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IGORR-1

Proceedings of the First Meeting of the International Group on Research Reactors

February 28–March 2, 1990
Knoxville, Tennessee

Compiled by
C. D. West
Oak Ridge National Laboratory

Advanced Neutron Source

OPERATED BY
MARTIN MARIETTA ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

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IGORR-I

PROCEEDINGS OF THE FIRST MEETING OF THE INTERNATIONAL GROUP ON RESEARCH REACTORS

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May 1990

Prepared by the
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Oak Ridge, Tennessee 37831-6285
Operated by
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PREFACE

Introduction

Many organizations, in several countries, are planning or implementing new or upgraded research reactor projects, but there has been no organized forum devoted entirely to discussion and exchange of information in this field. Over the past year or so, informal discussions resulted in widespread agreement that such a forum would serve a useful purpose. Accordingly, a proposal to form a group was submitted to the leading organizations known to be involved in projects to build or upgrade reactor facilities. Essentially all agreed to join in the formation of the International Group on Research Reactors (IGORR) and nominated a senior staff member to serve on its international organizing committee.

This committee organized the first meeting of IGORR, to be hosted by the Advanced Neutron Source Project of Oak Ridge National Laboratory. Meeting dates and a specific agenda were prepared in consultation with the wider research reactor community. Speakers were asked to provide manuscripts so that Proceedings could be made available to attendees and other interested parties.

The first IGORR meeting took place on February 28-March 2, 1990. It was very successful and well attended; some 52 scientists and engineers from 25 organizations in 10 countries participated in 2 1/2 days of open and informative presentations and discussions. Two workshop sessions offered opportunities for more detailed interaction among participants and resulted in identification of common R&D needs, sources of data, and planned new facilities.

The organizing committee and other attendees, in a summary session, decided that IGORR had served a useful purpose and should continue. Additional committee members were nominated, and a chairman was selected. A broad agreement was reached that the next meeting should be held in the Fall of 1991, and an offer by Bernard Farnoux of Saclay to serve as host was gratefully accepted.

IGORR is a specialized group, seeking to cooperate, to share knowledge, and to complement several broader, existing organizations. Our first meeting was an enjoyable success, with much information shared between colleagues from all over the world, as evidenced by these Proceedings. I believe that everyone who attended is looking forward to the second meeting, and I hope you will be able to join us there.

C. D. West
Chairman
International Group on Research Reactors

ACKNOWLEDGEMENT

Kathy Rosenbalm did a splendid job of coordinating and scheduling for this conference and in arranging for meeting rooms, accommodations, and meals.

Cynthia Chance should also be commended for her role as Editor on this report.

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.

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IGORR-1

1st Meeting of the International Group on Research Reactors
February 28-March 2, 1990
Knoxville Airport Hilton Inn

Agenda

February 28	Speaker	Time
■ Registration & Coffee		8:00 a.m.
■ Welcome, Purpose, & Agenda	C. D. West	8:30 a.m.
■ Project Descriptions - New Reactors		
Munich (FRM-II)	K. Boning	8:45 a.m.
Super Maple	R. F. Lidstone	9:15 a.m.
MITR III	O. K. Harling	9:45 a.m.
■ Break		10:15 a.m.
Project Descriptions		
ANS	C. D. West	10:30 a.m.
■ Project Descriptions - Major Reactor Facility Upgrades (recent, ongoing & proposed)		
HFR (Petten)	J. Ahlf	11:00 a.m.
HFBR	J. D. Axe	11:30 a.m.
■ Lunch Break		12:00 p.m.
■ Project Descriptions - Major Reactor Facility Upgrades		
Hahn-Meitner	A. Axmann	1:00 p.m.
BR2	J. M. Baugnet	1:30 p.m.
ORPHEE	M. Breant	2:00 p.m.
■ Break		2:30 p.m.
JRR-3	S. Matsuura	2:45 p.m.
MURR	J. C. McKibben	3:15 p.m.
NBSR	H. J. Prask	3:45 p.m.
HFIR	K. R. Thoms	4:15 p.m.

- Possible session for informal presentations and discussions (the floor will be open, and a projector available, for 10 to 15 min talks and discussion) if enough people wish to participate

- Hot buffet/w Cash Bar {United Room} 7:00-9:00 p.m.

March 1	Speaker	Time
■ Coffee		8:00 a.m.
■ The U.S. National Academy Study on the future of University research reactors	O. K. Harling	8:30 a.m.
■ Discussion panel - evolving regulations for new and upgraded reactor facilities	Chair: C. West	9:15 a.m.
European and Indonesian experiences	H.-J. Roegler	
Japan	S. Matsuura	
U.S.	Al Adams	
■ Break		10:15 a.m.
■ International sharing of information and procurements	T. E. Shannon	11:30 a.m.
■ Lunch Break		12:00
■ Workshop session (I): Worldwide Facilities	Chair: J. Hayter	1:00 p.m.
Plans for Various User Needs Neutron scattering Materials irradiation testing Materials analysis Isotopes production Nuclear physics Training		
■ Break		2:45 p.m.
■ Workshop session (II): R&D Needs of IGORR Members Materials data Physics and cross sections Thermal hydraulics Instrumentation and controls Other	Chair: K. Böning	3:00 p.m.
■ Adjourn		5:00 p.m.

March 2	Time
■ Coffee	8:30 a.m.
■ Organizing committee holds an open meeting to discuss the purpose and structure of IGORR, plans and schedules for the next meeting, and interim activities (e.g., exchanges of information)	
■ Adjourn	10:30 a.m.

THE PROJECT OF THE NEW RESEARCH REACTOR FRM-II AT MUNICH

K. Boning

The Project of the New Research Reactor FRM-II at Munich

K. Böning

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ABSTRACT

A new national research reactor is planned in Germany which shall replace the existing FRM reactor at Garching. The new FRM-II will be optimized primarily with respect to beam tube applications but it will also allow the irradiation of samples etc.. Because of the "compact core reactor concept", which provides for a particularly small H_2O cooled reactor core in the center of a large D_2O moderator tank, high values of the thermal neutron flux can be obtained at only 20 MW power. This paper also discusses some of the features of the technical concepts of the new reactor.

INTRODUCTION

The existing research reactor FRM of the Technical University of Munich at Garching - a conventional swimming pool reactor of 4 MW power - has become more than 30 years old now. Plans to modernize it or - finally - to completely replace it with a new one have been made for years. As a result of this design effort the so called "compact core reactor concept" has been developed which will be discussed in the next section of this paper. The new research reactor FRM-II is considered to become the main national neutron source of the coming decades. It is being designed as a high performance but nevertheless relatively small reactor - as may also be deduced by realizing the distinction between a national and an international project in Europe.

The scientific case has been often discussed in Germany during the last couple of years and strong support has always been obtained from the German scientific community and refereeing institutions. Recently further progress has also been achieved on the political level with respect to the funding of the project. At present we are working on the conceptual design and on the safety report of the facility.

COMPACT CORE REACTOR CONCEPT

The new neutron source shall be optimized primarily with respect to beam tube applications. That is, high flux levels and pure spectra of thermal neutrons have to be provided in a large useable volume outside of the core. All this has to be achieved at a relatively small value of the reactor power which has been established to be 20 MW. So our design studies have led to the concept of a particularly small reactor core cooled by light water and situated in the center of a large heavy water moderator tank /1,2/.

This "compact core" consists of a single, cylindrical fuel element with 113 fuel plates which have the shape of involutes. The inner and outer diameters of the two aluminum core tubes ("side plates") are 118 and 243 mm, respectively, and the active height of the fuel plates is 700 mm. The light water of the primary cooling circuit flows at a velocity of 17.5 m/s downwards through the cooling channels - which are 2.2 mm wide - between the plates. The full excess reactivity of the core will be controlled by a single hafnium absorber cylinder (with an aluminum filler) which moves upwards in the inner core tube during the cycle and which is followed by a beryllium inner moderator. A long, vertical core channel tube separates the core and its H_2O cooling circuit from the surrounding moderator tank which contains heavy water and which has dimensions of 2500 mm both in diameter and height.
In addition to all the experimental installations five safety shutdown rods are provided for in the D_2O tank which are, however, fully withdrawn during normal reactor operation. The D_2O tank with the core and all the equipment is placed in the center of the reactor pool containing light water.

The design value of the reactor cycle length is 45 - 50 days. In order to obtain the necessary excess reactivity with such a small core, high enriched uranium (93%) will be used in combination with the new high density U_3Si_2-Al dispersion fuel. With an active volume of only 17.6 liter the average power density in the compact core is 1.15 MW/liter. The power density profile in the core will be flattened radially by choosing an uranium density in the fuel of 3.0 g/cm^3 within a radius of 105.6 mm and of only 1.5 g/cm^3 outside of it. Axially the same goal will be achieved by installing a ring of boron burnable poison in the outer core tube just underneath of the edge of the fuel plates. In this way the maximum value of the heat flux density from the plates into the water can be kept below about 500 W/cm^2.

For the unperturbed case, i.e. for the moderator tank containing D_2O only and nothing else, neutron transport calculations have yielded a maximum of the thermal neutron flux of about 8 x 10^{14} $cm^{-2}s^{-1}$ in the D_2O at 20 MW power. The corresponding ratio of flux outside of the core to power ("rendement") is higher than at any other reactor /1,2/. The experimental installations to be realized in the moderator tank include, first of all, 11 big horizontal beam tubes. They all have directions tangential to the core in order to suppress background radiation, and some of them will be connected with a cold or a hot neutron source. However, although first priority will be clearly given to the beam tube applications, the new FRM-II will generally be a multipurpose reactor. That is, it will also provide for a variety of vertical channels for the irradiation of samples and also for a secondary fission target ("converter") to produce high energy neutrons for medical and computer tomography applications.

FEATURES OF THE PLANT DESIGN

A schematic vertical cut through the reactor pool with part of the equipment is shown in Fig. 1 /3/. The compact core (#1) is placed in the core channel tube which leads through the center of the D_2O moderator tank (#12). The central control rod with its hafnium absorber (#2) and its beryllium follower underneath is shown in its shutdown position. It can be moved upwards by the drive mechanism (#4) but can also be released quickly by an electromagnetic clutch (#3) in order to operate as an additional, independent shutdown system. The main shutdown system consists of 5 safety rods (#5) which are fully withdrawn during normal reactor operation but can be quickly inserted by spring or pneumatic forces if required.

The pumps of the primary cooling circuit are equipped with flywheels. The corresponding two primary pipes (#8) as well as the three pipes of the emergency cooling circuit (#10) are connected with check valves against reverse flow. They all end in a collector (#7) which leads to the header of the core channel tube (#6), both of them being designed in a way that rupture can be excluded. The pressure of the primary circuit at this position is about 9 bar and the flow rate about 320 kg/s. A few hours after shutdown all pumps can be switched off and the core will be cooled by natural convection since the two corresponding flaps (#11) open

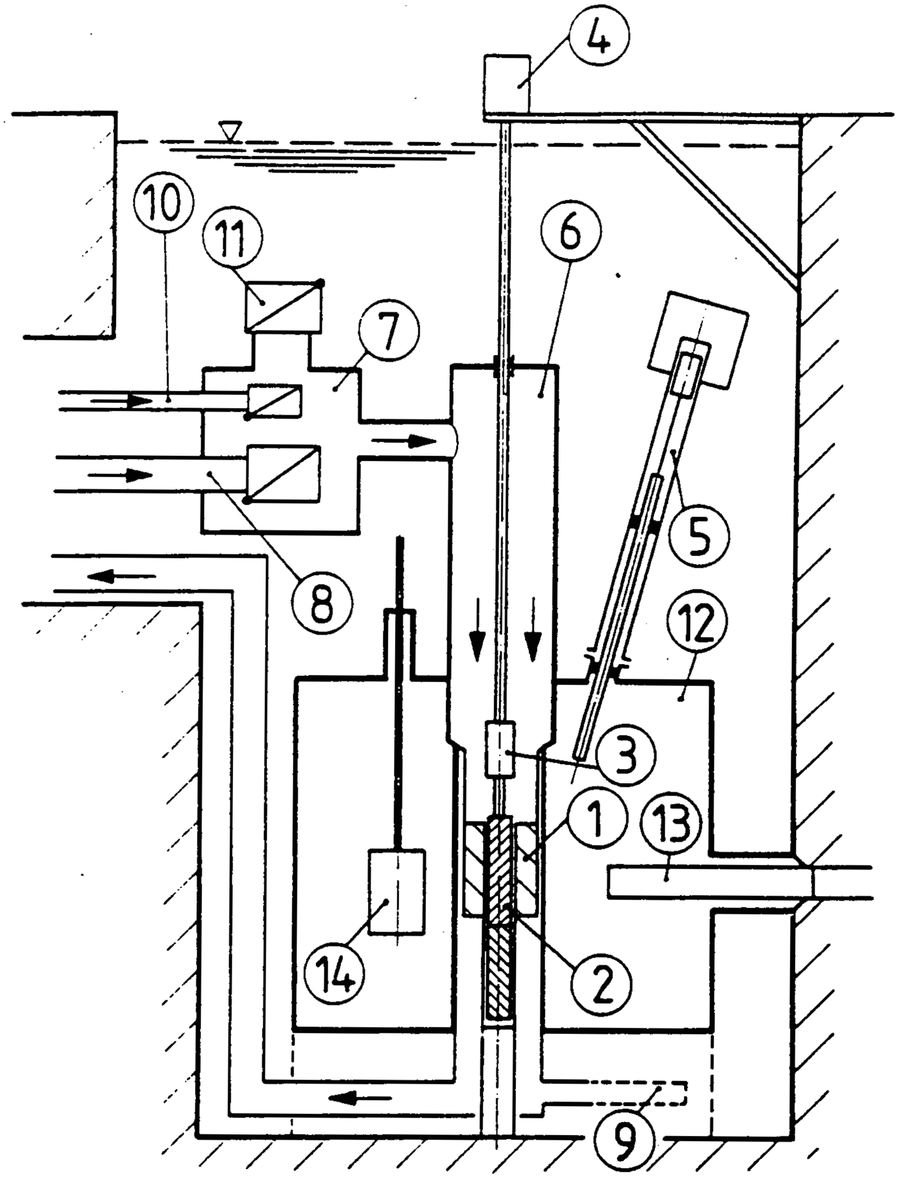

Fig. 1: Schematic vertical cut through the reactor pool of the FRM-II (Interatom GmbH; from /3/). The meaning of the various numbers is as follows:

1 fuel element (compact core)
2 central control rod (with beryllium follower)
3 electromagnetic clutch
4 drive mechanism of control rod
5 shutdown safety rod (5 x)
6 header of the core channel tube
7 collector of the primary pipes
8 primary cooling pipe (2 x)
9 opening sieve to the pool
10 emergency cooling pipe (3 x)
11 flap (valve) for natural convection (2 x)
12 D_2O moderator tank
13 horizontal beam tube (11 x)
14 cold neutron source

automatically and since the primary circuit is always connected with the pool through the sieve #9. - Finally, as an example of the experimental installations one of the 11 horizontal beam tubes (#13) and a vertical cold source (#14) are also shown in Fig. 1 /3,4/.

Some additional features of the present design status of the overall cooling concept become evident from Fig. 2 /3,4/. On the top of the figure the reactor pool with the compact core and the D_2O tank can be recognized on the right hand side, and the storage pool on the left. The pumps and heat exchangers of the primary core cooling circuit A are located in leak-tight chambers. Each of the two redundant primary systems would be sufficient to cool the core at full power. The closed secondary cooling circuit B serves as a further barrier against the release of fission products and allows the connection of the other cooling circuits through the heat exchangers #1 - #4. The ternary circuit C contains a number of air cooling aggregates which represent the nominal heat sink.

The power generated in the D_2O moderator tank is transferred from the D_2O cooling and purification circuit E through heat exchanger #3 into loop B. Similarly, the heat production originating from experimental installations as, e.g., the cold neutron source is discharged through the circuit F and heat exchanger #4. Finally, the pool water cooling and purification circuit G (not fully shown in Fig. 2) is connected with the heat exchangers #1 and #2.

The emergency core cooling system D is independent of the other systems and again consists of primary, secondary and ternary loops. The three battery-driven primary emergency pumps D - each of them being again sufficient to cool the core (after shutdown) - can feed pool water through the core and - by means of the sieve at the lower end of the core channel tube (Fig. 1) - back to the pool. In this way the thermal capacity of the whole reactor and storage pool water can be made use of as a further heat sink. Recooling of the pool water would be necessary only after a few days and can be performed by the secondary circuit D. The core decay heat would finally be transferred to a close-by river using water from existing wells. - The safety concept of the new FRM-II will be explained in more detail in ref. /4/.

A more realistic vertical cut through the reactor and storage pools has been published in ref. /5/. The reactor building of the

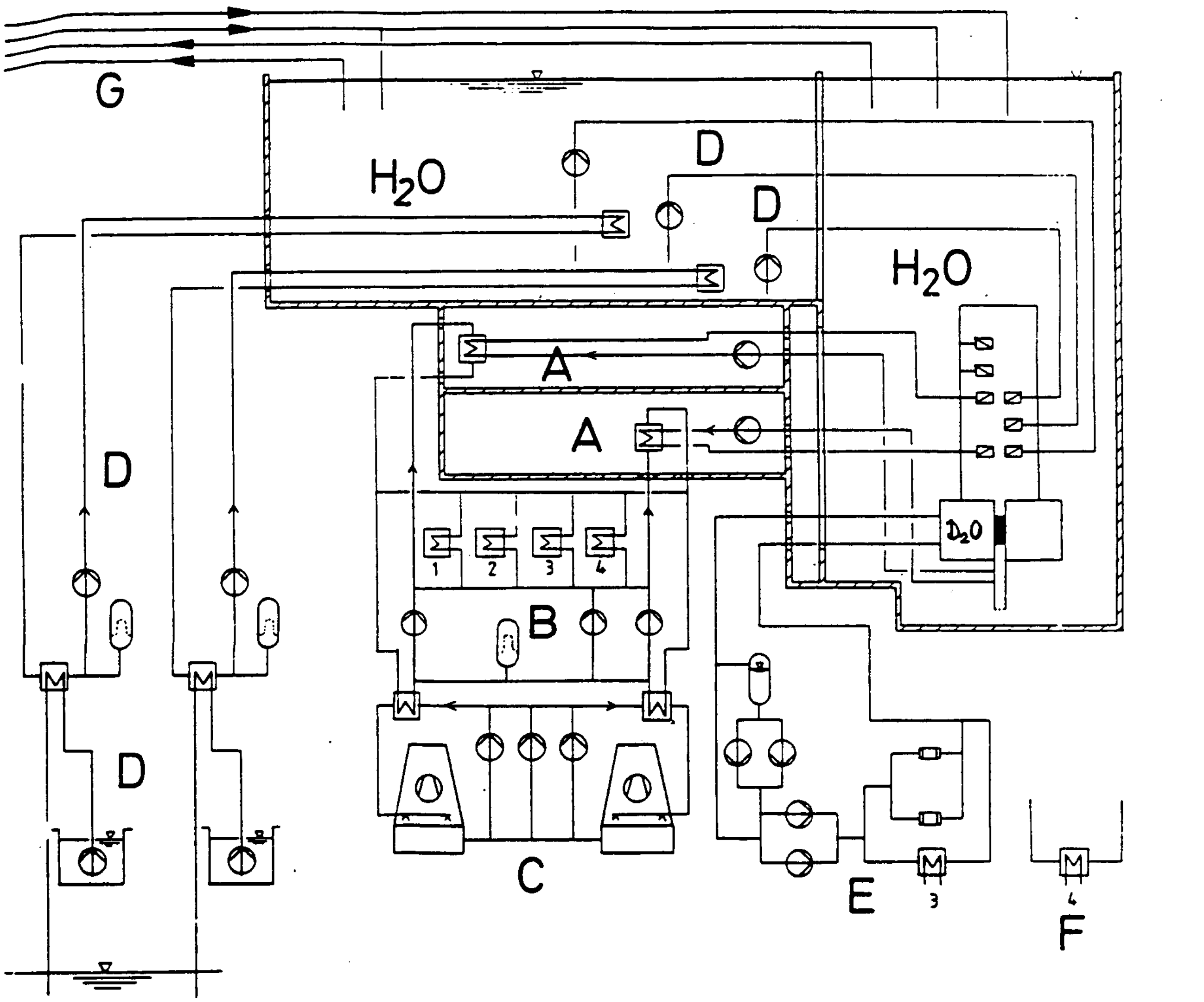

Fig. 2: Scheme of the preliminary cooling concept of the FRM-II (Interatom GmbH).

A primary core cooling circuit
B secondary circuit
C ternary circuit with air cooling aggregates
D emergency core cooling circuit
E D_2O moderator cooling circuit
F experimental installations cooling circuit
G pool water cooling and purification circuit
1-4 further heat exchangers

new FRM-II will have a quadratic cross section of about 40 m side length on the ground floor level where nearly the full area can be used for the beam tube experiments. This "experimental hall" will be completely separated from the "reactor hall" which extends above the pool water level and where fuel handling and irradiation experiments can be performed. On this level the cross section of the building will be an octogon. A vertical cut through the reactor building has also been presented in ref. /5/.

Finally, a view of the whole research reactor facility, according to the present design status, is shown in Fig. 3. The new FRM-II reactor building is to be seen on the left. It is connected with a low "neutron guide hall" where beam tube experiments with cold neutrons can be performed. This experimental area even extends into the egg-shaped building of the existing reactor FRM. This old FRM has been the first nuclear reactor in Germany and will be shut down and decommissioned shortly before the new FRM-II goes into operation.

ACKNOWLEDGEMENTS

This paper is a summarizing report on a project which many colleages and coworkers from various institutions have contributed to. These include numerous members of the Faculty of Physics E21 and of the operation group of the existing research reactor FRM of our University. Most of the technical engineering work has been performed by the company Interatom GmbH. The architectural design comes from Prof. Angerer and his group at our University.

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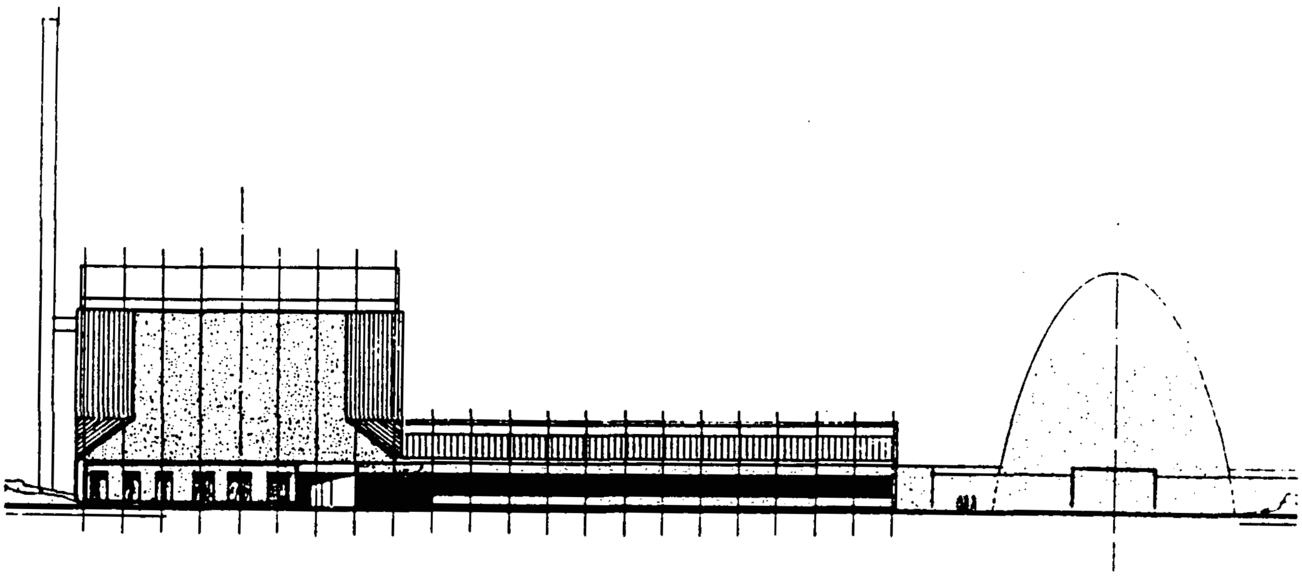

Fig. 3: View of the new research reactor facility according to the present (preliminary) design status. The building of the new FRM-II on the left will be constructed in about 100 m axial distance from the egg-shaped building of the existing FRM, with the new neutron guide hall in between. The architectural design is from Prof. Angerer, TUM.

MITR-II

O. K. Harling and D. D. Lanning

MITR-III

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ABSTRACT

This paper outlines the successful MIT research project which is currently based on a compact core 5 MW neutron source. In anticipation of the license expiration for the current MITR-II in 1996, studies have been initiated to define the user needs and the reactor design which could meet these needs. An overview of current activities relating to a new or upgraded reactor, MITR-III, are presented in this paper.

I. INTRODUCTION

The MIT Research Reactor has been, is and is expected to continue to be a valuable research tool at the Institute. The initial MITR-I, and now the upgraded MITR-II, has served the research, teaching, and service needs of students and staff from various MIT departments and laboratories as well as users from other universities, teaching hospitals, and industries for thirty-two years. Some appreciation of the value of this facility can be gleaned from the following statistics. A total of 180 doctoral, 271 masters', and 78 bachelors' theses have been completed using the MITR. A total of 800 papers in reviewed technical journals, 231 major technical reports and a total of ~2000 written publications of all types have been based upon work done with this reactor. No less than 90 major technical highlights, i.e. major achievements such as in-pile loops to simulate LWRs or highly accurate measurements of limits on the neutron's charge, have resulted from research at this reactor.

In 1958 the first MITR was made operable;[1] in 1975 the significantly upgraded MITR-II achieved full power;[2] and in 1989 planning for MITR-III was initiated. This paper provides some of the background on the MIT research reactor project, including a description of current facilities. The desired characteristics for an upgraded reactor and various considerations related to MITR-III are outlined in this paper.

II. BACKGROUND OF THE MIT RESEARCH REACTOR

The MIT Research Reactor Project was conceived in the first part of the decade of the 1950's. It was during this period that the "atoms for peace" initiative was enunciated by the US Government and the Atomic Energy Act of 1954 facilitated peaceful uses of the atom. Universities responded to this initiative by the creation of academic departments which specialized in nuclear engineering and by the design and construction of university research reactors (URRs). The construction of URRs was encouraged by the US Government, in particular, by the Atomic Energy Commission and the National Science Foundation.

At MIT, the Institute's president, Dr. James R. Killian was a leader in the effort to establish a department of Nuclear Engineering and to build a first-class research reactor. The organization of an academic department of Nuclear Engineering was put into the able hands of Manson Benedict, who after receiving a doctorate in Chemical Engineering from MIT, had distinguished himself through his contributions to the Manhattan Project. Professor Benedict recruited Dr. Theos J. Thompson from the Los Alamos Laboratory to take charge of the design and construction of the MIT Research Reactor. Dr. Thompson had an excellent background for this task since he had been involved with the design of the Omega West Research Reactor at Los Alamos.

The detailed design and construction of the MITR-I required about three years. This is a short time compared to the time it currently takes to complete projects of comparable complexity. In 1958, when MITR-I went critical, it was the fourth URR to go into operation in the USA and it was the largest or most ambitious of the initially constructed URRs. MITR-I had similarities to the Argonne National Laboratory's CP5 tank type reactor. It was designed to be capable of 5 MW thermal and to be heavy water cooled and moderated. A graphite reflector was used. Fuel elements were MTR type plate elements using a fuel meat which had fully enriched U-235.

The MITR-I was provided with a wide range of experimental facilities, including the following major facilities:

Eleven horizontal beam ports, 4 to 12 in. in diameter, radial to the core

Two horizontal thru ports, 4 and 6 in. in diameter, tangential to the core

Six pneumatic tubes for rapid transport of small samples

Four thimbles in the reflector, 3-1/2 in. i.d. for longer term thermal neutron irradiations

Fission neutron irradiation facility

Thermal column, hohlraum and blanket neutronics facility

Medical irradiation room with a large vertical beam below the core

In-pile irradiation positions based on removal of fuel elements

Many research projects including neutron scattering, blanket neutronic studies, medical irradiations, capture gamma-ray studies, organic reactor coolant stability, nuclear chemistry, neutron activation analysis, etc., were successfully carried out thanks to the availability of MITR-I.

From the beginning of the MIT reactor project, it had been recognized that the research reactor would require eventual rebuilding and upgrading. This farsighted approach included provision for funding a reactor upgrade. After several years of planning, which involved a number of graduate student thesis projects, the decision to rebuild the MITR was made in the early 1970's. To direct the major renovation, David D. Lanning was brought to MIT from the Hanford Laboratories. Professor Lanning, a Ph.D. graduate from the MIT Nuclear Engineering Department, already had considerable experience with the MITR-I while he was Superintendent of Operations.

Major design goals for the upgraded reactor included:

Improved thermal flux intensity and quality at the horizontal beam ports.

Enhanced safety, especially with regard to potential loss of core cooling.

Improved operational characteristics, including fuel life, reliability of components and control systems.

The upgraded MITR-II, see Fig. 1, features a completely redesigned core and reflector. The compact hexagonal core, 15 inches wide by 2 ft. high, operates at 5 MW with a power density of ~ 0.1 MW/liter. Core cooling is with H_2O while the reflector tank, which wraps around the lower part of the core tank, uses D_2O. To remove the 5MW of heat from the much compacted core without major changes to the heat removal systems, the aluminum fuel plates were grooved, resulting in a doubling of the heat transfer coefficient. Uranium loading in the fuel elements was substantially increased and each of the elements in the 27-element core has ~ 500 gms of U-235. Fuel elements can be rotated to three different radial positions and are constructed so that they can be flipped upside down to optimize fuel burnup. Burnups of ~ 45 percent are achieved, resulting in an average fuel element life of approximately four and one-half years or 900 full power days.

VIEW OF M.I.T. RESEARCH REACTOR, MITR – II, SHOWING MAJOR COMPONENTS AND EXPERIMENTAL FACILITIES

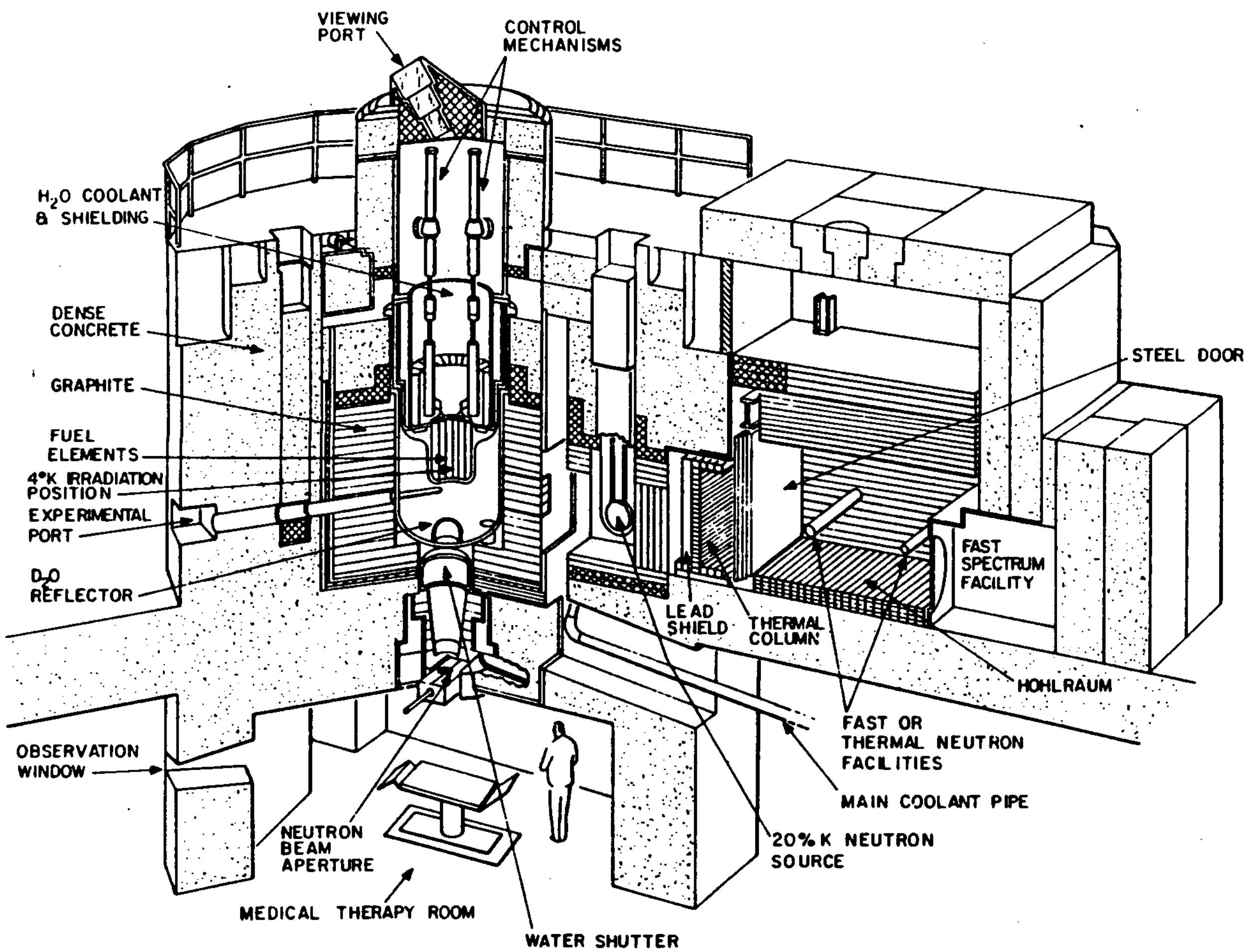

Figure 1

Another feature of the MITR-II core is the available option of poisoning the upper half of the core. This still permits operation at a full 5 MW and provides significantly increased flux, e.g. 30-50 percent increases of the beam tube flux.

Beam tubes in MITR-II are tangential to the bottom of the core and view a well moderated thermal flux where it peaks in the D_2O reflector. The tangential arrangement, the compact core, and the good moderation provided major improvements in beam tube thermal neutron flux intensity, and decreases in undesired backgrounds from fast neutrons and gamma rays..

A summary of important performance changes achieved by upgrading MITR-I to the MITR-II design (both reactors at 5 MW) is given below.

Neutron Flux

1. Increased thermal flux at horizontal beam ports, a factor of 3X for the total thermal flux and a factor of 9X at 3 Å, $\phi_{th} = 10^{14}$ n/cm^2-sec.
2. Fast neutron and gamma ray contamination of the thermal beams was greatly reduced.
3. In-core fast neutron flux was increased to more than 10^{14} n/cm^2-sec, $E_n > 0.1$ MeV.
4. The medical beam contamination by fast neutrons and gamma rays was significantly decreased.

Safety Improvements

1. No loss of coolant due to pipe rupture; pipes enter core tank near top of tank.
2. Natural convection demonstrated.
3. Seismic safety enhanced.
4. Light water primary system, resulting in reduced tritium problems.
5. Heavy water reflector, provided backup scram, and an extra tank wall to defend against loss of core cooling.
6. Remote shutdown and cooling capability.

Operational Improvements

1. Control rod magnets moved away from core radiation.
2. Elimination of tritium contamination on in-core components and experiments.

3. Access to core and in-core facilities through pool.

4. Fuel assemblies with improved heat transfer and long burnup lifetime.

III. MITR-III

General Considerations, Issues, and Options

The following considerations or issues must be understood and dealt with in planning for the MITR-III. We have begun to deal with these but are nowhere near a final resolution of some of these issues.

1. MITR-II's license expires July 1996.
2. What type of reactor is best suited to MIT's and the other users' needs? For example, should it be a general purpose neutron source, such as the MITR-I and MITR-II, or should there be strong emphasis on developing specific nuclear technology, e.g., gas cooled reactor design, or specific scientific applications such as neutron scattering.
3. Options include: a new reactor, an upgrade, relicensing, decommissioning.
4. Location of MITR-III
5. User base
6. Funding for MITR-III: a) construction, b) operation.

Desired Reactor Characteristics

Our current list of desired reactor characteristics is based upon discussions with a variety of reactor users. A distilled list of user requirements modified by the review of the authors in order to assure reasonable compatibility and to give consideration to the current and projected levels of utilization is presented below:

1. In-core Irradiation Experiments

Current and projected needs for in-core irradiation space for radiation damage studies and in-pile test loops is significant. Therefore, a somewhat larger core, with space in and above the core to accommodate three or four complex rigs or loops is desired. A fast flux somewhat higher than 10^{14} n/cm^2-sec E > 0.1 MeV and larger in-core test holes, ~ 3 in. in diameter compared to the current maximum of 2 in. diameter per fuel element, are desirable.

2. Medical Irradiation Beams

Increased epithermal flux for treatment of deep-seated tumors by neutron capture therapy is desired. Alternate sites for medical beams, e.g. from the side of the reactor, should be considered. Fluxes of 3-10 x 10^9 n/cm^2-sec 1 eV < E < 20 keV are reasonable goals.

3. Neutron Scattering

Tangential beam tubes, with beam-tube source flux of at least ϕ_{th} = 1 x 10^{14} n/cm^2-sec. are desirable. Provision for a rethermalizer at liquid helium temperature to produce a cold neutron source should be considered. Even though these spectrometer beams are not as intense as national laboratory beams, there is a broad research program that can be carried out in solid state physics, chemistry and material science, see, e.g. "New Materials," Science, Vol. 247, 9 Feb 1990.

4. Neutron Activation Analysis

All current short and long term thermal neutron irradiation facilities should be maintained. A thermal flux of greater than 5 x 10^{13} n/cm^2-sec is desired for at least one facility. Low flux gradients and cooling for sensitive samples is important.

5. Other Thermal Neutron Irradiation Facilities

Thermal neutron irradiation facilities for larger sample volumes, e.g. > 6 in. diameter samples at flux levels of 5 x 10^{13} - 10 x 10^{13} n/cm^2-sec, are needed. Cooling using reactor or moderator water is desirable.

Reactor Design Considerations and Goals

The general design goal for MITR-III can be stated, to improve the facility such that its usefulness is enhanced while allowing safe and economic operation of the facility with a high availability for all users.

More specific goals include:

1. Improving certain experimental capabilities, see above.
2. Improving the safety and operation.
3. High availability, ≥ 80%.
4. Licensability.

5. Plant personnel exposures < 10% of 10CFR20 limits.
6. No sheltering or evacuation plans for general public.
7. Investment protection, risk of exceeding design conditions not to exceed 1×10^{-5} events/year.
8. Yearly operating costs less than $1 million.

Some design restrictions:

1. Urban siting on the current site.
2. Power level less than 10 MW.
3. Use of LEU fuel.
4. Use of the existing reactor block and containment building.
5. Use of existing heat removal capabilities; or limits on pipes, pumps, heat exchangers or flow rates for increased heat transfer.
6. Restrictions on power or power density for safety considerations such as natural circulation cooling after shutdown.
7. License limit considerations such as existing containment design limits.

The full listing of the design requirements will be developed by the use of an integrated design approach. This approach includes a functional analysis and the use of probablistic risk assessments (PRA) for the assessment of design options.

Status of the MITR-III Project

In 1989 a preliminary study of MITR-III options was initiated using student resources and some faculty and professional staff. After examining a wide variety of options including the entire range from a new reactor on a new site to decommissioning and elimination of a research reactor at MIT, the tentative conclusion was reached to:

Upgrade the current research reactor and maintain the general philosophy of a general purpose neutron source which has served the users' needs well in the past.

To move this concept toward realization, a detailed design study effort, requiring several man years of effort, must be carried out. Currently, due to resource limitations we are only able to mount a modest study effort which cannot meet the requirements for the development of a detailed design and cost estimates. We intend to make strong efforts to obtain the resources needed for the design of MITR-III and for its realization in the period when the current MITR-II license expires.

IV. SUMMARY

This paper provides some background on the successful MIT Research Reactor Project which was initiated in the early 1950's. We have also outlined the considerations involved and the options available for an MIT reactor after the current license expires in 1996. These considerations include the projected user needs and reactor design aspects including core design, safety and licensability.

In the present climate of a dwindling number of university research reactors, we believe that every effort should be made to keep the MITR facility operating in order to continue the teaching and research that will be required for the future in the nuclear sciences and in nuclear technology.

V. ACKNOWLEDGEMENTS

The authors wish to acknowledge the assistance of L. Clark, Jr., and J. A. Bernard in preparation of this paper.

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THE ADVANCED NEUTRON SOURCE (ANS) PROJECT

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THE ADVANCED NEUTRON SOURCE (ANS) PROJECT

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ABSTRACT

The Advanced Neutron Source (ANS) is a new user experimental facility for neutron research planned at Oak Ridge. The centerpiece of the facility will be a steady-state source of neutrons from a reactor of unprecedented flux. In addition, extensive and comprehensive equipment and facilities for neutron research will be included. The scientific fields to be served include neutron scattering with cold, thermal, and hot neutrons (the most important scientific justification for the project); engineering materials irradiation; isotope production (including transuranium isotopes); materials analysis; and nuclear science.

I. MISSION, PERFORMANCE GOALS, AND OVERALL REQUIREMENTS

The Seitz-Eastman Committee, in a 1984 National Academy Study,[1] was most influential in defining the scientific justification for an advanced steady neutron source and in specifying the broad performance capabilities required. The committee debated the performance capabilities of present research reactor technology and selected an achievable, but challenging, performance goal to meet the scientific needs of the user community. The Seitz-Eastman report was studied, and its findings endorsed, by the U.S. Department of Energy's (DOE's) own Energy Research Advisory Board (ERAB).[2]

Subsequently, workshops[3,4] and the National Steering Committee for an Advanced Neutron Source (NSCANS) defined the performance requirements in greater detail and also in quantitative terms. NSCANS continues to guide the ANS Project and to review the project team's work to ensure that user requirements are being met. The DOE

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* Based on work performed at Oak Ridge National Laboratory, operated for the U.S. Department of Energy under contract DE-AC05-840R21400 with the Martin Marietta Energy Systems, Inc.

orders, and other appropriate codes and regulations, further define design requirements (see Tables 1 and 2).

II. TECHNOLOGY AND BASIC DESIGN FEATURES FOR THE ANS REACTOR

The user requirements, particularly the need for a large, accessible volume of very high thermal neutron flux, determine the main features (high power and small size) of the reactor core design. To minimize technical risk, the project has adopted the involute, aluminum-clad cermet fuel plates and the annular core arrangement, common in existing high-performance beam reactors [e.g., High Flux Isotope Reactor (HFIR) and the Institut Laue-Langevin (ILL) reactor at Grenoble]. A new geometrical arrangement using fuel elements of different diameters separated axially, offers many safety and performance advantages and forms the basis for the final preconceptual core design (see Figs. 1 and 2).

The basic concept is conventional: a heavy-water-cooled and reflected reactor. The large reflector tank places ~1.5 m of heavy water around the core and provides space for two cold sources, beam tubes and guides, rabbits, and isotope production targets (Fig. 3).

The coolant flows upward through the core, leading to a quicker, and more predictable, transition from forced-to-natural convection and also to a reduced probability of core flow blockage, because foreign objects or debris falling onto the core would be swept up and caught by the primary coolant screen when the flow is started.

There are two independent scram systems; one that is also used for control, is inside the central hole of the annular fuel elements and within the primary coolant circuit. The other scram system is outside the pressure boundary (Fig. 4). Either system alone can safely shut down the reactor, even if one rod were stuck.

III. FACILITIES

The reactor is housed in a large containment dome, with floor space for beam tube experiments. The experimenters are physically separated from the operating areas. A large guide hall provides space for cold neutron beam experiments, and office space is

Table 1

Quantitative Expression of Performance Goals

<u>Neutron beams</u>	
Peak thermal flux in reflector $m^{-2}.s^{-1}$	$5 - 10 \times 10^{19}$
Thermal/fast flux ratio	≥80
<u>Materials irradiation</u>[a]	
Fast flux, $m^{-2}.s^{-1}$	$\geq 1.4 \times 10^{19}$
Fast/thermal flux ratio	≥0.5
<u>Transuranium production</u>[b]	
^{252}Cf Production rate, g/year	1.5
^{254}Es Production rate, μ/year	40

a To match or exceed the capabilities of the irradiation positions in the HFIR flux trap.

b To match or exceed production capabilities at HFIR.

Table 2

Overall Requirements

Appropriate codes and regulations	User Needs
10CFR50, Appendix A (Design Criteria)	Neutron flux and spectrum
Pressure boundary integrity Two diverse scram systems Decay heat removal	Access for experiments
ASME Section III, Class 1	
DOE 6430.1a, Safety Policy	

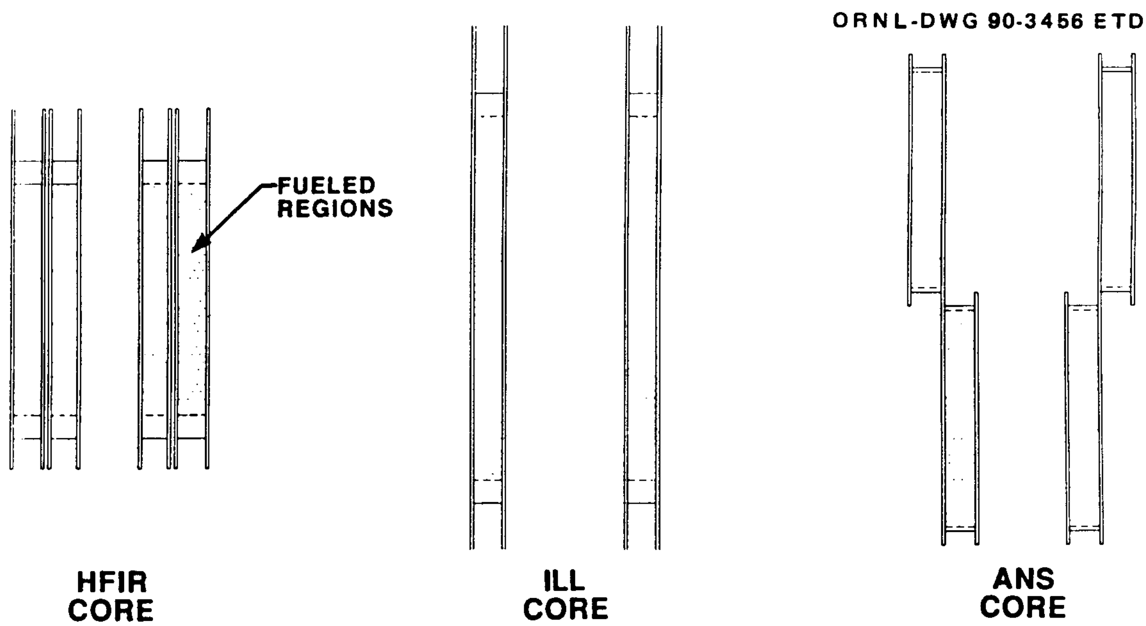


Fig. 1. Scale comparison of three involute plate, annular fuel element, research reactor cores.

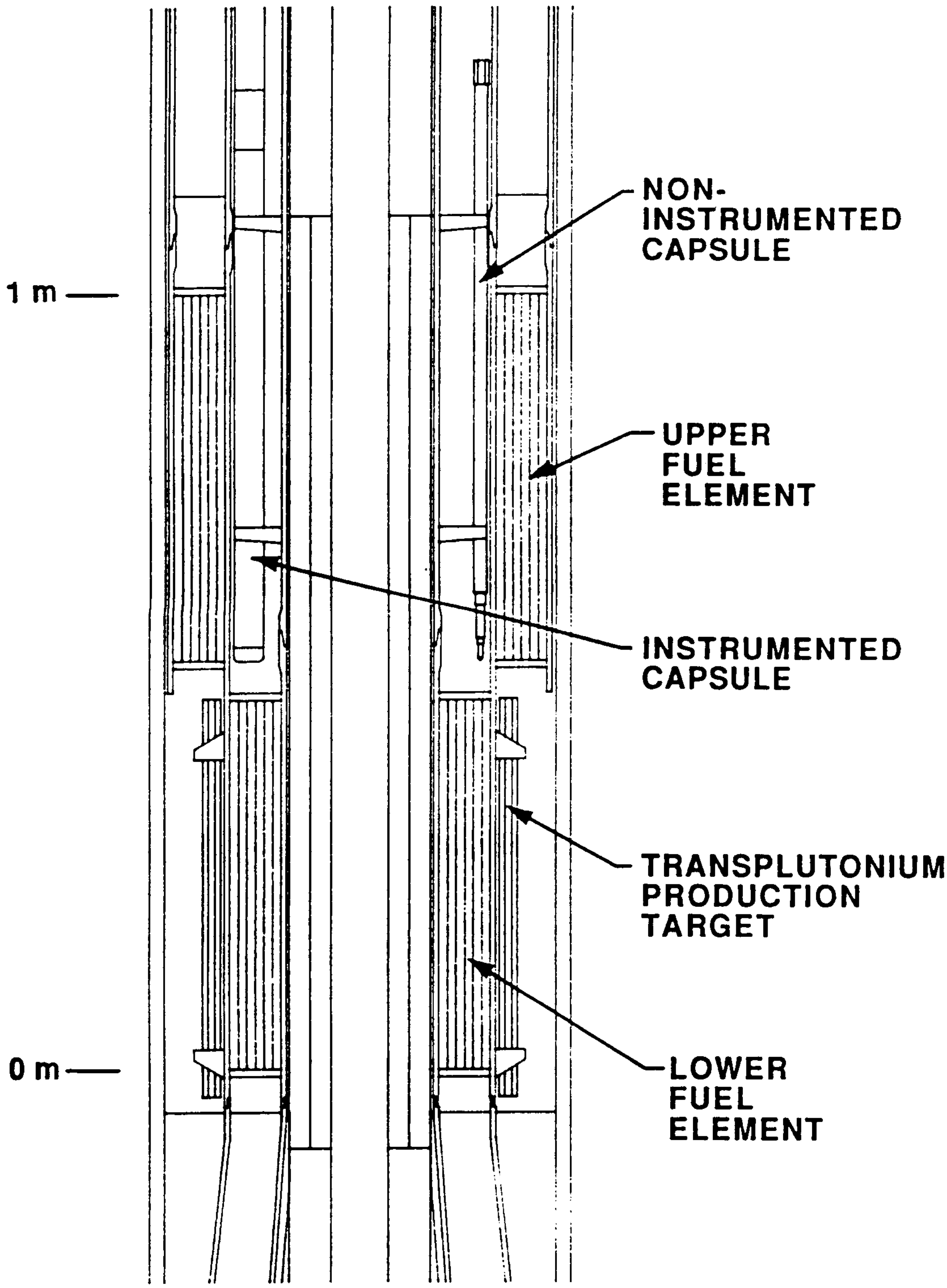


Fig. 2. Final Preconceptual Core Design

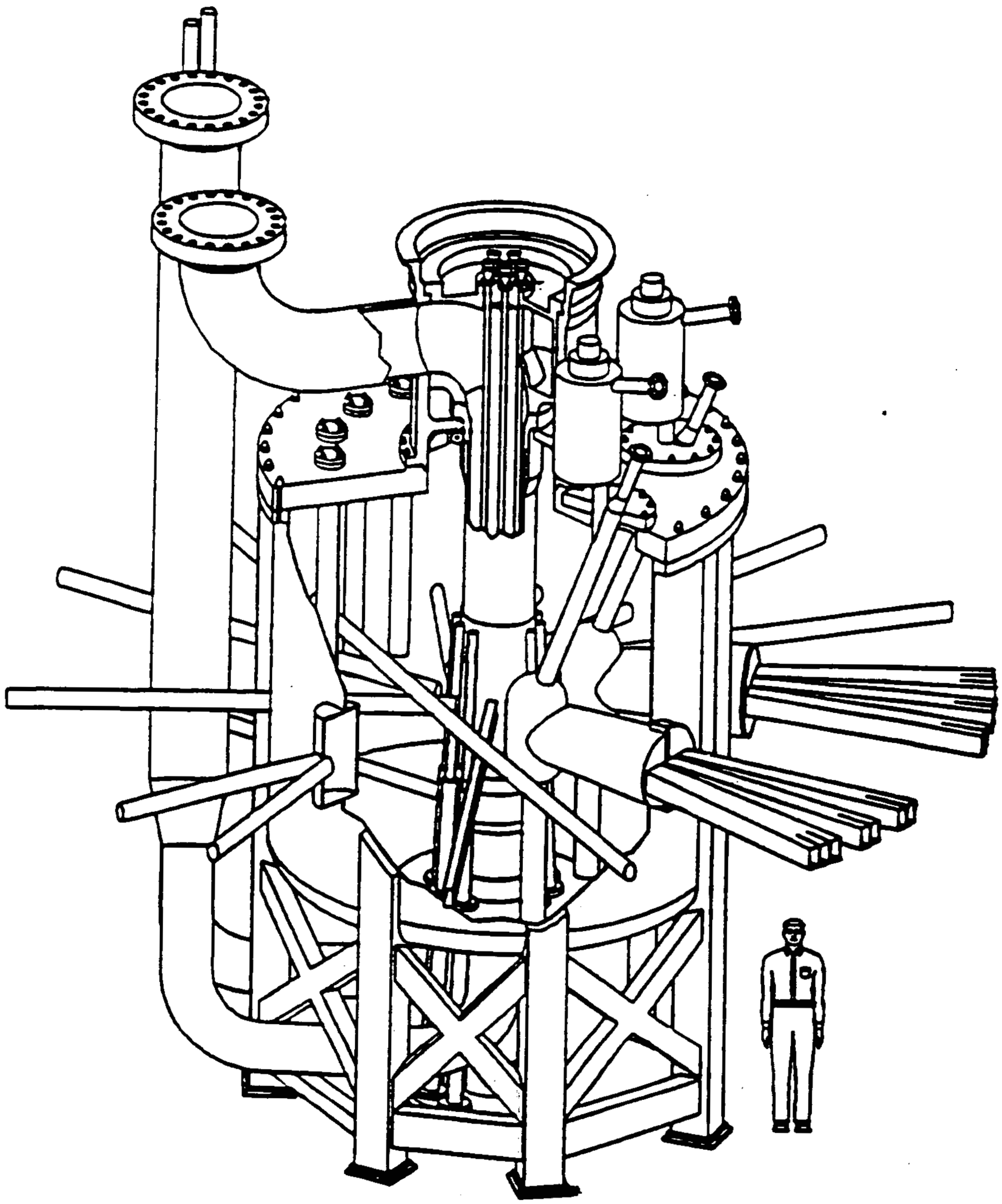
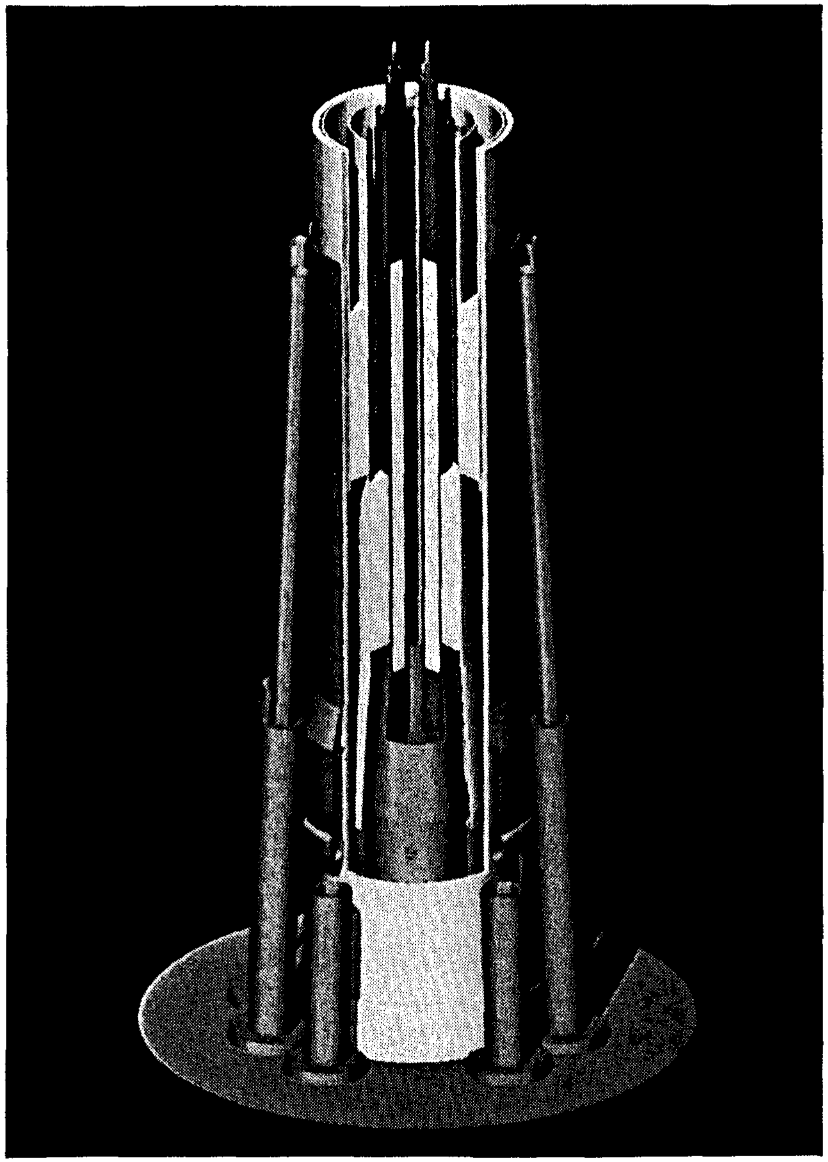


Fig. 3. ANS Reactor Systems



The ANS Reactor Core Arrangement, Using Two Separate Fuel Elements of Different Sizes, Offers Many Safety and Performance Advantages.

