IGORR 5

Photo Philippe Benet & Renata Holubachova

INTERNATIONAL GROUP ON RESEARCH REACTORS

NOVEMBER 4– 5– 6, 1996
AIX–EN–PROVENCE / FRANCE

technicatome

IGORR 5

INTERNATIONAL GROUP ON RESEARCH REACTORS

5th MEETING

NOVEMBER 4-5-6, 1996
AIX-EN-PROVENCE / FRANCE

GRAND HOTEL ROI RENE

PROCEEDINGS

PREFACE

Overview

The fifth meeting of the International Group on Research Reactors (IGORR-V) was held in Aix-en-Provence, France, on November 4-6, 1996. Attendance was excellent (between 70 and 80 participants from 21 countries). Thirty-one papers were presented in four sessions over the two-day meeting, and written versions of the papers or hard copies of the viewgraphs used are published in these Proceedings. A book of abstracts was published and handed out at the beginning of the meeting, which proved to be very useful to the participants.

The meeting was a huge success, and we look forward with much anticipation to IGORR-VI.

I should like to congratulate Klaus Böning on his election to Chairman of IGORR. I am certain that he and his colleagues will do an excellent job. It has been my pleasure to be associated with the IGORR membership, and I hope to meet you all again some day.

Kathy F. Rosenbalm

IGORR 5

OPENING SESSION

Colin D. West - ORNL
Yannick Le Corre - Technicatome
René Ginier - CEA

Welcome to the fifth meeting of the International Group on Research Reactors (IGORR). I should like first to thank our French colleagues, and the Commissariat A l'Energie Atomique (CEA) for their splendid efforts on organizing this meeting, which has a very strong international participation. I should also like to thank our Technical Program Coordinator, Kathy Rosenbalm, whom most of you know but who, unfortunately, could not be present.

IGORR has become stronger, over the years since it was founded. Clearly, our organization has a useful role to play within our community of people working on new and improved research reactor facilities. Its' vitality is evidenced by the increased number of participants in this meeting and by the continued evolution and changes in our agenda. For example, in making the preparations for this meeting we found so much interest in and so many contributions concerning cold neutron sources that we have included for the first time a whole session devoted to papers on that topic. In addition, this year there are reports on two separate surveys that were initiated by the last IGORR meeting: one, by Albert Lee, on containment design criteria and one, by Doug Selby, on cold neutron cross sections.

Another innovation this year, thanks to the CEA organizers, is the preparation, before the meeting, of a book of abstracts of the talks: I think we will all find this helpful in the next couple of days. Of course, the full Proceedings will be produced and published as soon after the meeting as practicable: please submit your contributions to the organizers on time.

I should particularly like to welcome, for the first time, representatives of some of the institutes in former Eastern European countries. I am sure their experiences and ideas will be interesting and helpful to us all.

After the technical sessions, there is to be a visit to Cadarache, which I know will be a new experience for many of us. In addition, I should like to thank very much the mayor of Aix, who has kindly and graciously invited us to a reception on Monday evening: a very nice welcome and gesture from our host city.

The agenda is a full one, and full of interesting papers. The session chairmen have been instructed to keep speakers strictly to the time limit, so that their colleagues later in the session will not be deprived of their fair share of time!

Let us begin IGORR-V.

IGORR 5 MEETING NOVEMBER 4, 1996

WELCOME ADDRESS

M. YANNICK LE CORRE

(TECHNICATOME Chairman)

¤¤¤¤¤¤¤¤¤

Dear Colin WEST,
Dear Bertrand BARRE,
Dear audience,

I am really happy to welcome you for the 5th meeting of the International Group on Research Reactors :

I am really happy because it's the second time since the creation of IGORR in Nineteen Ninety (1990), that TECHNICATOME has the pleasure to organise in a close collaboration with the Commissariat à l'Energie Atomique the IGORR meeting in France. In Paris in June Nineteen Ninety Two (1992) IGORR Two (2) gathered Forty Five (45) participants. Today, in Aix-en-Provence, we are more than Eighty (80), including for the first time new representatives of some Eastern Europe countries.

This shows the interest of the Research Reactor Community for IGORR activities. Thanks to Colin WEST for having created this group.

I am also happy because many of you, coming from foreign countries will have the opportunity to discover the fascinating city of Aix-en-Provence, the «City of Arts ».

The Nuclear Research Reactors have also an important past.

They have played a **key role** in the history of the Nuclear Energy and my opinion is that they will still play an important part.

More than Three Hundred and Seventy (370) Research Reactors and critical mock-ups were erected during the past Fifty (50) years in the world. More than Two Hundred and Eighty (280) are still in operation, and among them Eighteen (18) in France.

Let me remind you some aspects of their contribution to the Nuclear Energy development :

- with the first critical mock-ups, the feasibility of the fission nuclear chain reaction announced in August Nineteen Thirty Nine (1939) by Albert EINSTEIN to Franklin ROOSEVELT, on the basis of the work performed by Frédéric JOLIOT, Enrico FERMI and Leo SZILLARD was demonstrated : the first reactor CP1 went on criticality in Chicago on December Nineteen Forty Two (1942) . In France, ZOE started on December Nineteen Forty Eight (1948) ; It was the beginning of French Nuclear Reactor history which led to celebrate in Nineteen Ninety Five (1995) the Fifty (50) years of existence of the « Commissariat à l'Energie Atomique » with the background of one of the world biggest nuclear programs and a major contribution to the independence of our Nation.

- The first generation of Research Reactors enabled the determination of numerous physical parameters and gave the confirmation of the behaviour of fission Nuclear Reactors, therefore giving the basis for the development of large nuclear power plants.

- Afterwards many irradiation programs had to be performed in Research Reactors to qualify fuel and material technology for the future reactors. For this purpose sophisticated in core loops were also introduced in Research Reactors allowing fuel testing in representative thermodynamic conditions.

- Up to now in the world, Research Reactors have demonstrated their extended possibilities :

  ◊ in the field of fundamental and applied research,
  ◊ in the field of material, component and fuel testing,
  ◊ in the field of radioisotope production,
  ◊ in the field of teaching and training.

- Very early safety aspects have also arisen, leading to specific programs on dedicated facilities, like CABRI and PHEBUS in France. CABRI allowed the study of the cinetics of cores in case of sudden reactivity introductions. On PHEBUS, the consequences of severe degradations of fuel are studied under an international collaboration. I know that you will visit these facilities in Cadarache on November Six (6).

- For the scientific community, Research Reactors appeared to be also powerful tools for fundamental research and medicine applications.

In the Nuclear developed countries, the high level of demands led to the specialization of the Research Reactors. In France, it's the reason why after the first generation of multipurpose reactors like MELUSINE, SILOE and then OSIRIS, specialized high flux reactors were built, in Nineteen Seventy Two (1972) the RHF in Grenoble, and in Nineteen Eighty (1980) the ORPHEE reactor in Saclay. In a complementary way, OSIRIS and SILOE activities were focused on irradiation experiments and radioisotope production ; both still operate with heavy programs.

What's future for the Research Reactors ?

Today we have to acknowledge than most of existing Research Reactors are of pool type. They proved their versatility and their easy adaptability to new irradiation programs. High flux reactors obtain higher fluxes with the help of an additional reflector made of Beryllium or heavy water.

Pool type Research Reactors appear to be a good investment on the condition that enough attention is payed to their design, taking into account the experience acquired by the years of operation of the previous Reactors.

In the countries who developed for many years a nuclear program, a high level of demand exists with a resulting specialization of the research facilities and with even more high neutron flux requirements. There are needs for :

- on one hand, irradiation reactors for qualification of fuel and materials including fast reactors and fusion programs, and radioisotope for medicine and industry,

- on the other hand, reactor for neutron beam delivery with a growing demand for cold neutrons.

In a complementary way, safety assessment studies will certainly need the continuation of ongoing programs on dedicated facilities.

For the countries who recently decided to develop a Nuclear Research or Energy program, a multipurpose pool type Research Reactor is certainly a good choice if it is done in association with a coherent development policy for the national scientific community. This kind of Research Reactor will allow access to most of the applications I mentioned before.

The France who successfully managed a huge electronuclear program has developed the highest level of expertise in Research Reactor design, as well in CEA as in the associated industrial partners. Numerous collaborations with nuclear developing countries are on the way now and will benefit from this high skill level.

TECHNICATOME, with CEA, is willing to contribute to this effort.

Mister Chairman, I do believe that Nuclear Energy will remain a key technology for the future. Research Reactors can be in the next decades useful and versatile tools, contributing to this future in the frame of qualification programs for fission and fusion energy.

The Reducing Enrichment in Research and Test Reactors program, I mean RERTR program, aiming at demonstrating that most of existing Research Reactors can be converted to low enriched uranium silicide fuel, with a comparable level of performances, is of a great interest, opening the field for wide peaceful applications.

New projects will of course benefit from this new fuel technology. In France, we are preparing the necessary renewal of the existing irradiation Research Reactors.

Mister Chairman, dear audience, there are still a lot of things to do ; doubt and humility are necessary in any scientific approach. After years of success the major error would be to believe that all technical aspects have been solved. Chernobyl and TMI are here to remind us this reality.

Fighting against high economical constraints, our countries need to maintain for the future a scientific and engineering community with the highest skill level in the nuclear field.

I am sure that IGORR, facilitating exchanges between countries, is working in that direction for the world scientific development.

I wish you three interesting working days of technical debates and visits.

Mister Chairman, Ladies and gentlemen

I am very glad to have the opportunity to say some words for introducing this IGORR meeting. But before that, I have to apologize Mr Barré for not being here as it was anticipated. To day he has to be in Paris and as Deputy Dr I will do my best to represent him here.

Our Direction , the Nuclear Reactor Division is in deed strongly interested in the IGORR works, firstly because we are operating a certain number of research reactors, OSIRIS and ORPHEE in Saclay, SILOE in Grenoble, PHENIX in Marcoule, as well as critical mock-up like EOLE and MASURCA, in Cadarache, but also because we are great users of results obtained with all these Research Reactors.

"Nuclear energy will continue to play a central role in french energy policy". This strong declaration is not from me, ladies and gentlemen, although I quite agree with it, but was made recently by the french industry minister.

To support this declaration there is a lot of good reasons but there is to my feeling two basic conditions : public acceptance and competitivity

Public acceptance, great efforts are made to-day in direction of the public, in conferences, through media, but one of the best way to get the public acceptance is to show that power reactors can be operated without any significant accident. To prevent such event a lot of works has been done and is still under way on experimental loops or experimental reactors like Phébus or Cabri that some of you will visit after- to-morrow. There, in the frame of international programmes , are performed tests to determine the limits of service of fuel elements under accidental transients.

The second condition, **competitivity** will be, more than ever, the key factor. Due to the progress made by other sources of energy, nuclear energy has to improve its own competitivity. All the constructors are giving a high priority to these aspects : extending the life of the reactors to 40, 50 even 60 years, getting better availability through longer fuel cycles and reducing fuel costs with higher burn up rates. All possibilities are explored and, nuclear designer will demand more and more to the materials : fuel, cladding, internals material. We must better understand their comportement under irradiation, better predict their evolution under transients and make the better choice for these materials. New need appear to-day related to the necessary stabilization of plutonium stock-piles, transmutation of minor actinides. To achieve these goals important programmes are under way in experimental reactors like OSIRIS and SILOE and will be pursued in the next years.

However the neccesity of economy applies also to research reactors and our Division will concentrate all the programmes in OSISRIS in Saclay, the Siloe reactor being shut down at the end of the next year.

The importance of nuclear energy in France justifies that CEA continues to have to its disposal research reactors specially for material testing. It's the reason why we are soon thinking to the following reactor after OSIRIS which will be in operation still about a decade.

You will have to-morrow a first presentation of this future reactor ; it has soon a name : Jules Horowitz Reactor, from the name of a prominent physicist of CEA who died last year.

You see that we are confident in the future of nuclear energy : nothing is definitly acquirred, we have to compete but we will give us the best tools in order to maintain the economic advantages of nuclear energy

In that perspective, research reactors have an important role to play and I am sure that IGORR will contribute to draw a better profit of our tools.

I wish to IGORR a fruitful meeting

Thank for your attention.

IGORR 5

AGENDAS OF THE MEETING

IGORR 5

OVERALL AGENDA IGORR 5

SUNDAY NOV 3		
18H00	20H00	*Registration*

MONDAY NOV 4				
08H30	09H30	*Registration*		
09h30	10h15	*Opening session*		
		• Opening IGORR 5	C.WEST	ORNL
		• Welcome speech	Y. LE CORRE R. GINIER speaking for B. BARRE	TA CEA/DRN
		Practical information		
10h15	10h30	*Coffee break*		
10h30	12h30	Session 1A	Operating research reactors	(6 papers)
12h30	14h00	*Lunch*		
14h00	16h00	Session 1B	Operating research reactors	(5 papers)
16h00	16h15	*Coffee Break*		
16h15	17h00	Session 1C	Operating research reactors	(2 papers)
17h00	17h45	Session 2A	Research Reactors in design or construction	(2 papers)
19h30		*Reception at the City Hall*		

TUESDAY NOV 5				
08h30	10h30	Session 2B	Research Reactors in design or construction	(5 papers)
10h30	10h45	*Coffee Break*		
10h45	12h30	Session 3A	Cold Neutron Sources	(5 papers)
12h30	14h00	*Lunch*		
14h00	15h00	Session 3B	Cold Neutron Sources	(4 papers)
15h00	16h00	Session 4	Workshop on containment survey	(2 papers + discussion)
16h00	16h15	*Coffee break*		
16h15	17h15	Session 5	Workshop on R & D needs	
17h15	18h00	*Business meeting and Closing session*		

IGORR 5

CEA CADARACHE TECHNICAL VISIT
PHEBUS - CABRI - CASCAD - TORE-SUPRA FACILITIES

Wednesday November 6, 1996

8.45 a.m.	◆	Meeting point at the hotel
9.00 a.m.	◆	Departure to Cadarache
9.45 a.m.	◆	Arrival at Cadarache
9.45-10.00 a.m.	◆	Formality requirements
10.00-10.30 a.m.	◆	Welcome speech
10.30-10.40 a.m.	◆	Bus trip to Phebus

<table>
<tr><th>Group 1</th><th>Group 2</th></tr>
<tr><td>10.40-11.30 a.m. ◆ Visit of Phebus

11.45-12.30 p.m. ◆ Visit of Cabri</td><td>10.40.10.50 a.m. ◆ Bus trip from Phebus to Cascad
10.50-11.25 a.m. ◆ visit of Cascad
11.25-11.35 a.m. ◆ Bus trip from Cascad to Tore Supra
11.35-12.20 p.m. ◆ Visit of Tore Supra
12.20-12.30 p.m. ◆ Bus trip from Tore Supra to Cabri</td></tr>
<tr><td colspan="2">12.35-12.45 p.m. ◆ Bus trip to lunch
13.00-14.30 p.m. ◆ Lunch
14.30-14.40 p.m. ◆ Bus trip from lunch to Phebus</td></tr>
<tr><td>14.40-14.50 p.m. ◆ Bus trip from Phebus to Cascad
14.50-15.25 p.m. ◆ Visit of Cascad
15.25-15.35 p.m. ◆ Bus trip from Cascad to Tore Supra
15.35-16.20 p.m. ◆ Visit of Tore Supra
16.20-16.30 p.m. ◆ Bus trip from Tore Supra to Cabri</td><td>14.40-15.30 p.m. ◆ Visit of Phebus

15.45-16.30 p.m. ◆ Visit of Cabri</td></tr>
<tr><td colspan="2">17.00 p.m. ◆ Departure (Arrival in Aix en Provence at 17.45 p.m.)</td></tr>
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IGORR 5

DETAILED AGENDA

1	OPERATING RESEARCH REACTORS						4h45
	SESSION 1A - Chairman Bernard Farnoux						
15	HFIR UPGRADE PROPOSALS	***CD WEST*** - MB. FARRAR	OAK RIDGE NATIONAL LABORATORY	ORNL	USA		
6	OPERATION EXPERIENCE AND CURRENT STATUS OF HANARO	***KH. LEE***, HR. KIM - BJ. JUN - JB. LEE	KOREAN ATOMIC ENERGY RESEARCH INSTITUTE	KAERI	KOREA		
13	CURRENT STATUS OF RESEARCH REACTORS RA AND RB AT THE VINCA INSTITUTE OF NUCLEAR SCIENCES	***M. KOPECNI*** - M. MATAUSEK - D. STEFANOVIC	VINCA INSTITUTE OF NUCLEAR SCIENCES	VINCA	YUGOSLAVIA		
9	STUDSVIK'S R2 REACTOR - REVIEW OF RECENT ACTIVITIES	***M. GROUNES*** - C. GRÄSLUND- M. CARLSSON - T. UNGER - A. LASSING	STUDSVIK NUCLEAR AB	STUDSVIK	SWEDEN		
28	MAIN EXPERIENCES IN RENOVATION OF THE DALAT NUCLEAR RESEARCH REACTOR	TRAN H.A. - ***PHAM V.L.*** - NGUYEN N.D. - NGO P.K.	INSTITUT DE RECHERCHE NUCLEAIRE VIETNAM	IRN	VIETNAM		
2	UPDATE ON THE BR2 REFURBISHMENT	***E. KOONEN***	STUDIE CENTRUM VOOR KERNENERGIE	SCK/CEN	BELGIUM		
	SESSION 1B - Chairman Christian Desandre						
27	RESEARCH AND SERVICES OF LVR-15 REACTOR IN REZ	***J. KYSELA*** - O. ERBEN - V. KNOBLOCH - J. BURIAN -V. BROZ	NUCLEAR RESEARCH INSTITUTE REZ PLE	NRIR	CZECH REPUBLIC		
1	HIFAR MAJOR SHUTDOWN REPORT INSPECTION OF REACTOR ALUMINIUM TANK AND REPLACEMENT OF SECONDARY COOLING WATER CIRCUIT PIPEWORK	***S. KIM***	AUSTRALIAN NUCLEAR SCIENCE AND TECHNOLOGY ORGANIZATION	ANSTO	AUSTRALIA		
26	MODIFICATION OF JRR-4	***T. NAKAJIMA*** - M. BANBA - Y. FUNAYAMA - Y. HORIGUTI - M. ISSHIKI	JAERI	JAERI	JAPAN		
23	OVERVIEW ON THE MAIN ENGINEERING WORKS PERFORMED ON FRENCH RESEARCH REACTORS THESE LAST YEARS	***P. ROUSSELLE*** - G. DE SAINT OURS J. GUIDEZ - C. JOLY - M. MAZIERE H. GUYON	TECHNICATOME/CEA	TA/CEA	FRANCE		
19	UTILIZATION OF THE BUDAPEST RESEARCH REACTOR	***I. VIDOVSZKY***	KFKI ATOMIC ENERGY RESEARCH INSTITUTE	KFKI	HUNGARY		
	SESSION 1C - Chairman Kir Konoplev						
7	DESIGN MODIFICATION OF HANARO REFLECTOR COOLING SYSTEM	***J.S. WU*** - S.Y. HWANG - Y.K. KIM	KOREAN ATOMIC ENERGY RESEARCH INSTITUTE	KAERI	KOREA		
35	IAEA ACTIVITIES ON RESEARCH REACTOR SAFETY	***F. ALCALA RUIZ***	INTERNATIONAL ATOMIC ENERGY AGENCY	IAEA	VIENNA		

Bold & italic name = speaker

2	**RESEARCH REACTORS IN DESIGN OR CONSTRUCTION**						**2h45**
SESSION 2A - Chairman Albert Lee							
21	STATUS OF THE TRR II PROJECT	LF. LIN - WM. CHIA - CY. YANG - HS. SHEU - CC. WANG - ***DJ SHIEH-***	INSTITUTE OF NUCLEAR ENERGY RESEARCH	INER	TAIWAN ROC		
10	THERMOHYDRAULIC AND MECHANICAL ANALYSIS OF A SCALE FRM-II CORE DUMMY	***J. ADAMEK*** - H. UNGER	RUHR UNIVERSITÄT BOCHUM	RUB	GERMANY		
SESSION 2B - Chairman Jean-Luc Minguet							
4	A STATUS REPORT ON THE PROPOSED CANADIAN IRRADIATION RESEARCH FACILITY	***A.G. LEE*** - W.E. BISHOP - G.E. GILLESPIE	ATOMIC ENERGY OF CANADA, Ltd	AECL	CANADA		
20	PRELIMINARY STUDY OF CORE CHARACTERISTICS FOR TRR-II	JT. YANG - LS. KAO, HM. HSIEH - DY. YANG JA. JING - ***SK CHEN***	INSTITUTE OF NUCLEAR ENERGY RESEARCH	INER	TAIWAN ROC		
12	STATUS OF THE FRM-II PROJECT	***HJ. DIDIER*** - G. WIERHEIM	SIEMENS AG KWU	SIEMENS	GERMANY		
14	CONSTRUCTION OF THE HTTR AND ITS IRRADIATION PROGRAM	***M. ISHIHARA*** - T. KIKUCHI, J. AIHARA - H. MOGI -T. ARAI, T. TANAKA	JAPAN ATOMIC ENERGY RESEARCH INSTITUTE	JAERI	JAPAN		
18	THE JULES HOROWITZ REACTOR (R.J.H.) THE CEA FUTURE TOOL FOR TECHNOLOGICAL IRRADIATIONS	S. FRACHET - P. MARTEL, B. MAUGARD - P. RAYMOND - ***F. MERCHIE***	COMMISSARIAT A L'ENERGIE ATOMIQUE CADARACHE	CEA	FRANCE		

Bold & italic name = speaker

3	**COLD NEUTRON SOURCES**						**02h45**
	SESSION 3 - Chairman Hans Joachim Röegler						
3	EXPERIMENTS WITH COLD SOURCES FOR NEUTRON PHYSICS ANALYSIS	***K.A. KONOPLEV*** - V.A. KUDRJASHOV - G.D. PORSEV - I.A. POTAPOV - V.A. TRUNOV - G.JA VASILEV - A.S. ZAKHAROV	PETERSBURG NUCLEAR PHYSICS INSTITUTE	PNPI	RUSSIA		
8	THE COLD NEUTRON SOURCE AND OTHER IN-PILE EXPERIMENTAL FACILITIES OF THE NEW RESEARCH REACTOR FRM-II IN GARCHING	***K. GOBRECHT***- E. STEICHELE	TECHNICAL UNIVERSITY OF MUNICH	TU MÜNCHEN	GERMANY		
11	POST IRRADIATION EXAMINATION OF Z6 NCT 25 STAINLESS STEEL FROM SF2 COLD SOURCE CELL OF THE REACTOR ORPHEE	M. MAZIERE	COMMISSARIAT A L'ENERGIE ATOMIQUE SACLAY	CEA	FRANCE	paper presented by ***B. FARNOUX***	
33	SUMMARY OF HFIR COLD SOURCE PROJECTS	***D. SELBY***	OAK RIDGE NATIONAL LABORATORY	ORNL	USA		
37	PELLETIZED METHANE COLD SOURCE CONCEPT	A.T. LUCAS - PERFORMED BY ***D. SELBY***	OAK RIDGE NATIONAL LABORATORY	ORNL	USA		
29	PERFORMANCE OF THE NIST LIQUID HYDROGEN COLD SOURCE	JM. ROWE - P. KOPETKA - ***RE. WILLIAMS***	NATIONAL INSTITUTE FOR STANDARDS AND TECHNOLOGY	NIST	USA		
32	DESIGN REVIEW IN CONCEPT PHASE OF CNS AT HANARO	***CO. CHOI*** - KN. PARK, JM. SOHN, SH. PARK, MS. CHO	KOREAN ATOMIC ENERGY RESEARCH INSTITUTE	KAERI	KOREA		
34	COLD NEUTRON CROSS SECTIONS - HISTORICAL REVIEW	***D. SELBY***	OAK RIDGE NATIONAL LABORATORY	ORNL	USA		
36	GKSS - CNS : THE POSSIBILITY OF NATURAL CONVECTION OF THE GASEOUS HYDROGEN MODERATOR	***W. KNOP*** - W. KRULL	GKSS	GKSS	GERMANY		

4	**SESSION 4 -WORKSHOP ON CONTAINMENT SURVEY - Chairman Robert Williams**						**01h00**
5	RESULTS OF A SURVEY ON THE DESIGN BASIS FOR RESEARCH REACTOR CONTAINMENT/CONFINEMENT BUILDINGS	***A.G. LEE***	ATOMIC ENERGY OF CANADA Ldt	AECL	CANADA		
22	FRENCH RESEARCH REACTORS - DESIGN OF REACTOR BUILDING IN ACCORDANCE WITH SAFETY APPROACH AND IAEA RECOMMENDATIONS	P. ROUSSELLE - ***JL. MINGUET*** - F. ARNOULD	TECHNICATOME	TA	FRANCE		

5	**WORKSHOP ON R & D NEEDS AND RESULTS- Chairman Klaus Böning**	**01h00**
6	**BUSINESS MEETING & CLOSING SESSION**	**01h00**

Bold & italic name = speaker

IGORR 5

SESSION 1

OPERATING RESEARCH REACTORS

PAPERS

HFIR UPGRADE PROPOSALS

C. D. West and M. B. Farrar
Oak Ridge National Laboratory
Oak Ridge, Tennessee, USA

presented to the
5th meeting of the
International Group on Research Reactors

November 4, 1996

Why Upgrade The HFIR?

- There are important scientific and isotope production needs that can only be met with a high-powered research reactor
  - This has been recognized by all major reviews of the national needs for neutron sources
- The HFIR has the highest neutron flux (the single most important measure of a neutron source's capabilities) of any research reactor in the western world
- The HFIR is a multipurpose research reactor, and its upgrade will benefit many areas of science
  - Neutron scattering
  - Isotope production
  - Neutron activation analysis
  - Materials irradiation testing
  - And others
- With minor improvements, the HFIR can operate for many more years (at least until 2030)

Besides Possible Improvements To Existing Capabilities, What Major Facilities Are Lacking At HFIR?

- Cold neutron beams
  - Cold neutrons (low energy, long wavelength neutrons) have become increasingly important as neutron scattering science has tackled more and more subtle and complex problems
  - HFIR is the only major research reactor in the world without a cold neutron source (which would boost the flux of cold neutrons by an order of magnitude)
- Instrument space
  - All new and recently modernized research reactor facilities have large halls or beam rooms to accommodate many big scattering instruments and their shielding
  - HFIR was originally designed and optimized for isotope production and has only very limited space, inside the reactor building, for instruments

Other Desires In A HFIR Upgrade Program

- Schedule the upgrades to minimize risk, cost and down time

- Coordinate with other neutron facility improvements to provide continuous service to user communities

- Capitalize on our ANS experience (reactor and instrumentation design, cold source development, thermal neutron guides, etc.)

HFIR Futures Group

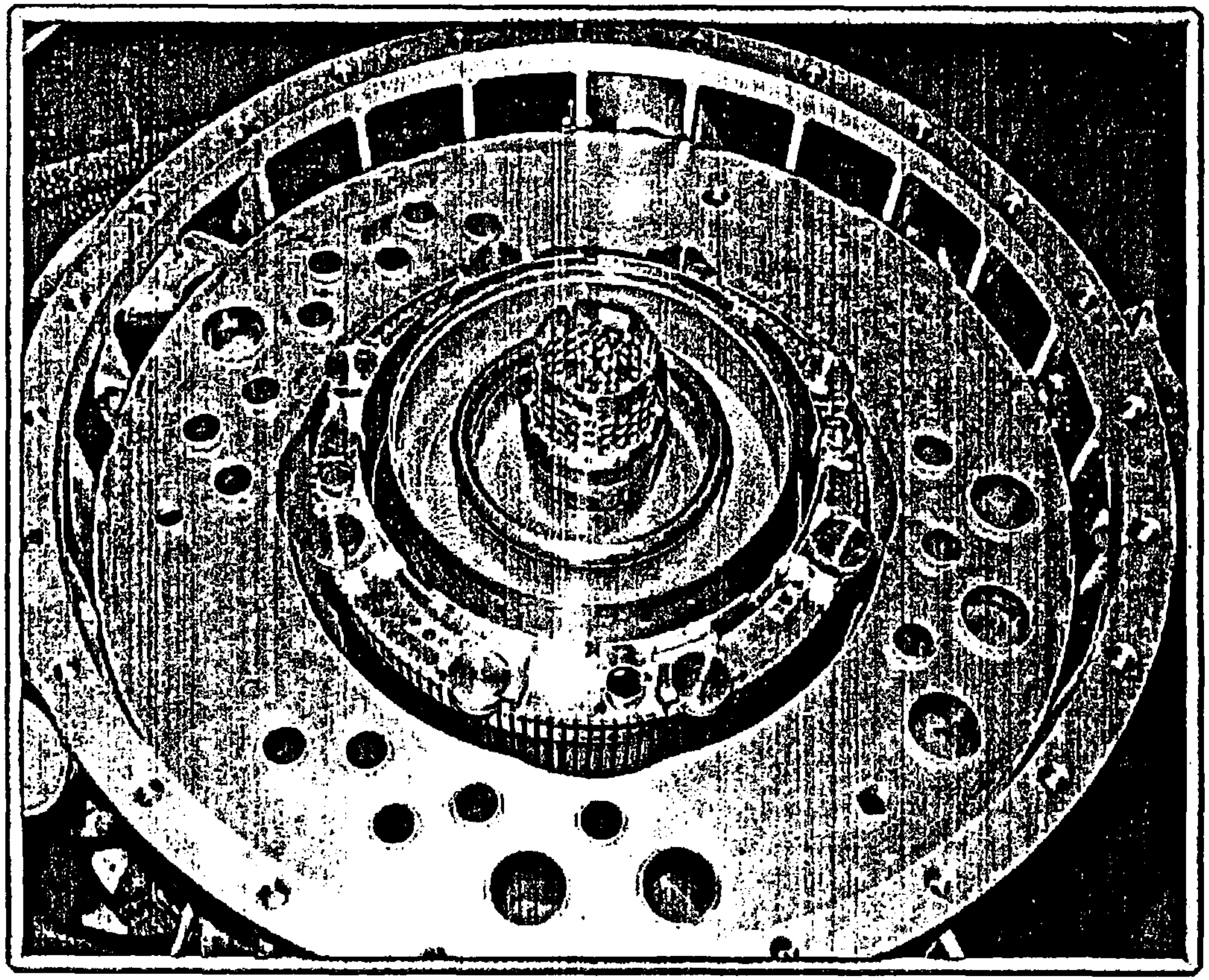
- In the spring of 1995, a HFIR Futures Group was formed to consider enhancements to the reactor's performance and capabilities
- The group, representing the interests of all parties at the laboratory, met over a period of several months
- The group, with much helpful input from many colleagues, prepared the basis of an upgrade proposal package that ORNL later presented, in a slightly modified form, to DOE's Basic Energy Sciences Advisory Committee
  - The committee endorsed these proposals and recommended their implementation

Summary of Proposed HFIR Upgrades

- Return to 100 MW operation
- Life extension improvements: Be replacement/upgrade, etc.
- Cold source: New beams & new beam hall: cold neutron intensities $\geq$ ILL
- Thermal neutrons: 5 new thermal beam guides & new guide hall. Thermal neutron flux at samples $\geq$ 5xILL (using new ideas to match the area and shape of the source to the acceptance angles of the guides)
- Isotope production: Improved spectrum and access
- Materials irradiation and testing: Improved access
- Neutron activation analysis: Improved access and handling, more irradiation space
- Remote handling facility for processing, maintenance and packaging

ORNL-DWG 96Z-6973

Top Side of Reactor and Beryllium Reflector



BRIEF DESCRIPTIONS OF UPGRADE, COLD SOURCE, AND BERYLLIUM CHANGEOUT/ISI PROPOSALS

Create Detailed Monte Carlo Model

A detailed Monte Carlo (MCNP) model of the High Flux Isotope Reactor (HFIR) is essential for the evaluation of any proposed modifications that impact the reactor assembly. The model has been completed and will be modified and updated as appropriate in the future.

Prepare Safety Analysis for 100 MW Operation

By selection of fuel elements with favorable channel dimensions from the present inventory, and later by adopting improved fuel plate inspection/selection procedures that were developed during the ANS Project, it will be possible to operate HFIR at 100 MW again with the present coolant inlet pressure and temperature while maintaining safe operating conditions for the embrittled reflector vessel. This will increase the neutron flux by 18%, benefiting all the missions of HFIR.

Cold Neutron Source

HFIR is the only high-power research reactor in the world whose capabilities do not include a source of cold (i.e., very low energy, long wavelength) neutron beams: such beams are now used for much of the most important neutron scattering research and basic nuclear physics experiments at the reactors. The liquid hydrogen cold source to be installed in HB-4 will provide gain factors of 5 to 30, depending on the wavelength, in the low energy neutron flux available at HFIR. The intensity of the new beams will be as high as or higher than those available anywhere in the world, although geometrical and space limitations will only allow three or four beams, at most, to be extracted, compared with at least eight at ILL.

New Beam Tubes

The present beam tubes at HFIR are about 4 in. in diameter, subtending only a small solid angle at the core, which greatly restricts the number of neutrons that make it all the way down the tube without being lost in the walls. Most scattering experiments would benefit if the tubes were flared between the reactor pressure vessel and the concrete shield around the pool to give a larger area and therefore pass more neutrons. Enlarging the beam tubes inside the reactor pressure vessel, to the extent allowed by the vessel openings and the core support structures, will also increase the number of neutrons available for scattering experiments.

Harden Spectrum in Flux Trap

The central hole in the HFIR core, a region of about 5 in. in diameter and 20 in. deep, is called the flux trap or target region and has the highest steady-state neutron flux in the world. Hydrogen in the light water coolant effectively slows down (moderates) neutrons released from fission reactions in the core, so that the neutron energy distribution in the trap is that of any material at a temperature of around 160°F. However, it is now known that a slightly higher energy (faster, "harder," or more epithermal) spectrum would be more effective in producing some of the transuranic elements. Displacing some of the water in the trap with solid aluminum rods - aluminum is a less effective moderator than hydrogen - would give a harder spectrum and, hence, increase the transplutonium production rate. This is a very simple, low-cost enhancement, and experiments are already underway to measure the benefit. Another approach under consideration is to fabricate the transplutonium production capsules from stainless steel, which would absorb thermal neutrons, leaving a harder spectrum within the capsule, instead of aluminum as at present.

Multiple Hydraulic Rabbit Tubes

The optimal production of some specialized isotopes requires that the target material be irradiated for only a fraction of a normal operating cycle. The existing hydraulic rabbit facility provides space for up to nine rabbits (2-in. long capsules) in the highest flux region of the reactor; its primary purpose is the production of medical R&D isotopes and some production medical isotopes. More such facilities are needed as demand increases, and a new magazine-fed multiple-tube hydraulic rabbit facility design will allow the irradiation of rabbits in six positions. This will increase the production capability from 9 to 54 rabbit capsules.

Rabbit Holders in Target Region

Special "peripheral target assemblies" to be inserted in one of the highest thermal flux regions of the reactor, are to be designed and fabricated to hold up to eight "hydraulic rabbit capsules." These targets will allow the irradiation of isotopes that are needed in some small quantities without the fabrication of a full-size target: the method will be most useful in the production of medical isotopes, such as Sn-117, that require irradiation for more than one operating cycle. The shipping and processing costs of the rabbits are lower than for full size targets. The rabbits are also much easier to handle and ship than larger targets, resulting in a quicker response to demand, which will result in a more cost-effective and responsive medical isotopes distribution program.

Neutron Activation Analysis Pneumatic Rabbit Tubes

There are currently two NAA rabbit tubes located in the HFIR core. The current facility is limited in the volume of the sample that can be irradiated. A new design will provide larger flight tubes and loading stations to handle larger rabbits. This will allow the irradiation of larger samples or even multiple samples in each rabbit. Additionally, provisions for delayed neutron counting will be added.

Neutron Radiography/Tomography Facility

In tomography, images or radiographs of successive planes in an object are combined to give a complete, three-dimensional view (CAT scanning is a familiar example from the medical world). The laboratory directed research and development (R&D) funding has been granted to demonstrate neutron tomography, which will add to the existing X-ray tomography technique the penetrating power of neutrons and their ability to "see" light materials such as lubricants or inclusions. For example, it will be possible to observe, by electronic gating of the signals, the changes taking place inside an operating engine at different times during the cycle. Assuming the demonstration is successful, a world-beating operational facility will be installed as part of the upgrade program.

New Gamma Irradiation Facility

The spent fuel elements from HFIR are highly radioactive, producing copious gamma rays, and even though the activity declines fairly quickly, they have been used as a source for testing the effects of gamma irradiation on small quantities of various substances. The control cylinders, movable neutron absorbers between the core and the beryllium reflector to control or shut down the reactor, are somewhat less radioactive at the end of their useful life than a fuel element, but their radioactivity decays more slowly. A new facility, to be inserted into a spent control cylinder, will provide a high flux (10^6 R/hr) irradiation capability for tests on larger samples.

New Neutron Scattering Instruments

Existing scattering instruments will be improved and new instruments (e.g., a small angle neutron scattering machine and a residual stress analyzer) constructed to take full advantage of existing and new beams from HFIR.

Monochromator Shields

Monochromators are used (like a prism or diffraction grating for light) to select, and send in a desired direction, only neutrons with the energies or wavelengths desired. Other wavelengths are deflected in other directions, and so the monochromator location must be heavily shielded. We have been awaiting replacement of the present shields, which are 20 years old and have a very restricted aperture, for ten years. In addition to the beam size restrictions, the mechanical components, such as bearings, are worn. The monochromator shields need to be replaced regardless of any other upgrades.

HB-2 Guide Hall

This project will put five thermal neutron guides into the existing HB-2 beam port to bring neutrons from close to the core out to a new guide hall that will have space for up to 15 instruments. HFIR already has the world's most intense beams of thermal neutrons, and this upgrade will make them brighter still and with lower background because of the inverse square weakening of the gamma and fast neutron contamination down the long guides.

Remote Handling Facility

At present, there is no remote handling (hot cell) at HFIR, and irradiated materials, (which include isotope targets, materials irradiation capsules, and assorted scrap materials) must be loaded directly into shipping casks positioned underwater and suspended from a crane.

With a remote handling facility at HFIR, irradiated materials could go directly into a hot cell from the pool, and specimens or target materials could be retrieved and prepared for their final destination, thus bypassing the shipment to a disassembly cell. Excess hardware and scrap materials could be shipped directly to the ORNL burial ground facilities. If possible within the cost limits, the remote handling facility would also be designed to accommodate examination of control plates and canning of spent fuel prior to dry storage.

The addition of this facility would improve safety and reduce person-rem exposures attributable to handling radioactive materials at HFIR and allow faster and more cost-effective handling of materials irradiation experiments and isotopes.

Beryllium Redesign for Plutonium Production

DOE has a need for an isotope of plutonium, Pu-238, to be used as a heat source for the production of electricity in remote locations and in space. Funding has already been received to redesign the vertical holes in the beryllium reflector to accommodate production of this, or other, isotopes. When the beryllium is next replaced (in FY 1999), the new reflector will be built to this modified design.

In-Service Inspection and Beryllium Changeout

This task must be carried out, regardless of upgrades, to continue operation beyond FY 1999. DOE Orders require an in-service inspection (ISI), and radiation embrittlement will necessitate another routine changeout of the beryllium reflector. A delay in the ISI may be requested so that it can coincide with the reflector replacement, minimizing cost and reactor downtime.

Neutron Sciences Support Building

During every changeout of the "permanent" beryllium reflector, research equipment in the HFIR building must be physically removed to a safe climate-controlled storage place. In the past, trailers and other temporary storage have been used, but with present concerns and regulations concerning contamination, that is no longer an attractive option. We propose to erect a preengineered building to serve this purpose for the 1998-1999 changeout and for later ones. In between these operations, the building would be used as space for additional beam scattering instruments on the HB-4 (cold neutron) beam line.

Replace Transformers

The existing transformers are oil filled and were identified at least as long ago as a 1990 DOE Oak Ridge Operations appraisal as in need of replacement as an environmental and fire hazard. Funding requests for this action have been made but were preempted by other, higher priority needs. Replacing these transformers with others, placed at another side of the HFIR electrical building, will also free up valuable space that can be added to the equipment support facility/experimental area.

Other Reactor Improvements

Given that the HFIR is to be upgraded and to give a further scientifically useful life of 35 years or so, aging electrical, mechanical, and control systems will be replaced or refurbished to keep the facility operating in a safe and reliable manner.

General Notes

All of the above improvements are independent of each other, and each would be beneficial on its own (although there are synergies - the whole upgrade is naturally worth more than the sum of the parts). Likewise, the two buildings (the Neutron Sciences Support Building, available for experimental space between beryllium changeout operations, and the HB-2 guide hall) and the remote handling facility are valuable improvements on their own but are also synergistic with other enhancements.

Operation Experience and Current Status of HANARO

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ABSTRACT

It has been almost two years since the first fuel loading at HANARO. Presented in this paper is the operation experience through the reactor commissioning and power operation during that period. The current status of utilization and installation of utilization facilities is summarized and some of research projects are introduced.

INTRODUCTION

To be a safe and useful research reactor, a lot of activities are going on in HANARO. The physics parameter measurement continues following the nuclear commissioning for safe operation and understanding reactor characteristics in parallel with the calculations for comparison. Maintaining operator resources would be a basis to have operators with high quality. In a sense that operator training program is valuable for safe operation, groupwise self-education and procedure review program not only broaden operator's knowledge but also revise operation procedures from experience. Due to the decommissioning of TRIGA II & III, the radioisotope production in HANARO, only remaining research reactor, for domestic demand is absolutely required. Neutron beam facility, irradiation facility for NTD (Neutron Transmutation Doping), and FTL (Fuel Test Loop) facility installation are in progress although those works sometimes require reactor shutdown. The first trial of reactor material test using non-instrumented capsule has been completed and is under PIE (Post-Irradiation Test) at IMEF (Irradiated Material Examination Facility). The design of instrumented capsule will be completed by July 1997. As well as the safe operation of HANARO, the reactor operation team involves in various projects for better environment of HANARO users and supports the utilization and development team. BNCT (Boron Neutron Capture Therapy), DUPIC fuel irradiation test analysis, IPS (In-Pile Section) design for fuel pin irradiation test, TRIGA decommissioning, etc. are the present project items. The number of visitors to HANARO is over five thousands in 1996, which shows that HANARO contributes to nuclear public acceptance.

CHRONOLOGY

After the successful approach to the initial criticality with 17 fuel bundles on Feb. 8, 1995, the first operation core was configured by loading seven additional fuel bundles on May 10, 1995. Everytime a fresh fuel bundle was loaded, the control absorber rods (CARs), shut-off rods (SORs), and fuel bundle worth as well as the critical CAR position were measured and compared with the predictions. With this first cycle core, zero power tests, power ascension tests, and long-term operation tests were followed consecutively. Zero power tests include noise analysis, and the measurement of void coefficient, fuel bundle worth, thermal and fast flux distribution, assemblywise power distribution, gamma dose rate distribution, photoneutron effect, and isothermal temperature coefficient. Starting with natural circulation test, thermal power, transfer function, loss of electric power test, decay heat, power defect and xenon effect measurement were carried out in power ascension tests.

Loading of four more fuel bundles led to the second cycle core on June 25, 1996. In order to have an appropriate excess reactivity and CAR position for the measurement of CAR worth by swap method, four fuel bundles were loaded in two times two at a time. In September, two more fuel bundles were loaded and the third cycle started with the operating power of 24 MW. The fourth cycle is scheduled to start in January of 1997. At each BOC and EOC, several important physics parameters were measured. The tests include the measurements of the control rod worth, isothermal temperature coefficient, xenon load, and power defect. Whenever the fresh fuels are loaded, the critical CAR position is predicted by diffusion code with burnup calculation. The operation power in first and second cycle was 15 MW and raised to 22 MW in the middle of 2-2 cycle. The core average burnup at the end of cycle 2 was 39.11 GWD/MTU (24.16 %U-235).

The water conductivities of primary cooling system, spent fuel pool, and heavy water system are measured everyday. The conductivity is maintained much lower than the technical specification limit value. The water quality is analysed weekly and the pool water radiation is measured weekly by HpGe detector. The radiation level around beam tubes was below 3.0 μSv/hr and 2.5 μSv/hr in the control room at 22 MW. The radiation area is categorized in four level; clean, normal operation, temporary operation, and contaminated area. The control room is in clean area and the radiation limit is 6.25 μSv/hr and the access time is unlimited. Around beam tube belongs to normal operation area and the radiation limit is 12.5 μSv/hr and the access time is limited to 40 hr/wk.

The first overhaul was carried out from March 4 to March 30, 1996. The Seismic Monitoring System surveillance was performed separately for 2 week from Feb. 27,

1996. The Neutron Flux Monitoring System linearity was measured and the position of each detector was calibrated accordingly.

Since January of 1996, radioisotope production has been tested for isotopes such as I-131, Ir-192, Mo-99, Tc-99m, Ho-166. Test of air balance and system performance test on Radioisotope Production Facility have been completed and will be licensed for isotope production facility in this November when the normal production is started. From that time, the domestic demand on I-131, Ir-192, Ho-166, P-32, etc. will be fulfilled. The feasibility study of Fission-Moly production is in progress. Seventeen lead hot cells and four concrete hot cells are currently in operation and eleven more lead hot cells will be ready by 1997.

The radiation emergency plan has been established to ensure the adequate response to the emergency which would cause a significant risk to the KAERI staff and the inhabitant neighbouring the HANARO site. The emergency organization consists of spot support team, administrative support team, and technical support team. The emergency state is divided into four levels. Whenever the emergency occurs, the emergency headquarter notifies KINS (Korea Institute of Nuclear Safety) and City Administration and reports MOST (Ministry of Science and Technology). The second emergency drill was performed on May 3, 1996 following the first one last year. The Safety Action against war has been also prepared for the security of HANARO fuel.

IAEA inspection is being carried out on in-core, fresh, and spent fuels and monitoring camera film is replaced annually for the transparency of HANARO operation.

EXPERIMENTS AND FACILITIES

Sixteen hexagonal fuel bundles irradiated during fast flux distribution measurement were scanned by in-pool gamma scanner installed in the reactor service pool for the assemblywise power distribution measurement. The photo peak of long half-life radioisotope, La-140 was selected and measured by NaI(Tl) detector due to more than 3 months of cooling time. For more accurate measurement, the gamma scanning system was designed, which can rotate and move up and down the fuel bundle with very high precision. The same sixteen fuel bundles were scanned by this system on Zr-95 and Nb-95 photo peaks with NaI(Tl) detector for axial power distribution measurement at IMEF. Pin power distribution analysis is in progress.

Non-instrumented capsule was inserted into the central trap for the lifetime evaluation of existing nuclear material and its comparison with the newly developed material. The test materials were stainless steel (core structural material), Zr-2.5%Nb

(CANDU pressure tube material), and low alloy steel (reactor pressure vessel material). Six eutectic alloys as a temperature monitor and Fe wire in Cd tube as a fluence monitor were also included in the capsule. The evaluation of neutron irradiation effects include tensile, impact, fracture toughness, hardness, thermal annealing recovery tests, magnetic property measurements, micro structure analysis by optical microscope and TEM (transmission electron microscope), component analysis, etc. Instrumented capsules will be loaded in CT, IR, OR, or IP next year.

To verify the HANARO fuel at high linear power, two types of test fuel assemblies were loaded at IR or CT. Each test fuel assembly has six fuel elements at the peripheral positions between the corners and 27 Al dummy and 3 hollow elements. Two type-A fuel assemblies were loaded at IR1 & 2 and the discharge burnup will be 40 and 65 a/o. A type-B fuel assembly with three instrument tubes and thermocouples were loaded at CT and the discharge burnup will be 90 a/o. In type-B fuel bundle, Rhodium SPND (Self-Powered Neutron Detector) and Platinum SPGD (Self-Powered Gamma Detector) were installed for power history and gamma flux measurement. In the pre- and post-irradiation test, dimension measurement, fine structure investigation, mechanical/thermal characteristics, and bending test of fuel element are performed. Gamma scanning, blistering test, and chemical burnup analysis of fuel element are carried out additionally in the post-irradiation test.

Neutron beam facilities are installed in three phases. At the end of the first phase (1993~1997), four units of neutron spectrometers (High Resolution Powder Diffractometer, Four Circle Diffractometer, Polarized Neutron Spectrometer, Small Angle Neutron Spectrometer) and a neutron radiography facility will be available. In the following phases, the neutron beam research will further extend its scope and depth such as in advanced technological materials, polymer science and biological substances by installing a cold neutron source and guide hall neutron beam facilities.

Fuel test loop is on manufacturing and IPS will be installed at LH in mid 1998. It is a steady state fuel test loop for irradiation test of a fuel assembly such as CANFLEX, DUPIC, and advanced PWR fuel bundles. The safety analysis report is on review by KINS for installation license. After the completion of this FTL, it is planned to install a transient fuel test loop for fuel pin test in power ramping.

OPERATOR PROGRAM

Three shift six group system has been established by operators. Each group has four personnel; one senior reactor operator, one reactor operator, and two system operators. Out of six group, one group is on education, and another supports the reactor management. There are 211 reactor operation procedures and those are assigned to six groups for review and revision. HANARO simulator is being used for the personnel to learn basic operation and reactor kinetics and will be further developed to include full reactor control algorithm. Operator training program is open annually in which six classes are with instructors who designed HANARO systems or who is an expert in his own field. The program of visiting to power reactors, foreign research reactor is prepared for operators annually. Intensive health checkup is carried out annually and radiation exposure is monitored quarterly.

PROJECTS

The design project of an IPS for the steady state irradiation of fuel pins has been initiated in collaboration with AEA recently. This IPS is for steady state irradiation of up to 3 fuel pins and will be coupled to the FTL out of pile system by means of pipework and flexible hoses within th HANARO pool. It is not intended that the main FTL IPS and this IPS will be operated at the same time. The preliminary design will be completed in about five months.

It is more than ten years since KAERI and its associate, Korea Cancer Research Hospital were interested in BNCT. With the operation start of HANARO, however, the interest in BNCT was vitalized and the feasibility study is in progress since last year. Boron compound survey, HANARO neutron beam feasibility study, gamma/fast neutron filter investigation, medical treatment technique, etc. are the items in BNCT project.

For the DUPIC fuel irradiation test in HANARO, the proper fuel assembly configuration with the instrumentation for monitoring was designed and the fuel enrichment to satisfy a required linear power condition was evaluated. The burnup calculation will be followed in next project year.

The low flow CHF test will be carried out using a flow loop in KAIST (Korea Advanced Institute of Science and Technology) at the end of this year. This test is required to validate the thermal margin of HANARO fuel at natural circulation condition and validate the accident analysis results. Thirty finned and 10 unfinned fuel element simulators are reserved for this test. Currently, the test section and the heater are being manufactured.

CURRENT STATUS OF RESEARCH REACTORS RA AND RB AT THE VINČA INSTITUTE OF NUCLEAR SCIENCES

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ABSTRACT

Both research reactors at the VINČA Institute of Nuclear Sciences, the heavy water moderated and cooled 6.5 MW research reactor RA and the heavy water moderated zero power research reactor RB, commissioned in 1959 and 1958 respectively, belong to the generation of research reactors which have played an important role in developing nuclear power technology, as well as different scientific, medical and industrial applications of nuclear energy and radioactivity.

This paper presents basic facts about the RA and the RB research reactors and describes the present status of their systems and components. Different options for the future status of the RA reactor are specified. Problems related to the RA reactor spent fuel storage and the long term safe disposal of irradiated fuel, which are identified as the most urgent ones, are discussed. Some experiments, as well as practical applications, performed at the RB reactor are displayed. Conclusions concerning future status of the research reactors, as well as some general conclusions about building and maintaining research facilities for the future of nuclear energy, are drawn.

1. INTRODUCTION

Both research reactors at the VINČA Institute of Nuclear Sciences, the heavy water moderated and cooled 6.5 MW research reactor RA, designed and constructed by USSR, and the heavy water moderated zero power research reactor RB, designed and built by Yugoslav scientists, commissioned in 1959 and 1958 respectively, belong to the generation of research reactors which have played an important role in developing nuclear power technology, as well as different scientific, medical and industrial applications of nuclear energy and radioactivity.

The research reactor RA was and still is the largest research facility in Yugoslavia and the most valuable installation in the VINČA Institute. It was first operated with 2% enriched uranium metal fuel of Soviet origin. In 1976 80% enriched uranium oxide fuel dispersed in aluminium was purchased from USSR. The decision to reconstruct and innovate the RA research reactor was made during the early eighties and a number of improvements, including complete exchange of electronic equipment, was initiated. Unfortunately, for both technical and political reasons, these actions have never been completed. Having in mind the long shut-down period of the RA reactor, together with the overall technological, political and economical changes, the status of the RA research reactor is now a subject of serious reconsideration.

In the RB reactor, besides its original natural uranium metal fuel in the form of solid rods, the RA reactor fuel can also be used. Since criticality is achieved by regulating the moderator level, a large variety of core configurations with different lattice pitches and fuel arrangements have been studied over the years, making possible experimental verification of different theoretical models and calculational schemes in neutron physics and neutron transport theory. Most recently, a coupled fast-thermal reactor system was realized, having a central zone where the neutron flux energy distribution is similar to the one in the fast research reactors. In the meantime, complete exchange of control, safety and radiation protection equipment of the RB reactor has been performed.

The following paragraphs present basic facts about the RA and the RB research reactors and describe the present status of their systems and components. Different options for the future status of the RA reactor are specified. Problems related to the RA reactor spent fuel storage and the long term safe disposal of irradiated fuel, which are identified as the most urgent ones, are discussed. Some experiments, as well as practical applications, performed at the RB reactor are displayed. Conclusions concerning future status of the research reactors, as well as some general conclusions about building and maintaining research facilities for the future of nuclear energy, are drawn.

2. RESEARCH REACTOR RA

2.1. Basic Facts about the Research Reactor RA Operation

The USSR designed and built 6.5/10 MW thermal heavy water moderated and cooled research reactor RA [1,2] started operation in 1959. Until 1976 the reactor was operated with 2% enriched uranium metal fuel, when 80% enriched uranium oxide fuel dispersed in aluminium was purchased from USSR. The new fuel, having the same geometry and the same amount of ^{235}U per a fuel element as the old one, was supposed to enable increase of the neutron flux due to diminished parasitic absorption in ^{238}U. From 1976 to 1979, the reactor operated with mixed 2% and 80% enriched fuel core.

In 1979, during regular refuelling, deposits were noticed on the new fuel cladding. The reactor was shut down and started operation again in 1981 with completely fresh 80% enriched fuel core. After a year of running the reactor at decreased power of 2 MW, deposits on the fuel cladding were noticed again, in spite of regular heavy water purification. After 1982 the reactor core was once again completely exchanged, the deposits were again detected, and since then the reactor has never been operated at full power and in the nominal working regime. In the meantime, serious problems were also encountered with some other reactor systems. The reactor was finally shut down in August 1984 in order to reconstruct and improve practically all vital reactor systems.

Examinations performed at that time showed that the reactor building, mechanical parts of the reactor and the reactor vessel, were in reasonably good condition. Institute VINČA decided to build new systems for emergency core cooling and emergency ventilation, a new irradiated fuel handling system, a system for purifying water in the spent fuel storage pool and a new power supply system. A complete exchange of electronic equipment and instrumentation had to be performed through the technical assistance program of IAEA. New equipment was purchased from Soviet suppliers. However, for a number of political, administrative and technical reasons, this reconstruction has not been completed. In the meantime, the old instrumentation of the RA reactor had been dismantled, awaiting for arrival of the new one, and the reactor RA was left in an intolerable situation without control, safety and radiation protection systems.

In the course of preparations for restarting the RA reactor, a significant amount of new fuel was purchased from the former USSR, sufficient for the life time of continuous operation of the reactor. The Institute VINČA built the new emergency core cooling system and the new emergency ventilation system. The new irradiated fuel handling machine was completed, but software for its remote operation is not installed. The new system for purifying water in the spent fuel storage pool is finished but not tested yet. Although the reactor facility is financially supported and basic expenses and salaries of the stuff are covered, during the last several years the reactor RA crew suffered serious losses in skilled and trained personnel.

With all this in mind, during the year 1995, the Institute VINČA, together with the Yugoslav largest consulting and engineering organization ENERGOPROJEKT, produced a document [3] describing scenarios of the possible options for the future status of the RA reactor: present state, restarting the reactor, conservation or decommissioning. The following main conclusions were reached: the present state of the research reactor RA can not be tolerated any longer; all other options would require international technical and/or financial support; the spent fuel storage pool is in a very bad condition and should therefore be the cause of considerable concern; the problem is primarily neither scientific nor technical, but it should be considered from the nuclear safety and radiation protection point of view. For this reason, Institute VINČA addressed both the national authorities and the IAEA, emphasizing the necessity for immediate action concerning safety, security and radiation protection problems at the RA reactor. Main results of this effort and actions taken so far are summarized in the following paragraphs.

2.2. Reactor RA Main Systems and Features

Research reactor RA is a thermal heavy water moderated and cooled reactor. Its main parts are: the central body, Figure 1, including the active zone situated in the inner aluminium reactor vessel; the graphite reflector surrounding the inner vessel and situated in the outer stainless steel reactor vessel; the biological shield of water and heavy concrete; the heavy water primary cooling system; the ordinary water cooling systems, i.e. the secondary technical water cooling system and the graphite cooling system; the He-system with He-atmosphere at the upper part of the reactor vessel; the control systems for technological parameters; the reactor power control system; the dosimetry system for measuring direct radiation and the water, He and air activity; the fuel transport system; the auxiliary systems like power supply system or special ventilation system.

The fuel element (slug) is an aluminium cladded hollow cylinder, 11 cm long, with an outer diameter of 3.7 cm, containing uranium in tubular form, Figure 2. Fuel elements are inserted in aluminium tubes (11 slugs/tube), forming a fuel channel. A maximum of 82 channels form a square reactor lattice having 13 cm pitch. The D_2O coolant, which is circulated by pumps into the reactor vessel from the bottom, flows upwards through the fuel channels on both sides of the fuel and flows down through the core moderator space into the outlet tube at the bottom of the reactor vessel. Water from the Danube river is used as a secondary coolant.

Two safety rods containing cadmium absorber are operated by two independent and identical safety mechanisms for automatic and immediate shut down (scram) of the reactor. Another two cadmium rods are used for automatic control of reactor power, while seven cadmium rods are used for compensating long term reactivity changes. Control and safety rods are inserted into the reactor core through aluminium channels, isolated from the primary circuit and filled with helium when the reactor is in operation. Since the existing control rods are partially burned-out, particularly their lower active parts, new stainless steel control rods, with gadolinium as a burnable absorber, were purchased together with the new electronic equipment.

The main characteristics of the present status of the RA reactor are the following: the core contains 480 fuel slugs, initially 80% enriched and with relatively low average burnup; safety rod, rods for automatic reactor power regulation and rods for compensating long term reactivity changes are all in the core in the lowest possible position; heavy water is drained from the primary circuit and placed in the heavy water tank; helium is drained from the inner reactor vessel and the gas system, so that the reactor vessel is filled with air under normal pressure; the automatic power control system, the system for measuring the reactor parameters and the relevant technological parameters, as well as the system for radiation control and protection do not exist, since the old equipment was dismantled and new equipment was not installed.

Visual inspection of the fuel from the core has been recently performed through the protective glass window on the biological shield of the reactor. A white layer of corrosion products and\or deposits was noticed on the fuel string taken from a fuel channel in the reactor core. Similar deposits, but at a much lower extent, could be noticed on the outer side of the fuel channel tube. Similar deposits were noticed and investigated many years ago, but no definite conclusions have been made about their origin, nor the way how to prevent their formation. Although the fuel in the reactor core has been left in the ambient temperature and humidity for more than ten years, no signs of new intensive corrosion processes were noticed, neither have any signs of fission-product contamination been observed in the reactor core and its surrounding. Some pitting corrosion could, however, take place under the layer of deposits. Whereas safety standards would require to remove the fuel from the reactor core, because of the very poor situation with the spent fuel storage pool, it was decided to leave this fuel as it is.

2.3. Reactor RA Irradiated Fuel Handling and Disposal

Closely related to the problem of the future use, or eventual decommissioning, of the research reactor RA is the problem of safe and reliable disposal of the so far irradiated fuel, as well as the newly irradiated fuel if and when the reactor is operated again.

Main components of the reactor RA fuel handling system are: the transporter on the central reactor body used for reactor fuelling, as well as for transferring the fuel channel with irradiated fuel into the water channel connecting the water reactor shield and the temporary spent fuel storage pool in the basement of the reactor building; the crane in the reactor hall and the crane in the spent fuel storage room; the irradiated fuel handling machine for distant automatic manipulation of irradiated fuel, which can be observed from the neighbouring control room through a special lead glass window.

The six meters deep temporary spent fuel storage pool, Figure 3, consists of four connected basins, having thick concrete walls cladded with stainless steel, and is filled with approximately 200 tons of stagnant ordinary water. 304 channel-type stainless steel fuel containers, Figure 4, receiving up to 18 spent fuel elements each, are placed vertically in the pool. Initially, it was planned to transfer spent fuel back to the Soviet supplier, after 4-5 years of cooling in the temporary storage pool. Since this has never happened, in order to increase the spent fuel storage capacity, some of the oldest metal uranium fuel has been taken out of the original stainless steel containers and repacked in sealed aluminium barrels, Figure 5, each containing 30 aluminium tubes receiving up to 6 irradiated fuel elements per a tube, placed in two layers in the annex of basin 4. Cadmium strips were placed in the barrels to provide the necessary subcriticality. Both the barrels and the channel-type fuel holders were filled with demineralized water, which is not supposed to mix with water from the pool.

The reactor RA spent fuel inventory is the following: 6656 fuel elements with initial enrichment 2% ^{235}U in the temporary storage pool; 884 fuel elements with initial enrichment 80% ^{235}U in the temporary storage pool; 480 fuel elements with initial enrichment 80% ^{235}U in the reactor core. The heavy nuclide inventory in the low enriched spent fuel is: 30 kg of ^{235}U, 2400 kg of ^{238}U, 5.20 kg of ^{239}Pu. Nuclear criticality safety studies performed for irradiated fuel at the reactor RA site proved that sufficient subcriticality was provided for all existing configurations [3].

Due to the obvious lack of attention paid to the conditions of the reactor RA spent fuel storage, presumably because it was considered only as a short term storage, it is now in very bad condition [5]. Water in the pool is dirty and its chemical parameters are not maintained so to minimize corrosion. New equipment for water purification is neither operable, nor adequate. At the bottom of the pool there is a lot of sludge which even hides one spent fuel slug lost many years ago. By measuring depth of the sludge at different points of the pool, which varied from 3 to 12 cm, its total amount is estimated to be roughly 3 m^3. Visual inspection showed that all steel construction elements are heavily corroded. Corrosion is also noticed on stainless steel walls of the basins and transport channels.

A string of spent fuel slugs, removed from RA reactor more than 17 years ago, before the reactor was shut down because of the corrosion problems for the first time, has recently been taken out from the channel-type spent fuel holder in the storage pool and brought to the hot room in the reactor shield to be visually inspected. A thick layer of corrosion products and/or deposits, covering completely the spent fuel slugs, pointed out the high probability that fuel cladding was penetrated and that fission products leakage took place in some of the spent fuel containers. It was not possible to examine the state of the spent fuel inside aluminium barrels, but it could be presumed that the corrosion is even worse.

Radiochemical analyses showed that the radioactivity content in the pool water was quite high, it contained about 3.8×10^7 Bq/m^3 Cs-137, i.e. about 0.25 Curies for the 200 m^3 of pool water. This could be attributed to the "lost" fuel slug, but suspicion can also be expressed that some contribution could also be due to corroded slugs in aluminium barrels whose leak tightness was questionable. Recent investigations show that water in some channel-type spent fuel containers, particularly those with very old fuel, is highly radioactive, what is an obvious indication that some spent fuel is leaking.

2.4. Present Activities Related to the Future Status of the RA Research Reactor and Safe Disposal of So Far Irradiated Fuel

Although the present state of the 6.5 MeV thermal research reactor RA systems and components is rather poor, it could still represent a valuable facility. In the country which does not have a nuclear power program, restarting the research reactor would help to preserve expertise in reactor use and operation and bridge the time gap until the moment when the attitude of the society towards the use of nuclear energy would be changed.

As the first step towards this goal, the VINČA Institute of Nuclear Sciences and the engineering organization ENERGOPROJEKT have recently started work on the Project of Restarting the RA Research Reactor, which should provide all the relevant technical and financial factors needed for making a final decision about the future status of this facility.

At this moment, however, the problem of the future status of the reactor is pushed back by the situation with the reactor RA spent fuel storage pool, which has been paid little or no attention to for years and has thus become a serious safety problem. Activities are now initiated in two directions. First, to improve the present situation of the existing spent fuel storage by reducing the water turbidity, by removing the failed fuel slug from the bottom of the pool and by establishing the water chemistry control so to reduce further corrosion as much as possible. Second, to obtain an expert assessment of the conditions in the stainless steel channel holders and the aluminium barrels and to consider transferring the irradiated fuel back to the supplier or constructing an independent installation, presumably dry storage space, for long term storage of spent fuel placed in special containers designed for this purpose. Immediate objective that these activities are expected to achieve is increased safety of the research reactor facility in the sense that the possibility of uncontrolled fission products release in the reactor building, and eventually to the environment, will be reduced to the minimum possible value. By storing the previously irradiated fuel of the research reactor RA in a newly built storage space, sufficient free space will be provided in the existing spent fuel storage pool for the newly irradiated fuel if the reactor starts operation again. In the case that it is decided to decommission the research reactor RA, the newly built storage space will provide safe disposal of the so far irradiated fuel.

3. RESEARCH REACTOR RB

3.1. Basic Facts about the Research Reactor RB

Research reactor RB at the Nuclear Engineering Laboratory of the VINČA Institute of Nuclear Sciences was the first critical facility in Yugoslavia, designed and built by Yugoslav scientists [6]. In its original form, as a natural uranium metal - heavy water critical assembly, having fixed 12 cm lattice pitch and two safety rods, it was commissioned in 1958. Later, the facility was reconstructed in order to improve its safety and flexibility. The improved reactor, with new control, safety and monitoring systems was critical again in 1962.

Besides its original natural uranium metal fuel in the form of solid rods (length 2.1 m, diameter 2.54 cm, 1 mm thick Al cladding), the RA reactor fuel can be used in the RB reactor, as well. Special Al support plates make it possible to change the reactor lattice pitch (7, 8, 9, 12 or 13 cm or some of the existing multiples). Since criticality is achieved by regulating the moderator level, a large variety of core configurations with different lattice pitches and fuel arrangements have been studied over the years, making possible to perform various experiments and to obtain valuable results.

The RB reactor power ranges from 10 mW to 50 W under normal operating conditions, but it can reach 10 kW or more under special conditions. The constant power level is maintained by adjusting the heavy water level. A Ra-alpha-Be neutron source of 17.5 GBq is used during the start-up procedure. The heavy water circulation system has two speeds of filling the tank (2.5 cm/min or 0.8 cm/min) and two draining speeds (11 cm/min or 1.7/cm/min). The dosimetry system comprises neutron and gama dose measuring devices in the reactor hall, the reactor control room and the reactor building. Besides being used for different kinds of experiments in rector physics and engineering, the RB research reactor is regularly used in training students studying nuclear engineering at the University of Belgrade.

3.2. An Application of the RB Research Reactor for Experimental Verification of Theoretical Models and Calculational Schemes

New reactor concepts and advanced in-core fuel management schemes are the subject of increased interest from both technical and economic reasons. However, techno-economical evaluation of some advanced reactor cores requires improved calculational procedures, either because of more complex core configurations, or because of more restrictive accuracy requirements. In most of the cases, mock-up or criticality experiments for new reactor cores are practically unfeasible, while studying burn-up effects in practice would be too much money and/or time consuming. Thus, experimental verification, on existing and available research reactors, of models, data and codes used for above purposes is particularly desirable and important. Here, an example of using the RB research reactor for verifying different calculational models and procedures [7] is given.

The problem of treating the reactor core having different composition in axial direction was encountered in a particularly pronounced form when analyzing possibilities and effects of using a natural uranium blanket in a PWR. Since the expected effects, although important from the economic point of view, are relatively small in absolute amounts, it is important to eliminate a possibility of drawing erroneous conclusions on the basis of inadequate results.

When calculating parameters of an axially inhomogeneous reactor core, procedures for determining the reactor lattice cell parameters and the overall reactor parameters can be combined in a number of ways. In order to validate different calculational approaches , a reactor core inhomogeneous in axial direction has been simulated on the RB reactor, using the fact that its fuel channel is formed of a number of fuel slugs. Three core configurations were established: (a) all fuel channels contain 80% enriched fuel only; (b) at both ends of all fuel channels there is one fuel slug containing 2% enriched uranium metal; (c) at both ends of all fuel channels there are two segments containing 2% enriched uranium dioxide. The cores (b) and (c) simulate the situation when a number of enriched uranium pallets at the ends of a PWR fuel element are substituted by natural uranium pallets. Besides the critical heights of the reactor cores, axial and radial distributions of the fast and thermal neutron flux were measured by activating Au foils in Al or Cd cladding.

Comparison of experimental and computational results suggests that axially inhomogeneous reactor cores should be treated in the following way: first, volume averaging of physical data should be performed for each axially inhomogeneous zone of an equivalent reactor lattice cell; group constants are to be determined for an axially homogenized reactor lattice cell; calculation of global parameters is to be performed supposing that the reactor is homogeneous in the axial direction. In other words, there is no justification to divide the reactor core into a number of axial zones, determine cell parameters for each of these zones and then perform global calculations, since in this case the basic concept of a reactor lattice cell and equivalent cell parameters is jeopardized. This conclusion is supposed to be true not only in the case when axial inhomogeneity originates from different initial fuel composition, but also in the most common situation when it is caused by nonuniform fuel depletion during the reactor operation.

3.3. Fast Neutron Fields at the RB Reactor

When 80% enriched uranium dioxide fuel became available in 1975, investigation of fast neutron fields at the RB reactor started with constructing of an external neutron converter which transforms the thermal neutron leakage flux into the fast fission neutron flux [6]. Advantages of this converter are easy accessibility to the large experimental space and the possibility of the fast neutron spectrum down-shifting by using different screens. Its main shortcoming is the low intensity of the fast neutron flux.

The intensity of the fast neutron flux was upgraded in 1982, when the inner neutron converter and the fast flux experimental fuel channel were constructed using the new highly enriched fuel. In this case, however, smaller experimental space and softer neutron spectrum are obtained. In 1983, in order to overcome problems encountered with the outer and the inner neutron converters, realization of a coupled fast - thermal system started.

Presently, a coupled fast-thermal system, HERBE, at the reactor RB is established, Figure 6. This system has a central zone with densely packed natural uranium fuel and 80% enriched fuel, and no moderator. In the central experimental channel the neutron flux distribution is similar to the one in the fast research reactors. During the realization of this system, complete exchange of control, safety and radiation protection equipment of the reactor RB has been performed.

4. CONCLUSION

Valuable results have been obtained and considerable experience has been accumulated during many years of successful operation of the RA and RB research reactors at the VINČA Institute of Nuclear Sciences. However, having in mind problems presently encountered with the reactor RA, it can be concluded that building, running and maintaining a research nuclear facility may become a too heavy burden for a country without a well defined nuclear research and development program, while absence or inadequacy of necessary regulatory, safety and control institutions, may finally lead to negligence of the important nuclear safety issues.

International cooperation, on multilateral or bilateral basis, is important in all phases of planning, building and use of nuclear research facilities. The ongoing and forthcoming activities on solving the problem of the research reactor RA future and on improving its spent fuel storage conditions will provide an opportunity to perform a comprehensive exchange of existing experience and to develop new management practices of general interest.

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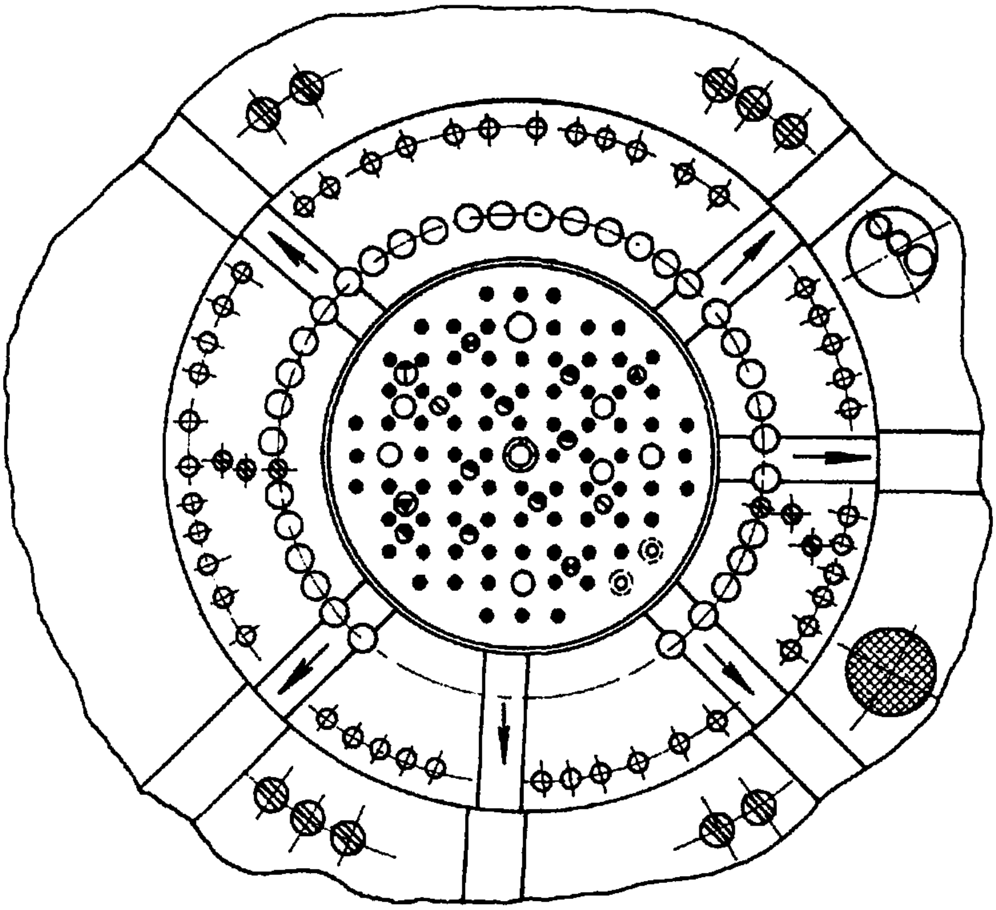

Fuel channel
Experimental channel
Compensation rod
Automatic rod
Safety rod
Biological channel

Figure 1. Main body of the RA research reactor. Horizontal cross section.

