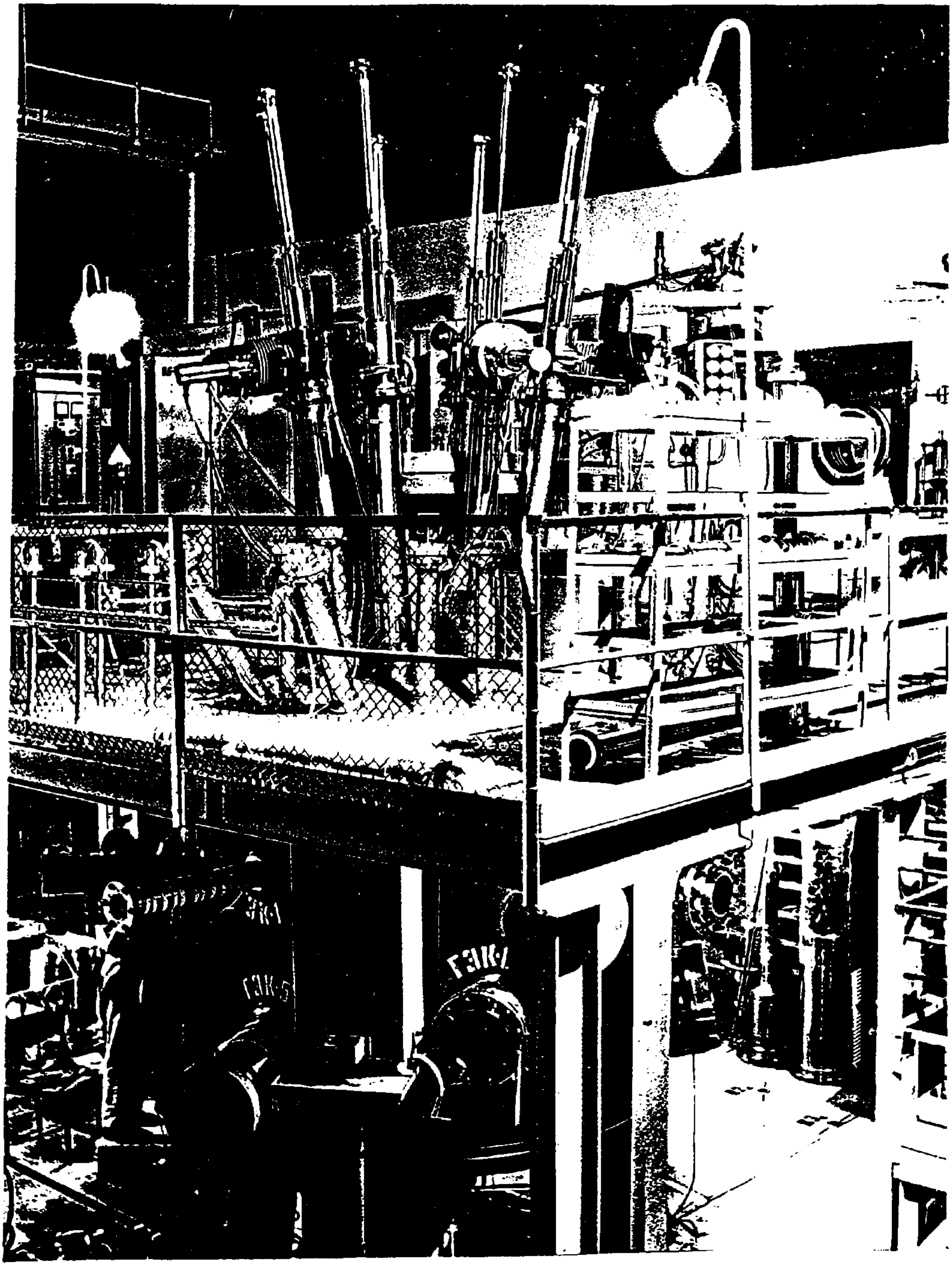


Fig.8: Mock-up of the PIK reactor

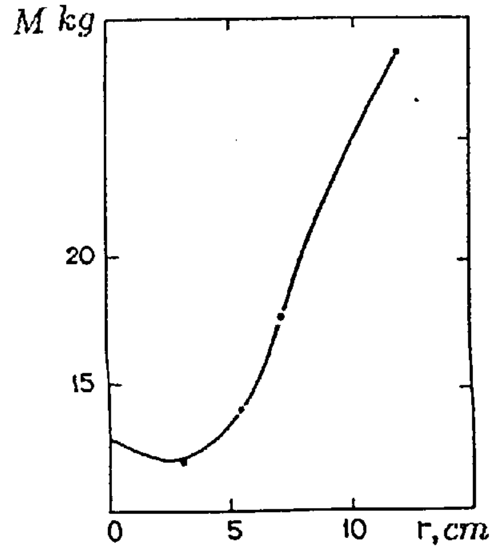


Fig.9: Critical mass M as a function of radius r of the inner light water trap. Full line: calculated values; dotted: experiment (error smaller than point size).

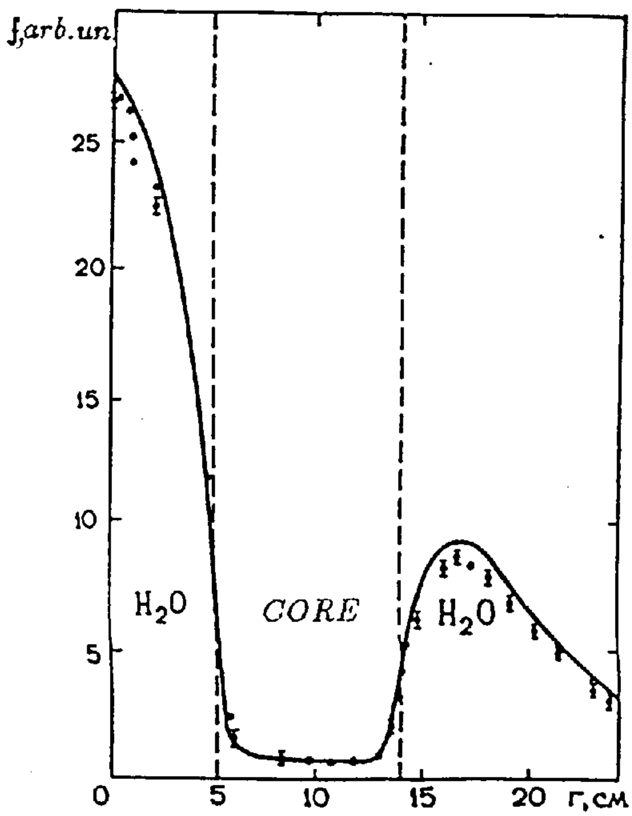


Fig.10: Fission rate as a function of the radius of the critical assembly. Full line: calculated values. Dotted: experiment

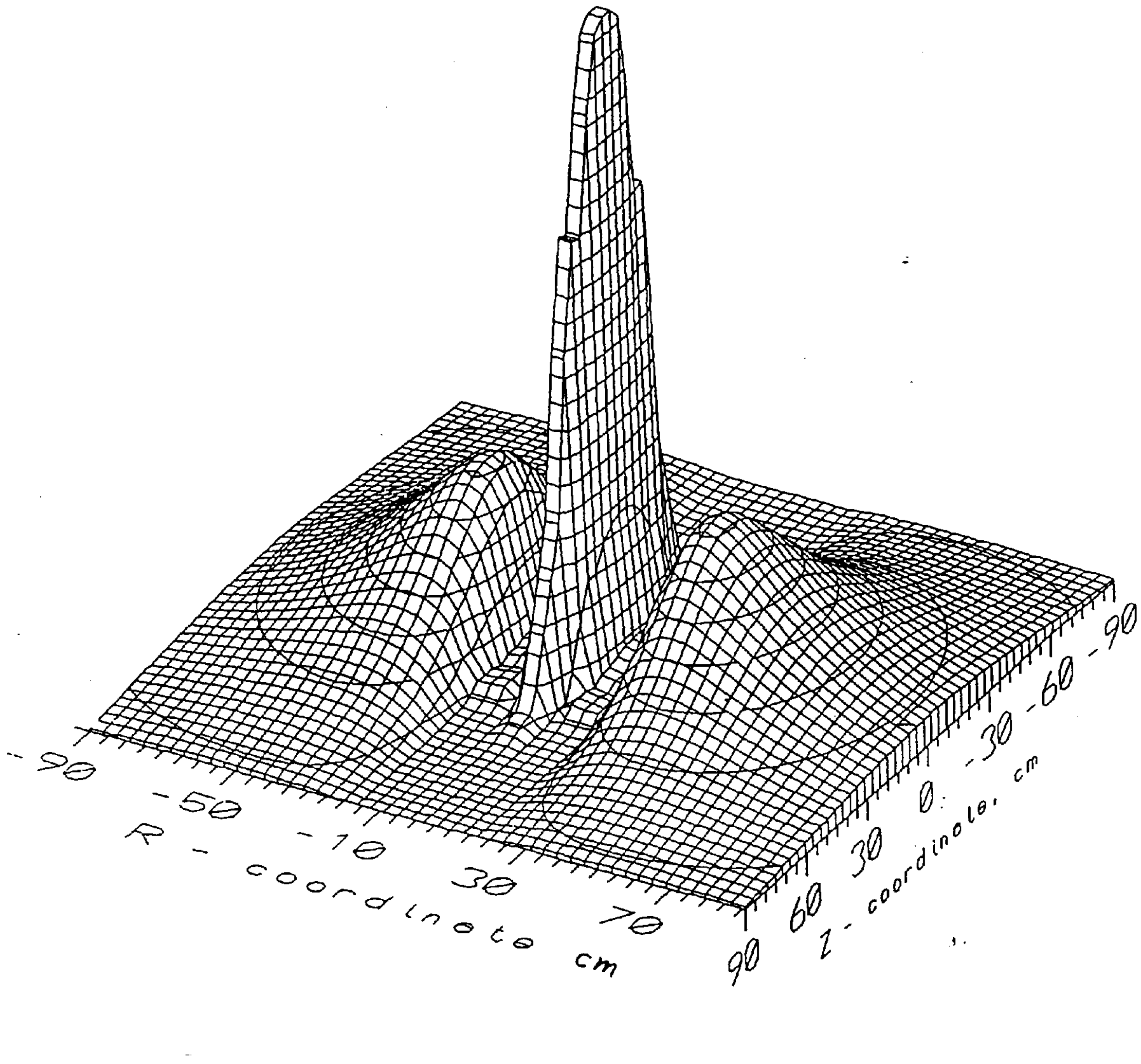


Fig.11: Distribution of unperturbed flux of thermal neutrons in the middle plane of the core

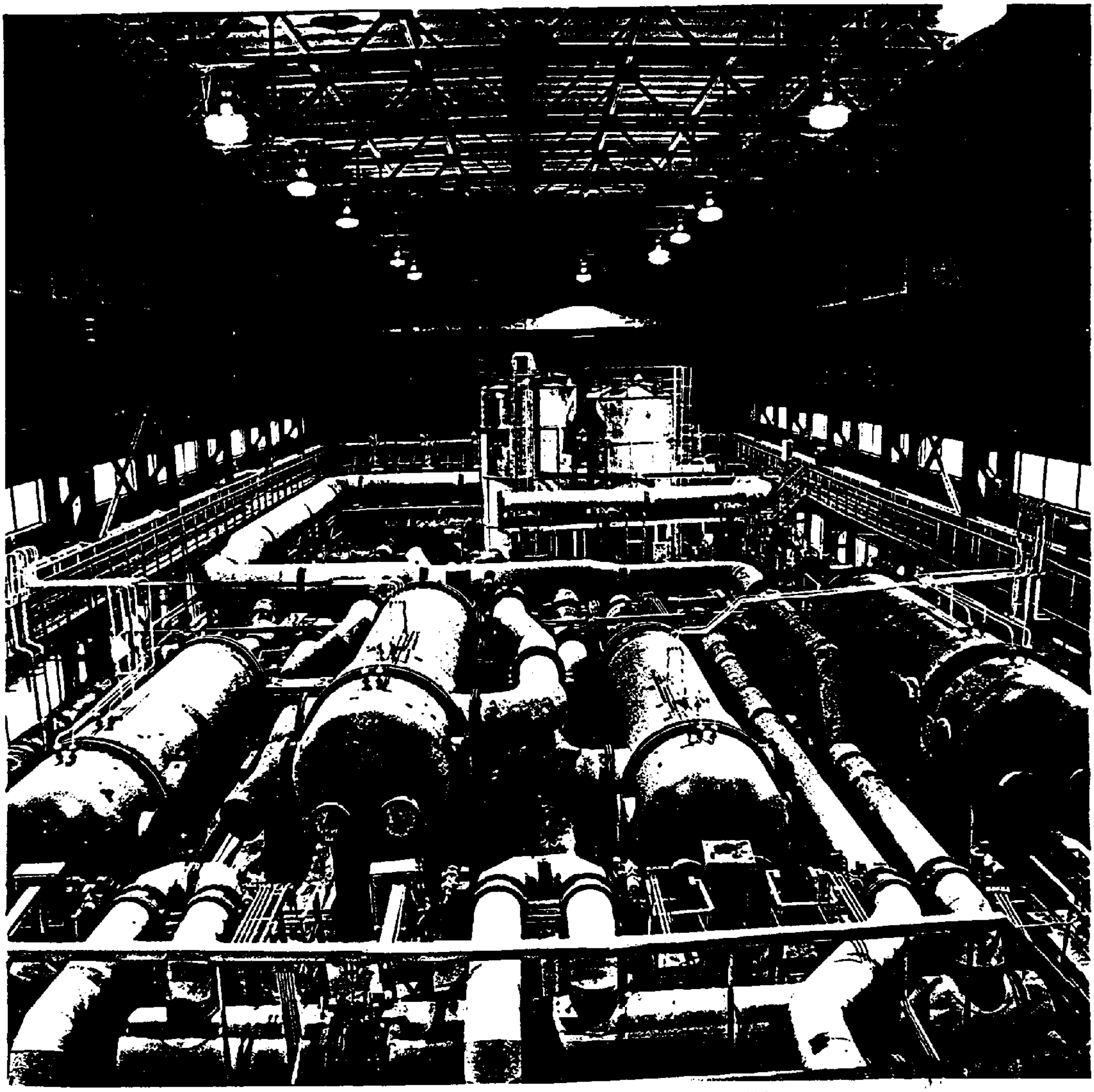


Fig.12: Pumping station of the intermediate (second) circuit.

Present Status of Reactor PIK*)

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*) The version of PNPI preprint: A.N. Erykalov, O.A. Kolesnichenko, K.A. Kopolev, V.A. Nazarenko, Yu.V. Petrov, S.L. Smolsky, "Reactor PIK", St. Petersburg, 1992, submitted to IGORR-2 meeting, May 18-20, 1992, Sacle, France

ПИК => Pressurized Beam Research ~~Reactor~~

Owner: Konstantinov ~~Leningrad~~ Petersburg Nuclear Physics Institute of the ~~USSR~~ Academy of Sciences of Russia

Location: Town Gatchina, ~~45~~ 30 ~~km~~ km south of ~~Leningrad.~~ St. Petersburg

Present Status: Under construction (start of construction 1976)

(slide 1 => general view of the reactor) (70% ready)

Goal of the reactor design:

To provide maximum number and minimum cost of the events in experimental apparatus.

II Reactor structure and parameters.

1. Reactor core. (slide 2 and 3)

a) Moderator and coolant: light water (H_2O).
small migration length $M \sim \sqrt{\tau}$
high specific heat C.

b) Core volume ($V \sim M^3$): $V = 50\ell$

c) Reactor power: 100 MW
(neutron intensity $\sim 0.8 \cdot 10^{19}$ n/s).

d) Average power density: $\bar{q} = 2$ MW/ℓ
($q_{max} = 4.5$ MW/ℓ large heat transfer area per volume)
$$\bar{q} \Rightarrow \frac{S}{V} \sim \frac{1}{\ell},$$ twisted cruciform pins
$\xi = 6.5\ cm^2/cm^3$ (Tr. 4)

e) Neutron leakage in reflector $\sim (1-\frac{1}{k_\infty})$
high k_∞ leads to high fuel enrichment $\zeta = 90\%$

f) Coolant inlet pressure: ≤ 50 bar
core vessel with double walls separates the core and reflector.

2. Reflector D_2O (no danger of radiation damage asin Be).

a) Large diffusion length $L \simeq 1m$ (0.2% H_2O),
reactor tank (height - 2m, diameter - 2.5m),
high flux of thermal neutrons at large distances and low background (slide 5 and 6)

b) Concentration of DTO $\Rightarrow$ 2 Cu/ℓ
(purification circuit)

c) Core vessel would be replaced every 2÷3 years (because of radiation damage).
It is possible to change the composition and dimensions of the core.
(Tr. 7)

3. Reactor control

a) Rapid control: 2 absorbing rings (enclosing the „trap" and moving apart simultaneously).

b) 8 Safety rods: europium plates (falling into the heavy water reflector) $\geq$5–6 for Xe-poisoning compensation.

c) Smooth control: gadolinium nitrate (dissolved in water in the gap between two walls of the core vessel). It is abolished now due to possibility of rapid insert $\Delta\rho > \beta > 0$.

4. Cooling system: 3-circuit cooling system
Closed intermediate circuit prevents the atmosphere pollution (because of possible demage of the heat exchanger in the first circuit) (Tr. 8)

4a Reactor safety (Tr. 9-10)

5. Reactor shielding

a) Reflector	1m
b) Heterogeneous iron-water shielding	0.5m
c) Heavy concrete	0.9m
d) Movable shielding (part of physics instruments)	1m

6. Full - scale reactor model
(for neutron-physics parameters measurements). (Reactor parameters Table 1)
(Tr. 11-14)

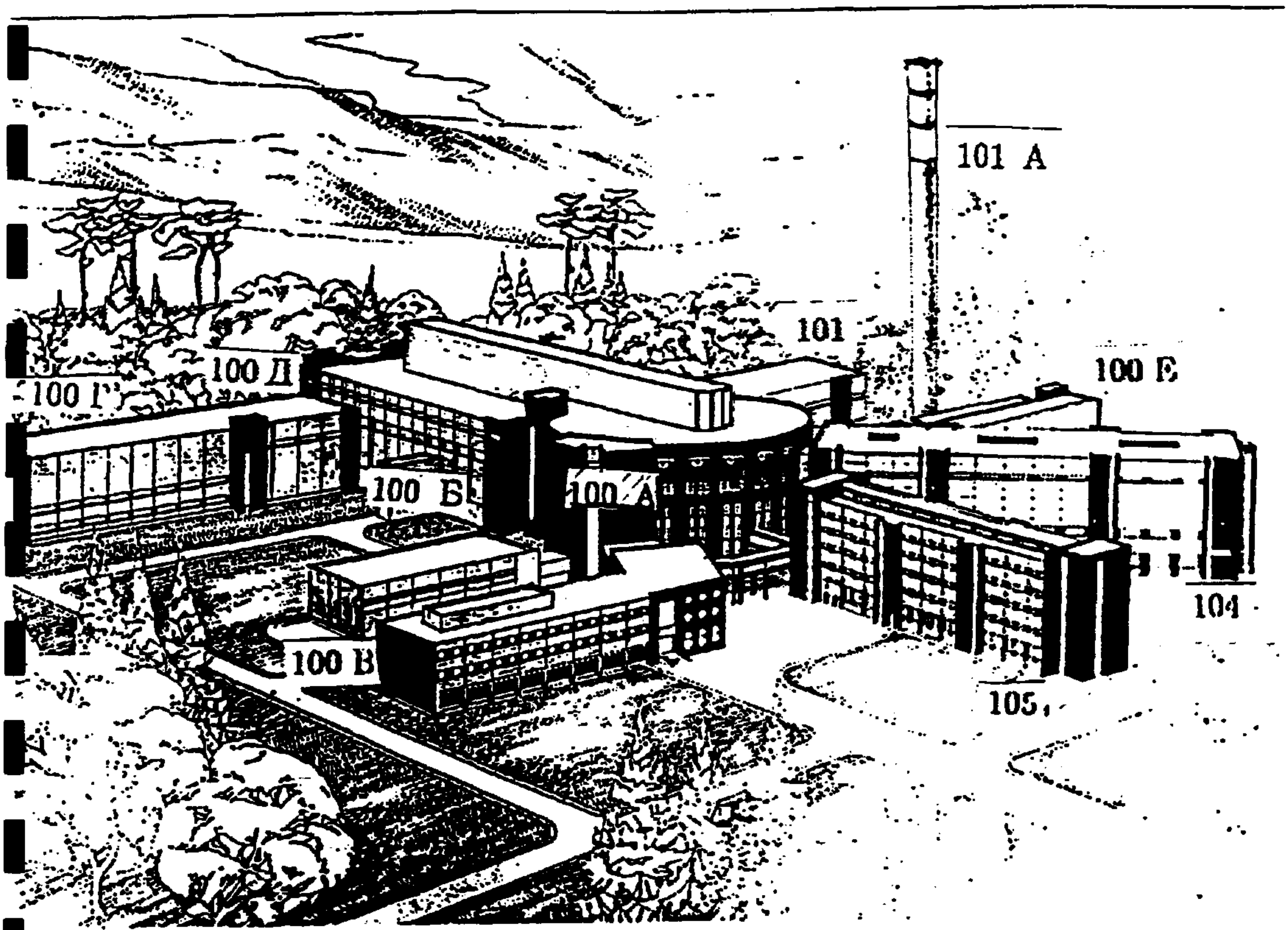


Fig. 1. General view to the main building of PIK complex.
100 A - reactor; 100 Б - hot (chambers), pumping block of the primary circuit and other equipment; 100 B,B' - administrative and recreative block; 100 Г - pumping block of the intermediate circuit; 100 Д - Electro substation for Б and Г($6kV$) blocks; 100 E - Apparatus of the hot neutron sources and isotopic purification unit D_2O; 104 - Neutron guides hall laboratories; 105 - Physics laboratories. 101 - ventilation center; 101 A - ventilation pipe.

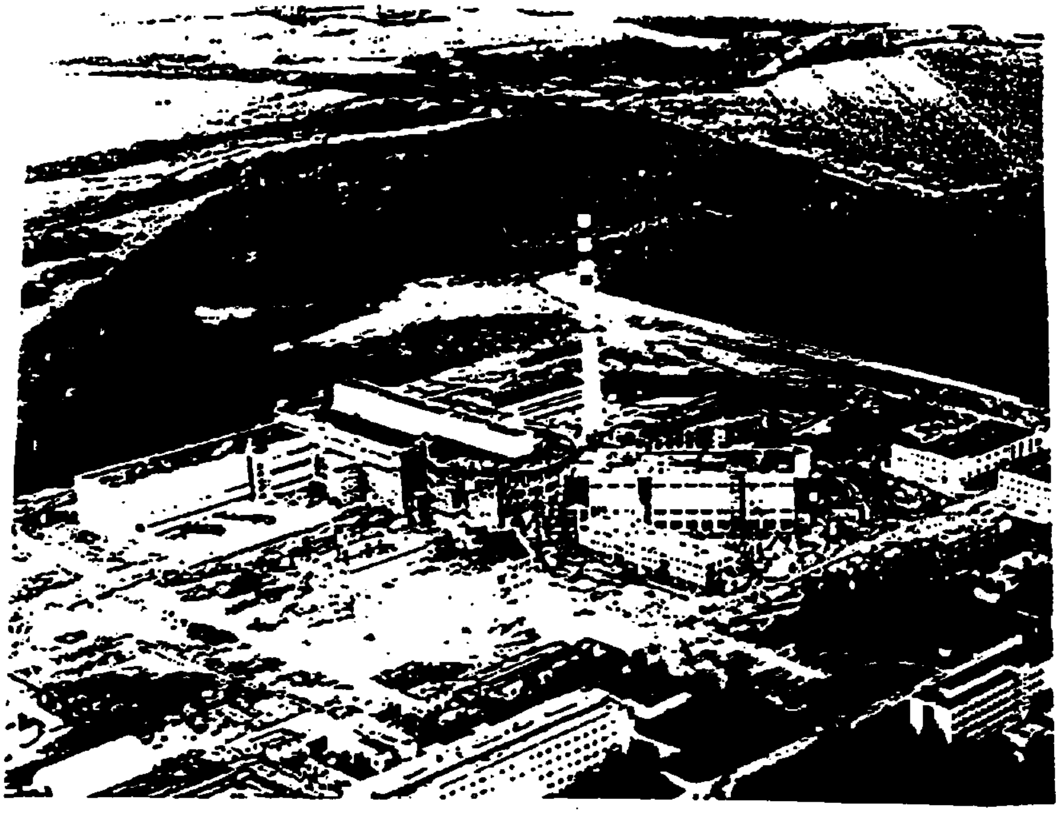


Fig. 2. View to PIK reactor complex (1989).

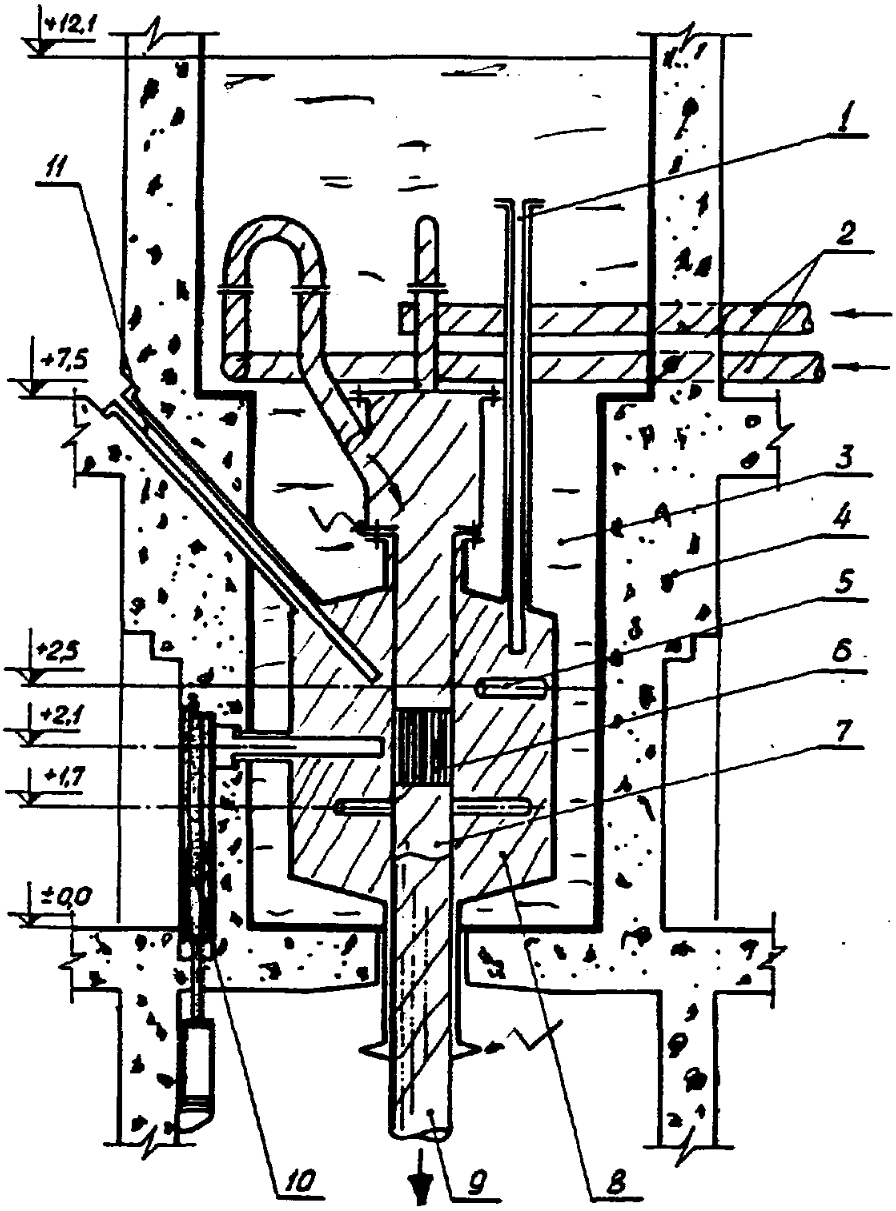


Fig. 3: Schematic vertical section of the PIK reactor.
1 - vertical channel; 2 - coolant inlet; 3 - water pool; 4 - biological shielding; 5 - horizontal experimental beam tube; 6 - core; 7 - replaceable vessel; 8 - heavy water reflector; 9 - coolant outlet; 10 - plug; 11 - inclined experimental beam tube.
Shown on the left are the distances from the floor of the experimental hall in meters.

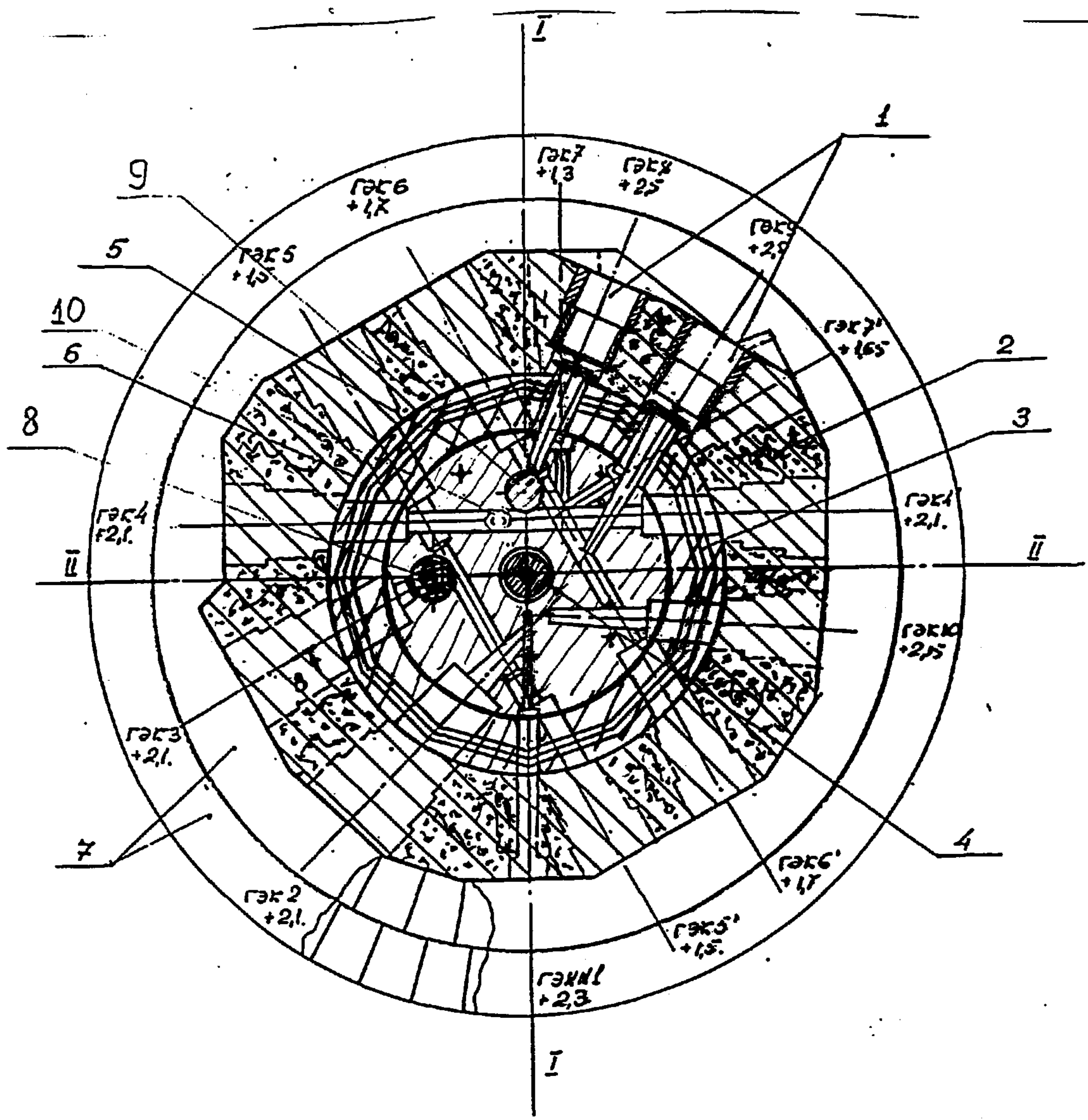
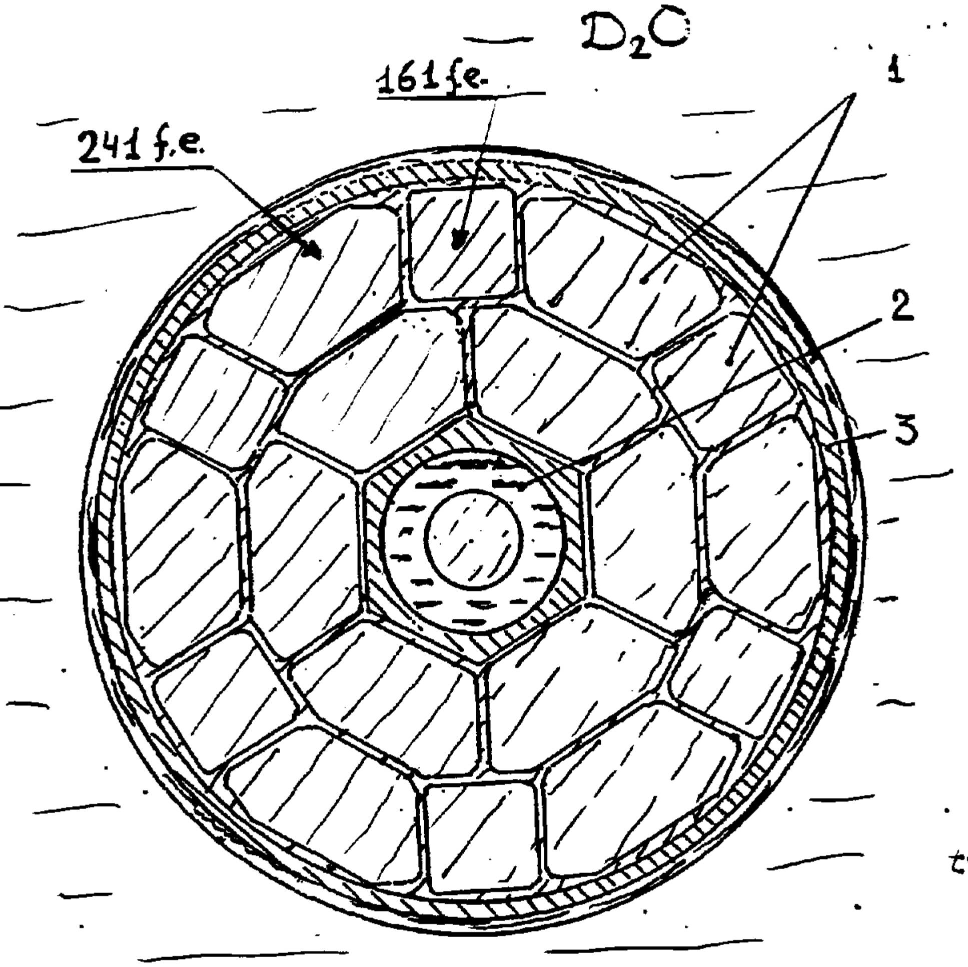


Fig. 4. Layout of arrangement of the experimental beams in the reflector of the PIK reactor.

1 - ГЭК plugs; 2 - biological shielding; 3 - iron - water shielding; 4 - core; 5 - heavy water reflector tank (D_2O); 6 - line - plates of the reactor pool; 7 - knock - down ring shielding; 8 - cold neutron source; 9 - hot neutron source; 10 - cold and ultracold neutron source.



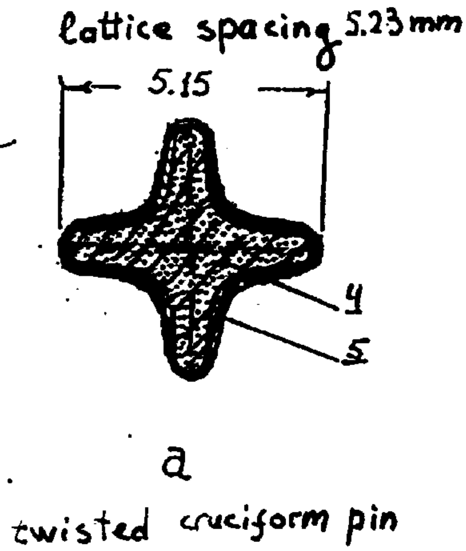


Fig .5. Section the core of the PIK reactor.
1 - fuel assembly; 2 - water trap with central vertical experimental channel; 3 - vessel.

Fig. 5a. Section of fuel element.
4 - fuel meet; 5 - cladding. 0.25 steel
2.2 $g^{235}U/cm^3$ in meet.
$\xi_r = 6.5\ cm^2/cm^3,\ \gamma_s = 540\ g^{235}U/\ell$

$q = 7.5$ MW/ℓ (tested at SM-2)

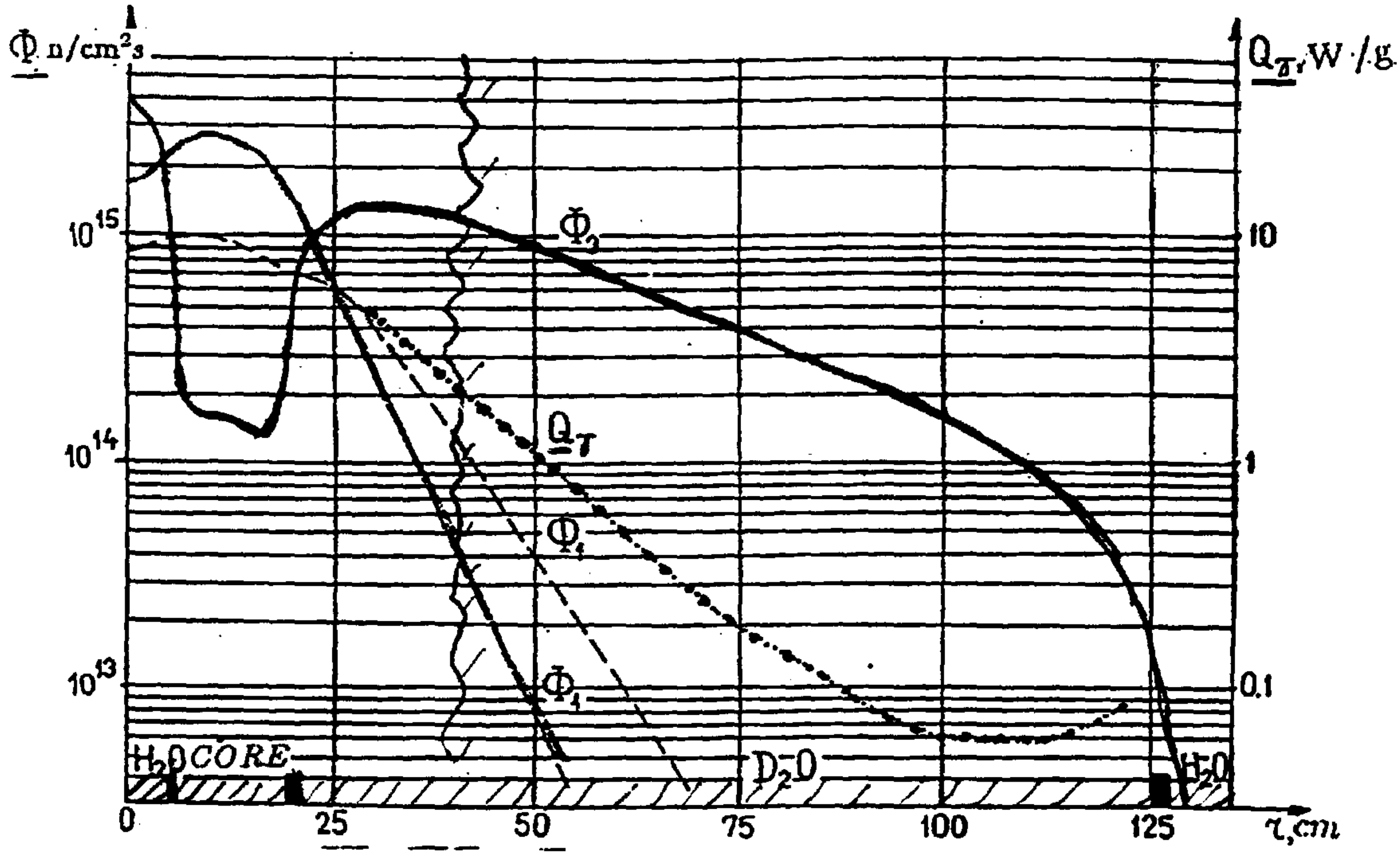


Fig. 6. **Distribution of unperturbed neutron flux Φ and energy output Q_γ for 100 *MW* reactor power.**

Φ_1 - fast neutron flux $E > 5\ keV$; Φ_2 - epithermal neutron flux $5\ keV > E > 0,6\ eV$; Φ_3 - thermal neutron flux $E < 0,6\ eV$.

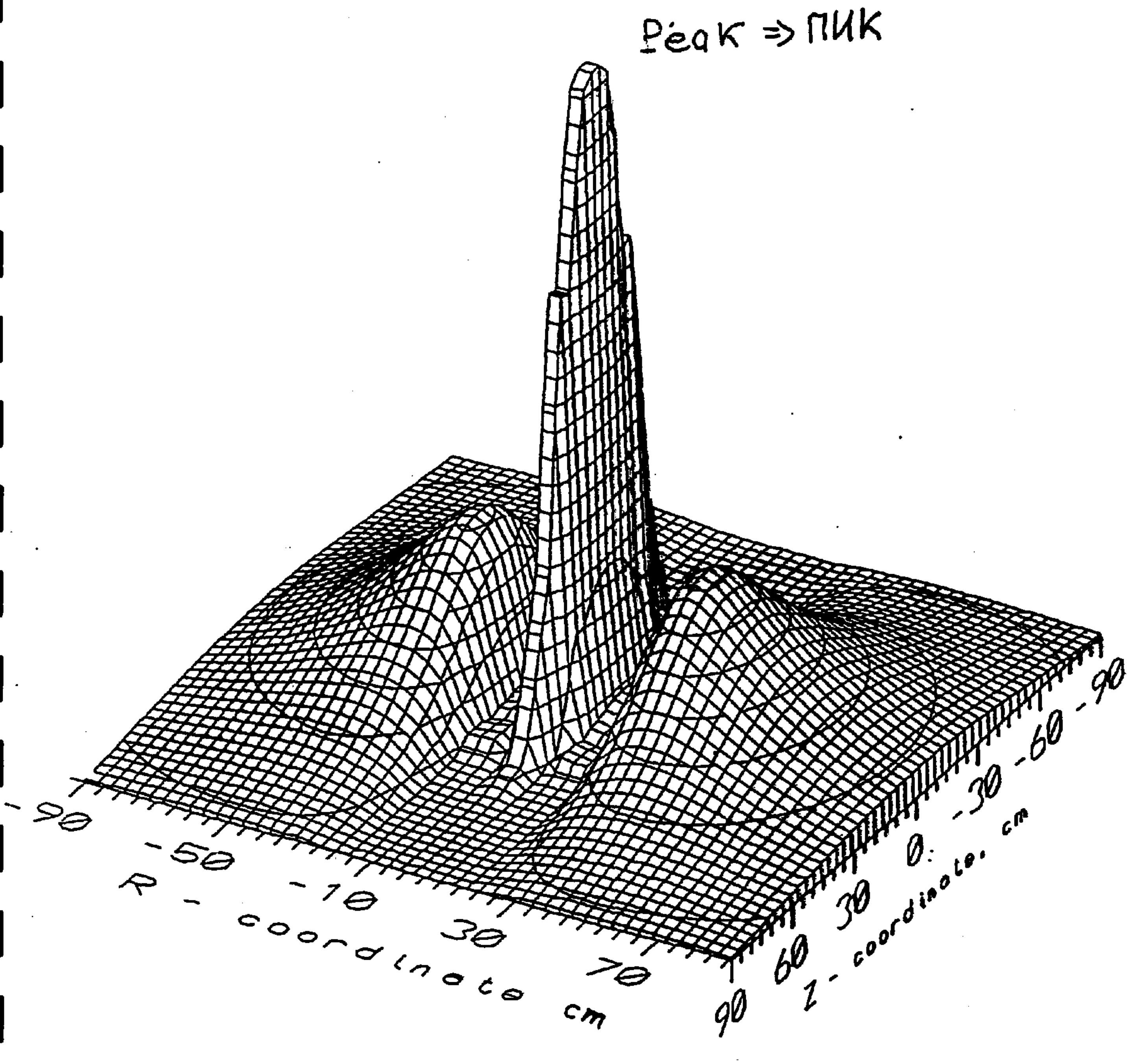


Fig. 16. Distribution of unperturbed flux of thermal neutrons along PIK reactor.

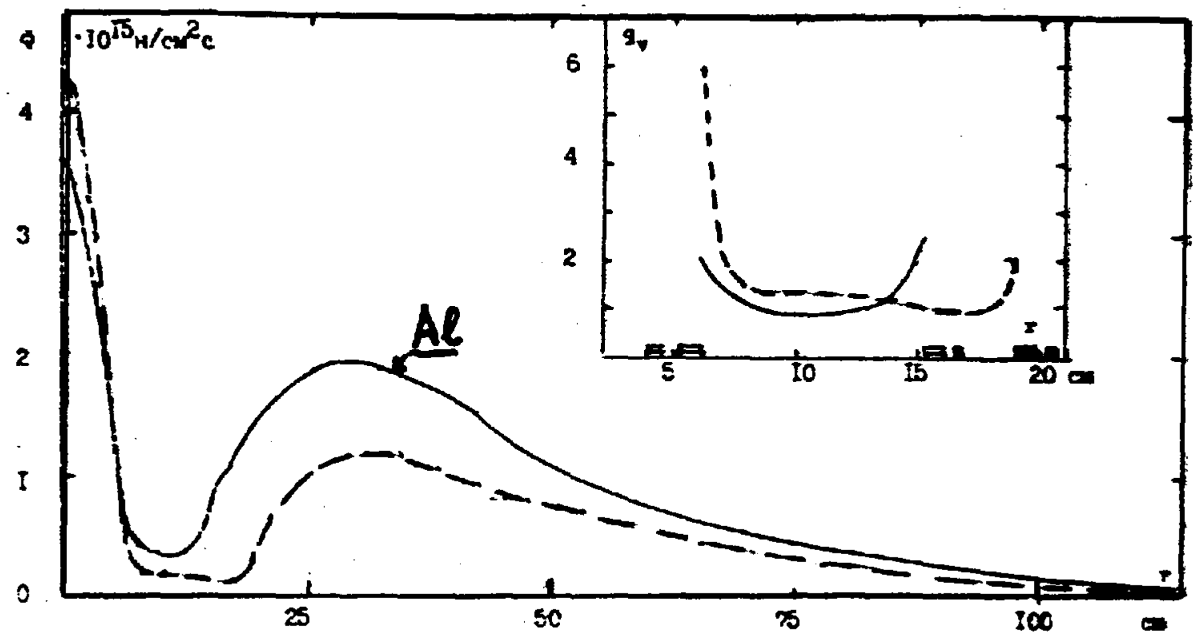


Рис.6. Распределение потока тепловых нейтронов Φ и неравномерности энерговыделения q по радиусу r в средней плоскости реактора при мощности 100 МВт. Сплошные кривые для водо-алюминиевой активной зоны ($w_{мет}$=0,413) с содержанием топлива 240 г ^{235}U/л, высотой 80 см и внешним радиусом 15 см. Пунктирные кривые для штатной конструкции активной зоны реактора ПИК. В обоих случаях в центре реактора размещена цилиндрическая легководная ловушка с алюминиевой направляющей.

	PIK-1	PIK-2 Al (WWR-M5)
H, cm	50	80
D, cm	40	30
ω_m	0.41	0.413
ξ_T, cm^2/cm^3	6.5	6.6
γ, ^{235}U g/l	540	240
$\Phi^{D_2O}_{max}$, 10^{15} n/cm^{2}s	1.3	2 ⇒ 1.5

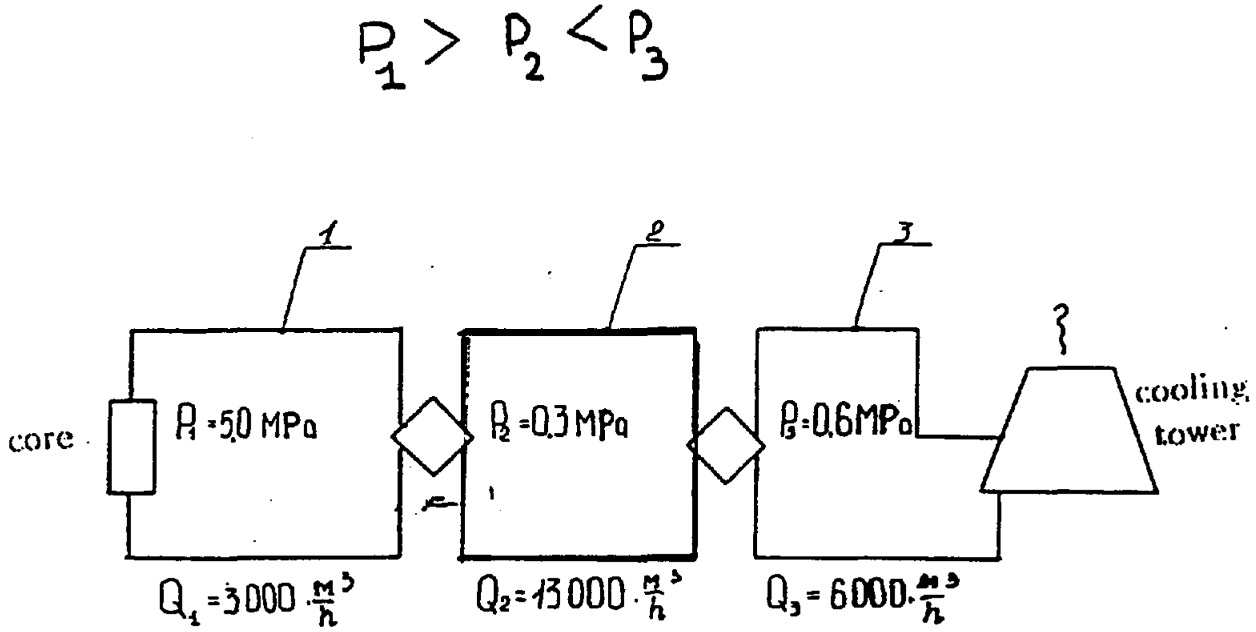


Fig. 8. Three circuit cooling system.

1 - core and primary sealed circuit; 2 - intermediate (secondary) circuit;
3 - third circuit with cooling tower.

		RBMK Chernobyl	PIK Gatchina	PIK/RBMK
1	Power, MW	3200 (th)	100	1/32 ⇒ ^{131}I
	Fission prod., kg	2000	6	1/300 ⇒ ^{137}Cs
2	Core mass, t	220	0.16	1/1400
	Therm. energy stored, MJ +)	$3-4\cdot10^5$	~20-100	$\sim 1/10^4$
	Reactor Build.	⇓ was destroyed	⇓ will not destroy.	
3	Graphite, t	1850	–	
	Energy $C+O_2$, MJ	$5\cdot10^7$	no fire	
	Reactivity void coeff.	positiv	negativ ++)	

+) $E_{st} = Mc\overline{\Delta T}$, $\Delta E_{rel} \simeq \alpha \Delta T \Delta S$

++) Fuel will not melt by inserting $\Delta\rho_0 \simeq 1\%$ within 0,1 sec.

1. Containment prevents contamination the enviroment

2. Containment can withstand the shock wave up to $P \simeq 4$ Kbar

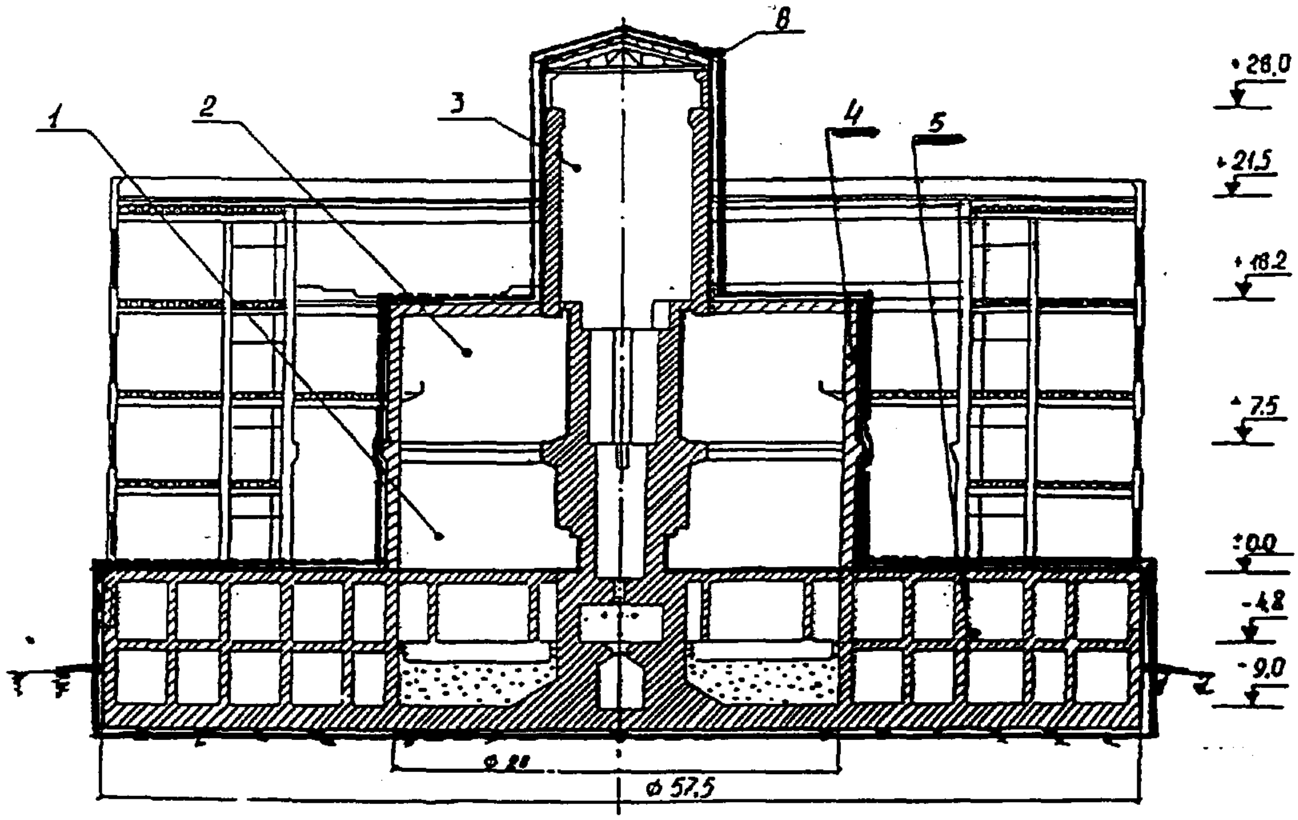


Fig. 7. Side section of the PIK reactor.

1 - horizontal beam experimental hall; 2 - inclined beam experimental hall; 3 - technological hall; 4 - reinforced concrete cylinder; 5 - containment contour.
The dimensions are in meters from floor experimental hall.

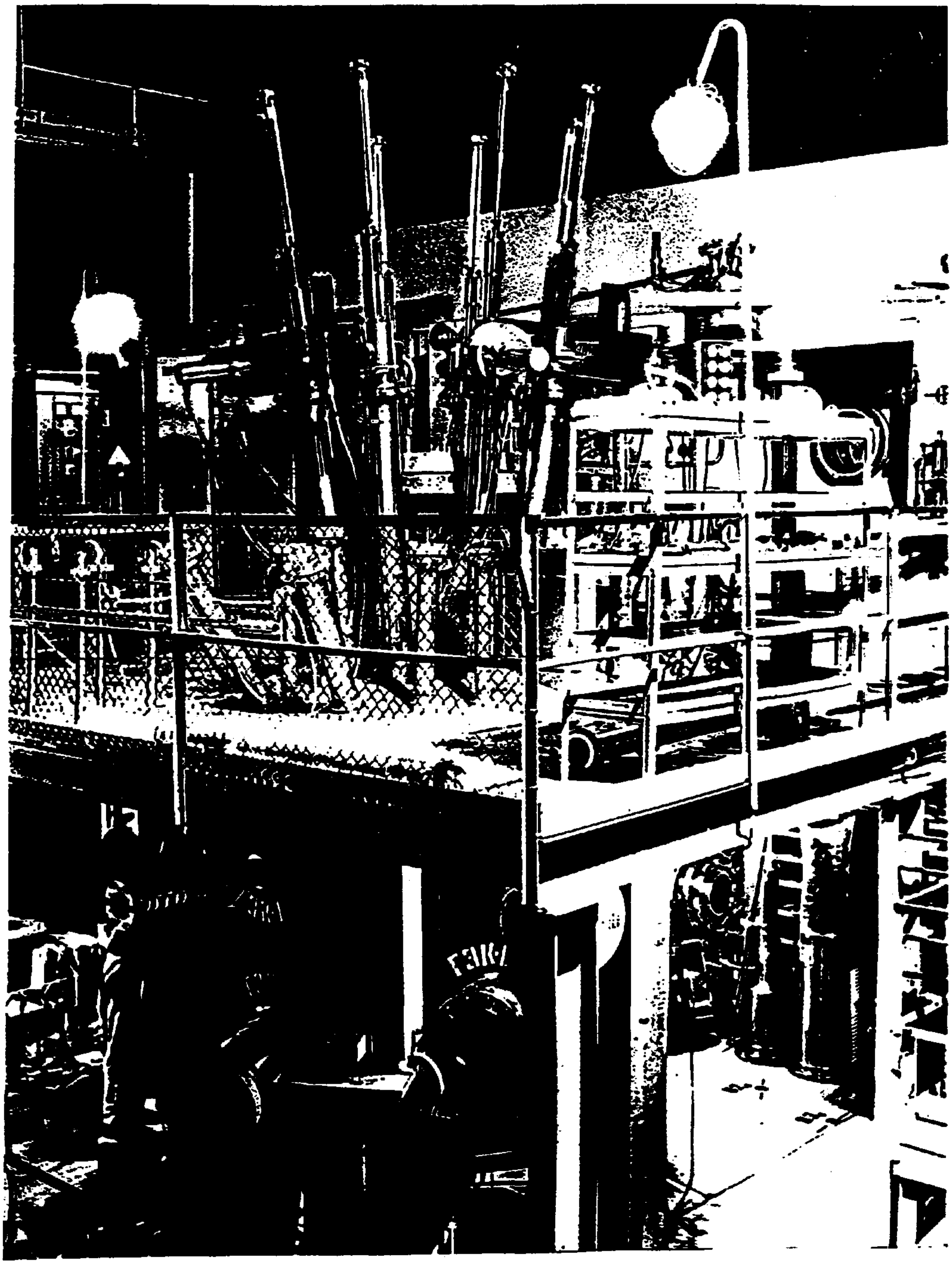


Fig. 9. Mock - up of PIK reactor (100 W).

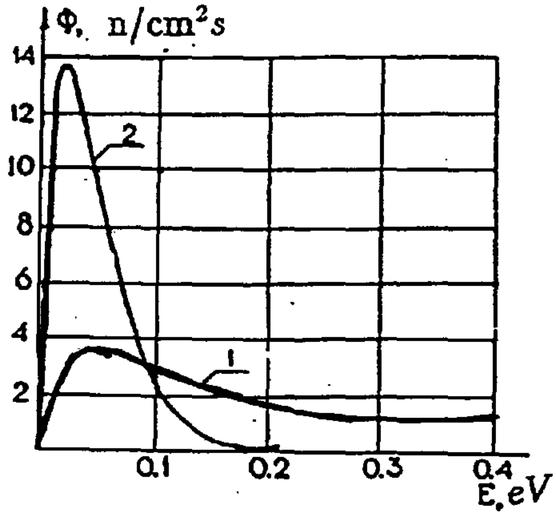


Fig. 10. Spectrum of thermal neutrons $\Phi(E)$.
1 - in PIK-04 core; 2 - in the water at 293K.

⇒ 2 overlapping thermal groups

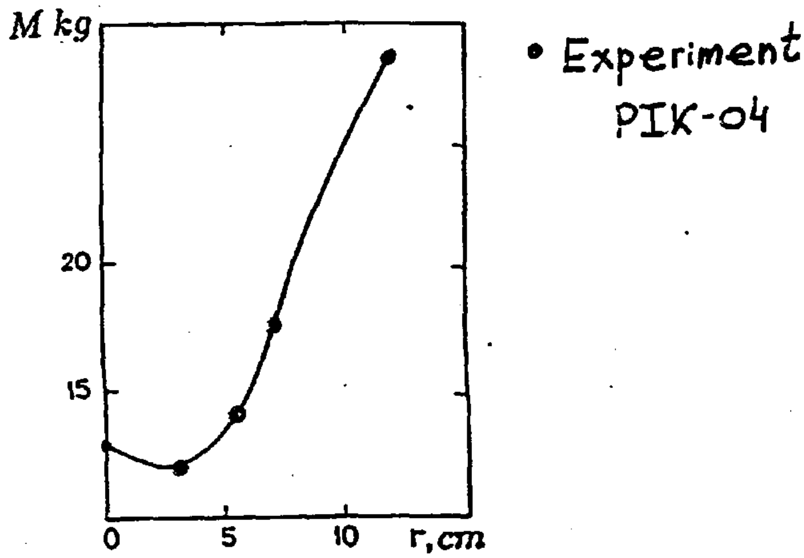


Fig. 11. Critical mass M as a function of r - radius of the inner light-water trap.
Continuous line - calculated values. Points - experiment (error is less than the points dimensions).

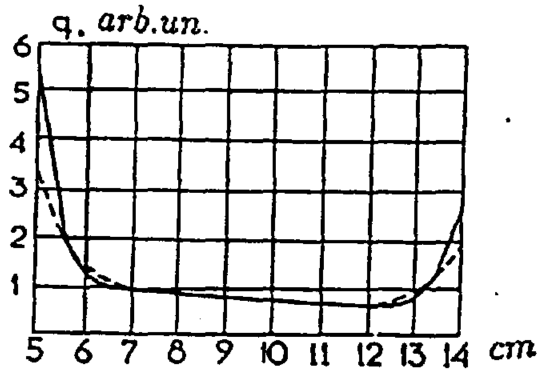


Fig. 12. **Distribution of energy output in medial plane along *PIK* – 04 core.**
Continuous line - 4 diffusion groups. Dash line - 3 diffusion groups.

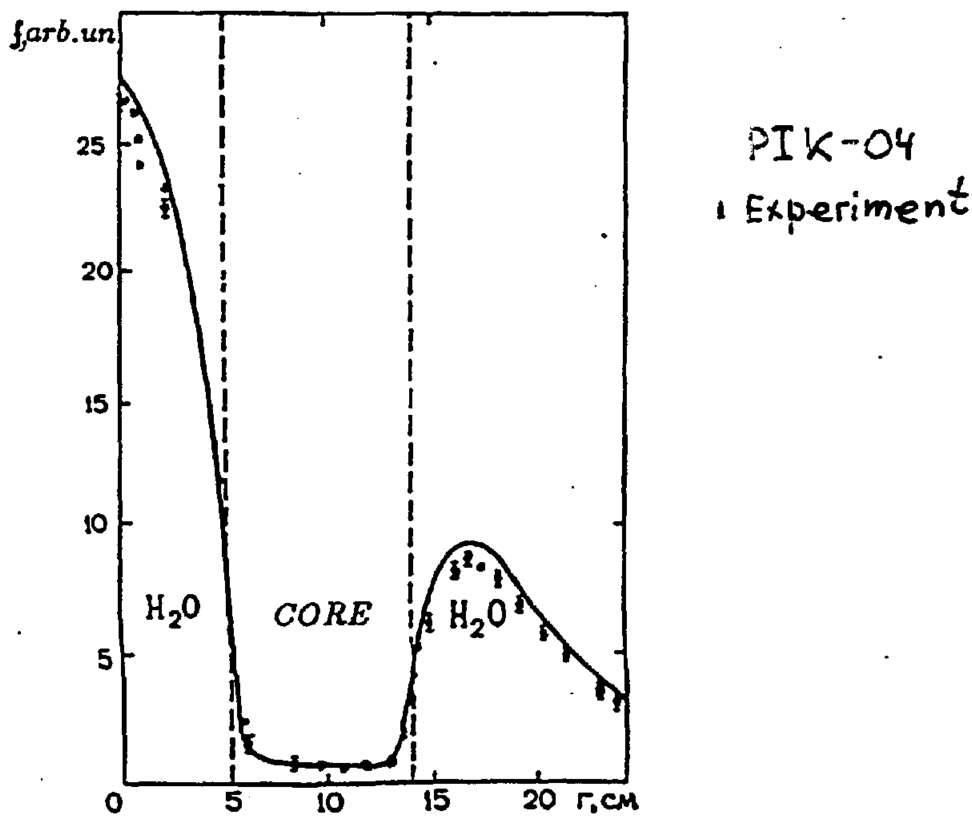


Fig. 13. **Fission rate as a function of the radius critical assembly.**
Continuous line - calculated values. Points - experiment.

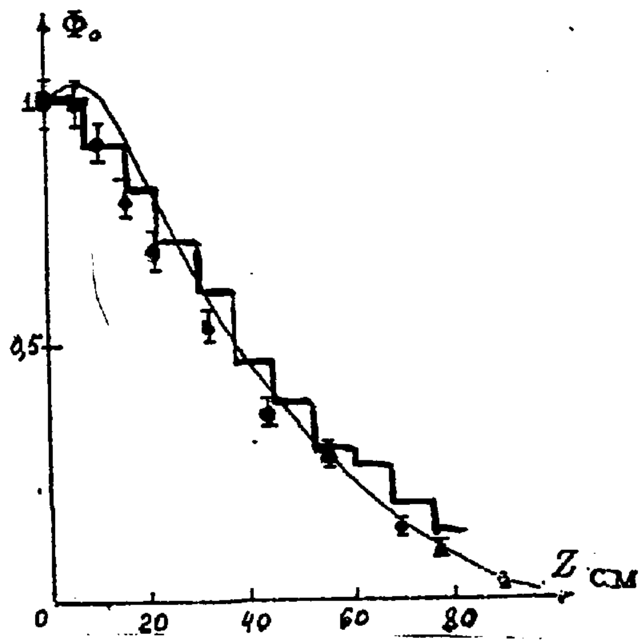


Fig. 14. Distribution thermal flux density Φ_o (in respect to l) along Z direction of beam tube ГЭК – 1.

Continuous line - calculated values for 4-group diffusion program without consideration of the channel; hystogramme - computation using Monte Carlo method for the channel 18x18 cm^2; ⊙ - experiment (by gold activation); △ - experiment (by semiconductor detectors).

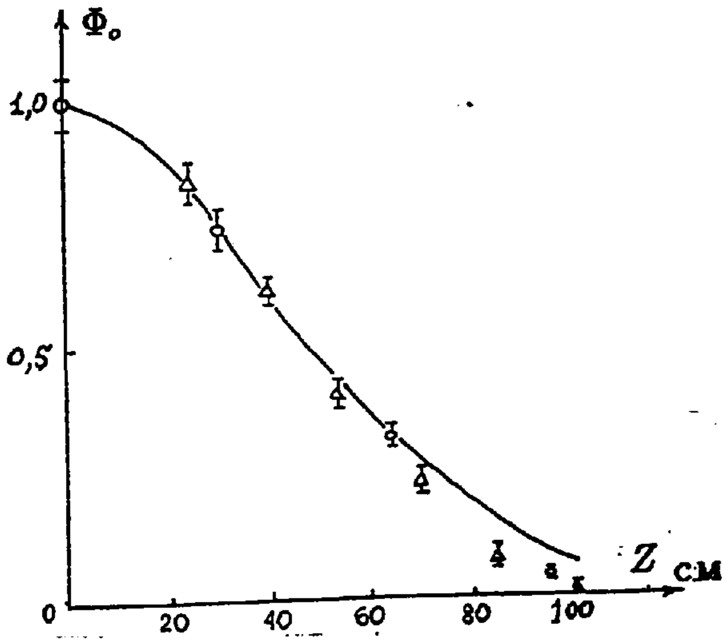


Fig. 15. Distribution thermal flux density Φ_o (in arbitrary units) along Z direction of beam tube ГЭК – 4.

Continuous line - calculated values for 4-group diffusion program without consideration of the channel; ⊙ - experiment (by gold activation); △ - experiment (by semiconductor detectors).

Table 2. Thermal neutron flux density and energy output in experimental channels at 100 MW reactor capacity. Experiment error is 15%.