Figure 4

provided to accommodate facility staff and the ~1000 users expected each year (Figs. 5 and 6).

IV. SAFETY GOALS AND RESISTANCE OF THE ANS PRECONCEPTUAL DESIGN TO ACCIDENTS

The ANS design goal for core damage probability is 10^{-5}/year, the same as that proposed in the DOE's draft safety objectives policy for the much larger production reactors. For the same core damage probability, personal risks from the ANS will naturally be lower than from the New Production Reactor (NPR), because the ANS fission product inventory (source term) is very much lower. For comparison, the HFIR and the Advanced Test Reactor (ATR) each have a core damage probability of ~10^{-4}/year.

Work began on probabilistic risk assessment (PRA) for the project very early, and the results have already led to design changes. By implementing an integrated safety, research and development (R&D), and design program from the beginning of the project, many safety issues were addressed during preconceptual design to minimize later, and more expensive, design changes or retrofits (e.g., Table 3). Continued attention to the areas identified in Table 3 for further work is expected to bring the 10^{-5} risk goal within reach.

V. R&D ISSUES RAISED BY THE DIFFERENCES BETWEEN THE ANS DESIGN AND HFIR

The ANS user requirements could not be met by a facility like the HFIR. However, many of the differences between ANS and HFIR (e.g., more beams and experimental stations, a higher heat removal capacity, and more effective scattering instruments) raise no new safety issues. The higher power density does mean that several operating parameters - including coolant velocity, fission rate in the fuel, and heat flux - are outside HFIR operating experience (Table 4). The tests and analyses necessary to verify that the new operating conditions can be safely achieved are either already under way or appropriately scheduled. Independent reviewers have endorsed the design approach, basing their judgement on available data and on the project team's plans to gather the necessary additional data.

Some of the planned research and tests are listed in Table 5. Results from this R&D program will be shared with the community.

PERSPECTIVE OF ANS FACILITY

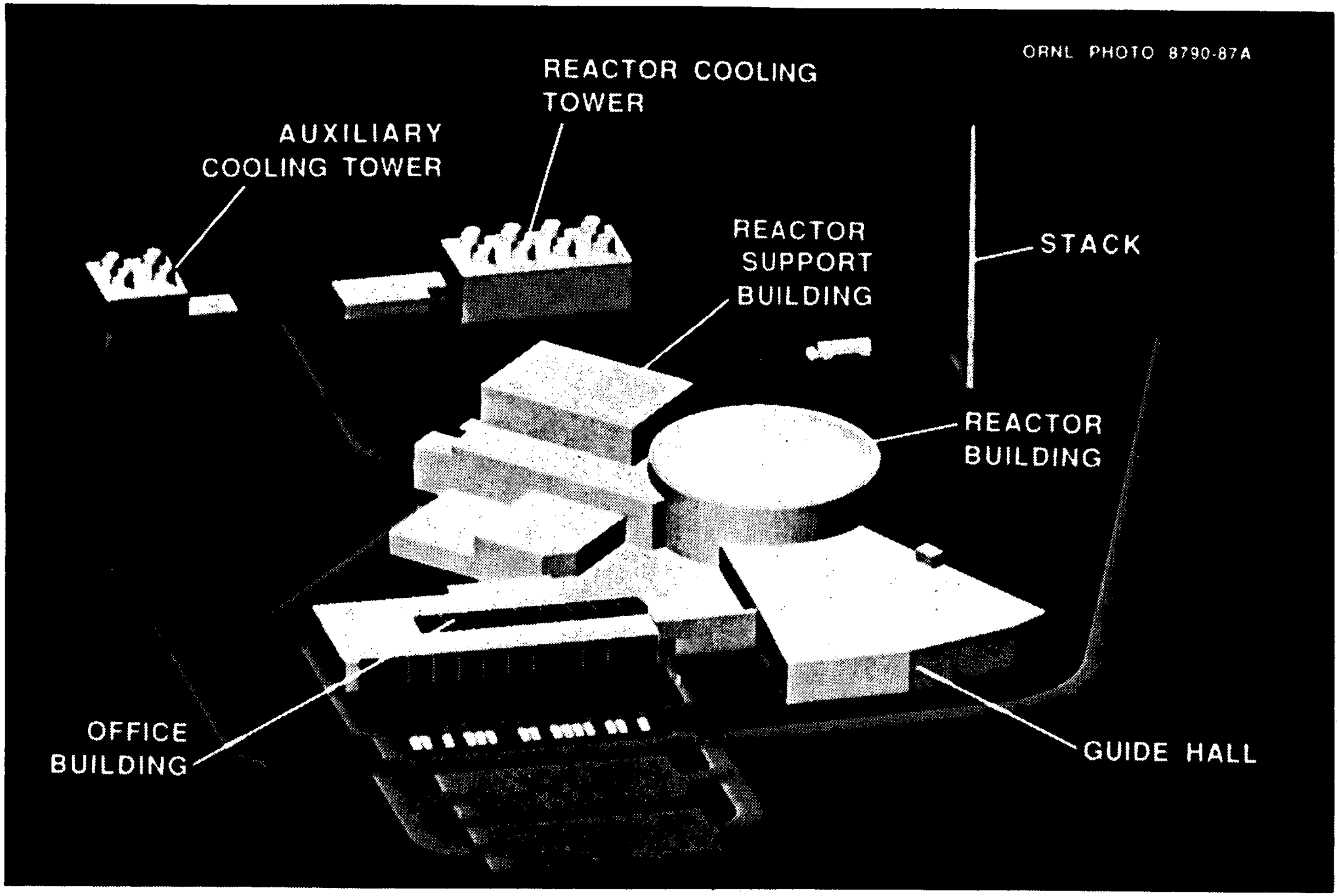


Fig. 5.

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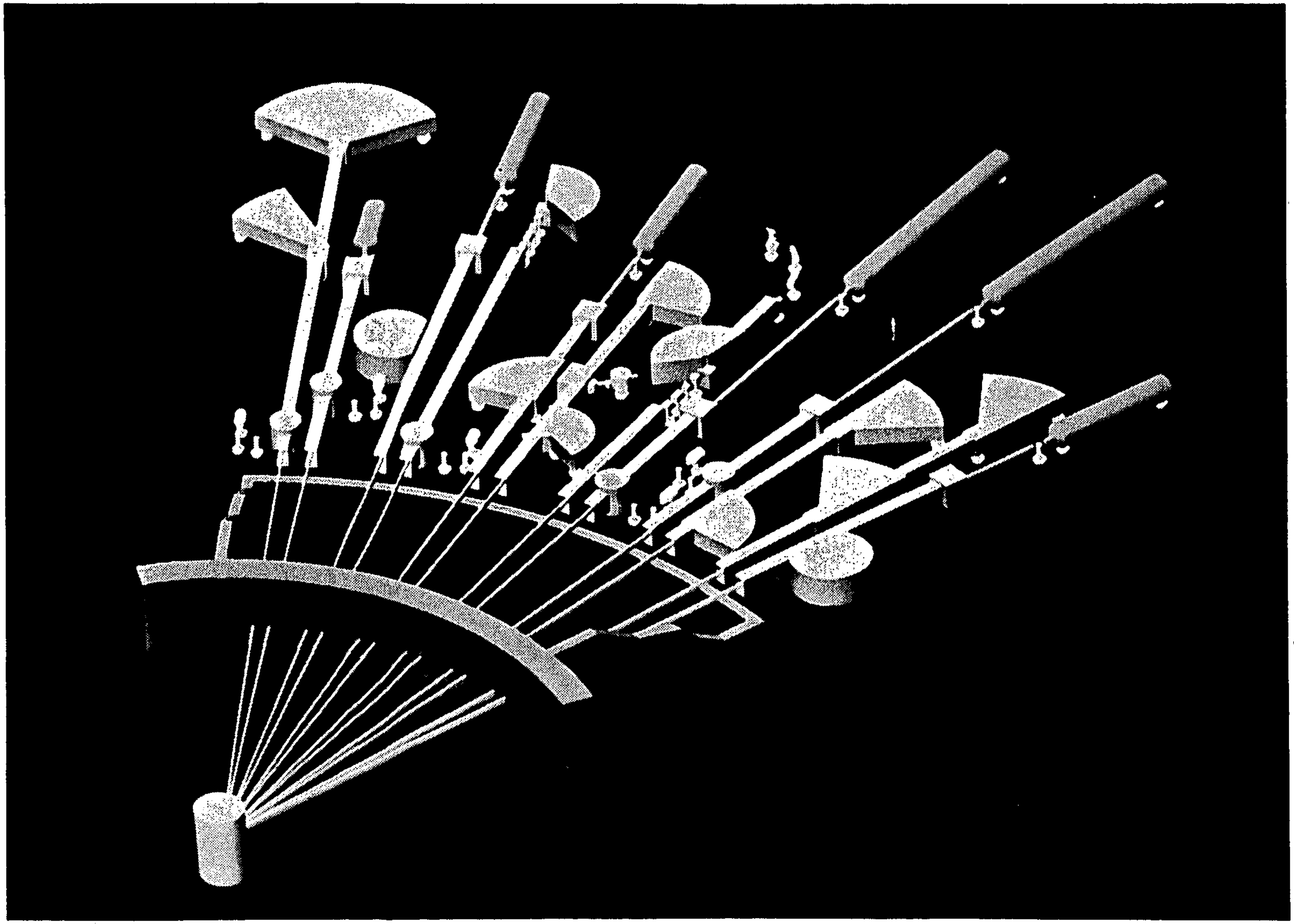

Fig. 6.

Table 3

HFIR AND ANS CORE DAMAGE PROBABILITIES FOR DOMINANT ACCIDENT SEQUENCES

INITIATING EVENT CATEGORY	CORE DAMAGE PROBABILITY (PER YEAR) HFIR[1]	ANS[2]	NOTES, REASONS FOR EXPECTED DIFFERENCES
FLOW BLOCKAGE	9.0×10^{-5}	7.2×10^{-6}	ADVANTAGE OF ANS REFUELING MACHINE AND UPFLOW
LARGE PIPE BREAK (COMPLETE SEVER-ANCE >51 mm PIPE)	3.3×10^{-5}	<3.0E-7	ANS USES FRACTURE MECHANICS-BASED LEAK BEFORE BREAK APPROACH IN THE DESIGN STAGE
SCRAM SCENARIOS (PRIMARILY MANUAL SCRAM FOR NORMAL SHUTDOWN)	2.3×10^{-5}	$<10^{-7}$	ANS DOES NOT HAVE AUTOMATIC[3] DEPRESSURIZATION OR CUTOFF OF SECONDARY COOLANT FLOW AFTER SCRAM
FUEL MANUFAC-TURING DEFECT	2.1×10^{-5}	ASSUMED SAME	ANS GOAL OF 10^{-5} TOTAL CORE DAMAGE PROBABILITY WILL REQUIRE ADVANCED FUEL INSPECTION TECHNOLOGY
LOSS OF INSTRUMENT AIR	1.7×10^{-5}	$<10^{-7}$	ANS DOES NOT REQUIRE AUTOMATIC DEPRESSURIZATION[3]
SMALL PIPE BREAKS (SEVERANCE OF ≤51 mm PIPE)	1.6×10^{-5}	ASSUMED SAME, IN ABSENCE OF FURTHER DESIGN WORK	ANS GOAL OF 10^{-5} TOTAL CORE DAMAGE PROBABILITY WILL REQUIRE SOME IMPROVEMENT BY DESIGN
PRESSURIZER PUMP FAILURES	7.3×10^{-6}	2.2×10^{-7}	
TOTAL	2.1×10^{-4} (92.5% OF TOTAL HFIR RISK)	4.5×10^{-5} (SHOULD BE ~90% OF TOTAL ANS RISK)	

[1]HFIR results based on Oct. 17, 1988 Update to the HFIR PRA.

[2]ANS results provided by Brookhaven National Laboratory (Fullwood and Shier) in the July 1989 ORNL/BNL PRA review meeting. The ANS results are very much subject to change since they are based, in part, upon preconceptual design information that will be modified and defined in great detail as the design effort progresses.

[3]ANS does not need the reactor vessel NDT-avoidance fixes (automatic depressurization of the primary and automatic post scram cutoff of the secondary coolant).

[4]HFIR has recently changed the failure mode of the automatic depressurization valves from fail-open to fail-closed, giving lower probability than listed here.

Table 4

ANS Reactor - Specifications and Comparison with HFIR

Quantity & unit	ANS	ANS notes	HFIR*
Fission power level, MW(f)	350		100
Power transferred to primary coolant, MW(c)	332	Heat convected away from fuel plates	97
Average power density, MW(c)/L	4.9		1.9
Max. power density, MW(c)/L	8.3	Estimated, fuel grading not yet optimized	4.4
Core life, d	14		20
Core active volume, L	67.4	Fueled volume	50.6
Fuel form	U_3Si_2		U_3O_8
Fuel matrix	Al		Al
Vol % of fuel in meat, %	15		12.5/18.0**
Fuel loading, kg U^{235}	14.9		9.4
Fuel cladding	6061 Al		6061 Al
Fuel plate thickness, mm	1.27		1.27
Clad thickness, mm	0.254		0.254
Coolant channel gap, mm	1.27		1.27
Coolant (and reflector)	$D_2O(D_2O)$		$H_2O(Be)$
Inlet pressure, MPa	3.7		4.1
Inlet temperature, °C	49		49
Heated length, mm	474		508
Coolant velocity in core, m/s	27.4	May be reduced after detailed analysis	16
Core pressure drop, MPa	1.6		0.7
Outlet pressure, MPa	2.1		3.4
Bulk coolant outlet temp., °C	81		73
Average heat flux, MW(c)/m^2	6.3		2.4
Max. heat flux, MW(c)/m^2	10.7	Estimated; fuel grading not yet optimized	5.6
Max. fuel centerline temp., °C	400	Design groundrule	327
Peak thermal flux in reflector, 10^{19} m^{-2}s^{-1}	>8	Unperturbed	1.5

*At 100 MW.

**Inner element/outer element.

Table 5. Some Planned Research and Tests

Area	Tests	Dates to begin (tentative)
Fuel plate stability	Epoxy plate tests (single plate)	Feb. 1990
	Aluminum plate tests (single plate)	Oct. 1990
	Epoxy plate tests (multiplate)	Nov. 1990
	Aluminum plate tests (multiplate)	Jan. 1992
	Thermal stress tests	May 1993
	Full core test	June 1993
	Plate vibration tests	Sept. 1993
Thermal hydraulics	T.H. limits narrow channels with coolant velocity	Sept. 1990
Fuel performance	Fuel tests by accelerator irradiations	underway at ANL
	In-pile sample tests at HFIR	awaiting HFIR full power operation
	Fabrication tests of two-dimensional grading	underway at B&W
	Test fabricate fuel plates to conceptual core design	Sept. 1990
	Miniplate tests in HFIR	Jan. 1991
Oxide formation	Out-of-pile tests with high coolant velocity and heat flux	underway at ORNL
	In-pile test at HFIR	Jan. 1992
Cold source	Liquid nitrogen simulations	July 1990
	Liquid H_2 or D_2 tests	Jan. 1993

VI. SUMMARY

The ANS Project has completed the preconceptual design phase. An evolutionary process has led to a new reference core design with greatly enhanced thermal-hydraulic margins and improved performance parameters in many areas. The reactor systems design has also evolved in response to input from the integrated safety analysis program and from HFIR studies and reviews. R&D is under way or planned in those areas important to safety, design, and performance improvement.

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THE HIGH FLUX REACTOR PETTEN, PRESENT STATUS AND PROSPECTS

J. Ahlf

The High Flux Reactor Petten, Present status and prospects

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ABSTRACT

The High Flux Reactor (HFR) in Petten, The Netherlands, is a light water cooled and moderated multi-purpose research reactor of the closed-tank in pool type. It is operated with highly enriched Uranium fuel at a power of 45 MW. The reactor is owned by the European Communities and operated under contract by the Dutch ECN. The HFR programme is funded by The Netherlands and Germany, a smaller share comes from the specific programmes of the Joint Research Centre (JRC) and from third party contract work.

Since its first criticality in 1961 the reactor has been continuously upgraded by implementing developments in fuel element technology and increasing the power from 20 MW to the present 45 MW. In 1984 the reactor vessel was replaced by a new one with an improved accessibility for experiments. In the following years also other ageing equipment has been replaced (primary heat exchangers, pool heat exchanger, beryllium reflector elements, nuclear and process instrumentation, uninterruptable power supply). Control room upgrading is under preparation. A new safety analysis is near to completion and will form the basis for a renewed license.

The reactor is used for nuclear energy related research (structural materials and fuel irradiations for LWR's, HTR's and FBR's, fusion materials irradiations). The beam tubes are used for nuclear physics as well as solid state and materials sciences. Radioisotope production at large scale, processing of gemstones and silicon with neutrons, neutron radiography and activation analysis are actively pursued. A clinical facility for boron neutron capture therapy is being designed at one of the large cross section beam tubes.

It is foreseen to operate the reactor at least for a further decade. The exploitation pattern may undergo some changes depending on the requirements of the supporting countries and the JRC programmes.

1. INTRODUCTION

The High Flux Reactor (HFR) in Petten, The Netherlands, is a multi-purpose research reactor operated with highly enriched uranium as a fuel at a power of 45 MW.

The HFR is property of the European Communities. It is operated under contract by the "Netherlands Energy Research Foundation (ECN)". The programme is managed by the "HFR Division" of the "Institute for Advanced Materials", one of the nine institutes of the "Joint Research Centre (JRC)" of the European Communities. The HFR programme is executed in four year periods. The present programme period covers the years 1988 to 1991. The running programme is funded by a "Supplementary Programme" shared by the Netherlands and the Federal Republic of Germany, and by "Specific Programmes" of the JRC. In addition a target is set for third-party contract earnings. The programme resources are outlined in Table 1.

The programme objectives are defined by Dutch or German institutions under the supplementary programme, by JRC institutes for the specific programmes and by interested institutes or companies from inside and outside the European Communities under third-party contract research.

The irradiation capacity of the HFR is utilized to a high degree. The largest share is still related to materials technology for fission reactors (structural materials and fuel irradiations for LWR's, HTR's and FBR's) and future fusion machines. The beam tubes are used for nuclear physics, but with a larger share for solid state physics and materials sciences. Radioisotope production, processing of gemstones and doping of silicon with neutrons, neutron radiography and activation analysis are actively pursued and offered to commercial customers. Design of a clinical facility for boron neutron capture therapy and related research is in progress. The facility will be installed at one of the large cross section beam tubes.

2. CHARACTERISTICS OF THE HFR

The HFR has been designed according to the principles of the Oak Ridge Research Reactor (ORR). It is a classical multi-purpose research reactor. Its nuclear and thermal characteristics are compiled in Table 2. The reactor core is housed in a closed tank, which together with the circulation pumps and the heat exchangers forms the primary circuit. Light water is used as a coolant and moderator. The reactor tank is submerged into a deep water filled pool with thick concrete walls. The pool is lined with an aluminum liner. Presently the HFR is operated at 45 MW using highly enriched uranium as fuel. The core lattice is a 9 x 9 array containing 33 fuel assemblies, 6 control members, 19 experiment positions and 23 beryllium reflector elements. The row at the east side of the core lattice, normally loaded with 9 beryllium reflector elements, is arranged outside the core box of the reactor vessel.

The fuel assemblies contain 23 fuel plates with an active length of 600 mm. The uranium is about 93 % enriched in U-235. The fresh fuel uranium content per assembly is presently increased from 420 g to 450 g U-235. The two side- plates of each fuel assembly contain together 1000 mg B-10 in fresh condition. The control elements consist of a cadmium section on top of a fuel section. The fuel section contains 19 fuel plates with a total fresh mass of 290 g U-235. Their drive mechanism is situated below the reactor vessel.

The new reactor vessel installed in 1984 has advanced characteristics with respect to accessibility for experimental equipment as compared to the original ORR-design. The vessel is shown in Fig. 1. It is an aluminium construction consisting of a lower cylindrical part embedded in the concrete floor of the pool and an upper part arranged in the reactor pool. Those parts are flanged and bolted together at the pool bottom level. The upper or pool part of the vessel is an all-welded construction and has apart from the support structure a rectangular cross section, which provides an easier access to the pool side facilities than the circular cross section of the older design. Direct vertical experiment access to the reactor core positions is through the holes of the central reactor top lid, which also supports the experimental tubes.

Adjacent to the reactor pool there are two smaller pools for storage and handling purposes. On top of one of these two pools a hot cell for the dismantling of irradiated capsules is placed. A simplified sketch of the installation is given in Fig. 2.

The arrangement of the irradiation possibilities at the HFR is shown in Fig. 3. There are 9 in-core positions in the fuel region of the core and 10 boundary and reflector positions. The 12 positions in the large pool side facility are highly valuable, mainly for transient tests and fuel rods, whereas the new small, low fluence rate pool side facility suffers from interference with some of the beam tubes. From the 12 horizontal beam tubes, two, namely HB11 and HB12, have been installed together with the new vessel, replacing the old thermal column. They have a very large cross section and advantageous characteristic for future use for boron neutron capture therapy. The vertical beam tubes and a pneumatic rabbit system are mentioned for completeness.

A detailed description of the operational characteristics of the reactor and the experimental facilities is given in Ref. [1].

3. MAJOR UPGRADING AND MODERNIZATION ACTIVITIES

Design of the HFR commenced in 1958 and first criticality was reached in November 1961. The major milestones of HFR's history are summarized in Table 3. Power increases and performance improvements with respect to the provision of more and higher flux density in-core positions was rendered possible mainly by fuel technology development which was characterized by increasing the mass of U-235 in the assembly from 120 g in 1961 to 450 g in 1990, and by the introduction of B-10 as a burnable poison.

In the mid seventies it was realized that embrittlement of the vessel material would become a licensing problem in the future. So it was decided to install a new vessel. The vessel replacement was carefully planned and prepared until action of removing the old vessel and replacing it by a new one took place in a shut-down period not longer than 15 months [2]. As mentioned above the new design incorporated major improvements with respect to experimental utilization. In addition provisions were made for an optional further power increase to 60 MW.

After the vessel replacement the process of upgrading and replacement of ageing equipment was accelerated, always under the consideration to keep the option open on further increase of operating power to 60 MW.

The old primary heat exchangers, being designed for 20 MW operation, gave some problems during summer conditions in heat removal capacity at 45 MW and made further power increase impossible. In addition the necessary increased flow at the secondary side, beyond design conditions, for a considerable period of time, led to degradation and increased vibrations at the penetrations of the baffle plates for the numerous pipes. As the available limited space in the bunkers did not allow to introduce tube/shell heat exchangers of the enlarged capacity, a new plate type heat exchanger with titanium plate was chosen. The number of maintenance and cleaning actions could be reduced considerably as a result of daily application of the built-in backflash option.

Also the pool heat exchanger was replaced to care for larger heat removal capacity. An additional reason for this replacement by a plate type was the need to be able to clean the primary side of the exchanger. This became necessary to avoid the growth of algae in the exchanger, which deteriorate heat removal capacity.

The original beryllium reflector elements are still in use with all indications of their extended use and handling damage after nearly 30 years of utilization. Embrittlement and dimensional changes caused by the high received neutron fluence as well as indications of reactivity loss after reactor shutdown due to the ingrowth of He-3 and Li-6 became a matter of concern. New elements were ordered with updated technical specifications, and after their recent delivery they will be inserted into the core in the near future.

In order to promote diversification and redundancy for the flux protection system, an extra set of three nuclear safety channels of different design will be introduced, which in the case of overpower will directly act on the magnet circuits of the control rods in the sub-pile room. The location will be completely outside the reactor control room to decrease the risk of common mode failures in case of fire.

Increased failure rates, and unavailability of spare parts led to the replacement of the major part of the nuclear channels. Also the uninterruptable power supply was replaced to avoid increasingly costly repairs of outdated equipment, and also to introduce redundancy and to relocate for diversity in cabling routes in search for additional fire protection measures and prevention of common mode failures. A complete upgrading of the control room is now under preparation. Outdated components have to be replaced, and the new design will incorporate modern ergonomic principles.

On request of the Dutch licensing authorities a complete reappraisal of the safety analysis is in progress which will replace the old HFR hazard report written more than 30 years ago. This safety analysis will be the basis for a renewal of the present HFR operating license. In parallel the set-up of a comprehensive quality assurance system is in progress which comprises a systematization and documentation of all the existing practices and procedures for reactor operation proper and all the activities connected to the exploitation of the reactor.

4. EXPLOITATION OF THE HFR PETTEN

The current programme of the HFR addresses a broad scope of applications of neutrons to science and technology which are briefly addressed in the following section [3].

LWR fuel rod behaviour is investigated under steady state and transient conditions. Both BWR and PWR operating conditions can be simulated. For transient scenarios in a wide range of power ramp rates, the pool side facility provides particularly favourable possibilities.

HTGR structural and reflector graphite is irradiated with emphasis on the behaviour under specified load conditions in the temperature range between 300 and 1200°C. HTGR fuel irradiations address mainly the fission product release behaviour in a wide temperature domain (600°C to 1500°C).

The HFR Petten is also participating in international R&D programmes on LMFBR fuel. Mixed oxide fuel as well as advanced concepts - carbide and nitride fuel - are tested under start-up and in-situ operational transients. FBR structural materials are irradiated to high fluences in order to assess mechanical properties, including creep and fatigue under irradiation.

Materials research for fusion has increased largely in recent years. The present tests in the HFR are embraced by the European Fusion Technology Programme. They mainly concern creep, fatigue and crack growth in austenitic stainless steel together with research on vanadium alloys and on structural ceramics as well as testing of ceramic and liquid metal candidate blanket breeder material with on-line tritium release measurements.

Present utilization of the horizontal beam tubes is shown in Fig. 4. The programme comprises crystal and magnetic structures, ordering in liquid and amorphous alloys, spin density distributions, phonons, spinwaves and residual stress measurements by diffraction and inelastic scattering of neutrons, further the study of inhomogeneities in technical materials by means of small angle neutron scattering, for which purpose a new facility has been brought into operation in 1989. One of the beam tubes is in permanent use for neutron radiography, methodology development as well as applications, mainly in the space and aircraft industry. At the large cross section beam tube 11 a filtered beam facility for boron neutron capture therapy with epithermal neutrons is being designed. This facility will be the principal research tool for BNCT research in Europe.

In view of decreasing irradiation capacity and in view of increasing demand for radioisotopes for medical and industrial purposes the radioisotope production services at the HFR are being upgraded presently. In the field of activation analysis the HFR offers several facilities over a wide range of irradiation times and sample volumes. Facilities for silicon doping and gemstone colouring are also in operation.

Efficient utilization of a research reactor is only possible, when it is embedded into the infrastructure of a large nuclear research centre. In this respect the close and fruitful co-operation between JRC and ECN is noteworthy. The HFR programme makes ample use of neutron metrology and reactor physics services and also of the well equipped ECN hot laboratories.

5. SUMMARY AND CONCLUSIONS

The High Flux Reactor Petten is operated as a multi-purpose research reactor and serves as a principal tool for a wide variety of applications. The programme is mainly sponsored by The Netherlands and Germany and by the European Communities, but the reactor is also offered to institutions and companies from Europe and abroad.

Being continuously upgraded and modernized since its first criticality in 1961, the reactor can be regarded as a modern and up-to-date research tool even after nearly 30 years of operation. Experience has developed and equipment is available for addressing the full scope of materials testing for nuclear energy deployment as well as for efficient utilization of the beam tubes for fundamental and applied research.

6. REFERENCES

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Table 1 :

PROGRAMME RESOURCES 1988 - 1991

Supplementary Programme		
a)	exploitation of the reactor	
	- Federal Republic of Germany	32.5 MECU
	- The Netherlands	32.5 MECU
b)	preparation of experiments	
	- Federal Republic of Germany	6.5 MECU
	- The Netherlands	p.m. *
JRC specific programmes		7.0 MECU
Third party contract earnings		5.0 MECU

Total		83.5 MECU +p.m. *

* work to be carried out directly by the Netherlands, valued by the Commission at 6.5 MECU

Table 2 : Summary of nuclear and thermal properties of the HFR

Reactor power	45 MW	
Specific power (averaged over fuel positions)	310 MW/m^3	
Number of fuel assemblies	33	
Number of control members	6	
Number of in-core irradiation positions	9	
Number of reflector irradiation positions	8	
Number of horizontal beam tubes	12	
Number of pool side facility positions	22	
Fuel charge of fresh fuel assemblies	420 g ^{235}U	
fresh fuel loading of the fissile		
control members followers : 290 g		
Boron charge in side plates of fresh fuel assemblies	1000 mg ^{10}B	
Total fuel charge	11 kg ^{235}U	
Volume of core	0.2 m^3	
Average thermal flux density in inner fuel position	$1.0 \times 10^{18}\ m^{-2}s^{-1}$	
Maximum thermal flux density in inner fuel position	$1.6 \times 10^{18} m^{-2}s^{-1}$	
Maximum fast flux density in in-core exp. position	$2.8 \times 10^{18} m^{-2}s^{-1}$	}
Maximum fast fluence rate in pool side facility I	$3.8 \times 10^{17} m^{-2}s^{-1}$	} equivalent fission fluence rate
Maximum fast fluence rate in pool side facility II	$1.6 \times 10^{16} m^{-2}s^{-1}$	}
Maximum thermal fluence rate in in-core exp. pos.	$1.5 \times 10^{18} m^{-2}s^{-1}$	
Maximum thermal fluence rate in pool side facility I	$2.7 \times 10^{18} m^{-2}s^{-1}$	
Maximum thermal fluence rate in pool side facility II*	$3.2 \times 10^{17} m^{-2}s^{-1}$	
Radiation heating graphite : in-core positions	6 to 12 W/g	}
reflector positions	2 to 6 W/g	} maxima in axial direction
pool side facility	<3 W/g	}
Flow rate of primary coolant through core	1.14 m^3/s (4100 m^3/h)	
Coolant speed in fuel assembly	7 m/s	
Coolant speed in filler element	0.2 to 7 m/s	
Inlet temperature of coolant	318 K (45°C)	

Table 2 (continued)

Outlet temperature of coolant	328 K (55°C)
Temperature difference across the reactor core	10 K
Average heat flux density in mid position	1.00 MW/m^2 (100 W/cm^2)
Maximum heat flux density in mid position	1.60 MW/cm^2 (160 W/cm^2)
Absolute pressure above reactor core	340 kN/m^2 (3.4 bar)
Pressure difference over the reactor core	110 kN/m^2 (1.1 bar)

• PSF 22 and PSF 27 only

Table 3 : HFR Petten, History

1958 - 1961	Design and construction
1961	First criticality of HFR (November 9)
	Maximum power 20 MW
1962	Transfer from RCN to EURATOM (October 31)
1966	Power increase to 30 MW (May 8)
1970	Power increase to 45 MW (February 20)
Mid 1972	Introduction of burnable poison
1974 - 1977	Feasibility study for replacement of reactor vessel
1978	Decision to replace reactor vessel
1980 - 1981	Design of new reactor vessel
Nov.'83 - Feb.'85	Period of shut-down for reactor vessel replacement
Jan. 1985	First criticality after vessel replacement
1987	Replacement of primary heat exchangers
1988/1989	Replacement of beryllium reflector elements
1989	Replacement of pool heat exchangers

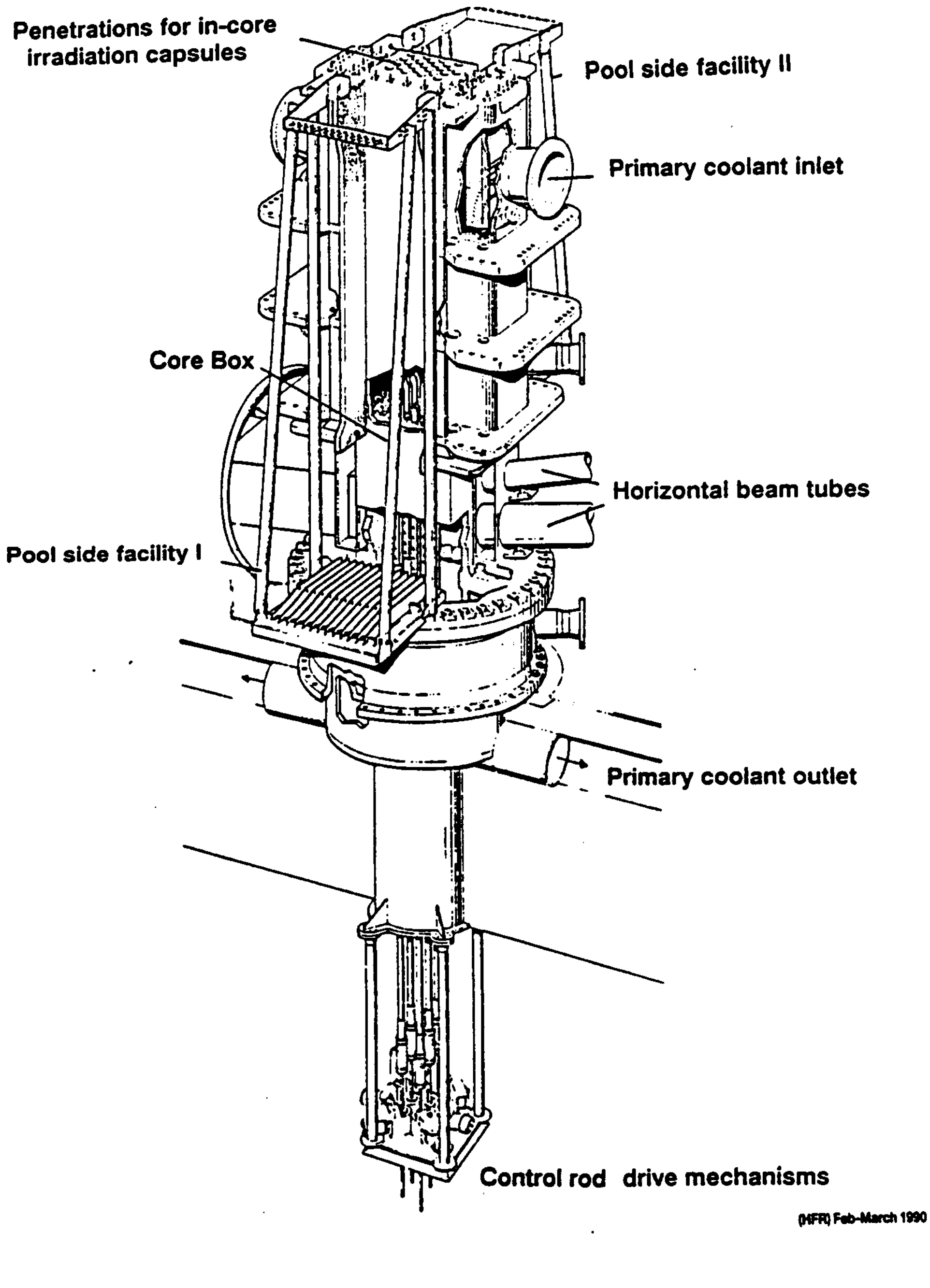


Fig. 1. New reactor vessel of HFR

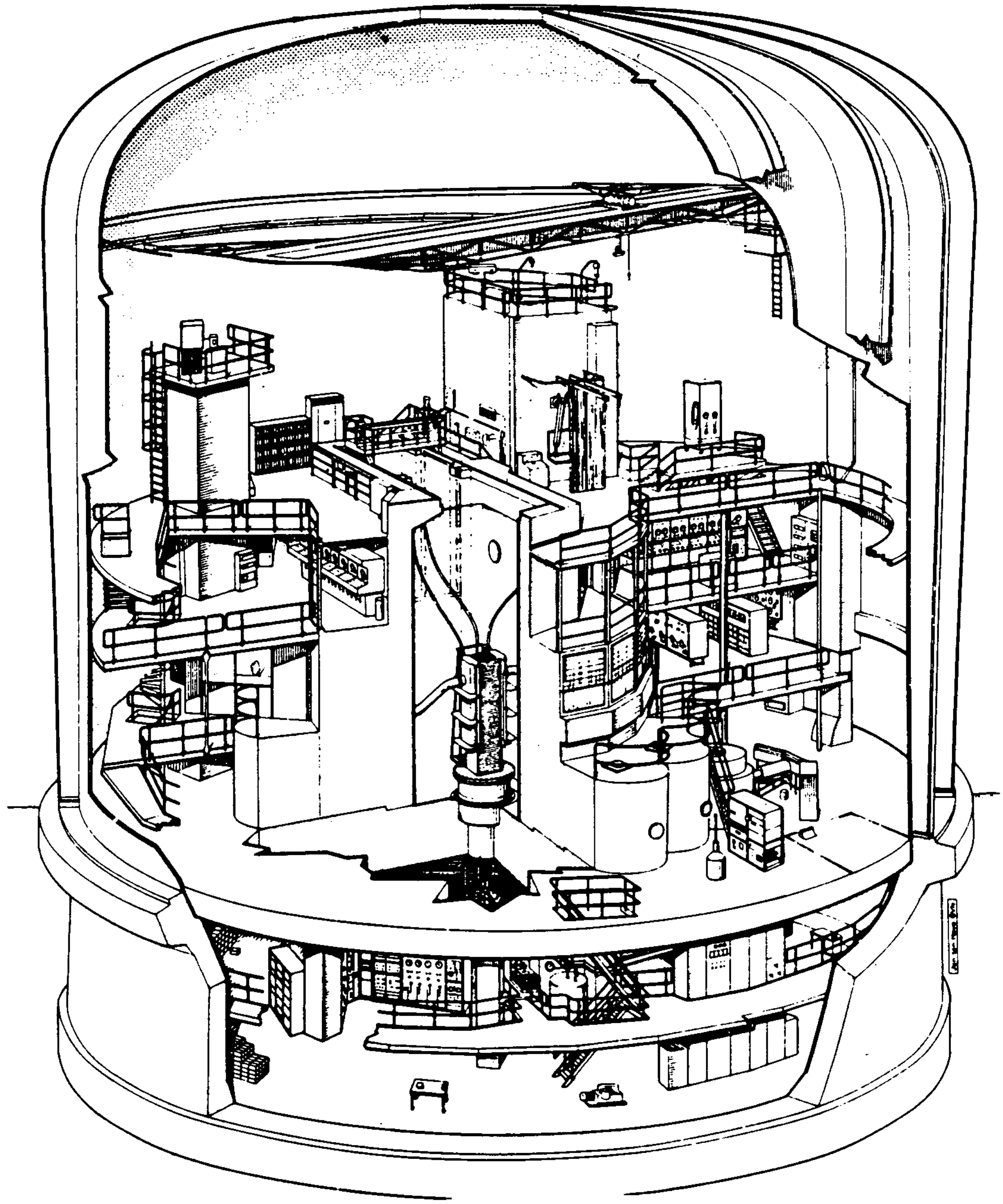


Fig. 2. Isometric drawing of the reactor building

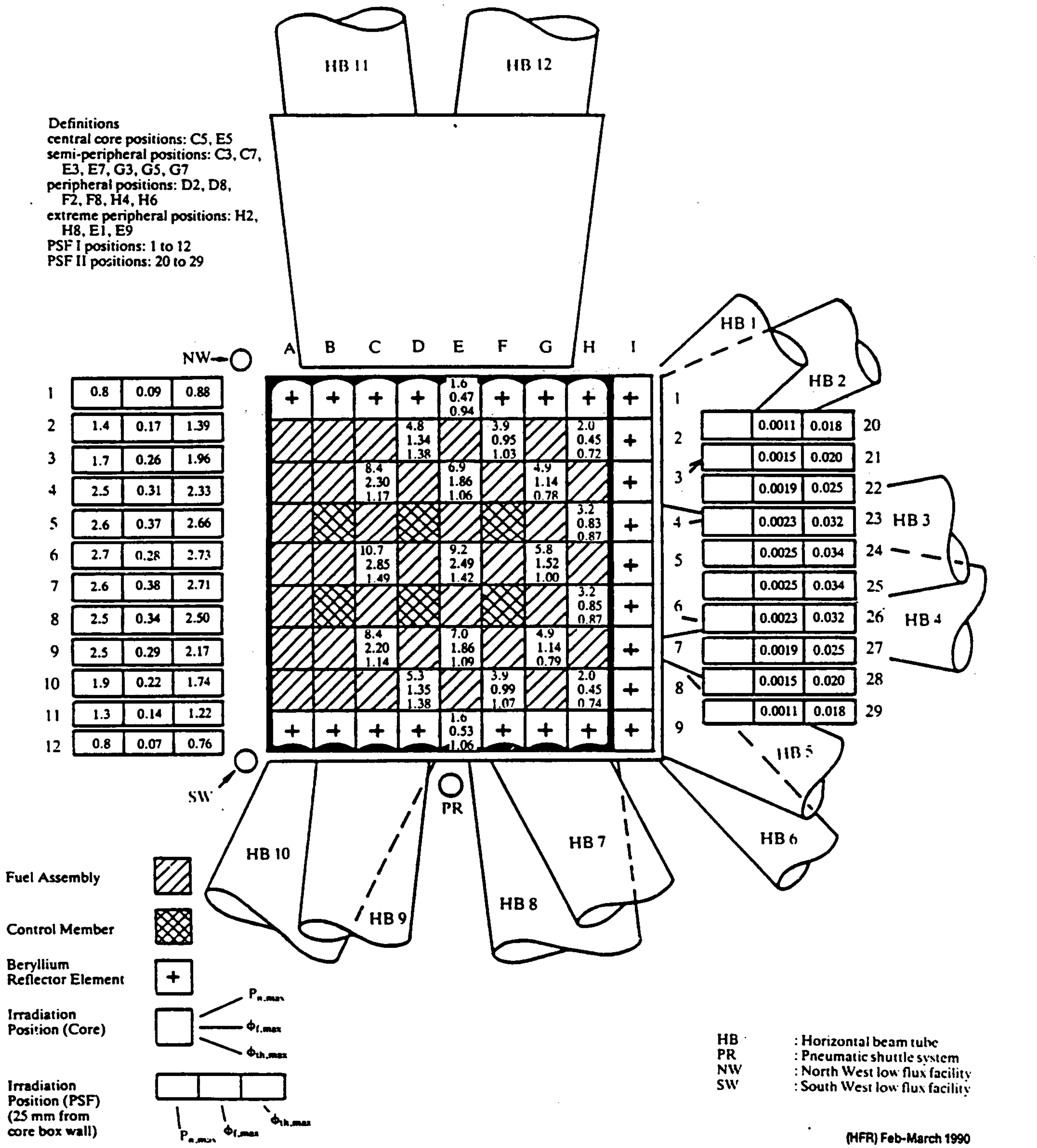
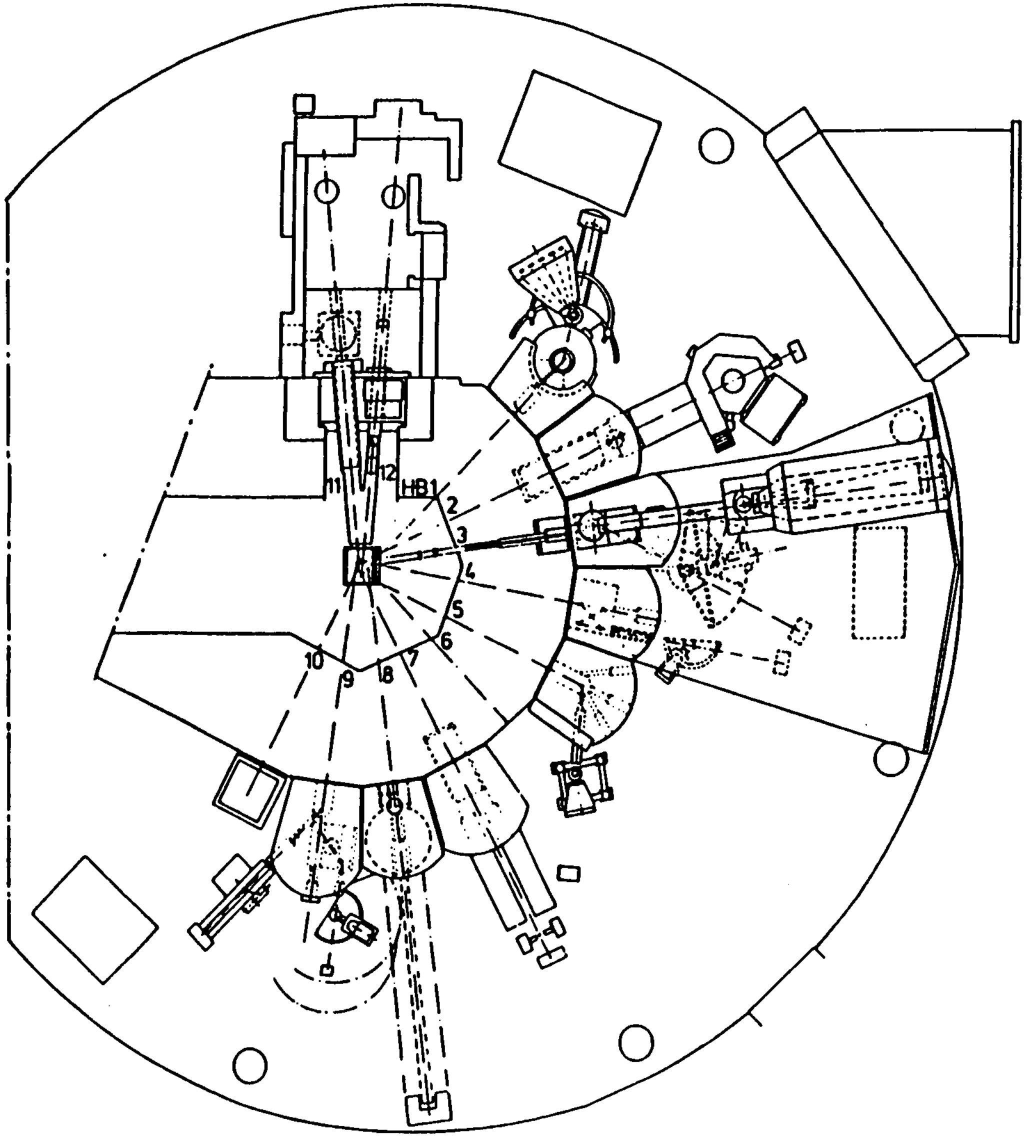


Fig. 3. Standard core configuration with nuclear values and permanently installed experimental reactor facilities



HB1 diffuse scattering/triple axis spectrometer

2 nuclear polarization set-up

3 A. triple-axis spectrometer
B. small angle neutron scattering facility (sans)

4 diffractometer for stress analysis

5 double-axis diffractometer

6

7 polarized neutron capture set-up

8 thermal/sub-thermal radiography

9 A. single-crystal diffractometer
B. activation analysis set-up

10 fasy

11 neutron capture set-up

12 filtered beam facility/(proton polarization set-up)

(HFR) Feb-March 1990

Fig. 4. Horizontal beam-hole neutron experiments

THE HIGH FLUX BEAM REACTOR INSTRUMENT UPGRADE

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The High Flux Beam Reactor Instrument Upgrade

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ABSTRACT

A brief overview is given of the current status of the High Flux Beam Reactor, including the ongoing safety review, and of plans for the upgrade of the instruments on the experimental level.

I. INTRODUCTION

Brookhaven's High Flux Beam Reactor (HFBR) is one of the major world centers for neutron scattering investigations in solid state and nuclear physics, chemistry and biology. The first reactor designed expressly for neutron beam research, the HFBR began operation in 1965 at 40 MW thermal power. It has operated since 1982 at 60 MW, providing a maximum thermal flux of 1×10^{15} n/cm^2-sec. It uses 9.8 kilogram of 93% enriched U-235 fuel configured in a compact core consisting of 28 curved plate elements. Heavy water serves as both coolant and moderator and the core is undermoderated to produce the peak thermal flux outside the core region with a minimum contamination by fast neutrons. The HFBR has a 24-day fuel cycle during which one half of the fuel elements are expended. It typically operates 10 cycles annually.

The HFBR is the only research reactor in the U.S. with a liquid hydrogen moderator, and it supports a major program of sub-thermal neutron investigations. Its nine ports serve 15 experimental facilities. Most of these facilities are operated by Participating Research Teams (PRT's) consisting of Brookhaven scientists and regular users of the HFBR from other government laboratories, industrial research laboratories and universities. A 25% fraction of the PRT instrument time is set aside for general users (non-PRT members), who obtain beam time through a peer reviewed proposal system.

The HFBR has been shut down since March 1989 for an extended safety review. The shutdown was initiated as the result of a BNL study of the consequences of a hypothetical major loss of coolant accident (LOCA). The conservative conclusions of this study could not rule out the possibility of some minor (1%) fuel damage at 60 MW operation accompanied by unacceptably high potential radiation exposures for small numbers of reactor personnel. Because no fuel damage was anticipated at 40MW operation, BNL proposed continuing at this reduced power level pending further study of possible problems attending 60 MW operation. DOE rejected this plan. In the months following shutdown several committees have examined and initiated studies of all vital aspects of HFBR operation. These studies have centered principally upon the thermal-hydraulic problems associated with LOCA's, containment vessel embrittlement and management and training of the reactor staff. The investigative phase of these investigations has wound-down and no problems have been identified which would preclude resumption of operation at 40 MW. The present schedule calls for 40 MW operation of the HFBR by August 1990. Although improved analytical calculations no longer suggest the likelihood of fuel damage at 60 MW operation, it appears that some physical modelling will be desirable before higher power operation is resumed.

II. PRESENT AND FUTURE ROLE OF THE HFBR

During its last full year of operation the HFBR had the largest neutron scattering program (263 users) of any U.S. neutron facility. The instruments available at the HFBR have been developed to cover a wide range of scientific studies. Major programs currently exist in protein crystallography and small angle scattering studies including biological assemblies (proteins, viruses and membranes) as well as metallurgical precipitates, micelles, polymer structures, etc. Single crystal structure determinations emphasize precise location of light atoms in carbohydrates and organometallic compounds, and on magnetically ordered structures. Inelastic neutron scattering studies are carried out on thermal excitations of solids and liquids using polarized and unpolarized neutrons. Recent studies have, for example, helped elucidate the structural and magnetic properties of high temperature superconductors. Nuclear physics center on neutron capture reactions and short-lived fission fragment studies using an on-line mass separator. Seven vertical thimbles provide a variety of neutron energies for sample irradiations, and also supports a major positron physics facility through the production of Cu-64 positron sources. Two new instrument (recently completed neutron reflectometer and a high resolution powder diffractometer still under construction) further increase the scope of the program.

Over the years there has been a steady increase in the number of scientists using the HFBR and it is fast approaching the limit of its research capacity. Moreover, in spite of the instrument building activities noted above, many of the existing instruments are 25 years old and are becoming increasingly outmoded.

Especially in view of the cost and complexity associated with building and licensing new reactors, well maintained high performance reactors such as the HFBR must be considered as renewable national resources. As has been noted in several high level reviews, the HFBR, with little modification is capable of supporting a considerably larger and broader-based research program than exists at the present time. The HFBR Upgrade, together with the Advanced Neutron Source, a next generation reactor presently under development, provide a coordinated plan for meeting our national needs for research reactors well into the next century.

The complement of new and refurbished instruments, many of which take advantage of the unique hydrogen cold neutron moderator at the HFBR, are summarized in Figure I, and will be built with very minor interruption of the ongoing HFBR programs. The improved performance of these instruments increase the scope of experiments susceptible to investigation, reduce the time necessary to perform a given experiment and thereby will consequently permit a substantial growth in the number of users that can be accommodated in the HFBR Program. Much necessary development and testing of instrumentation concepts relevant to the success of the ANS project will be performed. Moreover, wherever practical, these activities will be carried out in collaboration with the ANS staff and/or other appropriate neutron scientists in the U.S. Many of the new instruments could be transferred to the ANS in the future, should this prove desirable. Finally, the Upgrade will permit the HFBR to continue to accommodate the growing neutron scattering community until the ANS is in operation.

The reactor pressure vessel, beam tubes and core support structure are made of aluminum alloy that maintains its properties particularly well in high radiation fields. Recent metallurgical studies of samples taken from the highest flux regions indicate that radiation damage effects have not, as yet, become severe. On this basis it is expected that full power operation could continue for more than another decade before replacement of any in-pile components are necessary. And, should such replacement become necessary we have begun an engineering study for removal of the beam tubes by remote cutting, and replacement using a flanged seal. From a structural point of view, all indications are that the HFBR can support a world-class neutron scattering program well into the next century.

III. BRIEF PHYSICAL DESCRIPTION OF THE UPGRADE PROJECT

The HFBR Upgrade proposal is a $22M, four-year plan to build five new neutron scattering instruments at the HFBR and to rebuild six others to current standards. It was developed in response to several high level studies which called attention to the strong neutron instrumentation initiatives in Western Europe and to the aging and saturation of U.S. facilities. A major DOE review of all aspects of design, cost and management was satisfactorily performed in 1988, and the project now awaits funding.

It is proposed to make the following modifications to the reactor's external beam configuration: (i) Two additional thermal neutron beams would be brought out of the reactor. (ii) Existing beams would be enlarged wherever possible and modern focusing techniques applied to take fullest advantage of the flux available. (iii) Instrument shielding would be increased to keep the room background at its current low level. (iv) Neutron guides would be installed so that new instruments operating with sub-thermal neutrons could be placed well back from the reactor shield face where there is more floor space available and the background is lower.

Five new instruments for condensed matter and biological research would be added--one on a new thermal beam and other four on the neutron guides--and six existing instruments used for condensed matter research would be reconfigured to operate with bigger beams. Additionally, an existing neutron reflection spectrometer would be relocated on a neutron guide tube. The positron facility will be rebuilt to state-of-the art standards and minor changes and additions would also be made to operational equipment to support the expanded program.

<u>Instruments Employing Thermal Neutrons</u>. To increase the number of spectrometers operating with thermal neutrons (and improve performance of existing spectrometers) it is proposed that the present H-7 and H-8 single-beam plugs be replaced with double-beam plugs designed to bring out beams of the maximum practical size. Two new instruments would be installed on the satellite beams.

<u>Instruments Using Sub-thermal Neutrons from the HFBR's Liquid-Hydrogen Moderator</u>. For optimum performance of all spectrometers using sub-thermal neutrons it is proposed that the existing H9 beam plug and moderator assembly be replaced with one designed to provide three beams of the maximum practical size

and that new shielding be fabricated to accommodate the bigger beams. Additionally, three guides would be installed to conduct neutrons to four new spectrometers within the experimental hall but removed from the reactor shield face.

Instrument Upgrading. Most of the two and three-axis crystal spectrometers presently in use at the HFBR to investigate the structures and excitations of solids (both magnetic and non-magnetic) and liquids were designed and built in the mid-1960's when the reactor first became operational. They are in need of substantial modification to bring their performance up to current standards. New beam plugs will be installed to increase the vertical beam height (so that vertical focusing of the beams can be employed to fullest advantage) and the existing shielding will be replaced with new, thicker shielding to keep the room background at its present low level. There are six instruments which fall in this category.

A plan view of the physical layout of the instruments on the HFBR floor is shown in Figure II. All of the instruments will be accommodated in the existing reactor containment structure.

In spite of our current frustrations associated with the protracted shutdown, as the HFBR nears its 25th anniversary in October of this year, we are convinced that the HFBR not only has a distinguished history, but has an equally important role to play in the future of U.S. neutron scattering.

ACKNOWLEDGEMENTS

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Figure I.