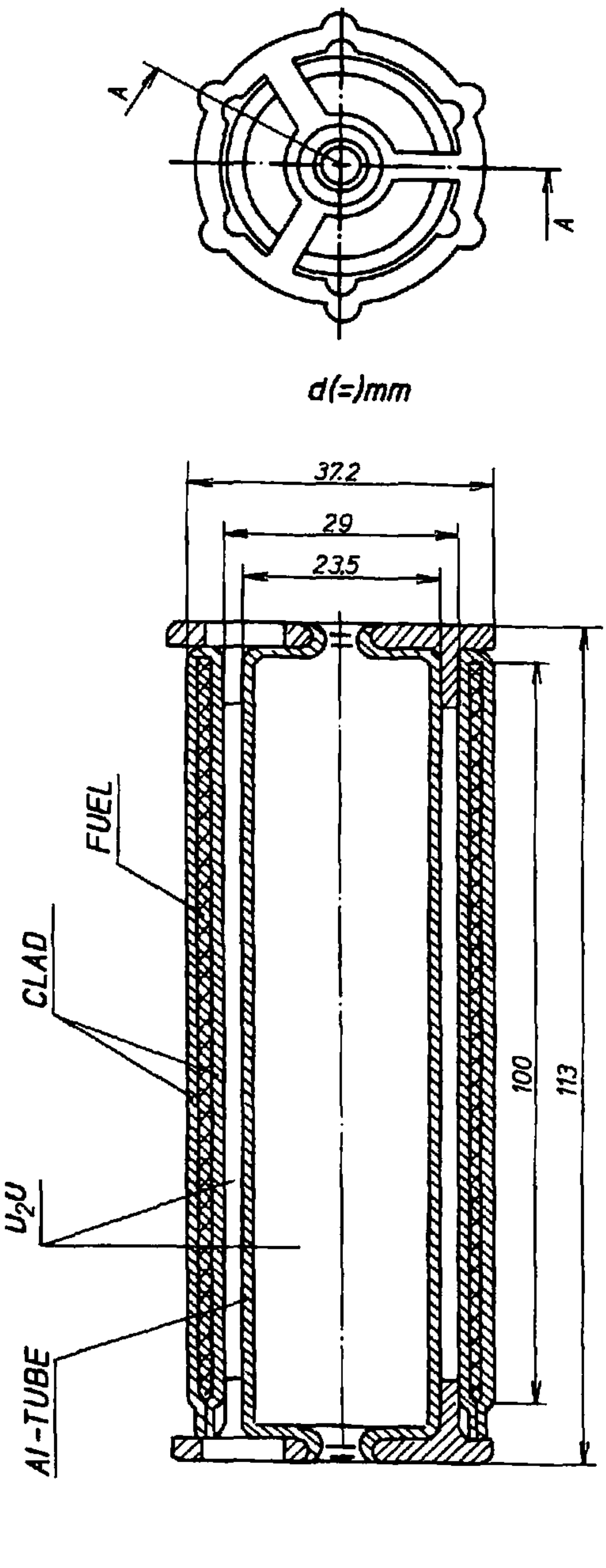

Figure 2. Cross section of the research reactor RA fuel element

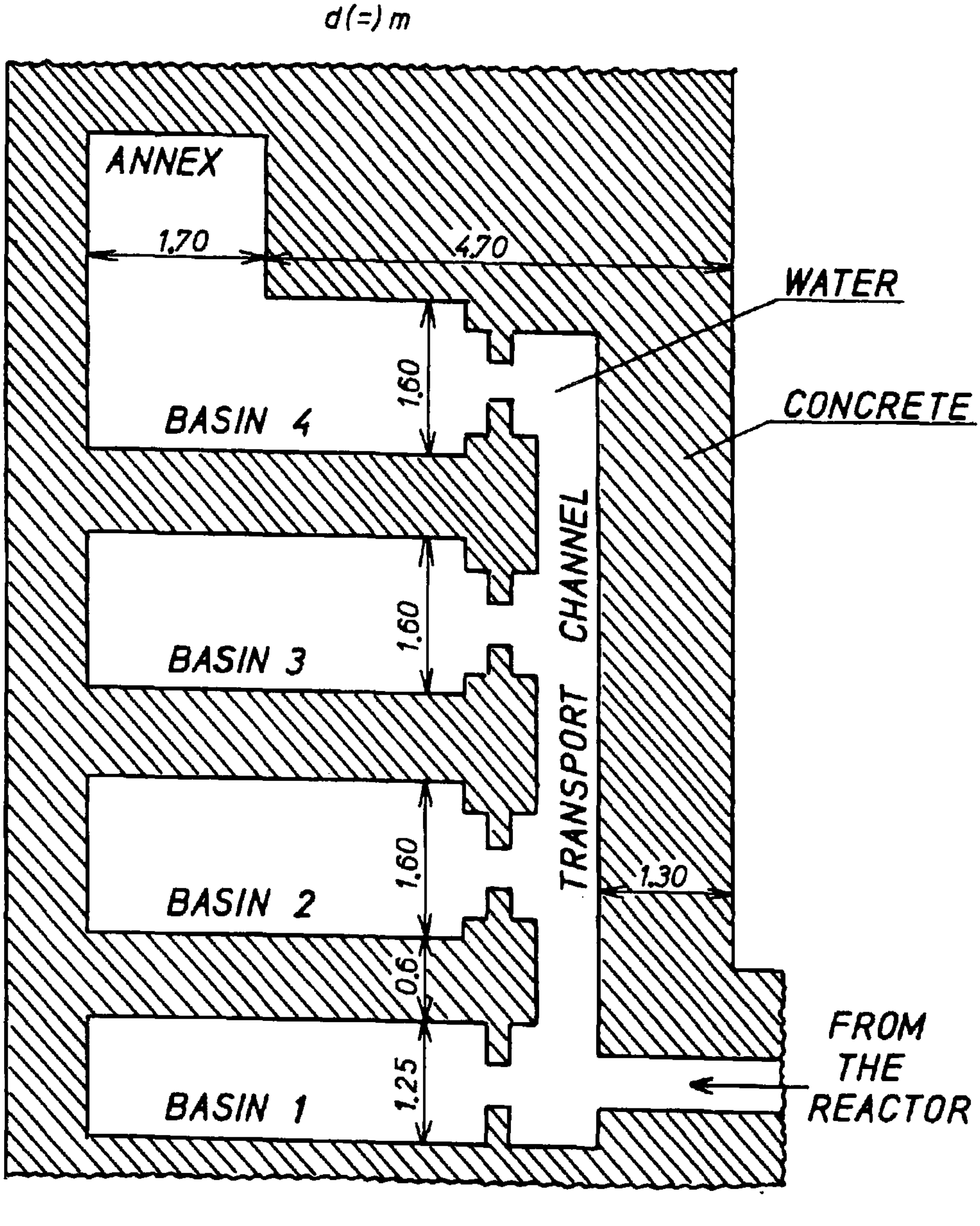

Figure 3. Scheme of the temporary spent fuel storage pool

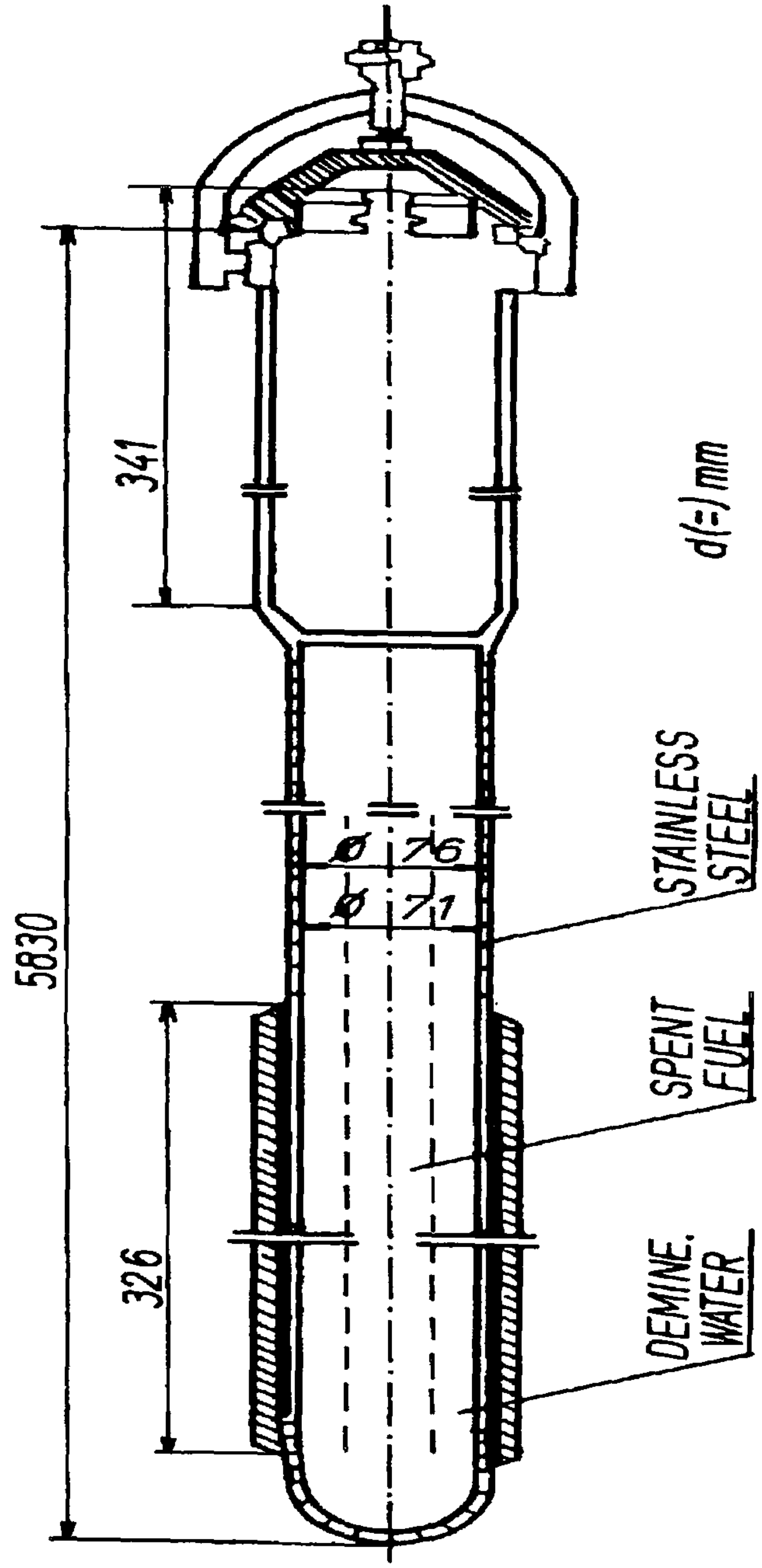

Figure 4. Stainless steel channel-type spent fuel holder

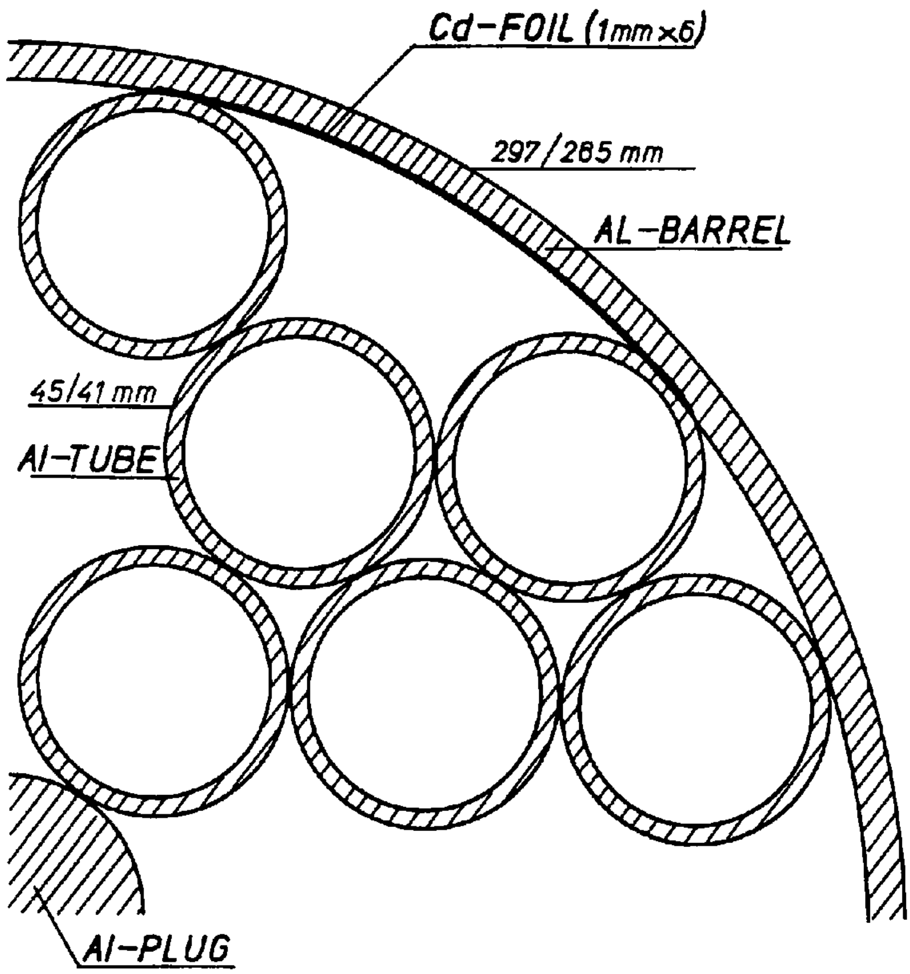

Figure 5. Aluminium barrel with spent fuel

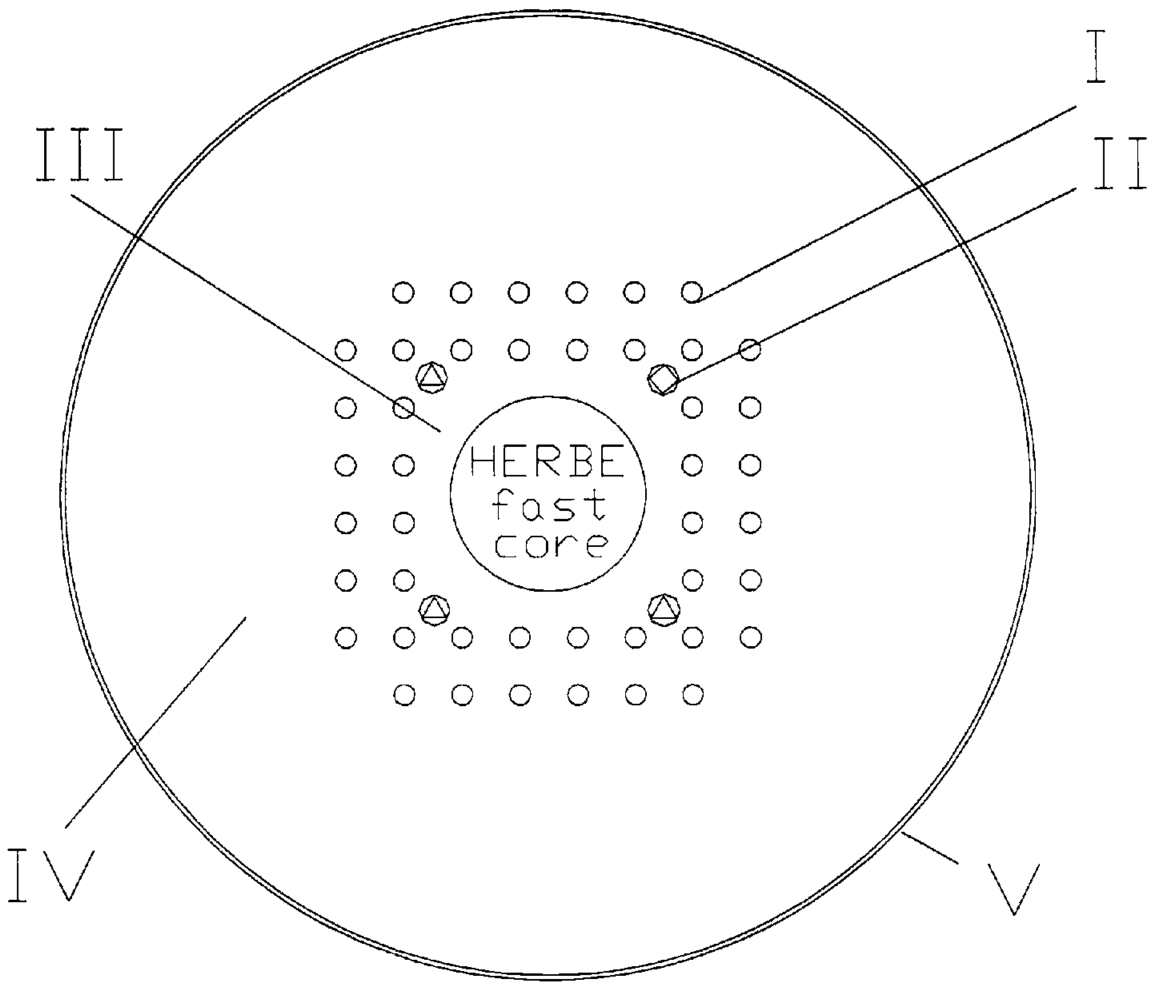

Figure 6. Horizontal cross section of the reactor RB core with the central HERBE zone
(I) fuel elements with 80% enriched UO_2; (II) safety rods; (III) inside reflector zone; (IV) outside reflector zone; (V) reactor vessel.

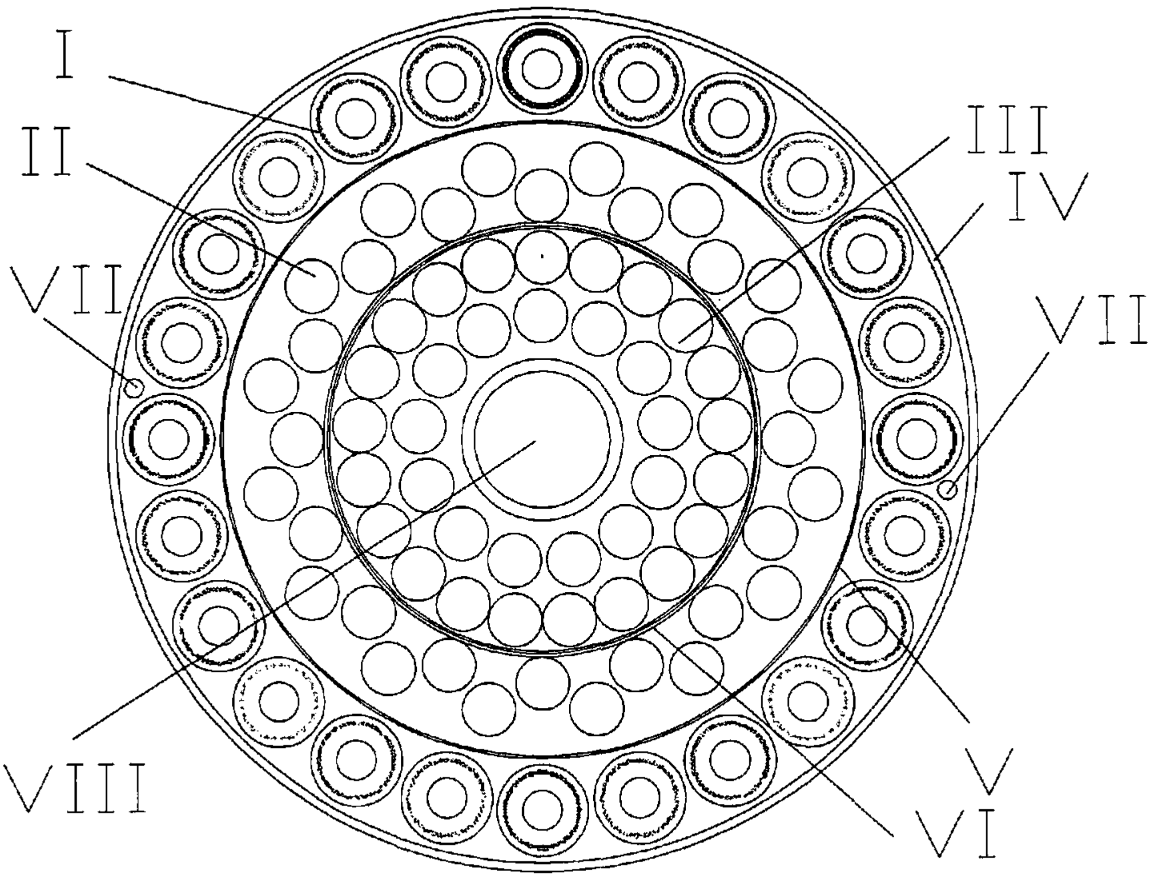

Figure 7. Horizontal cross section of the HERBE fast core
(I) convertor zone, 24 fuel elements with 80% enriched UO_2; (II) filter zone, Cd and 32 U metal fuel elements; (III) HERBE fast core, 32 U metal fuel elements; (IV) outer Al vessel; (V) intermediate Al vessel; (VI) inner Al vessel with Cd; (VII) indicator of moderator leakage; (VIII) experimental channel

STUDSVIK'S R2 REACTOR - REVIEW OF RECENT ACTIVITIES

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Abstract

A general description of the R2 test reactor, its associated facilities and its history is given. The facilities and range of work are described for the following types of activities: fuel testing, materials testing, in-pile corrosion studies, neutron transmutation doping of silicon, activation analysis, radioisotope production and basic research including thermal neutron scattering, nuclear chemistry and neutron capture radiography.

More detailed reviews are given of the following fuel R&D projects: DEFEX, ULTRA-RAMP, Ultra-High Burnup Fuel Irradiation Project, SUPER-RAMP III/10x10, TRANS-RAMP III, STEED I & II and the INCA facility.

1 Introduction

STUDSVIK AB is partly owned by the biggest Swedish utility, Vattenfall AB, and is performing R&D work and associated activities, primarily in the nuclear energy field. STUDSVIK AB is a commercial company, active in the areas of services, supply of special equipment and systems and also consulting. STUDSVIK NUCLEAR AB, which is the largest subsidiary within the STUDSVIK group, is one of the direct offsprings of AB Atomenergi, the origin of the STUDSVIK group, which was formed in 1947. The STUDSVIK group has about 580 employees and a turnover of about 425 MSEK/year.

During the 1950's and 60's, an ambitious nuclear program was launched in Sweden. The experience and competence gained from a large number of advanced projects constitutes the basis upon which the present activities of STUDSVIK NUCLEAR are based. Among the fuel- and materials-related experiences are the design, construction and operation of a fuel manufacturing plant for uranium dioxide fuel in Stockholm, a pilot line for mixed-oxide (MOX) fuel in Studsvik, the design and manufacture and initial operation of the Ågesta pressurized heavy water reactor, and the design, manufacture and irradiation proof testing for the Marviken boiling heavy water reactor. Ambitious R&D programs for heavy water reactor super-heater fuel and fast heavy water reactor fuel and materials were also carried out. Since the 1970's, the efforts have been concentrated on light water reactor fuel and materials, and the originally domestic R&D programs have been expanded so that a large fraction is now financed by non-Swedish sponsors.

Neutron activation analysis and radioisotope production as well as beam tube experiments for basic research applications were started in the 1960's. In 1977 neutron transmutation doping of silicon began.

The facilities of interest in this connection are the R2 Test Reactor, the Hot Cell Laboratory, the Lead Cell Laboratory and various other laboratories, all located at Studsvik, 100 kilometers south of Stockholm.

2 The R2 Test Reactor

2.1 General Description of the R2 Test Reactor

The R2 reactor is a tank-in-pool reactor, see Figure 1 and Table 1, in operation since 1960 and originally similar to the Oak Ridge Research Reactor, ORR (1). The reactor core is contained within an aluminum vessel at one end of a large open pool, which also serves as a storage for spent fuel. Light water is used as core coolant and moderator. The reactor power was increased to 50 MW(th) in 1969. In 1984-85 a new reactor vessel was installed.

The R2 reactor has a high neutron flux, see Table 1, and special equipment for performing sophisticated in-pile experiments. An important feature of the reactor is that it is possible to run fuel experiments up to and beyond failure of the cladding, which is not possible in a commercial power reactor.

Table 1 Technical Data for the R2 Test Reactor.

Power	50 MW(th)
Moderator/coolant	H_2O
Reflector	D_2O, Be
Fuel length	600 mm
Fuel assembly length	924 mm
Fuel assembly cross section	79x82 mm^2
Number of fuel plates per assembly	18
Neutron flux in experimental positions	
Thermal	(0.3-2.5) x 10^{14} n/(cm^2·sec)
Fast (>1 MeV)	(0.5-2.5) x 10^{14} n/(cm^2·sec)
Primary flow	1 300 kg /sec
Coolant temperature: Inlet	$\leq$ 40 °C
Outlet	$\leq$ 45 °C

The coolant water is circulated through the reactor vessel and flows through pipes and a large decay tank below the reactor hall to an adjoining building containing pumps and heat exchangers cooled with sea water.

The present core configuration is shown in Figure 2. The components of the core are arranged in an 8x10 lattice, typically comprising 46 fuel elements, 6 control rods, about 12 beryllium reflector assemblies and a number of in-pile loops, irradiation rigs and aluminum fillers. Rows Nos. 1 and 10 consist of beryllium reflector assemblies. The composition of the core can be altered to suit the experimental program.

The R2 driver fuel assemblies are, since the beginning of 1993, of the LEU type. They have 18 curved fuel plates containing an aluminum-clad aluminum silicide matrix. The initial fuel content is 450g ^{235}U per fuel assembly, enriched to less than 20 %. The burnup of the spent fuel of this type reaches about 65 %.

The control rods consist of an upper neutron absorbing section of cadmium and a lower fuel section. They are moved vertically by drive mechanisms placed below the reactor vessel.

The reactor vessel is 4.5 m high and 1.6 m in diameter. The design pressure is 3.3 bars.

Some of the irradiation facilities in the R2 reactor have been described in the literature (2-5), and details are given below. Most base irradiations of test fuel (irradiations at constant power, where fuel burnup is accumulated under well-defined conditions) are performed in boiling capsules (BOCA rigs). Some base irradiations and all ramp tests (irradiations under power changes) are performed in one of the two in-pile loops, which can be operated under either BWR or PWR pressure and temperature conditions. The ramp tests, simulating power transients in power reactor fuel, are achieved by the use of ^3He as a variable neutron absorber. Structural materials, such as samples of Zircaloy cladding, steels for pressure vessels and vessel internals and candidate materials for advanced reactors can also be irradiated in special rigs either in the loops or in special NaK-filled irradiation rigs in fuel element positions with a well-controlled irradiation temperature. Special equipment for in-pile corrosion experiments in the loops has recently been developed.

The R2 core has an active fuel length of 600 mm. Most fuel rods irradiated are segments of power reactor fuel rods, so-called rodlets, with lengths in the range 300 to 600 mm. However, tests have also been performed on full-size demonstration reactor fuel rods with up to 2.5 m length. In those cases only the lower 0.6 meters were irradiated. Non-destructive examinations of fuel rodlets can be performed in the R2 pool during short pauses in the irradiation program or between various phases of an experiment, see Section 3. All the handling and all the examinations are performed with the fuel rodlets in a vertical position; this is advantageous with respect to possible movements of fuel fragments etc.

Associated with the R2 reactor is the 1 MW(th) swimming pool R2-0 reactor, which is located in the same pool, see Figure 3. The basic research performed by use of R2-0 is briefly described in Section 10.

2.2 Boiling Capsules (BOCA Rigs)

The Boiling Capsule (BOCA) facility for irradiation of BWR and PWR fuel rods was introduced in 1973.

The in-pile part of a BOCA rig consists of a bare stainless steel pressure thimble containing a shroud with flow entrance ports at the bottom and exit ports at the top. The lower part of this shroud is located in the reactor core region. A fuel test rod bundle consisting of up to 6 rodlets is located inside the shroud. The BOCA is filled with highly purified pressurized water from a special pressurization system. Figure 4 shows a simplified BOCA flow diagram. BOCA system technical and operational data are given in Table 2 (page 4).

Coolant circulation and cooling is brought about by natural circulation, although no net boiling occurs. The water is heated mainly by the fuel rods. Buoyancy forces make hot water with lower density rise in the riser shroud, leave the exit port, meet the cold wall of the pressure thimble and be cooled down while flowing downwards in the annular channel between the pressure thimble wall and the riser shroud.

Table 2 BOCA System Technical and Operational Data.

Permissible operating pressure range:	
- Pressurizer System BOCA A	30 to 100 bars
- Pressurizer System BOCA B&C	30 to 170 bars
Permissible total power generation in one BOCA rig	≈ 90 kW
Permissible heat flux on test rod surface with regard to departure from nucleate boiling (DNB)	160 W/cm²
Possible range of thermal neutron flux achievable by:	
- Positioning in the core	up to 2.0×10^{14} n/(cm²·s)
- Internal hafnium shielding	Flux reduction to ≈ 50 %
- Internal hafnium shielding plus hafnium shielding of the rig adapter	Flux reduction total to 30 %

The circulation flow rate is low, about 0.2 kg/sec, but the coolant is substantially subcooled. When the heat flux at the fuel rod surface exceeds a certain value, between about 60 and 90 W/cm², subcooled nucleate boiling will occur at the rod surface, which implies a surface temperature equal to or slightly higher than the saturation temperature for the actual static pressure.

Power in the test rods is measured by a combination of nuclear measurements with the Delayed Neutron Detector (DND) technique, and coolant water temperature and flow measurements.

Up to five BOCA rigs can be operated simultaneously in the reactor. Two independent pressurization systems are available, each capable of supplying 3 to 5 BOCA rigs with water. Each BOCA rig is connected to a separate outlet circuit.

Each rig is constantly fed with a purging water flow in order to control the water chemistry in the self-circulating water volume of the pressure thimble. The same flow is let out to the drain. This inlet and outlet of water takes care of thermal expansion and contraction of the water in the thimble. During reactor start-up, when the water is heated up and expands, the flow is increased. The outgoing water is monitored for radioactivity (fission products), and the water chemistry is controlled.

In order to make it possible to irradiate power reactor fuel with standard enrichment in the in-pile loops and BOCA rigs in the R2 reactor, it is often necessary to decrease the neutron flux. This is achieved with hafnium absorbers in the form of tubes or plates.

2.3 In-Pile Loops

There are two in-pile pressurized light water loops used in the current LWR fuel R&D program in the R2 test reactor. They simulate realistic BWR and PWR temperature and pressure conditions. Data for the loops are shown in Table 3. Other loops, e.g. for the testing of HTR fuel, are described elsewhere (5) and will not be discussed here. Each LWR fuel loop utilizes two diagonally adjacent fuel element positions in the R2 test reactor, see Figure 2.

The two loops are pressurized in order that test fuel rods can be investigated at realistic operating conditions for either PWR (loop No 1) or BWR (both loops) type power reactors.

Table 3 Characteristics of the R2 In-Pile Loops.

Loop No.	1	2
Type	PWR/BWR	BWR
Pressure, bars	30-150	30-90
Coolant temperature, °C	220-325	220-285
Coolant flow rate, kg/sec	2.5-4.0	2.5-5.0
Max cooling capacity, kW	150	400
Neutron flux, 10^{14}n/[cm$^2\cdot$s]		
Thermal	0.6-2.2	0.5-1.5
Fast (>1 MeV)	0.8-2.0	0.7-1.4
Gamma heating in stainless steel, W/g	3-12	2-9

The loops can be used for irradiation at constant power of up to 4-5 test fuel rodlets simultaneously, and for power ramp tests of single rodlets. The test rodlets to be ramp tested are installed in the loop in special capsules, which in turn are inserted in a special test rig. The loops can also be used for irradiation testing of structural materials, e.g. Zircaloy test specimens and steel specimens of various types. A technique for in-pile corrosion tests in the loops with on-line corrosion potential measurements is discussed in Section 2.8 below.

The in-pile part of the loops are of a U-tube design, taking up two core positions and thus providing two test positions for each loop in the R2 core, one of which can be used for ramp tests (Figure 5). The U-tube is isolated from the reactor primary coolant by a gas gap containing CO_2. Heat losses from the tube to the reactor coolant are therefore quite small, which facilitates accurate test rod power measurement.

The working in-core inside diameter of these in-pile pressure tubes is 45.5 mm. The useful length in the core is 670 mm. The main features of the loops are presented in Figure 6.

2.4 Ramp Test Facility

Ramp testing in the R2 reactor began in 1969. Originally, the tests were performed by rapid power increases of the whole reactor. This mode of operation had two disadvantages: safety restrictions limited the maximum allowable power increase and financial restrictions limited the number of tests that could be performed since other experiments, which would have been disturbed by the ramp tests, had to be unloaded from the reactor at considerable expense.

In the present Ramp Test Facility, introduced in 1973, the fuel rod power during a ramp test in a loop is controlled by variation of the ^3He gas pressure in a stainless steel double minitube coil screen which surrounds the fuel rod test section. The principle of operation of this system is based on the fact that ^3He absorbs neutrons in proportion to its density, which can be varied as required by proper application of pressure.

The axial location of the minitube coil screen in the loop U-tubes is shown in Figure 6. The efficiency of the ^3He neutron absorber system makes it possible to increase test rod power by a factor of 1.8 to 2.2 (depending on the fissile content of the fuel). The ^3He absorber system is designed to achieve a 100 % power increase within 90 seconds, when operating with the normal pressure variation (bellows system).

In order to achieve a higher power increase than a factor of about 2, the reactor power must be increased before or simultaneously with the "^3He ramping". This technique with combined ramp systems is called "double step up-ramping", Figure 7. The technique makes it possible to increase the test fuel rod power by a factor of about 3.

An important advantage of the R2 Ramp Test Facility is that test rods, one at a time, can be loaded and unloaded during reactor operation. This is done by means of a lock vessel built onto an axial drive mechanism with about 3.5 m stroke. This lock vessel is bolted on top of a lock valve (ball type valve) fixed on top of the ramp rig. For BWR pressure conditions and normal rod lengths there is a 4-rod revolver lock vessel with a mechanical chain drive. For tests of PWR fuel rods and longer than normal rods, there exists a selection of rod lock vessels and dedicated hydraulic drives. Figure 6 shows a ramp rig with a hydraulic drive.

In the Ramp Test Facility ramp rates can be achieved in the range of 0.01 W/(cm·min) to about 3 000 W/(cm·min).

The maximum achievable ramp terminal level depends on the neutron flux in the experimental position and on the fissile content in the test rod. Tables 4 and 5 present the estimated ramp terminal level of some typical rods as function of burnup. The fissile content left in a BWR rod is highly dependent upon the axial location of the rod in the bundle, making the ramp terminal level estimation for BWR rods more uncertain than an estimation for PWR rods.

Table 4 Estimated Ramp Terminal Level of PWR Rods.

Burnup (MWd/kgU)	Ramp terminal level (kW/m) 17x17 PWR (4.5 %)[1]	15x15 PWR (3.4 %)[2]
35	70	63
55	50	40
65	40	35

Table 5 Estimated Ramp Terminal Level of BWR Rods.

Burnup (MWd/kgU)	Ramp terminal level (kW/m) 8x8 BWR (3.1 %)[3]	10x10 BWR (4.3 %)[4]
35	70	69
55	51	40
65	45	31

The power (linear heat generation rate in the fuel rod) is measured calorimetrically by the use of two inlet thermocouples, two outlet thermocouples, a venturi flowmeter and a pressure gage. The special calibration techniques employed have been described (3). The estimated uncertainty ($\pm 1\sigma$) is 2.3 % when the most common rod lengths (0.3-1.4 m) are used. The reproducibility obtained, when a fuel rod is irradiated several times in the same ramp rig, is ± 1 %. For fast ramps the discrepancy between the terminal power aimed at and the one obtained is less than ± 1 kW/m.

The axial thermal neutron flux distribution is measured by activation of cobalt wires in dummy rods and by gamma scanning of the ramp tested fuel rods. An axial movable SPN detector and gamma thermometer can also be used.

Each test rod is mounted in a separate stainless steel "capsule" (a shroud open at both ends), primarily as a safety measure and to facilitate the removal and handling of test rods that fail in the course of a power ramp. The "capsule" with the fuel rod is connected to the actuating rod which is used to move the fuel rod axially between the rod changing device and the in-pile section of the rig. There is a small floating push-rod built in at the bottom guide plug of the capsule. This push-rod transmits the elongation movements of the test rod to a LVDT type elongation detector built into the bottom of the ramp test rig.

2.5 Fuel Rod Failure Detection System

Fuel rod failures in the loops are detected by a Čerenkov-type radiation sensor which monitors the activity of the loop coolant water. The Čerenkov detector is installed in a by-pass circuit in order to increase the detection system sensitivity by decreasing the background ^{16}N activity produced in the loop coolant water. The ^{16}N background activity is decreased by the introduction of a delay time due to the fact that the Čerenkov detector is positioned in the by-pass circuit.

[1] Typical 17x17 PWR rod: Initial enrichment = 4.5 %
[2] Typical 15x15 PWR rod: Initial enrichment = 3.4 %.
[3] Typical 8x8 BWR rod: Initial enrichment = 3.1 %. Void = 60 %.
[4] Typical 10x10 BWR rod: Initial enrichment = 4.3 %. Void = 60 %.

The system detects fuel rod failure after 155 ± 10 seconds. This degree of failure detection capability has been verified experimentally. An example of system operation during a ramp test where the rod failed is shown in Figure 8.

However, the moment of failure is also registered instantaneously by the rod elongation measurement system as a sudden rod contraction and also often by the power measurement system as a small thermal "spike" (3,6,7).

2.6 Noise Analysis

Noise analysis can be a useful tool for monitoring changes in the fuel thermal response as well as the mechanical state during a rod irradiation. Usually the fuel thermal time constant is evaluated by analyzing the coolant temperature response and the clad elongation response to the rod power variation. The rod power is proportional to the neutron flux.

Natural R2 reactor power noise (± 0.5 %) is not sufficient to permit any determination of a test fuel rods thermal time constant. It is necessary to generate an artificial reactor power noise with a power variation of about ± 4 %.

Figure 9 presents the differences between natural and artificial noise; the rod elongation response to the nucleonic signal is clearly revealed.

The noise measurements performed in connection with studies of the rod thermal performance have been described elsewhere (8-10).

2.7 Data Acquisition

Three independent systems are used for data acquisition and storage. All three systems use the same sensors.

Most reliable is a high accuracy system, with a scanning rate of 1.0 Hz, linked to the STUDSVIK Hewlett-Packard 9000 computer. During experiments a PC data sampling system is added, and working with a scanning rate of up to 20 Hz it describes the time behavior very well. The data from the PC system are calibrated towards the high accuracy system and are used both as a backup system and as a source for rapid time-dependent event studies. A chart recorder with variable paper speed also describes the change of data with time.

2.8 INCA - A Facility for Materials Testing, Water Chemistry and Corrosion Studies

The experimental program at Studsvik has recently been extended to include investigations on cladding and structural materials in an in-pile corrosion rig, INCA (In-Core Autoclave).

A number of different conditions influence the corrosion behavior of reactor materials:

- Water chemistry
- Crud deposits
- Material characteristics, e.g. annealing treatment
- Neutron dose
- Hydrogen pick-up
- Boiling conditions.

The new INCA facility has been developed in collaboration with our sister company STUDSVIK MATERIAL AB. The main feature of the new facility is the ability to control and monitor the water chemistry. Therefore the facility is of the once-through type, which means that the rig is supported by a water supply system of its own and that the water passes the rig only once. The desired water chemistry is created by adding impurities and additives to a purified water flow, close to the test area. This technique has successfully been used out-of-pile by STUDSVIK MATERIAL AB to obtain well-characterized conditions. The facility has a flexible design and can easily be modified to suit different types of corrosion and water chemistry experiments.

The corrosion test rig, which can be seen in Figure 10, is the in-pile part of the INCA facility and is installed in one of the main in-pile loops in the R2 test reactor. It consists of two major parts, the rig tube and the electrode rod.

The rig tube separates the water system of the test rig from the in-pile loop main flow. The inlet water to the rig, which is degassed high purity water, is fed from a separate water supply system. It is heated by the loop water in a preheater coil and subsequently led into the rig tube. Additives and impurities (oxygen, hydrogen peroxide, zinc etc) can be added both before the rig and inside the rig just before the in-core test area in order to establish the desired water chemistry. The presently available inner diameter of the rig tube is 21 mm, but it is possible to vary the diameter or to have different diameters in different axial sections of the core.

The electrode rod is installed in the rig tube and is a carrier for the test specimens, the reference electrodes etc. The tube for the injection flow is also assembled on the rod. This arrangement makes it possible to change the electrode rod from one reactor cycle to another. For the moment the rod is bolted on top of the rig tube, but it could easily be rebuilt to be movable up and down in the core section and below the core during operation.

Figure 11 is a principle flow diagram showing the feed water system and the analysis system. In order to control the water chemistry the water is analyzed before and after the rig. The oxygen content is measured by an Orbisphere instrument and the content of hydrogen is measured by a gas chromatograph.

The inlet water is degassed deionized high purity water with a conductivity less than 0.08 µS/cm. A diaphragm pump with a variable flow is used to feed the water into the rig. Hydrogen is added to the water both at the inlet and outlet side of the main circulating pump by a dosage pump or by mixing the high purity flow and water saturated with hydrogen. The dosage of additives and impurities such as oxygen, hydrogen peroxide, boron, lithium, zinc etc is done by a controllable piston pump and is mixed with the main flow before or in the rig as described before. This arrangement gives a possibility to vary the flow through the rig, and to create the desired water chemistry.

The INCA facility can operate under both BWR and PWR conditions. Fast (> 1 MeV) and thermal neutron fluxes up to 1.9 and $2.0 \times 10^{14} n/(cm^2 \cdot s)$, respectively, can be achieved.

The facility is suitable for different kinds of experiments, for instance materials irradiations, waterside corrosion studies and in-core materials testing, all under controlled water chemistry. It has been in operation since March 1995. One of the objectives has been to develop reference electrodes for long term in-pile use. A radiolysis study where experimental measurements and computer model results were compared has also been performed (50).

2.9 Defect Fuel Degradation Studies

The Ramp Test Facility is utilized with a new test technique to investigate the potential degradation due to hydriding of test fuel rodlets with primary defects. A test of this type, to be used at the R2 test reactor, has to fulfil certain experimental constraints, i.e.

1 the length of the test fuel rodlets should not exceed the active core height (600 mm) of the R2 test reactor

2 the release of fission products and fuel material from the defect fuel must be minimized in order that the contamination of the in-pile loop circuit water be tolerable.

Both these requirements can be met with the unique but simple test technique adopted. Instead of an artificial primary defect (drilled hole etc) the primary defect is now simulated by a special device at the top end of the fuel rodlet (12). This device contains an enclosed small reservoir of liquid water in the "extension plenum", Figure 12. This extra plenum is connected with the normal plenum volume of the rodlet through a tiny tube in such a manner that only water in the form of steam will communicate with the plenum volume and the interconnected void space of the interior of the rodlet. Heat is transferred to the enclosed water reservoir from the surrounding pressurized loop water at the start of the operation until the pressure and temperature of the whole system are in balance, simulating either BWR or PWR operating conditions.

An important experimental feature of this arrangement is the possibility of post-irradiation puncture and subsequent collection of released fission gases and hydrogen from the rodlets which have been irradiated without failure of the cladding.

3 Pool-Side Examination

The general appearance of the irradiated fuel rods can be studied by visual inspection in the R2 pool (3). The following phenomena can also be investigated in detail:

Dimensional changes, ridge formation, rod bow and creep-down can be investigated with equipment for profilometry and length measurements. The existence and location of fuel rod defects can be established by means of eddy current testing.

The axial distribution of certain nuclides is determined by axial gamma scanning of fuel rods or of cladding samples. Data obtained before ramp tests are used as a check on the burnup profile during the base irradiation. Data obtained after ramp testing are used to check the power profile during the R2 irradiation and for studies of the fission product redistribution.

Neutron radiography can be used to study the general appearance and dimensions of the fuel, the extent of filling out of pellet dishing, of center porosities and of center melting (13, 14). This type of examination also reveals the presence of special fuel cracks, hydrides in the cladding, inter-pellet gaps etc. Indications of cladding failure and of structural changes in the fuel can also be observed. In cases where there is no apparent leakage of fission products from failed fuel rods neutron radiography is an important tool since minute cladding leaks are indicated by the existence of hydrides in the cladding or by the presence of water.

4 Post Irradiation Examination

Post-irradiation examination of irradiated fuel is performed in STUDSVIK's well-equipped Hot Cell Laboratory, which has been described in separate publications (15, 16).

Post-irradiation examinations of structural and cladding materials are performed in STUDSVIK's versatile Lead Cell Laboratory, which contains equipment for mechanical testing, corrosion testing and metallography. Among the different types of post-irradiation tests performed are tensile tests, fatigue tests, low-cycle fatigue tests, creep tests, impact test, stress corrosion tests and microscopy (optical microscopy, SEM/EPMA, both with image analysis, and TEM).

5 Fuel R&D

5.1 General Information

Much of STUDSVIK NUCLEAR's R&D work in the fuel area has been concentrated on fuel testing, which can be made in the R2 test reactor with high precision under realistic water reactor conditions. This type of work was started in the early 1960's. In a very general sense the purpose of fuel testing can be described as follows:

- Increasing of reactor availability by decreasing fuel-related operational power restrictions, defining the operational power limits.
- Acquisition of experimental data for fuel-related safety considerations.
- Decrease of fuel costs by making increases in fuel burnup possible.

The fuel testing activities can be divided into a number of well-defined steps as follows:

- Base irradiation, performed
    - in a power reactor, or
    - in STUDSVIK's R2 test reactor.
- Power ramping and/or other in-pile measurements, performed
    - in STUDSVIK's R2 test reactor.

- Non-destructive testing between different phases of an experiment, performed
    - in STUDSVIK's R2 test reactor pool, or
    - in the Hot Cell Laboratory.
- Destructive post-irradiation examinations, performed
    - in STUDSVIK's Hot Cell Laboratory, or
    - in the sponsor's hot cell laboratory.

Fuel examination can also be performed on standard (full-size) fuel rods from power reactors, which can be investigated in the Hot Cell Laboratory. If required, some types of tests could also be performed on such fuel rods in the R2 test reactor. However, due to the rather large initiation costs, such tests have not yet been performed. It should be noted, however, that short fuel rodlets, suitable for ramp testing and other on-line measurements in the R2 test reactor, can now be fabricated from irradiated full-size power reactor fuel rods by the STUDFAB refabrication process.

Several new hot-lab techniques have also been introduced in recent years (15, 16). The STUDFAB process for fabrication of rodlets from full-size fuel was mentioned above. Fuel ceramography can include scanning electron microscopy (SEM) and electron probe microanalysis (EPMA).

Descriptions of the fuel testing facilities and the techniques used were given in Section 2 above. Several other novel testing techniques have also been introduced. A very fast ramp rate, up to 3 000 W/(cm·min) can be used to obtain fast power transients and to determine the pellet-clad interaction/stress-corrosion cracking (PCI/SCC) failure boundary. The double step up-ramping technique was described in Section 2.4. On-line elongation measurements can be performed during ramp tests, Figure 8. Test fuel rods can be fitted with on-line pressure transducers through a refabrication process.

5.2 Types of R&D Projects

Ramp tests incorporating a very fast-responding test rod power measuring system and associated on-line measurements, such as rod elongation and noise measurements for studies of the rod thermal performance, are performed in the pressurized water loops.

The ramp tests are a form of integral performance tests where the complex interplay between the pellets and the cladding of a power reactor fuel rod is reproduced. The primary test objectives are:

- Determination of the failure boundary and the failure threshold, see Figures 13 (17) and 14.
- Establishing of the highest "conditioning" ramp rate that safely avoids failure occurrence.
- Study of the failure initiation and progression under short time over-power transient operation beyond the failure threshold.
- Proof testing of potential pellet-clad interaction (PCI) remedies.

Other, more specific test objectives have also been pursued in some projects.

The rod overpressure experiments utilize the on-line measurements associated with the ramp tests combined with non-destructive examinations between reactor cycles and destructive examinations after the irradiation. When LWR fuel is used at higher and higher burnups the question of how the fuel might behave when the end-of-life rod internal pressure becomes greater than the system pressure attracts a considerable interest. On one hand end-of-life overpressure might lead to clad outward creep and an increased pellet-clad gap with consequent feedback in the form of increased fuel temperature, further fission gas release, further increases in overpressure etc. On the other hand increased fuel swelling might offset this mechanism. In connection with such considerations the Rod OverPressure Experiments (ROPE) were initiated.

The defect fuel degradation experiments also utilize the on-line measurements associated with the ramp tests and combine these with non-destructive and destructive examinations after the irradiation. Fretting type failures are predominant causes of the very few fuel failures that have occurred in recent years in LWRs. These primary failures are sometimes followed by secondary failures which frequently cause considerably larger activity releases. In such cases the subsequent degradation of the defect fuel rods by internal hydriding of the cladding and by oxidation of the fuel are the common destructive mechanisms. In these tests an irradiation test scheme, adapted to the experimental conditions in the R2 test reactor was introduced. This scheme offers the possibility of executing comparative investigations of the process of degradation of commercial types of LWR fuel under simulated primary defect conditions as well as of the mechanisms involved.

5.3 Sponsorship

The fuel testing projects executed at Studsvik have been organized under three different types of sponsorship:

International (multilateral) fuel projects

- Jointly sponsored internationally on a world-wide basis.
- Project information remains restricted to the project participants throughout the project's duration and some predetermined time after project completion.

Bilateral fuel projects

- Sponsored by one single organization, or a few co-operating organizations.
- Project information remains restricted to the sponsor, sometimes published later.

In-house R&D work

- Sponsored by STUDSVIK NUCLEAR.

5.4 International R&D Programs

Since 1975, a series of international fuel R&D projects addressing the PCI/SCC failure phenomenon have been conducted under the management of STUDSVIK NUCLEAR (17-20). These projects are pursued under the sponsorship of different groups of fuel vendors, nuclear power utilities, national R&D organizations and, in some cases, licensing authorities in Europe, Japan and the U.S. In most of the projects the cladding failure occurrence was studied under power ramp conditions utilizing the special ramp test facilities of the R2 reactor. The recent projects have not been limited to PCI/SCC studies but also included other aspects of fuel performance, such as end-of-life rod overpressure (21-23) and defect fuel degradation (24-27, 52). Overviews of the projects that have been completed and those that are in progress or planned have been published (20, 28), see also Appendix 1 and 2. In most cases, the test fuel was base irradiated in commercially operating light water power reactors. In some instances the base irradiation took place in BOCA rigs in the R2 reactor.

In general, the international fuel R&D projects can be divided into two main categories:

- Projects aimed at decreasing the fuel costs by increasing fuel utilization and reactor availability.

- Projects providing data for fuel-related safety considerations.

A summary of some of the data from the former category of projects is shown in Figure 13. Reviews of the projects in the latter category have also been published (29-30).

Over the past years STUDSVIK has studied the behavior of defect fuel in a series of defect fuel experiments. The international DEFEX Project, which was concluded in 1995 addressed hydriding of fresh 8x8 BWR type fuel. Some results from this type of work have been published (24-27, 52).

A new defect fuel degradation experiment, DEFEX II, is under discussion. The main program will include:

- Studies, analogous with the DEFEX Project, of the degradation process in irradiated fuel rods, but in this case with medium (around 20 MWd/kgU) burnup with simulated primary defects

- Testing of potential remedies against secondary failure by hydriding in order to supplement the experimental data base

- Modeling of defect fuel behavior

- Studies of cracking behavior of the hydrided cladding.

The primary defect would be simulated by a technique entailing delayed intrusion of water into the rodlet, thus simulating a case when a power rod is defected during operation.

A two-year program is envisaged, starting in 1997. The primary goal is to study the behavior of BWR fuel. A PWR program is also planned but would have to be separately funded. It is assumed that the scope of DEFEX II would be approximately the same as the DEFEX I Project, i.e. 6-8 irradiation tests would be run, and destructive and non-destructive examination would include: neutron radiography, eddy current testing, dimensional measurement, visual inspection, gap squeeze measurement, gamma scanning, SEM investigation, internal gas analysis, metallography/ceramography, fracture toughness testing of the cladding and stoichiometry determination.

A study of the high-burnup behavior of PWR fuel exposed to power ramps, the Ultra-High Burnup Fuel Irradiation Project, is under discussion. In this project a few fuel rodlets, previously irradiated in a power reactor to a burnup of at least about 60 GWd/tU, will be irradiated further in the R2 test reactor to a total burnup of about 80 GWd/tU. The properties and behavior of the fuel after different types of ramps will be studied.

A new investigation of the ramp resistance, ULTRA RAMP, would address the question of the ramp behavior at high burnup (above 50 MWd/kgU). This question has been widely discussed in recent years as regards both normal and off-normal ramp rate conditions. However, only limited experimental information seems to be available so far (31).

The concern addressed appears to relate to the impact of changes in the physical properties of the fuel pellets at high burnup and their effects on the ramp behavior of the fuel rods. The fuel pellets seem to crack up in minor fragments and may no longer behave as solid bodies. The fission gases will be entrapped in a magnitude of small bubbles and might cause fuel rod swelling on up-ramping. Other concerns relate to the loss of thermal conductivity and the impact of the rim zone on fuel ramp behavior,

The prospective ULTRA-RAMP project would constitute a combination of three groups of ramp experiments: one series concentrated on the PCI phenomena under normal operating conditions in different types of fuel and the other two series concentrated on more safety-oriented issues. Thus the ramp resistance in current fuel types would be studied both under normal operating conditions ("slow" ramps) and under off-normal operating conditions ("fast" ramps corresponding to ANSI Class II and III events and "ultra-fast" ramps corresponding to some ANSI Class IV events).

A few recent simulated RIA experiments (Reactivity Initiated Accidents) with high-burnup fuel (55 and 65 MWd/kgU) have focused interest on Class IV events. STUDSVIK is proposing a new type of "ultra fast" ramps, faster than the fast ramps performed in earlier safety-related projects (TRI, TRII) but slower than the simulated RIA experiments. These new "ultra-fast" ramps could reach e.g. 120 kW/m during an 1 sec effective ramp time, corresponding to an enthalpy increase of 30 cal/g.

An MTR type reactor typically gives high neutron fluxes due to its compact design with a high power density in the core. The thermal neutron flux can be further increased by overmoderation in a flux trap in the test rig. This method gives a test position in the reactor with a thermal flux level of more than $4x10^{14}$ n/(cm^2·s), which is sufficient to give a PWR test rod (with a burnup of 45 MWd/kgU) a steady maximal LHR of over 100 kW/m. The flux trap also functions as a stabilizing tool for the test position, resulting in a reactivity balance for the rod in - out of less than 50 pcm.

Small test rodlets of about 10 cm length refabricated by the STUDFAB technique, are to be tested in the R2 flux trap facility. The test rod will be pushed from a non-flux position to a high-flux position by a mechanical device, achieving high power levels in the high flux position. The following measurements of interest can be performed during the experiment: Elongation measurements, internal pressure and diametral changes. The energy deposition is controlled by the time the test rod is located in the high-flux position in the flux-trap facility.

Different operational parameters can be selected by choice of temperature (from cold to operational temperature), pressure (from low to operational pressure) and power (from zero to operational power).

Thus in the ULTRA-RAMP project a number of high burnup fuel rodlets of BWR or PWR types, or both, would be exposed to fast and slow ramp rates to preselected terminal power levels and to "ultra-fast" ramp rates to preselected enthalpy increases, all in the R2 test reactor (31). The main objective is to identify any adverse or inadequate fuel rod behavior as for example abnormal fuel rod swelling, fast failure of the cladding or loss of fuel integrity on clad fracturing causing dispersion of fuel particles in the coolant water. Detailed non-destructive and destructive examinations (including advanced types of ceramography) would follow.

A more "conventional" PCI resistance study, SUPER-RAMP III/10x10, is also under discussion. In the new types of 10x10 BWR fuel the Linear Heat Rate is lower than in earlier types of fuel and the PCI resistance is presumed to be correspondingly improved. However, some utilities using zirconium liner with earlier types of fuel have raised the question whether the lower linear heat rates in 10x10 fuel really make the added resistance against PCI failure, achieved with zirconium liner fuel, unnecessary. As far as is known no ramp tests have ever been performed on 10x10 fuel.

Thus the proposed SUPER-RAMP III/10x10 project would be similar to the earlier SUPER-RAMP II/9x9 project (32). It is planned to start in 1997 and be completed in 1997.

A new type of test is planned where the stored energy (enthalpy) of different types of fuel rods will be investigated, the STEED project. The stored energy (the enthalpy) in fuel rods during operation depends on the fuel design (dimensions, materials), operating conditions and burnup. Experimental determinations of the stored energy provide a valuable alternative to fuel code calculations. Depending on the test technique used the accuracy can be better than the corresponding code calculations, especially with increasing burnup. The amount of stored energy is an important parameter in connection with safety considerations such as LOCA evaluations. The experimental results for unirradiated fuel can also serve as excellent bench-marking opportunities for fuel modellers.

STUDSVIK's R2 test reactor is well suited for scram experiments where the thermal response of different types of fuel can be compared. The measurements for the STEED project will be performed by analyzing the heat release from a test rod after scram, thereby using the R2 test reactor's calorimetric rod power measurement system. This system is the very same as that used in ordinary ramp experiments.

A demonstration experiment, STEED-I, is in progress on unirradiated fuel rodlets. The objective is to verify the test technique and to evaluate the accuracy. An international project, STEED-II, based on tests of irradiated fuel rodlets, will be discussed in 1997. An example of a "pre-project" demonstration test is shown in Figure 15.

A safety-oriented PCI study, TRANS-RAMP III, is also under discussion. Experience from the earlier safety-oriented STUDSVIK projects, as well as from power reactor operation, shows that non-penetrating cladding cracks form during certain short-time power transients and cracks initiate already within 5-10 seconds. However, it is conceivable that these non-penetrating (incipient) cracks will not propagate further during continued operation owing to some passivation effect. The main purpose of the TRANS-RAMP IV project was to investigate experimentally the propensity for through-wall crack penetration in PWR fuel rodlets of such initially non-penetrating (incipient) PCI cracks following a second power transient (29-30, 51).

In the TRANS-RAMP III (TR III) project the influence of non-penetrating (incipient) cladding cracks in BWR fuel rodlets on the PCI failure resistance during an anticipated subsequent transient occurring later in life will be studied. A project proposal will be circulated when data from the TR IV project will be available, in late 1996.

Test data from the international R&D projects are often used as "benchmarking" data in the project participants' own fuel modeling work. In recent years many ramp test data have also been analyzed with the INTERPIN code, developed by STUDSVIK (33-35). INTERPIN is a fuel performance code which satisfies real-time simulation requirements when implemented on a minicomputer.

5.5 Bilateral Programs

European, Japanese and U.S. fuel manufacturers and research organizations have also for many years been utilizing the R2 test reactor and the associated hot-cell laboratories for bilaterally sponsored research[1]. ABB Atom AB has made many series of ramp tests. General Electric Co. has executed several series of ramp tests at R2, as part of the efforts to develop the zirconium barrier fuel concept. Some of the ramp techniques requested were innovative, for example the "double ramping" of the test rods. Exxon Nuclear Co. (later Advanced Nuclear Fuels Corporation, later Siemens Nuclear Power Corporation, now Siemens Power Corporation) is another major customer as well as Hitachi Ltd. and Toshiba Corporation. Mitsubishi Heavy Industries, Ltd. has performed combined power cycling and ramp tests. B&W Fuel Company (now Framatome Cogema Fuels) performed ramp tests on high-burnup PWR fuel. Tests have also been performed on behalf of other organizations but the results have not always been published.

During the 1970's extensive series of HTR fuel irradiations were performed in a special HTR gas loop system operating with on-line measurements and analyses of the released fission gas and of the fuel temperature (5).

5.6 In-House R&D Programs

STUDSVIK NUCLEAR's in-house R&D work is mainly associated with improvements of test irradiation techniques, instrumentation and post-irradiation examination, all in support of ongoing or upcoming irradiation projects. Progress in these areas has made it possible to achieve important progress in fuel research. For example, the characterization of the PCI failure progression in some recent projects was only made feasible through a combination of several new techniques. These included a very fast ramp execution using a ramp rate of up to 3 000 W/(cm·min), compared to the previous maximum of 200 W/(cm·min), a prompt detection of the through-failure event using the on-line elongation detector and a subsequent special clad bore inspection technique. Another result of the in-house R&D work is the noise measurement technique mentioned above.

STUDSVIK NUCLEAR has also been carrying out an in-house R&D program aimed at improving the performance of LWR fuel by the utilization of a design concept with cladding tubes which have been "rifled" on a micro-scale (36). Results of R2 irradiations of such fuel and the associated modeling work have been published (37-44).

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[1] A list of available publications can be obtained upon request from the authors.

6 Materials R&D

The R&D work in this area consists of studies of irradiation effects in structural materials. These types of studies have been concentrated on pressure vessel steels, stainless steels and nickel-base alloys (for super-heater and fast reactor cladding) during the 1960's and on zirconium alloys since the 1970's. The early pressure vessel steel work comprised investigations of different potential pressure vessel materials, of different materials variables, and of the influence of different irradiation conditions (neutron fluence, irradiation temperature etc). Recent work has been concentrated on accelerated irradiation of materials actually used in pressure vessels under as realistic conditions as possible. More recent work on stainless steels was to a large extent concentrated on fusion reactor materials within the Next European Torus (NET) program, where tensile tests, fatigue tests, and stress corrosion tests have been performed after irradiation to displacement doses of up to 10 dpa. Some work, including post-irradiation creep and fatigue tests and crack propagation studies (including CT tests) has also recently been performed on potential FBR vessel materials. Among the in-pile tests performed stress-relaxation tests can be mentioned. In-pile corrosion tests in the R2 test reactor with on-line corrosion potential measurements were discussed in Section 2.8. The work on zirconium alloys is continuing and is being expanded in order to include in-pile corrosion studies.

Originally structural materials (pressure vessel steels, stainless steels, nickel-base alloys and aluminum alloys) were to a large extent irradiated in rigs in fuel element positions either in contact with the reactor coolant water (temperature about 60 °C) or in rigs where the heat-transfer conditions were closely defined and the specimen temperatures were measured but not regulated. Later rigs containing in-pile furnaces for constant temperature irradiations were introduced. Besides numerous detailed publications review papers were published on the early pressure steel work (45) and on the corresponding work on the other structural materials (46). During this period extensive work was also pursued on cladding materials for superheated reactors and for steam-cooled fast reactors. Some work was also performed on early Swedish candidates for FBR pressure vessel materials (47, 48).

During the 1970's the in-pile loops in the R2 test reactor became available for irradiations of specimens of structural materials. Such specimens are now irradiated either in rigs that only allow irradiations during whole 400 hour reactor cycles or in rigs where shorter irradiations, down to less than an hour, are possible. The specimens are either in direct contact with the loop water (temperature selected in the range 230-350 °C) or in some cases specimens of pressure vessel steels have been nickel plated in order to avoid corrosion problems during longer irradiations. Later new types of in-pile rigs for fuel element positions have also been developed where the specimens are heated by gamma heating. In these rigs close temperature control (about ± 10 °C in the range 250-350 °C) has been maintained by placing the specimens in specimen holders filled with a NaK alloy. The temperature is monitored by thermocouples placed in some specimens or in the NaK adjacent to the specimens. Temperature control is achieved by changes of the He/Ne gas mixture in the gap between the capsule containing specimens and NaK and the rig secondary containment. Varieties of these rigs are also used up to a temperature of 550 °C.

Current material irradiations are partly within the frame of the ITER project and partly within the future DEMO project. For the ITER project, work is carried out mainly on stainless steel and copper joined by hot isostatic pressing (HIP). Irradiations related to the DEMO project mainly comprise a special reduced activation stainless steel, type F82H. Long time irradiations up to 2.5 dpa are performed in the in-pile loops, where the specimen temperature is 290 °C. High-temperature irradiations at temperatures up to 400 °C are performed in NaK capsules placed in a central position in the core. In the latter case the specimen doses will range up to 0.7 dpa.

7 Neutron Transmutation Doping of Silicon

Neutron transmutation doping of silicon for industrial use in electric power components is done in a facility which is shown in Figure 16. Three shelves with silicon ingots can be irradiated simultaneously, but only the bottom shelf is shown in the figure. The silicon ingots are loaded manually onto the irradiation facility, which is situated in the R2 reactor pool, in front of the reactor vessel. The ingots perform a horizontal helical movement on the shelves in front of the core. The neutron flux is monitored through self-powered neutron detectors and the velocity of the ingots and hence the neutron fluence is controlled by a computer. After completion of the irradiation the ingots are removed from the shelves to a conveyor which slowly transports them to the pool surface. A permanent radiation instrument monitors the dose rate and in order to avoid hand doses to the operators the ingots are lifted with a crane. They are then stored for a few days in order to let the ^{31}Si and ^{32}P activities decay. The decontamination is done by rinsing in demineralized water.

Silicon ingots with lengths up to 600 mm and diameters from 60 to 152 mm are treated routinely. The target resistivity of the resulting conducting material usually lies in the 30-300 ohmcm range. The high uniformity and precision of the irradiation guarantees less than 5 % axial variation and a radial gradient which is better than 2 and 4 % for the minimum and maximum diameters, respectively. The day-to-day constancy of the operation of the facility is monitored by means of cobalt monitors attached to some of the silicon ingots.

The irradiated material is shipped in compliance with IAEA regulations. The bulk material must have a specific activity of less than 7.4 Bq/g before it is shipped as non-radioactive material. For the nonfixed surface contamination a limit of 0.4 Bq/cm^2 is maintained taking the ALARA principle into consideration. Application of these values to the process normally gives a turnover time of three weeks at Studsvik.

8 Neutron Activation Analysis

The present set-up for neutron activation analysis (NAA) permits multi-element determination by instrumental and radiochemical NAA of around 50 elements in trace concentrations. The samples which are investigated are of biological, environmental, industrial, or geological origin.

The samples usually require little or no pre-treatment, and after weighing they are pneumatically transported in plastic capsules to a position close to the reactor core. Having been irradiated the samples are left to decay in a storage unit for a suitable period of time, before the gamma radiation is registered with a Ge(Li) or a high purity Ge detector. The thermal and epithermal fluxes are measured with a Zr monitor. Thus the automatic data evaluation system gives an absolute determination of the composition of the sample from the intensity of the gamma spectrum, which is characteristic of each element.

Neutron activation analysis in theory permits determination of around 70 elements. In practice the number is limited to some 30 elements when using instrumental NAA. Application of radiochemical NAA increases this number.

Many elements can be determined at sub-ppb(10^{-9}) levels, but high concentrations of disturbing elements may be troublesome because of spectrum interferences. Sometimes corrections have to be made for other reasons, for instance due to interfering nuclear reactions. Neither the lightest elements nor lead and sulphur can be detected. Some examples of STUDSVIK NUCLEAR projects for which instrumental NAA has proved to be an efficient method are:

- Multi-element studies of geological samples, with special interest in rare earth elements and iridium.
- Uranium, thorium and other elements in sediment.
- Trace elements in metallurgical products.
- Trace elements in food-stuff.

9 Radioisotope Production

Radioisotopes can be produced over a wide range of conditions in several irradiation positions in and around the R2 vessel. The operational cycle of the reactor, however, to some degree limits the number of isotopes that are produced routinely.

There are six permanent rigs in the reactor core which are used for radioisotope irradiation. One of them can be loaded and unloaded during reactor operation. The maximum flux which can be obtained for irradiation is as high as 3×10^{14} n/(cm$^2 \cdot$s). The permanent rigs can be supplemented with temporary installations.

^{192}Ir is produced by irradiation in the core positions. The specific activity of the resulting product is higher than 10 TBq/g. The encapsulated isotope is used industrially for gamma radiography. ^{169}Yb, which is also produced, has the same application.

Eu is irradiated with the aim to extract ^{153}Gd, which is used for sources in bone scanners. ^{32}P and ^{35}S are examples of isotopes produced in R2, which are mainly used for biological research. There is also a wide variety of radioisotopes being produced for medical research and therapy, such as ^{85}Sr, ^{89}Sr, ^{86}Rb, ^{155}Cd, ^{110}Ag, ^{51}Cr, ^{59}Fe, ^{45}Cs, ^{47}Cs, ^{90}Y, ^{186}Re and ^{63}Ni.

A few other isotopes are produced routinely. ^{24}Na for example is delivered to the Swedish defence forces for training purposes.

10 Beam Tube Experiments

The R2 and R2-0 reactors serve as sources of thermal neutrons for a wide variety of basic research applications. The beam tubes at the R2 reactor are used for thermal neutron scattering experiments, see Figure 17 and Table 6. The R2-0 reactor, which is mobile in the pool, is in one position used as source for a boron neutron capture radiography facility and in the other position as source for a facility for nuclear physics and nuclear chemistry experiments based on an on-line isotope separator. Researchers from the universities have easy access to the facilities through the Studsvik Neutron Research Laboratory (49).The laboratory is organized as a department at the University of Uppsala but serves users from all Swedish universities. The instruments are also available for outside users, partly through a program financed by the European Community.

In connection with the replacement of the reactor vessel, a large D_2O box was installed on the outside of the core box. The reentrant beam tubes end at positions close to the thermal flux peak in the D_2O. Two tangential beam tubes were installed through a region in the biological shield which was previously inaccessible. The new beam tubes are rectangular with height 18 cm and width 8 cm. The larger vertical divergence of the beams increases the flux at the experimental positions. With these modifications substantial improvements in the thermal flux and in the ratio of thermal to fast flux at the experimental positions are achieved. The research performed involves, for example, structure determinations in crystals and amorphous systems, studies of magnetic phenomena in condensed matter, excitations in disordered systems and determination of residual stress and texture.

Table 6 Neutron Scattering Instrumentation at the R2 Reactor.

Neutron beam tube	Instrumentation
H1	Test beamline
H3 REST	Diffractometer for residual stress and texture measurements
H5 NPD	High-resolution powder diffractometer
H6	Vacant
H7 SLAD	Diffractometer for disordered materials
H8 SXD	Single/crystal diffractometer
H9 PREFECT	Polarized neutron reflectometer under construction
H10 TTOF	Time-of-flight spectrometer (under construction)

The above mentioned facility for boron neutron capture radiography (BNCR) is shown in Figure 18. A very pure thermal neutron field is produced by moderation of the fast neutrons from the reactor in a large D_2O volume positioned immediately outside the pool liner and adjacent to the reactor core, which is located immediately inside the pool liner. At the outer edge of the D_2O volume irradiations can be made in a thermal flux over a large area (30x30 cm^2). The thermal flux used in recent experiments has varied between 6×10^8 and 5×10^9 n/($cm^2 \cdot s$), corresponding to a reactor power of 25 to 200 kW. The facility is used extensively for biomedical research and has proved to be an efficient tool for studying the distribution of boron loaded compounds with a specific affinity for certain tumors.

A variety of nuclear physics and nuclear chemistry research programs are based on the on-line isotope separator OSIRIS at the R2-0 reactor. The main activity is aimed at studies of the properties of short-lived neutron-rich nuclides. The programs include determination of fission yields including branching ratios for gamma decay from fission products and determination of the antineutrino spectrum at a nuclear reactor. The system utilizes a novel method for plasma creation and allows higher temperatures, up to 2500 °C, and thereby shorter delay times for the released fission products. This has increased the number of nuclides available considerably and has increased the production yields of many short-lived isotopes by factors of 10^2-10^3.

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Appendix 1
Overview of STUDSVIK NUCLEAR's International Fuel R&D Projects 1975-1991

Project (duration)	Fuel Type (No of rods)	Base Irradiation (MWd/kgU)	Research Objectives	Data published
INTER-RAMP (1975-79)	BWR (20)	R2 (10-20)	Failure threshold Failure mechanism Clad heat treatment Modeling data	Yes (Ref 1,2) *)
OVER-RAMP (1977-80)	PWR (39)	Obrigheim (10-30) BR-3 (15-25)	Failure threshold Design parameters Modeling data	Yes (Ref 3,4)
DEMO-RAMP I (1979-82)	BWR (5)	Ringhals I (15)	PCI remedies (Annular, niobia doped pellets)	Yes (Ref 5)
DEMO-RAMP II (1980-82)	BWR (8)	Würgassen (25-29)	Failure threshold PCI damage by overpower transients	Yes (Ref 6,7)
SUPER-RAMP (1980-83)	BWR (16)	Würgassen (30-35) Monticello (30)	Failure threshold High burn-up effects PCI remedies Safe ramp rate Gd fuel	Yes (Ref 8)
	PWR (28)	Obrigheim (33-45) BR-3 (28-33)	Design parameters Modeling data	Yes (Ref 8)
SUPER-RAMP EXTENSION (1984-86)	BWR (9)	Oskarshamn 2 (27-31)	Safe ramp rate	No
	PWR (4)	Obrigheim (30-35)	Resolve unexplained failure resistance	No
TRANS-RAMP I (1982-84)	BWR (5)	Würgassen (18)	Failure boundary Crack init. and prop. Structural changes Fission gas release Modeling data	Yes (Ref 6,9)

*) References, see p. 31-32

Appendix 1, cont.

Project (duration)	Fuel Type (No of rods)	Base Irradiation (MWd/kgU)	Research Objectives	Data published
TRANS-RAMP II (1982-86)	PWR (7)	Zorita (30)	Failure boundary Crack init. and prop. Structural changes Fission gas release Modeling data	Yes (Ref 10)
ROPE I (1986-89)	BWR (4)	Ringhals (36)	Investigate clad creep-out as a function of rod overpressure	Yes (Ref 11)
SUPER-RAMP II/9x9 (1987-91)	BWR (4)	Dresden (30)	PCI performance	Yes (Ref 12)
TRANS-RAMP IV (1988-95)	PWR (7)	Gravelines (20-25)	Influence of non-penetrating cracks on PCI failure resistance	Yes (Ref 13)
ROPE II (1990-94)	PWR (6)	Ringhals Obrigheim (> 40)	Investigate clad creep-out as a function of rod overpressure	No
DEFEX (1993-95)	BWR (6)	Initially unirradiated rodlets	Study secondary damage formation in fuel rods with simulated fretting defects	No

Appendix 2
Overview of STUDSVIK NUCLEAR's Ongoing and Planned International Fuel R&D Projects

Project (duration)	Fuel Type (No of rods)	Base Irrad (MWd/kgU)	Research Objectives
DEFEX II (NS*), 1997-99)	BWR + PWR?	ND*) (10-20)	Study secondary damage formation in fuel rods with simulated fretting defects
ULTRA-RAMP (NS, 1997-	BWR PWR	ND > 50	Study PCI performance
UHBUP	PWR (3)	Power Reactor to > 60 then R2 (80)	Fuel behavior during ramp tests
SUPER-RAMP III /10x10 (NS, 1996-98)	BWR (ND)	ND	Study PCI performance
TRANS-RAMP III (NS, 1997-99)	BWR	ND	Influence of non-penetrating cracks on PCI failure resistance
STEED II	BWR PWR	ND	Stored energy (enthalpy) measurements

*) NS = Not yet started
ND = Not decided

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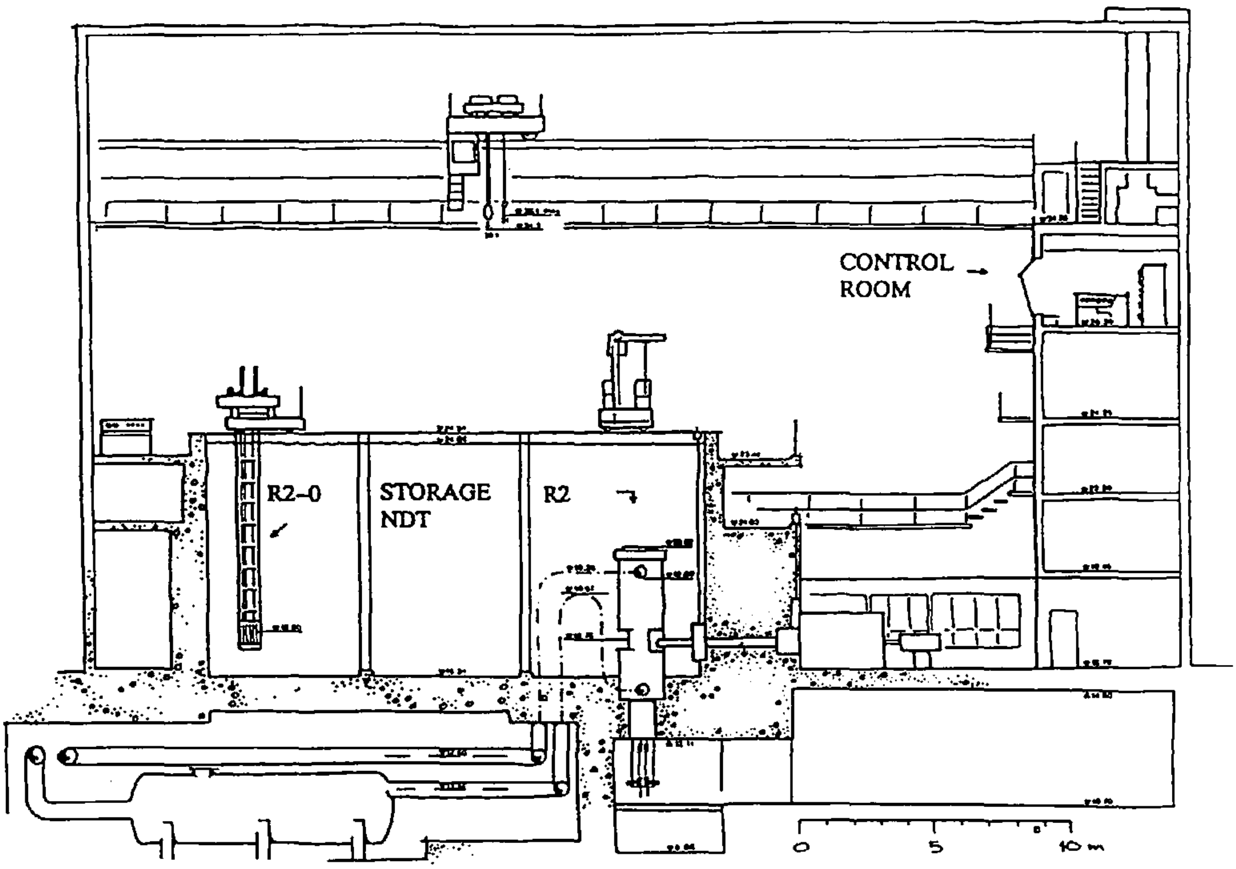


Figure 1
The R2 Test Reactor - General Arrangement.