Designation (see fig.4)	Channels	Coordinates of the channel botton R cm	Z cm	Channel diameter cm	Calculated unperturbed flux $\Phi_{th,c}$ $10^{14} n/cm^2s$	Measured perturbed flux Φ_{th} $10^{14} n/cm^2s$	Φ_f for E>1,2 MeV $10^{14} n/cm^2s$	q_γ W/g
ЦЭК	central	0	0		41	45	5	45 ÷ 53
ГЭК 1	radial	30	21	9	7,6	5,7	0,21	3,5÷4,7
ГЭК 2	tangential	52	1,5	25,2	7,7	4,6	$2,2\ 10^{-2}$	
ГЭК 3	tangential	97	1,5	35,0		0,4		
ГЭК 4	going through	48	1,5	8,2	9,1	9	$2,6\ 10^{-2}$	2,6÷3,2
ГЭК 5	going through	53	-59		2	1,8		
ГЭК 6	going through	53	-38	8,3	4	3,6	$2,6\ 10^{-3}$	
ГЭК 7	V-shaped	40	-80	19,4	1	0,73		
ГЭК 8	tangential	63	44,5		2,6	2,2	$9\ 10^{-3}$	1÷0,68
ГЭК 9	tangential	39	40	8,2	4,4	3,6		
ГЭК 10	tangential	49	7	25	8	4,8		
НЭК 1÷6	inclined	37÷79	-10÷27	4÷9	~ 6	~ 5	$\sim 5\ 10^{-3}$	
ВЭК 1÷6	vertical	74÷103		6,1		0,6÷1,4	$\sim 10^{-2}$	

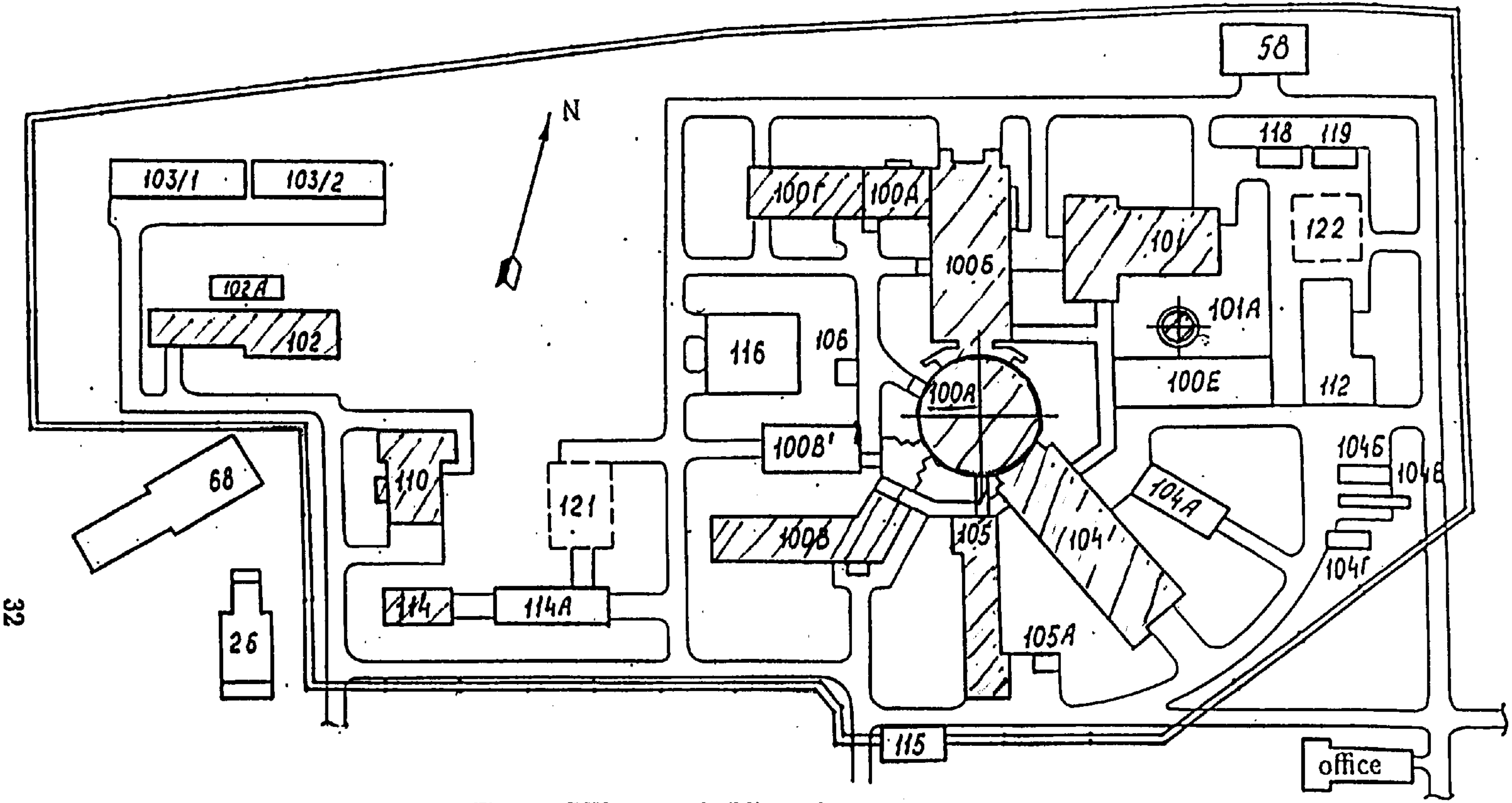


Fig. 17. PIK reactor buildings plan.

100 A - reactor and laboratories; 100 Б - pumping block of the primary circuit and hot chambers; 100 В - disinfection post and ventilation; 100 Г - pumping block of the intermediate circuit; 100 Д - energy block; 100 Е - cryogenic station; 101 - ventilation center; 101 A - ventilation pipe; 102 - water circulation pumping block; 102 A - cold water chamber; 103 (1,2) - water - cooling tower; 104 - neutron guide hall and laboratories; 104 A - technological block; 104 Б - water circulation pumping block; 104 В - cold water chamber; 105 - physics laboratories; 105 A - storehouse; 106 - carbon acid block; 110 - compressor block; 112 - isotopic purification block; 114 - storehouse; 114 A - storehouse; 115 - hall; 116 - emergency diesel power station; 118 - nitrogen station; 121 - training equipment; 122 - drain tank for irradiated liquids; 26 - chemical water cleaning; 58 - electric substation; 68 - emergency tank.

15

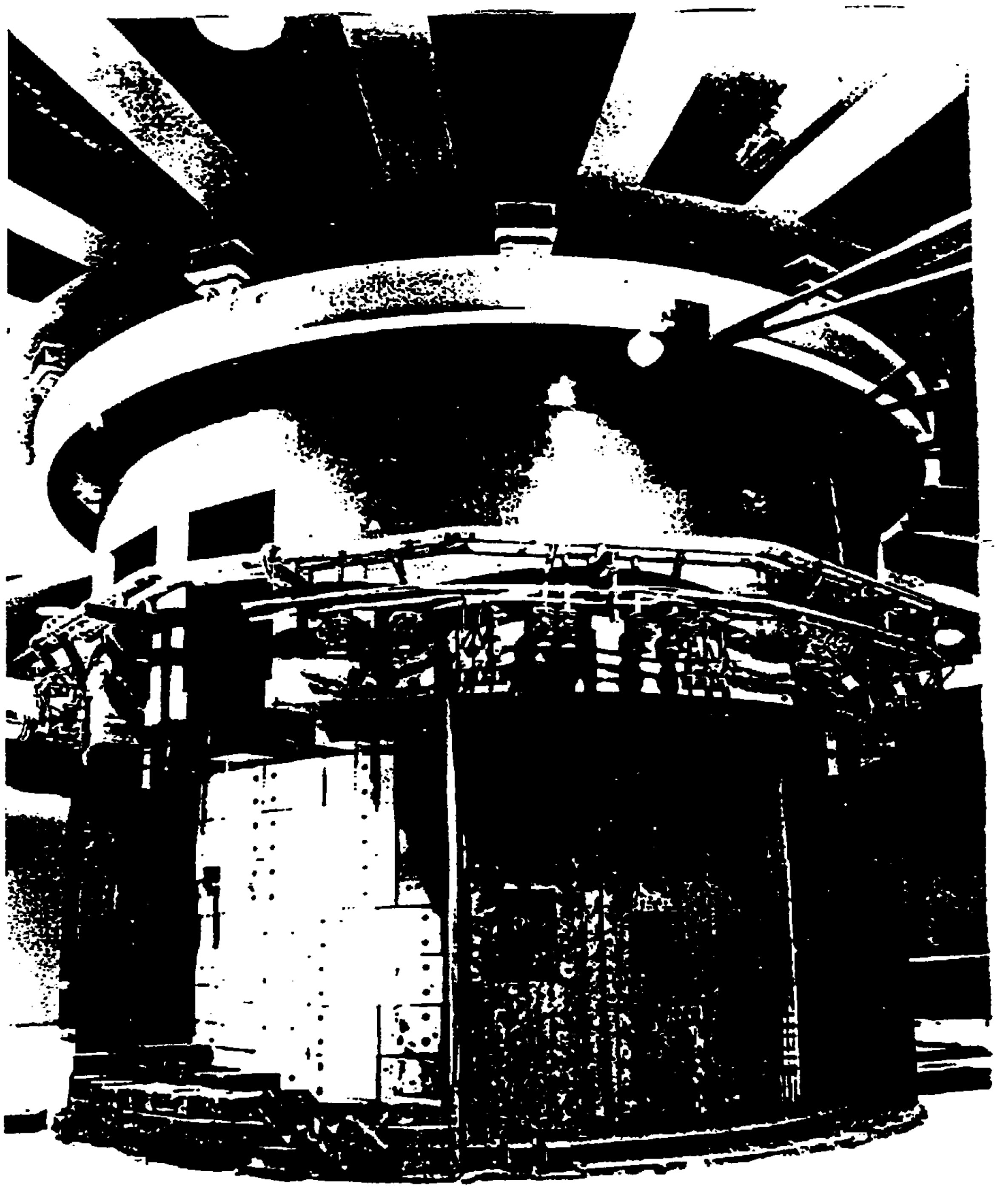


Fig. 18. Biological shilding of the reactor.

Fig. 19. Technology hall.

18

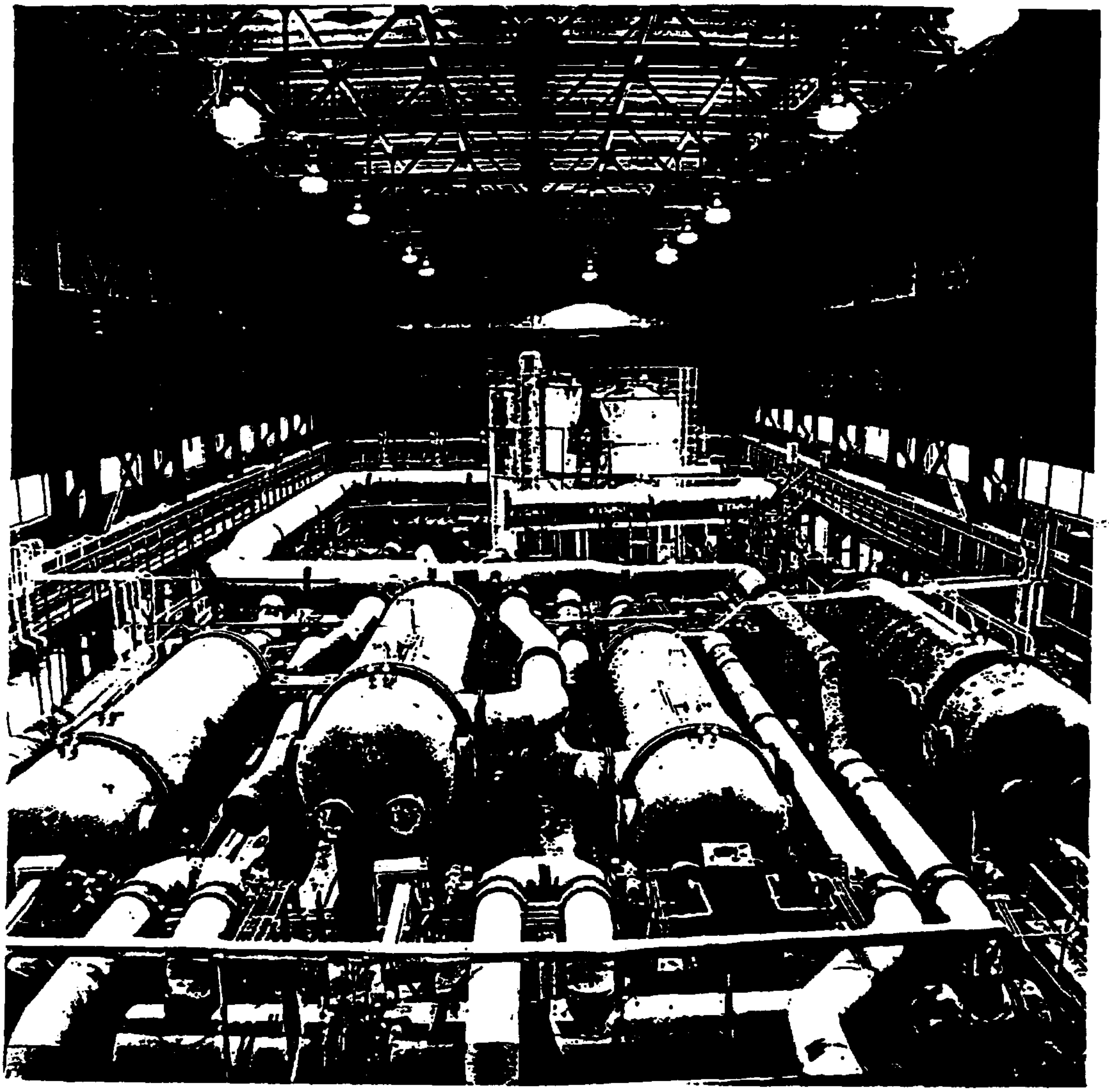


Fig. 20. Reactor equipment in pumping block of intermediate (second) circuit.

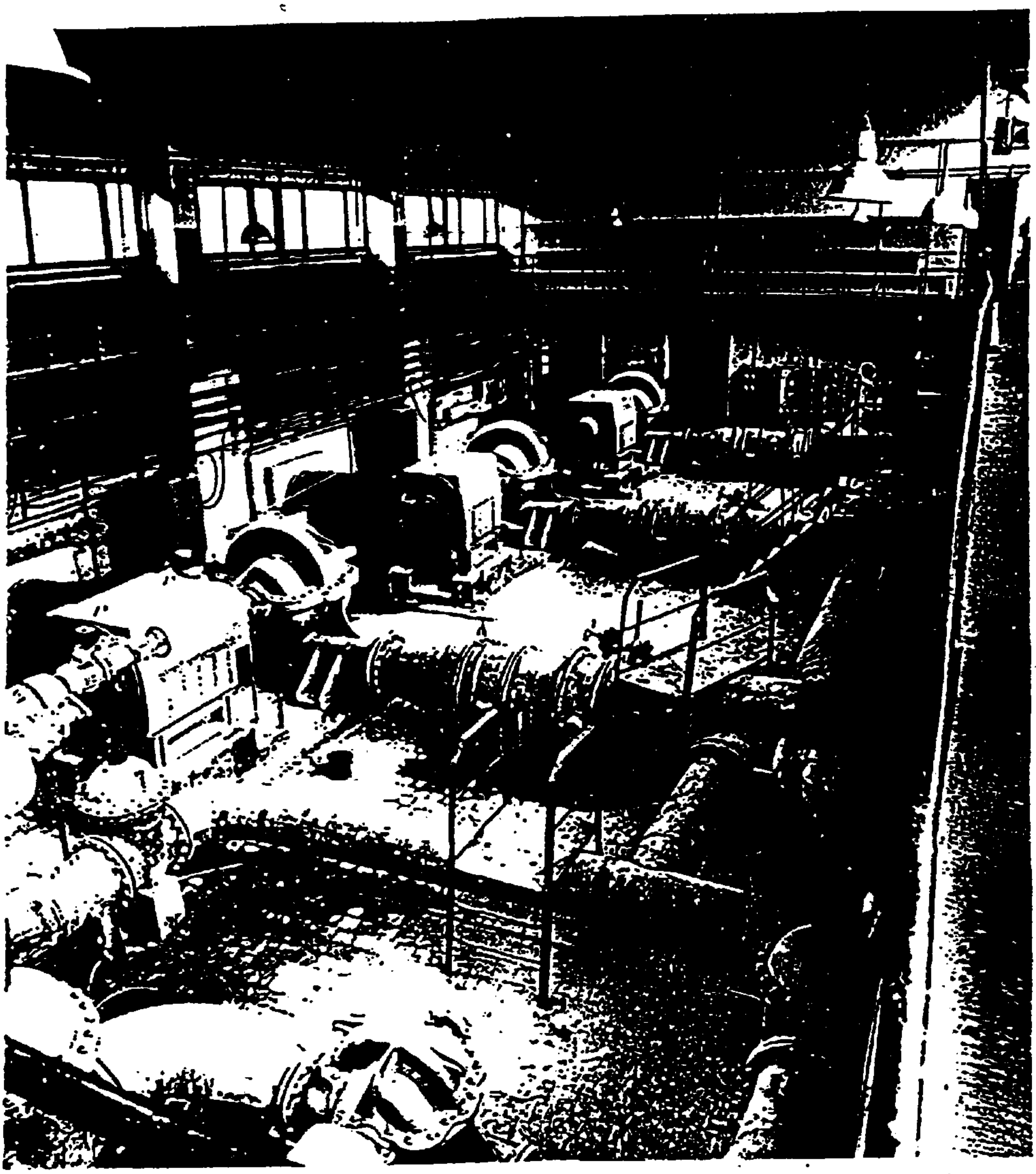
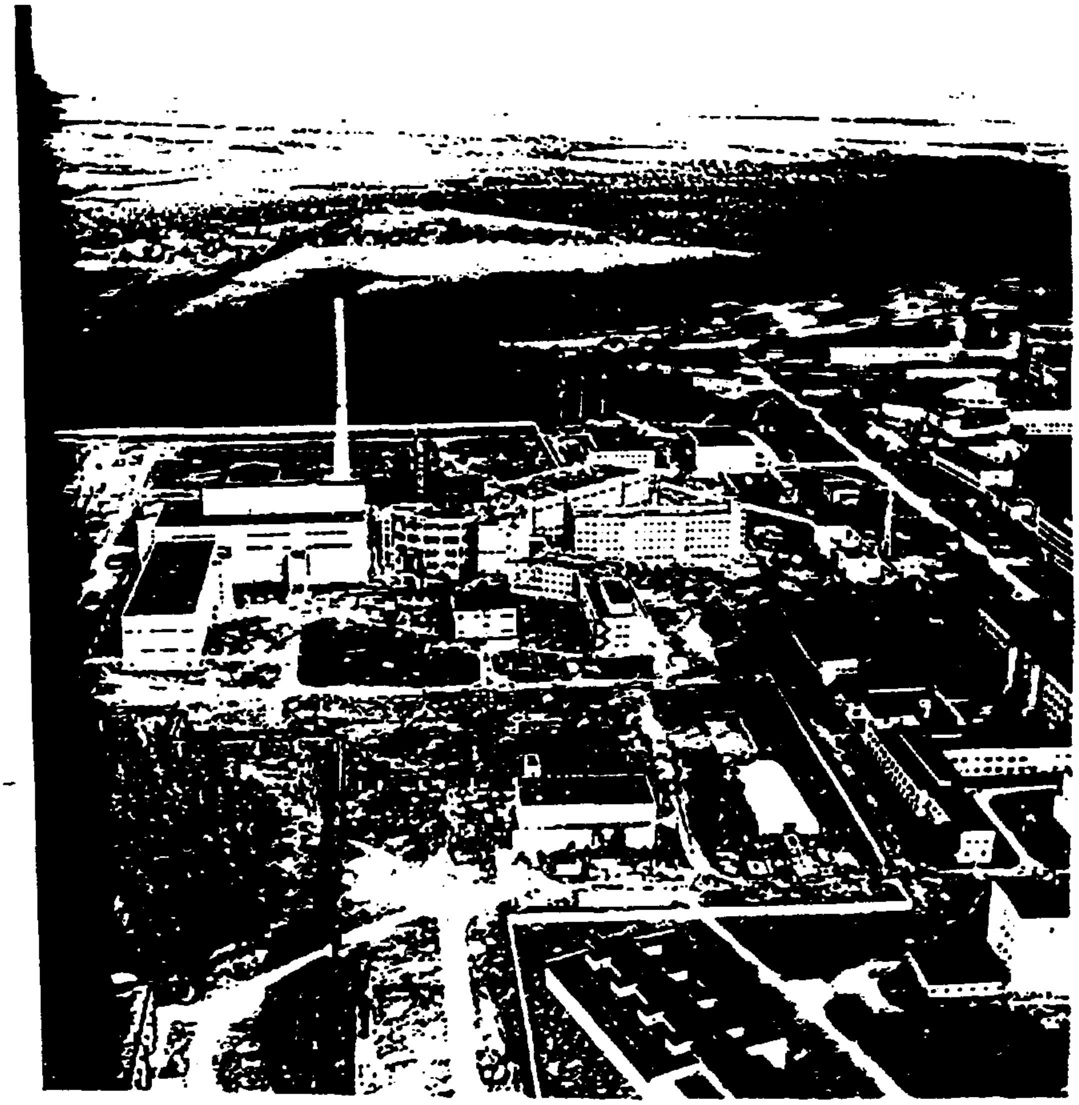


Fig. 19. Reactor equipment in third circuit water circulation pumping block.



View to PIK reactor complex (1991).

Reactor PIK and GHFR (table 4) are similar
The neutron fluxes in reflector are the same
But PIK has some advantages

1. The outlet beam intensity is 3 times higher (due to more compact shielding)
2. PIK has neutron trap. Thermal neutron flux in trap is 3 times higher than in the reflector.
3. The core and beam tubes could be replaced after reactor would be in operation. Using core with WWR-M type fuel elements increases neutron flux in reflector at a factor 1.5.
4. For special precise measurements PIK has vibration shielding.

PNPI proposal:

1 International Research Institute at Gatchina.

2 International cooperation in building and ranning the Research Reactor PIK

PIK status => similar to ILL reactor

3 Russian contribution => 70 % ready reactor

We need about $30 \cdot 10^6$ doll.

4. We are open for proposals.

CURRENT STATUS OF RESEARCH AND TEST REACTORS IN JAERI

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INTRODUCTION

JAERI was founded as a national center of research and development for the nuclear energy in 1956. In next year, the first research reactor JRR-1 reached the first criticality and was finally shut down in 1969. Research and test reactors were constructed sequentially in accordance with research requirements following JRR-1. At present, five reactors, JRR-2, JRR-3M, JRR-4, JMTR and NSRR, are in operation. Table 1 shows outlines of research and test reactors in JAERI.[1),2)]

JRR-2

The JRR-2 is a tank type (CP-5 type) heavy water moderated and cooled reactor of 10MW. It reached the first criticality in 1960. The total operating time is approximately 72,000 hours, and integrated power is 640,000 MWh. In 1987, the core was converted from HEU (93% enriched uranium) to MEU (45% enriched uranium) fuel. Normally, the annual operation consists of 12 cycles, and in each cycle the reactor is operated continuously 12 days.

The main utilization fields are neutron scattering, irradiation of fuels and materials, radioisotopes (RIs) production and activation analysis. Recently, the reactor is being used for the boron neutron capture therapy (BNCT) and related studies.

In July, 1990, the JRR-2 was shut down for reparing the main cooling pump, and will start it's operation in July, 1992.

JRR-4

The JRR-4 is a pool type research reactor of 3.5 MW. It reached the first cliticality in 1965, and total operating time is approximately 26,000 hours. The reactor is operated 4 days in a week and 6 hours in a day. The reactor was used for reactor shielding experiments at the early stage, and now it is used for nuclear educational activity, activation analysis, RI production, Si doping and so on.

The reactor operation with HEU (93% enrichment) U-Al alloy fuels is expected to continue up to the end of 1993, and the operation with LEU (less than 20% enrichment) fuels will start in 1996. In this conversion work, the utilization facilities are also planned to be upgraded for BNCT.

JMTR

The JMTR is a light water moderated and cooled MTR type reactor of 50 MW. The reactor reached the first criticality in 1968. Total operating time is approximately 43,000 hours, and integrated power is 86,000 MWd. Normally, the annual operation consists of 5 cycles, and in each cycle the reactor is opetated continuously 24 days. It is used for irradiation of reactor materials and fuels, and for RI production.

The JMTR has been operated with MEU fuels since July, 1986. According to the reduced enrichment program, the conversion work to LEU fuels (U_3Si_2-Al) is in progress. The safety review by regulatoly authority was finished in February, 1992. The operation with LEU fuels will be started in November, 1993.

NSRR

The NSRR is a TRIGA type reactor, and reached the first criticality in 1976. Since then, fuel behaviors at reactivity initiated accident have been studied by an intense pulsed irradiation. More than 2,200 pulsed operations have been done.

At the beginning, fresh fuels for light water reactor were tested. In 1989, irradiated fuel tests were started with some improvements of the experimental facilities. Also, combined and, sequential pulsed operations (with low flat top pulse and high peak pulse) were begun in the irradiation. In 1990, silicide plate type fuel tests were started.

Some of the results of the studies have been employed as the technological basis of the standards of reactor safety analysis by Japanese Government.

In future plans, irradiated mixed oxide fuel tests for advanced thermal reactor are planned, and fast reactor fuel tests are expected in the succeeding stage.

JRR-3M

The JRR-3, which was 10 MW, heavy water and slightly enriched uranium reactor, was modified and upgraded to JRR-3M.

The JRR-3M is a light water moderated and cooled, and beryllium reflected swimming pool type reactor with 20 MW. It erached the criticality in March, 1990, and started full power operation in November, 1990.[3),4),5),6)]

Figure 1 shows the reactor components of the JRR-3M and surrounding installations. The core components consist of reactor core, heavy water tank and structual components. They are set up at the bottom of the reactor pool. The core is composed of 26 standard fuel elements, 6 control rod elements, 5 irradiation elements and beryllium reflector. Each control rod element consists of a neutron absorber and a follower fuel element. Both fuel elements are UAlx-Al dispersed, MTR type fuels. Their enrichment is less than 20%. Table 2 shows the main paremeters of reactor.

A typical operating cycle consists of 4 weeks for operation (26 days continuous operation) and 1 week for refueling, irradiation sample handling and maintenance. Normally, the annual operation consists of 8 cycles. Five or six standard fuels are exchanged in each operating cycle.

Figure 2 shows the arrangement of experimental holes. Nine vertical irradiation holes are arranged in the core, and nine holes in the heavy water tank. Table 3 shows the irradiation facilities in JRR-3M. These facilities are used for the irradiation tests on nuclear reactor fuels and materials, Si doping, RI production, activation analysis and so on.

Figure 3 shows the arrangement of beam experimental facilities. Nine horizontal beam tubes installed in the heavy water tank are arranged tangentially to the core center to reduce fast neutrons and gamma rays in the thermal neutron beam. These facilities lead neutron from the core to the experimental equipments for neutron scattering experiments, neutron radiography, prompt gamma ray analyzing and so on. Table 4 shows the neutron flux at the end of beam tubes.

The beam tubes 8T and 9C are followed by two and three neutron guide tubes, respectively. Neutron guide tubes lead neutrons into the beam hall so that a sufficient number of beam experimental apparatus can be provided to users. 8T leads out thermal neutrons and 9C leads out cold neutrons supplied by a cold neutron source. Table 5 shows the beam experimental facilities in JRR-3M.

CONCLUSION

In JAERI, five research and test reactors are currently in operation. According to research requirements, they go on upgrading. JRR-3M was modified and started full power operation in 1992. JRR-4 has a plan of upgrading, including a facility for BNCT.

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5) K.Kakefuda, M.Tani and M.Issiki, "Completion of Reconstruction for Japan Research Reactor NO.3", Proceedings of the Third Asian Symposium on Research Reactor, Hitachi, Japan, November 11-14, 1991.

6) Y.Murayama and M.Issiki, "Full Core Operation in JRR-3 with LEU Fuels", Proceedings of the Int. RERTR Meeting, Jakarta, Indonesia, November 4-7, 1991.

Table 1 Research and Test Reactors in JAERI

Name	Type and Enrichment			Max. Power	Max. Neutron Flux		Start-Up Date
					Thermal	Fast	
JRR-2	D_2O (CP-5)	U-Al	93%	10MW			1960.10
		UAlx-Al	45%	10MW	2.0×10^{14}	6.0×10^{13}	1987.11
JRR-3	D_2O (tank)	MNU		10MW			1962.9
		UO_2	1.5%				
JRR-3M	H_2O (pool)	UAlx-Al	20%	20MW	3.0×10^{14}	2.0×10^{14}	1990.3
JRR-4	H_2O (pool)	U-Al	93%	3.5MW	3.5×10^{13}	8.7×10^{13}	1965.1
JMTR	H_2O (MTR)	U-Al	93%	50MW			1968.3
		UAlx-Al	45%	50MW	4.0×10^{14}	4.0×10^{14}	1986.8
NSRR	H_2O (TRIGA)	U-ZrH	20%	300kW (Pulse 23,000MW)	1.3×10^{12}	4.0×10^{12}	1975.6

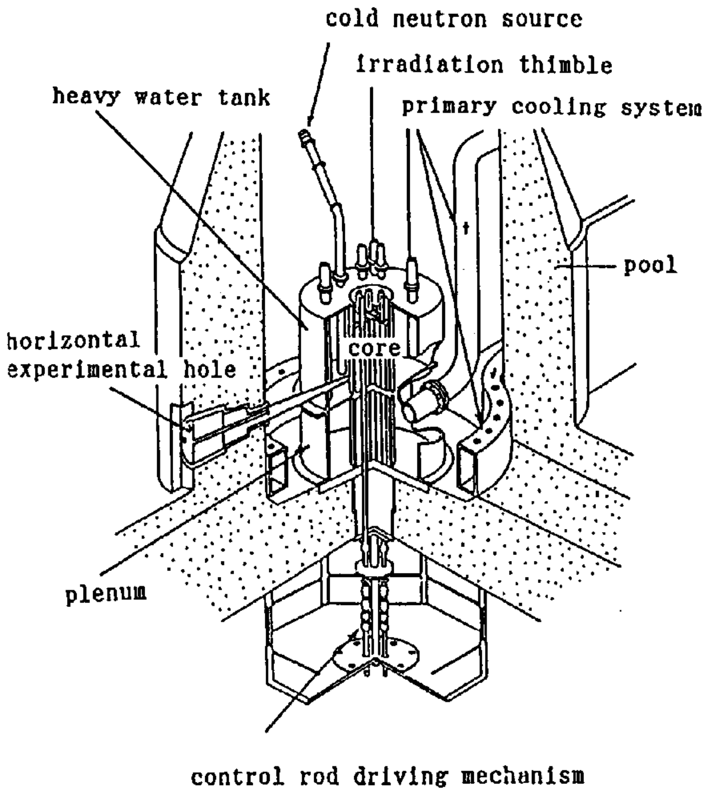
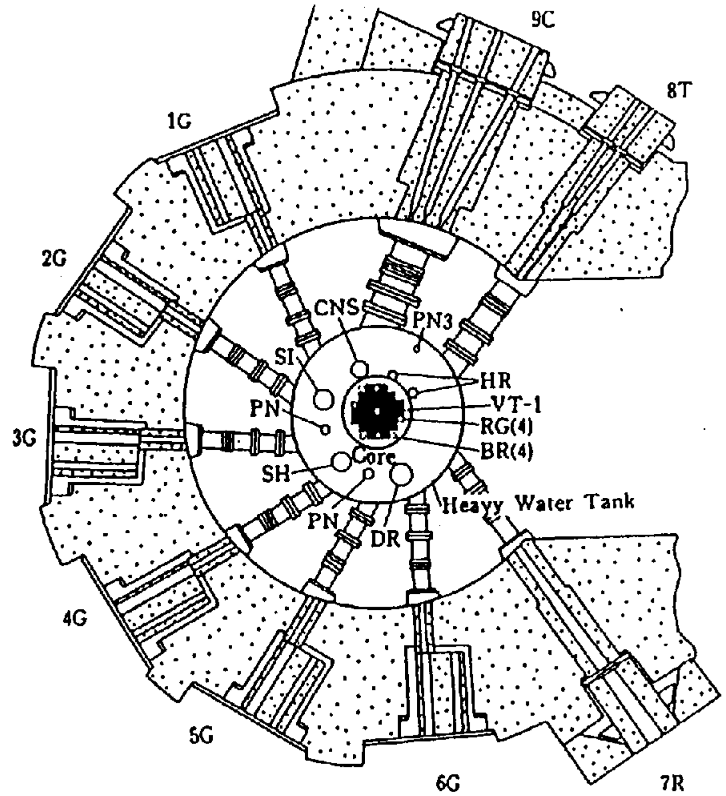


Figure 1 Reactor Components of JRR-3M



(Vertical Irradiation Holes:
HR, PN, PN3, SI, DR, RG, VT-1, BR, SH
Horizontal Beam Tubes:
1G~6G, 7R, 8T, 9C)

Figure 2 Experimental Facilities in JRR-3M

Table 2 Main Characteristics of JRR-3M

		Measured Value	Designed Value
Nuclear Properties			
Excess Reactivity	[%Δk/k]	15.9	16.3
Shutdown Margin (at One Rod Stuck)	[%Δk/k]	4.1	3.7
Control Rod Worth	[%Δk/k]	29.2	31.0
Moderator Temperature Coefficient	[%Δk/k/°C]	-1×10^{-2}	-2×10^{-2}
Moderator Void Coefficent (2% Void)	[%Δk/k/%void]	-0.2	-0.4
Average Thermal Neutron Flux in Core Region	[n/cm^2/sec]	8×10^{13}	8×10^{13}
Neutron Flux at Thermal Neutron Guide Tube Outlet	[n/cm^2/sec]	2×10^{8}	2×10^{8}
Neutron Flux at Cold Neutron Guide Tube Outlet	[n/cm^2/sec]	3×10^{8}	3×10^{8}
CNS Cold Neutron Gain		8	$\geqq 5$
Thermal-hydraulic parameters			
Flow Rate of Primary Cooling System	[m^3/h]		2400
Core Outlet Temperatrure (Max.)	[°C]		35
Core Inlet Temperatrure (Max.)	[°C]		42
Perturbation Factor of Channel Flow Rate		1.12	1.13
Maximum Fuel Temperature	[°C]	99.6	101.0
Shielding Characteristics			
Dose Equivalent Rate on Upper Shielding	[μSv/h]	12	$\leqq 60$
Dose Equivalent Rate in Beam Hall	[μSv/h]	0.2	$\leqq 6$

Table 3 Irradiation Facilities in JRR-3M

Name	Size and Number	Neutron Flux ($n/cm^2 \cdot sec$)		Application
		Thermal	Fast	
Hydraulic rabbit (HR)	ϕ37x2	$1x10^{14}$	$1x10^{12}$	·RI production
Pneumatic rabbit (PN)	ϕ37x2	$6x10^{13}$	$1x10^{11}$	·RI production
Activation Analysis (PN3)	ϕ20x1	$2x10^{13}$	$4x10^{9}$	·Activation analysis
Uniform irradiation (SI)	ϕ170x1	$3x10^{13}$	$3x10^{11}$	·Material irradiation ·Silicon doping
Rotating irradiation (DR)	ϕ140x1	$8x10^{13}$	$2x10^{11}$	·Large material irradiation
Capsule irradiation (RG,BR,VT-1,SH)	ϕ60x5 ϕ45x4 ϕ100x1	$3x10^{14}$ $2x10^{14}$ $8x10^{13}$	$2x10^{14}$ $5x10^{13}$ $2x10^{11}$	·Exposure test ·RI production

( ) shows an abbreviation of facility in Fig. 2

Table 4 Measured Thermal Neutron Flux at the End of Beam Tubes at clean core

BEAM TUBE NAME	Thermal Neutron Flux ($n/cm^2 \cdot sec$)
1 - G	1.2×10^9
2 - G	(not measured)
3 - G	3.3×10^9
4 - G	3.0×10^9
5 - G	(not measured)
6 - G	3.5×10^9
7 - R	1.2×10^9
8 - T	7.4×10^9
9 - C	1.5×10^9

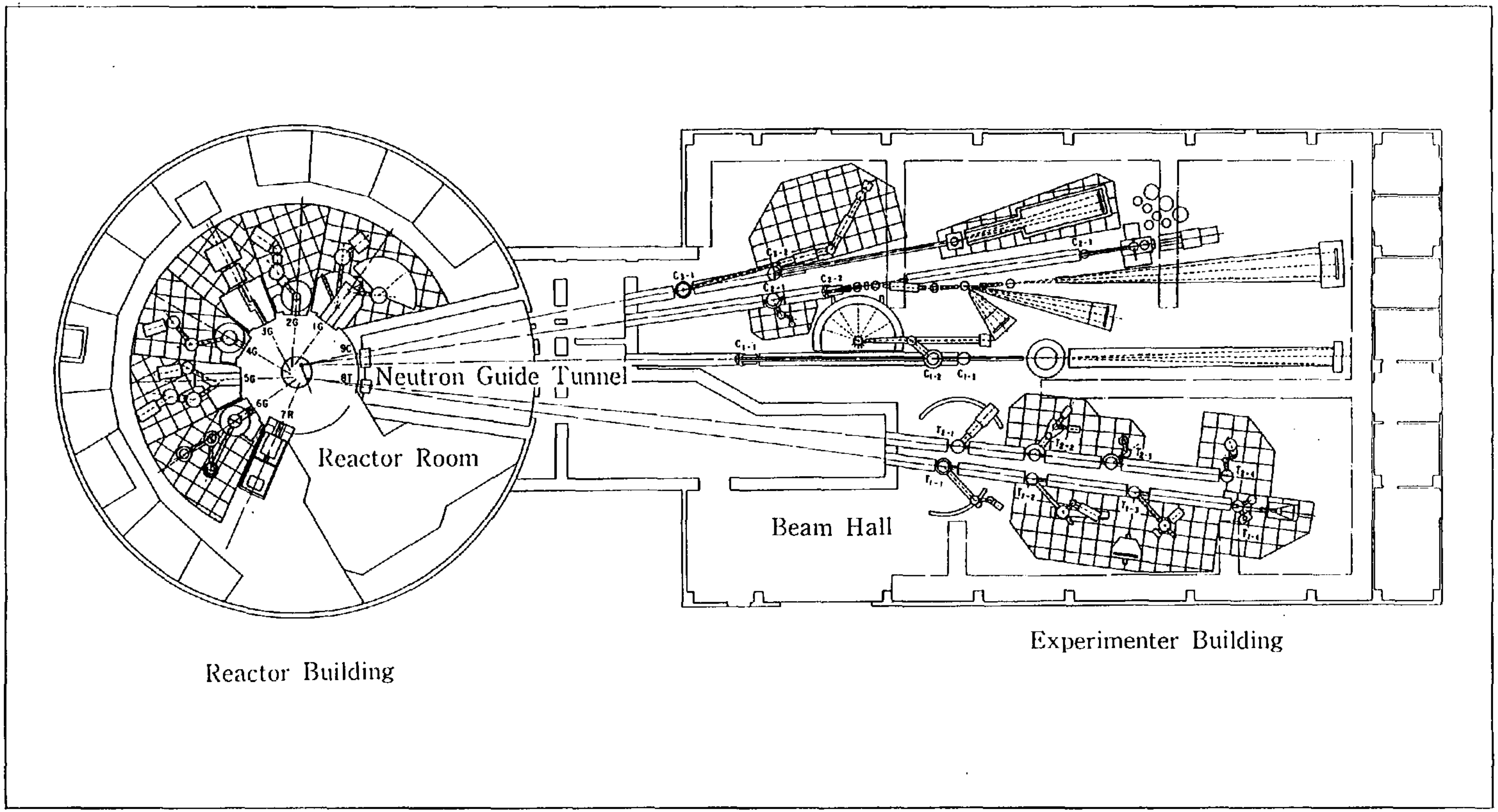


Figure 3 Beam Experimental Facilities in JRR-3M

Table 5 Current Assignment of the Neutron Beam Experimental Facilities in JRR-3M

BEAM TUBE NAME OR BEAM PORT NAME	INSTRUMENTS	LOCATION
1G	HRPD : High Resolution Powder Diffractometer	Reactor Building
2G	TAS-1 : Triple Axis Spectrometer	
3G	PNO : Neutron Topography and Precise Neutron Optics	
4G	TAS-4G : Triple Axis Spectrometer	
5G	TAS-5G : Triple Axis Spectrometer	
6G	TOPAN : Triple Axis Spectrometer	
7R	TNRF : Thermal Neutron Radiography	
T1-1	TAS-T1 : High Resolution Triple-Axis Spectrometer	Beam Hall (for thermal neutron)
T1-2	KSD : KINKEN Single Crystal Diffractometer	
T1-3	KPD : KINKEN Powder Diffractometer	
T1-4	PGAT : Prompt Gamma-ray Analysis	
	NDC : Neutron Diffraction Camera	
T2-1	DIVE : Diffraction Image Visualization Equipment	
T2-2	NDIO : Neutron Diffractometer Interfermetry and Optics	
T2-3	—— not assigned yet ——	
T2-4	TAS-2 : Triple Axis Spectrometer	
C1-1	TAS-C1 : High Energy Resolution Triple-Axis Spectrometer	Beam Hall (for cold neutron)
C1-2	SANS-U : Two Dimensional Small Angle Neutron Scattering	
C1-3	ULSANS : Ultra Low-Angle Small Angle Neutron Scattering	
C2-1	LTAS : Low-Energy Triple Axis Spectrometer	
C2-2	NSM : Neutron Spectral Modulation Instrument	
C2-3	CNRF : Cold Neutron Radiography	
	PGAC : Prompt Gamma-ray Analysis	
	UCN : Ultra Cold Neutron Cryostat	
C3-1	NSE : Neutron Spin Echo Spectrometer	
C3-2	SANS : Small Angle Neutron Scattering	

CURRENT STATUS OF RESEARCH AND TEST REACTORS IN JAERI

S. MATSUURA AND E. SHIRAI

Tokai Research Establishment
Japan Atomic Energy Research Institute

Research and Test Reactors in JAERI

Name	Type and Enrichment			Max. Power	Max. Neutron Flux		Start-Up Date
					Thermal	Fast	
JRR-2	*D_2O (CP-5)*	*U-Al* *UAlx-Al*	*93%* *45%*	*10MW* *10MW*	*2.0×10^{14}*	*6.0×10^{13}*	*1960.10* *1987.11*
JRR-3	*D_2O (tank)*	*MNU* *UO_2*	 *1.5%*	*10MW*			*1962.9*
JRR-3M	*H_2O (pool)*	*UAlx-Al*	*20%*	*20MW*	*3.0×10^{14}*	*2.0×10^{14}*	*1990.3*
JRR-4	*H_2O (pool)*	*U-Al*	*93%*	*3.5MW*	*3.5×10^{13}*	*8.7×10^{13}*	*1965.1*
JMTR	*H_2O (MTR)*	*U-Al* *UAlx-Al*	*93%* *45%*	*50MW* *50MW*	*4.0×10^{14}*	*4.0×10^{14}*	*1968.3* *1986.8*
NSRR	*H_2O (TRIGA)*	*U-ZrH*	*20%*	*300kW (Pulse; 23,000MW)*	*1.3×10^{12}*	*4.0×10^{12}*	*1975.6*

JRR-2

Type: Heavy Water Type (CP-5 Type)
Heavy Water Moderated and Cooled

Fuel: MTR Type, U-Al Dispersion,
45 wt% Enrichment

Thermal Power: 10MW

First Critical: October 1960

Operation Pattern:
24 hours/day, 12 days/cycle, 12 cycle/year

Utilization:
- Beam Experiments
- Activation Analysis
- Medical Irradiation (BNCT)
- Fuel and Material Irradiation
- R.I. Production

JRR-4

Type: Swimming Pool Type
Light Water Moderated and Cooled
Graphite Reflector

Fuel: MTR Type, U-Al Alloy,
93wt% Enrichment

Thermal Power: 3.5MW

First Critical: January 1965

Operation Pattern:
Daily Operation
6 hours/day, 4 days/week, 43 weeks/year

Utilization:
Radiation Shielding Experiments
Silicon Doping
Activation Analysis
R.I. Production
Training Operation

Future Plans:
Reducing the Enrichment to 20 wt%
Medical Irradiation (BNCT)

NSRR

Type: TRIGA ACPR Type
Light Water Cooled

Fuel: TRIGA Type, U-ZrH, 20wt% Enrichment

Thermal Power: 23,000MW(Pulse)
0.3MW(Steady)

First Critical: June 1975

Operation Pattern:
- Pulse Operation
- Steady State Operation

Utilization:
- IRA Experiments
  - LWR Fuels
  - Plate Type Silicide Fuels

Future Plans:
- Irradiated Mixed Oxide Fuel Test
- FBR Fuel Test

JMTR

Type: Tank Type
Light Water Moderated and Cooled

Fuel: MTR Type, U-Al Dispersion,
45wt% Enrichment

Thermal Power: 50MW

First Critical: April 1968

Operation Pattern:
24 hours/day, 24 days/cycle , 5 cycle/year

Utilization:
Fuel Irradiation
Material Irradiation
R.I. Production

Future Plan:
Reducing the Enrichment to 20 wt%
(U_3Si_2-Al Dispersion)

JRR-3M

Type: Swimming Pool Type
Light Water Moderated and Cooled
Heavy Water Reflector Tank

Fuel: MTR Type, U-Al Dispersion,
20 wt% Enrichment

Thermal Power: 20MW

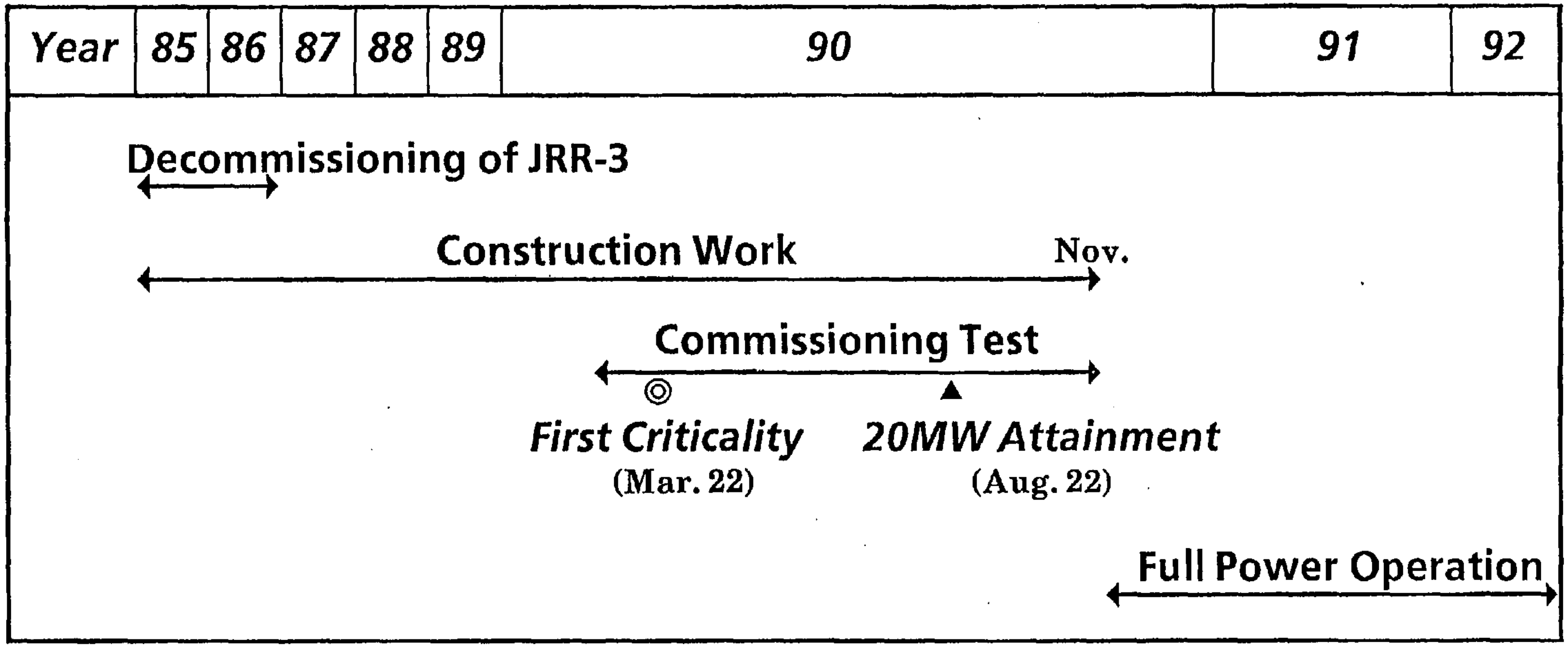
First Critical: March 1990 (after upgrading)

Operation Pattern:
24 hours/day, 7 days/week, 4 weeks/cycle ,
8 cycles/year

Utilization:
Beam Experiments
R.I. Production
Activation Analysis
Silicon Doping

Future Plan:
Fuel Conversion to Silicide Fuel

History of New JRR-3



Operation Cycle
4weeks ; continuous operation
1 week ; refueling, etc

Total Operation Time ; 7,000h
Total Integrated Power ; 130,000MWh
(Mar. 1992)

Major Specifications of the New JRR-3

Reactor Type	Low Enric[illegible] Light W[illegible] Moderated, [illegible]
Rated Power	20 MW
Size of Core	Approx. [illegible] high (with Be[illegible] Reflector).
Fuel	UAlx-Al D[illegible] MTR Plate Type 20% Enric[illegible] 26 Standar[illegible] and 6 Follower [illegible] Elements
Control Rod	6 Control Rods, Box Type Absorber, followed with Follower Fuel Element.
Reactor Pool	Swimming Pool Type 4.5m dia. [illegible] 8.5m deep [illegible]
Experimental Facilities	9 Horizont[illegible] 17 Vertica[illegible] 1 Cold N[illegible]

Isometric View of Reactor Pool

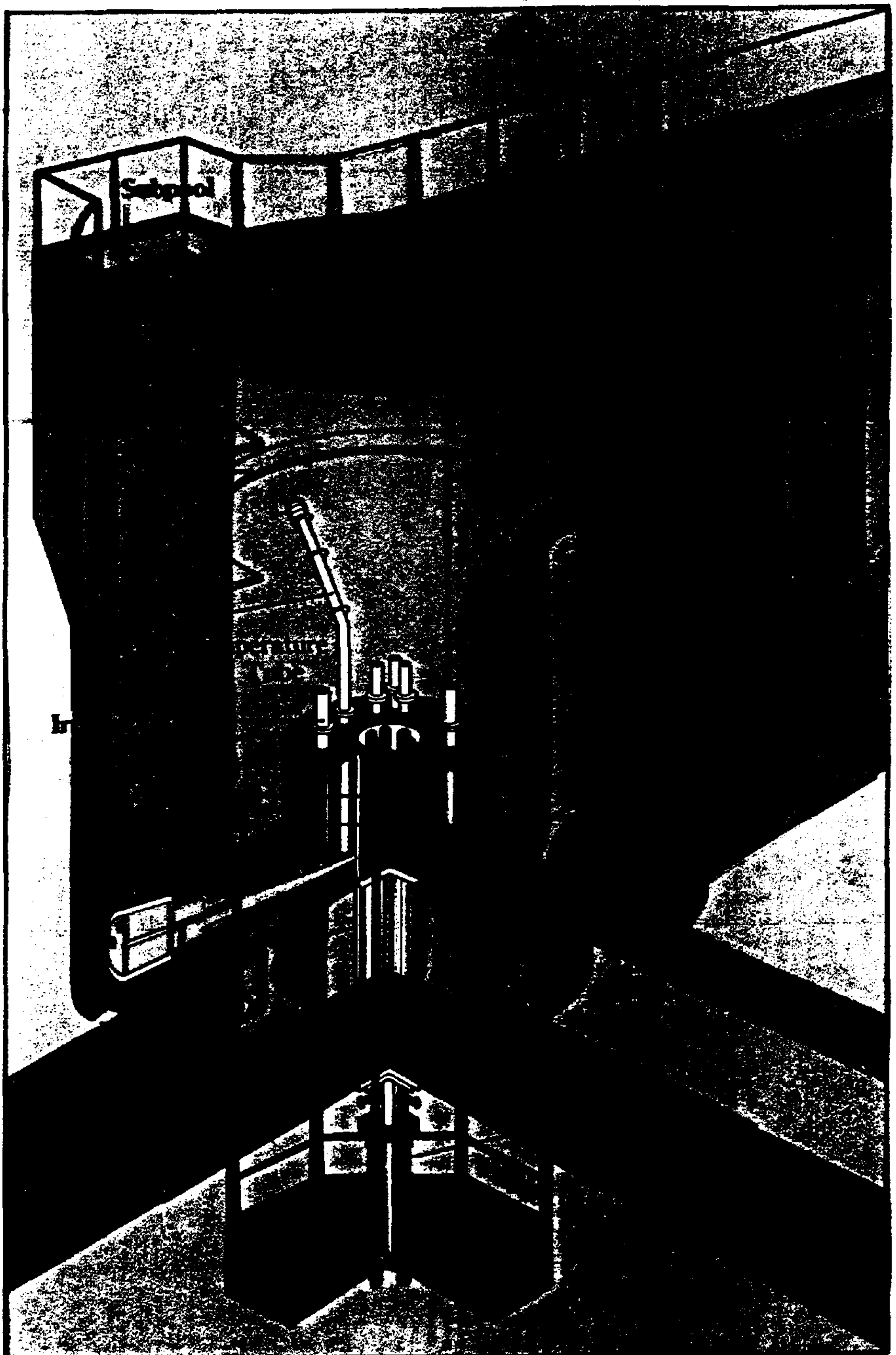
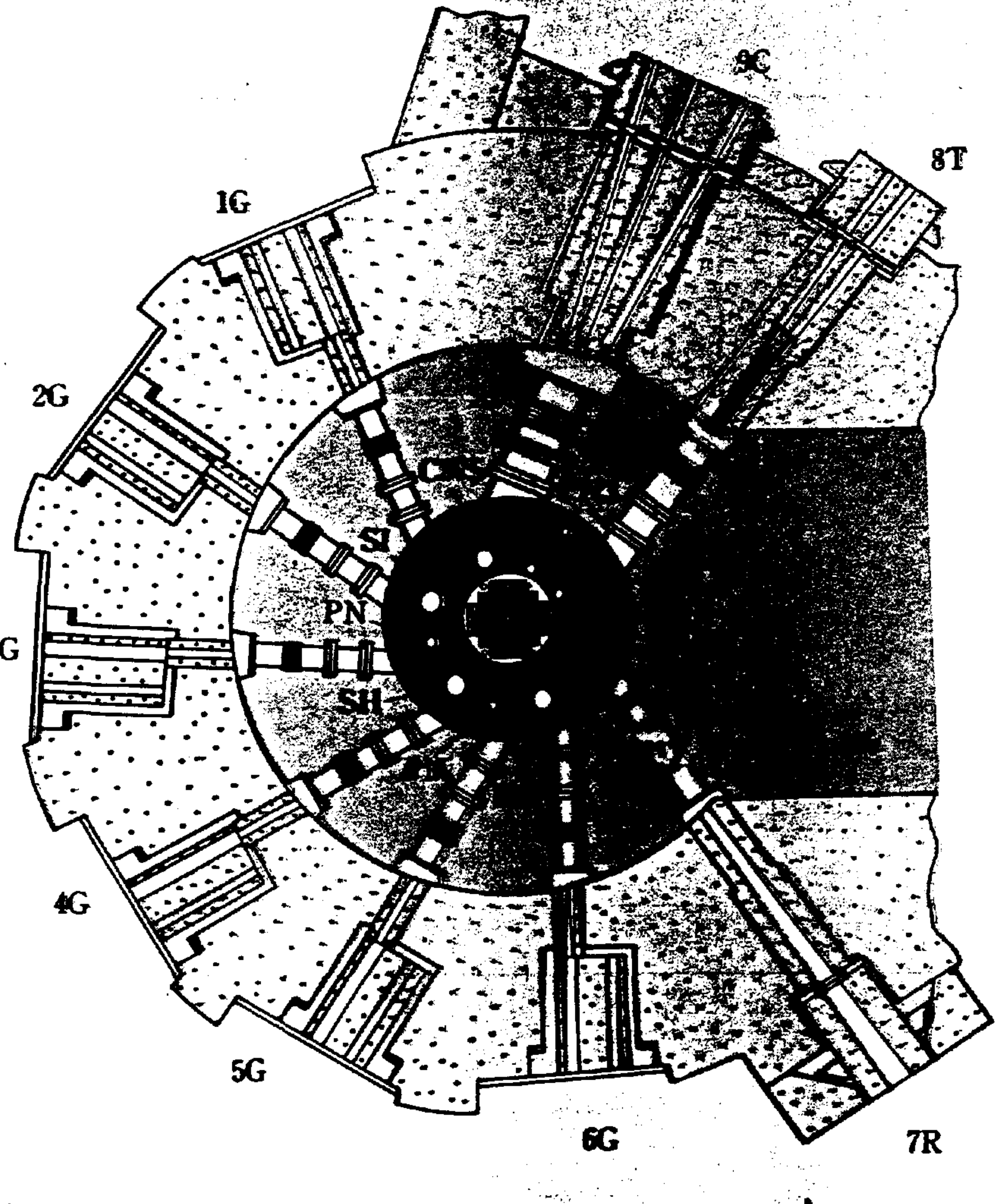


Table 2 Main Characteristics of JRR-3M

		Measured Value	Designed Value
Nuclear Properties			
Excess Reactivity	[%Δk/k]	15.9	16.3
Shutdown Margin (at One Rod Stuck)	[%Δk/k]	4.1	3.7
Control Rod Worth	[%Δk/k]	29.2	31.0
Moderator Temperature Coefficient	[%Δk/k/°C]	-1×10^{-2}	-2×10^{-2}
Moderator Void Coefficent (2% Void)	[%Δk/k/% void]	-0.2	-0.4
Average Thermal Neutron Flux in Core Region	[n/cm²/sec]	8×10^{13}	8×10^{13}
Neutron Flux at Thermal Neutron Guide Tube Outlet	[n/cm²/sec]	2×10^{8}	2×10^{8}
Neutron Flux at Cold Neutron Guide Tube Outlet	[n/cm²/sec]	3×10^{8}	3×10^{8}
CNS Cold Neutron Gain		8	≧5
Thermal-hydraulic parameters			
Flow Rate of Primary Cooling System	[m³/h]		2400
Core Outlet Temperatrure (Max.)	[°C]		35
Core Inlet Temperatrure (Max.)	[°C]		42
Perturbation Factor of Channel Flow Rate		1.12	1.13
Maximum Fuel Temperature	[°C]	99.6	101.0
Shielding Characteristics			
Dose Equivalent Rate on Upper Shielding	[μSv/h]	12	≦60
Dose Equivalent Rate in Beam Hall	[μSv/h]	0.2	≦6



(Vertical Irradiation Holes:
HR, PN, PN3, SI, DR, RG, VT-1, BR, SH
Horizontal Beam Tubes:
1G~6G, 7R, 8T, 9C)

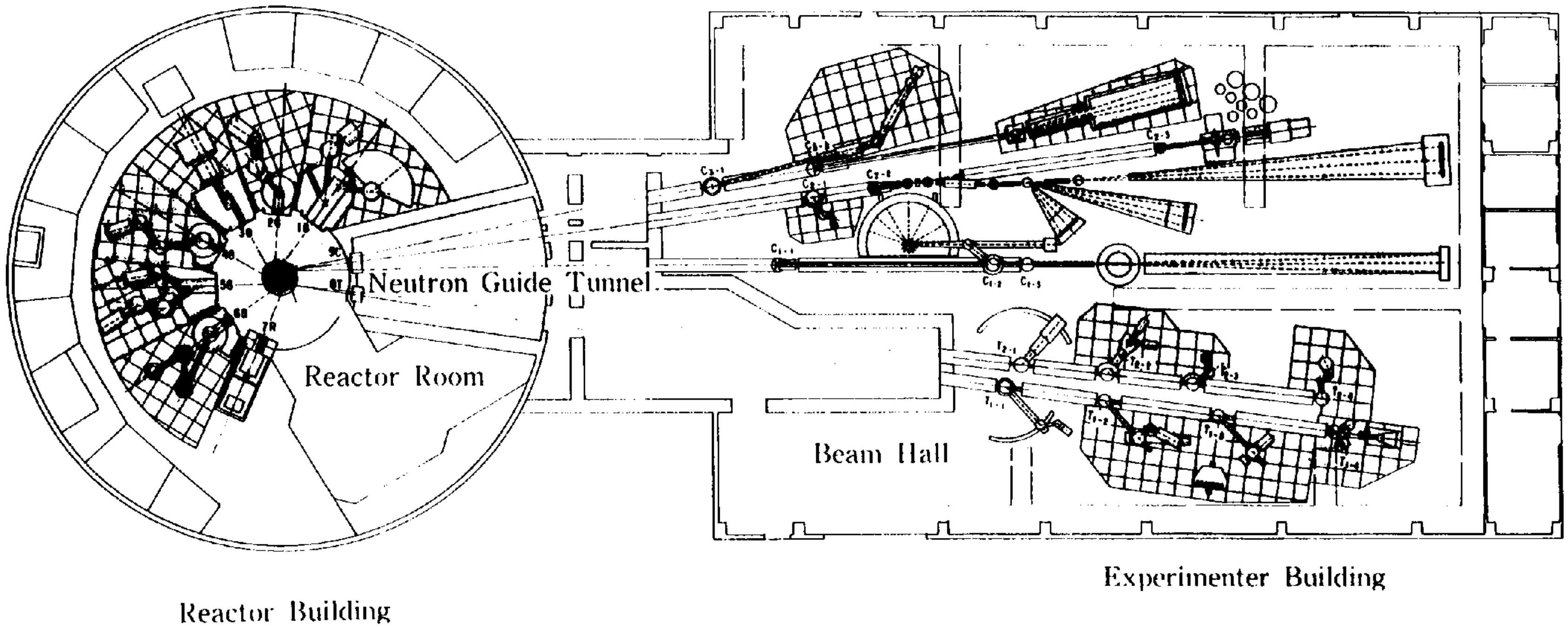
Arrangement of Experimental Holes

Irradiation Facilities in JRR-3M

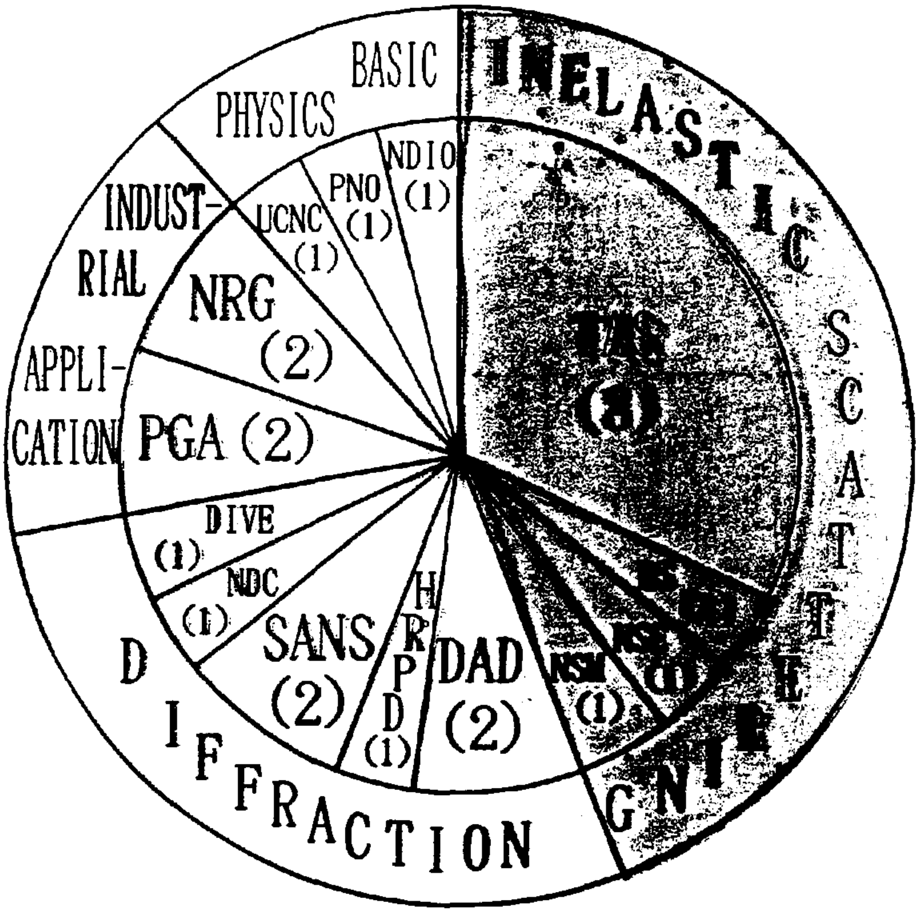
Name	Size and Number	Neutron Flux ($n/cm^2 \cdot sec$)		Application
		Thermal	Fast	
Hydraulic rabbit (HR)	ϕ37x2	1×10^{14}	1×10^{12}	RI Production
Pneumatic rabbit (PN)	ϕ37x2	6×10^{13}	1×10^{11}	RI Production
Activation Analysis (PN3)	ϕ20x1	2×10^{13}	4×10^{9}	Activation Analysis
Uniform irradiation (SI)	ϕ170x1	3×10^{13}	3×10^{11}	Material Irradiation Silicon Doping
Rotating irradiation (DR)	ϕ140x1	8×10^{13}	2×10^{11}	Large Material Irradiation
Capsule irradiation (RG,BR,VT-1,SH)	ϕ60x5 ϕ45x4 ϕ100x1	3×10^{14} 2×10^{14} 8×10^{13}	2×10^{14} 5×10^{13} 2×10^{11}	Exposure Test RI Production

Table 4 Measured Thermal Neutron Flux at the End of Beam Tubes at clean core

BEAM TUBE NAME	Thermal Neutron Flux ($n/cm^2 \cdot sec$)
1 - G	1.2×10^9
2 - G	(not measured)
3 - G	3.3×10^9
4 - G	3.0×10^9
5 - G	(not measured)
6 - G	3.5×10^9
7 - R	1.2×10^9
8 - T	7.4×10^9
9 - C	1.5×10^9



Beam Experimental Facilities



TAS :triple-axis spectrometer
BS :back scattering
NSE :neutron spin echo
NSM :neutron spectral modulation
DAD :double-axis diffractometer
HRPD:high-resolution powder diffractometer
SANS:small-angle neutron scattering
NDC :neutron diffraction camera
DIVE:diffraction image visualization equipment
PGA :prompt gamma-ray analysis
NRG :neutron radiography
UCNC:ultra-cold neutron cryostat
PNO :precise neutron optics
NDIO:neutron diffractometer interferometry and optics

Classification of Beam Experimental Facilities

Current assignment of the Beam Experimental Facilities in the Reactor Room

Beam	Instrument Name	
1G	HRPD	high-resolution powder diffractometer
2G	TAS-1	triple-axis spectrometer
3G	PNO	neutron topography and precise neutron optics
4G	TAS-4G	triple-axis spectrometer
5G	TAS-5G	triple-axis spectrometer
6G	TOPAN	triple-axis spectrometer
7R	TNRF	thermal neutron radiography facility

Current assignment of the Beam Experimental Facilities in the Beam Hall (Cold neutron)

Beam	Instrument Name	
C1-1	TAS-C1	high energy resolution triple-axis spectrometer
C1-2	SANS-U	two-dimensional small-angle neutron scattering
C1-3	ULSANS	ultra low-angle small-angle neutron scattering
C2-1	LTAS	low-energy triple-axis spectrometers
C2-2	NSM	neutron spectral modulation instrument
C2-3	CNRF	cold neutron radiography
	PGA	prompt gamma-ray analysis
	UCNC	ultra cold neutron cryostat
C3-1	NSE	neutron spin echo spectrometer
C3-2	SANS	small-angle neutron scattering

Current assignment of the Beam Experimental Facilities in the Beam Hall (Thermal neutron)

Beam	Instrument Name	
T1-1	TAS-T1	high-resolution triple-axis spectrometer
T1-2	KSD	KINKEN single-crystal diffractometer
T1-3	KPD	KINKEN powder diffractometer
T1-4	PGA	prompt gamma-ray analyzing system
	NDC	neutron diffraction camera
T2-1	DIVE	diffraction image visualization equipment
T2-2	NDIO	neutron diffractometer interferometry and optics
T2-3	— not assigned yet —	
T2-4	TAS-2	triple-axis spectrometer

PLANNING A NEW RESEARCH REACTOR FOR AECL: THE MAPLE-MTR CONCEPT

by

A.G. Lee, R.F. Lidstone and J.V. Donnelly

AECL RESEARCH

1. INTRODUCTION

AECL Research is assessing its needs and options for future irradiation research facilities. A planning team has been assembled to identify the irradiation requirements for AECL's research programs and compile options for satisfying the irradiation requirements. The planning team is formulating a set of criteria to evaluate the options and will recommend a plan for developing an appropriate research facility. Developing the MAPLE Materials Test Reactor (MAPLE-MTR) concept to satisfy AECL's irradiation requirements is one option under consideration by the planning team.

AECL is undertaking this planning phase because the NRU reactor is 35 years old and many components are nearing the end of their design life. This reactor has been a versatile facility for proof testing CANDU components and fuel designs because the CANDU irradiation environment was simulated quite well. However, the CANDU design has matured and the irradiation requirements have changed. Future research programs will emphasize testing CANDU components near or beyond their design limits. To provide these irradiation conditions, the NRU reactor needs to be upgraded. Upgrading and refurbishing the NRU reactor is being considered, but the potentially large costs and regulatory uncertainties make this option very challenging.

AECL is also developing the MAPLE-MTR concept as a potential replacement for the NRU reactor. The MAPLE-MTR concept starts from the recent MAPLE-X10 design [1] and licensing [2, 3] experience and adapts this technology to satisfy the primary irradiation requirements of AECL's research programs. This approach should enable AECL to minimize the need for major advances in nuclear technology (e.g., fuel design, heat transfer).

The preliminary considerations for developing the MAPLE-MTR concept are presented in the following sections of this report. A summary of AECL's research programs is presented along with their irradiation requirements. This is followed by a description of safety criteria that need to be taken into account in developing the MAPLE-MTR concept and a brief description of MAPLE technology. This report concludes by describing how the irradiation requirements are interpreted in terms of experiment facilities and how the MAPLE-X10 design has been adapted for the MAPLE-MTR concept.