OUTLINE OF HFBR PROPOSAL

BEAM PORT	NEW PLUG	GUIDE TUBES	SPECTROMETER	SHIELDING	SUBTOTAL ($1000)	TOTAL ($1000)
A. ENGINEERING, DESIGN, INSPECTION AND ADMINISTRATION						2162
B. SCIENTIFIC EQUIPMENT, FABRICATION, ASSEMBLY AND TEST						11573
POSITRON FACILITY	NO	NO	BUILD NEW SLOW POSITRON BEAMLINE	NO	396	
H5	YES	NO	REBUILD 3-AXIS SPECTROMETER	NEW	718	
H6 M	YES	NO	REBUILD DIFFRACTOMETER	NEW	721	
H6 S			NEW DIFFRACTOMETER WITH 2D DETECTOR	NEW	878	
H7 M	YES	NO	REBUILD 3-AXIS SPECTROMETER	NEW	874	
H7 S			FUTURE EXPANSION	NEW		
H8 M	YES	NO	REBUILD 3-AXIS SPECTROMETER	NEW	907	
H8 S			NEW BACKSCATTERING SPECTROMETER	NEW	802	
CNM			NEW MODERATOR AND PLUG ASSEMBLY		1211	
H9-A	YES	NO	REBUILD 3-AXIS SPECTROMETER (H9-A1)	NEW	425	
			RELOCATE REFLECTION SPECTROMETER (H9-A2)	NEW	94	
			NEW TIME-OF-FLIGHT SPECTROMETER (H9-A3)	NEW	1950	
H9-B	YES	YES	NEW MEDIUM RESOLUTION SANS (H9-B1)	NEW	609	
			NEW HIGH RESOLUTION SANS (H9-B2)	NEW	1634	
H9-C	YES	YES	NEUTRON INSTRUMENT R&D STATION	NEW	354	
C. GENERAL FACILITIES						1032
D. CONTINGENCY						2796
			GRAND TOTAL		($FY88)	17563
			GRAND TOTAL		($ESCALATED)	20782

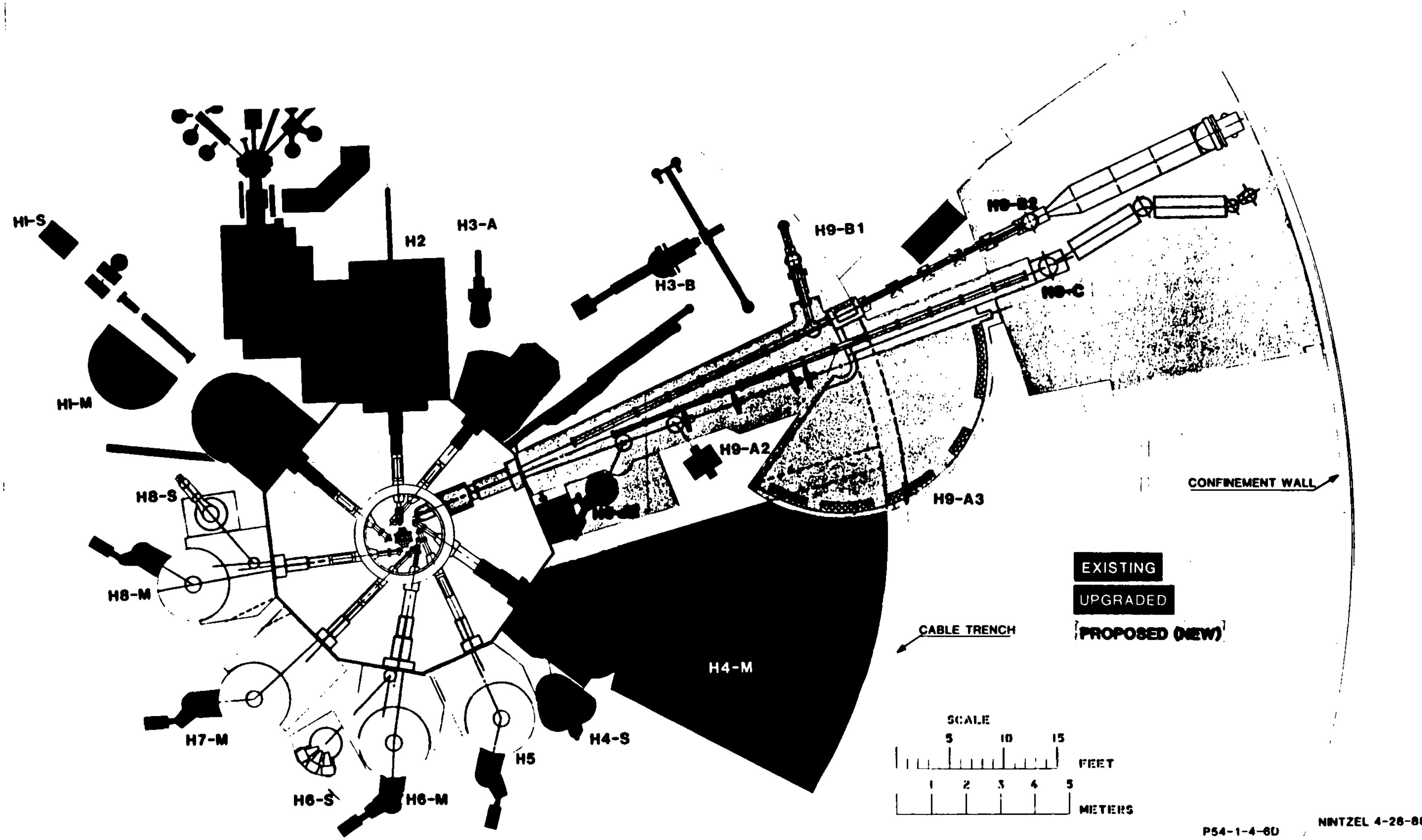

FIG. II

REACTOR FLOOR PLAN (PROPOSED)

BER-II UPGRADE

A. Axmann, H. Buchholz, C.-O. Fischer,
and H. Krohn

BER II Upgrade

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ABSTRACT

The paper gives a brief information on the physical improvements and the technical data of the Berlin Research Reactor BER II, which was upgraded from 1985 until 1990. The licensing procedure is described and the regulations applied.

General

The Hahn-Meitner-Institut is one of 13 national research centers in the Federal Republic of Germany. Basic research in close cooperation with the universities of Berlin is the main task of the institute. Thus, the Berlin Research Reactor BER II is an important facility in this program. The upgraded BER II will be a medium-flux neutron source (with a usable reflector flux of about 10^{14} n/cm^{2}sec) for standard applications, such as neutron scattering, material research and activation analysis. The BER II is a swimming pool reactor and was operated at a power of 5 MW from December 1973 until it was shut down in August 1985 for a general physical and technical improvement and upgrading.

The physical aim of the BER II upgrade is to increase the usable thermal neutron beam flux by a factor of 10 and the cold neutron flux by a factor of 130. This is the results of the following:

- The power of the reactor is increased from 5 to 10 MW
- The fission density is increased by a smaller core. The core grid has 7 x 6 = 42 MTR fuel element positions. The smallest

core is the so-called compact core, which uses only 4 rows of positions. The two rows of positions closest to the beam tubes R2 and R3 are filled with beryllium (see fig. 1).

- The reflection efficiency is increased by a beryllium reflector. The beryllium reflector, which is 30 cm thick, surrounds the core and produces a well pronounced flux peak about 5 cm from the core edge. To this flux maximum 10 beam tubes are adapted (a horizontal and vertical cut through the reactor is seen in fig. 1 and fig. 2). The flux gain of beam tubes ending in a beryllium reflector instead in a light water reflector is due to the well pronounced flux peak and due to the lower flux depression factor.
- The cold neutron flux density is increased by a cold source of hypercritical hydrogen at about 25 K. The cold source is installed in the conical beam tube (see fig. 1) and is feeding (via 7 neutron guides) a new neutron guide hall of approximately 1200 m^2.

The reactor and the cold source are ready for nuclear operation, but the Hahn-Meitner-Institut is still waiting on the operation license, which is expected in May 1990.

Materials of in-pool structures

The in-pool structures, which are in the neutron radiation field, are of the aluminum alloy $AlMg_3$ (core structures, beam tubes, primary cooling ciruit pipes, pool liner, inlet and outlet pipes of the purification circuits). Other in-pool parts are of stainless steel (inlet and outlet pipes of the hot water layer), the heat exchanger, the primary pumps, the pipes of the secondary cooling circuit).

Data

	Standard core	Compact core
- Reactor power [MW]	10	
- Thermal neutron flux [n/cm^{2}sec] (unperturbed in Be)	10^{14}	$1,5 \times 10^{14}$
- length of typical cycle [days]	60	33
- Fuel	Uranium-Aluminum	
- U-235 enrichment [%]	89 - 93	
- Type of fuel element	MTR (Material Test Reactor)	
- Weight of U-235 per plate [g]	7,83	
- Active length [cm]	60	
- Inner plate thickness [cm]	0,127	
- Outer plate thickness [cm]	0,15	
- Number of fuel elements (23 plates)	30	18
- Number of control elements and absorbers (17 fuel plates)	6	
- Type of absorber (In, Ag, Cd)	fork	
- Max. absorber speed [cm/sec]	0,05	
- Total absorber efficiency [%]	-18,2	-22,6
- Reflector (rectangular, 30 cm thick)	Beryllium	
- Coolant	H_2O	
- Total flow [m^3/h]	1050	
- Reflectorflow [%]	18	
- Number of primary pumps	3	

Experimental Facilities:

- Vertical facilities in core (1 MTR position each)	2
- Vertical facilities in Be	3
- Horizontal beam tubes	10
- Fast rabbit (beam tube T1)	1
- Cold Source (conical beam tube)	1
- Neutron guides	7

Accident Analysis

The maximum credible accident is a flow blockage in a fuel element (which may occur because of mechanical failures of fuel plates or because of particals in the cooling water). The plate melting is detected by the gamma dose rate due to fission production activity of the primary cooling water and the limits of the time derivative of the neutron flux density of the power instrumentation due to the void effect. The reactor will be shut down and the confinement of the reactor hall will be closed. The melting will never exceed one MTR fuel element and the resulting doses in the neighborhood of the HMI would be small against the limits set by the German regulations for accidents in power reactors [1].

The loss of primary coolant water (pool-water) is prevented by a double barrier system at all pool wall penetrations (double armatures in pipes or double walls in beam tubes for example). This and the following failures lead to a reactor shut down and are not connected with an activity release:

- The failure of one of the three primary coolant pumps (detected by pressure drop in the primary cooling circuit and by the change of the rotating speed).

- The failure of the secondary coolant systems (detected by an increase of the water temperature in the primary coolant system).
- Reactor power excursions due to a complete withdrawal of all control rods during the start-up procedure (detected by the period limit and the neutron flux density limit of the medium power instrumentation and the neutron flux density limit in the power instrumentation).
- The reactor power excursion due to a complete withdrawal of all control rods during power operation (detected by the limits of the absolute and differential neutron flux density of the power instrumentation).

Licensing procedure

In the Federal Republic of Germany, the regional state governments are responsible for the licensing procedures of nuclear reactors. Appropriate licensing authority in Berlin was at first the "Senator für Wirtschaft und Verkehr", later called the "Senator für Wirtschaft und Arbeit" (meaning "administration for economy and labor"). Since June 1989 the licensing authority has been shifted to the "Senatsverwaltung für Stadtentwicklung und Umweltschutz" (meaning "administration for city development and environmental protection"). In 1979, the HMI proposed at first the license of the BER II improvement connected with a power increase to 10 MW. Until 1982 the safety report of the upgraded research reactor BER II was submitted to an expert, the "Gesellschaft für Reaktorsicherheit" (Cologne). During this time, the safety report was rewritten twice to make it equal to the German regulations of safety reports of nuclear power plants [2]. The last two editions of the safety report were written by the Interatom GmbH, Bensberg, the later builder of the BER II upgrade.

In autumn 1982 the safety report was presented to the public and in October 1983 the public hearing was held within 4 days.

Due to public objections and a demand of the authority, the HMI was obliged to present a study on consequences of destructive interactions from outside the reactor.

The result of this study was that only events such as a crash of a fast flying military aircraft on the reactor has the potential to destroy the concrete wall of the reactor pool in a way that the core falls dry. The occurrence probability of such an event is 2×10^{-7} per year. The dose (integrated over 50 years, but with limited ingestion [3]) in the neighborhood of the HMI is about 50 rem. The calculations are in correspondence with the assessment of accident risks in German power plants [4].

Core melting accidents without the loss of the pool water but with loss of the integrity of the reactor building lead to a radiation dose which is a factor of more than 100 below this value. Sport planes and helicopters may have the potential for such a damage. The occurrence probability of these types of events is 2×10^{-7} per year as well.

Because of the low occurrence probability of these hypothetical accidents and the relatively small consequences of it, the licensing authority decided that the small research reactor BER II has not to be protected by an outer concrete shell against aircraft crashes. The construction permission for the BER II upgrade was issued in August 1985. The cold source was explicitely excluded of this licence, it was shifted into a consecutive procedure.

Against the construction license was sued at law. The legal procedures are still continuing. By preliminary decisions of the court the continuance of the construction was guaranteed. But the construction of the interface structures of the reactor to the cold source (the plug carrying the cold source and the conical beam tube in fig. 1) was suspended until the HMI had also presented the safety analysis of the cold source to the public and had held a public hearing about it. This was done between March and June 1987. In late autumn 1988 the licensing authority issued the construction permission for the cold source. Also against this license the plaintiffs sued at law. Meanwhile, the upgraded research reactor and the cold source are ready for the nuclear operation. Experts for the erection of the reactor and the cold source were the "Technischer Überwachungsverein Berlin e.V." (meaning Technical Surveillance Association Berlin) and the "Technischer Überwachungsverein Norddeutschland e.V., Hamburg". The experts ensured the quality control of materials and executions and ascertained the equality of the constructions with the rules of German nuclear safety regulations. The final experts report is disposed and we hope to get the operation license in May 1990.

Regulations

The German licensing authorities require backfitting according to current regulations for science and technology for technical approvement. A large number of guidelines and regulations concerning design, construction and operation of nuclear power plants have to be followed. The most important of these regulations are given below. Thus, planning permissions had to cover:

- The double energy supply system with double Diesel emergency power station and double D.C. supply for safety relevant systems [5, 6, 7, 8, 9, 10]
- Lifting equipments [11, 12]
- The fire protection [13]. An automatic sprinkler system is installed in the reactor building.
- Activity control [14]

- Instrumentation and reactor protection [15, 16, 17, 18, 19, 20, 21, 22]. The transdusers and actuators of the control system are of double redundancy, the measuring circuits of the safety system are of diversity (two different measuring methods of each variable) and of triple redundancy.
- Design against lightning effects [23]
- Communication devices [24]
- Radiological protection [25, 26]
- Requirements for the operating manual [27]
- Requirements for the testing manual [28]
- Quality assurance [29, 30, 31, 32]
- Work protection [33, 34]
- Requirements for the documentation [35, 36]
- The safeguard system [37]
- Disaster control [38, 39, 40, 41]
- Spent fuel disposal [42]

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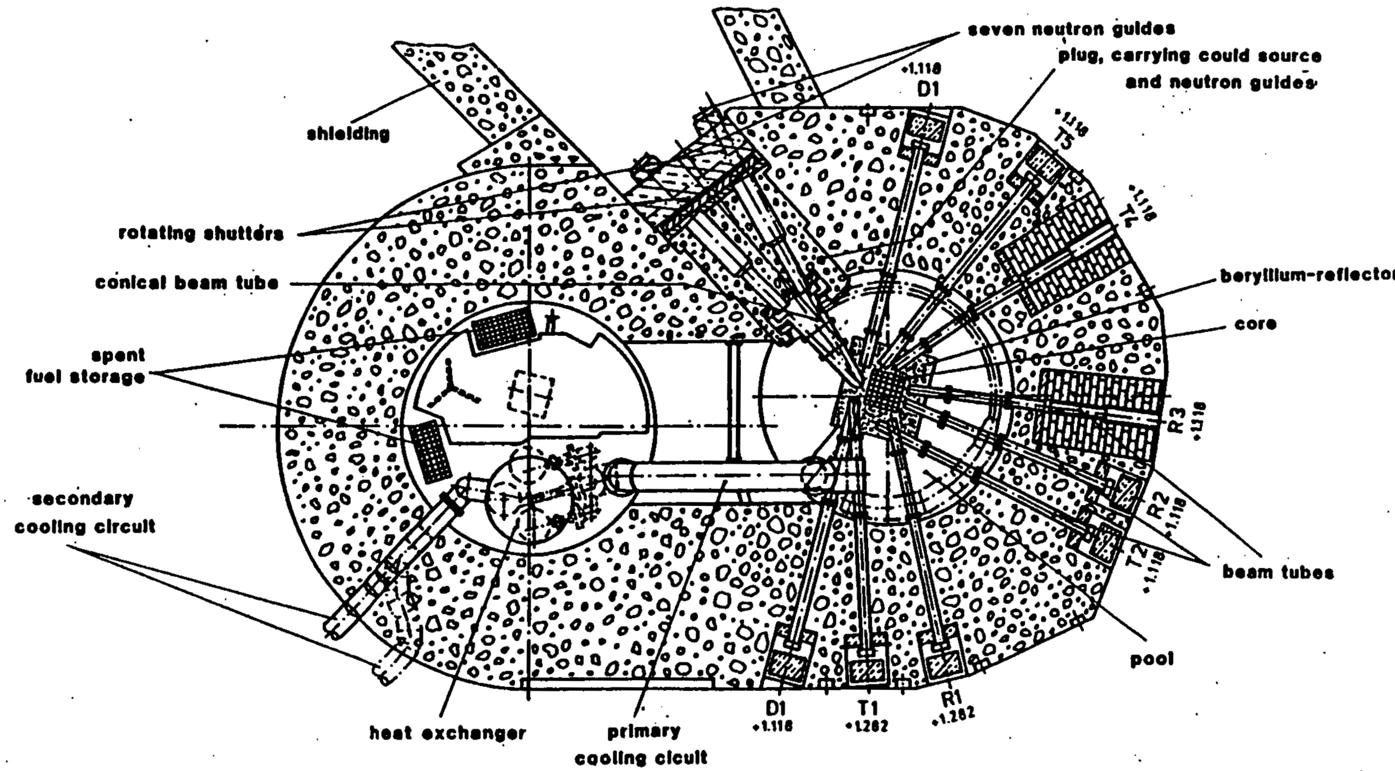

Fig.1 **BER II - horizontal cut**

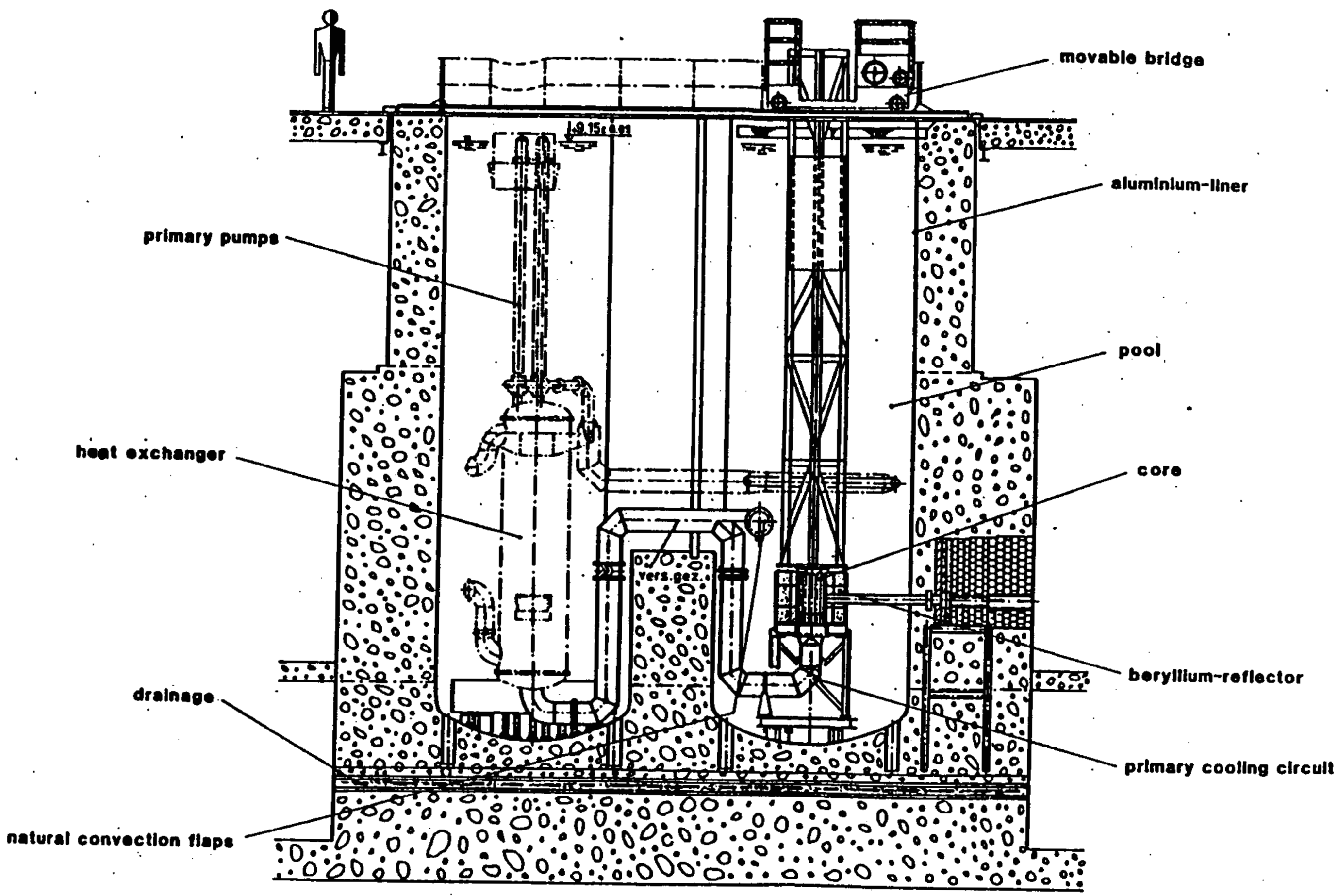

Fig.2 **BER II vertical- cut**

THE BR2 MATERIALS TESTING REACTOR PAST, ONGOING AND UNDER-STUDY UPGRADINGS

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The BR2 Materials Testing Reactor
Past, ongoing and under-study upgradings

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ABSTRACT

The BR2 reactor (Mol, Belgium) is a high-flux materials testing reactor. The fuel is 93% ^{235}U enriched uranium. The nominal power ranges from 60 to 100 MW. The main features of the design are the following :

- maximum neutron flux
  - thermal : 1.2×10^{15} n/cm^2 s
  - fast (E > 0.1 MeV) : 8.4×10^{14} n/cm^2 s;
- great flexibility of utilization : the core configuration and operation mode can be adapted to the experimental loading;
- neutron spectrum tailoring;
- availability of five 200 mm diameter channels besides the standard channels (84 mm diameter);
- access to the top and bottom covers of the reactor authorizing the irradiation of loops.

The reactor is used to study the behaviour of fuel elements and structural materials intended for future nuclear power stations of several types (fission and fusion). Irradiations are carried out in connection with performance tests up to very high burn-up or neutron fluence as well as for safety experiments, power cycling experiments, and generally speaking, tests under off-normal conditions. Irradiations for nuclear transmutation (production of high specific activity radio-isotopes and transplutonium elements), neutron-radiography, use of beam tubes for physics studies, and gamma irradiations are also carried out. The BR2 is used in support of Belgian programs, at the request of utilities, industry and universities and in the framework of international agreements.

The paper reviews the past and ongoing upgrading and enhancement of reactor capabilities as well as those under study or consideration, namely with regard to
- reactor equipment,
- fuel elements,
- irradiation facilities,
- reactor operation conditions,
- long-term strategy.

I. INTRODUCTION

The BR2 reactor at Mol (Belgium) went critical for the first time on June 29, 1961; it was put into service with an experimental loading in January 1963. On December 31, 1978 the reactor was shut down to replace the beryllium matrix. Routine operation of the reactor was resumed in July 1980.

Many upgradings and enhancements of reactor capabilities have been carried out; other ones are going on or are under study.

The BR2 reactor is part of a techno-scientific experimentation and production complex which comprises also general and peripheral support facilities, designed to optimize its utilization. A complete irradiation service can be provided, from design study to post-irradiation examination.

II. BRIEF DESCRIPTION AND SPECIAL FEATURES OF BR2

The BR2 reactor is a high-flux materials testing reactor of the thermal heterogeneous type [1]. The fuel is 93% ^{235}U enriched uranium sandwiched between aluminium plates. The moderator consists of beryllium and light water, the water being pressurized (1.25 MPa) and acting also as coolant. The pressure vessel is of aluminium, and is placed in a pool of demineralized water.

The BR2 has the following main features :

- The experimental channels are skew, the bundle presenting the form of a hyperboloid of revolution (Fig. 1). This original configuration gives easy access to the core, allowing the loading of complex instrumented devices, and it results in a compact core, source of very high neutron fluxes.

- The access to the top and bottom covers of the reactor authorizes irradiation of devices measuring up to 11 meters long, some of them being able to contain fuel rods up to 4 meters.

- Besides the standard channels (84 mm diameter), five large diameter (200 mm) irradiation channels are available, with the possibility of loading large experimental irradiation devices such as sodium, gas or water loops.

- Although BR2 is a thermal reactor, it is possible to achieve neutron spectra very similar to those obtained in other reactor types, e.g. fast reactors and fusion reactors (neutron spectrum tailoring).

- A remarkable flexibility of utilization : the core configuration and the operation mode of the reactor are adapted to the experimental requirements. Fig. 2 shows a typical core configuration.

III. OPERATION CHARACTERISTICS

The main operational characteristics are summarized in Table 1.

The reactor operation is carried out on the basis of an operating cycle. The present nominal cycle length is 4 weeks and consists of 2 weeks shut-down for loading and unloading and normal maintenance work, followed by 14 days of operation. Each year, two shut-down periods are extended for survey tests and special maintenance work. In addition, special irradiation campaigns are organized in order to carry out particular experiments such as safety tests. The total number of days of operation per year is presently 180.

The present maximum nominal heat flux at the surface of the reactor fuel elements is 470 W/cm², 500 W/cm² having been reached during special campaigns (programme MOL 7C) and 600 W/cm² being the maximum admissible heat flux (probable onset of nucleate boiling). The 470 W/cm² heat flux was tested under the circumstances of pressure loss incidents, 600 W/cm² having been tested for the nominal cooling flow rate.

The nominal full-power level depends on the core configuration used; at present with the configurations 10 or 12, it ranges from 60 to 80 MW, the maximum reached being 106 MW. The ultimate cooling capacity, initially foreseen for 50 MW, has been increased in 1971 to 125 MW.

IV. NUCLEAR CHARACTERISTICS

For a BR2 core operating at a hot spot heat flux of 470 W/cm^2, the maximum available neutron fluxes are :

- *in the axis of the central channel H1 with a Be plug*
  - thermal : 1 x 10^{15} n/cm^2 s

- *in a fuel element channel*
  - total : 1.4 x 10^{15} n/cm^2 s
  - fast (E > 0.1 MeV) : 7 x 10^{14} n/cm^2 s
  - fast (E > 1.0 MeV) : 3.5 x 10^{14} n/cm^2 s.

Typical neutron spectra in a reflector position and in a fuel element position are shown in Fig. 3.

It is possible to irradiate in BR2 fissile and structural materials intended for reactors of several types (fission and fusion) in such a way that irradiation effects correspond to those expected in these reactors.

V. REACTOR UTILIZATION

V.1. Facilities Available

At BR2, about 100 irradiation positions are available. It is possible to irradiate :

- *in the pressure vessel*
  - core
    - within the standard fuel elements (diameter of the experimental cavity : 17.4 mm to 51.6 mm)
    - in a driver fuel element or in a special plug (in the large 200 mm diameter channels)
  - reflector
    - in beryllium or aluminium plugs (diameter of the experimental cavity : up to 200 mm)
    - in the hydraulic rabbit
    - in the self-service thimbles.

- *outside the pressure vessel*
  - in the beam-tubes (radial or tangential)
  - in the reactor pool.

Fig. 4 shows examples of irradiation devices loaded in a beryllium plug or a standard fuel element containing 6 plates; by reducing the number of concentric fuel plates, it is possible to increase the useful diameter of the experimental cavity. Fig. 5 gives an example of a loop surrounded by a driver fuel element and loaded in a 200 mm diameter channel.

In addition to the irradiation itself, the BR2 Operating Group can provide a complete high flux irradiation service from the planning stage up to the interpretation of the final results :

- assistance in the design of experimental devices

- determination of the neutronic characteristics of the irradiation by means of 1-D and 2-D neutron transport or diffusion calculation codes, gamma heating calculations being also performed when required

- design and fabrication of irradiation equipment :
  - high performance loops,
  - instrumentation capsules for fissile and non-fissile materials irradiations at high temperatures and high power ratings,
  - retractable and reloadable devices,
  - test on pre-irradiated fuel pins,
  - power cycling devices,
  - capsules for the production of isotopes and transplutonium elements.

- testing and commissioning of irradiation equipment

- dosimetric analysis :
  - determination of optimum irradiation conditions with experimental mock-ups in the BR02 reactor, the zero-power nuclear model of BR2, or in BR2 operating at low power,
  - thermal and fast neutron detector measurements.

- post-irradiation examination and analysis :
  - dismantling of equipment
  - metallurgical and physical tests in hot cells,
  - chemical operations and analysis.

V.2. <u>Irradiations Carried out</u>

Purpose of the irradiations [1, 2] :

- *research activities*
  - study of the behaviour of fuel elements and structural materials intended for the reactors of future nuclear power stations (sodium or gas cooled fast reactors, high temperature gas cooled reactors, light water reactors, fusion reactors)
  - basic physical research within the beam-tubes (nuclear physics and solid state physics)

- *in-pile safety experiments (particularly related to fuel pin cooling and transient overpower)*

- *production activities*
  - production of high specific activity radioisotopes
  - silicon doping
  - colouration of gems (diamond, topaz)

- *peripheral activities*
  - neutron-radiography in the reactor pool
  - gamma irradiations within spent fuel elements (5×10^7 rad/h or 140 W/kg).

VI. PARTICULAR UPGRADINGS AND ENHANCEMENTS OF REACTOR CAPABILITIES

VI.1. Reactor Equipment

VI.1.1. Major Equipment Replacement

The unloading and replacement of the first beryllium matrix of the BR2 reactor took place in 1979-1980.

The main steps of the replacement operation are described in reference [3].

At unloading time the maximum fast fluence in the hottest channel had reached about 8 x 10^{22} n/cm^2 (E > 1 MeV). Dimensional stability and swelling of the beryllium matrix have been investigated. The swelling, mainly due to the formation of gas atoms, was found a nearly linear function of the fast fluence up to a value of ≈ 6.4 x 10^{22} n/cm^2 (E > 1 MeV) at the temperature of ≈ 50°C normally existing in the matrix. For higher values of the fast fluence, several observations showed an accelerated increase of the swelling. Consequently the maximum allowed fast fluence for the second beryllium matrix has been limited to 6.4 x 10^{22} n/cm^2 (E > 1 MeV). A surveillance programme of the second BR2 beryllium matrix has been set up; it mainly concerns direct observations and measurements on the beryllium matrix itself. Dimensional measurements allow comparison of the relative swelling in axial and radial directions with the dilatation coefficients obtained for the first matrix. Visual inspections are performed on the inner surfaces of the reactor channels in order to record the beginning and the evolution of cracks and to measure the total length of cracks in function of the fast fluence. Irradiations and measurements are also performed on test samples coming from the heats which served for the manufacturing of the second matrix.

In 1971, the nine original heat exchangers of the reactor primary cooling circuit, of the classical straight tube pattern, were replaced by three units, of a helical tube pattern.

In January 1972, it was found that one tube of the new exchangers was perforated, by vibration of the unsupported straight lead-in section against a weld. After removal of the faulty tube, extensive and systematic vibration measurements were carried out, and it was found that, in two of the heat exchangers, several tubes experienced excessive vibration. As an initial corrective measure, the manufacturer installed aluminium strips around the lead-in tube bundles, to reduce the vibration of the enter tubes. Subsequently, a woven stainless steel mat, 5 cm thick, was wrapped around the bundles to break up the impinging jet of primary water which enters at right angles to the tube bundle. These modifications were carried out on all three exchangers.

Also in 1971, the original wooden cooling tower packing was replaced by plastic material. This modification of the towers and the replacement of the heat exchangers have led to a nominal cooling capacity of the system exceeding 120 MW.

VI.1.2. Nuclear Instrumentation

From the beginning of the reactor life, a great effort has been devoted to the maintenance and the improvement of
- the reactor control equipment
- the control commands and mechanisms of the safety and regulating rods
- the instrumentation of the experiments and irradiation devices
- the radiation monitoring system.

Many mechanical improvements have been made in order to increase the reliability : balling screw for the regulating rod, improved scram mechanism, position sensors,...
The first generation nuclear instrumentation of BR2 was installed in 1960. All the electronics was then driven by tubes. Therefore, the reactor control electronics has been nearly completely renewed. About 80 racks completely transistorized are now controlling the power of the reactor. More chambers have been installed around the reactor with a view of redundancy.

The radiation monitoring system comprises more than 80 ionization chambers, many GM, NaI crystals, GeLi and spectrometers. It also includes radiation monitoring of the experiments.

Three new racks improving the performance and the safety of the reactor control (neutron measurement by linear chambers) are now ready to be installed. The replacement of the very last racks still working with tubes is under study. So the reactor control will be completely transistorized by the end of 1991.

A new phase will begin with the digitalizing of many signals. During 1990 the position of some chambers controlling the reactor will be fully digitalized using absolute position encoders. The next step will be the digitalizing of the position of the control rods.

Another project concerns the monitoring of fission products in primary water by on line spectroscopy.
During 1990, fission counters and completely renewed ionization chambers will be installed around the reactor in order to increase sensitivity resulting in an enforced safety of the reactor.

The safety of the reactor and the reliability of its instrumentation have been considerably improved during the last 30 years. In 1989, only one unscheduled shut-down (scram) occurred due to an electronic failure. The goal is to keep all the nuclear instrumentation as up-to-date as possible and to keep it ready for the future.

VI.1.3. Maintenance and upgrading

A great effort is devoted every year to the maintenance and upgrading of the installations in order to maintain an optimal availability and safety during the working of the reactor.
The accent is now put on the regular maintenance, automatization, renovation, modernization of the equipments during the scheduled reactor shut--downs.

The main items which were recently covered, or are still in progress, concern :

- the complete overhaul of the three emergency power generators and their associated equipments;

- the complete overhaul of all the main electric motors of the primary and secondary cooling circuits; also the complete overhaul of the main electric motors of all the reactor auxiliary circuits;

- the renewal of the high-voltage switching equipments for the main secondary pumps;

- the renewal of tens of low voltage electric breakers for auxiliary equipments;

- the replacement of the main ventilation pipings between the process and ventilation buildings;

- the complete refurbishment of the cooling towers of the main secondary cooling circuit;

- the upgrading and duplication of the control commands for the reactor building ventilation valves.

The following items are now planned or under study for the near future (1990-1991) :

- the inspection of the main electric transformers;

- the replacement of the heat exchangers of the reactor and storage pools;

- the internal examination of the main primary heat exchangers;

- the extension of the spent fuel storage capacity to cover the reactor working period until end of 1995;

- the erection of a new shipping and decontamination area for transport containers.

VI.2. <u>Fuel Element</u>

Since 1971, the majority of fresh fuel elements loaded have been of the Cermet type. The standard six tube elements used contain either 330 g ^{235}U (50 mg/cm^2) together with 2.8 g natural boron (in the form of B_4C) and 1.4 g natural samarium (in the form of Sm_2O_3) as burnable poisons, or 400 g ^{235}U (60 mg/cm^2) and 3.8 g boron and 1.4 g samarium, compared with 244 g ^{235}U for the alloy elements previously used, without poison.

The generalized use of these Cermet elements has led to an increase in the reactor cycle length, a reduction in the variation in irradiation conditions during the cycle caused by the control rod movement, the possibility of overcoming increasing reactivity absorption effects of the rigs and a reduction in fuel costs by decreasing the number of fresh elements used each cycle and by increasing the mean burn-up (Fig. 6).

The fuel is presently 93% ^{235}U enriched uranium. The uranium density in the meat of the 400 g ^{235}U standard six tube elements is 1.3 g/cm^3 corresponding to an uranium concentration of 37 wt %.

Since 1978, C.E.N./S.C.K. follows carefully the RERTR activities (Reduced Enrichment Research and Test Reactor program) and has offered its collaboration. But, under the present circumstances (the BR2 experimental program, state-of-the-art LEU fuel technology), BR2 has to continue to use highly enriched uranium (90-93% ^{235}U). [4]

Increase of the uranium density from the present value of 1.3 to 1.7 g/cm^3, allowing a load of 500 g ^{235}U per standard fuel element, is under consideration to improve the fuel cycle economics.

VI.3. Irradiation Facilities

In the challenging field of irradiation testing and neutron-based production activities, thorough experience has been gained from the start of the BR2 reactor [1]. Particular features of this are the following :

- a wide range of irradiation devices for LWRs, LMFBRs, HTRs, GCFRs and fusion reactors and for radioisotope and transplutonium elements production;

- long established use of BR2 for safety experiments (Mol 7C, PAHR) and power transients (VIC loop, TRIBULATION program);

- irradiation campaigns on pre-irradiated fuel pins (without length limits) or on fuel pins with fuel cladding defects;

- remote assembly of bundles of irradiated fuel pins and remote sodium filling of in-pile sections;

- handling of circuits contaminated by fission products or tritium.

Recent developments :

- in-pile fatigue tests on fusion reactor first wall candidate materials, FAFUMA 3 experiment (Fig. 7);

- the PRF device for the irradiation of highly enriched uranium targets, with loading and unloading capability during reactor operation (Fig. 8);

- modification of the beryllium plug loaded into the central channel H1 in order to increase the number of irradiation positions (from 3 to 6) with very high thermal neutron fluxes and to allow the loading of a BR2 standard fuel element in the centre (Fig. 9).

Under development :

- the high pressure water loop CALLISTO (Capability of Light water fuel testing In Steady state and Transient Operation) for the irradiation of 3 x 9 fuel rods with a broad range of experimental conditions representative of PWR reactors (Fig. 10);

- the SIDONIE device for silicon doping, ingot diameter up to 5" (Fig. 11).

Under consideration :

- the SOLISTE loop (Sodium Loop for Incidental and Safety Transients Experiments) for the epithermal irradiation of high burn-up fuel pins pre-irradiated in fast reactors such as PHENIX with simulation of different kinds of transient incidents such as a control rod withdrawal incident (CRW) (Fig. 12);

- PAHR (Post Accident Heat Removal) type experiments in support of LWR safety analysis studies;

- to provide, on the occasion of the next beryllium matrix replacement, an experimental cavity with a diameter up to 400 mm in the central region of the reactor (Fig. 13).

VI.4. Reactor Operation Conditions

From January 1963 till December 1989, 78 different core configurations have been used. This large number of variants utilized is due to continued attempts to use the flexibility of loading to the best advantage of the greatly varying experimental charge, while still maintaining the maximum operating cycle length and keeping fuel consumption to the minimum.

The nominal full power level was regularly adapted to the experimental loading demands : 34 MW in 1963, 60 to 80 MW at present.

The highest power ever reached by BR2 (and by a research reactor in Europe) is 106 MW; this power level was attained during a special campaign in order to meet the irradiation conditions required for a particular experiment of the MOL 7C safety program (study of local blockage of the sodium flow in a fuel bundle of 30 LMFBR fuel pins).

At each routine start-up, it is customary to operate for short periods at 1%, 40% and 80% of nominal full power, in order to carry out various checks, mainly on the irradiation conditions for the experiments, to determine the optimum power level and to find the best control rod alignment, having regard to flux perturbations and imbalance. In addition, during each cycle, the reactor power is modified, if necessary, to give the optimum irradiation conditions for the experimental load as a whole, while still respecting safety limits.

Under study or consideration :

- the loading of a standard 6 tube element in the central position of the new beryllium plug loaded in the H1 channel in order to compensate for the loss of reactivity caused by the radioisotope production baskets loaded in the other positions and to cope with the ^3He poisoning of the beryllium matrix during the shut-down periods;

- the increase of the maximum hot spot specific power from the present 470 W/cm^2 level to 550 or 600 W/cm^2 if requested by particular safety experiments on high burn-up fuel pins.

VII. LONG-TERM STRATEGY

A major milestone in the presumable future of the BR2 reactor will be somewhere around 1995, when the present beryllium matrix has reached its maximum allowed fast fluence.
Beyond this point, one can think of different scenarios ranging from a simple replacement of the matrix up to major modifications of the reactor.
The event-tree given in Fig. 14 can give some guidance to the decision-making process.
It clearly appears that decisions will have to be made quite in advance, if operations are to be conducted successfully. The choice of a scenario to be followed will be made at latest by the end of 1991.

VIII. CONCLUSION

The BR2 reactor was first put into service with an experimental loading in 1963. Since then it has contributed greatly to the development of many large nuclear and non-nuclear projects within the European Community and other countries such as the U.S.A. and Japan. This successful result has been obtained thanks to the continuous process set up for the upgrading and enhancement of the reactor capabilities.

As far as the future is concerned, efforts will be continued in order to maintain an optimal availability and safe operation of the reactor and to adapt the irradiation capacity and the operating conditions of BR2 so as to accommodate the future expected experimental loading. In addition, consideration is given to the long term strategy after 1995, time at which the present beryllium matrix has to be replaced.

TABLE 1 : BR2 MAIN DATA

Beginning of utilization	*1963*
Maximum heat flux • *nominal* • *admissible*	 *470 W/cm²* *600 W/cm²*
Nominal power	*60 to 100 MW*
Maximum neutron flux (for 600 W/cm²) • *thermal* • *fast (E > 0.1 MeV)*	 1.2×10^{15} *n/cm² s* 8.4×10^{14} *n/cm² s*
Irradiation positions	*up to 100*
Fuel enrichment	*90 - 93 %* ^{235}U
Fissile charge at start of cycle	*9 to 13 kg* ^{235}U
Cycle of 4 weeks *presently*	 *2 weeks shut-down* *14 days operation*
Total operation days per year • *presently* • *possible*	 *180* *200 to 250*

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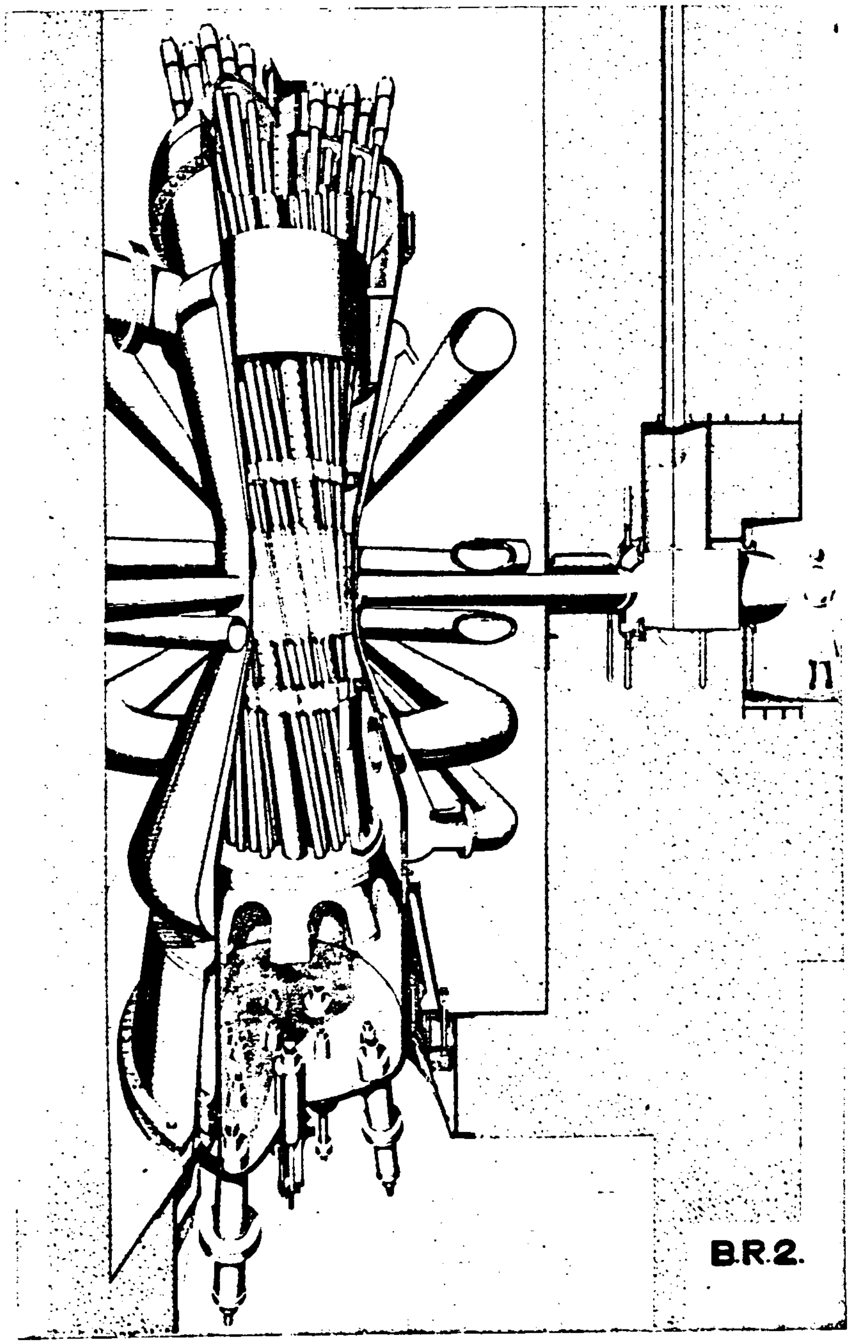

Fig.1 General view of the BR2 reactor.

CONFIGURATIE 10N

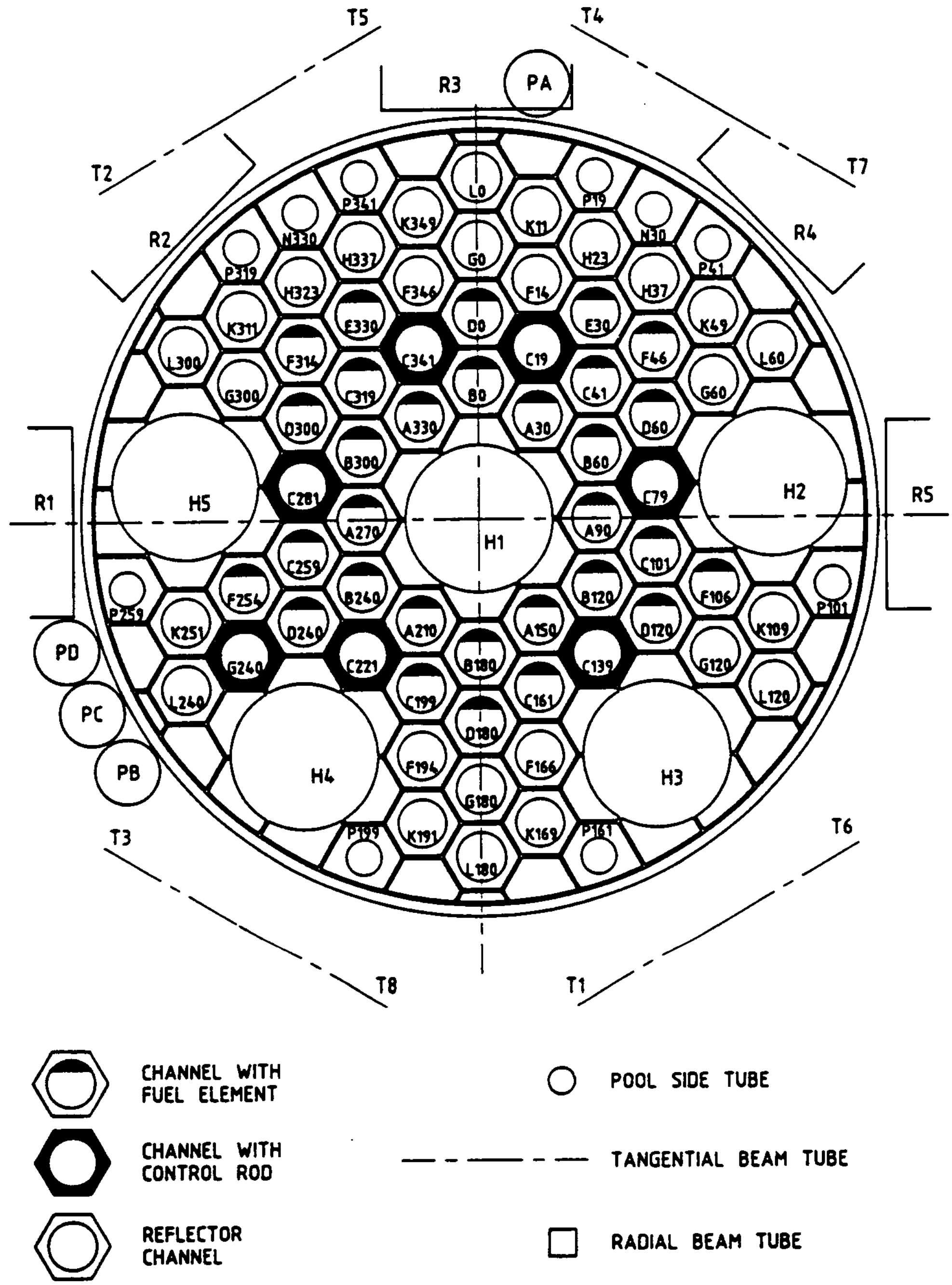

Fig.2 BR2 CORE
Cross section at mid plane
TYPICAL CONFIGURATION

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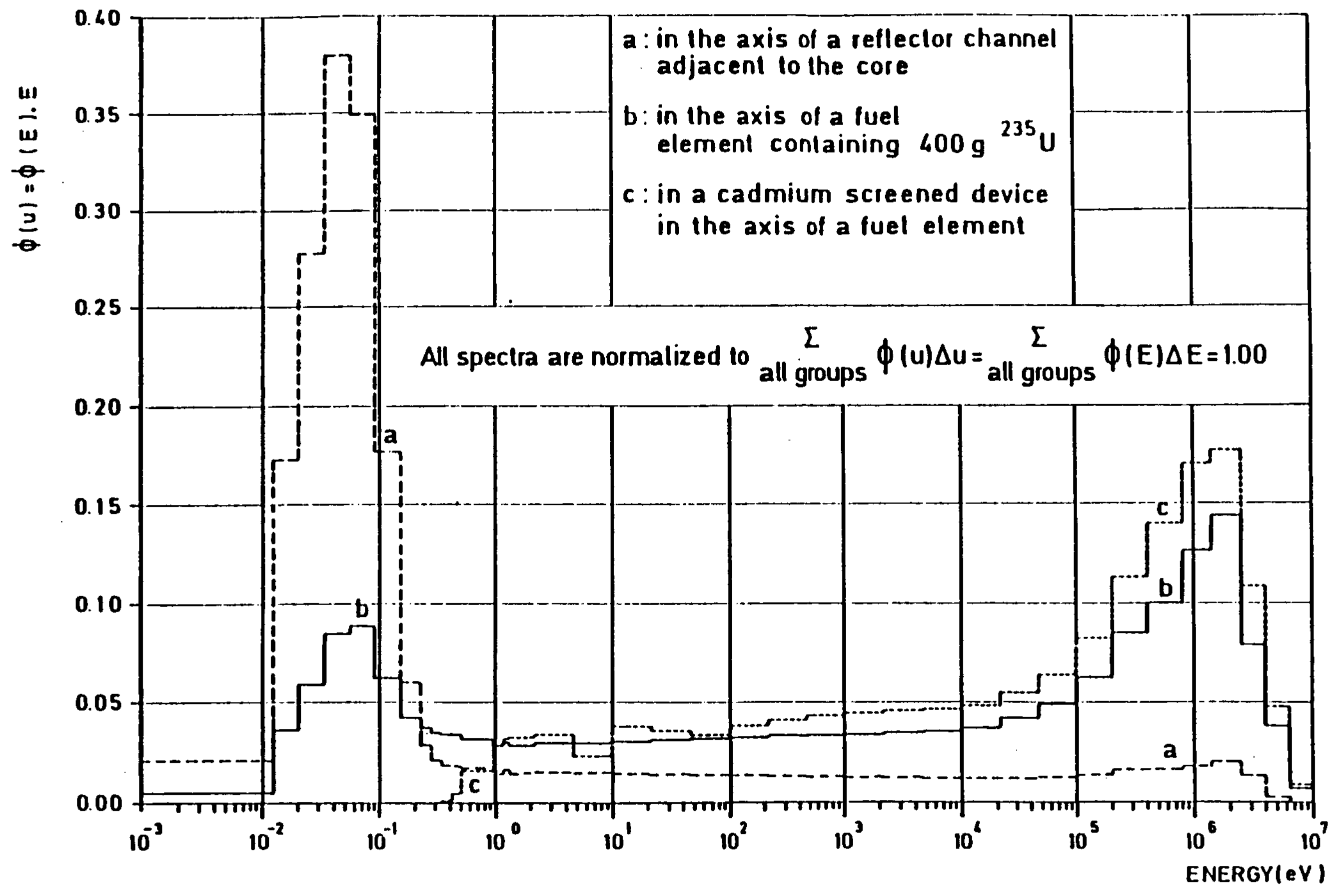
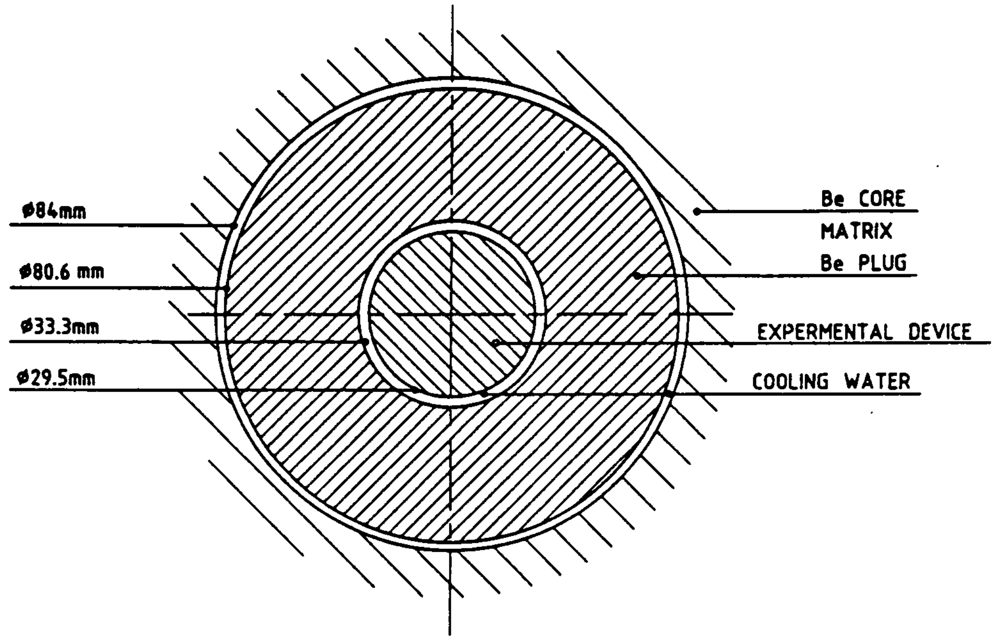
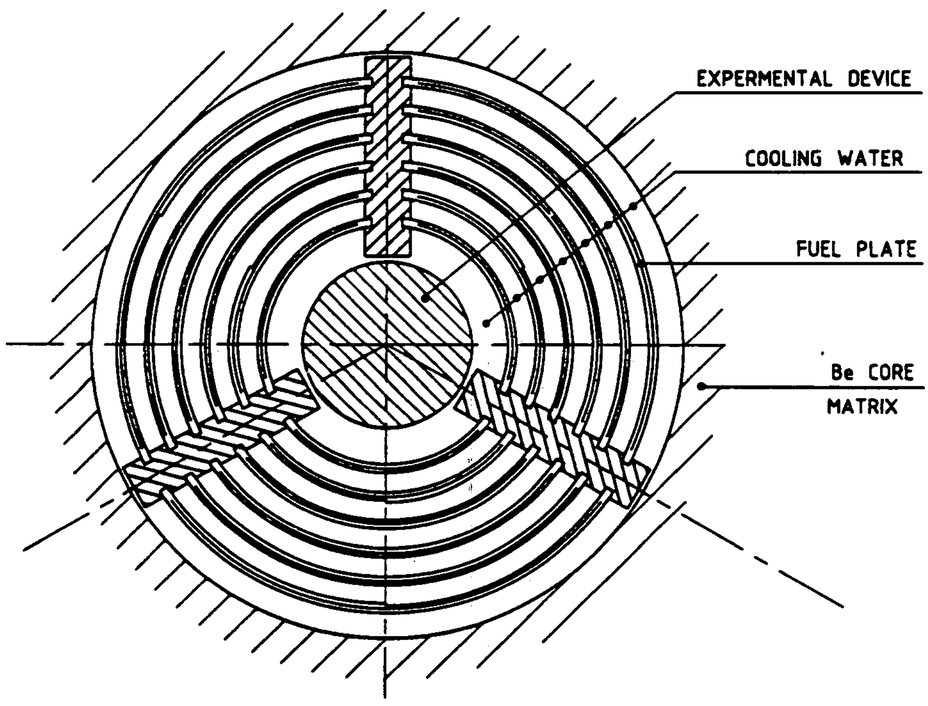


Fig.3. TYPICAL NEUTRON SPECTRA IN BR2 SCK-CEN F 602



With a Be FILLING PLUG (Cross section)



With a FUEL ELEMENT (Cross section)

Fig.4 BR2 Irradiation in a standard channel

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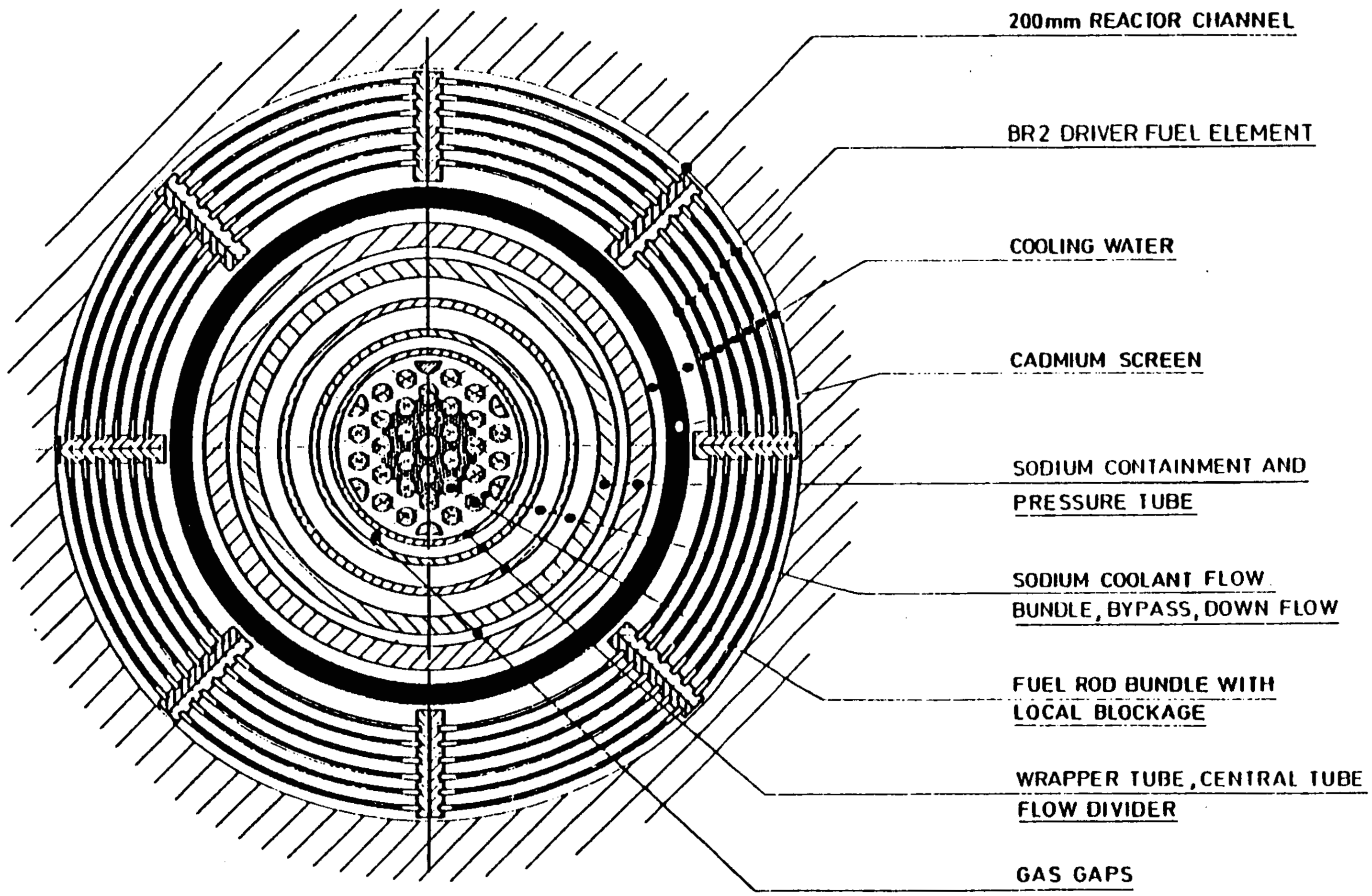


Fig.5. MOL 7C SODIUM LOOP

CROSS-SECTION AT REACTOR MID-PLANE

SCK/CEN TECHN. F. 901.