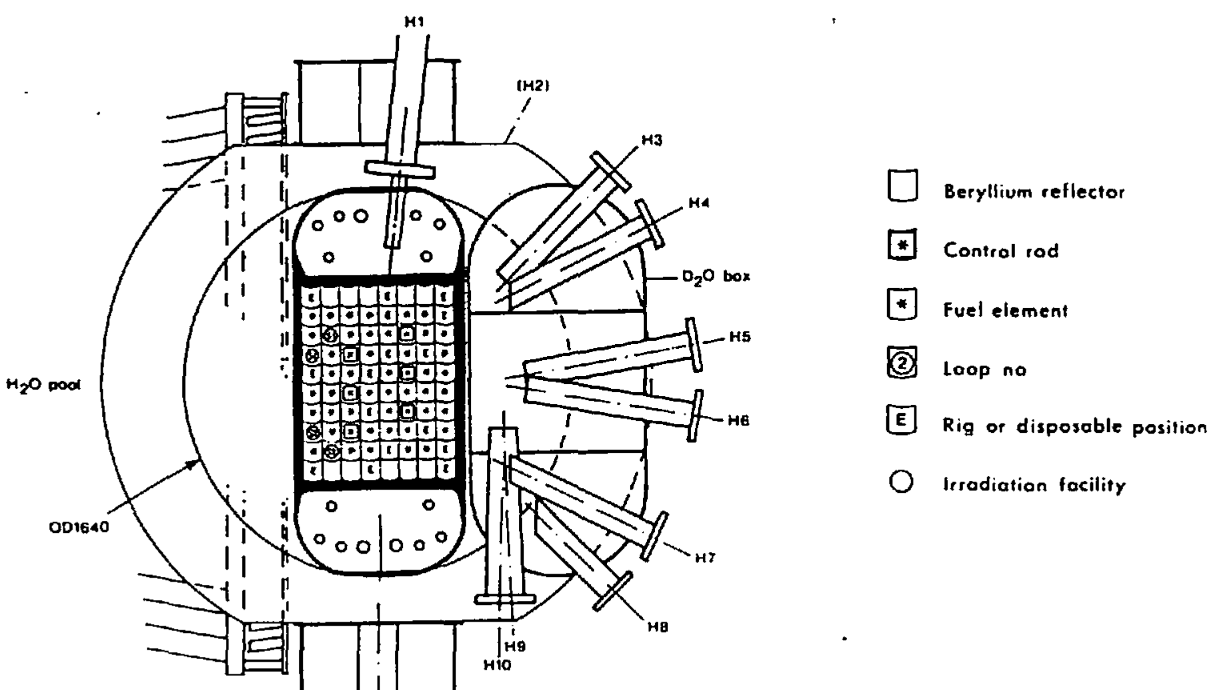


Figure 2
The R2 Test Reactor - Core Configuration.

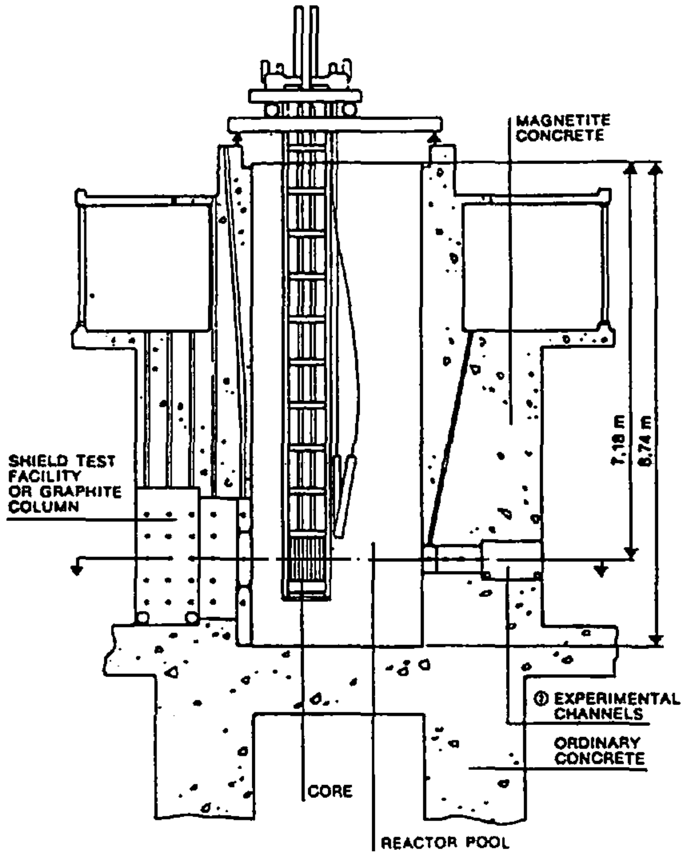


Figure 3
The R2-0 Reactor.

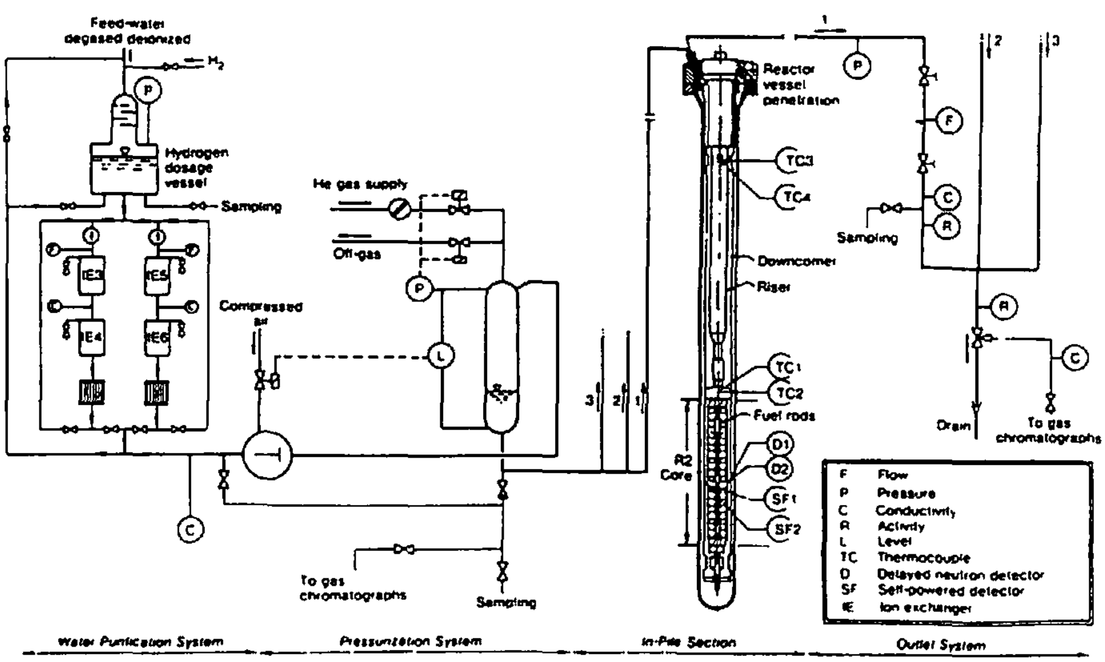
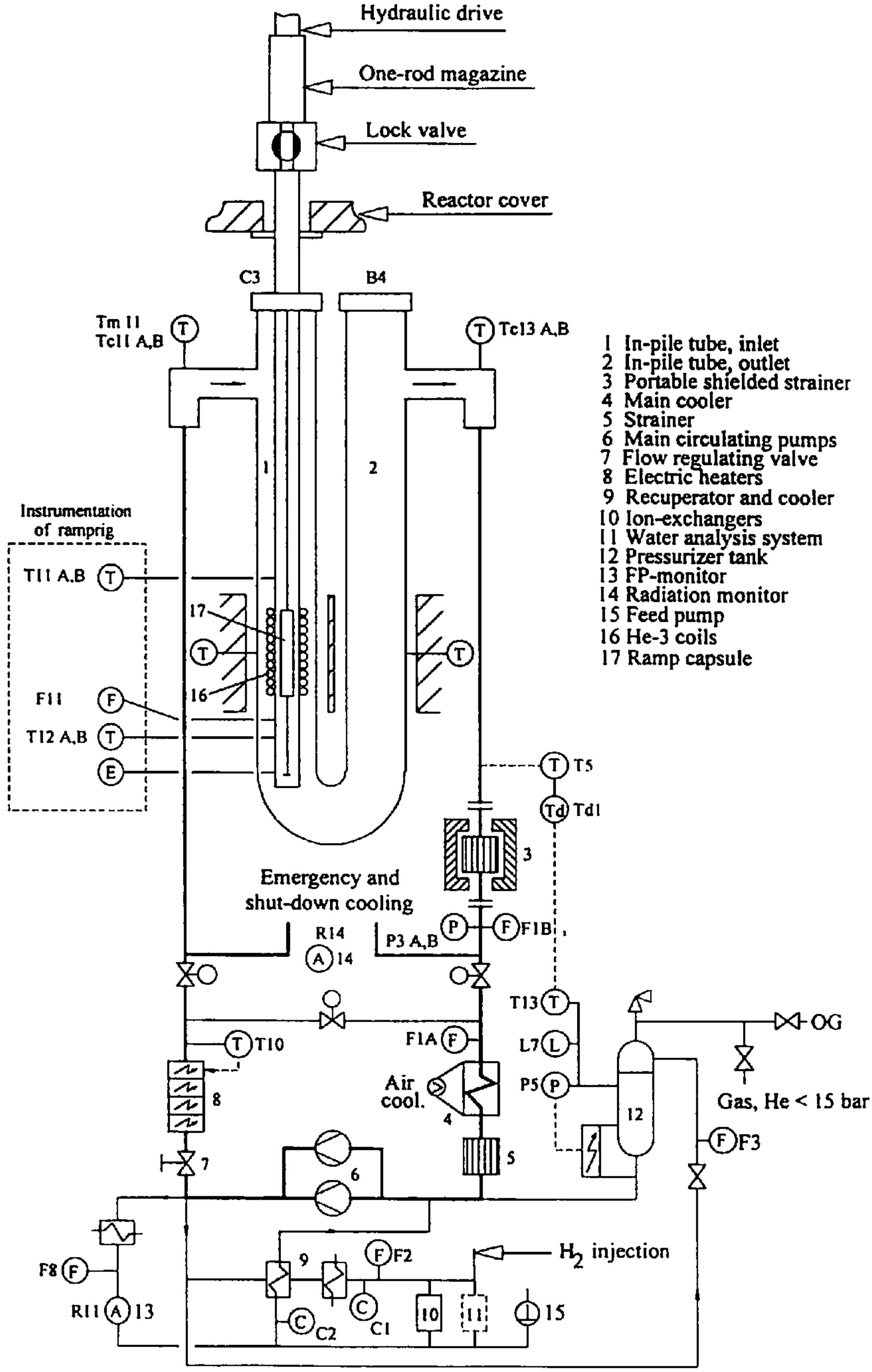


Figure 4
BOCA Rig in R2 - Simplified Flow Sheet and In-Pile Part of a Boiling Capsule.



C - Conductivity meter
F - Flow meter
L - Water level indicator
P - Pressure sensor
E - Elongation meter

T - Temperature sensor
Tc - Thermocouple
Td - Temperature difference measurement
Tm - Resistance thermometer

Figure 5
Loop No 1 in R2 with Ramp Test Rig - Simplified Flow Diagram.

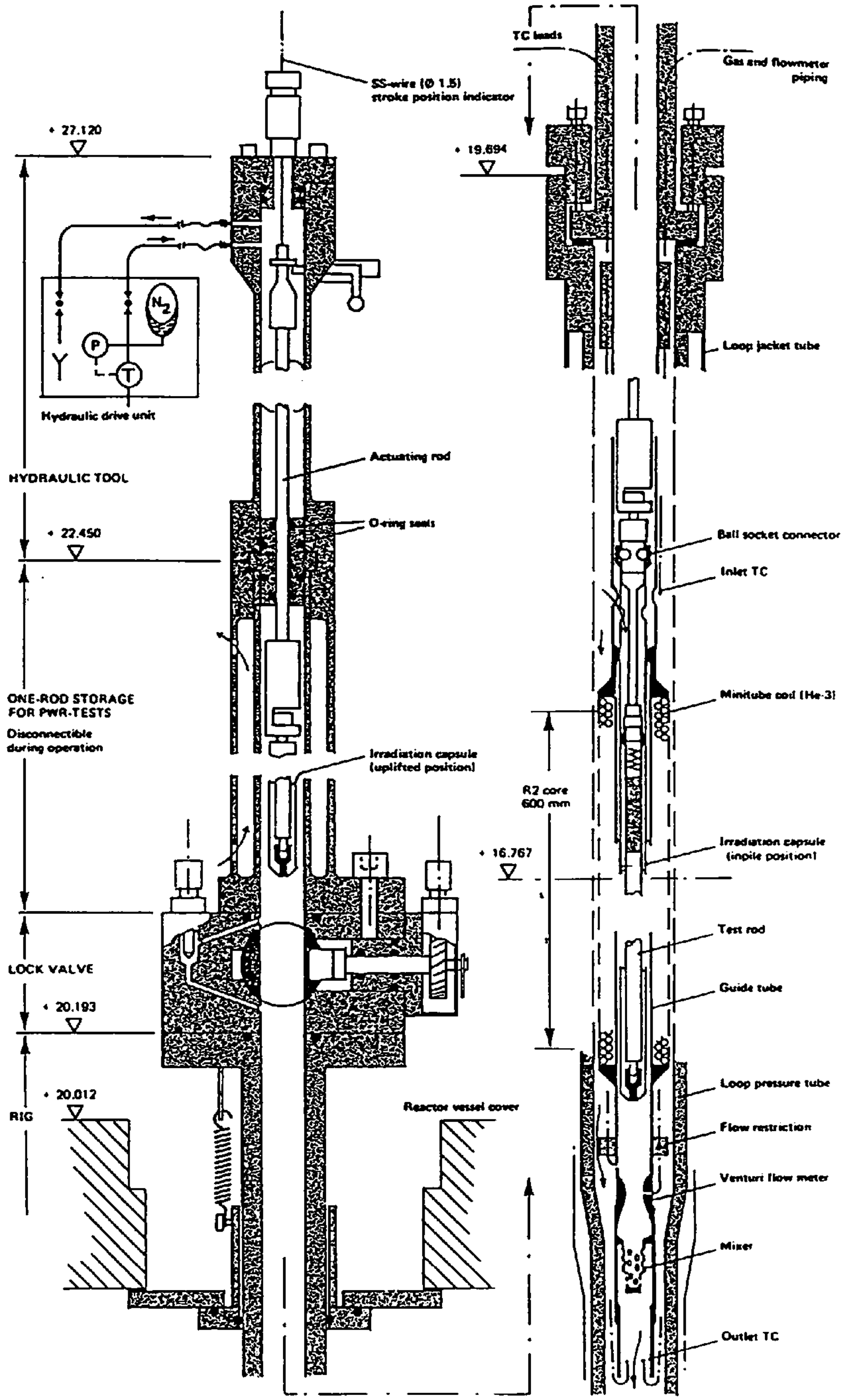


Figure 6
Loop No. 1 in R2 - PWR Ramp Testing Facility.

Recent versions include a transducer in the lower part of the rig. This transducer is used to monitor the length changes in the fuel rod. These changes are monitored continuously, see Figure 8.

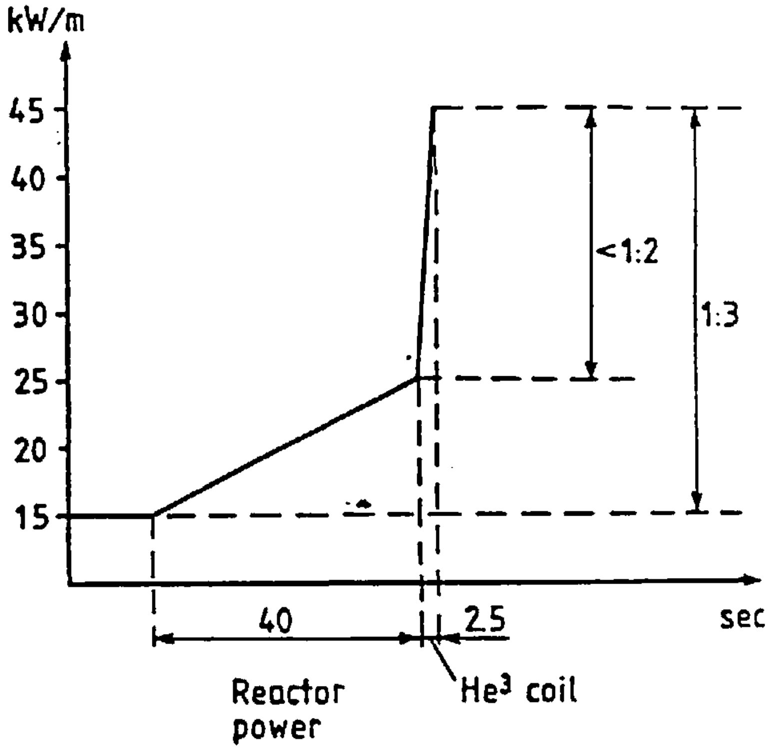


Figure 7
Schematic Rod Power vs. Time Diagram for Double Step Up-Ramping.

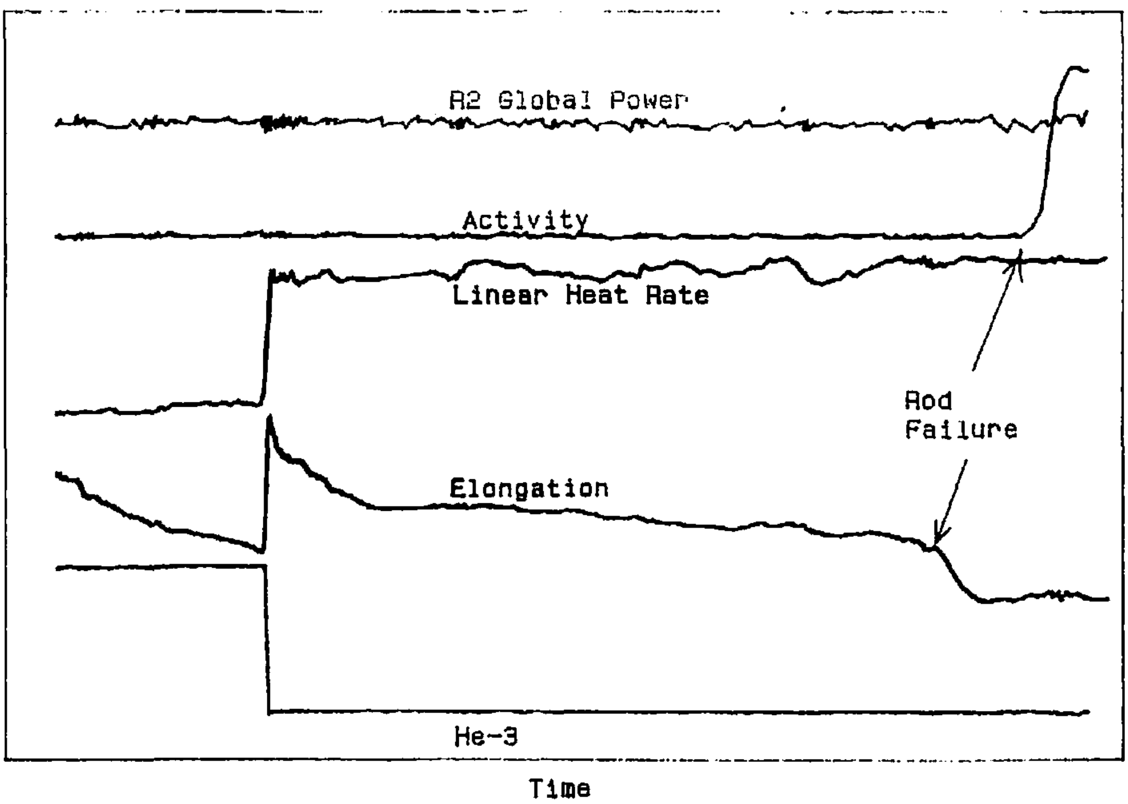


Figure 8
On-line Measurements During a Ramp Test Showing a PCI Failure Event.

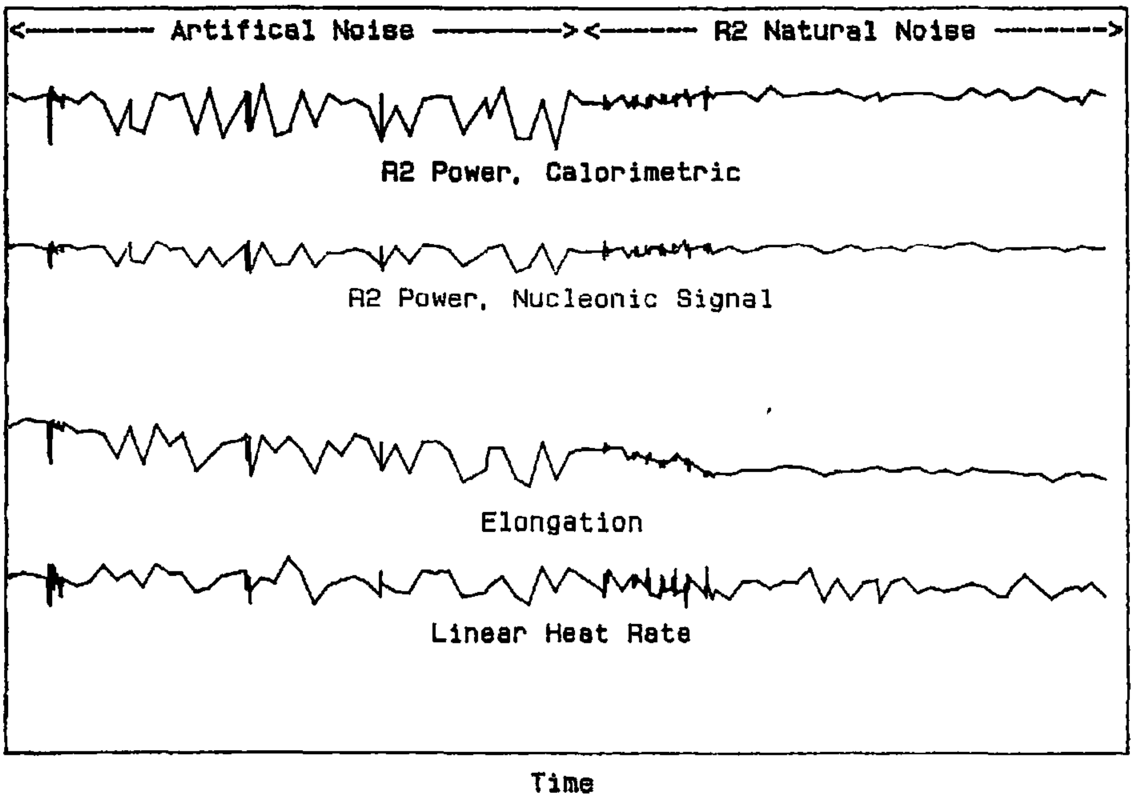


Figure 9
Comparison of Signals from Random Noise and Background Noise

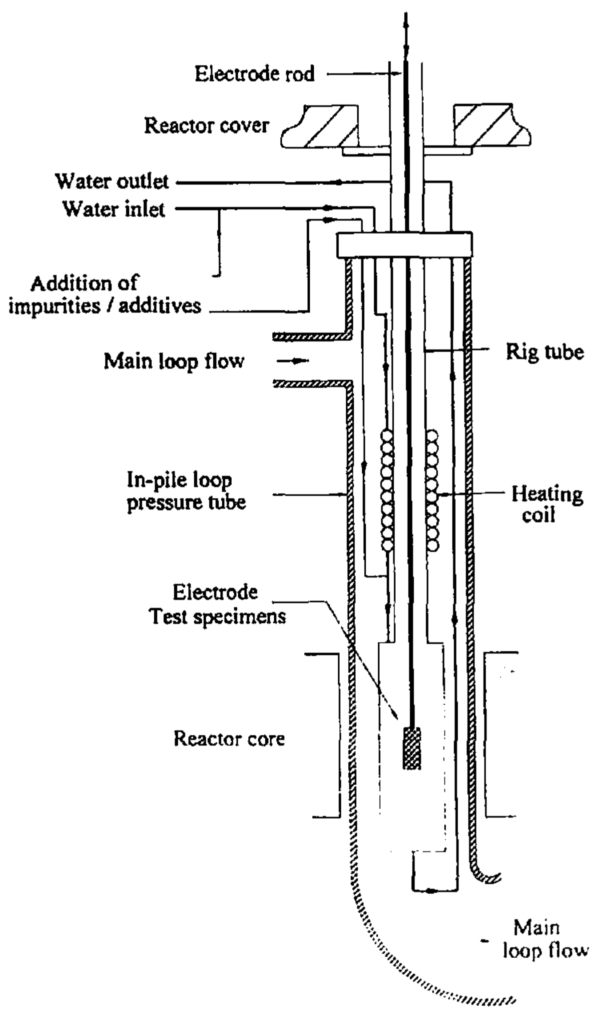


Figure 10
INCA Test Rig

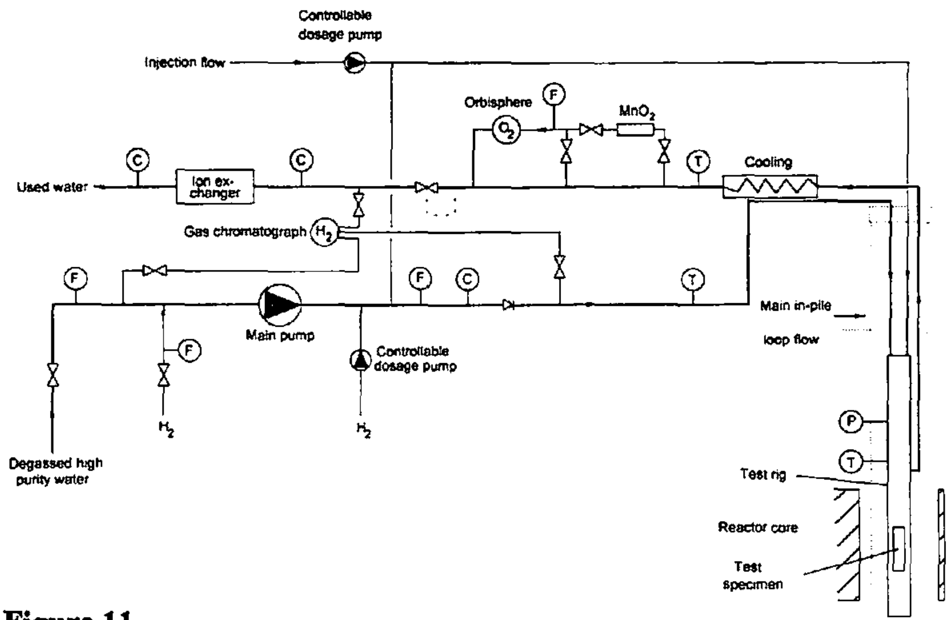


Figure 11
INCA Feed Water System and Analysis System

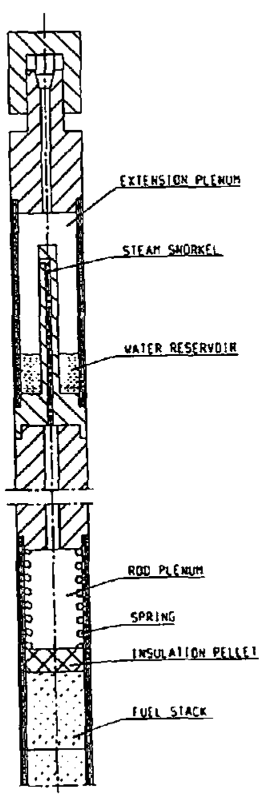
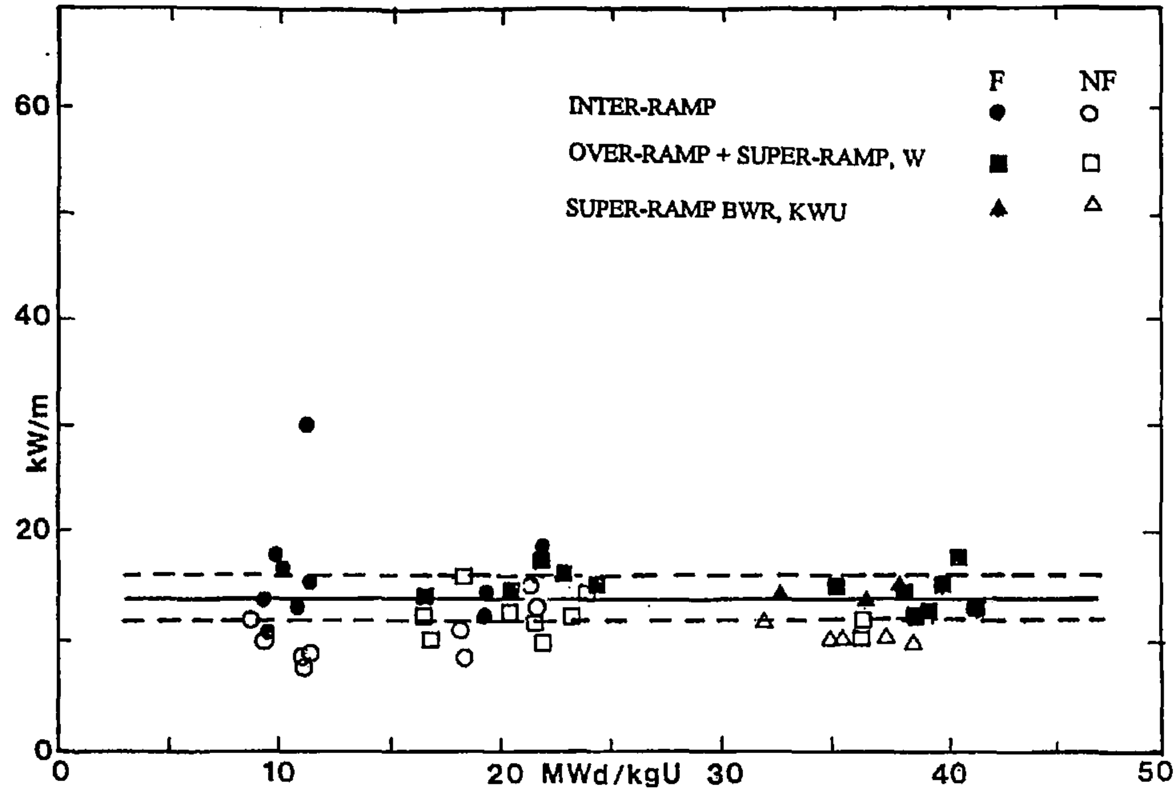


Figure 12
Array Simulating Primary Defects in Fuel Rods (Patent Pending).



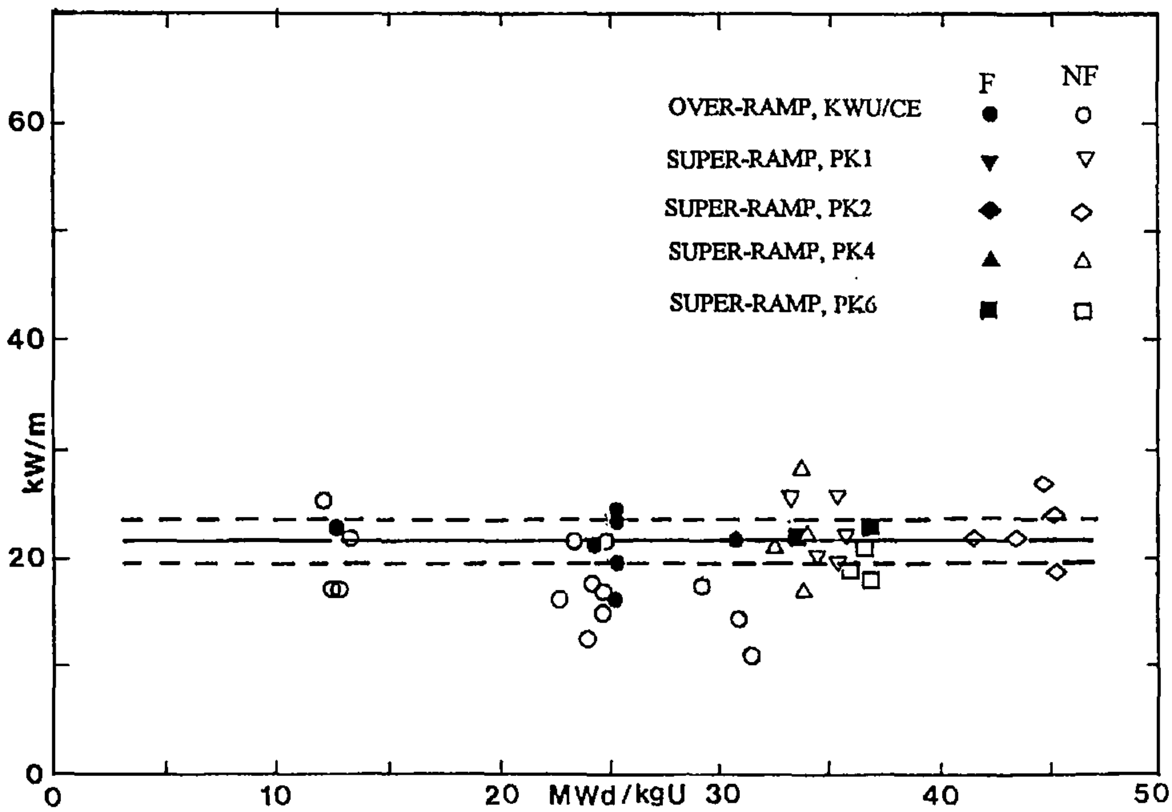


Figure 13
Summary of Some Data From the INTER-RAMP, OVER-RAMP and SUPER-RAMP Projects. The Incremental Failure Threshold as a Function of Burn-Up for Different Groups of Fuel Rods.

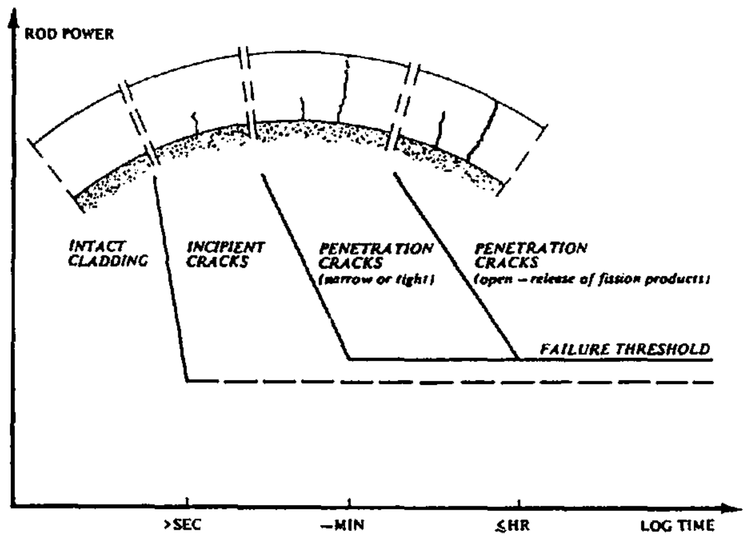


Figure 14
Schematic PCI Failure Progression Diagram.

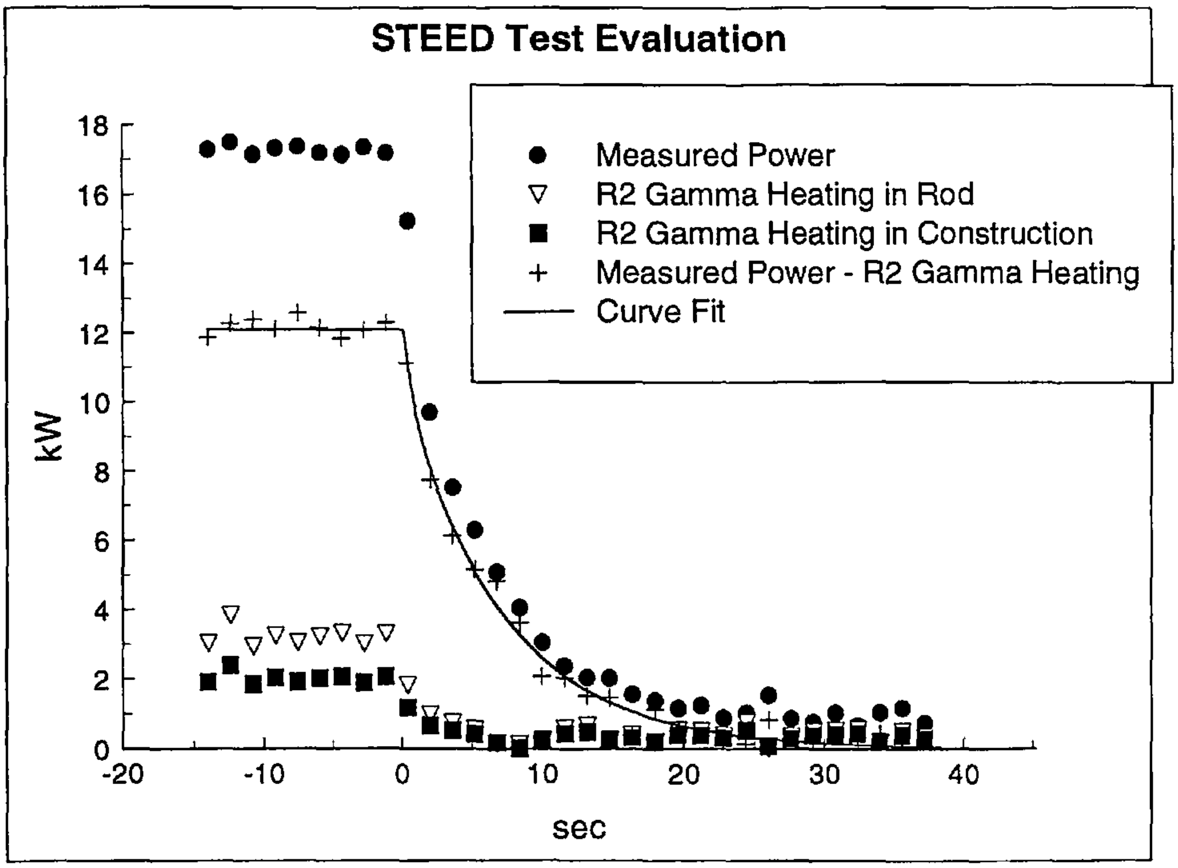


Figure 15
STEED Project - Example of Stored Energy Evaluation.
(the top curve and the two bottom curves are measured,
the second one from the top is evaluated from the others).
The stored energy is obtained from the area under the derived curve.

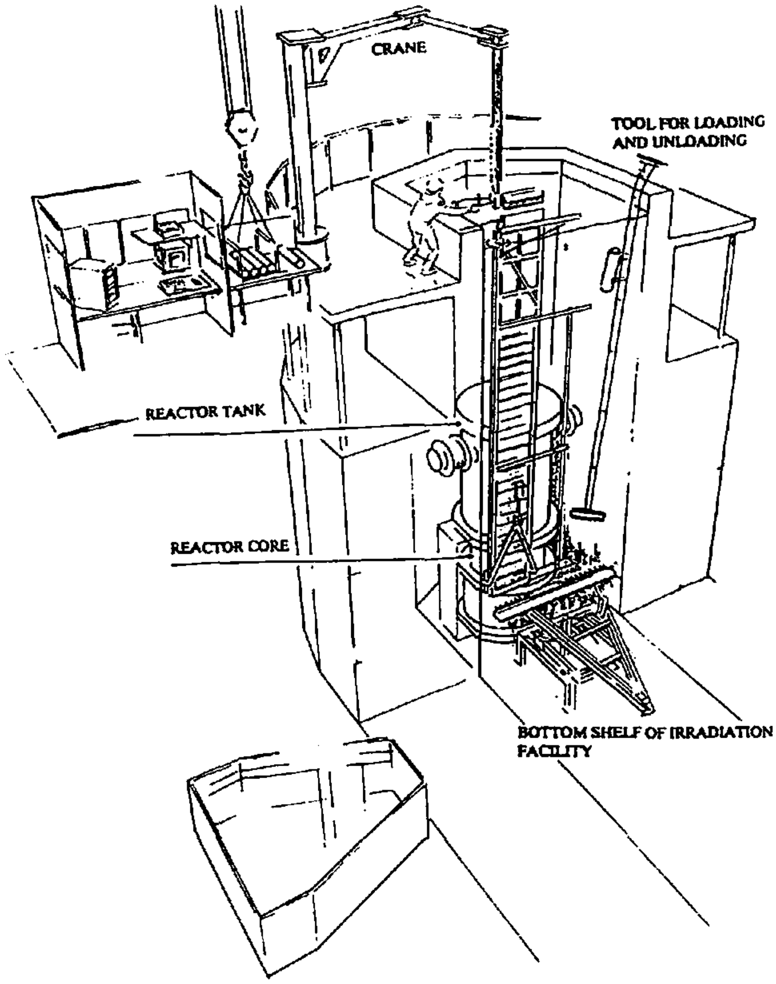


Figure 16
Silicon Irradiation Facility.

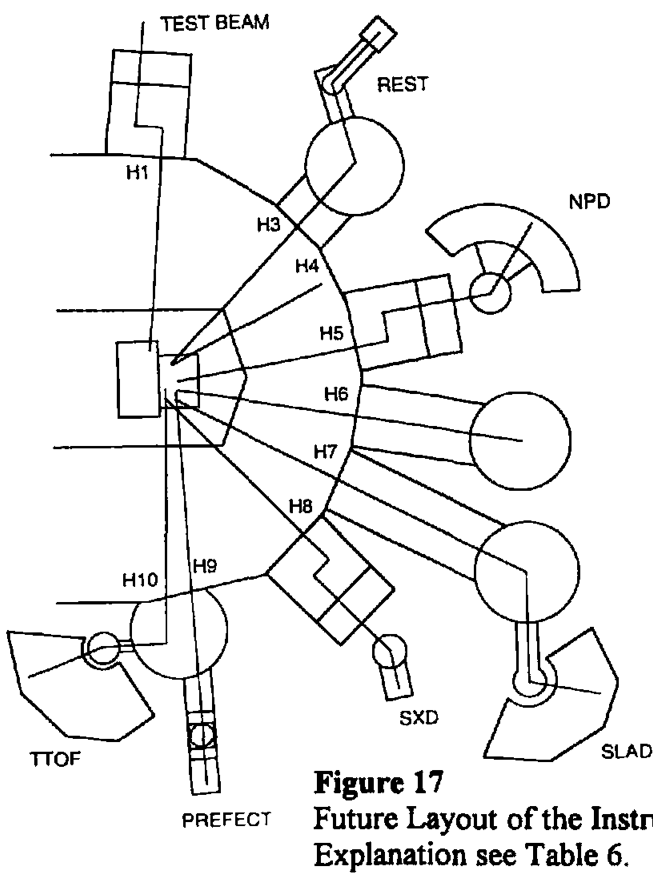


Figure 17
Future Layout of the Instruments at the R2 Test Reactor. Explanation see Table 6.

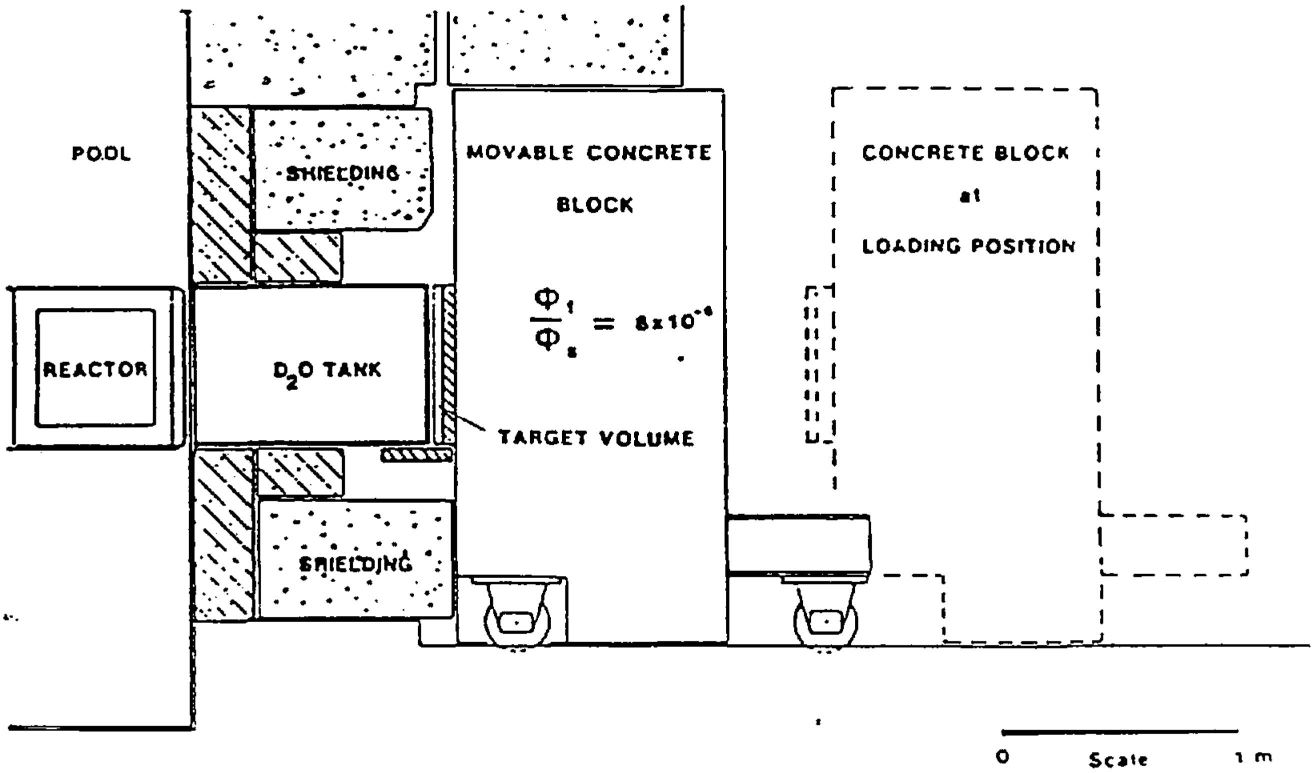


Figure 18
The R2-0 Facility for Neutron Capture Radiography.

MAIN EXPERIENCES IN RENOVATION OF THE DALAT NUCLEAR RESEARCH REACTOR

Tran Ha Anh, Pham Van Lam, Nguyen Nhi Dien and Ngo Phu Khang

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ABSTRACT

The Dalat Nuclear Research Reactor (DNRR) is a pool type research reactor which was reconstructed in 1982 from the old 250 kW TRIGA-MARK II reactor. Some structures of the former reactor such as the reactor aluminum tank, the graphite reflector, the thermal column, the horizontal beam tubes and the radiation concrete shielding were retained. The reactor core, the control and instrumentation system, the primary and secondary cooling systems as well as other associated systems were newly designed and installed. The renovated reactor reached its initial criticality in November 1983 and attained its nominal power of 500 kW in February 1984. Since then DNRR has been operated safely. It is mainly used for research, isotope production, neutron activation analyses and training.

During the recent years, the reactor control and instrumentation system has been renovated due to ageing and obsolescence of its components. Reactor general inspection and refurbishment were carried out in order to ensure and strengthen reactor safe operation. Thanks to the renovation works, DNRR has been continuously operated and effectively utilized. This paper presents main experiences acquired in the renovation of the Dalat nuclear research reactor.

1. INTRODUCTION

The Dalat Nuclear Research Reactor was reconstructed in 1982 from the old 250 kW TRIGA-MARK II reactor. The latter was put into operation in 1963. In 1975 all the fuel elements were removed from the core and transferred elsewhere. During 1982-1983, the reactor was reconstructed. Some structures of the former reactor such as the reactor aluminum tank, the graphite reflector, the thermal column, the horizontal beam tubes and the radiation concrete shielding were retained. The reactor core, the control and instrumentation system, the primary and secondary cooling systems as well as other associated systems were newly designed and installed. The natural convection mechanism of light water for reactor core cooling was kept unchanged. The core is loaded with VVR-M2 fuel elements with 36% enrichment. The renovated reactor reached initial criticality in November 1983 and attained its nominal power of 500 kW in February 1984. The maximum thermal neutron flux was 2.1E13 n/cm^2/sec. Since then DNRR has been operated safely. It is mainly used for research, isotope production, neutron activation analyses and training.

The configuration of active core was changed several times, improving the experimental condition of the reactor. During the recent years, the reactor control and instrumentation system has been renovated due to ageing and obsolescence of its components. Reactor general inspection and refurbishment were carried out in order to ensure and strengthen reactor safe operation. Thanks to the renovation works, DNRR has been continuously operated and effectively utilized. This paper presents main experiences acquired in the renovation of the Dalat nuclear research reactor.

2. RENOVATION OF THE DALAT NUCLEAR RESEARCH REACTOR [1-2]

2.1 Aim of the renovation

The main aim of the renovation was to redesign the reactor in keeping its old geometrical structure with an utterly different kind of fuel material, whereas requesting higher technical features such as doubling the power to 500 kW but conserving the natural convection mechanism for reactor core cooling. The most significant feature is obtaining as high neutron fluxes as possible for the given power, thus allowing the most effective use of the reactor.

2.2 Renovation works

Reactor reconstruction was carried out during 19 month period. During reconstruction, some structures of the former reactor such as the aluminum reactor tank, the graphite reflector, the thermal column, the horizontal beam tubes and the radiation concrete shielding were retained.

The rather high level of radiation dose rate inside the old reactor tank had been unfavorable for the assembling process of the reactor core structures. Thus the solution of an entirely suspended structure was adopted for the reactor core. A cylindrical chemise is installed inside the aluminum tank. It has 1.90 m diameter and 3.56 m height and is hung from the top of the reactor to serve as a suspender of the core structures. The upper part of the suspender has some structures to support the control rod, the neutron detector or temperature sensor tubes, a tube for loading samples into the rotary specimen rack, a channel for transferring spent fuel bundle. At the bottom part of the suspender is a supporting base. The base has a height of 20 cm and is used for suspending the extracting well and the core.

The fuel elements use 36% enriched U235 fuel made of uranium - aluminum alloy. Each element is laid out in three concentric layers to increase the heat exchange superficies. Fuel elements are of the VVR-M2 bundle type and composed of 3 concentric layers : 2 circular inner tubes and a hexagonal outer tube. The fuel layer with a thickness of 0.7 mm is wrapped between two aluminum alloy cladding layers of 0.9 mm thickness.

The core is located in distilled water at a depth of 5 m. Water is used as moderator, coolant, reflector and biological protection.

The neutron trap is situated at the center of the core. It is made of beryllium and has a diameter of 65 mm. The region between the active core and the old graphite reflector is also lined with beryllium.

One of the most important technological systems of the reactor is the control and instrumentation system, which controls the reactor during normal operation and scrams it if a threat of accident appears. For determining the neutron flux, the reactor control system uses 9 detectors placed in dry channels at different heights and positions at the outside of the graphite reflector. The detector measure neutron flux in three ranges : source, intermediate and energy ranges. Seven control rods of the reactor are located symmetrically in the core : two safety rods, four shim rods and one automatic regulating rod. The safety and shim rods are made of boron carbide and the automatic regulating rod of stainless steel. These rods are driven by a quite complex system. During reactor operation all important information from the rector technological systems is received and processed by the control and instrumentation system and finally displayed on the control console.

In other to cope with higher power, the cooling efficiency was increased by the installation of an extracting well placed above the core. The cooling system of the reactor has two loops : a primary loop between the reactor tank and the heat exchanger, and a secondary loop to extract heat from the heat exchanger and release it into the atmosphere via a cooling tower. The cooling system were improved, the heat exchanger was replaced to increase the heat transfer areas, the cooling tower of the secondary cooling system was also improved to increase the cooling capacity, so finally to raise the reactor power.

To ensure radiation protection for the staff working inside the reactor building, a layer of heavy concrete was added to the lateral shielding of the reactor at intermediate level to protect against gamma radiation from the reactor core, and also from strengthened upward water flow bearing activated elements produced in the reactor core. The cover of the reactor tank is made of a steel plate with thickness of 15 cm. It is used for radiation protection during reactor operation.

The former bulk shielding has been modified into a storage tank for spent fuel elements. Depth of the tank is 3.7 m. The tank has covers and is filled with distilled water. It has 300 holes to contain spent fuel elements. A description of reactor components, section view is shown in figure 1.

The reactor building is equipped with ventilation systems, which ensure good working conditions and allow safe gas release to the atmosphere after being purified via 40 metre-height stack.

The first working configuration of the core was obtained on February 1984 with the 72 hr successful test operation of the reactor. The configuration consisted of 88 fuel elements with neutron trap in the center of the core and one wet channel in position 7-1.

The irradiation facilities in this core comprised of 2 vertical pneumatic irradiation channels together with the central neutron trap and the wet irradiation channel. Surrounding the reactor core, a rotating tray containing 40 irradiation holes was arranged at the same position as the former "Lazy Suzan". The 4 horizontal neutron beam ports and the thermal column were retained from the old reactor.

Refueling operations were carried out in April 1994, which consisted of adding 11 new fuel elements in the core periphery at previous beryllium element locations. This fuel reloading ensured the reactor to be exploited for 3 - 4 years before another refueling is needed. The working configuration of reactor core with 100 fuel elements is shown in figure 2.

The introduction of a steel cover for the reactor tank, the suspended structure of the reactor core, the use of a reactor control system made of 7 control rods as well as the additional irradiation facilities inside the reactor core make the reactor's internal structure rather complicated in comparison with the former reactor. Handling operations in the reactor core become more subtle, and so do the operations aimed at inspecting periodically the reactor tank and other internal structures. These operations are even not practicable in some areas inside the reactor tank.

This is a problem of great concern, because structures kept from the old the reactor are now more than 30 years old and their periodical inspection must be performed as rigorously as possible in order to detect any unacceptable defect if exists, and thus to prevent any incident leading to the loss of integrity of the second barrier.

3. RENOVATION OF CONTROL AND INSTRUMENTATION SYSTEM [3-4]

3.1 Aim of the renovation

After ten years of reactor operation, the system has revealed some features of ageing and obsolescence. As its components and equipment were produced in the 70's, some integrated circuits (IC) and equipment were hard to find out on the market and as a consequence maintenance work of the system became difficult. To solve this problem the Dalat Nuclear Research Institute has asked for the assistance of the IAEA through the TC Project VIE/4/010 - "Renovation of the Dalat Reactor Control and Instrumentation System".

The design obeyed some principles such as keeping unchanged the old mechanical standard and the technical specification of the old boards, but making them more reliable.

Some electronic boards/blocks/sub-systems which mainly affect the system reliability will be renovated by using more modern IC and equipment.

3.2 Renovation works

The instrumentation and control system of the Dalat research reactor can be divided into four main sub-systems as follows :

* Neutron Flux Control sub-System - NFCS
* Control Console Display sub-System - CCDS
* Control Logic sub-System - CLS
* Process and Instrumentation sub-System - PIS

Some important boards of the NFCS were renovated.

Because of low reliability, low quality of indication, the reactor control console display system was totally designed and constructed according to the project program.

All electronic boards of the system were designed on the EUROCARD standard. The software of the system was written in Pascal language. Data are saved in hard disk 120 MB or in floppy-disk 1.2 MB.

To increase the accuracy and the reliability of rod position indication, the indicators and related electronic board have been replaced. The indicators are of a digit and bar-graph type - INDICOMP A2000.

All relays in intermediate relay board and transistors in the amplifier boards of control logic block for safety rods have been replaced by items with higher quality and reliability. Decreasing currents fed to LED indication and consequently decrease in power consumption in all electronic boards has an important meaning in the reliability improvement of the CLS.

Intermediate relays, power amplifier boards of control logic block for compensating rods have been renovated or replaced.

Also, intermediate relays and power amplifiers of the AR control logic block have been renovated or replaced.

The PIS has been totally renovated in the frame of the project VIE/4/010.

The renovated instrumentation and control system of the Dalat reactor has been commissioned and put into operation since November 1993. The system has been working with high reliability during the three years of its operation.

4. CONCLUSION

After renovation, the maximum power of DNRR was increased to 500 kW. The maximum thermal neutron flux was 2.1E13 n/cm^2/sec. Some structures of the former reactor were retained. The technological systems of the reactor were newly designed and installed. The natural convection mechanism of light water for reactor core cooling was kept unchanged. The renovated reactor attained its planned nominal power in February 1984. Since then DNRR has been operated safely.

During the recent years, the reactor control and instrumentation system has been renovated due to ageing and obsolescence of its components. Thanks to the renovation works, the DNRR has been continuously operated and effectively utilized.

REFERENCE

[1] TRAN HA ANH et al., Main Safety Lessons from 5-year Operation of the Renovated Dalat Nuclear Research Reactor, IAEA/AECL Research Reactor Symposium, Chalk River, Oct. 1989. IAEA-SM-310/78

[2] VU HAI LONG et al, Dalat Nuclear Research Reactor - Safety Analysis Report, Dalat, 1989

[3] TRAN HA ANH et al., Eight-year Operation of the Dalat Nuclear Research Reactor and the Necessity of Renovation of Its Control System, IAEA Regional Nuclear Research Seminar, Bangkok, May 1992. IAEA-SR-179/15C

[4] NGUYEN NHI DIEN, The Dalat Reactor Instrumentation and Control System after Renovation, IAEA technical Committee Metting on Technology and Trends for Research Reactor Instrumentation and Controls, Ljubljana, December 1995

Table 1. Main characteristics of the Dalat Nuclear Research Reactor

Parameter	Nature/Value
Reactor type	Swimming pool
Power, kW	500
Fuel	U-Al alloy
Critical mass, g U-235	2781
Number of fuel elements	100
Uranium loading, g	4015
Enrichment, % U-235	36
Moderator and coolant	Light water
Core cooling mechanism	Natural convection
Excess reactivity, $	6.5
Xenon poisoning effect, $	-1.65
Temperature and power effect, $	-0.36
Control rod worth, $:	2.97
- Shim rod No. 1	3.09
- Shim rod No. 2	2.70
- Shim rod No. 3	2.50
- Shim rod No. 4	0.50
- Regulating rod	5.36
- Two safety rods	
Thermal neutron flux, $n/cm^2/sec$:	
- Neutron trap	2.2E13
- Wet channel 1-4	1.28E13
- Dry channel 7-1	5.1E12
- Dry channel 13-2	4.2E12
- Rotary specimen rack	4.27E12
- Thermal column	5.5E9
- Horizontal beam tube No. 3	2.5E6
- Horizontal beam tube No. 4	5.8E7
Maximum surface temperature of fuel, ^0C	97.2
Maximum water temperature in the core, ^0C	54.5

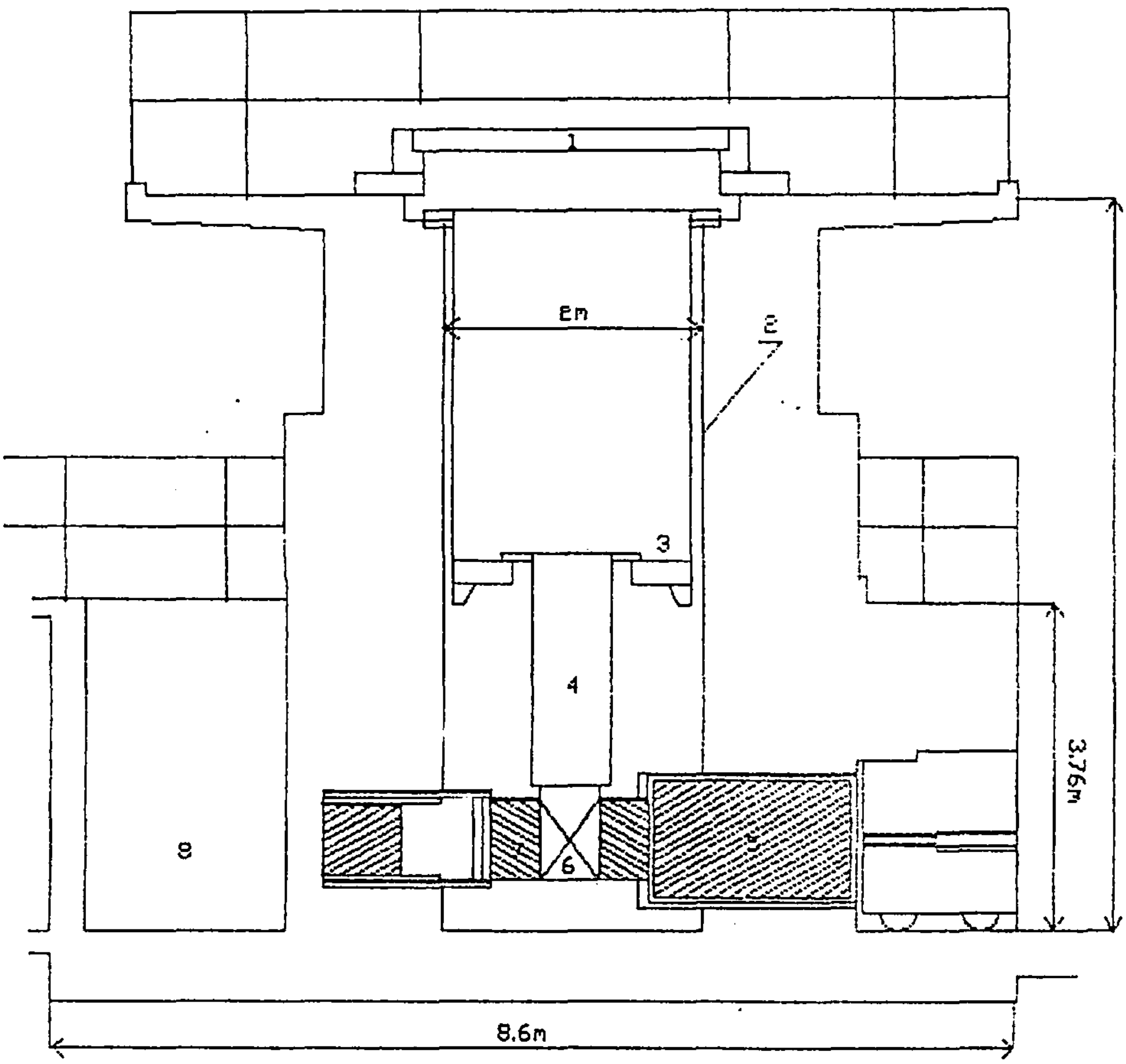

Fig. 1 DESCRIPTION OF REACTOR COMPONENTS, SECTION VIEW
1-COVER, 2-TANK, 3-SUSPENDING SUPPORT, 4-EXTRACTING WELL
5-THERMAL COLUMN, 6-CORE, 7-REFLECTOR, 8-SPENT FUEL STORAGE

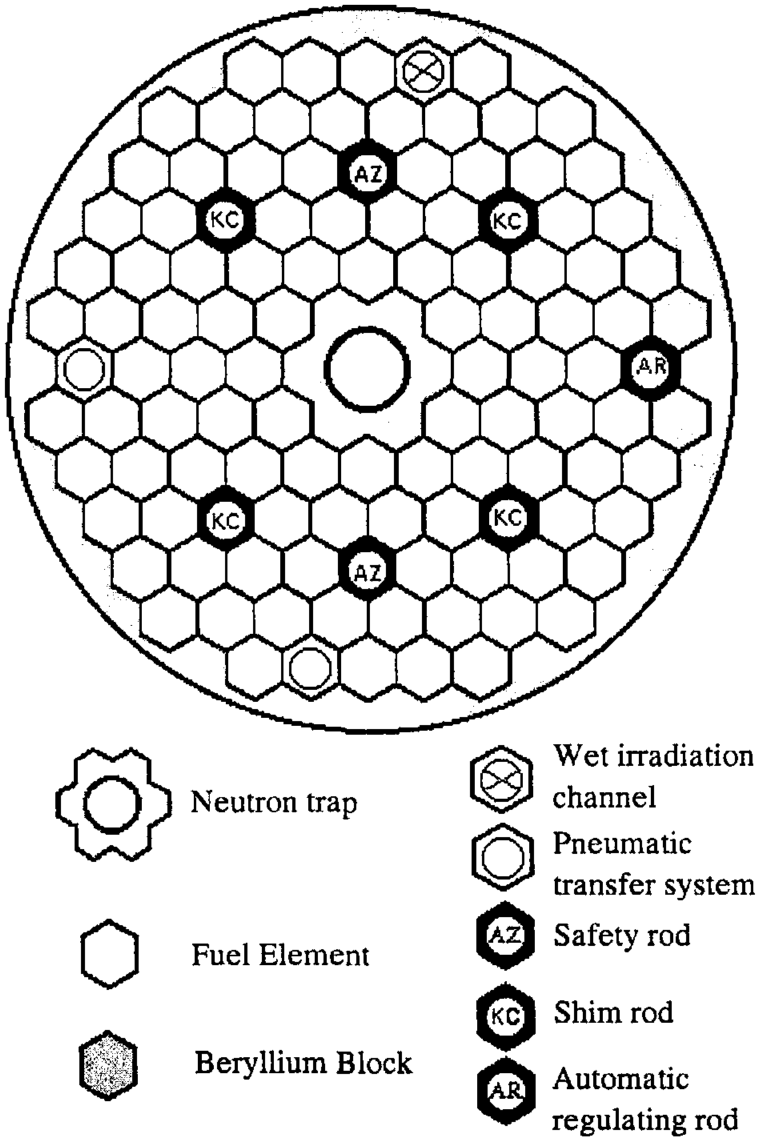

Fig. 2 WORKING CONFIGURATION OF REACTOR CORE

SCK•CEN

Ongoing Refurbishment and Future Utilization of BR2

E. Koonen
1996-09-09

SCK•CEN

BR2 beryllium matrix

- The Be-matrix is the life-limiting component of BR2
- Phenomena :
  - swelling ----------> cracking
  - ^{3}He poisoning ----------> operational constraints

$$^{9}\mathrm{Be}\,(n,\alpha)\,^{6}\mathrm{He} \xrightarrow{\beta^-} {}^{6}\mathrm{Li}\,(n_{th},\alpha)\,^{3}\mathrm{H}$$

$$\xrightarrow{\beta^-} {}^{3}\mathrm{H} \longrightarrow {}^{3}\mathrm{He}\,(n_{th}, {}^{1}\mathrm{H})$$

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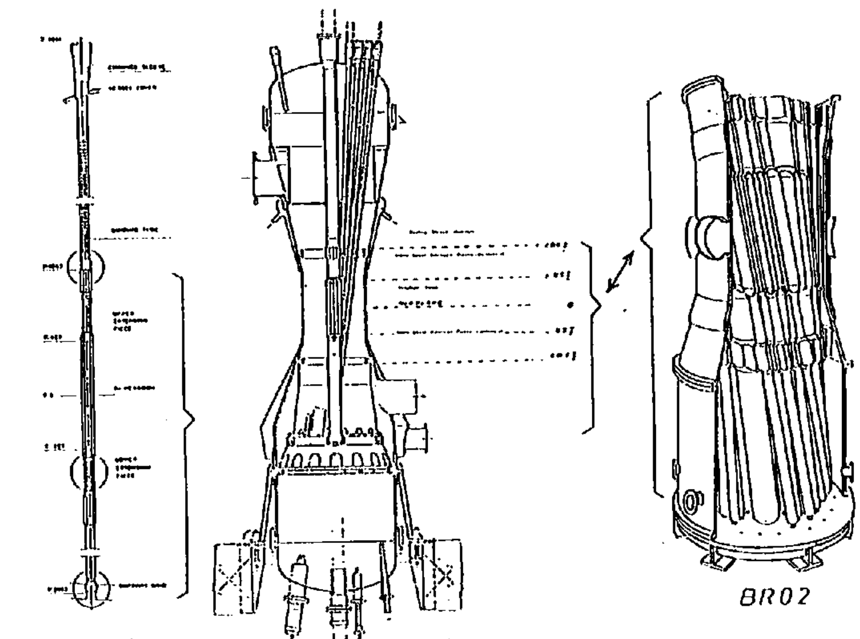
BR2 refurbishment programme

- **BR2**
  - = essential research tool for CEN·SCK
  - = optimized design for MTR-use and safety experiments

  ==> Refurbishment = **PLEX**
- Requirements :
  - replacement of Be matrix
  - new safety case in compliance with modern standards
  - availability and reliability of installation
- Future use will be focused on Engineering R&D



Standard channel

BR 2
FIG. 1

BR02

SCK•CEN Major Refurbishment Issues

Safety assessments

- Vessel integrity
- PSA -> LOCA assessments
- Instrumentation and support systems
- Reliability isolation systems
- Structural and seismic requalifications
- Industrial safety

Operational safety

- Lessons learned from past operation and incidents
- Ergonomics
- Operator training and requalification

SCK•CEN BR2 aluminium tank

Issue : prospects for life extension

- evaluation of consequences of vessel rupture
- determination of fluences distributions
- evaluation of life-limiting phenomena :
  - fracture toughness
  - low cycle fatigue
- In-Service Inspection

SCK•CEN Fluence distribution in vessel wall

Methods: 1. Calculations

2. Samples from first Be-matrix

3. Samples from shroud surrounding the vessel

Dosimetry:

- thermal fluence : $^{27}Al(n,\gamma)\ ^{28}Al \xrightarrow{\beta^-} {}^{28}Si$

  $^{9}Be(n,\gamma)\ ^{10}Be$

- fast fluence : $^{9}Be(n,2n)\ ^{8}Be \longrightarrow 2\ ^{4}He$

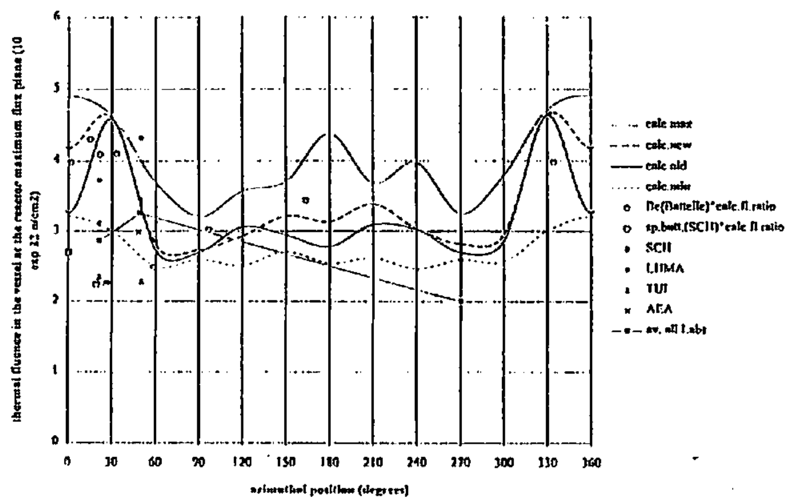


FIG. 2 Azimuthal distrib. of the thermal fluence (nv_0t) in the BR2 vessel (at the max. flux plane) at mid-1995 as calculated, as deduced from measurements on the first Be matrix and as deduced from the shroud measurements

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Fatigue analysis of Al tank

Areas of concern :

- central cylindrical shell
- inlet & outlet nozzles

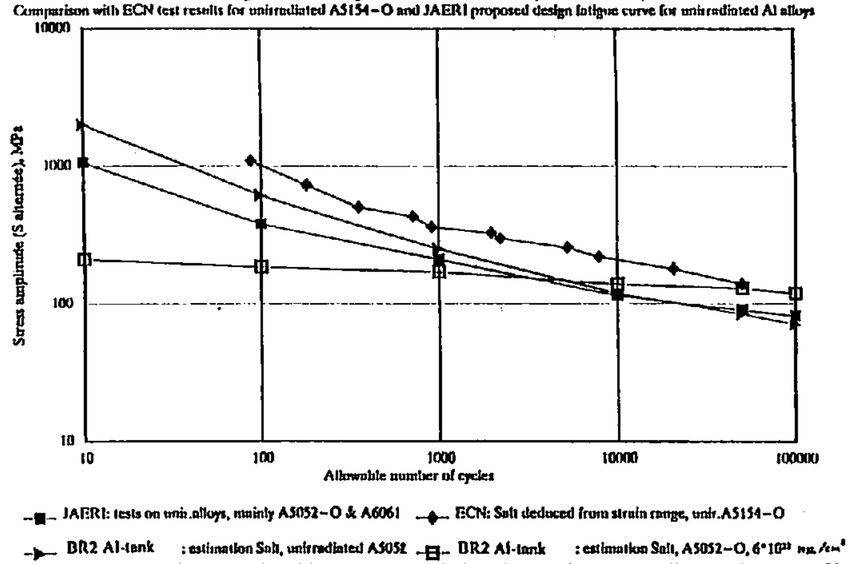
Methodology :

- ASME VIII, div. 2, class 1 components
- Fatigue curves for irradiated material theoretically established

Conclusions :

the damage factors for the contemplated life extension are acceptable provided the ISI doesn't reveal unacceptable flaws

Fig. 5 Modelled fatigue life curve for A5052-O (unir.& irradiated)
Comparison with ECN test results for unirradiated A5154-O and JAERI proposed design fatigue curve for unirradiated Al alloys



ECN-tests: triangular push-pull cyclic straining with constant total axial strain ranges, Internal report ECN-CX-91-049, July 91
JAERI: Proposed design fatigue curve for Al alloys, Nuclear Engineering and Design, volume 155 (1995) No. 3 pp.547-557

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Fracture Toughness (1)

Objectives :

- determination of static initiation fracture toughness of BR2 vessel weld and base metals
- extrapolation for the contemplated life extension period -> trend-curves

Materials :

- BR2 shroud material, obtained by electro-erosion cutting
- BR2 initial control rod guide tubes
- remnants from first HFR-Petten vessel

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Fracture Toughness (2)

Experimental programme :

- uniaxial tensile tests
- three point slow bend precracked Charpy
- Charpy V-notch impact tests

\+

- microstructural characterisation
- chemical analysis

Two successive modelling steps :

1. dislocation theory guided by transmission electron microscopy
2. two complementary micromechanical models: relate KIc to the flow properties

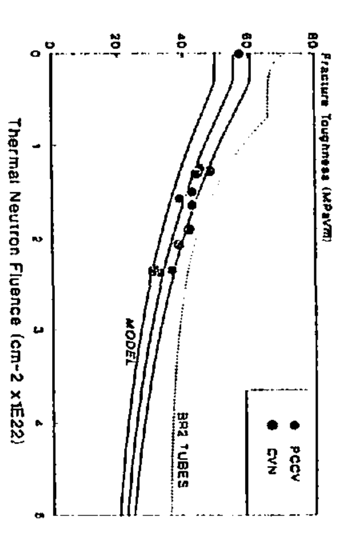

Fig. 4.a Embrittlement Trend Curve for BR2 Aluminium Base Metal

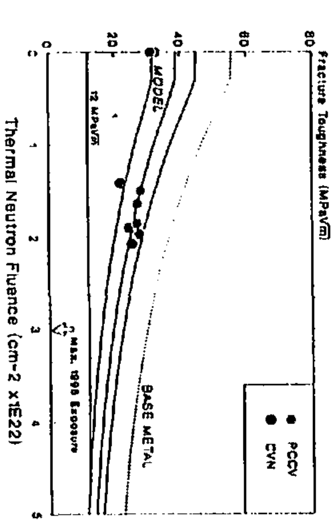

Fig. 4.b Embrittlement Trend Curve for BR2 Aluminium Weld

SCK•CEN Fracture Toughness - Conclusions

- The weld is controlling in terms of vessel integrity concerns
- The lower bound fracture toughness in the thermal fluence range of 4 - 4,5 E22 n/cm² is projected to exceed

  $12\ MPa\sqrt{m}$

- No significant future operation risk if inspection of the vessel does not reveal any surface-breaking crack of depth exceeding 3 mm, nor any embedded flaw causing a local KI in excess of

  $6\ MPa\sqrt{m}$

  2 methods, finite elements and the R6-method, have given similar results for the stress intensity factor associated with an infinite axial crack of 5mm depth: about 7,43 MPa$\sqrt{m}$

SCK•CEN In-service inspection of Al-tank

Methods:

- direct visual inspection
- remote visual inspection by means of cameras
- remote mechanized geometrical control
- penetrant testing
- ultrasonic testing
- Eddy current testing

Results:

- no flaw with unacceptable dimensions
- the Al-tank has not undergone any significant alteration since the 1980 inspections

SCK•CEN PSA results

Major concern : Large LOCA outside C-B
-> possibility of containment by-pass

Analysis :

- RELAP model of BR2 primary circuit

Actions :

- reengineering of control & actuation systems of major automatic block valves
- reduction of closing times of those valves
- enhanced capability of pool-communication valve
- new fast reactor-trip on loss of pressure

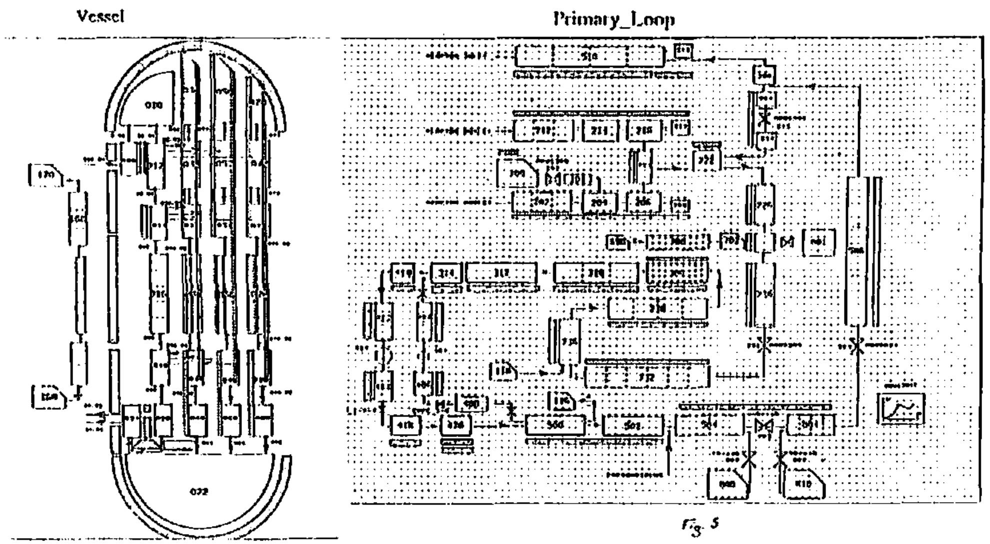

Fig. 5

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BR2 instrumentation & control

Instrumentation :

- nuclear : fault-tree analysis of trip-lines
  -> minor modifications to eliminate single and common mode failures
- process : FMEA & common mode analysis
  -> renewal of all safety related instrumentation (redundancy, diversity and testability)

Support systems : FMEA & common mode analysis

- 110 V dc : decentralisation & separate batteries
- 220 V ac - vital net : increase of reliability
- compressed air : renewal of compressors and redesign of distribution network

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BR2 seismic evaluation

Reference earthquakes :

- historical records most significant earthquakes
  max. intensity hypocentre: VII MSK
- safe shutdown earthquake:
  horizontal PGA = 0.1 g
  vertical PGA = 2/3 * 0.1 g

Methodology : ASCE4-86 standard

Calculated structures :
Reactor building, pool, vessel, isolation systems, damage by fall of crane/heavy structures, fuel storage pool, primary pipework outside C-B

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Operational safety

- Ergonomic study -> new reactor control desk
- Emergency control panel outside C-B
- Data acquisition system, integrating reactor control, experiments and radiation-protection
- Lessons learned from past experience and incidents -> action plan
- Accident management procedures
- Operator training and requalification programme

SCK•CEN — Operation licence

- needs only to be prolonged, not renewed
- procedure : decennial safety reassessment
  (same procedure as Power Stations)
- basic documents :
  updated SAR, including results from studies, inspections, improvements and upgradings of refurbishment programme
- regulatory body is represented in the refurbishment review committee

SCK•CEN — Future utilization of BR2

Base load : **internal R&D programmes**

- PWR fuel (MOX) :
  thermal conductivity, fission gas release, PCMI
- Pressure Vessel Steel Embrittlement
  RPV surveillance, modelling efforts
- PWR structural components
  embrittlement, IASCC

SCK•CEN — Standard irradiations devices at BR2

- PWR-loop **CALLISTO**
  - 3 IPS in standard channels
  - up to 9 fuel rods in each IPS
  - fresh and pre-irradiated fuel
  - realistic PWR conditions
- **PWC** capsules associated with ^{3}He-screens
  single-rod transient tests
- Reflector rigs : steel and structural materials
- Gamma irradiation facilities

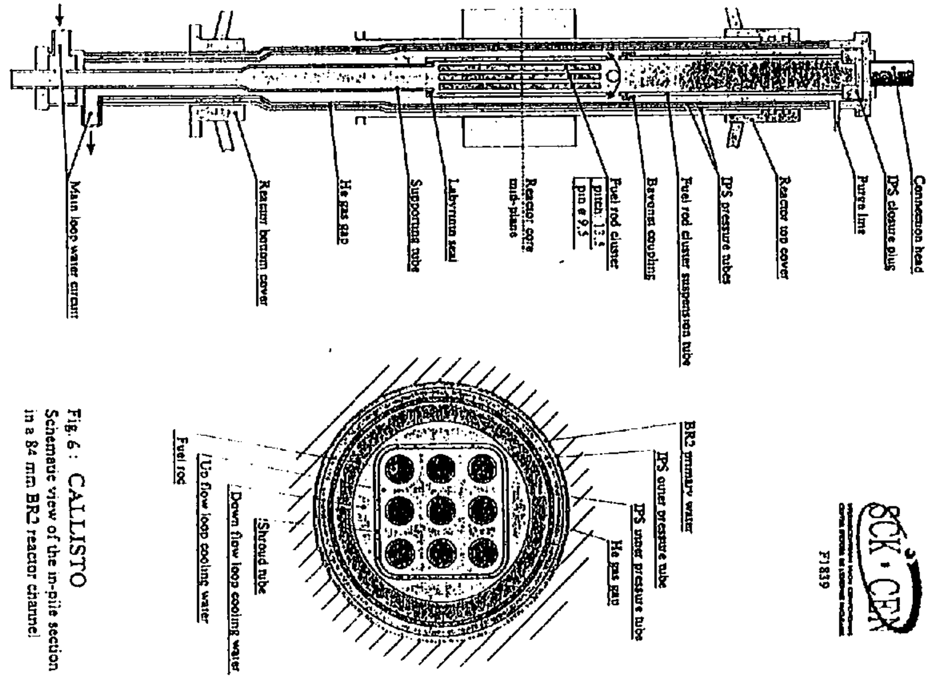

Fig. 6 : CALLISTO
Schematic view of the in-pile section in a 84 mm BR2 reactor channel

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New irradiation devices for BR2

- Pool Side Facility : **PSF**
  - for vessel-steel & remote handling programmes
  - space available by removal of 2 tangential beam-tubes
- Feasibility studies :
  - ► PWR-type integrated loop : **DESTIN**
    - in central 200 mm flux trap
    - up to 25 fuel rods (fresh or pre-irradiated)
    - steady-state and transient tests
  - ► solid breeder blanket test-module : **CEPHEID**
    - in central or peripheral 200 mm channel
    - integrated loop

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Special capabilities of BR2

BR2 is designed to accept complex experiments and large loops, as required for severe-accident simulations

- Power transients :
  - ramping up to fuel pin failure, with possibility for post-transient irradiation
  - non-adiabatic RIA tests, e.g. control-rod withdrawal
- Core blockage formation
- Core debris coolability

An international context is required for these programmes

SCK•CEN

Production activities in BR2

- **Radioisotopes**
  - Mo-99, I-131, Xe-133 (PRF and DGR devices)
  - Ir-192 (up to 700 Ci/g)
  - Co-60 (300 Ci/g)
  - Sr-89, Re-186, Sm-153, Eu-154 ...
- **NTD - Si**
  - SIDONIE irradiation device
  - mainly 5 inch ingots or wafers

SCK•CEN

Conclusions

- Refurbishment underway => ready early '97
- Future utilization :
  mainly Engineering R&D
- Base load : internal programmes
- BR2 available for :
  - ► scientific collaborations
  - ► irradiations on request

RESEARCH AND SERVICES OF LVR-15 REACTOR IN ŘEŽ

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ABSTRACT

Main features and characteristics of the LVR-15 research reactor are presented. Reactor is used for different purposes including material testing, radioisotope production, silicon doping, neutron beam channels and neutron capture therapy. Material testing in realised at rigs and loops which involves PWR or BWR chemistry conditions. Main radioisotopes produced at LVR-15 is for medicine. The two in-pile section is mounted in a reflector for doping of silicon where 3"crystal are irradiated. Thermal column of the reactor is used for the intensive study and development of neutron capture therapy. At the reactor nine horizontal beam channels are installed, five of them is utilised for material structure research by neutron diffraction methods.