2. RESEARCH REACTOR IRRADIATION REQUIREMENTS

A reactor-based irradiation facility is required by the CANDU reactor support, condensed matter science and radioisotope production programs. Several other programs, such as neutron radiography and fusion breeder blanket fuel development also require a source of neutrons; however, their requirements are captured by the major programs.

All of the programs share a common requirement for a high availability factor for the reactor. A goal of 90% availability has been set.

2.1 REQUIREMENTS FOR CANDU SUPPORT

The design of the CANDU reactor is supported by three programs: fuel channel materials research, fuel development and reactor safety research. Fuel development and reactor safety research have largely the same irradiation requirements.

2.1.1 Fuel Development and Reactor Safety Research

The CANDU fuel bundle consists of relatively short (0.5 m) lengths of fuel elements that are held together by thin end plates. The fuel development program supports this design through irradiation testing, which includes investigating the interactions between reactor coolant, reactor physics, materials properties, heat transfer, and fabrication variables. In addition to studying the performance of individual aspects (i.e., fuel elements) of the fuel bundle, the overall performance is studied under conditions that closely approximate the CANDU environment.

The reactor safety program involves establishing the limits of performance, and the behaviour and consequences if the fuel is exposed to extreme conditions. This program requires the capability to operate fuel beyond the performance limits in a controlled environment where the consequences of fuel failure can be managed safely and studied to provide data for normal plant operation.

2.1.1.1 Fuel Element Test Requirements

For fuel element tests, the requirements are the ability to irradiate from one to seven elements to produce peak linear power ratings of about 70 kW/m. The fuel test loop must be able to operate at representative CANDU temperature (e.g., 300°C), pressure (e.g., 10 MPa), flow, and coolant chemistry conditions. H_2O will be the normal coolant used in these small diameter loops, but other coolants, such as D_2O, may also be required.

The reactor safety research program also requires the use of small-diameter fuel test loops with the above mentioned irradiation conditions. The additional requirement is the ability to induce rapid loss of coolant and loss of flow transients in a selected test loop.

2.1.1.2 Fuel Bundle Test Requirements

For fuel bundle tests, the irradiation requirements include 0.5-m length of relatively uniform flux and to produce peak linear power ratings of 70 kW/m in some elements. These fuel test loops must also operate at representative CANDU temperature, pressure, flow and coolant chemistry conditions. The normal coolant used in these loops will be H_2O.

Although the current reactor safety research program relies on use of small multi-element assemblies, future programs may require off-normal tests to be performed on full-diameter CANDU bundles. Since the fuel channels in a CANDU reactor are oriented horizontally, the capability of testing multi-element assemblies and CANDU bundles in a horizontal orientation is also desirable.

2.1.2 Fuel Channel Materials Requirements

The CANDU reactor design features an array of horizontal fuel channels, each containing a series of short fuel bundles. The fuel channel (i.e., pressure tube and calandria tube combination) is the main pressure retaining boundary and separates the high-temperature coolant from the low-temperature moderator. The fuel channel materials program involves gaining an understanding of the ongoing and end-of-life behaviour of the pressure tubes and calandria tubes. This understanding requires exposing fuel channel materials to conditions that both match and significantly exceed the fast-neutron fluxes of CANDU reactors.

The fuel channel materials program has several types of irradiation requirements. To define the life of fuel channel components, small (~30-mm diameter) samples of fuel channel materials need to be exposed to high fast ($E > 1.0$ MeV) neutron fluxes (i.e., 2 to 3 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$) and need to accumulate annual fast-neutron fluences of 3 to 4 x 10^{25} $n \cdot m^{-2}$. These material samples need to be maintained at typical CANDU temperatures.

The fuel channel materials program also irradiates sections of full diameter pressure tubes for proof testing of new materials and for gaining further information on the life of the pressure tubes. These need to be maintained at typical CANDU temperatures, pressures and fast-neutron fluxes (i.e., 0.2 to 0.3 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$).

Samples used in corrosion studies need to be maintained at typical CANDU temperatures and pressures in a controlled D_2O water-chemistry environment and a typical CANDU irradiation environment (i.e., fast-neutron fluxes of 0.6 to 0.7 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$).

2.2 NEUTRON BEAM REQUIREMENTS

The condensed matter science program at AECL involves performing neutron scattering studies such as investigations of residual stress and texture in components. In addition, active programs in neutron radiography and the application of neutron scattering to industrial research are pursued. The users of neutron beams, for neutron scattering and neutron radiography, acknowledge that the primary uses for the MAPLE-MTR will be for fuel and materials irradiations. Efforts will be made to match the beam-tube capabilities of the NRU reactor in any new reactor by providing six to eight beam tubes with ~2 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$ for the thermal-neutron fluxes. Provisions will be made to add a cold neutron source at some future date if it is not initially included.

2.3 RADIOISOTOPE PRODUCTION REQUIREMENTS

AECL considers it prudent to have a facility available to produce ^{99}Mo as a fission product and ^{125}I during the periods of time when the MAPLE-X10 reactor is out of service. Production of ^{99}Mo requires at least matching the power density (i.e., ~160 kW/L) in the MAPLE-X10 ^{99}Mo target assemblies. For ^{125}I production, a thermal-neutron flux of about $1 \times 10^{18}\ n \cdot m^{-2} \cdot s^{-1}$ is needed.

3. SAFETY PRINCIPLES AND CRITERIA

The successful development of the MAPLE-MTR concept requires not only satisfying the irradiation requirements of the research programs but also satisfying the safety and licensing requirements of the regulatory authorities. To ensure that safety and licensing considerations are adequately addressed, a set of safety principles and criteria are being prepared. These safety principles and criteria will provide the safety basis for the design of the MAPLE-MTR and for assessing the design.

For the design of the neutron source, it is proposed to minimize the consequences of abnormal or accident conditions by using inherent safety features such as:

- negative fuel temperature coefficient,
- negative coolant temperature coefficient,
- negative coolant void coefficient,
- negative D_2O temperature coefficient, and
- negative D_2O void coefficient.

Other proposed safety principles and criteria for the reactor design include:

- the ability to shut down the reactor and maintain it in a subcritical state for all operational states and accident conditions;
- the withdrawal of any single control rod or largest worth experimental assembly in the shutdown reactor will not make the reactor core critical;
- the inadvertent dropping of a control rod will shut down the reactor;
- the maximum permissible design limits for fuel elements, reactivity control mechanisms, experimental assemblies, etc., are not exceeded;
- significant fuel damage is prevented during severe accident conditions; and
- the maximum positive reactivity worth of an experimental assembly will be 5 mk.

4. MAPLE REACTOR TECHNOLOGY

The basis for developing the MAPLE-MTR concept is the technology developed to support the design, licensing, construction and commissioning of the MAPLE-X10 reactor. This technology consists of

- a design verification and qualification program to test components used in the MAPLE-X10 reactor,
- a thermalhydraulics experiment program to develop heat transfer correlations and to validate the thermalhydraulics codes, and
- the computer codes used to model and analyze the characteristics of the reactor.

4.1 MAPLE-X10 CONCEPT

The MAPLE-X10 reactor has 19 core sites surrounded by a D_2O reflector. Each core site can accommodate a hexagonal 36-element fuel assembly, a cylindrical 18-element fuel assembly or a target assembly. The reference core configuration includes 10 hexagonal 36-element fuel assemblies and up to nine cylindrical 18-element fuel assemblies. To produce ^{99}Mo, the 18-element fuel assemblies are replaced by ^{99}Mo target assemblies. The MAPLE-X10 reactor has an average power density of 159 kW/L at a power output of 10 MW.

The MAPLE-X10 D_2O reflector is an annular zirconium-alloy vessel, filled with D_2O. There are 18 small diameter vertical sites for other radioisotope targets and four large diameter vertical sites. Further details about the MAPLE-X10 design can be found in References 1 and 4.

4.2 FUEL

The reference fuel is the low-enrichment (about 19.7 wt% ^{235}U in total uranium) U_3Si-Al that AECL developed for use in the NRU and MAPLE-X10 reactors. The fuel meat is composed of U_3Si particles dispersed in an aluminum matrix and coextrusion clad with aluminum to form finned rods. This fuel has demonstrated excellent performance with up to 93 percent burnup of initial fissile material and with linear power ratings up to ~120 kW/m without any defects. Further information about the fuel can be found in Reference 4.

4.3 DESIGN VERIFICATION AND QUALIFICATION PROGRAM

The MAPLE-X10 project has undertaken an extensive test program to verify and qualify the design of systems and components. A full-scale hydraulic test rig [5] has been constructed utilizing reactor components and prototype assemblies to demonstrate that the safety and reliability requirements of the safety-related, underlying concepts are met. This test program has led to many design improvements and has provided data for validating computer codes used in support of the design.

4.4 THERMALHYDRAULICS EXPERIMENT PROGRAM

The thermalhydraulics experiment program includes heat transfer experiments to derive correlations for finned fuel operating in low temperature, low pressure water and to validate the computer codes used to predict MAPLE reactor behaviour. The experiments use fuel element simulators consisting of heaters sheathed in aluminum. The single fuel-element experiments have yielded data to develop forced convection and nucleate boiling heat transfer correlations, boiling curve criteria correlations and void fraction correlations. A heat transfer software package has been developed and implemented in the computer codes used to determine design parameters for the primary cooling system and thermal safety margins during normal and transient conditions. Further tests involving multi-element assemblies are in progress. These multi-element assembly tests will be used to validate the computer codes.

4.5 COMPUTER CODES USED TO MODEL MAPLE REACTORS

Several computer codes are being used or developed to analyze the normal and transient behaviour of MAPLE reactors. These codes are briefly described in the following sections.

4.5.1 Thermalhydraulics Modelling

The CATHENA [6] computer code is used to simulate steady-state and transient reactor behaviour. This code uses a one-dimensional two-fluid thermalhydraulics model to describe steam-water/noncondensable gas flow in a pipe. A point-kinetics model is available in CATHENA to simulate transient reactor behaviour.

The CATHENA computer code is used to determine the basic thermalhydraulic parameters (i.e., inlet and outlet temperatures, inlet and outlet pressures and coolant flows) and the thermalhydraulic design limits. The design limits are the sheath temperature at onset of nucleate boiling (ONB), onset of significant void (OSV) and critical heat flux (CHF). For transient analyses, the CATHENA code is used to determine the variation of reactor power at which the hottest point on the hottest fuel element would reach ONB, OSV and CHF as a function of core inlet temperature.

4.5.2 Physics Modelling

Several computer codes, WIMS-AECL [7], 3DDT [8], MCNP [9], are used to investigate the static behaviour of the MAPLE-X10 reactor and to determine the neutronic performance of the reactor. A two-dimensional neutron kinetic code, TANK [10], has been developed to analyze reactivity insertion transients.

4.5.2.1 WIMS-AECL

The Winfrith Improved Multigroup Scheme (WIMS [11]) is a multigroup neutron transport code that solves the neutron transport equation using either discrete ordinates methods or collision probability methods to prepare cell-averaged macroscopic reaction cross section data and to perform fuel burnup calculations. The AECL version of this code, WIMS-AECL, has been extensively modified from the original code and utilizes an 89-group cross-section data library which has been compiled from the ENDF/B-V data library. Each of the cells in the 3DDT model are analyzed with the WIMS-AECL code to obtain the macroscopic reaction cross section data that are flux and volume weighted for the various materials.

4.5.2.2 3DDT

AECL uses the 3DDT three-dimensional neutron diffusion code to perform reactor calculations. A three-dimensional model is created by representing the reactor as a series of rectangular cells in the XY-plane and superimposing many XY-planes to represent the Z-direction. The 3DDT calculations are used to determine:

- neutron flux distributions in five energy groups,
- power distributions in the fuel assemblies,
- fuel temperature reactivity coefficient,
- water temperature reactivity coefficient,
- void reactivity coefficient,
- fuel burnup and
- reactivity worths of the fuel assemblies, hafnium absorbers and radioisotope targets.

4.5.2.3 MCNP

MCNP is a generalized geometry Monte Carlo code used to perform neutron and photon transport calculations. The generalized geometry capabilities make it possible to construct very detailed and realistic reactor models. The MCNP code is used to analyze specialized aspects of the MAPLE reactors which are not well-suited to analysis with WIMS-AECL and 3DDT, e.g., reactivity worth of D_2O dump. As well, this code is used to independently verify some of the results obtained from the WIMS-AECL/3DDT computations by determining the same characteristics with a different calculational method. The MCNP code is used to determine:

- heat deposition in the structural materials (e.g., flow tubes, inner wall of the reflector tank, absorbers, etc.),
- power distributions in the individual fuel rods,
- reactivity worths of the reactivity-control devices and
- neutron flux distributions.

4.5.2.4 TANK

TANK is a two-dimensional, two neutron-energy-group, space-time reactor kinetics code developed to simulate reactivity insertion transients in MAPLE reactors. TANK uses the heat transfer package developed from the single element heat transfer experiments to determine the cladding-coolant heat transfer coefficients for sub-cooled and saturated boiling. Work is in progress to couple TANK to CATHENA to form a code package that can simulate reactor transients where spatial effects are important to the neutron kinetics.

5. MAPLE-MTR CONCEPT

To develop the MAPLE-MTR concept, AECL has started from the underlying MAPLE-X10 design concepts and adapted them to form a reactor configuration which provides many of the irradiation requirements discussed in Section 2. These irradiation requirements are summarized in Table 1.

TABLE 1

SUMMARY OF IRRADIATION CRITERIA

	DESIGN CRITERION
Fuel Development	
Location	D_2O
Power Rating	peak 70 kW/m
Thermal-neutron flux	~4 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$
Axial average to peak ratio	0.9
Fuel Channel Materials (Small sample)	
Location	Core
Fast-neutron flux	1.8-3 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$
Location	FN-site
Fast-neutron flux	0.6-0.7 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$
Fuel Channel Materials (Large sample)	
Location	Fuel test loop
Fast-neutron flux	0.2-0.3 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$
Beam Tubes	
Location	D_2O
Thermal-neutron flux	~2 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$
Thermal/fast-neutron ratio	~80
Backup Radioisotope Production (^{99}Mo)	
Location	Core
Average power density	250-300 kW/L
Backup Radioisotope Production (^{125}I)	
Location	D_2O
Thermal-neutron flux	~1 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$

5.1 MAPLE-MTR REFERENCE CORE POWER

An initial core power output of 15 MW has been chosen for the MAPLE-MTR concept. This power output is based on several considerations. At 15 MW, the average power density in the core will be ~250 kW/L. This is very similar to the power densities of the HFR-Petten (45 MW) and OSIRIS (70 MW) reactors, which provide fast-neutron fluxes of 2 to 3 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$ at power densities of about 210 and 280 kW/L, respectively. Since the MAPLE-MTR core is similar in composition to these reactors, similar fast-neutron fluxes can be expected. Preliminary physics calculations indicate the perturbed fast-neutron fluxes are ~1.4 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$ in realistically modelled samples. Also 15 MW is considered to be a modest increase in power over the MAPLE-X10 reactor and is not expected to result in linear power ratings beyond AECL's operating experience. The reference core power will be adjusted if the need for higher powers is identified.

IV.6. WATER CIRCUITS

MAIN PRIMARY COOLING CIRCUITS

PRIMARY CORE COOLING SYSTEM MAIN CHARACTERISTICS	
Fluid :	Demineralized water
Decay tank :	
number	1
volume	100 m³ (approximately)
Pump* :	
type	centrifugal
number	2
flow rate	1200 m³/h (each train)
Heat exchanger :	
type	tubular
number	2
exchange thermal power	12.5 MWth
flow rate, primary side	1200 m³/h (each train)
inlet temperature	49°C (primary side)
outlet temperature	40°C (primary side)
flow rate, secondary side	1800 m³/h (approx.)
maximum inlet temperature	32°C (secondary side)
outlet temperature	39°C (secondary side)

* Pumps are equipped with Flywheels designed to guarantee 85 % of Core cooling flowrate 7.5 seconds after loss of electrical power, but Reactor is automatically shut-down 3 seconds after loss of power.

AUXILIARY PRIMARY CIRCUITS

- PRIMARY WATER PURIFICATION / HOT LAYER

. 2 identical files 100 % (flowrate : 15 m^3/h)
Connections between both purification trains allow each component of a train to be backed up by the corresponding one of the other line.

. regenerative heat exchanger (normal operation) and electrical heater (start-up) : ~ 90 kW

. hot layer : height - 2.3 m - $\Delta t = +5°C$

- CLAD FAILURE DETECTION

. 2 pumps (flow rate : 4 m^3/h) backed-up powered.

- PRIMARY DRAINING CIRCUIT

One draining tank located in the basement, inside the containment, can store active primary water from one of the three pools.

- POOLS SKIMMING

Dusty water is sent either to liquid wasts tanks or, through appropriate filtration, to the draining tank.

- ADDITIONAL POOL COOLING SYSTEM

It is made of a single one heat exchanger located on the purification hot layer circuit. Heat removal is performed by connecting the secondary to the circuit city water outside Reactor building. This residual heat removing mean could be useful occuring a long duration loss of off-site electrical power stopping for more than a week normal ventilation and normal primary and secondary cooling systems.

IV.7. ELECTRICITY

- see : Single line diagram

  . Off site power : two separate lines (1.500 kVA)
  . two transformers (1250 kVA) for non backed-up grid;
  . two transformers (400 kVA) for backed-up power supply and uninterruptible electrical sources;
  . three uninterruptible electrical power units for reactor protection system (autonomy : 1 hour);
  . two more units for other consumers (experimental devices, data system...);
  . two stand-by diesel generators (400 kV) for keeping power on ventilation system, I&C and back-up uninterruptible power sources.

IV.8. INSTRUMENTATION AND CONTROL

- The main features of I&C systems are :

  . compliance with modern safety requirements for Reactor Protection System or safety related components (segregation of ways, one single failure criteria..;control room outside containment, emergency control panel...)

  . availability : * shut down logic in a 2/3 vote.
  * high reliability and first choice technology.

  . digital technology : is chosen when appropriate; especially for core nuclear measurement and reactor protection system.

  . ergonomy : Man-machine dialogue.
  * traditional methods are used for control rods and a synthetic mimic panel provides information on the general status of the facility.
  * screens and printers provide information on other operations.

- I&C systems include mainly :

  . the Reactor protection system : core nuclear measurement, safety related sensors, shut down system. All this equipment is of the highest safety level (1E safety class). There are 3 lines that are physically and geographically segregated as far as possible.

  . the control system for the reactor and its main auxiliary circuits. There are 2 redundant digital processing units.

  . the reactor data processing system which is designed to process in real time all data related to the reactor cycle, and experiments.

  . the health physics monitoring system.

  . intercoms means (TV, telephone...)

V. OPERATION

For one year's operation at the rated power of 25 MW :
10 cycles - 210 EFPD

FUEL :	20	Standard elements
	10	control elements
	10	irradiation elements
CONTROL RODS :	1	every 2 years
ELECTRICAL POWER :		1.500 kW at 25 MW or < 10^7 kW.h/year
WASTES :		- liquid : activity < 10^{-5} Ci/m³ (3.7.10⁶ Bq) = ~80 m³; activity > 10^{-5} Ci/m³ = ~ 1 m³
		- Mixed bed I.E.R : 2,000 to 2,500 litres
		- Solid : activity < 10^{-5} Ci/m³; incinerable : ~ 5 to 10 m³; other wastes : < 3 m³

REACTOR OPERATION STAFF : 39 to 59 operators (including 8 engineers)

- Head, safety, QA	6
- Operators . Shift agents	5 or 6 teams x 4
. Others	10
- Maintenance groups	20
- Health physics agents	5

SIRIUS 2 - DESIGN CRITERIA

REACTOR

* Fuel : MTR plate type (SILOE) 76.1 x 80 x 873 cm
  U_3Si_2/Al : 4.8 g/cm^3
  Enrichment : 19.75%
  At equilibrium : 33 elements (22 + 5 +6)

* Moderator/Coolant : Demineralized water

* Reflector : Beryllium blocks (+ heavy water tank *)

* Absorber : Hf
  Fork type control rods

* Cycle duration : 21 days

FLUXES

* In core positions : Fast flux (E> 100 kev) = 3.8 10^{14} n.cm^{-2}.s^{-1}

* In reentrant positions : Thermal flux (E< 0.625 kev) = 3.5 10^{14} n.cm^{-2}.s^{-1}

* 1st raw : Thermal flux = 3.10^{-14} n.cm^{-2}.s^{-1}

AVAILABILITY = 99% (MELUSINE = 99.3, SILOE = 98.7%)

LIFETIME = 30 Years

THERMAL HYDRAULICS

* Thermal power : 25 MWth (extensible to 30 MW*)

* Primary coolant flowrate : 2,400 m^3.h^{-1} (2 loops 50%)

* Average heat flux : 45 W.cm^{-2}

* Core inlet temperature : 40°C

* Core outlet temperature : 49°C

* Core pressure drop : 7.4 mWG (at nominal speed ~5.2 m.s^{-1})

* Cold source wet bulb temperature : 32°C

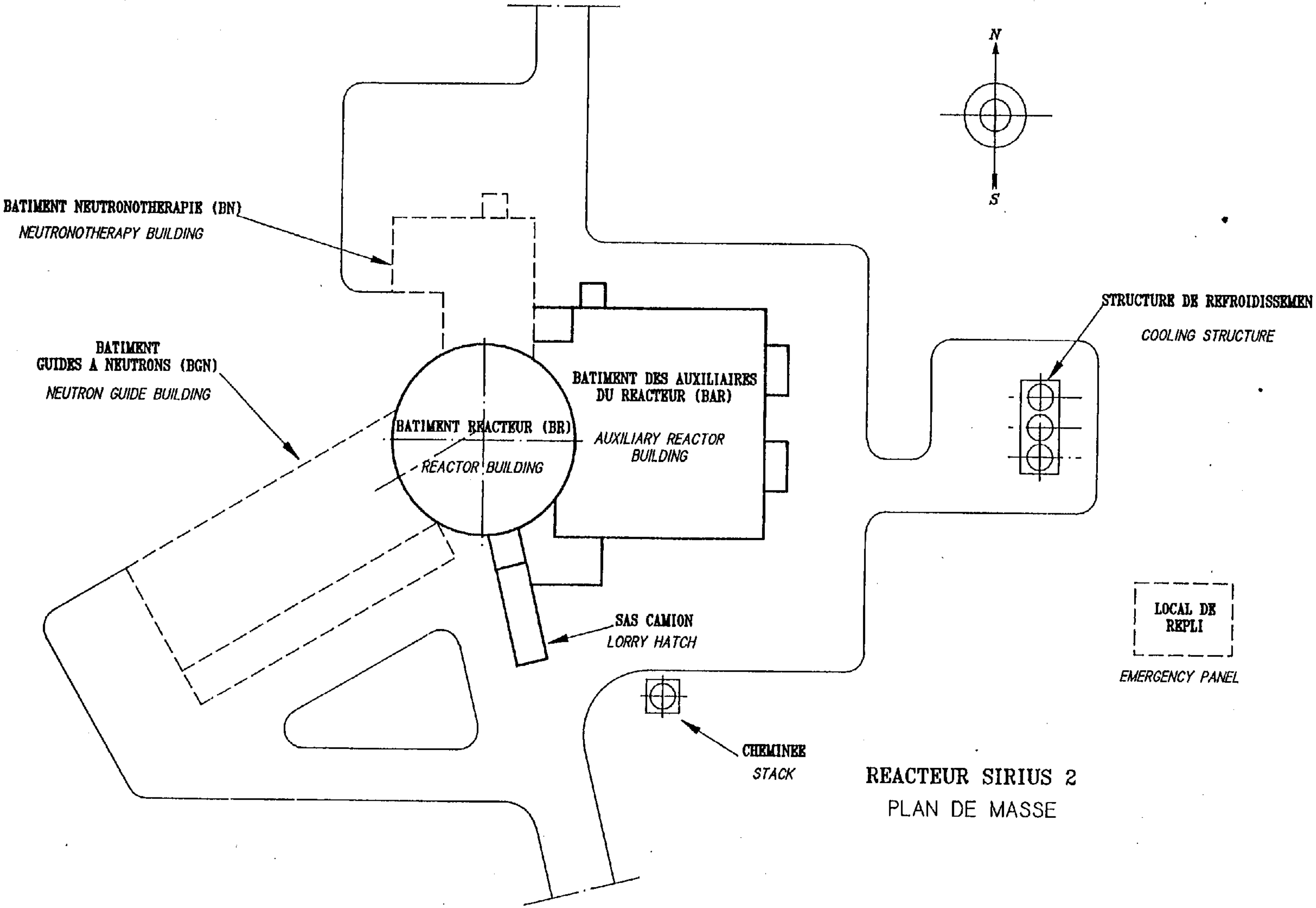
D.B.A.

* BORAX type accident with complete core meltdown under water

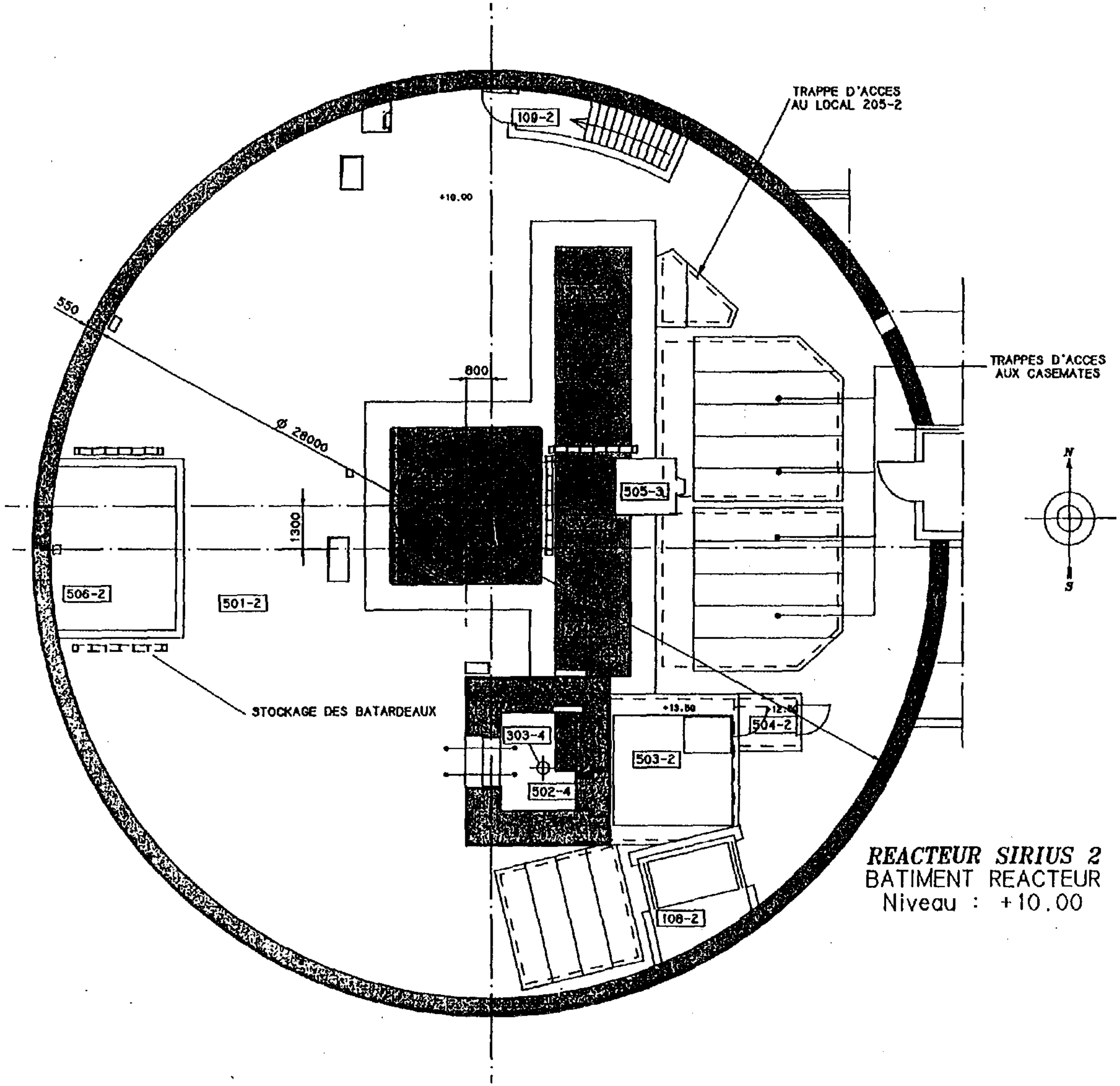
* Underwatering of Core by passive means

* The primary cooling system have no safety related functions

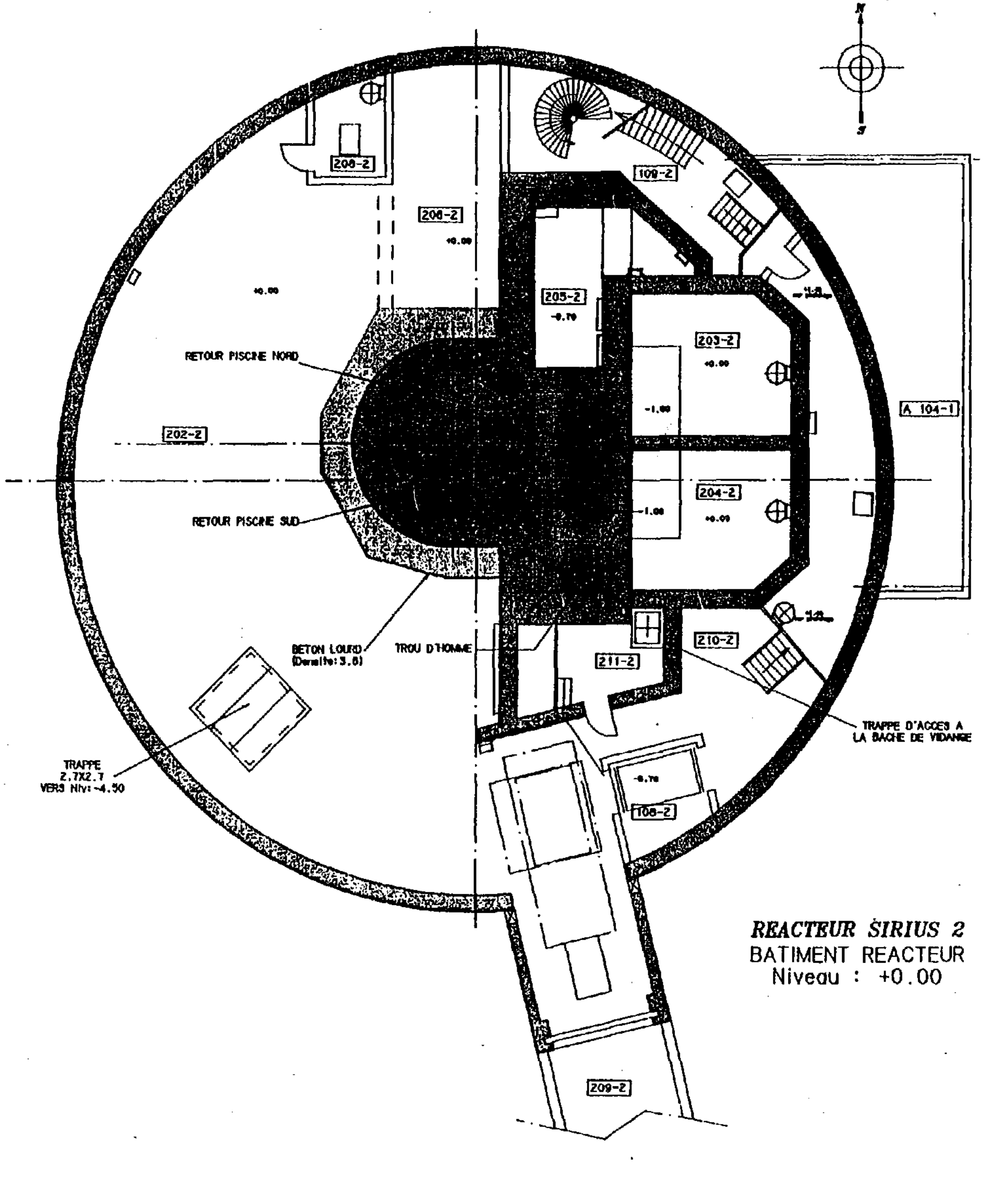
(*) Option



REACTEUR SIRIUS 2
PLAN DE MASSE



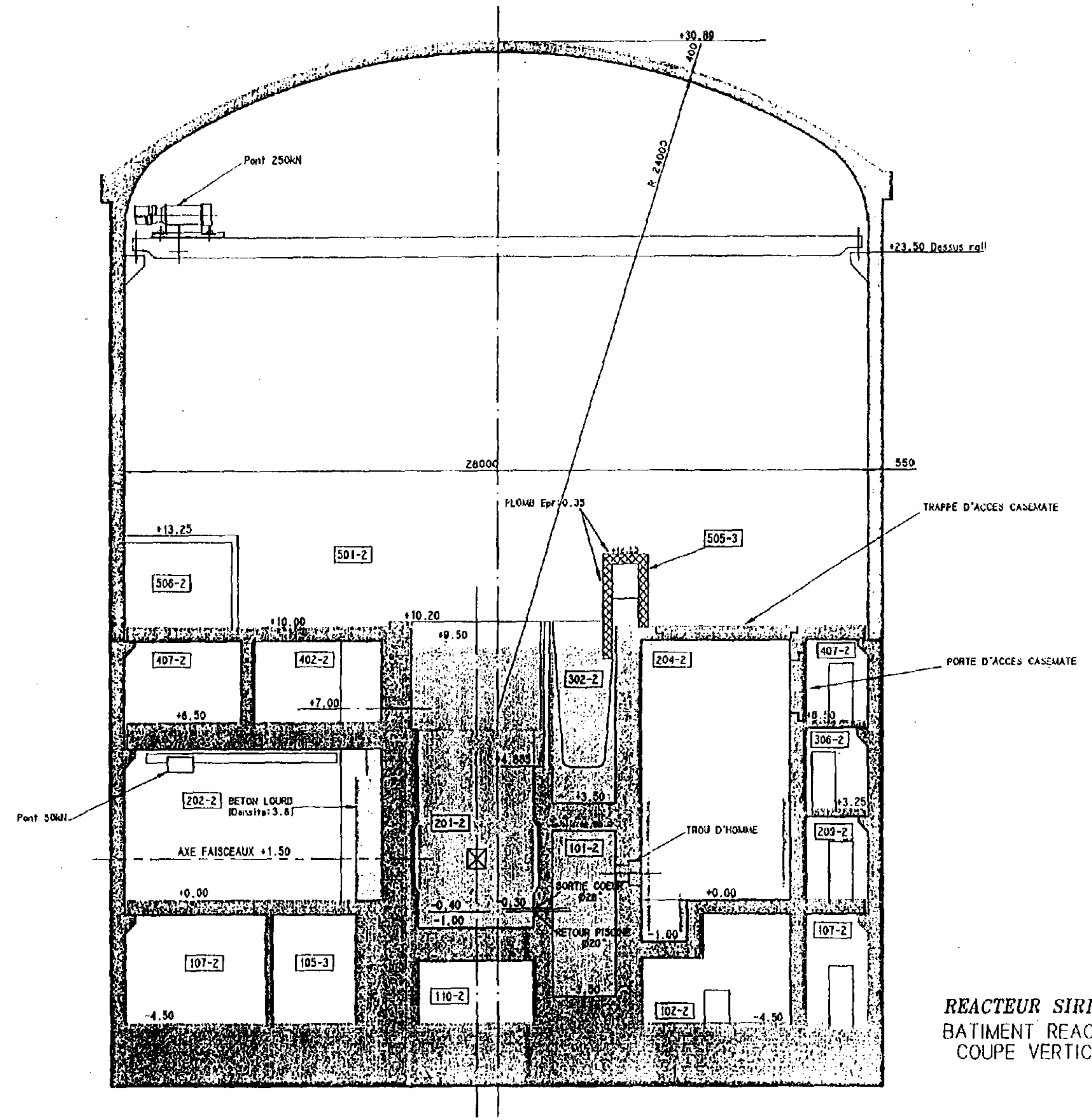
REACTEUR SIRIUS 2
BATIMENT REACTEUR
Niveau : +10.00



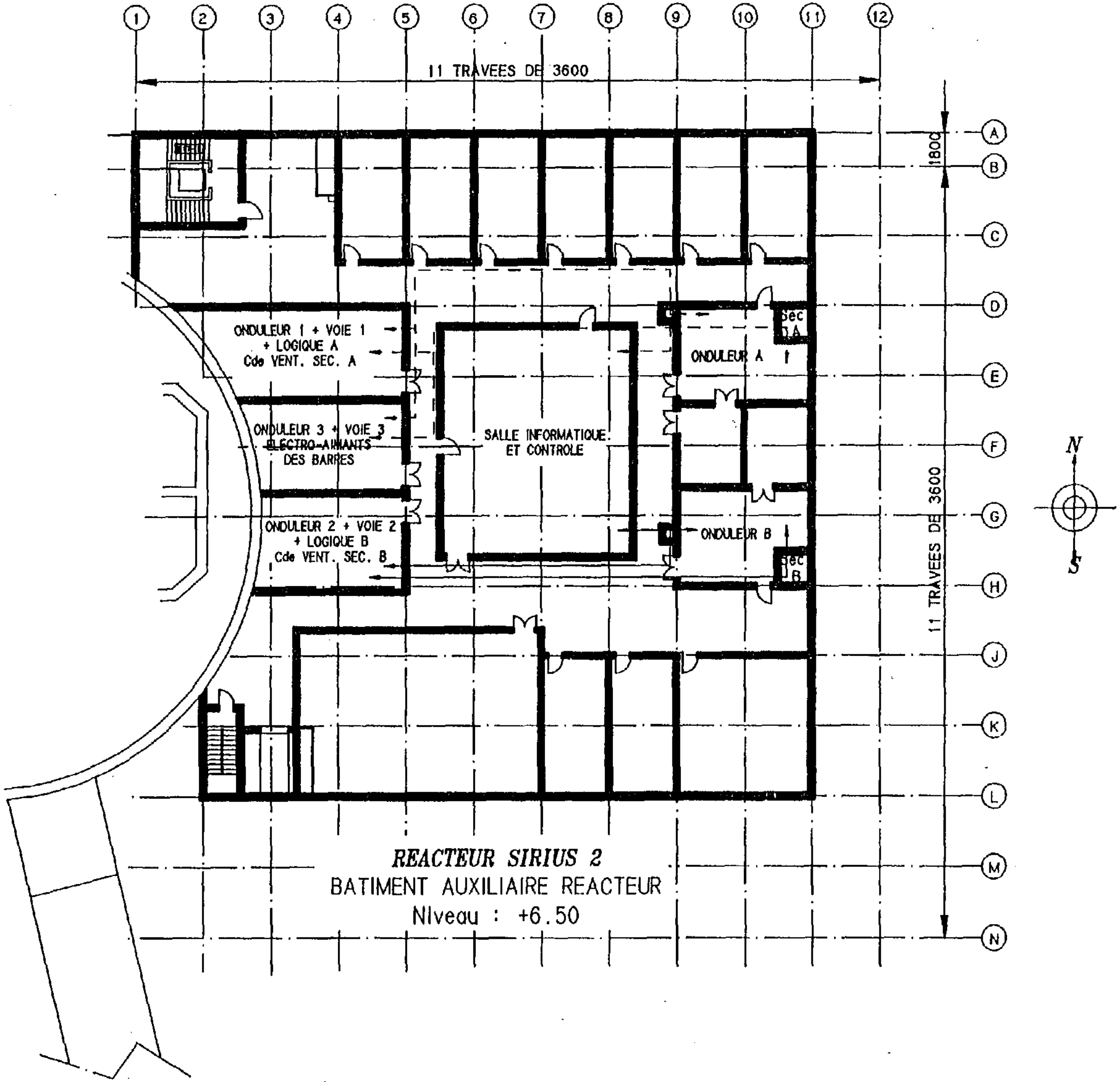
REACTEUR SIRIUS 2

BATIMENT REACTEUR

Niveau : +0.00



REACTEUR SIRIUS 2
BATIMENT REACTEUR
COUPE VERTICALE



REACTEUR SIRIUS 2

BATIMENT AUXILIAIRE REACTEUR

Niveau : +6.50

FIGURE 1

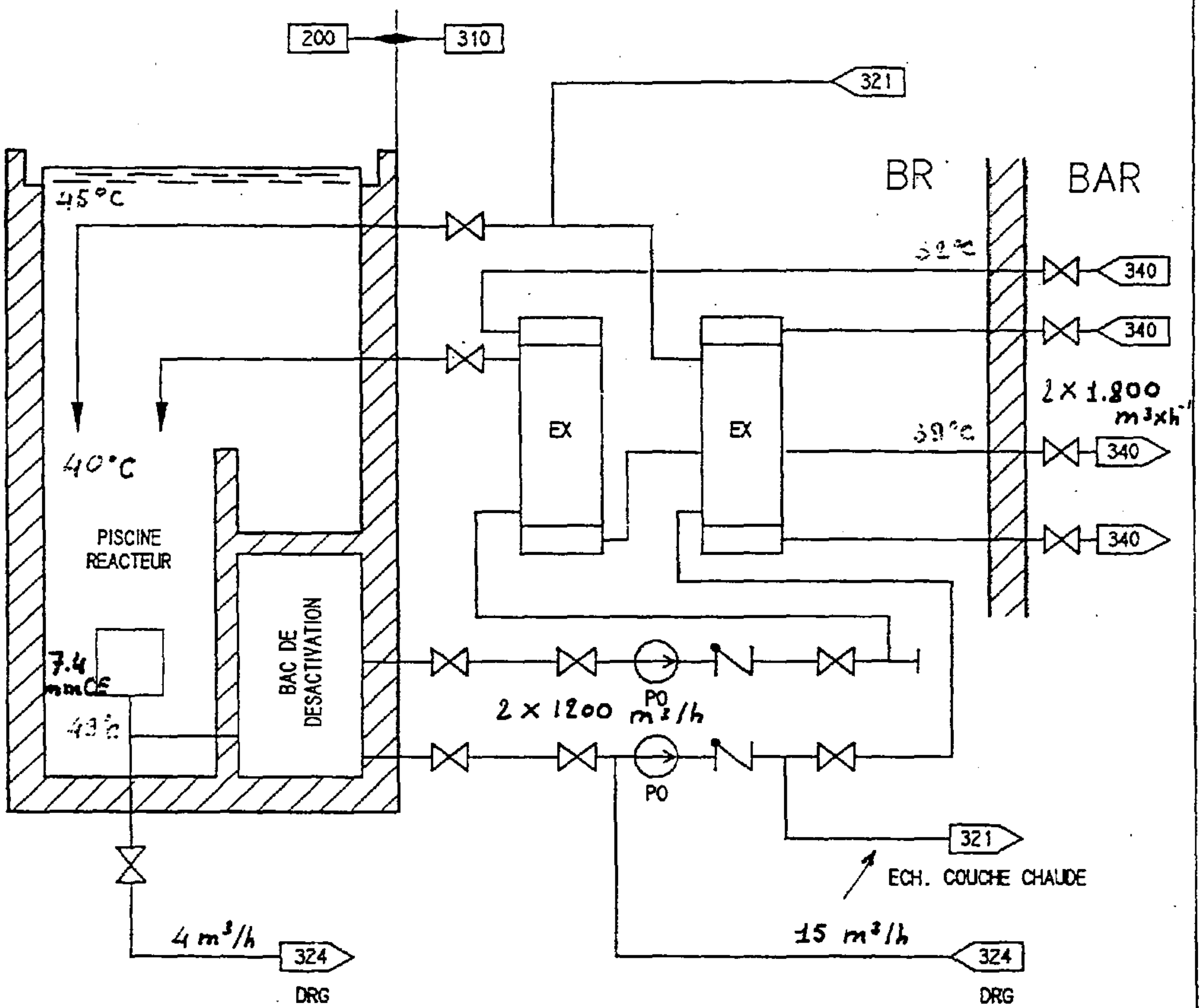


FIGURE 1 :
CIRCUIT PRIMAIRE PRINCIPAL 310

00	01	02	03	04	05	06	07	08	09
	0,4	0,9	1,3	1,6			1,3	0,9	0,4
10	1		B E	R Y	L L	I	U M		1
20	1,6								1,6
30	1,8								1,9
40	1,9	3,1						3,1	1,9
50	1,5								1,5
60	1								1
70	0,6	1,1	1,6	2	2.2	2,1	1,7	1,1	0,6
80	FLUX MOYEN DE NEUTRONS THERMIQUES EN 10^{14} n/cm^2 s								
90	PUISSANCE 25MW			TOUTES BARRES HAUTES					

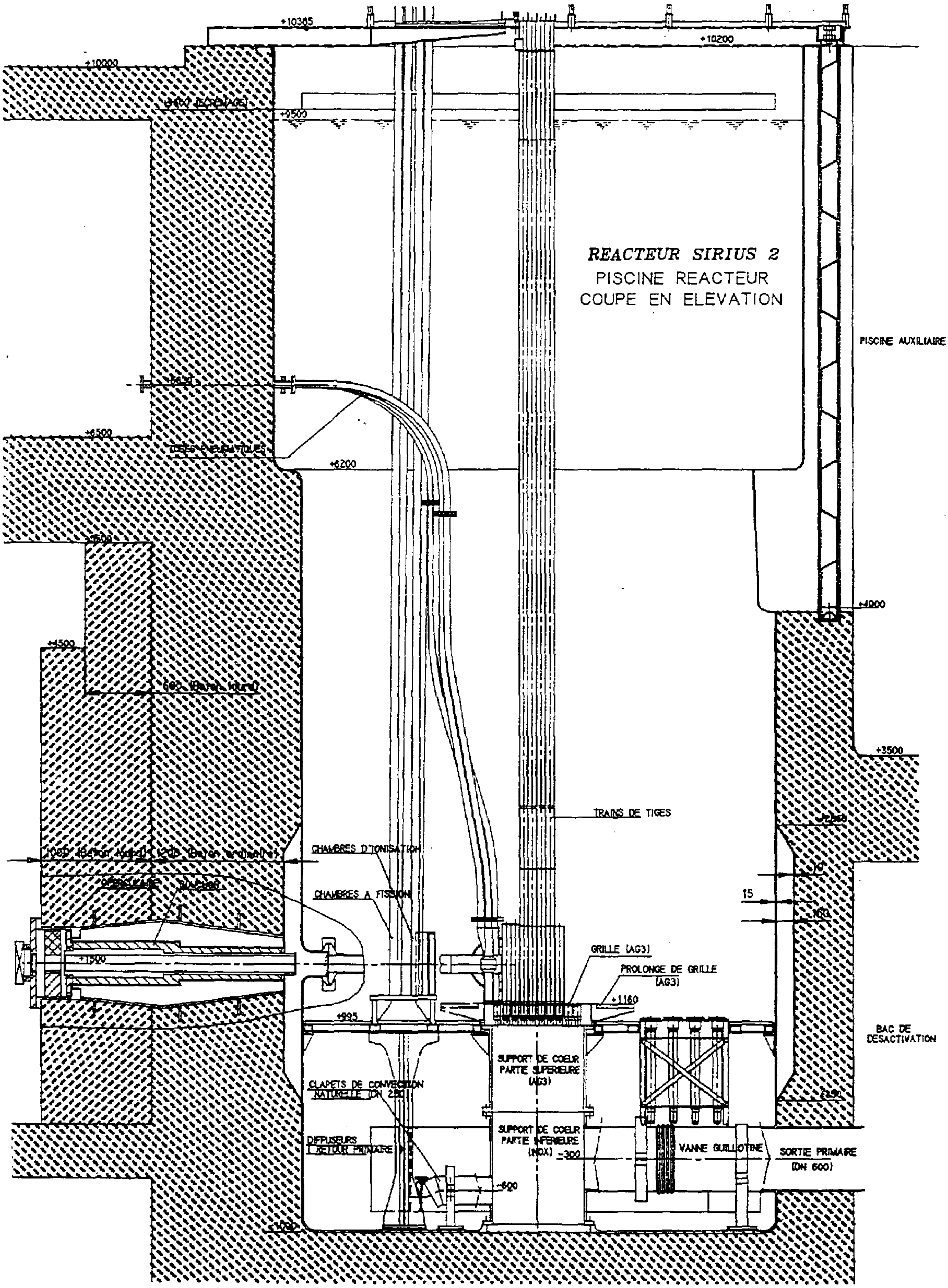
ELEMENT STANDARD

ELEMENT DE CONTROLE

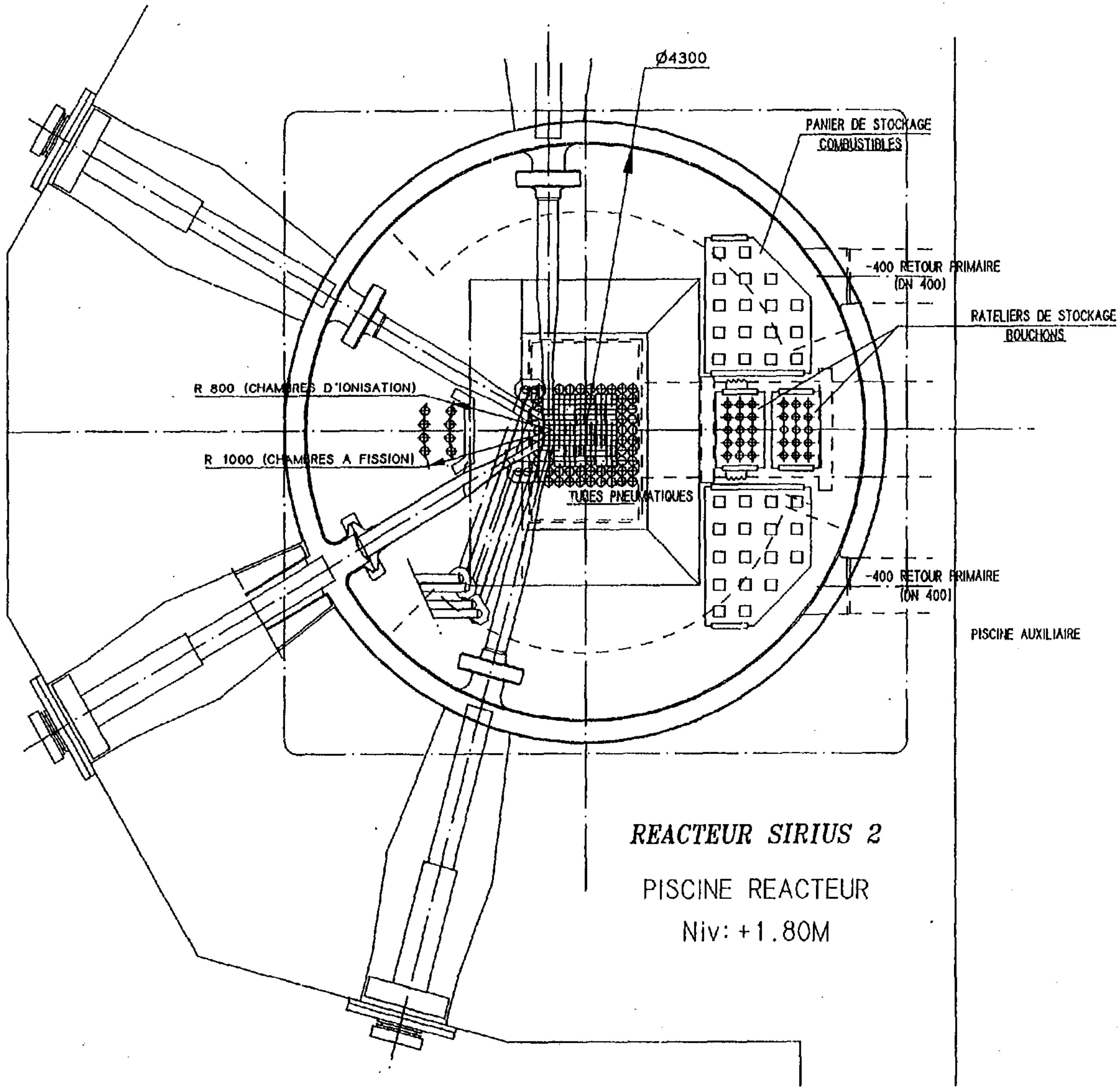
ELEMENT D'IRRADIATION

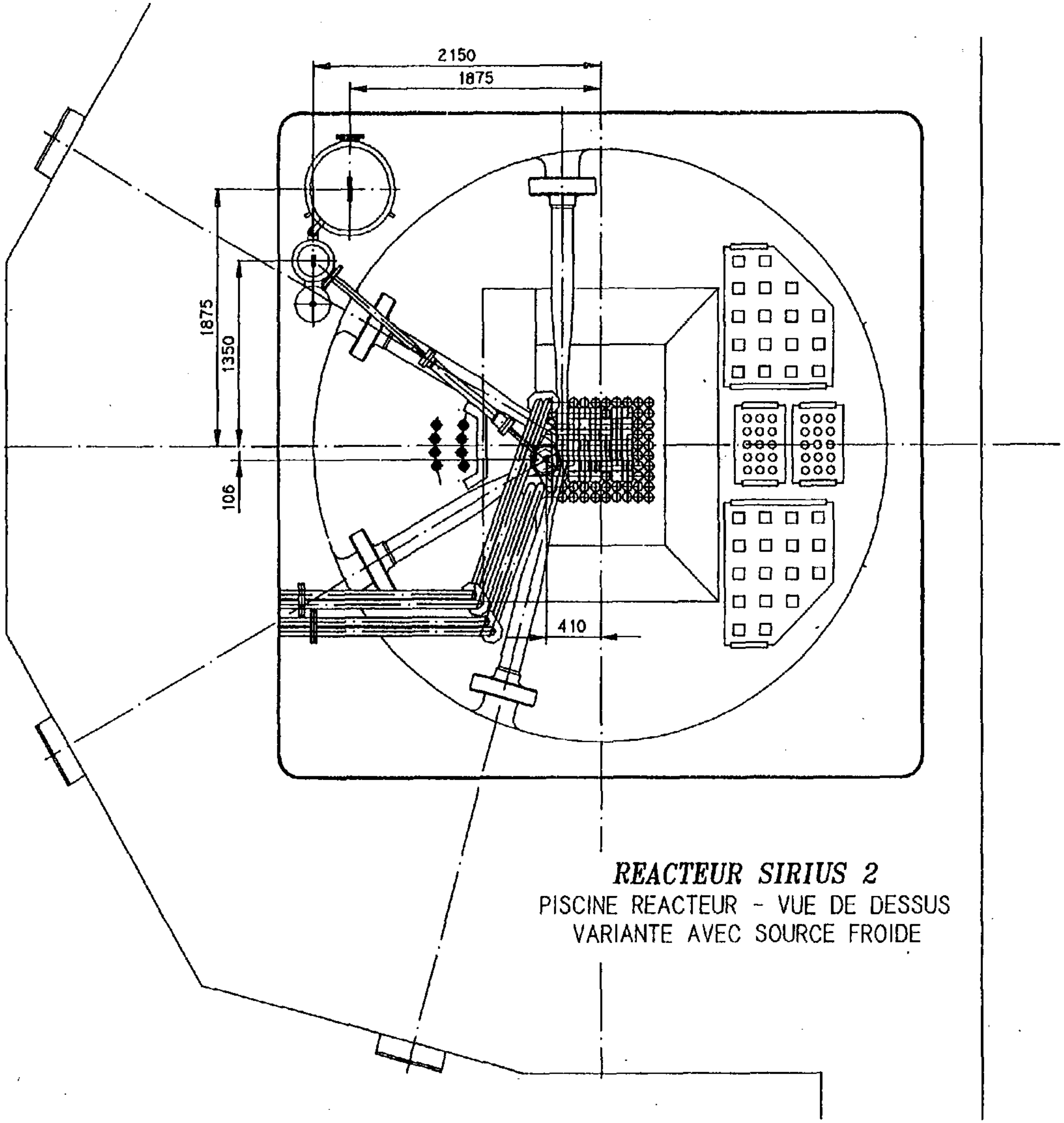
COEUR 10A1

FIGURE 2



REACTEUR SIRIUS 2
PISCINE REACTEUR
COUPE EN ELEVATION
+10385
+10200
+10000
+9500
PISCINE AUXILIAIRE
+6500
+6200
TIGES PNEUMATIQUES
+4900
+4500
+3500
TRAINS DE TIGES
CHAMBRES D'IONISATION
CHAMBRES A FISSION
GRILLE (AG3)
PROLONGE DE GRILLE (AG3)
+1160
+995
15
SUPPORT DE COEUR PARTIE SUPERIEURE (AG3)
SUPPORT DE COEUR PARTIE INFERIEURE (INOX)
-300
CLAPETS DE CONVECTION NATURELLE (DN 250)
DIFFUSEURS RETOUR PRIMAIRE
-600
VANNE GUILLOTINE
SORTIE PRIMAIRE (DN 600)
BAC DE DESACTIVATION


Ø4300
PANIER DE STOCKAGE COMBUSTIBLES
-400 RETOUR PRIMAIRE (DN 400)
RATELIERS DE STOCKAGE BOUCHONS
R 800 (CHAMBRES D'IONISATION)
R 1000 (CHAMBRES A FISSION)
TUBES PNEUMATIQUES
-400 RETOUR PRIMAIRE (DN 400)
PISCINE AUXILIAIRE
REACTEUR SIRIUS 2
PISCINE REACTEUR
Niv: +1.80M



REACTEUR SIRIUS 2

PISCINE REACTEUR - VUE DE DESSUS

VARIANTE AVEC SOURCE FROIDE

OTHER NEUTRON SOURCES

PAUL SCHERRER INSTITUT

Mai 1992/HI82

BAUER 30.04.92.TXT

SINQ - The Future of Neutron Scattering in Switzerland

G.S. Bauer

In early 1988 the Swiss parliament approved the proposal forwarded by the then Swiss Institute of Nuclear Research, to upgrade the current of its 590 MeV isochronous proton cyclotron from 250 μA to 1.5 mA and to use the beam passing throught the pion production targets to drive an intense souce of neutrons for beamhole research.

SINQ, as the new project was named, clearly was the most ambitious project for a spallation neutron source so far approved world wide. By and large its proton current would be an order of magnitude higher than that of the closest facility, ISIS at Rutherford-Appleton Laboratory in the UK. The challenge to produce, accelerate and transport the beam to the target station was taken on by the accelerator development group and a major rebuild of the whole target system for pion production and parts of the proton transport system was necessary to cope with the planned 6-fold increase in beam power. This task was finished in 1991 and at present a proton current of about 500 μA can be accelerated. The necessary increase in RF power on the accelerator to achieve the full current will be implemented in steps over the next two years.

General description of SINQ

The cylotron delivering an essentially continuous stream of energetic protons, SINQ will be a cw-neutron source and for the user it will ressemble very much a research reactor. The proton beam is injected into a cylindrical target of about 18 cm diameter from underneath. Depending on the target material used, up to about 10 neutrons will be set free by each proton over a distance of some 30 to 50 cm. Thus the primary neutron source of SINQ will be rather compact. It will be surrounded by a large tank of D_2O (Fig. 1) to moderate the neutrons down to thermal energies for extraction through four pairs of twin-tube beam holes or for further moderation by a cold moderator of 20 litres of liquid D_2.

One pair of beam tubes plus a bundle of 5 neutron guides will view the cold moderator. In order to avoid partial rethermalization of the cold neutrons in windows or layers of D_2O between windows, the beam tubes and the source insertion tube are connected together to form a T which is filled with He gas. A similar T is also formed by two other tube pairs and a seccond source insertion tube. The use of the latter one is not yet definitively decided. It could simply hold a H_2O-scatterer, or a H_2 cold source or a graphite hot source.

Since the lateral shielding of a spallation neutron source must be rather massive to protect the experimental floor against high energy neutrons form the nuclear cascade part of the spallation process, great care was taken to insert as much as possible of the beam tube gear into the shielding. This includs a system of rotatable shutters, a filter holder and a drum to exchange collimators. In this way, neutrons scattered from these pieces of equipment can be taken care of by the main source shield.

A 35 x 50 m^2 neutron hall and a 25 x 50 m^2 neutron guide hall have been built to house the facility.

The target development programme

The target is the most critical component of SINQ. Safe containment of the radioactivity produced, high material density, efficient heat removal, low neutron absorption and resistance to intense radiation are only the most prominent ones of a large number of requirements to be met.

The original concept was to use a large quantity (ca. 100 litres) of molten Pb-Bi eutectic as target material and to remove the heat from the reaction zone by natural convection. This concept had a number of attractive features:

- good neutron yield from the heavy metal
- low neutron absorption in the target material
- highest possible material density in the reaction zone
- good self-shielding properties in beam direction against high enery neutrons
- no coolant in the direct proton beam
- low melting point (125 °C) allows direct water cooling

However, Bi being rather corrosive to all nickel-containing steels and many other materials, the choices for the canning materials are very limited and no one with low neutron absorption could be qualified. Furthermore, Bi, although of very low absorption cross section, produces 210-Po by thermal neutron capture which is a much feared α-toxic and volatile material. It was mainly this fact together with the losses to be envisaged from the target containment that led to a final change in opinion about the target concept.

It is still true that a liquid metal target, preferrably lead, would be the most desireable option, but in order to create enough confidence that such a target can be operated safely, a multiple-stage programme has been initiated:

- step one will consist of a heterogeneously cooled rod target of zircaloy in a safety container with a separately cooled beam entry window. The coolant will be heavy water to minimize moderation and absorption in the target itself. Although the neutron flux from this target is only 35 % of the optimum (see Table 1), it amounts to about 70 % of the engineered Pb-Bi-target. The main purpose of this step is to qualify the safety container with a target which in itself has little or no risk of failure and to produce enough neutron flux for the start-up phase of the source and its instruments.

- step two will use a similar target design but the zircaloy rods will be replaced with lead-filled zircaloy tubes. This change is expected to increase the thermal neutron flux by about 50 % over the value given for the zircaloy target, i.e. to what one would obtain from a Pb-Bi target with steel container. Since there is a certain risk of the tubes to be damaged by the higher expansion of lead during thermal cycling, prior qualification of the safety container is desireable.

- step three will be prepared in parallel to step one and two by a design and testing programme to develop a liquid lead target with a low absorption container. It is likely that part of the liquid metal may be pumped to produce an unambiguous cooling situation for the beam entry window. The liquid metal target would then be surrounded by the safety container qualified in steps 1 and 2 with the space between surveyed for radioactivity, pressure changes or moisture. This step will aim at coming as close as possible to the optimum performance of SINQ.

Neutron scattering instruments

As can be seen from Fig. 4,with the exception of a four circle diffractometer and a high resolution powder diffructomer located at one beam tube pair, all first generation instruments will be situated at neutron guides. A brief summary of the guides' specifications and their planned instruments is given in Table 2. These spectrometers constitute a rather conventional basic set of "production tools" and occupy less than half of the possible instrument positions in the neutron guide hall, the remaining ones being open for more advanced machines to complement the suite at a later date.

Expected performance

The 100 %-value of the relative flux levels given in Table 1 corresponds to $2x10^{14}$ $cm^{-2}s^{-1}$ per mA of proton current on target.. This value is derived from model calculations but agrees well with measured data obtained from mockup-experiments.

The calculated flux distribution in the D_2O-moderator tank is shown in Fig. 2. The beam tube noses and the D_2-cold neutron modertaor will be located at about 25 cm from the target centre line, i.e. in a region where the flux has fallen of only little from its peak value. Numbers are also given in Table 1 for the various target options. Fig. 3 shows the calculated radial dependence of the different neutron energy groups in the moderator. Apart from the presence of an energy group of more than 15 MeV this is of course very similar to a reactor.

The angular distribution of this high energy is very anisotropic, decreasing by about 5 orders of magnitude between a viewing angle of 0° to 90° relative to the target (radial versus tangential beam tube).

It is the presence of this small fraction of neutrons above 15 MeV which causes some concern about background levels on a cw spallation neutron source. The bulk shield of SINQ is designed to cope with these neutrons but some of them will also be present in the extracted beams. This gives rise to an uncertainly factor in the monochromator shielding requirements on the beam tubes. It is for this reason that it is planned to have only one pair of beam tubes equipped on day one and put most of the effort in initial instrumentation on the cold neutron guides which are considered the real strong part of SINQ.

Prospects

SINQ is a new development with little relevant experience to draw from world-wide. It will be the most powerful spallation neutron source, using novel concepts and operating on a proton beam intensity, where only minor losses can be tolerated without causing excessive activation or damage to components. Therefore, despite good expertise at the laboratory in the field, one must be aware of a certain risk in any prediction one might wish to make. According to present planning, construction work on the facility will be essentially completed in late 1994 or early 1995. The year of 1995 will be mostly devoted to commissioning and startup procedures. Since, together with SINQ, also the other targets and the beam transport line will be exposed to the full beam intensity for the first time, careful planning and measurements are mandatory. However, there is every reason to believe that, after this initial period, the facility can be run as reliably and efficiently as it used to run in the past 20 years. In this situation SINQ will be among the most powerful and most highly performing neutron sources in Europe, especially in the field of cold neutrons.

The project, therefore, does not only deserve the efforts of its highly motivated staff, but also at least some outside interest.

Reference

|1| F. Atchison: "Neutronic Considerations in the Design of Targets for SINQ". Paper presented at the International Workshop on Target and Moderator Technology for Spallation Neutron Sources" PSI, Villigen, Feb. 1992

Target system	Peak thermal flux ($cm^{-2}s^{-1}$)	Th. flux at 25 cm ($cm^{-2}s^{-1}$)	Loss factor at 25 cm radius
Pb with low absorption container	2×10^{14}	1.3×10^{14}	$\underline{1}$
Pb-Bi with steel container	0.9×10^{14}	0.85×10^{14}	0.65
W-plate in Al-container	0.8×10^{14}	0.6×10^{14}	0.46
Ta-plate in Al-container	0.55×10^{14}	0.45×10^{14}	0.35
Pb-spheres in Al-container	1.5×10^{14}	1×10^{14}	0.76
Zircaloy	0.8×10^{14}	0.59×10^{14}	0.45

Table 1 Calculated relative performance of different target concepts for SINQ. Numbers for unperturbed fluxes are per mA of proton current on target. (after ref. |1|).

Guide designation	Angle relative to centre line of bundle	Dimension Width x Height (mm^2)	curved length L_{curv} (m)	Radius of curvature R_{curv} (m)	critical wave length λ^* (Å)	Position at guide Beginning	Length	End
1RNR11	-6°	50 x 120	20	1445	2.4	NN (BSS)	NN (DSS)	<u>TOF</u>
1RNR12	-4.8°	20 x 120	20	3612	1.7 (NiC)	<u>High res. powder diff.</u>	Test	<u>Opt. bench</u>
1RNR13	-4.0°	30 x 120	20	2408	1.5	<u>Triple axis</u> ("Drüchal")	NN	NN (Laue-Camera)
1RNR14	+3.2°	35 x 120	20	2063	1.7		NN	<u>Pol. triple axis</u>
1RNR15	+4°	30 x 120	20	2408	1.5	NN	NN	NN
1RNR16	+6°	50 x 50	20	1445	4.2 (NiC)			<u>SANS</u>
1RNR17	+6°	two times 24.5 x 50s	24	1234	1.9	Reflectometer		

____ = First generation instruments   NN positions available for future instruments

Table 2 Specifications of neutron guides and instrument allocatins at SINQ. Guide coating is with supermirrors unless indicated otherwise (NiC)

HI82/TEXT TAB1-BQ.TXT

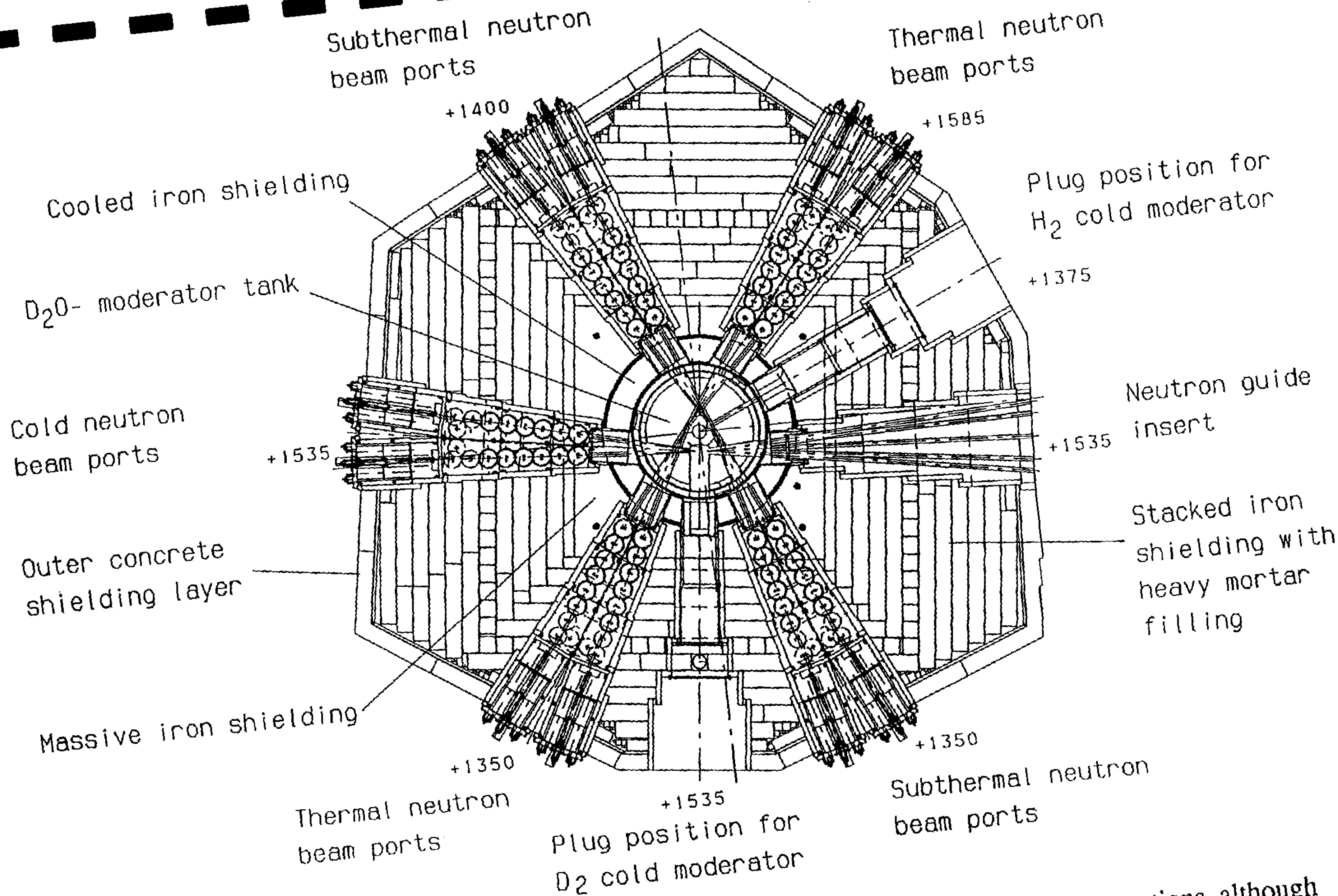


Fig.1 Horizontal section through the SINQ target block showing all horizontal penetrations, although actually on different levels. The height above the floor is indicated.