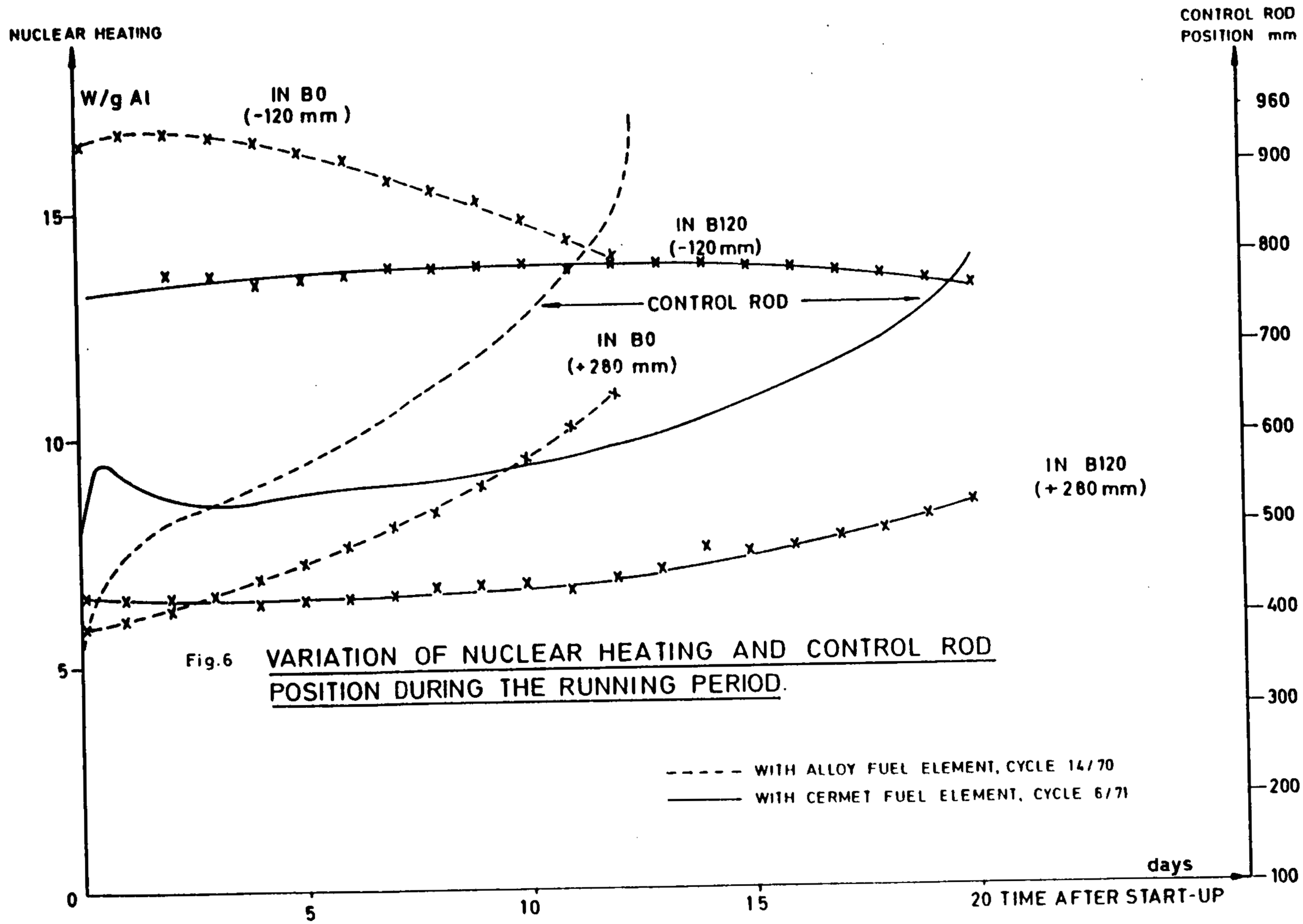
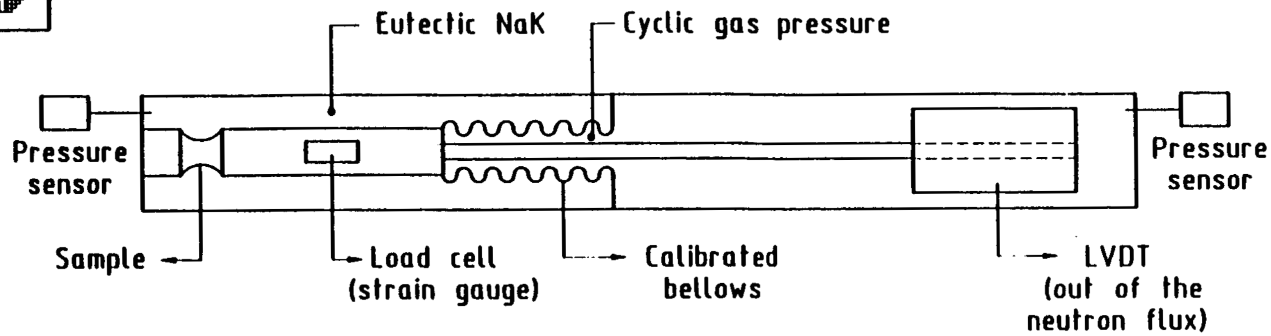


Fig.6 VARIATION OF NUCLEAR HEATING AND CONTROL ROD POSITION DURING THE RUNNING PERIOD.



MEASUREMENTS :

Load : -Pressure Δ + calibrated bellows
-Load cell

Deformation : -LVDT (Linear Variable Differential Transformer: displacement sensor)

CALIBRATIONS :

1. Bellows : before irradiation (in the actual mounted situation)
2. Pressure sensors : can be calibrated every time it is needed by connexion between static-cyclic pressure circuits
3. LVDT : during irrad. by the known clearance (Zeroing system)

Fig.7. FAFUMA-3 : MODULE FOR IN-PILE FATIGUE TESTS.
SCHEMATIC VIEW

F1370

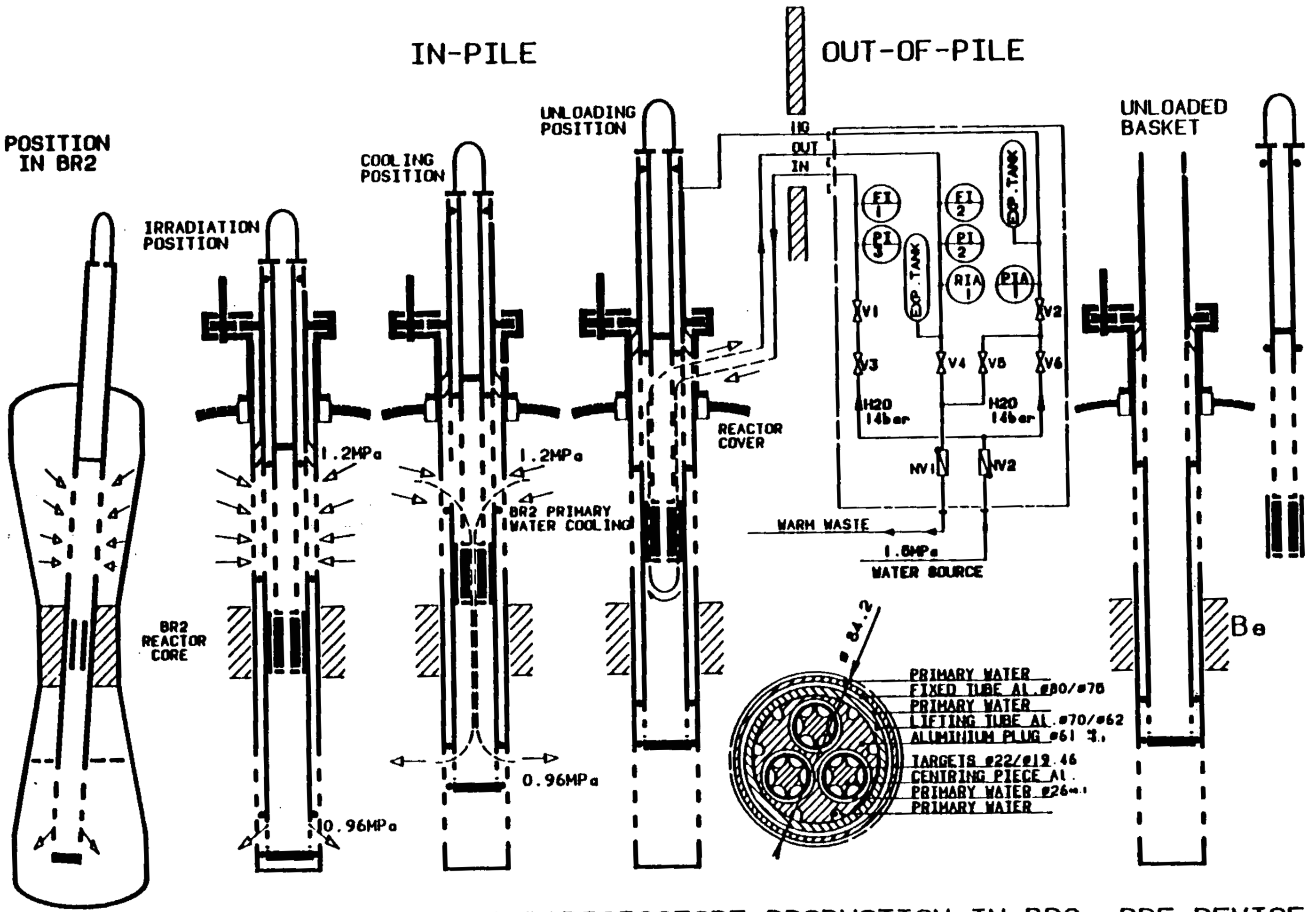


Fig.8. Xe-133 Mo-99 I-131 RADIOISOTOPE PRODUCTION IN BR2 PRF DEVICE

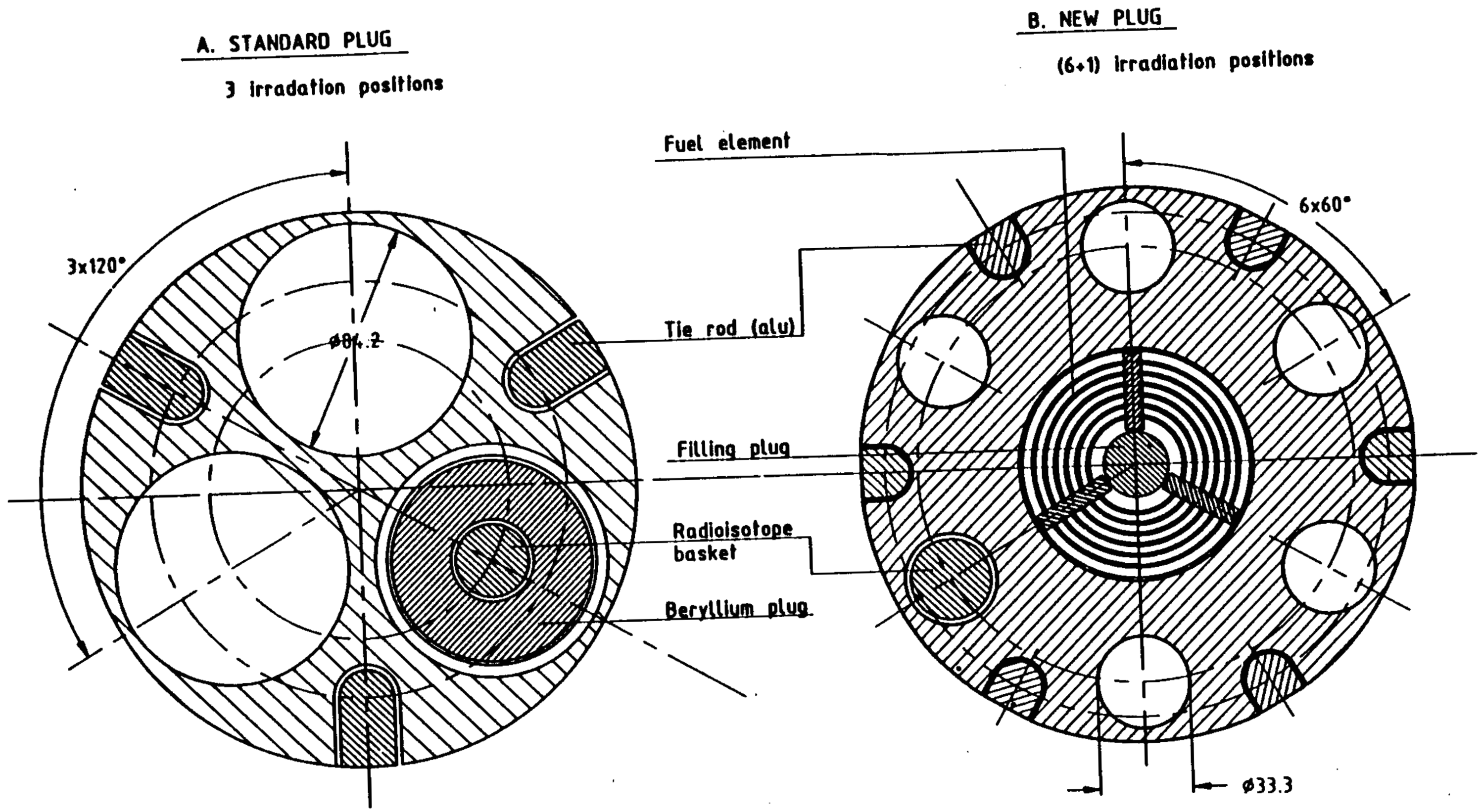


Fig.9 Central channel H1 with a 200mm beryllium plug (cross section)

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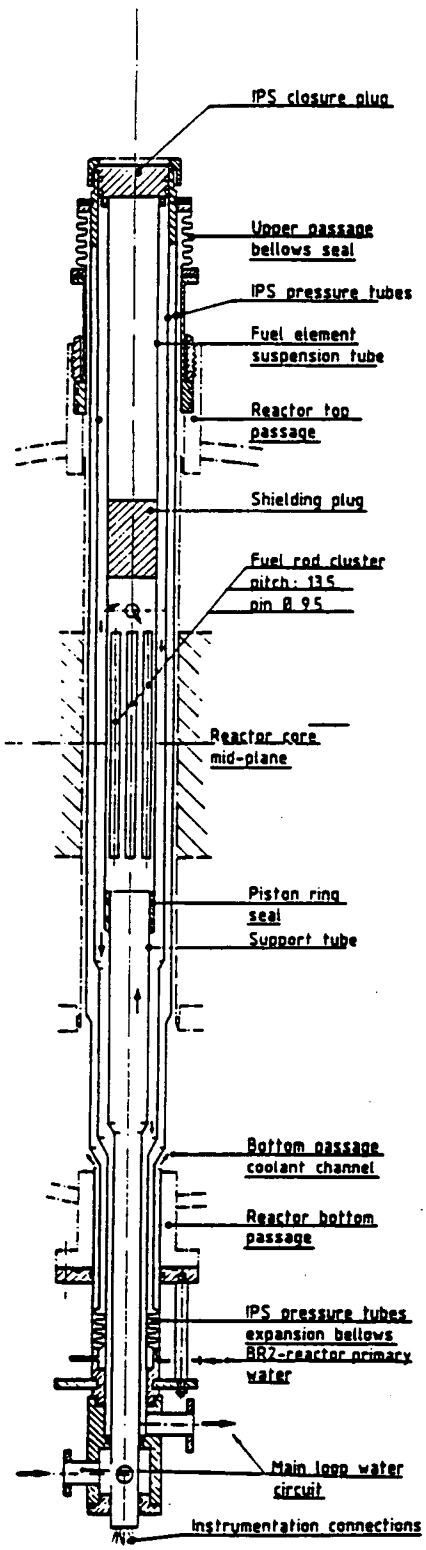
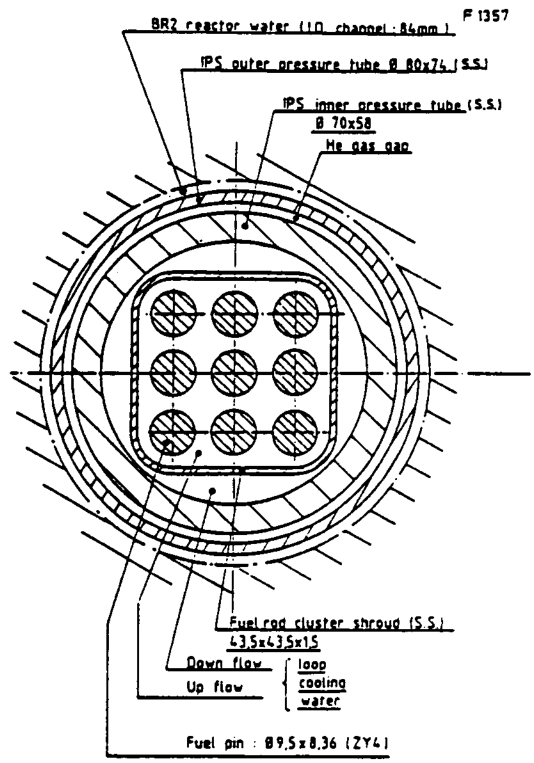




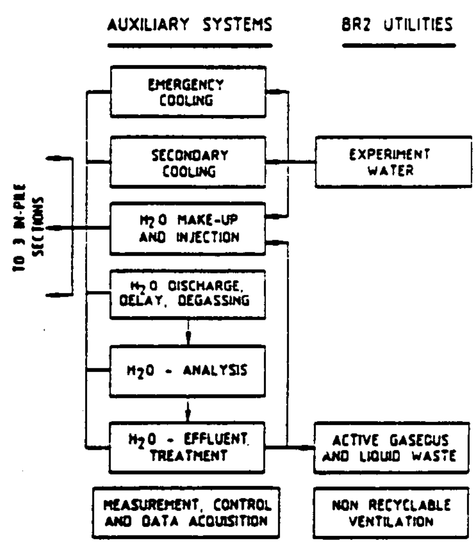


Fig 10. CALLISTO WATERLOOP
Schematic view one in-pile section in Ø 84mm BR2 reactor channel

SIDONIE

POSSIBLE LOADING SCHEMES OF THE BASKET

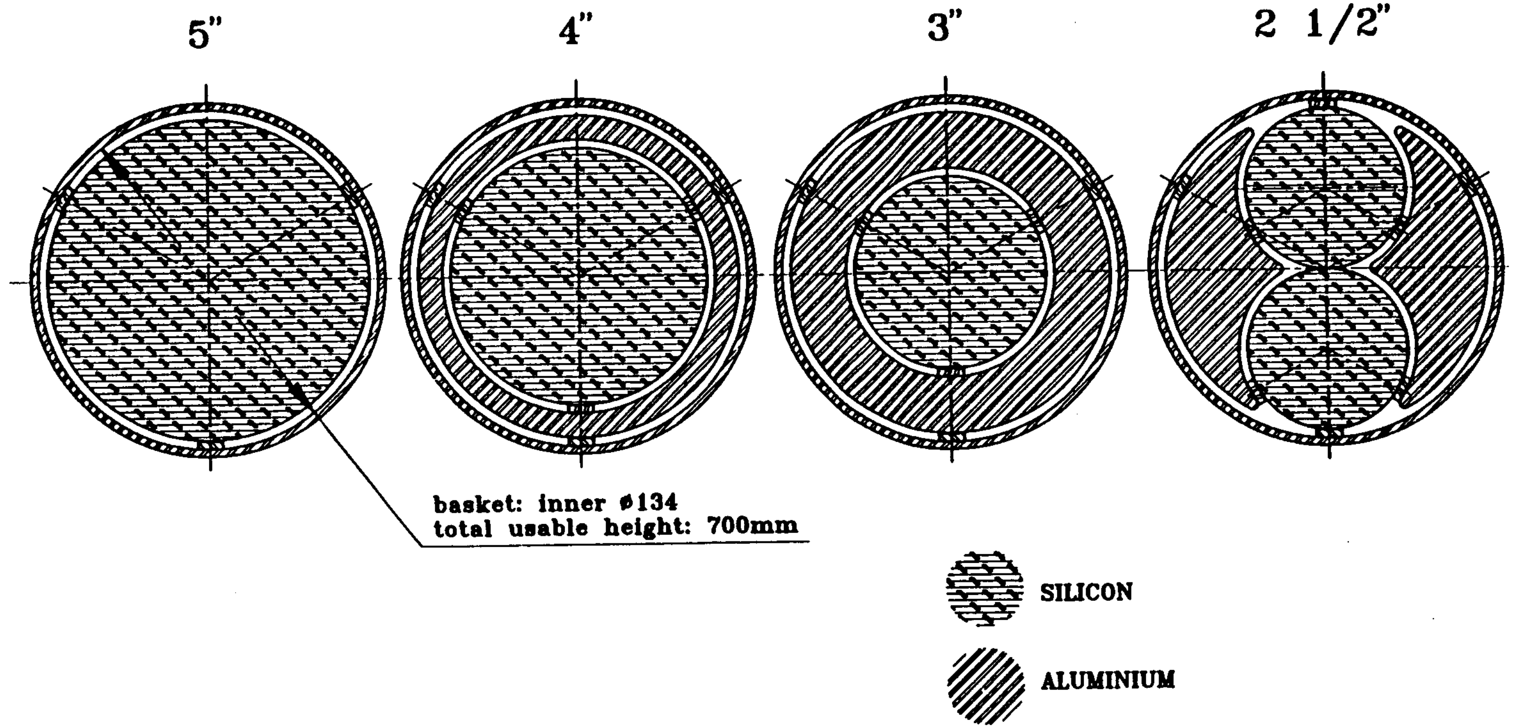


Fig.11 Silicon doping

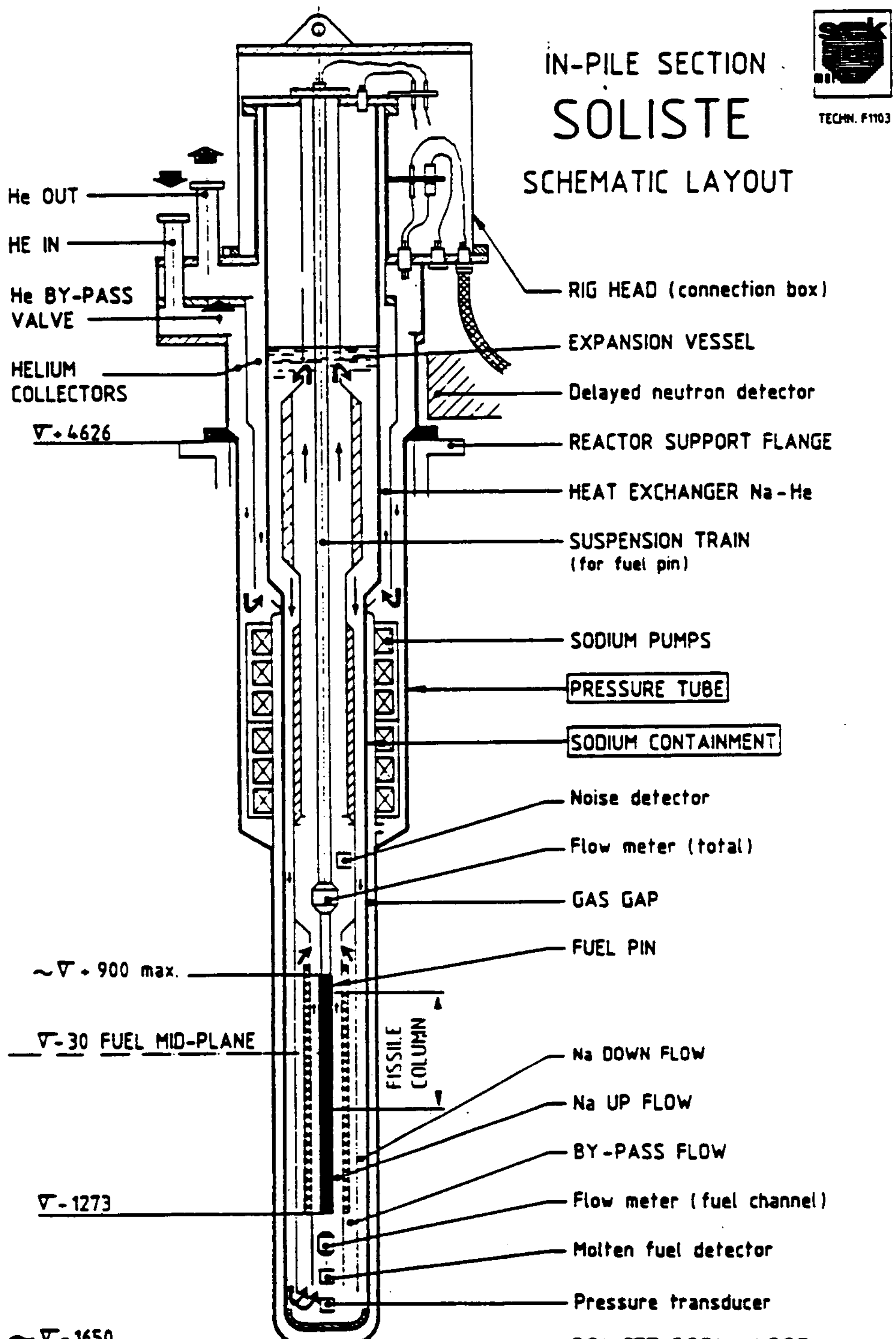
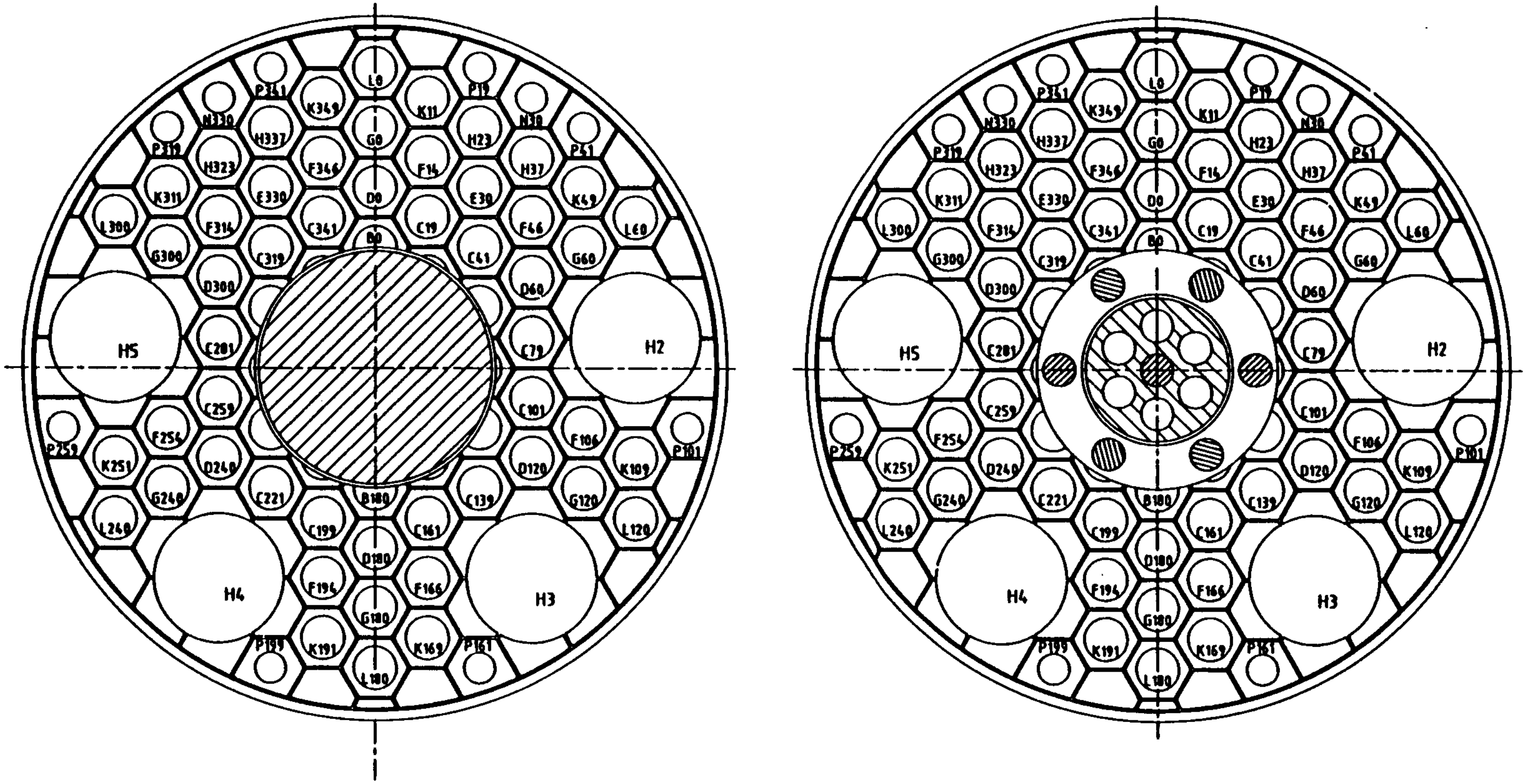


Fig.12. SOLISTE SODIUMLOOP



A. loading: large experimental device

B. loading: filling plug with
-fuel elements
-as Ø200mm case

Fig.13 BR2 with a central hole of 400 mm diameter

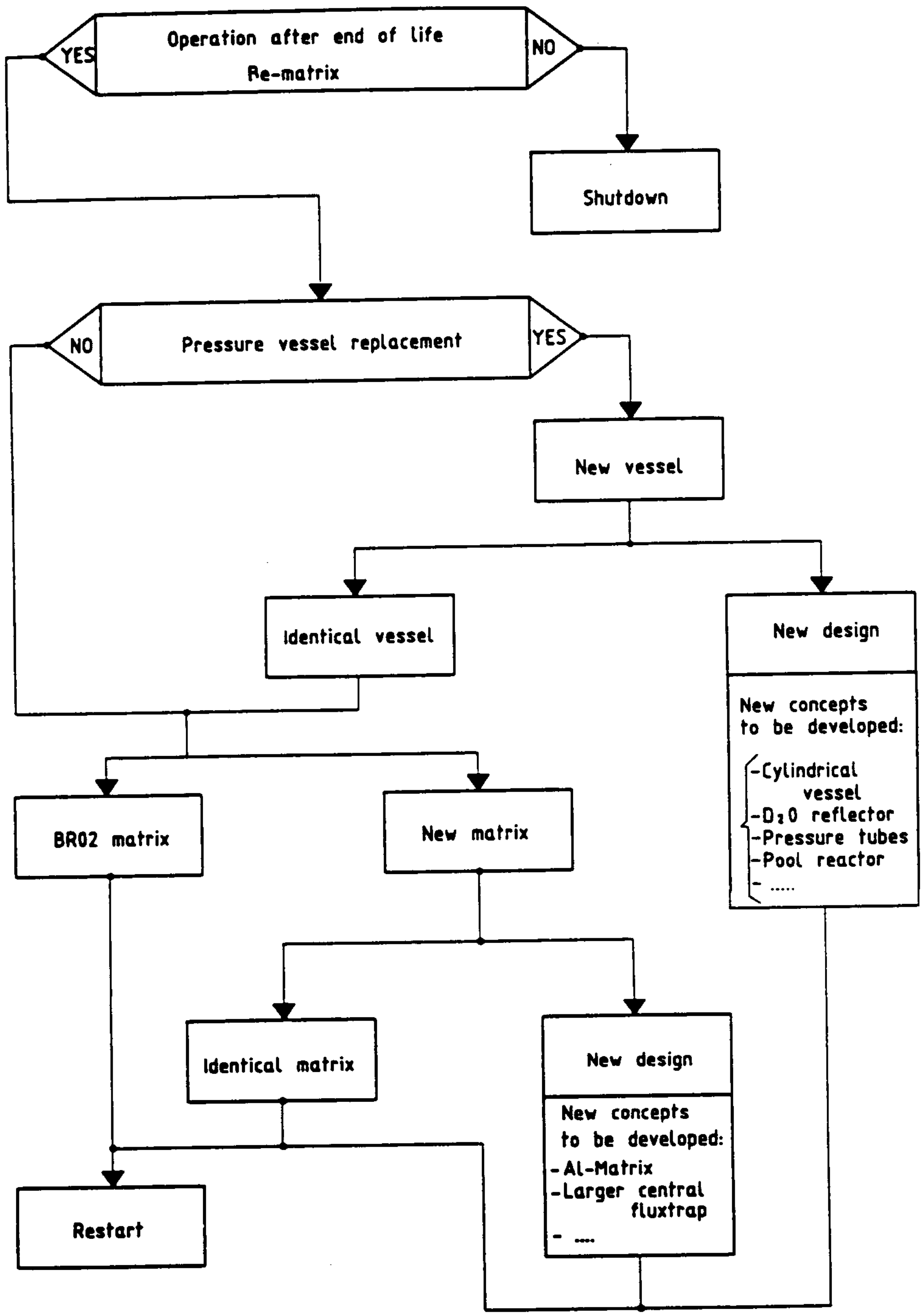


Fig.14 BR2 Long-term strategy

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THE ORPHEE REACTOR CURRENT STATUS AND PROPOSED ENHANCEMENT OF EXPERIMENTAL CAPABILITIES

P. Breant

The Orphée reactor current status and proposed enhancement of experimental capabilities

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Saclay - FRANCE

ABSTRACT

This report provides a description of the Orphée reactor, together with a rapid assessment of its experimental and research capabilities. The plans for enhancing the reactor's experimental capabilities are also presented.

I. REACTOR GENERAL DESIGN PRINCIPLES

Orphée is a swimming pool-type reactor cooled with light water and moderated with heavy water. It was designed and built between 1975 and 1979. Initial criticality was achieved in December 1980 after a one-year testing period to verify proper operation of the reactor's functional and safety systems.

Low-power testing was carried out from December 1980 to July 1981, followed by high-power testing from July 1981 to September 1981. It was thus possible to verify neutron characteristics, core thermal performances, available neutron fluxes and their distribution, power distributions, efficiency of the control rods, reactivity effect, and efficiency of radiological protection features.

The following main criteria were used for design of the reactor :

a) Continuous, independent verification of the effectiveness of three protective barriers :

- the fuel cladding (first barrier),
- the reactor primary system and swimming pool (second barrier),
- the reactor building and its ventilation system associated with its isolation valves (third barrier).

b) Redundancy of assessment and monitoring facilities for the main reactor protection parameters with total separation of instrumentation channels.

c) Design, construction and testing of systems in accordance with a quality assurance plan employing qualified teams, facilities, suppliers and codes.

d) Definition of a design basis accident (worst-cas hypothetical accident) and assessment of the maximum allowable consequences for the environment. All reactor protection facilities are designed to effectively maintain these consequences within the prescribed limits.

e) Capability to control the reactor in case of an accident from an emergency control panel that can ensure safe reactor shutdown, i.e. :

- insertion of the control rods and verification of their position,
- core cooling with reinjection of makeup water, if necessary, to prevent core dewatering,
- reactor containment with, if necessary, planned, controlled depressurization of the containment.

II. DESCRIPTION

II.1 Reactor block (core and moderator)

As shown in Figure1, the core is relatively compact. It consists of 168 identical fuel plates contained in two types of fuel assemblies : standard (24-plate) assemblies and control (18-plate) assemblies. Each fuel assembly has a square section of 82.4 mm per side.

The fuel is placed around a central beryllium element designed to optimize power distribution in the fuel.

The core, which has a life of 100 equivalent full-power days, is completely discharged at the end of one operating cycle and replaced by a new one. For this reason, space is provided along the centerline of the beryllium element to contain an antimony-beryllium neutron source required for reactor control during the core loading phase.

The four control assemblies are placed in the corners of a Zircaloy 2 core tank. The four standard assemblies are placed at the center in relation to the four sides of the tank.

In order to improve axial power distribution, the side plates of the fuel assemblies contain a partial load of boron. During two-thirds of the operating cycle, the control rods thus move only about $\pm$ 5 mm from their assigned position (Figure 2), which provides a highly constant neutron flux distribution in the moderator.

The moderator is enclosed in a stainless steel tank about 2 m in diameter and 2 m high. It contains about 6 t of heavy water that is usually between 99.7 and 99.9% pure. Tritium content is maintained between 2 and 4 Ci/l with an exceptional maximum of 6 Ci/l.

The reactor block is located at about mid-height of a 15 m deep swimming pool designed to always contain water. The pool is divided into three areas of about the same height.

From top to bottom (see Figures 3 and 4), these areas are:

- a 4.5 m deep basin that can be drained to a tank when the reactor is shut down,
- the core, reflector and channels area, which is 5.5 m high. This area is closed off by a double-shell tank assembly with the inner tank made of stainless steel. In the event of a design basis accident (135 Mjoule), part of the mechanical energy would be absorbed by deformation of the inner tank. This deformation would not cause the tank to rupture, as demonstrated by qualification tests performed on a model.
- a decay tank, which receives water circulated to cool the core and enables disintegration of the nitrogen 16 contained in the main reactor system water.

Adjacent to the main swimming pool is a transfer canal (or service pool) where all routine service operations are performed on the fresh and irradiated fuel, as well as on all the activated products from the reactor. No hot cell is provided for the Orphée reactor.

II.2 Core performance characteristics

When equipped as described above, the reactor core is highly undermoderated. It has the following characteristics :

- MTR-type fuel with 93 % U-235 enrichment,
- UAl alloy with 34 % enrichment,
- total U-235 fuel weight : 5.9 kg,
- U-235 per plate : 0.035 kg,
- fuel core working dimensions : 0.25 x 0.25 x 0.9 m or a volume of 56 dm^3,
- heat exchange area : 20.68 m^2,
- core mean specific power : 0.25 MW/l,
- mean heat flux : 71 W/cm^2,
- maximum heat flux : 172 W/cm^2,
- maximum hot channel heat flux : 206W/cm^2,
- maximum temperature at surface of fuel plates : 123.5°C,
- available undisturbed thermal neutron flux in the moderator at core maximum level : 3 x 10^{14} $n/cm^2/s$,
- working thermal neutron flux : 2.5 x 10^{14} $n/cm^2/s$,
- maximum gamma flux : 6 W/g,
- gamma flux in integrated vertical devices : hot and cold sources < 2 W/g,
- core cooling capacity : 835 m^3/h light water,
- water flow through fuel plates : 7.5 m/s,
- maximum coolant temperatures : 35°C inlet, 49°C outlet,
- hot channel factor : approx. 1.40,
- flow capacity redistribution margin (all uncertainties cumulated) : 2.12,
- moderator circulation capacity : 35 m^3/h,
- moderator circulation flow in heavy water tank : < 0.1 m/s,
- total reactor power : 14 MW

. distributed in :	core	12.4 MW
	pool	0.8 MW
	reflector	0.8 MW

II.3 Horizontal and vertical devices

Vertical devices

The reactor includes two types of vertical devices : two liquid hydrogen cold sources and one hot source. These devices are considered in the reactor safety analysis.

Four channels are used for activation analysis.

Six channels enable irradiation of various materials : radioisotopes, silicon for neutron irradiation doping.

Associated experimental rigs

These devices include nine channels, eight with two beams and one with four beams. They thus provide a total of 20 beams (eight cold beams, eight thermal beams and four hot beams).

Six cold beams are associated with neutron guides.

The reactor can accomodate 25 spectrometers.

One neutron guide provides capability for use of a neutron radiography facility.

II.4 Other systems

A detailed description is not given for the primary system, swimming pool system and heavy water system. These systems are all of conventional design and include : pumps, heat exchangers, purification systems, storage vessels, instrumentation and controls.

All or part of these systems are generally considered important to reactor safety. They are built to meet precise quality criteria.

The following sections provide a more complete description of the following :

- containment and ventilation system,
- cold sources,
- hot source,
- reactor protection system.

II.4.1 Containment and ventilation system

The reactor building is made of strongly reinforced 0.60 m thick concrete. It rests on a thick foundation raft and is capped by a dome of decreasing thickness form the periphery (0.60 m) to the center (0.30 m) (Figure 5).

The foundation raft is provided with a leaktight multilayer bituminous coating up to the ground level. Leaktightness integrity is verified by means of sumps that penetrate the containment either at the upper level of the multilayer coating or at the lower level of the swimming pool linings.

The straight part of the building is lined over part of its height by a leak recovery vessel. This double-shell vessel is built to cover the fraction of the reactor building circumference that is penetrated by fluid pipes or electric cable ducts.

The containment penetrations have double walls. Their leaktightness can be tested individually.

The leak recovery vessel is designed to recover any leakage from the penetrations. It is placed under negative pressure in the event of an accident causing an overpressure inside the containment.

The ventilation systems includes the following four subsystems (Figure 6) :

- an air supply subsystem equipped with three fans, one for standby duty, capable of providing suitably humidified and conditioned air. The air supply capacity is 38,000 m^3/h for a total containment capacity of approximately 20,000 m^3 or two renewals an hour,
- a non-active air exhaust subsystem with a capacity of 20,000 m^3/h. Air is removed through high-efficiency absolute particulate filters with an efficiency of a least 1000 for dust particles of 15-micron diameter and iodine filters with an efficiency of at least 100 for methyl iodide. The premises are maintained under an negative pressure of 1 cm of water relative to the outside.
- an active air exhaust subsystem for the "technical rooms", which are subject to atmospheric contamination hazards. This subsystem has a capacity of 18,000 m^3/h and is equipped with the same filtration system as above. The rooms are maintained under a negative pressure of 1 cm of water relative to the above premises (or 2 cm relative to the outside).
- an accident ventilation subsystem with a capacity of 200 to 800 m^3/h.

By regulating valves, following objective examination of the conditions for outside discharge, the accident ventilation subsystem enables :

- depressurization of the reactor containment after an accident that would have placed it under overpressure, to a level such that direct leakage through the reactor building concrete is negligible,
- pumping into the leak recovery vessel to "treat" leakage through the containment penetrations.

All accident ventilation discharges are also filtered through absolute and iodine filters.

The reactor containment is periodically and systematically tested for leaktightness integrity. The maximum leakage allowed by safety authorities is restricted to 1 % of containment capacity per hour.

II.4.2 Cold sources (Figure 7)

The reactor includes two could sources, each contained in tubes that pass directly into the heavy water reflector. Located at about 30 cm from the core centerline, the cold sources remain in the area of maximum thermal neutron flux, but where gamma heating is moderate (< 1 W/g on average).

The vertical arrangement of the source enables separation into two distinct areas the design constraints useful for operation of the source proper and the constraints that provide the researcher an experimental device suited to his needs. Utilization and operation of the source are thus completely independent.

The two sources are supplied with liquid hydrogen via two independent systems, each of which is cooled with a hydrogen-helium heat exchanger. All of these components are completely immersed in the reactor swimming pool, which ensures active safety in case of an accidental loss of leaktightness integrity.

The two He-H_2 heat exchangers are supplied with cold helium by a common cryogenerator for the two sources through a cold box. The cold box can cross over the systems of each of the two sources.

Cold helium is obtained by reducing pressure of the gas through two fluid-bearing gas turbines. The gas is previously compressed to standard temperature at 15 bar using two double-acting piston compressors.

The effective power output obtained for each cold source is 700 watts.

Source No. 1 (SF1) supplies cold neutrons to four neutron guides. Source No. 2 (SF2) supplies cold neutrons to two neutron guides and to a facility located in the reactor hall for use of two 3-axis spectrometers.

Figure 8 shows a graph of the cold neutron gain for various wavelengths.

II.4.3 Hot source (Figure 9)

The reactor block also includes a hot source which is considered in the overall safety analysis. This source is placed in the most heavily loaded area of the reflector tank, i.e. where flux is highest (around 1.5 to 2.5 W/g) and where the available thermal neutron flux is also close to maximum.

The hot source consists of a graphite block approximately 150 mm in diameter and 250 mm high. This block is surrounded by highly effective thermal barriers formed mainly by solid shields and graphite felt ; it is placed inside a double Zircaloy 2 housing. The space between the two housings provides a gas blanket with an atmosphere that can be helium, nitrogen or a mixture of these two gases. The internal housing that contains the graphite block can also be filled by either of these two gases, or placed under a vacuum. The latter operating mode is normally used with a nitrogen gas blanket.

The temperature reached in the graphite under these conditions when the reactor is at nominal power is approximately 1450 K.

The next table shows the numbered values of the hot neutrons gain for this device.

Wavelength Å	1.1	0.7	0.5
Multiplication factor	0.8	2.5	5

The hot source supplies hot neutrons to two channels (four beams). The neutrons are used directly on spectrometers installed in the reactor building.

II.4.4 Reactor protection system

The reactor is protected by a group of redundant facilities and measures that are verified and analyzed by three independent protection channels, which enable 2/3 majority voting for all parameters vital to reactor safety. These parameters mainly cover the following functions :

- neutron verifications
. maximum and minimum counting at high and low power for each instrumentaiton range,
. positive or negative doubling time,
- maximum coolant temperature at the core inlet,
- maximum temperature difference at core inlet-outlet,
- coolant flow,
- maximum gamma activity initiating reactor containment,
- loss of electric power supply,
- maximum pressure of cold sources (hydrogen and isolating vacuum),
- maximum pressure of hot source (internal compartment and gas blanket),
- fuel clad burst (activity and flow),
- maximum shift in control rod positions,
- negative pressure of the reactor building,
- accelerometer on the reactor building (seismic monitoring).

These parameters are all associated with a loop-type logic circuit that initiates a reactor scram command in the event of a continuity break. This command is sent to the reactor through two fully independent reactor scram channels. It can be initiated manually from several locations, particularly via two independent channels connected to an emergency control panel located about 400 m from the reactor in a direction different from that of the prevailing winds on the Saclay site.

The reactor protection system has been considered in a safety study. Using the failure rates of its components, and overall availability was determined and used to establish the frequency of systematic testing of the system. The unsafe failure rates were compatible with the recommendation of IEC standard 231 A (less than 10^{-5} over three months).

III. OPERATION

The reactor is operated by the French Atomic Energy Commission (CEA) with a group of 54 operators, 24 of whom are on continuous duty in six teams of four operators. Each surveillance team includes a shaft supervisor, a mechanic, an electronics technician and an electrical technician. All continuous duty operators are individually certified to operate the reactor through constant monitoring of their technical knowledge.

The reactor operates an average of 250 equivalent full power days a year or 2.5 cycles.

The main operating activities are conducted with an organization to meet the standards established by French law, which stipulates that the reactor operator must define the list of activities for which quality must be monitored.

These activities include:

- periodic tests to comply with precise operating rules for 45 systems. These tests are usually performed by the reactor operating teams or by specialized service teams. Their results are controlled by the operating team managers for technical conformance and by the local quality assurance officer for compliance with procedures.
- maintenance activities for all operating equipment. These activities are also performed in accordance with precise written procedures.

All operating documents are systematically recorded.

IV. UTILIZATION

The French Atomic Energy Commission (CEA) and the National Center for Scientific Research (CNRS) have established a joint laboratory, the Léon Brillouin Laboratory (LLB), for utilization of the Orphée reactor neutron beams. The LLB has a board of directors and a scientific board. Proposed experiments are reviewed every year during specialized working meetings (round tables), which assemble the reactor users and the equipment managers. The reactor users mainly come from the French community, but extensive collaboration has developed at the European level:

- the Federal Republic of Germany has built and manages two spectrometers (one triple-axis thermal neutron unit and one four-circle unit).
- Belgium has installed a time-of-flight spectrometer at the end of a neutron guide,

- Hungary has designed and built a high-resolution spin echo spectrometer in collaboration with LLB teams,
- Austria has built and manages a triple-axis spectrometer installed on a neutron guide.

A total of 25 spectrometers are not installed around the Orphee reactor (Figure 11), 11 in the reactor hall and 14 in the neutron guide hall. These units are listed in Table 1. Orphee is equipped with six cold neutron guides (beam cross section of 15 x 2.5 cm^2), which penetrate the reactor containment and are arranged in an experiment hall 50 meters long and 30 meters wide. In this hall they are extended by secondary neutron guides (two at present), thus increasing the possibilities for use of wide-spectral distribution beams.

In conclusion, it should be recalled that Orphee is provided with:

- four pneumatic channels connected to the Pierre Sue laboratory, which is specialized in activation analysis. Several hundred irradiation operations are conducted in this laboratory every year.
- a neutron radiography facility placed at the end of a neutron guide on beam G4. This facility is used for industrial applications, particularly in the aircraft sector. It is also employed for certain types of non-destructive examinations, notably for ensuring uniformity of the fuel plates and burnable poison plates,
- several vertical devices capable of irradiating large quantities of monocrystalline silicon for phosphorus doping, as well as supplying radioisotopes for industry and medicine.

V. FUTURE PLANS

Orphee will have been in service for 10 years at the end of 1990, which is not very long for a reactor. For the neutron sources, these 10 years have enabled precise determination of the exact capabilities of the reactor and assessment of its few shortcomings. For its utilization, the experimental equipment is now almost fully exploited. The quality of this equipment makes Orphee an ideal complement for the potential of the high flux reactor of the Laue Langevin Institute in Grenoble.

Since nearly all experimental capabilities are being exploited and requests are being received from the European Community, a study is being conducted to enhance the quality and increase the number of usable neutron beams. Three projects are being contemplated for implementation within the next five years:

- improvement of the existing beams. Transmission capabilities of the present cold neutron guides can be enhanced by replacing the most curved neutron guides with straight ones and by changing the reflecting material. The use of Ni58 or supermirrors is under consideration.
- the creation of new cold neutron beams. This project has already been carried out with installation of the two secondary neutron guides. A third secondary neutron guide is being designed and will be installed in 1991. This enhancement of cold neutron capability also involves the startup of an annular cold source. A prototype will be tested this year under actual service conditions. Such a source would increase the number of channels capable of being supplied with cold neutrons.
- installation of a new neutron guide using the containment penetration provided during construction. This guide, which has a large cross section (about 12 x 12 cm^2), will pass through the reactor hall and into a new neutron guide hall. The beam will then be separated into three secondary neutron guides. This equipment will increase the LLB's experimental capabilities by about 30 % (see Figure 12).

These projects will not exhaust the development potential of the Orphée reactor. The second spare containment penetration could subsequently be used to provide new capabilities, which have not yet reached the design stage.

VI. REFERENCES

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2. J. Safieh, Cold and hot sources study in a research reactor. Thesis presented at INSTN-CEN Saclay. October (1982).

TABLEAU I

30.8.89

LABORATOIRE LEON BRILLOUIN

—

RESPONSABLES D'APPAREILS

SECURITE

CANAL FAISCEAU	TYPE	TELEPHONE	EXPERIMENTATEURS
- DIFFUSION INELASTIQUE & QUASI ELASTIQUE			
1T1	Trois axes, neutrons thermiques	62 28	B. HENNION
2T1	Trois axes, neutrons thermiques	39 78	N. PYKA
4F1	Trois axes, neutrons froids	62 30	B. HENNION
4F2	Trois axes, neutrons froids, "TANGO"	62 30	D. PETITGRAND
8F-G3.2	Echo de spin, haute résolution, "MESS"	72 96	R. PAPOULAR
8F-G4.3	Trois axes, neutrons froids, "VALSE"	65 19	W. SCHWARZ
9F-G6.2	Temps de vol, "MIBEMOL"	63 55	G. CODDENS
- DIFFUSION ELASTIQUE (STRUCTURES)			
3T1	Deux axes, poudres	62 29	M. PINOT
3T2	Deux axes, haute résolution (poudres)	62 29	F. BOUREE
8F-G4.2	Deux axes, neutrons froids	65 19	M. PERRIN
5C1	Deux axes, neutrons polarisés, "POLDIF"	62 31	B. GILLON
5C2	Deux axes, quatre cercles, (monocristaux)	62 31	P. SCHWEISS
8F-G4.1	Deux axes, neutrons froids, "PYRRHIAS"	65 21	Y. ALLAIN-G. ANDRE
9F-G5.1	Laue neutrons (monochromatique)	65 19	A. DELAPALME
6T1	Deux axes, quatre cercles (textures)	62 32	R. PENELLE
- DIFFUSION DIFFUSE (SYSTEMES DESORDONNES)			
8F-G4.4	Diffusion diffuse, détecteurs multiples	65 20	R. CAUDRON
9F-G6.1	Diffusion diffuse, multidétecteur linéaire neutrons polarisés "DNPX"	65 17	G. PARETTE
7C2	Deux axes, multidétecteur linéaire, amorphes et liquides, "PHYLIKAM"	29 58	R. BELLISSENT
- DIFFUSION AUX PETITS ANGLES			
8F-G1.2	Diffusion isotrope, multicompteur à anneaux concentriques, "PACE"	62 79	L. AUVRAY
8F-G2.2	Deux axes, haute résolution, "PADA"	65 16	P. CALMETTES
8F-G2.3	Diffusion anisotrope, multicompteur XY, "PAXY"	62 22	A. BRULET
9F-G5.4	Diffusion anisotrope multicompteur XY, "PAXE"	85 18	J. TEIXEIRA
- APPAREILS SPECIAUX			
8F-G1bis	Prototype de réflectomètre "DESIR"	54 46	B. FARNOUX
9F-G5.3	Diffusion aux petits angles avec orientation nucléaire, "PAON"	62 25	H. GLATTLI
G3 bis	Réflectomètre en cours de montage	54 46	B. FARNOUX

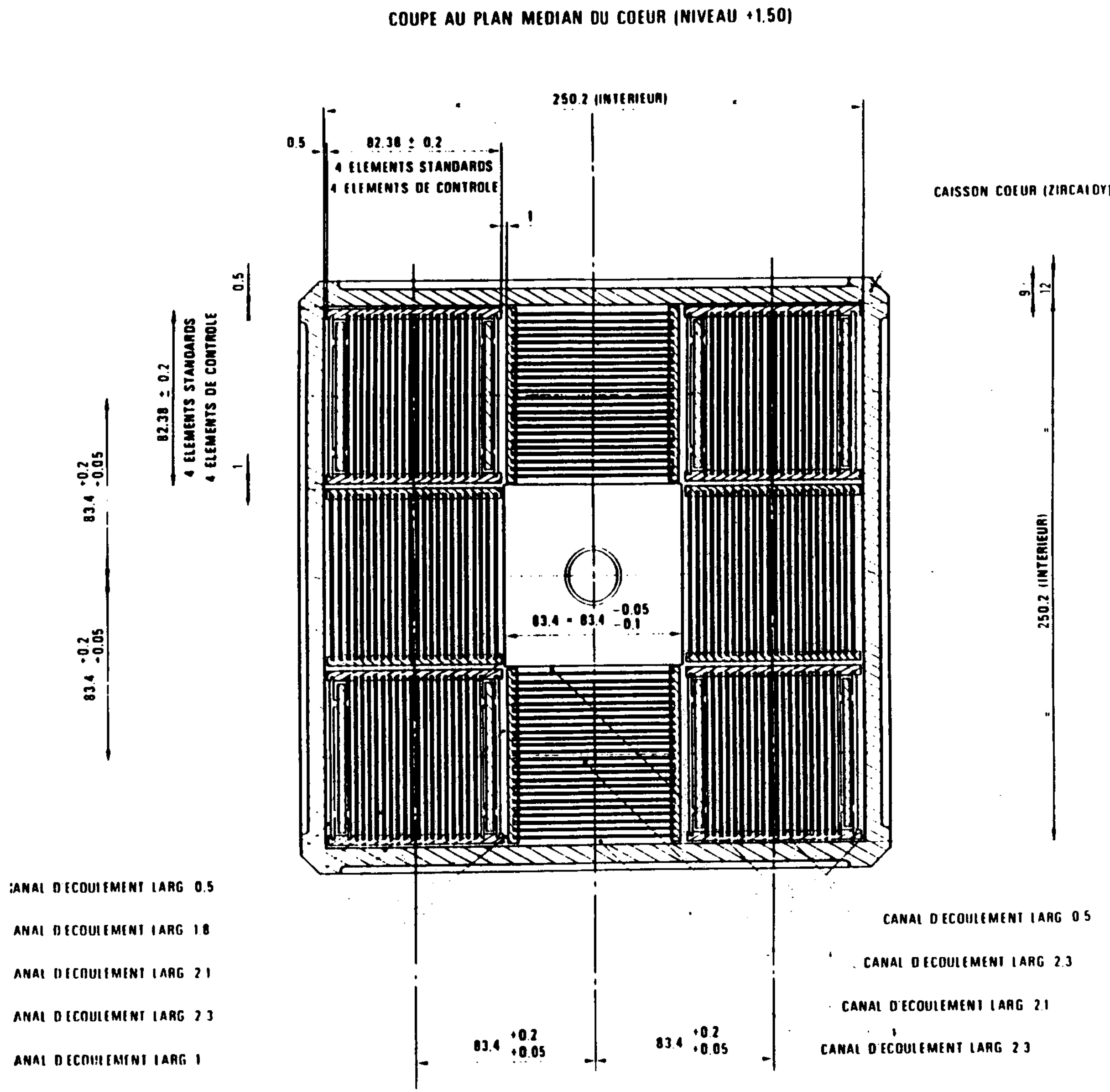

COUPE AU PLAN MEDIAN DU COEUR (NIVEAU +1,50)
250.2 (INTERIEUR)
0.5
82.38 ± 0.2
4 ELEMENTS STANDARDS
4 ELEMENTS DE CONTROLE
1
CAISSON COEUR (ZIRCALOY)
9
12
250.2 (INTERIEUR)
83.4 +0.2 -0.05
83.4 +0.2 -0.05
83.4 × 83.4 -0.05 -0.1
CANAL D'ECOULEMENT LARG 0.5
CANAL D'ECOULEMENT LARG 1.8
CANAL D'ECOULEMENT LARG 2.1
CANAL D'ECOULEMENT LARG 2.3
CANAL D'ECOULEMENT LARG 1
CANAL D'ECOULEMENT LARG 0.5
CANAL D'ECOULEMENT LARG 2.3
CANAL D'ECOULEMENT LARG 2.1
CANAL D'ECOULEMENT LARG 2.3
83.4 +0.2 +0.05
83.4 +0.2 +0.05

Figure 1.