DESCRIPTION OF REACTOR LVR-15

The LVR-15 is a light-water moderated and cooled tank nuclear reactor with forced cooling. The maximum reactor power is 10 MWth. The fuel type IRT-2M enriched to 80 % or 36 % and a combined water-beryllium reflector are used.

REACTOR CHARACTERISTICS

maximum reactor power	10 MWth
maximum thermal neutron flux in the core	$1{,}5 \times 10^{18}$ n/m^{2}s
maximum fast neutron flux in the core	3×10^{18} n/m^{2}s
at the end of beam tube	1×10^{13} n/m^{2}s
irradiation channel in fuel	1×10^{18} n/m^{2}s
irradiation channel in reflector	$3\text{-}5 \times 10^{17}$ n/m^{2}s

The reactor core is composed of IRT-2M type fuel assemblies produced in Russia with the enrichment of 80 % ^{235}U. This fuel has been changing for fuel with an enrichment of 36 % ^{235}U. The reactor core grid has a pitch of 71,5 mm and 80 cells. In the basis operation configuration, 28-34 cells contain fuel cells are dedicated to channels for experimental probes and 3-6 cells in the reflector and core periphery are dedicated to vertical irradiation channels (see Fig. 1).

The reactor core is situated in the reactor vessel (outer diameter 2 300 mm, total height of the vessel 6 235 mm), which is made of stainless steel, the internal parts of the reactor are made of an aluminium alloy. The reactor has a forced circulation of the coolant. The generated heat is transported via three cooling circuits to Vltava river (see Fig.2).

The primary circuit contains 5 main circulation pumps and 2 emergency pumps which provide the circulation of the coolant through the reactor core and 2 heat exchanges. The emergency pumps are connected with an emergency battery source and a diesel generator, which secure their operation when the electrical network in the reactor site fails. The coolant flow is 2100 m^3/hour. The coolant is a light demineralised water.

The protection and control system ensures

- Measuring of the neutron flux and of the rate of its change in all operational and extraordinary conditions
- Start up, operation, and shutdown of the reactor
- Control of the reactor with signalling of selected states
- Automatic and hand control of the neutron flux

The neutron flux is measured with 3 equal sets of measuring channels, realised by the electronic analog system SAKOR B. From the measuring channels, the measured values are transferred to the control panel, into the information system, and to the reactor recording devices. From excessive critical values of the neutron flux of the rate of the change of the neutron flux, the scrams are deduced in accident circuits (a choice „two out three").

Logic and accident circuits perform the emergency shut-down and control of the reactor in all operation conditions. The system is based on relays.
The LVR-15 reactor has 12 control rods of equal type. By selection of the rod function in the electric circuits and by location in the active core, 3 safety rods, 8 compensation rods as a automatic regulator are determined. The absorption part of the rod is made of stainless steel filled with B_1C.

FUEL

In the reactor LVR-15 IRT-2M type fuel assemblies are used in two modifications - 4 tubes assembly and 3 tubes assembly. The 3 tubes assembly is installed with the control rod in the centre of the fuel assembly.
Until the year 1995 the with enrichment of 80 % ^{235}U was used. From that year the use of the fuel with enrichment of 36 % ^{235}U been started.

The main characteristics of both types of the fuel IRT-2M are:

^{235}U enrichment		80 %		36 %
total length			882 mm	
section square - head				71,5x71,5 mm
section square			67x67 mm	
mass total of the assembly	- 4 tubes	3,27 kg		3,2 kg
	- 3 tubes	2,64 kg		2,6 kg
mass of ^{235}U	- 4 tubes	171 g		230 g
	- 3 tubes	147 g		198 g
tube wall thickness			2 mm	
cladding material			Al	
cladding thickness		2x0,8 mm		2xmin. 0,4 mm
fuel material		U-Al		UO_2-Al
fuel plate thickness		0,4 mm		0,64 mm
active length			580 mm	

The spent fuel is stores in the 2 stores on site of the reactor. In 1995 a new high active waste storage was commissioned. All spent fuel will be transported to the new storage facility in due course. The transport route is outside the NRI area through Řež village and back to the NRI but to a different area. For the transport the special container Škoda 1xIRT-M is used.

IN-PILE MATERIAL RESEARCH

Materials related problems cause a significant portion of power plant outage time. This is connected with environmental degradation processes (corrosion, mechanical and radiation effects) during power plant operation. At the same time the operators and reactor vendors are interested in plant life extension, higher fuel burnup and lower radiation fields. Towards these aims, specific investigations are performed in facilities simulating environmental parameters of light water reactors including reactor water loop performing in-pile tests in radiation environments.

Experimental test facilities allowing exposition of materials to radiation environment are used. The facilities enabl to carry out research on the behaviour of materials and water chemistry of PWR and BWRs. High pressure water loop RVS-3 is operated at test research reactor. The loop was designed as a universal facility providing wide experimental possibilities. Loop is designed for a pressure of 16.7 Mpa, temperature of 334 °C and 10 t/h flow rate. Water chemistry used corresponds to the PWRs with boric acid, pH control by potassium/lithium and hydrogen control by hydrogen gas or ammonia. BWR-1 loop is designed for investigation of structural materials and water chemistry in conditions of boiling water power reactors. BWR loop has the following parameters: 10 MPa pressure, 300 °C temperature and 2 t/h water flow rate. CHOUCA MT type irradiation rigs are used for the irradiation of reactor pressure vessel materials within the framework of reactor pressure vessel integrity, lifetime and reliability programme. Second boiling water loop BWR-2 is used for stress corrosion cracking of 50/20 CT specimens.

The facility for corrosion of zirconium alloys and cladding-coolant interaction consists of two experimental sections of electrically heated fuel rods with maximum power of 100 W/cm^2. One section is situated in the loop outside the reactor core (out-of-pile), the second section is situated in the irradiation channel (in-pile).
The facility enables to carry out research in the following fields:

- Corrosion of zirconium alloys with/without the effect of radiation fields,
- Deposition of corrosion products,
- Effects of water chemistry components including lithium, potassium, ammonia, hydrazine and boric acid.

PIE (Post Irradiation Examination) studies involve metallographic evaluation of the phase composition of alloys, degree of hydriding, thickness of oxidic layer, and corrosion damage.

The SSRT tests are used for the evaluation of irradiation assisted stress corrosion cracking (IASCC). The aim of the experiment is the determination of the effect of neutron radiation and flowing water in specimens loaded with small strain rate in the reactor water loop inside the active channel.

The IASCC experiment includes the following activities:

- Irradiation of tensile test specimens with a diameter of 6 mm in the irradiation probe CHOUCA-MT (French production) in an inert atmosphere at a temperature of 290-300 °C and neutron fluence up to $1x10^{24}$ n m $^{-2}$.
- Disassembling of the irradiation probe and removal of test specimens. Insertion of the test specimens into the carrier for SSRT.
- Carrying out the SSRT in the active channel of the reactor water loop at a temperature of 290-300 °C, pressure of 12.5 MPa and total neutron fluence up to approximately $5x10^{24}$ nm^{-2} (E>1 MeV). The tests will be performed at parameters corresponding to PWRs or BWRs.

RADIONUCLIDES PRODUCTION

A number of reactor radionuclies can be prepared in the Nuclear Research Institute's LWR-15 reactor. These radionuclides are applied either as unsealed sources in the nuclear medicine diagnostics and therapy or as sealed sources for commercial irradiation and in the radiation oncology. Some of these nuclides have been prepared in the NRI laboratories formerly already (^{99}Mo, ^{113m}In, ^{125}I, ^{131}I and ^{192}Ir), while the production of the other ones is the subject of current research (^{152}Sm, ^{60}Co). The possibilities are being explored of producing some other nuclides with the supposed application in nuclear medicine as radiotherapeutics, namely of ^{89}Sr, ^{111}Ag, ^{166}Ho, ^{169}Er, ^{186}Re, ^{188}W and ^{191}Os.

SILICON TRANSMUTATION DOPING

By the capture of a thermal neutron, the natural isotope ^{30}Si can be transmuted into the unstable isotope ^{31}Si which decays by beta emission into the stable isotope ^{31}P (half-life of ^{31}P is 2.6 hr). By means of this reaction, donors can produced in a silicon crystal.

The capsule with silicon crystals is held with the input device DORA from the preparation channel and inserted into the irradiation channel DONA. The DONA is placed in the outer region of the core. The maximum flux of thermal neutrons reaches $1.7x10^{13}$ neutrons/cm^2s in the irradiation channel at the reactor power of 8 MW. During the irradiation, DONA can perform two types of motion, a rotational motion of 2 rpm around his own axis and a translational one. During the trial operation it has been desisted from the translational motion.

Neutron flux is measured by a self-powered neutron detectors in five positions at the distance of 200 mm. The position of maximum thermal neutron flux is computed from the obtained data by the least squares method. The crystal center is placed into the maximum of the fluence. The course of the thermal neutron flux enables to irradiate crystals up to 200 mm in length.

The required dose is entered into the control computer. The neutron-flux detectors measure continuously the neutron flux and the moment, when the actual dose equals the required one, crystal automatically leaves the irradiation position and is pulled out to the so-called parking position. The capsule is gripped by DORA and transferred into hot cells where the crystal is removed from the capsule. One week after it is possible to handle the crystal without danger.

Following are produced:	Typical parameters of the DONA channel: (10 MW)
- max. crystal dimension: diameter 3" (4") length 200 mm (300 mm)	- thermal neutron flux density: $2x10^{17}n/m^2s$ - fast neutron flux density: $2x10^{16}n/m^2s$

- typical nominal resistivity: 330, 220, 130, 60 ohms

BORON NEUTRON CAPTURE THERAPY FOR CANCER

The ultimate goal of cancer therapy is to achieve a degree of selectivity that would spare normal cells and may lead to recurrence, metastases and death.

Nuclear reactors have been the exclusive source of neutrons for BNCT. Thermal neutron beams with energies in the region of 0,0025 eV and epithermal beams with energies of 1-10,000 eV may be used. Thermal neutrons are attenuated rapidly in tissues, and it is difficult to obtain sufficient flux of thermal neutrons at the depth without severely irradiating the surface tissues. An epithermal neutron beam may be produced by using a filter or moderator as a spectrum shifter to slow the fast neutrons to intermediate energy range. Epithermal neutrons are relatively non-destructive, when used to irradiate tissues, provide better penetration than thermal neutrons and produce thermal neutrons at the depth in tissue because of the moderating effects of the tissue itself. Such a beam of neutrons peaks the thermal neutron flux a few centimetres distance from the tissue surface.

There is a need to develop alternative neutron sources to overcome the limitation of BNCT given by its current dependence on nuclear reactors as a source of neutrons.

A filtered beam of epithermal neutrons at the LVR-15 reactor is used as a source for BNCT purposes. The filters are situated in the empty space of the reactor thermal column. The filter consists of cylindrical blocks of 100 cm in diameter, total length is 265 cm. The active part of the filter is composed of 55 cm of aluminium, 15 cm of sulphur and 1 cm of titanium. The outer diameter of the beam of 11,5 cm is adjusted by the final shutter. The central block is a graphite collimator. The techniques available to measure the neutron fluence include semiconductor Si-Li detectors, solid state nuclear track detectors, fission chambers, silicon diodes, for γ-rays thermoluminescence detectors, scintillation spectrometer and for neutron spectrum determination Bonner spheres, scintillation spectrometer, and proportional hydrogen spectrometer. It is important to know the dose distribution, both neutron induced and gamma, along the patient body during the treatment. This parameter was measured using a human whole body phantom. The phantom-skeleton was filled up with water. Measuring channels are situated along the back bone and trough the head.

The BNCT method offers an opportunity to treat certain types of tumours that are presently inoperable or highly metastasised. With appropriate quality of the neutron delivery system and boron compound it is now feasible that BNCT can be successfully demonstrated. The Czech Republic could be among a few centres where the procedure will be used for treatment as a standard one.

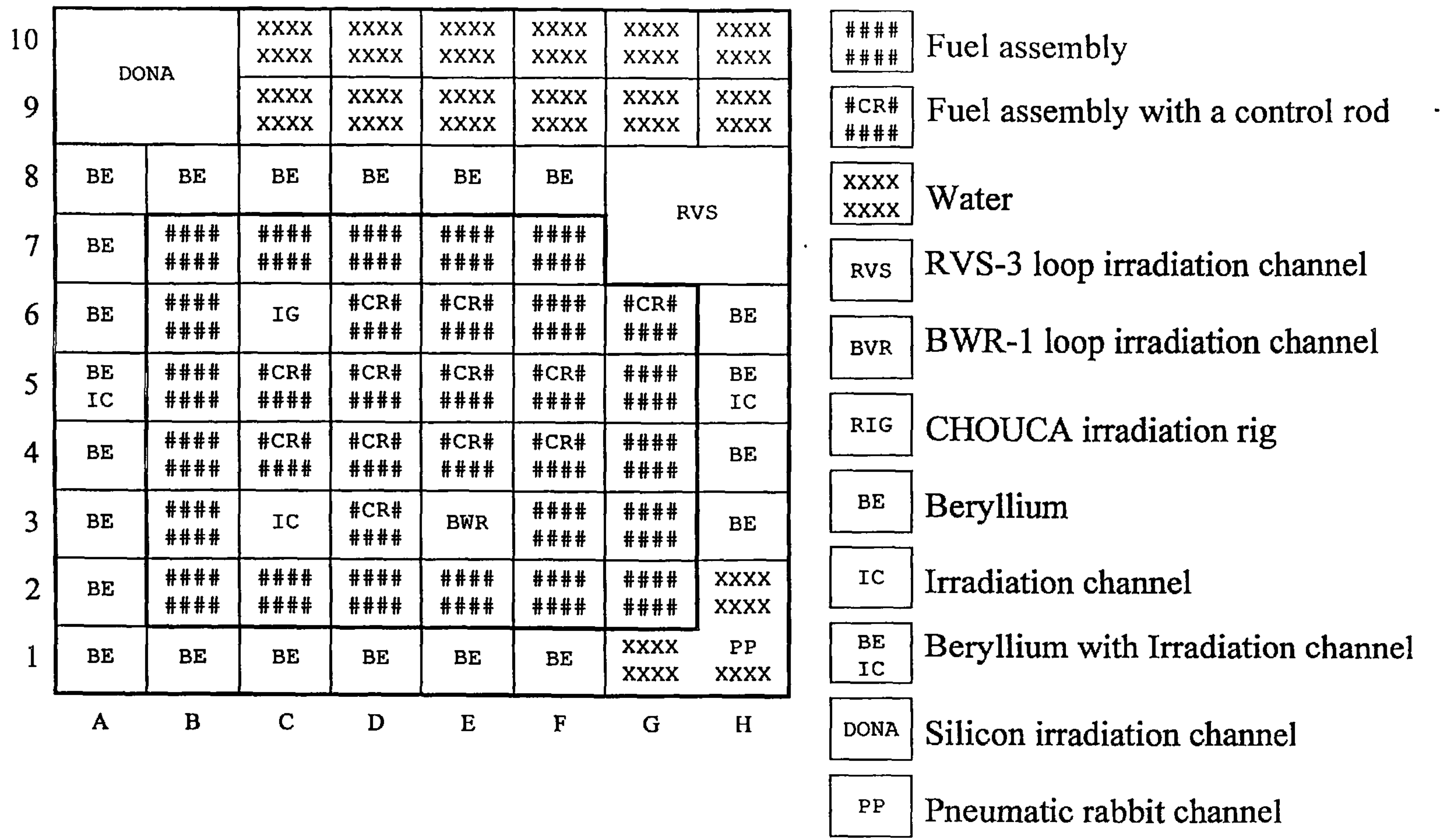

Fig. 1 LVR-15 reactor core configuration

REACTOR LVR - 15

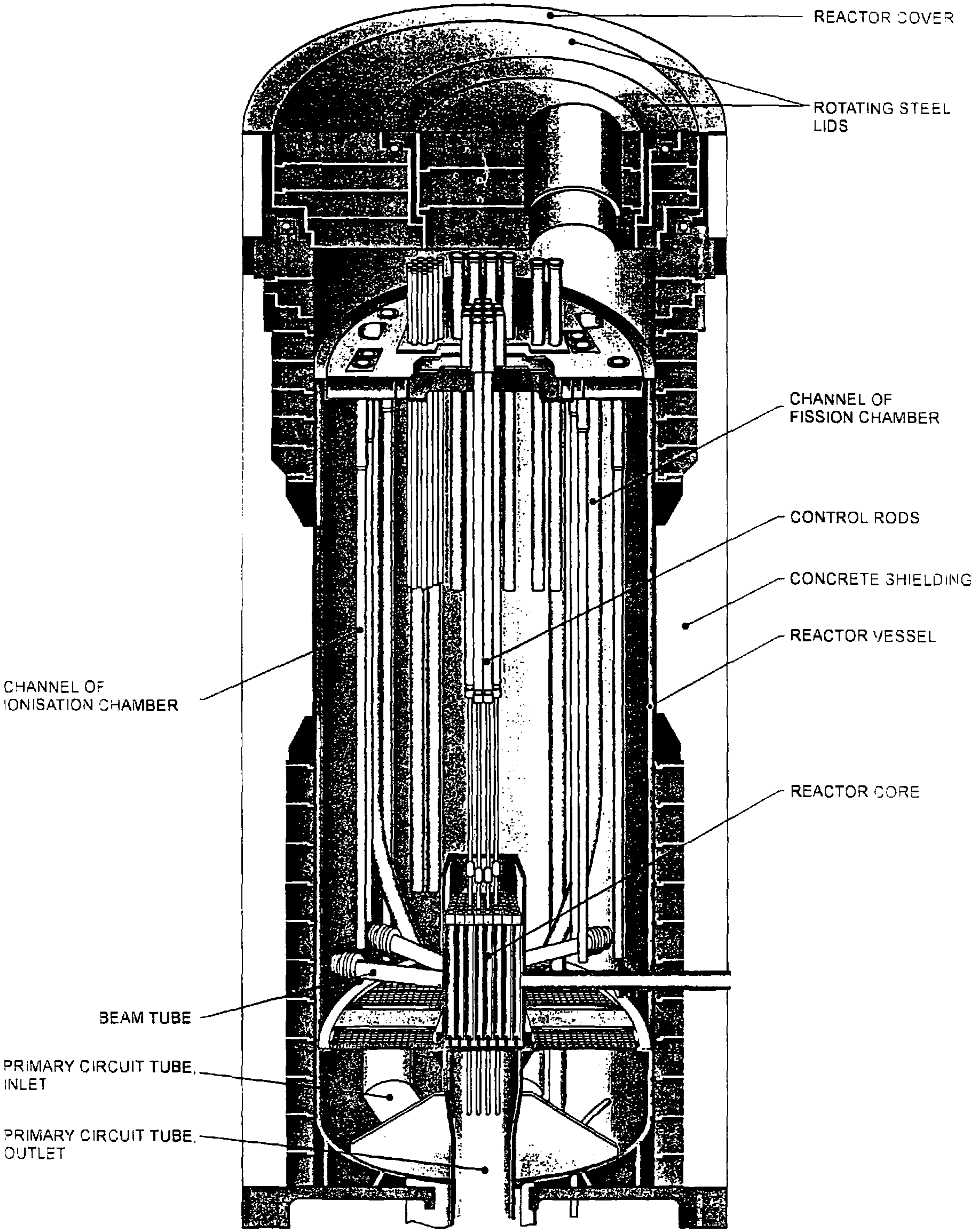

Fig. 2

HIFAR Major Shutdown Report

INSPECTION OF REACTOR ALUMINIUM TANK AND REPLACEMENT OF SECONDARY COOLING WATER CIRCUIT PIPEWORK

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ABSTRACT

HIFAR (High Flux Australian Reactor) is a 10-MW DIDO-class research reactor operated by the Australian Nuclear Science and Technology Organisation (ANSTO).

Every four years HIFAR has a scheduled major shutdown to undertake inspection, maintenance and upgrade activities that are not possible at other times. The last major shutdown was in late 1995 and lasted 10 weeks during which, and amongst other activities, the Reactor Aluminium Tank was inspected and the major part of the Secondary Cooling Water Circuit pipework replaced.

New cooling pipework was designed in accordance with Australian Standards Pressure Piping Code, employing NISA II computer software for finite element analysis where appropriate, with a view to ensuring safety under every conceivable operating and accident condition. Design, manufacture and installation activities were carried out according to ANSTO Engineering's Quality System Procedures (ISO9001-1994 accredited) and agreed by the Nuclear Safety Bureau, an independent organisation, through staged document submissions.

1. INTRODUCTION

HIFAR is a heavy water moderated, reflected (also by graphite) and cooled reactor with a maximum thermal flux of 1.4×10^{14} n cm^{-2} s^{-1} at its design power of 10 MW which makes it an ideal source of neutrons to irradiate materials for scientific, medical and industrial purposes. HIFAR is the only reactor operating in Australia.

HIFAR first went critical in 1958 and has been in successful operation at full power (10 MW) since 1960. One reason for its long and safe life is the vigorous maintenance program including, but not limited to, chemical control of the cooling water which has direct bearing on the state of the Reactor Aluminium Tank (RAT) and the cooling water circuits. Corrosion inhibitor and algaecide are added to the secondary cooling water (light water) to minimise the internal pipe corrosion and algae growth.

Prior to and during the major shutdown, selected sections of the Secondary Cooling Water Circuit (SCWC) pipework were surveyed using the 'Micro-Map' system which provided thickness maps of the pipe shell to determine degrees of internal corrosion. The results showed that, after 38 years of reactor operation, some of the pipework needed to be replaced.

2. REACTOR ALUMINIUM TANK (RAT)

The RAT has been regularly inspected at each major shutdown since 1970 to identify any corrosion or structural damage that could render it unfit for continued service. Previous inspections have been limited to a visual survey of the interior surfaces of the RAT, but recent improvements in the inspection equipment and procedures enabled greater precision and accuracy in identifying features during the 1995 major shutdown.

Six special jigs were designed and manufactured to facilitate inspection and testing of the RAT. These enabled obtaining cast impressions of the RAT internal surfaces and measuring the surface hardness. They were also used for holding ultrasonic probes and moving them inside the RAT in the vicinity of the main welds and inside the primary cooling water pipes. The jigs were held by a manipulator inserted through appropriate vertical facility holes in the top plate of the reactor when the fuel, rigs and heavy water were temporarily removed. Their movements inside the RAT were tracked by a video camera.

2.1 Visual Survey

A high resolution colour camera was connected to a Super-VHS video recorder and a laser disc image recording system. This enabled individual frames to be captured to the full resolution of the camera.

2.2 Hardness Testing

Remote hardness measurements were made on the internal wall and several internal facilities of the RAT to provide information regarding the current post-irradiation mechanical properties of the aluminium material. The hardness tester was positioned against the RAT wall by a remote manipulator. The tester was basically equipped with an 'Equotip' head which armed and propelled the spring loaded ball to hit the wall when triggered by a pneumatic remote control device. The velocity change of the ball after the impact was used to give a measurement of hardness. The hardness tester is shown in Figure 4.

More than 600 locations were tested to obtain the profile of the tank hardness as shown in Figure 2. It was found that the hardness varied as a function of distance from the core centre line, however, the maximum hardness was measured in a region slightly away from the core centre line. The hardness measured is the result of two counteracting processes; radiation annealing of the initial cold worked structure which results in a decrease in hardness and transmutation of aluminium to silicon which results in an increase in hardness. Due to these competing processes the position of maximum hardness is not necessarily at the core centre line. Overall, the hardness of the RAT has been found to be similar to that which is likely to have been the installed condition. It should be noted that the accuracy of the results and the phenomena causing these hardness changes are currently under study.

Estimation of tensile properties by hardness correlation revealed that the observed hardness changes are of no significance to the structural integrity of the RAT.

2.3 Wall Thickness Survey

An ultrasonic device was used for point measurements of the RAT wall thickness. Demineralised water was used as the coupling medium, with the jig being positioned using the in-tank manipulator. The results showed no measurable loss of wall thickness to date. See Figure 3 for histogram showing distribution of RAT wall thickness values.

2.4 Replication of Surface Features

Dental paste was employed to obtain cast impressions of the selected RAT internal surfaces. Uncured paste was pushed against the wall by a manipulator and held in position until hardened (7 minutes) to form a negative replica which was then recovered for production of a positive replica (a replica of the replica) for analysis. This method enabled true three dimensional interpretation of the surface conditions and accurate measurements of defects, e.g. deposits, tide marks, oxide flaking, peening, scuff marks and pitting. The results showed no significant defects. See Figure 5 and Figure 6 for examples of replica images.

3. SECONDARY COOLING WATER CIRCUIT (SCWC) PIPEWORK

The secondary cooling water is cooled by six cooling towers and circulated at an approximate rate of 350 kg/s to remove the heat from the primary D_2O cooling circuit and the rig heat exchangers inside the reactor. Water chemistry is monitored and controlled daily in order to minimise internal pipe corrosion and algae growth. Total volume of the water in the system is 332000 litres.

The SCWC pipework is made of carbon steel pipes, mainly 500 mm nominal bore with other smaller pipe branches. Some sections of the pipework are buried in sand 550 mm deep (ground to the centre of the pipe). See Figure 1 for general layout of the SCWC pipework with the replaced section identified.

3.1 Old Pipe Thickness Inspection

The SCWC pipework has been regularly inspected for residual wall thickness using conventional ultrasonic thickness gauges until the last major shutdown. This measured the local thickness and did not give information of the surrounding areas. An advanced ultrasonic imaging system, Micro-Map, was procured to enable more comprehensive surveys during the 1995 major shutdown. The Micro-Map contains a number of modules for performing specific ultrasonic inspections. The "See-Scan" module employs a video capture board capable of monitoring the position of a thickness gauging transducer so that readings of thickness and relative positions can be permanently recorded as the transducer is moved over the test areas until the complete area is covered. The survey results showed that the pipework was in fair condition, but a decision was made well before the major shutdown to replace it to be conservative. Figure 7 shows an example of the Micro-Map image of a test area (approx. 400 mm x 400 mm) and Figure 8 shows chemical deposits on the inside of the pipe.

3.2 Design

3.2.1 Design Standards

The new pipework was designed not only to comply with the current Australian Standards but also to withstand any abnormal conditions such as bushfire and earthquake experienced in Sydney in 1994 and 1989 (weak tremor only), respectively.

Australian Standards for Pressure Vessels (AS1210-1989, Class 1) and Pressure Piping Code (AS4041-1992, Class 1) were adopted for the design, manufacture, installation and testing of the new pipework. These standards are comparable with the American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section 8.

NISA II computer software for finite element analysis was employed to calculate stresses and strains in the pipework due to the hypothetical bushfire and seismic events. The original design had to be modified to satisfy both the seismic and bushfire conditions because the former required rigid pipework while the latter required the opposite.

The pipe modifications included the following:

- thicker pipes for rigidity and greater thermal mass - pipe temperature would not rise as much as the original pipes during bushfire, e.g. 90°C (9.5 mm thick) vs. 107°C (6.4 mm thick);
- installation of new pipe supports for the buried sections to control and guide the pipe movements (due to thermal expansion and seismic vibration) in such a way that over-stress in the pipework is avoided without losing flexibility; and
- reinforced pipe supports near the reactor building in order to minimise the stress in the reactor building wall penetrations (through which the SCWC pipework is connected to the circuit inside the reactor) due to the pipe movements.

The design, manufacture and installation were carried out according to ANSTO Engineering's Quality System Procedures (ISO9001-1994 accredited) and reviewed and agreed by the Nuclear Safety Bureau (NSB) through staged document submissions before the execution of the next stage. The NSB is an independent Australian Government organisation responsible for monitoring and reviewing the safety of the nuclear reactor operated by ANSTO.

A summary of the design basis conditions and criteria is shown in Table 1.

3.2.2 Reactor Containment Building (RCB) Penetration

The reactor is contained in a building made of 21 mm thick carbon steel plates, 21.3 m in diameter and approximately 25 m in height. The RCB is leak tight and classified as a pressure vessel 'Class 1' which requires stringent modification procedures and tests.

The RCB penetrations, through which the SCWC pipework enters and exits the RCB, were reinforced after the installation of the new pipework and hermetically seal welded.

Stress analysis on the penetrations was carried out to justify the design.

3.2.3 Corrosion Protection

The external surface of the exposed pipework was painted according to the HIFAR colour scheme. The external surface of the buried pipework was coated with polyethylene for protection against corrosion. This method has been proven to be reliable for the Sydney water supply pipes for many decades.

The internal surface of the pipework is not treated (bare metal) because the current water chemistry has proved to be satisfactory. With the increased pipe wall thickness, which provided an additional 3 mm of inherent corrosion allowance, the new pipework is anticipated to surpass the performance of the old pipework which lasted 38 years, provided that the water chemistry is maintained. It should be noted that a working group has been recently formed to review the water chemistry and to automate the chemical dosing system.

3.2.4 Flowmeter

A new in-line electromagnetic flowmeter with 500 mm nominal bore stainless steel body was selected to replace the old 'Dall Tube' venturi type flowmeter because,

- the electromagnetic meter is modern, reliable and much more accurate (±0.2%), and
- the electromagnetic meter is a full bore design, hence does not restrict the flow.

The principal of operation is that an electromagnetic flux is induced across the fluid when the flow inside the pipe cuts a magnetic field generated across the meter (Ø500 mm pipe with windings on the outside). The flow is an aggregate figure and not affected by the velocity profile of the flow across the pipe or the chemical contents in the fluid, hence yielding accurate readings.

The remote transmitter is located inside the flowmeter pit outside the reactor and its signal cables (typical 4-20 mA) are fed through the RCB penetration cartridge (socket and plug) to each of the display locations inside the reactor. The old analogue gauge in the reactor Control Room was replaced by a digital and graphic display unit for enhanced readability.

3.3 Tests

The welding test specimens were mechanically and chemically tested prior to manufacture as required by the 'Class 1' pressure piping standard.

The weld joints in pipes were tested by radiography during manufacture and by the magnetic particle method during installation.

After final installation but prior to filling the trench with sand (for buried pipes) and sealing the ground surface with asphalt, the new SCWC pipework was inspected by a sniffer dog for the presence of any explosives inside the pipework as a security measure, followed by a hydrostatic test.

The hydrostatic test was carried out at 460 kPa (gauge), ie, 1.25 times the design pressure (which also included pressure surge due to water-hammer effect), in accordance with the relevant Australian Standards. No structural weakness was observed during the test, however, a few minor leaks through flanged joints had to be corrected by re-seating the gaskets and tightening the flange bolts to the specification.

The RCB penetrations were leak tested by pressurising the whole reactor building (RCB) to 10.3 kPa (g) and holding for 24 hours. Pressure drop was used to give a measurement of the gross leak rate through all RCB penetrations and other seals. The test showed no measurable leak through the penetrations. This test is regularly carried out to ensure the leak tightness of the RCB at all times.

4. FUTURE PLAN

The rest of the old SCWC pipework (mainly inside the reactor) is planned to be replaced during the next major shutdown in 1999. Seismic analysis and other relevant calculations will be carried out as part of the design.

Meanwhile, the water chemistry will be changed to give better protection against internal pipe corrosion and algae growth. The chemical dosing system will be automated for more precise control of the water chemistry.

5. CONCLUSION

The objective of the inspection was to provide visual and other evidence of the conditions of the RAT and the SCWC pipework.

The results from the RAT inspection tasks showed that the RAT is in very good physical condition. There has been no measurable corrosion or loss of wall thickness to date and no evidence of any notable change in physical condition since the previous visual inspection. The only defects observed were insignificant and limited in length and depth. The hardness of the RAT has slightly changed from what is deduced to have been the value at manufacture, however, the observed hardness changes are of no significance to the structural integrity of the RAT or on the extent of conservatism embodied in the original design. No features were observed that would preclude the operation of the RAT until the next major shutdown in 1999.

The results from the SCWC pipework inspection showed that the pipework was in fair condition with some internal corrosion as expected from the 38 year old carbon steel construction even though corrosion inhibitor has been used in the cooling water. The main part of the SCWC pipework and its associated equipment (flowmeter, valves, headers and reactor building penetrations) were replaced by the upgraded components designed and manufactured with today's technology.

The 1995 major shutdown was brought to a successful conclusion when the reactor returned to full power operation on 28 November 1995, one week ahead of schedule.

6. ACKNOWLEDGMENT

The inspection and testing of the RAT were carried out by Advanced Materials (a division of ANSTO) with support from ANSTO Engineering (a division of ANSTO).

The SCWC pipework project was managed by ANSTO Engineering.
Many colleagues contributed to the successful completion of this project.

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TABLE 1: SUMMARY OF DESIGN BASIS CONDITIONS

PARAMETER	SCWC Pipework & Headers	Reactor Containment Building Penetration	REMARK
1. Process Fluid	light water	air	
2. Internal Pressure 2.1 Normal 2.2 Abnormal 2.3 Design 2.4 Test	gauge pressure 0 to 150 kPa 0 to 230 kPa (0 to 365kPa)* 460 kPa	gauge pressure 0 -6.9 to +10.3 kPa -7.6 to +11.3 kPa +10.3 kPa	* Design pressure includes the maximum abnormal pressure and pressure rise due to water hammer effects caused by valve and pump operation.
3. Temperature 3.1 Normal 3.2 Abnormal 3.3 Design	 1 to 45°C 1 to 90°C ** 1 to 100°C	 (1 to 45°C) + 1 to 107°C ** 0 to 200°C	+ Extreme air temperatures on site (southern Sydney). ** Calculated maximum pipe temperature due to bushfire near the reactor. It will be much lower if the water flows inside the pipe. N.B. Headers are inside the RCB and not affected by bushfire.
4. Flow Rate 4.1 Normal 4.2 Abnormal	 0 to 355 kg/s 0 to 390 kg/s	 0 0	Electromagnetic flowmeter replaced the 'Dall Tube' (venturi type) flowmeter.
5. Seismic Vibration 5.1 Peak Horizontal 5.2 Peak Vertical 5.3 Damping Value	 0 to 2.3 m/s^2 0 to 1.53 m/s^2 up to 3% of critical	 0 to 2.3 m/s^2 0 to 1.53 m/s^2 up to 3% of critical	Analysed by the NISA II Finite Element Analysis computer software.
6. Cumulative Radiation Dose 6.1 Normal 6.2 Abnormal	 0 to 10 Gy 0 to 10000 Gy	 up to 10 Gy up to 10000 Gy	Does not affect the design. Neoprene gaskets are suitable for these conditions.
7. Corrosion Protection	Externally painted. Corrosion inhibitor in cooling water.	Painted	Buried pipes are externally coated with polyethylene.
8. Material Spec. 8.1 Old system 8.2 New System	 Mild steel pipe Ø500 mm NB, 6.4 mm WT Mild steel pipe, ASTM A53 B, Ø500 mm NB, 9.5 mm WT	 Mild steel shell, Ø21 m, 21 mm thick, penetration hole for Ø500 mm NB pipe is reinforced with rings (Ø790 mm OD, Ø520 mm ID, 20 mm thick)	 Construction of the old pipework does not conform to AS4041 Class 1 standard. New materials for pipes and plates conform to AS1210 and AS4041 Class 1 constructions.

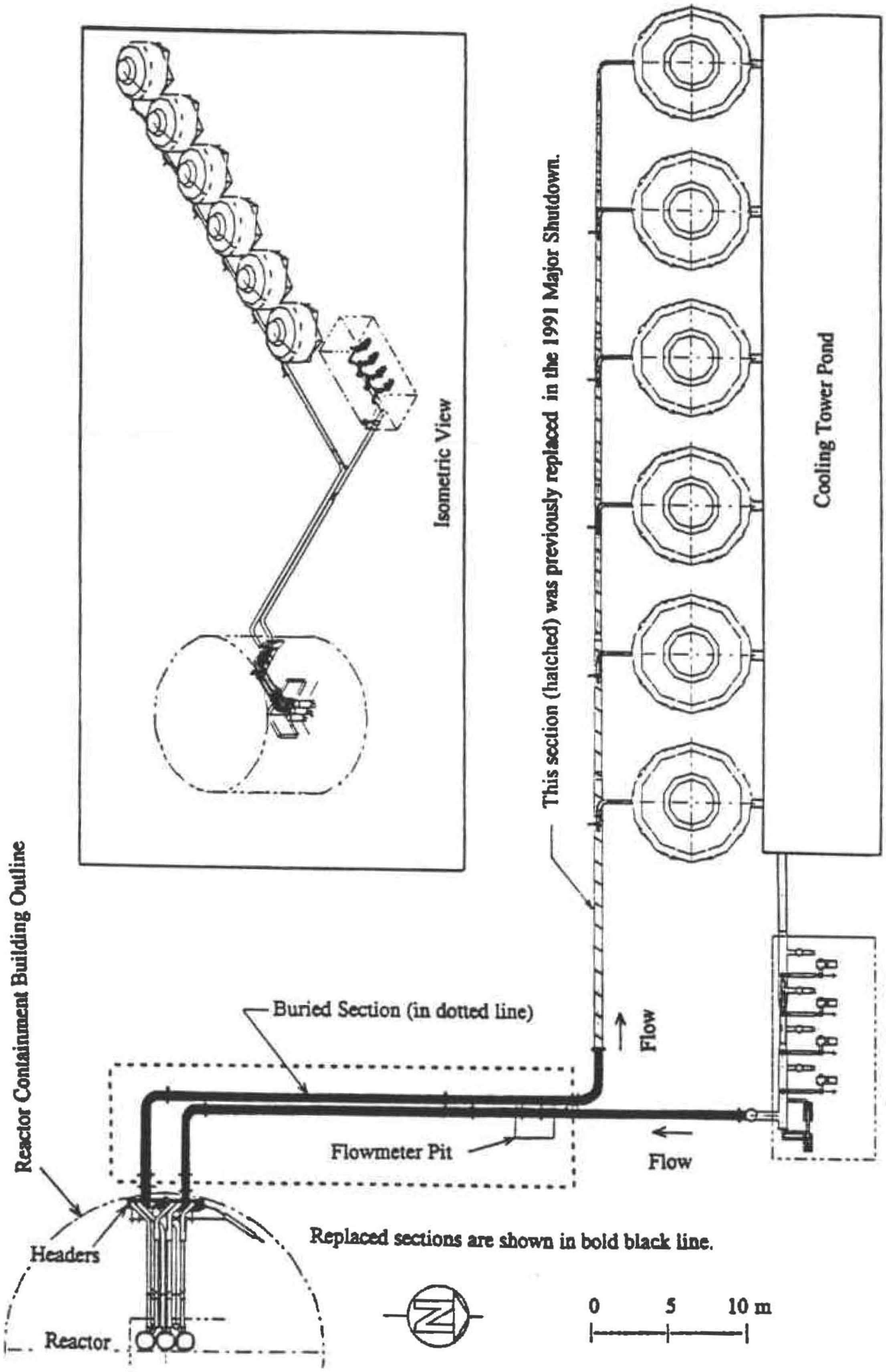

Figure 1. General Layout of HIFAR Secondary Cooling Water Circuit

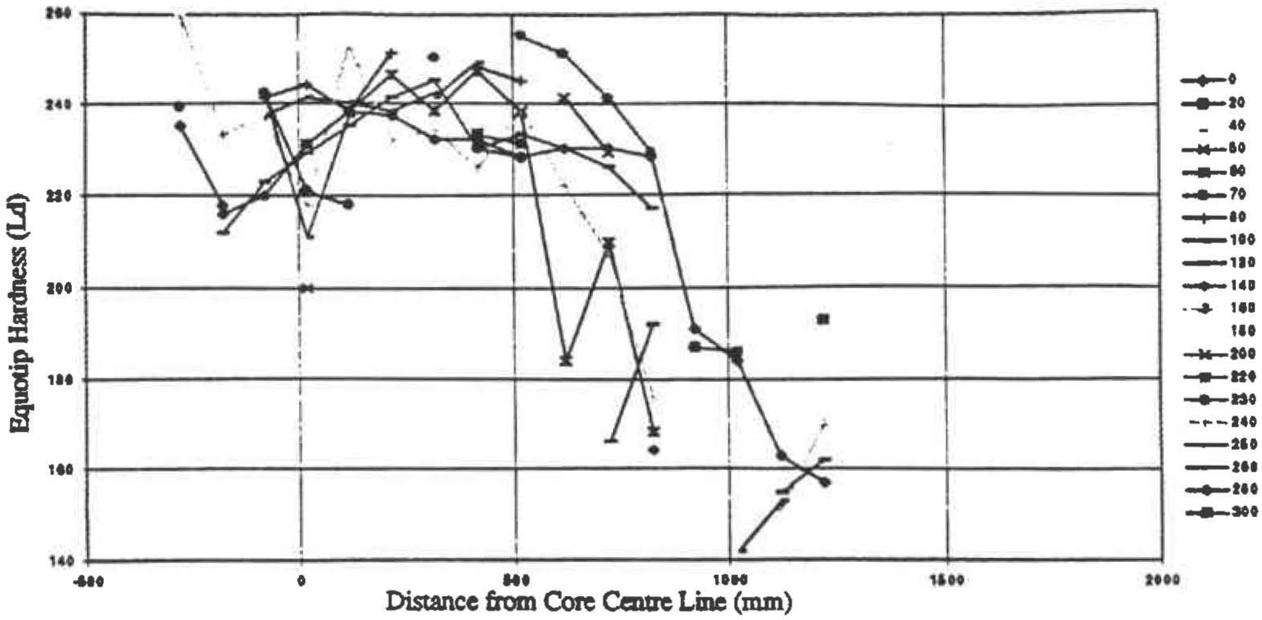

Figure 2. RAT Wall Hardness Profile at Various Angles

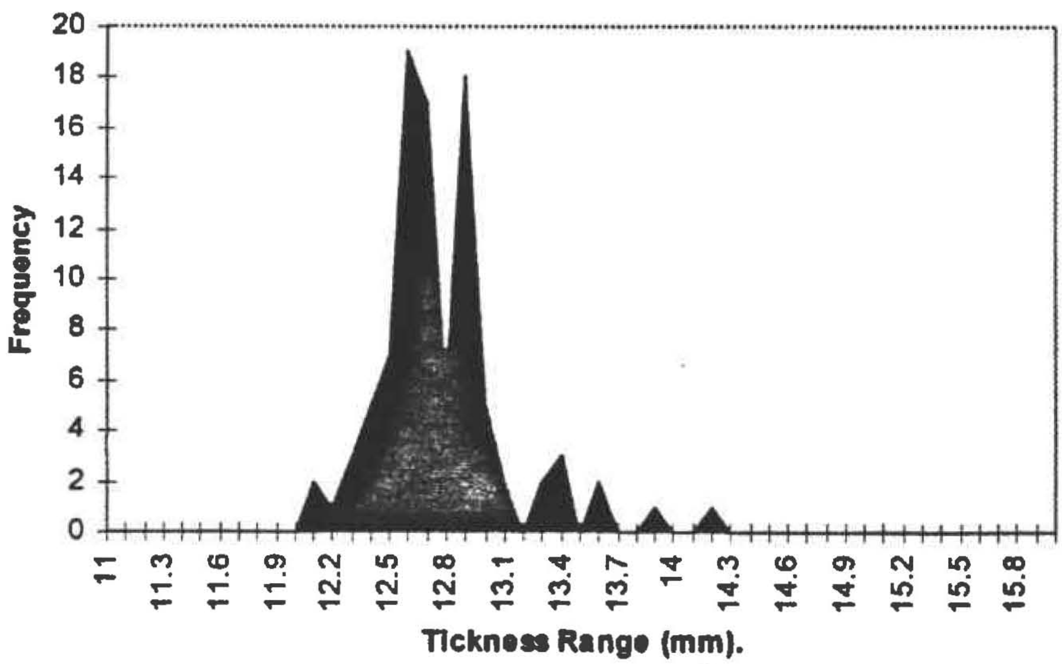

Figure 3. Histogram of RAT Wall Thickness Values
(High and low values from the vertical and horizontal welds contributed the observed spread, ie, greater than 13 mm.)

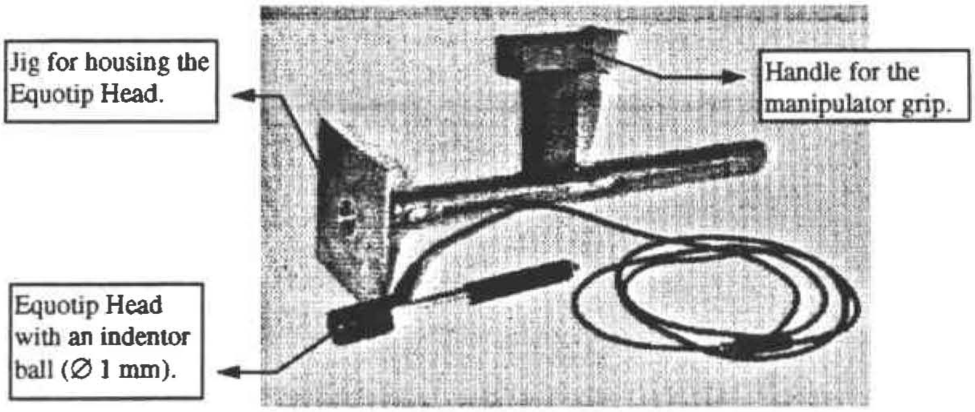

Figure 4. Hardness Tester - 'Equotip' Head and a Jig

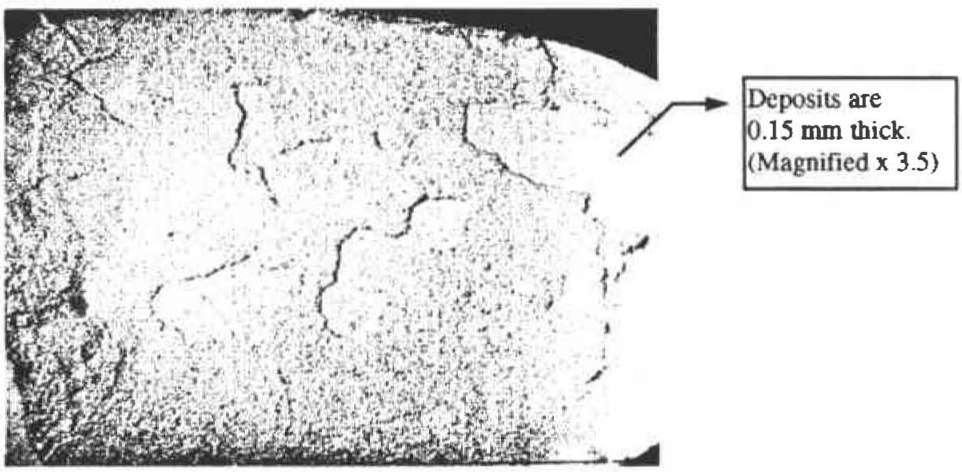

Figure 5. Replica Image of Oxide Flaking on RAT Wall

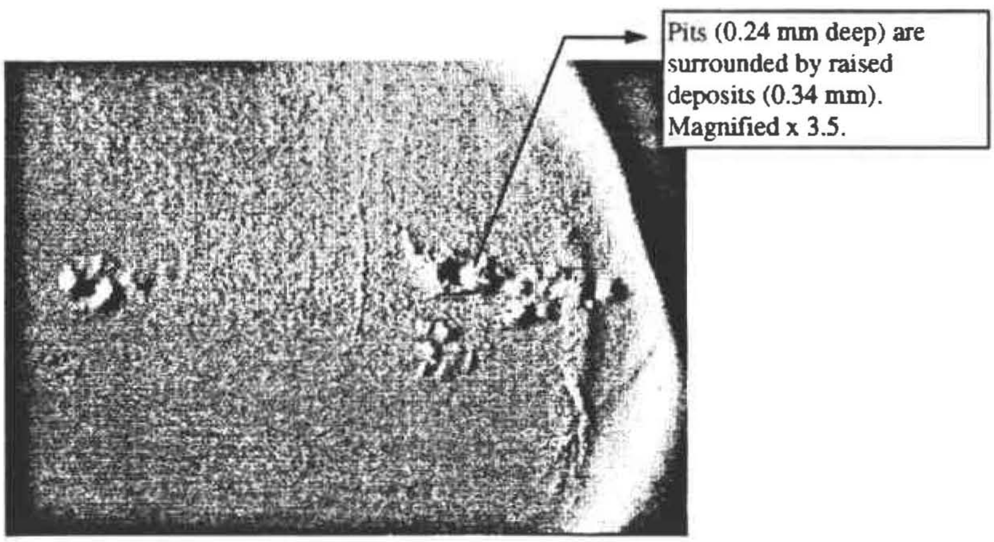

Figure 6. Positive Replica of Pits on RAT Wall

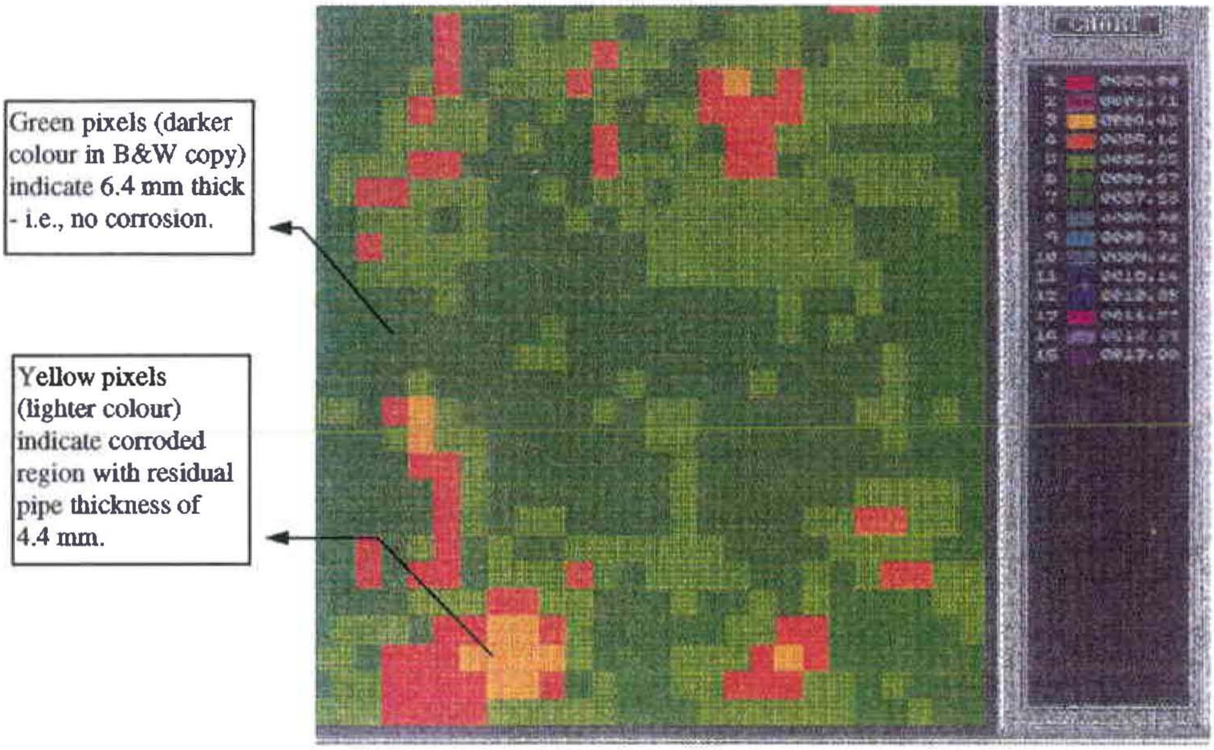

Figure 7. Micro-Map Image of the Old SCWC Pipe Thickness

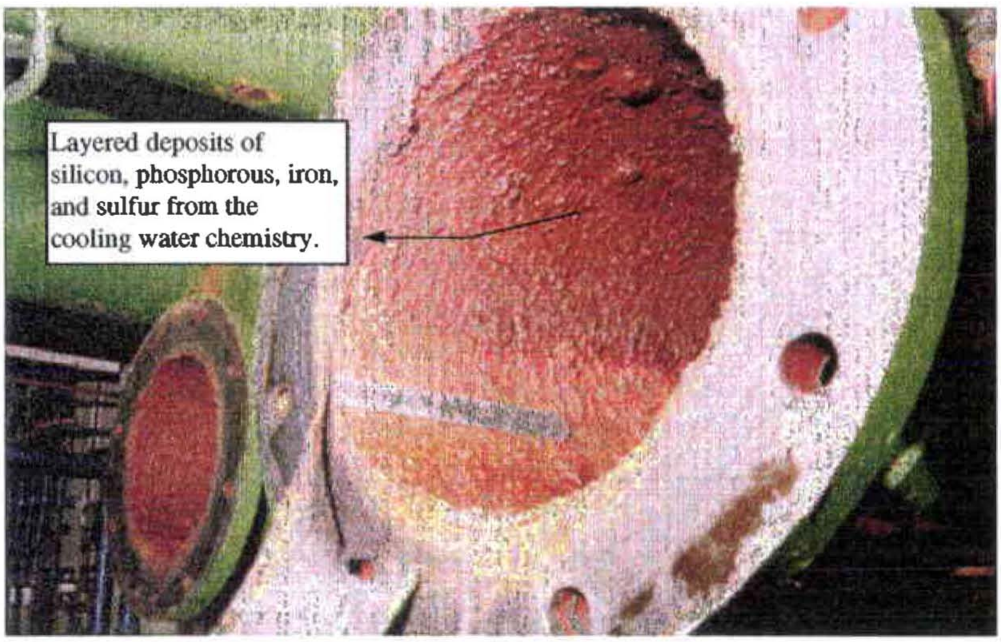

Figure 8. Corrosion Deposits inside the Old SCWC Pipework

Modification of JRR-4

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ABSTRACT

Japan Research Reactor No.4 (JRR-4) is a light water moderated and cooled, 93% enriched uranium ETR-type fuel used and swimming pool style reactor with thermal output of 3.5MW. Since the first criticality was achieved on January 28, 1965, general investigation has been continuously succeed so as shielding experiment, fuel and material irradiation experiment, RI production, nuclear activation analyses, silicon doping, reactor school training, etc..

The modification of JRR-4 is planned about core conversion according to the framework of reduced enrichment on research reactor program, utilization facilities upgrading, and renewal of some reactor systems. The fuel will be changed from aluminized uranium to 20% lower enriched uranium silicide fuel with 3.8g/cc, however the reactor performances will be calculated same thermal flux level as HEU core at same thermal power, without changing of structure, size and number of fuel in the core. The utilization facilities are installed a medical irradiation facility for BNCT and are modified a NAA system for short life nuclides ,and a large size pipe irradiation system. Furthermore to be operated long term safety and continuously, it is designed that are renewals of instrument and control system, installation of large scale fuel failure monitor system and emergency exhausting system, repairing of reactor building for reinforced seismic design. This paper described outline of the modification of JRR-4.

1. INTRODUCTION

In Tokai Research Establishment, JAERI, three research reactor as JRR-2, JRR-3M and JRR-4 is operated over thirty years. The JRR-3M which was reconstructed in 1990 as one of the highest performance research reactor in the world, today, is used by a great number of researchers and engineers. The JRR-2 which is CP-5 type research reactor and is operated for thirty-six years since 1960 will be terminated by reason of aging in December 1996.

In the such situation, it was necessary that the JRR-4 was converted LEU core because of getting new driver fuels. The JRR-4 has used over 90% enriched uranium plate-type fuel(total 120 elements) since first criticality in 1965, amount of driver fuel of hand durability disappeared finally at January 12th, 1996. The core conversion activities in JAERI were begun in 1980's. It is already done successfully that JRR-3M is used aluminide LEU fuels, JMTR is converted to MEU in 1986, and silicide LEU in February 1994. JRR-4 has been studied to convert to the LEU silicide fuel without major change of fuel structure and core arrangement. Fig.1 and Fig.2 show a bird's-eye view of the JRR-4 buildings and a view of the reactor.

The other, JRR-2 has been using 33 cases of medical irradiation for Boron Neutron Capture Therapy (BNCT) since 1990, but JRR-2 will be terminated at the end of 1996. And the mission of JRR-2 for BNCT will be transferred to JRR-4 after suspension of two years, a new medical irradiation facility will be installed at.JRR-4.

Furthermore, some components of JRR-4 was aging cause by operating long term since first criticality, and the reactor building will be reinforced against seismic design.

2. LEU CORE DESIGN

The core of JRR-4 is composed 8 x 8 array with 20 fuel elements, 7 control rods, 5 irradiation holes, a neutron source and many reflectors. The JRR-4 core arrangements is showed Fig.3. A fuel element has 15 fuel plates which is fuel meat (U_3Si_2) of 600x64x0.5mm with aluminum cladding. The fuel element is showed Fig.4. The design and evaluation for core conversion with LEU silicide fuel instead of LEU aluminide fuel was started in 1991. The comparison of LEU fuel with HEU fuel is showed table-1. A preliminary estimate showed that the new LEU core has been available performance for thermal flux level (7 x 10^{13} n/cm^2/s) as same as HEU core, without major change of structure and size of fuel element and core dimensions. Table 1 show the comparison of LEU fuel with HEU fuel.

As a result on the neutronic design , the excess reactivity of core with fresh 20 LEU fuel elements is 11.7% δ k/k at maximum,and the one rod stuck margin is about -2.3% δ k/k and the reactivity coefficient is negative value in the calculated all range. The new fuel elements will be able to use about six years longer than HEU fuel elements.

And on thermal-hydraulic design, the core coolant flow rate is increased from 7m³/min to 8 m³/min by three main pumps because a safety margin of fuel is increased in case of power cut. The fuel surface temperature is about 111℃ which value is not exceeded 125℃ as ONB(onset of nucleate boiling) temperature. The fuel meat temperature is about 113℃ at rated power 3.5MW operation. The minimum DNBR(departure from nucleate boiling ratio) on the normal operation and transient condition are about 3.1 and 1.6 respectively, and those are not below 1.5 safety limit. The dynamics-characteristic of core is keep on enough safety with reactor control system. Table 2 show the characteristic of LEU core.

Next, the safety evaluation according with the National Safety Guideline for research reactors showed that the fuel should not failed at any events in both abnormal transient conditions and accidents. But in the site evaluation events, two cases which are one fuel element failure and all fission product release from the core were evaluated, the results showed that JRR-4 site evaluation satisfied the judgment criteria of above guideline under the condition of installation of emergency exhausting system.

3. UPGRADING OF UTILIZATION FACILITY

In JAERI, the utilization of test and research reactors is contributing successfully so that JMTR can be mainly used for irradiation, JRR-3M can be mainly used for beam experiments and multi-purpose irradiation and JRR-4 is mainly used for simple and special experiments and irradiation, respectively. The utilization facility of JRR-4 are two big pool and dry shielding room for reactor shielding experiments, in-core irradiation holes for silicon doping, nucleate activation analyses, RI production, thermal column for special experiments, N-16 gamma ray irradiation system. In addition, a medical irradiation facility will be installed, the activation analyses system is modified, and a large pipe irradiation system is changed. The medical irradiation research in JAERI is conducted thirty-three cases for boron neutron capture therapy against brain cancer at JRR-2 since 1990. But after JRR-2 is terminated in December, 1996, the mission of BNCT is transferred to JRR-4 in reply to the request of medical researchers in national hospitals and universities. The general arrangements of the facility is shown in Fig.5. The irradiation room and the medical treatment room are located on the basement. The cross section of the medical irradiation facility which is shown in Fig.6 is composed with a heavy water tank and a beam experimental hole. The heavy water tank plays role of neutron beam filter with aluminum and heavy water. The heavy water layers can be changed for optimized performance both thermal and epi-thermal beams according to the need of each treatment. The beam experimental hole is composed bismuth and cadmium filters, lead and graphite reflectors, LiF and B_4C covers, LiF collimeter, and polyethylene and lead for shielding materials. As a result of beam design and analysis, the beam performance of facility will be more than two times of JRR-2 to short treatment for thermal neutron beam, furthermore the facility will be use epi-thermal beam for treatment without surgery operation, as is shown Table 3.

A neutron activation analysis system is used to analysis in widely for research of environment, geology, biology and so on, and the system is requested analysis for more shorter life(approximately 1 minute) nuclide recently. Then a new system with automatic analysis by repeating irradiation method is installed.

A large pipe irradiation system is mainly used for silicon doping, the diameter of pipe will be changed from 4 inches to 5.5 inches for a large sample irradiation.

The plan of utilization of new JRR-4 is proposed many experiments, example are a leaser beam instrumentation system, a in-situ measurement irradiation experiment, a neutron radio-graphy experiments, special reactor training.

4. WORKS AGAINST FOR SYSTEM AGING, SEISMIC DESIGN

JRR-4 is constantly operated longer than thirty years, and is used successfully. In those term, some reactor components and systems are remodeled as instrumentation system, thermal heat exchanger, secondary cooling system, ventilation system, but many components have been to keep safety by inspection and repairing for maintenance since the time of first criticality. In the modification works, it will be performed for long continuously operation for future that detailed inspections of core tank, two reactor pools and cooling systems , remodeling of instrument and control system including control rod drive mechanism, installation of emergency exhausting system, duplication of emergency power and so on.

Furthermore the JRR-4 reactor building was designed according to old guideline of the Building Standard Low of Japan in 1960's. Then as a result of new seismic design according to the Examination Guide for Seismic Design of Nuclear Power Reactor Facilities, it was necessary for seismic safety that the remodeling of roof structure, reinforced of some pillars and walls of the building.

5. CONCLUSION

The fuel will be changed a silicide fuel with 3.8 U-g/cc, the new core performance is similar with HEU core its without major change of structure and size of HEU fuel. And the activities of reduced enrichment works will be finished on research reactors in JAERI . The medical irradiation facility will be had higher performance than that in JRR-2, and will be able to use epi-thermal neutron beam too. In addition, the reactor system will be remodeling for safety operation and steady utilization for long term.

The licensing of the LEU core is already permitted by Japan government on September 1996. The JRR-4 modification work will be performed for about two years. After then, the new JRR-4 will be start to use many experiments as a boron neutron capture therapy etc. in 1998.

In the future, JRR-4 is expected to use for a wide variety utilization by many researchers as one of important reactors in JAERI.

6. ACKOWLEDGMENT

This paper represents a summarizing report on the JRR-4 LEU Project which many colleagues and coworkers from various departments have contributed to. Especially the authors wish to thank Mr. E.Shirai and Mr. N.Ohonishi who are former directors of department of research reactors for good direction of this project.

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Table 1 Comparison of LEU fuel with HEU fuel

Item	LEU fuel	HEU fuel
Enrichment, %	19.75	93
Uranium density, g/cc(outer plate)	3.8(1.9)	0.66(0.33)
Specific content of U-235 per element, g	204	166
Number of Fuel plate per element	15	
Fuel meat material	U_3Si_2-Al	UAlx
Cladding material	Aluminum-alloy	
Maximum burn-up ratio, %	50	20
Size of fuel element, mm	80 x 80 x 1025	

Table 2 Characteristic of LEU core

calculated

Item	value	remark
Excess reactivity、 %Δ k/k	max. 11.7	initial core
One rod stuck margin, %Δ k/k	min. -2.3	initial core
Temperature of plate surface,℃	max. 111	3.5MWt
Temperature of fuel meat, ℃	max. 113	3.5MWt
ONB temperature at hot spot,℃	max. 125	3.5MWt
Minimum DNBR	3.1	3.5MWt

Table 3 OBJECTIVES OF BEAM DESIGN FOR BNCT

1. Thermal flux; $\geqq 1 \times 10^9$ n/cm^2/s (Thermal beam Mode)
2. Epi-thermal flux; $\geqq 1 \times 10^9$ n/cm^2/s (Epi-thermal beam Mode)
3. Gamma dose contamination; $\leqq 3 \times 10^{-13}$Gy/n-cm^2
4. Fast neutron contamination; $\leqq 5 \times 10^{-13}$Gy/n-cm^2
5. Size of beam port; 200 x 200 mm
6. Estimated Irradiation time; $\leqq$ 2 h

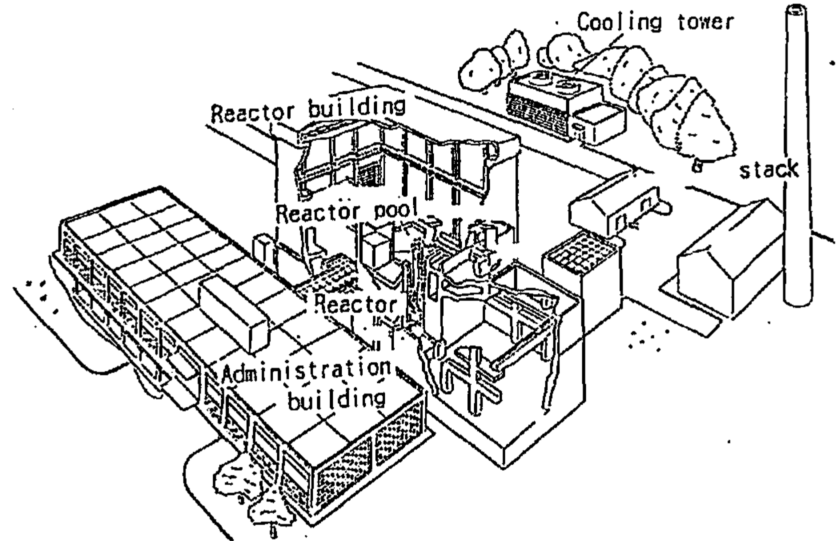

Fig. 1 BIRD'S-EYE VIEW OF THE JRR-4 BUILDINGS

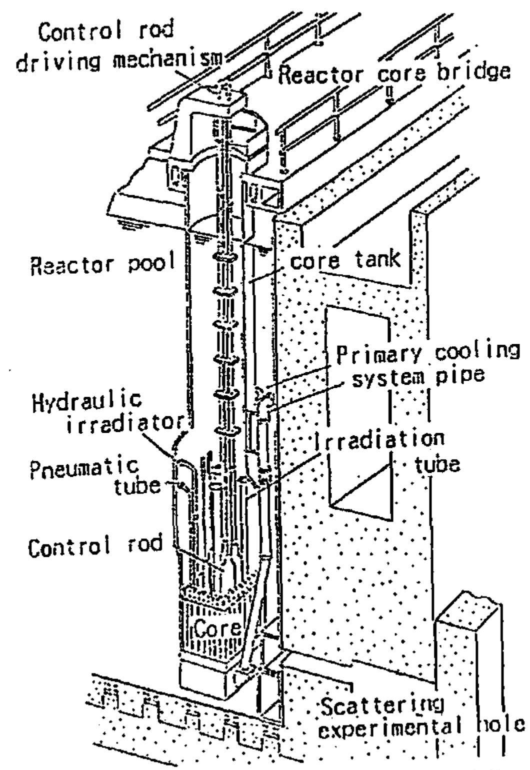

Fig. 2 A VIEW OF THE JRR-4 REACTOR

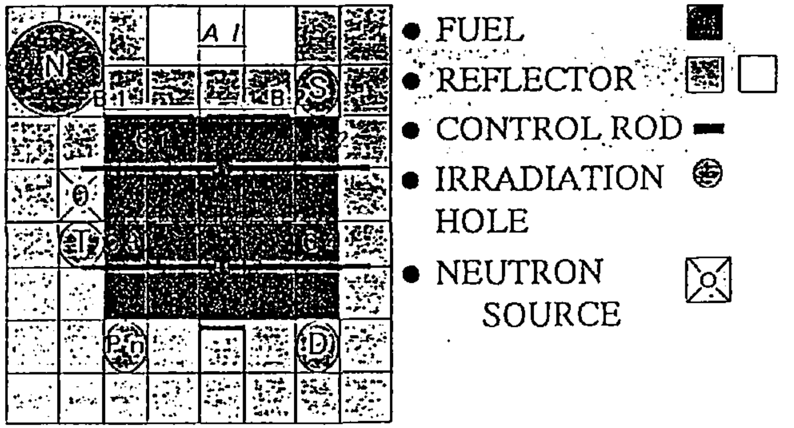

Fig. 3 JRR-4 CORE ARRANGEMENTS

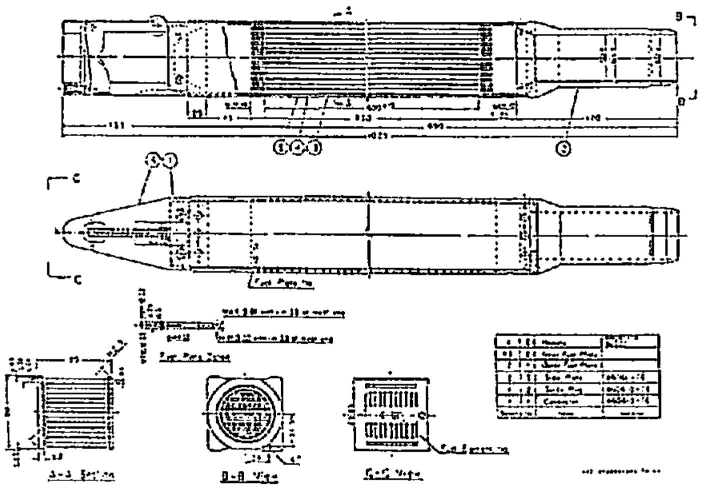

Fig. 4 JRR-4 FUEL ELEMENT

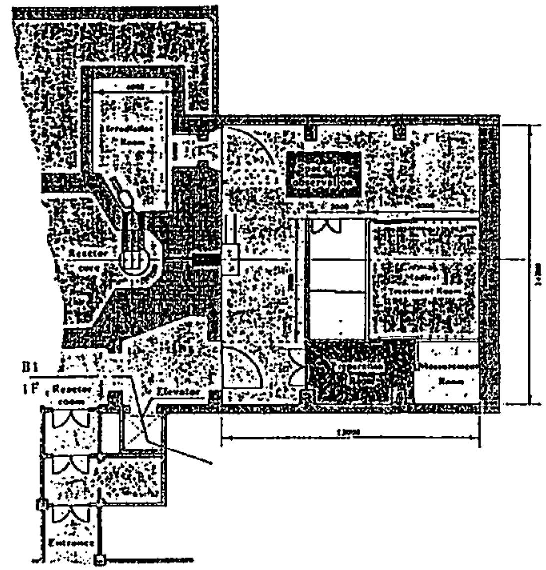

Fig. 5 GENERAL ARRANGEMENTS OF A NEW MEDICAL IRRADIATION FACILITY

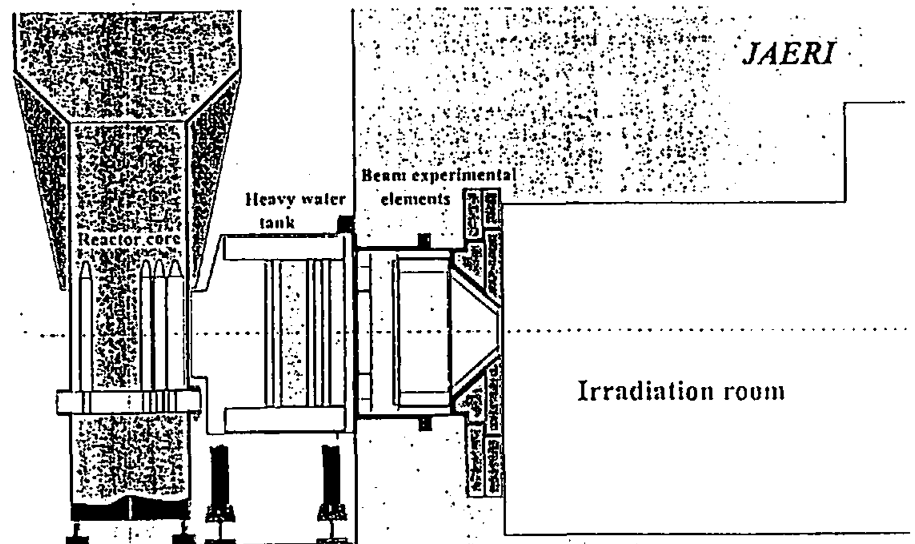

Fig. 6 CROSS SECTION OF MEDICAL IRRADIATION FACILITY

OVERVIEW ON THE MAIN ENGINEERING WORKS PERFORMED ON FRENCH RESEARCH REACTORS THESE LAST YEARS

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H. GUYON (CEA-SILOE)

Introduction

From the beginning of eighties, design, construction and commissioning of new neutron sources like research reactors have severely slowed down in western developped countries as well as worldwide.