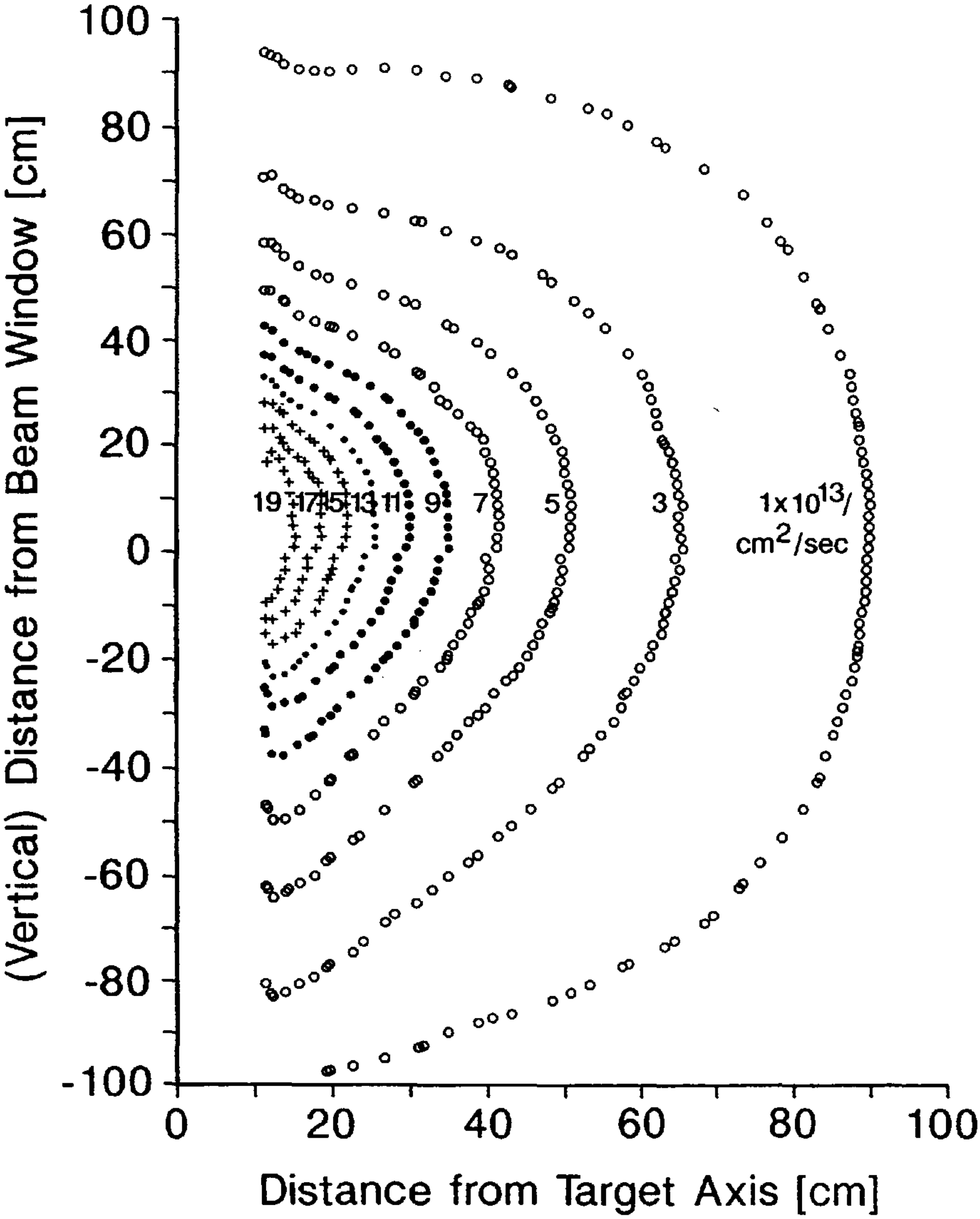


Fig. 2 Contours of undisturbed thermal neutron fluxes in the moderator for a Pb target with a low absorption container (ref. |1|).

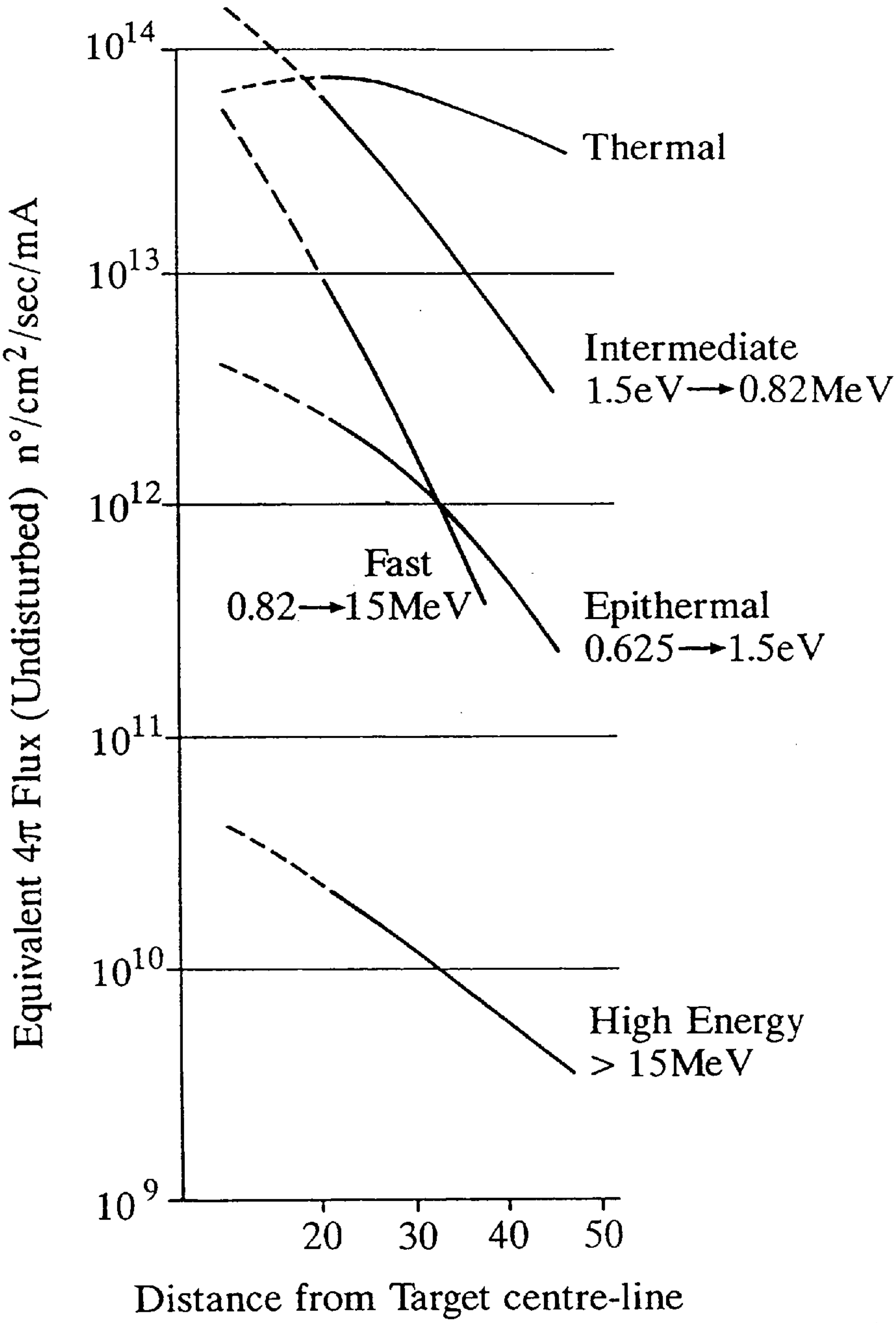


Fig. 3 Flux distribution for various energy groups in the D_2O moderator for a Pb-Bi target with steel container. The flux depression caused by absorption in the steel is clearly visible for the thermal group. (after ref. |11|)

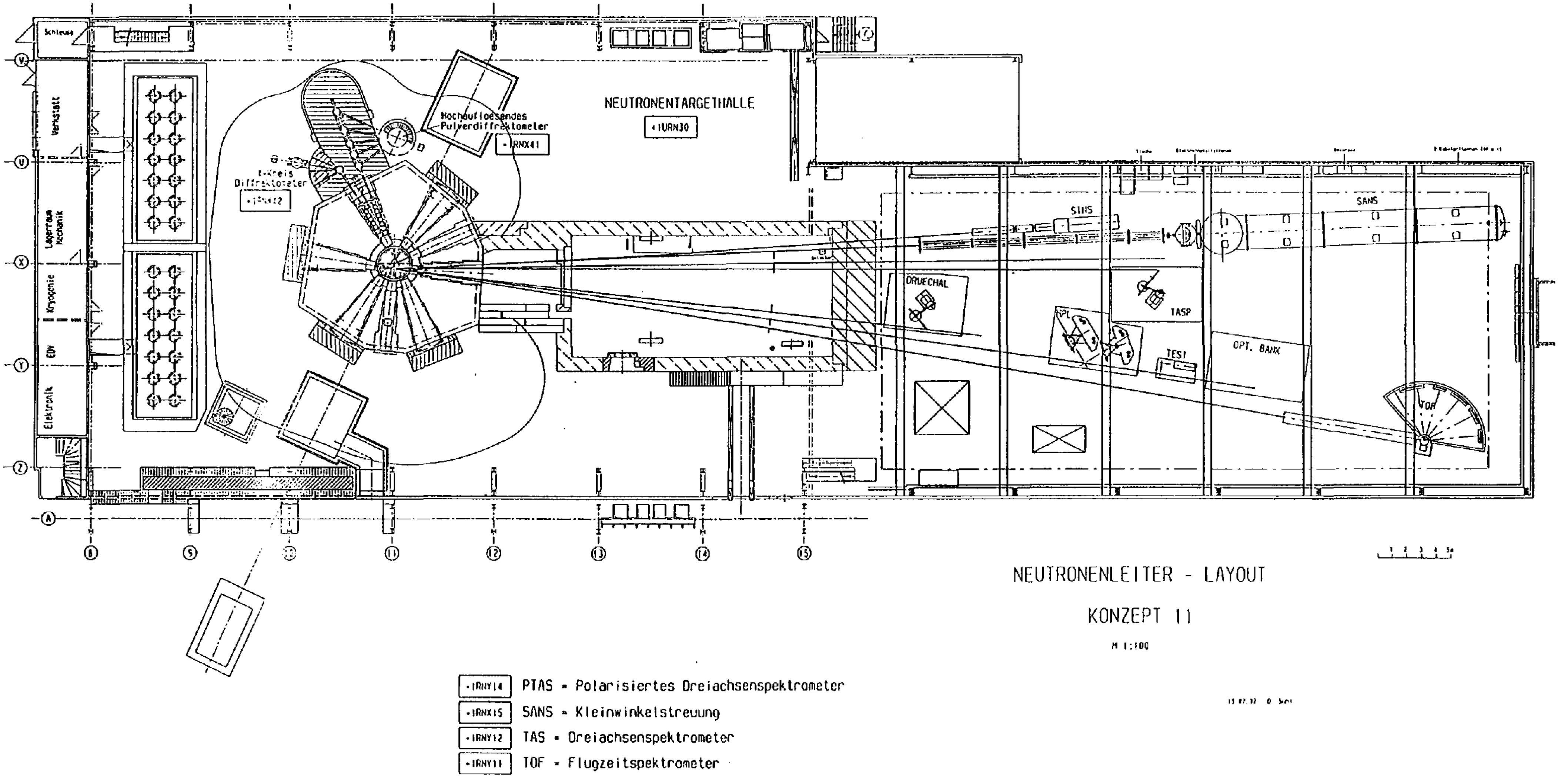


Fig. 4 Floor plan of the SINQ target and neutron guide halls showing the presently projected suite of instruments.

ABSTRACT *(Final Report not available)*

THE EUROPEAN SPALLATION SOURCE

A D Taylor
Rutherford Appleton Laboratory, UK

The Study Panel of the CEC on Neutron Beam Sources (Large Installations Plan), which addressed the short and long term prospects for neutron scattering research within the EC, recommended initiating study groups to investigate options for a new research reactor and a new spallation source to ensure Europe's continued pre-eminence in neutron scattering beyond the year 2000.

Together with KFA Julich, Rutherford Appleton Laboratory has recently taken the initiative to explore possible accelerator-based options for a future 'Next Generation' European Neutron Source. The CEC has provided support for two Expert Meetings, on Accelerators (at Simonskall near Julich in September 1991) and on Neutron Instrumentation and Techniques (at Abingdon near RAL in February 1992), and a further meeting on Target Technologies was held jointly with PSI at Villigen in early February.

The source specification was based on a proton accelerator producing 5 MW of beam power in 1 μs pulses at a repetition rate of 50 Hz. Two target stations were envisaged, one operation at 10 Hz for high resolution and long wavelength instruments and one operating at 50 Hz for high intensity. The conclusions of the three workshops were

- that an accelerator which meets this specification is technically possible;

- that an appropriate target-moderator system can be built;

- that such a source, having a performance substantially beyond what is available today at ILL and ISIS, will make significant advances in the existing fields of neutron scattering and will open up new scientific horizons.

These meetings defined the R&D programme on accelerators, target/moderator systems and neutron scattering instrumentation that is required to produce a full feasibility study (with preliminary costing) and outlined the scientific case.

WORKSHOP

R&D Results and Needs

PULSE IRRADIATION TEST ON LOW ENRICHED SILICIDE FUEL PLATES IN JAERI

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Japan Atomic Energy Research Institute
Tokai-mura, Naka-gun, Ibaraki-ken, 319-11 Japan

ABSTRACT

The low enriched, high density uranium aluminide/silicide fuels are becoming widely used in research and test reactors under the reduced enrichment program. In order to experimentally clarify the fuel behavior under transient/accidental conditions, and to establish a data base necessary for difining the safety criteria of such fuels, a series of transient irradiation experiments is being carried out using the NSRR in JAERI. This report presents the results of the first stage transient irradiation experiments.

INTRODUCTION

The low enriched, highly dense uranium aluminide/silicide fuels are becoming widely used in research and test reactors. In Japan, the operation of the modified Japan Research Reactor No. 3 (JRR-3M) of the Japan Atomic Energy Research Institute (JAERI) was started in November 1991, using low enriched (19.75wt%) aluminide fuels with uranium density of 2.2g/cc. Further, the conversion program of JMTR to low enriched silicide fuel core are being successfully progressed. A lot of efforts have been made internationally in order to understand the irradiation behavior and the safety characteristics of these high density fuels, including the fuel irradiation experiments with mini-plates/full-size fuel elements. However, regarding the fuel irradiation behavior under transient and accidental conditions, almost no experimental studies have been conducted on these fuels.

In JAERI, the transient irradiation experiment on the low enriched uranium silicide mini-plates was started from December 1990, by the pulse irradiation using the Nuclear Safety Research Reactor (NSRR) of JAERI-Tokai. This paper describes the results of the first stage transient irradiation experiment.[1),2),3)]

OBJECTIVES AND OUTLINE

The major objective of the transient irradiation experiment on the low enriched uranium silicide mini-plates are to experimentally clarify the fuel behavior under transient and accidental conditions, and establish a data base necessary to define the safety criteria for the silicide plate-type fuels. Such a data base will be more important for the design of higher flux/higher power density research reactors using the similar plate type fuels.

In the transient experiment, emphasis is made to understand the dimensional stability, the fuel failure threshold and the threshold of mechanical energy generation due to fuel melting and/or fragmentation under transient temperature conditions. Further, to know the behavior of fission products (FPs) release is also important.

The fuel transient temperature is one of the important parameters of the experiment. The fuel transient temperature depends on the deposited energy in the fuel mini-plate and ambient cooling condition. The deposited energy is proportional to the integrated value of the pulse power of the NSRR. The experiment was planned to be started from low pulse power and the pulse power to be increased step by step.

TRANSIENT IRRADIATION EXPERIMENT

Test Fuel Mini-plate

The fuel mini-plates used in these pulse irradiation tests are shown in Fig. 1 and its specifications in Table 1. The fuel mini-plates with uranium density of 4.8g/cc have been supplied to the tests at the first stage.

Instrumentations and Irradiation Capsules

Five Pt/Pt-13%Rh thermocouples (hereinafter abbreviated T/C's), of which melting point was 1,780°C, were spot welded directly to one side of the surface of each fuel mini-plate at five different locations, as shown in Fig. 1. After assembling to the supporting device, fuel mini-plate was contained in a irradiation capsule with stagnant water at the room temperature and at one atomospheric pressure. Two capsule pressure sensors and a water level sensor were also installed inside the capsule in order to monitor the pressure pulse and water hammer force caused by the melting and/or fragmentation of mini-plate.

Pulse Power History

The half-width of power of the NSRR pulse irradiation is a minimum of about 4.4ms at a maximum integral power of 110MW•s. The value of this width varies from 4.4 to 20ms depending on the magnitude of inserted reactivity. The integral value of the pulse power P (MW•sec) measured by micro fission chambers was used to estimate the deposited energy E_g (cal/g•fuel) in each test fuel mini-plate by

$$E_g = k_g \times P$$

where the power conversion ratio k_g (cal/g•fuel per MW•sec) was determine through fuel burn-up analysis on irradiated mini-plate at low pulse power taking the radial and axial power peaking into consideration.

RESULT AND DISCUSSIONS

Transient Temperature

Four pulse irradiation tests (#508-1, -2, -3 and -4) have been conducted in the first stage experiment. Table 2 shows a summary of fuel behavior derived from in-core measurements and post pulse irradiation examination (PIE). Figure 2 shows the typical histories of the reactor power, coolant and fuel temperatures. The measured plate temperatures are estimated to be lower (40°C in maximum) than realistic plate temperatures by the fin effect of T/C's themselves.

Figure 3 summarizes the relation between the measured fuel plate temperature and the given deposited energy. In this figure, solidus-liquidus temperatures from the phase diagram and observed phenomena are indicated. In first two experiments, in which maximum plate temperature were below 300°C, the silicide mini-plates kept thier intactness. In test #508-3, T/C #5 recorded the maximim temperature 544℃, and around this T/C local cracks were observed which were considered to be the intergranular crackings. Even in this experiment, no cracks were observed at the area where the temperature was below 400 ℃.

Dimensional Stability

Dimensional stability of mini-plates under pulse irradiation was studied using data from PIE and the summary is shown in Table 2. During the PIE, either logitudinal or transversal cutting of the mini-plates were made along T/C's, so that the dimensional stability of the mini-plates could directly be related to the measured local temperatures.

At the fuel temperature was below 400℃, the bowing was negligible, however, when the temperature exceeded 400℃, the bowing became evident. The bowing was enhanced significantly by necking, that is, a significant thinnig of plate thickness at the end peak locations where melting of Al-3%Mg alloy cladding and fuel seperation occurred simultaneously.

At higher temperature of close to 971℃, a significant bowing accompanying agglomerated and relocated molten aluminum were observed. During the quenching, intergranular crackings might have occured at the denuded fuel meat. In these manners, the dimensional stability of the silicide fuel mini-plate was degraded with increasing temperature from 400℃ to 971℃.

Microstructure of Fuel Core

Photograph 1 shows a result of SEM/XMA (Scanning Electron Microscope combined with X-ray Microstructural Analysis) for a specimen of the test #508-4. In this photograph, traces of metal to metal reaction were observed between aluminum matrix and silicide particles resulting in the formation of two additional phases at outer surface of the silicide particles. The outermost (first) phase consisted of aluminum riched U-(Al, Si) compounds with thickness of about 4 μm. The second phase consisted of Silicon riched (U, Si) compounds with thickness of about 1 μm.

CONCLUDING REMARKS

The transient irradiation behavior of the low enriched high density uranium silicide mini-plates have been studied by the pulse irradiation tests. Four mini-plates were irradiated by differnt reactor pulse power, and the results through the transient measurments and the PIE are as follows:

(1) Below 400 ℃ of surface temperature, neither failure nor degradation of dimensional stability of the mini-plates occured. They kept a good dimensional stability.
(2) At temperature beyond 400℃, the dimensional stability was degraded with temperature, and beyond 640 ℃, the melting point of cladding, mini-plate were damaged showing a bowing up to 7mm and increased cracking.
(3) Despite of the large degradation of the fuel mini-plate at temperature arround 970℃, no fuel fragmentation nor mechanical energy generation was observed.
(4) At higher temperature over 900℃, metal to metal reaction between aluminum matrix and silicide particles are observed.

The transient irradiation experiment on the low enriched uranium silicide mini-plates is planned to be continuously progressed for a couple years ahead, in which more detailed studies around the fuel failure threshold temperature, 400 ℃, will be conducted. Further, the pulse irradiation tests which provide more severe conditions to the mini-plates are also planned.

REFERENCES

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2) K.YANAGISAWA, "Post-pulse Detail Metallographic Examinations of Low-enriched Uranium Siliside Plate-Type Miniature Fuel", JAERI-M 91-152 (1991)
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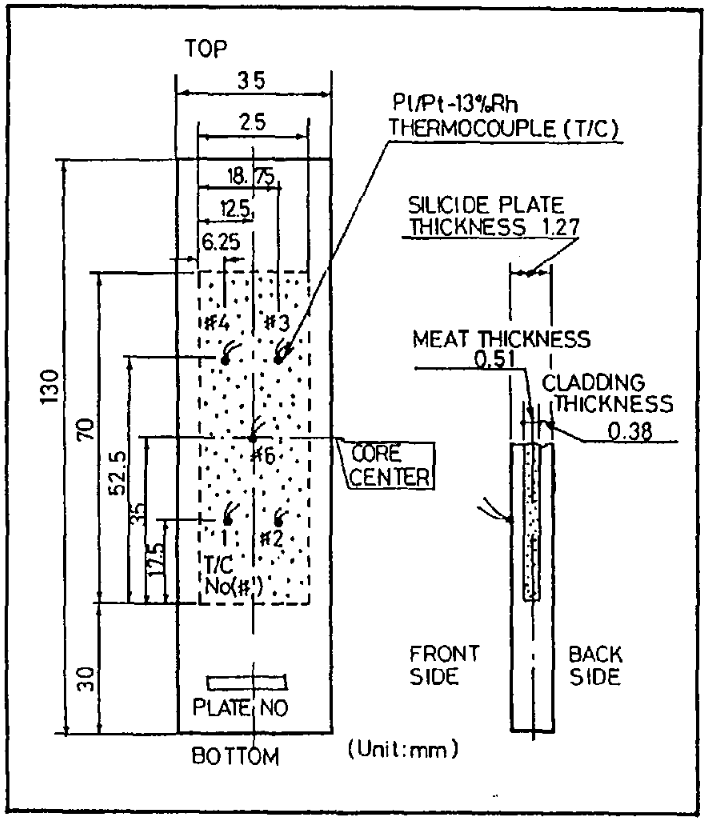


Fig. 1 Schematic Representation of the Tested Silicide Mini-plate

Table 1 Specifications of the Tested Silicide Mini-plate

ITEMS		Specifications
(1) CORE		
MATERIAL		U_3Si_2-Al
ENRICHMENT	, wt%	19.75
THICKNESS	, mm	0.51
WIDTH	, mm	25.0
LENGTH	, mm	70.0
URANIUM DENSITY	, g/cc	4.8
(2) CLADDING		
MATERIAL		AG3NE (AL-3%Mg)
THICKNESS	, mm	0.38
(3) PLATE		
THICKNESS	, mm	1.27
WIDTH	, mm	35.0
LENGTH	, mm	130.0

Table 2 Summary of Transient Irradiation Behavior of Low Enriched Silicide Mini-plate

TEST ID. NO.	#508-1	#508-2	#508-3	#508-4
Deposited Energy (cal/g·fuel)	62	77	116	154
Peak Sueface Temperature (℃)				
T/C #1	× (1)	200	350	971
#2	177	179	387	893
#3	216	183	414	652
#4	234	178	393	881
#5	178	195	544	957
Average±Standard Deviation (℃)	201±28 (2)	187±10	418±74	871±128
Coolant Temperature (℃)				
pre-pulse	20.4	21.6	17.8	17.2
peak	23.5	25.7	47.0	34.8
Capsule Pressure (MPa)				
bottom	(3)	0	0	0
top	(3)	0	0	0
Water Column Velocity	(3)	0	0	0
Max Bowing (mm)	None	None	1.50±1.15	4.11±1.60
Remarks			Pitting Cladding through cracks	Fuel seperation Molten Al relocation Plate through cracks Core-Al reaction

Note : (1) mulfunction
(2) Error band is σ_{n-1}
(3) not equipped

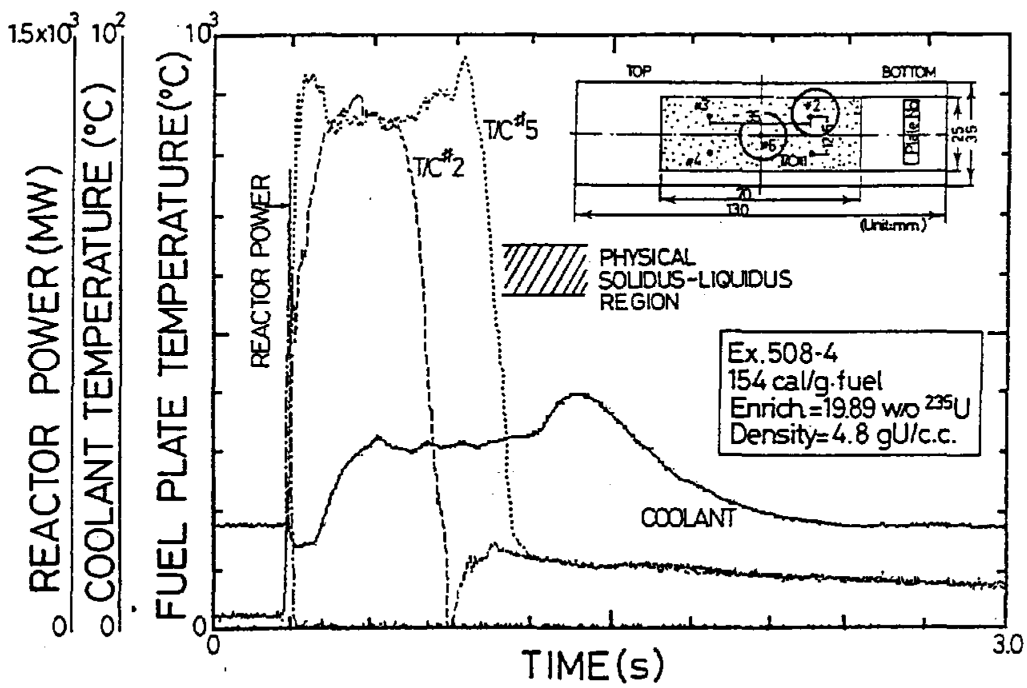


Fig. 2 Time-dependent Change of Fuel Plate Surface Temperatures, Coolant Temperature and Reactor Power in Test #508-4

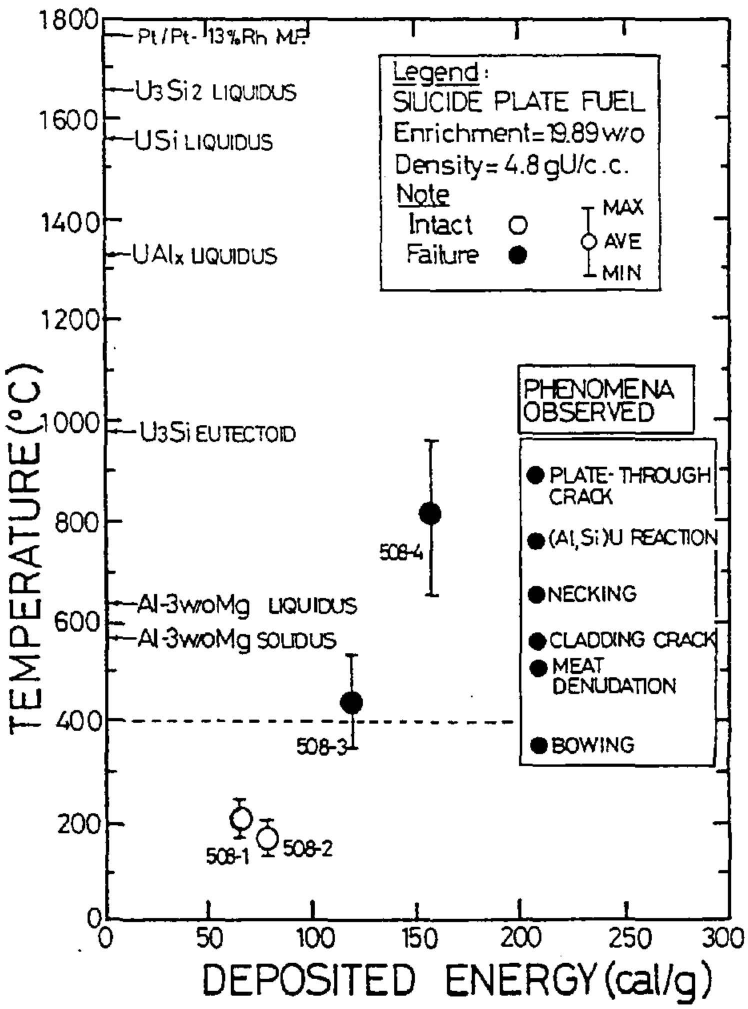


Fig. 3 Temperature of the Silicide Fuel Mini-plate as a Function of Deposited Energy by Pulse Irradiation

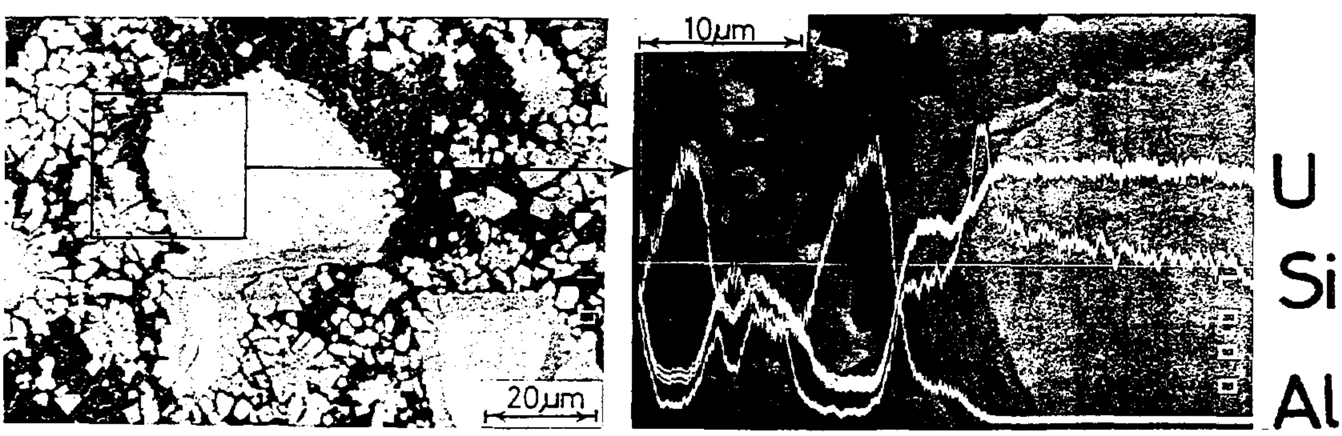


Photo. 1 XMA Linear Analysis against the Agglomerations of Si, U, and Al

PULSE IRRADIATION TEST ON LOW ENRICHED SILICIDE FUEL PLATES IN JAERI

S. MATSUURA, E. SHIRAI, T. FUJISHIRO
AND T. KODAIRA

Tokai Research Establishment
Japan Atomic Energy Research Institute

OUTLINE

Transient Irradiation Tests on the LEU silicide fuels have been conducted at NSRR of JAERI in order to:

- clarify the fuel behavior under transient/accidental conditions, and
- establish a data base necessary to define the safety criteria

OBJECTIVES

To know :

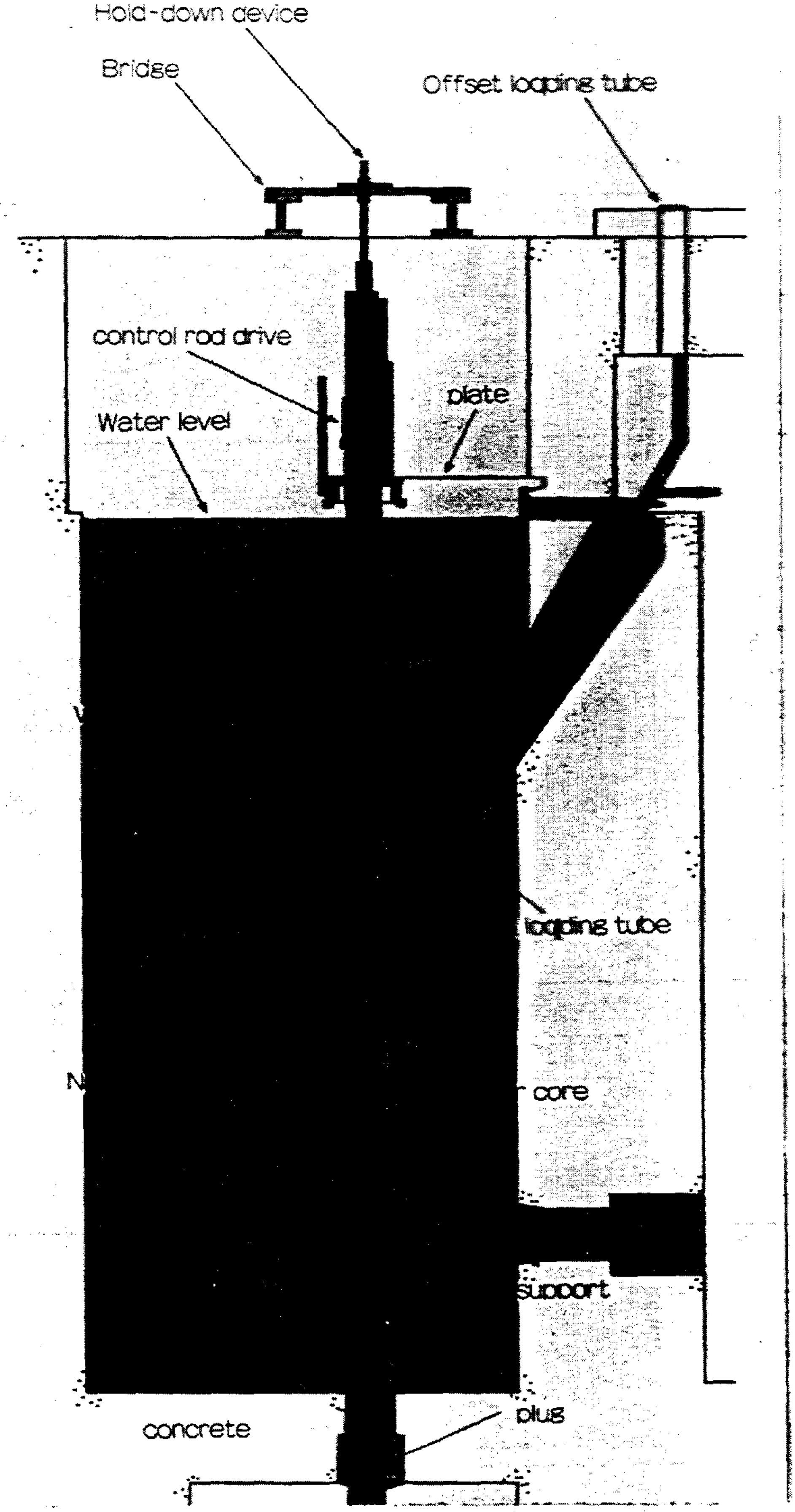
- Dimensional Stability
- Fuel Failure Threshold
- Threshold of Machanical Energy Generation
- Behavior of FPs Release

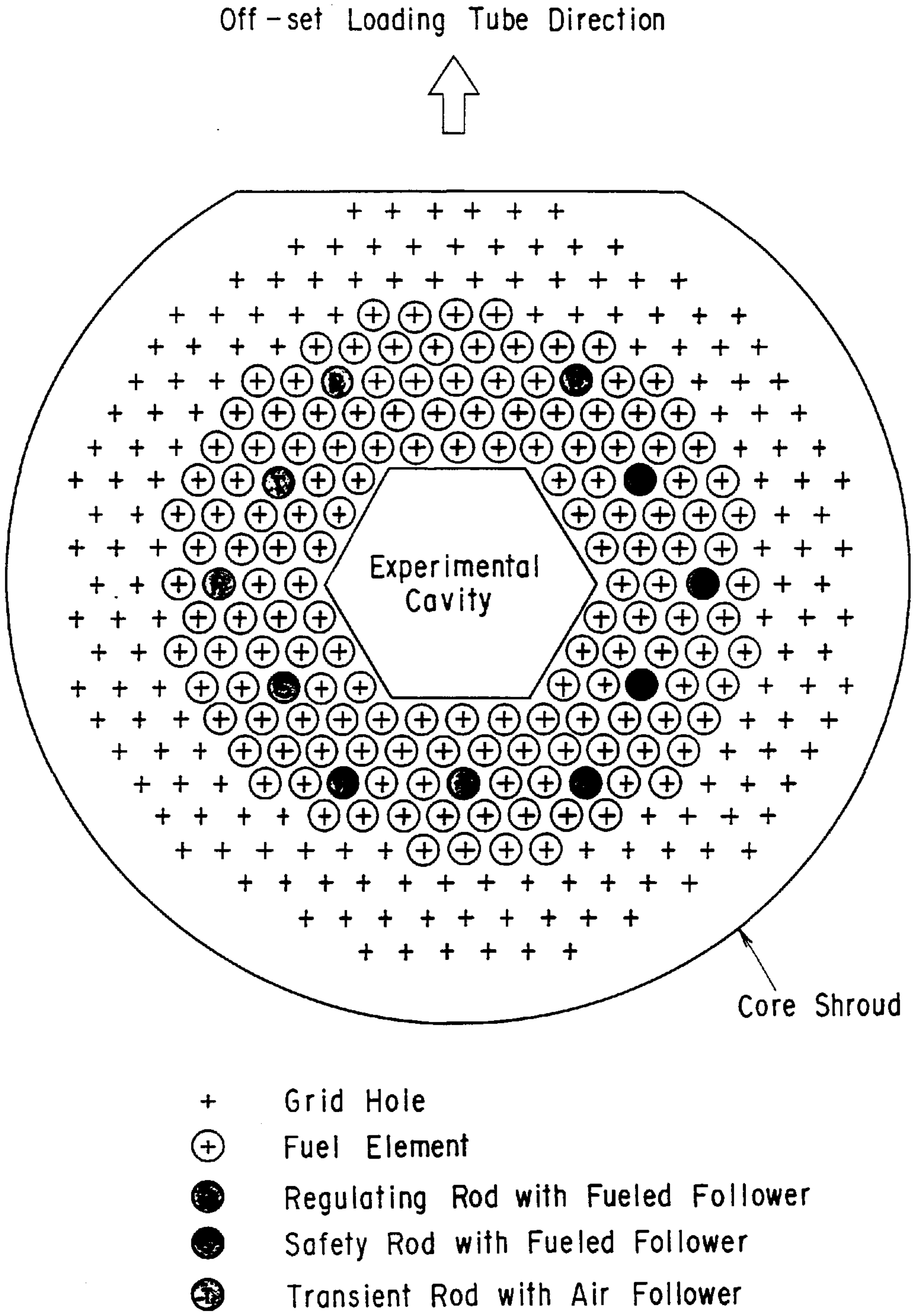
NSRR

The Nuclear Safety Research Reactor

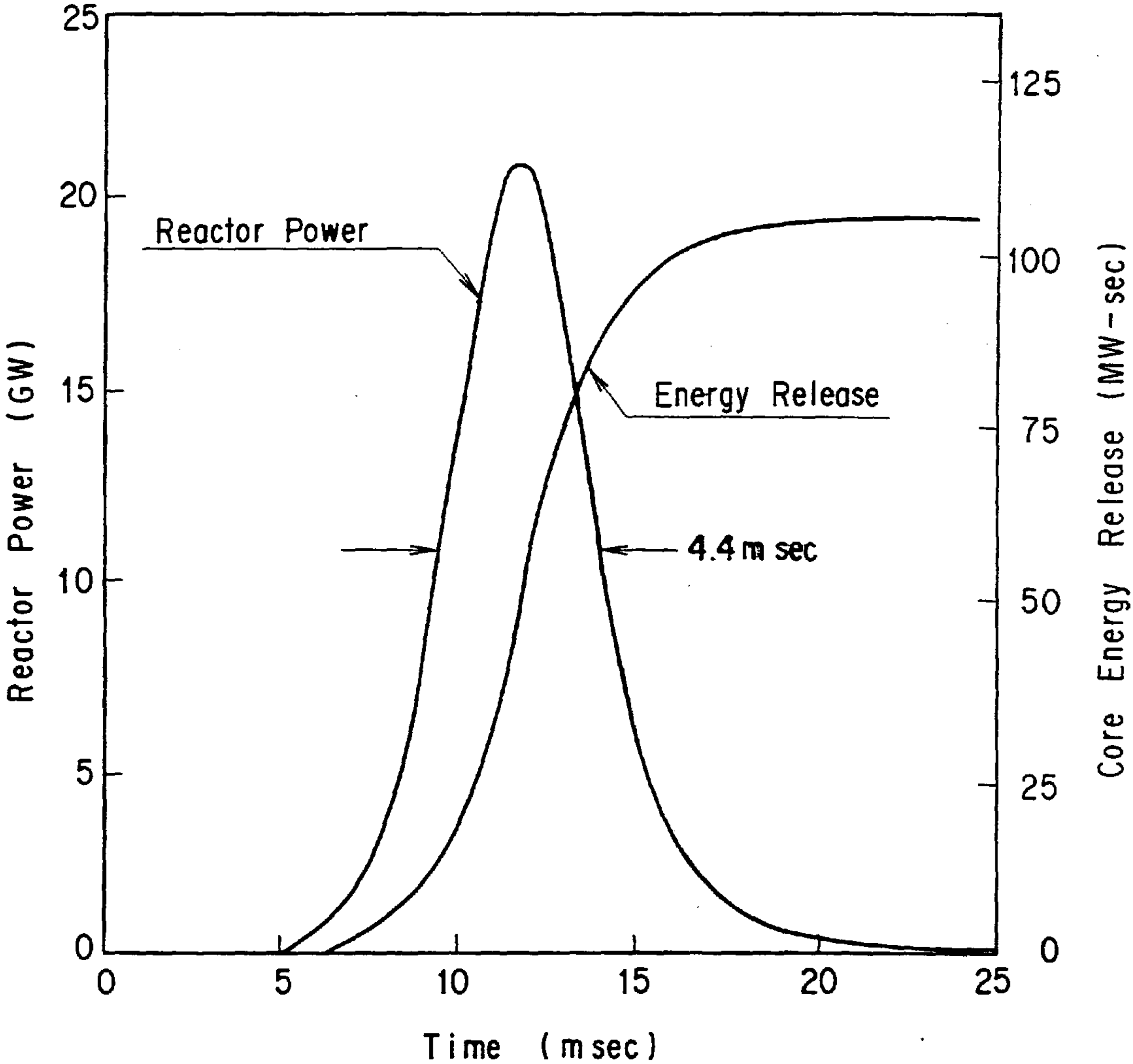
- is a modified TRIGA-ACPR
- has capability of the Pulse Irradiation Tests to investigate fuel behavior under RIA conditions

NSRR Facility



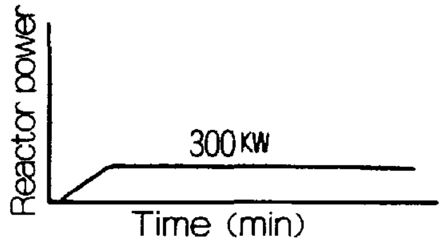
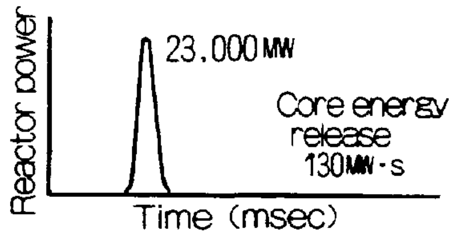
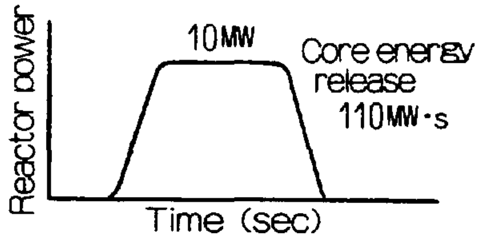
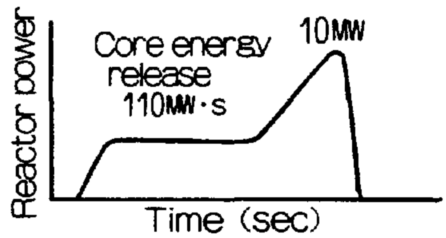
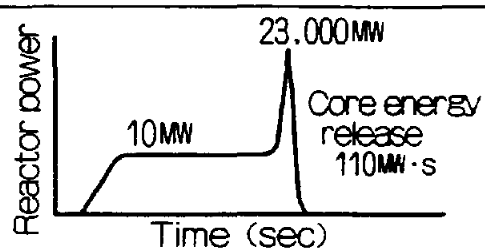
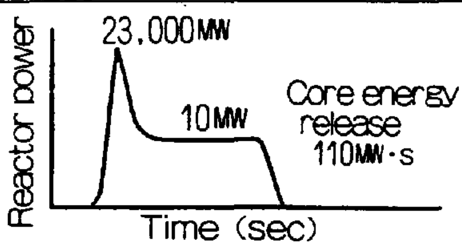


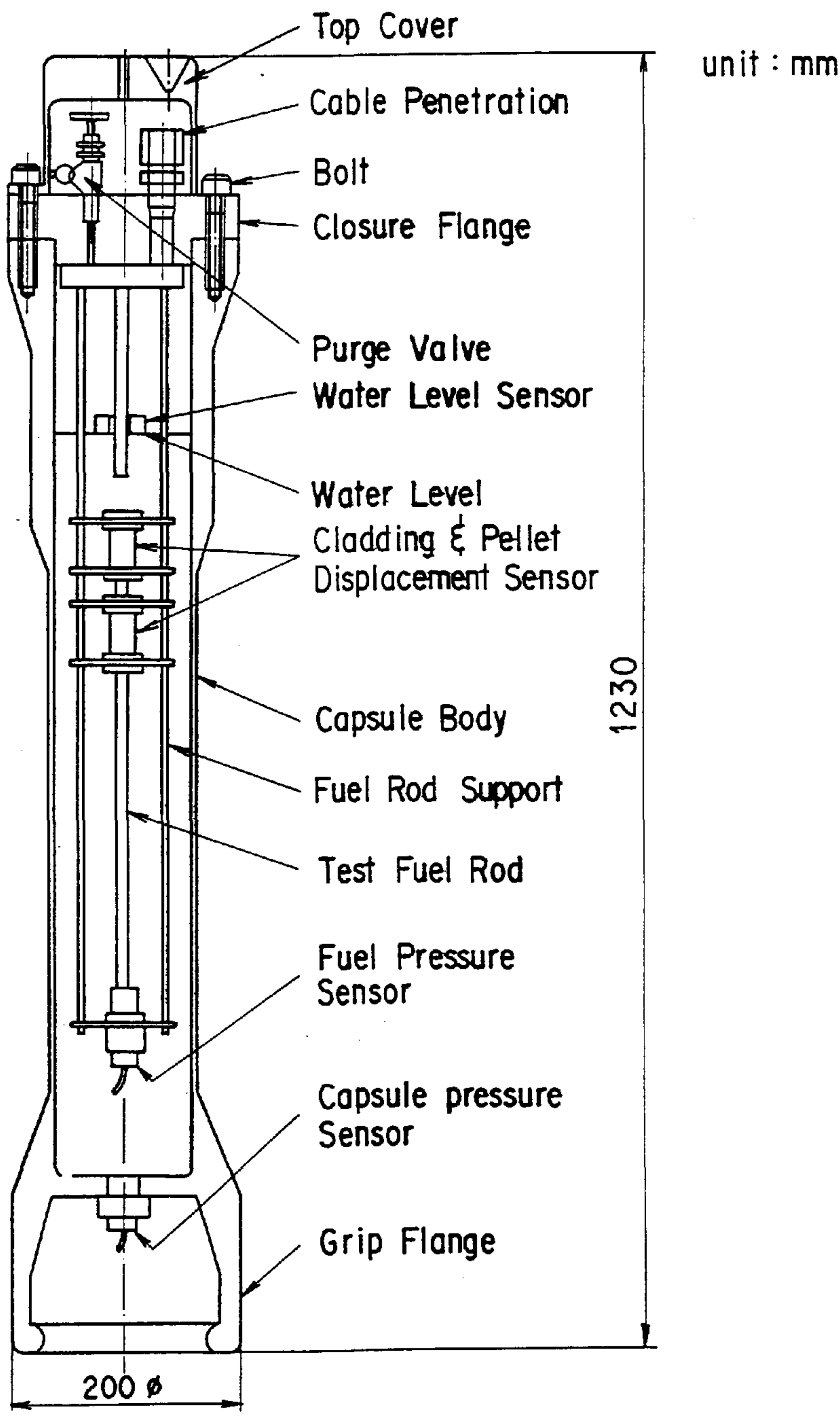
Operational Core Configuration



Reactor Power and Core Energy Release Histories
Obtained by 4.67S Pulse Operation in the NSRR

NSRR Operation Capability

Operation Modes		Research Items	
Steady State Power Operation	1. Steady state operation		·Fuel relocation
Pulse Power Operation	2. Natural pulse operation		·Fuel behavior in an RIA from cold start-up ·Molten fuel-coolant interation
	3. Shaped pulse operation		·Fuel relocation and coolability
			·Fuel behavior during power ramping
	4. Combined pulse operation		·Fuel behavior in an RIA from rated power
			·Debris coolability



Standard Water Capsule

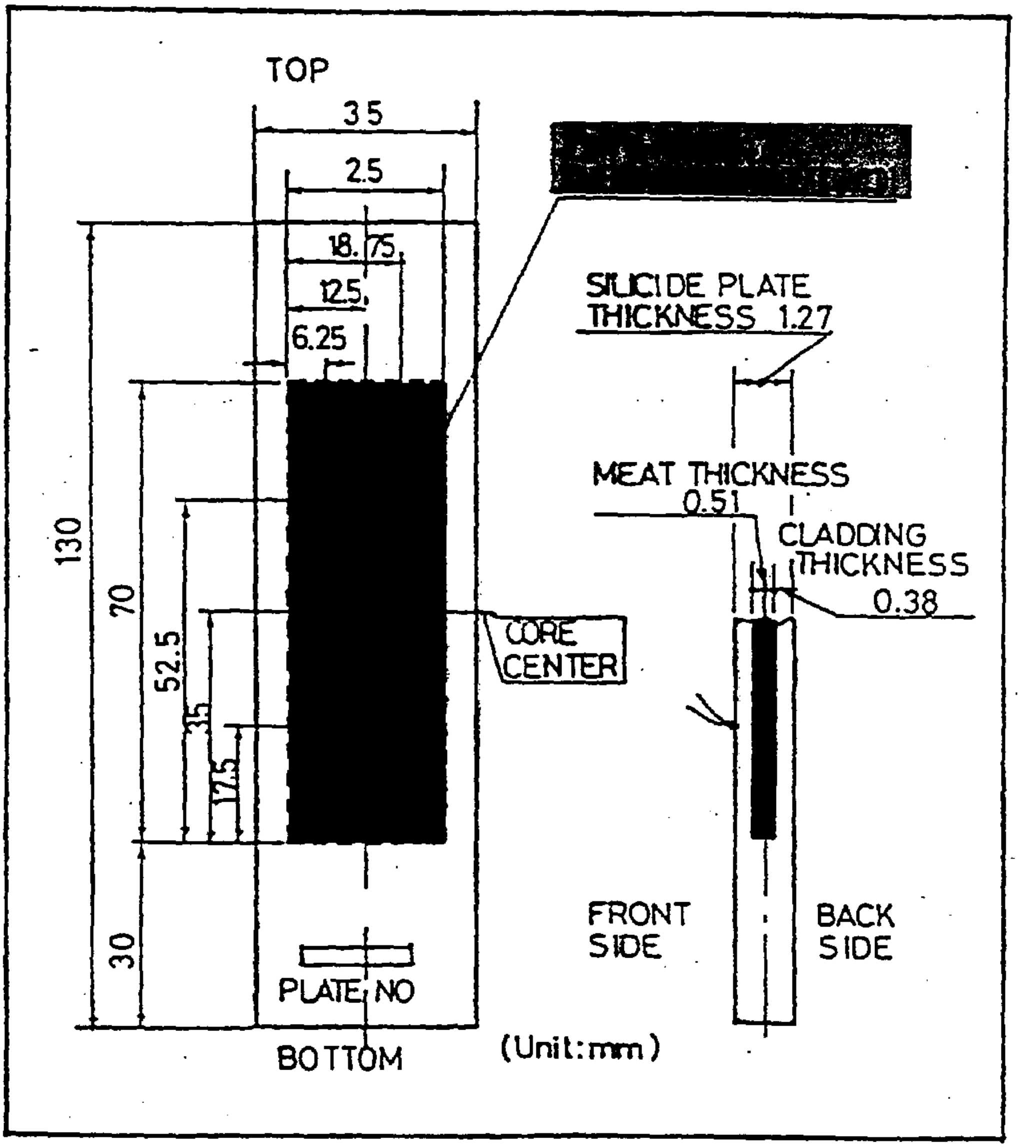


Fig. 4 Schematic Representation of the Tested Silicide Mini-plate

Table 2 Summary of Transient Irradiation Behavior of Low Enriched Silicide Mini-plate

TEST ID. NO.	#508-1	#508-2	#508-3	#508-4
Deposited Energy (cal/g·fuel)	62	77	116	154
Peak Sueface Temperature (℃)				
T/C #1	× (1)	[redacted]	350	[redacted]
#2	177	179	387	893
#3	216	183	414	652
#4	[redacted]	178	393	881
#5	178	195	[redacted]	957
Average±Standard Deviation (℃)	201±28 (2)	187±10	418±74	871±128
Coolant Temperature (℃)				
pre-pulse	20.4	21.6	17.8	17.2
peak	23.5	25.7	47.0	34.8
Capsule Pressure (MPa)				
bottom	(3)	0	0	0
top	(3)	0	0	0
Water Column Velocity	(3)	0	0	0
Max Bowing (mm)	None	None	1.50±1.15	4.11±1.60
Remarks			Pitting Cladding through cracks	Fuel seperation Molten Al relocation Plate through cracks Core-Al reaction

Note : (1) mulfunction
(2) Error band is σ_{n-1}
(3) not equipped

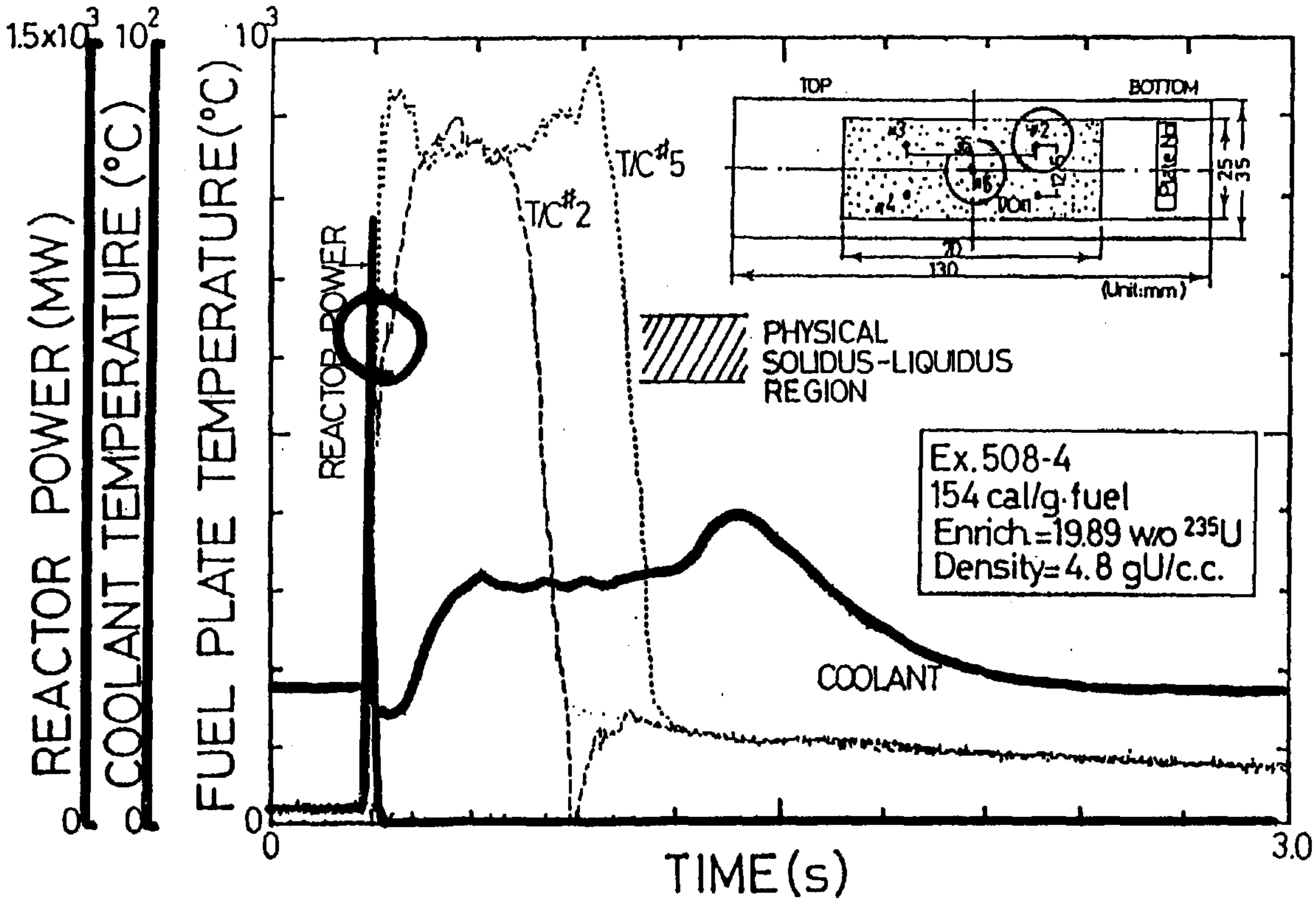


Fig. 3.4.2 Time-dependent change of the fuel plate surface temperatures measured by T/C's #2 and #5, that of coolant temperature and reactor power, where 19.89w/o enriched uranium silicide plate-type fuel having density by 4.8 gU/cm^3 was used in experiment 508-4 (154 cal/g·fuel)

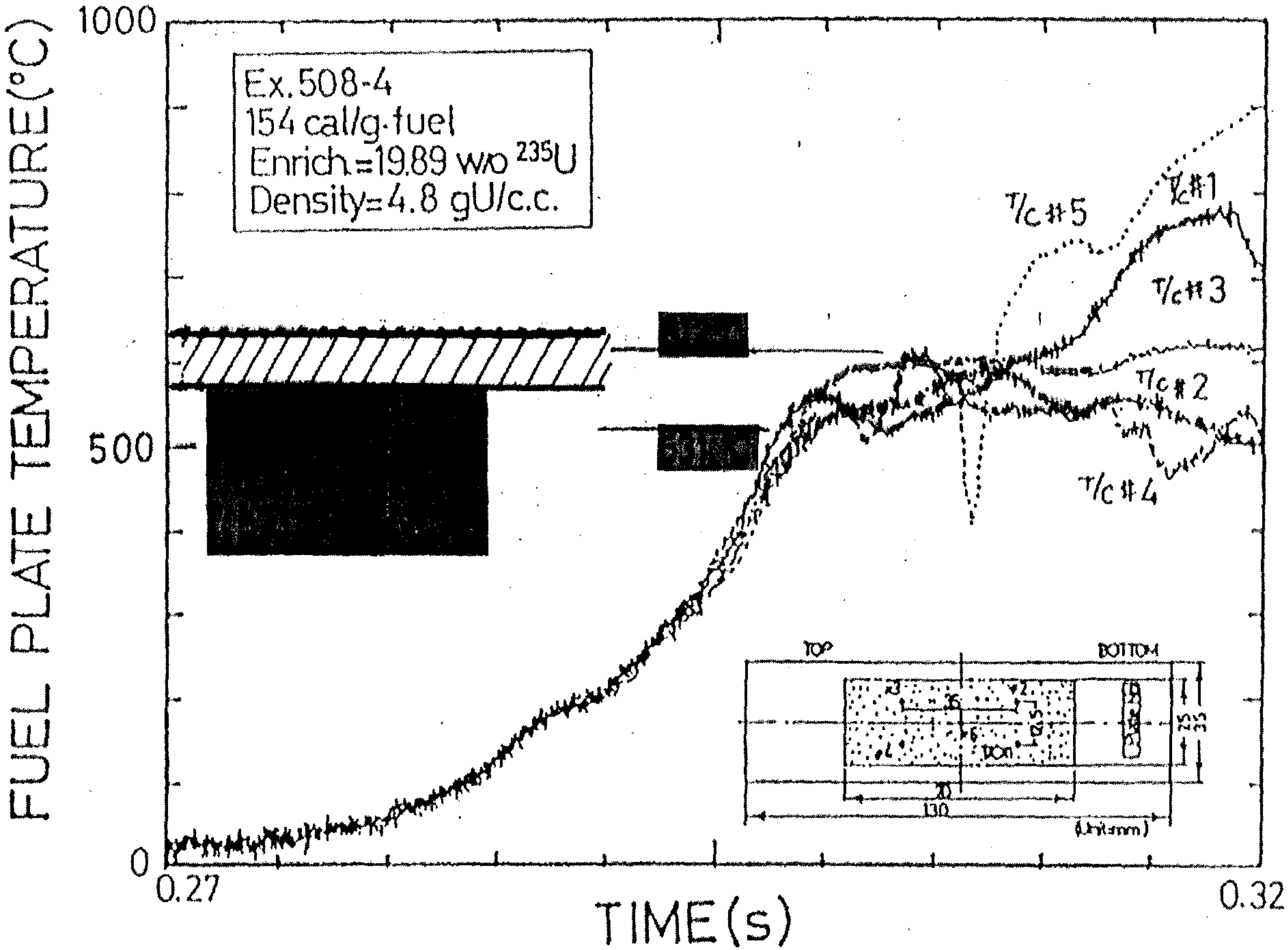


Fig. 5 Solidus-Liquidus Transformation (Latent Heat) Observed in Experiment #508-4.