ORPHEE EVOLUTION DE LA COTE DES B.C. PENDANT UN CYCLE

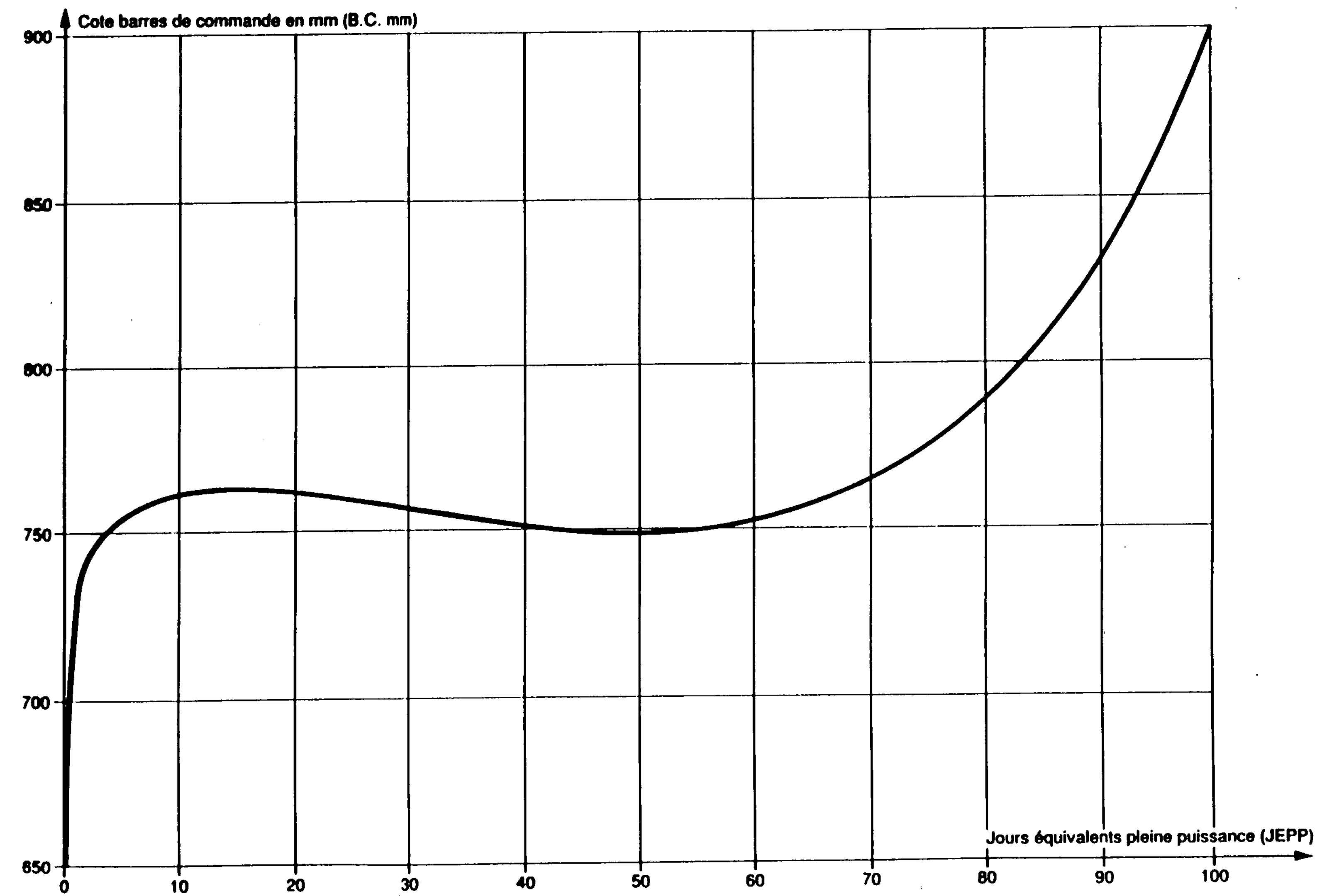
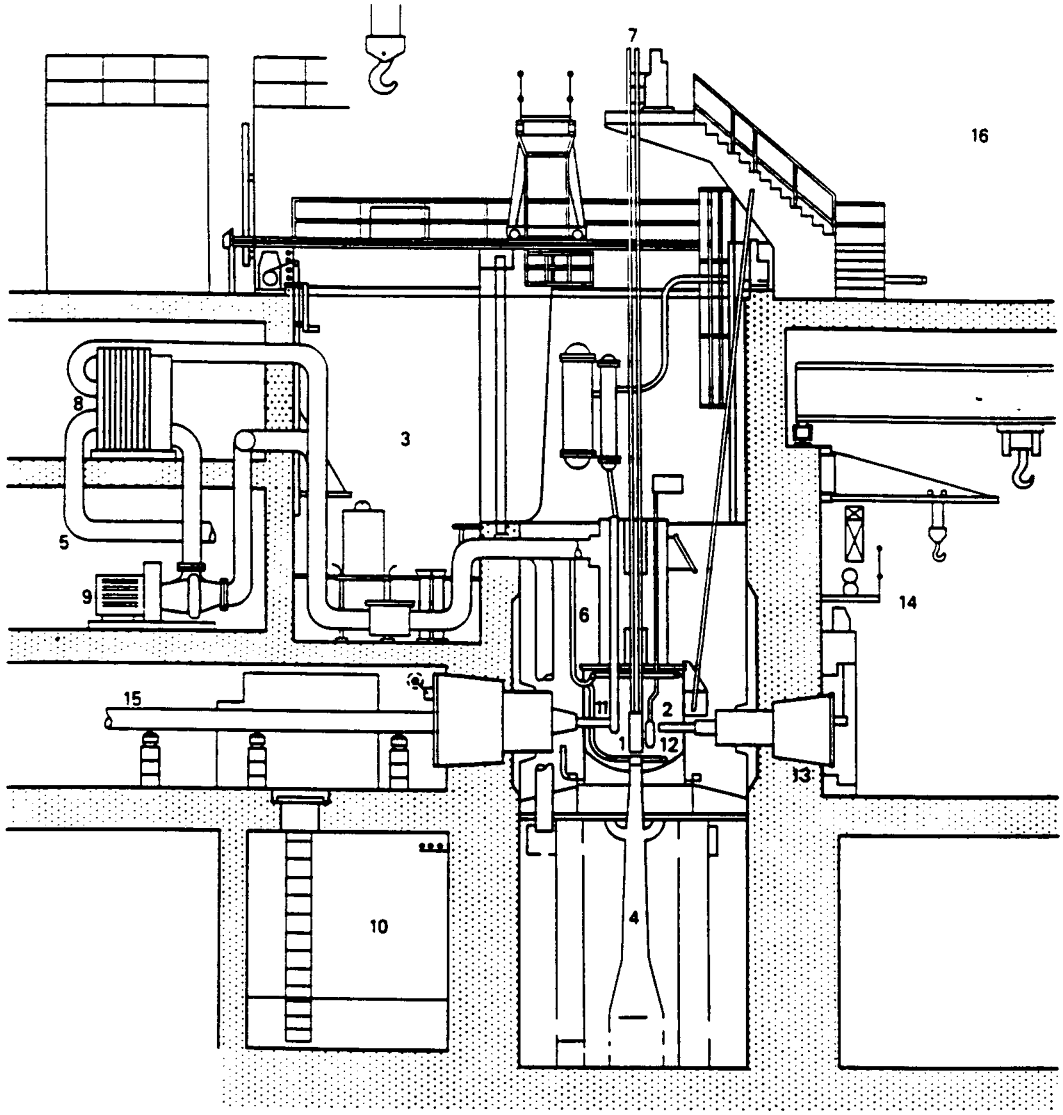
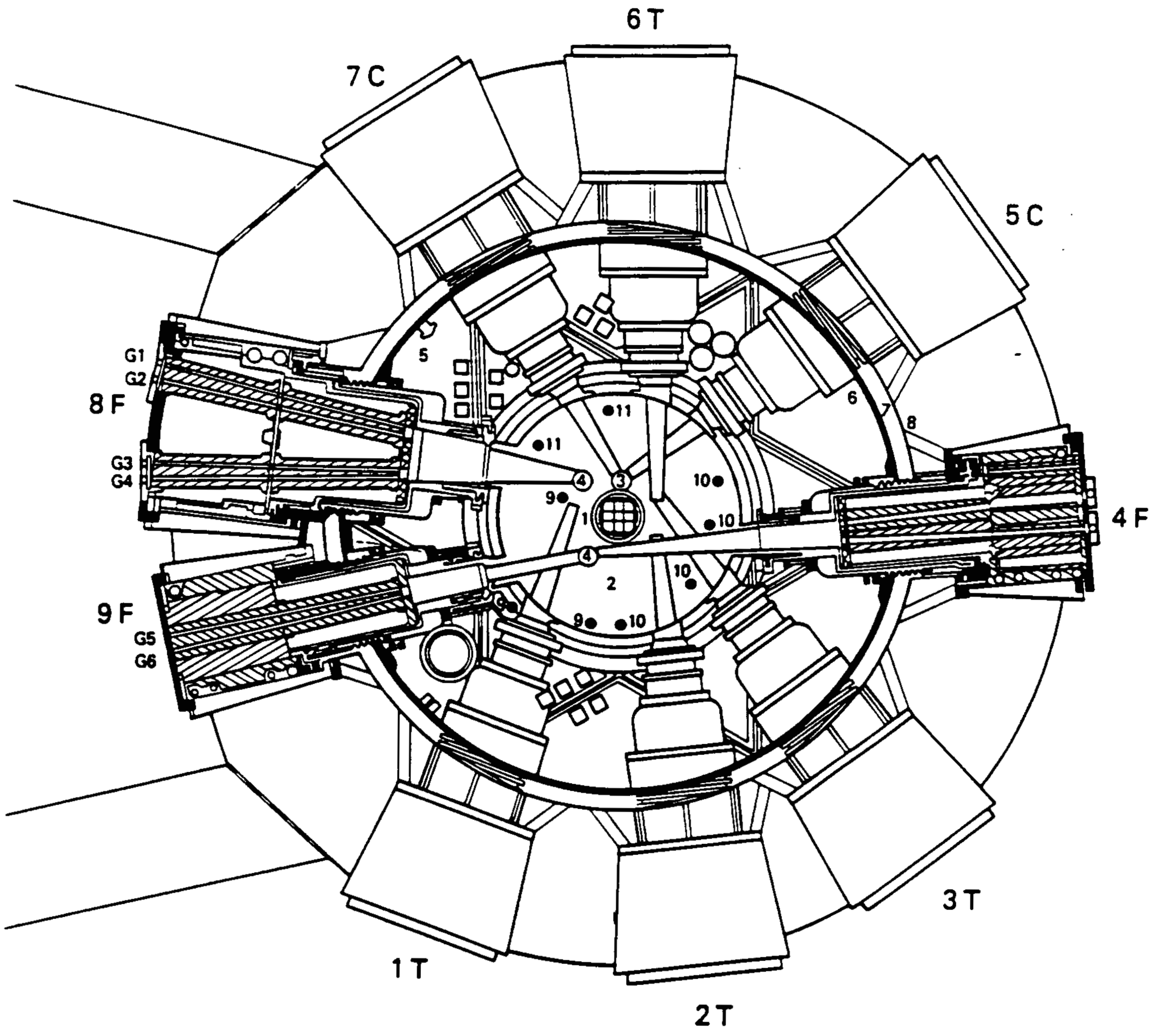


Figure 2.



1. core
2. heavy water reflector
3. pool
4. primary coolant system
5. secondary system
6. heavy water system
7. control rod drives
8. heat exchanger
9. pump
10. pool drain tank
11. low-temperature source
12. high-temperature source
13. tangential channel
14. experimenters' area
15. neutron guide
16. reactor building

Figure 3.



1. core
2. heavy water reflector
3. high-temperature source
4. low-temperature source
5. pool
6. pool inner wall
7. annular space
8. pool outer wall
9. radio-isotope production channel
10. shuttle tube
11. vertical irradiation channel

Figure 4.

SCHÉMA CONFINEMENT DU RÉACTEUR

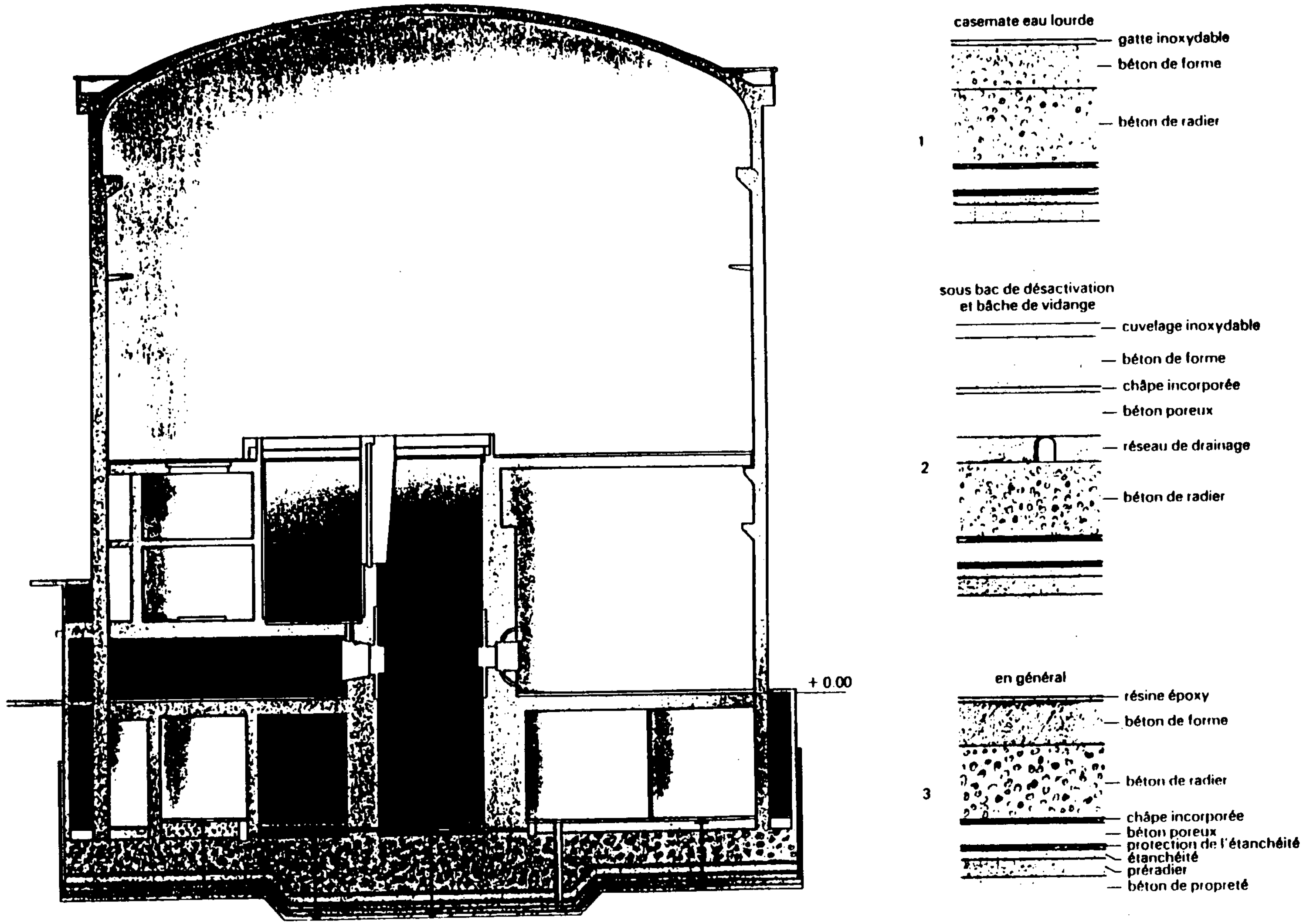


Figure 5.

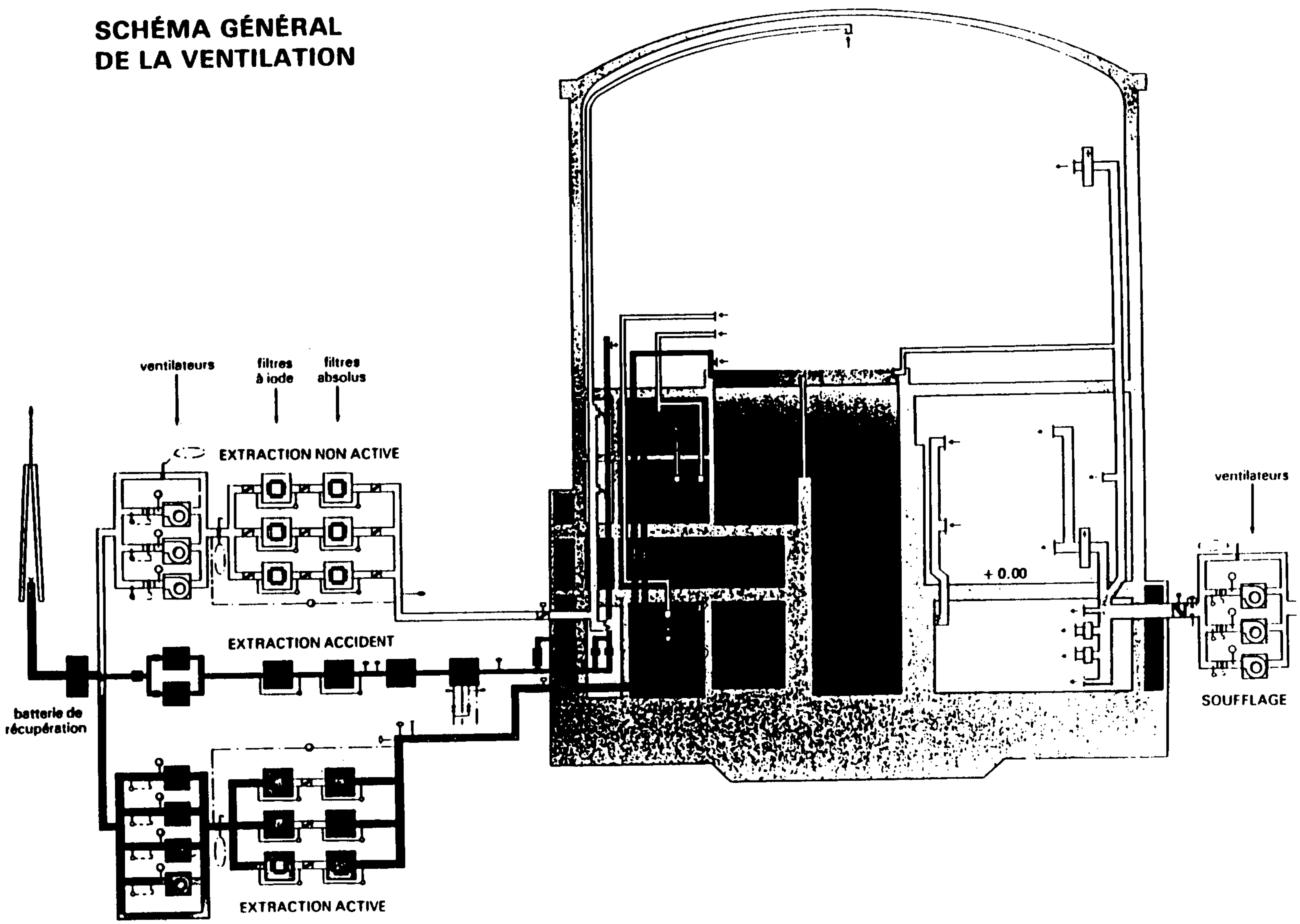
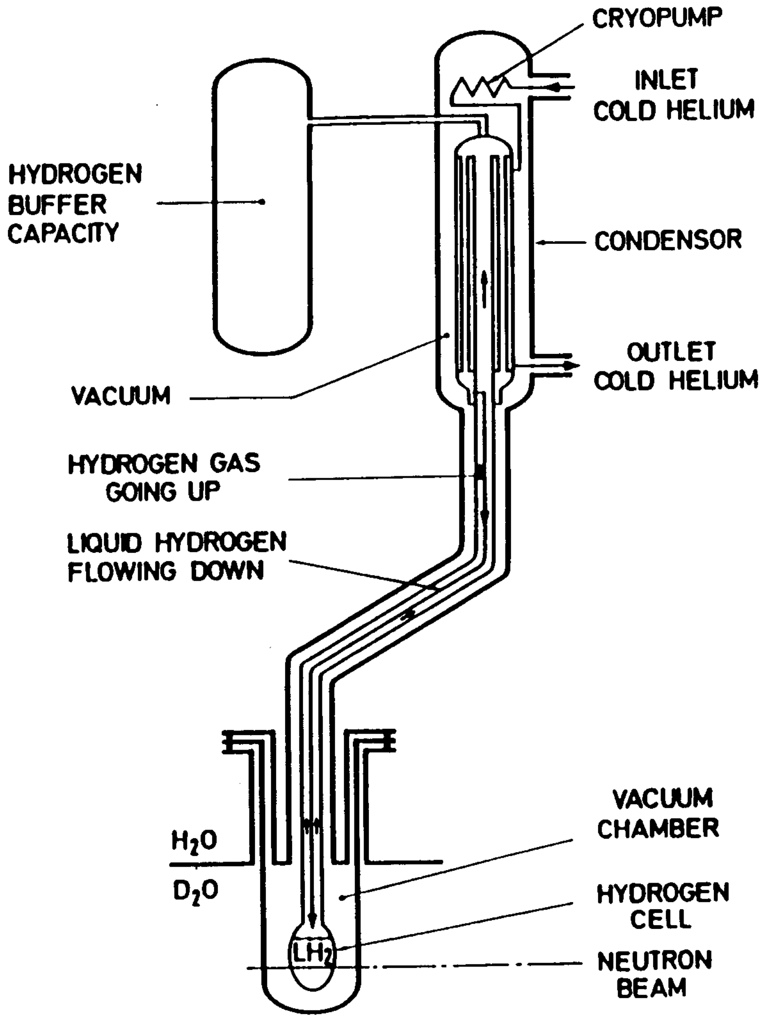


SCHÉMA GÉNÉRAL
DE LA VENTILATION
ventilateurs
filtres
à iode
filtres
absolus
EXTRACTION NON ACTIVE
EXTRACTION ACCIDENT
batterie de
récupération
EXTRACTION ACTIVE
+ 0.00
ventilateurs
SOUFFLAGE

Figure 6.

SOURCE FROIDE ORPHEE

Schéma de principe circuit hydrogène



Courbe de gain

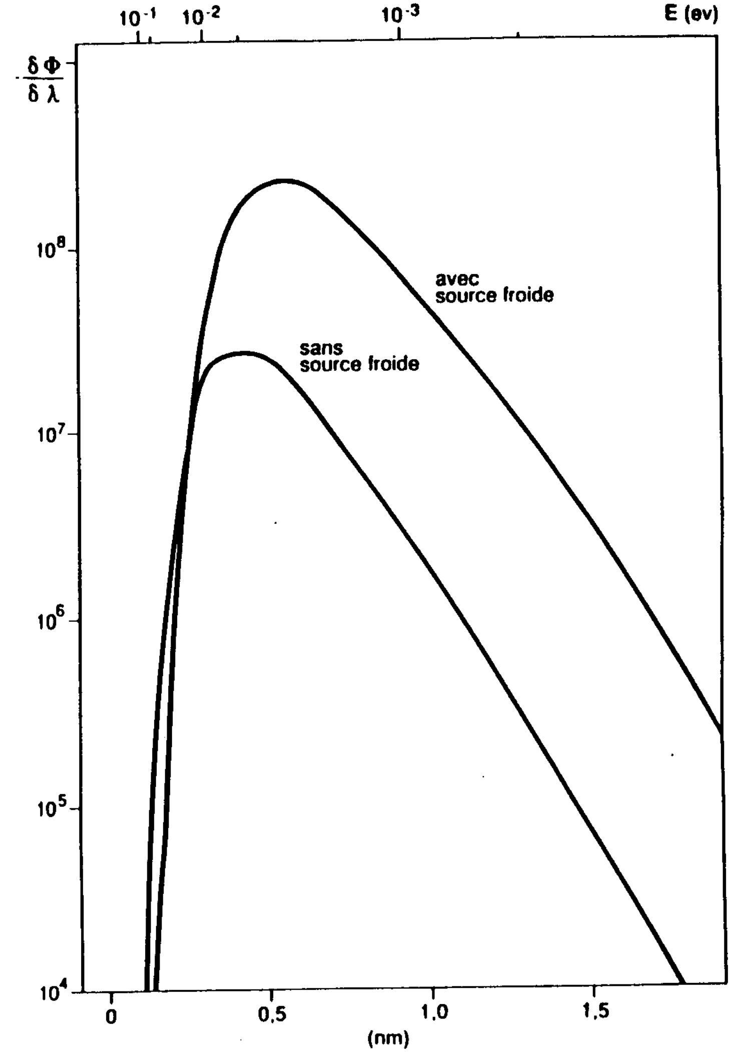


Figure 7.

SOURCE FROIDE ORPHEE
Schéma de principe circuit hélium

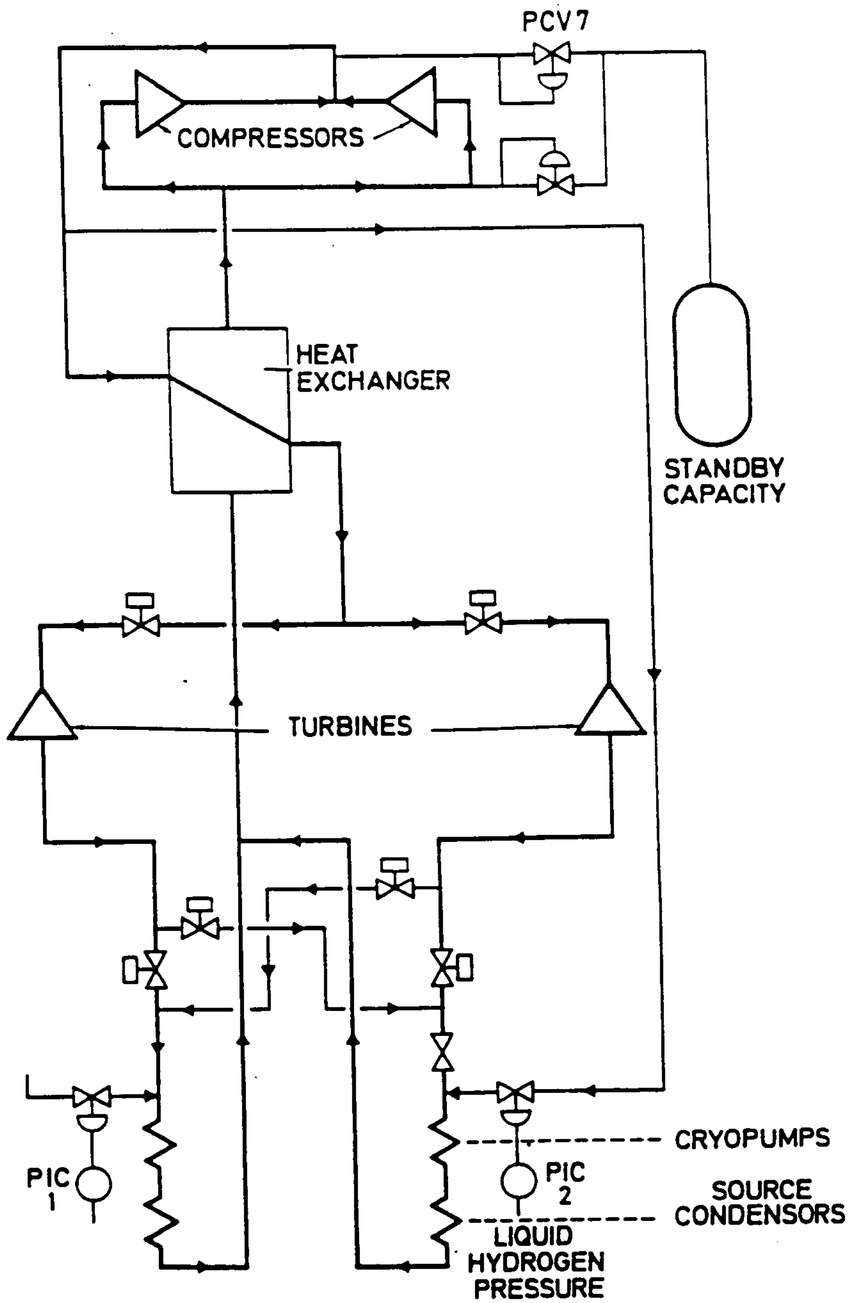


Figure 8.

SOURCES FROIDES
Courbes de gain

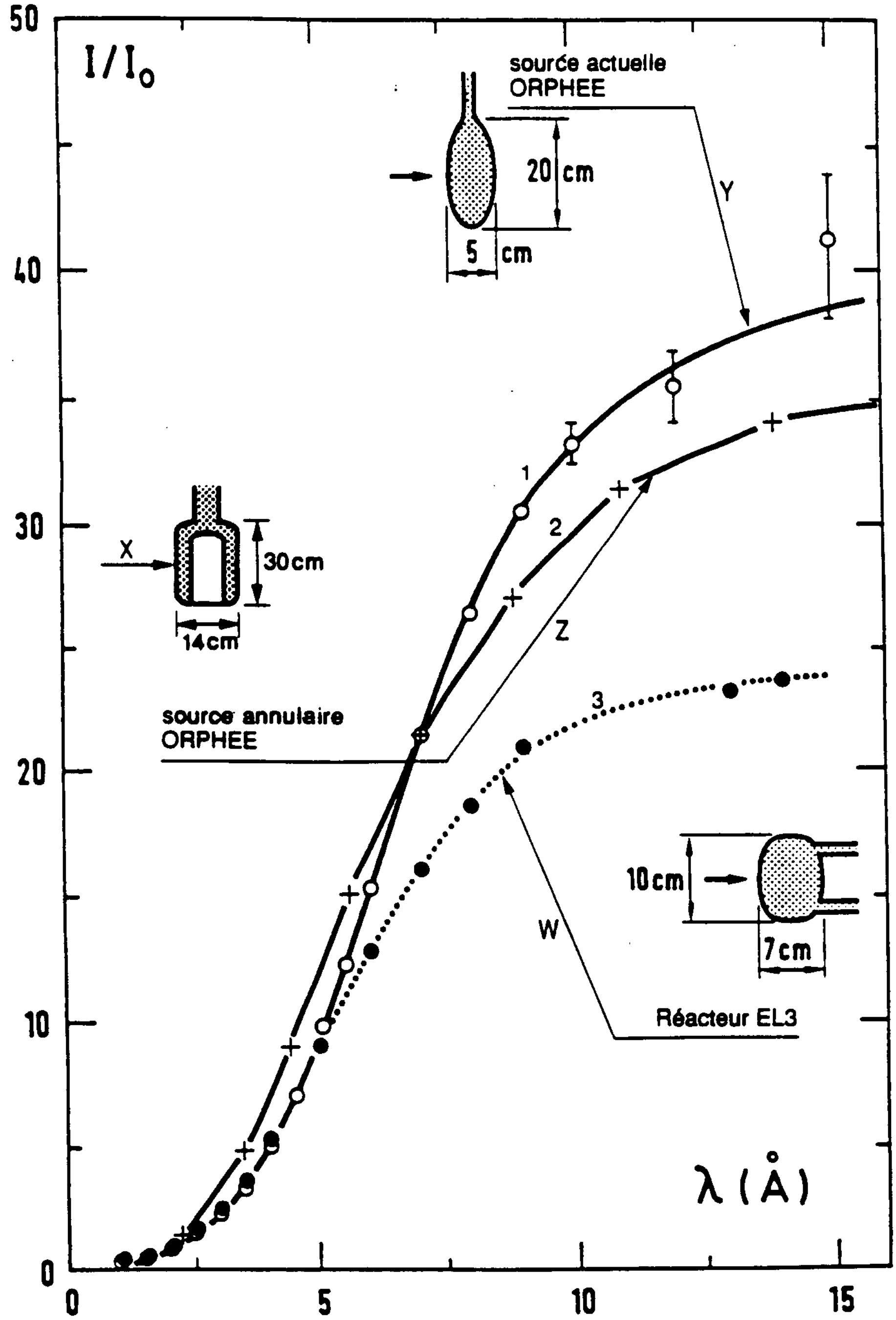
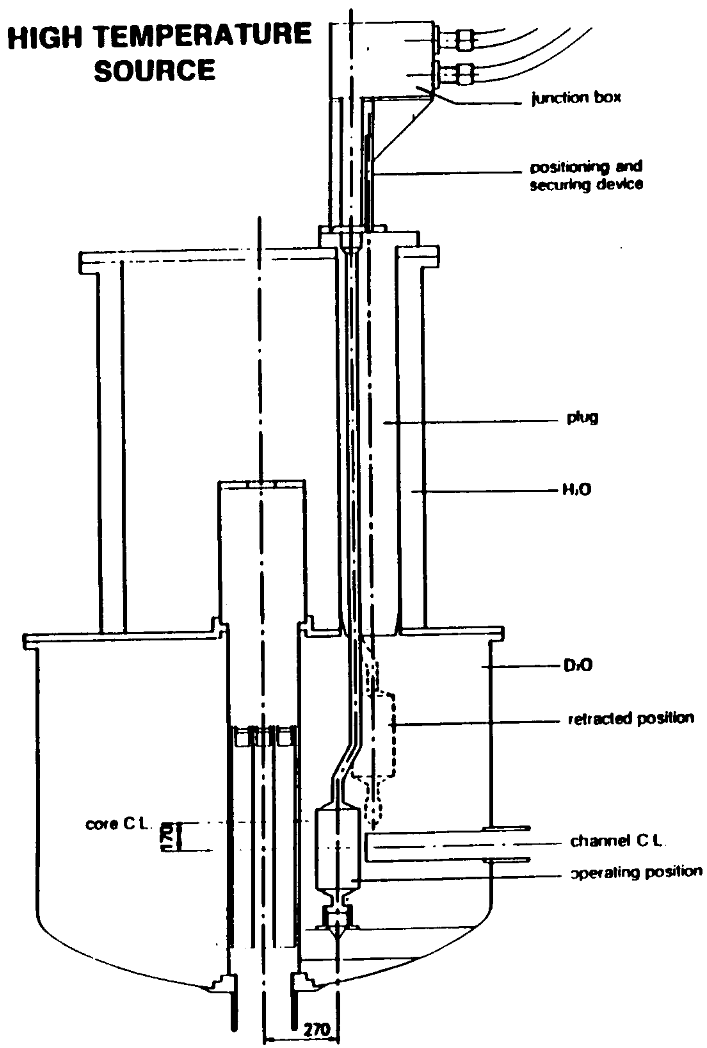


Figure 9.


HIGH TEMPERATURE
SOURCE
junction box
positioning and
securing device
plug
H₂O
D₂O
retracted position
core C L.
120
channel C L.
operating position
270

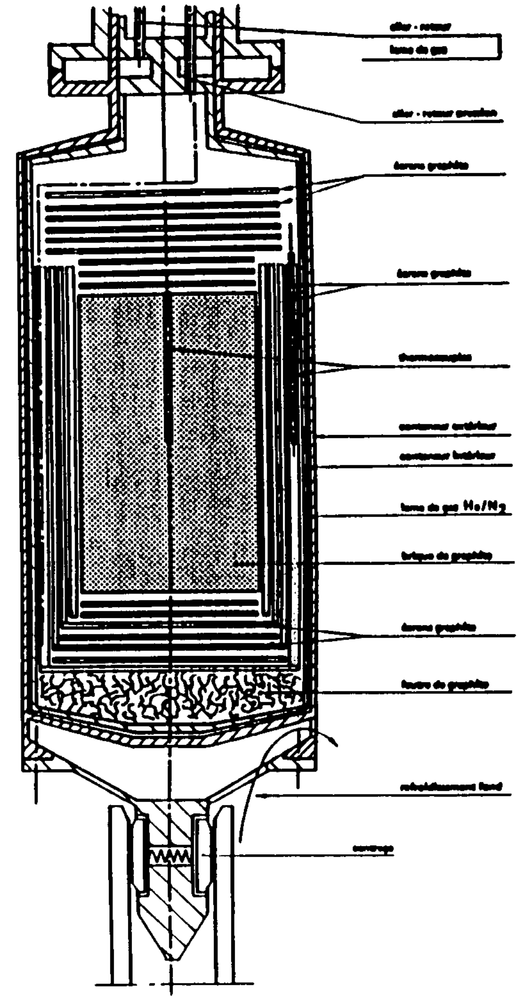

He/N2

Figure 10.

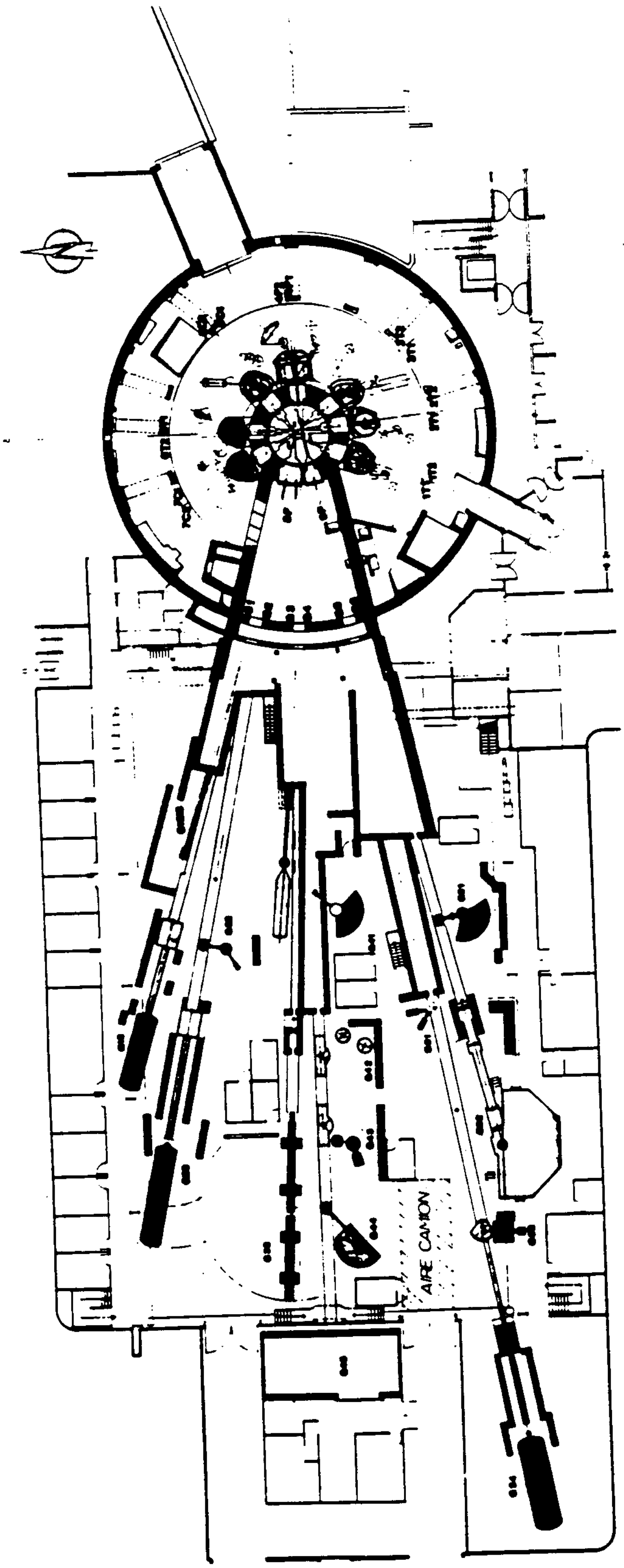


Figure 11.

PROJET DE HALL DES GUIDES EST

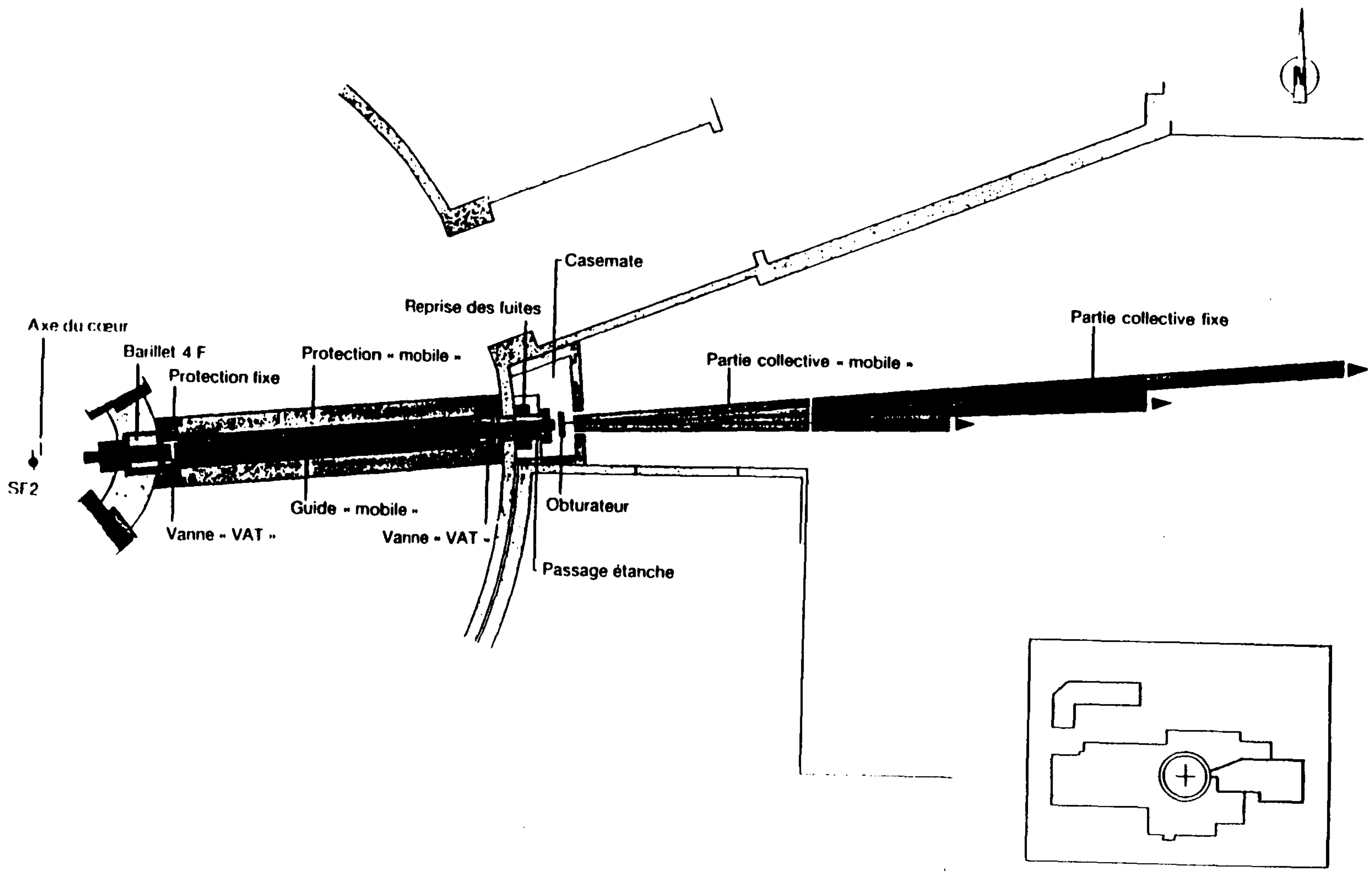


Figure 12.

CONSTRUCTION OF THE UPGRADED JRR-3

S. Matsuura, E. Shirai, and
N. Onishi

Construction of the Upgraded JRR-3

Shojiro MATSUURA, Eiji SHIRAI and Nobuaki ONISHI

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Tokai-mura, Ibaraki-ken,
Japan

ABSTRACT

The construction of the upgraded Japan Research Reactor No.3 (JRR-3) has almost completed. The old reactor was removed from the reactor room, and the new reactor is reconstructed at the place where the old core was. The upgraded reactor is a pool type, 20MW(th), light water moderated and cooled, beryllium and heavy water reflected. Fuels are 20% low enriched UAl_x-Al(LEU) plate type as part of the international Reduced Enrichment for Research and Test Reactors (RERTR) program. Maximum thermal and fast flux are expected to be more than 2×10^{14} n/cm^2s. The upgraded JRR-3 has several beam experimental holes and irradiation facilities for multipurpose utilization, including a cold neutron source facility.

1. INTRODUCTION

The old JRR-3 was the first domestically constructed reactor in Japan. It had been operated for 21 years and was shut down in 1983 for upgrading. Only the reactor building, the spent fuel storage room and the fresh fuel storage facility are reused, but every other facility and building are newly constructed. So we can say this upgraded reactor is essentially new one[1).

2. REMOVAL OF OLD REACTOR

The main body of the old reactor consisting of a core tank, graphite reflector, thermal shield tank and biological shield, was removed by the one-piece-removal method[2). The body was about 2200ton in total weight. It was separated from the reactor room floor by means of continuous core boring, and transferred to the large scale waste storage room. This body will be kept under close observation for a long time. This method has an advantage that the spread of the radioactive waste is avoided. As this is a established technique in the construction industry field, it was carried out in safety. This is the first time for this method to try in the nuclear field.

3. REACTOR FACILITIES

The upgraded JRR-3 is a pool type reactor and the thermal power is 20MW. The depth of the pool is about 8m and the cylindrical core is submerged in light water. The diameter of the core is 0.6m and the height is about 0.75m. Its specific power is rated at 156 kW/l. The fuel is MTR type (UAl_x-Al dispersion fuel) with an enrichment of 20wt%. There are two types of fuel elements, a standard fuel element and a follower fuel element. They have 300g ^{235}U and 190g ^{235}U, respectively. The neutron absorbing control rod is made of hafnium and connected to a follower fuel element.

The heavy water tank, as a reflector, is a double cylindrical type aluminum vessel. Its outer diameter is about 2m and height is about 1.6m. Irradiation thimbles, horizontal beam tubes and a cold neutron source are installed in this tank.

The operation cycle length will be 5 weeks, 4 weeks for a 20MW operation and 1 week for a shutdown work. Nine cycles are scheduled in a year and the operating efficiency will be about 70%.

4. EXPERIMENTAL FACILITIES

Nine horizontal beam tubes (1G-6G,7R,8T,9C) are arranged tangentially to the reactor core. Six tubes (1G-6G) are ready for the neutron scattering facilities in the reactor hall. 7R is for the neutron radiography facility.

There is one cold neutron source (CNS) on 9C. It is a vertical thermo-siphon type and the moderator is liquid hydrogen at about 21K. The gain of this CNS is estimated to be more than 5 for neutrons of 5meV (wavelength is 4Å).

Two thermal and three cold neutron guide tubes are installed in the beam hall which is 30m width and 50m length. Their neutron mirrors are natural nickel sputtered borosilicate glass. The longest guide tube is about 60m and the total length is about 230m.

5. SOME FEATURES ON DESIGN

The design of the upgraded JRR-3 has much contrivance. Some of them are as follows. 1) Maximum thermal and fast flux is rather high with low enriched fuels. 2) Horizontal beam tubes are arranged tangentially to the reactor core center. 3) The cooling system and the ^{16}N decay tank is designed very compactly to be stored in the old building. 4) The emergency exhaust system will ventilate the reactor room through a charcoal filter in accident conditions. 5) Two siphon break valves are installed independently on the primary cooling pipe to prevent the loss of coolant (pool water).

6. CONCLUDING REMARKS

The installation of the upgraded JRR-3 has finished and it will be critical in March, 1990. After the half year characteristic test, full power operation will start in October.

This upgraded JRR-3 will meet the expectations of scientists and engineers by its increasing experimental ability.

REFERENCES

[1] N.Onishi et al., Proc. Int. Symp. on the Utilization of Multipurpose Research Reactors and Related International Co-operation, (IAEA,Vienna, 1988) p.79, IAEA-SM-300/2.

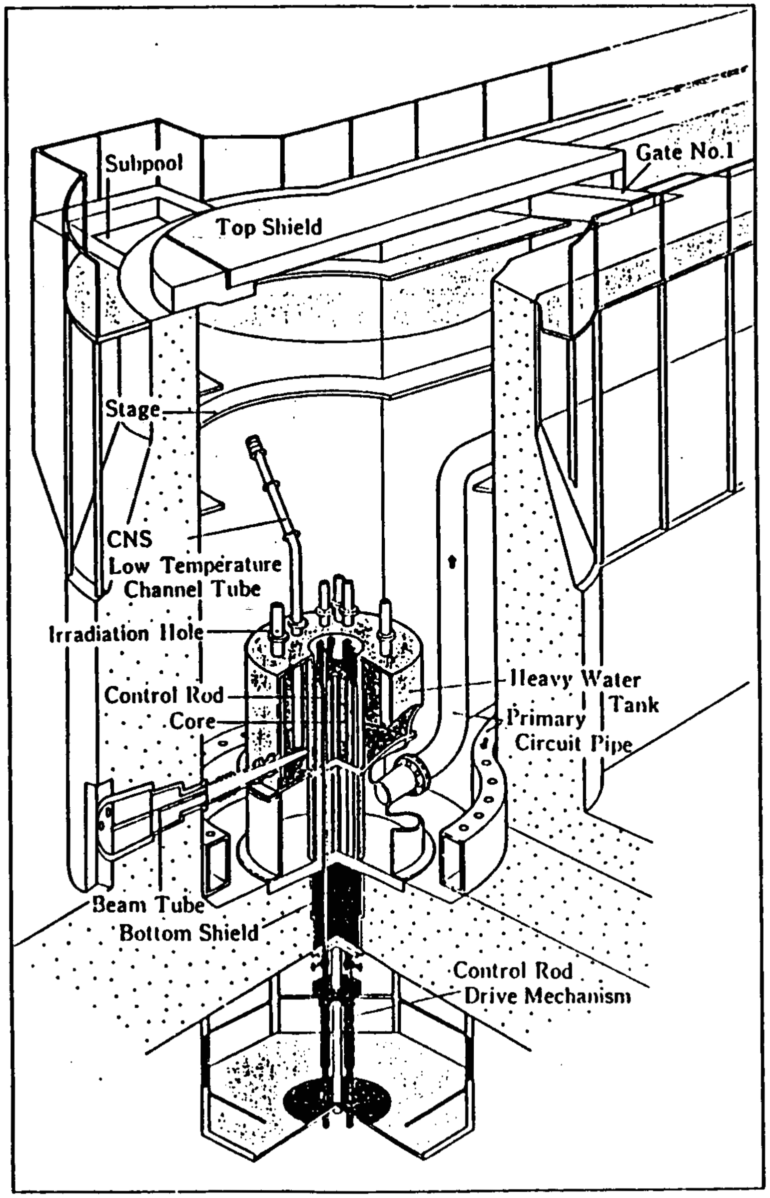
[2] N.Onishi et al., Proc. Int. decommissioning Symp., (Pittsburg, 1987) VI-95, CONF-871018-Vol.1.