On the contrary several facilities have been shut down and some of them decommissioned.

In France, disregarding reactors dedicated to safety studies and very low powers reactor (critical assemblies and training reactors) only four facilities are still operated, namely OSIRIS and SILOE for technological irradiations and multipurposes and HFR and ORPHÉE for fundamental research.

Abroad, the situation is not better : new projects are rare nowadays and subject to well known difficulties surch as lack of budget, green opposition, enrichment of fuel...

In this general framework what are the remaining activities for engineering companies specialized in the nuclear field ?

The purpose of this paper is to review the main engineering works performed in France on research reactors during these last years in strong cooperation between operators (CEA, ILL) and engineering companies (Technicatome).

Main factors involving engineering works in the field of RR

When facilities are under operation, operation teams have enough well qualified and trained staff to perform the operation of the reactor, itself its current maintenance as well as minor refurbishing and adaptations/modifications of limited importance.

But some factors necessitate more important works and the, operators have to call to engineering companies and cooperate strongly with them for the preparation, the execution and the tests of the interventions to be done on the facility.

The main factors involving such engineering help to main modifications/refurbishment are :

- ageing of main equipment and materials,

- the obsolescence of equipment and the apparition of new technologies (mainly I & C equipment),

- the evolution of safety considerations and rules, which may involve some adaptations of the facility,

- new projects and modifications necessitated by the evolution of research programs

and last but not least :

- the decommissioning questions.

The evolution of safety considerations and rules

As seen hereabove the main research reactors still operated in France have been designed from the early sixties (SILOE, OSIRIS) to the end of seventies (RHF, ORPHÉE). From this time safety considerations have moved towards more and more strictness and rigorism resulting that for each one of this reactors some design feature would not be nowadays accepted by French safety authorities.

For example :

At Siloé, the first design and the execution (1963) of the reactor pool and of the working pool which were ceramic lined did not allow the complete monitoring of the second barrier : checking eventual leakage through the bottom of pools was not possible. This point was put right with the reactor stainless-steel pool erected in 1988 and then, on the working pool which was stainless-steel coated in 1995.

Consequently some adaptations and/or modifications are made on the facilities taking the opportunity of major refurbishments or works where there is no actual urgency, in order to improve the safety level and to be in accordance with modern safety considerations.

There are, in France, only few written rules that are related to research reactors ; howerer these documents are worth to be quoted here because there were progressively put in applications in existing facilities these last years and involved some works :

- L'arrêté du 10 Aout 1984 relatif à la qualité de la conception (Design phase), de la construction (Construction phase) et de l'exploitation (operation phase) des installations nucléaires de base (basic nuclear installations),

- Fundamental safety rule on the installations on ventilation filters RR1 (August 4th, 1986),

- Fundamental safety rule on fire protection in research reactors RR2 (July 2nd, 1991).

Conclusions

To conclude this general overview of some of the more significative works performed these last years on the French research reactors the following can be underlined :

In France, the strong cooperation, and the mutual assistance between reactors operators and engineering staff allows :

for one part to provides operators of CEA the necessary assistance for important refurbishment and renovation works which overpass the job of reactor current operation and ordinary maintenance. On the other side the engineering staff takes advantages of these many and strong contacts with operation teams to incorporate operating experience feedback in new projects.

The number and the variety of interventions on French operating research reactors allow to maintain them in very good conditions, to adapt them to the evolutive research programs and needs and, last but not least, to improve their safety level, even for a reactor that is to be shut-down in a near future (like SILOÉ at the end of 1997).

So, the challenge of the future French RJH research reactor (design, construction, operation) can be considered with a high confidence, relying on qualified teams.

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MAIN FACTORS INVOLVING ENGINEERING WORKS

- AGEING OF MAIN EQUIPMENT & MATERIAL
- OBSOLESCENCE OF EQUPMENT, NEW TECHNOLOGIES
- EVOLUTION OF SAFETY CONSIDERATIONS
- NEW PROJECTS
- DECOMMISSIONING

MAIN CHARACTERISTICS

	SILOE (CEA)	OSIRIS (CEA)	HFR (ILL)	ORPHÉE (CEA)
First criticality	1963	1966	1971	1980
Power	35 MW	70 MW	57 MW	14 MW
Max. Flux th (10^{14} $n.cm^{-2}.s^{-1}$) f	4.4 4.5	4 4.5	15 (reflector) 5	3 (reflector)
Fuel	U.Al (93%)	U3 Si2 (19.75%)	U.Al (93%)	U.Al (93%)
Moderator	H2O	H2O	D2O	H2O - Be
Reflector	H2O, Be	H2O, Be	D2O	D2O
Coolant	H2O	H2O	D2O	H2O
Utilisation	Polyvalent : • Technological irradiation • RI production • Silicon doping • 4 neutron beams	Polyvalent : • Technological irradiation • RI production • Silicon doping • Neutron radiography	Fundamental research • 17 beam tubes • 2 cold sources • 1 hot source	Fundamental research • 9 neutrons beam tubes (20 beams) • 2 cold sources • 1 hot source • Silicon doping
Scheduled Shut-Down	End 1997			

AGEING OF MAIN EQUIPMENT

SILOE 1963	OSIRIS 1966	HFR (ILL) 1971	ORPHEE 1980
• <u>Reactor pool</u> (Ex : 1986/88) Ceramic liner → SS liner • <u>Neutron beams</u> (Ex : 1986/88) • <u>Auxiliary Pool</u> (Ex : 1993/94) Ceramic liner → SS liner • <u>Heat exchangers</u> (Ex : 1992/93) Replacement	• <u>Decay tanks</u> (ex: 1994/95) Pool cooling and core cooling (painted C.S.) • <u>Primary cooling, core exit main pipe</u> (Ex : 1994) Repairing • Auxiliary canals (St : 1995) • <u>Cellular core structure</u> Replacement (Ex : 1996/97)	• <u>Reactor block changing</u> (and all in-pool primary circuit equipment : heavy water collectors, connecting sleeves....) • expertise (Ex : 1991) • fabrication and erection (Ex : 1992)	• <u>Zircaloy core housing</u> Dismantling of the old/one (St 1996) Erection of a new one (St 1997) • Auxiliary water circuit (Ex : 1996) - arrangements - new air-coolers

Ex : Execution St : Study

EVOLUTION (IMPROVEMENT) OF SAFETY

SILOE 1963	OSIRIS 1966	HFR (ILL) 1971	ORPHEE 1980
• **New stainless steel reactor pool** Borax resistant : with shock absorbers at the bottom	Physical separation of different sections of low voltage		**Ventilation**
			• Improvements of installation of iodine traps on nuclear ventilation exhaust circuit
• **Nuclear measurement** : Geographical separation of redundant cable trays			• Installation of a test equipment for the period control of iodine traps
• **Logic of vote (RPS)** Improved performances			
• **Fire protection** Expertise • Siloe (1993) • Siloette (1995)	• **Fire protection**	• **Fire protection** (Expertise : 1992)	
	• Emergency control room (Ex : 1996)		

Ex : Execution St : Study

OBSOLESCENCE OF EQUIPMENT NEW TECHNOLOGIES

SILOE 1963	OSIRIS 1966	ORPHEE 1980
<u>I & C Equipment</u>	<u>I & C Equipment</u>	<u>Health physics equipment</u> St : 1994
• Reactor control processing system (Digital equipment) Ex : 1992/93	• Reactor control processing system (Digital equipment) Ex : 1992	
• Nuclear measurements (Digital technology : SIREX type) Ex : 1992/93	• Nuclear measurements (SIREX type) and RPS system Ex : 1992	
• New logic of vote and RPS St : 1995 Ex : 1996	• <u>Electrical</u> low voltage supply network Ex : 1989-1996	• <u>New cold neutron source (CNS)</u> Cylindrical type
• Electrical low voltage supply network Ex : 1989/94	• <u>Mixed bed resins</u> without regenerations Ex : 1994	• <u>Upgrading of neutron guides</u> Supermirrors
• Water Purification system Ex : 1992/93		

Ex : Execution St : Study

NEW PROJECTS

ISIS	Upgrading of power and feasibility of adaptation of the reactor for BNCT Application (1990)
ORPHEE +	Implementation of new cold guides on the beam port 4F : increase by about 30% the number of beam positions
SIRIUS	Preliminary design of multipurpose reactors derived from SILOE.
	SIRIUS 2 : 15 to 30 MW
	SIRIUS 3 : 5 to 10 MW
RES	New reactor for training of crews for submarines/air-craft carrier
REACTOR JULES HOROWITZ	New French Research Reactor for Irradiation

DECOMMISSIONING

(RESEARCH AND EXPERIMENTAL REACTORS)

PEGASE	(CADARACHE)	Engineering study & following up of works	1975-80
EL3	(SACLAY)	Engineering study & following up of works	1979-80
TRITON & NEREIDE	(FONTENAY AUX ROSES)	Engineering study & following up of works	1982-89
RAPSODIE	(CADARACHE)	Engineering studies	1982-90
EL4	(BRENNILIS)	General studies execution	1985-87 1995-96
MELUSINE	(GRENOBLE)	General studies	1994-95
SILOE	(GRENOBLE)	Preliminary studies	1990

others reactors : G1, G2/G3, VANDELLOS, SNLE

UTILIZATION OF THE BUDAPEST RESEARCH REACTOR

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ABSTRACT

The Budapest Research Reactor, the first nuclear facility of Hungary was first put into operation in 1959. The reactor operated for 27 years without any safety problem; it provided the country with neutron research possibility, supplied it with radioactive isotopes and served as a basis for training as well. After a major reconstruction and upgrading the start-up procedure began in 1992. The upgraded reactor serves for: basic and applied research, technological and commercial applications, education and training: (e.g. involving the IAEA). The reliability of the reactor is very good as e.g. in the 1995 operation period only two unexpected shut downs occurred. The main goal of the reactor is to serve neutron research, but applications as neutron radiography, radioisotope production, pressure vessel surveillance test, etc. are important as well. The neutron research will get a much improved tool, when the cold neutron source will be put into operation. The start-up of the cold neutron source is scheduled for late 1998.

Background

The research reactor in Budapest was in operation from 1959 to 1986. No incident occurred during the 27 years of reactor operation. In this period the reactor played an essential role in establishing nuclear research and technology in Hungary. It served as a basic facility for neutron scattering, nuclear and particle physics, radiochemistry, shielding investigations; for establishing nuclear medical applications providing radioisotopes; for performing pressure vessel surveillance programme for reactor safety studies; and it was an important school of university and postgraduate training.

The full scale reconstruction and upgrading project started in 1986 [1], aiming the substitution of aged components, the enhancement of reactor safety, the increase of reactor power to 10 MW. The reactor vessel and the primary piping turned out to be much less corroded than previously assumed. Reactor safety has been enhanced by adding some new safety systems and thus satisfying the "Defense in Depth" concept and also by applying more up-to-date and reliable systems. A new safety analysis report taking into account all relevant recommendations of IAEA has been prepared. The increase of reactor power was facilitated mainly by building new cooling towers, permitting a reactor power of 20 MW. However, the characteristics of the VVR-SM fuel do not allow a reactor power higher than 10 MW and the higher power can be achieved only by applying a new type of fuel elements.

Short Technical Description

The Budapest Research Reactor [2,3] is a tank type reactor, moderated and cooled by light water. The reactor is in a cylindrical reactor tank, made of a special aluminium alloy. The diameter of the tank is 2300 mm, the height is 5685 mm. The heavy concrete reactor block is situated in a rectangular semi-hermetically sealed reactor hall. The area of the reactor hall is approximately 600 m^2. It is ventilated individually.

The fuel of the research reactor is of the VVR-SM type (Russian product). It is an alloy of aluminium and uranium-aluminium eutectic with aluminium cladding. The uranium enrichment is 36%, the average U-235 content is 39 g/fuel element. The fuel elements contain three fuel tubes, the outer tubes are of hexagonal shape, while the two inner ones are cylindrical. The active length of fuel elements is 600 mm.

The equilibrium core consists of 223 fuel elements, with a lattice pitch of 35 mm. The core is surrounded radially by a solid beryllium reflector. The reactor is equipped with boron carbide safety and shim rods. There is a stainless steel rod for the purpose of automatic power control.

The reactor can be characterized by the following main technical data:

thermal power: 10 MW
mean power density: 61.2 kW/litre
approx. maximal thermal flux: 2.5 x 10^{14} n/cm^2s
approx. maximal fast flux: 1.0 x 10^{14} n/cm^2s
cooling water inlet temperature: 54°C
maximum cooling water outlet temperature: 60°C

The reactor has 10 horizontal beam tubes (8 radial and 2 tangential). Irradiations may be carried out by inserting samples into the 51 special vertical channels. The reactor staff has a long experience in assisting physical experiments and radioisotope production.

The reactor is supplied with a reasonable amount of fresh fuel for at least 6 - 7 years (depending on the future demands for annual operational hours), so this problem is solved up till the first years of the next century.

Operation

The reactor was operated according to the pre-determined timetable throughout its operation period. These timetables have always been revised and modified according to the changes in the demand.

The timetable was last modified for the year 1995. One reactor operation period was 11 days, which meant 270 hours of continuous operation and one cycle consists of 4 - 6 such periods. One cycle was 44-66 effective days, which was followed by a refuelling period of two weeks. The 1995 timetable was planned for 4476 operational hours for the year and this amount was exactly fulfilled. The total energy production was 1883 MW days. The reliability of the reactor was very good as in the 1995 operation period only two unexpected shut downs occurred. The equilibrium core size was reached by the end of the year 1995.

The timetable for 1996 has the same structure as that for 1995.

The Use of the Reactor

Use in research

The main goal of the reactor is to serve neutron research.

Investigations based on neutron scattering are essential in studying the condensed matter structural properties. These studies will get a much improved tool, when the cold neutron source will be put into operation (in 1998).

Investigations of biological objects (models or in vivo) affected complex irradiation from the reactor, dosimetric calibrations, development of chemical dosimeters, etc.

Besides the basic research some problems for possible applications are foreseen, two of them are mentioned here briefly:

- It is of great interest to investigate segregation processes, e.g. determining crystallite orientations by texture analyses and carrying out internal stress analysis by high resolution lattice parameter measurements.

- A very practical use can be the aging calibration of turbine blades by small angel neutron scattering and neutron diffraction.

The use of the reactor can not be considered as a finished matter, further ideas are welcome, a limited of excess space is still available.

Use for practical purposes

A lot of technical problems can be solved by means of the reactor[4]. These problems are radioisotope production, neutron radiography, activation analyses, and pressure vessel surveillance studies. Some other applications, as e.g. silicon doping, might be of some interest as well.

The production of radioactive isotopes can be considered, as one of the main applications of the reactor. The production restarted in 1993, 1994 was the first full year of isotope production. The quantity and variety of the isotopes produced increased during 1995, but a slight progress is still expected.

Neutron radiography is a well accepted method. Collimate neutron beam is used to investigate objects in closed volumes. In dynamic radiography moving objects or processes can be recorded. A new channel was constructed and put into operation in 1995, devoted to static radiography, where the resolution is better, than in the dynamic one, so the two radiographic procedures can now complete each other, providing the users with a full-possibility radiography. The current use of radiography is mainly related to refrigerators and fire extinguishers, but other fields, e.g. combustion engine investigations are foreseen as well. Two horizontal channels are used for neutron radiography, one for the dynamic investigations and the other for static pictures.

Activation by reactor neutrons is a very sensitive analytical method. About 70 various chemical elements can be detected in an extremely wide range of content, i.e. from percents to 10^{-10} g/g concentrations. A pneumatic rabbit system has been constructed to serve the laboratory of activation analysis. The laboratory is capable to analyze 100 - 150 samples per week. The moderate demand of users (e.g. medical industry) has been satisfied by neutron activation analyses; marketing activity was started, it may increase the use of this method significantly.

An extended national programme for the surveillance of the power plant's pressure vessels is going on. As regions, where the flux is higher, than the flux at which the pressure vessel of NPPs is exposed, can easily be found in the research reactor, the neutron induced embrittlement of 20 - 30 - 40 years can be studied after a few month irradiation. The possibilities will be extended by a heatable irradiation channel, providing with better simulation of the conditions of a power reactor pressure vessel. This channel will be put into operation in 1996.

Cold neutron source

The installation of a cold neutron source equipment is foreseen at one of the tangential beam tubes. The project of the cold neutron source is partially sponsored by the IAEA. Financing problems seem to be solved now, so there is real hope, that the cold neutron source may start to operate in September 1998. The cold neutron source will extend the use of the reactor, especially in the scientific field.

The cold neutron source will be the liquid hydrogen type. The moderator cell is relatively small, i.e about half litre volume. The relatively low estimated heat release (about 100W) makes feasible the direct cooling of the condensed hydrogen in a double walled moderator cell by helium gas (12 - 14 K).

The preliminary safety report of the cold neutron source is being prepared now. If this report will be accepted by the Hungarian regulatory body, construction works may start early 1997 and could be finished by August 1998. The scheduled start of the operation of the source is September 1998.

Organisation

The Budapest Research Reactor is operated by the KFKI Atomic Energy Institute (AEKI), which is responsible for reactor safety and utilisation as well. The institute is prepared for any reasonable cooperation.

For the utilization of the reactor in the field of basic research the Budapest Neutron Centre (BNC) has been set up, by three research institutes: KFKI Atomic Energy Research Institute, KFKI Research Institute for Solid State Physics and the Isotope Research Institute of the Hungarian Academy of Sciences. The BNC has an international scientific advisory council.

The utilization of the reactor for practical applications is organized by AEKI as well, it can be considered in international cooperation.

International Relations

The Budapest Research Reactor is considered as Centre of Excellence by the Central European Initiative. The international user system for neutron research will be started in the near future.

The co-operation between the operator of the Budapest Research Reactor (AEKI) and IAEA is complex and important. IAEA has sponsored the reconstruction (providing the Beryllium) and sponsors the construction of the cold neutron source. The safeguards is naturally controlled by the staff of the IAEA. The co-operation might be extended: e.g. training courses organized by IAEA could be performed at the reactor.

Personal contacts between the management of the Budapest Research Reactor and of other research reactors in Europe (Řež, Seibersdorf, Ljubljana, Petten, Risø, etc.) are excellent and beneficial.

Many co-operations are organized on bilateral basis, e.g. some pressure vessel surveillance investigations are performed in a co-operation between PSI (Switzerland) and AEKI.

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Design Modification of HANARO Reflector Cooling System

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ABSTRACT

HANARO, 30MWth is a light-water-cooled and heavy-water-reflected research reactor. The Reflector Cooling System(RCS) circulates the heavy water from the reflector vessel through the heat exchanger then back to the bottom of the vessel. Two different features are provided to protect the overpressure of the reflector vessel. One is the concept of the relief valve for air vent activated by a relatively small amount of pressure build-up during normal operation. The other is a rupture disc for large volumn release in case of the accident condition that the system pressure abnormally exceeds the predefined limit.
During the system function test, an unexpected transient peak pressure was detected when the operating pump was stopped. It was found out that this pressure peaking came from the water hammering phenomena in the piping and the overpressure protection mechanisms could not act, as intended against the transient peak pressure.
To resolve this problem, we introduced an additional extension pipe at the position of the rupture disc to give more room for absorbing the peak pressure from the water hammering. The test result for the modified system indicates that the transient peak pressure was almost absorbed into the air-cushion inside the extension pipe newly added. This paper concentrates on the basis of the design modification and the system behaviors before and after the implementation of the new mechanism.

1. Introduction

The purpose of the Reflector Cooling System(RCS) is to cool and maintain the purity of the heavy water which acts as the reflector in the annular cylindrical tank. The RCS consists of two 100% circulating pumps, a heat exchanger, two ion exchanger columns and all necessary piping and instrumentation as shown in Figure 1.

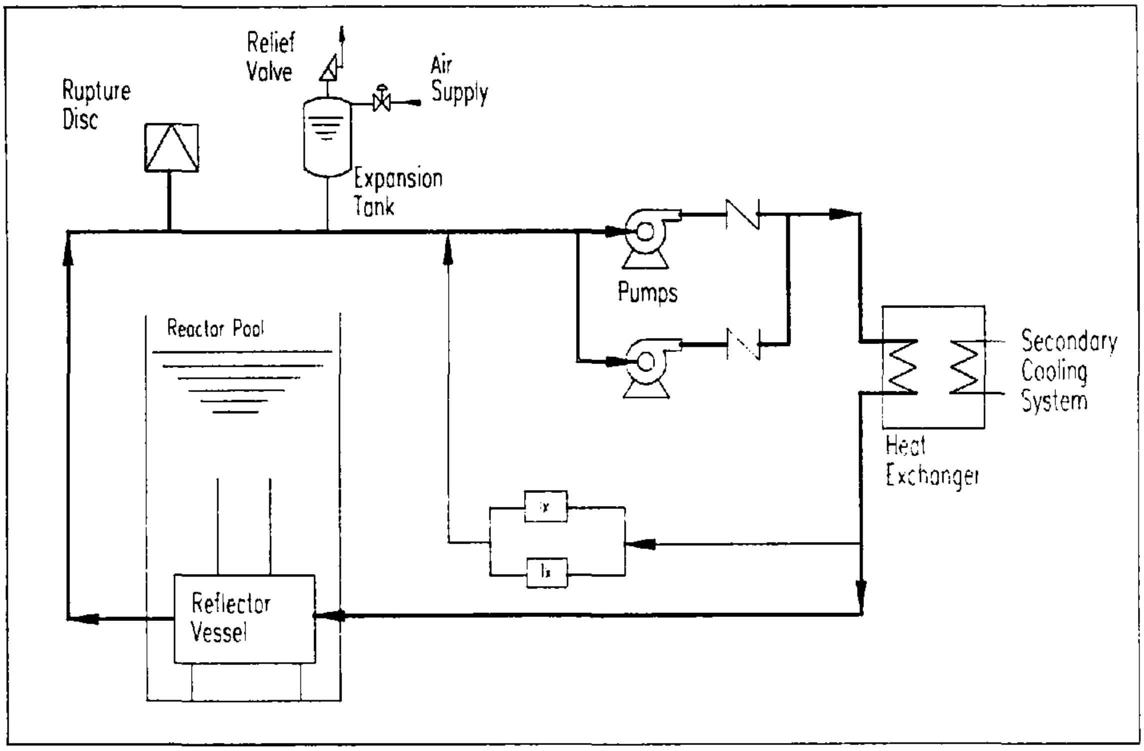

Figure 1. Flow Diagram for Reflector Cooling System

The system circulates the heavy water(D_2O) at a normal flow of about 43.2 l/s dissipating up to 2.5 MW of heat to the heat exchanger. A small bypass flow 0.1 l/s is taken through one of two ion exchange columns for purification.

An expansion tank is provided to control the system pressure coping with the volume change of D_2O and to purge D_2 gas from the heavy water. The top area of the expansion tank is filled with the controlled cover gas supplied by compressed air system. The internal pressure of the reflector vessel should be maintained below the design limit even during the anticipated transients such as loss of reflector circulation, loss of secondary cooling system, etc.

The overpressure protection for reflector vessel is provided by two mechanisms, a relief valve and a rupture disc. A relief valve is installed at the top of the expansion tank. It is an 1 1/2 inch glove type and the pressure set point is 35 kPa(g). A 6 inch rupture disc is installed at the pump suction line. According to the process design, the internal pressure of the reflector vessel is 134.9 kPa(g) in normal operation. The reflector vessel was tested over this pressure at the manufacturing shop. Then the maximum allowable pressure in the vessel is 245.1 kPa(g) considering the hydrostatic pressure of pool water, 110.2 kPa(g). The rupture disc limits the internal pressure of the reflector vessel below the maximum allowable pressure and its setpoint for burst was decided to 180 kPa(g) through RELAP5/KMRR simulation which is a computer code to analyze the thermal hydraulic behavior of RCS under the anticipated and postulated transients.

During the system function test, an unexpected transient peak pressure was detected when the operating pump was started and stopped. It was found out that the overpressure protection mechanisms could not absorb the transient peak pressure from the water hammering phenomena during pump stopping and starting. In design stage the RCS pump was selected with bellows mechanism to reduce D_2O leakage to the minimum. The RCS pump has a characteristic of a very small inertia. It means that the pump stops immediately and then acts like a check valve when the electric power is failed. The maximum pressure rise at the pump suction depends on the rate of decrease of the flow and the pressure wave reflections.

The RCS volume change was identified due to deflection of the heat exchanger plates. The maximum peak pressure was over 180 kPa(g) of setting level of rupture disc burst. Therefore, it is needed to provide any scheme for absorbing the transient peak pressure. At that time, the field works for example, pipe cutting and welding, were very limited to maintain the cleanness of the RCS and reflector vessel. Finally, we selected an additional stand pipe at the point of rupture disc with flange connection and carried out the design modification and the experiment.

2. Design Modification

The pressure variations were observed on pump start and stop operation. With the RCS pump operating, the RCS pressure of 344 kPa(g) at the heat exchanger(HX) exceeds the Secondary Cooling System(SCS) pressure of 263 kPa(g). With the RCS pump stopped, the RCS pressure at the HX of about 24 kPa(g) is lower than that of the SCS. When the RCS pump stops, the higher SCS pressure deflects the plates to reduce the D_2O volume in HX. The rapid volume change cannot be accommodated quickly enough when pump operation mode changed, because the small pipe to the expansion tank has high flow resistance. The cause of pressure transients is identified as water hammer wave with rapid RCS volume changes due to deflection of the heat exchanger plates when the pressure differential between the RCS and SCS is changed.

In order to relieve a maximum pressure rise on reflector cooling system, we added a length of 6 inch pipe to the existing rupture disc flange which raises the rupture disc location 5.275m above its previous location. The 6 inch standpipe will act as additional expansion tank to provide a fast source of D_2O supply on pump startup and to remove D_2O quickly from the existing RCS loop on pump stop.

The Figure 2 shows outline of the design modification on reflector cooling system. The cover gas system has a function to purge the D_2 gas concentration from radiolysis except giving a space to allow volumetric increase of heavy water from the temperature variation of RCS. The D_2 gas concentration in the standpipe cannot be measured by the spectrometer measurement of the expansion tank. The original system design for purging D2 gas in the expansion tank can not be applied to the new standpipe. Therefore, an air connection to the standpipe for purging D_2 gas concentration should be provided. It was done by connecting the air supply and exhaust line to the stand pipe as shown on the Figure 2.

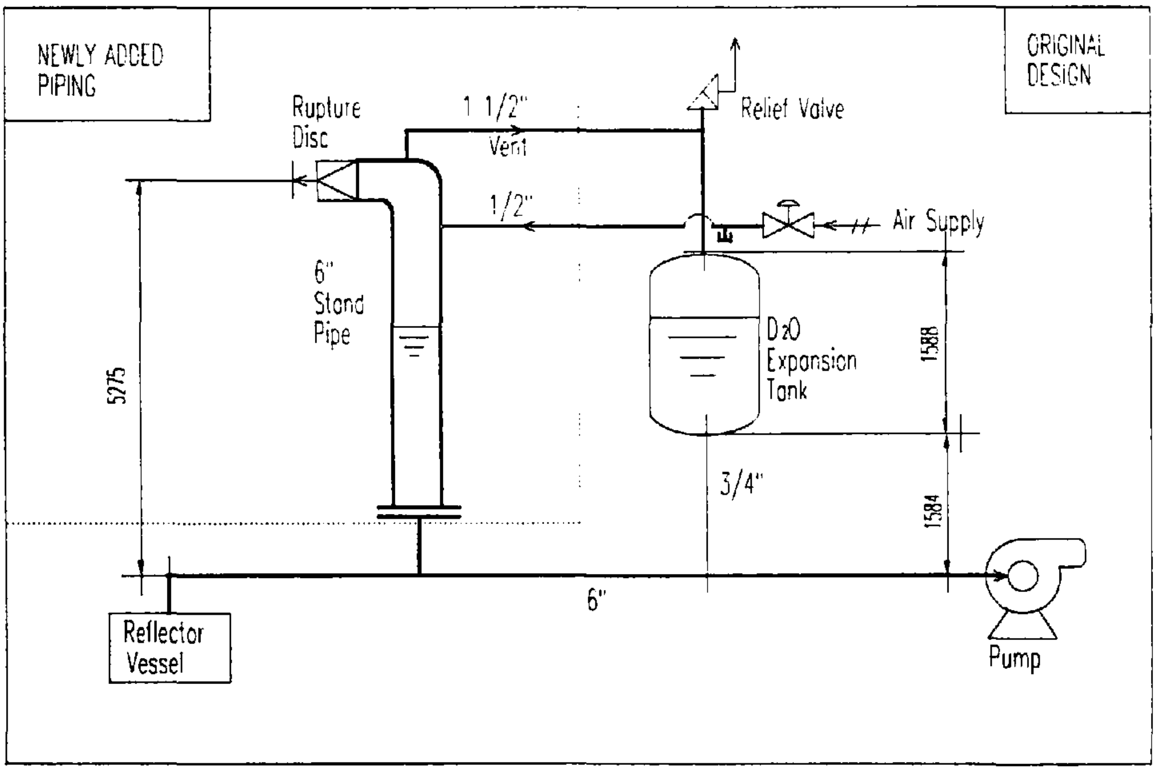

Figure 2. Overview of Design Modification

The purge air enters the standpipe and flows through the 1½ inch pipe which is provided from the top of standpipe to the expansion tank. Then it is exhausted to the ventilation duct. Since the flow goes through both the standpipe and the expansion tank, all of the air space can be purged.

3. Measurements And Results

3.1 Pressure Measurements

A 6 inch rupture disc was originally installed at the pump suction line between the reflector vessel and the branch of the expansion tank as shown in the previous drawings. Later on, 5 meter length of 6 inch upright standpipe was newly installed at the previous location of the rupture disc. Accordingly, the rupture disc moved to the higher position as much as the length of the standpipe. The pressure measurements were achieved under two different conditions-before and after the addition of the standpipe. A high resolution pressure transducer and a high speed X-Y recorder were used for .signal recording.

The pressure transducer was connected to the pressure tap provided at the pipe cap which was temporarily installed at the previous position of the rupture disc. After the installation of the standpipe, the pressure transducer was mounted on the same elevation as the rupture disc was originally installed. The pressure measurements were carried out in the following manner :

1) As a normal operating condition, two Secondary Cooling System(SCS) pumps are being operated during these measurements.
2) One out of two RCS pumps, pump No.1 starts.
3) After the stabilization of the system flow, the pump No.1 is intentionally turned off to simulate a pump failure. (Pump No.2 is at standby position.)
4) A few second later, the standby pump then starts automatically by the transfer control logic.
5) Finally, the pump No.2 stops.

3.2 Results and Analysis

The Figure 3 shows the pressure transient behaviors at the moment of the pump startup and stop in the sequences previously mentioned. For comparision purposes, two separate measurements-before and after the design modification are plotted on the same graph. The dotted line indicates the pressure measurement before the system modification, while the solid line means the measurement after the addition of the standpipe. As shown on the graph, on each pump start the pressure at the rupture disc location, which is pump suction side is abruptly decreased to -84 kPa(g) then recovered to the steady state in about 10 seconds. The cause of this negative pressure peak has been identified as rapid RCS volume changes. Before the RCS pump is started, the SCS pressure exceeds .the RCS pressure and the heat exchanger plates are deflected to reduce the RCS volume. As soon as the pump starts the RCS pressure now increases above the SCS pressure so deflects

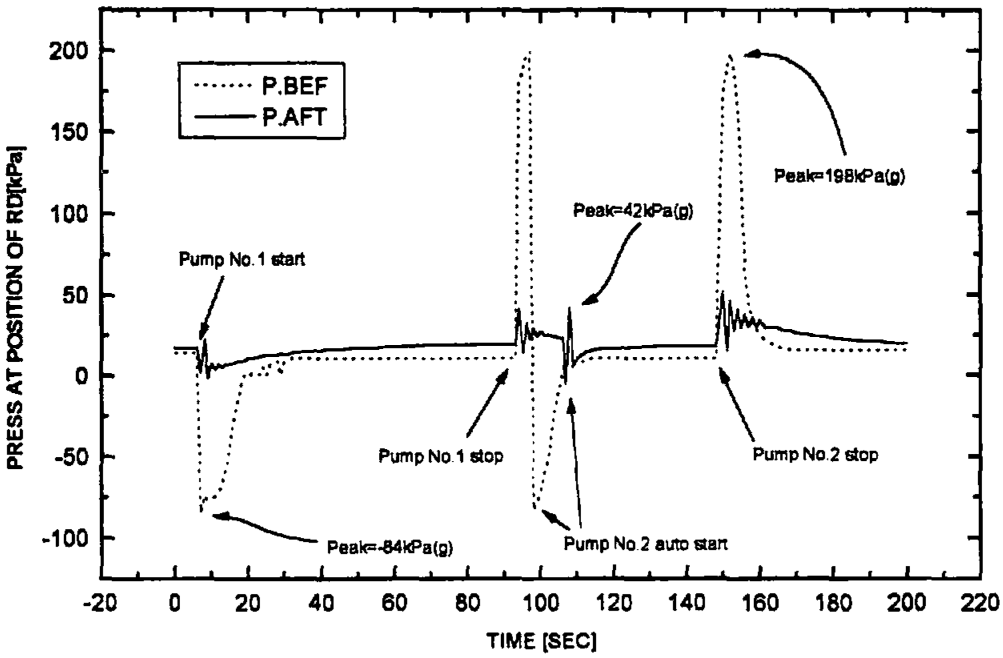

Figure 3. Pressure Transients Behaviors on RCS Pump Startup and Stop

the heat exchanger plates in the direction of increasing the RCS volume. This generates the negative pressure peak which lasts until the additional heavy water to accommodate the volume changes has been transferred from the expansion tank. Although the heat exchanger plates are thin and flexible and can be deflected within the first second by the pressure differentials, the small size and relatively long length of the connecting pipe to the expasion tank brought that it took more than 10 seconds for the required transfer of the heavy water to occur. On the contrary, when the RCS pump stops, the higher SCS pressure deflects the plates to reduce the RCS volume rapidly, which results in the positive pressure peak at the position of the rupture disc. According to the Figure 3, the measured peak pressure reached about 198 kPa(g) which is higher than the setpoint for bursting the rupture disc. Within few seconds, the standby pump starts by the automatic transfer control logic. Again, the heat exchanger plates deflect in the direction of increasing the RCS volume similarly to the previous case.

It is concluded that the change in sign of pressure differentials across the plates of the heat exchanger created very fast volume changes in the RCS. Since the absorption of the volume changes by flowing into or out of the expansion tank is

much slower than the heat exchanger plate can deflect, this causes the pressure decreases of up to -84 kPa(g) on RCS pump startup and the pressure increases of up to 198 kPa(g) at the position of the rupture disc.
After the implementation of the 6 inch standpipe at the previous rupture disc location, the pressure measurements were carried out in the same manner as we did before the design change. As indicated in the Figure 3, the magnitude of the pressure transients on pump startup and stop was significantly reduced. The positive peak is just about 42 kPa(g) and the negative peak is almost disappeared. This is achieved by permitting rapid inventory changes into or out of the standpipe instead of the expansion tank. That is, the 6 inch standpipe acts as another expansion tank to provide fast heavy water supply from the standpipe to the RCS loop on pump startup and to remove heavy water quickly from the RCS loop to the standpipe. This flow path has large area and low flowing resistance to reduce the pressure fluctuations due to the rapid volume changes that could not be managed by the existing long, small size connection line to the expansion tank.

4. Conclusion

Based on the test result with modified system of Fig. 3, the maximum peak pressure at the rupture disk position is 42 kPa(g) instead of the 198 kPa(g) before modification.
In conclusion it is verified that the additional expansion pipe can accommodate most transient peak pressure from water hammer wave and volume change due to pump stop and start and that the rupture disc is not blasted under transients condition on normal operation.

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IAEA ACTIVITIES ON RESEARCH REACTOR SAFETY[1]

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1. INTRODUCTION

IAEA activities on research reactor safety are determined by its statutory functions and responsibilities and on the current situation of research reactors around the world. In particular, they respond to the current situation of the 274 research reactors in operation in 58 countries around the world as per August 1996. General objectives of these activities are:

(a) To complete a set of safety publications, including Safety Standards, Safety Guides and Safety Practices[2], that would cover all safety aspects related to reactor siting, design, commissioning, operation, utilization and modification, and which would provide guidance to reactor management and regulatory supervision.

(b) To provide IAEA Member States with assistance, including training and to implement the IAEA Safety Standards and Safety Guides for research reactors to foster and enhance their safe operation.

(c) To collect and assess information on operational experience in research reactors by creating and maintaining a database on safety-related unusual events and to disseminate relevant information to Member States.

In accordance with the above objectives, these activities are grouped in the following three projects within the IAEA Programme and Budget for 1995-96[3]:

(1) Development of Safety Guidance for Research Reactors
(2) Integrated Safety Assessment of Research Reactors (INSARR) Services
(3) Incident Reporting System for Research Reactors (IRSRR): Collection of Data on and Systematic Analysis of Safety Relevant Events.

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[1]To be presented at the International Group on Research Reactors (IGORR) 5 Conference, Aix-en-Provence, France, 4-6 November 1996.

[2]This category may be discontinued in the future and replaced with Safety Reports in the new IAEA Safety Report Series.

[3]For the forthcoming two-year period 1997-98, these activities will be re-grouped in the following three projects:
(1) Research Reactor Regulatory Issues and Safety Assessments;
(2) Research Reactor Operational Safety; and
(3) Research Reactor Safety Advisory Services.

The research reactor statistics shows the following:

(a) The total number of operating research reactors in the world is decreasing, but the number is remaining constant for the group of developing countries;
(b) The number of developing countries operating research reactors is double that of industrialized countries;
(c) Ageing of research reactors can be considered an important concern;
(d) Many activities (safety reviews, in-service inspections, modification projects) are carried out to extend the life of research reactors; and
(e) Some new research reactors are planned or under construction.

These features determine to some extent the above programme, whose basic characteristics are development of documents, provision of technical advice and assistance, and fostering the exchange of operational experience.

2. DEVELOPMENT OF SAFETY GUIDANCE FOR RESEARCH REACTORS

In accordance with its statutory function of establishing safety standards and providing for their application,[4] the IAEA has developed a significant number of standards publications on nuclear safety.

The former Safety Series No. 35, Safe Operation of Research Reactors and Critical Assemblies, 1984 edition, while establishing general requirements and providing practical guidance on safe operation, did not deal with many other aspects which arose during the design, commissioning, licensing or modification of research reactors and which influence safety. To remedy this, basic principles and requirements for the safety of research reactors and critical assemblies were compiled in two Safety Standards, one on design and one on operation (Safety Series No. 35-S1 and No. 35-S2). These standards, which supersede Safety Series No. 35 of 1984, also include basic requirements for siting, quality assurance and regulatory control of research reactors. These two Safety Standards accompany the IAEA Safety Fundamentals, the Safety of Nuclear Installations (Safety Series No. 110), and are further developed by a number of Safety Guides which provide recommendations and guidelines, based on international experience, in relation to the fulfilment of the basic requirements, and Safety Practices, which give practical examples that can be used for the fulfilment of the requirements and recommendations of the standards and guides.

An overview of the present status of finalization of the set of documents in the RRSP is shown in Fig. 1. At present, there are drafts of all the documents in different status of finalization. Because of the recent changes in the procedures for preparation and review of documents of the IAEA Safety Series, the publication of various Safety Guides (on Commissioning, on Operational Limits and Condition, on Maintenance) and Safety Practices (on Operating Procedures, on Provision of Radiation Protection Services) may suffer some delays.

[4] IAEA Statues, Art III, A.6.

New documents on research reactor safety, such as a Safety Practice on safety in core management and fuel handling or a TECDOC on selected safety assessment issues, have been included in the IAEA Programme and Budget for 1997-98.

3. INTEGRATED SAFETY ASSESSMENT OF RESEARCH REACTORS (INSARR) SERVICES

Activities under this group respond primarily to functions and responsibilities of the Agency related to its projects (with Member States) to which the Agency's Safety Standards and Safety Guides apply. An example is the performance of safety missions to research reactors which are covered by a Project and Supply Agreement with the Agency. An account of the safety missions conducted so far is shown in Table I. The Agency has extended such missions to any type of research reactor. The objectives of these missions may be related to the design, the commissioning, the operation or the refurbishment of the research reactor.

Many of the missions included in the table have been conducted following the procedures developed for the so-called Integrated Safety Assessment of Research Reactors (INSARR) missions. These missions are offered as a safety service of the IAEA to all Member States operating research reactors.

Other activities in this group are those related to IAEA responsibilities such as encouraging and assisting nuclear research and fostering the exchange of technical information, scientists and experts and their training. In this regard, the IAEA has organized the following types of activities in relation to research reactor safety:

(a) Co-ordinated research programmes (CRP);
(b) International symposia (SM), inter-regional and regional seminars (SR); and
(c) Interregional and regional training courses (TC).

The above activities are normally organized in co-operation between the Agency's Departments. A CRP on Applications of Non-destructive Testing and In-service Inspections to Research Reactors is currently going on. An international SM on Management on Ageing of Research Reactors was held in may 1995 in Geesthacht, Germany, an inter-regional training course on Safety in the Operation of Nuclear Research Reactors took place in Chalk River, Ontario, Canada and Argonne, Illinois, USA, from 8 May to 2 June 1995, and a regional training course on Safety Documentation for Research Reactors was held in Cairo from 9 to 20 March 1996.

An important type of activity related to research reactors is organized by the Department of Technical Assistance and Co-operation. At present, there are on-going Technical Co-operation projects on research reactor safety in the following countries: Algeria, Bangladesh, Egypt, Ghana, Iran, Kazakstan, Nigeria,Morocco, Syria, Thailand and Zaire.

4. INCIDENT REPORTING SYSTEM FOR RESEARCH REACTORS (IRSRR)

Based on its statutory responsibilities of exchange of technical information and on Member States' requests, the IAEA has included in its current programme on research reactor safety the establishment of an Incident Reporting System for Research Reactors (IRSRR). The objective of the IRSRR is to improve the safety of operating research reactors through the exchange of safety-related information. The exchange of information on unusual events with safety significance is considered beneficial for the improvement of the operational safety. The IRSRR will collect, maintain and disseminate reports on unusual events which are received from Member States participating in the system. The IRSRR makes use of the existing IRS for nuclear power plants to the extent possible.

The IAEA has convened various meetings in the past years to prepare a basic working document for the establishment of IRSRR. The IAEA held a TCM in October 1996 to clearly define the types of unusual events which should be reported and to make recommendations on the implementation and operation of the proposed IRSRR. The TCM recommended the further implementation of the IRSRR and requested the IAEA to ask the Member States officially for their participation so that the IRSRR is operational by January 1997 for a trial period of two years.

The IRSRR makes use of the existing resources of the IRS for NPPs and has a similar organization. In particular, the nomination of the national co-ordinators should be the responsibility of the Member States. These national co-ordinators should preferably be officers of the regulatory bodies with technical background on research reactors.

FIG. 1 IAEA RESEARCH REACTOR SAFETY PUBLICATIONS (Schematic)

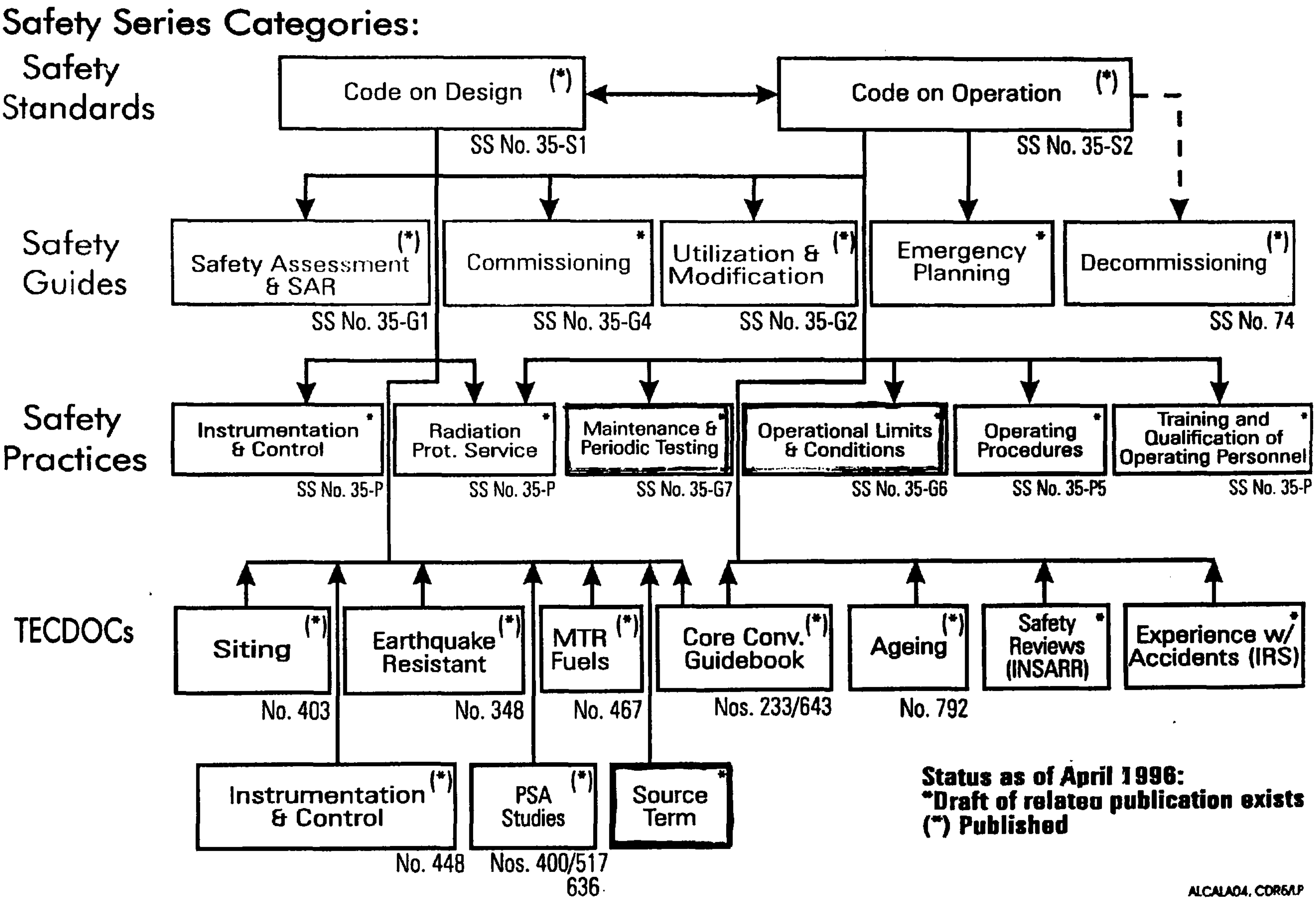

TABLE 1: INSARR MISSIONS

	1972-1976	1977-1981	1982-1986	1987-1991	1992-1996
(INSARR missions pursuant to project or supply agreements with the Agency)					
Argentina	1973 (2)	1978 (2)			1992(2)
Chile	1973	1977	1986	1991	
Finland	1976	1981		1987	
Greece	1972, 1976		1982, 1986		1993
Indonesia	1972, 1974	1978, 1979	1982, 1986		1994
Iran, Islamic Rep. of	1972, 1976			1990	
Jamaica			1986		1994
Japan	1976				
Malaysia		1977	1982, 1986		
Mexico	1972, 1975 (3)	1977(4),1981(4)	1986 (4)		1994
Norway				1987, 1988	
Pakistan	1976		1985		
Peru		1978, 1981		1987	1992
Philippines	1972, 1973, 1975	1978	1983		
Romania			1983		1992
Spain			1982, 1986		
Thailand	1974	1978	1982	1987	
Turkey		1977	1986		1992
Uruguay	1974	1978, 1979	1984		
Viet Nam			1985	1989	1995
Venezuela	1975	1979	1984	1988	
Yugoslavia (Slovenia)	1976		1985		1992
Zaire		1979	1984		1996
(INSARR missions at the request of Member States)					
Bangladesh					1995
Bulgaria				1990	
Brazil	1993	1977		1991	
Chile				1991	
Colombia		1977	1983	1987	
Egypt			1985		
Hungary			1983	1989	1993
Indonesia		1979	1982, 1986		
Iraq				1988(2)	
Korea, Rep. of	1976		1982(2)	1988(2)	
Peru					1992
Portugal					1992
Turkey Ukraine					1998
~~Ukrainian SSR~~				1991	
USSR				1990(2)	
Yugoslavia (Serbia)			1985		
Kazakstan					1993
Uzbekistan					1993
No. of missions in period	25	27	32	21	19

CHAIRMAN : B . FARNOUX

SESSION 1A

HFIR UPGRADE PROPOSALS (Colin West)

Question from Kir Konoplev of PNPI :

About the choice and configuration of the reflector.

A : OK. Did everyone hear the question? I'm only asking that to give myself time to think!

I proposed replacing the beryllium reflector, this outer - the larger piece of the beryllium reflector here with a tank of heavy water, which is a better kind of reflector as we all know, for neutron scattering work.

The consensus at the laboratory - there was a long debate - but the outcome of the debate was that that was risky. Not technically risky, but risky from a regulatory point of view. If we made such a major change to the reactor, then we would be into very many safety reviews and regulatory reviews, and it might take years - three years, four years, who knows? Whereas simply replacing the beryllium reflector with one with a slightly larger hole in it was a much easier process.

So that is why we did not adopt the heavy water reflector.

Now the other question I have forgotten! Oh, yes - the beam tubes penetrate essentially all the way through this outer reflector and into a small indent in the next layer of the reflector. So they are, in fact, closer to the thermal neutron peak than the beam tube tips at, for example, RHF/ILL Grenoble or, I think, ORPHEE.

By putting the cold source very close like that, we can get lots of neutrons, but also lots of heat, and Doug will tell you how we tackle that.

Question from Hans-Joachim Roegler of Siemens :

There are two reasons for stopping the ANS, one was the cost and the other was the enrichment. Now, with your statement that you will operate this year up to the year 2030, will you not have to justify that, for that long of a period, you will still use a high enriched uranium grade material in the reactor, and will you have similar studies as you had in the past with the ANS.
Secondly, is there any evaluation of the budget you will have available safely for upgrading that and do your technical proposals not have to check about whether they fit into that budget?

A : Yes, I have no doubt that we will have to debate the use of uranium with lower enrichment, although, at the recent RERTR meeting in Korea, the paper from Jim Matthaus from Argon said that at present they do not have a fuel that is suitable for the conversion of HFIR, and that is true. But, no doubt, they will keep reviewing that.

As to the budget, I do have a few copies only of a little newsletter that discusses the budget. The budget that we need is about 70 million dollars, and of that, almost 10 million is to change the beryllium, which we have to do anyway to keep operating. So the 60 million difference is for the upgrade. Of that 60 million, we already have spent or have available, we believe, about 10 million. We've got money already available for the cold source and we are working on it. So, we need about 50 million additional dollars over the next 4 years, which is not so very much. I mean it's much less than the operating budget, for example, of the reactor. So, I am hopeful that we will be able to get that, although the budget situation is very, very, very tight in the United States, and also here, I know; and so it's possible that we will have delays, but we can't have too many delays because the beryllium will need to be changed anyway regardless of Congress.

Question from Horst Hassel of Jecta Consulting :

I have two questions. The first is, is there any intentions of producing Mo-99, and the second has been partially answered already, the fuel element you use - is there any change besides the question of using the HEU, any change of the existing fuel element design and, in this connection, if you start with HEU, of course, is there any intention, later on, to change if there is a very high density fuel available without any change in the fuel element design, then will you use the LEU maybe?

A : The Mo-99 production : we did propose producing Mo-99 at Oak Ridge and we could do so, but that is not the US Department of Energy's plan at present. Obviously, we have enough flux and space to do that. The upgrade itself does not include any change to the fuel element, although I have proposed at the laboratory that we consider increasing the fuel loading, using existing fuel, maybe even U_3O_8, in order to increase the core life and save on operating costs. It's the US government's policy that if the suitable fuel is available to convert reactors to LEU then they should be converted when the fuel and the money is available. That has not been the case so far, for example for the NIST reactor or the University of Missouri reactor either.

Question from Edgar Koonen of CEN/SCK :

Concerning the beryllium reflector, what are the criteria for replacing it, what fluences have been reached, and is there a surveillance program?

A : There is a surveillance program - we have to look at the beryllium, I mean inspect it. We know from past experience that there has never been any problem at less than, I believe it's 279,000 megawatt days, and so, administratively, we say that we are required to replace the beryllium to operate after 279,000 megawatt days. It develops cracks. We have never had any problem from those cracks because we always change it out before we get a problem.

Anonymous question : Have you ever lost any material?

A : Yes, but not enough to cause problems anywhere else. I mean, not big pieces!

Question from Hans-Joachim Roegler of Siemens :

On the second side, the last sentence you didn't read - I don't know why - it says that you have minor modifications only to operate the reactor only up to the year 2030. It surprised me a little bit - how can you assess that because you have nearly 40 years to go. Does the history of this reactor really demonstrate that these 40 years will be accomplished without bigger changes?

A : We think so. The major problem that we have is the embrittlement of the pressure vessel. It's a carbon steel vessel with a stainless steel coating. As you irradiate it, at first it embrittles rapidly, and as time goes by less and less rapidly because all the embrittlement you can do is almost done. Our most recent results and calculations, and surveillance results, imply that at least over that time period the embrittlement will not be enough to pose a safety hazard. The only other things that seem like they might be problems are that the beryllium, which we have to change out anyway, and all of the other components like the pumps and the motors and the control rod drives, and that's minor routine maintenance - it's not so minor in money, but its a thing you would have to do anyway. That's why we believe that.

OPERATION EXPERIENCE AND CURRENT STATUS OF HANARO
(Kye-Hong Lee)

Question from Jean-Luc Minguet of Technicatome :

What is the reason for having chosen an outer core and what is the technology of this outer fuel elements? Is it the same than for the inner core ?

A: It's the same. The fuel assemblies are the same. Only the difference between the cores is we had different shapes, one circular with 18 elements, and the other fuel elements with 36 elements, and those fuel assemblies are the same in the inner core and the outer core.
We have eight holes in the outer core : four of them are used for fuel loading, the other four we use for the material we test.

Question from Jean-Luc Minguet of Technicatome :

Is there a special dedicated refueling program for these outer elements ?

A : No, during refueling, we refuel during the same refueling period.

Question from Klaus Böning of TU München :

You mentioned a Low Flow Critical Heat Flux experiment. Could you say a few more words about this? What are the details of these experiments ?

A : We got this low flow, low pressure critical heat flux data from the AECL; and they performed their experiment, but their point wasn't that many to satisfy the regulatory body, so we conditionally had put some more time together, some more data points in that low flow low pressure critical heat flux data, so one of our colleagues in the Hanaro Center is setting up the test loop - I think it's already done - and in December they are going to run the test experiment so soon we'll get some data out of it. But I can't go more into the details!

Question from Hans-Joachim Roegler of Siemens :

So far we have only heard rumors about the money you paid to AECL for that project, but we have never heard the total cost figure. Is there any internal assessment at KAERI of what the total plant has finally cost, meaning internal costs of KAERI's supplies from Korean industry plus money you pay to AECL? Can you at least mention an oral figure, if you don't print it?

A : I'm sorry I didn't catch his question very well... Can you rephrase it?

I'll try to repeat it slowly. The total cost of the plant, because it has contributions from KAERI internal efforts, from Korean industrial efforts, from imported things and from costs you pay to AECL. Can you give a total cost figure for your plant?

A : Mr. Chae, can you answer that question?

Answer by Mr. Chae : I didn't have the detailed data for that - if I had expected this kind of question I would have prepared. Anyway, I, as project manager, I control the whole budget and the schedule, I can say the simplest our total investment in the Hanaro project, except the site purchase is estimated at 130 million US dollars

When we purchased a significant part from the AECL, as you may know, for the extra structure like Zircalloy for the reflector tank and fuels, we had to purchase extra components, more material. The States didn't want to sell any more material to our country, so they set some limitations for us to do this kind of business. We had some collaboration with the AECL for this kind of significant great components so we purchased these components for about 16 million dollars from Canada. (just a public loan from the EDC of Canada). All the other parts were done with local companies. Does that answer your question?

CURRENT STATUS OF RESEARCH REACTORS RA AND RB AT THE VINCA INSTITUTE OF NUCLEAR SCIENCES (Miroslav Kopecni)

Question from Johannes Wolters of Jülich Research Center :

What is the reason that the pressure increase in your fuel barrels? Can you explain it?

A : Yes. This is only, of course, estimated pressure, and the first step we are going to do is to measure exactly the pressure which might be present in the barrels. The reason is hydrogen buildup due to the corrosion process. And then, if you consider the results which are available in the literature on the corrosion of aluminum in a spent fuel pool solvent storage; you can estimate roughly what the pressure could be.

Question from Klaus Böning of TU München :

If I understood you correctly, you mentioned that you used graphite in the heavy-water cooled reactor RA. What is the reason for using graphite in a heavy water reactor? Do you think it's better or what?

A: This is our inner core of the reactor which is situated in a stainless steel vessel and this is a reflector as far as I know. I'm here for the chemistry of the pool, so I'm not quite sure that I can give you a correct answer to that question.

STUDSVIK'S R2 REACTOR - REVIEW OF RECENT ACTIVITIES
(Mikael Grounes)

Question from Hans-Joachim Roegler of Siemens :

R2 is obviously widely used by these experiments, so of course the financing is also dependent on these experiments. Maybe I didn't get it - who is behind these experiments, who finances the experiments and who will use the results for improving the power reactors? Is that the power stations in Sweden or all over Europe or even more?

A : No, Sweden plays a rather minor part. When these experiments were started in the 70s, they were mainly financed by Swedish parties, but since then the international cooperation has increased very much. So I can say that there are three types of experiments here : the most widely known, and the ones I have been speaking about all the time are the international programs. This means that, for each individual program here that I have described, a sort of consortium is formed, internationally - from different countries - to a large extent the fuel vendors, power reactor fuel vendors, to some extant also power reactor utilities in many countries, in some cases, safety organizations. But we also have a much wider volume that I have not talked about at all, with about the same types of experiments, which are bilateral - that is, an individual company, very often a fuel vendor, approaches us and says, "We have a new type of fuel we would like to test but we don't want our competitors to know about it unless the results are very good."
So, in the international programs, Sweden usually participates but only to a small extent, some bilateral programs are of course made for the Swedish utilities and for the Swedish fuel vendors, but the very large majority of them are made for organizations in Europe, in the United States and in Japan.

Question from Jong-Sup Wu of KAERI :

You mentioned damages in the primary cooling system due to debris. I would like to know what kind of debris and what was the damage - I mean damage to the primary component or fuel?

A : Well, you see, what happens is that during maintenance, and in the western world one perhaps does not always have the discipline among the workers as you have in your countries, so this means that some swarf that should have been removed is left in the power reactor and since the water is circulating at very great speed, small pieces of metal can get stuck in the fuel - you know, power reactor fuel has spacers, and in the spacers, this debris can get stuck and then it sort of vibrates against one part of the fuel and finally it wears a small hole, which can only be a half a millimeter by one millimeter, and out of the hole one does not get very much fuel, so that is not a safety problem in itself, but if the water that intrudes into the fuel gives rise to cracks that are half a meter long, then so much fuel might come out and there have been cases where even the bottom part of the fuel rod has broken or there was one very well publicized event in the United States when the fuel was taken out, half of the fuel rod was left in the reactor because it had broken.
So these are the problems. It's not an acute safety problem, but in the long run one does not want to have a buildup of radioactivity in the power reactor.

Question from Francis Merchie of CEA :

I understand that the last INCA once-through loop is installed inside the core and the core vessel and has to cross the lead of the vessel and the bottom of the vessel, right?

A : Well, not the bottom of the vessel. It is installed in one of the in pile loops. That is, we have used one of the existing in pile loops, which is a hairpin loop - it looks like a U-tube. It takes two of the fuel element positions. So there is no penetration in the bottom of the reactor and what we have done is that we have put this device into one of the loops. It can be removed and we can have different types, but that's the way it operates.

Q : What is the diameter of the loop?

A : The inner diameter, in the present experiment, I think is 24 millimeters. That is, we can have samples that are 24 millimeters but it can be wider if there should be an interest need for that.

Question from Johannes Wolters of Jülich Research Center :

When you do the fast-ramp tests, is there any impact on the reactor? Is there any impact on the power of the reactor?

A : Are you speaking of the standard ramp test or are you speaking about this new ultra-fast ramp test?

Q : The fast test.

A : Well, the new ultra-fast ramps - the first one will be made very soon, it hasn't been made yet, but according to all the safety reviews and the calculations we have made, there will not be any impact that will be disturbing. That's one of the reasons why we have such a small piece of fuel for these experiments.

Question from Adrian Verkooijen of IRI, Delft T.U. :

I would like to ask you with respect to you ultra-high burn-up program - did I understand this correctly, that you use the standard industrial fuel which has a burn-up of more than 20 megawatt days, and then extend that to 60 'til 80.

A : Yes, we will take standard fuel that has been irradiated to 60. We will then refabricate short fuel rodlets, because, since the R2 reactor has a core of 60, this means that we can handle... the most convenient is to use so-called "rodlets", which are the fuel set lengths of perhaps 400 or 500 or so. So we would refabricate them. We have a remote refabrication capability. And then we would irradiate in a fuel element position. We have something called "boiling capsules" where several fuel rods can be irradiated together, just as a sort of continued base irradiation. We would then irradiate them up to 80. And then we would make ramp tests with them.

Question from JJ. Verdeau of Technicatome :

When do you intend to begin this ultra ramp project?

A : Well, we have always had the policy that we are not going to start a national program with a technique that we have not demonstrated in full scale, and for this ultra-fast ramps - the ones that will take one second - the first tests will be made in January or February of next year, I don't remember. So this means that shortly after that time we are going to invite prospective participants to a first pre-project meeting in Studsvik. We have on several occasions invited people to other pre-project meetings, and then we have also had the information meeting discussing this type of experiment, but we didn't want to start the actual attempts to get people committed to these experiments until we have shown that we really can make them in the way that we have intended to.

So, the ultra ramp consists of three parts, one can say. Two of them are ramps of a type that we have been pursuing for many years - that could have been started a long time ago. But since many people were interested in these "ultra-fast ramps", we didn't want to start the program until we had proven that the ultra-fast ramps could be made in the way that we had intended to. And that will take place early next year.

MAIN EXPERIENCES IN RENOVATION OF THE DALAT NUCLEAR RESEARCH REACTOR (Pham Van Lam)

Question from Hedi Ben Kraiem of CNSTN :

Could you give some indication please of the running cost for operation, the time of operation of the reactor and the kind of users of the reactor?

A : Now, our reactor is mainly used for radioisotope production, activation analysis and research and training. The main mode of operation now is one cycle of one hundred hours of continuous operation once in four weeks. We use the rest of the time for maintenance and some sorts of operations.

And the running costs?

Actually, I have not kept track of running costs at the time. Excuse me.

Question from Wolfgang Knop of GKSS :

What is the core size and where have you measured the neutron flux of $2.1 10^{13}$ neutrons per centimeter and second? In the neutron trap?

Yes, the high neutron flux is received in the neutron trap in the center. The diameter of the core is 40 cm. And the height is 60 cm.

Question from Francis Merchie of CEA :

What is the use of the thermal column?

A : At that time we put a pneumatic transfer system in the thermal column beside two pneumatic transfer systems in the positions 71 and 32 and in the thermal column we also put a pneumatic transfer system and from 4 horizontal channels we now use channel number three, it is a tangential channel, and channel number four - now we use two horizontal channels.

UPDATE ON THE BR2 REFURBISHMENT (Edgar Koonen)

Question from Kir Konoplev of PNPI :

Is it possible to use the concept of the "leak before failure" for your vessel?

A : No, no, this was not accepted

Question from Yang Tsing-Tyan of INER :

You mentioned that you will implement the ISCC program on the BWR stressed components, but now the ISCC of BWR is more severe than the PWR. Why don't you extend the program for the PWR?

A : I'm not sure I understand your question. It's mostly BWR and PWR - it's what you say.

Q: For example many cracks have been found in the BWR core shroud.

A : There are two aspects to this, first the internal program has to be for domestic power stations and we only have PWR reactors, so the internal program is focused on our reactors. Now, we discussed having some done nationally, but in that case we will build a new loop. As far as it's based only on our net program, we will do it for some PWR conditions.

So it depends on whether the negotiations come to an end or not.

Question from Chen shih-Kuei of INER :

You mentioned the PSA study for the reactor. My question is, first, why do you want to do a PSA study for that? Is that a regulatory requirement? The second is, are there any criteria for your results? Why do you want to perform modifications for your LOCA situation?

A : We started this PSA in '91. At that time it was not requested by our authorities, but our authorities have changed since then - we had another authority which specifically concerned with us, it was not the same people as those who deal with power stations. So we did the PSA and the aim was to prioritize items for the refurbishment program, so it was an internal effort.
Afterwards, when we had this new authority AVN they had already made some PSA studies for all our PWRs, so they were very happy that we had already done it! We had to make a presentation and they were quite satisfied with the way we had done it. They are not really looking at the figures. They are looking at relative importance of items.
Now, the LOCA is one of the reference accidents for the PWRs. Of course, that sort of study had not previously been done for BR2 so we had no relap model before that. The relap model was triggered by the PSA study. And there is, of course, some aging in the design of BR2 - part of the primary circuit is outside of the containment building.
So, even in an improbable case of core damage, we must show that we are absolutely certain that we can keep any activity inside this containment and that this primary circuit is somehow a containment bypass.

Question from Albert Lee of AECL :

Edgar, how did you solve the problem with RELAP5 being unable to model situations at very low pressure and low flow?

A : I didn't solve it, first of all! I'm afraid we would have to talk to the specialists. We started the model several times, changing the calculation step. I'm unable to give you the real details, but there is definitely a problem at low pressures. Sometimes the calculation stops. Also the values that you have in the tables don't go so low. I wasn't the one who made those calculations, this is some dedicated person who does only that.

Q : Without valid steam property tables and water property tables at low temperature and low pressure, RELAP5 gets into numerical instabilities and you can alter the time steps to try and adjust for it but you have to be very careful about matching up the answers between successive cases. There can be some unphysical discontinuities that appear.

A : We have, and are still busy by making some benchmark calculations using the 1963 thermohydraulic perturbations. Those were not really LOCA , but there were loss of flow, loss of pressure, and we still don't have all of the data, but most of the data, that RELAP would require. So we try to recreate these conditions. One of the problems that we have is that we don't have the really detailed time delays from those days of valve closing and how the scram was actually operating and so on. These are of course factors that you really have to know, so the course closing of the valves. But we are using these 1963 perturbations to benchmark the calculations. It's the best we can do.

Question from Jean-Luc Minguet of Technicatome :

I have just two questions. Would you remind me of the level of pressure of the primary circuit of your reactor? And then what are the scheduled research and irradiation programs after the startup of your reactor next year?

A : The pressure is between 10 and 12 bars. And I said that we'd try to start in April, or maybe if this doesn't work it will be June. Next year we will probably be starting three cycles and the next year 1997 will be five cycles. I can't really tell you exactly. Mainly, the standard cycle is 21 days, but sometimes we run very short cycles for transients, which are not counted there. So 25 times 5 is 125 - that's what I said about 100 days

CHAIRMAN : C. DESANDRE

SESSION 1B

RESEARCH AND SERVICES OF LVR-15 REACTOR IN REZ (Jan Kysela)

Question from Johannes Wolters of Jülich Research Center :

You mentioned that you operate the reactor with a staff of 32 people. What does this include? Radiation protection people and maintenance people? What is included?

A : answer not recorded

Question from Hans-Joachim Roegler of Siemens :

You mentioned that you convert the reactor from 80% to 36% - this is still HEU as we all know of course. Is it not possible for you to stick to the 80% until the 20% is ready, or do the Russians no longer supply the 80%? Is that the reason why you had to convert?

A : answer not recorded

Anonymous question :
Regarding the BNCT - what is your tentative schedule for completely implementing this BNCT facility?

A : answer not recorded

HIFAR MAJOR SHUTDOWN REPORT (Shane Kim)

Question from Doug Selby of ORNL:

I have two questions : Is the vessel pure aluminum or an aluminum alloy? And secondly, what was the pH quality of the water in the place where you saw flaking?

A : It's pure aluminum, 1000 series. I think its 1050, I'm not sure.
As for the pH quality, how can I answer that? The water chemistry in the heavy water is monitored every day. I think it's about 5.5, something like that. But it varies.

Question from Kir Konoplev of PNPI:
What is the fluence of the aluminum tank?

A : 1.4×10^{14} that's the flux. Is that too high?

Question from Francis Merchie of CEA :

Some time ago, I mean two or three years ago, ANSTO was considering the replacement of HIFAR by a new reactor. What is the situation now?

A : We're just waiting for it. In about three year's time we should know, but we have to make sure our Greenpeace is happy about it.

Question from Edgar Koonen of CEN/SCK :

Can you explain why, when you made the ultra-sonic inspection it was just used for wall thickness measurements and not for flaw detection or other indications?

A : Actually we used two different types of ultra-sonic measurements, one actually is to measure the thickness of the aluminum tank to make sure that thickness didn't change. Between the wall and the probe there is about a one millimeter gap filled with de-mineralized water.

Q : So you could have detected some flaws?

A : We are not finding any imperfections in the wall. To find those imperfections, we have time-of-light ultra-sonic measurement which is angled at 90%, and we did that along the weld joints.

Q : Is there any limit set for the life of this vessel?

A : That's what we tried to find out, and so far we haven't seen any reason why we couldn't continue operating the reactor with that aluminum tank. But we'll keep watching.

Q : You do this every four or five years. I just wonder what the dog does in between time? What does the dog do when he's not doing this?

A : Getting fed!

Question from C. Desandre : I think that the reactor in the range of power of 10 megawatts is certainly the oldest reactor in operation, isn't it? Do you know of any other reactors? It was critical in '58, so that's 38 years of operation. I think it's the oldest, isn't it?

Anonymous answer : No, there's one in Denmark.

MODIFICATION OF JRR-4 (Teruo Nakajima)

Question from Edgar Koonen of CEN/SCK :

What was the intensity of the earthquake that you had to consider in your calculation?

A : The study considered a level of 0.72 g. In Japan, the examination guide for seismic hazards in nuclear plants is 0.72 g.

Question from Johannes Wolters of Jülich Research Center :

Do you need a new license for the modifications and what are the costs?

A : Yes and we estimated about 35 million dollars.

Question from Wu Jong-Sup of KAERI :

You considered a larger bore size of the silicon doping system. Do you know the diameter of that?

A : Yes, we changed to a 5.5 inch round pipe. Now they are 4 inches and we are changing to 5.5 inches.

Question from Jean-Luc Minguet of Technicatome :

When do you intend to start the BNCT treatments for patients.

A : Maybe when the modification work is over in 1998 - at the end of 1998, in December.

OVERVIEW ON THE MAIN ENGINEERING WORKS PERFORMED ON FRENCH RESEARCH REACTORS THESE LAST YEARS (Pascal Rousselle)

Question from Hans-Joachim Roegler of Siemens :

Does the new porous concrete which you use below the bottom of the pool allow detection of where the leakage is or just does it just detect that there is a leakage?

A : You can detect leakage by conducting a potential leakage to sumps. But, of course we have to make a specific examination after to localize it.

Question from Klaus Böning of TU München :

You mentioned that you planned to exchange the pressure boundary around the core very soon, and my question is: is this the first exchange of the core shroud for ORPHEE?

A : Yes, it will be the first one.

Q : After 15 years?

A : Yes, after 15 years of operation.

Q : And what technique are you going to use? Do you have to fill the whole heavy water system with light water?

A : Yes, we have to fill it.

Q : No problems with pollution by contamination of the heavy water ?

A : We have done this operation many times, when we remove the tubes for instance.

Q : Do you have an on-line cleaning system?

A : We clean the heavy water tank before putting the water inside. So that is very easy to do. There are no problems - we've done it many times.

UTILIZATION OF THE BUDAPEST RESEARCH REACTOR (Istvan Vidovszky)

Question from Bernard Farnoux of CEA / DSM :

What kind of material do you plan to use for the cell of the liquid hydrogen cold source?

A : Aluminum

Question from C. Desandre : Is it a vertical type, is it a thermo-siphon type?

A : Liquid hydrogen loop.

Question from Jean-Luc Minguet of Technicatome :

Who performed the basic design of this CNS?

A : This is complicated. It was mainly done by PNPI in St. Petersburg, but this was in cooperation with many other partners. And some of the design work will be done by a Hungarian design company and even some tasks will done by members of foreign subsidiaries. It's complicated.

Q : After the startup of your reactor, how many years do you plan to operate it?

A : This question is very hard to answer. Of course, it depends mainly on financial problems,. So we have the fuel for six more years, maybe seven, and whether we will be able to buy fuel again is an open question now. Of course, we hope that from the technical point of view the reactor could be operated 30 more years, but whether it will be financed or not I can't say.

CHAIRMAN : K. KONOPLEV

SESSION 1C

DESIGN MODIFICATION OF HANARO REFLECTOR COOLING SYSTEM (Jong-Sup Wu)

Question from Shieh Der-Jhy of INER :

In your design, the reflector cooling system has a higher pressure compared with the secondary cooling system, right? In this case, do you have any concern about the diffusion of tritium through the cooling system and into the cooling tower?

A : We have a radiation monitoring system in the secondary cooling system to detect if there is some leakage from the primary or D_20.

Question from Hans-Joachim Roegler of Siemens :

If you had to design this as a new system, instead of improving an existing system, would you have taken another heat exchanger like a pipe-type heat exchanger instead of the plate type?

A : In our D_20 boundary, the space was very small, there was a limit to the operator heat exchanger type. But I think the tube and shell type is larger than the plate-type heat exchanger. The plate-type heat exchanger has the advantage of a higher efficiency and is highly compact, but it has the disadvantage of the plate heat deflections. So we would be very prudent when using this type of exchanger.

Comment from Guy Gistau of Air Liquide : I'm not used to water heat exchangers, I mainly deal with cryogenic heat exchangers. But we also use plate heat exchangers and there are some possibilities to avoid this volume change even when the pressure changes.

A : But in our case, the reflector cooling system is very sensitive to limit the pressure in the system, because we have an outside shell of the core which is very sensitive to high pressure from the reflector system, so it is limited.

IAEA ACTIVITIES ON RESEARCH REACTOR SAFETY
(Francisco Alcala-Ruiz)

Question from Hans-Joachim Roegler of Siemens :

Assuming that a country does not have its own rules code and standards on research reactors, would you then consider the schemes that IAEA has implemented is sufficient to run a licensing process for a nuclear plant or for a research reactor?

A : Well, the scope of the documents prepared by the agency for nuclear reactors is also limited. These documents are only intended for some research reactors, not all. We couldn't establish a clear limit for the scope of the documents. We say that they should be applied to research reactors up to several tens of megawatts. For high-power reactors, I think another standard protocol should be used. But just up to this distribution of power by a research reactor, I think that many, many reactors come under the potential application of these documents. But not all, of course.

IGORR 5

SESSION 2

RESEARCH REACTORS IN DESIGN OR CONSTRUCTION

PAPERS

STATUS OF THE TRR-II PROJECT

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ABSTRACT

The 40 MW Taiwan Research Reactor (TRR) operated by the Institute of Nuclear Energy Research (INER) went critical in 1973 but was permanently shut down in 1988. After the shutdown of TRR, remodeling of the reactor into a light water pool type research reactor, i.e., TRR-II, has been considered by INER. TRR-II project is currently working on environmental impact study, reactor dismantling and construction planning study, and some conceptual design. The design goal is to have a reactor with thermal as well as fast flux around 10^{14} cm^{-2} sec^{-1} and a nuclear thermal power of less than 20 MW to meet the utilization requirements.

1. Introduction

The Taiwan Research Reactor (TRR) was a natural uranium metal fueled, heavy water moderated and light water cooled reactor with a thermal power output of 40 MW. It first went critical in January 1973 and, after a little more than 15 years in operation, was permanently shutdown in 1988. An evaluation program was then launched to figure out: (1) how to deal with (i.g., dismantle) the TRR, (2) do we need a new research reactor, and (3) what type of the reactor is the most proper one in consideration of the technology localization and the domestic need of neutron sources for research, industrial and medical applications.