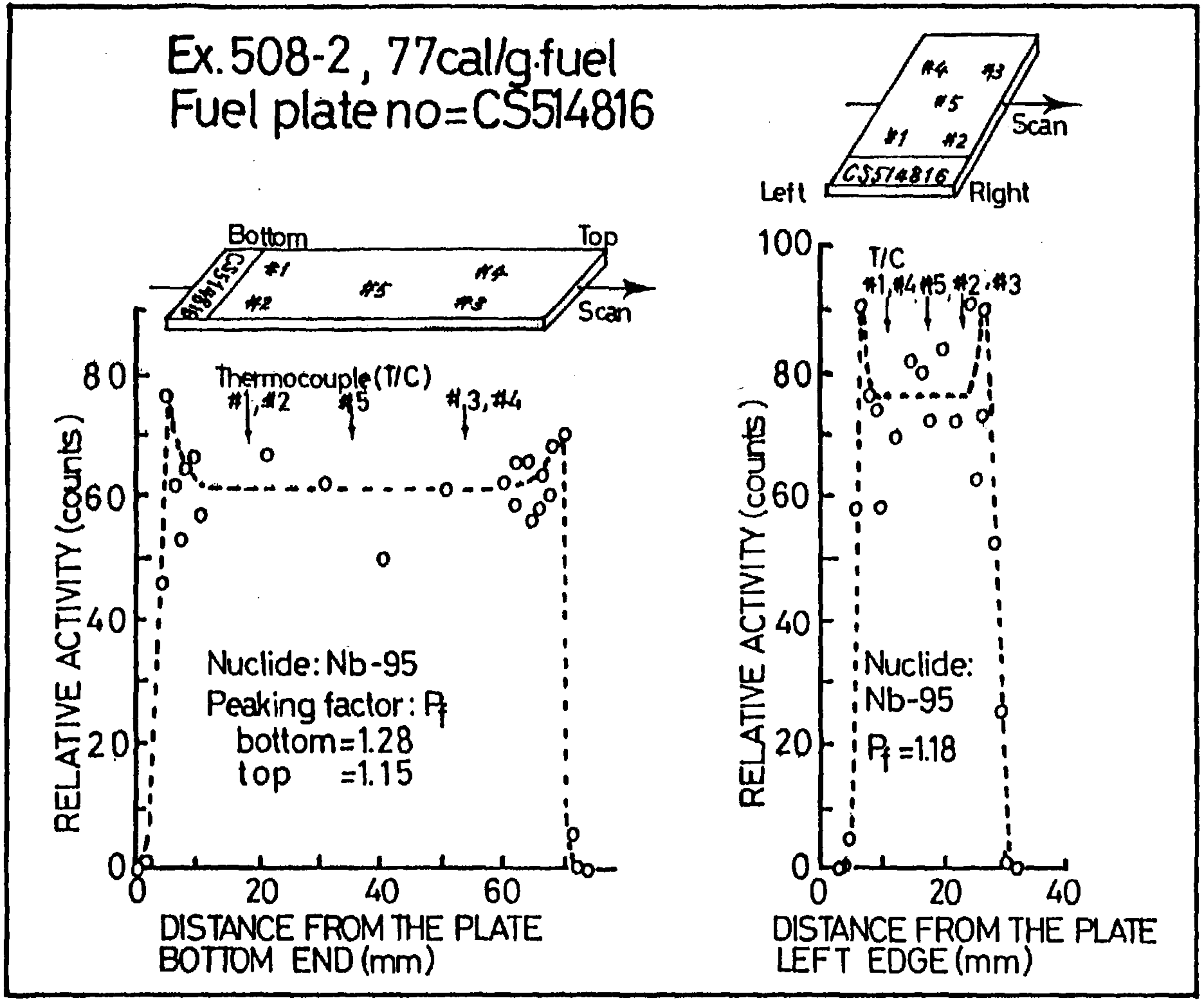
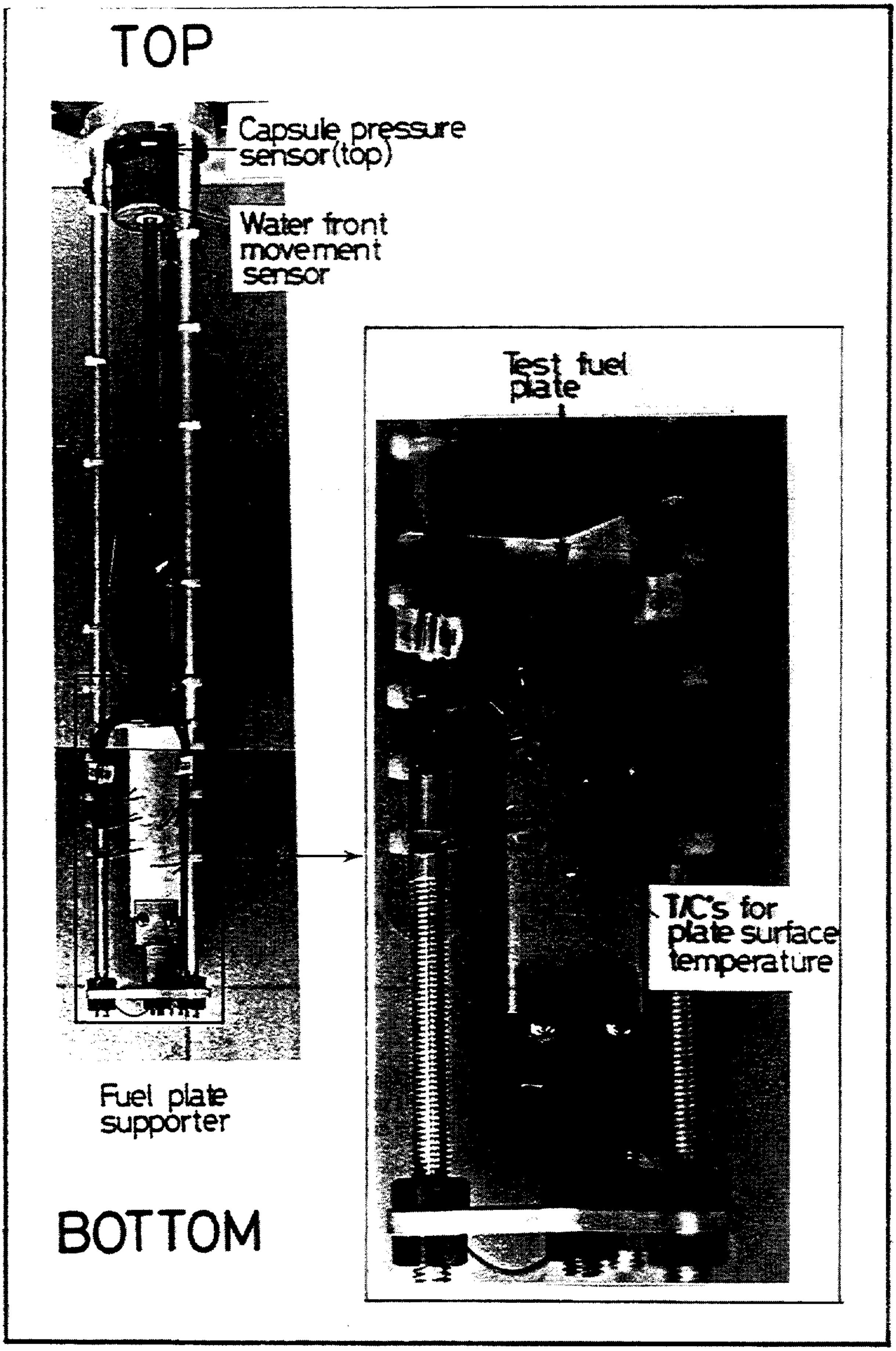
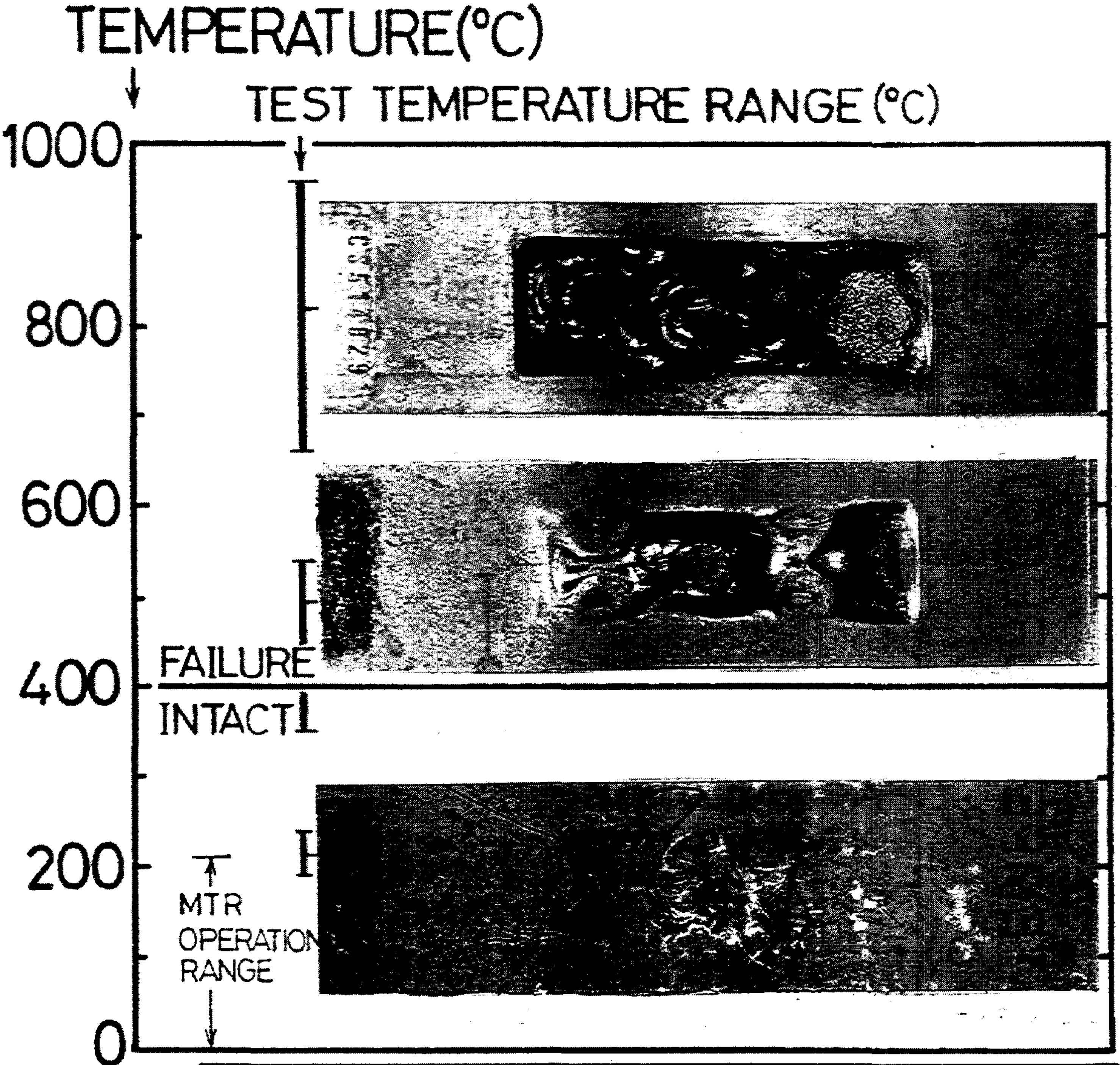
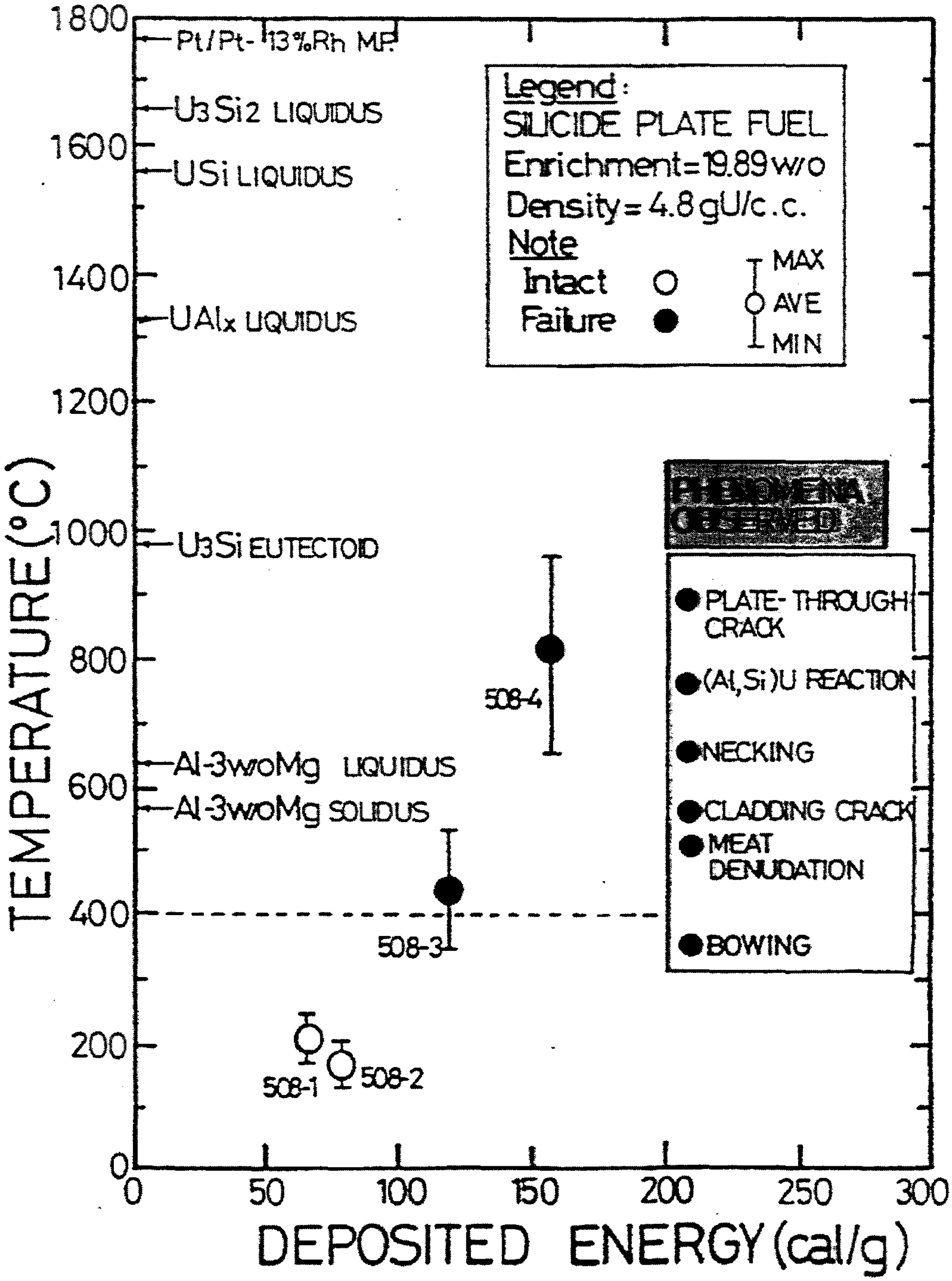


Fig. 2.6 Results of gamma scanning of the silicide plate-type fuel


TOP
Capsule pressure sensor(top)
Water front movement sensor
Test fuel plate
T/C's for plate surface temperature
Fuel plate supporter
BOTTOM



RESULT OF RIA EXPERIMENT ON LOW ENRICHED URANIUM SILICIDE PLATE-TYPE MINIATURE FUEL (ρ=4.8 g/c.c.)



Pt/Pt-13%Rh MP.
U3Si2 LIQUIDUS
USi LIQUIDUS
UAlx LIQUIDUS
U3Si EUTECTOID
Al-3w/oMg LIQUIDUS
Al-3w/oMg SOLIDUS
Legend:
SILICIDE PLATE FUEL
Enrichment=19.89 w/o
Density=4.8 gU/c.c.
Note
Intact
Failure
MAX
AVE
MIN
PHENOMENA OBSERVED
PLATE-THROUGH CRACK
(Al,Si)U REACTION
NECKING
CLADDING CRACK
MEAT DENUDATION
BOWING
508-1
508-2
508-3
508-4
TEMPERATURE (°C)
DEPOSITED ENERGY (cal/g)
0
200
400
600
800
1000
1200
1400
1600
1800
0
50
100
150
200
250
300

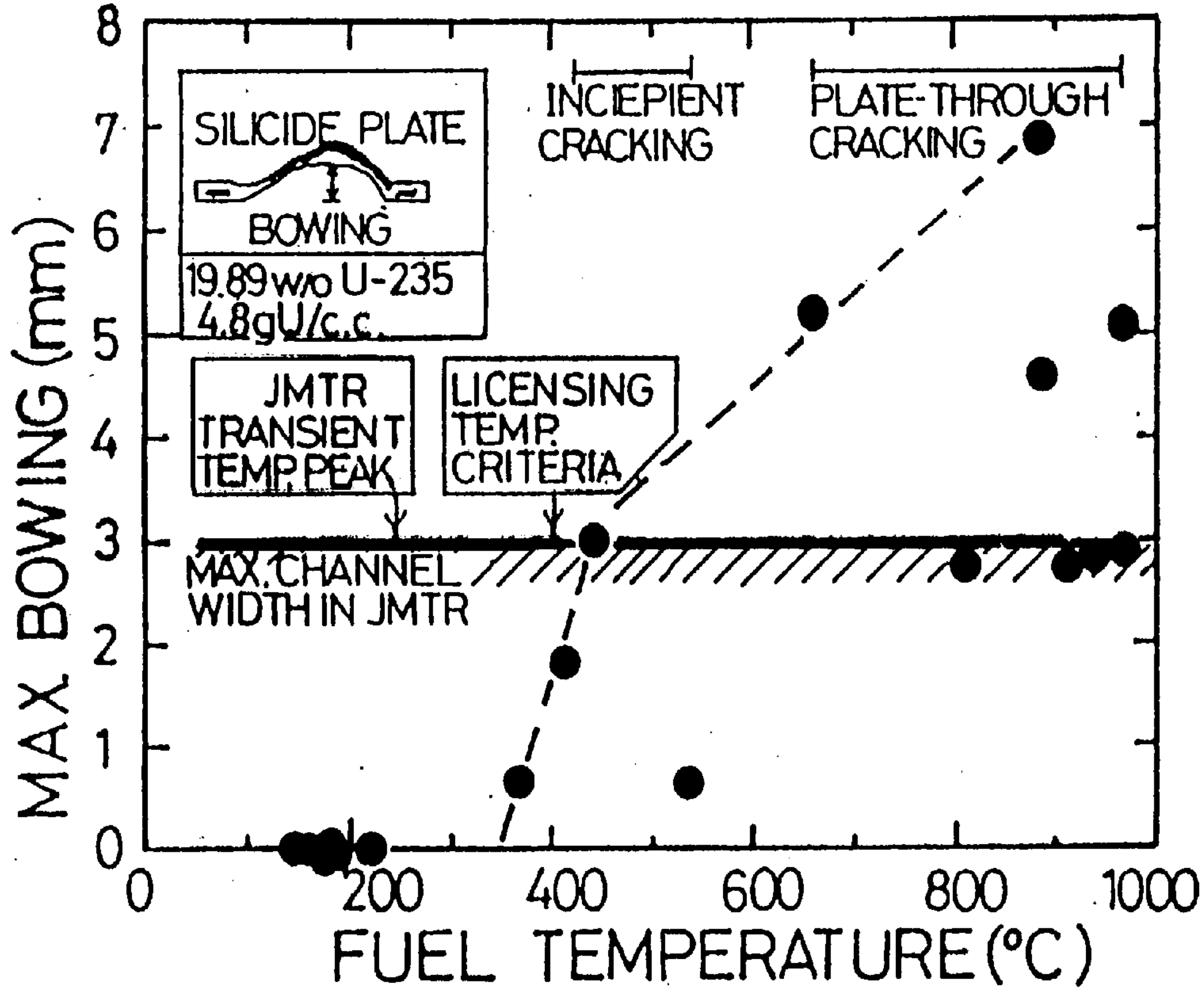
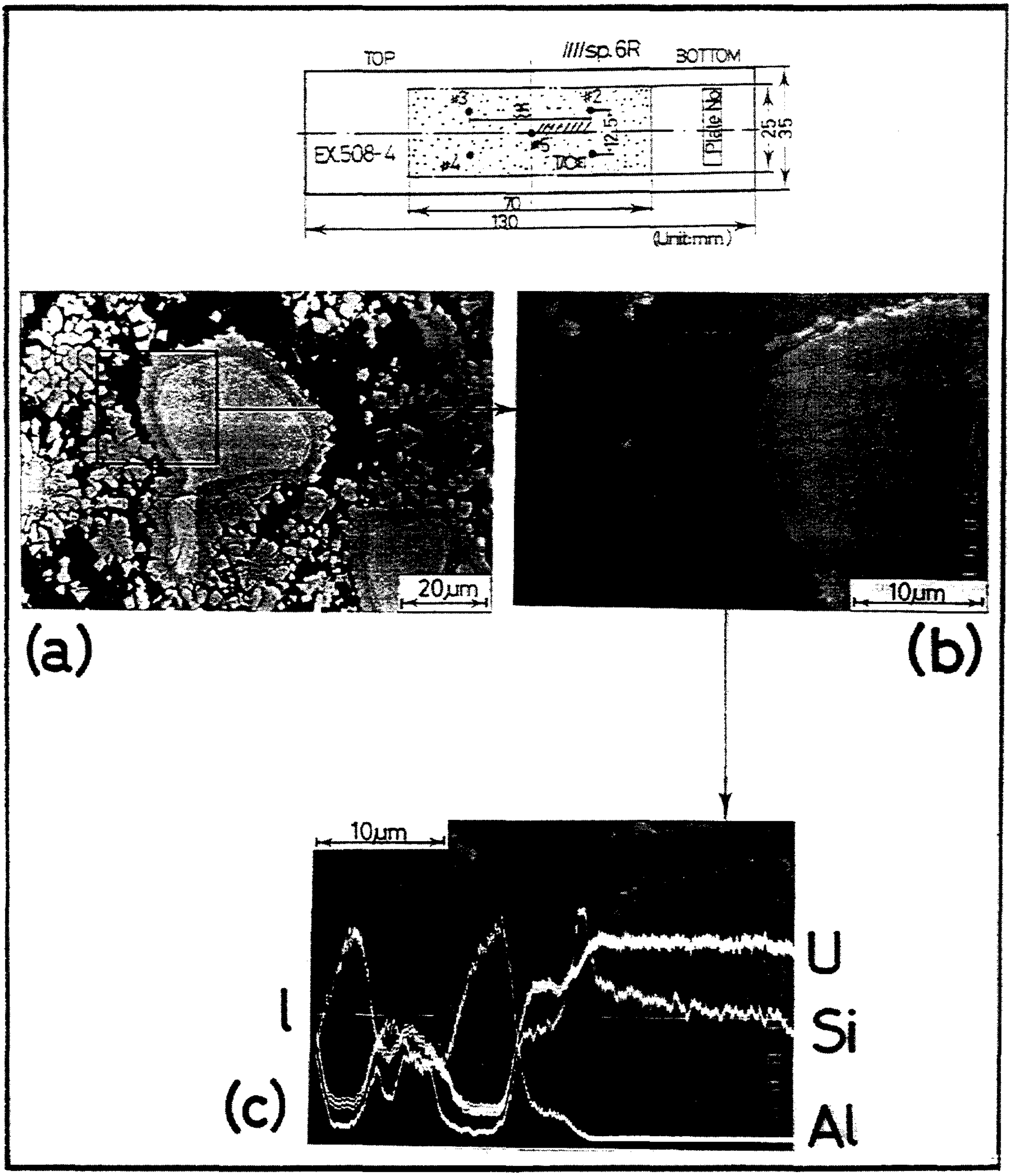


Fig. 7 Maximum Bowing of the Silicide Fuel Mini-plate as a Function of Fuel Temperature


TOP
sp. 6R
BOTTOM
#3
#2
#5
#4
12.5
EX.508-4
Plate No
25
35
70
130
(Unit:mm)
20μm
(a)
10μm
(b)
10μm
(c)
U
Si
Al

CONCLUSIONS

- At fuel temp. below 400℃, Mini-plates kept good dimensional stability

- Beyond 400℃, dimensional stability was degraded, and beyond 640℃, Mini-plates were damaged showing a bowing up to 7㎜

CONCLUSIONS
(continued)

- Despite of the large degradation around 970℃, <u>no fuel fragmentation/mechanical energy generation were observed</u>

- Metal to metal reaction between Al and silicide particles might be occured over 900℃

THE CNS FACILITY AND NEUTRON GUIDE TUBES IN JRR-3M

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ABSTRACT

A cold neutron source facility and five neutron guide tubes are installed in the upgraded JRR-3 in order to provide high quality neutrons for neutron beam experiments. The neutron fluxes and spectra were measured at the end of the neutron guide tubes using the foil activation method and the time-of-flight method. The gain of the cold neutron source is also found from these spectra. The measured neutron fluxes and spectra well agreed with their designed values, and the gain of cold neutron source also agreed with predicted value.

INTRODUCTION

One of the main purposes on the upgrading project of JRR-3 is to improve the performance of the neutron beam experiments using low energy neutrons. Therefore, a cold neutron source (CNS) facility and five neutron guide tubes were installed in the upgraded JRR-3 (JRR-3M). This paper introduces the CNS facility and the neutron guide tubes in JRR-3M and also presents their characteristics obtained since their beginning of operation in 1990.[1),2),3),4),5)]

THE FACILITY DESCRIPTION

The CNS Facility

The CNS facility in JRR-3M is a vertical thermosyphone type using liquid hydrogen at 20K as the moderator for production of the cold neutrons (ORPHEE type). It mainly consists of a hydrogen plant in the reactor building and a helium refrigerator plant in the compressor building which is located about 100m apart from the reactor building. From the safety point of view, the hydrogen plant has double containment structure with a vacuum blanket and is completely immersed in the reactor pool and the sub-pool. The main part of the hydrogen plant consists of a moderator cell, a vacuum chamber and a condensor. Schematic diagram of the CNS is shown in Fig. 1, and the general description of the moderator cell is shown in Fig. 2. The moderator cell which is an flask shape stainless vessel is located at the maximum thermal neutron flux area in the heavy water reflector. The designed specifications of the CNS are shown in Table 1.

The Neutron Guide Tubes

Two thermal neutron guide tubes and three cold neutron guide tubes take out thermal/cold neutron beams to the beam hall. The thermal neutron guide tubes, T1 and T2 guide tube, are designed to have a characteristic neutron wave length of 2Å, and the cold neutron guide tubes of 4Å (C1 and C2 guide tube) and 6Å (C3 guide tube). The main design parameters of the neutron guide tubes are shown in Table 2. The neutron mirrors are made of nickel (Ni) sputtered borosilicated glasses with the Ni layer thickness of approximately 2000Å. The neutron guide tubes consist of short straight units with a length of 85cm, that is short enough to make good polygonal approximation and also long enough to lessen the number of conjunctions. The guide tube units are connected to each other and inside the guide tubes are evacuated to avoid the neutron losses by air. Two thermal neutron guide tubes have a length of approximately 60m. One cold neutron guide tube has a length of 51m and the other two have a length of 30m. The layout of the neutron guide tubes in JRR-3M is shown in Fig. 3.

CHARACTERISTICS OF THE CNS AND THE NEUTRON GUIDE TUBES

The gain of the CNS is difined as the ratio of the neutron spectra when the CNS is operated and not operated. The neutron spectra measured by means of the time-of-flight method at beam port C2-3 are shown in Fig. 4 of both case that the CNS was operated and not. Figure 5 shows CNS gain depends on the neutron wavelength. For example, the gain of the CNS in JRR-3M is almost

8 for neutrons whose wavelength of 4Å and 20 for 6Å. The measured neutron spectra and the gain of CNS show good agreement with the designed values. The gain is large enough to satisfy the primary goal of the design and the construction of the CNS in JRR-3M.

Figure 4 also shows that the neutrons whose wavelength are much shorter than the characteristic wavelength of the guide tube are rejected. The thermal and cold neutron flux values measured at the end of the neutron guide tubes by the gold foil activation method are shown in Table 3. Table 4 shows the measured cross sectional flux distributions at the end of the neutron guide tubes.

CONCLUDING REMARKS

The neutron fluxes and spectra of the neutron guide tubes in JRR-3M were measured. The gain of the CNS is also found from these spectra. It is shown that the neutron fluxes and uniformity of the flux distribution at the end of neutron guide tubes are good enough for the neutron beam experiments, and they give good experimantal conditions for beam ports. The CNS of JRR-3M gives a high cold neutron flux so that it could expand not only neutron scattering research but also new research fields such as cold neutron optics, ultra/very cold neutron source development, cold neutron radiography and the prompt γ-ray analysis.

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1) Y. KAWABATA et al., "Transmission Efficiency of Neutron Guide Tube with Alignment Errors", J. Nucl. Sci. Technol., 27[5], 406 (1990)
2) Y. KAWABATA et al., "Construction of Neutron Guide Tubes in Upgradedi JRR-3", J. Nucl. Sci. Technol., 27[12], 1138 (1990)
3) M. SUZUKI et al., JAERI-M 86-037, (in Japanese) (1986)
4) T. KUMAI et al., "The Cold Neutron Source Facility of The JRR-3M", Proceedings of The 3rd Asian Symposium on Research Reactors (1991)
5) Y. KAWABATA et al., "Neutron Fluxes and Spectre of The Guide Tubes in The Upgraded JRR-3", Proceedings of The International Conference on Neutron Scattering (1991)

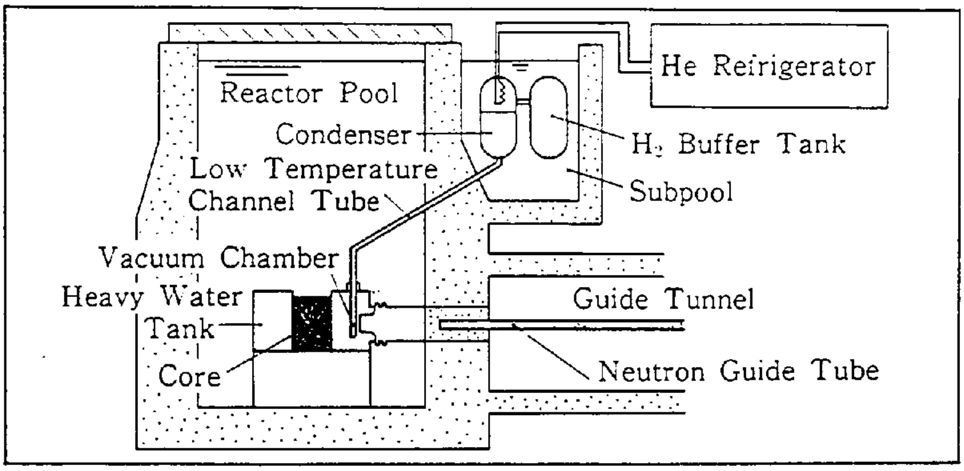


Fig. 1 The Schematic Diagram of JRR-3M CNS Facility

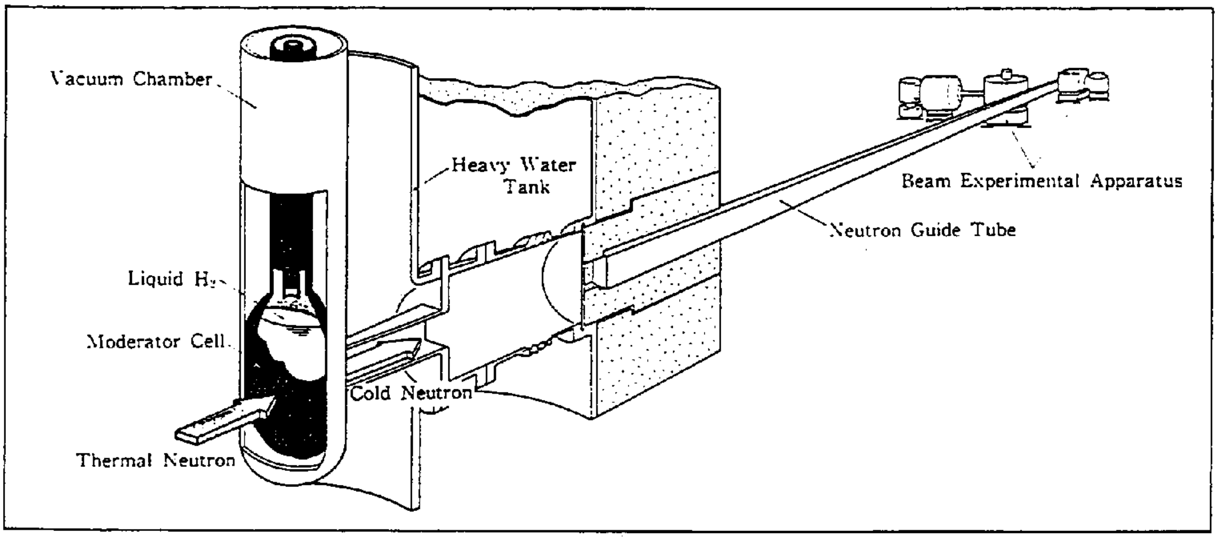


Fig. 2 General Description of Moderator Cell of JRR-3M CNS Facility

Table 1 Design Parameters of JRR-3M CNS Facility

TYPE	:	Vertical Thermosyphone type
MODERATOR	:	Liquid Hydrogen, 20K
COOLANT	:	Helium Gas
MODERATOR CELL	:	Flask Shape, $200mm^H \times 130mm^W \times 50mm^T$, 0.8 ℓ
		Stainless Steel, $0.8mm^t$
VACUUM CHAMBER	:	Φ154mm, $8mm^t$, Stainless Steel

Table 2 The Main Design Parameters of Neutron Guide Tubes in JRR-3M

Guide Tube No.	Characteristic Wavelength (Å)	Beam Cross Section (mm^2)	Radius of Curvature (m)	Total Length (m)
T1	2	20×200	3337	59.9
T2	2	20×200	3337	59.0
C1	4	20×120	834	30.8
C2	4	20×120	834	51.1
C3	6	20×120	371	31.4

Guide Tube Unit : natural Ni sputtered Borosilicated Glass
200mm×20mm×850mm (thermal)
120mm×20mm×850mm (cold)
Ni Layer Thickness, ave. 2000Å
Surface Roughness, ave. 80Å

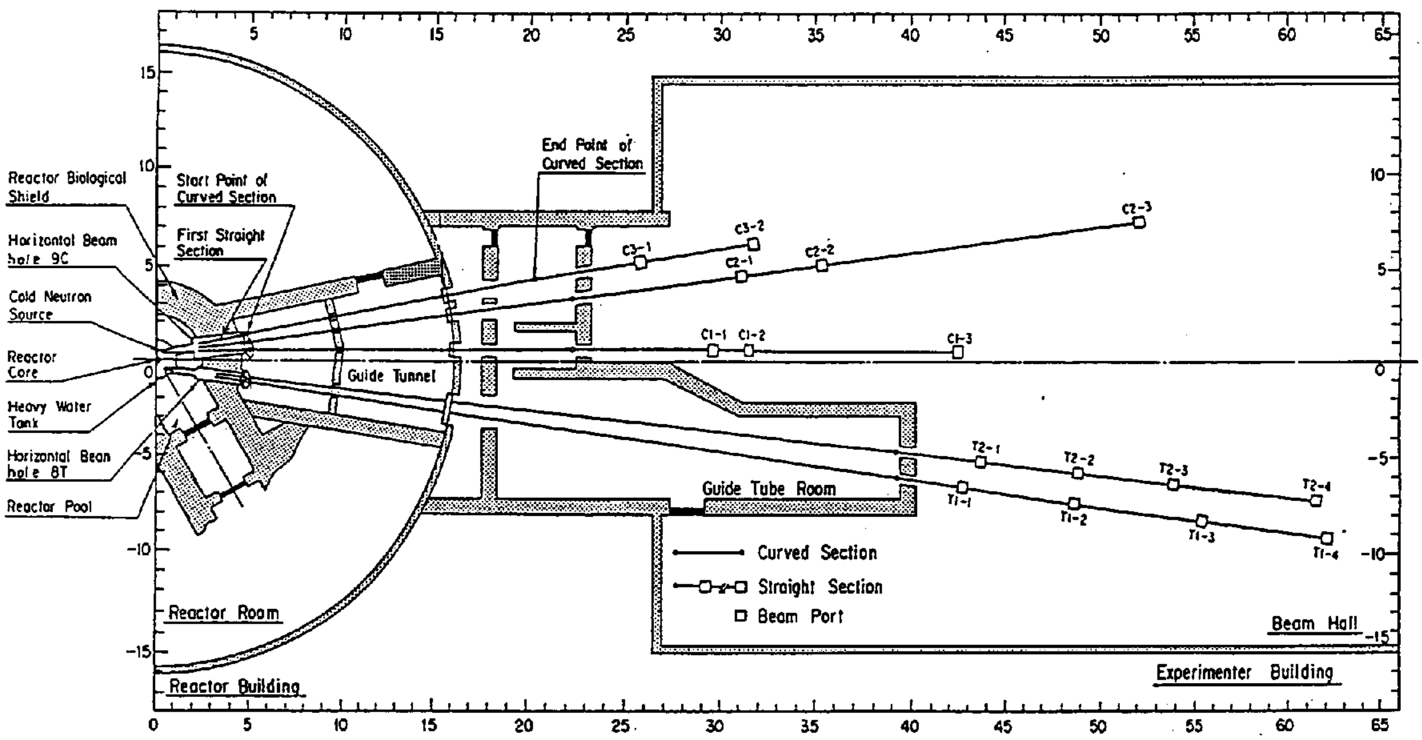


Fig. 3 Layout of the Neutron Guide Tubes in JRR-3M

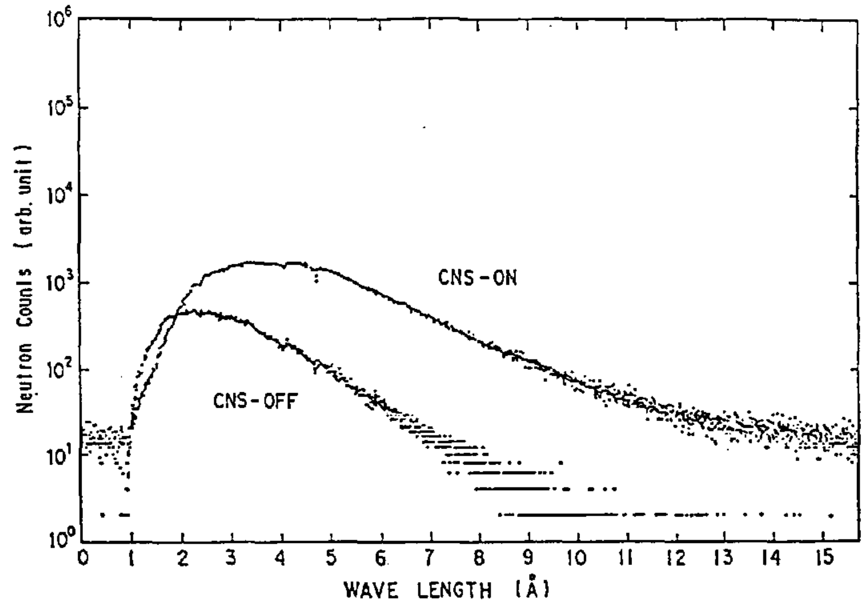


Fig. 4 Comparison of Neutron Spectra at the End of the Cold Neutron Guide tube C2 when the CNS was Operated (CNS-ON) and the CNS was not Operated (CNS-OFF)

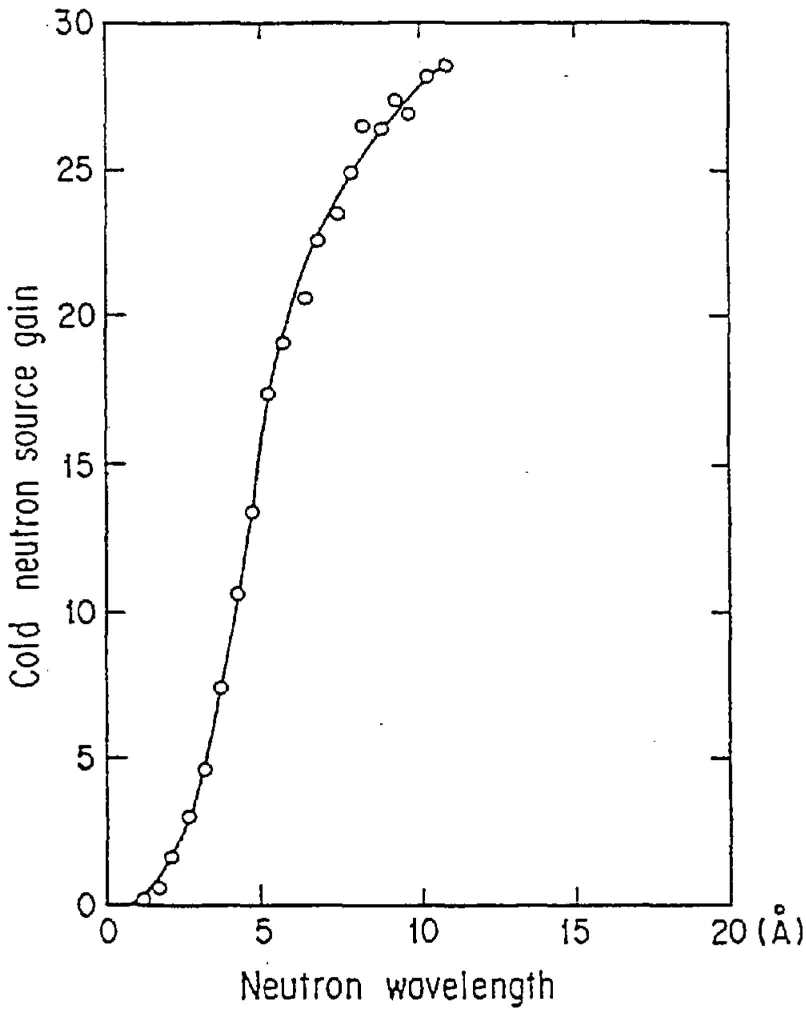


Fig. 5 The Cold Neutron Source Gain

Table 3　Measured and Calculated Value of Thermal Neutron Flux at the Entrance and Exit of Neutron Guide Tubes (NGTs)

NGT ID No.	Φ [Entrance] ($n \cdot cm^{-2} \cdot sec$)	Φ [Exit] ($n \cdot cm^{-2} \cdot sec$)	E/E Ratio
C − 1	1.5×10^9 2.1×10^9	2.0×10^8 3.3×10^8	0.133 0.157
C − 2	1.4×10^9 2.1×10^9	2.0×10^8 2.8×10^8	0.143 0.133
C − 3	1.5×10^9 2.1×10^9	1.4×10^8 2.3×10^8	0.093 0.109
T − 1	7.9×10^9 8.3×10^9	1.2×10^8 2.1×10^8	0.015 0.025
T − 2	6.8×10^9 8.3×10^9	1.2×10^8 2.1×10^8	0.018 0.025

・Upper:Measured, Lower:Calculated
・E/E Ratio = Φ [Exit] / Φ [Entrance]

Table 4　Cross Sectional Neutron Flux Distributions at the Exits of the Neutron Guide Tubes, C2-3 and T2-4, measured by Gold Foil Activation Method

T1-4

	1.1	1.3	1.1	
	1.1	1.1	1.1	
<I>	1.1	1.3	1.2	<O>
	1.2	1.2	1.2	
	1.2	1.3	1.2	

C2-3

	1.8	1.9	1.7	
<I>	2.1	1.9	1.9	<O>
	1.9	1.9	1.8	

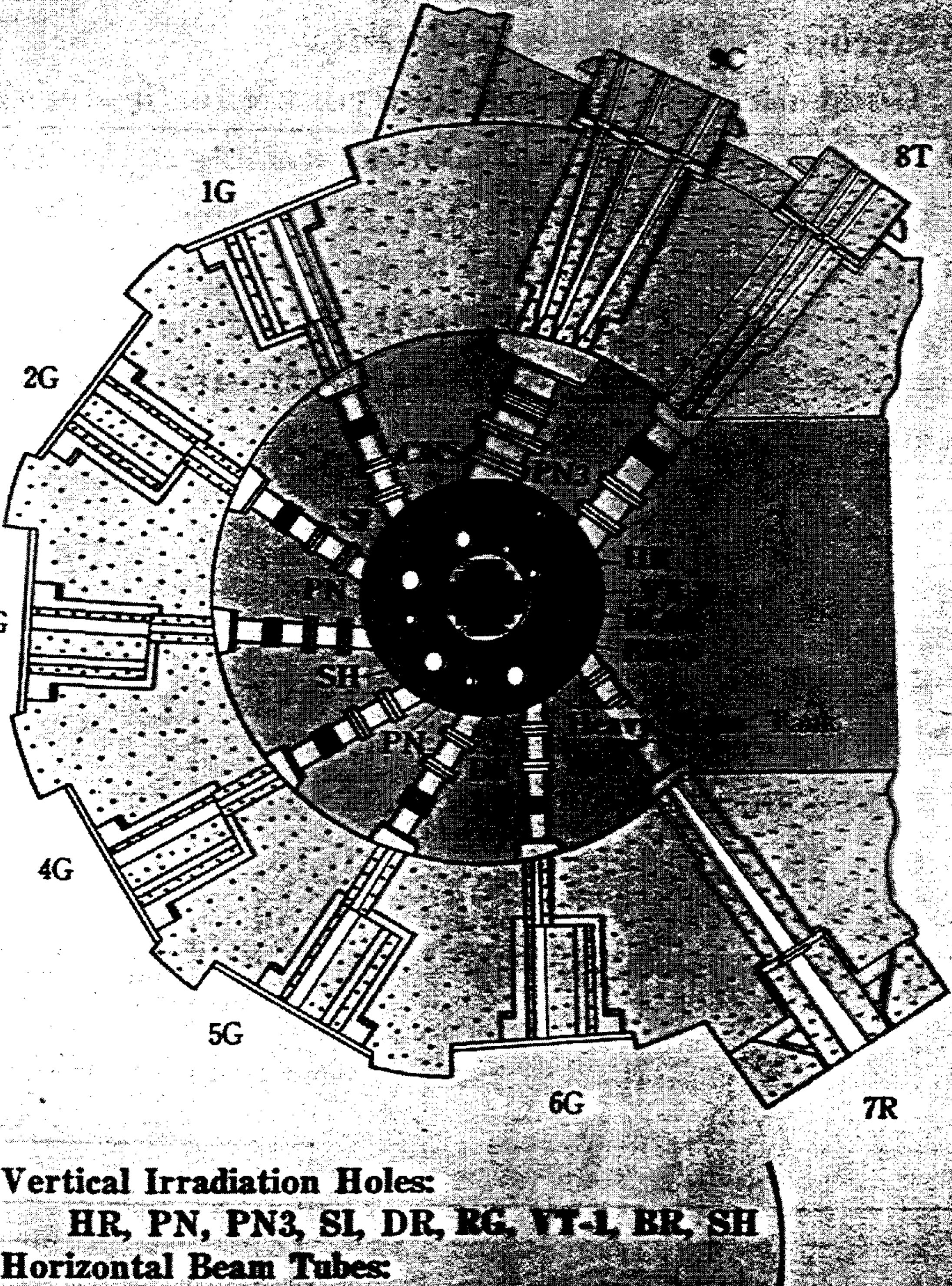
unit : $\times 10^8$ $n \cdot cm^{-2} \cdot sec^{-1}$

<I> : Inner side of curvature
<O> : Outer side of curvature

THE CNS FACILITY AND NEUTRON GUIDE TUBES IN JRR-3M

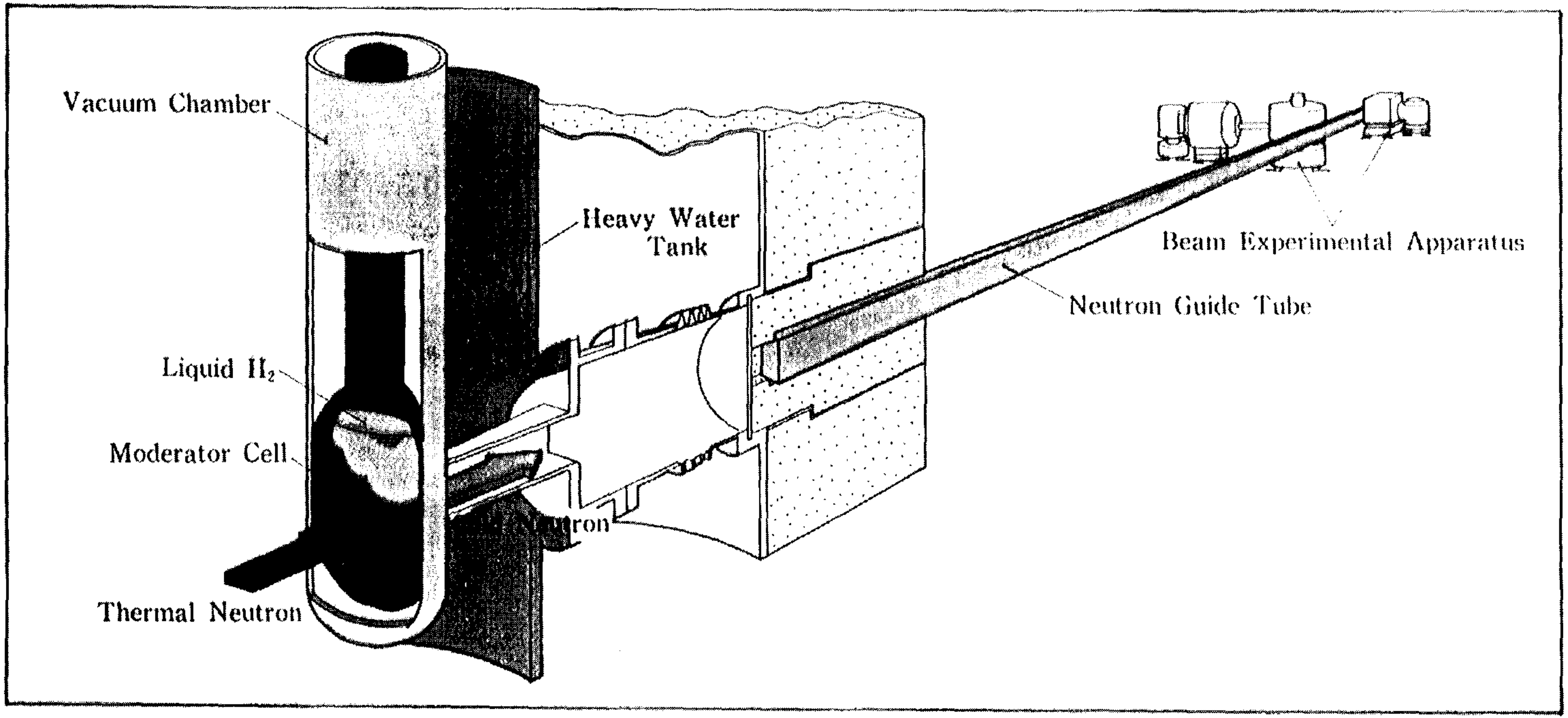
S. MATSUURA, E. SHIRAI AND M. TAKAYANAGI

Tokai Research Establishment
Japan Atomic Energy Research Institute

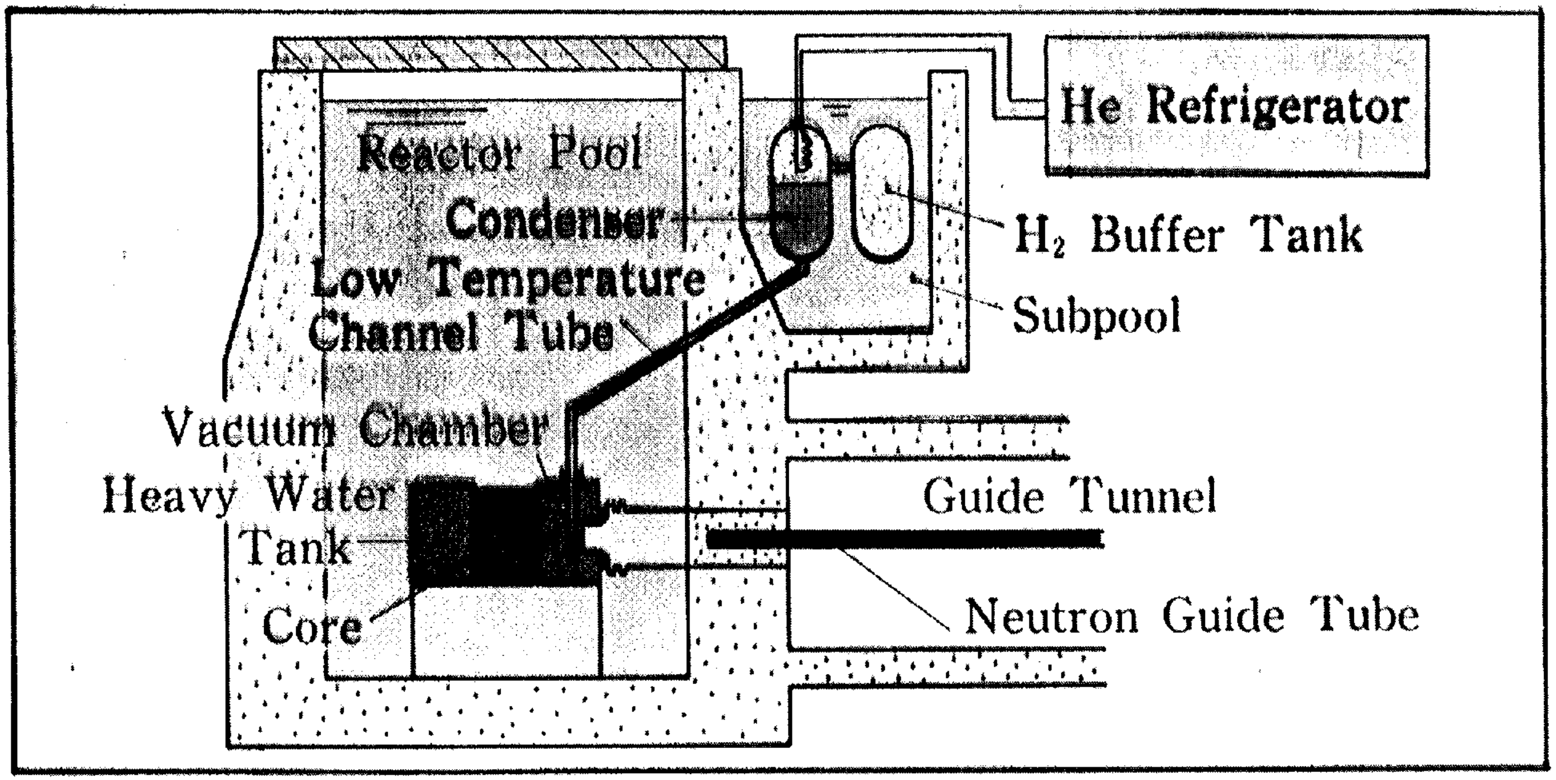


Vertical Irradiation Holes:
HR, PN, PN3, SI, DR, RG, VT-1, BR, SH
Horizontal Beam Tubes:
1G~6G, 7R, 8T, 9C

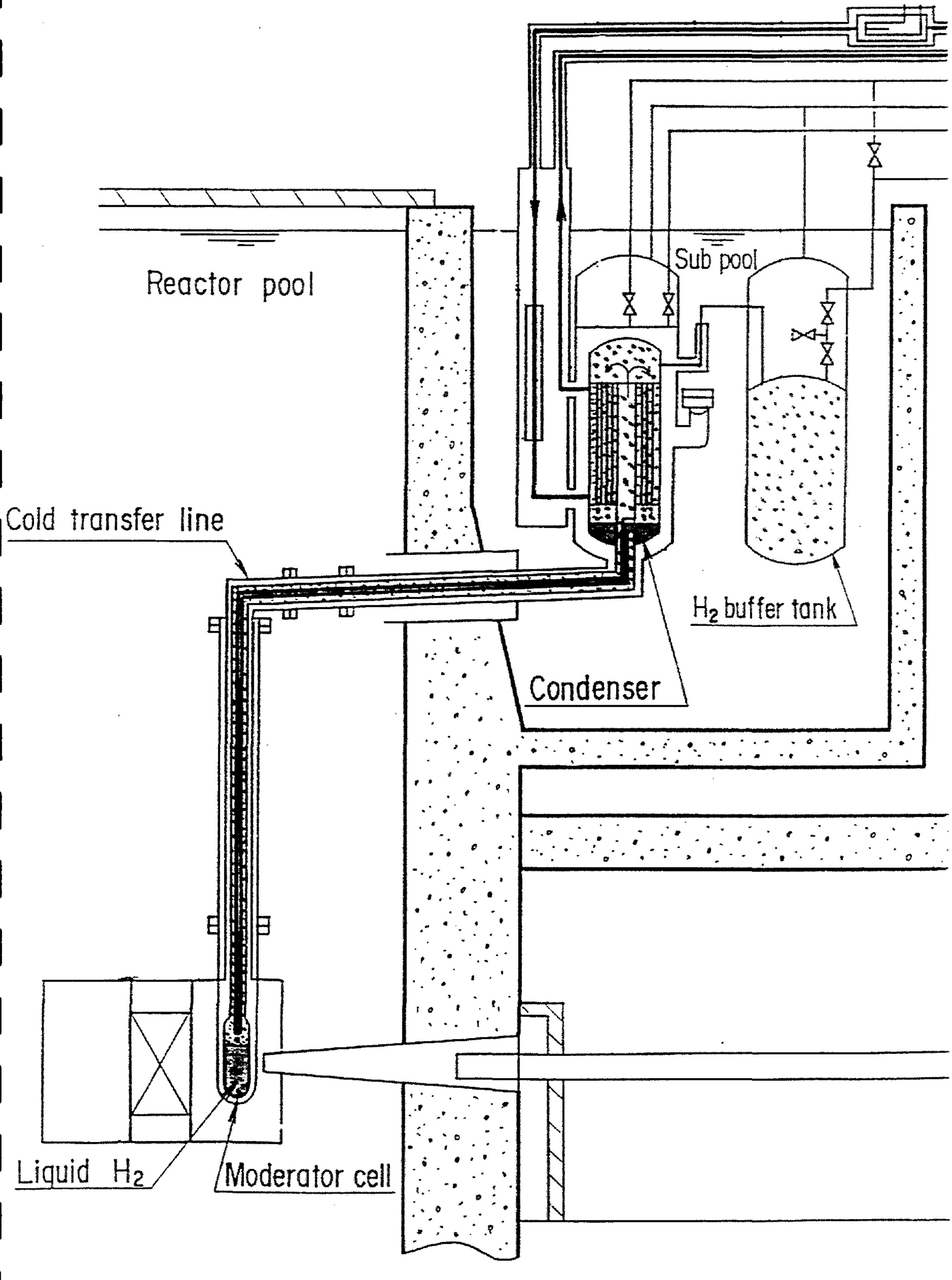
Arrangement of Experimental Holes



CNS and Neutron Guide Tube



Schematic Diagram

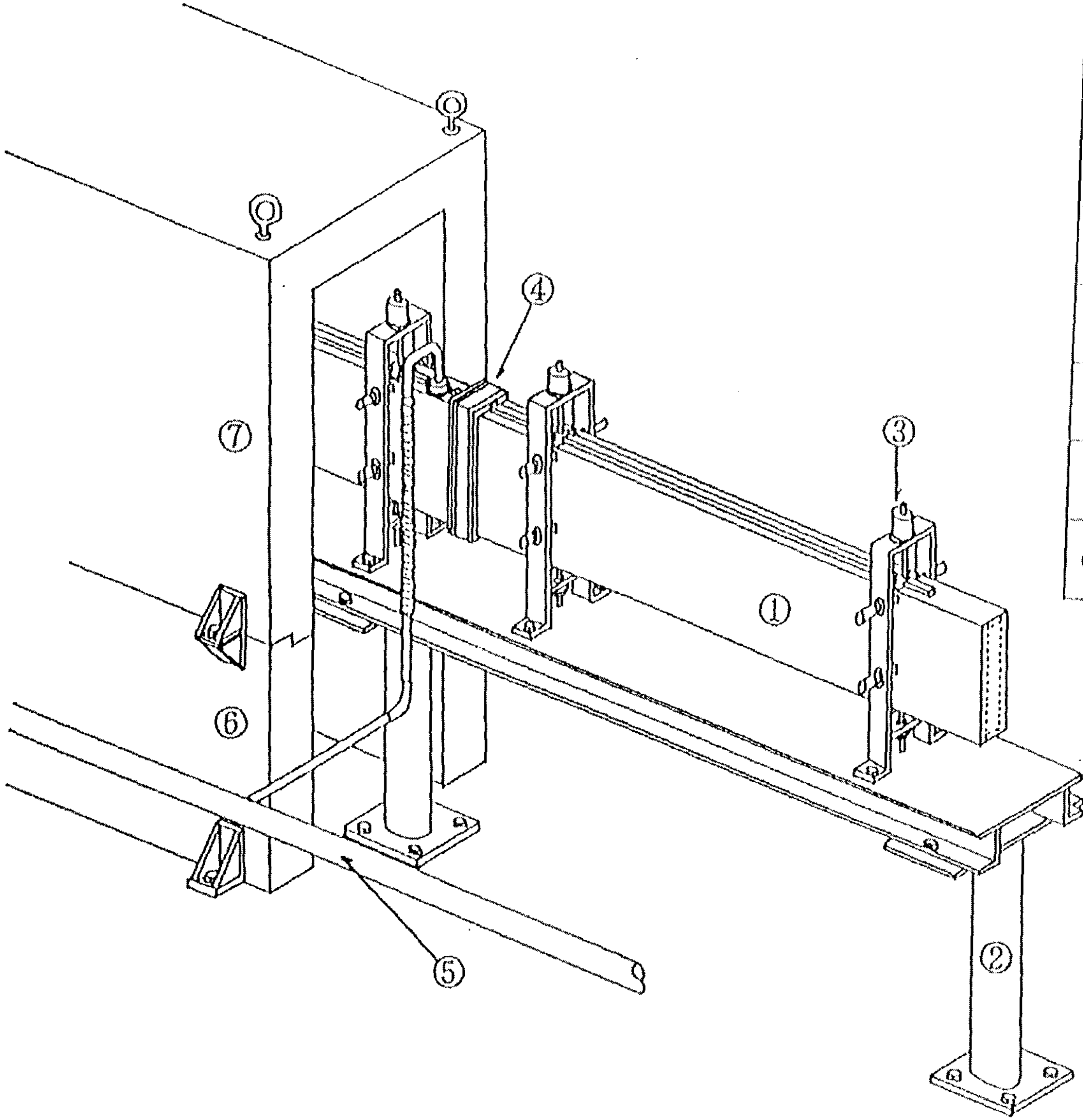


THERMOSYPHON

JRR-3M NEUTRON GUIDE TUBE

GUIDE TUBE UNIT

- nat. Ni sputtered Borosilicide glass
- 200mm × 20mm × 850mm (thermal)
- 120mm × 20mm × 850mm (cold)
- Ni Layer Thickness : ave. 2000Å
- Surface Roughness : ave. 80Å



①	Guide tube unit
②	Support
③	Position adjuster
④	Connection seal
⑤	Vacuum line
⑥	Shielding
⑦	Shielding

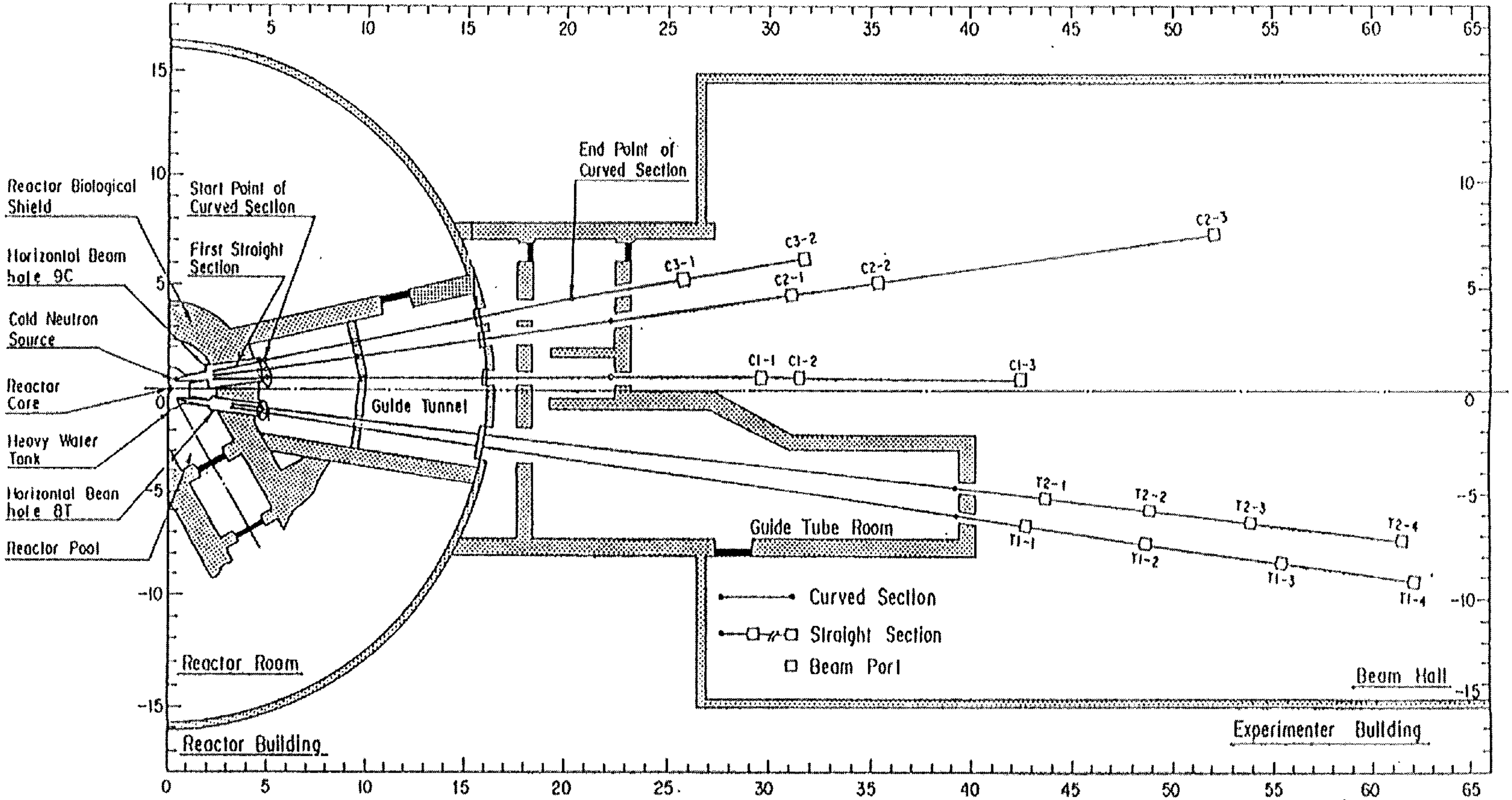


Fig. 3 Layout of the Neutron Guide Tubes in JRR-3M

Table 2 The Main Design Parameters of Neutron Guide Tubes in JRR-3M

Guide Tube No.	Characteristic Wavelength (Å)	Beam Cross Section (mm^2)	Radius of Curvature (m)	Total Length (m)
T1	2	20 × 200	3337	59.9
T2	2	20 × 200	3337	59.0
C1	4	20 × 120	834	30.8
C2	4	20 × 120	834	51.1
C3	6	20 × 120	371	31.4

Table 3 Measured and Calculated Value of Thermal Neutron Flux at the Entrance and Exit of Neutron Guide Tubes (NGTs)

NGT ID No.	Φ [Entrance] ($n \cdot cm^{-2} \cdot sec$)	Φ [Exit] ($n \cdot cm^{-2} \cdot sec$)	E/E Ratio
C − 1	1.5×10^9 2.1×10^9	2.0×10^8 3.3×10^8	0.133 0.157
C − 2	1.4×10^9 2.1×10^9	2.0×10^8 2.8×10^8	0.143 0.133
C − 3	1.5×10^9 2.1×10^9	1.4×10^8 2.3×10^8	0.093 0.109
T − 1	7.9×10^9 8.3×10^9	1.2×10^8 2.1×10^8	0.015 0.025
T − 2	6.8×10^9 8.3×10^9	1.2×10^8 2.1×10^8	0.018 0.025

・Upper:Measured, Lower:Calculated
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Table 4 Cross Sectional Neutron Flux Distributions at the Exits of the Neutron Guide Tubes, C2-3 and T2-4, measured by Gold Foil Activation Method

T1-4

	1.1	1.3	1.1	
	1.1	1.1	1.1	
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	1.2	1.2	1.2	
	1.2	1.3	1.2	

C2-3

	1.8	1.9	1.7	
<I>	2.1	1.9	1.9	<O>
	1.9	1.9	1.8	

unit : $\times 10^8$ $n \cdot cm^{-2} \cdot sec^{-1}$

<I> : Inner side of curvature
<O> : Outer side of curvature

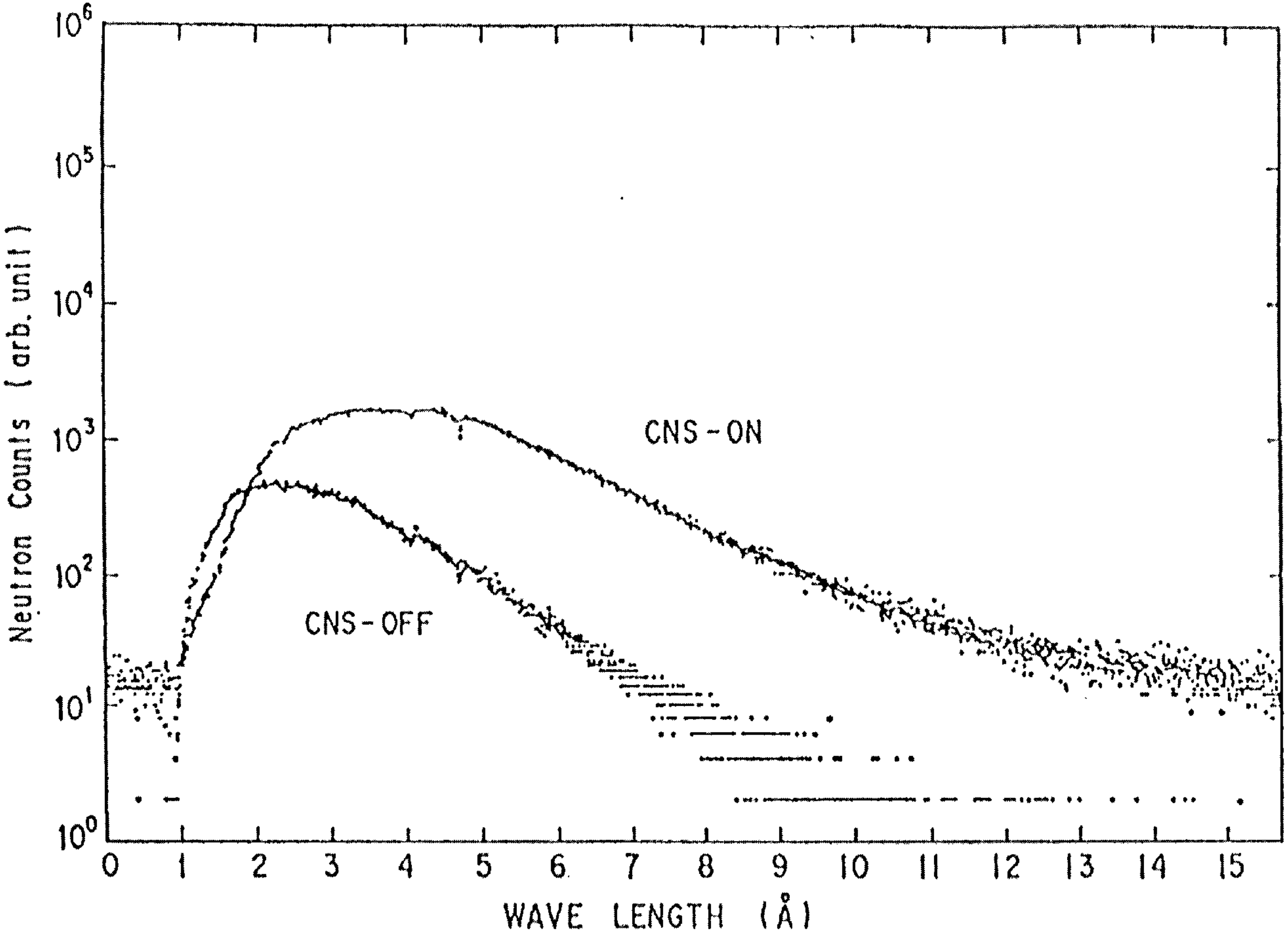


Fig. 4 Comparison of Neutron Spectra at the End of the Cold Neutron Guide tube C2 when the CNS was Operated (CNS-ON) and the CNS was not Operated (CNS-OFF)