History of JRR-3

Year	Major Events	Year	Major Events
1959	Beginning of JRR-3 construction	'76	
'60		'77	
'61	Reactor completion	'78	Integrated power 300,000 MWH achieved
'62	Reactor critical	'79	
'63		'80	
'64	Rated power 10,000 KW achieved	'81	
'65	Beginning of RI production	'82	The twentieth anniversary since the reactor
'66	Beginning of common use		critical
'67	Beginning of homemade fuel use	'83	Close of common use
'68		'84	Finish of safety review for new JRR-3
'69	Medical irradiation for a brain turmor	'85	Beginning of the construction work for
'70			new JRR-3
'71	Sample irradiation of nuclear fuel in LHTL	'86	↑
'72	Beginning of shift to UO_2 fuel core	'87	Construction
'73		'88	
'74	Integrated power 200,000 MWH achieved	'89	↓
'75	Completion of shift to UO_2 fuel core	'90	Completion of new JRR-3

Total operation time : 47,137 hrs 39 min
Total integrated power : 419,073.5 MWH

Isometric View of Reactor Pool



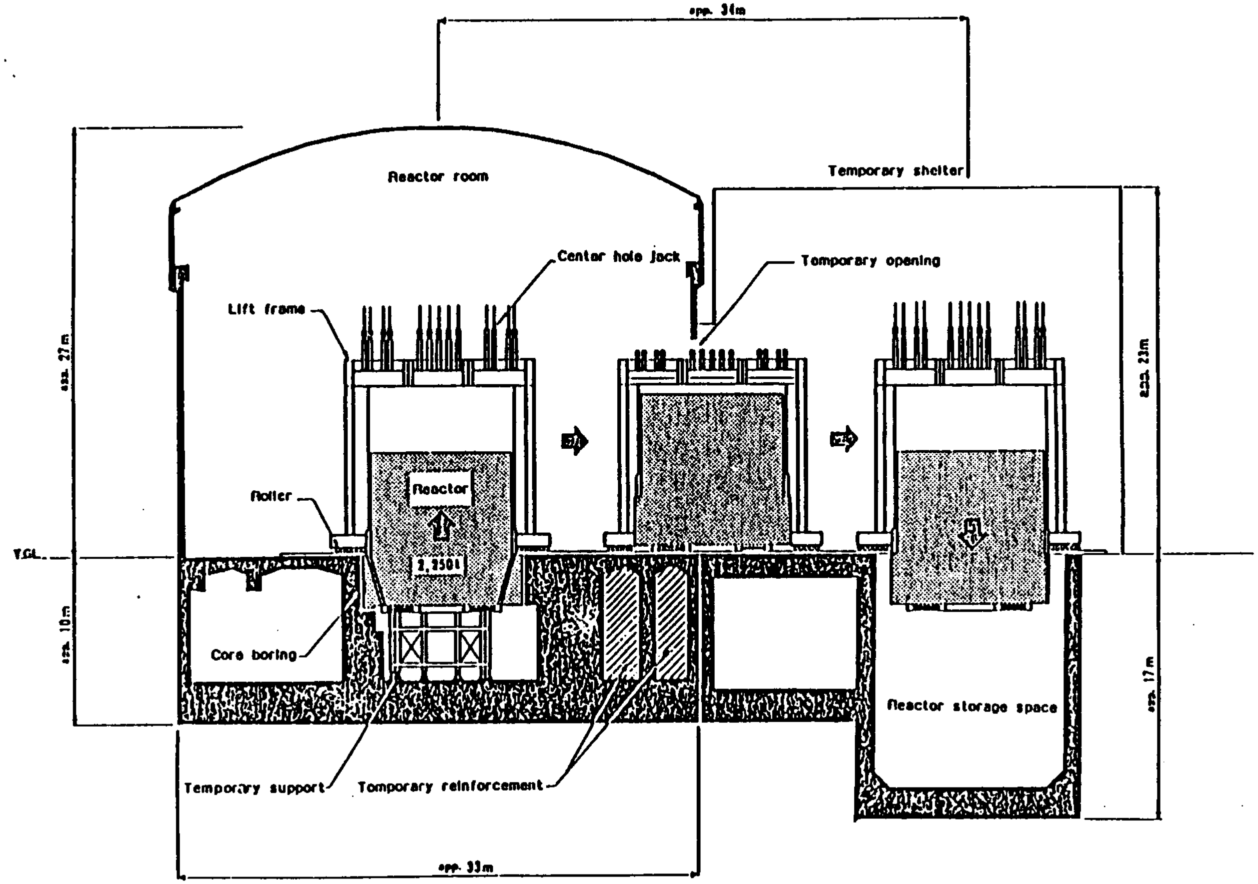
REACTOR

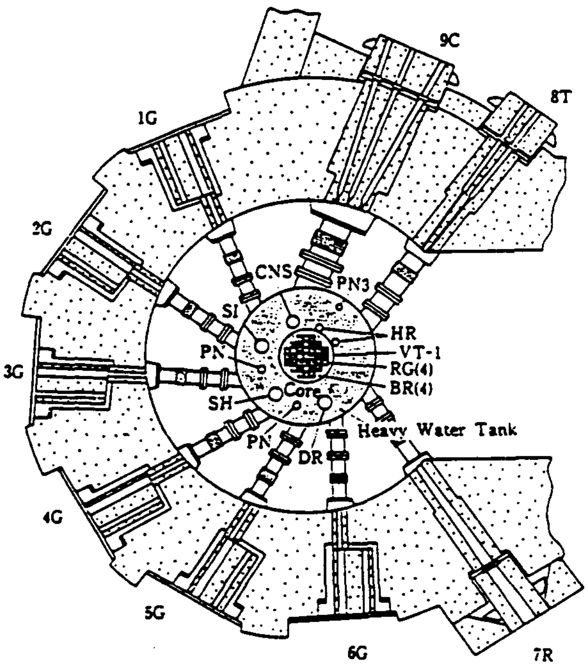
Major Specifications of the New JRR-3

Reactor Type	Low Enriched Uranium, Light Water Cooled and Moderated, Swimming Pool Type.
Rated Power	20 MW
Size of Core	Approx. 60cm dia. and 75cm high (with Beryllium Reflector).
Fuel	UAlx-Al Dispersed, MTR Plate Type 20% Enriched Fuel. 26 Standard Fuel Elements and 6 Follower Fuel Elements.
Control Rod	6 Control Rods, Box Type Absorber, followed with Follower Fuel Element.
Reactor Pool	Swimming Pool Type 4.5m dia. 8.5m deep
Experimental Facilities	9 Horizontal Beam Tubes, 17 Vertical Irradiation Holes, and 1 Cold Neutron Source

CONCEPT OF ONE PIECE REACTOR REMOVAL

Cross Section





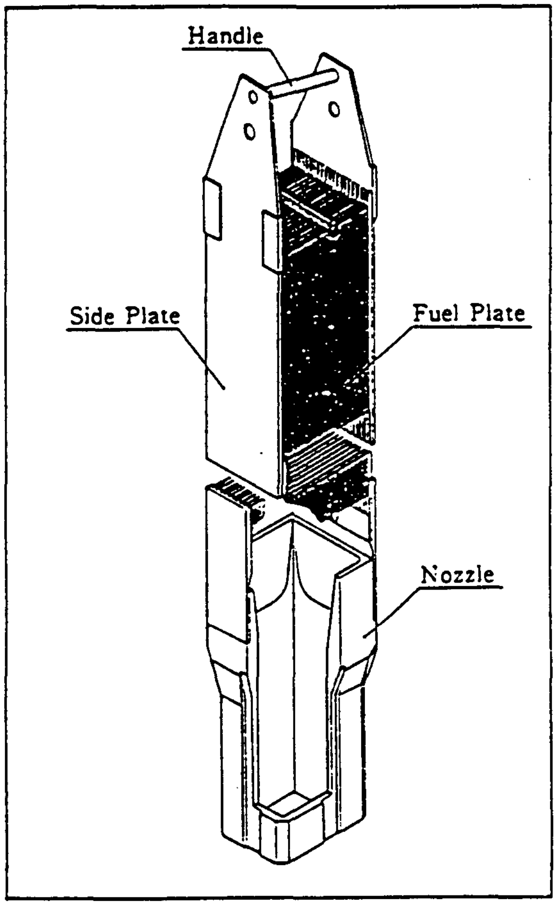
(Vertical Irradiation Holes:
HR, PN, PN3, SI, DR, RG, VT-1, BR, SH
Horizontal Beam Tubes:
1G~6G, 7R, 8T, 9C)

Arrangement of Experimental Holes

Summary

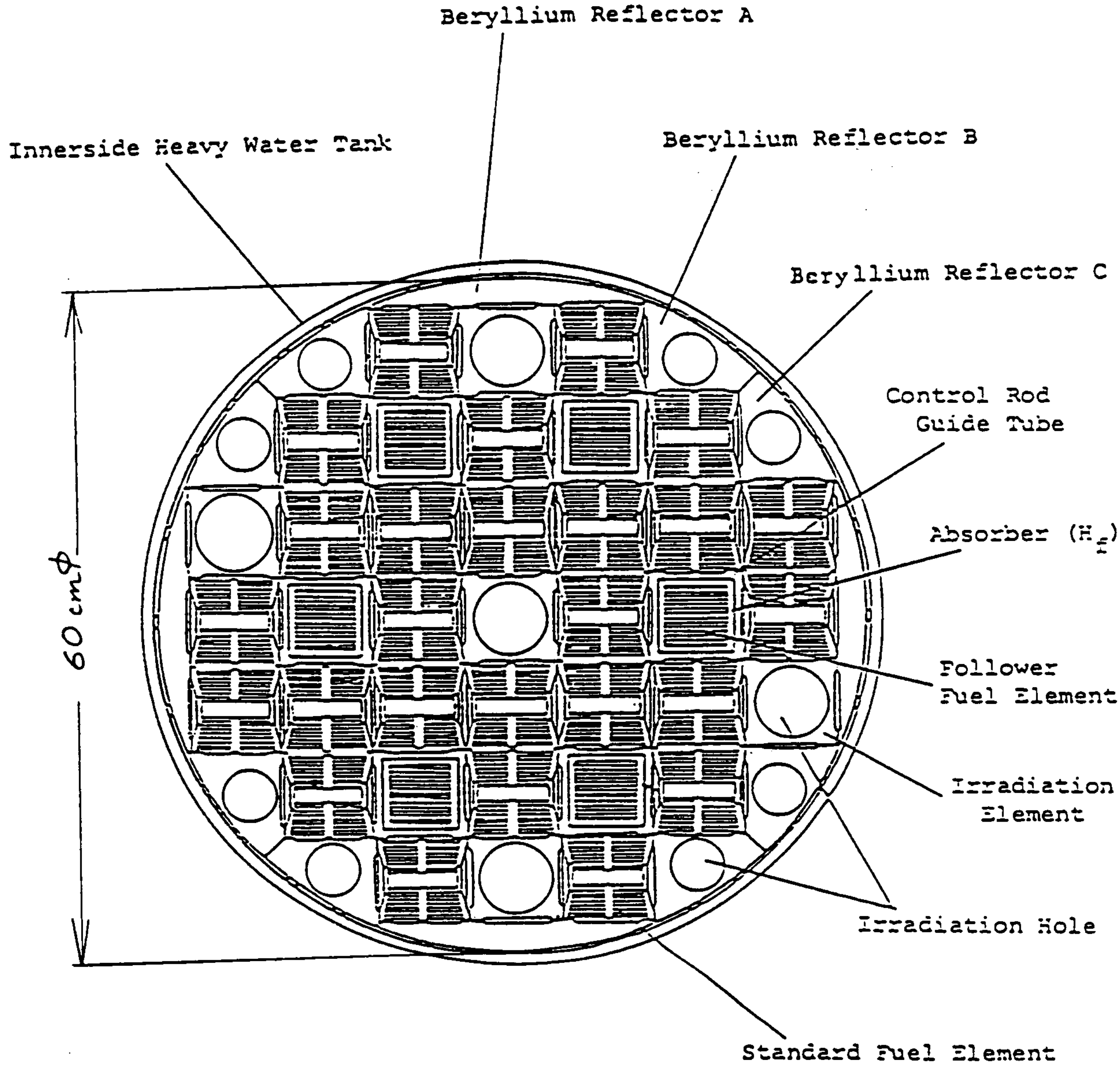
Name	No	Application	Feature
Hydraulic rabbit irradiation facility (HR)	2	General irradiation Radioisotope production	The rabbit is conveyed and cooled by water. This facility is used to irradiate the relatively heavy and high heat generating samples.
Pneumatic rabbit irradiation facility (PN)	2	General irradiation Radioisotope production	The rabbit is conveyed and cooled by N_2 gas. This facility is used to irradiate the light and low heat generating samples.
Activation analysis irradiation facility (PN3)	1	Activation analysis of the short life radio nuclides	The radiation measurement is started immediately after the irradiation. This facility is used to analyze the short life radio nuclides.
Uniform irradiation facility (SI)	1	Material irradiation Silicon irradiation	The sample is rotated and moved up and down during the irradiation. This facility is used to irradiate the sample uniformly.
Rotating irradiation facility (DR)	1	Large material irradiation	The sample is rotated during the irradiation. This facility is used to irradiate the sample uniformly in the radial direction.
Capsule irradiation facility (RG, BR, VT-1, SH)	10	Exposure test Radioisotope production	This facility is used to irradiate for long period or control the sample temperature in response to the irradiation condition.

Standard Fuel Element

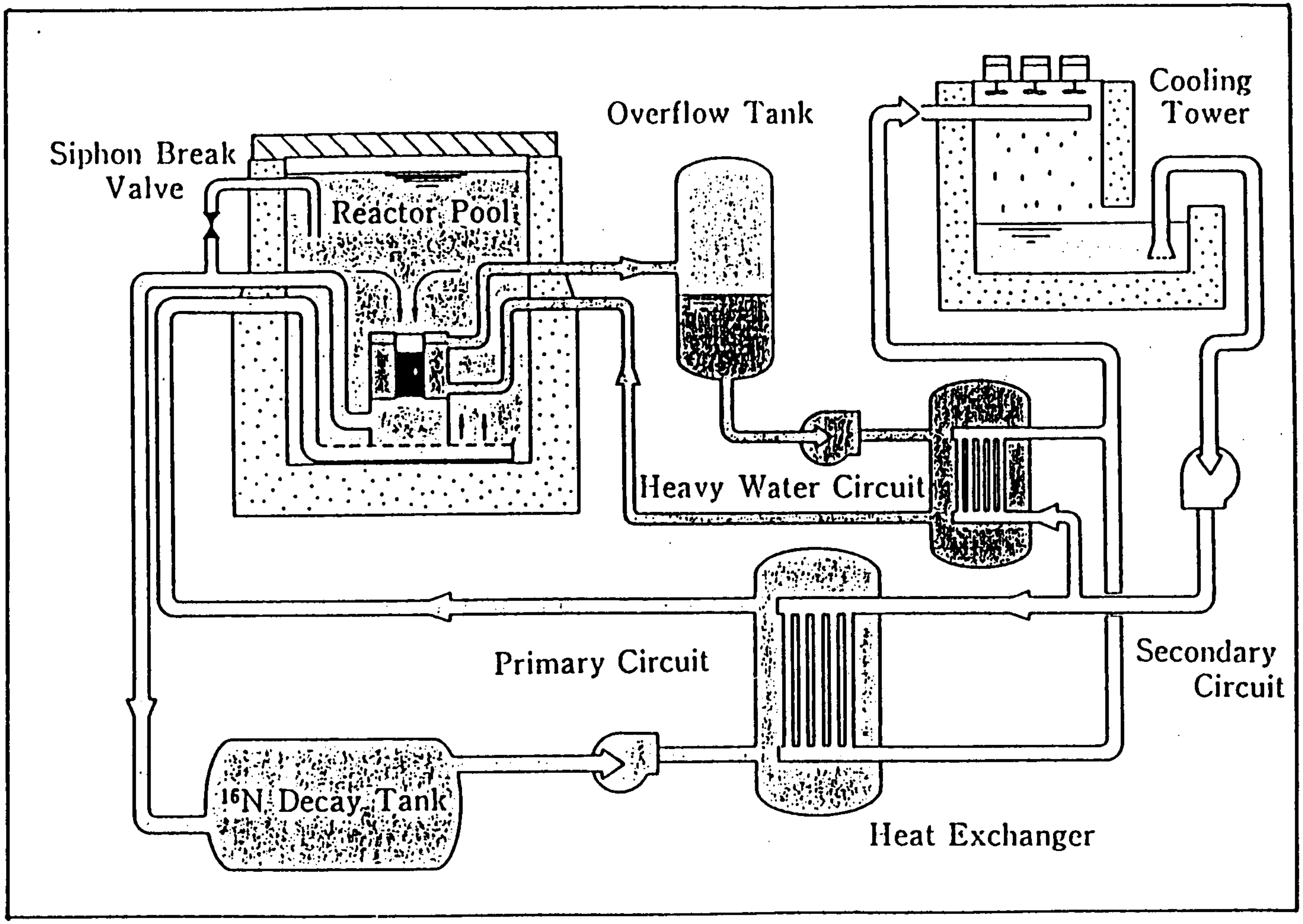


Summary of Fuel Specification

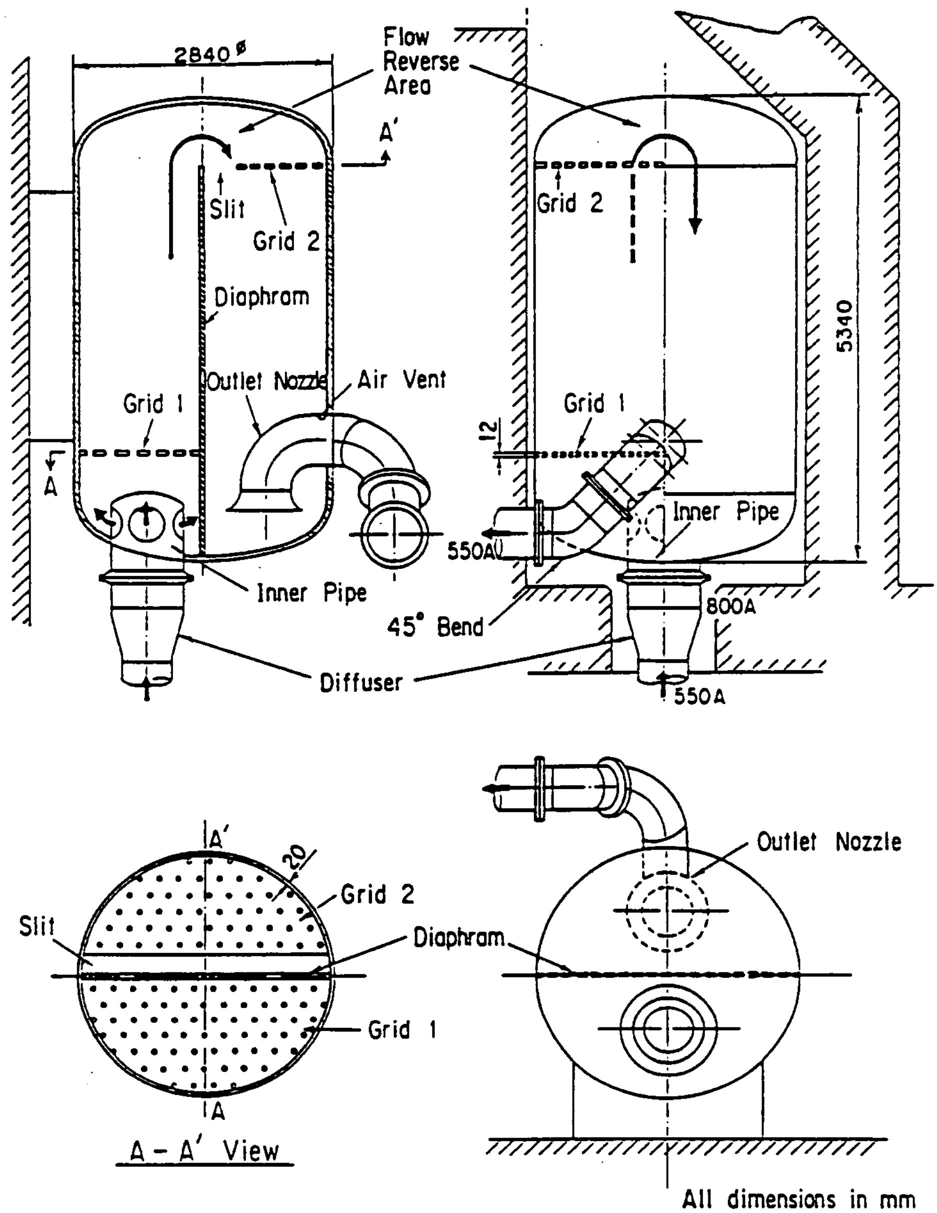
Item	Standard Fuel Element	Follower Fuel Element
Size of Fuel	$76 \times 76 \times 1150$mm	$64 \times 64 \times 880$mm
U-235 Enrichment	20%	20%
U-235 Contents	300g	190g
Size of Fuel Plate	$1.52^t \times 71^w \times 770^L$mm	$1.52^t \times 60^w \times 770^L$mm
Fuel Plate Number	20/Element	16/Element
Fuel Meat Material	Dispersed UAlx-Al	
Cladding Material	Aluminium Alloy	
Maximum Burn-up	50% (on the average)	



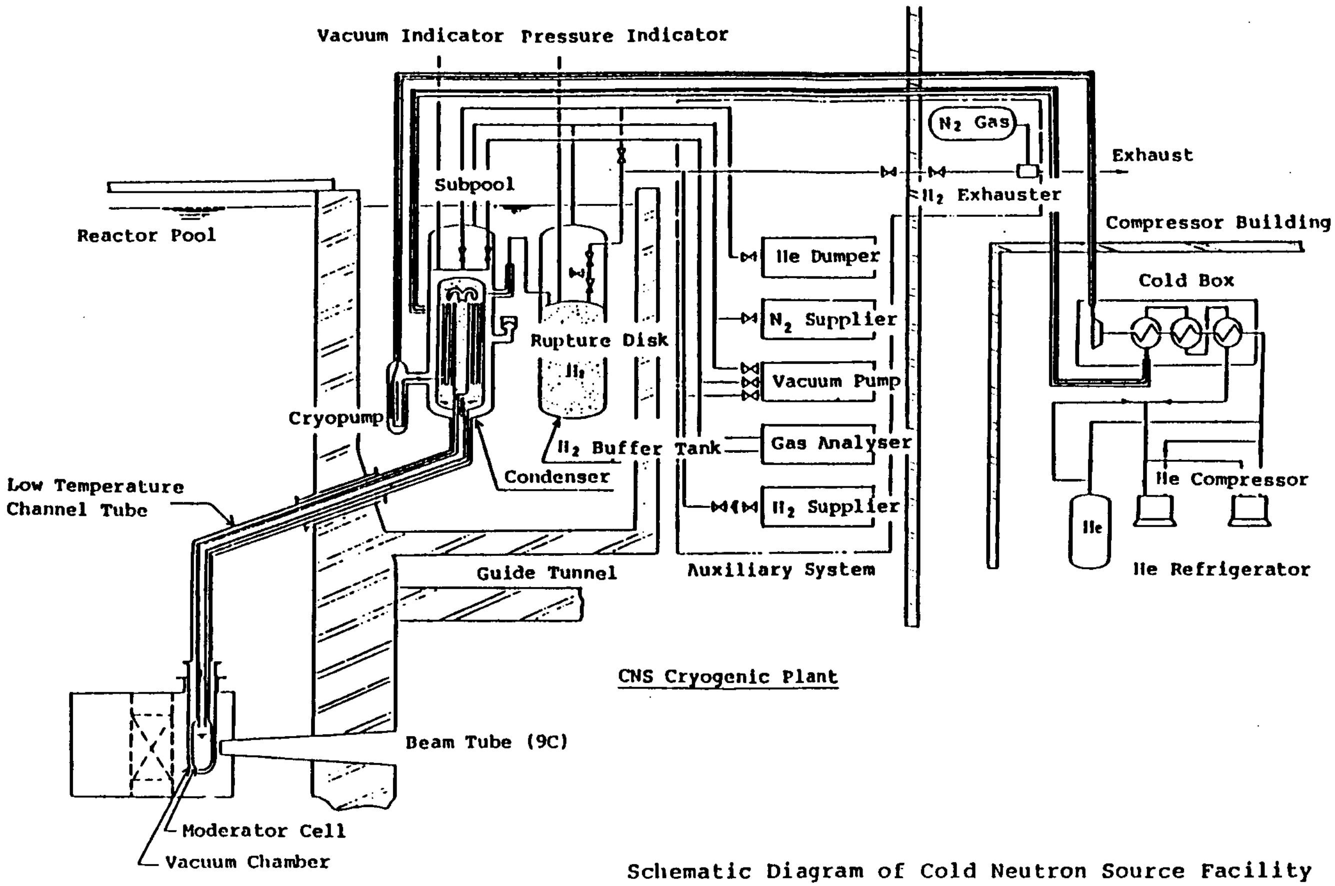
Horizontal Section of Reactor Core



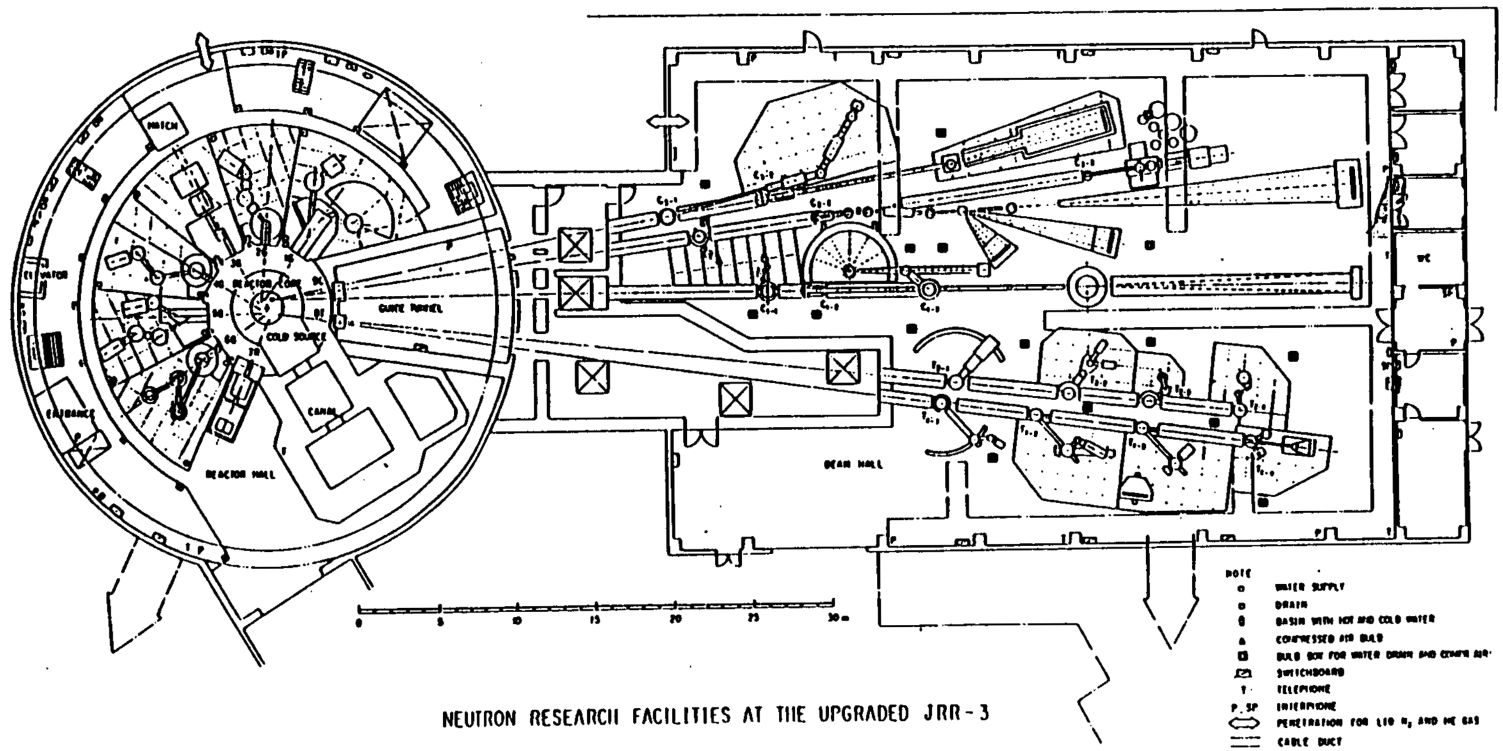
Flow Diagram of Cooling Systems



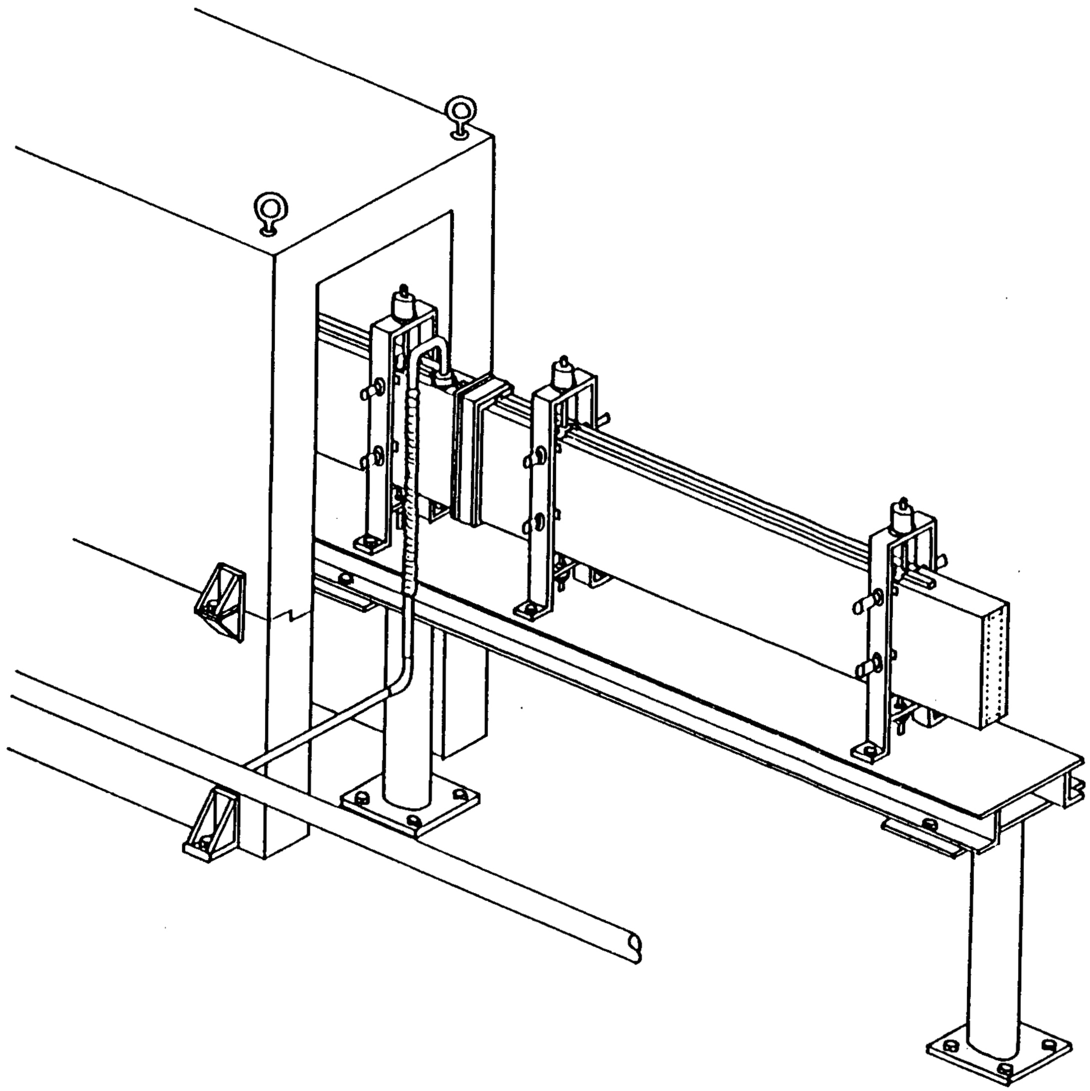
Description of ^{16}N Decay Tank



Schematic Diagram of Cold Neutron Source Facility



NEUTRON RESEARCH FACILITIES AT THE UPGRADED JRR-3



Isometric View of Neutron Guide Tube

JRR－3 Neutron Guide Tubes

	CHARACTERISTIC WAVELENGTH (A)	BEAM CROSS-SECTION (cm)	CURVATURE (m)	LENGTH (m)	NEUTRON FLUX *) ($cm^{-2}s^{-1}$)
T1	2	2 X 20	3337	60	2.1×10^8
T2	2	2 X 20	3337	59	2.1×10^8
C1	4	2 X 12	834	31	3.1×10^8
C2	4	2 X 12	834	51	2.8×10^8
C3	6	2 X 12	371	31	2.3×10^8

*):calculated results for perfect guides

total length = 232 m

NEUTRON MIRROR : NATURAL NICKEL COATED BOROSILICATE GLASS

GUIDE TUBE UNIT : 85cm length

FABLICATION ERROR OF UNIT : 0.005 mm (average)

INSTALLATION ERROR (average)

IMPERFECTION SPACIAL ALIGNMENT : 0.008 mm

IMPERFECTION ANGULAR ALIGNMENT : 5.3×10^{-6} rad

STATUS OF THE UNIVERSITY OF MISSOURI-COLUMBIA RESEARCH REACTOR UPGRADE

J. C. McKibben, C. B. Edwards, Jr.,
W. A. Meyer, Jr., and S. S. Kim

Status of the University of Missouri-Columbia Research Reactor Upgrade

J. Charles McKibben
Chester B. Edwards Jr.
Walter A. Meyer Jr.
MU Research Reactor
Columbia, Missouri USA

Dr. Soon Sam Kim
Idaho Nuclear Engineering Laboratory
Idaho Falls, Idaho USA

ABSTRACT

The University of Missouri-Columbia (MU) Research Reactor Facility staff is in the process of upgrading the operational and research capabilities of the reactor and associated facilities. The upgrades include an extended life aluminide fuel element, a power increase, improved instrumentation and control equipment, a cold neutron source, a building addition, and improved research instrumentation and equipment. These upgrades will greatly enhance the capabilities of the facility and the research programs. This paper discusses the parts of the upgrade and current status of implementation.

I. INTRODUCTION

The MU Research Reactor (MURR), the highest steady state power university research reactor, is enhancing the research, education, and service capabilities of the facility through a five part upgrade. The existing reactor is a 10 MW, pressurized loop type, already operating at a greater than 90 percent availability to support demand from researchers and industry. The upgrade is focused to meet the increasing demand for more neutrons and higher specific activity radioisotopes, especially for the biomedical community, cold neutrons, and expanded laboratory and office space. This kind of upgrade is especially important at a university research reactor to help meet the need for increased education and training in the nuclear sciences in the USA. The upgrade includes an extended life fuel element, power increase, instrument and control and electrical system upgrade, cold neutron source (CNS), building addition, and modernization of research instrumentation and equipment.

MURR has been working on this upgrade since 1983, when work started on the fuel element conceptual design and power upgrade feasibility study. The fuel design and safety analyses were performed by the MURR staff. They are also working on the power upgrade safety analyses with support from the MU Nuclear Engineering Program faculty and students. Stone and Webster Engineering Corporation was architect/engineer for the conceptual design, completed in 1987, of the fluid system modifications for power increase, instrumentation and control, and electrical package and the building addition. Stone and Webster acted as MU's agent in determining the CNS options. In March 1989, Sverdrup Corporation was selected architect/engineer for the detail design of the building addition with the design to be completed after a new MURR Director is hired.

II. DISCUSSION

Extended Life Fuel Element

MURR based the new fuel element design on the extended life aluminide fuel (ELAF)(1) with the goal of reducing the fuel cycle cost and providing a core capable of operating at a higher power level. The ELAF is aluminide fuel with UAl_2 the primary constituent in the UAl_X powder with 50 volume percent UAl_X powder in the fuel meat. This gives a loading density of 3 gm/cm^3 of high enriched uranium (HEU). The advantage of the ELAF is the coupling of the high uranium loading with the ability to maintain good fuel plate integrity in high burnup conditions. Test plates were run in the Advanced Test Reactor (ATR) in Idaho with peak burnups of 3.0×10^{21} fissions/cm^3.(2)

The uranium loading per unit volume is varied to flatten the radial power density, with a total element loading of 1.244 kg of U-235. Boron carbide is used in some of the inner and outer plates as a burnable poison. The combination of heavy uranium loading, high burnup limit, and flattened power density provide the characteristics needed to more than double the megawatt days (MWD) of energy obtained per element to 300 MWD.(3-4)

Since submitting the fuel license amendment in September 1986, MURR has answered three sets of questions concerning the requested license amendment to cover the ELAF fuel. The last set (submitted February 1989) was strictly concerning reactivity transient analysis and some clarification of wording in previous submittals. The Nuclear Regulatory Commission (NRC) asked four more question in November 1989 requesting MURR modify the requested license amendment to better fit the analysis completed for the first three NRC question sets. Response to these latest questions will be completed in March 1990, and the NRC's approval for use of ELAF may be given shortly thereafter. If the DOE university fuel budget can support the startup cost for fabrication of a new fuel design, the MURR ELAF fuel element will be used starting in 1990.

Power Increase

The goal of the power upgrade is to operate the reactor at the maximum safe technical limit as is often done in European reactors. With its flattened power distribution, the new fuel element design provides the greatest increase in the safe technical limits. The physical dimensions are identical to the current fuel element, requiring no change in the geometry of the reactor core.

The only modifications planned for operating at a higher power are the slight increases in operating pressure and primary coolant flow rate, and installation of new heat removal equipment. The upgrade power level has been targeted at 30 MW. The safety analyses performed have shown no problem operating at 30 MW.[5-9] However, the loss of coolant accident results are very sensitive to changes in power level around 30 MW.[10] To show confidence in the RELAP5/M0D2 analysis for a low pressure/low temperature fuel plate type research reactor, MU's NE department and MURR submitted a proposal to DOE for a benchmark experiment. This was funded in November 1988 and the work is in progress.[11]

Reactor Instrumentation & Control and Electrical System Upgrade

The reactor instrumentation and control (I&C) system was reviewed to determine necessary modifications and recommendations for equipment replacement and improvements. An evaluation was performed both for reliability/availability improvements and for reactor upgrade requirements.

During FY89, the first phase of I&C/electrical upgrade began. A small building addition adjoined to the facility houses the new 275 KW Cummins diesel emergency generator. It includes an empty bay sized for the future installation of a 2500 KW electrical substation required for the power upgrade, CNS, and expansion of facilities. The electrical upgrade also included the installation of a Solidstate Controls Inc. (SCI) uninterruptable power supply (UPS) to handle the reactor I&C system. The emergency generator was put in service in August 1989, the UPS in November 1989. Upgrade of two I&C subsystems was started during Spring 1989--the area radiation monitoring system (ARMS) and the exhaust stack radiation monitor. The installation of an Eberline analog ARMS will be completed in March 1990 and of a Nuclear Measurements Corporation (NMC) stack monitor in May 1990. MURR is planning to complete the next task in the I&C upgrade during 1990 by replacing the nuclear instrumentation systems (replace six 1960 model channels with three 1990 model channels).

Cold Neutron Source

To meet the rapidly increasing demand for long wavelength neutrons, the MURR upgrade includes adding a cold neutron source (CNS) facility. The CNS will be in the reflector near the core, and this section of the reflector will be modified to enhance the CNS effectiveness. With the reactor operating at 30 MW, the thermal flux around the CNS would be approximately 3×10^{14} neutrons/cm^2 sec with gamma heating of 3.6 watts/gram. A small part of the analyses have been completed to evaluate what combination of reflector materials (beryllium, D_2O, bismuth, etc.) and which CNS design will give the optimal combination of long wavelength neutron beam intensity, capital and operating costs, and minimum impact on other users of the reactor.(12) With 30 MW reactor power, a lower core position, and a CNS, the intensity of 8 Å (8×10^{-10}m) wavelength neutrons could increase by a factor of 80.

The MURR staff also evaluated CNS designs from three suppliers of this equipment, Technicatome, Interatom, and AECL. As part of this evaluation, Stone and Webster assisted MU in obtaining conceptual design information and cost data from the three vendors. The design information requested was for a total package including cryogenic and control systems that would be applicable to installation and operation of a CNS at MURR's beamport "E" position. The information was to be based on previous experience, and to include the design parameters and estimate of the expected capability and performance characteristics.

Facilities Expansion

The MURR upgrade includes a 44,000 sq ft (4087 m^2) building addition, the conceptual design of which was completed in January 1987. The addition centers around a cold neutron guide hall that allows a 60 m flight path for experiments using the CNS. Several electronics laboratories will be relocated in the new addition, freeing up their current location (laboratories with fume hoods) for expanding the radioisotope applications research. The building will also contain a 1600 sq ft (148 m^2) counting room, a clean room, culture room and sample preparation laboratory for neutron activation analysis (NAA).

The State of Missouri appropriated $250,000 for planning of the new addition. Sverdrup Corporation was selected to work on the detail design of this facility in March 1989, but the design will be worked after the new MURR Director is hired. With the July 1989 administrative shift of MURR to the MU campus from the University of Missouri System level, the 1987 conceptual design will be restudied to see if it provides for the best coupling of the combined strengths of MURR and the research strengths of other MU departments.

III. CONCLUSIONS

MURR has designed a fuel element that will cut the fuel cycle cost at least in half and provide a flatter power distribution allowing for a power increase up to 30 MW. A power increase to 30 MW will provide a peak thermal flux of 1.8×10^{15} neutrons/cm^2 sec in the flux trap and a beam port flux of 3.5×10^{14} neutrons/cm^2 sec.

The cold neutron source coupled with other improvements will provide an increase up to a factor of 80 in 8 Å wavelength neutrons, which are needed for the fastest growing area of neutron scattering research. The laboratory space available to radioisotope applications for finding ways to cure cancer will at least double. The NAA group will have a major increase in facilities to aid in their trace element research in areas such as nutrition studies, epidemiology studies, archaeometry studies, etc. The 44,000 sq ft (4087 m^2) building addition will provide a guide hall and new research facilities to meet the demand for increased utilization of the reactor. Estimated to cost $23,000,000 in FY88 dollars, the total project will take four to five years to complete.[13-14] Missouri Governor Ashcroft has pledged the State will provide one-third of the project cost when the matching two-thirds are obtained. The upgrade will expand the world-class capabilities of the best research reactor located on a university campus, and will enable the USA to educate and train the researchers needed by national laboratories and the proposed Advanced Neutron Source.

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THE REACTOR AND COLD NEUTRON FACILITY AT NIST

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The Reactor and Cold Neutron Facility at NIST

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ABSTRACT

The National Institute of Standards and Technology Reactor (NBSR) is a 20 MW research reactor located at the Gaithersburg, MD site, and has been in operation since 1969. In the reactor hall there are 26 experimental facilities which are used for materials science, chemical analysis, nondestructive evaluation, neutron standards work, and irradiations. The reactor had built into it from the beginning a provision for a large volume cold neutron moderator. Utilizing this capability, the Department of Commerce and NIST have begun a project to develop a major national facility for cold neutron research, the CNRF. This $30M project will, when fully completed, provide fifteen new experimental stations with capabilities currently unavailable in this country. It will be operated as a National User Facility, open to all qualified researchers on the basis of scientific merit. One-third of the experimental stations will be provided by sources outside of NIST.

The NBSR

The reactor is D_2O cooled and moderated; the core is comprised of thirty, enriched-uranium fuel elements of a unique, split-core design, in which beam tubes "look" at a 17 cm gap between fuel-element halves. The reactor operates 24-hours a day on monthly cycles, followed by approximately a week of shutdown for refueling and maintenance. Specifications and reactor capabilities are listed in Table 1.

The experimental facilities in the reactor hall are allocated among the following activities:

- neutron scattering and diffraction; neutron radiography;
- trace analysis and depth profiling; nondestructive evaluation;
- neutron standards development; fundamental neutron physics;
- long-term irradiations and isotope production.

A plan view of the reactor hall is shown in Figure 1. The NBSR utilizes the flux available in a very efficient manner through the incorporation of relatively short core-to-instrument distances and large-diameter beam tubes. As a reference, the flux on the sample at the BT-4 and BT-9 spectrometers is 10^7 n/cm^2-s, measured at 14 meV incident energy, with 40′ collimation before and after the monochromator. Programs and instrumentation of the thermal beams have been described recently.[1]

The CNRF

Ground breaking for this laboratory took place in November 1987; it was dedicated by Secretary of Commerce C. William Verity in January 1989. The guide hall (30 m wide by 61 m long) and associated office and laboratory space has more than tripled the area available for neutron beam research, much of which is reserved for guest researchers.

This new national user facility utilizes the cold neutron source indicated in Figure 2. This source is a block of D_2O ice (with 8% H_2O added) cooled to 30-40 K by recirculating helium gas. The gas is circulated by a compressor through a refrigerator capable of removing 1 kw of heat at 25 K. The ice block is 36 cm in diameter by 22 cm long, with an 18 cm diameter by 10 cm long reentrant hole to increase the cold neutron flux. Integrated cold neutron flux (λ>3.95Å) is expected to be 1.8×10^8 n/cm^2-s at the entrance to the guide hall. The increase in neutron flux with the present cold source has been measured to be a factor of five, source full to empty, in the 4-8Å range.[2] An advanced liquid hydrogen source is now being designed for future installation.

The cold source will be viewed by eight neutron guide tubes (NG 0-7), one of which ends inside the reactor hall. The neutron guides consist of a thin coating (1000 Å) of Ni deposited on optically flat glass, of 15x6 cm^2 cross-section. The initial three guides will be coated with Ni^{58} to increase the useable solid angle by 30% over normal nickel. There is also a provision for installing supermirror guide elements on subsequent guides to further enhance available flux. The guides are completely evacuated to reduce neutron losses due to air scattering and go through holes in the reactor confinement building wall, with shutters provided at the wall to allow work on the beam lines in the guide hall while the reactor is operating.

On completion in 1993, the CNRF will include up to 15 new cold neutron instruments, the categories of which are listed in Table 2. In Figure 3 is shown the planned layout for the main floor of the new construction, and the first three guides to be instrumented. Also indicated is the neutron depth-profiling instrument which will be on NG-0, the new beam-line in the reactor hall. Installation is staged to minimize reactor down-time. Some features of particular interest are described below for the initial instruments.

Two 30m SANS instruments are under construction: the first to be installed (1990) will be an NIST/Exxon/U. of Minn. instrument on NG-7; the second, on NG-3, will be one of two instruments in the NSF/NIST Center for High Resolution Neutron Scattering (CHRNS). The NIST/Exxon/U. of Minn. SANS instrument will be the first to use a doubly curved mirror as a focal element in a long flight path to provide angular resolution and beam intensities which compare favorably with any SANS instrument in the world. In addition, each instrument will have provision for two large-area position-sensitive detectors to extend the angular range to cover both the small and intermediate angle regions. An optional feature for these instruments will be the ability to utilize a polarized neutron beam to study materials with magnetic constituents.

A neutron reflectometer to probe surfaces and interfaces in a wide variety of materials will be installed in early 1991. The low background and high intensity of cold neutrons in the new guide hall will permit the measurements of reflectivities down to levels of $\approx 5 \times 10^{-7}$. The proposed instrument will have the capability to produce polarized neutrons to study magnetic and superconducting materials. The sample geometry will be horizontal to facilitate the study of liquid samples. There will also be provisions to do grazing-angle surface neutron diffraction experiments, the feasibility of which was recently demonstrated at NIST.

Two experimental stations for chemical analysis are planned. A neutron depth profiling (NDP) instrument is being designed to take advantage of the factor of twenty or more increase in signal intensity expected over the existing thermal neutron beam facility. The new instrument will include several new features.

Neutron capture prompt-gamma-ray activation analysis has been developed at NIST and elsewhere as a reliable, often uniquely sensitive, method of elemental analysis with wide application in materials science, geochemistry, and environmental monitoring. Through a combination of greater neutron intensity, lower gamma-ray background, and advanced detectors and coincidence-counting instrumentation, the new instrument on NG-7 will provide 100 times more sensitivity for this method than at any thermal neutron instrument in the world.

Two of the stations will be devoted to fundamental neutron physics, including neutron interferometry. A variety of interferometer geometries will be investigated for several different experiments. The anticipated experimental investigations will include long baseline neutron interferometry with separated crystals, delayed choice experiments and, possibly, a neutron Michelson-Morley experiment. The other experimental station will provide an intense cold neutron beam for basic investigations in nuclear and particle physics. Anticipated experiments on this beam include studies involving neutron decay, nucleon-nucleon weak interactions and tests of basic symmetry principles.

Not shown in Figure 3 is the new cold neutron triple-axis spectrometer to be installed on NG-5. It is expected to be operational early in 1991.

Development of NG-1, 2, 3 and 4 will proceed as quickly as possible and will include the NSF/NIST 30m SANS and the NSF/NIST spin-polarized inelastic neutron scattering spectrometer (SPINS). The latter will be a triple-axis type instrument, but with high resolution and high intensity achieved through the use of supermirror polarizers and an energy-dependent flipper. Other instruments are listed in Table 3, among which will be a conceptually-new, very high-resolution multichopper time-of-flight spectrometer and a state-of-the-art back-reflection spectrometer.

The National-User Facility

Administratively, this facility is located within NIST's Materials Science and Engineering Laboratory, which is comprised of the Polymers, Ceramics, Metallurgy, Fracture and Deformation Divisions, and the

Center for Nondestructive Evaluation, as well as the Reactor Radiation Division. As indicated in Figure 3, office and laboratory space for users of the facility is provided in an addition to an existing office/lab wing of the reactor building. This space provides 36 additional offices for users and staff, and 12 laboratories for sample preparation and equipment maintenance.

CNRF facilities are divided into two classes: CNRF instruments and Participating Research Team (PRT) instruments. For CNRF instruments, 2/3 of available time will be scheduled by the Program Advisory Committee (PAC) and 1/3 reserved for NIST use (out of which proprietary research is allocated). For PRT instruments, 1/4 of available time will be scheduled by the PAC and 3/4 reserved for PRT members. As mentioned above, two of the instruments at the CNRF are being funded by the NSF as a Center for High Resolution Neutron Scattering (CHRNS). The NSF-funded portion of CHRNS will be scheduled entirely by the PAC, aside from some time for instrument improvement, "breakthrough" experiments, and a small allotment of time for instrument-responsible scientists. Other PRT members include Exxon Research and Engineering, Eastman Kodak, AT&T Bell Labs, Sandia Labs, and the University of Minnesota.

Full cost recovery will be required for all proprietary research, whether performed on a CNRF instrument or by a PRT member during PRT-reserved time. No fees will be charged for non-proprietary research. Unless formally described as proprietary research, all research is required to be published in the open literature or made accessible in the public domain.

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Table 1. NBSR Specifications

Power	20 MW
Peak Thermal Neutron Flux	4×10^{14} n/cm^2-sec
Peak Fast Neutron Flux	10^{14} n/cm^2-sec
Core:	
Radius	55 cm
Height	74 cm
Loading	6 kg U^{235}
Life	30-35 weeks
Moderator/Coolant	5×10^4 L D_2O
Fuel Elements:	
Type	Split MTR curved plate
Enrichment	93% in U^{235}
Shielding:	
Thermal	5 cm lead and 20 cm iron
Biological	1.8 m magnetite concrete
Neutron Ports	
Beam tubes:	
Radial	4 with 15 cm diam.
Radial	3 with 13 cm diam.
Radial (truncated)	2 with 13 cm diam.
Tangential	2 with 10 cm diam.
Vertical	1 with 10 cm diam.
Cryogenic Facility	8 15x6 cm^2 guides viewing cold source
Thermal Column	137x132x94 cm^3 graphite
Vertical Thimbles:	
In-core	6 with 9 cm diam.
In-core	4 with 6 cm diam.
In reflector	7 with 9 cm diam.
Rabbit tubes (2.5cm IDx7.5cm long):	
Near-core	3 at (3-10)$\times 10^{13}$ n/cm^2-s
Thermal column	1 at 3×10^{11} n/cm^2-s

Table 2. CNRF Instrumentation

Materials Structure
- SANS
- Reflectometer
- Grazing-Incidence Diffractometer

Materials Dynamics
- Triple-Axis Spectrometer
- Spin-Polarized Inelastic Neutron Spectrometer (SPINS)
- Time-of-Flight Spectrometer
- Back-Reflection Spectrometer
- Spin-Echo Spectrometer

Chemical Analysis
- Depth-Profiling Facility
- Prompt-Gamma Facility

Neutron Physics
- Neutron Interferometer
- Fundamental Physics Station

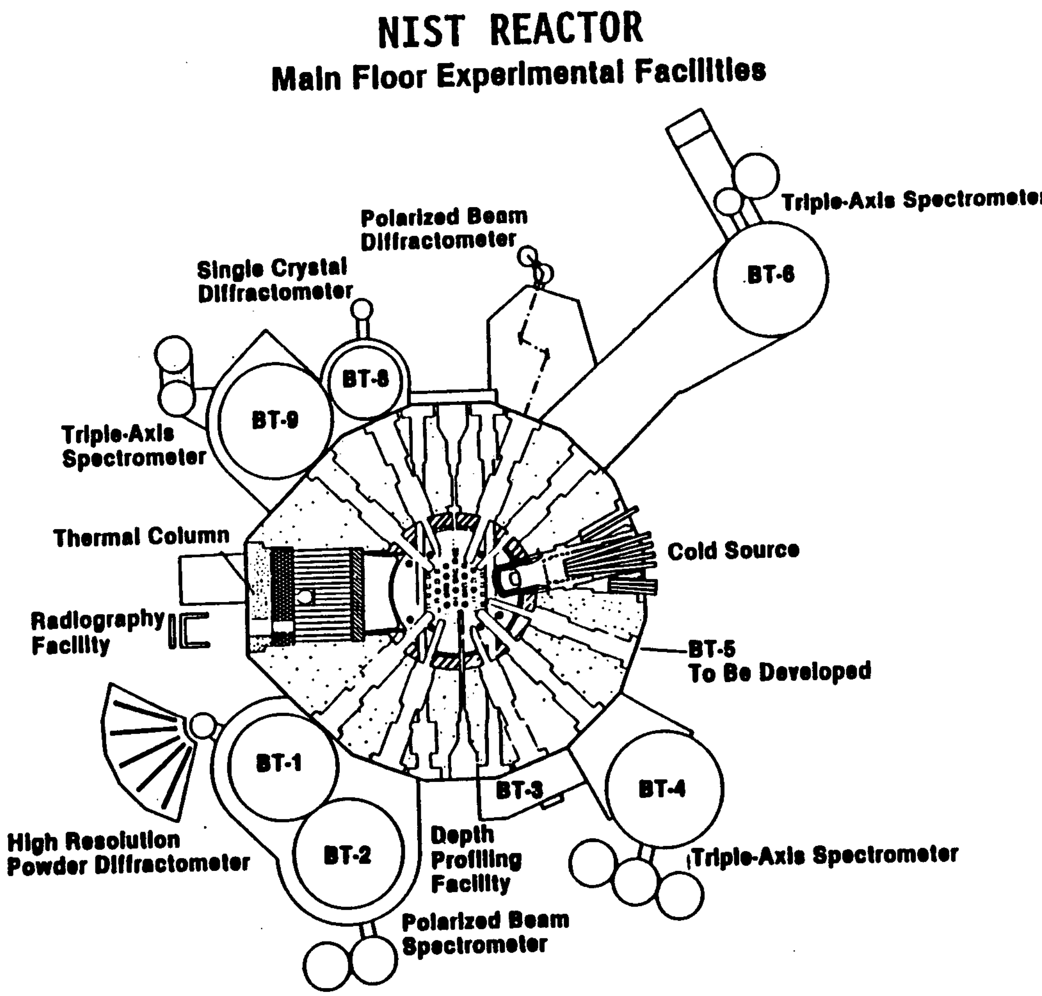

Fig. 1. The floor plan of the NIST Reactor hall.

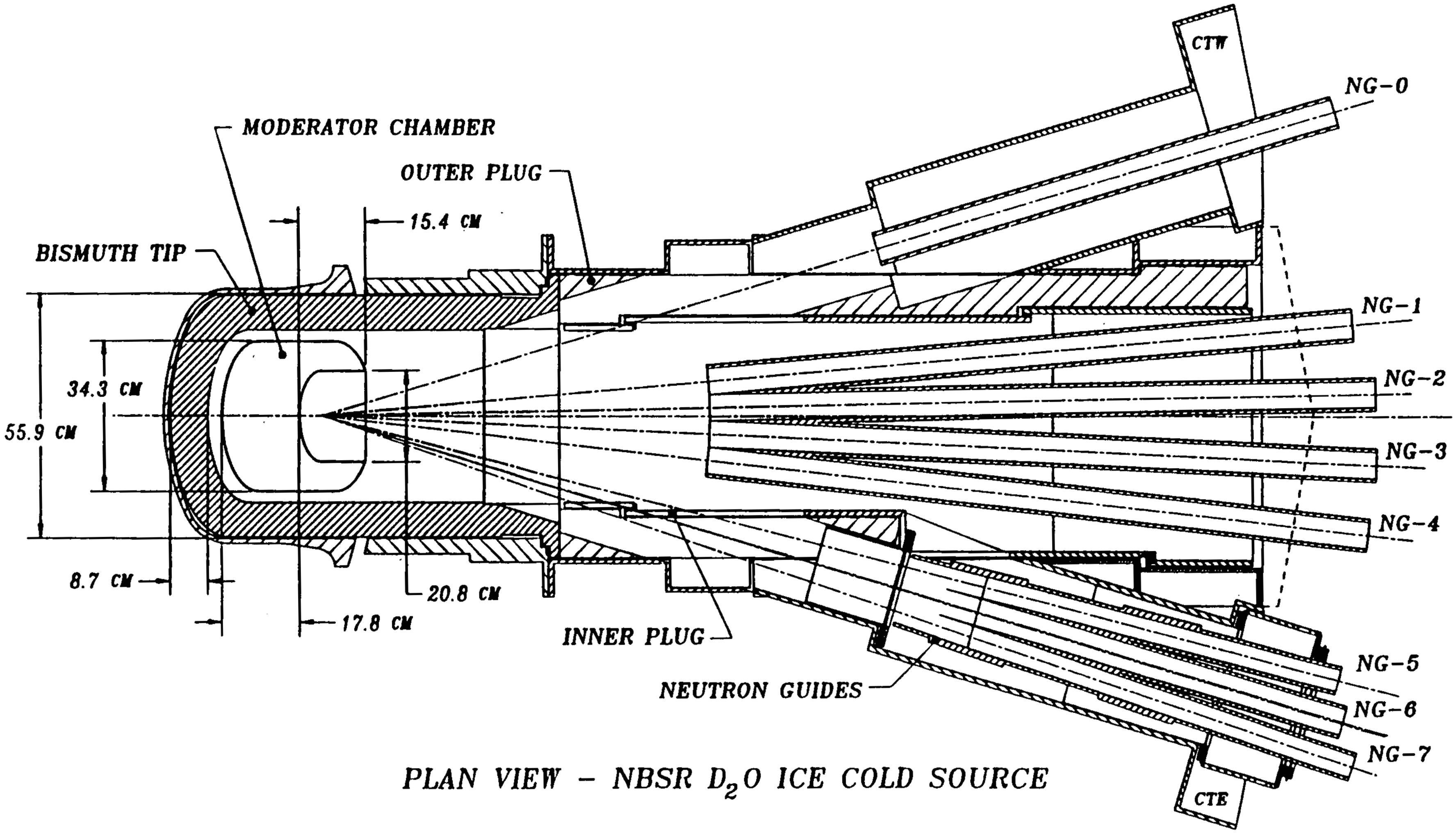

Fig. 2. The NBSR cold source.