Initially, a proposal of dismantling the original core block by one piece removal and procuring a multipurpose research reactor through turn-key project has been discussed for years and revised several times. However, most of the reviewers had suggested that instead of purchasing a research reactor from foreign vendors, INER should build one by the institute itself in order to promote the local technical capabilities. In 1995, a new proposal based on comments from reviewers got an approval from Atomic Energy Council (AEC). Then, the proposal was submitted to the National Science Committee (NSC) of Executive Yuan. Due to the lack of consensus on environmental impact assessment of TRR-II and on the priority for TRR-II construction, NSC reviewers only approved a limited amount of budget for the following studies in 1996:

(1) environmental impact assessment,
(2) basic engineering design and safety analysis,
(3) project schedule and detailed cost estimation,
(4) cost benefit analysis, and
(5) user training program.

After completing the studies, the TRR-II Project will be submitted to Executive Yuan for an approval again.

2. Status of TRR-II Project

2.1 Objective

After the shutdown of TRR, not a neutron source with flux higher than 10^{14} $cm^{-2}sec^{-1}$ is available in Taiwan. For the purpose of continuing and promoting the peaceful atomic applications, by taking a careful survey of needs and considering the constraint conditions, it is decided to build a multipurpose research reactor with a thermal flux around $10^{14}cm^{-2}sec^{-1}$

Survey of need was conducted through numerous face-to-face discussions with domestic universities, research organizations, hospitals and industries as well as considering the experimental requirements of INER itself. It was realized that the local utilization requirements of interest to use TRR-II as a neutron source include the followings:

- neutron transmutation doping (NTD),
- neutron beam experiments (NBE),
- nuclear fuel and material development (NFMD),
- neutron activation analysis (NAA),
- neutron radiography (NR),
- boron neutron capture therapy (BNCT),
- water and radio-chemistry (WRC), and
- radioisotope research and production (RIPP).

Among these applications, INER and other institutes in Taiwan already have some experiences in isotopes production, neutron activation analysis, neutron radiography, and BNCT. The rest actively worldwide studied applications, such as NTD, fuel and material tests, and neutron beam applications, will be somewhat new to Taiwan. TRR-II will contribute greatly in promoting these new applications. Currently, seven users' planning groups were formed and are working actively in recruiting potential users.

2.2 TRR-II Design

Because a thermal as well as fast neutron flux around $10^{14}cm^{-2}sec^{-1}$ is sufficient for majority of users and INER technical strength is UO_2 type of fuel, therefore, UO_2 fuel is selected as the first choice for TRR-II core. In the future, if a higher neutron flux will ever be in demand, the core can be readily converted with other types of fuel.

For core design, two core configurations are under study. One design is a rectangular core and the other design is a circular core. All fuel rods will contain low enrichment UO_2 pellets. The core will use light water as moderator and coolant, and the core is surrounded by beryllium reflectors and a heavy water tank. The reactor core is located at the bottom region of the reactor pool. Control rod drive mechanisms (CRDMs) are installed below the reactor core. Fig.1 shows the cutaway view of the TRR-II.

For the reactor cooling and shutdown systems, the key features are as followings:

(1) The core is submerged in a pool. The depth of water is about 8 meter. The pool is covered with top shield to reduce the radiation level.

(2) The coolant flow is downward for the purpose of reduction of the radiation dose of ^{16}N.

(3) The primary coolant pipes penetrates the pool wall at a level higher than the core to avoid core uncover caused by coolant pipe break.

(4) The primary coolant loop is equipped with siphon break valves to avoid loss of pool water due to siphon phenomenon.

(5) The primary coolant pumps are backed up with emergency auxiliary pumps to assure decay heat removal after reactor shutdown.

(6) The lower plenum of the core is equipped with two flap valves so that decay heat can be removed with natural circulation for long term cooling after shutdown.

(7) The pool water clean up and heat removal system will play the role of ultimate heat sink when the main coolant systems shutdown.

(8) In addition to the control rod shutdown system, the heavy water dump system will be used as a backup shutdown system in case of

anticipated transient without scram (ATWS). All of the control rods will drop into the core by gravity, also is the dump of heavy water .

2.3 Dismantling of the TRR core

The new reactor will be constructed after dismantling the old one. For the sake of minimizing radwaste generated from dismantling of the old reactor, the possibility of keeping the biological shields and other usable equipment is still under study and the TRR reactor building and stack will be retained. The conceptual study of the piece by piece dismantling work for the TRR core is elaborated below.

Fig.2 shows the general arrangement of the old TRR reactor. The zircaloy calandria is the central component of the reactor. The calandria is a cylindrical vessel. It is supported by the lower axial thermal shields. The vessel is 106 inches diameter (inside) and approximately 129 inches high. It is penetrated by 199 vertical tubes arranged on a hexagonal lattice. The calandria is surrounded by the graphite reflector , and then the thermal shields and the biological shields.

Before developing the idea of TRR piece by piece dismantling method, specific activities of the major components have been calculated using ORIGEN-II and QAD-CG codes. The weight, volume and material for the major components are shown in Table 1. Table 2 shows the radioactivities of these components. Referring to these results and according to the strategy of TRR remodeling project, the piece by piece dismantling method was developed. We compared the dry process and wet process for dismantling by taking lots of factors into account such as cost, schedule, waste production, etc. Finally, it is dicided to use wet process. The dismantling work will be done under the water. To ensure no leakage will occur, all the openings of reactor will be sealed, and a layer of cylindrical cover plats surrounding the biological shield will be installed to act as a second barrier. The principal dismantling step are shown in Fig.3. Since no final disposal place is available in Taiwan, on site storage for high radiation level waste generated from the dismantling work is the preferred choice.

2.4 Environmental impact assessment

A formal environmental impact assessment for research reactors is not a regulatory requirement in Taiwan now. But today, people are getting more concerned with the quality of environment and a construction project of nuclear facility surely will attract great attention of the environmental group. Therefore, following the suggestion of National Science Committee of Executive Yuan, a complete environmental impact study similar to the one for a nuclear power plant is conducted. The task is awarded to the Environmental Department of E&C Engineering Corporation.

Because strong protest from the surrounding inhabitants is foreseeable if the LPZ is beyond the boundary of INER, one of the TRR-II policy is to design a reactor with the Low Population Zone (LPZ) inside the boundary of INER,

3. Concluding Remarks

TRR-II Project team is currently working on environmental impact study, reactor dismantling and construction planning study, and some conceptual design. In the design process, the most difficult problem comes from the uncertainty of regulation for research reactors. In our study of the design of research reactors around the world, we find that there are lack of consensus with regard to some design principles, such as:

- the postulated accident for LPZ calculation, which is varied from the assumption of a single fuel bundle handling accident to the BORAX type accident,
- the necessity of a secondary shutdown system,
- the extent of redundancy and physical separation,
- the applicability of the standards of nuclear power plants, such as ASME Section III, to research reactors, etc.

In our design, the safety codes of IAEA[(1)(2)(3)] are followed. However, many subjects still are remained to be discussed with the regulatory authority, i.e., ROCAEC.

A good research reactor is an essential tool for peaceful atomic applications in Taiwan. Therefore, we look forward to receiving a final approval for construction of the reactor in 1997.

4. References

(1) IAEA, Code on the Safety of Nuclear Research Reactor: Design, No. 35-S1, 1992.

(2) IAEA, Code on the Safety of Nuclear Research Reactor: Operation, No. 35-S2, 1992.

(3) IAEA, Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report, No. 35-G1, 1994.

Table 1. Weight, volume and material of reactor components

Part	Weight (t)	Material	Volume
1. Removal biological shields	70.86	concrete	3.8ø×0.5t(m)
2. Fixed biological shields outside removal biological shields	61.24	concrete	
3. Upper thermal shields	32.80	cast iron	
4. Embedded thermal shields	56.47	cast iron	
5. Lower thermal shields	71.22	cast iron	5ø×0.42t(m)
6. Floor plate	24.49	cast iron	
7. Side thermal shields inner / outer	138.49	mild steel	62×250×36(cm)
8. Reflector inner / outer	60.67	Graphite	
9. Thermal column	19.59	Graphite	
10. Biological shield	1574.42	Concrete	
11. Calandria	3.12	Zircaloy	
12. Revolving floor	18.45	Cast iron	
13. W Shapes	5.75		
14. Master plate	——		
Total of 1～14	2137.57		
15. Floor concrete block	588.86		
Total	2726.43		

Table 2. Calculation result of the activation of the major components in reactor core

Structure Components	2 yr	5 yr	10 yr	30 yr	Total Weight	Specific Activity (2 yr)
Reactor Calandria	3.921E2	1.256E1	1.356E1	1.575E1	3.12	4 μCi/g
Graphite Reflector	1.726E1	1.660E1	1.573E1	1.390E1	60.67	0.3 μCi/g
Upper Thermal Shield	1.859E6	8.351E5	2.208E5	1.165E3	32.80+56.47	9E3 μCi/g
Lower Thermal Shield	4.038E6	1.814E6	4.798E5	2.531E3	71.22+24.49	2E4 μCi/g
Iron Thermal Shield	2.395E5 +3.510E3	1.075E5 +1.602E3	2.835E4 +4.204E2	1.392E2 +2.046E0	138.49	7E2 μCi/g
Thermal Column	6.127E-5	6.125E-5	6.121E-5	6.106E-5	19.59	3E6 μCi/g
Reactor Biological Shield	1.466E-1	1.057E-2 1.466E-1	1.057E-2 1.057E-2	1.057E-2 1.057E-2	1574.42 1.057E-2	1E-5 μCi/g
Removable Biological Shield	1.045E-1	1.081E-2	1.080E-2	1.080E-2	70.68	8E-5 μCi/g

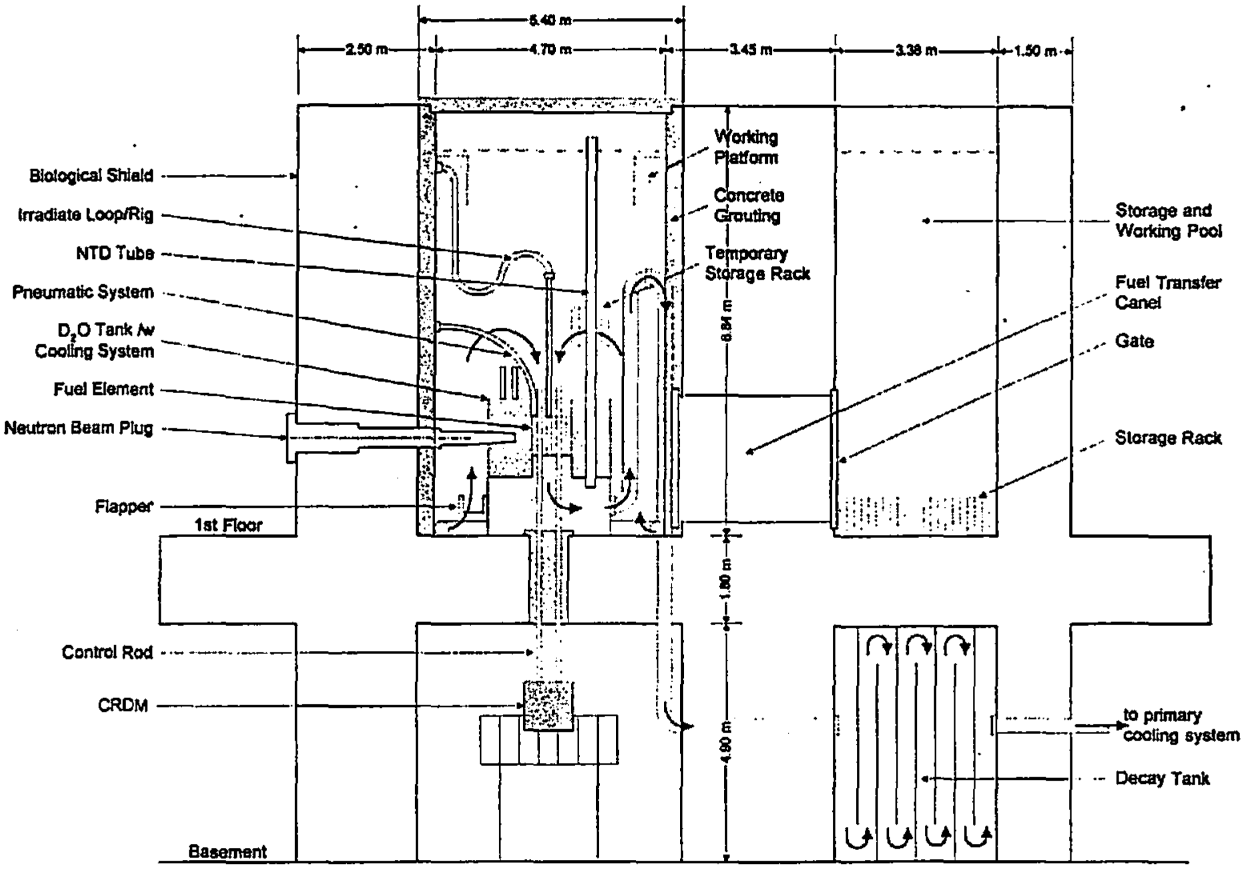
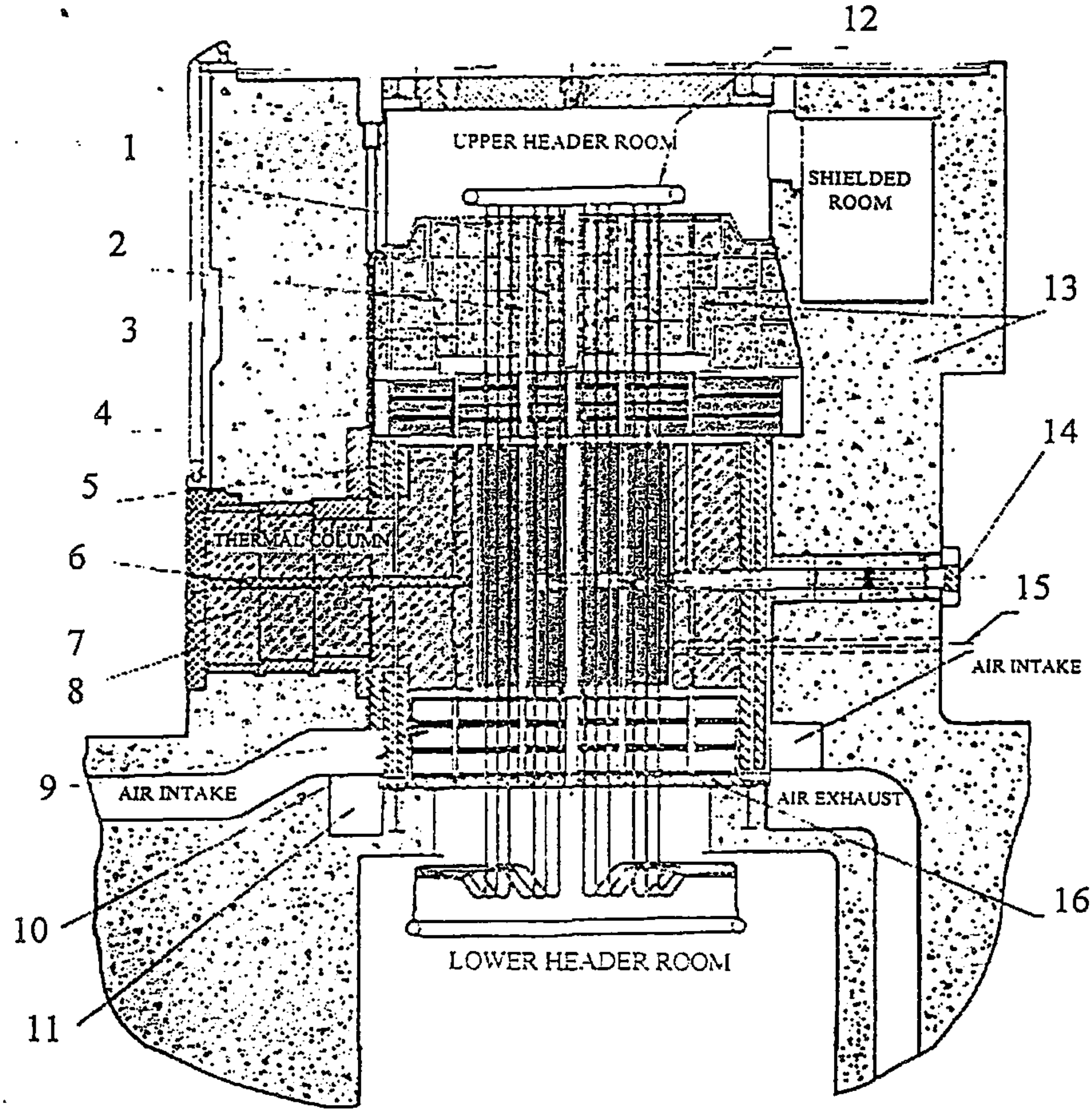


Fig. 1 A vertical cross section view of TRR-Ⅱ



1-CENTRAL THIMBLE
2-EXPERIMENTAL LOOP POSITIONS
3-J-ROD POSITION
4-TOP THERMAL SHIELD
5-GRAPHITE REFLECTOR
6-REMOVABLE PLUG
7-REMOVABLE GRAPHITE BLOCKS
8-LEAD DOOR
9-BOTTOM THERMAL SHIELDS
10-SIDE THERMAL SHIELDS
11-AIR EXHAUST
12-INLET HEADER
13-BIOLOGICAL SHIELDS
14-EXPERIMENTAL HOLES
15-ION CHAMBER HOUSINGS
16-FLOOR PLATE

Fig. 2 Reactor general arrangement

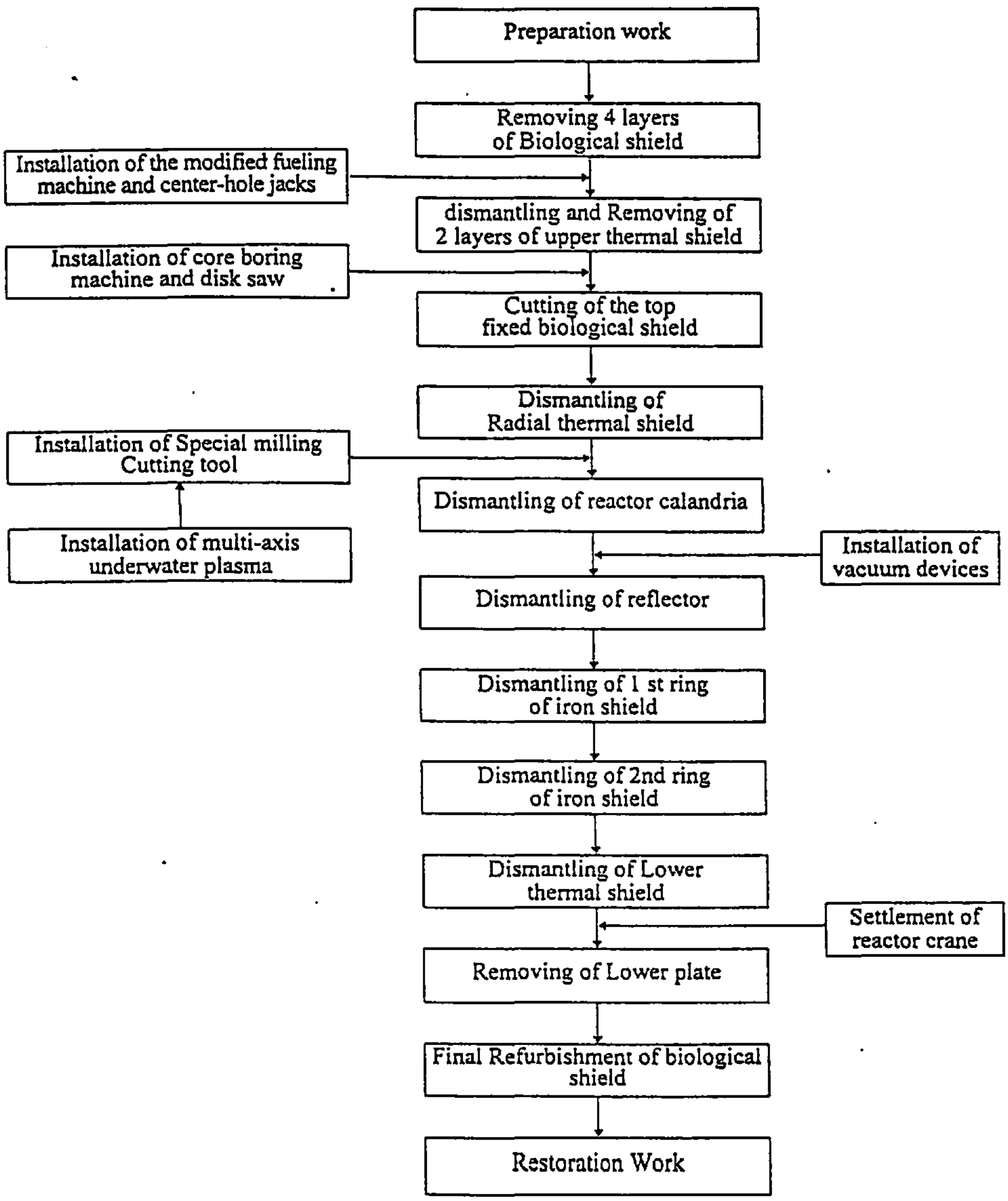


Fig. 3 Principal steps of TRR dismantling

UNIVERSITY OF BOCHUM

Department for Nuclear
and New Energy Systems
Prof. Dr.-Ing. H. Unger

Thermohydraulic and Mechanical Analysis of a Scale FRM-II Core Dummy

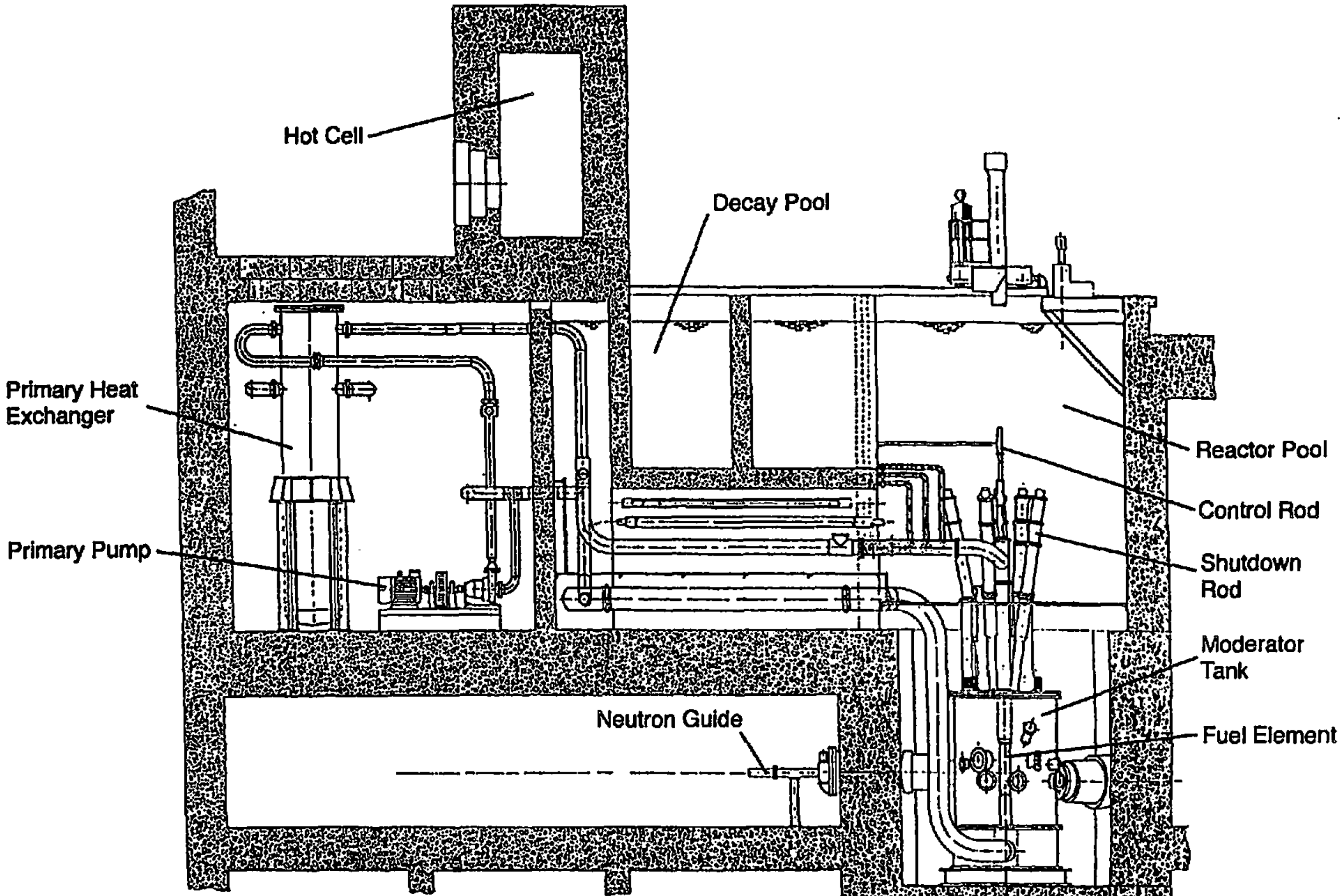
J. Adamek, H. Unger

IGORR 5 Meeting
Aix-en-Provence, November, 4. – 6. 1996

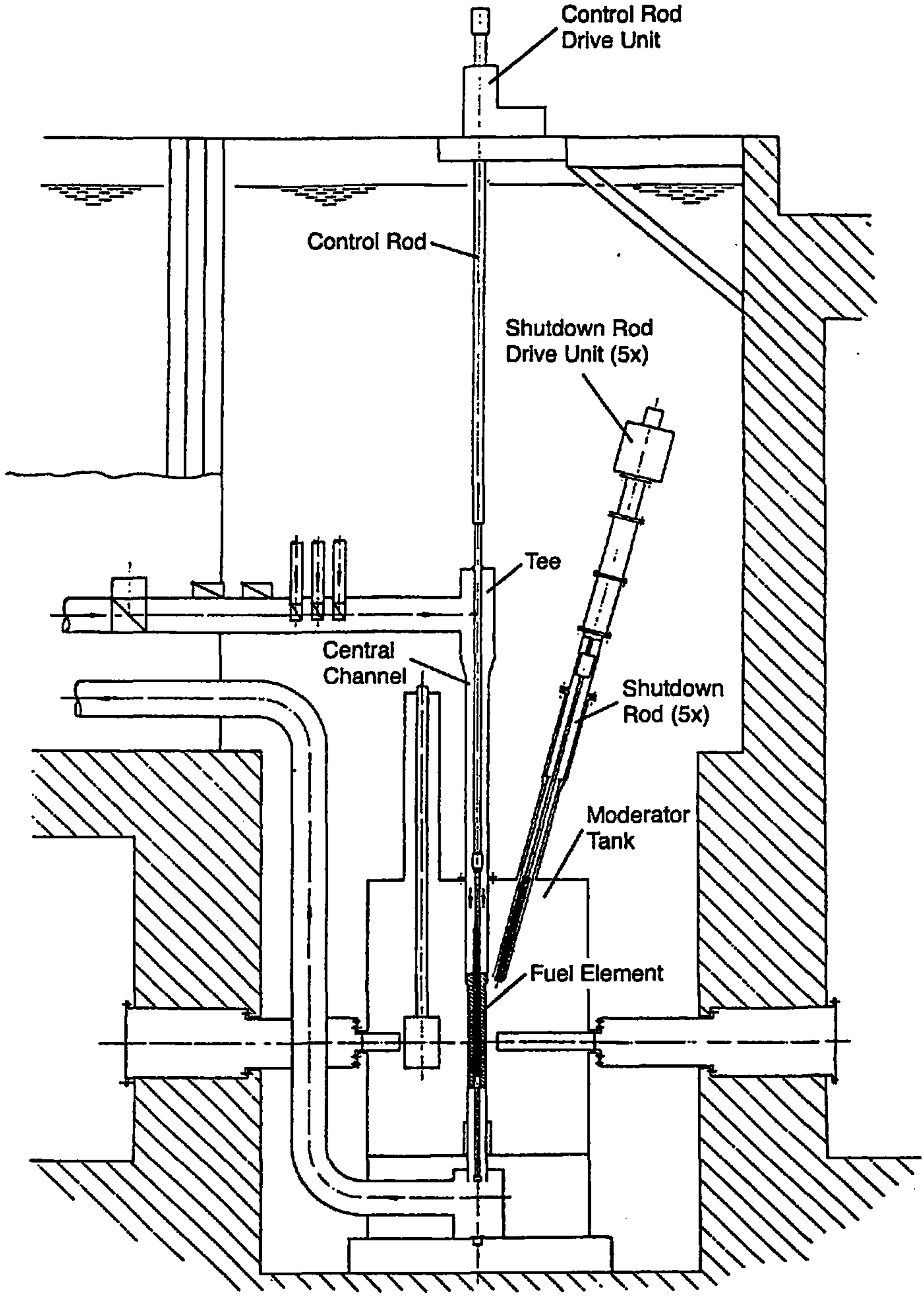
Contents

- Characteristics of the FRM-II,
- Objectives of the Experiments and the Test Program,
- Test Facility for the Qualification of the FRM-II Fuel Element,
- First Experimental Results:
  - Vibration Tests,
  - Pressure Drop Measurements,
  - Start-Up and Shutdown Tests of the Main Pump,
- Summary.

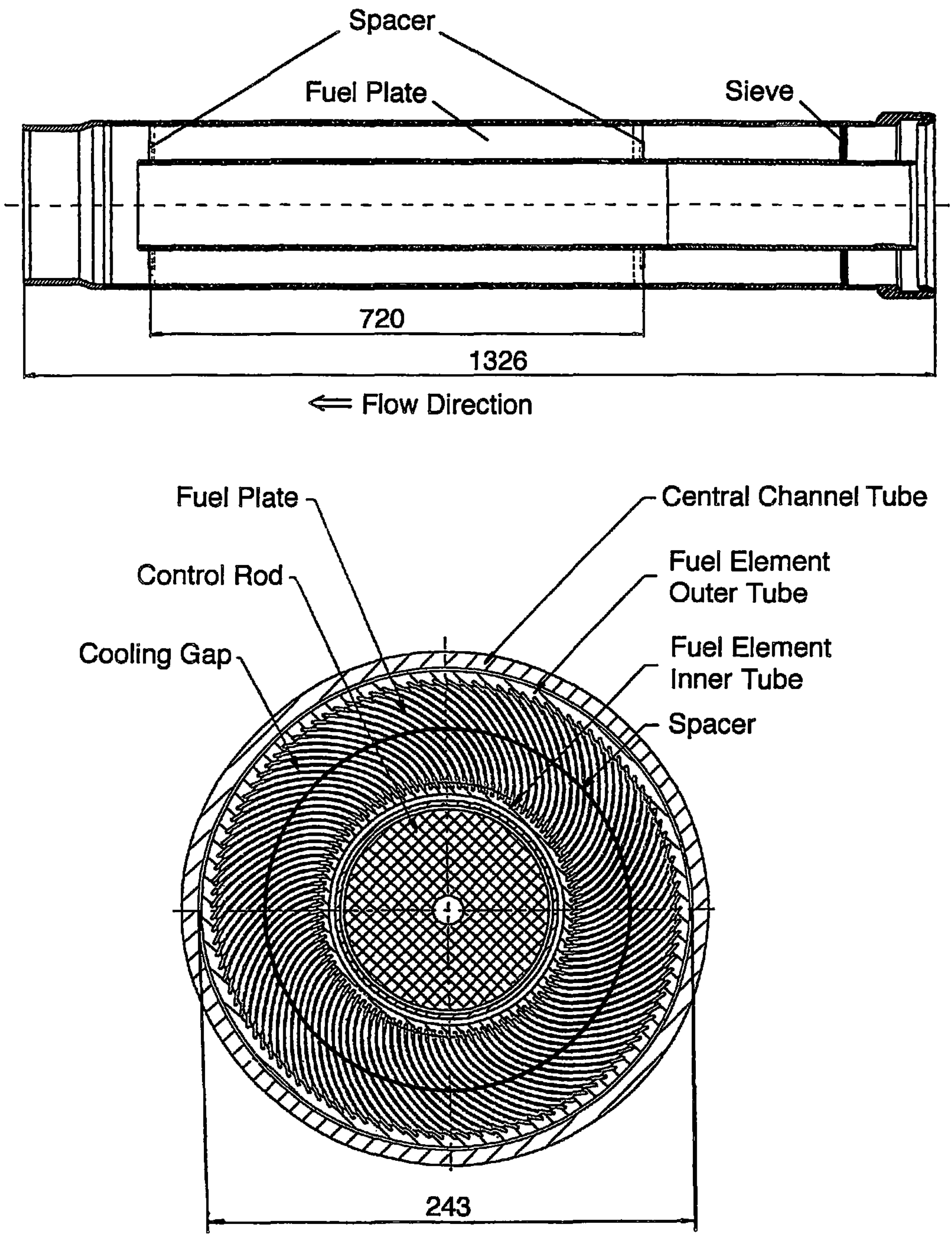
View of the FRM-II Reactor Building



Partial View of the Inner Section of the FRM-II



Fuel Element KKE-7 of the FRM-II



Objectives of the Experiments and the Test Program Performed

- Objectives of the experiments:
  - Verification of the hydraulic design of the FRM-II fuel element,
  - verification of the feasibility of the core cooling in case of a partly obstructed sieve,
  - verification of the fatigue strength of the fuel element by means of an endurance test.

- Test program:
  The tests are performed with a 1 : 1 scale dummy fuel element, with depleted uranium.
  - Vibration behaviour of the fuel element in dry and operating conditions,
  - pressure drop measurements at different flow rates,
  - start-up and shutdown tests of the main pump,

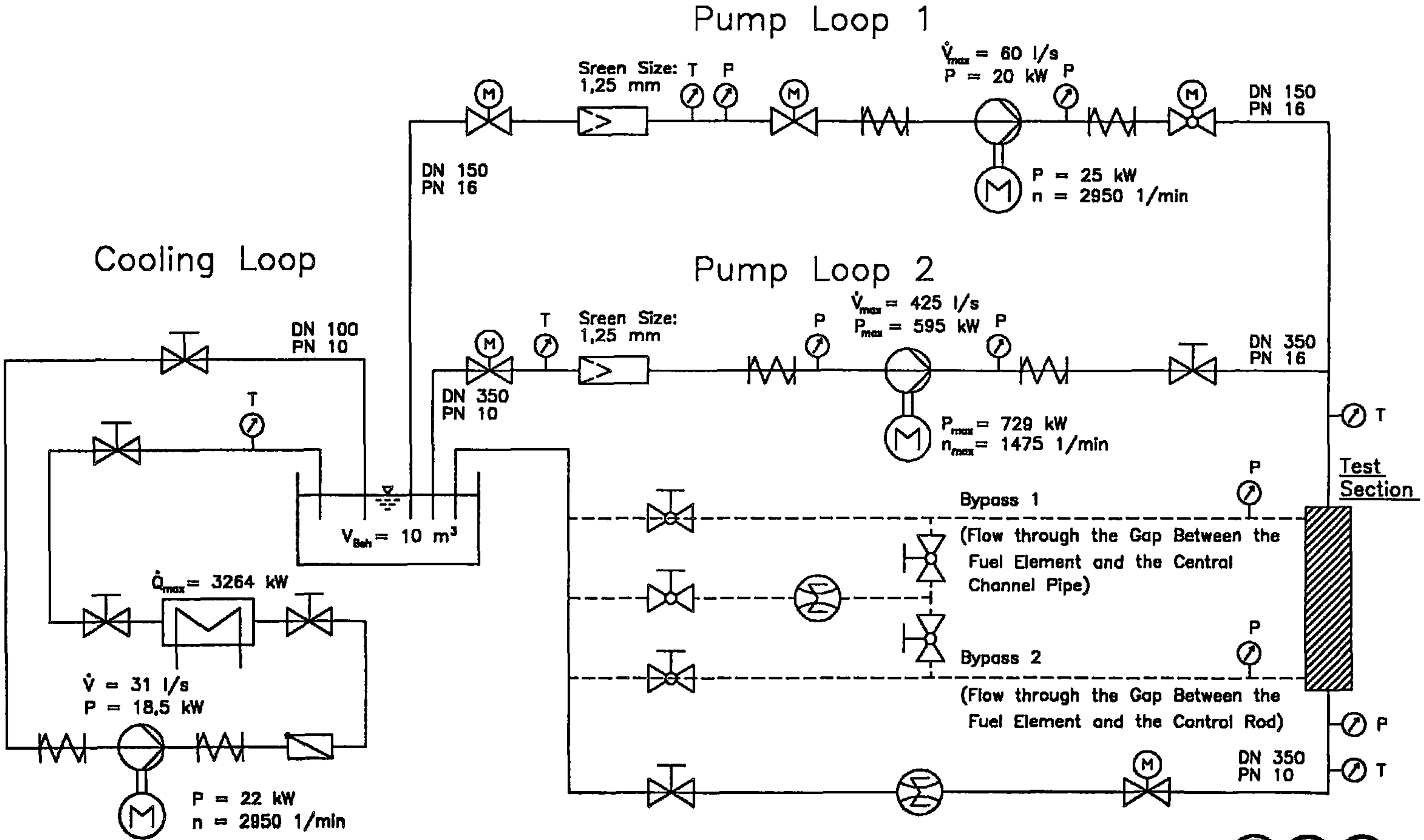
Objectives of the Experiments and the Test Program Performed (Continued)

- Test program (Continued):
  - Flow division in the region of the fuel element,
  - flow profile at the fuel element outlet in case of free sieve,
  - flow profile at the fuel element outlet in case of partly obstructed sieve,
  - endurance test of the fuel element, lasting 60 days (equivalent to 1.2 operating cycles).

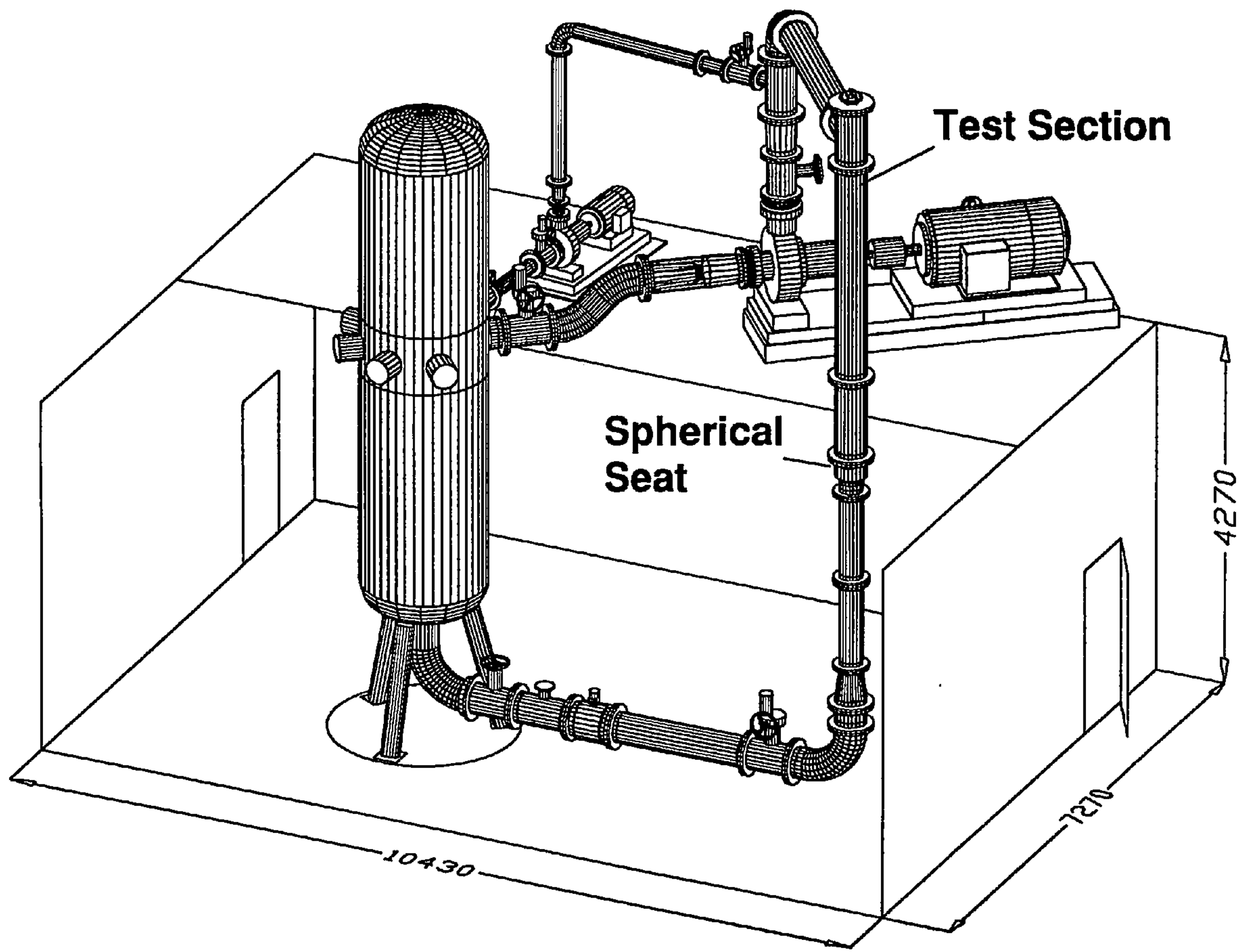
The Test Facility for the Qualification of the FRM-II Dummy Fuel Element

- A test facility was built at the Ruhr University of Bochum (Germany).
- The test facility mocks up the central region of the reactor coolant system in a 1 : 1 scale.
- The two pump units enable flow rates from 0 l/s to 60 l/s (small pump) and from 200 l/s up to 425 l/s (large pump).
- Presently tests are performed at a flow rate of 60 l/s (emergency core cooling), 300 l/s (nominal flow rate of the FRM-II) and 425 l/s (double hydraulic load relative to the nominal flow rate).

Block Diagram of the FRM-II Fuel Element Test Facility



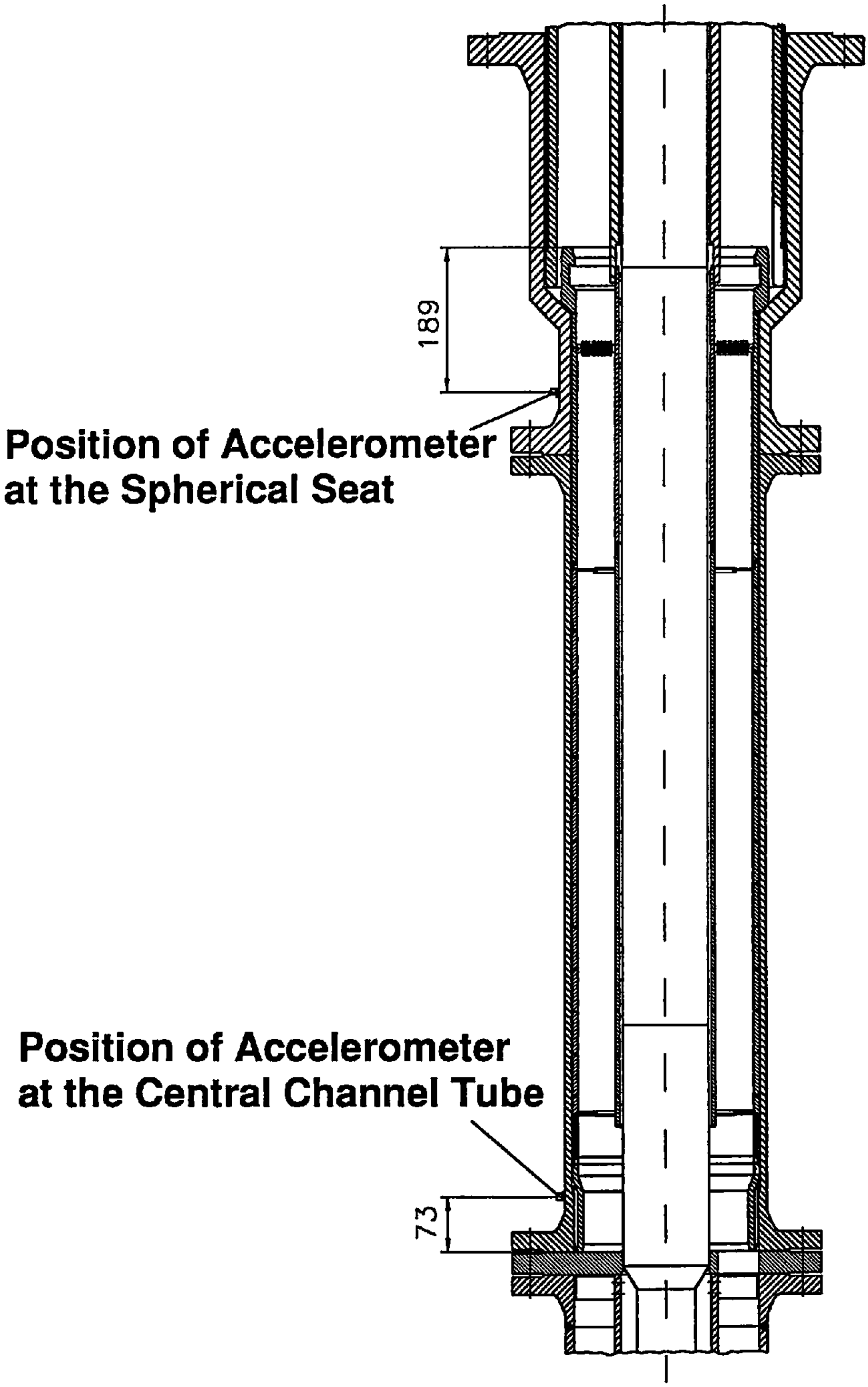
View of the FRM-II Fuel Element Test Facility



Vibration Measurements in Operating Conditions

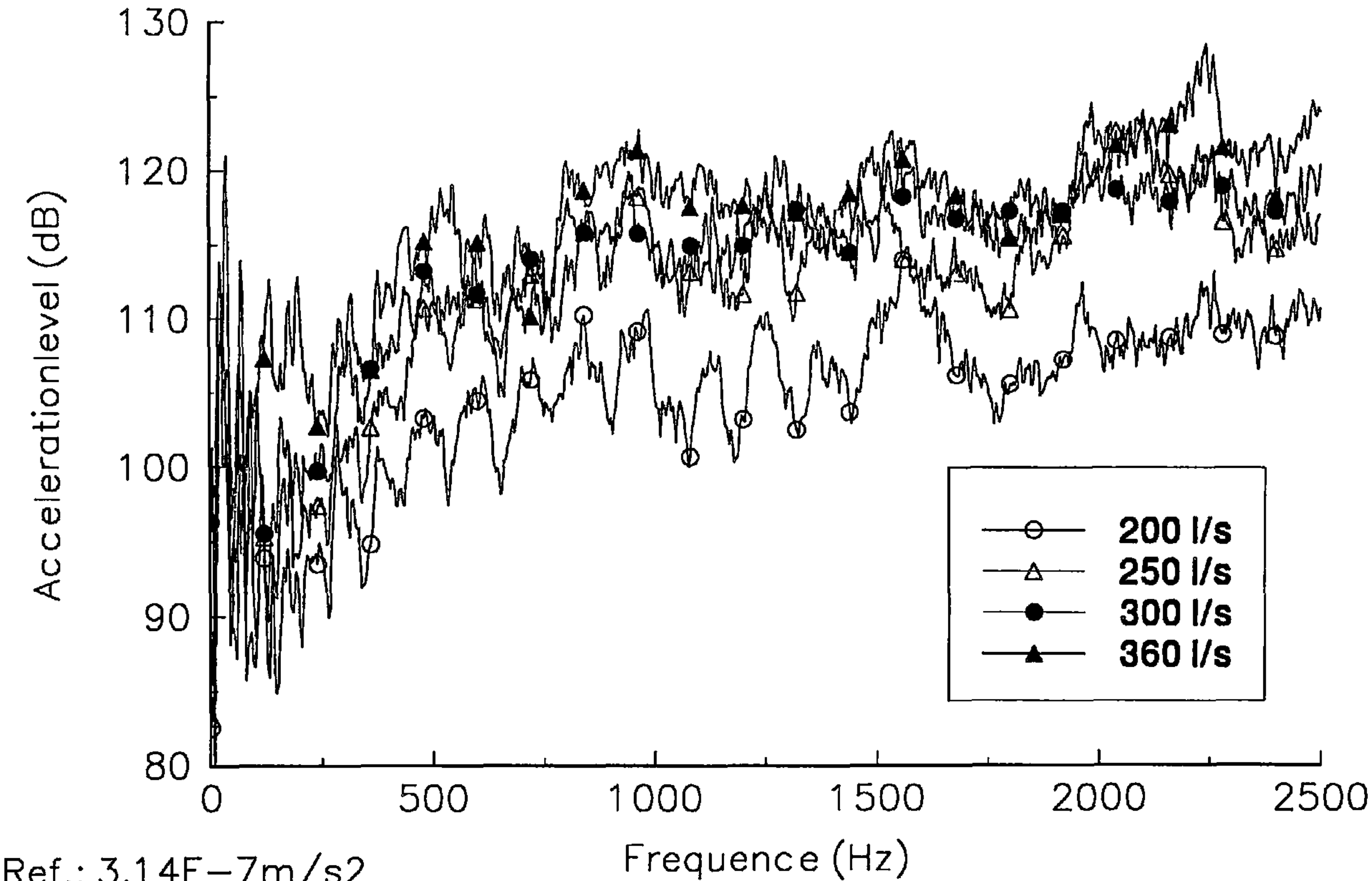
- Preliminary experimental modal analysis has proofed:
  Vibration measurement of the fuel element is possible at the outside of the central channel tube also.
- The vibration measurements in operating conditions are performed with accelerometers placed on different locations on the outside of the central channel tube.
- These tests are performed at flow rates up to 360 l/s.

Locations of the Accelerometers for the Vibration Measurement



Acceleration at the Spherical Seat

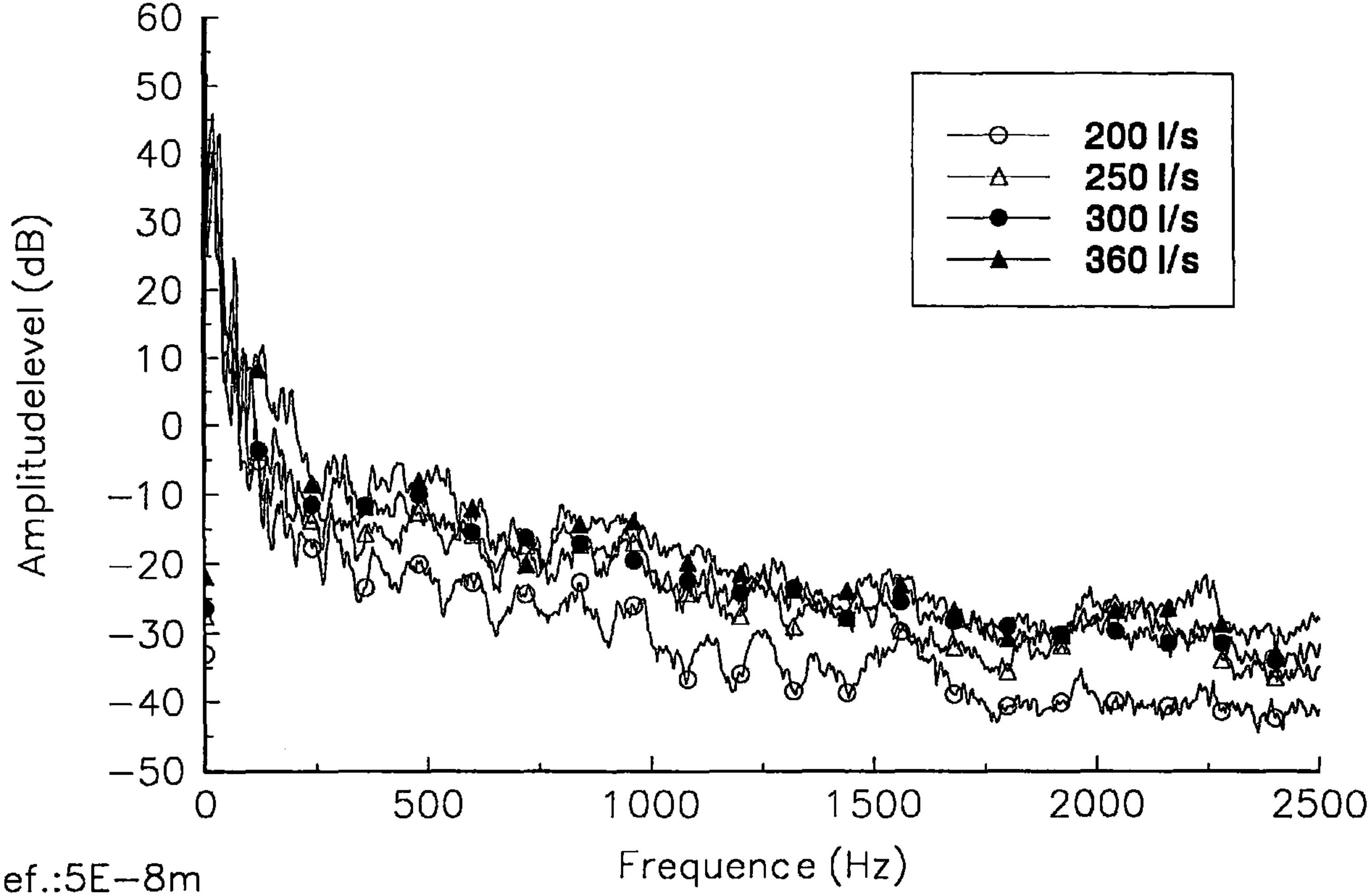
Flow Rate 200-360 l/s, Uranium Dummy Fuel Element



Ref.: 3.14E−7m/s2

Amplitudes at the Spherical Seat

Flow Rate 200-360 l/s, Uranium Dummy Fuel Element

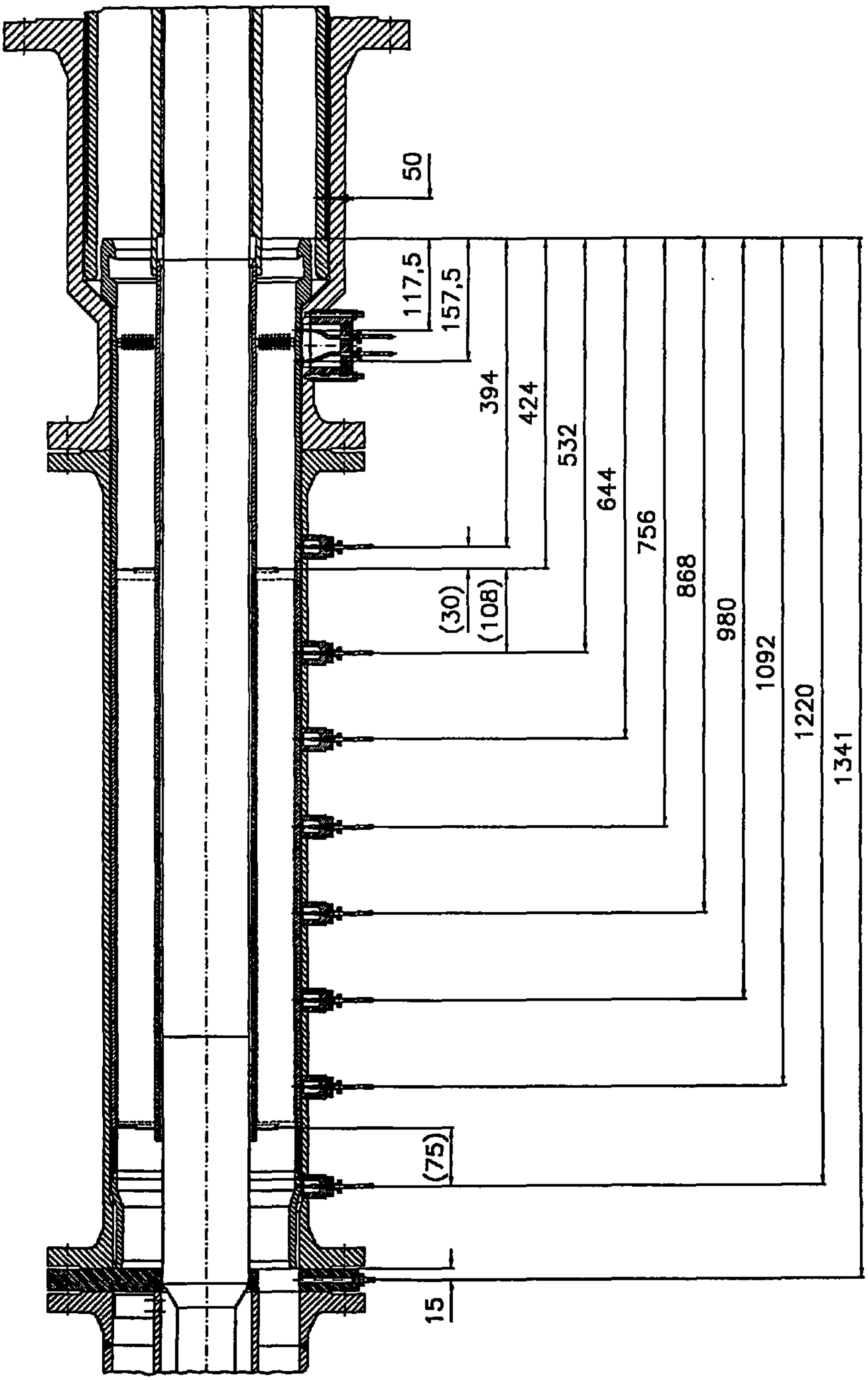


Ref.:5E-8m

Pressure Drop Measurements

- ☐ Measurements of the pressure drop within the fuel element, with particular interest in:
  - Pressure drop at the sieve,
  - pressure profiles within the plate zone of the fuel element.
- ☐ The tests are performed at flow rates up to 360 l/s.

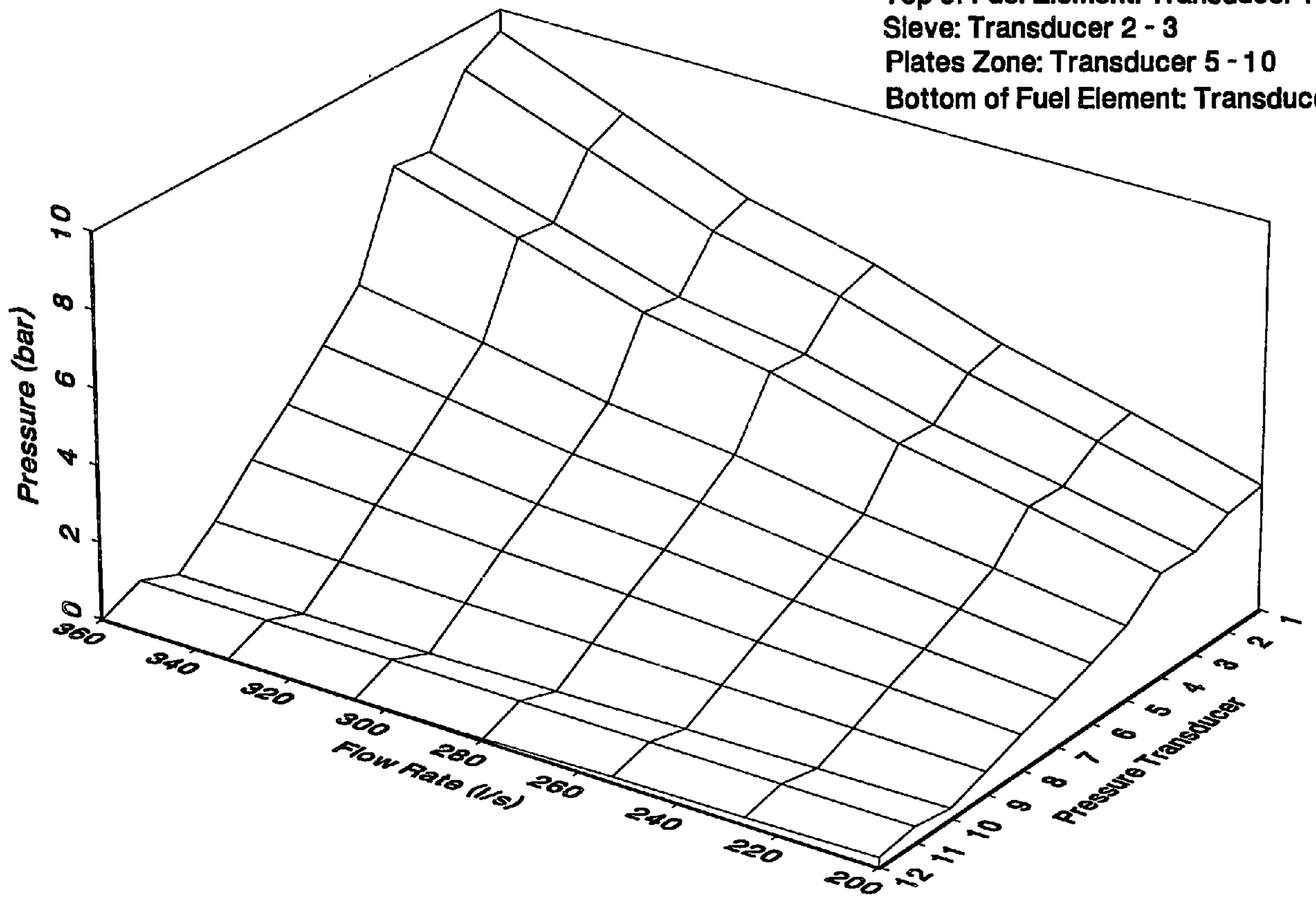
Locations of the Pressure Transducers for the Pressure Drop Measurements



Pressure Drop Within the Uranium Dummy Fuel Element

Flow Rate: 200 - 360 l/s

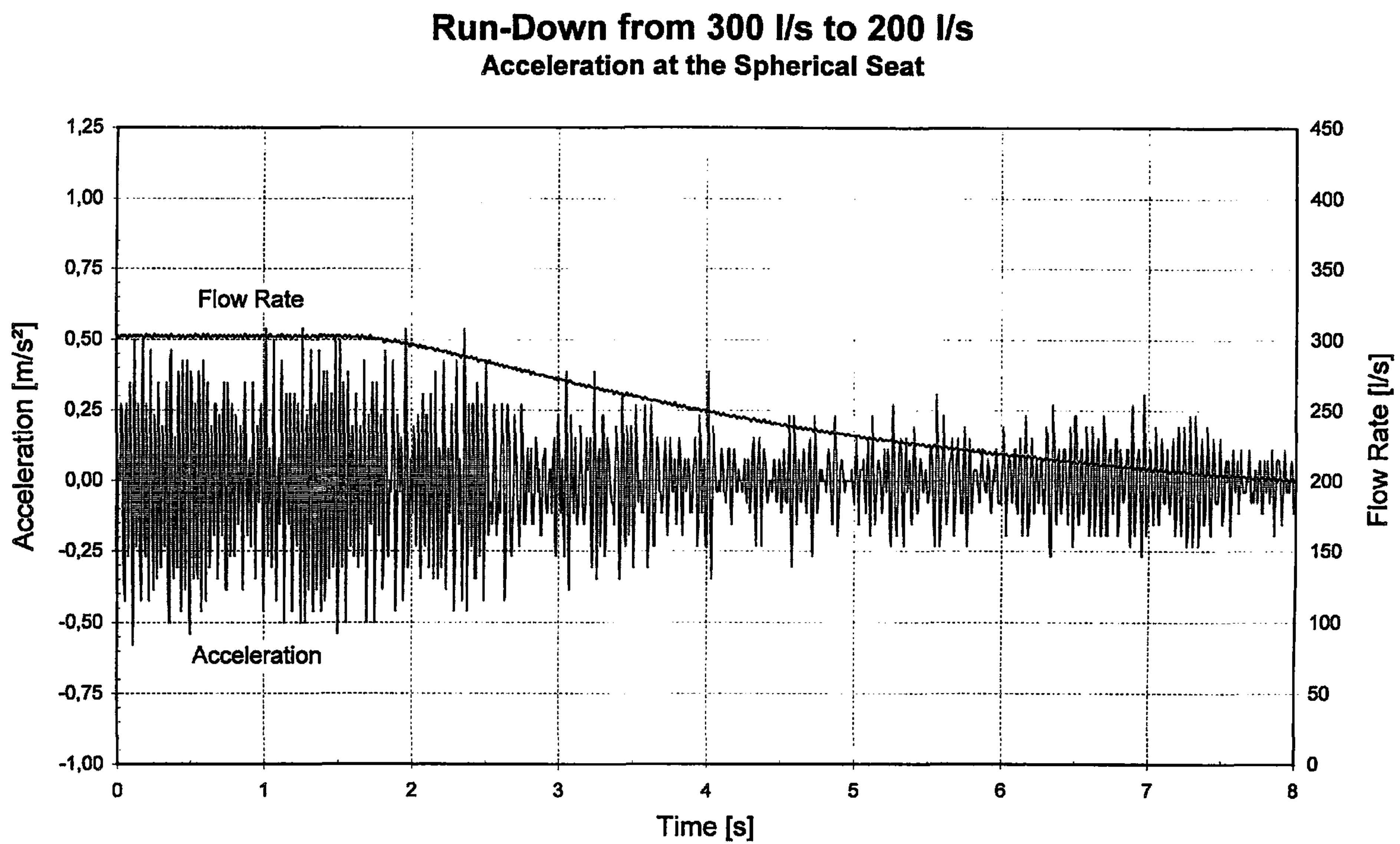
Top of Fuel Element: Transducer 1
Sieve: Transducer 2 - 3
Plates Zone: Transducer 5 - 10
Bottom of Fuel Element: Transducer 12

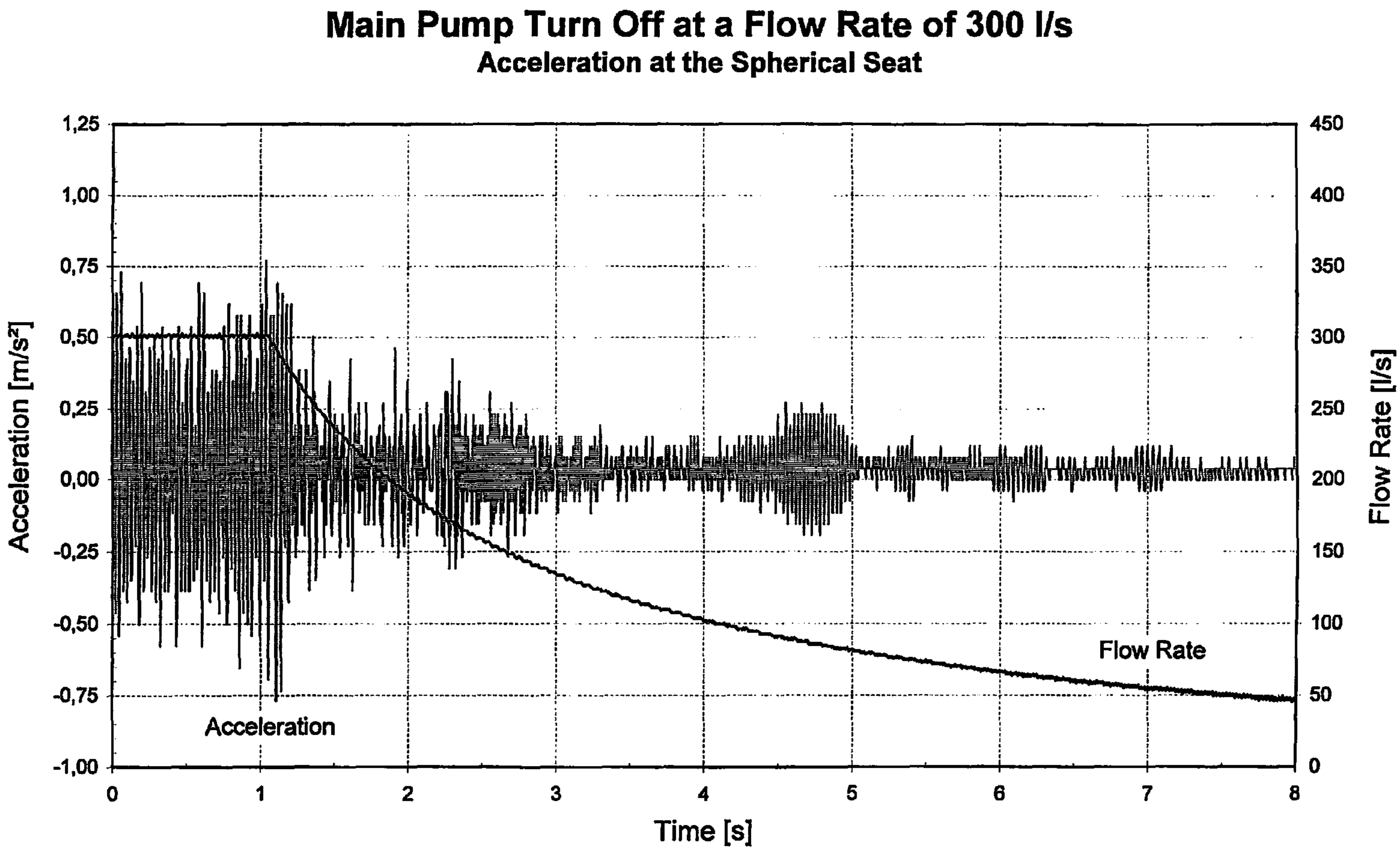


Start-Up and Shutdown Tests of the Main Pump

- ☐ Acceleration measurements at the spherical seat of the fuel element and its bottom region during pump speed run-up and shutdown.

- ☐ Test program:
  - Main pump speed run-down from a flow rate of 300 l/s simulating the run-down of the primary pumps of the FRM-II,
  - shutdown of the main pump at a flow rate of 300 l/s simulating the blockage of one of the four primary pumps of the FRM-II,
  - starting of main pump, test of transient phase from idling conditions (200 l/s) up to the nominal flow rate of 300 l/s.


Run-Down from 300 l/s to 200 l/s
Acceleration at the Spherical Seat
Flow Rate
Acceleration
Acceleration [m/s²]
Flow Rate [l/s]
Time [s]
1,25
1,00
0,75
0,50
0,25
0,00
-0,25
-0,50
-0,75
-1,00
450
400
350
300
250
200
150
100
50
0
0
1
2
3
4
5
6
7
8



Main Pump Turn Off at a Flow Rate of 300 l/s
Acceleration at the Spherical Seat
Acceleration [m/s²]
1,25
1,00
0,75
0,50
0,25
0,00
-0,25
-0,50
-0,75
-1,00
Flow Rate [l/s]
450
400
350
300
250
200
150
100
50
0
Time [s]
0
1
2
3
4
5
6
7
8
Acceleration
Flow Rate

Summary

- ☐ A test facility for the qualification of the FRM-II fuel element was built at the Ruhr-University Bochum (Germany).
- ☐ The test facility mocks up the central region of the reactor coolant system in a 1 : 1 scale.
- ☐ First experimental results:
  - Experimental modal analysis:
    Vibration measurements of the fuel element are possible at the outside of the central channel tube also.
  - Vibration measurements during operation:
    - Acceleration and amplitude levels increase with increasing flow rates; qualitative vibration behavior in relation to the frequency is maintained,
    - acceleration levels show maxima at frequencies which coincide with fuel element eigenfrequencies determined by the experimental modal analysis.

Summary (Continued)

- First experimental results (Continued):
  - Pressure drop measurements:
    - Pressure drop increases steadily with increasing flow rate,
    - pressure drop increases constantly within the plate zone of the fuel element.
  - Main pump speed run-up and shutdown tests:
    No major resonance regions can be investigated during transient pump operation.