JRR-3M COLD NEUTRON SOURCE

TYPE : VERTICAL THERMOSYPHONE TYPE

MODERATOR: LIQUID HYDROGEN, 20K

COOLANT : HELIUM GAS

VESSEL : FLASK SHAPE, 0.8ℓ

200mmH × 130mmW × 50mmT

STAINLESS STEEL, 0.8mmt

VACUUM CHAMBER: 154mmΦ×8mmt

STAINLESS STEEL

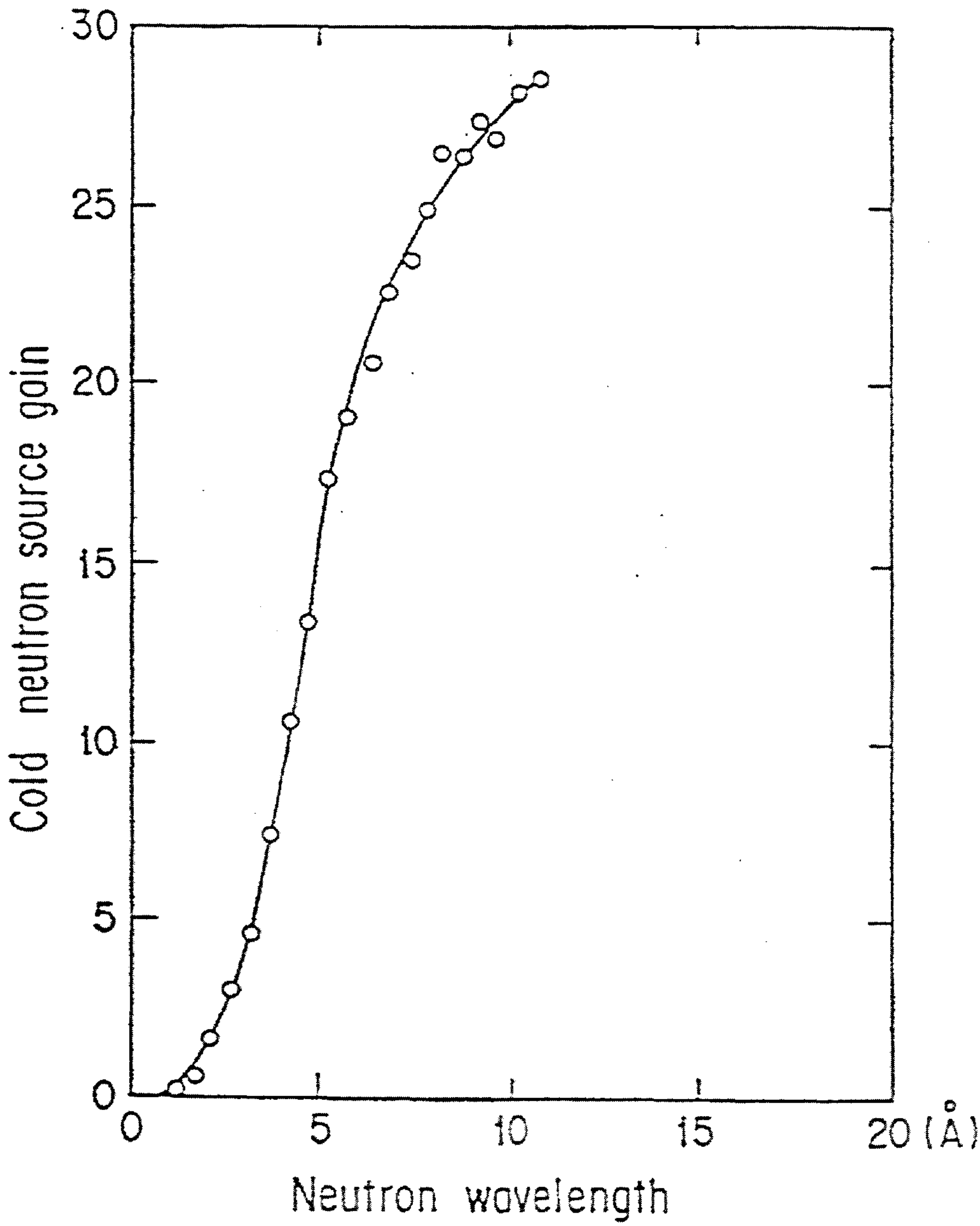
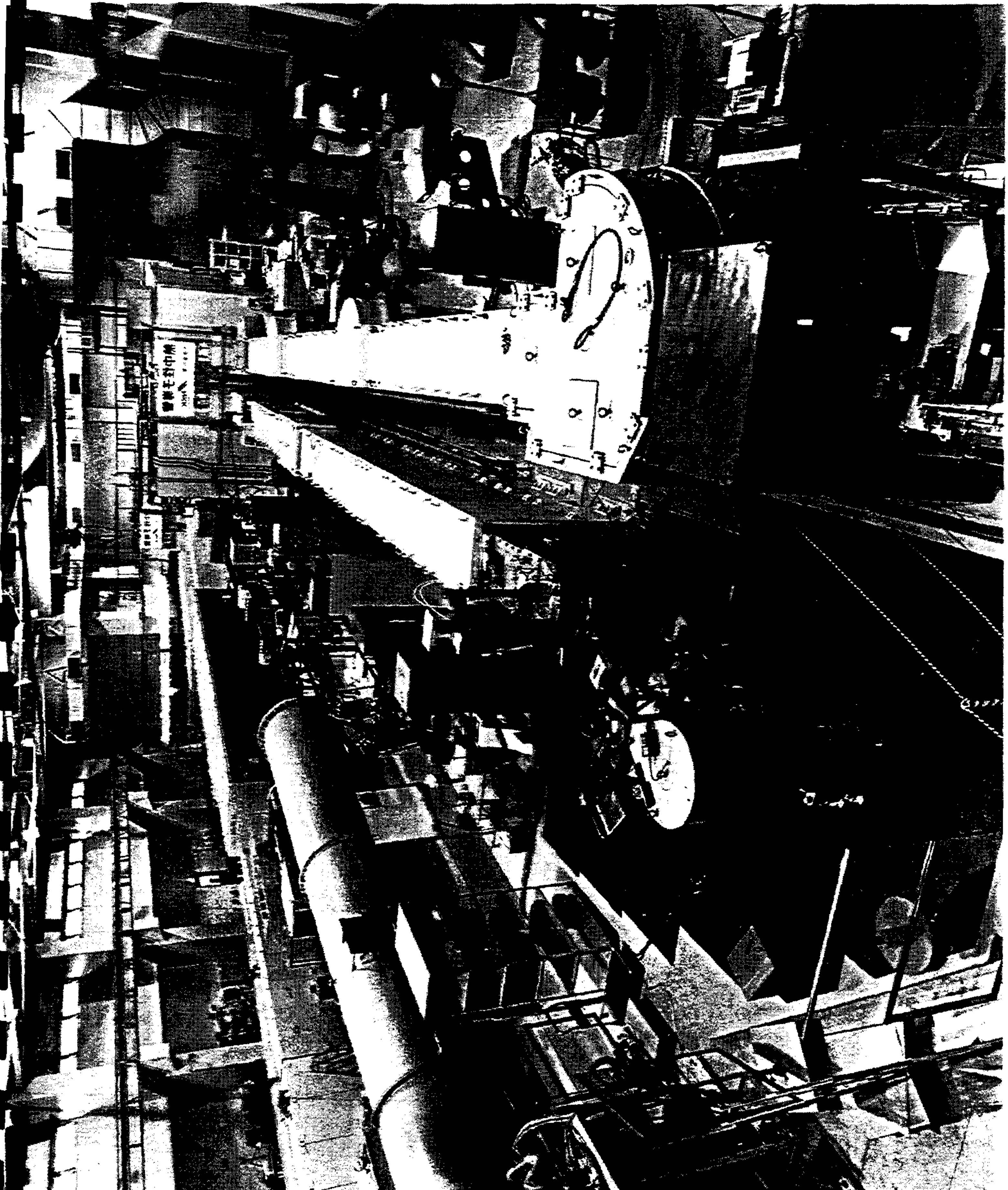


Fig. 5 The Cold Neutron Source Gain

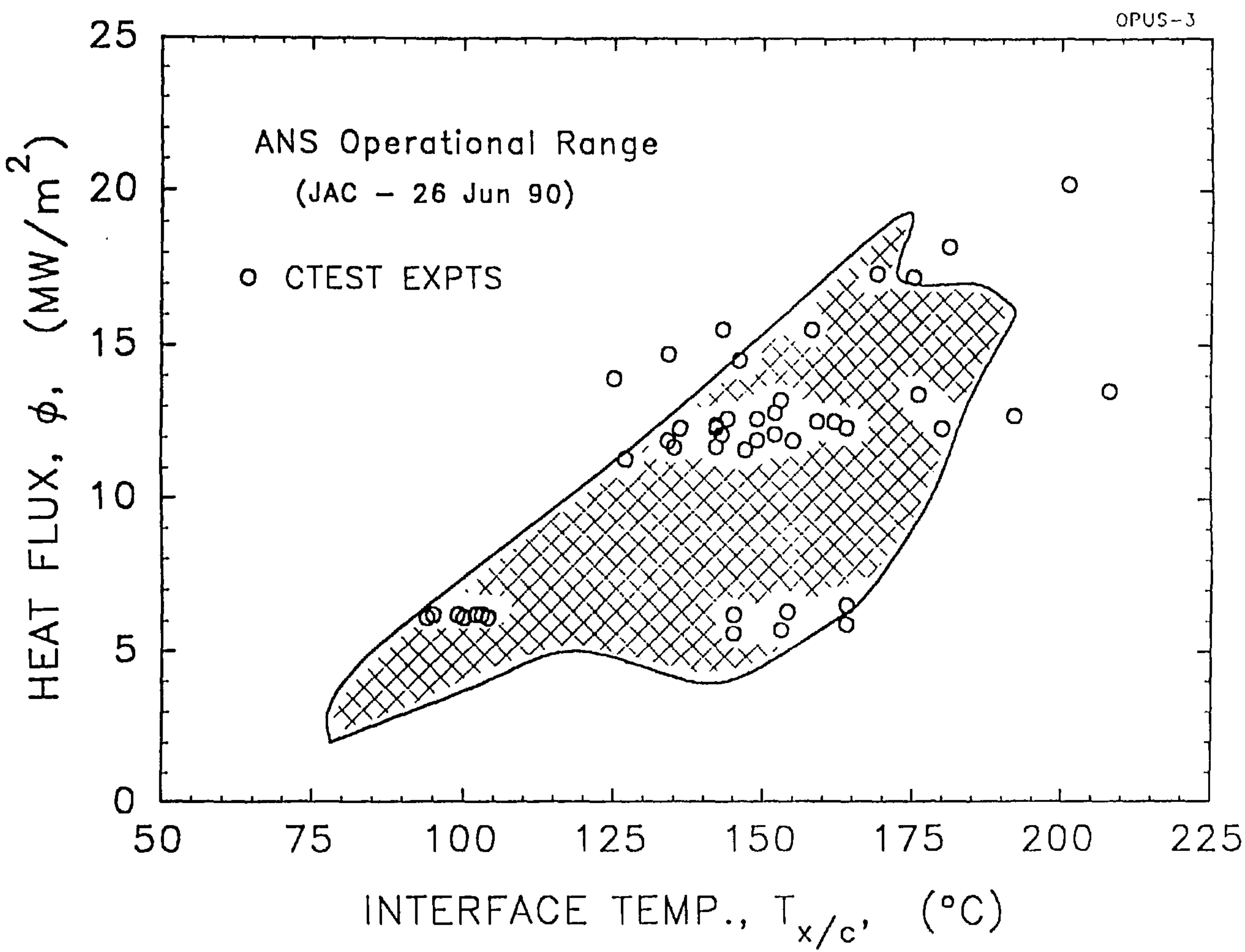


PROGRESS REPORT ON R&D RESULTS FROM THE ADVANCED NEUTRON SOURCE

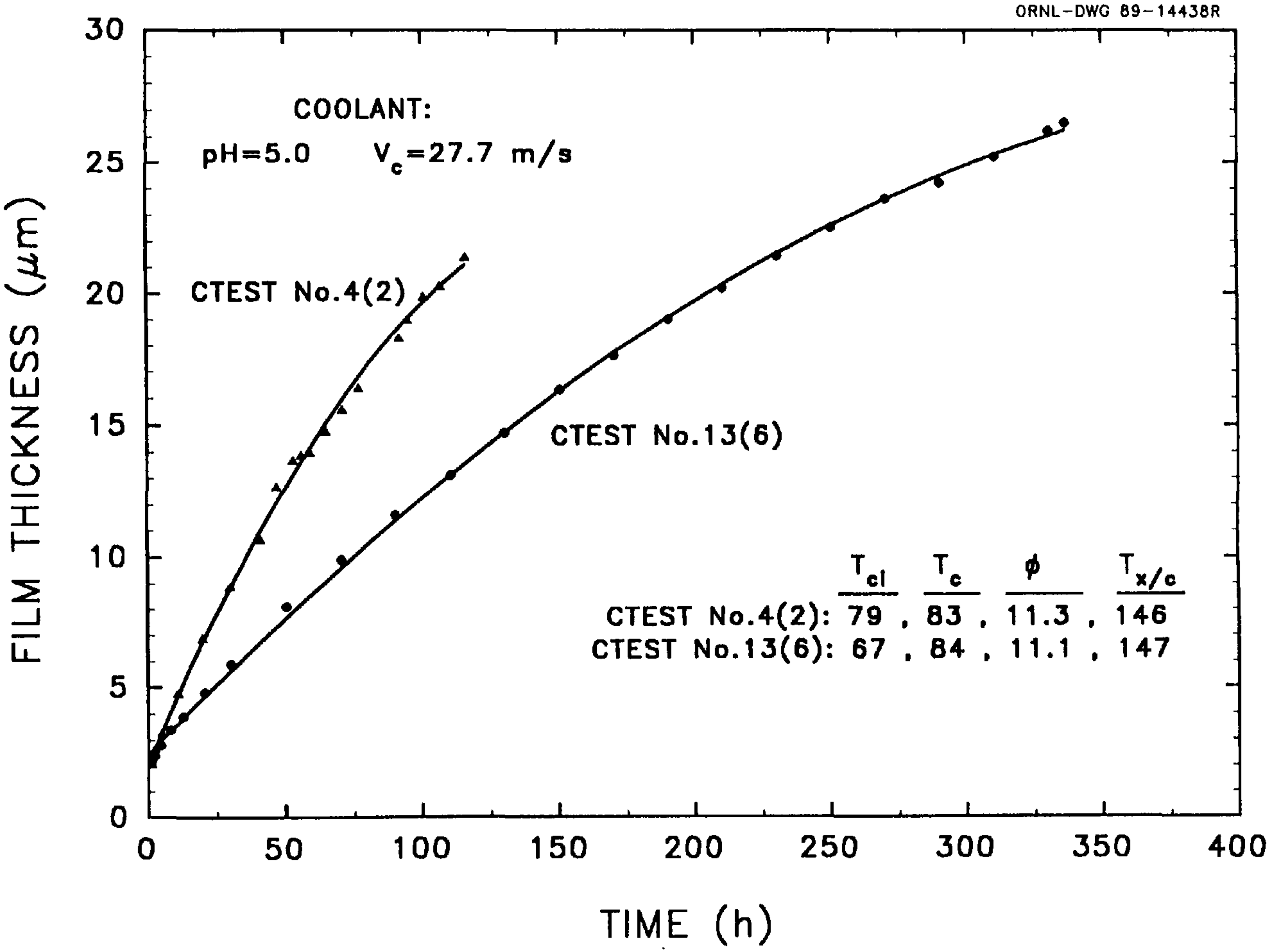
C. D. West
Oak Ridge National Laboratory

May 19, 1992

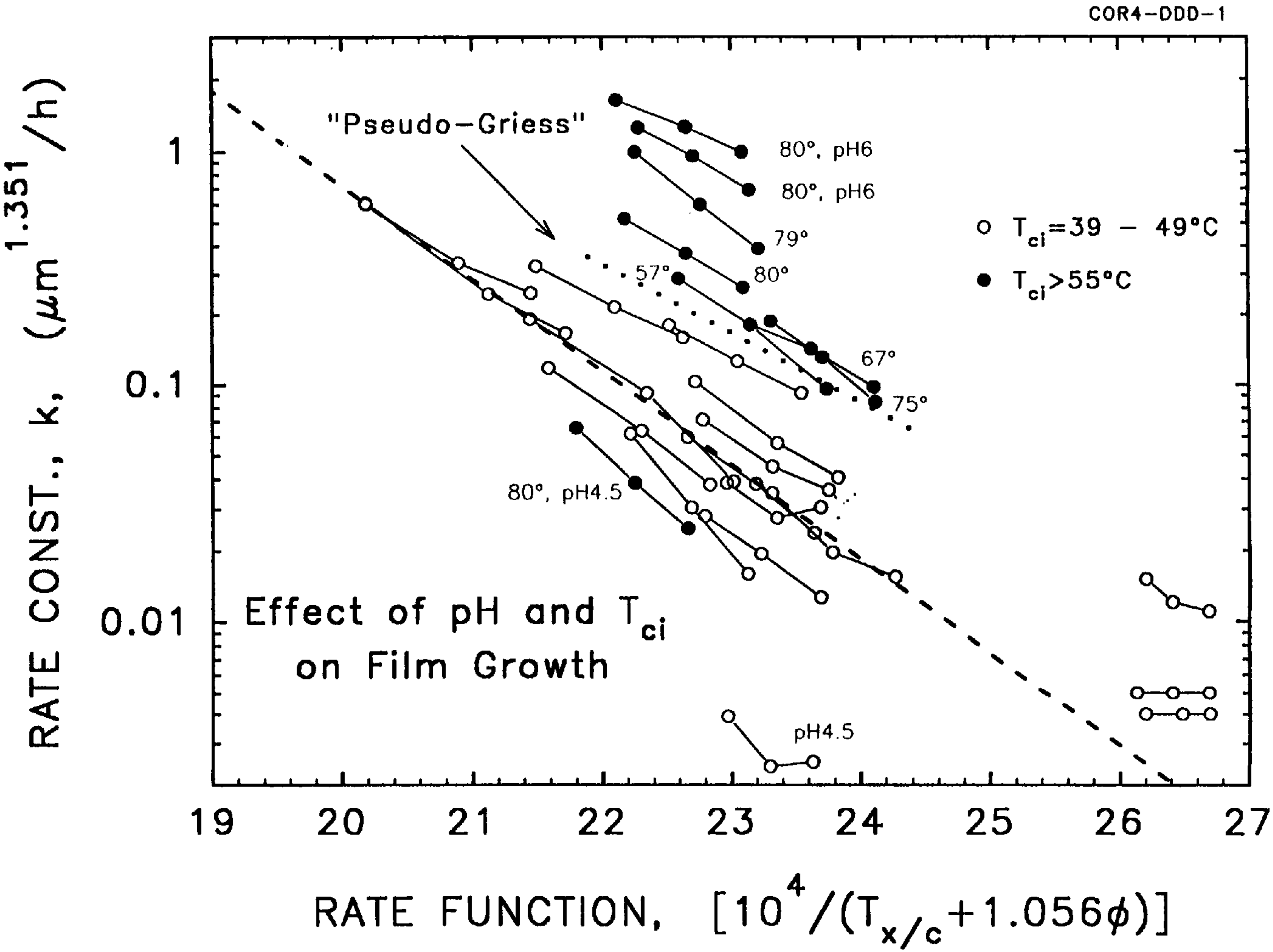
Contents	Task Leader	Fax No.
Oxide Formation	R.E. Pawel	615/574-5118
U_3Si_2 Fuel Performance	G.L. Copeland	615/576-3894
Aluminum Irradiation Properties & Code Case	G.T. Yahr	615/574-0740
Fuel Plate Hydraulic Stability	G.T. Yahr	615/574-0740
Thermal-Hydraulic Test Loop	D.K. Felde	615/574-2032
Cold Source Design Concept	A.T. Lucas	615-574-2032

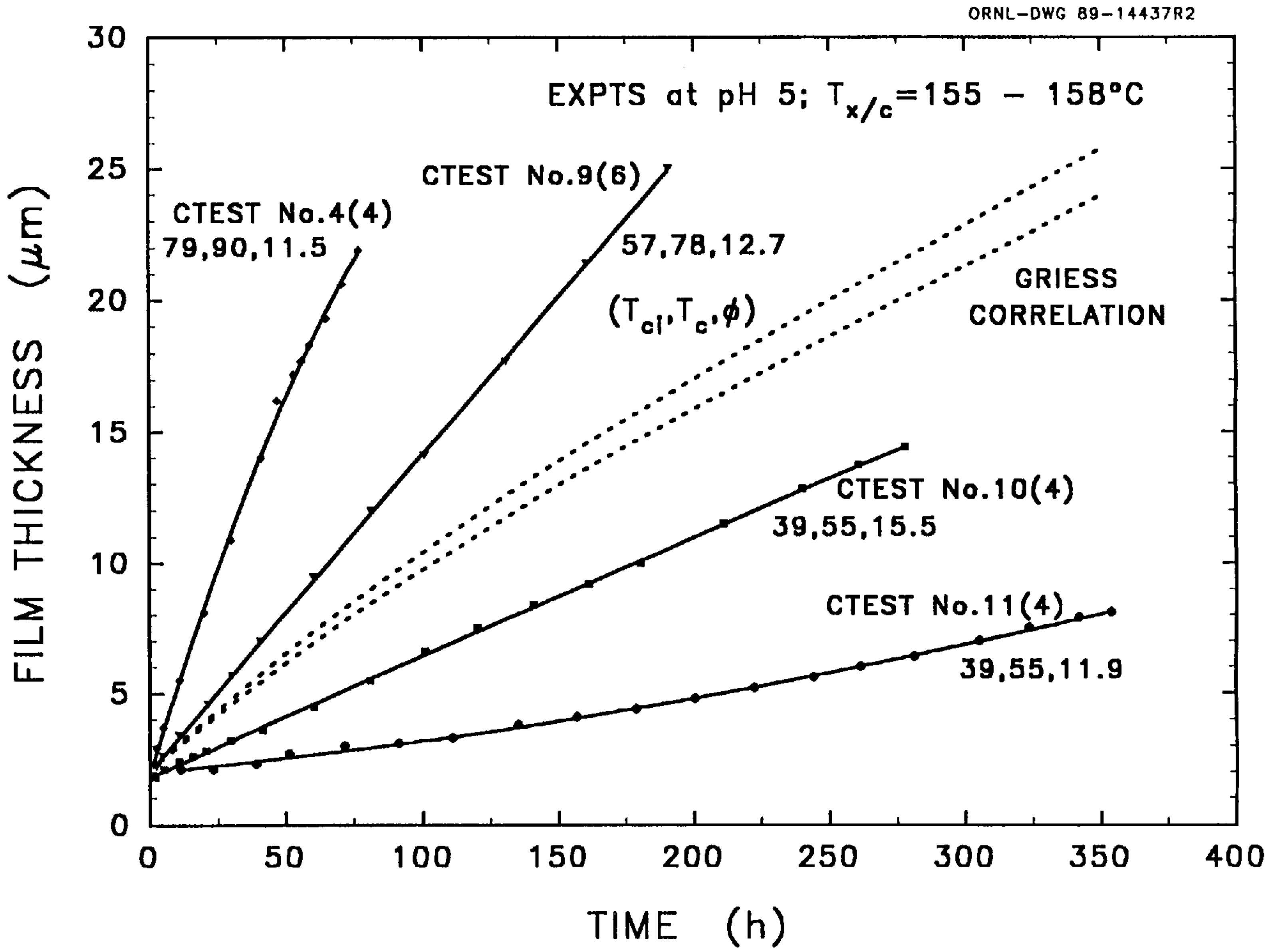


OPUS-3
ANS Operational Range
(JAC - 26 Jun 90)
o CTEST EXPTS
HEAT FLUX, φ, (MW/m²)
INTERFACE TEMP., $T_{x/c}$, (°C)
0
5
10
15
20
25
50
75
100
125
150
175
200
225



Coolant inlet temperature/loop water temperature is an independent variable in determining oxide growth rate


COR4-DDD-1
"Pseudo-Griess"
80°, pH6
80°, pH6
79°
80°
57°
67°
75°
80°, pH4.5
pH4.5
T_{ci}=39 – 49°C
T_{ci}>55°C
Effect of pH and T_{ci} on Film Growth
RATE CONST., k, ($\mu m^{1.351}$/h)
RATE FUNCTION, [$10^4/(T_{x/c}+1.056\phi)$]



Oxide growth rate with less coolant inlet temperature is slower than Griess predictions, even for high heat flux conditions

RESULTS FROM FIRST FUEL IRRADIATION TEST IN HFIR

- Minimal fuel-Al interdiffusion indicating poor bonding between fuel and aluminum particles

  - therefore, fuel temperatures may have been even higher than the calculated 425° C maximum

- Temperature gradient across fuel particles due to high power density resulted in variable bubble size across fuel particles

- Strong temperature effect on bubble morphology in U_3Si^2

  - small, stable bubbles in cooler U_3Si_2 confirmed
  - consistent with our improved models of fuel behavior

- New model of fuel behavior includes radiation induced recrystallization to ultra-fine grain size

  - predicts/explains experimentally-observed bubble sizes

THE FIRST TWO OF A SERIES OF IRRADIATION CAPSULES HAVE BEEN INSERTED IN HFIR TO PROVIDE DATA ON ALUMINUM 6061 BEHAVIOR FOR DESIGN OF ANS

- Occupies 4 HFIR target positions

- Fracture toughness, tensile specimens, and specimens for microstructural examination

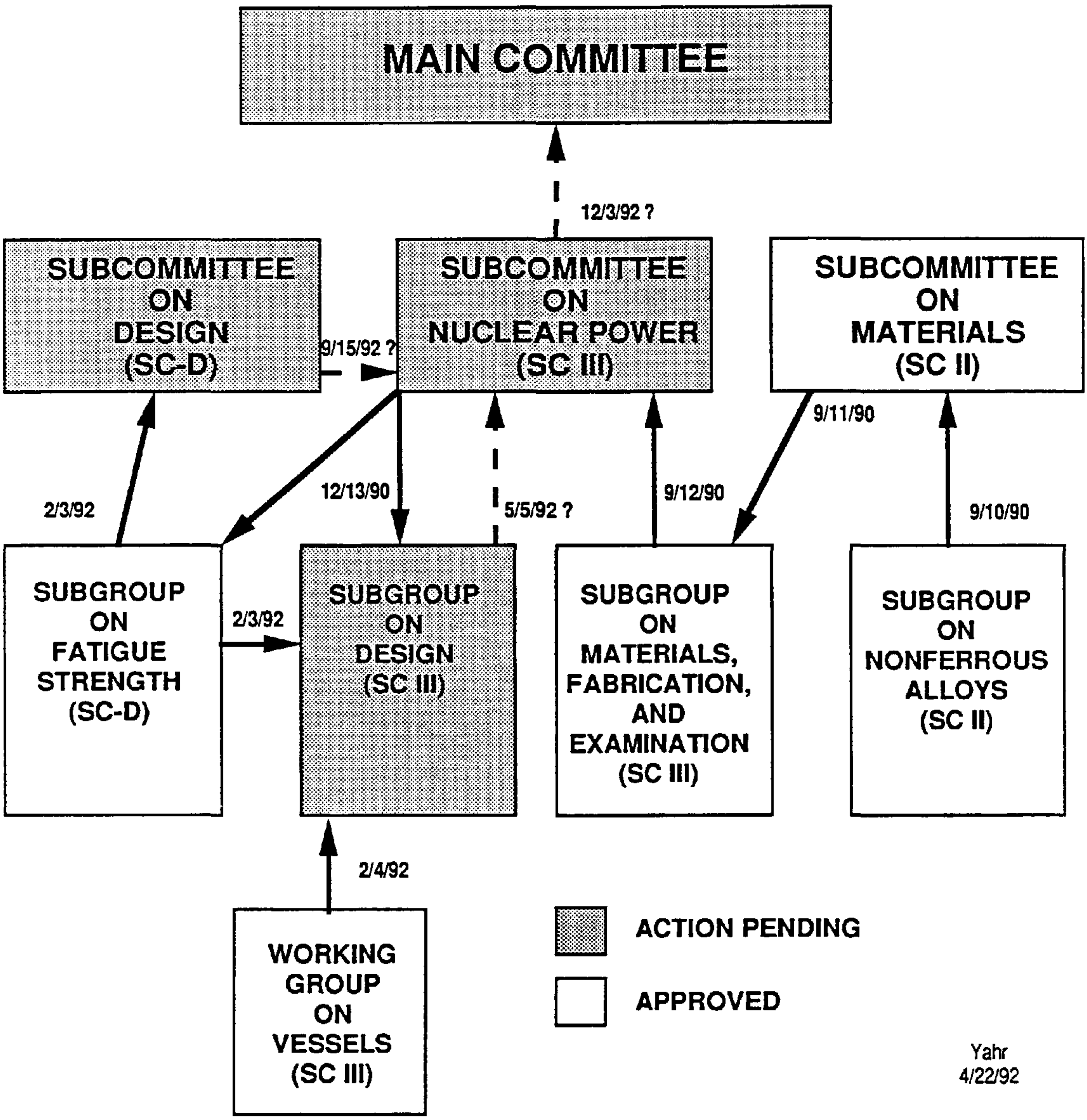
- 16 compact and 15 tensile specimens

- Irradiated to a fluence of 10^{26} m^{-2}

  - PIE in progress

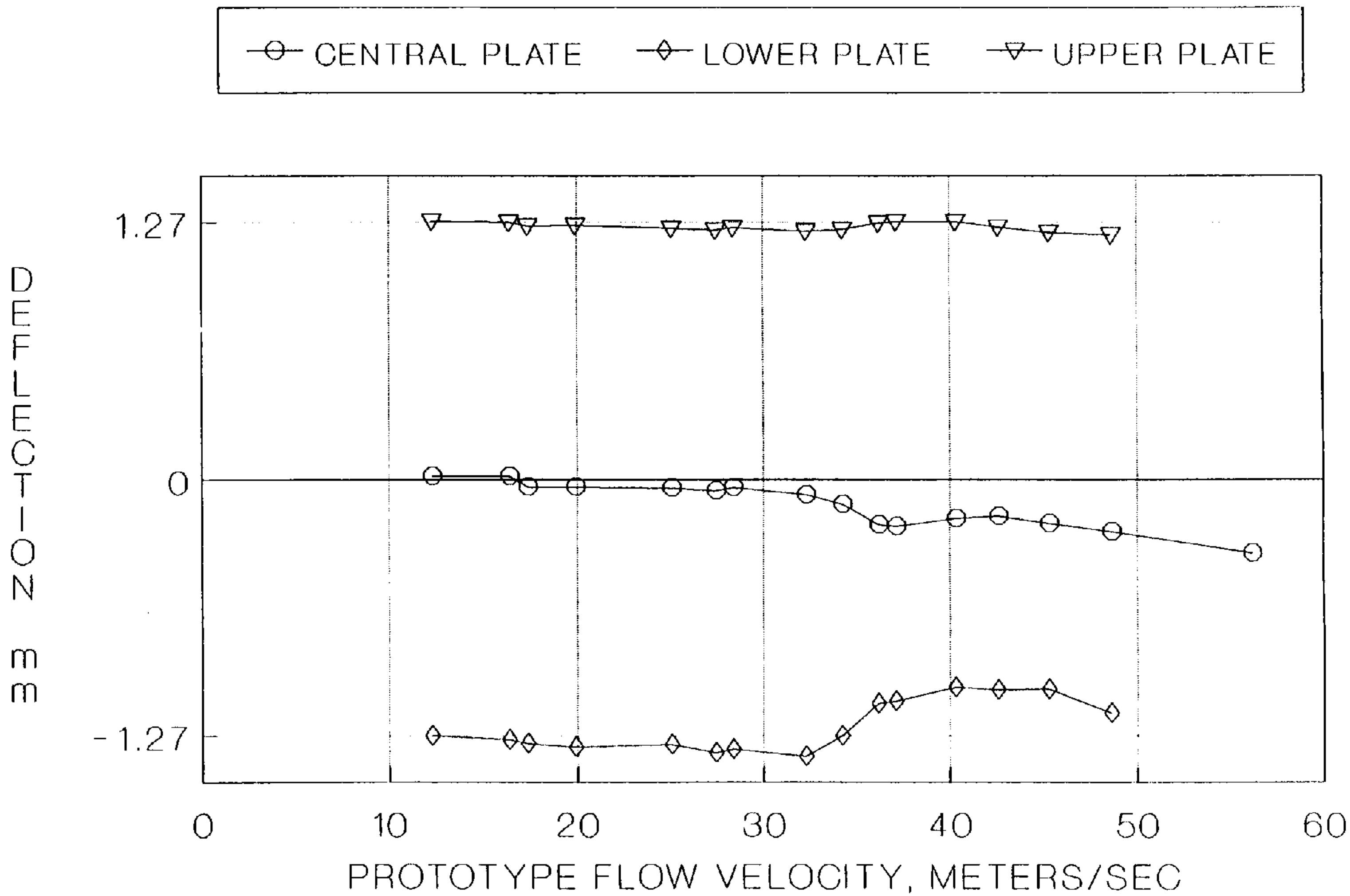
- Identical capsule inserted 10/30/91 with a target fluence of 10^{27} n_{th}/m^2 (2 years)

REQUEST FOR APPROVAL OF 6061-T6 ALUMINUM IS PROGRESSING THROUGH THE ASME B&PV CODE COMMITTEES



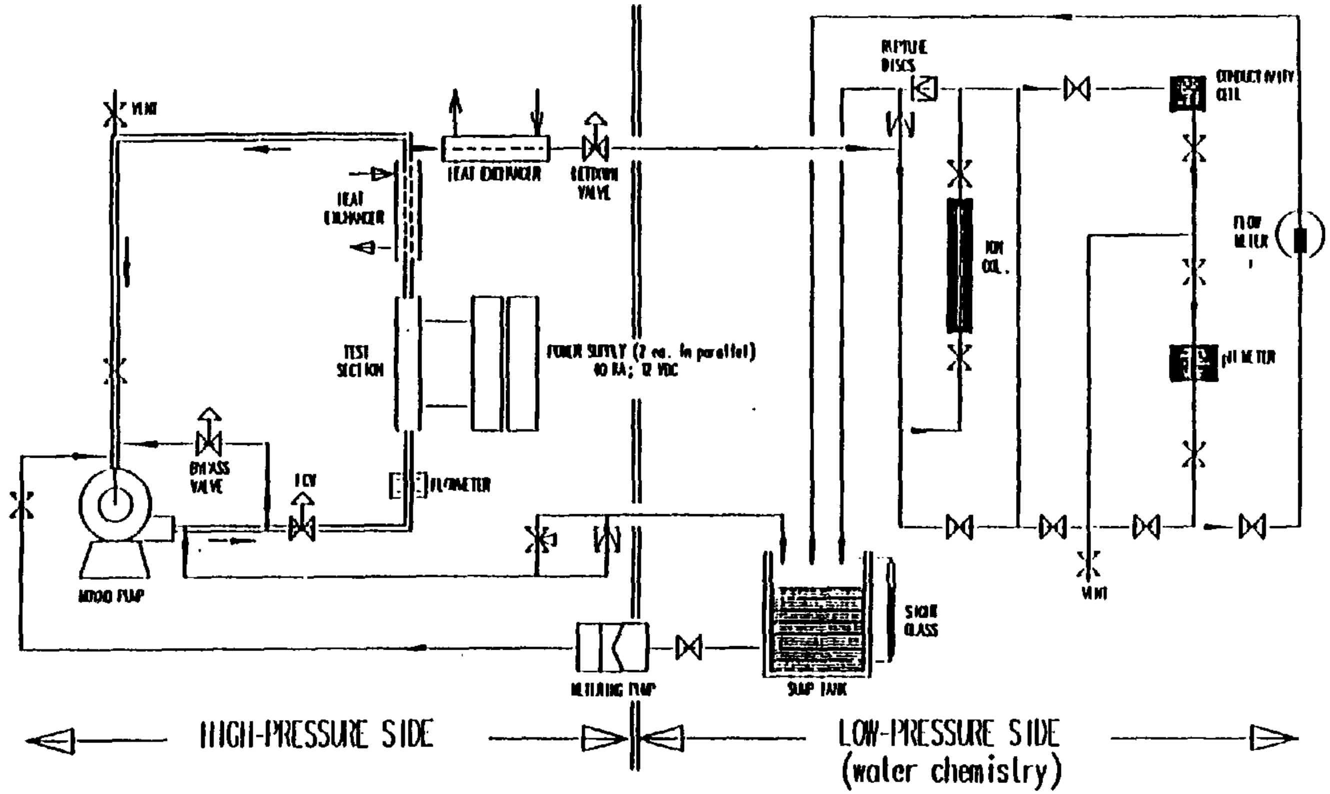
Yahr
4/22/92

ENTRANCE PLATE DEFLECTIONS FOR THREE PLATES VS PROTOTYPE FLOW VELOCITY



DATA TAKEN NOV. 1991

ANS THERMAL-HYDRAULIC TEST LOOP
FLOW SCHEMATIC



FEATURES:

- STAINLESS STEEL THROUGHOUT
- 6.2-MPa (900-psi) OPERATION
- 35-m/s COOLANT VELOCITY
- 35-MW/m² HEAT FLUX
- 369-kW SPECIMEN POWER
- CONTROL OF WATER CHEMISTRY
- COOLANT TEMPERATURE CONTROL
- COMPUTERIZED CONTROL, SAFETY AND DATA ACQUISITION

THE THERMAL-HYDRAULICS LOOP IS DESIGNED TO EXAMINE THE CHF/FLOW INSTABILITY LIMITS AND THE THERMAL-HYDRAULIC CORRELATIONS OF THE ANS CORE

- 50 tests are presently planned with additional testing to be added as needed
  - parametric tests
    - characterize loop operation
    - benchmark against existing data
  - normal operating envelope
    - examine behavior around normal operating point
  - low flow tests
    - simulate shutdown/refueling conditions
  - oxide tests
    - determine effect of oxide on CHF or other thermal limits
  - D_2O proof tests
    - confirmatory testing to prove H_2O/D_2O interchangeability

THTL STATUS

- Loop construction has been completed

- Loop characterization and benchmark testing is currently in progress

    - Characterization of test channel flow instabilities as a function of heat flux, mass flux, exit pressure, and inlet temperature

    - Evaluation and development of test channel design under prototypic conditions, including testing to failure

- Initial tests under conditions representative of the Costa flow excursion show results in good agreement with the predictions

HORIZONTAL VERSUS VERTICAL COLD SOURCE LAYOUT

- The horizontal source has an estimated 30% flux gain over the earlier vertical one due to the absence of one heavy wall and a D_2O layer between the LD_2 and the guide tubes

    - could take advantage of high flux, or keep the same flux with a lower heat load by moving further from the core

- The horizontal source should require neither an ASME III Code design for the moderator nor a pipe containing deuterium running through the reflector tank

    - simplifies safety issues in the design

- Vertical cold source could be removed without disturbing beam guides and associated experimental equipment

    - disadvantage unless cold source replacement can be scheduled to coincide with necessary replacements of the beam guide thimbles

5-10 BAR PUMPED LIQUID MODERATOR SYSTEM FOR HIGH HEAT FLUX SOURCE

(Compared with ILL-type boiling system)

Advantages	Disadvantages
Known moderator density	Requires moving parts in the D_2 system
Less density variation over viewed area	Higher deuterium operating pressure
About 50% lower mass of moderator in the high flux region	More parts requiring maintenance
	More complicated startup

ANS Design Will Require Characterizations of Thermal Limits Over a Range of Conditions

- Incipient Boiling
- Critical Heat Flux
- Flow Instability

  - Normal Operation - 3.7 MPa
    27.4 m/s
  - Pony Motor Flow - 0.1-2 MPa
    4.1 m/s
  - Natural Circulation - 0.1-2 MPa
    $<$ 1 m/s

CAPSULES 3 AND 4 WILL BE IRRADIATED IN HFIR RB POSITIONS

- Approximately 40 compact and 38 tensile specimens plus specimens for microstructural examination
  - base and weld metal
- 10^{25} n_{th}/m^2 will require only about half of one cycle
  - this capsule will be irradiated in FY 1993
- 10^{26} n_{th}/m^2 will require 5 cycles
  - this capsule will be irradiated in FY 1994
- Further tests are planned for FY 1995 and FY 1996

CONCLUSIONS

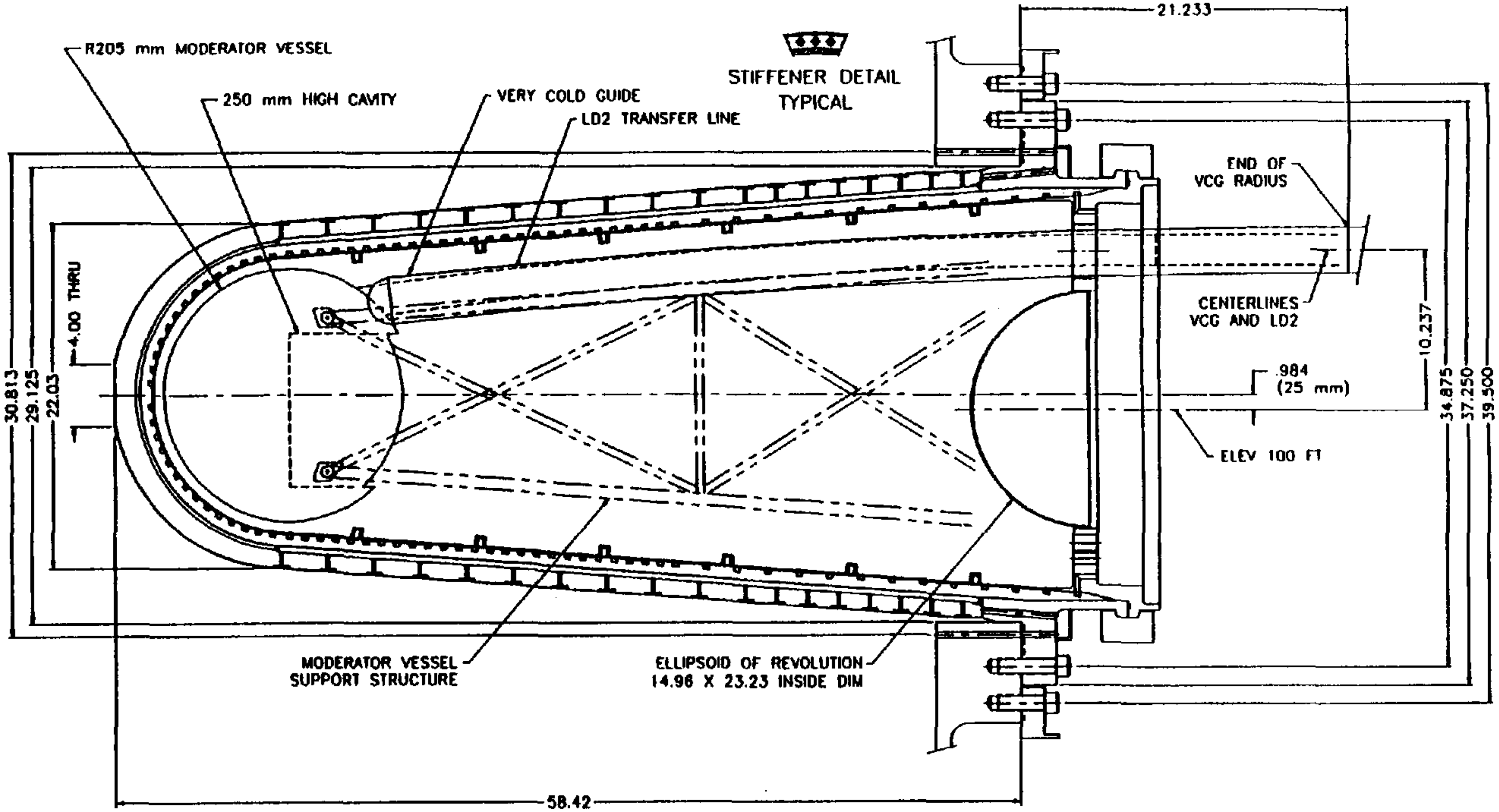
- HFIR results indicate good performance of U_3Si_2 for ANS conditions

- HFIR and ORR data are consistent with our current models

- Additional data from HFIR tests, RERTR fuel, and simulation experiments are expected to improve understanding of basic behavior

FURTHER R&D PLANS FOR THE COLD SOURCE

- Test circulator system
- Test beryllium fabricability and properties
- Develop and test modified Ageron pressure-balanced cryostat
- If possible, design for safety, continued operation of the reactor even if cold source refrigeration is lost



SOME DETAILS OMITTED FOR CLARITY

PRELIMINARY 3/24/92

DO NOT USE FOR FABRICATION

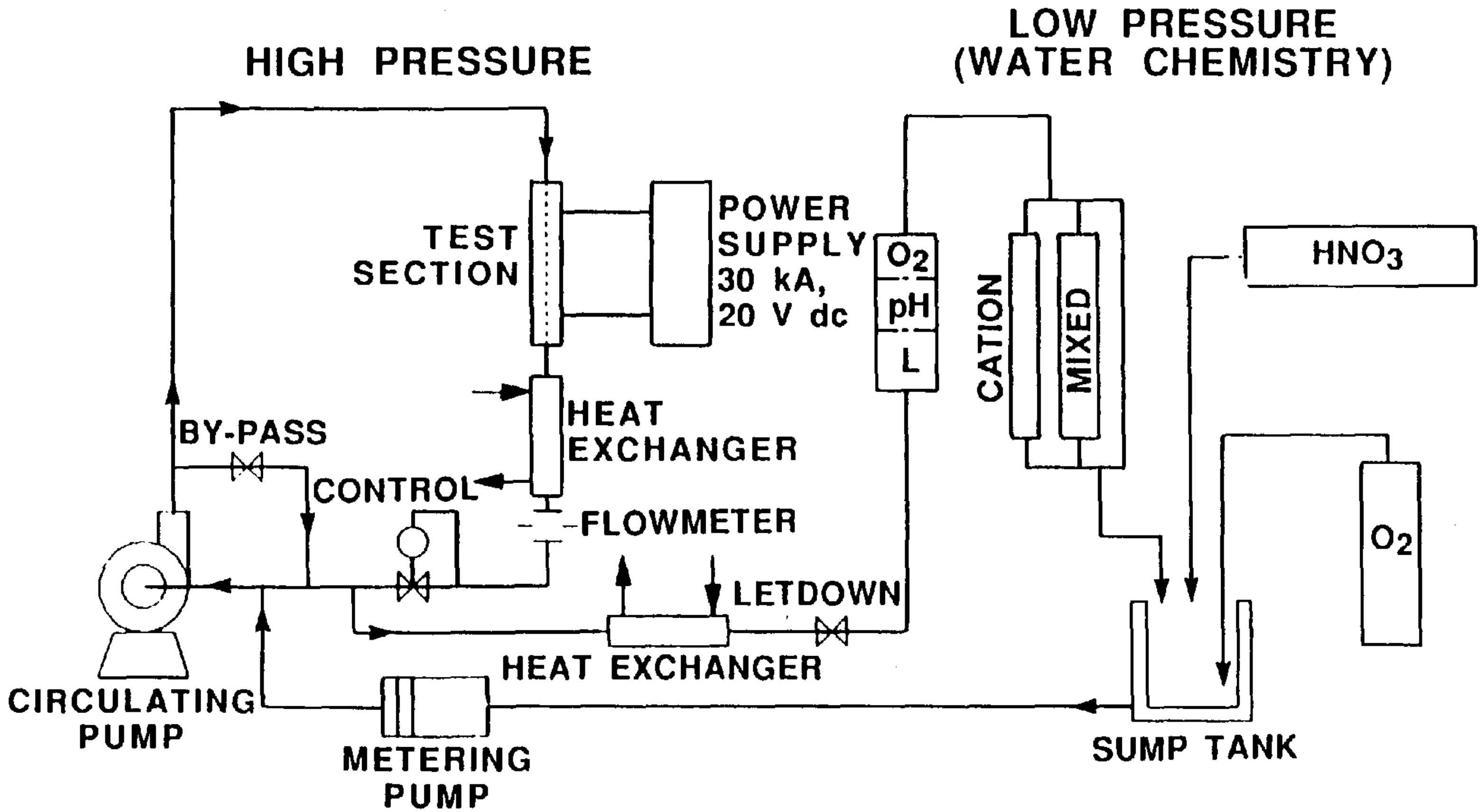
MARTIN MARIETTA

COLD SOURCE ASSEMBLY
SIDE ELEVATION VIEW

ADVANCED NEUTRON SOURCE

ORNL-DWG 88-4810 ETD

CORROSION TEST LOOP IS IN OPERATION



FEATURES:

- 304 L SS THROUGHOUT
- 7-MPa (1000-psi) OPERATION
- 35-m/s COOLANT FLOW
- 20-MW/m^2 HEAT FLUX FOR CORROSION TESTS (EST. 32 MW/m^2 FOR THERMAL-HYDRAULIC TESTS)
- 65-kW SPECIMEN POWER (325 kW FOR T/H)
- CONTROL OF WATER CHEMISTRY
- CONTROL OF COOLANT TEMPERATURES
- COMPUTERIZED CONTROL, SAFETY, AND RECORDING

ornl

FUEL PLATES DEVELOPMENT FOR FRM II CORE

A. TISSIER - Y. FANJAS

IGORR 2 MEETING
MAY 18-19, 1992
SACLAY - FRANCE

Fuel plate Development for FRM II Core

A. Tissier - Y. Fanjas
CERCA
Les Bérauds - BP 1114
26104 Romans sur Isère, France

ABSTRACT

The new FRM II design is based on a compact core. The uranium fuel is silicide U_3Si_2.
A particularity of the fuel plates relates to the uranium distribution in the meat which consists of 2 areas of different uranium densities : 3 g/cm^3 and 1,5 g/cm^3.

In the first part, this paper describes the main characteristics of the fuel plates.

In the second part, the results of preliminary fabrication tests are given. Full size fuel plates have been produced using depleted uranium and adjusting the parameters of the production processes. These tests show that the fuel plates can be produced on industrial scale.

Additional developments are carried out to set up series inspection technics adapted to this double density product.

1. INTRODUCTION.

The Research Reactor of Münich, so-called FRM, will be replaced by a new, high performance neutron source named FRM II.

The new reactor design has previously been presented by Pr. Böning in other international meetings [1] - [2].

The reactor core consists of a single cylindrical fuel element containing 113 involute shape fuel plates. It is very compact. (Slide 1)
The outer radius is 121,5 mm and inner radius 59 mm only. The main characteristics of the fuel plates are summarized in slide 2.

Fuel material is U_3Si_2-Al. The cladding material is AlFeNi which is specially adapted to high temperature use.

The particularity of the plates lies in the fuel meat which is composed of two parts of different uranium densities. One part is loaded to 3 g U_T/cm^3, the other one to 1,5 g U_T/cm^3.
These two parts are positioned side by side along the plate length according to the sketch of slide 3. The low density part occupies one sixth of the total meat width. In the finished plate, it corresponds to the outer part of the fuel element. This disposition allows to flatten the power density profile accross the reactor core [1].

Due to the presence of this double density in the meat, it was necessary to check the industrial feasibility of such plates according to stringent specifications.

Two different densities create an heterogeneity in the plate meat which induces a mechanical behaviour during rolling or bending different from the one usually observed on classical plates.

It also implies adaptation of the inspection techniques for bonding (UT) and U distribution homegeneity (X-ray scanning).

Therefore, we started an investigation program in order to study the double density plate behaviour on the following points in particular :

- Quality of bonding
- Straightness of the meat and core location
- Homogeneity of uranium distribution in both cores
- Curving ability.

The purpose of this paper is to present the development status of these plates.

2. FUEL PLATES FABRICATION TESTS.

The production process was adapted and the fabrication parameters adjusted in order to take into account the previously explained particularity of the fuel plates.
This preliminary work done, 12 full size depleted uranium fuel plates were manufactured.
The observed results are described herebelow :

Bonding quality :

Blister test and UT inspection showed no evidence of lack of bonding between the meat and the cladding on the one hand and between the two cores on the other hand.

The good bonding quality is also visible on the micrographs of slide 4. The three micrographs were taken from a transversal cut of the plate. They respectively correspond to the high density, the transition between high and low density, and the low density zones.
The darker particles are fuel particles.
It can be seen that the thickness of the meat is quite the same on both sides and therefore the cladding thickness is regular. It can also be noticed that the borderline between the two cores is very sharp and straight.

Dimensions - Radiographic inspection

The external dimensions of the plates were measured as for ordinary plates.
To check that the dimensions of the cores met the specifications, X-Ray films of the plates were taken.
Slide 5 shows one of these X-Ray films. The difference of uranium density is imaged by different grey levels. It can be noticed that the borderline between both parts of the meat is very straight.
Therefore the parameters used for rolling were appropriate.

Other defects such as U stray particles in forbidden aluminium areas can be revealed thanks to the X-Ray films. In our case, they did not show any tendancy to present stray particles.
The radiograph also shows qualitatively the good homogeneity of the uranium distribution in the two parts of the core.

Homogeneity of Uranium distribution

Specially designed X-Ray scanning machines are used to control quantitatively the homogeneity of uranium distribution in fuel plates. A focused beam of X-Rays goes through the thickness of the plate. By measuring the absorption of the beam across the plate, the uranium density variations are determined.
The whole plate surface is inspected by successive scanning along its core length.

Slide 6 shows the results of this control on a scanning length on each part of the core. The machine was successively calibrated to inspect each uranium density zone.
It can be noticed that each profile is very linear in both low and high density zones, which proves the good homogeneity of U distribution.

Curving ability

The previous inspection carried out on the flat products had shown that the fuel plates could be produced according to specifications.

It remained to be checked that the double core was not detrimental to the plate curving ability. Therefore, 6 fuel plates were curved to obtain the appropriate involute shape. The results of the bending test were satisfactory; the involute shape was correct. In particular, no lack of continuity was observed along the borderline separating the two cores.

3. FUTURE WORK

3.1 Manufacturing.

Previous work corresponds to preliminary tests to check the feasibility of FRM II double core plates. It was carried out on a small quantity of full size plates. As for any new product, reproducibility tests have to be done. For this purpose, in the frame of the cooperation between FRM and CERCA, full size depleted uranium plates will be manufactured in quantity large enough to allow the production of a complete dummy fuel element.

3.2 Inspection.

The test plates have been inspected by adjusting the calibration of inspection equipment for each uranium density area.
For industrial production, inspection procedures will have to be adapted to minimize the time length required for plate inspection.

CONCLUSION.

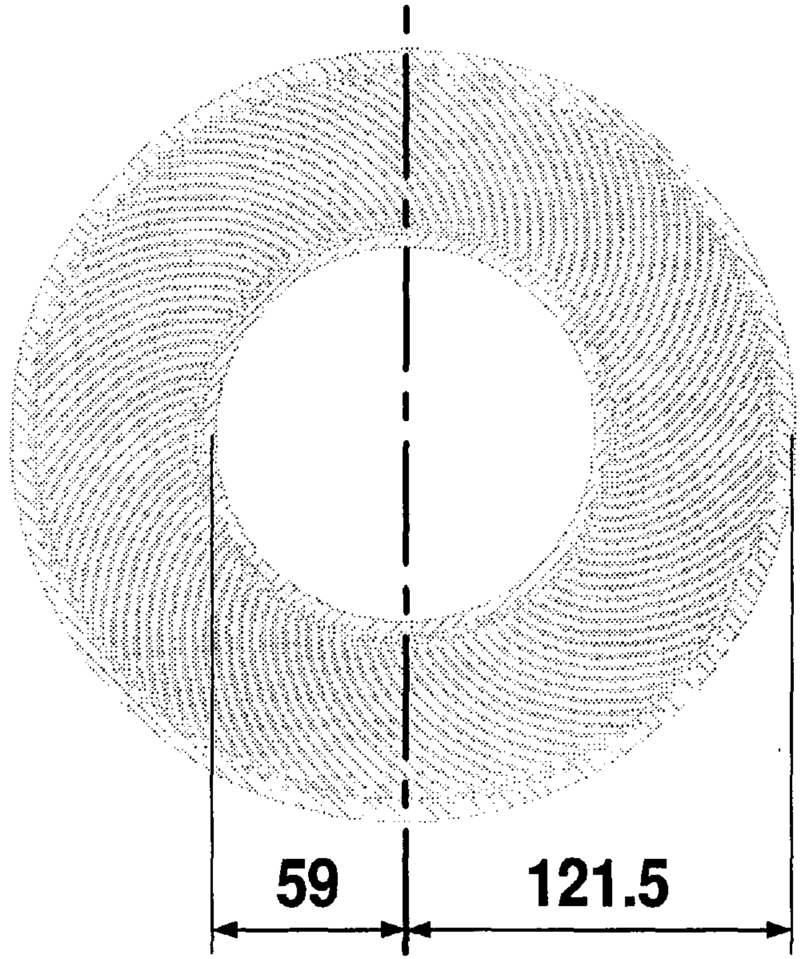
The test results which have been exposed previously show the feasibility of full size fuel plates for FRM II. The adaptation of the existing production processes permits the fabrication of fuel plate containing a double core, insuring a good quality of the bonding between the cladding and each uranium part, a good homogeneity of the uranium distribution, and meeting the geometry requirements of the double core and the final involute plate.

REFERENCES.

[1]K. Böning, J. Blaubach : "Design and Safety features of the planned compact core research reactor FRM II" International meeting on Reduced Enrichment for Research and Test Reactors, Jakarta (Indonesia) Nov 4-7, 1991.

[2]K. Böning, W. Gläser, J. Meier, G. Rau, A. Röhrmoser, E. Steichele : "Status of the Munich Compact Core Reactor Project". Proceedings of the XII. Int. Meeting on Reduced Enrichment for Research and Test Reactors 1989, Berlin Germany (FRG), September 10-14, 1989; Report of the KFA Jülich GmbH, Konferenzen Band 4/1991, ISBN 3-89336-063-8, page 473-483 (1991)

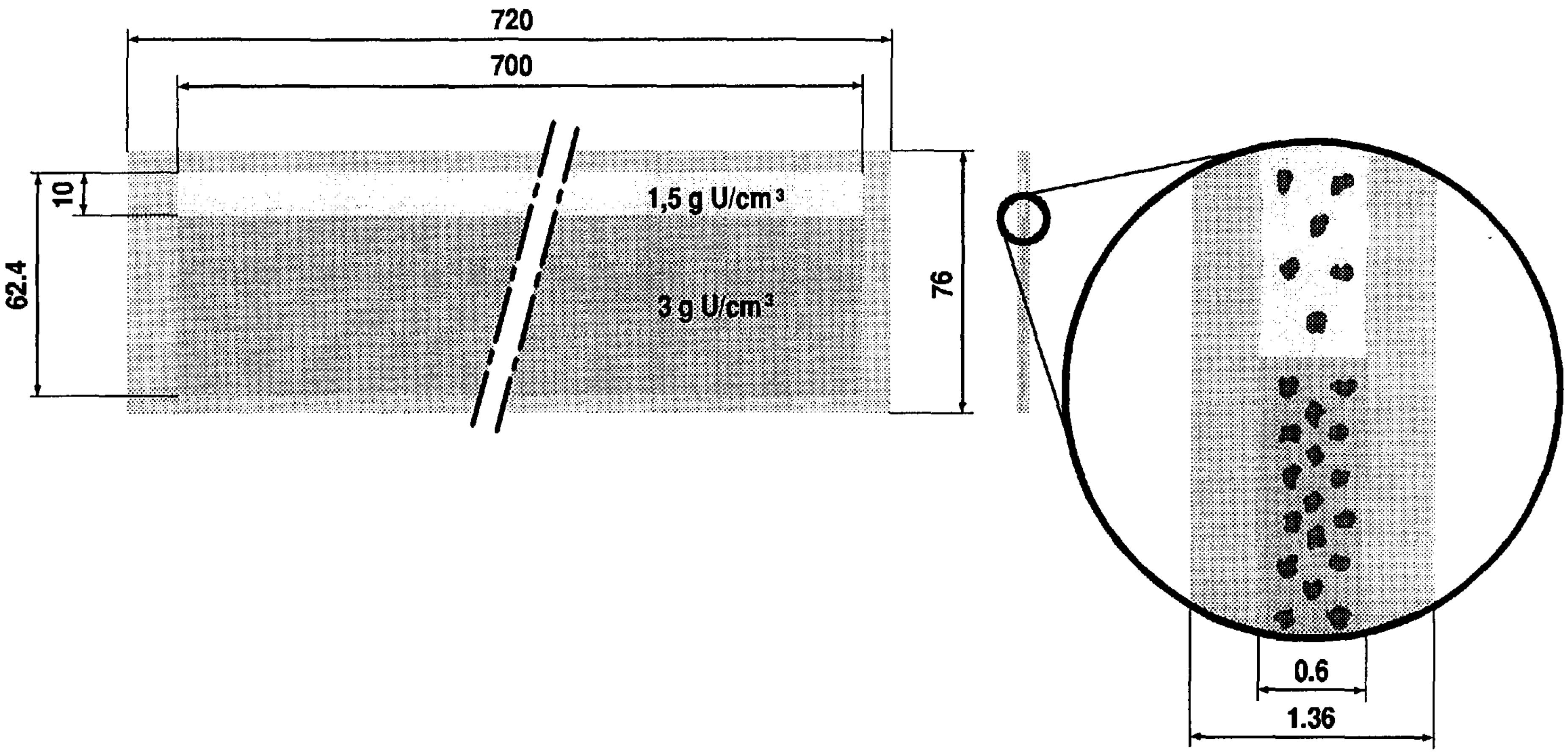
FRM-II
FUEL ELEMENT SECTION



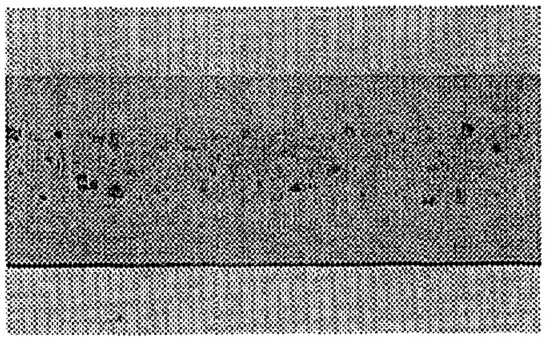
MAIN CHARACTERISTICS OF FRM-II FUEL PLATES

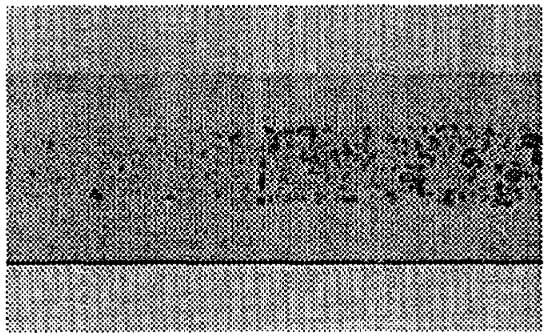
U DENSITY	:	1.5 and 3.0 g U_T/cm^3
FUEL MEAT	:	$U_3 Si_2$ + Al
CLADDING	:	Al Fe Ni/AG_3 NE
CLADDING THICKNESS	:	Nominal : 0.38 mm Mini : 0.25 mm
MEAT THICKNESS	:	Nominal : 0.60 mm
PLATE SHAPE	:	INVOLUTE

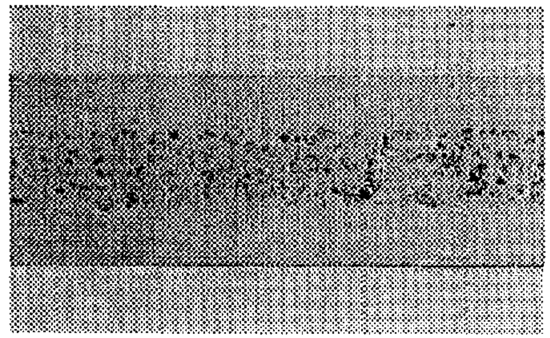
FRM-II DOUBLE-CORE FUEL PLATE (flat condition)



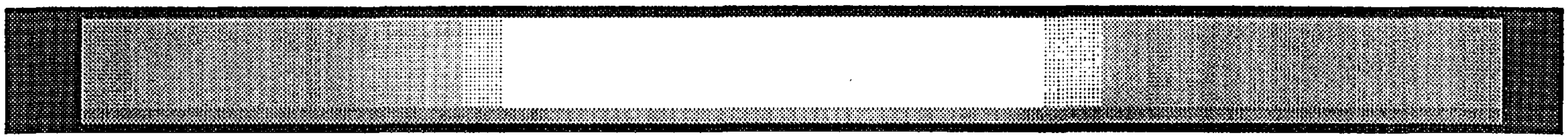
FRM-II DOUBLE-CORE FUEL PLATE TRANSVERSE SECTION


Low density zone


Transition zone

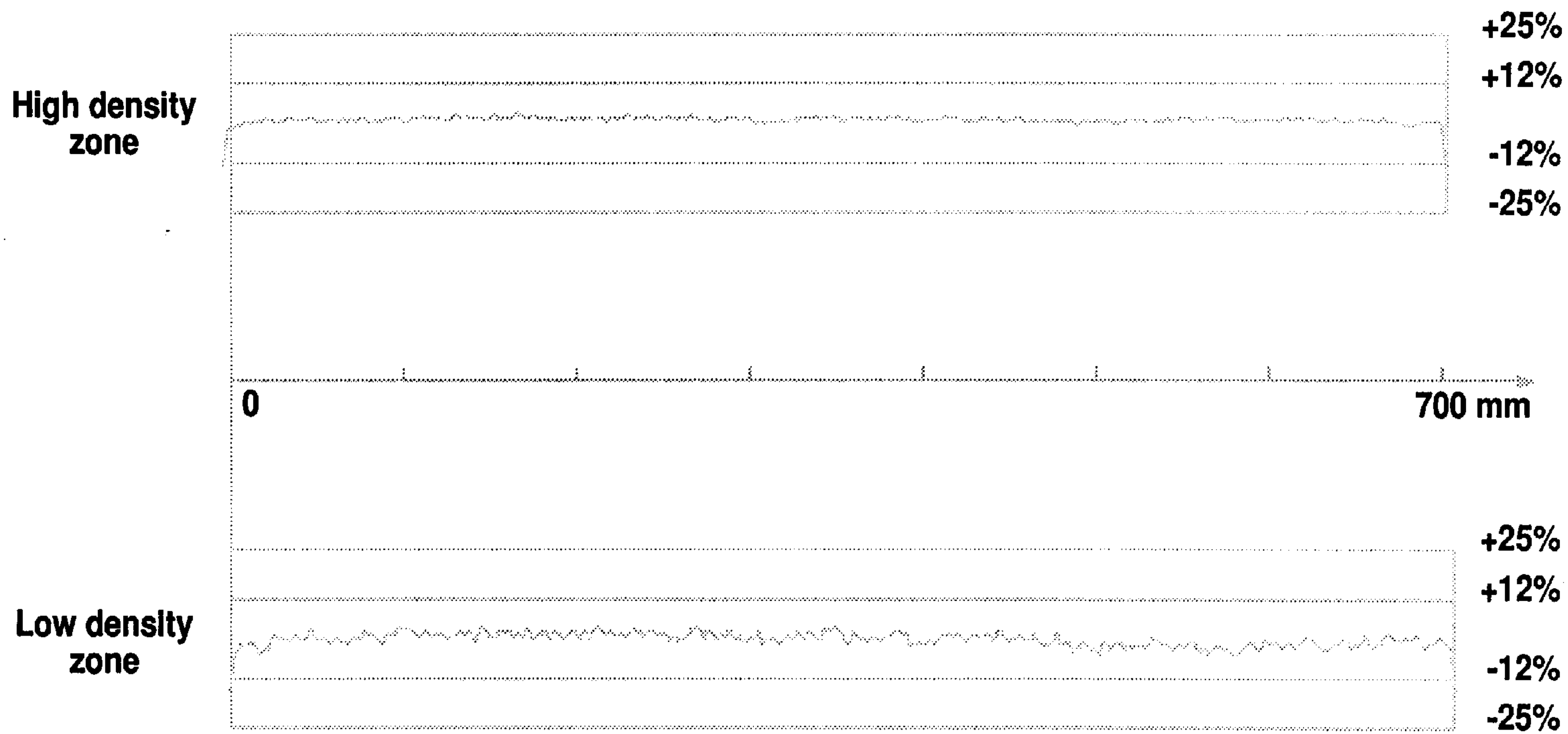

High density zone

FRM-II PLATE FUEL CORE LOCATION



CERCA

U DISTRIBUTION HOMOGENEITY ALONG FRM-II PLATE HIGH AND LOW DENSITY ZONES



IGORR-II

2nd Meeting of the International Group on Research Reactors,

May 18-19, 1992 - Saclay, France

IRRADIATION EFFECTS ON ALUMINIUM AND BERYLLIUM

M. Bieth

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The High Flux Reactor (HFR) in Petten (The Netherlands) is a 45 MW light water cooled and moderated research reactor. The vessel was replaced in 1984 after more than 20 years of operation because doubts had arisen over the condition of the aluminium alloy construction material.

Data on the mechanical properties of the aluminium alloy Al 5154 with and without neutron irradiation are necessary for the safety analysis of the new HFR vessel which is constructed from the same material as the old vessel.

Fatigue, fracture mechanics (crack growth and fracture toughness) and tensile properties have been obtained from several experimental testing programmes with materials of the new and the old HFR vessel.

- Low-cycle fatigue testing has been carried out on non-irradiated specimens from stock material of the new HFR vessel.

  The number of cycles to failure ranges from 90 to more than 50,000 for applied strain from 3.0% to 0.4%.

- Fatigue crack growth rate testing has been conducted :
  - with unirradiated specimens from stock material of the new vessel
  - with irradiated specimens from the remnants of the old core box.

Irradiation has a minor effect on the sub-critical fatigue crack growth rate. The ultimate increase of the mean crack growth rate amounts to a factor of 2. However crack extension is strongly reduced due to the smaller crack length for crack growth instability (reduction of K_{IC}).

- Irradiated material from the core box walls of the old vessel has been used for fracture toughness testing.

  The conditional fracture toughness values K_{IQ} ranges from 30.3 down to 16.5 $MPa\sqrt{m}$.

  The lowermost meaningful "K_{IC}" is 17.7 $MPa\sqrt{m}$ corresponding to the thermal fluence of 7.5 $10^{26} n/m^2$ for the End of Life (EOL) of the old vessel.

- Testing carried out on irradiated material from the remnants of the old HFR core box shows an ultimate neutron irradiation hardening of 35 points increase of HSR_{15N} and an ultimate tensile yield stress of 589 MPa corresponding to the ductility of 1.6%.

Besides, due to the effects of embrittlement and swelling induced by irradiation, the HFR beryllium reflector elements had to be replaced after more than 25 years of operation.

Operational and practical experiences with these reflector elements are commented, as well as main engineering features of the new reflector elements : upper-end fittings of both filler element and insert in stainless steel, no radially drilled holes and no roll pins.