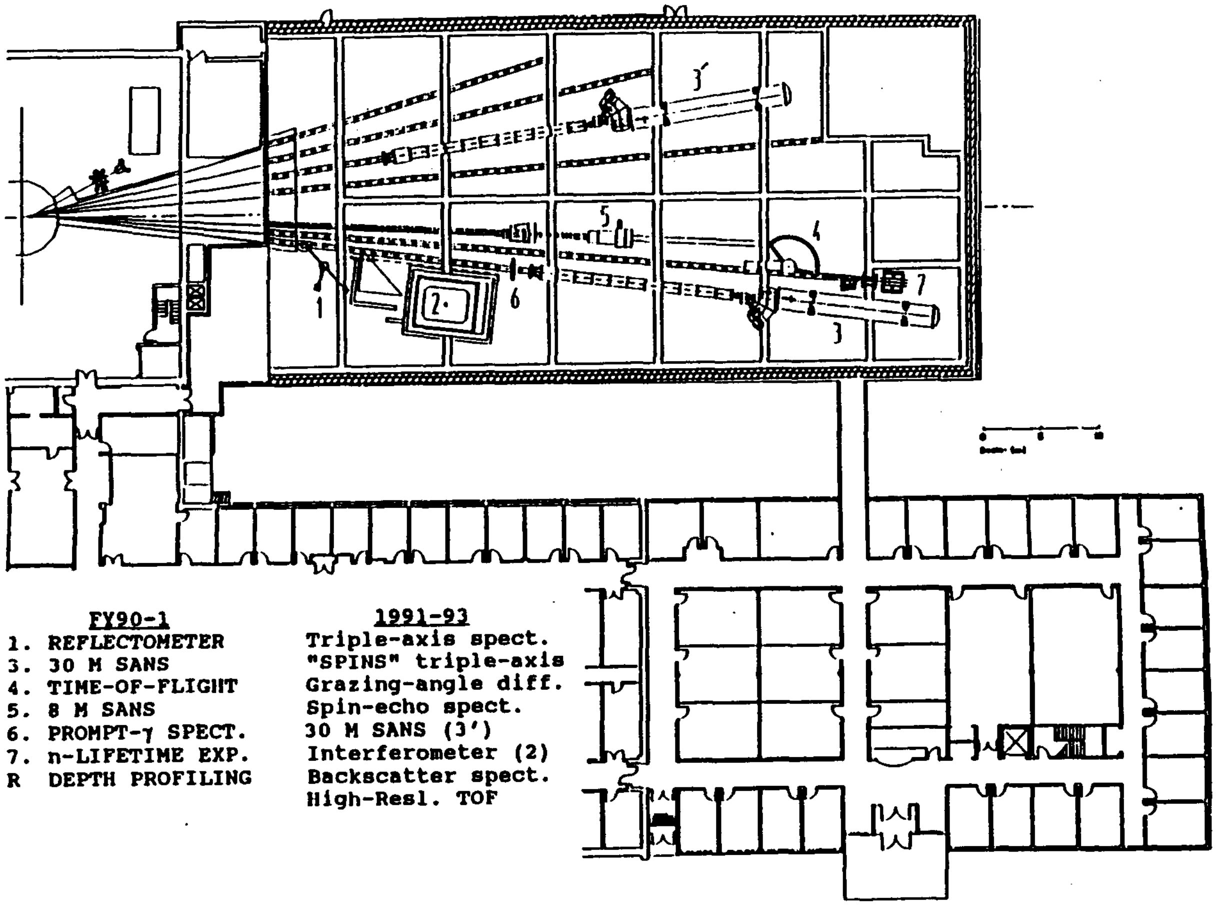

Fig. 3. The floor plan of the CNRF with instrumentation indicated for the initial three guides.

UPGRADE OF MATERIALS IRRADIATION FACILITIES IN HFIR

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UPGRADE OF MATERIALS IRRADIATION FACILITIES IN HFIR*

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ABSTRACT

Modifications have been made to the High Flux Isotope Reactor (HFIR) which permit the operation of instrumented irradiation capsules in the target region, and more and larger capsules in the removable beryllium region. As many as two instrumented target capsules can now be accommodated and positions for up to eight 46-mm-diam instrumented capsules are now available in the removable beryllium region. One instrumented target capsule has already been irradiated and new capsules have been prepared for irradiation in the removable beryllium region.

1. INTRODUCTION

The High Flux Isotope Reactor (HFIR) is a pressurized, light-water-cooled, beryllium-reflected, 85-MW reactor. The HFIR was designed for the production of isotopes, particularly transuranium isotopes. This production requires high thermal and epithermal fluxes; indeed, the HFIR target region (the cylindrical space inside the two concentric annular fuel elements) has the highest steady-state thermal neutron flux in the world. The relatively high reactor power and power density leads to a high fast neutron flux near the core, so that the HFIR is also used for materials irradiation experiments.

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While the HFIR had outstanding neutronics characteristics for materials irradiations, some relatively minor aspects of its original mechanical design severely limited its usefulness for that purpose. In 1984, an ad hoc committee was established at the Oak Ridge National Laboratory (ORNL) to "... consider and recommend changes and improvements to the Laboratory's facilities for materials irradiation testing." The committee's report[1] included recommendations for certain modifications to the HFIR that would significantly enhance the number and value of materials irradiation experiments that could be accommodated by the reactor.

The basic improvements that were needed to provide better materials irradiation facilities at the HFIR were in two areas. The highest flux positions in the target region could not be instrumented, and the removable beryllium (RB) positions were few and much smaller than those of general purpose reactors. These deficiencies have been remedied through the HFIR Irradiation Facilities Improvement (HIFI) Project which has provided two instrumented target region facilities and larger and additional RB irradiation positions with straight-line access penetrations through the pressure vessel.

2. DESCRIPTION OF FACILITIES

A general arrangement of the new materials irradiation facilities with typical instrumented target and RB capsules in place is shown in Fig. 1. The characteristics of these facilities are presented in Table 1.

Providing instrumented target facilities required newly designed and fabricated components from the bottom to the top of the reactor "stack." These components included a fuel grid, target holder, target tower, target hole plug, quick-access hatch, rabbit facility U-bend, and several in-pool tools for removing and replacing components. The target tower extends upward from the target region to a quick-access hatch and target hole plug in the pressure vessel lid. The tower houses three guide tubes - one for the hydraulic rabbit facility and the other two for the instrumented target facilities.

With these components modified, at least two small target capsules of 16-mm-diam may be instrumented. The guide tubes in the target tower are large enough such that by

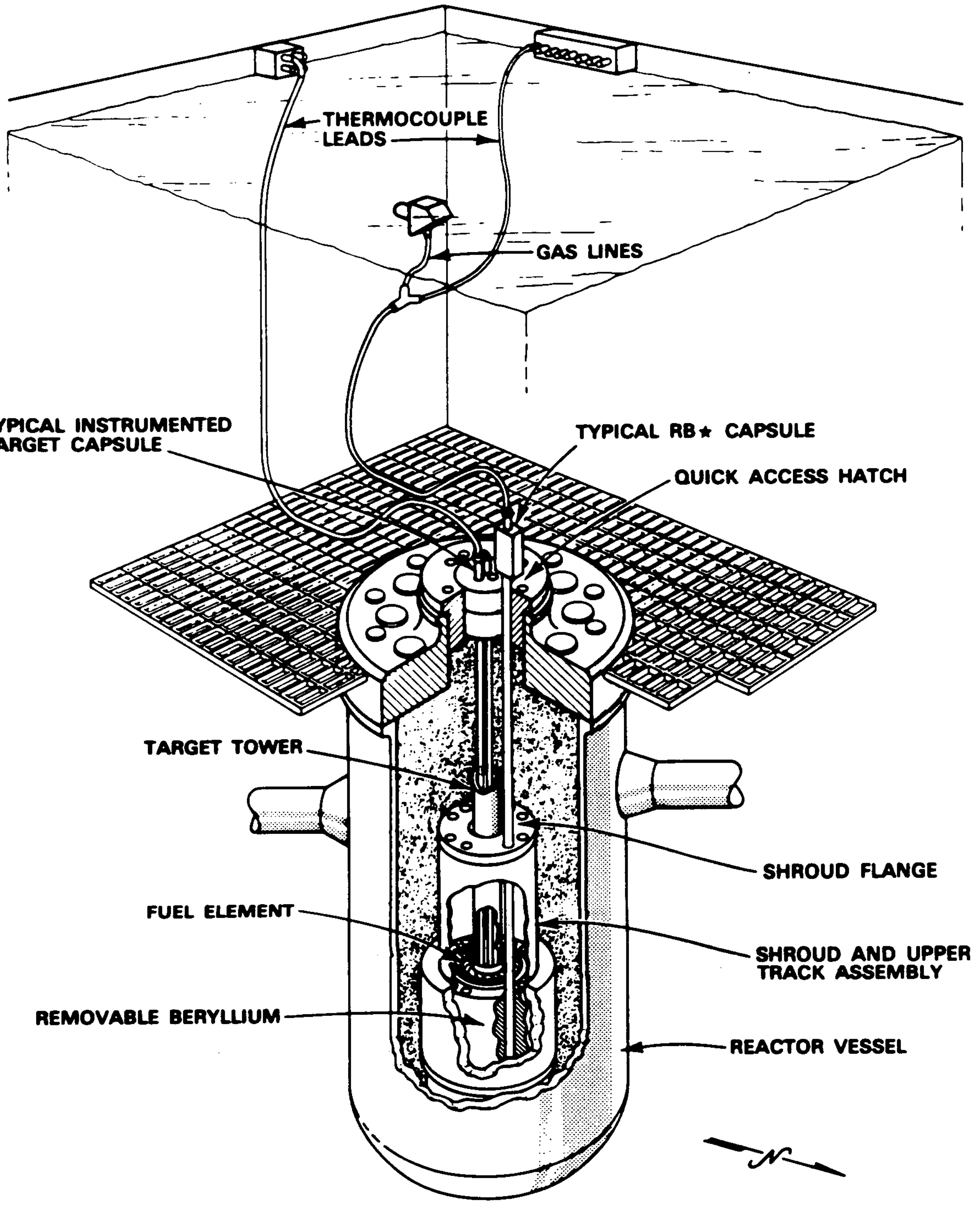

Fig. 1. The High Flux Isotope Reactor (HFIR).

Table 1. Characteristics of primary HFIR materials irradiation facilities

Characteristics	Irradiation position	
	RB	Target
Fast neutron flux, E > 0.1 MeV (10^{18} $m^{-2}s^{-1}$)	6	12
Thermal neutron flux (10^{18} $m^{-2}s^{-1}$)	13	24
Maximum displacements per atom per calendar year, stainless steel	8	25
Gamma heating (W/g SS)	14	47
Typical capsule diameter (mm)	46	16[a]
Typical capsule length (mm)	500	500
Number of available positions	8[b]	>20[c]
Minimum specimen temperature (°C)	60	60
Instrumentation	Yes	Yes[d]
Typical fuel cycle length (days)	25	25

[a]By occupying up to seven positions, 25-mm-diam can be accommodated.

[b]Plus four smaller positions, approximately 12-mm diam.

[c]A total of 37 target positions exist. The number available depends on the number being used for transuranium isotope production.

[d]Two target positions can accommodate instrumented capsules.

occupying up to seven target positions, capsules of up to 25-mm-diam can be accommodated (Fig. 2).

The new RB facilities required a modified design for the reflector, replacing the four 37-mm-diam positions with eight holes, each with a 48-mm diam. This change increased the total experimental volume available within the removable beryllium by a factor of greater than 3. These new positions are referred to as the RB Star (RB*) facilities.

In addition, several components above the beryllium and the core were modified to provide straight-line access to all eight of the RB* positions. The straight-line access permits rotation and vertical relocation of irradiation capsules during the course of an experiment and facilitates experiment interchangeability.

Recording and control equipment is in place to operate two singly-contained capsules and two doubly-contained capsules, with space readily available to expand the equipment for the operation of a total of eight fully instrumented capsules.

3. TYPICAL EXPERIMENTS

Significant funding for the necessary modifications was provided by the Magnetic Fusion Energy (MFE) program. The first instrumented target capsule was the target temperature test (TTT) capsule. It was a part of the US/Japan fusion materials program and was irradiated to determine more accurately the probable temperature in the uninstrumented target capsules previously irradiated as part of that program. Two thermocouple array tubes (TCATs), each having seven thermocouple junctions, were used to measure the centerline temperature of mock specimens. The experiment performed well, and revealed (Fig. 3) that the gamma heating decreases much more rapidly at the ends of the capsule than had previously been thought. A general configuration of the TTT capsule is shown in Fig. 4.

The larger reflector positions permit spectral tailored experiments, similar to those previously performed in the Oak Ridge Research Reactor (ORR), to be performed in the HFIR where fluence can be achieved in about half the time. Indeed, MFE specimens irradiated in ORR spectral tailored capsules have been retrieved and are being

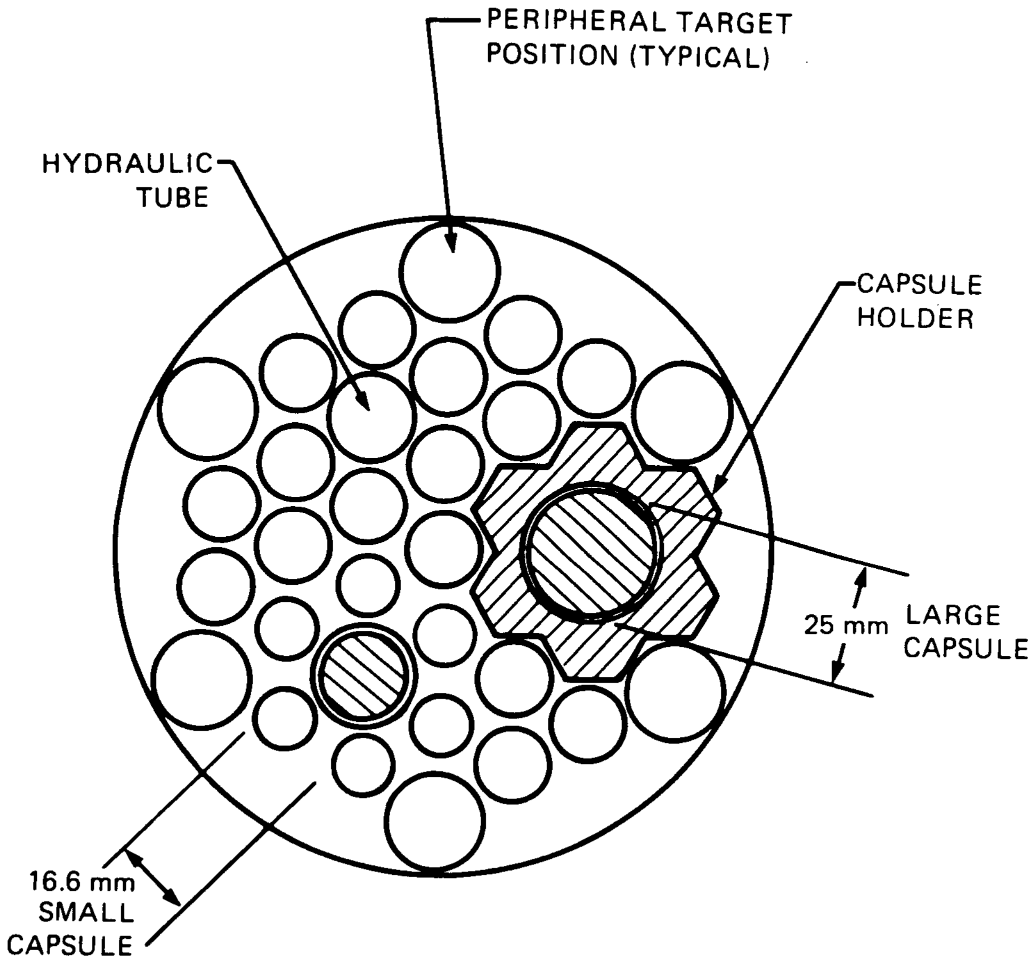

Fig. 2. Instrumented target positions illustrating capability of accommodating 25-mm capsules.

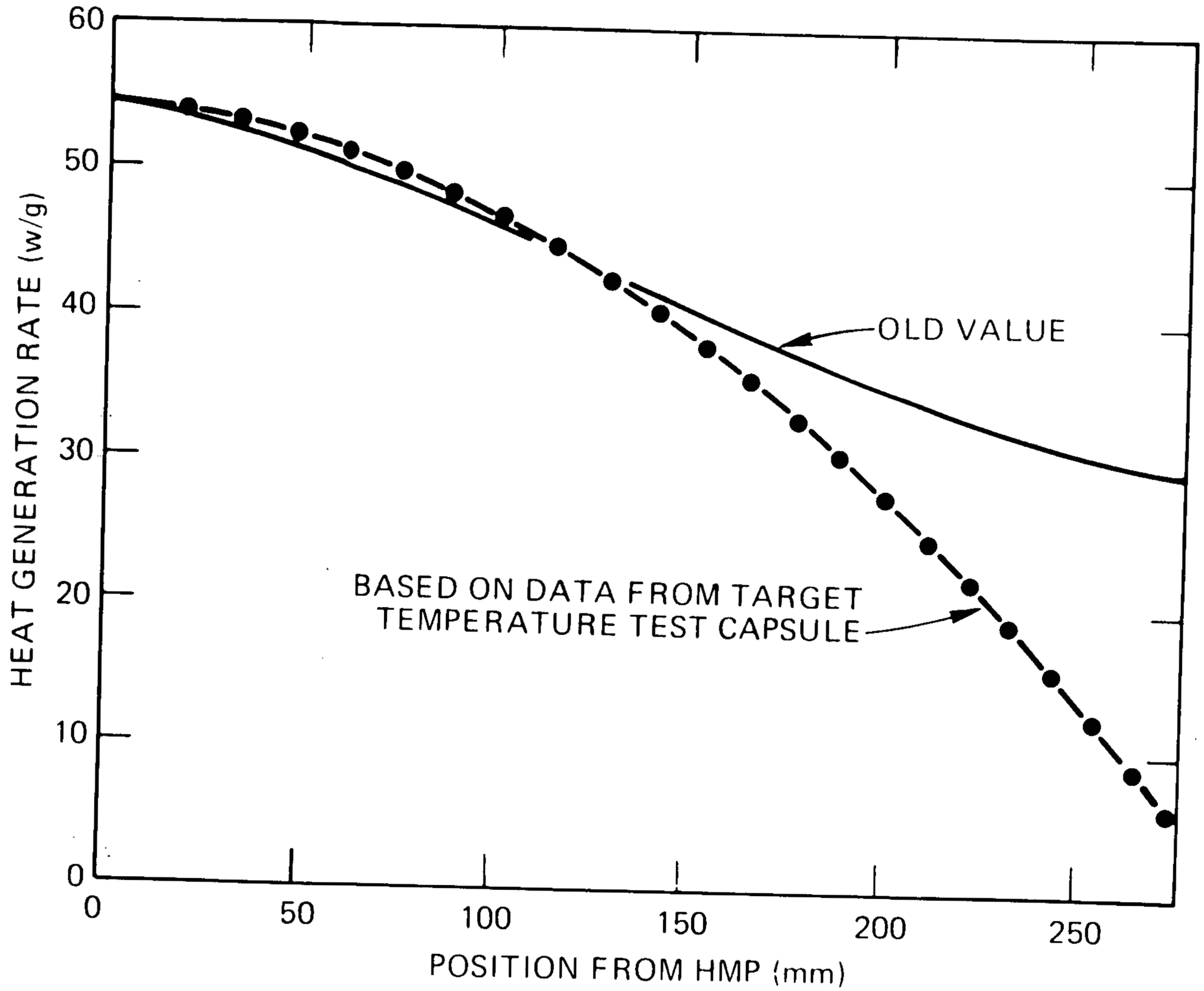

Fig. 3. Heat generation in HFIR target region (data is for 100 MW and should be multiplied by 0.85 to obtain present values).

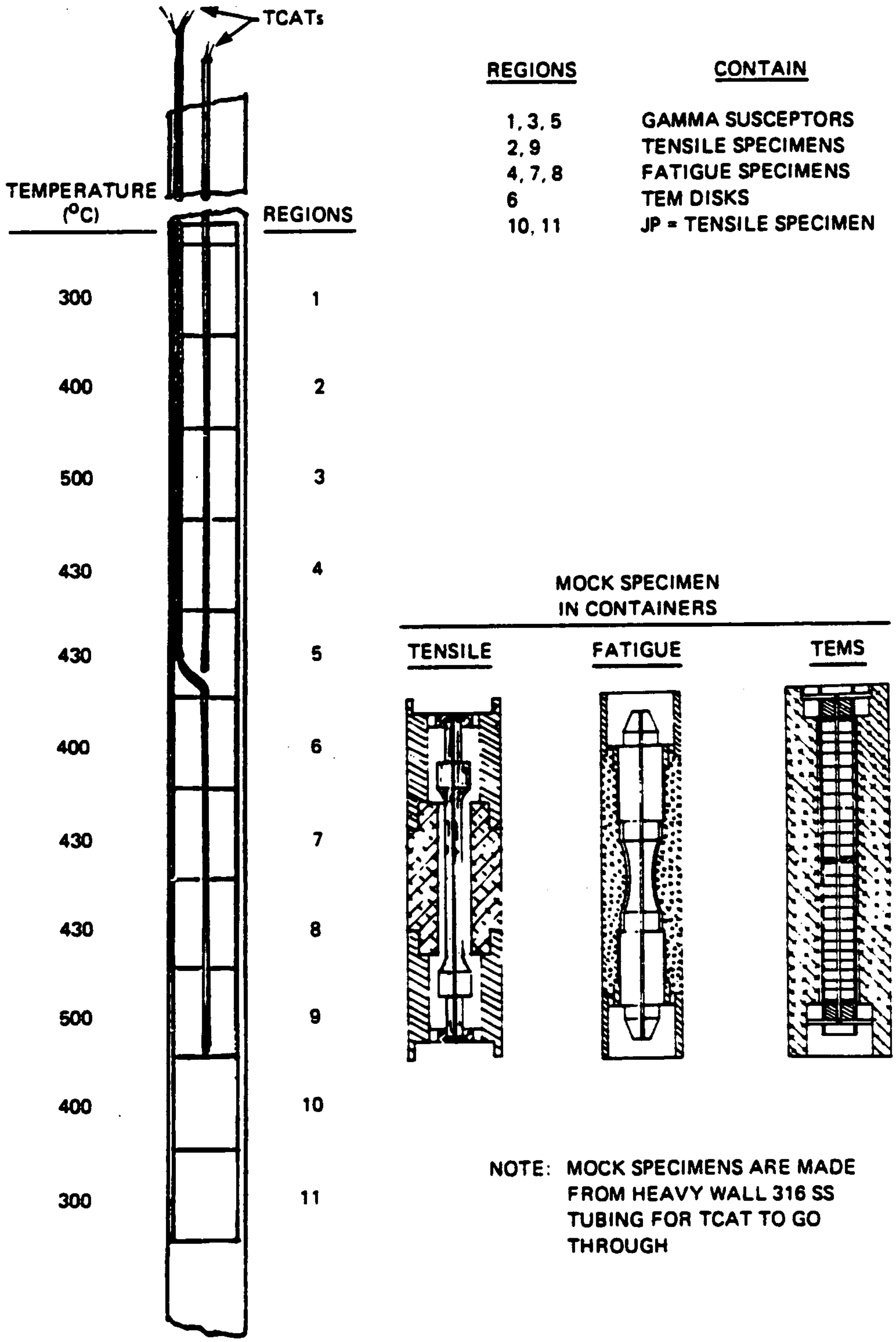

Fig. 4. General configuration of TTT capsule.

reencapsulated for continued irradiation in HFIR RB* positions. Toward this end, series of experiments have been designed to irradiate up to 250 mechanical property specimens each at temperatures of 60, 200, 330, and 400°C.[2] Around each of these experiments will be a 4.2-mm thick Hafnium sleeve which will reduce the thermal neutron flux by about 85%, thus permitting the specimens to receive the same helium production-to-displacements per atom (He/dpa) ratio as is expected in the first wall of a MFE device. Horizontal and vertical cross sections through the 330°C capsule are shown in Fig. 5.

New RB* capsules have also been assembled for the High Temperature Gas-Cooled Reactor (HTGR) program. These will irradiate coated particle fuel compacts in a graphite fuel body. A horizontal cross section through a typical HTGR fuel capsule is shown in Fig. 6.

4. SUMMARY

These new HFIR facilities provide the materials irradiation community with very powerful tools with which to carry on its work. The HFIR should now be considered a world-class materials testing reactor; in the case of the instrumented target capsules, it surpasses any reactor for the magnitude of neutron flux available in instrumented irradiation experiments.

5. REFERENCES

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2. A. W. Longest, J. E. Corum, D. W. Heatherly, and K. R. Thoms, "Design of spectrally-tailored fusion reactor materials experiments in the HFIR RB* capsule irradiation facility," J. Nucl. Mater. 155-157 (1988) 1340-1345.

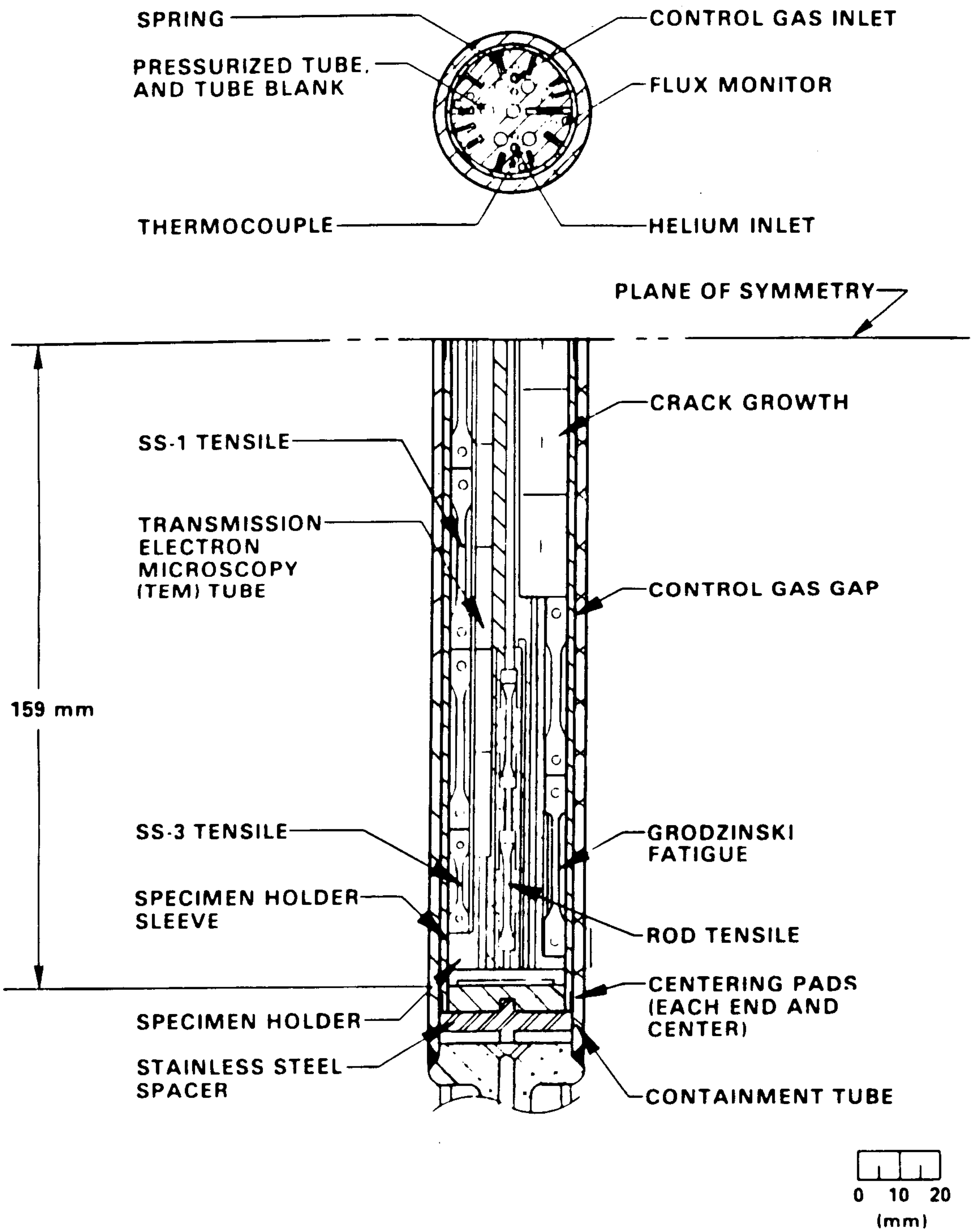

Fig. 5. Lower half of the HFIR-MFE-330J-1 capsule.

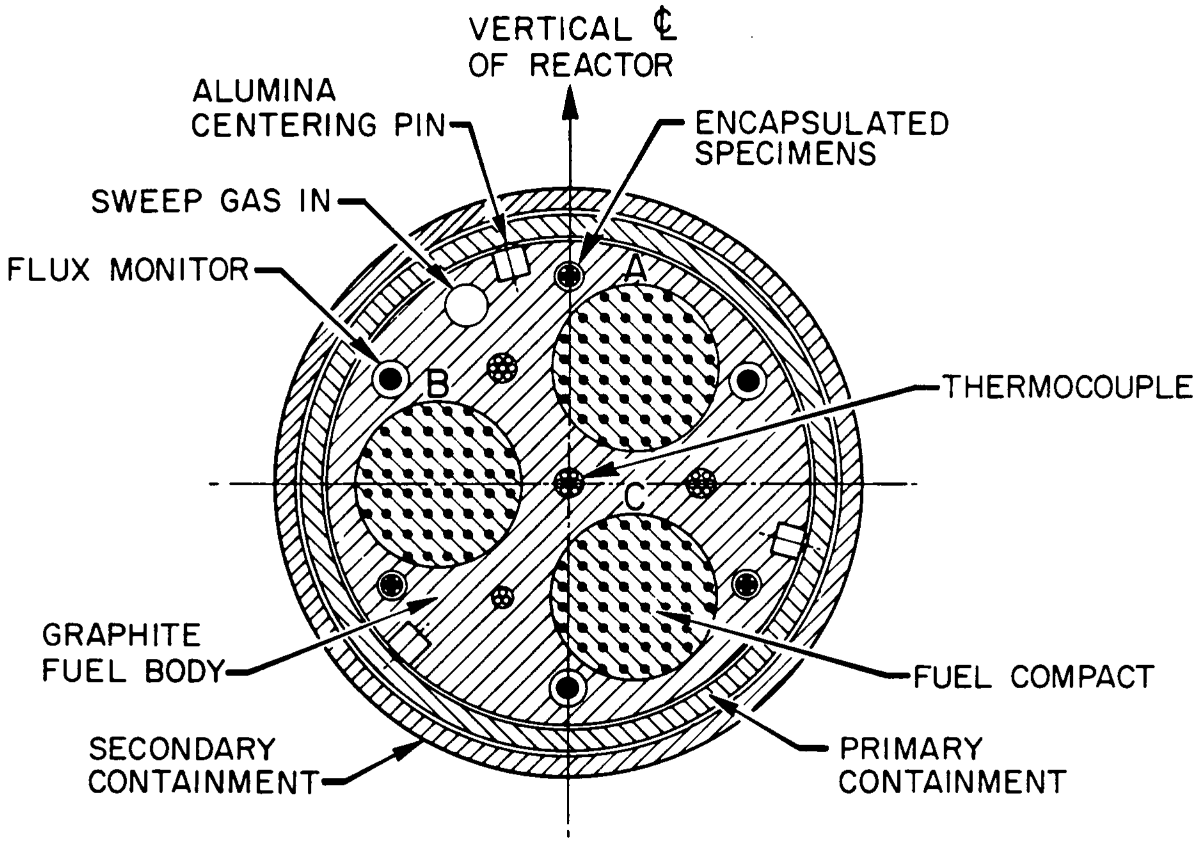

Fig. 6. Horizontal section through irradiation capsule HRB-21.

BACKFITTING OF THE FRG REACTORS

W. Krull

Backfitting of the FRG-reactors

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1 The FRG-research reactors

The GKSS-research centre is operating two research reactors of the swimming pool type fueled with MTR-type type elements. The research reactors FRG-1 and FRG-2 having power levels of 5 MW and 15 MW are in operation since 31 a (FRG-1) and 27 a (FRG-2). They are comparable old like others, too. The reactors are operated at present at approximalety 180 d (FRG-1) and between 210 d and 250 d (FRG-2) per year. Both reactors are located in the same reactor hall in a connecting pool system (fig. 1).

2 Backfitting reasons

Backfitting measures are needed for our and other research reactors to ensure a high level of safety and availability.

Generally reasons for backfitting in our case have been:

1. operating experience;
2. main modifications related to e.g. power increase, changes in utilization or lack of spare parts;
3. changes in safety philosophy;
4. updated risk analyses, recent research results and political impacts;
5. necessary repairs.

3 Summary of the main backfitting activities within the last ten years

Midth of the seventeeth there were plans to increase the power of the FRG-2 from 15 MW to 21 MW. This was the reason for the following activities e.g.:

1. comparison of the existing design with today demands (criterias, guidelines, standards etc.).

2. probability approach for events from outside like aeroplane crashes and earthquakes.

   It should be mentioned, that the FRG-reactors are located in an area where aeroplanes flights are restricted as this location is close to the boarder of the German Democratic Republic and that this location is an extreme quiet region for earthquakes, too.

   Therefore: the risk coming from events from outside is acceptable for the operation of FRG-1 and FRG-2.

3. The main accidents had to be rediscussed like startup from low and full power, loss of coolant flow, loss of heat sink, loss of coolant and fuel plate melting.

4. We were forced to install a new reactor protection system following today demands on redundancy (2 out of 3), diversity etc. from the relevant power reactor standard. Fig. 2 gives a general overview of the design of the reactor protection system. The design principle can be seen in the table. It should be known that there are comparators between each channel of the redundant analog signals.

probable accident	diverse chain 1	diverse chain 2
startup from low power	startup chambers (1 out of 2) safety chambers (2 out of 3)	linear chambers (2 out of 3)
startup from high power	safety chambers (2 out of 3) N^{16}-chambers (2 out of 3) primary temperature (2 out of 3)	linear chambers (2 out of 3)
fuel plate melting	N^{16}-chambers (2 out of 3)	γ-chambers on ceiling of reactor hall (2 out of 3)
loss of primary coolant flow	primary flow rate (2 out of 3)	pressure primary circuit (2 out of 3 in max. and min.)
loss of coolant	water height in pool (2 out of 3)	water height in basement (2 out of 3) γ-chambers on ceiling (2 out of 3)
loss of secondary coolant flow	primary coolant temperature (2 out of 3)	slow time response therefore only standard actions
experiments (FRG-2)	2 out of 3	2 out of 3

5. In the reactor hall a new crane has been installed (fig. 3). Before designing and installing the new crane a risk estimation has to be made giving the demands on the design coming up from power reactor standards. The crane load has been increased at that time, too.

6. An operation manual and inspection manual has to be written in the meantime not only for the research reactors FRG-1 and FRG-2. We have such manuals now for the cold neutron source and for the hot cells. A lot of work and a lot of paper but if these manuals are on hand they are very useful.

4 Backfitting activities within the last two years

Especially within the last two years larger backfitting and modernization activities have been made to enable reactor operation for the following ten and more years.

4.1 Installation of a cold neutron source

A cold neutron source has been installed to increase the flux of cold (5 Å) neutrons by a factor of 14. The source is in operation since June 1989. A report will be given next week at the workshop in Los Alamos.

4.2 Enrichment reduction to LEU fuel for FRG-1

The FRG-1 is being converted from 93 % enriched U with UAl_x fuel to 20 % enriched U with U_3Si_2 fuel. We have the license and the fuel elements on hand. This has been reported already, too, at an earlier meeting and present results will be presented this year at the RERTR meeting.

4.3 Measures for fire protection

The gap between an old building and new standards on fire protection has to be closed

1. all three stairways have to be separated from the floors
2. the fire resistance for some walls and doors has to be increased
3. not used cables from cable channels must be removed
4. smoke flaps and smoke ventilators have to be installed
5. cable penetrations must be fire protected
6. a fire detection monitoring system in all technical rooms and a fire alarm control panel was installed (fig. 4).

4.4 Installation of a double tubing for parts of the primary piping of the FRG-1

The piping, valves, pumps etc. of the FRG-1 are located in the cellar below the FRG-1 reactor pool. Between ceiling and the first valve there was nothing to stop leaking water. For this reason between the ceiling and the first automatic operated valve a double tube has been installed for the water inlet and water outlet pipings.

In fig. 5 the double tubing and an auxiliary construction can be seen. This auxiliary construction is necessary to fix the double tubing in place in case of a leak in the first piping. Then the double tubing has to withstand the pressure of 10 m water above. In the space between the tubes 3 water detectors (heated thermocouples) have been installed giving an alarm in 2 out of 3 mode.

4.5 Repair of both cooling towers

The internals of both cooling towers consisted of wooden materials for the distribution and spraying of the cooling water. The following actions were taken

1. In the FRG-1 cooling tower all wooden internals have been replaced by polystyrol and the coolant capacity has been slightly increased

2. In the FRG-2 cooling tower the water distribution system has been renewed totally.

4.6 Modernization of the ventilation system

Old flaps were partially removed and hose flaps were installed. Within the exhaust air channels the conventional filtering system has been replaced totally. Now all inspections can be made leaving the filters in place and the main flaps can be operated automatically.

4.7 Measures against water leakages

As I know that water leakage is a problem for some other older research reactors, too, I will describe these repairs in more details.

Two kind of damages were known before starting the repair action: Water leakage from the ceiling in the cellar below the reactor pools and some defect tiles (fig. 6) at the walls of the FRG-1 pool.

The licensing authorities demand to present a repair program for the pool and for avoiding water leakage into the cellar in the future.

To understand the considerations and the proposed repair program a brief design description of the biological shield must be taken from fig. 7.It is clear that there were rised the following questions:

1. quality of the internal part of the 60 cm concrete (ρ = 2.3 g/cm^3)
2. quality and γ-resistance of the epoxy resin layer
3. status of the steel liner (5 mm!)
4. status of the thiokol waterproofing between steel liner and Al beam tubes.

Steps of the repair program were the following:

1. The reactor bridge of FRG-1 including core and grid plate was moved to an other pool.

2. Radioactive components like the inpile parts of the beam tubes, etc. were removed by three divers. A maximum whole body dose of 0,9 mSv

was achieved by the divers. It should be mentioned that at the front part of the beam tubes the dose level was between 100 and 600 mSv/h around 8 weeks after shutdown.

3. After removing parts of the tiles, of the epoxy resin and the internal concrete the situation looks like it can be seen in the fig. 8.

   This shall demonstrate that after removing all radioactive components we were able to stay there. At most places the dose level was below 10 μSv/h after shielding the flanges of the beam tubes.

4. Inspection of the internal concrete was done by the consultants of the licensing authority. This includes optical inspection, compression measurements on selected samples and hardness measurements. The result of all these inspections were that the concrete is in good condition.

5. The steel liner has been inspected at two different positions: In both cases the liner was found to be in an excellent condition (fig. 9).

6. The polyurethan has been injected with high pressure to tighten the thiokol seal between steel liner and Al beam tubes. For this purpose holes have been drilled through the concrete near to the thiokol. In these holes polyurethane was pressed. There are 20 of such penetrations of the liner and concrete and per penetration as a minimum 2 holes must be drilled to get a high confidency in this work.

7. After repairing the concrete, sealing the concrete with - at this time - unsaturated polyester (as this polyester is better γ-resistant than epoxy resin), the tiles have been placed and the pool cleaned up.

The next fig. 10 shows the repaired pool:
- the beam tubes in place
- a Be block reflector installed
- and the inpile part of the cold neutron source.

Now I am coming to an absolutely different kind of damage and repair: In a separate part of the cellar (3 x 12.5 m^2) which is below pool 3 and 4 there were a few small cracks within the concrete of the ceiling. This looks not dangerous but it may become a severe problem.

Removing these damaged parts in a small region and to a depth of ca. 2 cm we checked the pH of the concrete which gives good information about the quality of the concrete. With a great surprise to the involved parties a carbonization depth of ca. 12 cm was found. This carbonized conrete must be removed totally and new concrete has to be placed there as otherwise the whole structure may loose its stability.

Finally on these repairs: All repair actions were fully accepted by the licensing authorities and their consultants and til today the status of the repaired parts is excellent and no water leakage into the cellar was seen again. You may believe, we are happy about this.

5 Ongoing and planned backfitting activities

5.1 Replacement of instrumentation etc.

As it could be seen that there will be within the next future increasing difficulties for maintaining and repairing the process control system and for getting new spare parts it was decided to renew the instrumentation, the process control system and the alarm system. The order was placed end of 1988. Fig. 11 shows the principal design of the instrumentation and monitoring system. The system is being implemented. We are hoping to go into operation again in March 1990.

5.2 Renewal of the emergency power supply

At present we have since approximately 25 years in operation a flying wheel diesel generator and a diesel generator for the emergency power supply (1 out of 2). The capacity of these generators is large so that they are not only used for the needs of an emergency power supply. Probable faults can be caused by other reasons.

Considering this situation the decision was made to build a new station for (1 out of 2) diesel generators to be used only for the emergency power supply for our two reactors. The principal design work is being finished and we are going to ask for inquiries from competent suppliers. The principal design can be taken from fig. 12.

5.3 Lightning protection

The standard conventional lightning protection is present and inspected annually by consultants. Due to research results taking into account damages in modern electronics (IC) arising from induced voltages and currents a by far more increased lightning protection is necessary. A report is being made by consultants we hired ourselfes.

6 Resumee

The GKSS research centre intends to operate their research reactors safe to prevent undue risks from the public and the operational staff. Therefore many actions have been made

- to follow present safety philosophies
- to replace old equipment to have an installation which is near the state of the art
- to learn from operation experience got in our and other facilities.

These efforts will continue to allow safe operation of our research reactors over their whole operational life.

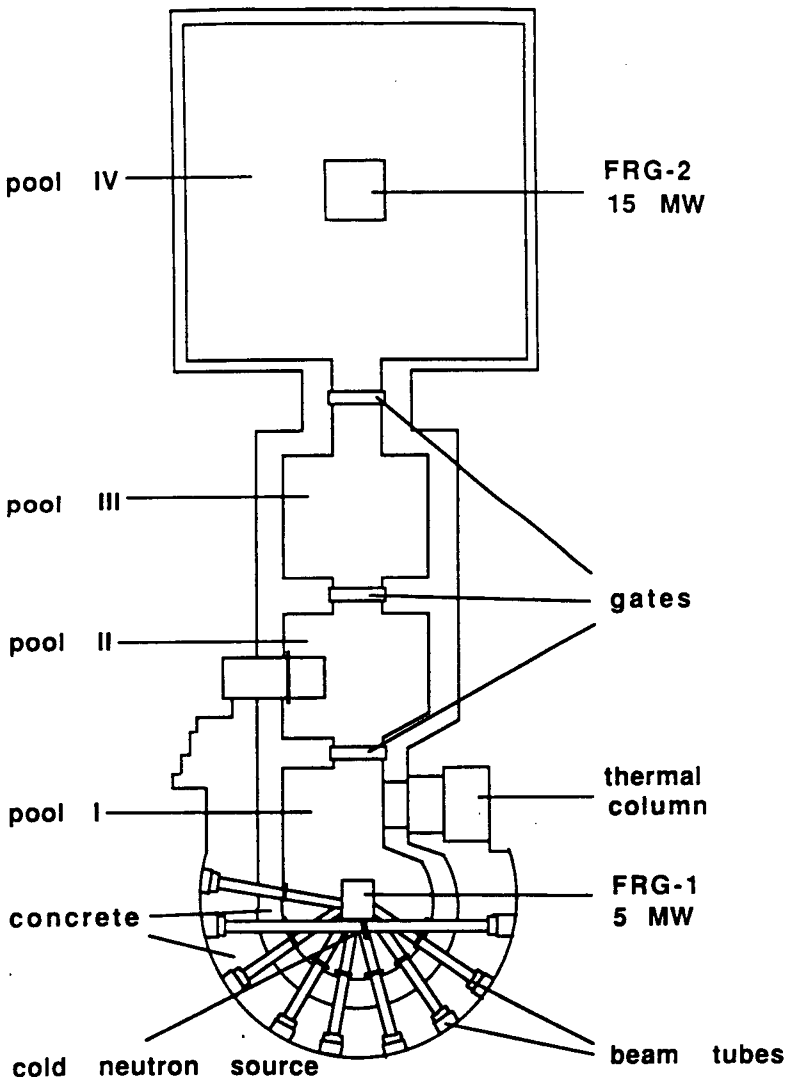

pool IV
FRG-2
15 MW
pool III
gates
pool II
thermal
column
pool I
FRG-1
5 MW
concrete
beam tubes
cold neutron source

Fig. 1

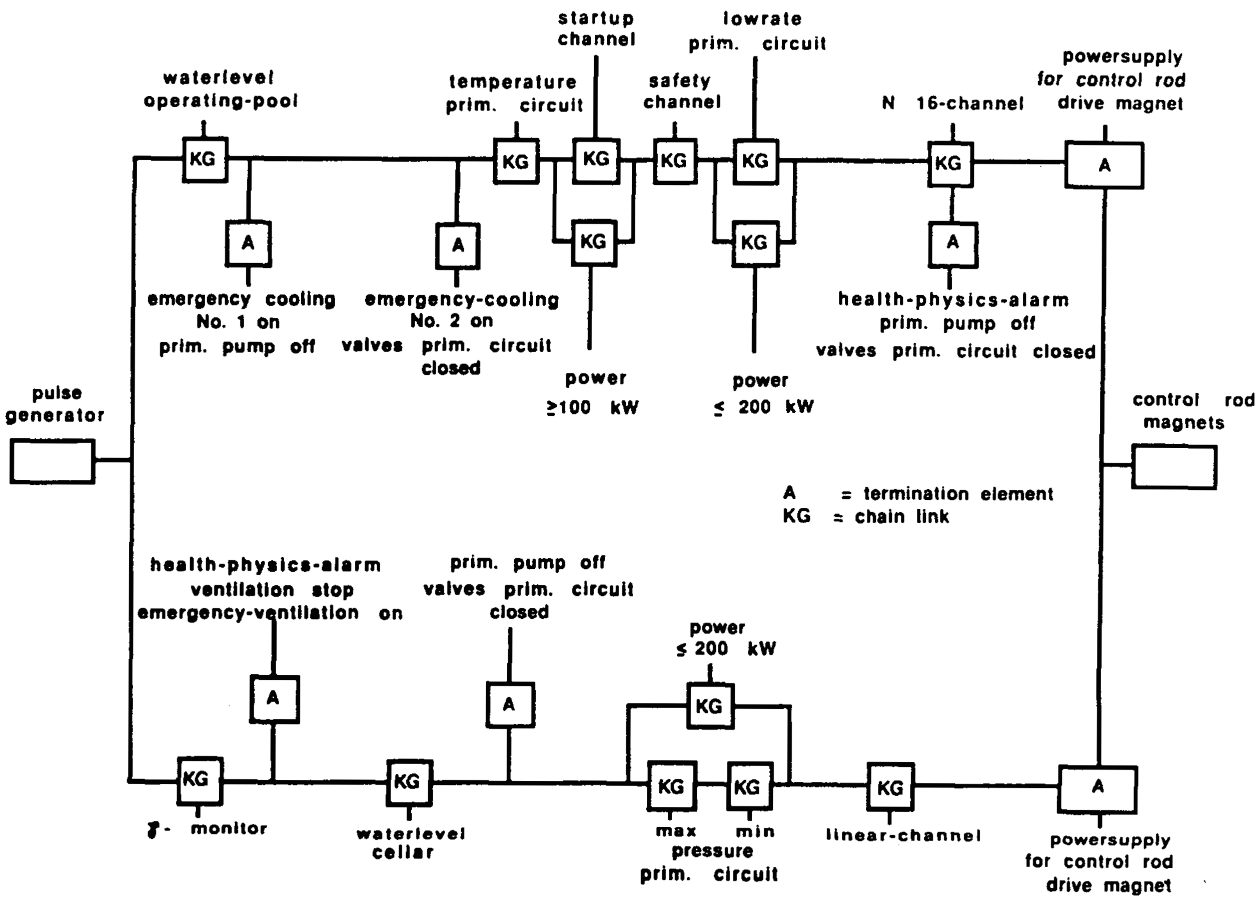

startup channel
lowrate prim. circuit
waterlevel operating-pool
temperature prim. circuit
safety channel
N 16-channel
powersupply for control rod drive magnet
KG
A
emergency cooling No. 1 on prim. pump off
emergency-cooling No. 2 on valves prim. circuit closed
power ≥100 kW
power ≤ 200 kW
health-physics-alarm prim. pump off valves prim. circuit closed
pulse generator
control rod magnets
A = termination element
KG = chain link
health-physics-alarm ventilation stop emergency-ventilation on
prim. pump off valves prim. circuit closed
power ≤ 200 kW
γ- monitor
waterlevel cellar
max min pressure prim. circuit
linear-channel
powersupply for control rod drive magnet

Fig. 2

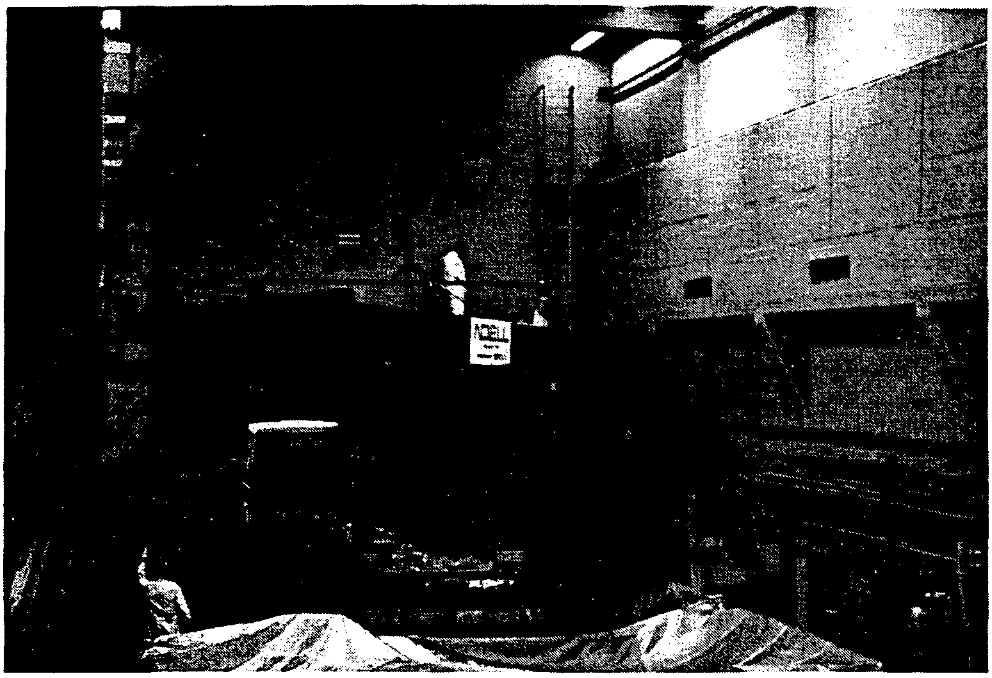


Fig. 3

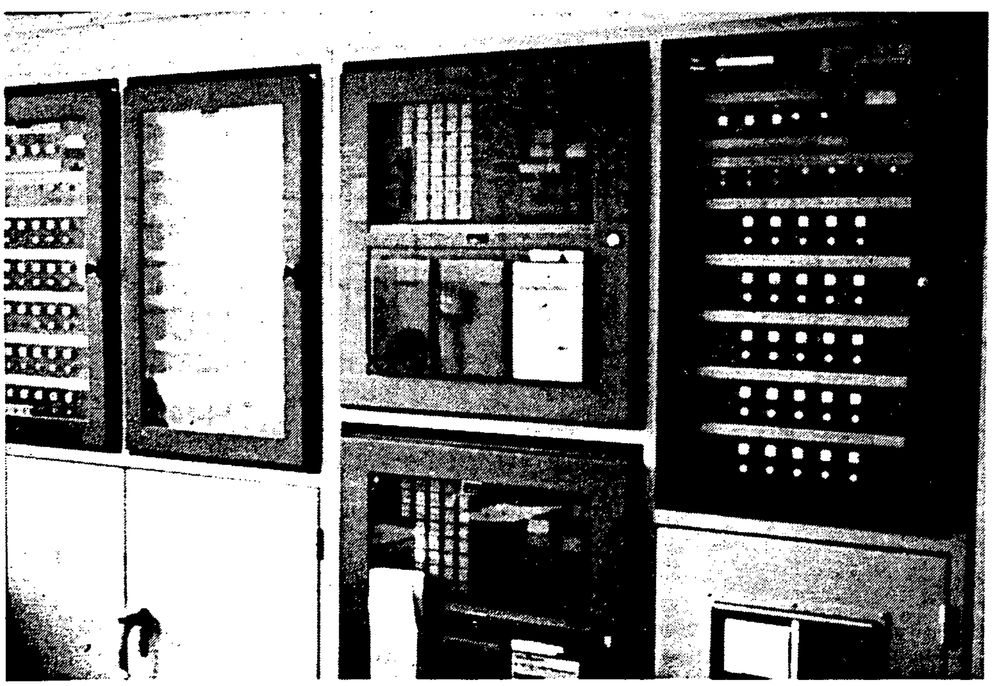


Fig. 4

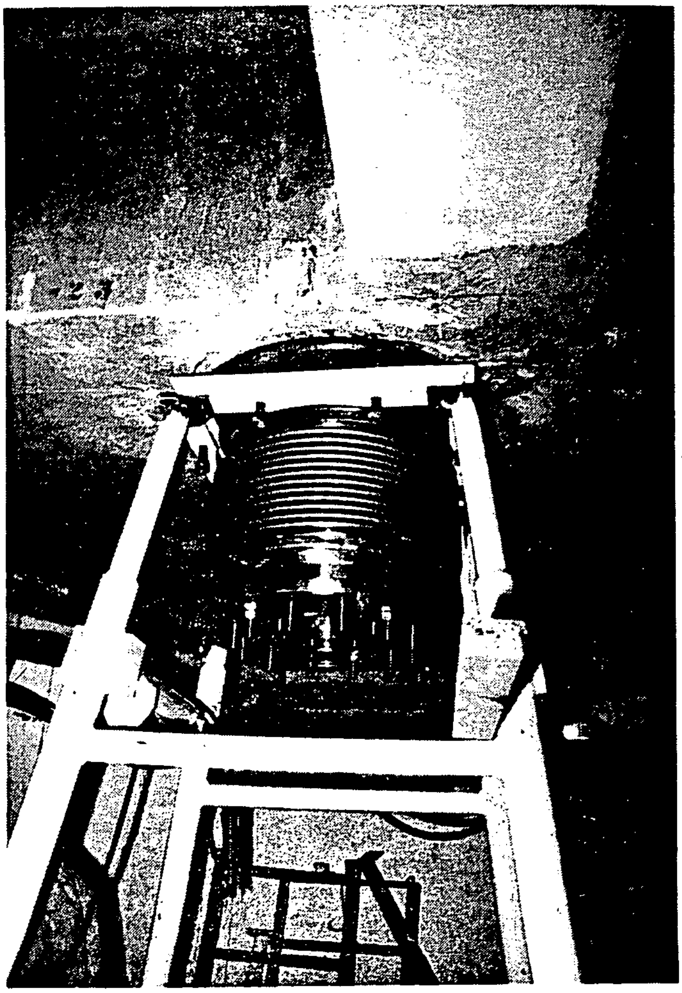


Fig. 5

Fig. 6

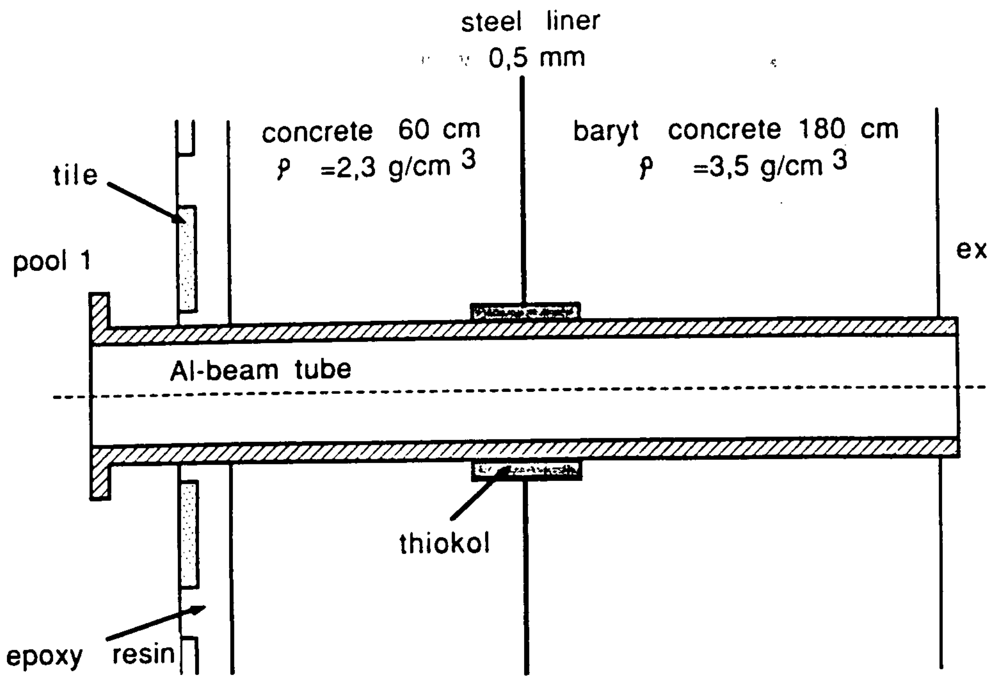

steel liner
0,5 mm
concrete 60 cm
ρ =2,3 g/cm 3
baryt concrete 180 cm
ρ =3,5 g/cm 3
tile
pool 1
experimental hall
Al-beam tube
thiokol
epoxy resin