A STATUS REPORT ON THE PROPOSED CANADIAN IRRADIATION RESEARCH FACILITY

A.G. Lee, W.E. Bishop and G.E. Gillespie

IGORR 5
1996 November 4-6

Topics

- Overview of IRF Pre-Project Activities
- Update on IRF Concept Description
- Summary

Pre-project Phase: 95/11/01-97/03/31

- **Advance the Project Schedule**
- **Reduce Risk to Cost and Schedule**

by

- **Advancing the Analysis**
- **Advancing the Testing**
- **Advancing Fuel Development**
- **Developing the Concept**
- **Starting the Licensing Process**

IRF Pre-Project Activities

Advance Development of System Concepts for IRF

- identifying technical feasibility issues and developing feasible system concepts
- confirming feasibility for manufacturing components
- improving definition of project work scope
- developing concepts for novel systems (e.g., fuel handling for test loops)
- updating requirements for experimental facilities
- advancing system design
- investigating alternative IRF building concepts

IRF Pre-Project Activities

Up Front Licensing Activities

- prepare IRF Project QA Program - *Complete*
- prepare Licensing Plan - *AECB Review*
- prepare Licensing Basis Document - *AECB Review*
- prepare Safety Analysis Program
- prepare Safety Design Guides - *75% Complete*
- define requirements for environmental assessment
- prepare software validation documents
- develop criteria for containment design
- prepare preliminary assessment of fuel disposal options - *Complete*

Up Front Licensing Objectives

- Familiarize AECB staff with IRF concept
- Reach agreement with AECB on IRF QA Program
- Identify major safety analysis, design and licensing issues
- Reach agreement with AECB on licensing basis
- Define requirements for environmental assessment
- Define design criteria for containment

IRF Pre-Project Activities

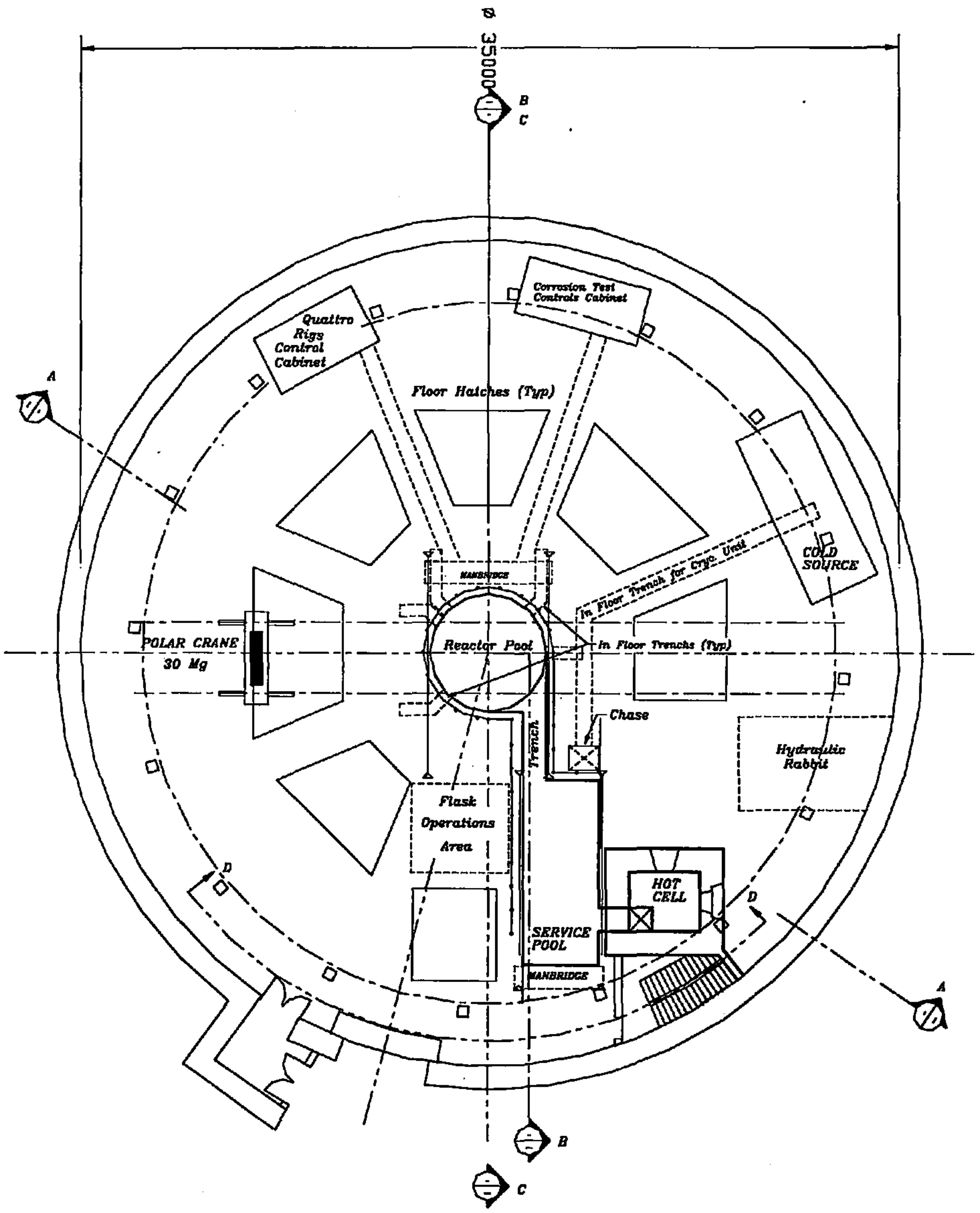
Analysis and Testing

- Review thermalhydraulics and physics methods and actions added to IRF Pre-project plan - *Complete*
- Building detailed thermalhydraulics and physics models - *75% Complete*
- Performing calculations to support safety analysis and design
- Assessing requirements and code capabilities for 3-D kinetics
- Preparing validation of thermalhydraulics and physics codes - *Technical Basis Document in review*

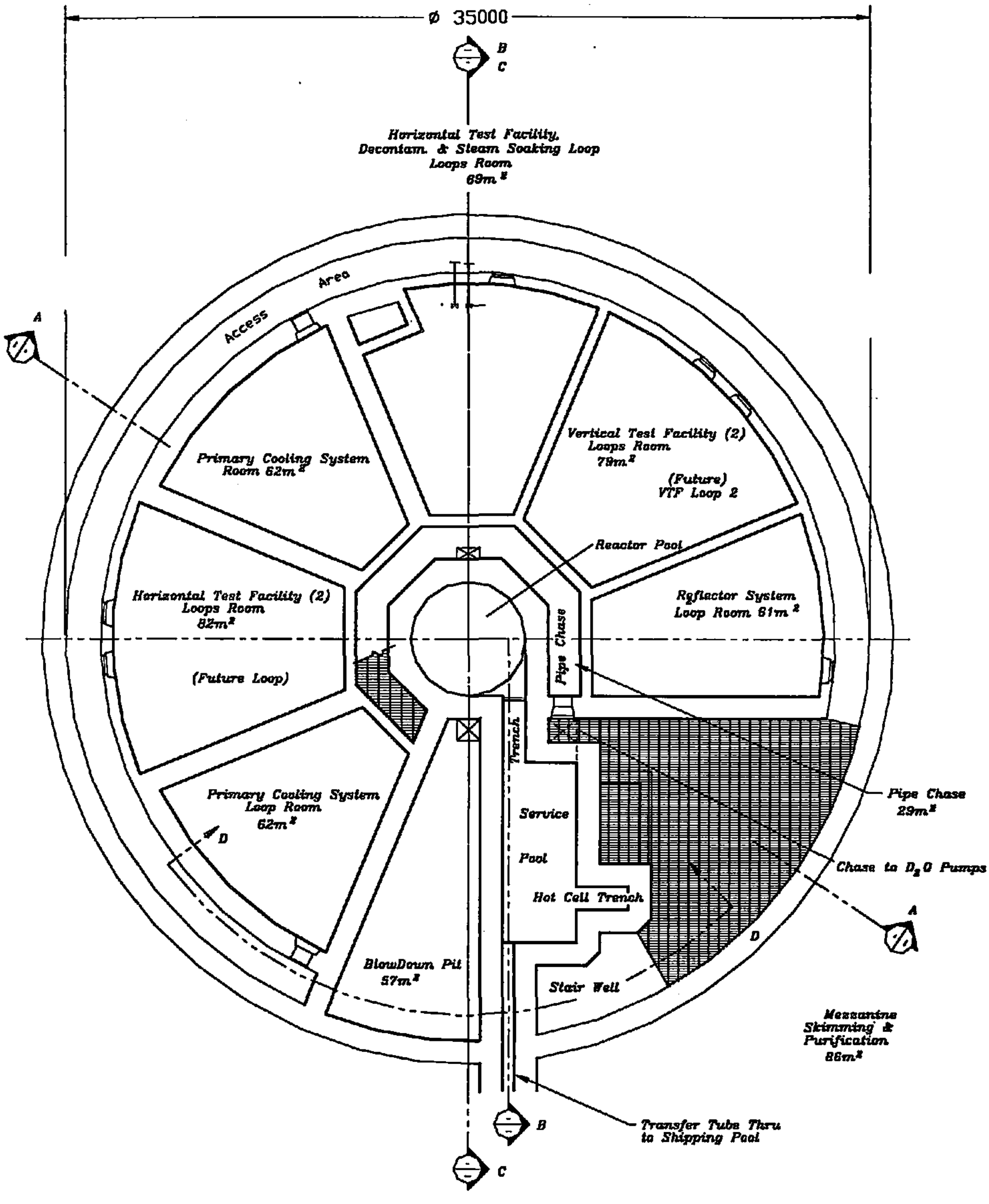
IRF Pre-Project Activities

Analysis and Testing

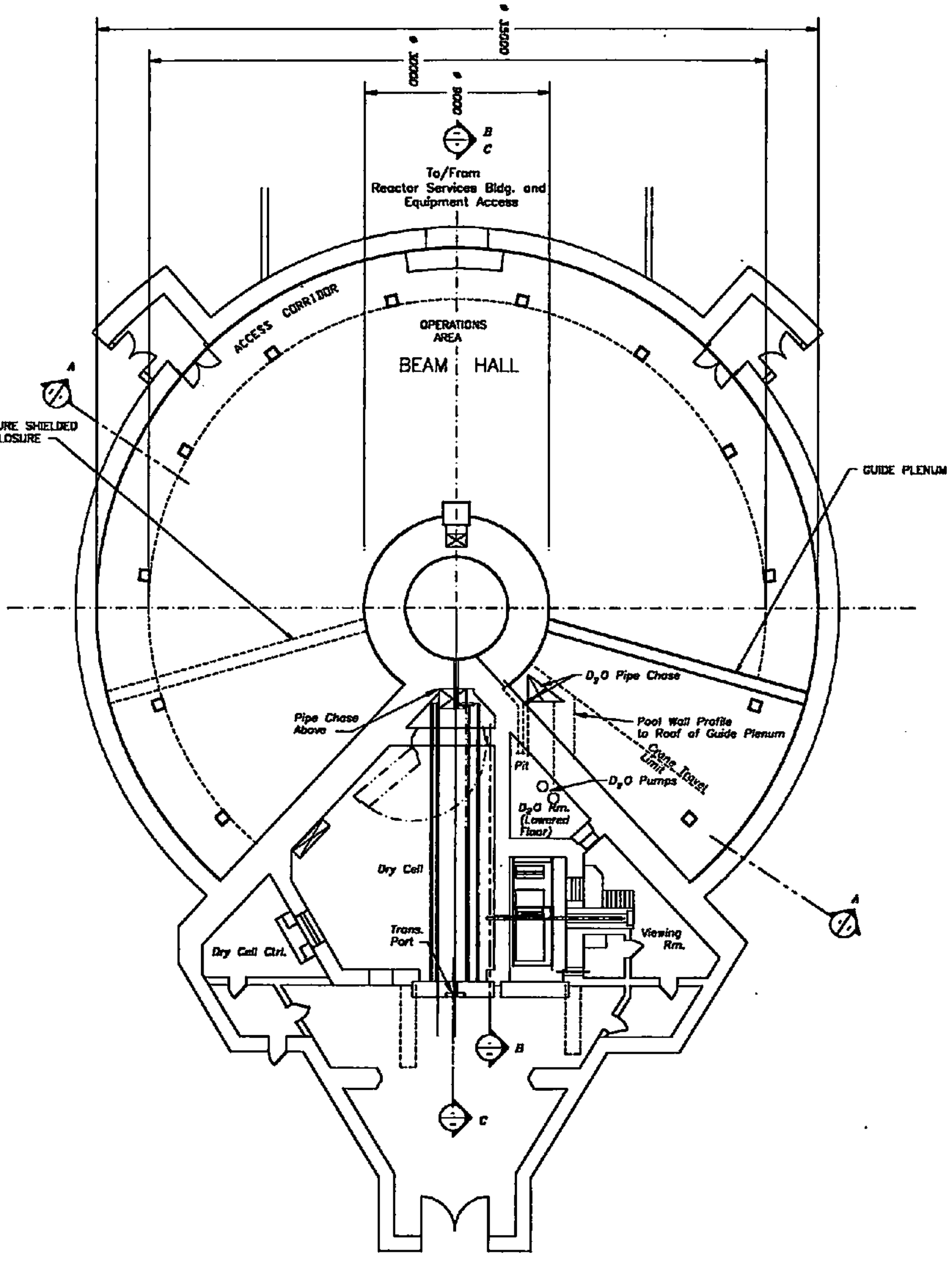
- **Performing heat transfer measurements to improve correlations for CATHENA - *Tests complete***
- **Performing bundle CHF tests for CATHENA validation**
- **Measuring thermal conductivity for irradiated fuel**
- **Measuring oxide layer on irradiated fuel**
- **Developing techniques for fabricating U_3Si_2-Al fuel rods with burnable poison**
- **Assessing need for physics parameter measurements using ZED-2 - *Complete, experiments not required***



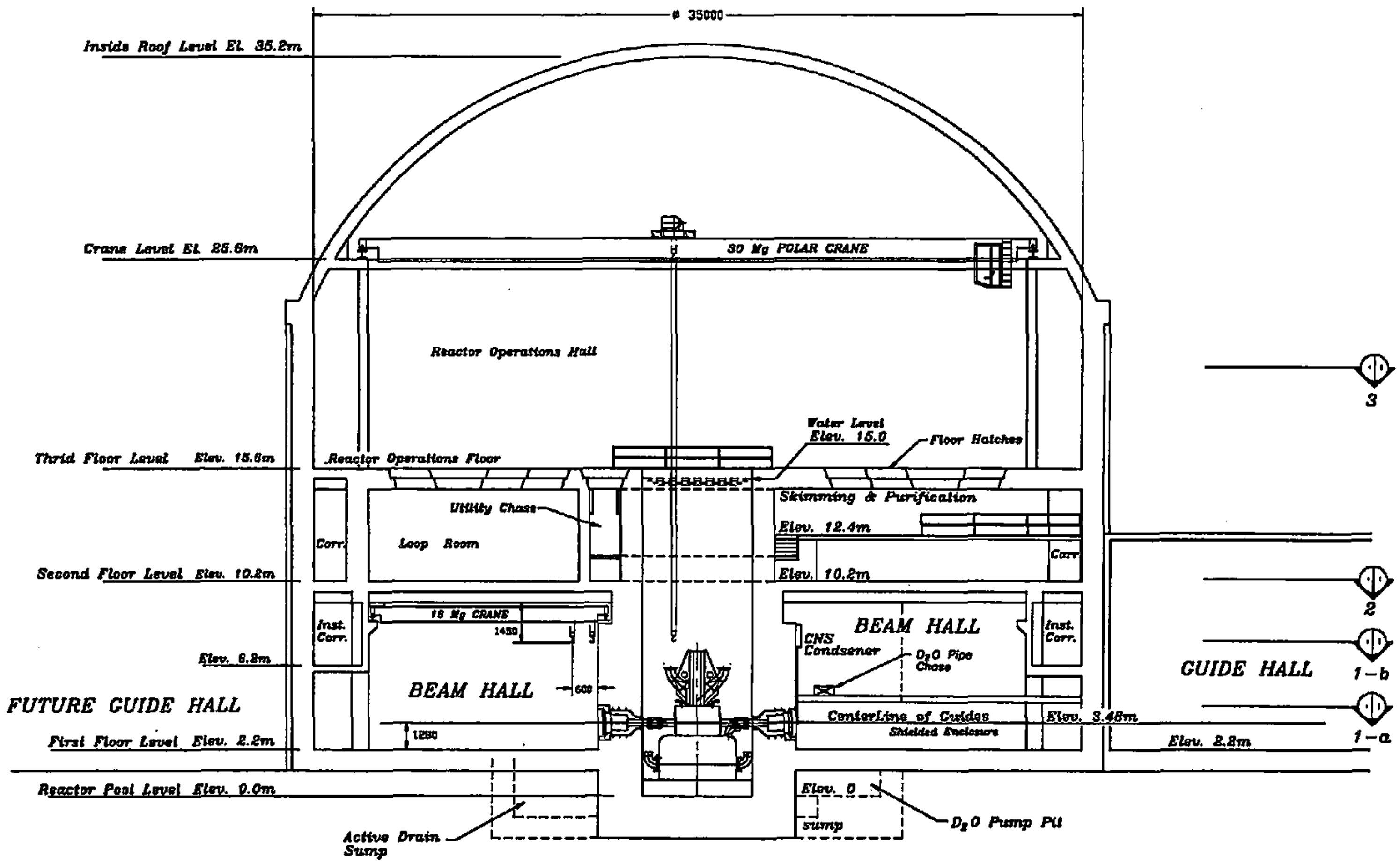
ø 35000
B
C
A
Corrosion Test
Controls Cabinet
Quattro
Rigs
Control
Cabinet
Floor Hatches (Typ)
MANBRIDGE
In Floor Trench for Cryo. Unit
COLD
SOURCE
POLAR CRANE
30 Mg
Reactor Pool
In Floor Trenchs (Typ)
Chase
Hydraulic
Rabbit
Trench
Flask
Operations
Area
D
HOT
CELL
SERVICE
POOL
MANBRIDGE



Ø 35000
B
C
Horizontal Test Facility,
Decontam. & Steam Soaking Loop
Loops Room
69m²
Access Area
A
Primary Cooling System
Room 62m²
Vertical Test Facility (2)
Loops Room
79m²
(Future)
VTF Loop 2
Reactor Pool
Horizontal Test Facility (2)
Loops Room
82m²
Reflactor System
Loop Room 61m²
Pipe Chase
(Future Loop)
Trench
Primary Cooling System
Loop Room
62m²
Service
Pool
Pipe Chase
29m²
Chase to D₂O Pumps
D
Hot Cell Trench
A
BlowDown Pit
57m²
Stair Well
D
Mezzanine
Skimming &
Purification
86m²
B
Transfer Tube Thru
to Shipping Pool
C



To/From Operations Bldg.

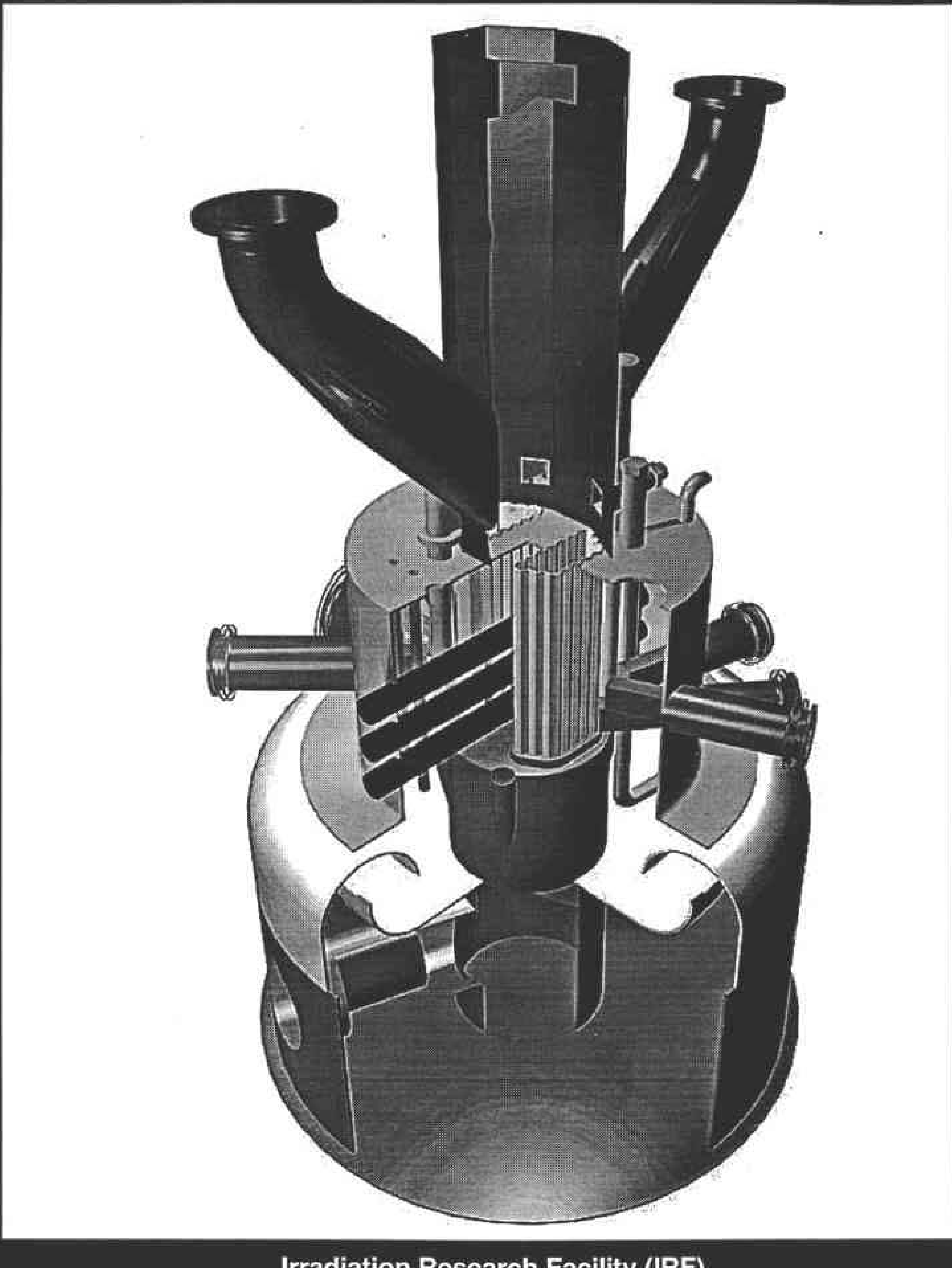


ø 35000
Inside Roof Level El. 35.2m
Crane Level El. 25.6m
30 Mg POLAR CRANE
Reactor Operations Hall
Water Level
Elev. 15.0
Floor Hatches
Thrid Floor Level Elev. 15.6m
Reactor Operations Floor
Utility Chase
Skimming & Purification
Elev. 12.4m
Corr.
Loop Room
Corr
Second Floor Level Elev. 10.2m
Elev. 10.2m
18 Mg CRANE
1450
Inst. Corr.
BEAM HALL
Inst. Corr.
CNS Condsener
D_2O Pipe Chase
Elev. 6.2m
BEAM HALL
600
GUIDE HALL
FUTURE GUIDE HALL
CenterLine of Guides
Shielded Enclosure
Elev. 3.48m
1280
First Floor Level Elev. 2.2m
Elev. 2.2m
Reactor Pool Level Elev. 0.0m
Elev. 0
sump
D_2O Pump Pit
Active Drain Sump
3
2
1-b
1-a

Performance

- Operating Power, Nominal 40 MW
- Horizontal Test Sections
  - Test Section Power- 1300 to 2000 kW
  - CANDU Bundle Power -670 to 1200 kW
  - Outer Element Power -53 to 77 kW/m
- Beam Tubes
  - Unperturbed Flux 2.7 to 3.8 $\times 10^{18}$ $n/m^2/s$
  - Perturbed Flux at Cold Source Chamber 2.2 $\times 10^{18}$ $n/m^2/s$

AECL EACL



Irradiation Research Facility (IRF)
Reactor Structure - Sectional View

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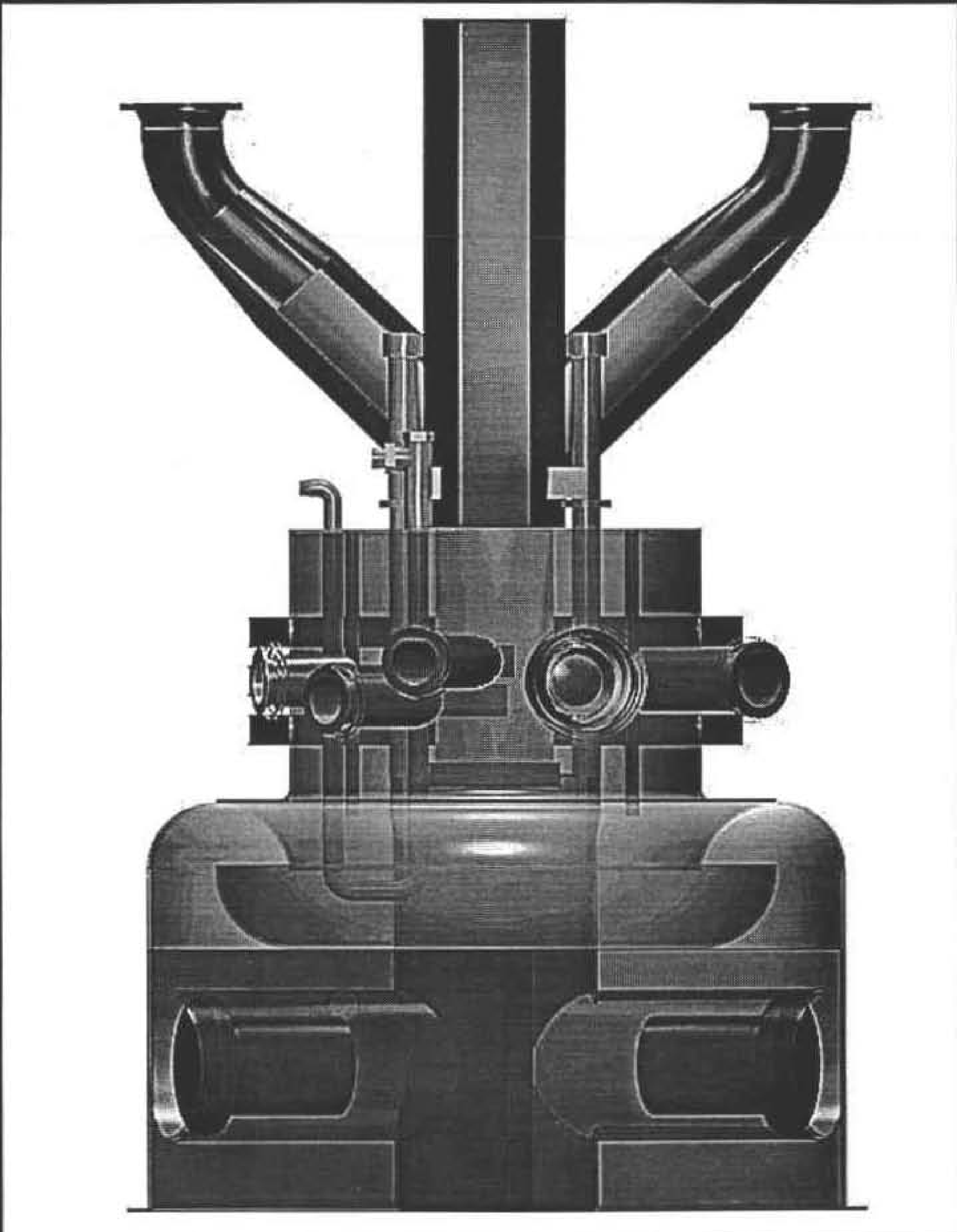

Irradiation Research Facility (IRF)
Reactor Structure - Front View

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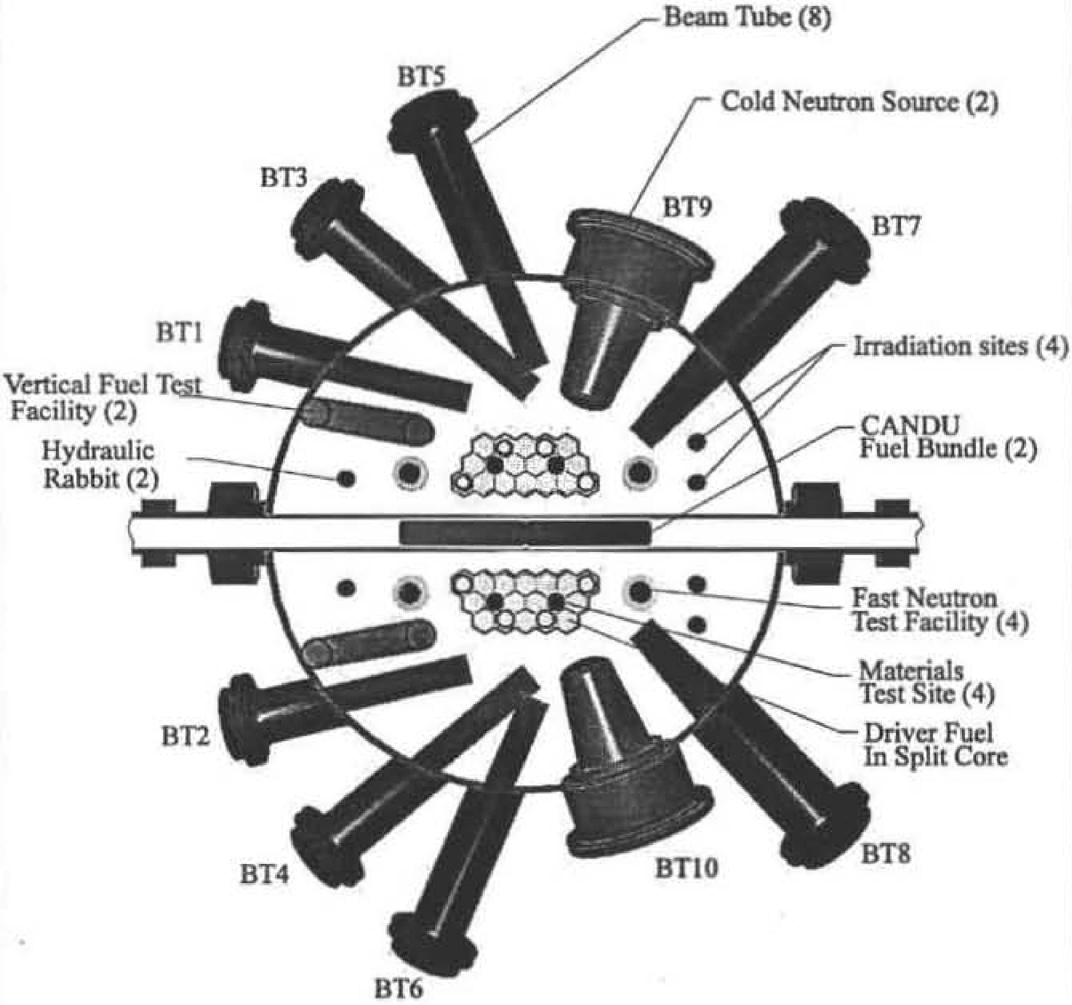


Irradiation Research Facility (IRF)

Reactor Plan View

Major Features

- Maple Reactor, Open Tank in Pool, Vertical Core
- Two Vertical Core Segments
- LEU U_3Si_2 in AL Fuel, Al Clad
- Light Water Coolant, Upward Flow
- Heavy Water Reflector
- Hafnium Metal Control Absorbers
- Shutdown System 1 -Absorber Drop
- Shutdown system 2 -Reflector Dump
- Reactor Building

Update on IRF Concept

- Revise IRF Complex layout and concepts
- Reactor based on a MAPLE-type reactor assembly
- Design experimental facilities for CANDU support
- Design experimental facilities for beam research programs
- Revise cost and schedule estimates
  - reference cost estimate of $500 million
  - reference schedule of 87 months
  - risk of schedule delays of 12-24 months

Validation Methodology

Steps 3, 4 & 5 are code dependent

- **Validation Plan**
  - **Demonstrate that code version accurately represents governing phenomena**
- **Validation Exercises**
  - **Compare model predictions with selected data sets**
- **Validation Manual**
  - **Summarize code accuracy, sensitivities, and uncertainties for specific applications**

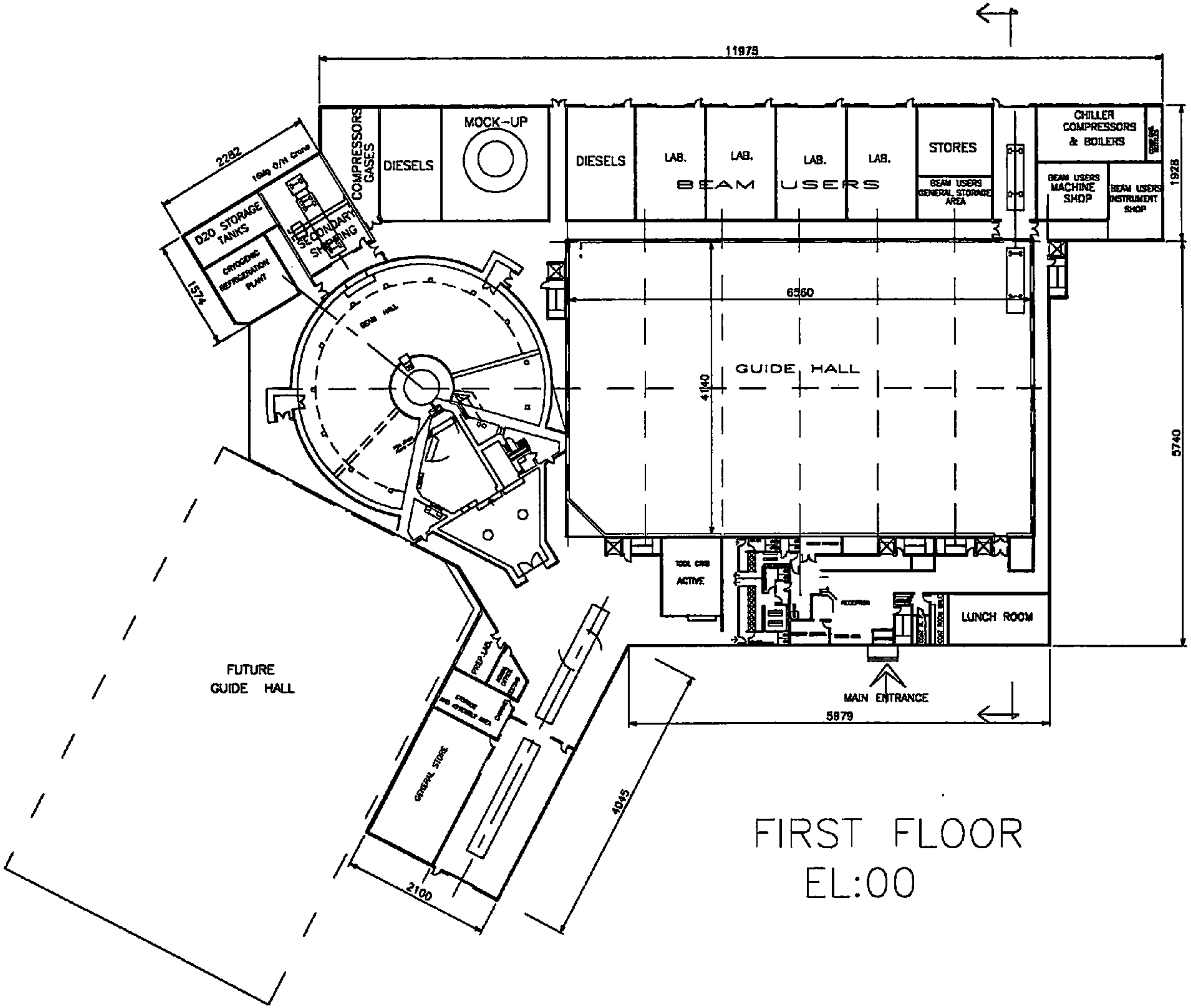
Validation of Computer Codes Used in Safety Analysis for IRF

- Adopted validation-matrix methodology
- Based validation framework on NEA/OECD work for LWR community and CANDU industry work
- Three major disciplines into which physical phenomena can be grouped
  - physics
  - thermalhydraulics
  - safety-related analyses (e.g., containment behaviour, fission product release & transport)
- Five major steps in validation

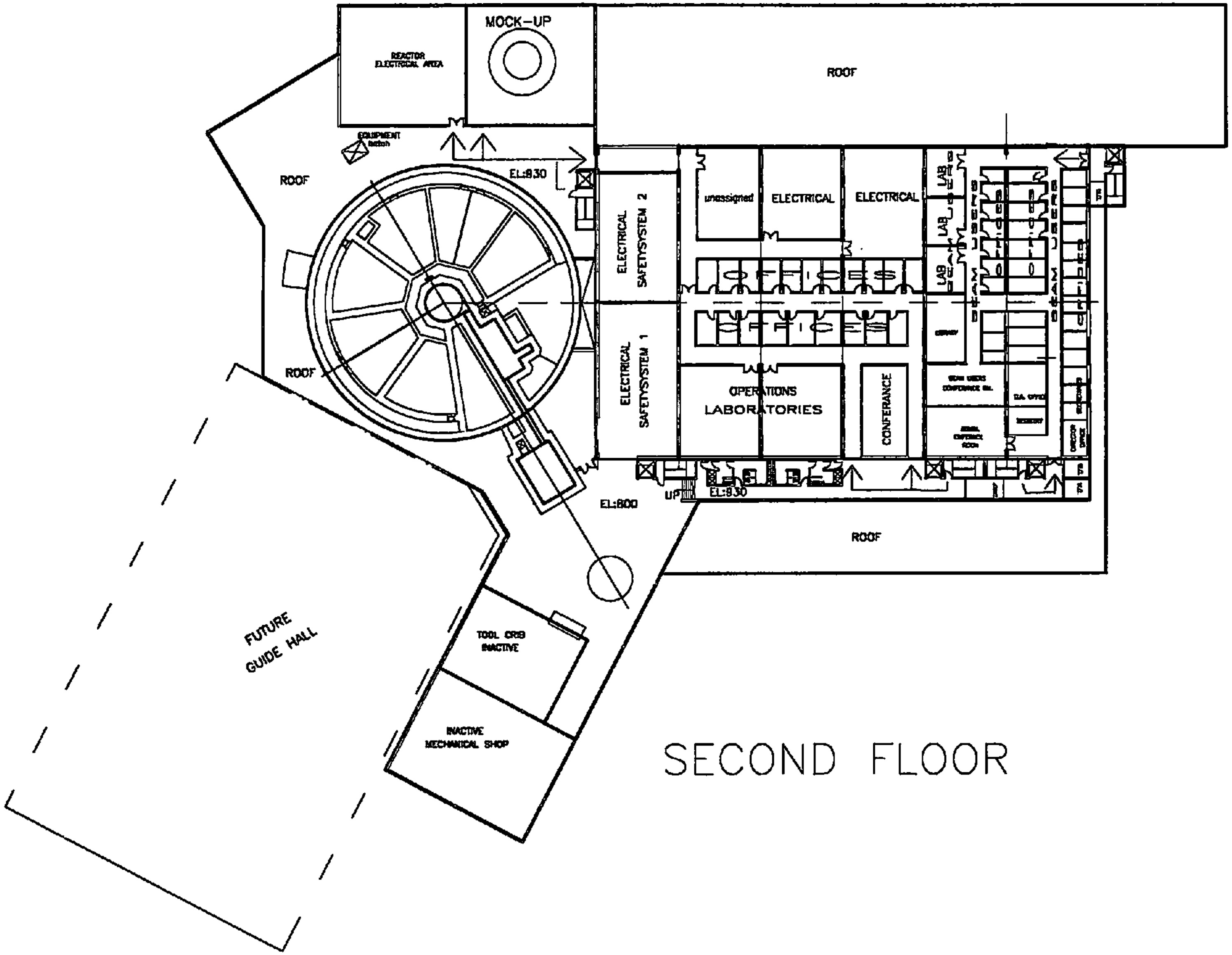
Validation Methodology

Steps 1 & 2 are code independent

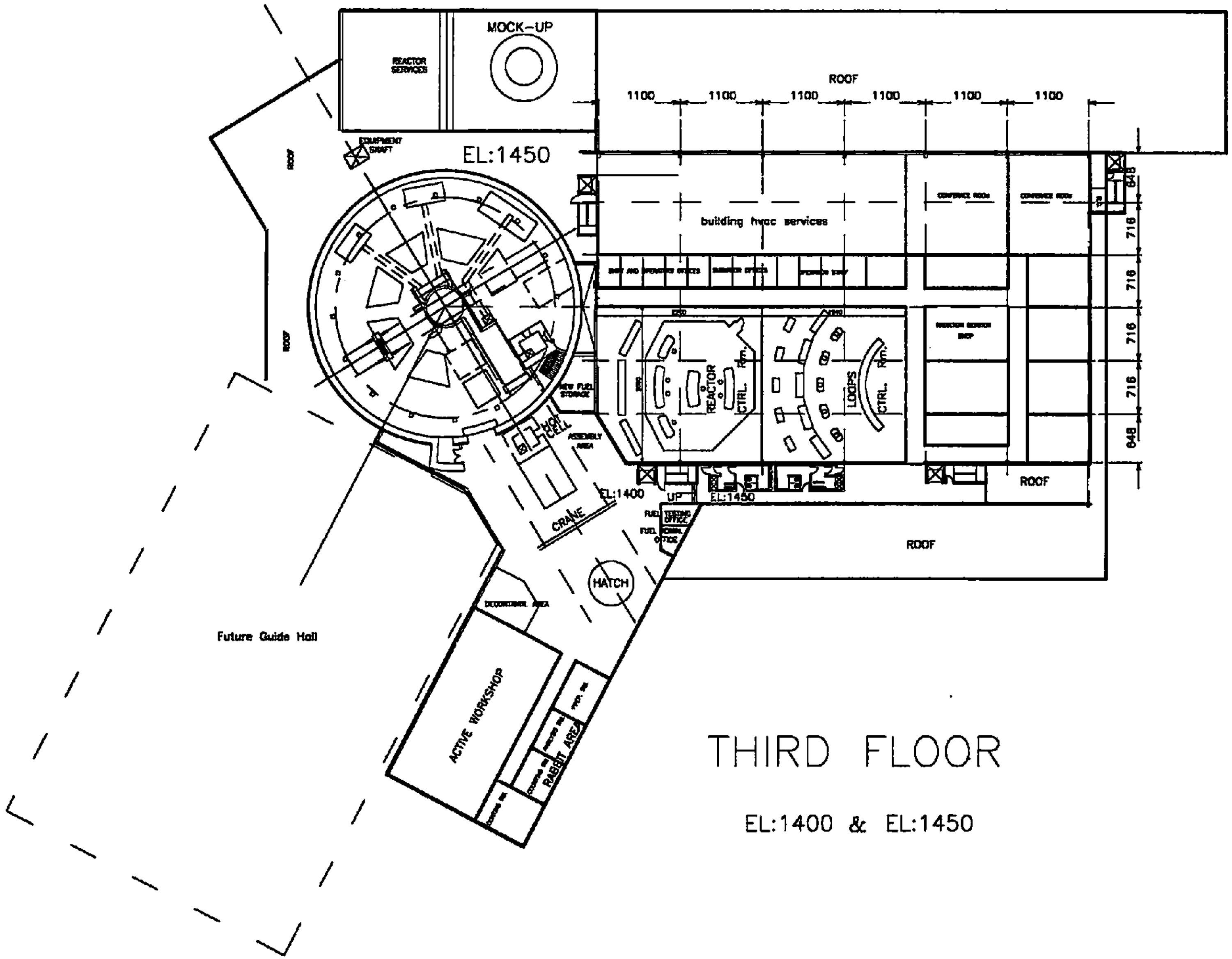
- **Technical Basis Document**
  - **Relate safety concerns, behaviours of plant subsystems and main phenomena during specific accidents**
- **Validation Matrices**
  - **Relate all relevant phenomena to accidents and data sets**



FIRST FLOOR
EL:00

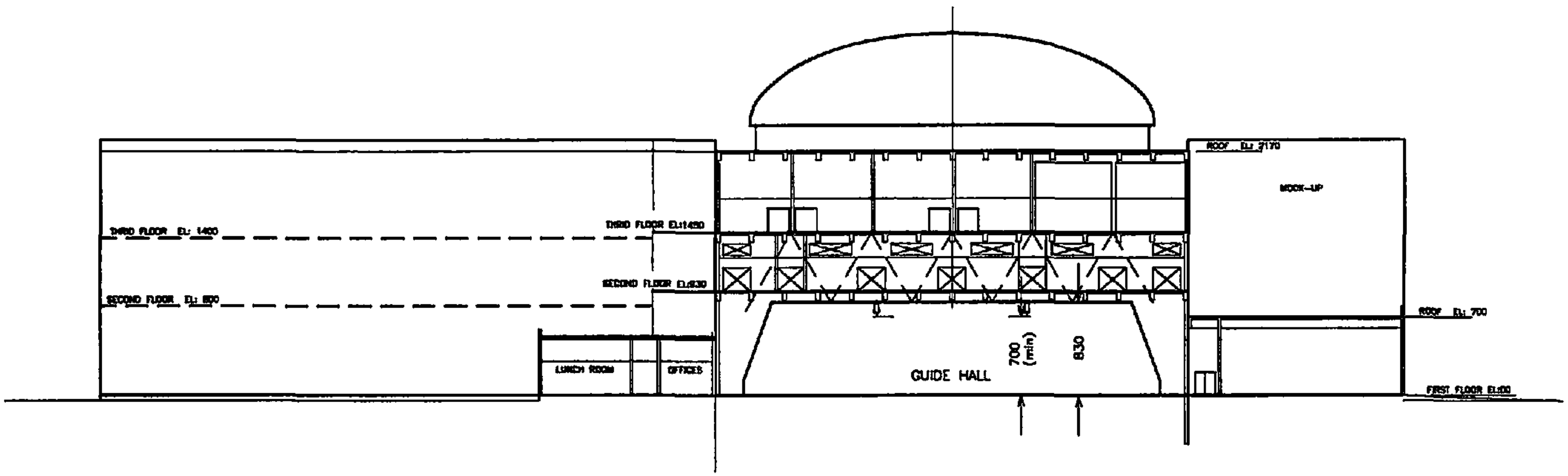


SECOND FLOOR



THIRD FLOOR

EL:1400 & EL:1450


ROOF EL: 2170
MOCK-UP
THIRD FLOOR EL: 1400
THIRD FLOOR EL:1450
SECOND FLOOR EL:830
SECOND FLOOR EL: 800
ROOF EL: 700
LUNCH ROOM
OFFICES
GUIDE HALL
700 (min)
830
FIRST FLOOR EL:00

SECTION A-A

Preliminary Study of Core Characteristics for TRR-II

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H. M. Hsieh, D. Y. Yang, J. A. Jing, and S. K. Chen

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Fax 886-3-4711404

TRR-II is the Taiwan Research Reactor Remodeling Project. The proposed multipurpose reactor is a water-moderated and cooled pool type of 10 MW to 20 MW design. Enough irradiation space will be provided for radioisotope production, silicon transmutation doping, material and fuel tests, etc. Neutronic and thermal-hydraulic performances were studied for a core using typical PWR UO_2 rod fuel configurations. The core is surrounded by a concentric cylindrical aluminum heavy water tank as the reflector. A large fraction of the fission neutrons will leak out the core surface and slow down in the reflector to form a high flux peak with very pure spectrum of thermal neutrons. The coolant is designed to be down flow. Calculation results show that the maximum unperturbed thermal flux in heavy water reflector is about 1×10^{14} n/cm^2 • s. The design concept of the core is that it has great flexibility of using alternative fuel designs for core (i.e., neutron flux) upgrade.

1。 Introduction

The old Taiwan Research Reactor (TRR) was a 40 MW(th), heavy water moderated and cooled, graphite reflected research reactor. It was permanently shutdown in 1988 and will be remodeled, designed and constructed domestically as a multi-purpose research reactor to meet national demand in the fields of radioisotope production, basic research in neutron applications, and industrial and medical uses.

To satisfy the above utilization goals, the design of Taiwan Research Reactor Remodeling (TRR-II) has to meet certain performance requirements as well as economic aspect. The design principles are as follows.

a. The maximum available thermal neutron flux in the flux trap should be at least 1×10^{14} n/cm^2.sec
b. Low enriched uranium fuel is chosen by taking into consideration the easiness of obtaining.
c. The aluminum alloy is chosen for the core components to reduce neutron absorption and radiation dose under the maintenance work.
d. A conventional swimming pool design is chosen and control rods are driven by the control rod driving mechanism installed below the core for easy access to the core.
e. The coolant flow is planned to be downward for the purpose of reduction of the radiation dose cause of ^{16}N which is produced by the (n,p) reaction in ^{16}O, and a ^{16}N decay tank is installed in the primary cooling system.

The main goal of this report is to introduce the current progress in the preliminary study of core characteristics for TRR-II. The brief description of reactor is given in section 2. Section 3 dipicts the calculation models regarding to nuclear design, thermal-hydraulic design and accident analysis. Section 4 summarizes the nuclear and thermal-hydraulic characteristics of TRR-II. The conclusions in this study is given in section 6.

2。 Reactor Description

2.1 Fuel

One of the unique feature of TRR-II is to use domestic made UO_2 rod. It is very well known for their extensive use in LWR. Their physical properties, in pile behavior, manufacturing features etc, are well mastered. The high density 10.3g/cm^3 of UO_2 matrix leads to a higher uranium content, making it possible to reduce the enrichment about 3 to 10% 。

The fuel assembly of TRR-II consists of 6 × 6 UO_2 fuel rods. The diameter of the fuel rod is 9.5mm with zircloy as cladding.

2.2 Core

The core is cylindrical in shape, about 0.6m in diameter and about 0.6m in height. It is composed of fuel assemblies, control rods, irradiation elements and beryllium reflectors. Its average power density is 110 kw/l at the rated power operation. A configuration of the core is shown in Figure 1.

The control rods are made of hafnium, and connected to the aluminum follower elements. They are driven by the control rod driving mechanism installed beneath the core.

The irradiation element is similar to a fuel assembly in its outer dimension and has an irradiation hole of 60mm diameter at the center. There are five elements in the core.

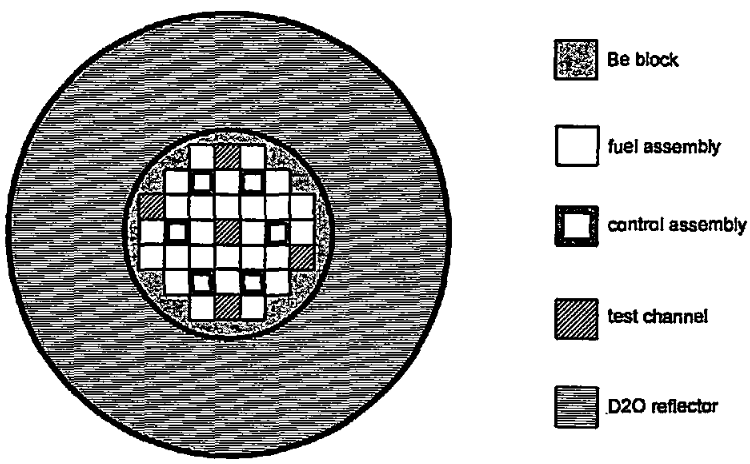

Figure 1. TRR-II core configuration

Beryllium reflectors are installed between the fuel region and the inner wall of the heavy water tank. There are four irradiation holes in the beryllium reflectors.

The heavy water tank is a double cylindrical type aluminum vessel, with a height of about 1.2m and an outer diameter of about 2m. Irradiation thimbles, horizontal beam tubes and cold neutron source facilities are installed in it. A cooling system is designed to remove heat generated in it. Also, a heavy water dumping system which can shutdown the reactor by discharging heavy water from the tank is considered.

The coolant flow in the core is downward for the purpose of reduction of the radiation dose. The total coolant flow rate in normal operation will be about 10,000 gpm and the coolant velocity in the core is about 5.7m/s.

3. Calculation Model

3.1 Nuclear Design

The major objective of the nuclear design for TRR-II lies in ascertaining the reactor safety and performance characteristics along with fuel burnup such as the flux distribution in the irradiation holes and the power distribution of the fuel. Importance is given to the reactivity worth of a variety of materials such as fuel assembly, control rod and shutoff rod. The nuclear design should confirm the reactivity effect induced by variation of reactor operating conditions such as the temperature and density of fuel, coolant, moderator and reflector.

The TRR-II nuclear design was conducted by using a combined system of CASMO-3 [1] and VENTURE[2]. CASMO-3 with its library was benchmarked by comparing some key parameters against the Monte Carlo Calculation and some other critical experiments. The two-dimensional characteristics solution in CASMO-3 has been performed extremely well against all other comparisons. The global core characteristcs were searched using the cross section set from CASMO lattice modeling and the core model for VENTURE. VENTURE is 3-dimensional multi-group neutron diffusion code based on the mesh-centered finite difference method. Using VENTURE, the reactor criticality, flux/power distribution reaction rate, and detector response, etc. can be obtained. MCNP4A [3] is a Monte Carlo code using continuous energy cross section data and has no limit to simulate any kinds of geometrically complicated configurations. Thus, the geometrical approximation and energy condensation are not required. This benefit enable MCNP4A to be widely used as a benchmark tool in case no experimental data are available. Since no experimental data similiar with TRR-II were available during its design period, MCNP4A is used for this purpose. Several kinds of artificial and real core configuration for comparison were assumed. For those reactor models, the assembly power distribution and the reactivity effect from CASMO-VENTURE are examined against the MCNP4A results. MCNP4A model for TRR-II describes the core geometry explicitly as far as possible. After validation, the fuel management was carried out up to the equilibrium core.

3.2 Thermal-Hydraulic Design and Accident Analysis

The major activities of the thermal-hydraulic(T/H) design were focus on giving the basic information for configuring the cooling system for reactor and reflector, on the basis of the selected basic concept. T/H analysis have been carried out to determine the important T/H parameters such as coolant flow rate, coolant temperature, system pressure, onset of nucleate, the boiling temperature, minimum departure from nucleate boiling ratio(MDNBR), and cladding temperature etc.

To investigate the T/H behaviors during steady state and transients including postulated accident conditions, calculation models with COBRA/RERTR [4], and RELAP5/MOD3 [5] are developed. Since TRR-II is going to be operated at low pressure and low temperature conditions, adequate heat transfer correlations are incorporated into the TRR-II T/H design codes. COBRA/RERTR, a modified version of COBRA/IIIC/MIT-1, is a subchannel analysis code to calculate fuel and coolant temperature, local flow rate and DNBR, while RELAP5/MOD3 is a system code used for the thermal hydraulic response calculations. The power peaking factors used are from CASMO-VENTURE with quantified uncertainties and benchmarked from the MCNP4A results.

4. Design Characteristics

The aim of the TRR-II design is to achieve as high thermal neutron flux as possible. For this purpose, the core is designed to be compact but the reflector is spacious to accommodate various experimental holes inside. The high thermal flux, which is most important to the reactor users, is avaliable in a wide region of reflector with minimum noise of fast neutron flux. Table 1 summarizes the nuclear and thermal-hydraulic design properties of the TRR-II core obtained from the observed calculation results.

Table 1 Nuclear and Thermal-Hydraulic Characteristics of TRR-II Core Design

Power (MWth)	10
Fuel Temperature Coeff.(mk/°C)	-0.038
Coolant Temperature Coeff.(mk/°C)	-0.064
Cold w/o Poison Excess Reactivity(mk)	119
Hot with Poison Excess Reactivity(mk)	82
6 Control Rod Capability(mk)	240
5 out 6 Capability(mk)	183
Max. thermal neutron flux(n/cm^2.sec)	1.0×10^{14}
Max. fast neutron flux(n/cm^2.sec)	1.4×10^{14}
Total Peaking Factors	3
Average heat flux(w/cm^2)	58.3
Mass flux(kg/m^2-s)	5425
MDNBR	>2.4
Core $\triangle$ T(°C)	5
Tin(°C)	40
Inlet pressure(psia)	23.0

5. Conclusion

The design of TRR-II is still under its feasibility study stage. The inverse neutron trap and the surrounding D_2O reflector were employed as core concept in order to obtain high pure thermal neutron flux and to secure enough irradiation space. The design flexibility of using the advanced silicide uranium dispersion fuel were to be studied, so as to improve the reactor performance in the future.

6. References

(1) M. A. Edenius, "CASMO-3 Benchmarking", Trans. Am. Nucl. Soc., 56 (1988)
(2) D. R. Vondy, "VENTURE-A code block for solving multi-group neutronics problems by applying the finite difference diffusion theory approximation to neutron transport", ORNL-5602 (1975)
(3) J. F. Briesmeister, "MCNP-A general Monte Carlo code for neutron and photon transport." LA-7396-M, Rev. 2 (1986)
(4) Jason Chao, "COBRAIIIC/RERTR A Thermal-Hydraulic Subchannel Code with Low Pressure Capability", CP-0326, Argonne National Laboratory, Dec. (1980).
(5) K. E. Carlson, etal, "RELAP5/MOD3 Code Manual," NUREG/CR-5535, (1990)

Construction of the HTTR and its Irradiation Program

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Abstract

The HTTR (High-Temperature engineering Test Reactor) is a high-temperature gas-cooled reactor with a maximum helium coolant temperature of 950°C at the reactor outlet.

The construction of the HTTR started in March 1991 aiming at the first criticality in 1997 after the commissioning test. The main components in the HTTR, such as a reactor pressure vessel, an intermediate heat exchanger, hot gas piping and graphite core structures, had been installed in the reactor building. At present the functional test of the HTTR has been carried out, and fuel elements are being fabricated at works with their loading targeted in 1997.

The HTTR project is intended to establish and upgrade the technology basis necessary for HTGR developments. At the same time materials and fuel irradiation tests are also planned for an innovative basic research after safety demonstration tests in the HTTR. Preliminary tests on selected research subjects which should be performed as irradiation tests, such as new development of semiconductors, superconductors and optical fibers, have been carried out at high temperatures with/without irradiation.

This paper describes present status of the construction, major features of its irradiation condition and prospects on irradiation test programs using the HTTR.

1. Introduction

It is essentially important in Japan, which has limited amount of natural resources, to make efforts to obtain more reliable and safety energy supply by extended use of nuclear powers including high temperature heat from nuclear reactors. Hence efforts are to be continuously devoted to establish and upgrade HTGR technologies and to make much of human resources accumulated so far. It is also expected that making basic researches at high temperature using HTGR will contribute to innovative basic research in future.

The HTTR has been, therefore, so designed as to be an engineering test reactor which aims to establish and upgrade the technological basis necessary for advanced HTGR development, and to conduct various irradiation tests for innovative basic researches(1).

The reactor building, which has two stories aboveground and three stories underground, is a size of about 48mX50m in the plane. A reactor pressure vessel, an intermediate heat exchanger and other heat exchangers in the cooling system are installed in the containment vessel, as illustrated in Fig.1.

The HTTR is a helium–gas–cooled and graphite–moderated reactor with 30MW thermal power and outlet coolant temperature of 950°C for high temperature test operation. The major specification of the HTTR is shown in Table 1. The reactor pressure vessel made of 2 1/4 Cr–1Mo steel, which contains core and core support structures, is 13.2m in height and 5.5m in diameter. The core structure is composed of fuel blocks, control rod guide blocks, and replaceable reflector blocks, and the core support structure is composed of large graphite components, carbon components and metallic components as illustrated in Fig. 2. The reactor cooling system is composed of a main cooling system (MCS), an auxiliary cooling system (ACS) and two reactor vessel cooling system (VCS) as shown in Fig. 3. The ACS is in the stand–by condition during the normal reactor operation and is operated to remove the residual heat from the core when the reactor is scrammed. The VCSs are operated during the normal operation in order to cool biological shielding concretes. It is also used as a residual heat removal system when the forced circulation in the primary cooling circuit is no longer available.

The construction of the HTTR is proceeding in the Oarai Research Establishment, JAERI. The first criticality will be attained in the end of 1997, and the reactor performance test will follow to achieve its full power operation in FY1998. Various kinds of operation and irradiation tests are also planed, such as experimental researches for the development of advanced fuels and materials and demonstration of the nuclear process heat utilization. The operation and test plan in the HTTR is shown in Table 2.

2. Construction of the HTTR

The installation permit of the HTTR was issued from the Government in November 1990. The design and construction methods of the HTTR was approved from the Science and Technology Agency in January 1991 and the construction work of the HTTR was started in March 1991. The excavation of ground was completed in August 1991 and the construction of concrete base-matt was also completed in May 1992.

The reactor containment vessel was installed and the pressure-proof and leak tests were successfully conducted in November 1992. The reactor pressure vessel, the intermediate heat exchanger, primary helium circulators and the primary pressurized water cooler were installed in the reactor containment vessel in 1994. The first pressure-proof test for the primary cooling system was carried out successfully in October 1995. The reactor building was completed in December 1995 by closing the temporary opening for the large components.

The JAERI obtained the uranium material for the first loading fuel in September 1994 and the fuel fabrication started in Jun 1995. The first criticality will be attained in the end of 1997. The HTTR construction schedule is shown in Table 3.

3. Irradiation Test Program of the HTTR

3.1 Irradiation Characteristics of the HTTR

The HTTR has the capability not only to conduct the R&D on the advanced technologies for the HTGRs but also to irradiate large sized samples at a high temperature with a uniform neutron flux.

The core of the HTTR consists of prismatic fuel elements of hexagonal blocks, with 580mm in height and 360mm in across flats, and therefore the maximum size of an irradiation sample is 500mm in height and 300mm in diameter.

The samples can be irradiated at high temperatures ranging from 400°C to 1100°C at maximum. Various irradiation tests are possible to use five irradiation regions as shown in Fig. 4. The irradiation conditions for these regions are summarized in Table 4. Irradiation tests at a high temperature (400 to 1100°C) and high neutron flux (about $2x10^{17}n/m^2 \cdot s$ (E>0.18 MeV)) can be performed in the center column region. Central four columns, the center column region and the test fuel loading region, are permitted to load fuel elements. Irradiation tests at a lower temperature (400 to 600°) and lower neutron flux (about $8x10^{14}$ $n/m^2 \cdot s$ (E>0.18MeV)) can be performed in the permanent reflector region.

3.2 Preliminary Design of Irradiation Facility

A Preliminary design of a new irradiation capsule (Rotation drive type capsule,RDC) has been carried out for use in the material irradiation test at high temperature with neutron fluence controlled uniformly. The RDC capsule is to

- Development of in–core neutron and gamma ray detectors in service at high temperature up to 950°C.
- Development of heat–and radiation–resistant optical fiber for an instrumentation of high temperature reactors.

(5) Other researches

- Study on radiation shielding
- Development of fuels for an advanced reactor

Some research themes are of great significance in view of irradiation test candidates for the HTTR but need to be subjected to preliminary tests (out–of pile or in–pile) to assess the scientific effectiveness.

Preliminary out–of pile experiments are being undertaken currently for the following themes.

- Neutron transmutation doping of low atomic number high temperature semiconductor
- Improvement of high temperature oxide superconductors by irradiation
- Pecuriality chemical reaction at high temperature irradiation condition
- Development of heat– and radiation–resistant optical fiber
- Development of in–core neutron and gamma ray detectors for use at very high temperature

Here, typical preliminary test result is shown in Fig.6[2]. This figure shows a test result for optical fiber. Two kinds of SiO_2–based optical fibers, the high purified and OH doped fibers, were tested at 800°C in an air condition; measuring wavelength was 850nm. Both fibers show stable optical intensity during about 300 hours heat treatment, furthermore the high purified optical fiber shows a lower optical intensity. From this preliminary test high purified optical fiber was selected as a good heat–resistant one. At present an atmosphere effect, e.g. contained H_2, O_2, etc. elements effect, on the optical intensity is investigating at high temperature using this optical fiber.

4. Concluding Remarks

The HTTR is a high temperature gas cooled reactor which has various aims and operational modes. The construction of the HTTR has progressed on schedule and its first criticality is foreseen at the end of 1997.

The various tests planned in the HTTR will make a great contribution to confirm salient characteristics of HTGRs and reliable supply of heat as high as 950°C. Moreover the HTTR has a unique and superior capability for carrying out high temperature irradiation tests for innovative basic researches.

be installed in the replaceable reflector region A in Fig.4. This capsule is schematically shown in Fig. 5. The RDC is composed of capsule units, drive gear, rotation shaft and so on. Irradiation test specimens are inserted into the capsule unit made of graphite. The maximum specimen size is about 100 mm in height and 80 mm in diameter. The capsule unit is moved to certain longitudinal position in the core by the drive gear, and then rotated at the same position by the rotation shaft until a specified neutron fluence. Capsule units are turned at both ends of the RDC by the drive gear. Two types of capsule units are considered in the preliminary design; the one is an irradiation capsule unit in which the irradiation specimen is contained, the other is a dummy capsule unit made of graphite for the purpose of a neutron shielding. Both temperature and neutron fluence are measured nearby irradiation position in the capsule.

3.3 Preliminary Study on Innovative Basic Research

Sixty eight themes have been proposed as the innovative basic research at a high temperature under neutron irradiation conditions until March 1994; 34 themes for new material development, 5 themes for radiation chemistry research, 8 themes for nuclear fusion research, 8 themes for high temperature irradiation technique development and 13 themes for other researches. Typical research themes proposed are as follows:

(1) New material development

The proposals involve of development of the new material by a high temperature irradiation and explication of the material behavior under a high temperature irradiation condition. As for the new material development, the following research themes were proposed.

- Development of high temperature semiconductors
- Improvement of high temperature superconductors

(2) Radiation chemistry research

- Research for the decomposition reaction of a high polymer
- Research for the peculiar reaction of radiation chemistry
- Research for radiation assisted synthesis of fullerenes containing actinoids

(3) Nuclear fusion research

The proposals in nuclear fusion research consist of irradiation tests of several types of blanket components for high temperature application and development of the first wall materials for a prototype/commercial fusion reactor.

(4) High temperature irradiation technique development

REFERENCE

(1) S.Saito et al.,Design of the High Temperature Engineering Test Reactor(HTTR),JAEI 1332(1994).
(2) T.Shikama,private communication.

Table 1 Major specification of the HTTR.

Thermal power	30MW
Outlet coolant temperature	850/950℃
Inlet coolant temperature	395℃
Fuel	Low-enriched UO_2
Fuel element type	Prismatic block
Direction of coolant-flow	Downward, flow
Pressure vessel	Steel
Number of main cooling loop	1
Heat removal	Intermediate heat exchanger, Pressurized water cooler
Primary coolant pressure	4MPa
Containment type	Steel containment
Plant lifetime	20years

Table 2 HTTR operation and test plan.

Item \ Fiscal year	1997	1998	1999	2000	2001	2002	2003
1. Establishment of HTGR technologies	criticality	Initial core					Advanced core
		Reactor performance tests	30MW/850℃~950℃				30MW/950℃
2. Upgrading of HTGR technologies			Irradiation tests				
			Safety demonstration tests	Construction and demonstration tests of nuclear heat application system			
3. Innovative and basic technology				• Development of very high temperature heat resistant materials • Tritium production and recovery tests, etc.			

Table 3 HTTR construction schedule.

Fiscal Year	1990	1991	1992	1993	1994	1995	1996	1997	1998
MILESTONE	▽ Construction start							Fueling ▽ ▽ Criticality	
Approval of design and construction methods	1st	2nd, 3rd	4th, 5th						
Site renovation									
Excavation for reactor building									
Reactor building									
Reactor containment vessel									
Cooling system									
Reactor pressure vessel and core internal									
Fuel									
Criticality and power up test									

Table 4 Irradiation condition in the HTTR.

Irradiation position / Items	Fuel regions		③Replaceable reflector region A	④Replaceable reflector region B	⑤Permanent reflector region
	①Center column region	②Test fuel loading region			
Configuration	Hexagonal block	Hexagonal block	Cylindrical	Cylindrical	Cylindrical
Maximum size	Accross flat 360 mm Height 580mm	Accross flat 360mm Height 580mm	Diameter 300mm Height 500mm	Diameter 130mm Height 500mm	Diameter 100mm Height 2900 mm
Number of samples (max)	1 column	3 columns	3 columns	12 columns	4 holes
Fast neutron flux (>0.18Mev) $(n/m^2 \cdot s)$	$\sim 2\times10^{17}$	$\sim 2\times10^{17}$	$\sim 2\times10^{16}$	$\sim 2\times10^{16}$	$\sim 8\times10^{14}$
Themal neutron flux (<2.38eV) $(n/m^2 \cdot s)$	$\sim 7\times10^{17}$	$\sim 5\times10^{17}$	$\sim 4\times10^{17}$	$\sim 4\times10^{17}$	$\sim 3\times10^{17}$
Intensity of gamma ray (Gy/s)	$\sim 1\times10^{2}$	$\sim 1\times10^{2}$	$\sim 3\times10^{1}$	$\sim 3\times10^{1}$	~3
Temperature (℃)	400 to 1100	400 to 1100	400 to 800	400 to 800	400 to 600
In-core instrumentation	Temperature instrument, Neutron flux instrument, Neutron fluence instrument, Fuel failure detection system				

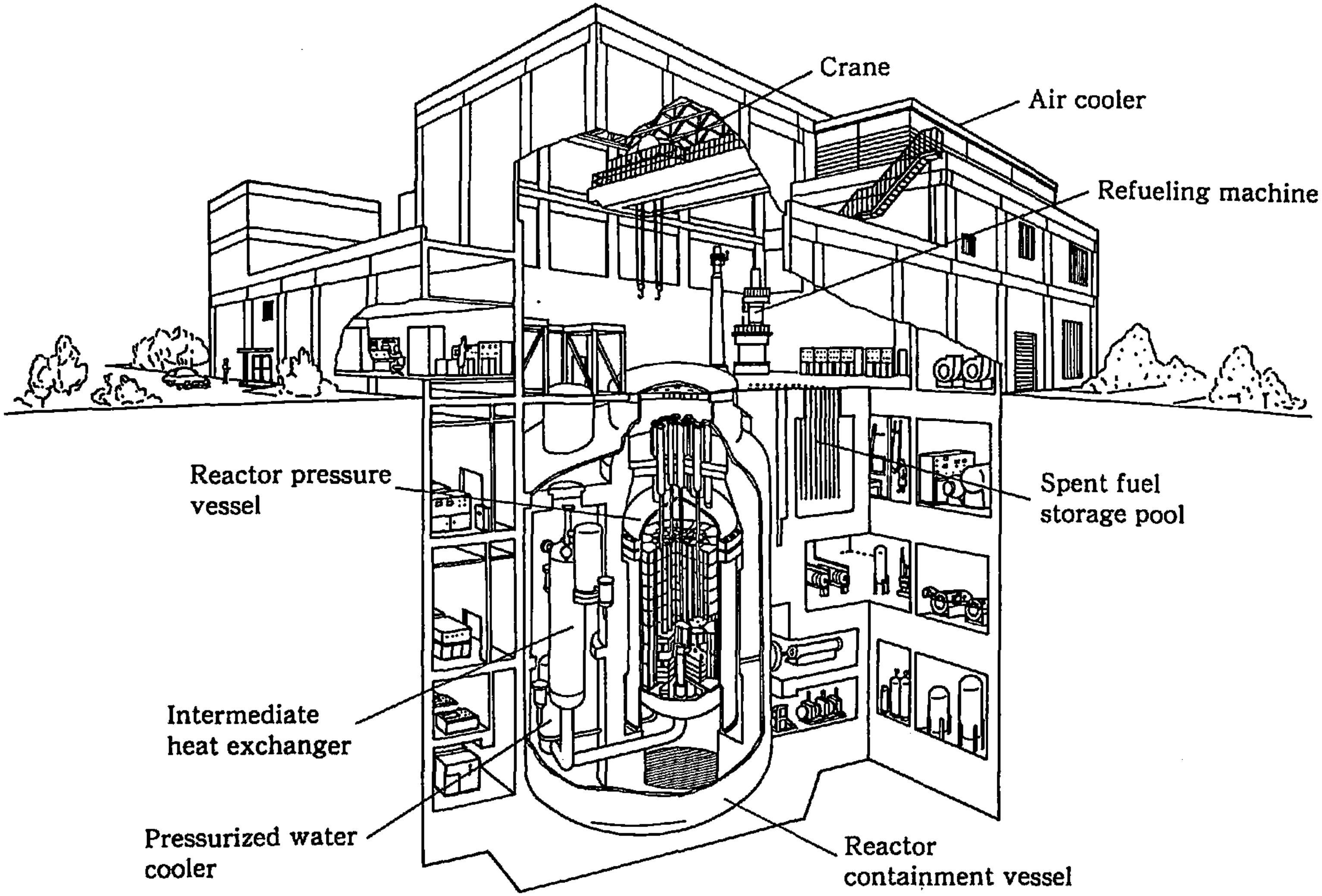

Fig.1 Bird's-eye view of the HTTR reactor building.

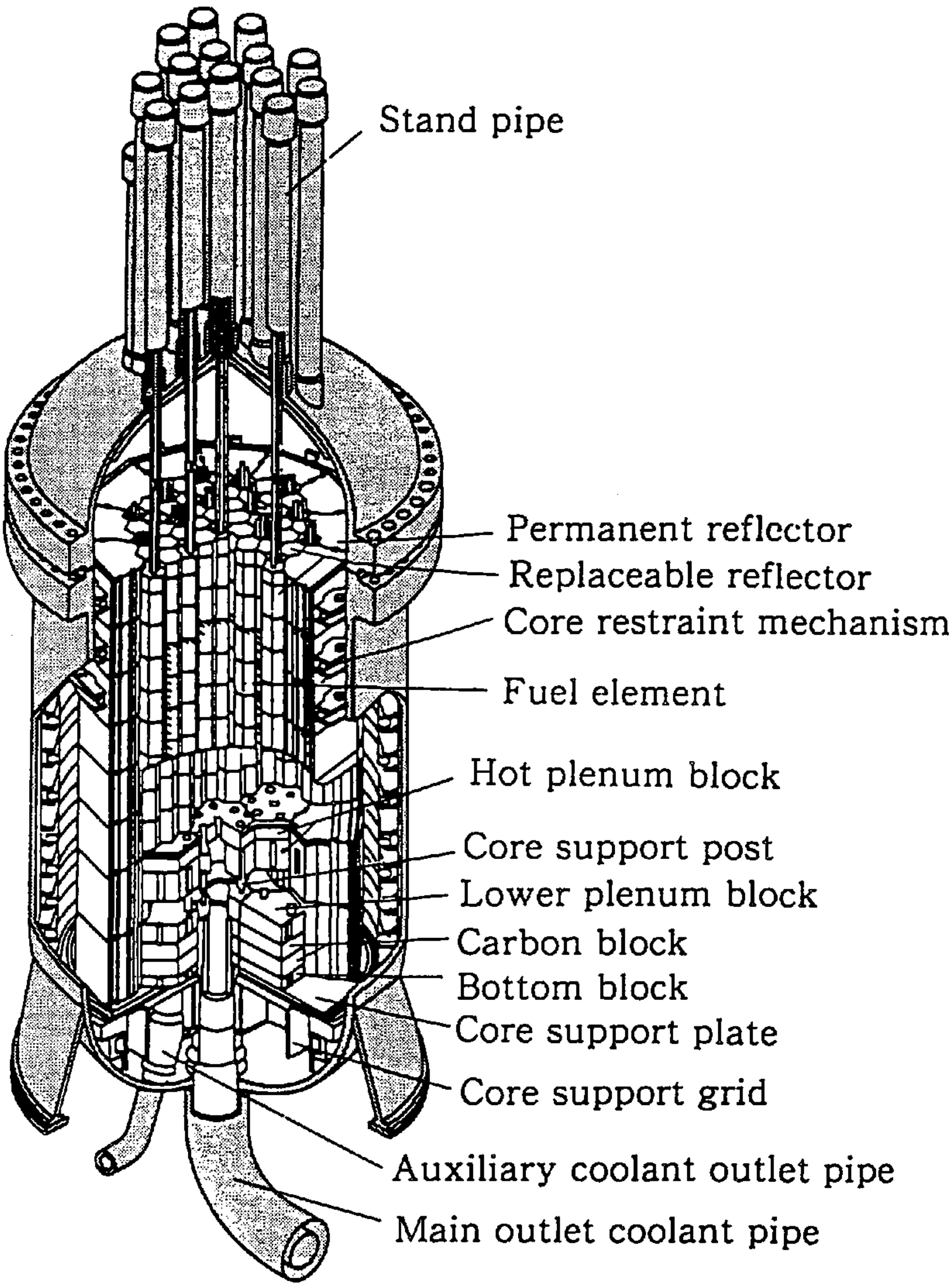

Fig.2 Bird's–eye view of the pressure vessel and the core of the HTTR.

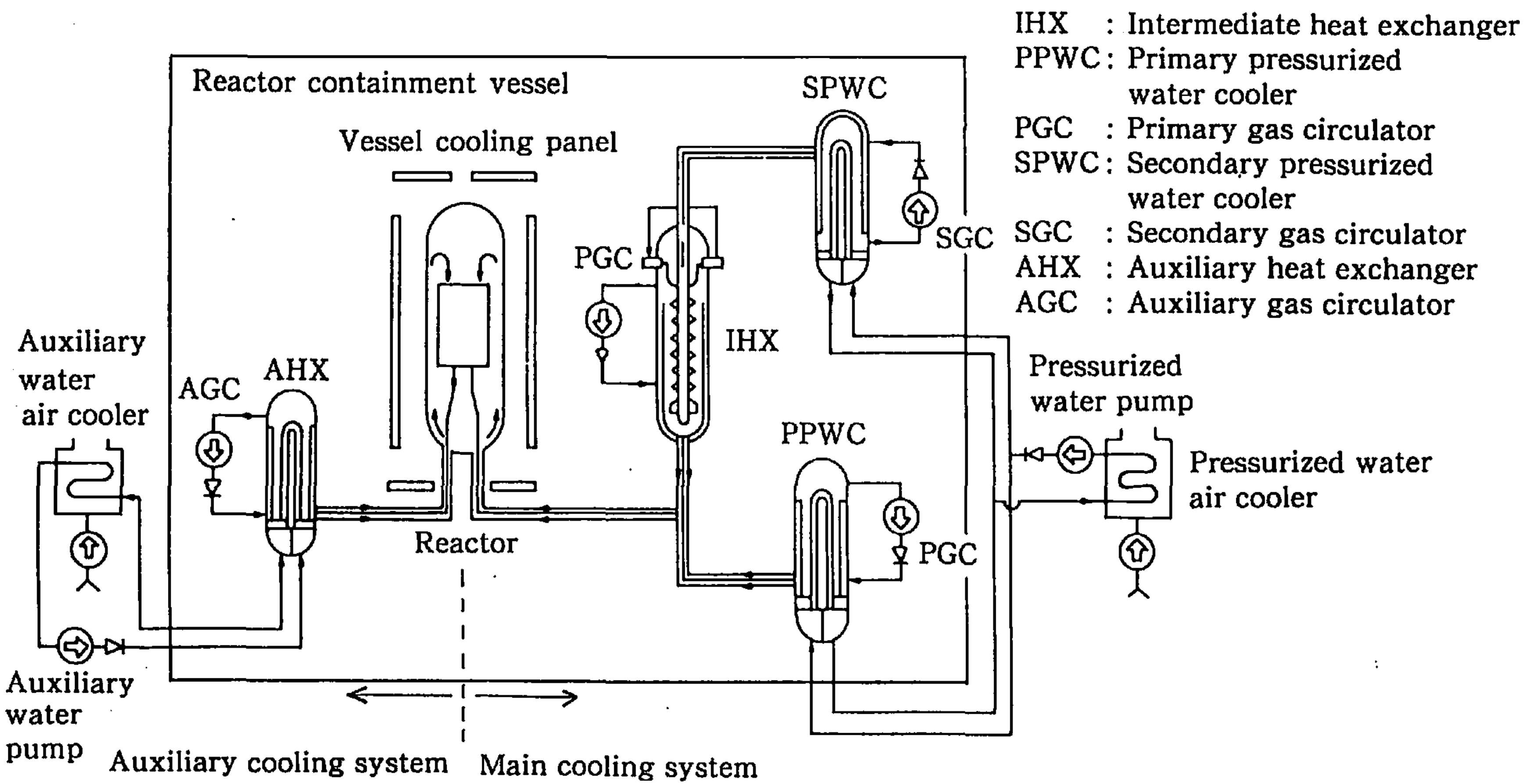

Fig.3 Cooling system in the HTTR.

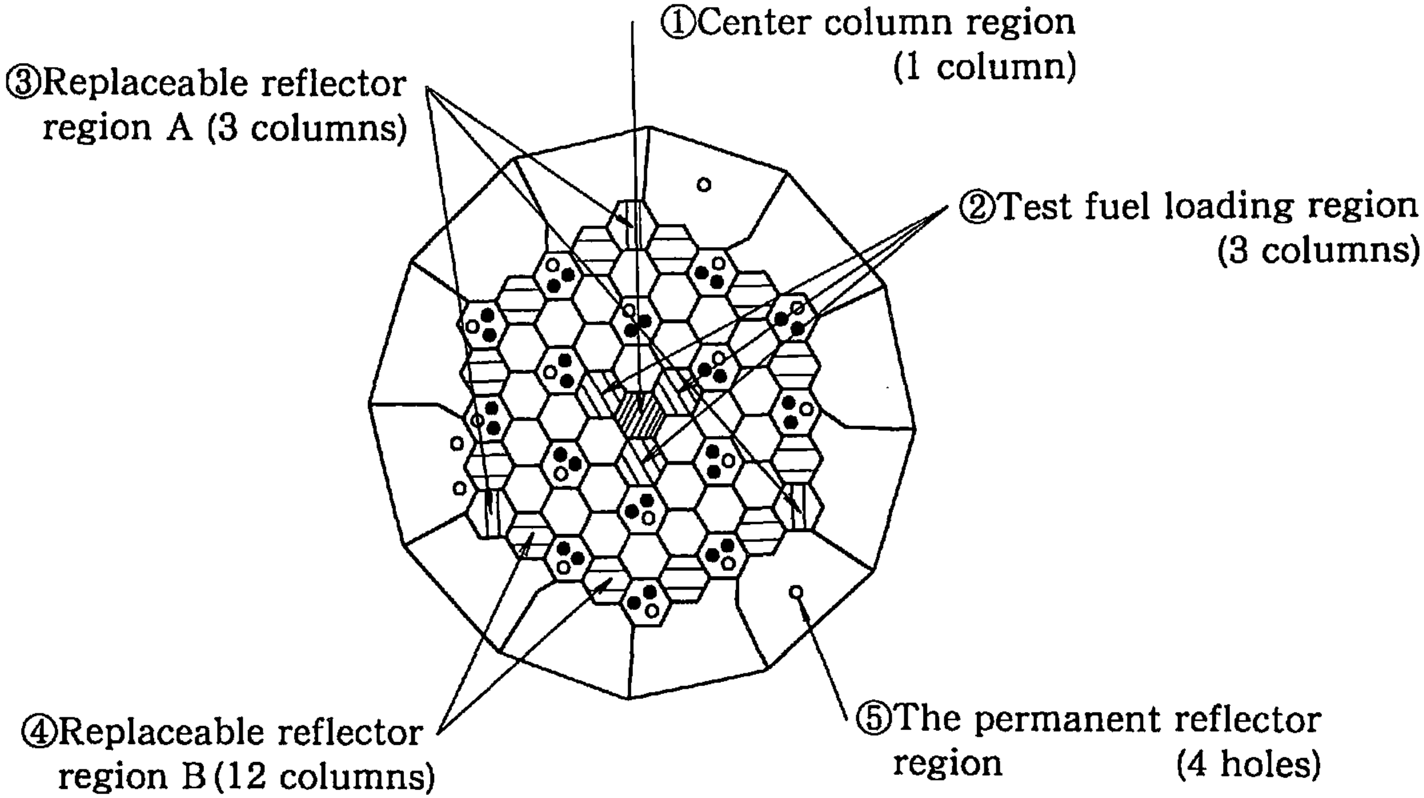

Fig.4 Irradiation location in the HTTR.