IGORR - II

2nd meeting of the International Group on Research Reactors,
May 18-19, 1992 - Saclay, France

IRRADIATION EFFECTS ON ALUMINIUM AND BERYLLIUM

Michel BIETH

COMMISSION OF THE EUROPEAN COMMUNITIES

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HFR DIVISION, PETTEN SITE, THE NETHERLANDS

INTRODUCTION

- THE HIGH FLUX REACTOR (HFR) IN PETTEN (THE NETHERLANDS) IS A 45 MW LIGHT WATER COOLED AND MODERATED RESEARCH REACTOR

- IN OPERATION DURING MORE THAN 30 YEARS

- INSTALLATION KEPT UP-TO-DATE BY REPLACING AGEING COMPONENTS

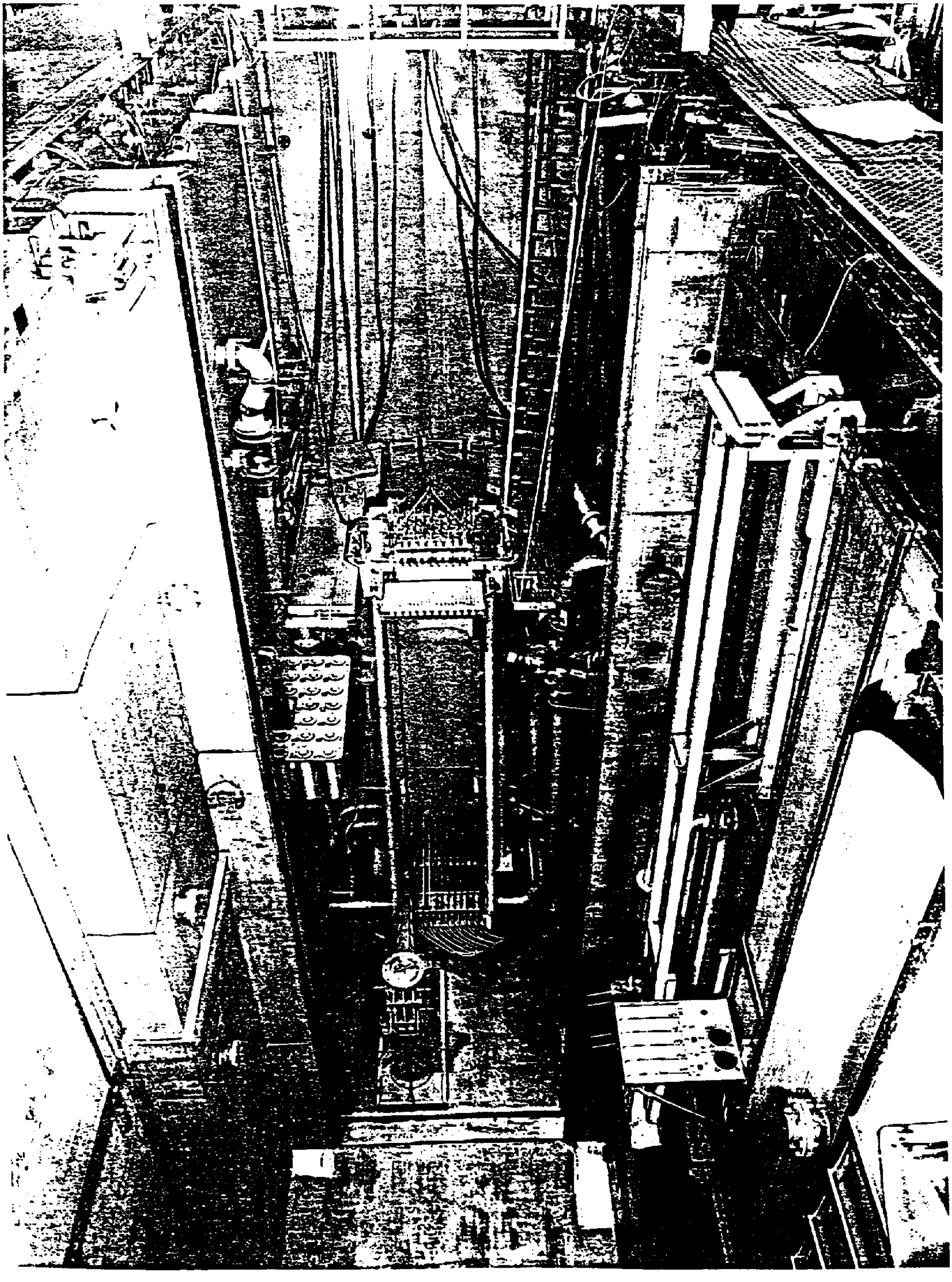
- REPLACEMENT IN 1984 OF THE ALUMINIUM REACTOR VESSEL AFTER MORE THAN 20 YEARS OF SERVICE

- DESIGN OF THE NEW ALUMINIUM 5154 ALLOY REACTOR VESSEL IS BASED ON THE DEMONSTRATION OF EVIDENCE THAT THE VESSEL CONTAINS NO CRITICAL DEFECTS

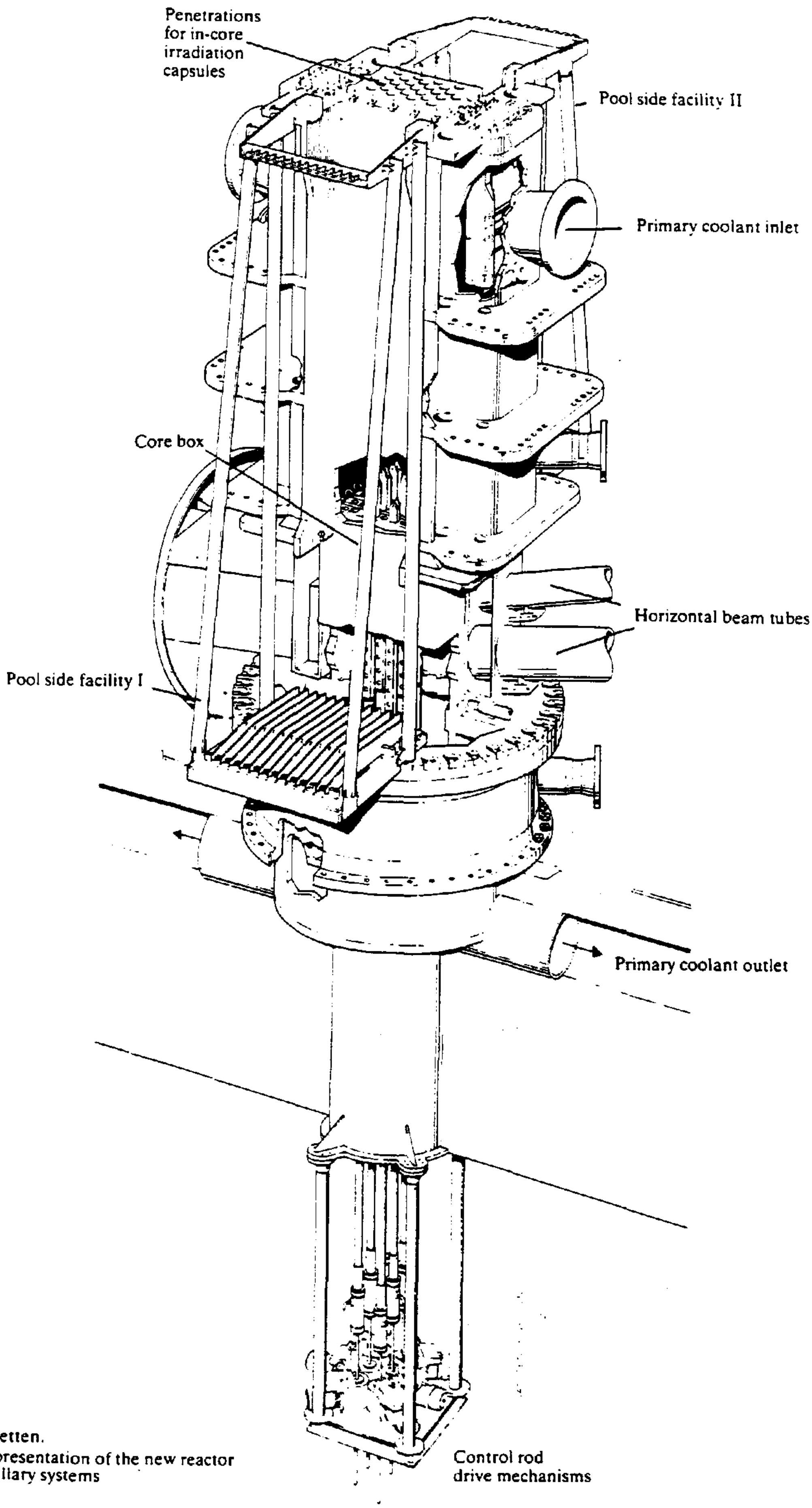
- KNOWLEDGE OF MATERIAL PROPERTIES AND LIKELY DEFECT PRESENCE AND SIZE IS REQUESTED FOR THE ASSESSMENT OF THE HFR VESSEL INTEGRITY

- REPLACEMENT OF THE BERYLLIUM REFLECTOR ELEMENTS AFTER MORE THAN 25 YEARS OF OPERATION, DUE TO EMBRITTLEMENT AND SWELLING INDUCED BY IRRADIATION

VIEW INTO THE REACTOR POOL OF HFR PETTEN



HFR REACTOR VESSEL



HFR Petten.
Perspective representation of the new reactor vessel with ancillary systems

CHEMICAL COMPOSITION OF THE MATERIALS OF THE OLD AND THE NEW HFR VESSELS

	Mg	Si	Cu	Mn	Fe	Zn	Cr	Ti	Al
Old core box	3.78	0.14	0.04	0.33	0.28	0.01			bal.
New vessel	3.21	<0.25	<0.05	<0.10	<0.40	<0.20	0.24	0.11	bal.

CALCULATED NEUTRON FLUENCE VALUES FOR THE MID-CENTRE POSITIONS OF THE OLD CORE BOX WALLS

Fluence, 10^{26} n/m^2	West wall	East wall
thermal ($E < 0.414$ eV)	7.5	3.2
fast ($E > 0.1$ MeV)	6.9	0.8
ratio thermal/fast	1.1	4.0

OLD CORE BOX TIME EXPOSURE TO NEUTRON IRRADIATION = 4,24 10^8 SEC.
FROM NOVEMBER 1961 UNTIL NOVEMBER 1983

MATERIAL PROPERTY TESTING PROGRAMMES

- DATA ON THE MECHANICAL PROPERTIES OF ALUMINIUM ALLOY AL 5154 WITH AND WITHOUT NEUTRON IRRADIATION ARE NECESSARY FOR THE SAFETY ANALYSIS OF THE HFR VESSEL AT BOL AND EOL

- FATIGUE, FRACTURE MECHANICS (CRACK GROWTH AND FRACTURE TOUGHNESS) AND TENSILE PROPERTIES OBTAINED FROM SEVERAL EXPERIMENTAL TESTING PROGRAMMES WITH MATERIALS OF THE NEW AND THE OLD HFR VESSEL

- TESTING CARRIED OUT BY ECN

FATIGUE EXPERIMENTS WITH ALUMINIUM ALLOY 5154 SPECIMENS

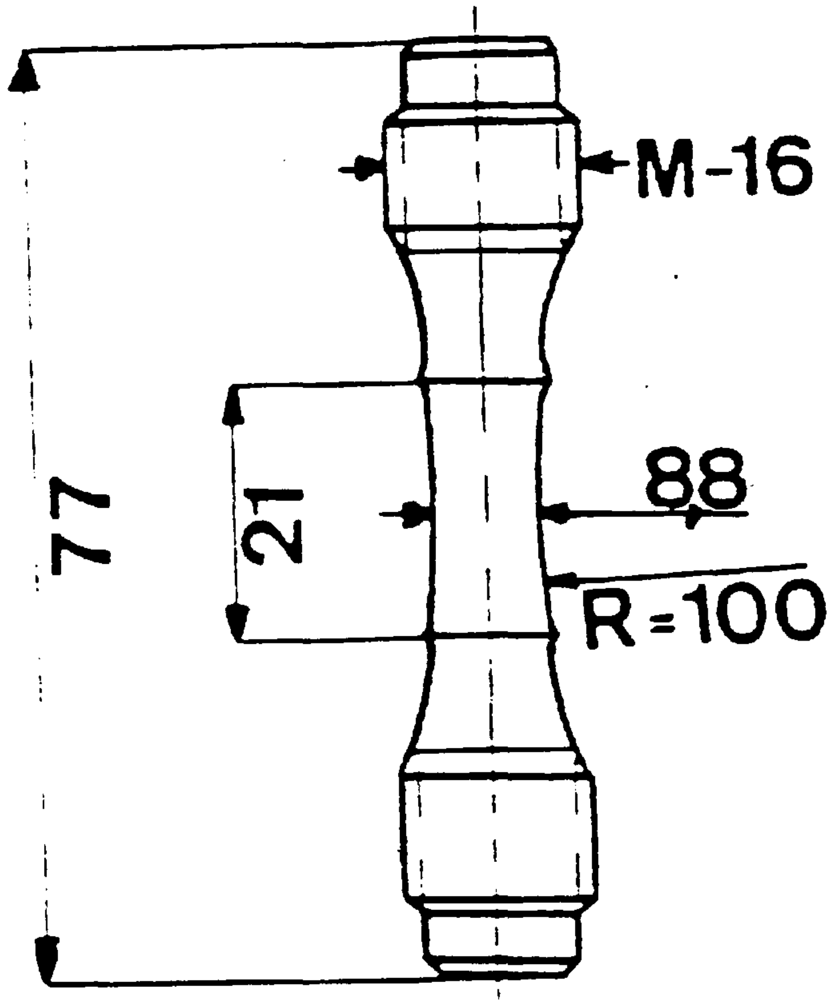
° INFORMATION ON THE FATIGUE PROPERTIES OF THE CONSTRUCTION MATERIAL OF THE NEW VESSEL (ALUMINIUM ALLOY TYPE AL 5154) REQUESTED BY THE NETHERLANDS LICENSING AUTHORITIES

° TESTING PROGRAMME OBJECTIVES:

- TO PERFORM SPOT-CHECKS ON THE FATIGUE DESIGN CURVE
- TO MEASURE FATIGUE CRACK GROWTH RATES IN IRRADIATED AND NON-IRRADIATED SPECIMENS

° TESTING CARRIED OUT BY ECN:

- LOW CYCLE FATIGUE TESTS AND FATIGUE CRACK GROWTH MEASUREMENTS WITH UN-IRRADIATED SPECIMENS FROM STOCK MATERIAL OF THE NEW HFR VESSEL
- FATIGUE CRACK GROWTH TESTS WITH IRRADIATED SPECIMENS FROM MATERIAL OF THE OLD HFR CORE BOX



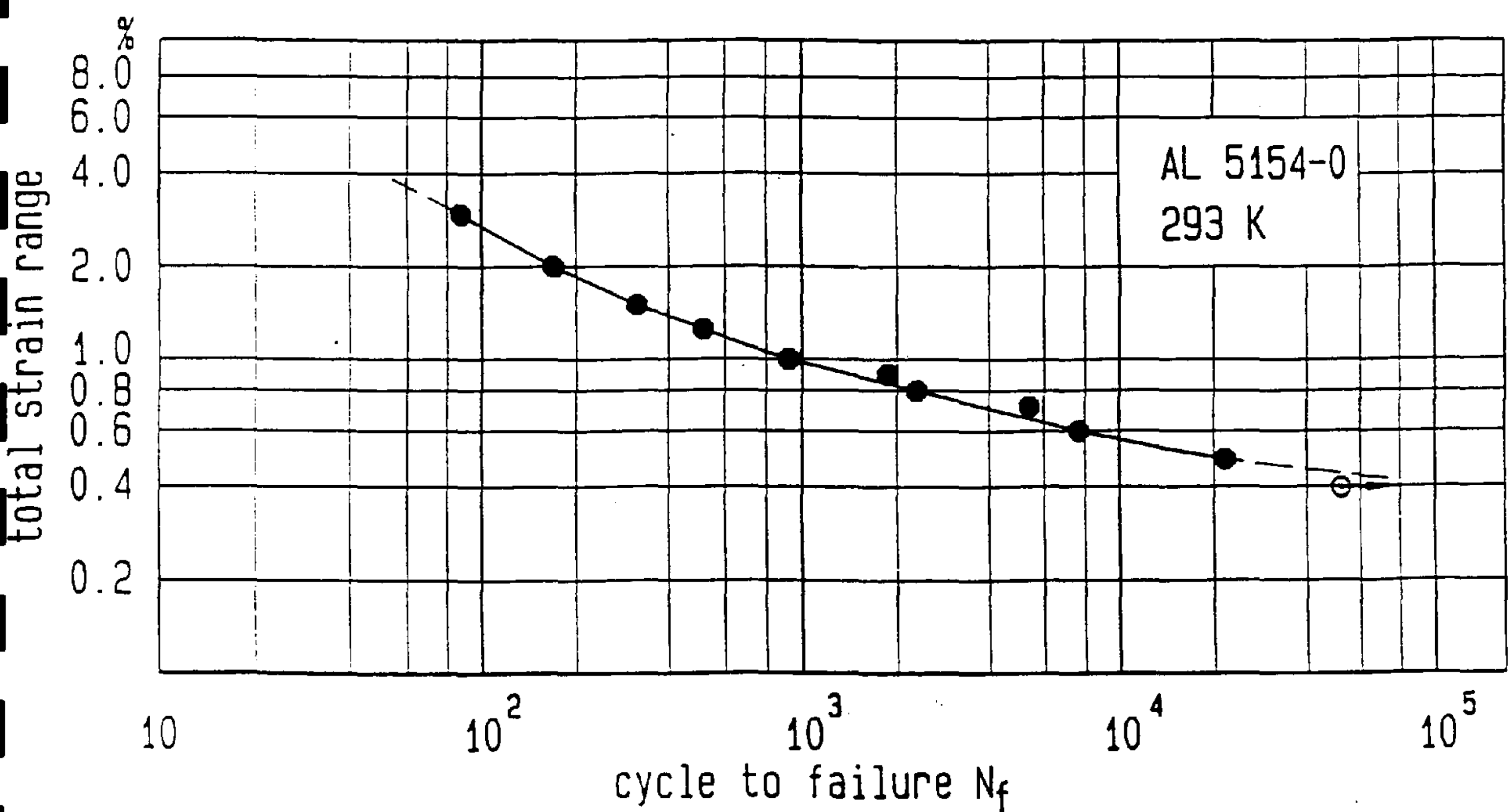
Dimensions of the Low Cycle Fatigue Specimen

LOW CYCLE FATIGUE TEST RESULTS

- THE NUMBER OF CYCLE OF FAILURE RANGES FROM 88 TO 50.000 FOR APPLIED STRAIN FROM 3,0% TO 0,4%. THE CORRESPONDING ULTIMATE CYCLIC STRESS VARIES FROM 580 MPa TO 305 MPa.

- THE STRESS AMPLITUDE INCREASES WITH THE NUMBER OF FATIGUE CYCLES FROM 106 MPa FOR THE FIRST LOOP AT 0,4% STRAIN RANGE TO 289 MPa FOR THE SATURATION LOOP AT 3,0% STRAIN RANGE.

FATIGUE LIFE CURVE OF NON-IRRADIATED Al ALLOY 5154



FATIGUE CRACK GROWTH MEASUREMENT

- FATIGUE CRACK GROWTH TESTS HAVE BEEN PERFORMED ACCORDING TO ASTM E 647-83

- TWO SERIES OF EXPERIMENTS:
  - WITH UN-IRRADIATED SPECIMENS FROM STOCK MATERIAL OF THE NEW HFR VESSEL
  - WITH IRRADIATED SPECIMENS FROM THE OLD CORE BOX REPRESENTING DIFFERENT NEUTRON RATIO RANGING FROM 1 TO 5

- CONTINUOUS CRACK MONITORING BY DIRECT CURRENT POTENTIAL DROP TECHNIQUE
- CRACK EXTENSION CHECKED BY OPTICAL MEANS

- FATIGUE CRACK GROWTH TEST CONDITIONS:

Loading type:	cyclic tensile-force loading
Specimen type:	CT-specimen, W = 50 mm
Specimen thickness:	B = 10.0 mm (East wall) or 12.5 mm (others)
Mechanical notch:	a_n = 12.5 mm
Control parameter:	constant load amplitude (ΔP)
Wave form:	triangular
Environment/humidity:	air/51% - (88% * For the 2 tests in water)
Temperature:	room temperature (about 300 K)
Pre-crack-extension (Δa_i):	1 - 3 mm
Crack growth loading:	constant ΔP
Crack growth (Δa):	10 - 15 mm
Pmax:	4.0 kN - 2.4 kN
Number of tested specimens:	8* un-irradiated (including 2 in water)
	12 irradiated (8 west wall and 4 East wall)

FATIGUE CRACK GROWTH

- FC GROWTH EXPERIMENTS TO PROVIDE INFORMATION ON THE LINEAR ELASTIC CRACK GROWTH BEHAVIOUR OF AL 5154 ALLOY

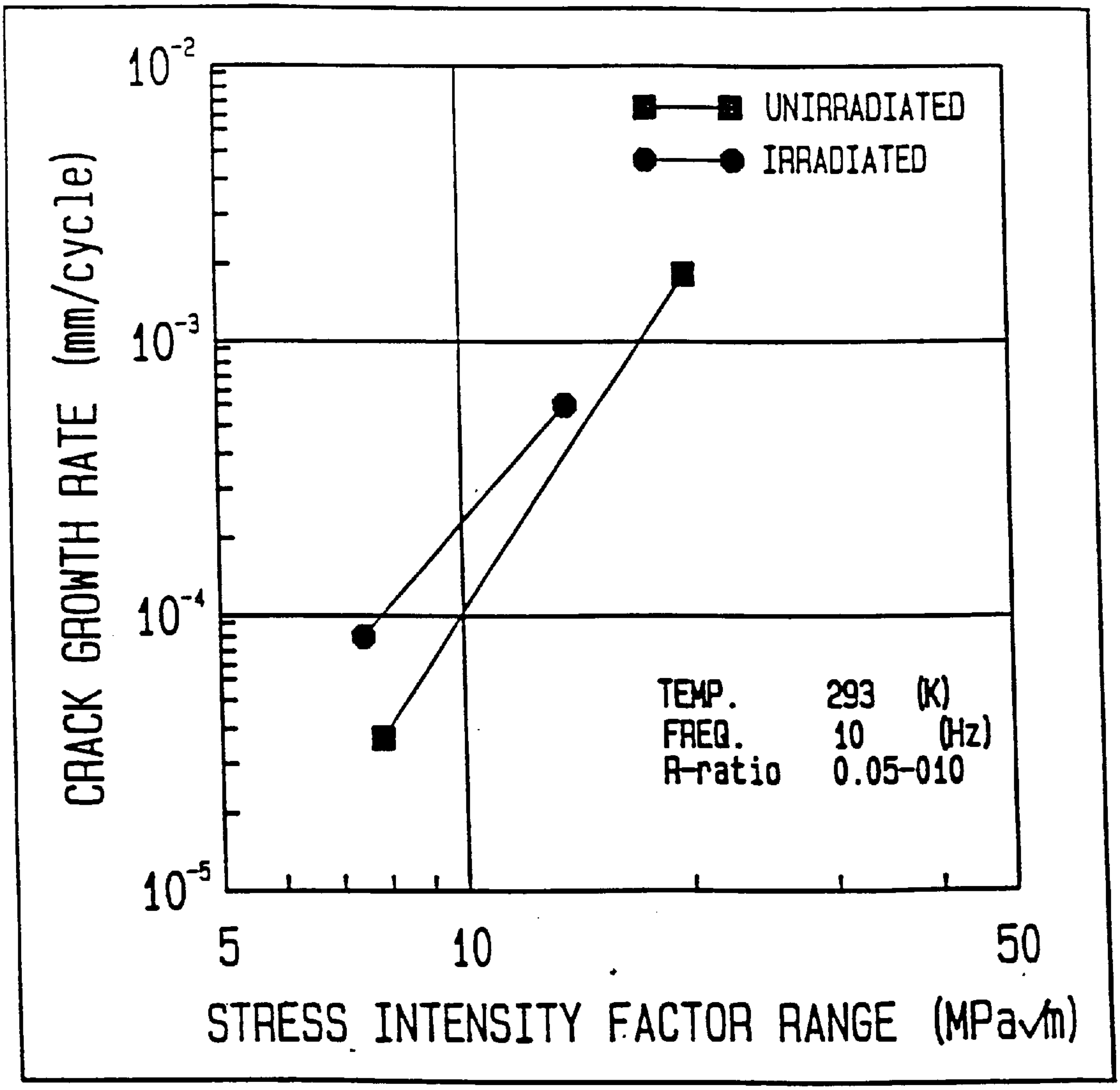
- EMPIRICAL "PARIS RELATIONSHIP" BETWEEN CRACK GROWTH RATE da/dn AND THE CRACK TIP STRESS INTENSITY FACTOR RANGE ΔK

  $$da/dn = C\,(\Delta K)^n$$

- IRRADIATION HAS A MINOR EFFECT ON THE SUB CRITICAL FATIGUE CRACK GROWTH RATE. THE ULTIMATE INCREASE OF THE MEAN CRACK GROWTH RATE AMOUNTS TO A FACTOR OF 2

- CRACK EXTENSION IS STRONGLY REDUCED DUE TO SMALLER CRACK LENGTH FOR CRACK GROWTH INSTABILITY (REDUCTION OF K_{IC})

SUBCRITICAL FATIGUE CRACK GROWTH "PARIS CURVE" OF IRRADIATED AND NON-IRRADIATED AL 5154



FRACTURE TOUGHNESS

- IRRADIATED MATERIAL FROM THE CORE BOX WALLS OF THE OLD VESSEL HAS BEEN USED FOR FRACTURE TOUGHNESS TESTING

- THE CONDITIONAL FRACTURE TOUGHNESS VALUES K_{IQ} RANGES FROM 30.3 TO 16.5 MPa$\sqrt{m}$

- THE LOWERMOST MEANINGFUL "K_{IC}" IS 17,7 MPa$\sqrt{m}$ CORRESPONDING TO THE THERMAL FLUENCE OF 7.5 10^{26} n/m^2 FOR THE END OF LIFE OF THE OLD VESSEL

CONDITIONAL FRACTURE TOUGHNESS (K_{IQ}) OF IRRADIATED AL 5154

K_{IQ} MPa$\sqrt{m}$	thermal neutron fluence 10^{26} n/m^2 (E<0.414eV)
25.3	5.4
22.0	5.8
24.9	6.1
25.3	5.8
23.9	6.1
24.0	5.0
27.0	3.2
28.9	2.6
27.8	2.6
30.3	1.7
23.0	6.2
24.2	5.3
24.3	5.0
26.6	3.7
26.5	4.4
24.4	5.0
22.3	6.9
* 17.7	7.5
16.5	7.5
25.7	5.0

HARDNESS, STRENGTH AND DUCTILITY

- TESTING CARRIED OUT ON IRRADIATED MATERIAL FROM THE OLD REACTOR VESSEL

- HARDNESS IS STRONGLY INCREASED BY NEUTRON IRRADIATION. TESTING SHOWS AN ULTIMATE HARDENING OF 35 POINTS INCREASE OF HSR_{N15} AND AN ULTIMATE TENSILE YIELD STRESS OF 589 MPa AND WITH A DUCTILITY OF 1.6%

SURVEILLANCE PROGRAM (SURP)

- AN IRRADIATION DAMAGE SURVEILLANCE PROGRAMME HAS BEEN SET UP IN 1985 FOR THE NEW VESSEL MATERIAL TO PROVIDE INFORMATION IN FRACTURE MECHANICS PROPERTIES.

- COMPACT TENSION AND TENSILE HAVE BEEN PLACED IN AN IN-CORE AND A POOL SIDE FACILITY (PSF) POSITION FOR THE PROPER SIMULATION OF THE NEUTRON FLUX CONDITIONS OF THE DIFFERENT WALLS OF THE CORE BOX.

- NEUTRON MONITOR SETS ARE REMOVED EVERY 3 YEARS.
  THE SPECIMENS ARE SCHEDULED TO BE REMOVED FROM IRRADIATION AND TO BE TESTED AT DIFFERENT INTERVALS OF THE REACTOR VESSEL LIFE.

- THE THERMAL NEUTRON EXPOSURE REACHES ABOUT $2.5 * 10^{26}$ n/m^2 IN 1992.

CONCLUSION

- ROUTINE IN-SERVICE INSPECTION HAS BEEN PERFORMED IN 1991 IN ACCORDANCE WITH THE OPERATING LICENCE REQUIREMENTS AND LEADS TO THE CONCLUSION THAT THERE HAS BEEN NO CHANGE IN THE VESSEL SINCE ITS INSTALLATION IN 1984.

- INFORMATION ON LIKELY DEFECT SIZES TOGETHER WITH KNOWLEDGE OF THE MATERIAL PROPERTIES INCLUDING THE EFFECTS OF IRRADIATION ALLOWS THE DEMONSTRATION THAT THERE IS NO CRITICAL DEFECT AND THEN TO ASSESS THE INTEGRITY OF THE REACTOR VESSEL.

IRRADIATION EFFECTS ON BERYLLIUM ELEMENTS

- INDUCED BY NEUTRON IRRADIATION, NUCLEAR REACTIONS IN BERYLLIUM LEAD TO GAS FORMATION CONSISTING OF MAINLY 4 He

- THE BERYLLIUM ELEMENTS WERE DELIVERED BY ALLIS CHALMERS MANUFACTURING IN THE LATE FIFTIES TO THE HFR

- A BERYLLIUM ELEMENT CONSISTED OF A BERYLLIUM FILLER ELEMENT WITH A CYLINDRICAL BORE (ø 52,37 mm) IN WHICH A BERYLLIUM INSERT IS POSITIONED

- SEVERE PROBLEMS DURING CORE LOADING BACK IN 1975 OCCURRED WITH BERYLLIUM ELEMENTS DUE TO RADIATION INDUCED EFFECTS (SWELLING AND EMBRITTLEMENT) AND TO OPERATIONAL HANDLING DAMAGE ON ALUMINIUM PARTS.

- SINCE VESSEL REPLACEMENT, THERE ARE THREE CATEGORIES OF Be ELEMENTS

- REPLACEMENT OF Be ELEMENTS OCCURRED AFTER MORE THAN 25 YEARS OF OPERATION

- DESIGN IMPROVEMENT OF THE NEW Be ELEMENTS BASED ON EXPERIENCE WITH THE OLD Be ELEMENTS

° THE CORE OF THE HFR PETTEN CONSISTS OF 9 X 9 ARRAY CONTAINING 33 FUEL ASSEMBLIES, 6 CONTROL RODS, 19 EXPERIMENT POSITIONS AND 23 BERYLLIUM REFLECTOR ELEMENTS

° 3 CATEGORIES OF BERYLLIUM ELEMENTS SINCE REACTOR VESSEL REPLACEMENT
- IN-CORE REFLECTOR ELEMENTS
- I ROW REFLECTOR ELEMENTS
- CORNER ELEMENTS

DESIGN IMPROVEMENT OF Be ELEMENTS

- NO RADIALLY DRILLED HOLE AND NO ROLL PIN

- INCREASING OF THE CENTRAL BORE DIAMETER

- UPPER-END FITTINGS IN STAINLESS STEEL

REFLECTOR ELEMENTS

IN-CORE REFLECTOR ELEMENT:

A. BERYLLIUM FILLER ELEMENT
- * BERYLLIUM BODY WITH Ø 52.37 BORE
- * ALUMINIUM UPPER- AND LOWER END PIECES

B. BERYLLIUM INSERT
- * SOLID BERYLLIUM BODY (Ø 47.55)
- * ALUMINIUM UPPER AND LOWER END PIECES

I-ROW REFLECTOR ELEMENT:

A. BERYLLIUM FILLER ELEMENT
- * IDENTICAL TO IN-CORE ELEMENT

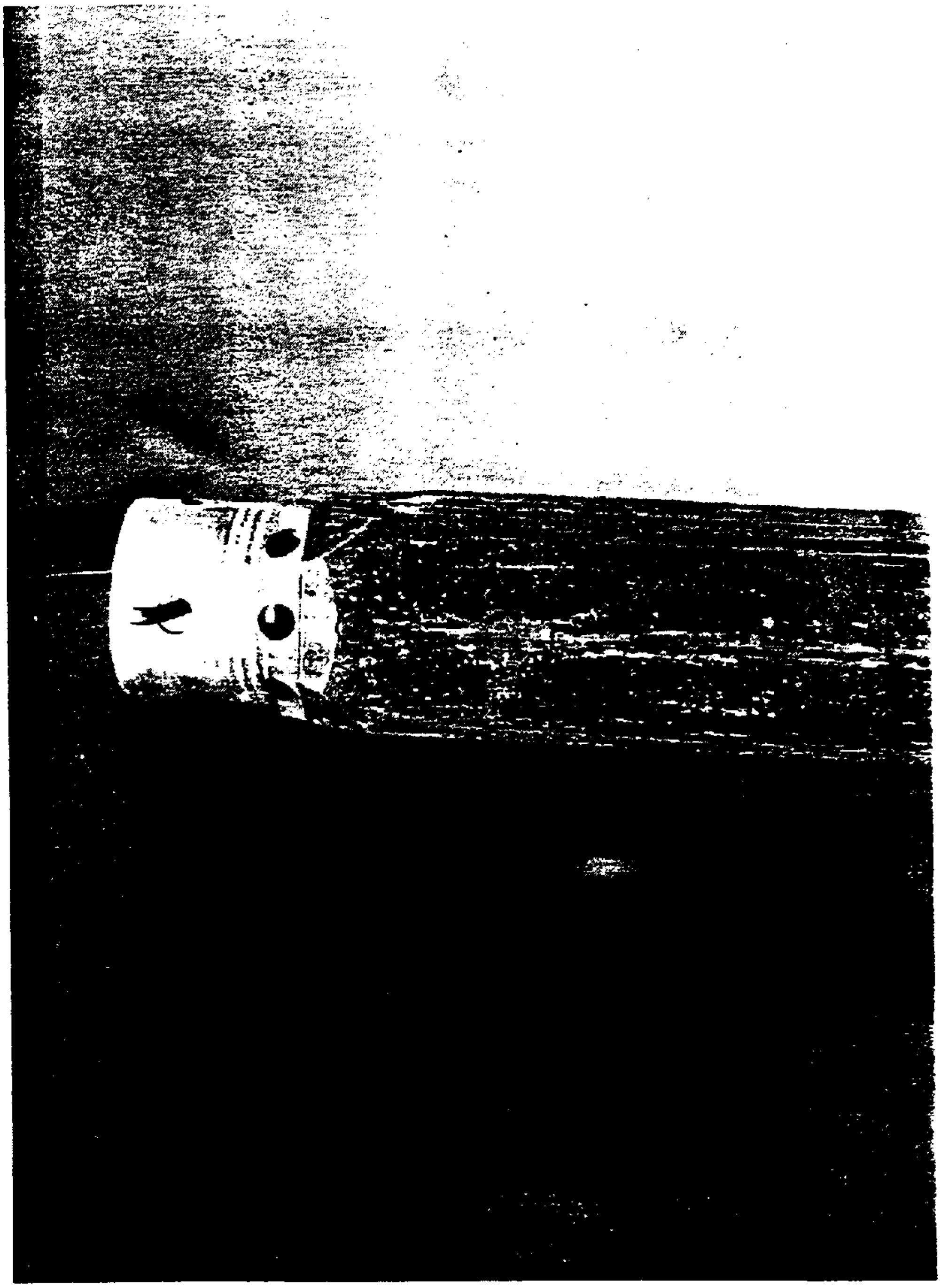
B. BERYLLIUM INSERT
- * MODIFIED IN-CORE INSERT (RESTRICTOR)

CORNER ELEMENT:

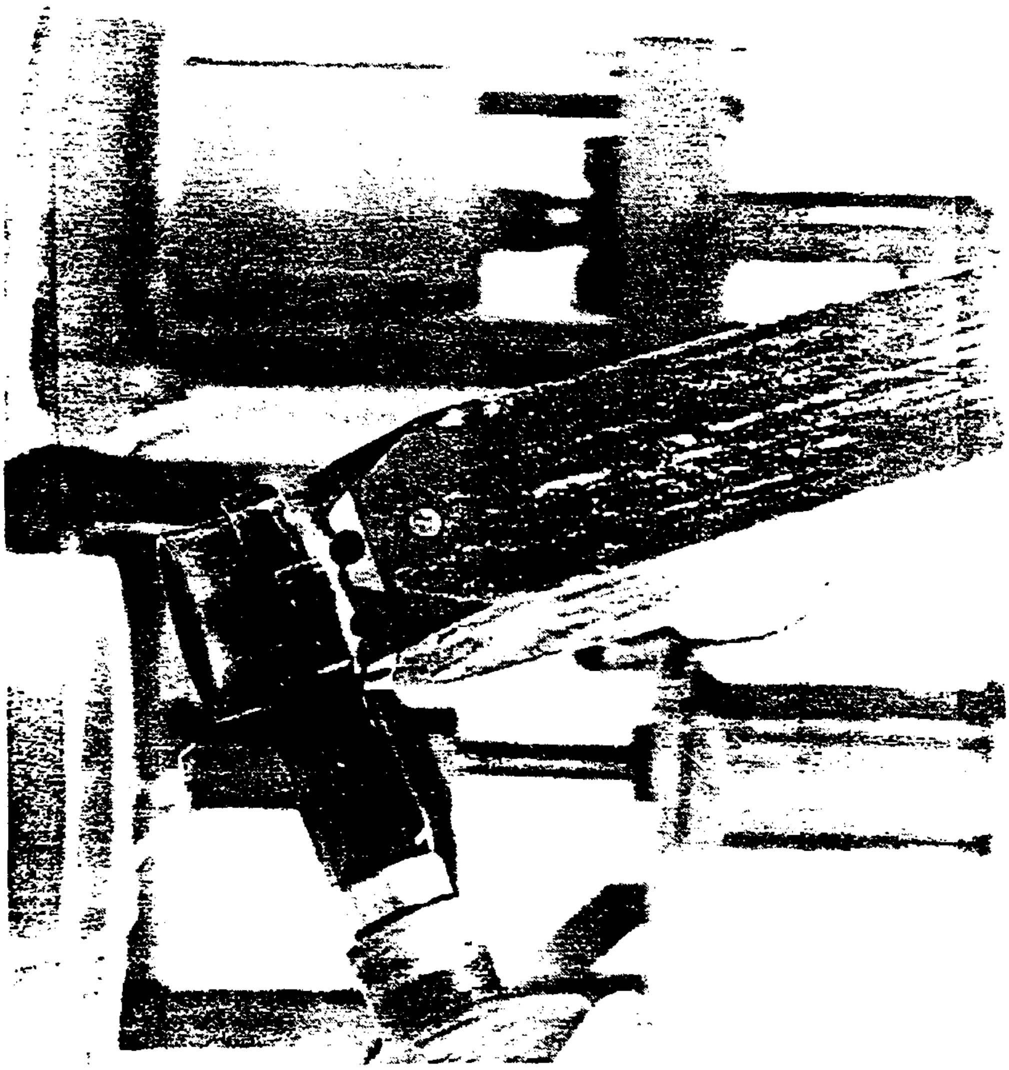
A. ALUMINIUM FILLER ELEMENT
- * ALUMINIUM BODY WITH Ø 75.0 BORE
- * ALUMINIUM LOWER END PIECE
- * STAINLESS STEEL UPPER END PIECE

B. BERYLLIUM INSERT
- * SOLID BERYLLIUM BODY (Ø 70.2)
- * ALUMINIUM UPPER AND LOWER END PIECES

Beryllium Reflector Element
after 25 years of utilization



Beryllium Reflector Element after 25 years of utilization



CONCLUSIONS

- AT THE HFR, REFLECTOR MATERIAL HAS BEEN USED MORE THAN 25 YEARS OF REACTOR OPERATION CORRESPONDING TO A MAXIMUM NEUTRON FLUENCE OF 2.56 10^{26} m^{-2} (E > 1.0 MeV) WITH REGULAR INSPECTIONS AND RE-MACHINING

- RADIALLY DRILLED HOLES AND MOUNTED ROLL PINS ARE THE WEAKEST POINTS ON FILLER ELEMENTS AND INSERTS. SEVERE CRACKING WAS OBSERVED IN THESE LOCATIONS

- PRESENCE OF BLISTER FORMATIONS DUE TO RADIATION INDUCED HELIUM

- HANDLING DAMAGE CAN BE REDUCED BY A PROPER CHOICE OF END FITTING MATERIAL

- THE AVERAGE SWELLING OF THE REFLECTOR ELEMENTS AFTER 21 YEARS OF HFR OPERATION (1962-1983) WAS 0.9% WITH 1.39% AS MAXIMUM AND 0.61% AS MINIMUM

(For Presentation at the Conference on Expert System Applications for the Electric Power Industry, September 9-11, 1991, Boston, Massachusetts)

DESIGN, INSTALLATION, AND INITIAL USE OF A SMART OPERATOR AID

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ABSTRACT

This paper reviews the design, installation, and initial use of a smart operator aid that was developed to assist licensed personnel with the performance of power adjustments on the 5-MWt MIT Research Reactor (MITR-II). The aid is a computer-generated, predictive display that enables console operators to visualize the consequences of a planned control action prior to implementing that action. The motivation for the development of this display was the observation that present-time control decisions are made by comparing a plant's expected behavior to the desired response. Thus, an operator will achieve proper control only to the degree that he or she is capable of anticipating plant response. This may be difficult if the plant's dynamics are non-linear, time delayed, or counter-intuitive. A study was therefore undertaken at the MITR-II to determine whether operator performance could be improved by using digital technology to provide visual displays of projected plant behavior. It was found that, while the use of predictive information had to be learned, operator performance did improve as a result of its availability. Information is presented on the display's design, the computer system utilized for its implementation, operator response to the display, and the possible extension of this concept to the control of steam generator level.

DESIGN, INSTALLATION, AND INITIAL USE OF A SMART OPERATOR AID

INTRODUCTION

Since the late 1970s, the Massachusetts Institute of Technology has been engaged in a program to develop and demonstrate experimentally advanced technologies for the control and instrumentation of nuclear reactors. Much of this work has been focused on the closed-loop digital control of reactor neutronic power. The reactivity constraint approach, which is a supervisory method to ensure the absence of challenges to a reactor's safety system as the result of an automated control action, and period-generated control laws, which are a means of accurately tracking a demanded trajectory, are two results of this effort [1-4]. In parallel with the development of these and other analytic control techniques, studies have been performed of the man-machine interface with emphasis on operator acceptance of digital technology. To that end, an examination was conducted in 1987-1988 of the possible benefits of providing the licensed operators of the 5-MWt MIT Research Reactor (MITR-II) with a computer-generated predictive display that would enable them to visualize the consequences of a planned control action prior to implementing that action. A series of five such displays were designed and evaluated [5]. It was found that the licensed reactor operators preferred a display that provided them with a full record of the power history as well as three power projections, each corresponding to one of their three possible control options which were to withdraw the control device, to maintain its position constant, or to insert it. Operator response to this display was positive. However, at the time of the original study, the limited capacity of the MITR-II's existing digital control system made it impractical to maintain the display as a regular feature in the reactor control room. The computer equipment has recently been upgraded and both predictive displays as well as other types of smart operator aids can now be made available to MITR operators on a routine basis.

The specific objectives of this paper are to (1) review the design of the predictive display that was developed to assist operators with power maneuvers, (2) describe the digital control system on which the display is generated, (3) report initial operator reaction to the display, and (4) discuss the possible extension of this technology to commercial reactors.

assist them in the conduct of power increases on the MIT Research Reactor. It combines derivative, current, and predictive information. The operator is provided with a full record of the power history as well as three power projections. The point from which these projections emanate is the current power level. Each projection corresponds to one of the operator's three possible control options. The predictive information is obtained by solving a reactor model (the alternate dynamic period equation [10]) at a rate faster than real time. The transient shown in Figure Four is a power increase from 1000 kW to 2000 kW. The display has a projection time of twenty seconds. The power level is currently 1650 kW, the regulating rod is at 4.42 inches, and the reactor is on a positive period. Eighty seconds have elapsed since the start of the power maneuver. The existence of a positive period implies that the control device has already been withdrawn beyond the critical position. The uppermost trajectory of the display should therefore be interpreted as meaning that if the console operator were to continue withdrawing the rod for the next twenty seconds, then the power level would be 2200 kW. Similarly, the middle trajectory means that if the operator were to hold the regulating rod at its now existing position of 4.42 inches, then at the twentieth second, the power level would be 1950 kW. (Note: The reactor is already on a period at time zero. Hence, power will rise even if the rod position is kept constant.) Finally, the lowermost trajectory indicates that if the console operator chooses to insert the regulating rod for the next twenty seconds, then the power level will reach 1770 kW.

Figure Six illustrates the utility of displaying power projections in addition to the derivative and current information that is typically available from conventional strip chart recorders. As shown here, the power is being raised to 2000 kW and the current power is 1500 kW. The lowermost projection indicates that if the operator now inserts the control device and continues that insertion for the next twenty-five seconds, the power level will attain the demanded value, 2000 kW. Any other action will result in an overshoot. The availability of this predictive capability provides the operator with information that he could not otherwise obtain and which he might not be able to infer given the non-linear nature of reactor dynamics to say nothing of control rod strengths that vary as a function of position and xenon distribution. The operator is still responsible for the safe operation of the reactor, but the display provides additional insight that may facilitate his discharge of that responsibility.

DIGITAL COMPUTER SYSTEM FOR DISPLAY IMPLEMENTATION

Figure Seven shows the configuration and function of the MITR-II's Advanced Control Computer System (ACCS). The ACCS was recently installed on the MITR-II and it is now in routine use for experimental work on the digital control of reactor power and temperature [11]. It consists of five separate computers linked in a multiple-

PREDICTIVE DISPLAY DESIGN

The motivation for the development of the smart operator aid described here was a study on the approach used by licensed MITR operators to raise and lower the reactor power [6,7]. Specifically, it was noted that present-time control decisions are made by comparing a plant's expected behavior to the desired response. This observation was in agreement with the results of far more extensive studies such as those by Kelly and Sheridan [8,9]. The question therefore arose as to whether or not operator performance could be improved by using digital technology to provide visual displays of projected plant behavior. The operator could then see the consequences of a particular control action prior to actually implementing it. It was recognized at the outset that the basic premise of this work was somewhat controversial. Predictive displays might only hinder the operator's normal decision process. Or, worse yet, an error in the analytic model used for the predictions might result in erroneous information. Or, such displays might be so successful that operators would gradually lose their learned ability to predict plant behavior. The uncertainty of the outcome was of course the reason for undertaking the experiments.

Five displays were developed and evaluated as part of this project. These ranged in complexity from a simple scheme showing only the current plant status to ones that provided the operator with various combinations of derivative, current, and predictive information. Relative to the design of these displays, it should be noted that the MITR-II is equipped with six shim blades and one regulating rod. These are for coarse and fine control respectively. During routine operation of the reactor, use is normally made only of the regulating rod. Thus, at any given moment, the console operator has only three control options. These are to withdraw the regulating rod thereby inserting positive reactivity, to hold the regulating rod's position constant thereby leaving the reactivity unchanged except for inherent feedback effects, or to drive the regulating rod inwards thereby inserting negative reactivity. These operations are designated as OUT, HOLD, and IN respectively. The display format assumes that the option selected (OUT, HOLD, or IN) will be maintained continuously. This is unrealistic because the operator will most likely use some combination of withdrawal, hold, and insertion signals to accomplish the power change. Nevertheless, it is an extremely useful approach because the trajectories shown bracket all possible sequences of control signals. The extent of the trajectories in time is a selectable quantity with operators being able to specify any duration from ten to thirty seconds.

Figures One-Five show the five displays that were developed as part of the original MITR-II study. Descriptions of each and the rationale for its design have previously been given [5]. Figure Four is the display that the licensed operators preferred as a smart aid to computer/single-task architecture. The role of one of these computers, an IBM-XT 8088, is to receive validated signals from the data acquisition computer and to display model-based predictive information. Display projections of as much as 60 s are possible with the screen being updated once per second. Given below is a summary of the major features of the ACCS.

The Advanced Control Computer System consists of five interconnected computers, each responsible for a different set of tasks. In addition to permitting flexible operation and allowing testing of computation-intensive concepts, this system enhances safety because safety-related functions such as the reactivity constraint algorithm can be run separately from control law calculations. Software for the former is well-established, is based on first-principles, and is invariant [1]. In contrast, software for the latter will probably always be under development because one objective of the MIT program on reactor control and instrumentation is to identify new methods of control and to make their use a practical reality. The function of each of the ACCS's components is as follows:

1. Rack-Mount 80386: This data acquisition IBM-AT computer is assigned three major tasks. First, it collects data from a maximum of thirty-two sensors, performs signal validation on the collected data, outputs the validated information to up to four other computers, and displays the validated information on the console CRT monitor. Second, it computes the maximum allowed control signal using the supervisory reactivity constraint algorithm [1] as well as limits of other MITR technical specifications, receives the requested control signal from the other computers, compares that signal with the one calculated by the supervisory algorithm, outputs the more conservative signal to the control rod motors, and displays the control decision on the screen. This computer's third function is to write the desired data to the permanent disk. Changes to this computer's software are rare.

2. MicroVAX-II: The VAXstation II/GPX is a machine dedicated for intensive floating-point computations. Engineering and control calculations such as are required for the MIT-SNL minimum time control laws [2], are performed on this machine. The MicroVAX-II receives validated information on the reactor from the data acquisition system (IBM-AT). It then calculates the demanded control signal from whatever control law is being tested, and exports that signal to the data acquisition system for output to the control rod motors. Changes in this system's software are expected to be frequent.

3. IBM-Compatible 80386: This is a high-speed machine on which computer programs are first edited, compiled, and finally linked to form an executable module. This machine is capable of supporting automated reasoning using PROLOG, LISP, or C. It is designed to be compatible in all details with the Rack-Mount 80386 data acquisition system.

4. IBM-XT 8088: This computer's role is to receive validated signals from the data acquisition computer and to display model-based predictive information or a safety parameter display on its screen.

5. LSI-11/23: This unit is connected to the MicroVAX-II for the purpose of providing an independent machine on which a model of a reactor can be run. This permits new controllers to be programmed on the VAXstation II/GPX and tested against a simulation model running on the LSI-11/23 prior to the

performance of actual closed-loop runs on the reactor. This approach has the advantage that new programs are tested under realistic conditions. In particular, signals must be passed between two computers as is done for actual implementations. Previously, simulations were done with the new control law and the model both running on the same unit.

Integration of components within the data acquisition computer was accomplished through a passive back plane which is basically a non-intelligent bus that allows only lines such as data, status, and timing to be passed. Integration of the five separate computers was achieved through use of RS-232 serial communication.

In addition to the above five computers, the ACCS is equipped with a broad-range power sensor that spans about seven decades. The output of this chamber is connected to a KEITHLEY Model 485 auto-ranging picoammeter which is equipped with a Model 4853 IEEE-488 interface. This unit is basically a 4-1/2 digit (4 significant digits with sign) auto-ranging picoammeter with seven DC current ranges. The design of this system allows the Rack-Mount 80386 data acquisition system to receive an on-scale reading of the output of the neutron-sensitive compensated ion-chamber in a digitized form. This makes the system less prone to electrical interference or noise than if a signal were first obtained in analog form and then converted for digital operation. A further advantage is that the output of the Model 485 is digitized.

In summary, the ACCS's multiple-computer/single-task architecture with auto-ranging picoammeter offers several advantages over single computer systems. These include separation of safety-related software that is in finished form from control software that is under development, transmission of both the scale and reading of a power instrument so as to permit operation over many decades, the acquisition of digitized signals directly from the neutron sensors, and the performance of interactive simulations.

OPERATOR RESPONSE TO PREDICTIVE DISPLAY

Operator response to the predictive display has, for the most part, been either neutral or positive. As a general rule, experienced operators do not refer to the display when performing power adjustments under standard plant configurations. However, they do refer to it under other circumstances. For example, the reactivity worth of the MITR-II's regulating rod decreases rapidly once it is withdrawn more than half of its range of travel. Hence, operators normally reshim the reactor so that the regulating rod is close to its in-limit before using that rod to perform a power increase. Under these conditions, experienced operators do not need the display. However, if a power increase is initiated without first positioning the regulating rod so that it is low in the core, then reference may be made to the display. Another observation is that novice operators make more use of the display than experienced ones. This is to be expected because the former are using knowledge-based learning while the latter rely more on pattern recognition skills. These observations are consistent with those obtained by others during more elaborate tests of smart tools. For example, Sun gives a discussion of the expert system Emergency Operating Procedures Tracking System (EOPTS) in which operator performance was studied [12]. He reports that:

> "In terms of time response, the expert system typically allowed the operator to respond faster. The exception is in the case of fast, simple transients with well-trained operators, where the system slows operator response." ... "Operating crews using the expert system tend to be rule-based and therefore faster responding, while those using the flow charts tend to be knowledge-based which results in slower response. Use of the expert system improved operator response times, decreased actuation errors, and thus improved operational reliability."

It is instructive to note operator attitudes concerning both the derivative and the predictive information depicted in the display. In this regard, comments obtained during the original MIT study in which a comparison was made of the five displays shown in Figure One are germane. Specifically, both verbal and written operator comments suggested that the derivative information contained in schemes II and IV and, to a lesser degree in scheme V, was reinforcing their existing approach to the control of reactor power. Namely, they were using the power history traces to judge the rate at which the power was changing and from that determining the proper time for initiation of reactivity removal. Relative to the predictive information, operators reported the opposite result. That material did not mesh smoothly with their existing concepts for exercising control and the effective use of the predictive displays had to be learned. Specifically, the three power projections that represented the OUT, HOLD, and IN control options did not correspond to their thought processes. Operators stated that they projected the consequences of only the current control action and did not think in terms of the three simultaneous options provided by the displays. Moreover, operators indicated that they did not actually predict the expected power level. Rather they recognized limiting conditions for maintaining proper control and restricted operation to those bounds. For example, an operator's mental model might consist of the realization that a power transient could be halted provided that the reactor period was longer than a certain value and that the regulating rod was below a certain position. In this respect, the lower of the three projections was the more useful because it clearly showed the conditions under which a transient could be terminated.

As noted above, operators had to learn the proper usage of the predictive displays. Several errors were noted during the learning phase. For example, operators would occasionally treat the endpoint of one of the predictor lines as the current power level.

Also, they sometimes mistook one projection for another. This occurred most often after a change was made in the control signal. Once the predictive display was understood, operators were unanimous in their favorable opinion. They reported that the predictive information was of particular value during the final stages of a transient. Evidence supporting these statements was obtained by initiating transients with only a ten second prediction horizon. Operators would invariably increase the extent of the projections in time as the transient neared completion. Also, each operator formulated a strategy for utilizing the power projections. The first strategy adopted was typically to withdraw the rod continuously until the OUT predictor first crossed the target power line. The regulating rod's position was then kept constant until the HOLD predictor crossed the target power line. The regulating rod was then inserted. As they became more experienced with the use of the displays, most operators began to recognize that there was no particular rationale for this approach. Moreover, it was observed to result in overshoots should the regulating rod be initially at the higher end of its range of travel. The strategy ultimately adopted was to withdraw the regulating rod until the derivative information indicated that the maximum allowed rate of rise had been achieved. Rod insertion would be initiated when the IN predictor became tangent to the target power line. In this regard, it was crucial that the extent of the prediction in time be such that the IN predictor was always concave downwards. Otherwise, the time required to halt the power transient could not be inferred.

In summary, it is worth emphasizing that operator response to a smart tool such as the predictive display described here is in large measure determined by the degree to which that tool is designed to support human cognitive needs. In this respect, the tool should reinforce both the operator's understanding of the plant and his or her mental approach to the analysis of plant behavior. Displays that show trends and predictions satisfy the first of these two criteria because such information will assist operators in anticipating plant response. As for the second criteria, graphics should be emphasized so that an operator need only look at a display to comprehend it. This approach allows experienced operators to continue using their pattern recognition skills. In contrast, were text to be displayed, an operator would have to switch to a deductive mode of reasoning in order to make sense of the information. Additional discussion of these criteria has previously been given [13].

EXTENSION TO NUCLEAR INDUSTRY: STEAM GENERATOR LEVEL DISPLAY

Aspects of pressurized water reactor (PWR) operation for which predictive displays might be of benefit include maintenance of pressurizer level during plant heatups, adjustments of the soluble boron concentration, and the damping of xenon oscillations. Relative to boiling water reactors (BWRs), the coordination of recirc pump speed with adjustments of turbine load might benefit from predictive displays. Described here is work that has recently been completed at MIT concerning the development of a predictive display for steam generator level.

Steam generator level is monitored by narrow and wide range indicators that are located in the steam generator downcomer. These are used by plant operators to maintain level in the steam generator within an allowed band thereby ensuring adequate heat removal from the reactor's primary system and the absence of carryover to the turbine. Unfortunately, the control of steam generator level is difficult because the level sensors are subject to the counterintuitive effects of shrink and swell. The former refers to the temporary reduction of the water level in the steam generator during an increase in mass inventory. It can be induced either by increasing the feedwater flow to the steam generator or by decreasing steam flow to the turbine. An increased feedwater flow adds cold water to a steam generator which decreases the average enthalpy of the water inventory and results in both a reduction in the average quality and a collapsing of vapor bubbles. The vapor bubble collapse in turn causes the downcomer water level to drop even though the mass inventory is rising. A decrease in the steam flow to the turbines has the same effect as increasing the feedwater flow, but through a different mechanism. Namely, when steam flow is reduced, pressure increases in the steam generator. This pressure increase causes vapor bubble collapse that in turn leads to the shrink effect. Swell is the reverse of the shrink effect. Namely, a decrease in the mass inventory leads to a temporary increase in the water level. Again, this effect can be brought about either through a decrease in feedwater flow or an increase in steam flow with the mechanisms being as described above.

Analog, and more recently some digital, controllers are used to adjust the position of the main feedwater valve and thereby maintain a balance between steam demand and feedwater flow. However, the analog controllers sometimes malfunction during operation at less than 15% of rated plant power. The result is that manual control is used during plant startup with a human operator maintaining level by adjusting a small valve that allows some feedwater flow to bypass the main valve. This bypass valve is used because, being smaller than the main valve, it is more responsive. Also, the relation between its position and flowrate is more linear than that of the main valve. Unfortunately, plant trips still do occur on occasion. In part, these happen because human operators sometimes have difficulty estimating both the magnitude and duration of shrink and swell effects. For example, on a power increase, the void fraction in the tube bundle region increases causing a temporary rise in the liquid level of the downcomer region. An operator might therefore cut back on feedwater flow. However, such an action would be a mistake because the increased steam flowrate will soon necessitate additional feedwater flow. A predictive display would make this obvious and such a display was therefore developed as part of the MIT program on advanced instrumentation and control for nuclear reactors [1,2].

The MIT Steam Generator Level Display Program (SGLDP) is a predictive display that operates in real time on an IBM-compatible PC. The basis of its projections is a simplified model of steam generator dynamics. The program projects the steam generator narrow gauge level signal for three cases: control valve being closed, control valve position maintained constant, and control valve being opened. The operator can select the speed at which the valve is to be opened or closed. The demanded reactor power is also displayed so that the operator can observe the correlation (or lack thereof) between power demand and change in anticipated level in the steam generator. The model gives the narrow-range steam generator level in the downcomer as a function of the steam and feedwater flowrates. Four terms are included. The first is a mass capacity term that reflects the net difference between the steam and feedwater flowrates. The second allows for shrink/swell effects associated with changes in feedwater flow. The third is similar except that it is for changes in steam flow. The fourth allows for short-lived mechanical oscillations that can be caused by the addition of feedwater to the generator. This model's accuracy was verified by comparison with a much larger and more rigorous model that had been benchmarked against plant data. The simplified model used in the display is believed to be accurate to within 2 cm. Projections of up to 200 s are possible with a display update frequency of 1 s.

Figure Eight shows the predictive display for steam generator level. The upper portion shows the reactor power and the lower portion depicts steam generator level. Reactor power was initially at 10% of rated and it is being raised at 2.5% of rated per minute. Derivative information, the steam generator level for the previous 100 s, is shown together with the current level. Emanating from the current level are the three projections, each corresponding to a possible control option (valve opened at selected rate, held constant, or closed at selected rate). In the actual display, each option is shown in a different color. The advantage of this display is that an operator can visualize the effects of adjusting the position of the feedwater control valve before doing so. This capability should result in more reliable operation because even though operators are trained to and do understand the counterintuitive nature of shrink and swell, they may have difficulty quantifying those effects. Thus far, no trials of this display have been conducted either by simulation or in an actual plant.

CONCLUSIONS

The concept of providing predictive displays as an aid to those responsible for controlling complex processes is not new. Early applications of the approach concerned the diving controls for submarines and the landing system for Apollo spacecrafts [8]. More recently, the technique has been utilized for air and railroad traffic control [14,15]. Relative to the nuclear industry, the use of predictive display technology as an operator aid for the control of steam generator level in pressurized water reactors (PWRs) has been previously suggested though not implemented [16]. The work done at MIT on the provision of predictive information in the form of a smart operator aid has achieved the following. First, a display format has been devised that licensed MITR operators have judged to be of benefit. Second, accurate, fast-running models have been developed for use in projecting both neutronic power and steam generator level. Third, the neutronic power display has been demonstrated to be of advantage for the conduct of some types of power increases on the 5-MWt MIT Research Reactor.

Much remains to be done before the use of predictive information will become routine in the nuclear industry. In particular, regulatory issues remain to be explored. These have not yet been broached as part of the MIT trials because all use of the neutronic power display has been under an approved experimental protocol [5]. Issues such as those raised at the outset of this paper concerning operators becoming overly dependent on a display or the consequences of inaccuracy in a projection remain to be addressed. While not minimizing those challenges, predictive displays may offer a means of gradually incorporating digital technology in reactor control rooms and thereby bridging the gulf that now exists between manual and fully automated control of nuclear power facilities.

ACKNOWLEDGMENTS

The contributions of Professors David D. Lanning and John E. Meyer to the development of the steam generator level display model are gratefully acknowledged. Appreciation is expressed to Ms. Georgia Woodsworth for typing the manuscript. Mr. Ara Sanentz performed the layout and prepared the graphics. This research was supported by the U.S. Department of Energy under Contract No. DE-FG07-90ER123.90.

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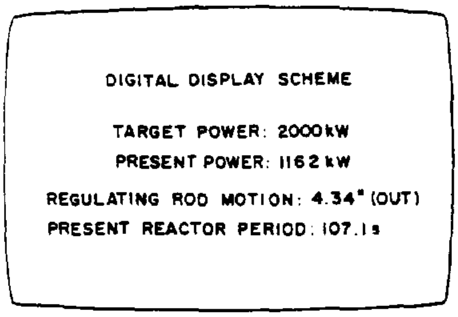

Figure 1. Current Information in Numeric Form (Scheme I)

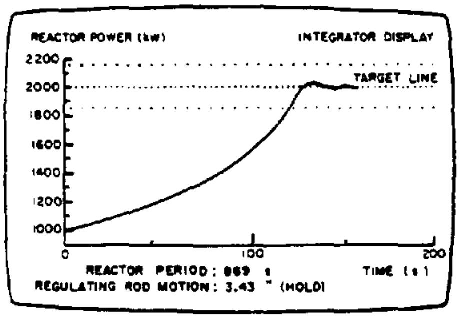

Figure 2. Current and Derivative Information (Scheme II)

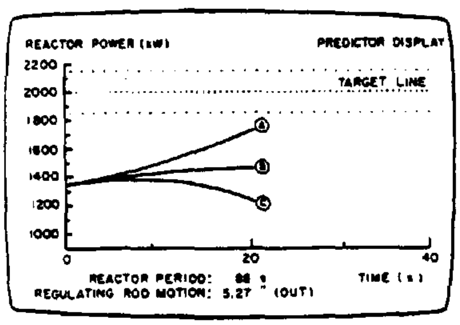

Figure 3. Current and Predictive Information (Scheme III)

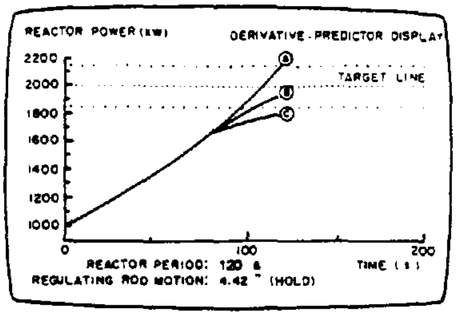

Figure 4. Derivative, Current, and Predictive Information (Scheme IV)

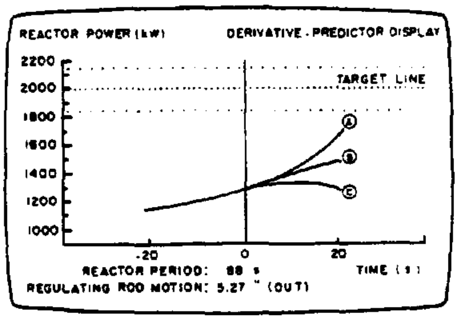

Figure 5. Limited-Derivative, Current, and Predictive Information (Scheme V)

KEY

- (A) PREDICTED RESPONSE IF CONTROL ROD IS WITHDRAWN.
- (B) PREDICTED RESPONSE IF CONTROL ROD IS HELD CONSTANT.
- (C) PREDICTED RESPONSE IF CONTROL ROD IS INSERTED.

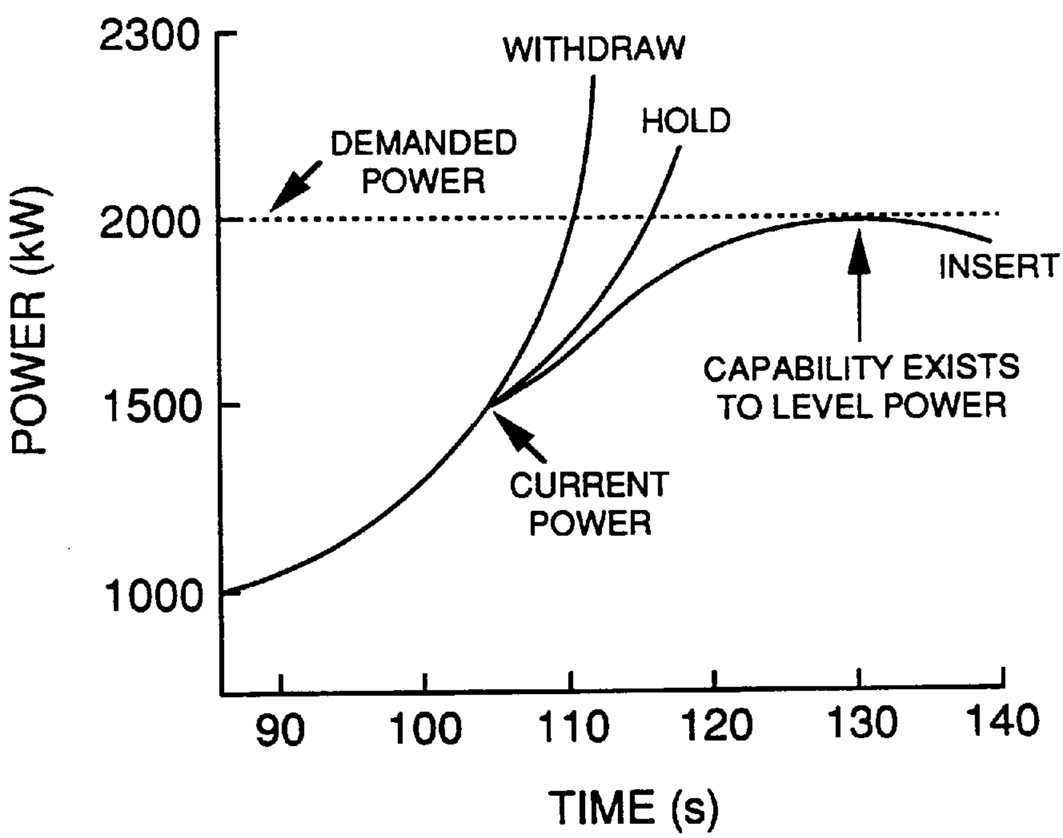

Figure 6. Predictive Display

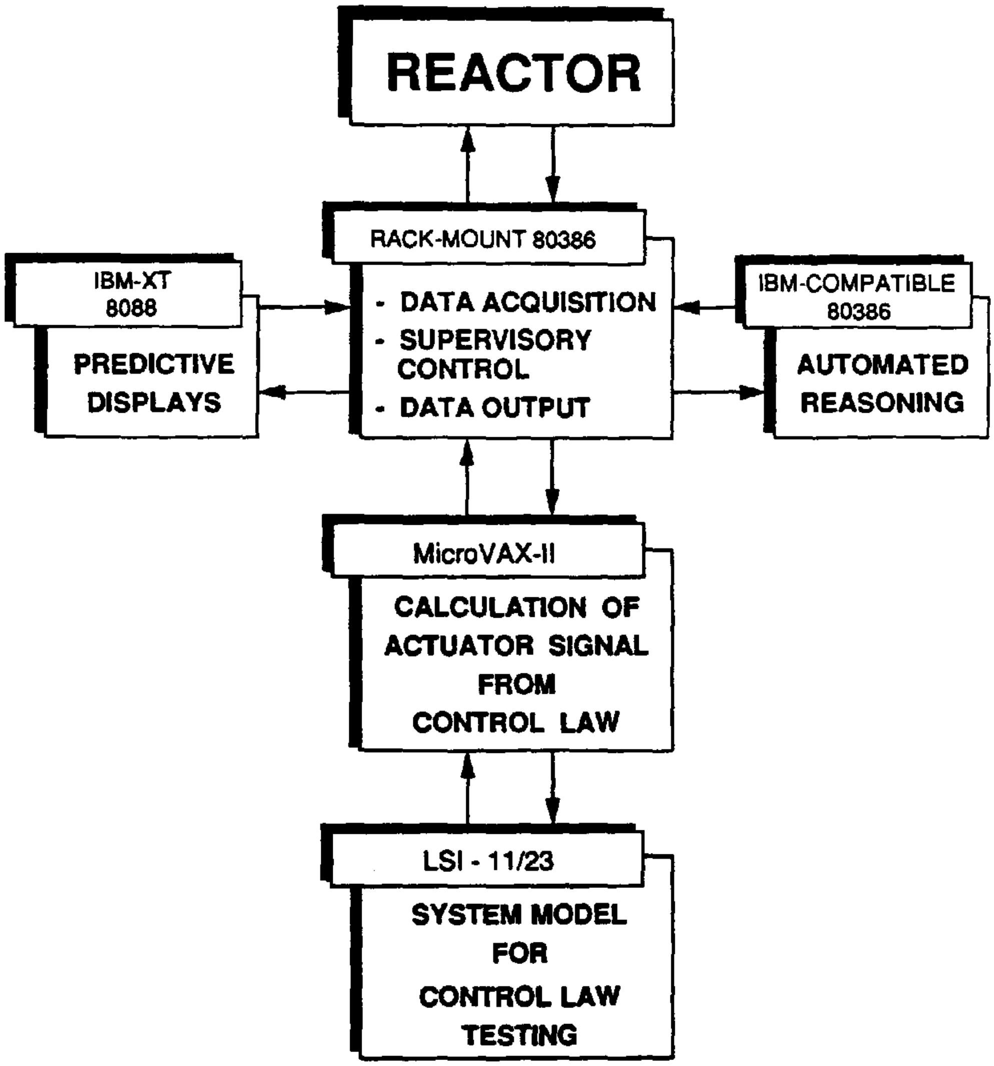

Figure 7. Configuration and Functions of MITR-II Advanced Control Computer System

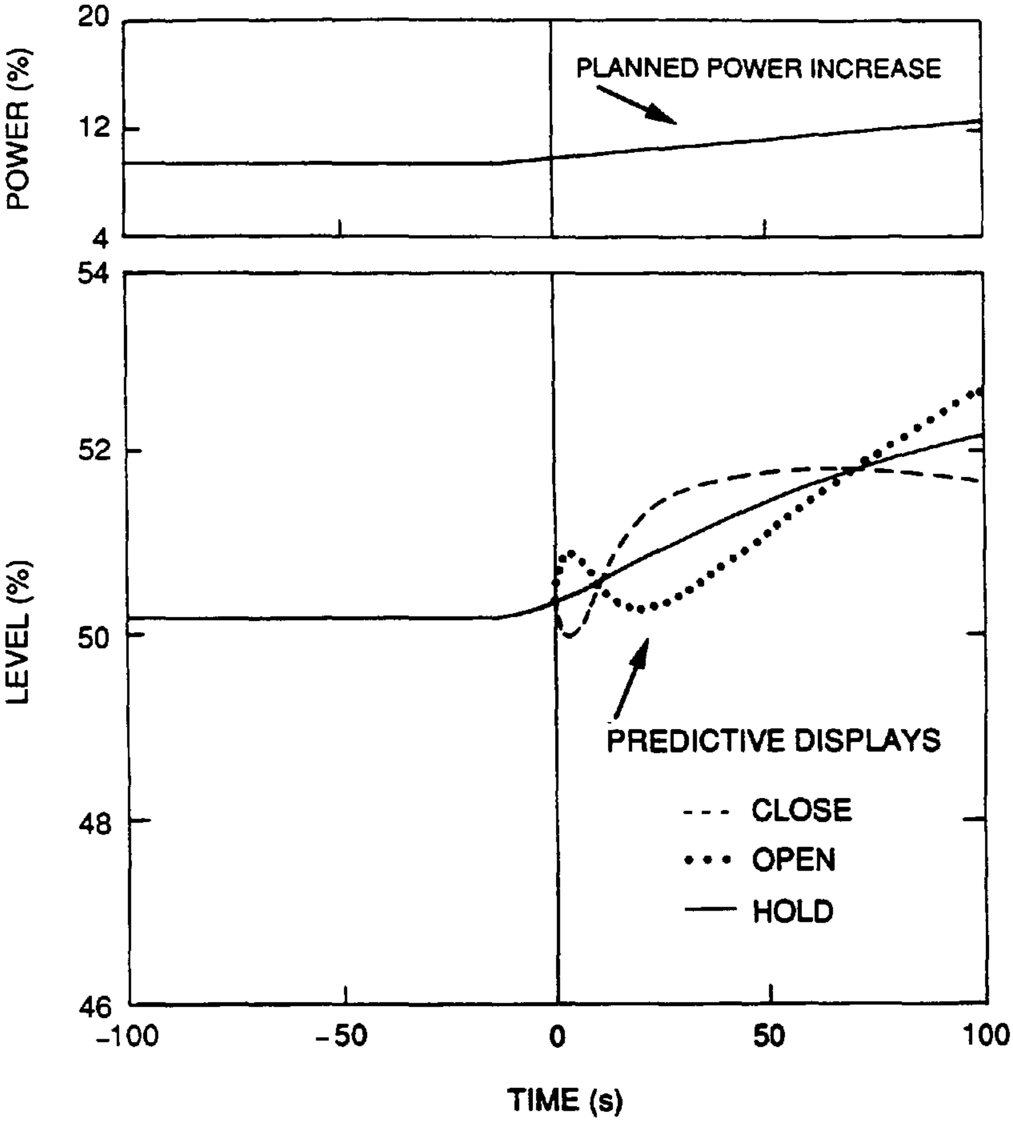

Figure 8. Steam Generator Level Display After Start of Power Ramp with Ramp Rate of 2.5% of Full Power Per Minute with Reactor Initially at 10% of Rated Power

DESIGN, INSTALLATION, AND INITIAL USE OF A SMART OPERATOR AID

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Objectives

1. Review design of predictive display.

2. Describe digital control system on which the display is generated.

3. Report initial operator reaction to the display.

4. Discuss extension of this technology to commercial reactors.

Rationale for Predictive Displays

- Control decisions are made by comparing a plant's expected behavior to the desired response.