Fig. 7

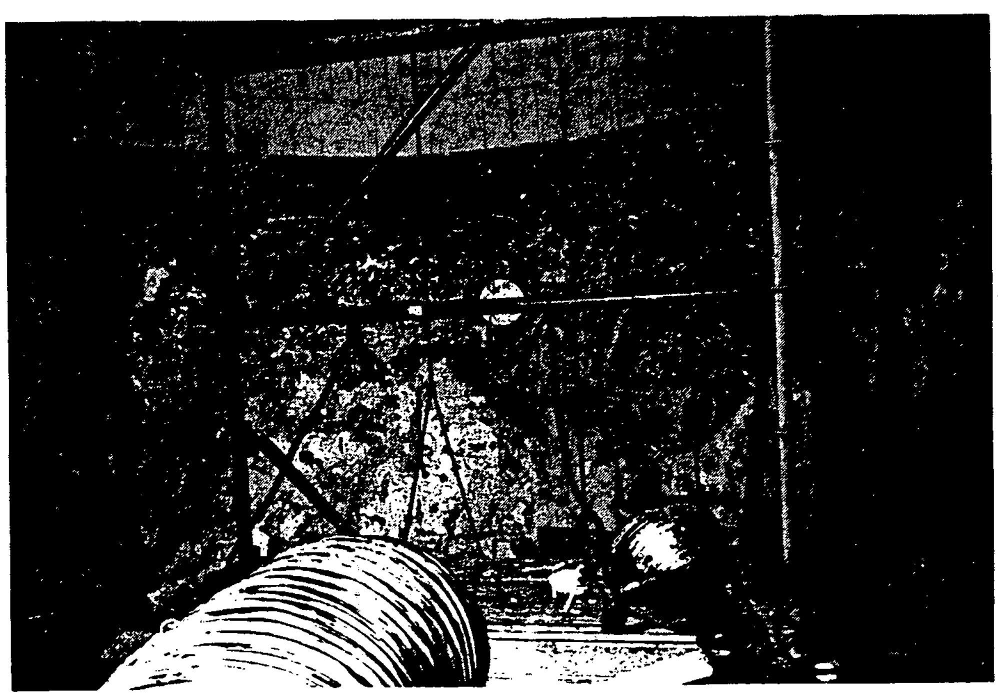


Fig. 8

Fig. 9

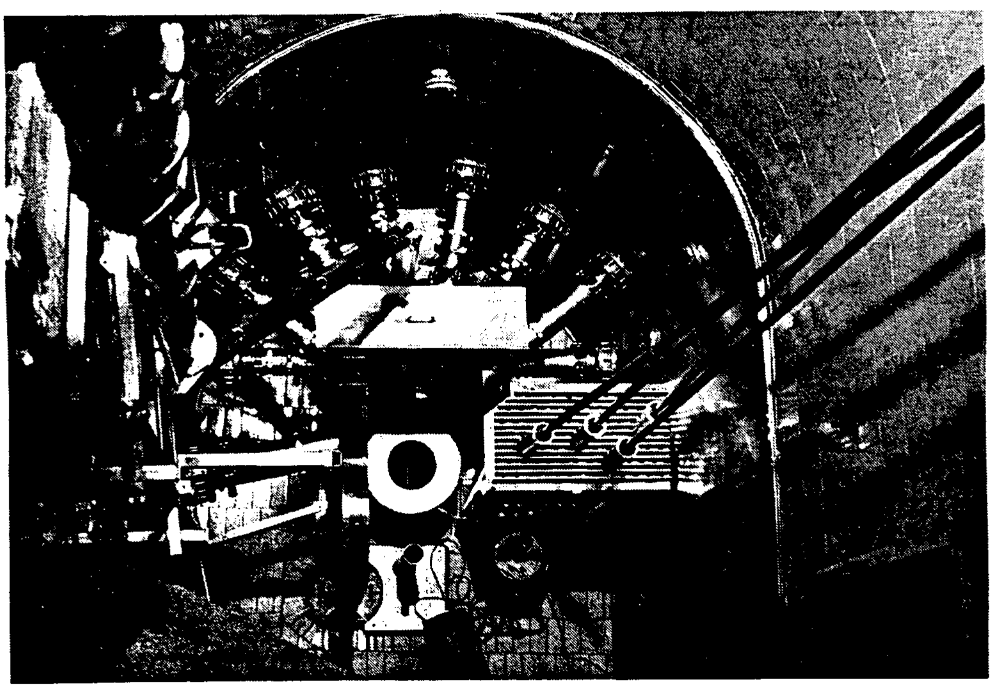


Fig. 10

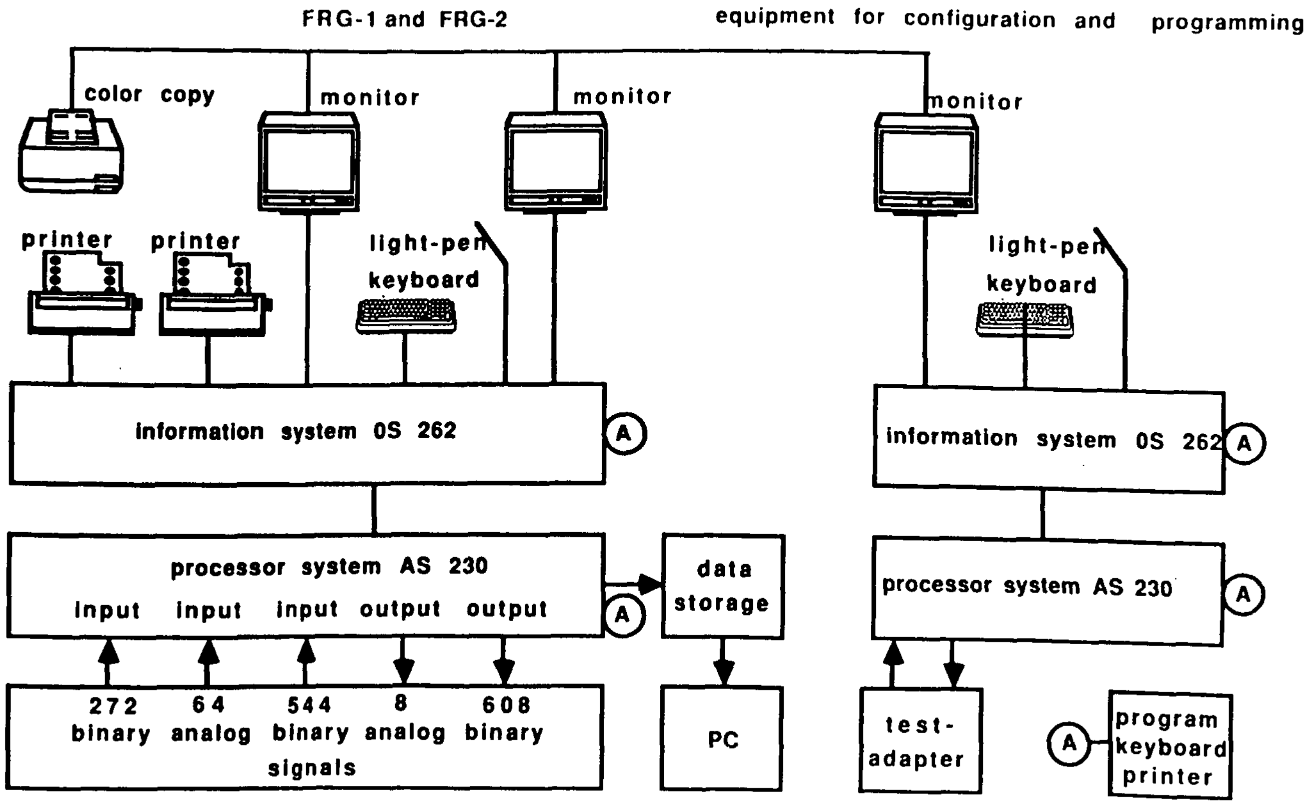
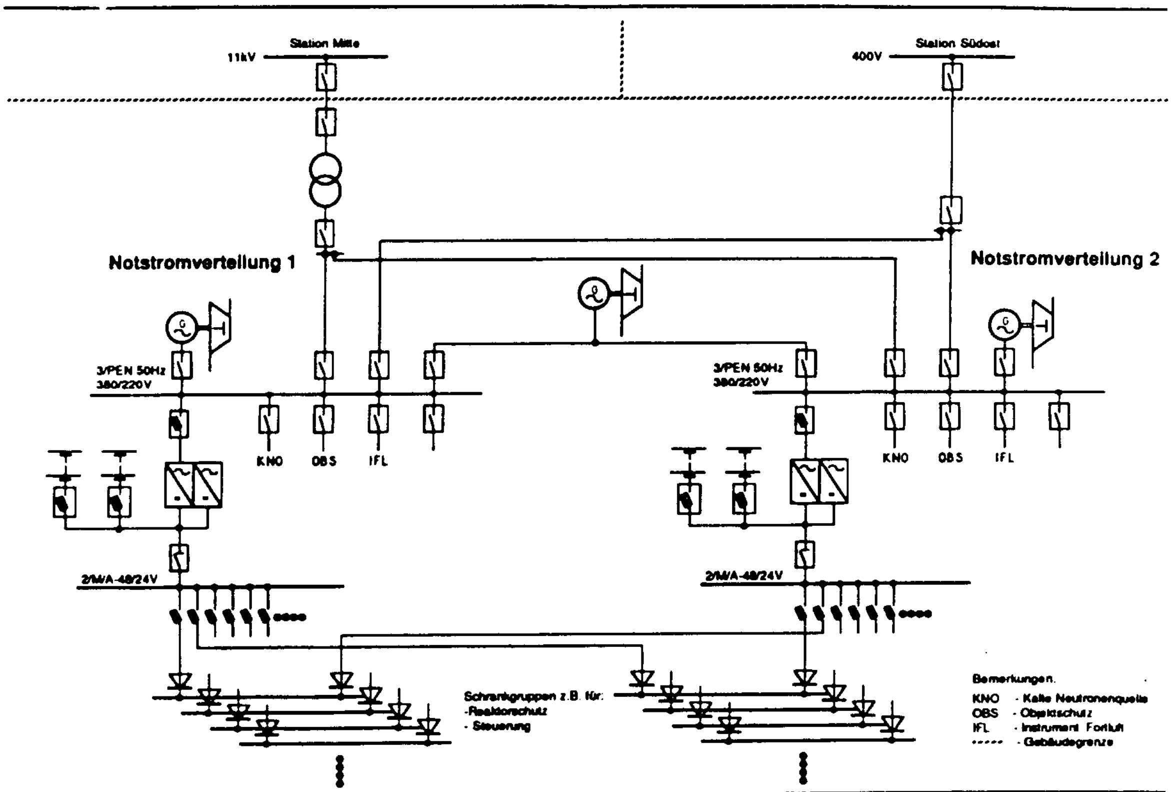

FRG-1 and FRG-2
equipment for configuration and programming
color copy
monitor
monitor
monitor
printer
printer
light-pen
keyboard
light-pen
keyboard
information system OS 262
A
information system OS 262
A
processor system AS 230
input
input
input
output
output
A
data
storage
processor system AS 230
A
272
binary
64
analog
544
binary
8
analog
608
binary
signals
PC
test-
adapter
A
program
keyboard
printer

Fig. 11



GKSS- Geesthacht, Übersichtsschaltplan-Notstromanlagen

Fig. 12

UNIVERSITY RESEARCH REACTORS IN THE UNITED STATES; THEIR ROLE AND VALUE - FROM THE 1988 STUDY BY THE NATIONAL RESEARCH COUNCIL

O. K. Harling

University Research Reactors in the United States; Their Role and Value – From the 1988 Study by the National Research Council

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Cambridge, Massachusetts USA

ABSTRACT

This paper provides a brief overview of the 1988 National Research Council study of the role and value of university research reactors in the US.

I. INTRODUCTION

There are currently 35 operating university research reactors (URR's) on 33 sites in 24 states. Figure 1 shows the distribution of URR's by power level and Figure 2 shows their location in the USA. Concern about these facilities has been widespread in the URR user community, in university administrations, in government, and often among the public, especially individuals residing relatively close to these research facilities. The concern about the cost (including safety) and the value of URR's has been heightened by the increasing number of URR shutdowns. The number of these facilities has diminished by a significant fraction during the last several decades.

A result of heightened concern in the federal government was the USDOE's 1985 questionnaire to URR operators. This was designed to assess URR productivity. As a result it was clearly shown that URR's were continuing to make major contributions to education and research despite their inadequate and diminishing funding base. To obtain an unbiased picture of URR costs and value by a blue ribbon national committee, the USDOE in 1986 requested the National Research Council to evaluate the contributions of university-based research reactors to research and education in nuclear science and engineering. Consideration was to be given to:

a) Increasing costs at universities

b) Decreasing enrollments and research in nuclear science and engineering programs

c) Anticipated increases in URR regulations

d) Concerns about reactor safety and security

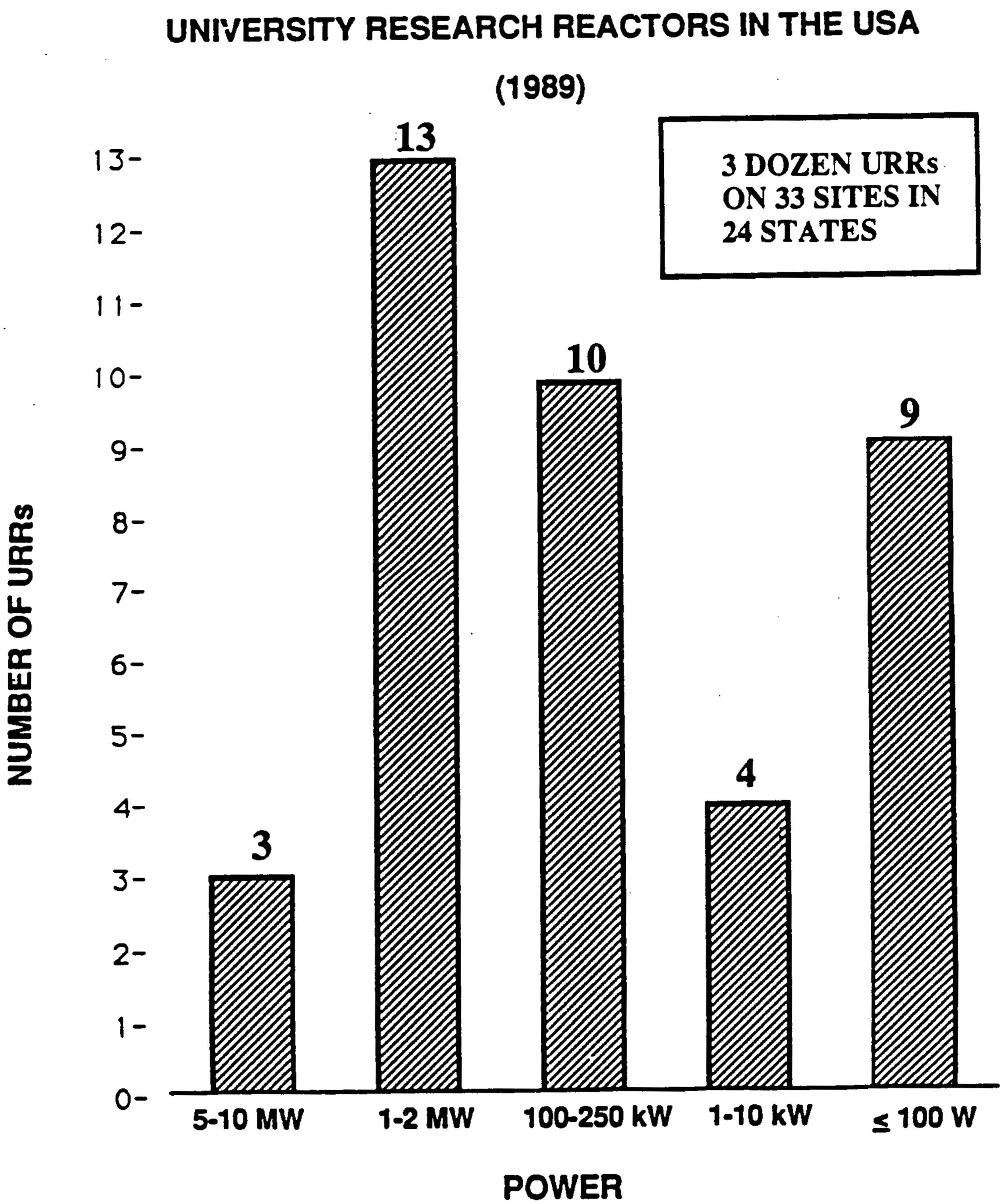

UNIVERSITY RESEARCH REACTORS IN THE USA
(1989)
3 DOZEN URRs ON 33 SITES IN 24 STATES
NUMBER OF URRs
0
1
2
3
4
5
6
7
8
9
10
11
12
13
3
13
10
4
9
5-10 MW
1-2 MW
100-250 kW
1-10 kW
≤ 100 W
POWER

Figure 1

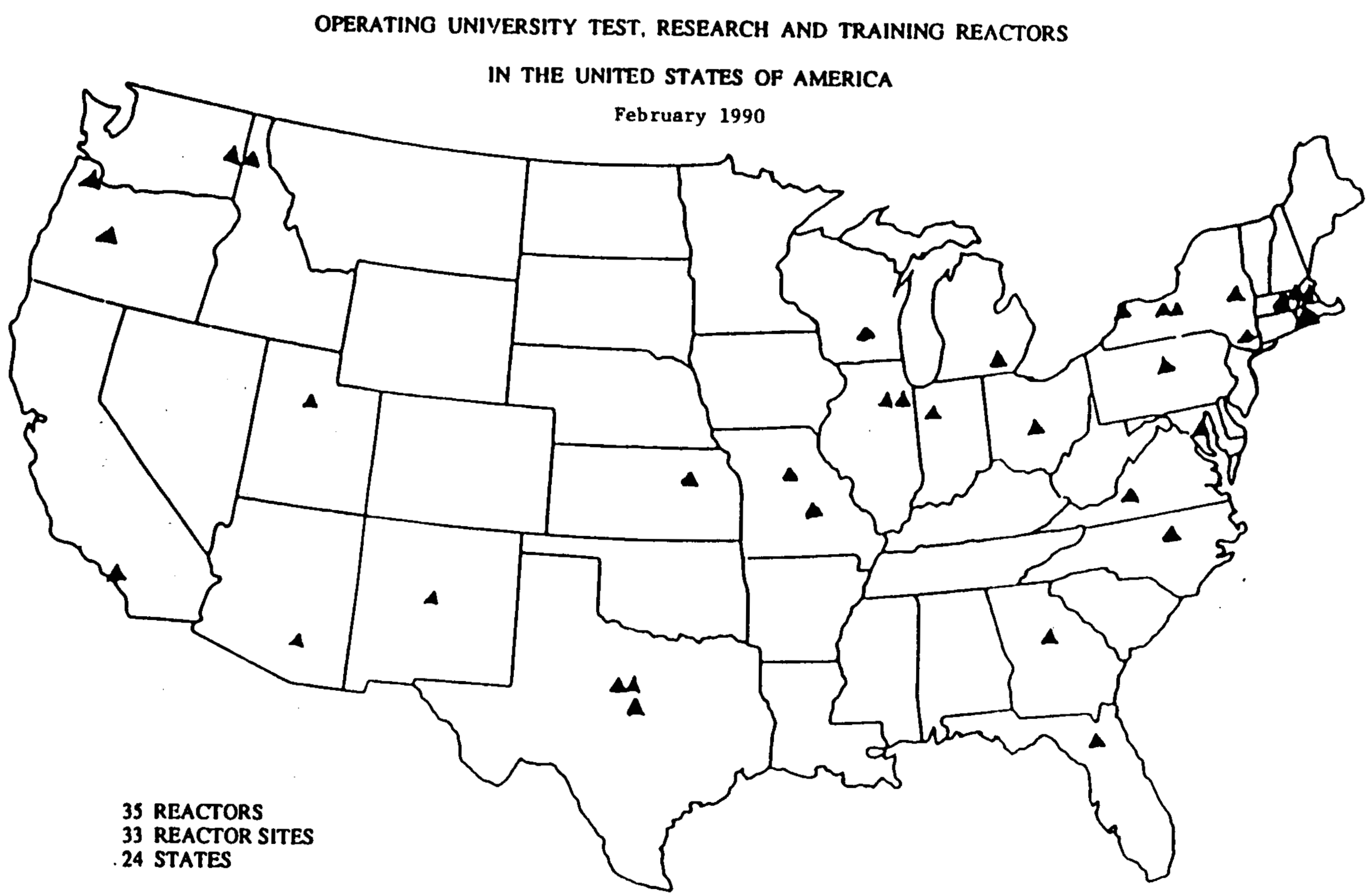

OPERATING UNIVERSITY TEST, RESEARCH AND TRAINING REACTORS
IN THE UNITED STATES OF AMERICA
February 1990
35 REACTORS
33 REACTOR SITES
24 STATES

Figure 2

I. THE NATIONAL RESEARCH COUNCIL STUDY

The National Research Council (NRC) accepted the DOE-sponsored study and decided to divide their efforts into the following tasks:

1. Review and evaluate existing university research reactors to determine their role in meeting the needs to education, training, research, and service in relevant fields of science and engineering.
2. Evaluate the specific mandates and interests represented by academic, government, and industry organizations with respect to university research reactors.
3. Review and evaluate the use and support of similar reactors elsewhere, in Western Europe, for example.
4. Review security and safeguard issues involving university research reactors.
5. Evaluate the role of university administrations and other entities in support of URR programs.
6. Evaluate the role of the federal government in support of URR programs.
7. Provide recommendations and/or options for federal and other support of university research reactors.

An interdisciplinary group of experts, with strong representation from outside the university reactor community, was appointed to conduct the study. The members of the NRC committee are listed below:

David A. Shirley (Chairman), Director, Lawrence Berkeley Laboratory, Berkeley, California

Robert M. Brugger, University of Missouri, Columbia, Missouri

Geoffrey L. Greene, National Bureau of Standards, Gaithersburg, Maryland

John S. Laughlin, Memorial Sloan-Kettering Cancer Center, New York, New York

Mihran J. Ohanian, University of Florida, Gainsville, Florida

John Poston, Texas A&M University, College Station, Texas

Clifford G. Shull, Massachusetts Institute of Technology, Cambridge, Massachusetts

Bernard I. Spinrad, Iowa State University, Ames, Iowa

Anthony L. Turkevich, University of Chicago, Chicago, Illinois

Edwin L. Zebroski, Electric Power Research Institute, Palo Alto, California

During the study the NRC committee reviewed the following areas and reached the following conclusions:

1. The spectrum of research at university reactors:

   - Neutron activation analysis – extremely wide range of applications, bulk of US work is done at URR's.
   - Neutron scattering – a large fraction of neutron scattering experts were trained at URR's.
   - Neutron radiography – a wide range of uses, including industrial.
   - Medical diagnostics and therapy – the bulk of the USA's innovative research in this area is done at URR's.
   - Radiation effects in materials – radiation damage studies prepare students for work in the national facilities.
   - Nuclear engineering and reactor physics – a major contribution to the national programs through research, covers a wide range.

2. Research reactors in education and training:

   - URR's of all sizes are an important part of the educational process in a broad multi-disciplinary sense. URR's are useful in teaching nuclear science at all levels, from high school through graduate school.
   - URR's often provide a unifying theme for nuclear engineering programs in presenting reactor behavior in a realistic way, and in being a primary research tool.

- Those universities which have nuclear engineering programs and a broad educational program in applications of the nuclear sciences will be best equipped for this purpose if they have a URR on campus.

3. Research reactor services for other users (outreach):

   - URR's play a vital role as a service facility in the nuclear sciences – related disciplines. Service is provided to other universities, industry, government laboratories and departments. Service is a significant component of the reactor utilization at several URR's.

   Typical services include:

   - radioisotope production and application
   - neutron activation analysis
   - neutron radiography
   - neutron gauging
   - neutron scattering
   - gamma-ray scattering
   - standardization assays
   - radiation shielding testing
   - radiation damage testing in structural materials
   - personnel training
   - radiation chemistry
   - safety analysis

4. Research reactors in other countries, especially Europe:

   - There is a strong perception that western Europe occupies a position of leadership in reactor based science. This is strikingly evident in neutron beam tube research.
   - The Europeans support a strong network of reactors which include the major national facilities and the university class research reactors.

- Many techniques now used at the national facilities originated at the URR's.
- University reactors in Europe receive base support for operation and a base in-house research program. Typically, several million dollars per year for a reactor of several MW.
- The user base is considerably larger at the European URR's compared to the large US URR's.
- Success of the European URR's is possible because of a national community-wide effort that encourages efforts at reactors of all sizes and encourages cooperation between them.
- Whether US URR's as a national resource are adequately employed is germane to the planning of our national program in the neutron sciences, including the advanced neutron source. URR's are also particularly germane to the question, whether there will be an adequate community of younger scientists and engineers to use the advanced neutron source and other national facilities.

5. Safety and safeguards of URR's:
    - Safety hazards:
        - damage to the fuel core and consequent spread of radioactivity in or beyond the reactor containment building
        - spread of small amounts of radioactivity or medical isotopes from experimental programs
        - spread of radioactive coolant in the event of leakage
        - injury to personnel from weapons, fire, and explosive devices
    - Safeguards hazard:
        - theft and diversion of nuclear material
        - intrusion and theft of materials or equipment other than nuclear materials
        - intrusion, sabotage, and vandalism

- Conclusion of NRC study (related to safety):
  - the safety records of URR's are excellent
  - the safety hazards are small relative to large power reactors
  - the effective functioning and continued operation of the present URR's are more affected by disproportionate public and institutional perceptions of risk than be actual physical or nuclear hazards.

6. Institutional and federal support of URR's:
   - Based on its deliberation, the NRC committee believes that a national program of support for URR's is justified by their educational, research, and service value to the nation.
   - There is no consistent, dependable pattern of support for URR's at either the local or national level.
   - Total need of URR's is estimated at ~ $ 35M annually for a healthy program with high utilization.

III. CONCLUSIONS OF THE STUDY

- URR's merit a base of federal support because of national benefits that accrue from a healthy URR program.
- URR's merit support from state and local governments and industry because they train workers and provide services of direct benefit.
- If URR's are to play a vital role in research and in the education of scientists and engineers, they need immediate funds to:
  - bring current operations up to a level adequate to maintain vital programs
  - purchase instrumentation and equipment needed to modernize reactor operations, research, and teaching programs
  - the federal government should consider committing up to $20 million per annum to assist in funding URR operations and upgrades.

IV. PRINCIPAL RECOMMENDATIONS OF THE STUDY COMMITTEE

- The federal government, in partnership with the universities and the national laboratories, should develop and implement a national research reactor strategy, the elements of which should include:
  - development of university and national laboratory centers of excellence in specific areas of the neutron sciences and reactor technology for world-class research as well as for education
  - anticipation that as some university reactors are upgraded and a user's network is created (see below), others are likely to close
  - mechanisms to assure that such closures do not go so far as to damage the national interest related to research and educational capabilities in the nuclear sciences and engineering
  - development and support of a reactor network to provide enhanced utilization and productivity of U.S. research reactors involving researchers from universities with and without on-campus reactors, and from the national laboratories
- To implement the above strategy:
  - a single federal agency should be designated to administer programs in support of the national research reactor programs
  - the federal government should create a standing advisory structure to advise on a continuing basis on all aspects of this program
- In pursuit of this strategy, the federal government should:
  - adopt the goals of meeting U.S. research reactor needs, and regaining a position competitive with Europe and Japan in the neutron-based sciences
  - study, in detail, the approaches of other advanced countries to operating research reactor networks such as that of linking the major facility at Grenoble with smaller reactor research centers in Europe
  - establish and support such a network, adapted to U.S. needs

  - make up to $20 million available annually (as a preliminary estimate to be modified as improved data becomes available) to universities through the designated federal agency, specifically for operational support and facility upgrades of university research and educational reactors
  - create a peer review mechanism to assist the designated agency in making grants to universities
- The Nuclear Regulatory Commission should examine its current approach to the licensing and regulation of university research reactors in terms of the following issues:
  - the small nuclear materials inventories and low power densities of university research reactors, which result in risk factors related to safety and safeguards are considerably lower than commercial power reactors
  - avoiding unnecessary exposure of small university reactor operators to costly hearing and litigation procedures as a condition for licensing upgrades and improvements
- Finally, the Nuclear Regulatory Commission should consider grants of technical and financial assistance to help university reactor operators to comply with upgraded safety and safeguard requirements, including and continuing beyond the current program of assisting with the conversion to low-enriched fuels.

V. REFERENCES

[1] "University Research Reactors in the United States – their Role and Value," National Academy Press, Washington, D.C., 1988.

ORGANIZATION OF THE ITER PROJECT - SHARING OF INFORMATIONAL PROCUREMENTS

T. E. Shannon

T. E. Shannon, Manager
Fusion Engineering Design Center

Organization of the ITER Project Sharing of Information and Procurements

MOTIVATION FOR INTERNATIONAL COOPERATION IN FUSION ENERGY DEVELOPMENT

- **Fusion has the potential for a safe, environmentally attractive and practically inexhaustible source of energy, worldwide**

- **Very large and expensive experimental facilities are needed to prove the technical feasibility of magnetic confinement fusion**

- **Four large worldwide programs of about equal size have similar goals and objectives for constructing an experimental reactor as the focal point for the development program**
  - **European Community**
  - **Japan**
  - **U.S.S.R.**
  - **U.S.A.**

- **Cooperation among the four countries to build a single experimental reactor would reduce the cost for each country and provide an international pool of scientific and engineering resources**

COUNTRIES PARTICIPATING IN INTERNATIONAL AGREEMENTS WITH THE U.S.A. IN FUSION ENERGY DEVELOPMENT

Countries and Organizations

Australia	1
Canada	2
European Community	9
IEA	9
Soviet Union	2
Peoples Republic of China	1
Israel	2
Spain	1
Japan	7

LARGE COIL TASK

- The goal of the Large Coil Task (LCT) was to demonstrate reliable operation of large superconducting coils, to gain experimental data, and to prove the design principles and fabrication techniques proposed for the magnets in a tokamak experimental power reactor.

- The LCT is an example of effective multinational collaboration in an advanced technology project involving large-scale hardware produced in several countries and operated as a tightly integrated system.

- Participants of LCT were the United States, Japan, Switzerland, EURATOM, including six industrial organizations.

- ORNL designed and constructed the test facility and was responsible for overall integration and coordination of the entire test effort.

- The LCT successfully accomplished its intended purpose (September 1987) and the world-wide participants in fusion research are reaping benefits from the multinational effort.

ITER

The International Thermonuclear Experimental Reactor

Background

- 1985 — Government leaders in summit meetings call for cooperation in fusion energy development

- 1988 — Four government organizations: the European Community, Japan, Soviet Union, and United States begin Conceptual Design Activity (CDA) under auspices of IAEA

- 1990 — CDA to be completed in December 1990, will produce a conceptual design including:
  - Cost and schedule estimates
  - Site requirements
  - Validating R&D plan
  - Plan for operation

ITER ORGANIZATION FOR CONCEPTUAL DESIGN ACTIVITY

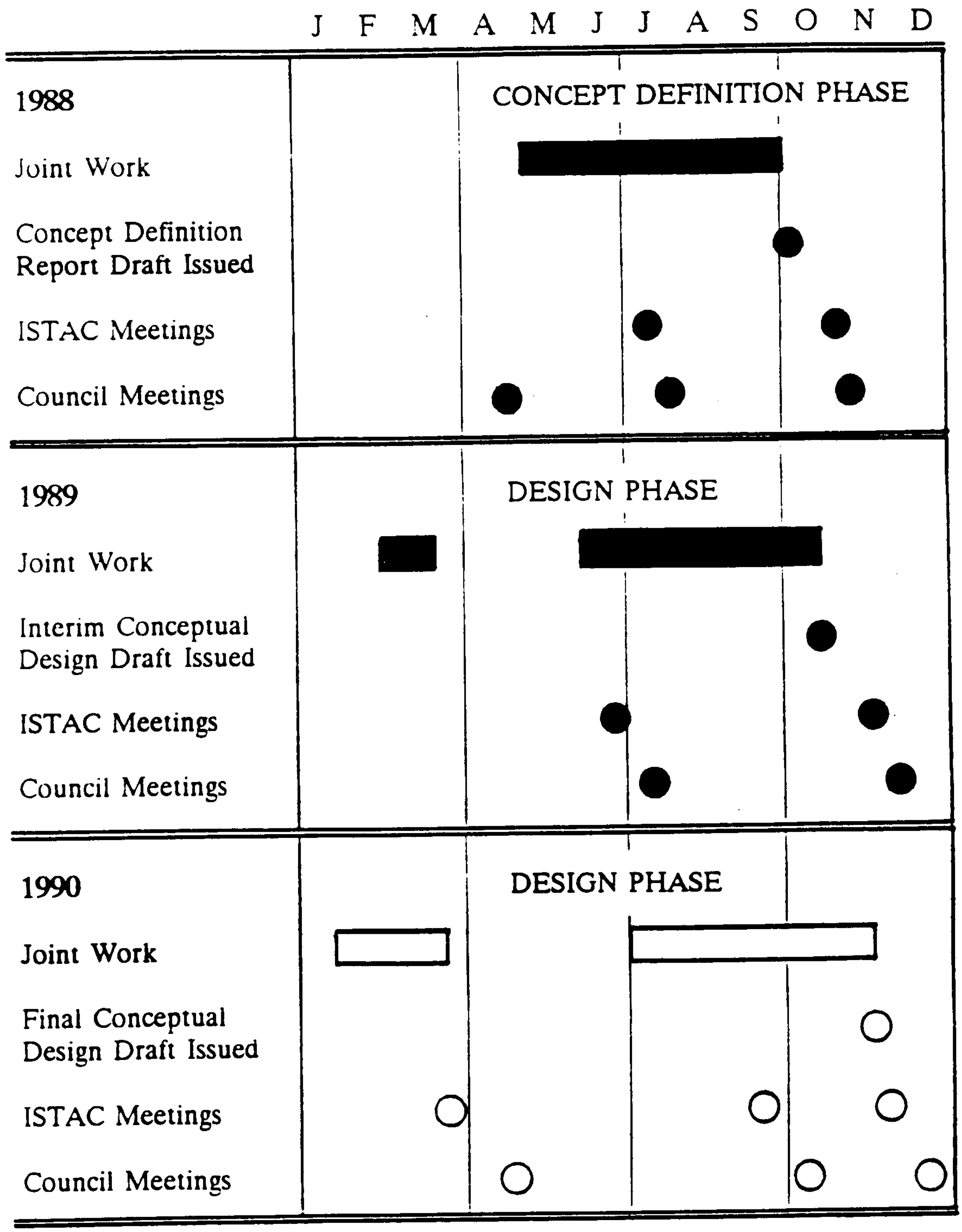
- Technical site for design effort at Garching, West Germany
  - Small staff of 8–10 are in permanent residence
  - Primary technical work is performed at home countries
  - Working sessions of 2–6 months duration with 40 or more individuals on site
  - Design issues are resolved by consensus
  - Supporting R&D at home sites $10 M/year, each country

- ITER Council (IC)
  - Responsible to IAEA for overall direction of activities

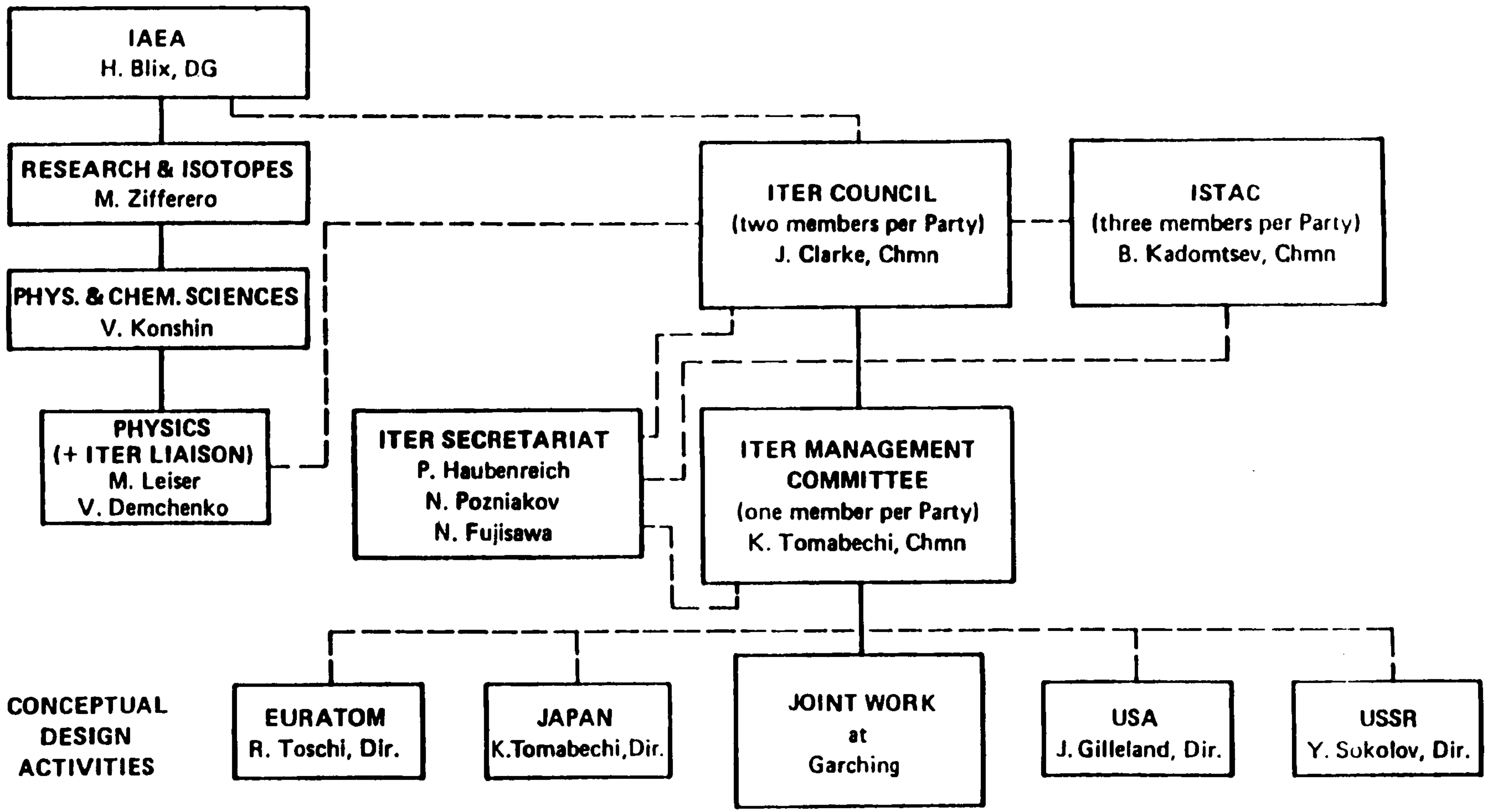
- ITER Management Committee (IMC)
  - Responsible for execution of activities
  - Manages work at Garching

- ITER Scientific and Technical Advisory Committee (ISTAC)
  - Consists of eminent scientists and engineers
  - Advises the IC

SCHEDULE FOR ITER CONCEPTUAL DESIGN ACTIVITIES



ITER ORGANIZATION



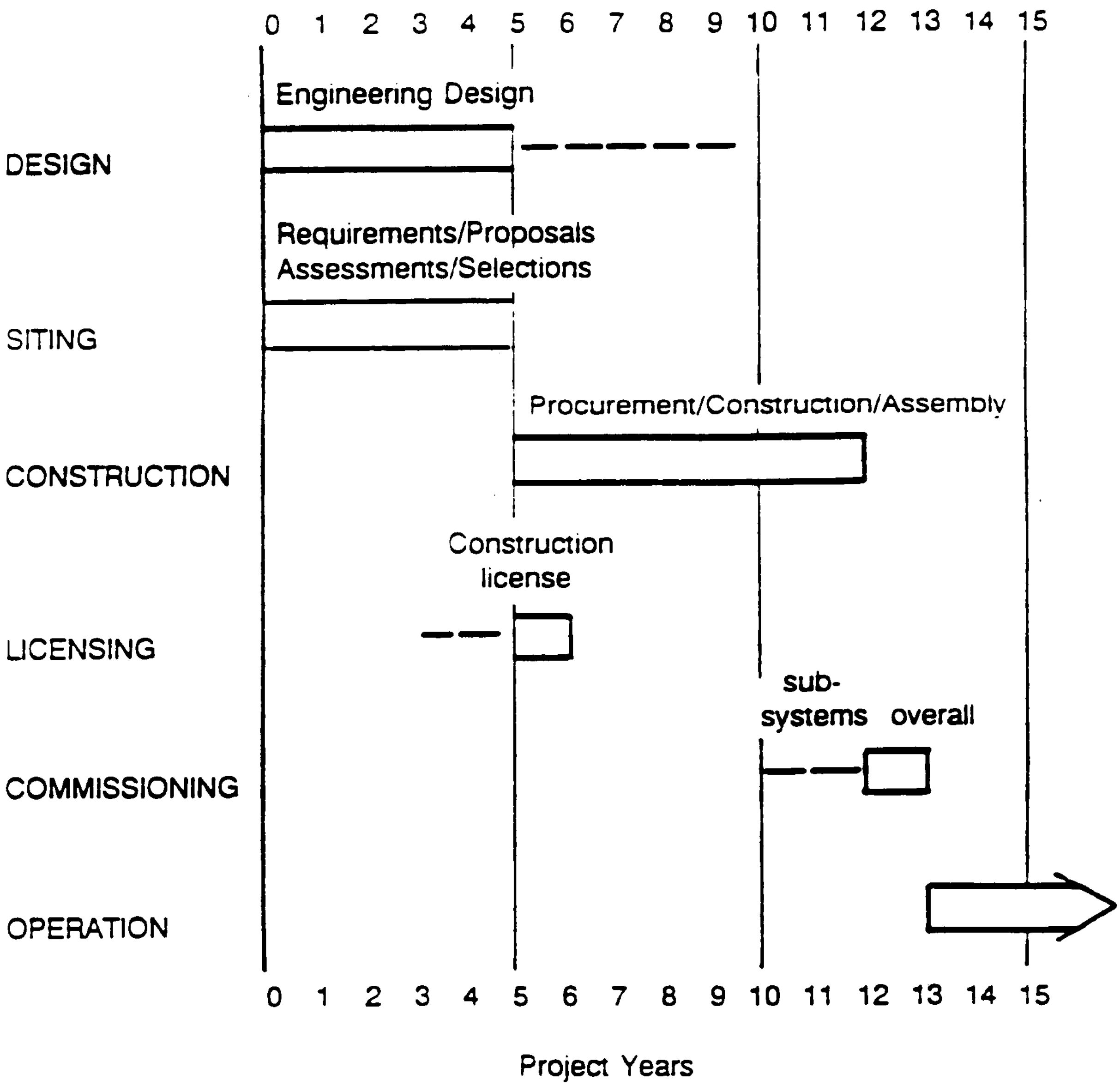
ITER OBJECTIVES

- ITER is expected to fully confirm the scientific feasibility and to address the technological feasibility of fusion power. Consequently, the machine must be designed for controlled ignition and extended burn of deuterium-tritium plasma. It must also demonstrate and perform integrated testing of components required to utilize fusion power for practical purposes.

ITER OPERATING PARAMETERS (PHYSICS PHASE)

Major radius, R (m)	5.8
Minor radius, a (m)	2.25
Toroidal field on axis, B (T)	5.0
Current, I_p (MA)	25
Fusion power, P_{fus} (MW)	1000
Neutron wall load (MW/m^2)	1.0

CONCEPTUAL PROJECT SCHEDULE AND COST



Cost Estimate — $4.9 B (January 1989)

FUTURE PLANS

- **Engineering Design Activity (EDA)**
  - **Working Party on Ways and Means established**

- **Key issues identified for EDA**
  - **Treatment of intellectual property (information and know-how)**
  - **Approach to task-sharing**

- **Evolutionary solutions based on development of further detail and experience appear to be required rather than determining solutions adopted at the outsdet due to complexity of issues**

- **Issues require innovative approaches for both design and construction**

BASIC PRINCIPLES RELATED TO TASK SHARING

- Two distinctly different approaches possible — centralized and decentralized
- A hybrid approach is most likely
- A strong central team is required with primary responsibility for overall design integration and the facility
- A desirable procurement system is one that does not require transfer of funds across the parties' boundaries
- Standards are required to assure equity among the parties

QUESTIONS RELATED TO TASK SHARING

- **What is the responsibility of the central team vs the home team**
  - **Design**
  - **R&D**
  - **Procurement**
  - **Installation**

- **Do home teams assume full or partial responsibility for systems**

- **At what level can private industry be involved**

- **How to divide tasks among home teams**

- **How is equity obtained**

- **When is the best time to divide**

- **Who will decide on all of the above**

- **How to ensure compatibility with domestic needs**

- **How to handle significant changes or new tasks**

- **How to take into account the different areas of concentration among the four parties**

- **How do "other" countries become involved**

- **Relationship of internal divisions to other parties**

HANDLING OF INTELLECTUAL PROPERTY

- Basic principles
- Ownership
- Transfer of intellectual property
- Protection of proprietary information

BASIC PRINCIPLES IN HANDLING OF INTELLECTUAL PROPERTY

- **Approach based on recognition of mutual benefit, need for equality, respect for international and domestic arrangements and legislation**

- **Technical success is top priority to ensure broad and prompt exchange of information while ensuring intellectual property rights of parties**

- **Intellectual property includes data, results and methods in the form of drawings, models, calculations, reports, equations, instructions, inventions, etc.**

- **Each party will receive regular reports on the project status**

OWNERSHIP OF INTELLECTUAL PROPERTY

- Property and copyrights of reports created by central team is owned equally by the four parties

- Each party will determine ownership between its national groups based on domestic requirements

- Inventions by central team members will be the responsibility of the director

- Inventions by home teams will be owned by the party and shared openly with other parties

- Inventions by home team members will be the responsibility of home team leaders who will report to the director

- In questions of ownership, the director with concurrence of home team leaders will recommend ownership to the Council

TRANSFER OF INTELLECTUAL PROPERTY

- **Inventions by home team members will be licensed for use at no charge to other parties**

- **Pre-existing property will be licensed for ITER purposes to other parties over a finite time**

- **Protected or sensitive, pre-existing information that a party wants to contribute could be used confidentially under agreed-upon rules**

- **Inventions by the central team will be transferred to "third persons" after approval by all parties**

PROTECTION OF PROPRIETARY INFORMATION

- **Information and know-how will not appear in reports produced by the central team**

- **Information used will be handled by parties in accordance with laws, rules, and administrative practices throughout a time period mutually agreed upon**

WORKSHOP SESSION I REPORT: WORLDWIDE FACILITIES PLANS FOR VARIOUS USER NEEDS

J. B. Hayter

Plans for Various User Needs

John Hayter (ANS, Chairman) presented an overview of neutron scattering and other beam research techniques, from the viewpoint of their impact on reactor design. The major change in emphasis in recent years has been the need for specialized sources of cold, very cold and ultracold neutrons, in addition to the more conventional thermal neutron beams. Research reactors now handle experiments on an incredibly wide range of subjects, from the most fundamental elementary particle physics through materials science and engineering to chemistry and biology. This leads to very specialized requirements, not only in the reactors, but in the surrounding infrastructure, such as the provision of sample-handling laboratories for, *e.g.*, biochemical materials. Some beam instruments, such as those used for studies of neutron optics, now have such extraordinary sensitivity that stringent anti-noise and anti-vibration requirements are imposed on certain experimental positions at reactor facilities.

In almost all areas of activity, conventional methods are expanding into new areas of application. In the case of materials irradiation, testing of materials for fission and fusion reactors continues apace, but considerable production irradiation also takes place, for example to dope silicon with phosphorus for the semiconductor industry, or to generate color centers in synthetic gemstones. Activation analysis is becoming an even more important tool than in the past, as more emphasis is placed on finding traces of pollutants in the environment. Another analytical tool, depth profiling, is also finding widespread use, particularly by the semiconductor industry.

Radioisotope production, both low-Z and transuranic, remains a necessity, providing sources for industrial radiography and cancer therapy, and as tracers in medical research. The need for explosives detection at airports has increased the need for californium, and this need may increase further if epithermal sources based on ^{252}Cf prove useful for neutron capture therapy. Special sources are also being requested. These are often of a very "hot", short-lived isotope, for example for positron production or for Mossbauer spectroscopy, and require special facilities in place.

Nuclear physics research has made steadily increasing use of reactors in the past two decades. Studies of neutron-rich nuclides by fission product spectroscopy requires local hot cell handling of small (mg) quantities of fissionable materials. Thermal neutron beams without background-creating material in the reflector ("through" tubes) are required for precision γ-spectroscopy, with other types of access needed for β-spectroscopy. Finally, in-pile loops of various types are always needed for testing, and for R&D on new ideas. These are often related to a primary function of research reactors, namely education and training.

Medium flux reactors ($\phi < 10^{19}$ m^{-2}.s^{-1}) are able to perform many of the tasks discussed above, and they are irreplaceable as training centers. Almost all new ideas have been conceived and taken through R&D at medium flux reactors, for well-understood reasons: time on high flux reactors is so scarce, relative to the number of users requesting it, that there is rarely the opportunity to undertake preliminary development projects at reactors such as HFBR, HFIR or ILL.

One of the most immediate needs for high flux reactor centers is thus a healthy community of medium flux reactors. The fundamental role of high flux sources is to push the state-of-the-art, especially in neutron scattering, which will always be signal-limited on any currently conceivable neutron source. Most of the new methods which have been initially developed at small reactors have led to improved resolving power, for which the cost is always a need for higher flux. As a result, the practical implementation of such new techniques as backscattering, contrast-matched SANS, or neutron spin echo, has required the highest flux possible. There is thus an essential symbiosis between medium and high flux reactor centers.

Otto Harling (MIT) continued the session with an number of comments on in-pile loops, irradiation techniques and medical therapeutic uses of neutrons, especially in the epithermal range. MITR is an example of the type of facility which can be designed for the latter purpose, and details will be found in his article elsewhere in these proceedings. Some other aspects of medical isotope use were discussed by Charles McKibben (MURR), and Ken Thoms (HFIR) spoke on materials irradiations for users from the nuclear industry. Finally, Bernard Farnoux (Saclay) and Hank Prask (NIST) offered commentaries on practical aspects of running user facilities, and the extent to which industries could be interested in joint instrumentation projects through the *Participating Research Team* (PRT) concept; many of these ideas will be found in the articles by these authors.

WORKSHOP SESSION II REPORT: R&D NEEDS OF IGORR MEMBERS

K. Boning

Report on the Workshop Session II:

R & D Needs of IGORR Members

K. Böning

On the first day of this meeting a variety of reports has been given on projects covering the range from major upgrades of existing research reactors or experimental installations up to totally new research reactor facilities. From these project descriptions it was clear that a strong need of all these groups for R & D (research and development) work had to be anticipated. So the purpose of the present workshop was - on the one hand - to identify those fields in which the results of R & D work would be required by IGORR member groups.

On the other hand - there is an appreciable amount of R & D work either being planned for the near future or being performed right now (or having just been completed) by IGORR member groups. These groups would - quite generally - be happy to share their results with other groups interested. So the purpose of this workshop was also to identify those fields in which R & D work actually is or will be (or has been) performed by IGORR members.

It seems obvious that the problem so far is one of communication, i.e. of bringing together those IGORR member groups which have common R & D interests. So it has been attempted during the workshop to establish a matrix in which the various fields of R & D activities could be compiled together with the various IGORR groups being either performing such R & D work or only interested in its results. It was hoped that in this way correlations would show up which could allow groups of common interests to come into contact with each other, to exchange information and perhaps to avoid duplicating efforts.

Having this in mind, overviews were first given on the workshop of the R & D work planned or required for the two projects aiming at totally new research reactors. Here R & D work is particularly important in the case of the Advanced Neutron Source ANS, a big and challenging project requiring new technology in many areas, whereas this demand is not so strong in the case of the new Research Reactor Munich FRM-II, a much smaller project relying essentially on established technology. Although these reports concentrated more on experimental R & D work, since this is usually most budget- and time-consuming, it was generally understood that non-experimental work (e.g. evaluation of data, check of correlations, etc.) is of comparable importance and should not be excluded from our considerations here.

After these two presentations a matrix has been set up as argued above, in which the various fields of R & D work were listed on the vertical and the IGORR member groups interested in this work on the horizontal axis. The symbol "+" has been introduced for that R & D work which a given group was not only interested in but determined to perform actively by itself, and the symbol "o" in those cases where a given group was only interested in being informed about the results of such work but did not intend to perform it by itself. Having established this matrix a lively discussion set in among the participants of the workshop which led to the matrix growing significantly in size. The final form of the matrix, in which the various IGORR member groups are listed in the order of their contribution to the discussion, is shown at the end of this paper.

A few comments should be given with respect to this table. First, it was generally regretted that the representatives of the High Flux Reactor at Grenoble (ILL) had to cancel their participation in the IGORR Meeting shortly since important contributions to the subject could be expected from this center. Further, it goes without saying that the matrix only represents a very rough, summarizing description of the actual situation since space does not allow to include any details here.

Nevertheless, some more specific comments shall be given with respect to the following topics of the matrix:

To 3.: It was felt that for small, undermoderated reactor cores which are surrounded by a high-quality moderator the distinction between the short lifetime of prompt neutrons remaining within the core and the long life of prompt neutrons diffusing back from the moderator into the core should be taken into consideration more explicitely.

To 6.: Includes the behaviour under irradiation of highly enriched silicide (ANS, FRM-II) and aluminide (MURRI) fuel.

To 7.: Refers mainly to boron poison in the form of either a cermet or an aluminum alloy.

To 8.: Interest seems to concentrate on Al 6061 (ANS, BNL) and zircaloy (ORPHEE, MAPLE) both representing well established research reactor materials. The available data include those of irradiated hafnium absorbers (MAPLE).

To 9.: Neutron transmission tests on irradiated guides.

To 10.: The data from Riso refer to Al 5052 cold source material.

Further, it was pointed out from the Canadian side (MAPLE) that - as far as commercial interests are being involved - one would have to distinguish between free and proprietory R & D results.

How shall we proceed for the future to make the best out of this evaluation? It would certainly not be a proper approach to distribute all available material on R & D results unspecifically among the IGORR members. It is rather being suggested here that those IGORR members having common R & D interests - according to the matrix - try to get into contact with each other. It has further been suggested during the workshop that an IGORR News-letter should be established which should appear in regular time intervals; this newsletter should contain a "R & D section" in which the titles and abstracts of reports on recent R & D work could be published so that interested IGORR members could directly contact the authors to ask for copies.

R & D work

topics: / being planned (+) or results needed (o):	ANS	FRM-II	MURRI	BNL	Riso	JAERI	Petten	Berlin	ORPHEE	MAPLE	MIT
1. Thermal-hydraulic tests and correlations	+	o	+	o		+					
2. Corrosion tests and analytical models	+	o	o								o
3. Multidimensional kinetic analysis for small cores	o	o								+	
4. Fuel plates fabrication	+	+									
5. Fuel plates stability	+	+	o								
6. Fuel irradiation	+	o	+			+					
7. Burnable poison irrad.	+	+									
8. Structural materials irradiation	+	o	+	+			+	o	+	+	
9. Neutron guides irradiation	o	o				+					
10. Cold Source materials irradiation	o	o			+			o	+		
11. Cold Source LN_2 test	+										
12. Cold Source LH_2-H_2O reaction (H or D)	o	?		+		+					
13. Instrumentation upgrading and digital control system	+		o	o			+		+		
14. Man-machine interface	o	o									+

comments:
+ results needed and own work/tests planned
? results needed, but own tests not decided yet
o results needed, but own work not planned

REPORT ON IGORR ORGANIZING COMMITTEE MEETING

On the final day of the First IGORR Meeting, the original Organizing Committee held an open meeting to discuss the purpose and structure of IGORR, possible plans and schedules for the next meeting, and plans for interim activities.

The members present decided that IGORR is indeed an organization with a useful role and should continue. A charter was agreed upon and is printed at the beginning of the Proceedings. The Group plans to concentrate on new reactors or planned upgrades, a field that did not previously have any specialist international forum, although IGORR will continue to seek interaction with other organizations representing research reactor operators and users, including university reactors, the IAEA, and the European research reactor community. Future IGORR meetings will emphasize new designs, new equipment, safety, and new data (such as materials data) of primary interest to members.

To further the sharing of knowledge among members, B. Farnoux volunteered to produce an IGORR newsletter. The newsletter will include announcements of proposed upgrades or new projects and will list recent publications of research and other results of importance to the community.

The group proposed to hold the next meeting in Fall 1991, perhaps in conjunction with a meeting on irradiation effects scheduled at that time. B. Farnoux volunteered to host the meeting at Saclay.

Two new members were added to the organizing committee - Monsieur C. Desandre (Technicatome) and Professor H. Nishihari (Kyoto University). Colin D. West was elected Chairman.

John Axe kindly offered to publish a short summary of the meeting in *Neutron News.*

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