- Operator performance might be enhanced if Operator could visualize consequences of a particular control action prior to implementing it.

- But certain caveats exist:

  - Might a display hinder the normal thought process?

  - What if there were an error in the predicted information?

  - Might operators lose their learned ability to predict plant behavior?

Predictive Display Design

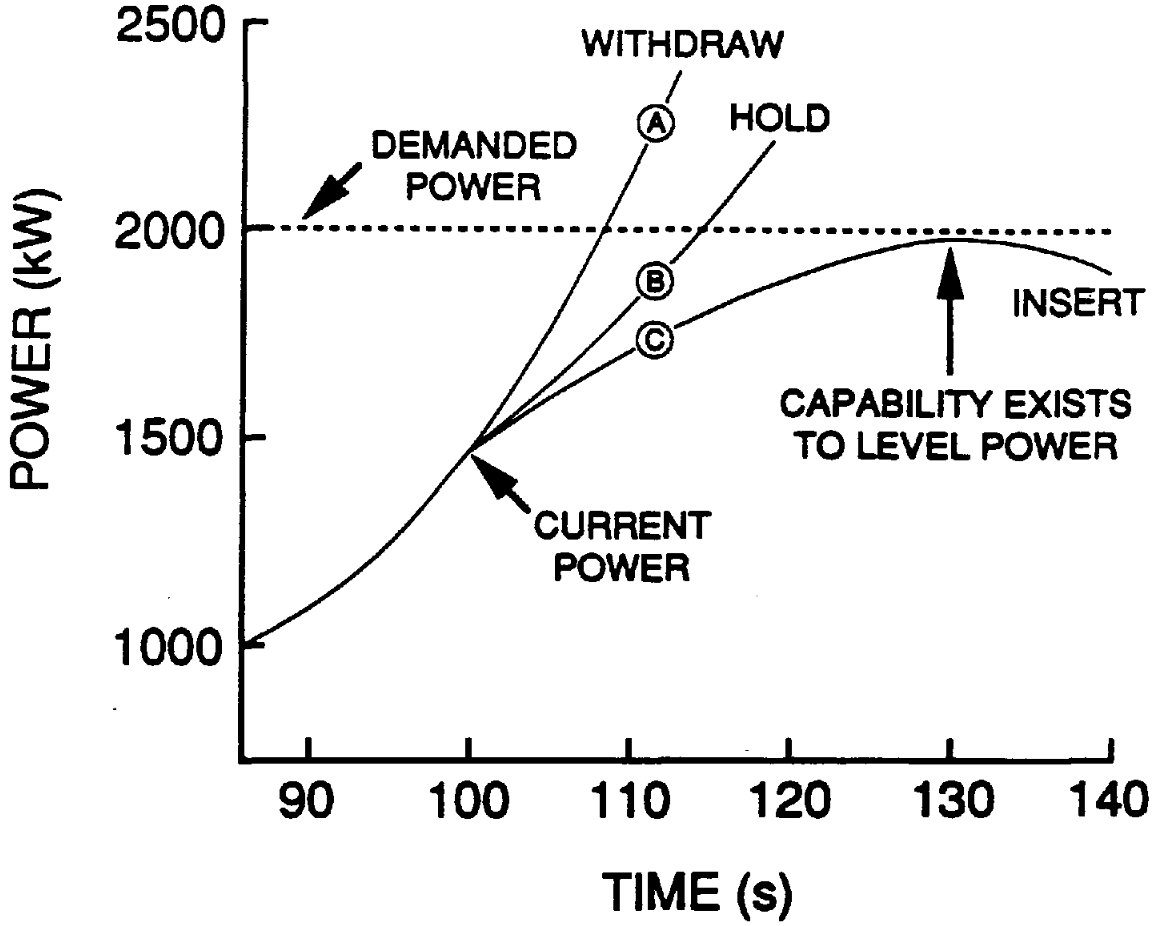
- Original display was developed to assist in the control of reactor neutronic power.

- Operator has only three control options: Withdraw, maintain constant, or insert a control device.

- The display shows three projected power trajectories, one corresponding to each control option.

- Extent of the displays is selectable in time.

- Operators expressed preference for a display that provides a full record of the power history as well as the three power projections.



Display for Man-Machine Interface

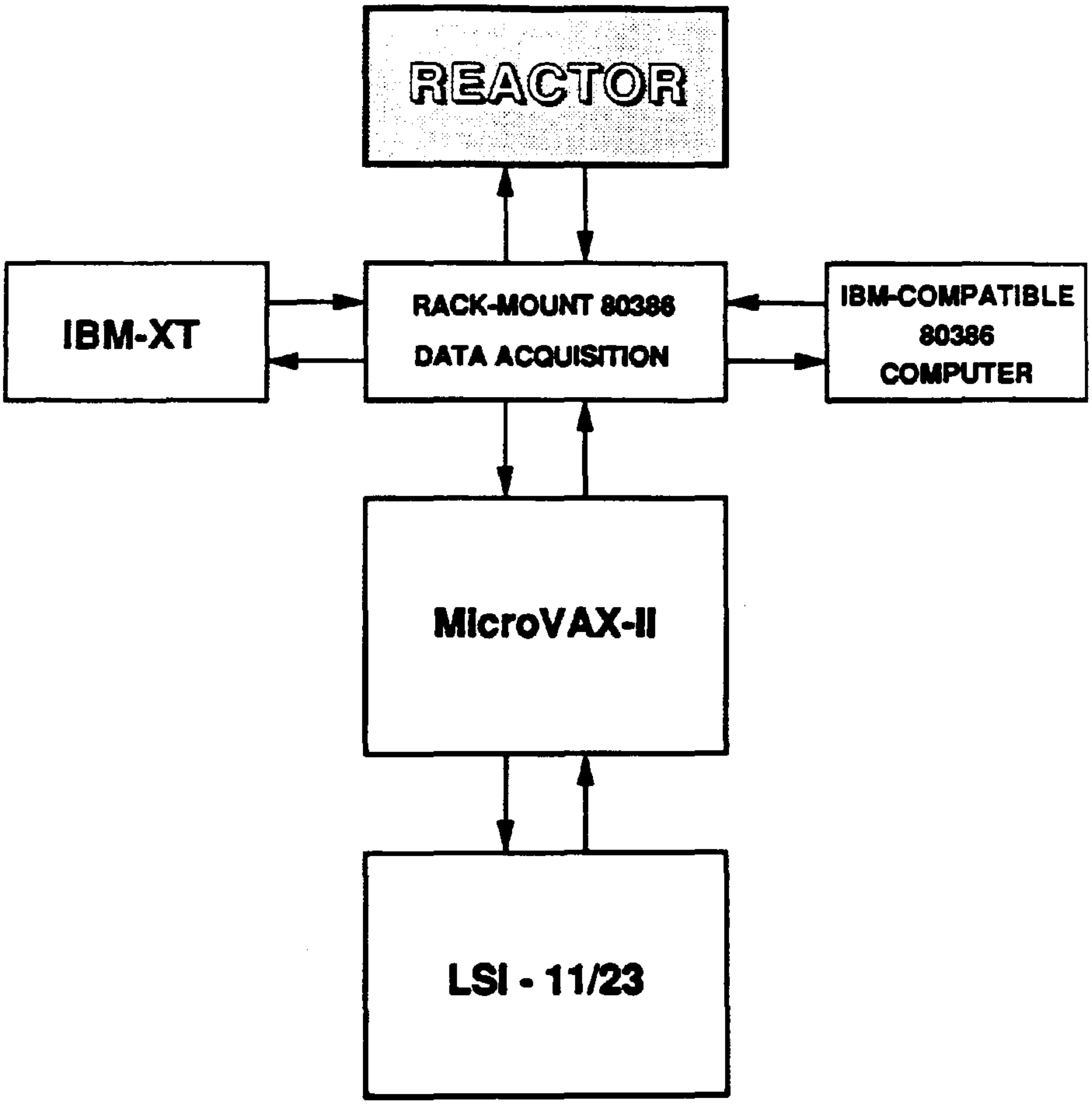
Advanced Control Computer System

- Permit digital operation over many decades to support automated startups.

- Separate software essential to safety from control software subject to periodic change.

- Provide capability for improved man-machine interface, automated reasoning, predictive displays, and interactive simulation.

- Real-time operation and high numerical throughput.



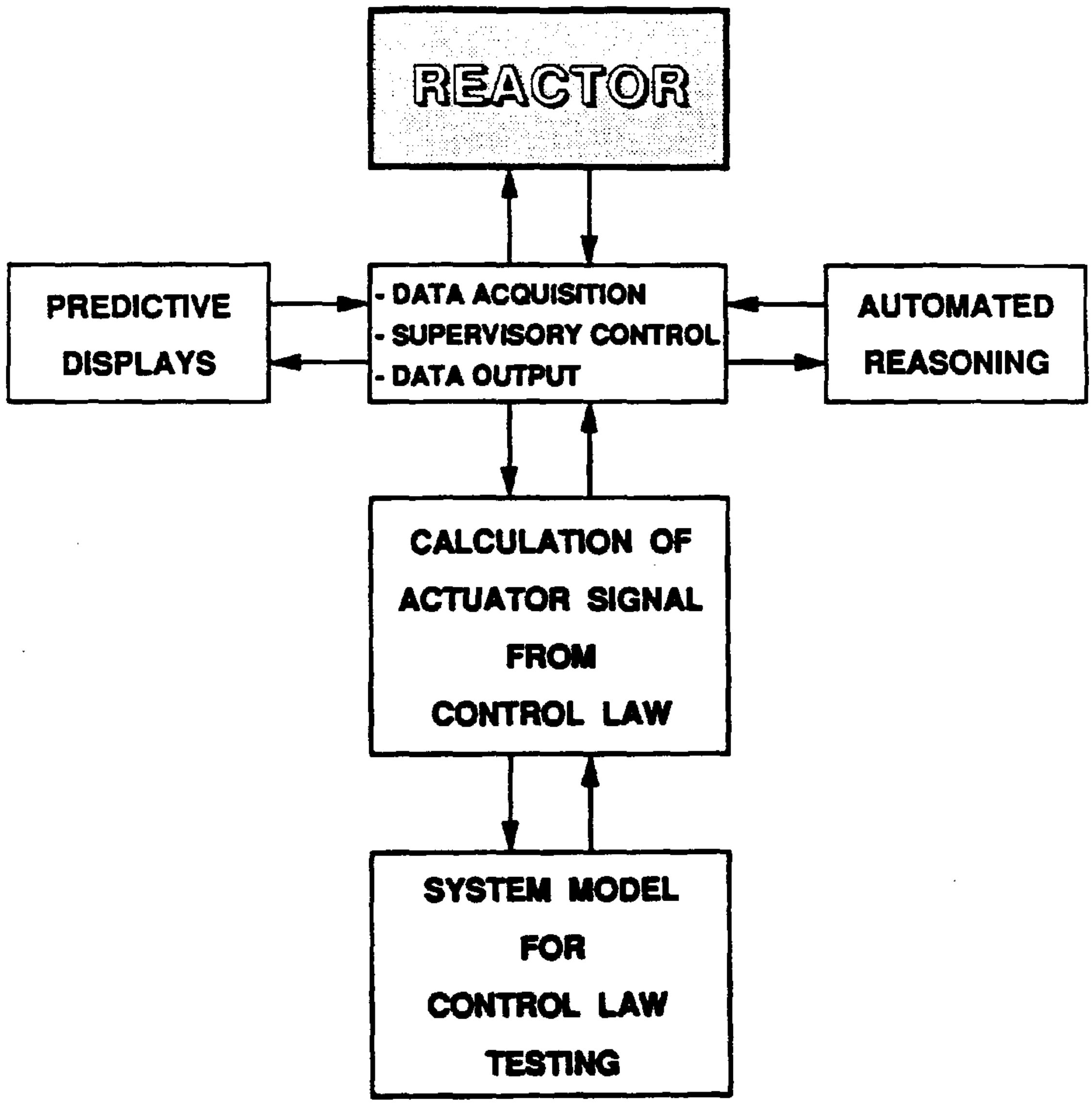
Hardware Configuration of MITR-II Digital Controller

Description of ACCS

Five separate computers form a single-task system through use of passive back plane.

- Rack-Mount Data Acquisition System:
  - Collect data, process supervisory software, and display control decision.
- MicroVAX-II:
  - Engineering and control calculations.
- IBM-Compatible 80386:
  - Code development and automated reasoning.
- IBM-XT:
  - Man-machine and predictive displays.
- LSI-11/23:
  - Interactive simulation.

Design of a passive back plane was required for integration of otherwise incompatible components.



Functional Configuration of MITR-II Digital Controller

Advantages of ACCS

- Both level indication and scale of power instruments are transmitted to computer. This permits automated startups over many decades.

- Software essential to safety is on the Data Acquisition System Computer while software for control is on the MicroVAX-II. Hence, as reactor procedures or control strategies change, the latter can be updated without affecting the former.

- Dedicated computers available for:

  - Automated reasoning
  - Improved man-machine interface
  - Predictive displays
  - Interactive simulation

- Real-time operation and high numerical throughput achieved by combining best available products from several different vendors.

- Conventional method would have been to use a single computer in a multi-tasking environment. The advantages noted above were achieved by taking the opposite approach which was to create a multiple-computer/single-task system.

Operator Response to Predictive Displays

- Experienced operators do not use the display when performing power adjustments under standard configurations. They may refer to it at other times.

- Novice operators make more use of displays than experienced ones.

- Operator comments indicate that:

  - Derivative information (the power history) reinforces their existing approach to reactor operation.

  - The proper use of predictive information had to be learned.

- The above findings are based on a very small sample (~10 operators). But, they are consistent with results of larger studies such as the EPRI evaluation of EOPTS.

Extension to Nuclear Industry

For Pressurized Water Reactors:

- Maintenance of pressurizer level during heat-up.
- Adjustment of soluble boron.
- Damping of xenon oscillations.
- Steam generator level control.

For Boiling Water Reactors:

- Coordination of recirc pump speed with adjustment of turbine load.

Steam Generator Level Display

- Predictive display that operates in real time on an IBM-compatible PC.

- Uses a simplified, but accurate model of steam generator dynamics:

  - Mass capacity term reflecting net difference in steam and feedwater flowrates.

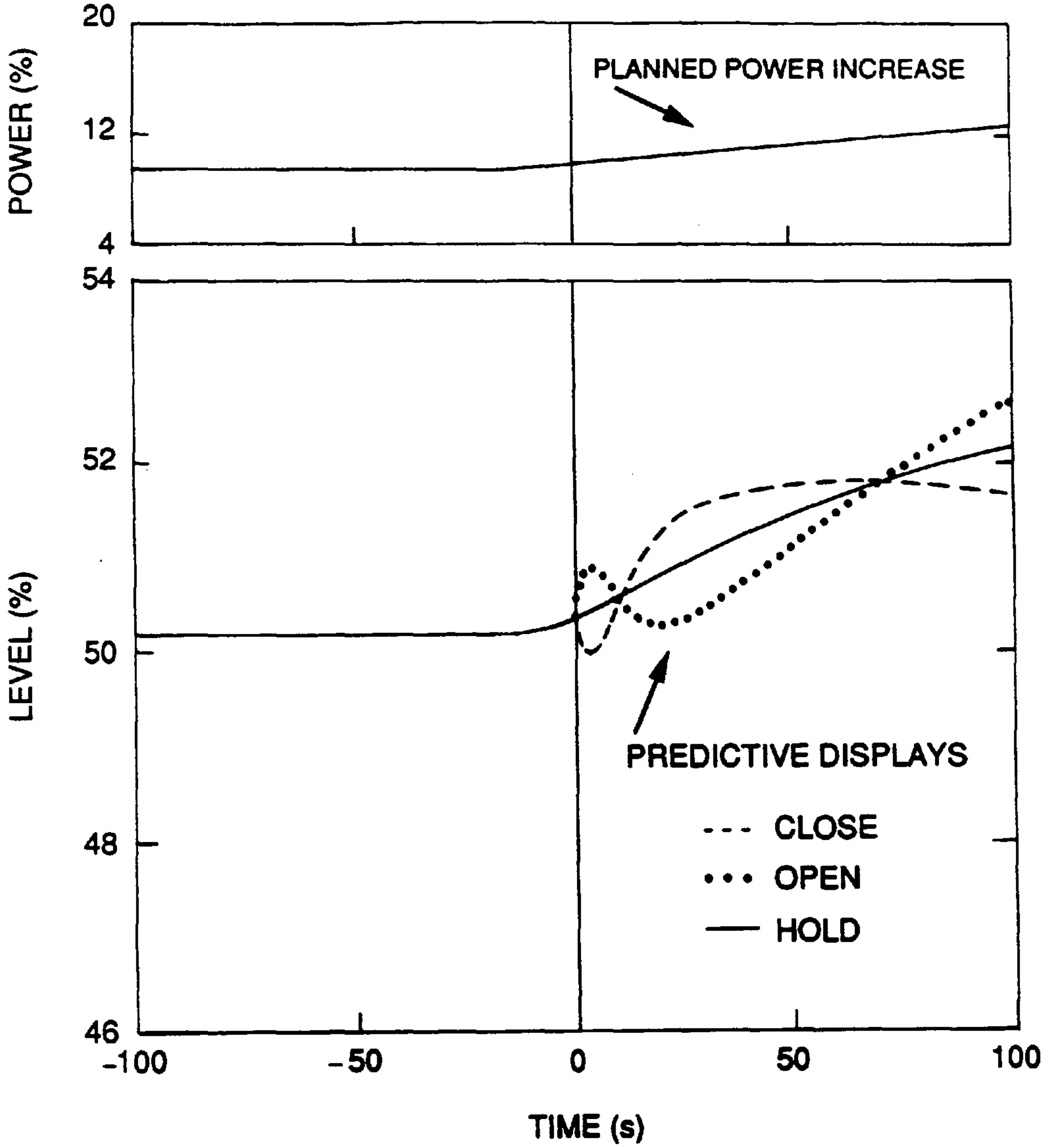
  - Shrink/swell effects for changes in feedwater and steam flow.

  - Short-lived mechanical oscillations caused by addition of feedwater to the generator.

Steam Generator Level Display

(cont.)

- Gives history of level plus three projections;
  - Valve opened.
  - Valve position constant.
  - Valve closed.
- Projections of 200 s possible with a 1-s update frequency.



Steam Generator Level Display After Start of Power Ramp with Ramp Rate of 2.5% of Full Power Per Minute with Reactor Initially at 10% of Rated Power

Conclusions

- Predictive displays used for diving controls of submarines, the Apollo landing system, and air & railroad traffic control.

- The MIT work has achieved the following:

  - Devised a display format that MIT Research Reactor Operators judged to be of benefit.

  - Accurate, fast-running models developed for predicting both neutronic power and steam generator level.

  - Neutronic display evaluated on MIT Research Reactor.

Conclusions
(cont.)

- Many issues remain to be explored including:
  - Regulatory concerns.
  - Operator dependency on displays.
  - Inaccuracies in predictive models.
  - Calibration of predictive models.
- Predictive displays may be a means of bridging the gulf that now exists between manual and automatic digital control.

Report on the Workshop
R & D Results and Needs

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By definition, the audience of the first and of the present IGORR meetings was mostly composed of individuals from research reactor institutions who are actively working to design, build and promote new research reactors or to make significant upgrades to existing facilities. These initiatives were reflected in the status reports which have been given in the first sessions of this meeting. In many cases these IGORR groups have performed their own research and development work in specific fields which might have been too detailed or too special for a full presentation at the meeting, but which the authors would usually be ready to share with other research reactor institutions interested. On the other hand, these groups would also be eager to benefit from R & D work going on at other places.

The IGORR-1 Meeting has shown that the IGORR initiative could, indeed, offer the appropriate forum to meet this demand of communication, i.e. to identify and bring together those IGORR member groups which have common R & D interests. So a matrix has been established during the IGORR-1 Meeting (see those Proceedings, page 258) in which various fields of R & D needs have been compiled versus IGORR groups which are either performing such work by themselves or which are only interested in results. By bringing groups of common interests into contact with each other it was also hoped that the efforts could be intensified, the chances of obtaining most comprehensive results could be promoted, and perhaps also that unnecessary duplication of work could be avoided.

The IGORR-2 Meeting has succeeded in obtaining the first presentations of such R & D results as will be outlined below. Additionally, new fields of R & D needs have been identified by the participants. As a consequence, the previous matrix could be further developed and has now adopted the form as shown in the Table. The various R & D topics are compiled on the horizontal lines and the various research reactor institutions on the vertical columns. Full labels indicate that a group would need the corresponding R & D results but would also plan or already perform their own work in this field, whereas open labels mean that a group would only be interested in these R & D results but would not plan to perform their own work for the time being.

Reports were given on the IGORR-2 workshop covering those R & D topics which are printed in italics in the Table. In the following we will briefly mention these contributions listing them in alphabetic order as in the matrix (Table), whereas the full papers can be found elsewhere in these Proceedings.

- C.D. West (ANS) gave a comprehensive survey on the results of R & D studies as performed for the Advanced Neutron Source project: oxide film growth rate on Al 6061 cladding material as measured with the new corrosion test loop; first irradiation tests of high enriched uranium silicide fuel in the HFIR; first irradiation tests of Al 6061 structural material also in the HFIR; preliminary fuel plate hydraulic stability measurements as performed on epoxy dummies; begin of operation of the thermal-hydraulics test loop; and, finally, the status of the 5-10 bar liquid deuterium cold source design for the ANS.

- A. Tissier (Cerca) reported on fabrication studies of fuel plates with uranium silicide fuel being density-graded in two zones as required for the FRM-II project.

- S. Matsuura (JAERI) gave two reports: first, on pulse irradiation tests of fresh, low enriched uranium silicide fuel plates up to surface temperatures of 971 °C and, second, on the performance (neutron fluxes and spectra, gain factor) of a liquid-hydrogen cold source with neutron guides as recently installed at the upgraded JRR-3.

- K. Kwok (MIT) informed the auditory of reactor controls research at MIT including a computer-generated predictive display for the operators.

- M. Bieth (Petten) reported on the results of mechanical properties measurements on Al 5154 irradiated in the HFR up to $7.5 \cdot 10^{22}$ n/cm^2 and of Beryllium.

After these presentations of results, some of the participants in the workshop got up to put out new need of R & D work or to put more emphasis on fields which were already established earlier. These requests or reports, again listed in alphabetic order of the institutions as has been done in the matrix (Table), were as follows:

- K. Böning (FRM-II) pointed out that more data seems to be necessary for the Whittle and Forgan "bubble detachment parameter" to assess the excursive flow instability in the coolant channels, in particular for water velocities exceeding about 10 m/s.

- K. Böning (FRM-II) also mentioned that the use of silicide fuel with high enrichment leads to high fission densities in the particles; a cooperation with the ANS project is planned to obtain more experimental information.

- J. Wolters (Jülich) said that data on the release of fission products during the melting of uranium silicide fuel would be urgently needed; experimental data seem to exist but not to be accessible.

- K. Kwok (MIT) stated that the MIT would continue to be strongly engaged in the field of instrumentation upgrading, digital control system and man-machine interface.

- J. Ahlf (Petten) mentioned that they would be interested in data on irradiation effects and the longterm corrosion of aluminium but that they are also planning their own investigations.

In conclusion, the workshop "R & D Results and Needs" of the IGORR meetings has proven to represent a valuable forum for the exchange of information on existing needs and available results of R & D work as well as on the groups requiring or being able to supply this information.

WORKSHOP

User and R&D Needs

REACTORS AND PHYSICS EDUCATION

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Abstract

This paper will discuss some ideas for using neutrons in physics education, including experiments which demonstrate diffraction and optical refraction, divergence imaging, Zeeman splitting, polarization, Larmor precession, and neutron spin-echo.

Introduction

There are about 325 test, research, and training reactors currently in service. One of the main roles which has evolved for these reactors is research and training in nuclear engineering and reactor operation. However, it is relatively rare to see them used for education and training in other branches of science and engineering, even though many are now used for research encompassing practically all scientific disciplines. Thermal neutron beams of quite modest flux (10^9 $m^{-2}.s^{-1}$) make excellent educational tools for teaching some basic aspects of physics, in particular. With a simple chopper and fairly inexpensive optical bench equipment of the type typically found in undergraduate physics laboratories, dual wave-particle experiments may be used to give some "hands-on" reality to quantum mechanics, for example. These types of pedagogical experiments have been pioneered, in particular, by Professors Cliff Shull (at the M.I.T. reactor), Peter Egelstaff (using an Am/Be source at the University of Guelph), and Sam Werner (at the Missouri University Research Reactor).

Examples of Introductory Experiments

The basic kitset consists of a thermal neutron beam running parallel to a length of optical bench on which devices such as diaphragms can be placed using standard optical mounts. The beam may be white or monochromatic. Diaphragms can be made from plasticized boron carbide sheets with holes of various sizes punched in them. Several

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simple but instructive experiments can be done with this basic kit, using a scintillator and Polaroid camera as the imaging device. A series of photographs of the beam with different attenuators (such as stacks of 1 mm plexiglass) or absorbers (such as Cd) in place gives an introduction to neutron transmission and absorption. The addition of a pinhole images divergence, rather than intensity, and shows the difference between flux and brightness. The divergence in the beam may be made anisotropic and varied by using pairs of rectangular diaphragms spaced by different distances; moving the pinhole allows the divergence to be observed at different points in the beam. A more interesting variation with a monochromatic beam is to place a Stern-Gerlach magnet around the beam and observe the Zeeman splitting of the beam into two beams; since $2s+1$ beams would be expected, this confirms that the neutron spin is $s = ½$. The use of a video camera makes these survey experiments much easier and is probably cheaper than instant film in the long run.

A monochromatic beam finely collimated in one dimension allows neutron refraction to be demonstrated. Figure 2 shows the arrangement. For a material with average number density N of scatterers with scattering amplitude b, the neutron refractive index is

$$n = 1 - \frac{Nb\lambda^2}{2\pi} \tag{1}$$

where λ is the wavelength. This is an interesting formula because it is an optical (wave) quantity which may be derived using particle considerations, starting with $\frac{1}{2} mv^2$ as the neutron energy, where m is the mass and v the velocity of the neutron[1]. A simple application of Snell's law shows that there will be an angle of critical reflection

$$\theta_c = \lambda \left(\frac{Nb}{\pi} \right)^{\frac{1}{2}} \tag{2}$$

below which all neutrons will be totally reflected. For Ni, this angle is 17.45 milliradian per nm of wavelength.This may be demonstrated by setting a Ni mirror to an angle smaller than this and scanning the detector through the reflecting position.

This effect is the basis of neutron guides.

More Advanced Experiments

The reflection experiment demonstrates refraction at low angles, where the variation in speed of the neutron in different materials generates the effect. If the mirror is replaced by a crystal (pyrolytic graphite or mica, for example), high angle diffraction may be demonstrated. For a given incident angle, θ, between the incident beam and the crystal, scanning the detector will show strongly scattered intensity whenever the Bragg condition

$$n\lambda = 2d\sin\theta \qquad (3)$$

is satisfied. Here, n is the order of the reflection and d the spacing between atomic planes in the crystal; if d is known the wavelength may be deduced. Repeating the measurement for different values of θ allows the intensity in the beam, $N(\lambda)$, to be measured as a function of wavelength for each wavelength band $\Delta\lambda$. After correcting for standard instrumental effects such as geometrical area of the beam intercepted by the crystal and for the (known) crystal reflectivity as a function of wavelength, this measurement allows verification of the fact that the beam spectrum from the reactor is Maxwellian:

$$N(\lambda)\,\Delta\lambda = \lambda^{-6}\exp(-h^2/2\lambda^2 mkT)\,\Delta\lambda \qquad (4)$$

Here, h is Planck's constant, k is Boltzmann's constant, and T is the absolute temperature.

Figure 3 shows the arrangement for the Bragg diffraction measurement. Also shown in the figure are a simple chopper and an oscilloscope whose timebase is synchronized with the chopper. The chopper is a 400 mm diameter aluminum disk painted with a neutron absorber such as Gd_2O_3 paint, leaving just one or two clear windows which pass pulses of neutrons as they pass through the beam; only modest speeds (1500 - 2000 RPM) are required. With the detector in the correct position to detect Bragg scattering, the oscilloscope will show the time each detected pulse arrives relative to the time of chopping. This may be converted to a particle velocity, v, using the distance from chopper to detector. The experiment thus simultaneously uses a wave phenomenon to measure the wavelength, λ, and a particle phenomenon to measure the momentum, mv, of the neutron. The value of Planck's constant may be derived from the experiment by using the de Broglie relation

$$h = mv\lambda \tag{5}$$

Finally, polarized neutrons open the way to many simple and fascinating experiments that demonstrate fundamental physical ideas related to magnetism and magnetic resonance. Thin-film multilayer magnetic mirrors which act as efficient neutron polarizers or analyzers are now available and may be mounted in a collimated beam as an insertion device. It is worth noting that, for the experiments to be considered here, very high polarization efficiency is not needed, and a simple setup giving a polarization of order 0.9 (easily achievable) is satisfactory. Linear polarization effects familiar from optics are easily demonstrated, but the most interest experiments use precessing polarization.

Figure 4(a) shows a typical setup in which the beam entering from the left is already polarized. The spin-turn device labelled $\pi/2$ is a flat coil wound from 1 mm wire on an aluminum plate about 5 mm thick and larger and wider than the beam. Such devices are easily made and set up[2]. When appropriately energized it interchanges the x and z components of the neutron spin, where z is the direction of the magnetic field in which the polarized beam is travelling, leaving the y component alone. This is equivalent to rotating the spin direction from parallel (or antiparallel) to z to perpendicular to z:

$$(x,y,z) \rightarrow (\pm z, -y, \pm x) \tag{6}$$

The spin will precess about the field at the Larmor frequency so that, as the neutron travels in the field, the polarization direction will have rotated to a phase angle

$$\psi = (4\pi|\gamma|\mu_n m\lambda/h^2)\int \boldsymbol{H}\cdot d\boldsymbol{l} \tag{7}$$

after traversing a path with field integral $\int \boldsymbol{H}\cdot d\boldsymbol{l}$. As the analyzer/detector combination is moved along the beam, the detected intensity will be modulated as $\cos\psi$. However, the further the distance, the lower the modulation depth, since different velocities in the beam will precess at different rates, eventually washing out the signal.

The concept of spin echo can be demonstrated by adding two more spin-turn coils to the arrangement, as shown in figure 4(b). The π coil is physically the same as the $\pi/2$ coil but is energized differently[2]. Its action on the spin components is

$$(x,y,z) \rightarrow (x,-y,-z) \quad (8)$$

which is equivalent to the action of a spin flipper. (These coils alone may be used between a polarizer/analyzer combunation to demonstarte linear polarization.) The arrangement is symmetric about the π coil, so that whatever precession takes place in the first part is reversed exactly in the second part. A spread in velocity in the beam no longer matters and the full initial polarization is recovered at the analyzer.

Many other experiments are possible with the simple equipment described in this paper, which is intended only to stimulate further thought on the use of neutron beams in physics education.

Acknowledgements.

I am indebted to Professor Peter Egelstaff for sharing with me his laboratory notes for students at the University of Guelph Physics Department. Oak Ridge National Laboratory is managed by Martin Marietta Energy Systems, Inc., for the U.S. Department of Energy under Contract No. DE-AC05-84OR21400.

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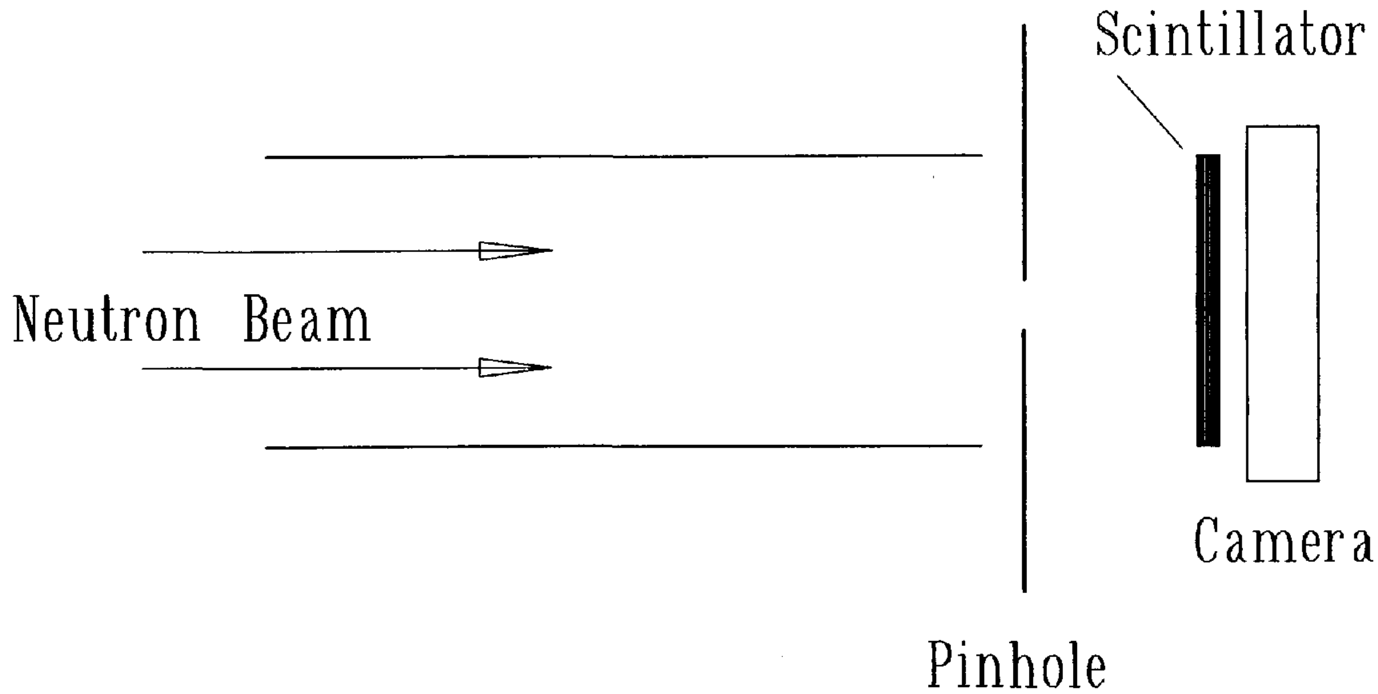

Figure 1

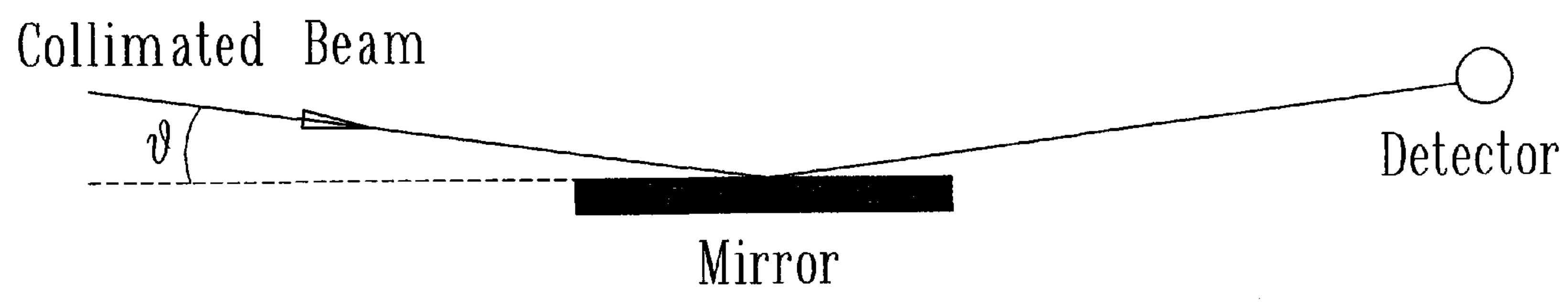

Figure 2

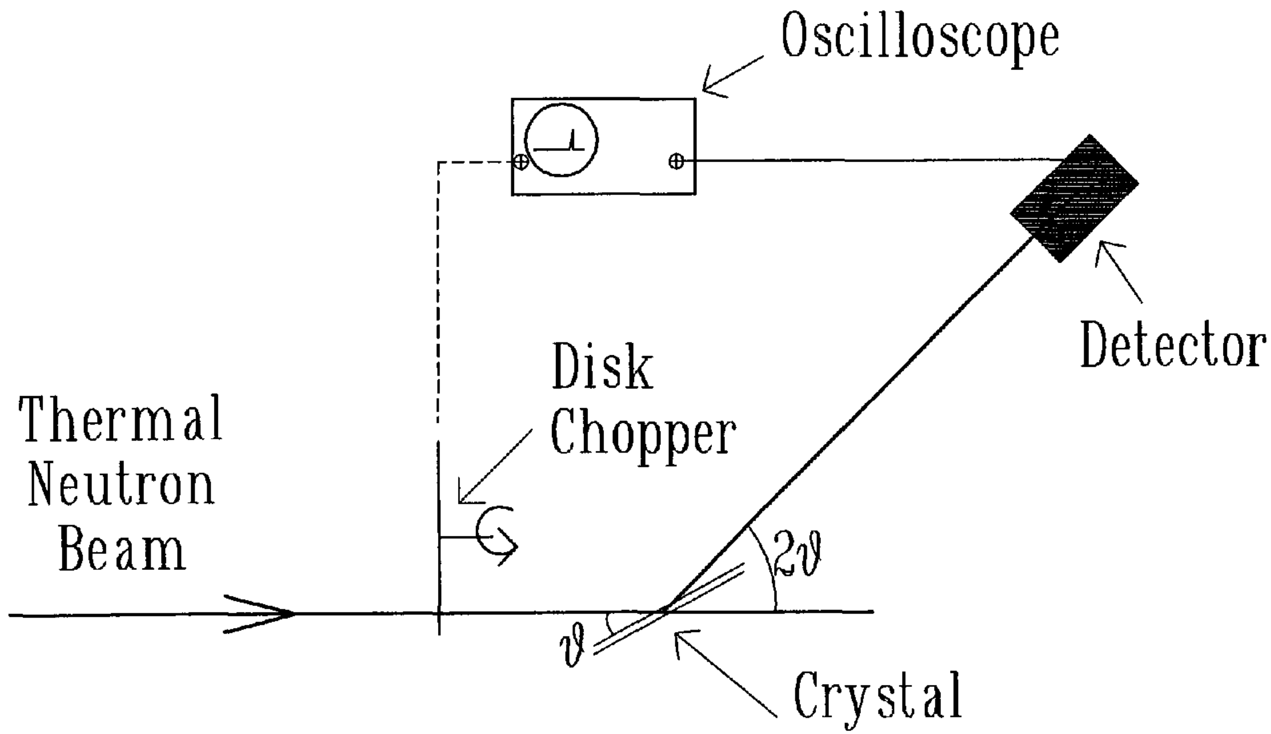
Oscilloscope
Detector
Disk
Chopper
Thermal
Neutron
Beam
2ϑ
ϑ
Crystal

Figure 3

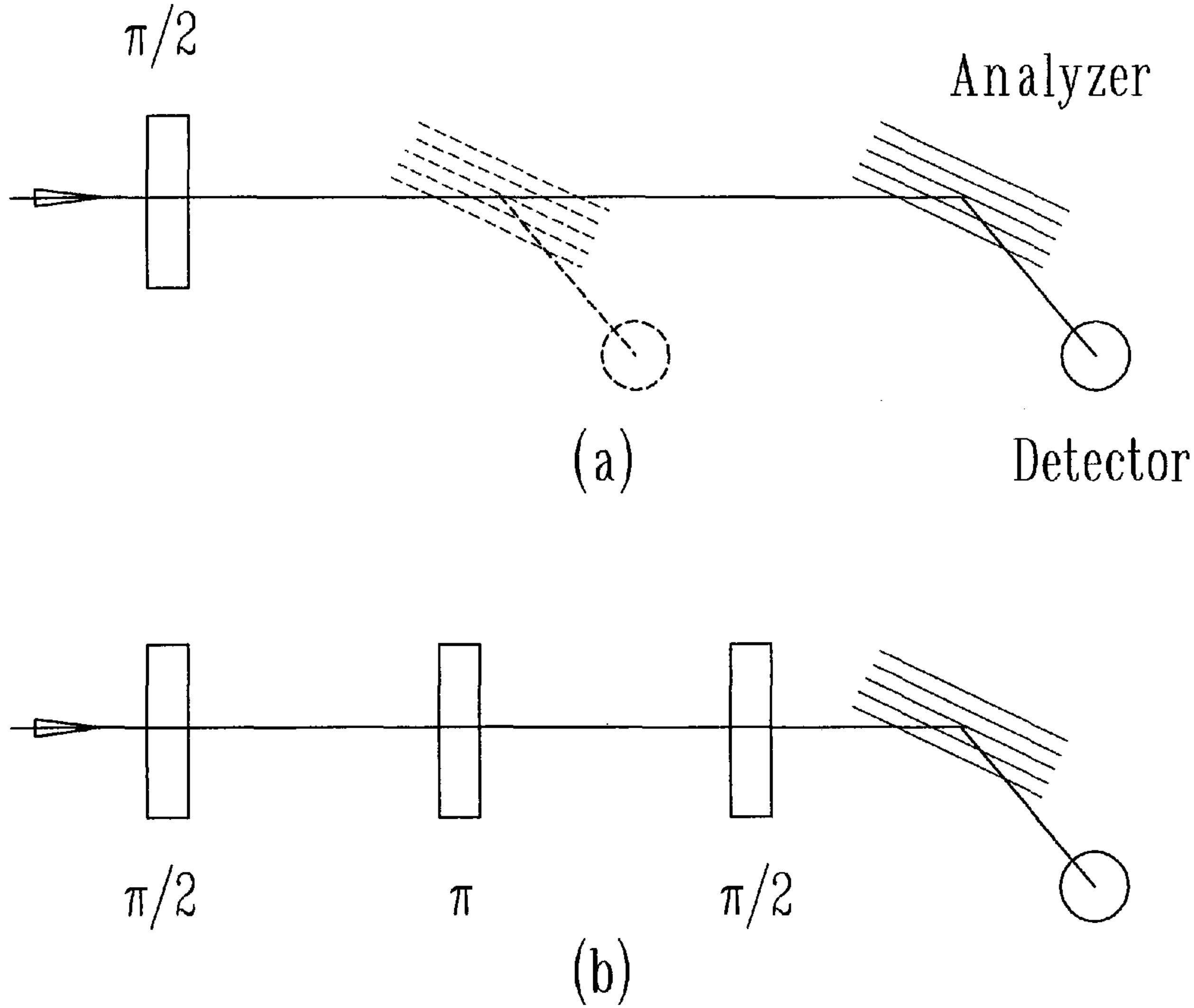

Figure 4

ORNL DWG 92M-8315R

R&D needs identified at IGORR

Topics	ANS	BERLIN	BNL	FRM-II	JAERI	JÜLICH	MAPLE	MIT	MURRI	ORPHEE	PETTEN	RISO
Thermal-hydraulic tests and correlations	●		○	○	●				●			
Corrosion tests and analytical models	●			○				○	○			
Multidimensional kinetic analysis for small cores	○			●			●					
Fuel plate fabrication	●			●								
Fuel plate stability	●			●					○			
Fuel irradiation	●			○	●	○			●			
Burnable poison irradiation	●			●								
Structural materials irradiation	●	○	●	○			●		●	●	●	
Neutron guides irradiation	○			○	●							
Cold source materials irradiation	○	○		○						●		●
Cold source LN_2 test	●											
Cold source LH_2-H_2O Reaction (H or D)	○		●	○	●							
Instrumentation upgrading and digital control system	●		○					●	○	●	●	
Man-machine interface	○			○				●				

● Results needed and work already underway or planned
○ Results needed but work not already planned

Note: Italicized text – results to be reported at IGORR-II

ADDITIONAL REPORT

(NOT PRESENTED DURING SESSION)

PROGRESS IN KOREA MULTI-PURPOSE RESEARCH REACTOR (KMRR)

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ABSTRACT

This paper gives the latest progress in Korea Multi-purpose Research Reactor (KMRR), a 30 MW open-tank-in-pool type reactor designed and being constructed by the Korea Atomic Energy Research Institute (KAERI), expected to take the central role of national nuclear R&D activities beyond the nineties. In this paper, background and necessity of this new research reactor are described. The progress in R&D, construction and commissioning of KMRR follow.

BACKGROUND [1]

Development of nuclear science and technology in Korea at the present is found briefly through three eras. Genesis of it goes back to 1959 when the Korea Atomic Energy Research Institute was established. However, genuine research activities regarding to peaceful use of atomic energy, e.g. fundamental neutron physics experiments, isotope production and its application research, and neutron activation analysis as well as reactor characteristics measurements, launched in 1962 when the novel key of 100 kW TRIGA MARK II was placed to KAERI's hands. The second era happened to open with the first shoveling of a nuclear power plant, Kori Unit 1 in 1971. The principal goal of nuclear R&D effort at these days inclined to supporting safe operation of nuclear power plant. Training of qualified operation staffs and qualified engineers was one of essential tasks assigned to Korean nuclear society. Under the circumstances, it is natural that less emphasis to the fundamental research was evident. The third era, substantially materializing localization of nuclear power technology, was geared up with successful development of CANDU fuel led by KAERI. Indepted to the success in localization of CANDU fuel, Korea evolved the localization program of PWR fuel, and technology involved in NSSS design and major component fabrication. International agreements associated with technology transfer and joint R&D were taken as the effective route to achieve these goals.

Meanwhile, in spite of its economical merit, it became a serious concern that the technology transfer approach not fully backed up with national R&D supports would have been ultimately hampered by restrictions. Consequently, Korean government wisely decided the long term investment plan for stimulating scientific research and development activities in order to take measures to break from the anticipated obstruction in future national growth. The KMRR project is one of visible examples regarding to this subject to the nuclear science and technology field.

NECESSITY OF NEW RESEARCH REACTOR [1]

The TRIGA reactors located in Seoul, which have contributed to the nation's nuclear research for thirty years are to be decommissioned in late nineties. The KAERI moved to Daeduk science center away from Seoul in 1984. Together with many neighboring research institutes in this area, it is natural that a new research reactor is planned. Following reasons briefly explain its necessity.

1. The need for performance test of nuclear reactor material and fuel
2. The need for radioisotope production
3. The need for neutron beam research facility

When the construction of KMRR is completed, it will take the central role of national nuclear R&D activities in neutron activation analysis, neutron radiography, and fundamental/applied science as well as the above area.

DESIGN CHARACTERISTICS AND OPERATION OF KMRR [1]

There are five design requirements for KMRR.

1. The maximum thermal neutron flux should be greater than 5×10^{14} n/cm^2sec.
2. The spatial and time rate of neutron flux change at the experimental

site is less than 20%.

3. The reactor should be operable at least for four weeks without refueling and with at least 25mk of available excess reactivity for experiments.
4. It should have a capability of performing several experiments at the same time without any interferences.
5. It must have the inherent safety features.

In achieving these goals, the well validated codes have been used in physics, thermalhydraulics, shielding and structure design. The important design parameters of KMRR are given in Table 1. Fig. 1 shows the plan view of the reactor and the schematic of primary cooling system is given as Fig. 2.

As for the operation, the core heat is removed by forced convective flow maintained by two pumps in parallel and then discharged to the secondary cooling system through two plate type heat exchangers, at normal operation. When the core power is less than 50% of full power(FP) or one of the PCS pumps is failed, KMRR can be operated with only one pump and one heat exchanger at the reduced power level. During reactor shutdown, the core decay heat is removed by the natural circulation through primary cooling system when the secondary flow is available. Otherwise, the core decay heat is dumped to the pool by a gravity driven circulating flow via the flap valves inside the pool.

RECENT R&D ACTIVITIES

Physics Design

WIMS-KAERI and VENTURE-KAERI have been used for the lattice design and the fuel management, respectively. For the detail evaluation of the local peaking in the fuel assembly, monte-carlo method and the super cell model is being applied.

Thermalhydraulic Design

Subchannel analysis to evaluate the flow distribution within the fuel assembly is being performed in depth with the subchannel velocity measurements to validate the code calculation. Also, the statistical uncertainty analysis is being performed to quantify the uncertainties clearly in the design calculation.

The design of the flap valve is one of the challenges, which aims to achieve the inherent safety feature for decay heat cooling.

Fuel Design

The design criteria for the fuel in KMRR was determined based on the early research for the U_3Si-Al fuel. Thus, the studies to relieve the over conservatism in design criteria is being done in depth.

The initial core of KMRR will be loaded with the fuel supplied by AECL. But, eventually, the core will be loaded with that fabricated by KAERI's own technique. For that, the authentic fabrication process - ATOMIZATION METHOD - was developed and is studied in depth [2]. ANS people in USA showed their interest in that process and the cooperation between KAERI and ORNL for the development of ANS fuel is being made.

Safety

The PSA study for the process systems such as primary cooling system, reflector cooling system and secondary cooling system was completed. The results showed that KMRR is safe from the severe fuel failure in the internal and external events.

CONSTRUCTION AND COMMISSIONING

Construction

Excavation work of the KMRR facilities started from Feb. 1989, and the first concrete was poured in July 1989. As of May 1992, concrete structure work including embedments has been completed and the installation of pool liners, consisting of reactor pool, service pool and spent fuel storage pool is approaching completion. Shop fabrication work of piping and duct system is on going and mechanical equipment installation has started. Reactor and reactivity control unit are scheduled to be installed from 2nd quarter of 1993. Construction will be completed by end of 1993.

Commissioning

When the construction of KMRR is completed, commissioning test will be performed to check whether it operates well as designed prior to normal operation. The activities related to the commissioning started in Jan. 1990. In the first year, the start-up plan was set up. Also, several engineers participated in the start-up test of JRR-3 in Japan for training. In 1991, the start up manual was developed and the test equipments were purchased. The preparation of test procedures and the training of the engineers will be the main activities before the start-up test. At this stage, it is expected that non-nuclear test will start from Jan. 1994 and the first criticality will be achieved in Dec. 1994.

COOPERATION WITH OTHERS

AECL engineering company have been working with KAERI as the major supplier of the reactor structure, fuel and RCU (Reactivity Control Unit). These mission has become a development type of work involving a completely new detailed design due to the difficulties found during fabrication and functional test. AECL is putting their best man powers to complete this challenge.

KOPEC (Korea Power Engineering Company) and HCC (HYUNDAI Construction Company) is working hard with KAERI to solve the problems revealed during construction.

REMARKS

First critical was supposed to be achieved by the end of 1992. Due to budget shortage, delay in construction, and that in fabrication of key equipments, the first critical is expected to be achieved by end of 1994. As the project is in final stage, almost of the new findings are critical and challenging. But, all the participants ensure that KMRR will play a central role in Korean nuclear research and development for next decades.

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Table 1. Important Design Parameter of KMRR

Type of Reactor	Open-Tank-In-Pool
Thermal Power	30 MW
Fuel	U_3Si-Al (20w/o U^{235})
Coolant	Light Water
Reflector	Heavy Water
Moderator	Light Water/Heavy Water
Core Cooling	Upward Forced Convection
Secondary Cooling	Cooling Tower
Max. Thermal Neutron Flux	5.3×10^{14}
Max. Fast Neutron Flux	1.53×10^{14}
Coolant Temperature(In/Out)	35/45 °C
Core Coolant Flow Rate	653 kg/sec
Reactor Pressure(In/Out)	0.4/0.2 MPa
Max. Fuel Temperature	177.8 °C (Center)
	120.0 °C (Surface)

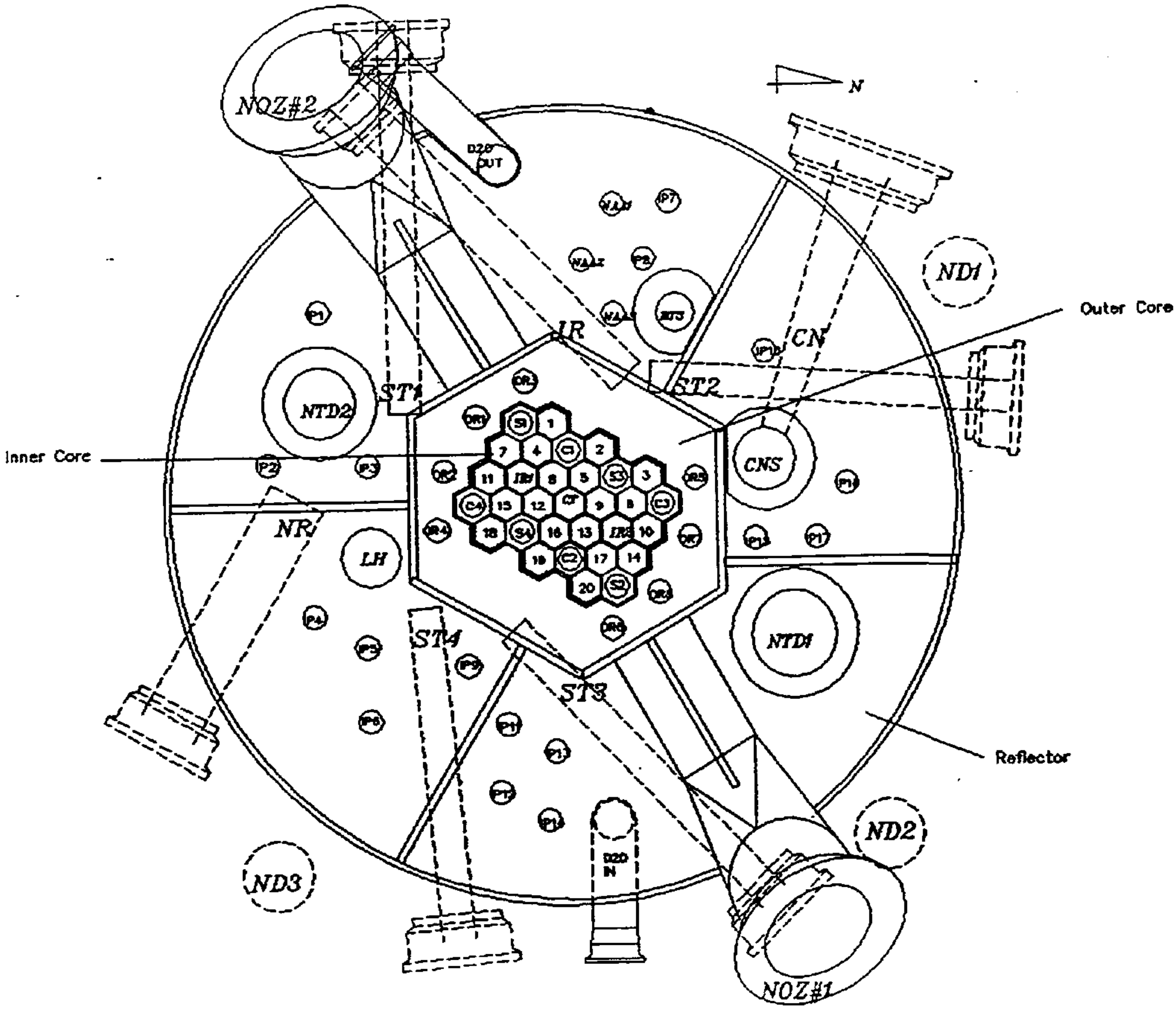

Fig. 1. Plane View of Reactor Experimental Facilities

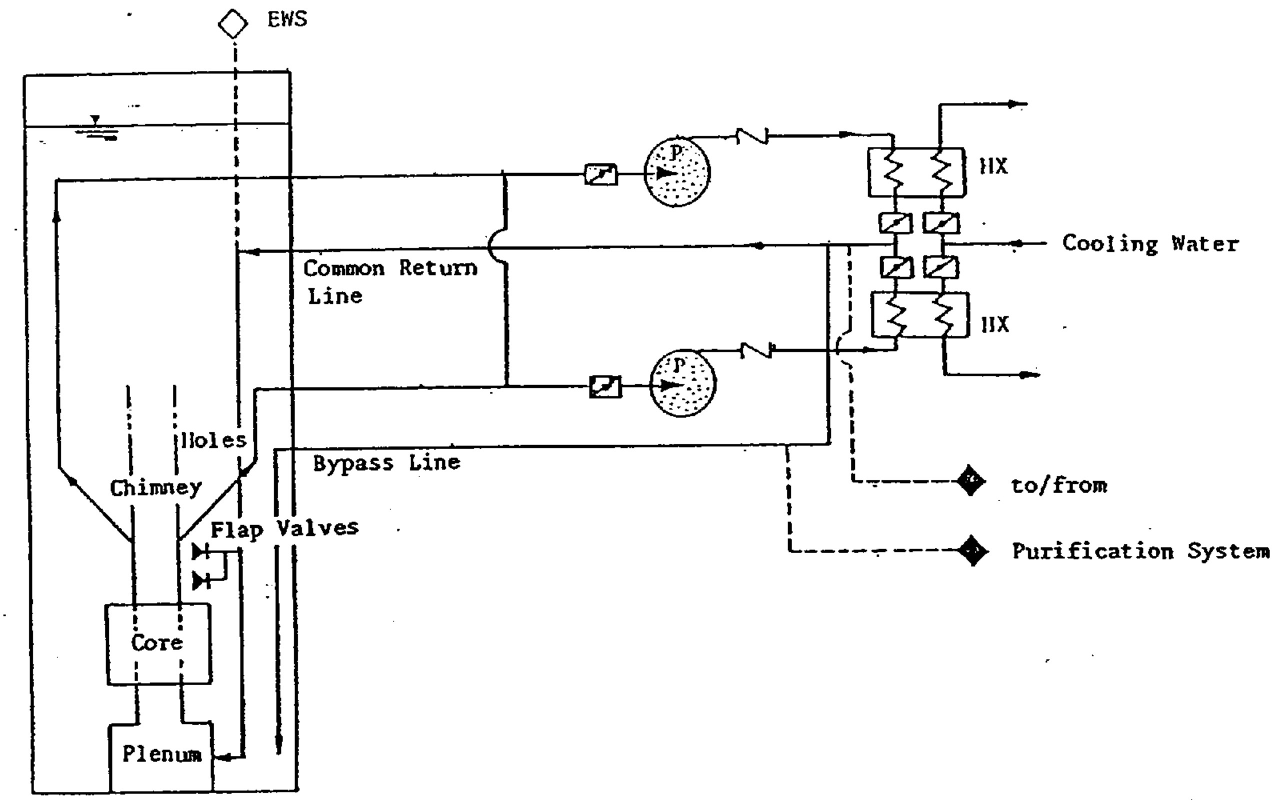

Fig. 2. Primary Heat Transport System

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AEA Engineering is one of the nine businesses of AEA Technology, the trading name of the United Kingdom Atomic Energy Authority.
For over 30 years AEA Technology successfully operated two multi-purpose 25 MW Materials Testing Reactors DIDO and PLUTO. They provided research support to the UK's nuclear power and defence programme and also the production of radioisotopes and neutron-transmutation-doped silicon for the home and international market. Research reactor operation dates back to 1947 when the GLEEP reactor, the first in Western Europe, was commissioned. With the closure of these reactors and the restructuring of the AEA, this considerable expertise is now available to customers through AEA Engineering's MTR Agency. The Agency's principle services include :

- -Consultancy on nuclear engineering, science and operations
- -Carrying out feasibility, design, technical and safety assessments
- -Applying reactor physics, heat transfer and fluid flow analysis, stress and fracture analysis
- -Control and instrumentation design etc to the solution of reactor problems
- -Gas and water reactor fuel test loops
- -High temperature creep, corrosion and power transient experimental rigs
- -Neutron beam instruments
- -Mechanical plant - containment, shielding, ventilation and cooling systems
- -Production irradiation rigs for radioisotope and silicon doping equipment
- -Decommissioning and radioactive waste services
- -Post Irradiation Examination services

Agency staff have close links with many reactor centres world-wide. It has also set up consultancy agreements with a number of reactor centres to provide a rapid response to design and operational problems. The Agency offers a silicon neutron-transmutation-doping service by the establishment of a successful joint venture with SCK/CEN at Mol in Belgium to irradiate silicon in the BR2 reactor.

AEA Engineering also has expertise in process plant provision for all aspects of the nuclear fuel cycle -fuel fabrication, fuel recover/reprocessing and waste processing and packaging. Its experience ranges from fixed facilities through to transport flasks. The Centrifugal contactor is a typical project which demonstrates our expertise in the area of Nuclear Plant design. The project comprises a major addition to the highly active cycle of an existing irradiated fuel reprocessing facility. It incorporates novel solvent extraction plant. The principal features of the facility are :

- .full alpha/beta/gamma primary and secondary containment
- .remote handling and viewing
- .minimum maintenance design for fluid handling systems
- .shielding to modern standards
- .bagless waste posting systems
- .incorporation into an existing operational facility with constraints on space and freedom of programming
- .preparation of safety documentation in support of nuclear licensing.

AEA Engineering's design capability has been brought to bear on all types of reactor systems including design of reactor fuel, handling, protection and shutdown sytems and coolant circuits. Considerable experience has also been gained on conventional power generation systems, including turbine, ancillary plant, steam raising plant, electrical power systems and instrumentation. Work often involves plant life extensions, updates and modifications. Reactor systems :

- .repair of austenitic pressure vessels
- .boron ball shutdown devices for magnox reactors
- .tools for in-core handling of fuel and other components

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AEA TECHNOLOGY


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- TARGETS FOR RADIOELEMENTS PRODUCTION FOR MEDICAL USES.

URANIUM FUEL SERVICES

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- STRUCTURAL PARTS FOR RESEARCH AND POWER REACTOR FUEL ELEMENTS.

R & D

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- FRESH AND SPENT FUEL ELEMENT EXAMINATION STANDS.
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- NEUTRONIC TARGETS.
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- IRRADIATION DEVICES.
- HANDLING EQUIPMENTS.
- CONTROL ROD DRIVE MECHANISMS (CRDM).
- CONTROL RODS.
- MISCELLANEOUS : COLD NEUTRON SOURCES.

HIGH ENERGY PHYSICS

- SUPERCONDUCTING CAVITIES BULK OR COATED METAL.

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- CLASS 1, 2, 3.
- ACCORDING TO STANDARDS ISO 9000 50 CQ A, OR 10 CFR 50 APPENDIX B.
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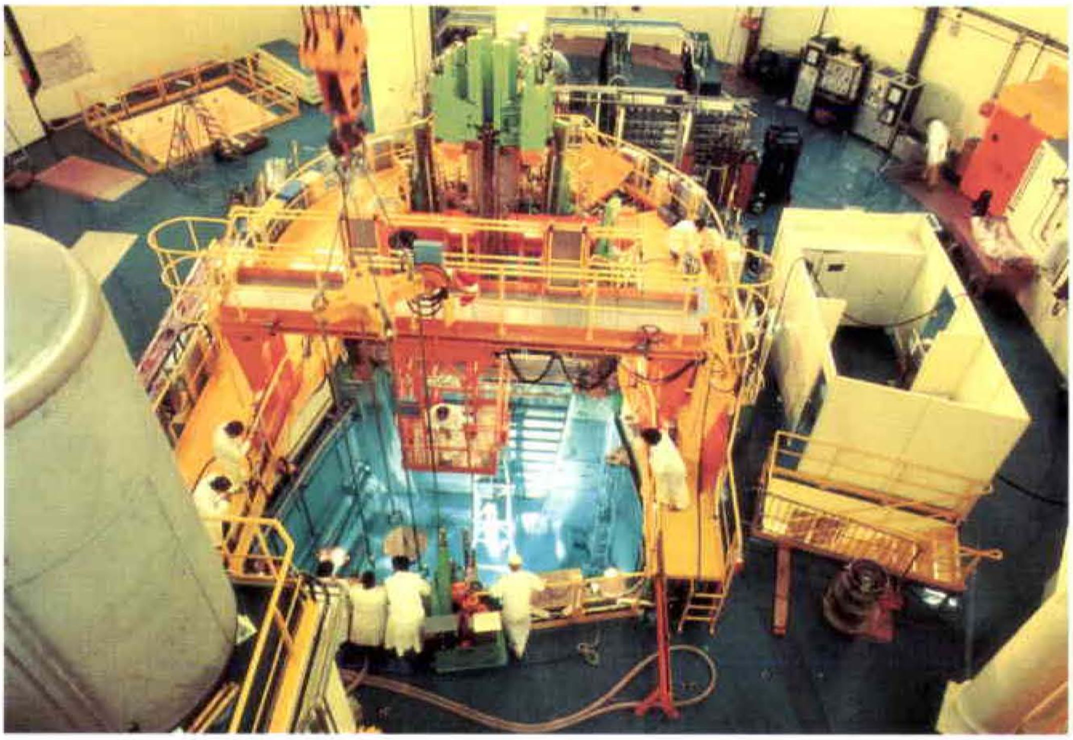

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