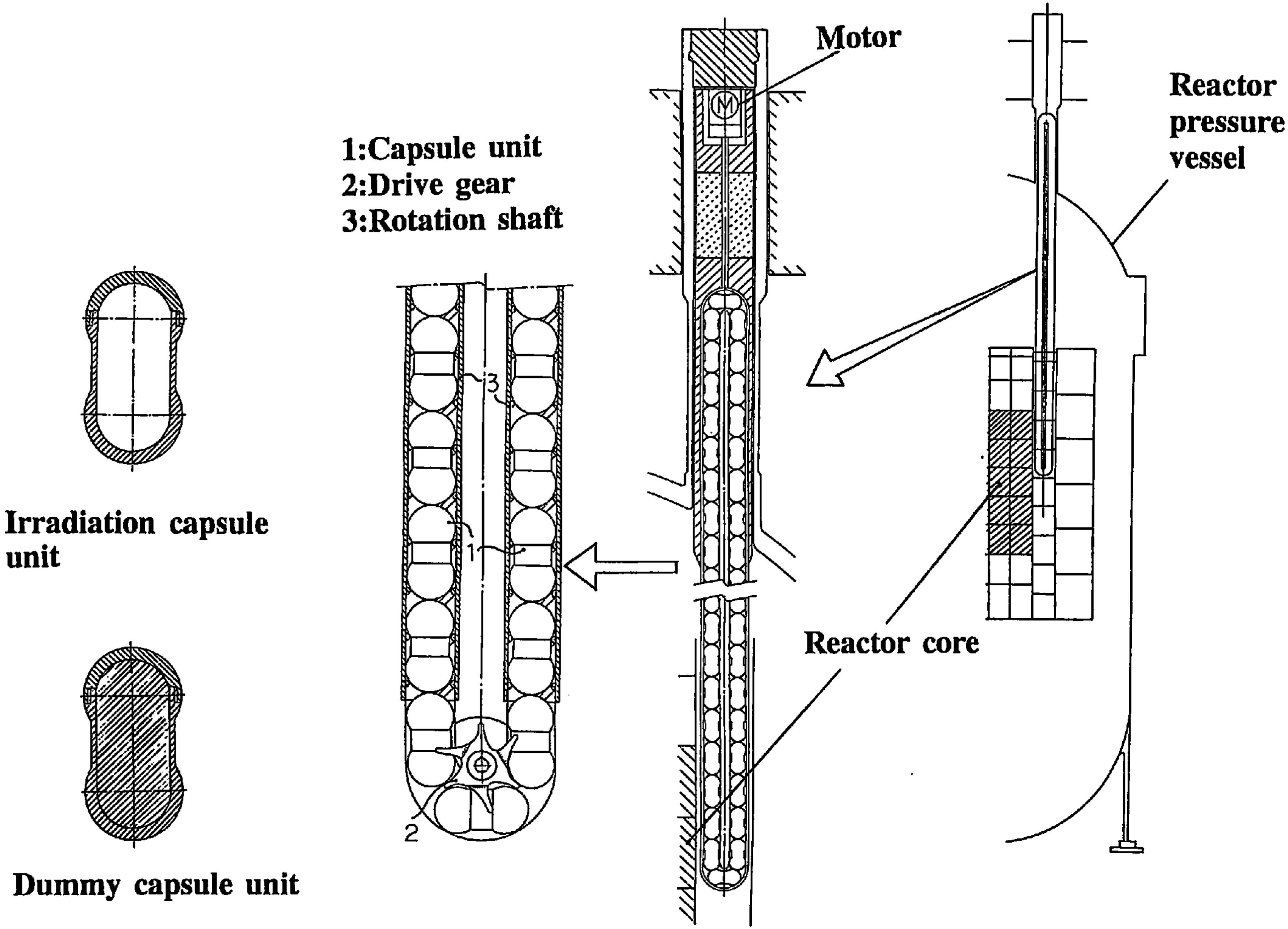

Fig.5 Rotation drive type capsule in the HTTR.

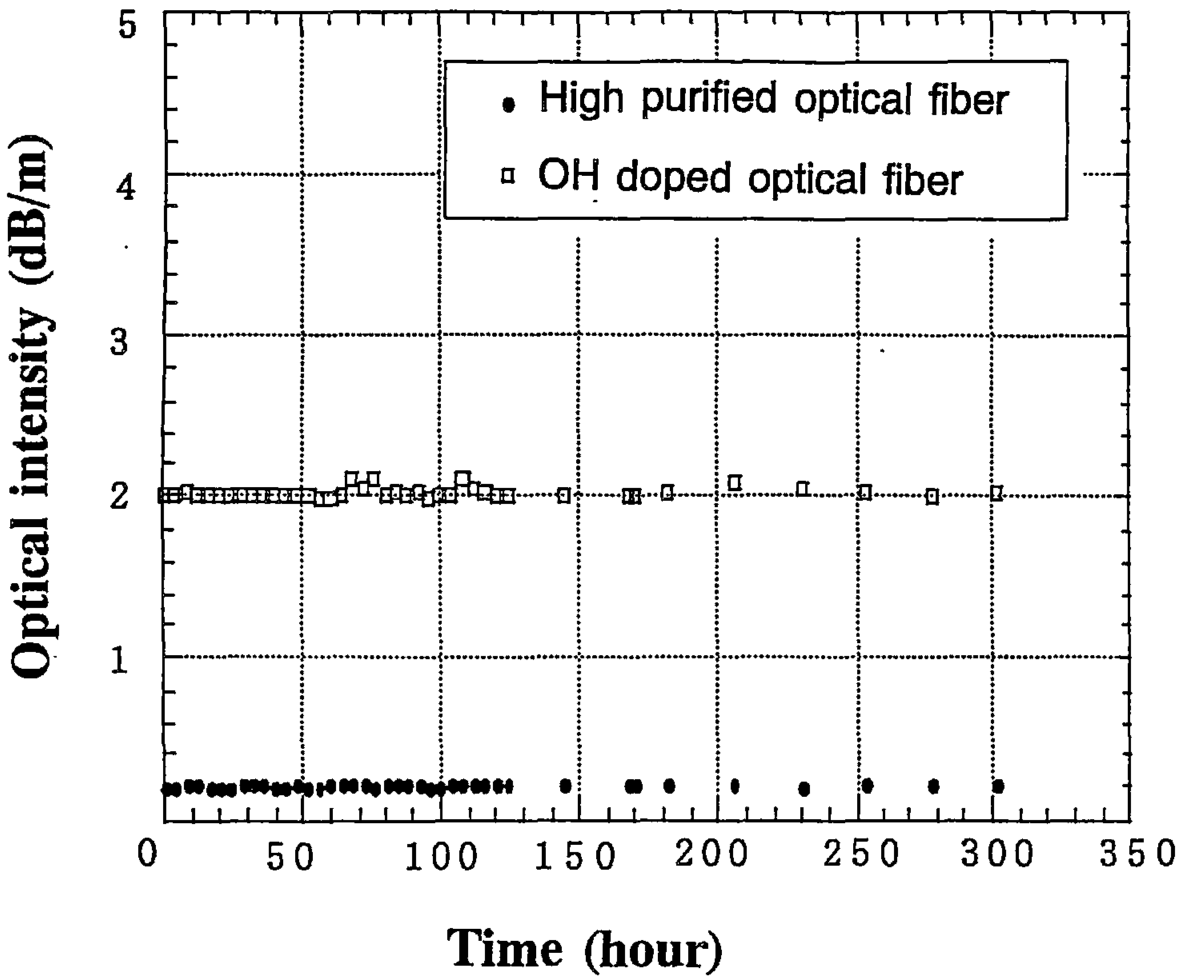

Fig.6 Preliminary test result for the optical fiber.
(This figure shows a optical intensity change at 800°C heat treatment condition; measuring wavelength 850 nm)

November 8, 1996
KWU NLFM / Didier

5th meeting of the International Group on Research Reactors (IGORR V)

November 4-5-6, 1996, Aix-En-Provence - France

Agenda

1. **Introduction**

2. **Status of the Project**

   Barrier removal

   Site

   Installations

   Project management, tools

   Action against the first partial license

3. **Outlook**

4. **Technical Highlights**

5th meeting of the International Group on Research Reactors (IGORR V)

November 4-5-6, 1996, Aix-En-Provence - France

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1. Introduction

Since our last IGORR-IV meeting in Gathingburg (May 1995) the „high flux neutron source" FRM-II has developed rapidly.

As we have reported at our last meeting this project was still in the design and licensing phase. It had to overcome some important barriers before start of civil construction.

Those barriers have been removed !

On August 1st, 1996, the first sod procedure has taken place under the attendance of representatives from

- the German Federal Republic (BMBF)
- the local municipalities
- Science and Industry — **Fig. 1**
- Licensy authorities and experts
- and finally the Bavarian State Government headed by the Prime Minister Dr. Edmund Stoiber — **Fig. 2**

2. Status of the Project

- Barrier removal

  - Project funding is settled, 720 Mio. DM for the whole project incl. 15 % tax
  - First partial license has been issued April 9th, 1996
  - Environmental impact examination has been concluded successfully
  - Use of fuel with high enriched uranium (HEU) was accepted by the licensing authority
  - General suppliers contract, TUM-Siemens, has become effective April 9th, 1996

- Site

  Fig. 3 | Fig. 4 |

  - Site preparation is concluded, including protection measures to control the permission of staff to enter on the site
  - Foundation of reactor building in the excavation (8 m deep and almost 2000 m^2 wide)

- Installations

  - The metallic cladding (stainless-steel) of the reactor pool is ordered, the supplier is preparing to manifacture this liner in such a way that it can be used as „lost form" in spring 1997
  - Two dummy fuel elements have been manufactured (one aluminium, one depleted uranium) which are being zested in a hydraulic loop at the University Bochum

- Project management, tools

Fig. 5 | • TUM projectgroup is revalorized to a central organisation reporting directly to the president

- The Siemens FRM-II project-team has been settled and set up for operation
- Electronic management tools have been implemented for:
  - time scheduling and follow up
  - cost planning and follow up
  - data management system
  - 3 dimensional computer aided design system linked with a logistic system for procurement and installation
  - reporting
- Implementation of a professional public relation group exclusively operating for this project.

- Action against the first partial license

- 1. action for the restriction of suspensive effect
- 2. action against substantial matters of licenses
- Main complaints:
  - liability of fuel element
  - radiation exposure in case of a beyond design accident to high
  - provisions to avoid a spontaneous rupture of primary piping
- 8 actions by communities and individuals
- Withdrawal of action by 4 parties because of low hope to succeed

3. Outlook

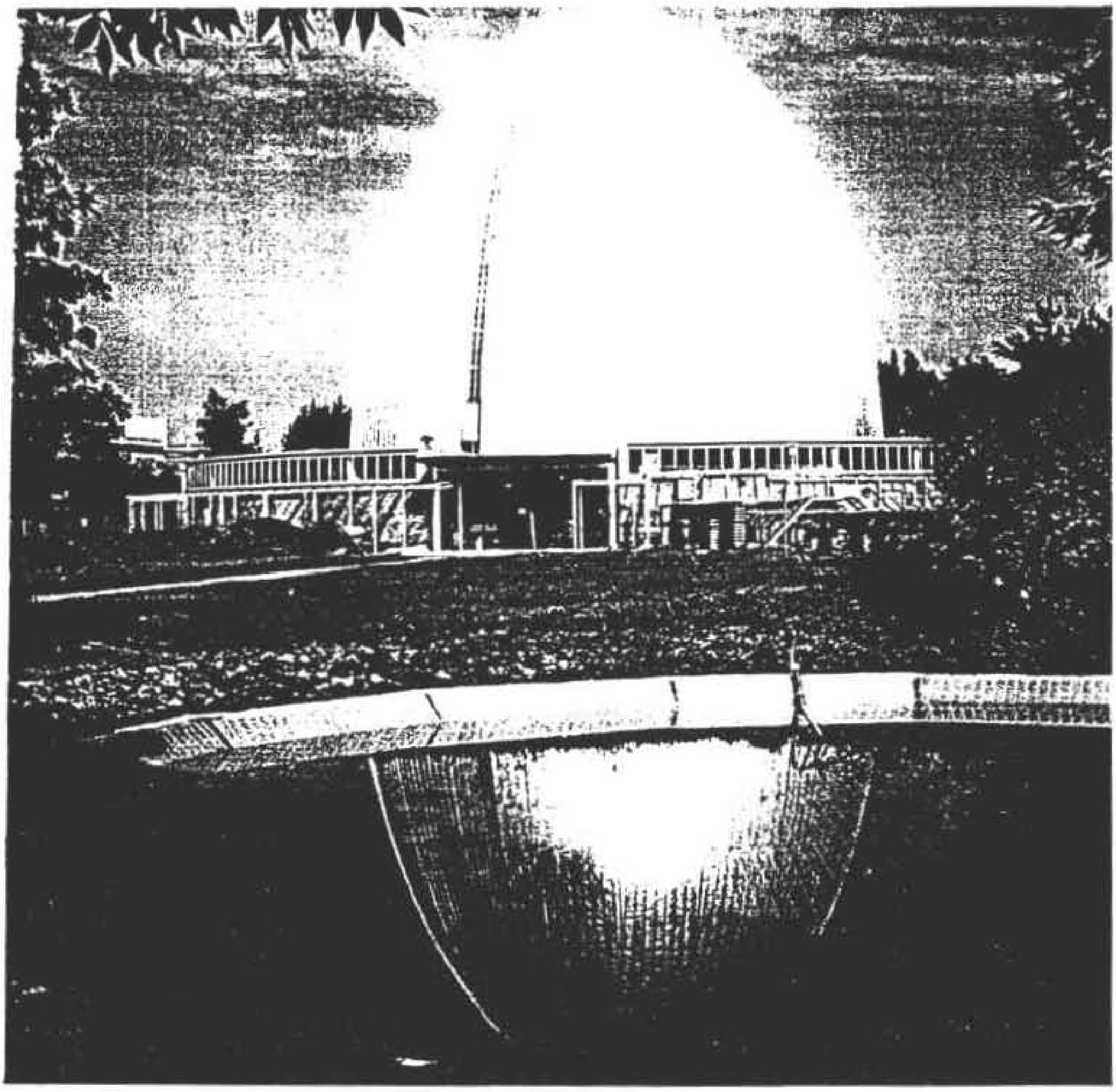
Fig. 6 | Timetable

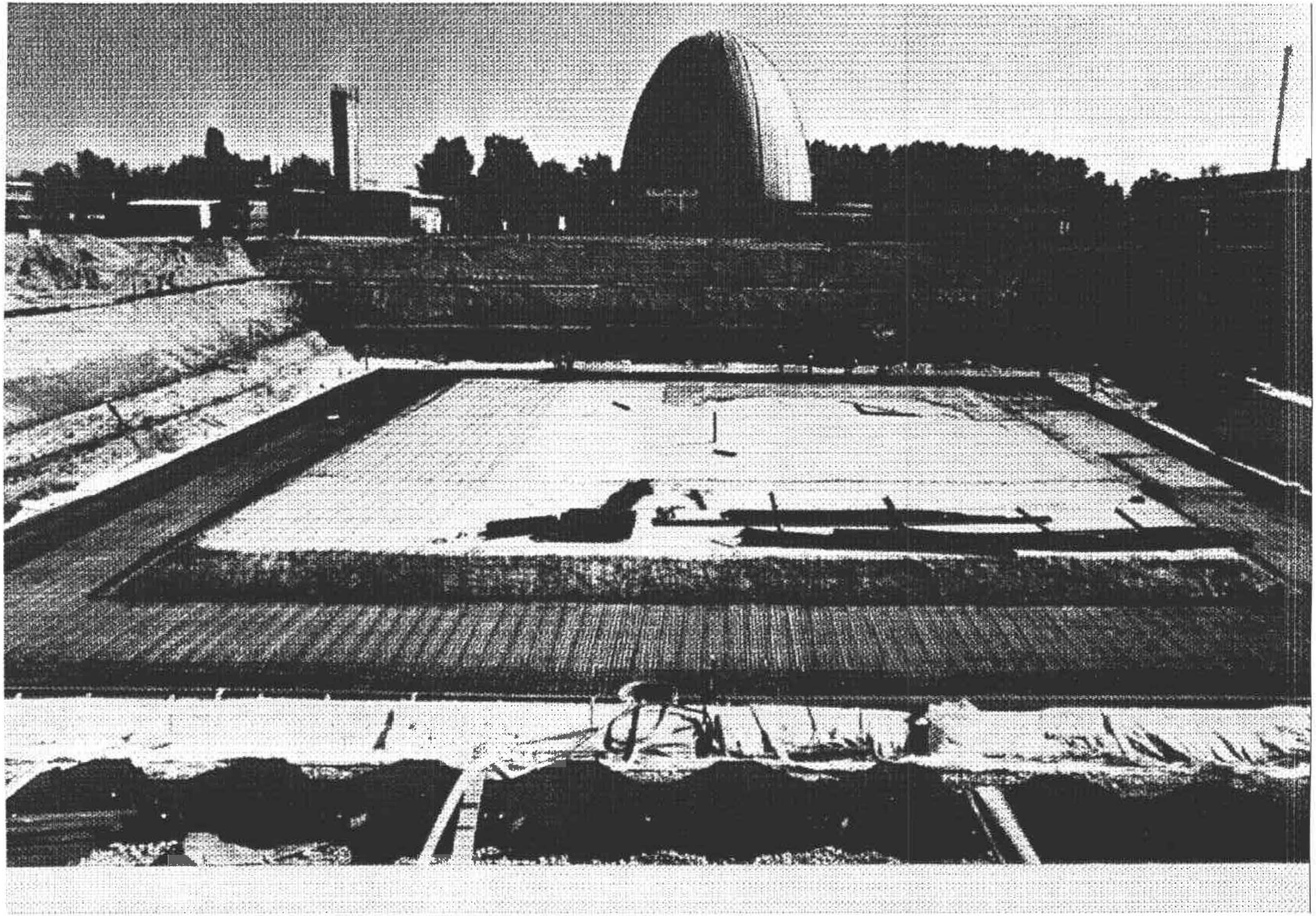
4. Technical Highlights

Fig. 7, 8 |
- Moderator tank
- Overall building sections

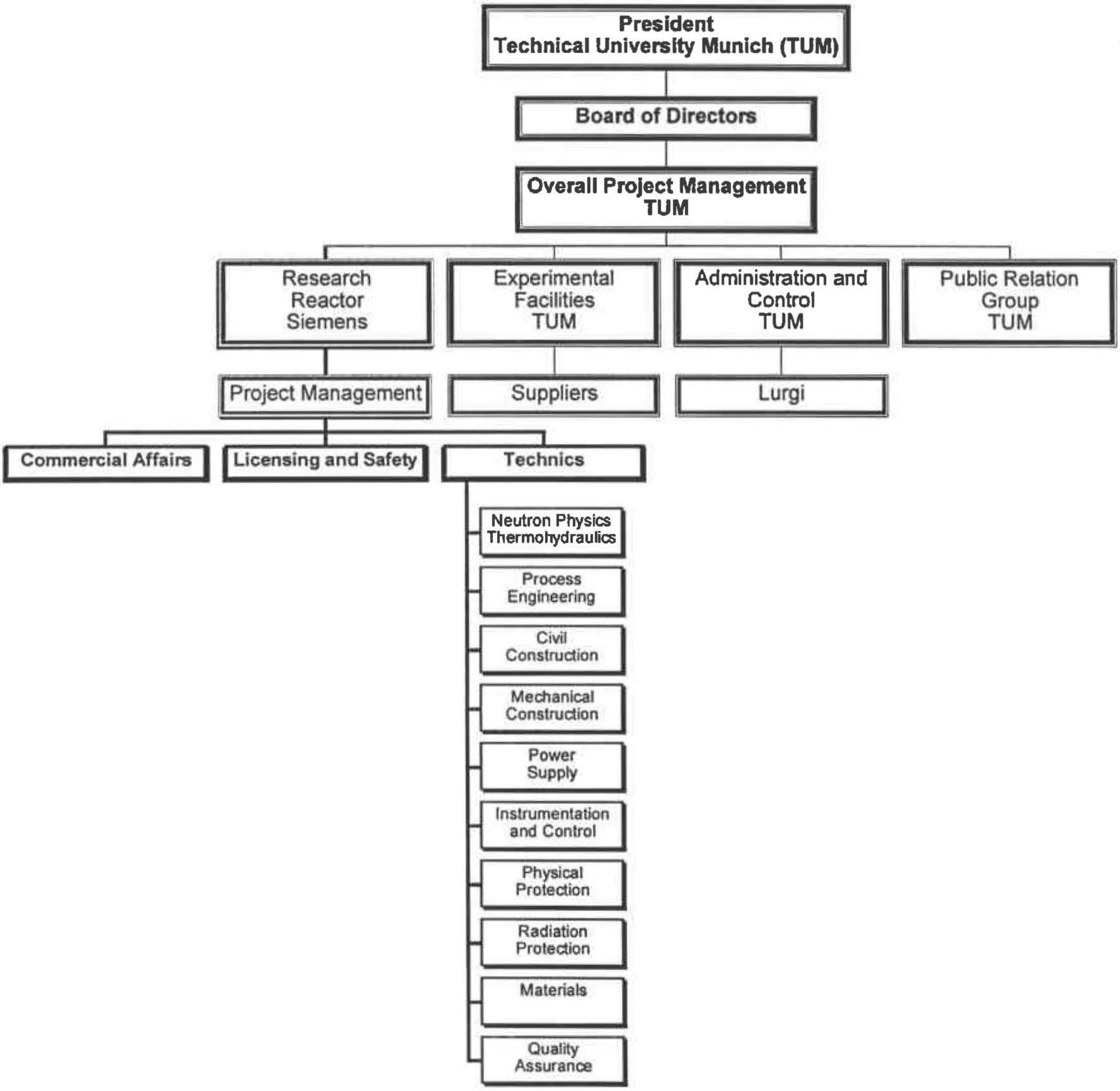

FRM ~ II
Garching


FRM II - 1. SPATENSTICH

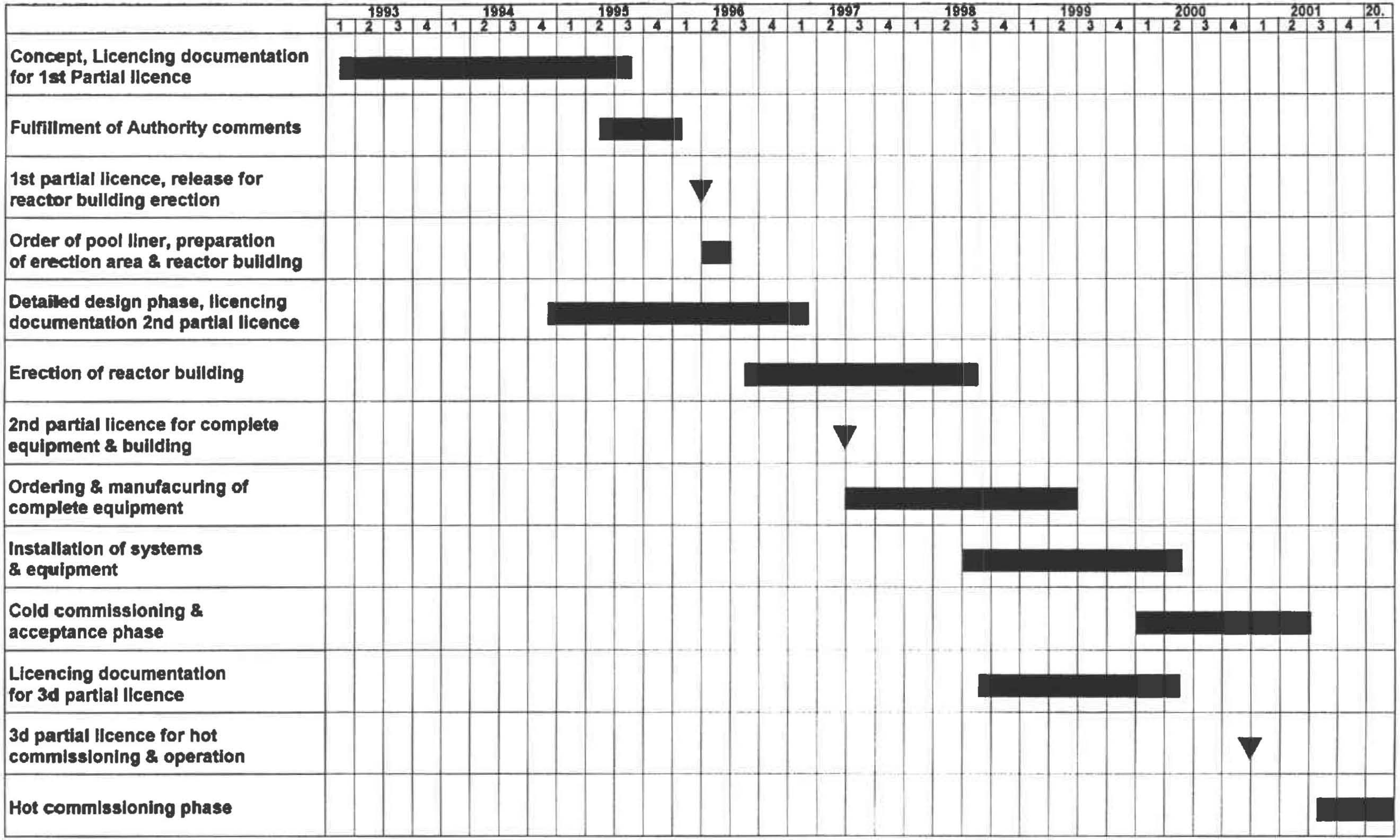




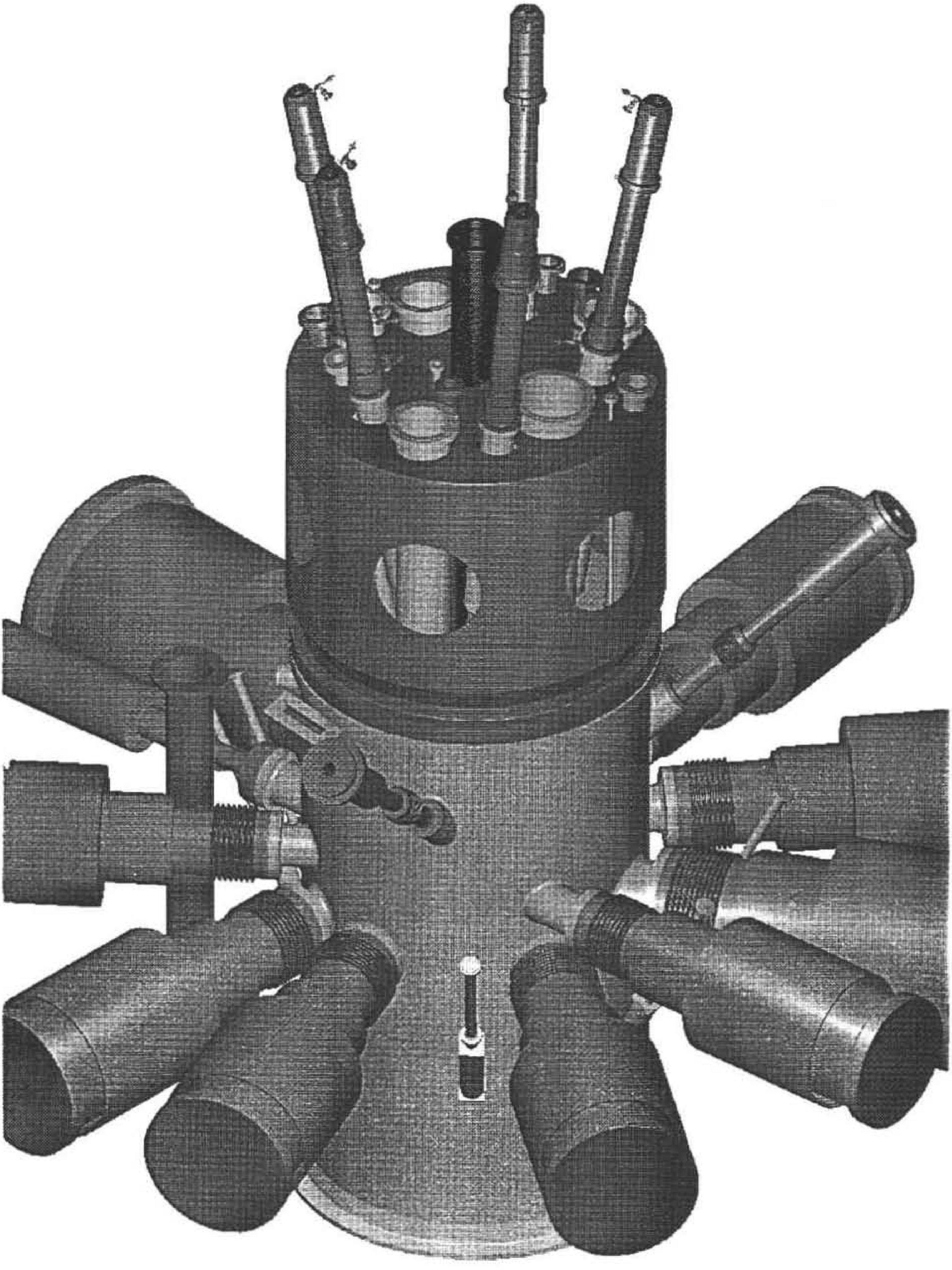
Organisational Chart of FRM-II-Project

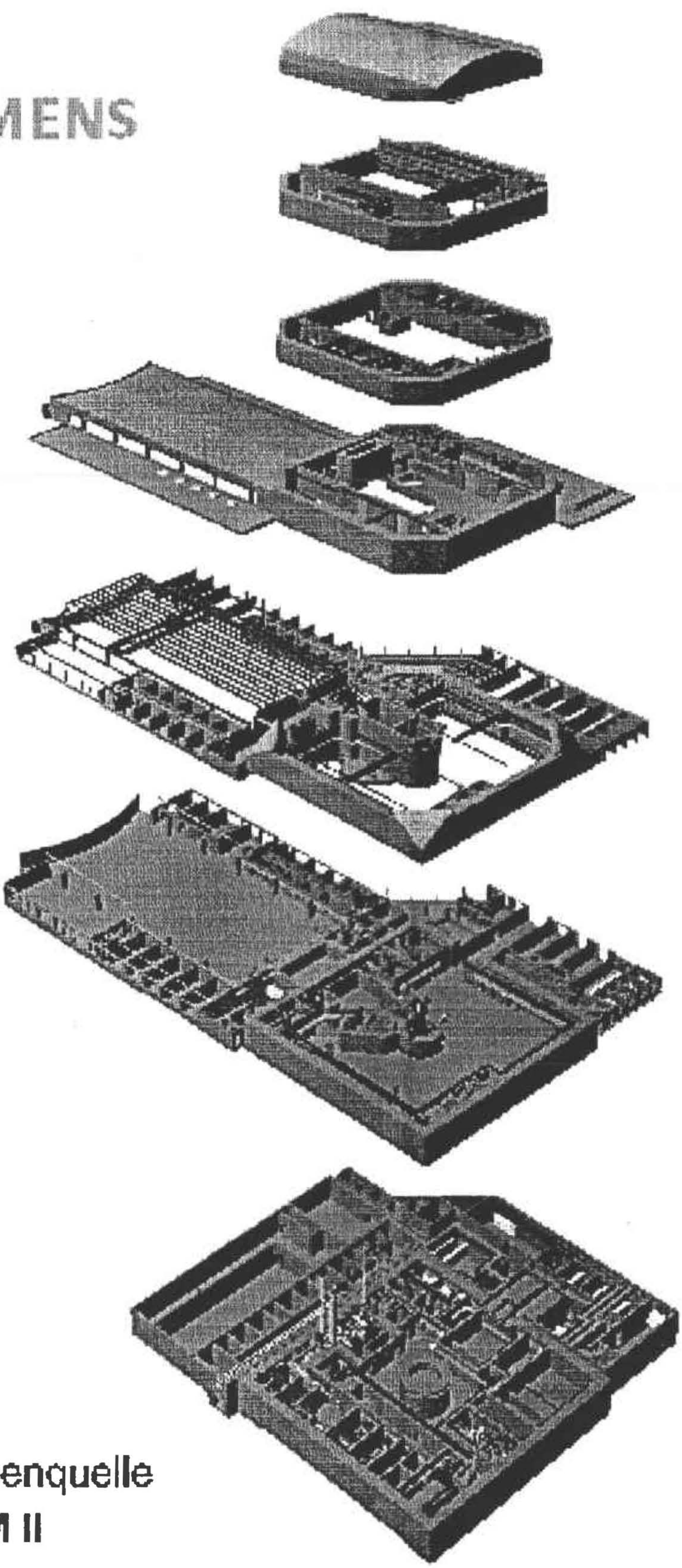


SIEMENS FRM - II Neutron Source Munich - Overall time Schedule



KWU-NLFM





Neutronenquelle
FRM II

THE

JULES HOROWITZ REACTOR

(R.J.H.)

THE C.E.A. FUTURE TOOL FOR TECHNOLOGICAL IRRADIATIONS

The R.J.H. reactor

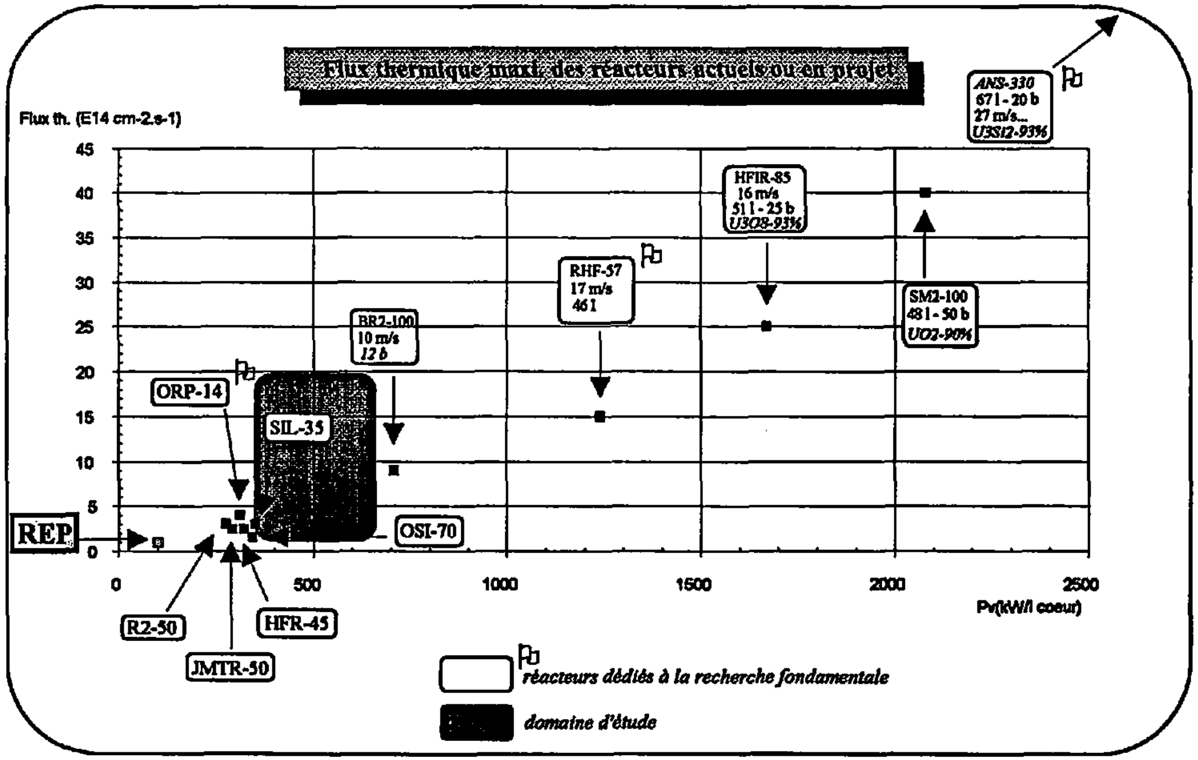
- GENERAL BACKGROUND
- EXISTING EXPERIMENTAL REACTORS DEDICATED TO TECHNOLOGICAL IRRADIATIONS
  - SILOE
  - OSIRIS
  - European Reactors

  All these reactors were designed and constructed in the 1960's

- NEED FOR A NEW TOOL DEDICATED TO TECHNOLOGICAL IRRADIATIONS IN THE NEXT CENTURY

The R.J.H. reactor

■ GENERAL BACKGROUND



The R.J.H. reactor

- GENERAL OBJECTIVES

- FUEL QUALIFICATION
  - > THERMAL NEUTRON REACTORS
  - > FAST NEUTRON REACTORS
  - > POWER TRANSIENTS
  - > OTHERS

- NUCLEAR MATERIAL QUALIFICATION

- Radio-isotope production

The R.J.H. reactor

❑ TECHNICAL REQUIREMENTS

- HIGH FLUX
  - > 5.E14 $cm^{-2}.s^{-1}$ in the thermal range (<0.625 eV)
  - > 1.E15 $cm^{-2}.s^{-1}$ in the fast range (>1MeV)
- WIDE USE SPECTRUM (various, evolutive and changing needs)
- ADEQUATE EXPERIMENTAL VOLUME
- EXPLOITATION FLEXIBILITY (accessibility,adequate cycle length ...)

The R.J.H. reactor

□ CONSTRAINTS

- FUEL (LEU)

- SITE : CADARACHE
  Wet Temperature : high in summer
  Seismic level

The R.J.H. reactor

- SAFETY OBJECTIVES

- Risks for environment and peoples will be as low as possible during normal operating conditions and accidents

- Radiation doses will be as low as possible for reactor workers

    > CIPR 60
    > ALARA principle

  –

The R.J.H. reactor

- EXPERIMENTAL DEVICES

- E1 = "capsules" (D=30 to 80 mm) in core
- E2 = "small loop" (D=80 mm) in reflector
- E3 = "loop in displacement box" (D=80 mm) in reflector
- E4 = "large loop" (D=150 mm) in reflector
- E5 = "central loop" (D=80 to 200 mm) in core center

The R.J.H. reactor

- OPERATING CONDITIONS

- "S" MODE = Without penetrating central loop
  (Nominal Experimental Charge)

- "B" MODE = With penetrating central loop
  (FBR irradiations - PWR irradiations and power transients)

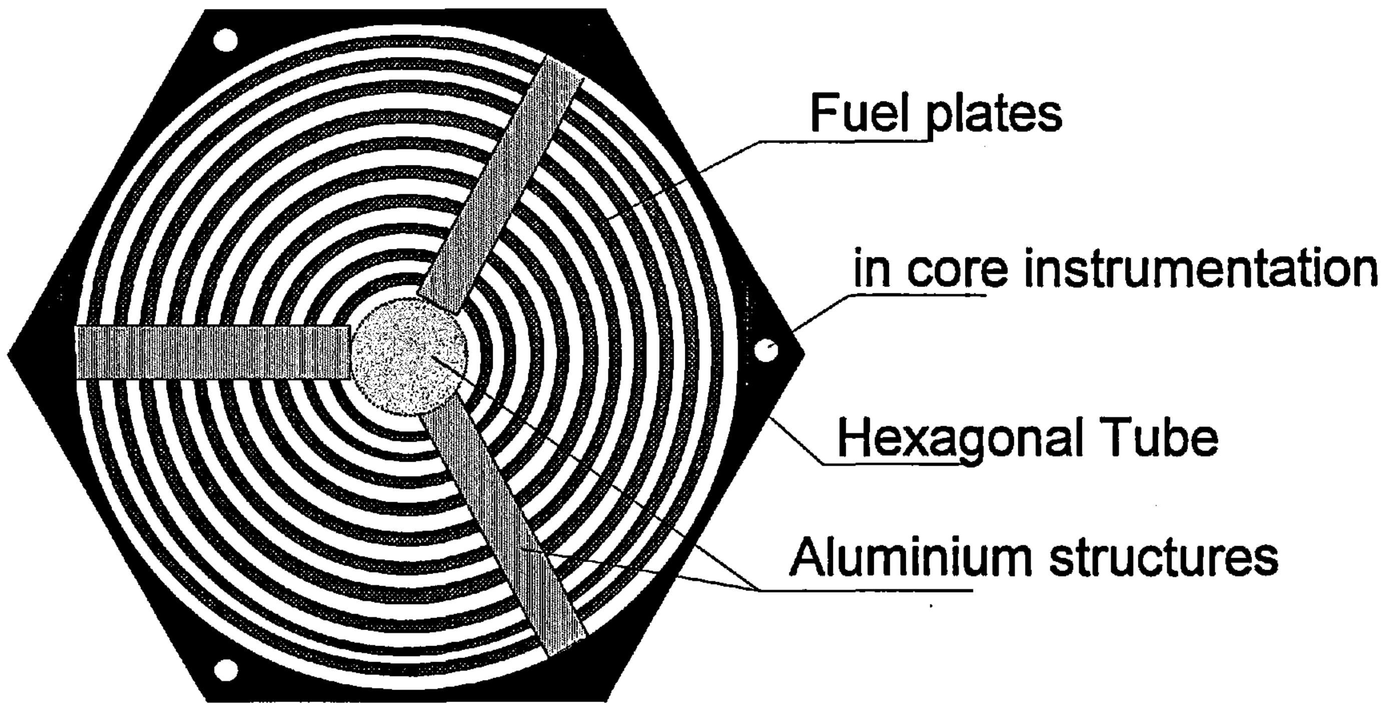
- "M" MODE = Mock up conditions
  (pre-irradiation neutron measurements)

The R.J.H. reactor

- NOMINAL EXPERIMENTAL CHARGE

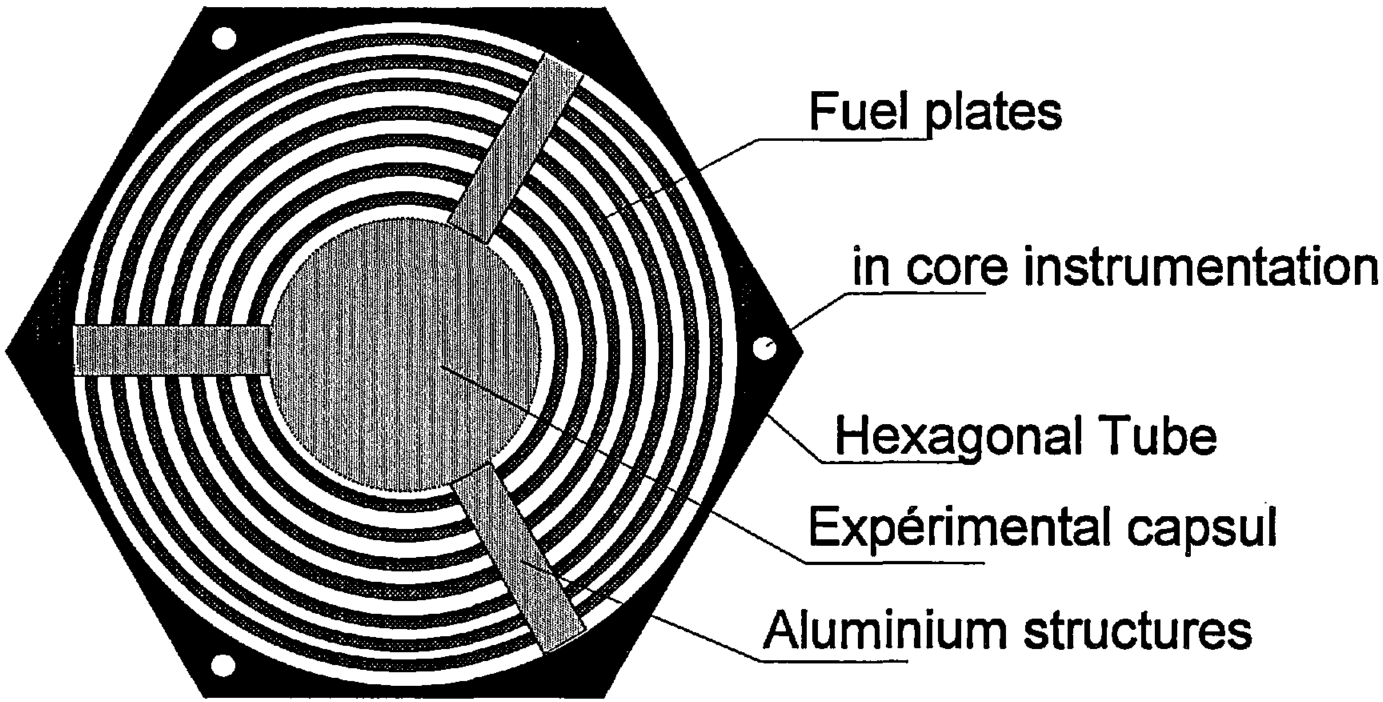
  10 E1 + 10 E2

The R.J.H. reactor



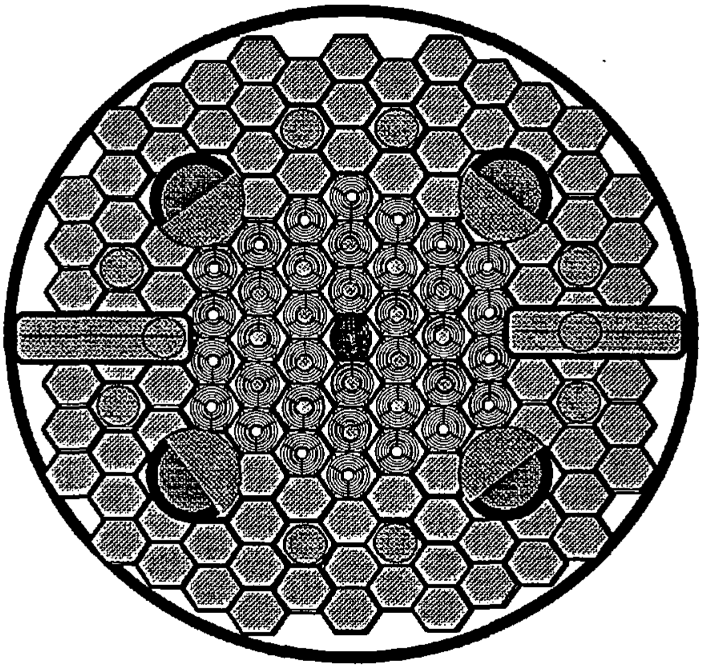
STANDARD ASSEMBLY

The R.J.H. reactor



EXPERIMENTAL ASSEMBLY

The R.J.H. reactor



Standard Fuel Assembly

Control Fuel Assembly

Mobile Absorber of compensation-regulation

Reflector (Be)

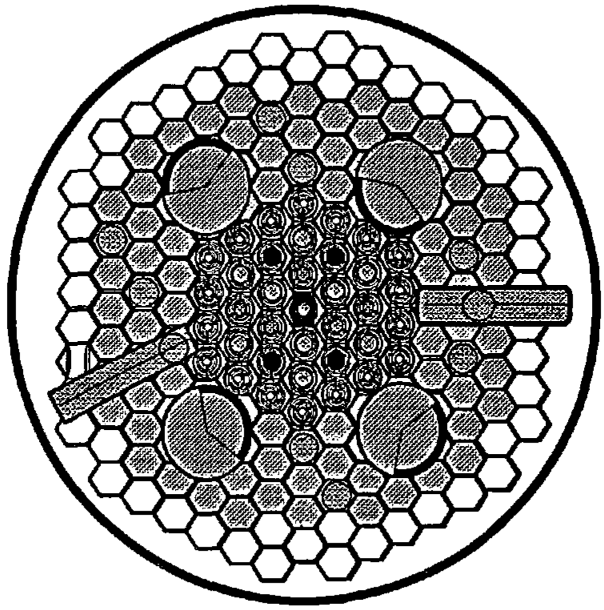
Experimental location in experimental assembly

Experimental location in reflector

Experimental location, displacement box

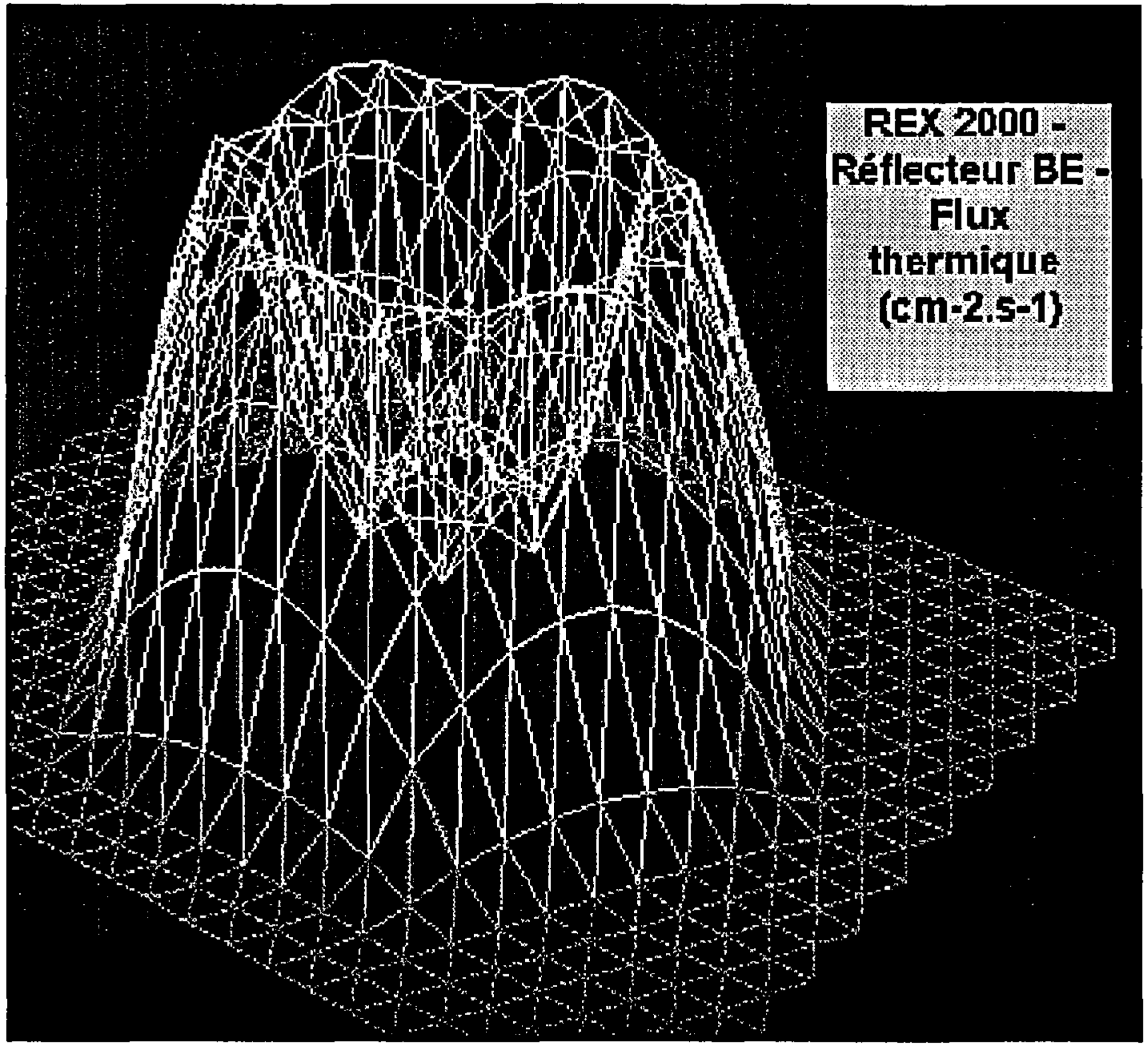
penetrating central loop
(FBR loop -PWR loop)

The R.J.H. reactor

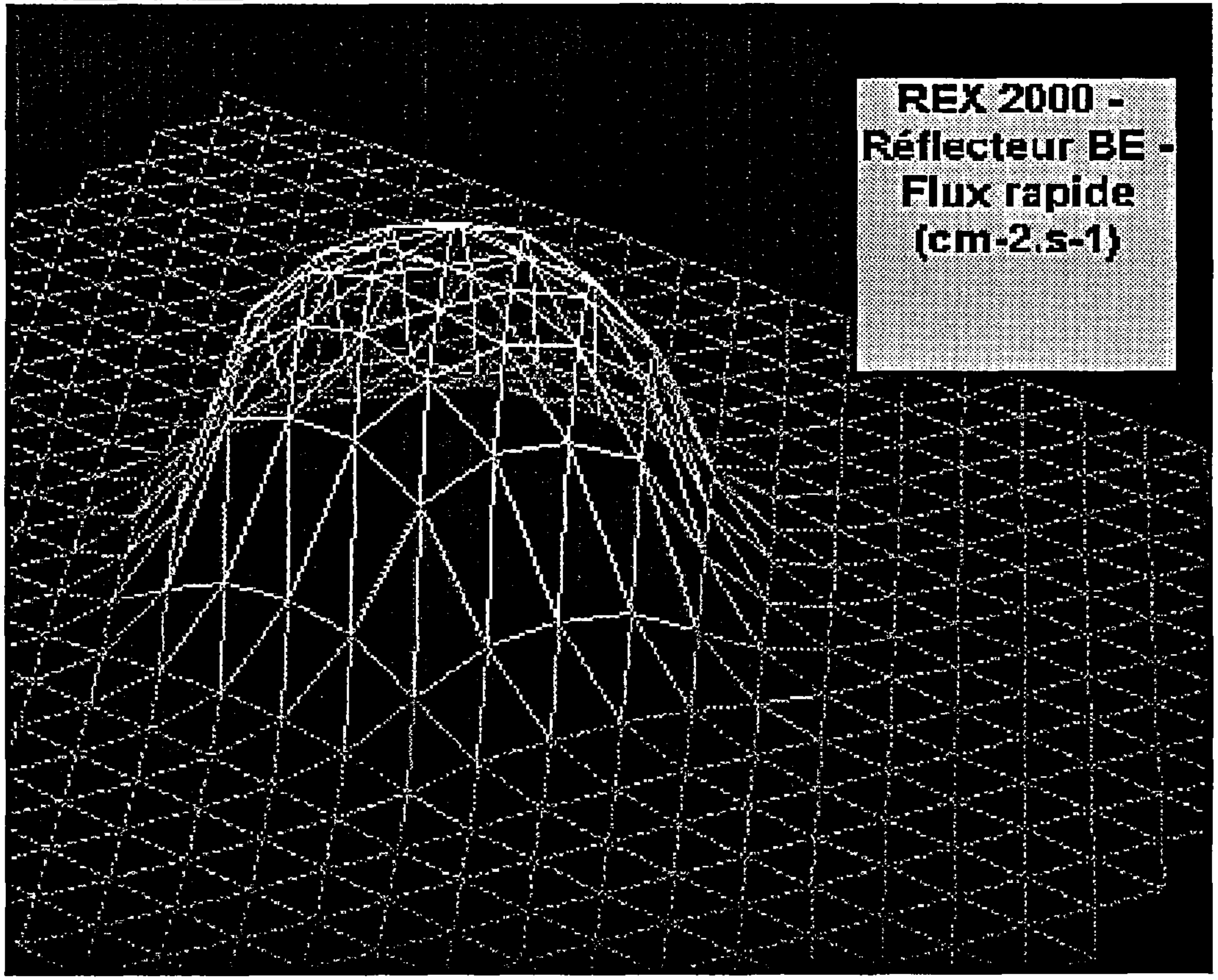


- Standard Fuel Assembly
- Control Fuel Assembly (Security absorbant)
- Control Fuel Assembly (Compensation absorbant)
- Mobile Absorber of regulation
- Reflector (Be)
- Experimental location in experimental assembly
- Experimental location in reflector
- Experimental location, displacement box
- penetrating central loop (FBR loop -PWR loop)

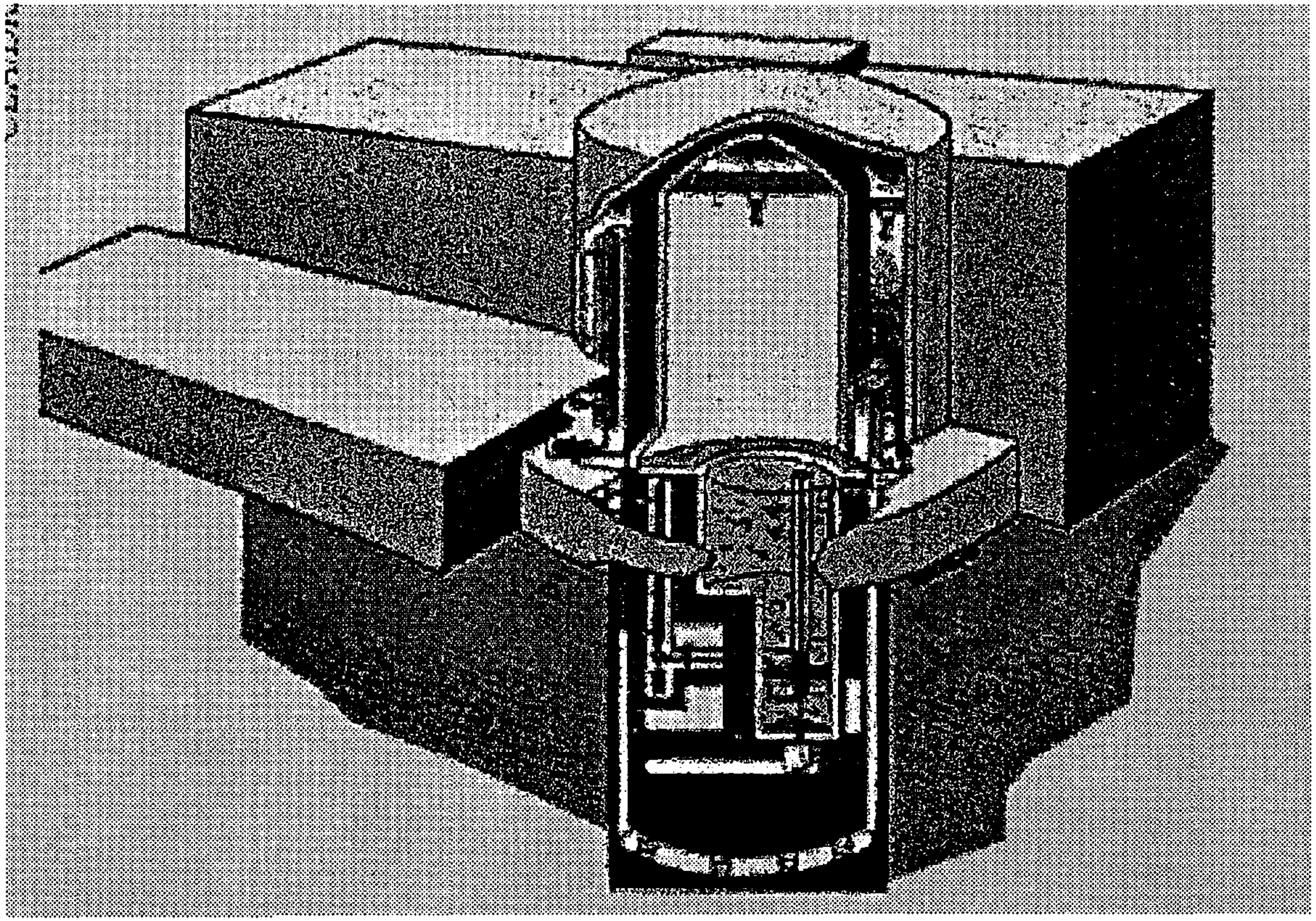
The R.J.H. reactor



The R.J.H. reactor



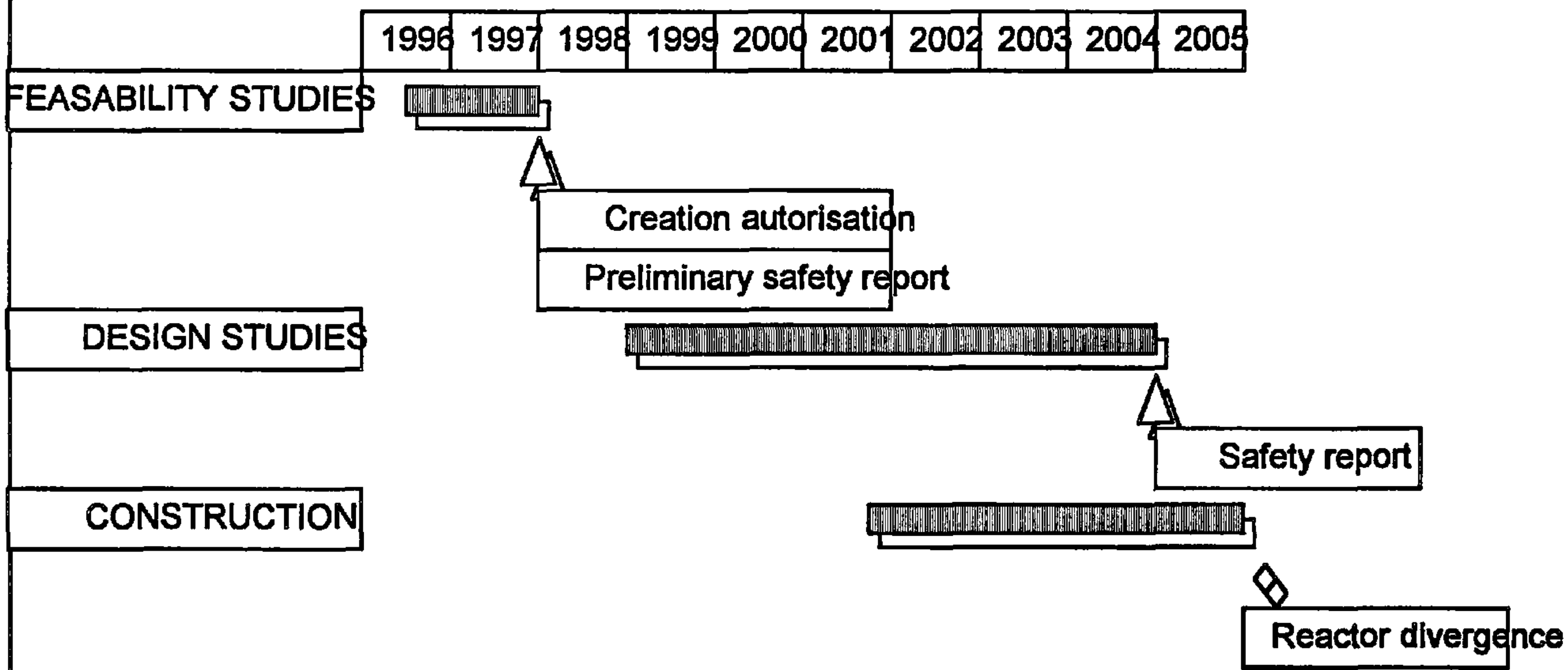
The R.J.H. reactor



The R.J.H. reactor

❑ PLANNING

PLANNING R.J.H.



CHAIRMAN : A. LEE

SESSION 2A

STATUS OF THE TRR-II PROJECT (Der-Jhy Shieh)

Question from Klaus Böning of TU München :

In the case of the Munich research reactor, we had to face a similar problem as what you said when you mentioned that at first you would like to dismantle the old reactor and then install the new structure. In the case of Munich, we changed our minds. There are many reasons for this. First of all, as we found out, it was more time consuming, of course, just to dismantle the old structure and then put in this new one. Second, it was difficult to settle a time schedule. It's very complicated - you never know what you will find when you begin to dismantle old reactors. You never know. You never can tell in advance. Third, we even thought it would be more costly to go this way. Finally, although our safety authority tried to convince us to change our mind because they said it was difficult to apply all the safety standards which had come during the last ten or twenty years. It's difficult to implement all this on an old structure, in the old building. For example, you would require a thicker wall structure, a thicker building, or the basement may not be sufficient. So we in Munich decided to build a new reactor just beside the old one.

A : Yes, I thank you very much for your comments. I think a lot of people here are making the same comment, though. But I think the reason for choosing to use the original site for building a new research reactor is just for political reasons, not for technical reasons.

Question from Balarko Gupta of AECL :

It wasn't very clear to me whether you have one loop on the primary cooling or two.

A : We intended to have just the one loop. The reason we thought this was that we tried to follow the design concept of JRR 3M in Japan and they have only one loop. For such kind of a low pressure system it is impossible to have a double ended break on the primary coolant pipe. So one loop is enough - that is their concept. I think that it is acceptable.

Additional comment :

Well, let me pass on to you our experience. We spend a lot of time - a lot of time - in human resources and talking to the regulators about one loop or two loops. Our experience has been that the more money you spend on hardware, pumps and heat exchangers, the less you'll spend in analyzing and trying to convince the regulators that one loop will do.

A : Again, I thank you very much for your comments. I think that maybe the trouble will come from the regulatory authorities. You have two loops and it is easier to get permission from the regulatory body. That may save a lot of time; that may be the case. Thank you very much.

Question from Jan Kysela of NRI :

You concluded that the new reactor should be a multi-purpose reactor. Have you considered some overlapping or constraints from one experiment to the other experiments?

A : You mean that with multi-purpose the different applications will have conflicts. When we started out, when we talked about this, we said, OK, problem applications - we asked the experts about problem applications. A 10 % power fluctuations would not affect their experiments. In this case I think that the conflicts will not be so serious. That's my opinion. You have a lot of experts here - maybe you çan give me some suggestions. That's clearly the case. Because, in our minds we said, OK, you can do the BNCT simultaneously with the neutron activation, and then use neutron isotope production while running other experiment. You can do all of this simultaneously. Isn't this the case?

Question from Edgar Koonen of CEN/SCK :

I was just wonder if anybody has ever used the ASME-III code to build a research reactor? In my opinion it's only used for constructing a high-pressure loop, but never for a low-pressure, low-temperature research reactor, so I don't think it's really an open question. And it will cost you a lot if you really apply ASME-III to all your materials.

A : Yes, thank you very much for your comments. We also think so because this is an operation system and the ASME Section III is mainly for the pressure boundary. But you talk to the vendors, and they all tell you that they use the Section III code, believe it or not.
All right, how about some comments? Are you using the Section III code for research reactors? No? Do you use Section VIII?

Answer from Jean-Luc Minguet of Technicatome :

No, in fact, the question is not so simple, we have a specific system of technical specifications..

Are they equivalent to Section III?

It can't be answered in that way. There are also specific materials.

Additional comment from AECL :

The question is what are your regulators like? If they will take what you give them, that's OK. If they really want to put a lot of difficulties for you, then if all systems are non-nuclear, you may face a lot of difficulties. That's been the experience at AECL in Canada.

Question from Francisco Alcala-Ruiz of IAEA :

I would also like to say something about this. In the late '70s, in a reactor of 3 MW, we had to weld a small tube to the primary pipe and we were obliged to homologate the welder following Section IX of ASME code. We were obliged by the checking people.

A : Yes, I think that the key point is what the regulatory bodies think. If they say you must apply ASME Section III, then you have no choice. You have to follow their suggestion. I think that that's the case in most countries, right?

Comment from Albert Lee of AECL :

We're running a little late, but I think he's hit on a lot of key issues that have faced all of us at one time or another in terms of uncertainties with how a regulatory body reacts and, I know Francis has one more comment to make, so I'll let one more comment happen and then we'll get on with the next paper.

Question from Francis Merchie of CEA :

Thank you, Albert. You have chosen to reduce the radiation level at the top of the reactor by installing a lid. Why don't you consider installing a warm water layer system at the top of the pool instead of a lid to reduce the radiation level?

A : I know that some research reactors have a hot layer, but the layers don't work very well. Even if a hot layer is still needed on top, sure, I think some reactors are working like that. So we don't know if the hot layer really works.

I can tell you that the hot water layers at OSIRIS and SILOE in France are working very well, and probably in many other facilities, so I think that it is an alternative for reducing the radiation level at the top of the pool reactor.

A : OK, Thank you very much for your suggestion.

OK, it's good to see a lot of discussion stimulated by the paper, but I'd like to now move on to the next paper since what used to be a lot of time until the end of the session has now become very short.

THERMOHYDRAULIC AND MECHANICAL ANALYSIS OF A SCALE FRM-II CORE DUMMY (Jürgen Adamek)

Question from Francis Merchie of CEA :

What is the water velocity along the fuel plates and what is the fabrication tolerance of the water channel between two fuel plates?

A : For the first part of your question, velocity depends on the flow rate, and that's 425 liters with a velocity of about 25 meters per second. And the tolerance in the width of the cooling gap is 0.25 millimeters by a width of 2.2 millimeters of the cooling gap.

Comment from Mr Böning of TU München :

One can say that 17.5 meters per second is the nominal flow rate at 300 liters per second.

Question from Kir Konoplev of PNPI :

Do you suppose that if you immerse the core in the tank, surrounded by the water, will the vibration results be the same?

A : I don't think that they will be exactly the same. But on the other hand, I don't think that there will be major changes that will affect the normal behavior of the fuel element.

Question from Johannes Wolters of Jülich Research Center :

I was a little bit surprised by the figure about the pressure loss on the flow rate. Can you show us it again.

Yes, of course.

To me it seems that there is no quadratic dependency on the flow rate. It looks linear, but the problem is perhaps...

A : The problem is the viewpoint on this graph.

Question from Doug Selby of ORNL :

Yes, I have two questions. One, have you looked at any fuel plate deflection at all? Distortion of the plate from the flow, and how that might affect these tests? Secondly, I noticed that you have a very long flow path before entering your test section, which produces a well-developed linear flow. Is that section prototypic also, including screens and things of that nature before entering the fuel element?

A : OK, for the first part, we make the measurement at the nominal flow rate and at flow rates which go beyond that, and we look at test element after this measurement. We haven't yet investigated any deflection of any fuel plate.
The second part of your question : we've mocked up the vertical region of the reactor cooling system of the FRM-II and we have got a long straight between the upper tee and the inlet into the fuel element. Anything which is before the upper tee and below the diffuser,

corresponding to the diffuser in the lower communicating chamber, does not correspond exactly to the FRM-II.

Question from Albert Lee of AECL :

I've got one question as well. These measurements that you've made with cold water, right? At 20° or 30°C?

Yes, that's right.

Have you got some feel for how the frequency response will change as water heats up and the density changes?

A : As we have determined the eigen frequencies in dry conditions, and also in operating conditions, we haven't investigated any major changes in these eigen frequencies. Therefore, I don't think there are further changes when the water temperature rises.

Question from Albert Lee of AECL :

The plate deflection issue could, I think, become more important if you get into a situation where you start to develop a small amount of sub-cooled void on the surface of the plates. It can become quite significant depending on how much of sub-cooled void is developed. You get thermal gradients in the plates.

A : That may be possible. But in this case, the fuel element isn't heated because, of course, it's a depleted dummy, and all we can do is raise the water temperature by using our pumps, which is possible up to a temperature of 60 to 70° C, which is above the normal operating temperature of FRM-II. Further tests, of course, will be performed.

Any more question? If not, I'll close the session and turn it over to Colin. I'd like to thank all the speakers.

Question from Jean-Jacques Verdeau of Technicatome

What is the cost of this reactor?

A :That is very important question - The overall cost, including 15% tax, is 720 million Marks. This is for the complete reactor, complete experimental facilities, that means the complete plant which is necessary to use it as a neutron source ; 720 million marks.This budget is available (secured). We are free to use it as it has been passed by Parliament.

CHAIRMAN : JL. MINGUET

SESSION 2B

A STATUS REPORT ON THE PROPOSED CANADIAN IRRADIATION RESEARCH FACILITY (Albert Lee)

Question from Klaus Böning of TU München :

It seemed to me that out of the 10 horizontal beam tubes, some are more or less "radial", i.e. looking toward the core, which enhances background radiation.

A :Yes, it is not desirable. Unfortunately, what we're having to live with in this facility are some very difficult compromises and one set of compromises which means that I can't spread the beams out as far as I want are those horizontal test loops. The horizontal test loops will penetrate through the reactor pool on either end, with a shielded vault at one end, where I locate a fueling machine. But I have to have access from both ends for the kinds of experiments that people want to do, which means that I can't spread the beam tubes 330° around the periphery of the reactor pool, which is what the beam researchers would really want. I have to constrain it to segments that are more like 120° on each side. If you think about it, with five beam tubes on each side and a spread of about 120°, you can't get them all to be tangential. It's a compromise one has to live with in a multi-purpose facility.

Question from Doug Selby of ORNL :

Albert, concerning the zirconium vessel - I believe that at one time you were looking at possible usage of a relatively new zirconium niobium. Is that the material you are using already or is it a zircalloy alloy?

A : No, what we expect to do is to build the upper part of the tank out of zircalloy 4, build the lower part, the dump vessel, out of a conventional chemical-grade zirconium alloy. We can live with higher impurity levels and the presence of some hafnium in the lower part of the vessel because it is out of flux and is not going to be subjected to creep and growth and swelling, or embrittlement. But we want the vessel material at the upper part to be made out of a material that we know won't creep, grow or embrittle rapidly over, say, the first twenty years. The choice between zircalloy 4 and zirc-niobium, right now it's half and half - you can go either way, but in the cold temperatures, low temperature-low pressure, there is no reason to use zirc-niobium.

Question from Horst Hassel of Jecta Consulting :

I understood that this project is completely independent from that your friendly and gentle customer wants to have, these two other MAPLE 10-Megawatt production reactors

A : Correct. This reactor concept is designed to replace the materials testing and the fuel testing and the beam tube capabilities of NRU. It will not replace - I'll never say never -but at the moment it does not have any isotope production capability built into it, although in practice it's not hard to install. The reason for that is we have a commercial contract with NORDION International to provide them with two dedicated radioisotope production reactors to be built at Chalk River and to be owned by NORDION.

Q : The background to my question is, do you have engineering staff and construction staff? You have three reactors to be constructed, designed, licensed and so on at the same time. And you reduced the time for this project, and there is big pressure from NORDION to reduce the time for the two other projects. Do you have enough staff for that?

A : We do not intend to use AECL staff for the construction. We expect that all the civil engineering for this project, the civil engineering for the two reactors for NORDION to be contracted out to private companies. We expect that the majority of the project management staff for both projects will be contracted out to private companies. AECL engineering, physics, thermo-hydraulics staff, safety and licensing staff will be used to do the nuclear part of the work. And we will contract out for the rest.
Having said that, the answer is, in the last six months I have hired 11 people to train them to be thermo-hydraulics analysts and reactor physicists, and I expect to be hiring somewhere in the neighborhood of another 10 in the next six months.

Question from Patrick Martel of CEA/DRN/DER/SIS :

What kind of Quality Assurance have you chosen?

A : The design work is done to what is called the Canadian Standards Association N286.2 program. It's the same quality assurance program that's used for the CANDU power reactor for engineering design.
The commissioning program - and this has been subject to negotiation with our regulators - will be performed to the intent of CSA N286.4, which is again is the quality assurance program for commissioning for the power reactors. We only intend to meet the intent of it, not to meet it to the letter, bearing in mind that there are sufficient differences between this kind of reactor and a power reactor, so it's not possible to meet everything.
In terms of the component and equipment specifications, depending on the system - the high temperature/high pressure systems are all ASME section 3 class 1, they are high temperature high pressure components for the fuel test systems that operate at 300 °C, 10 MPA pressure. The vessel itself, depending upon the accident analyses that will be done, we will decide on the quote class. We haven't decided yet.

PRELIMINARY STUDY OF CORE CHARACTERISTICS FOR TRR-II
(Hans-Jürgen Didier)

Question from Klaus Böning of TU München :

Your preliminary core design uses very unusual fuel elements (UO_2, enrichment = 4%, rod type), so you will have to accept penalties which you would not have to if you used typical MTR fuel elements (U silicide, enrichment 20%, plate types). Since then, you would obtain more excess reactivity and better cooling conditions, i.e. you could make your core smaller (with the same power) and produce higher thermal fluxes in the D_2O reflector. But you would have to make the decision now before starting construction.

A :Yes, as I said, this study is very preliminary. We don't have any experience in designing a research reactor, so we will start from what we are familiar with, for example the PWR fuel rods. So this study will show whether we have the capability to do some necessary calculations from the beginning, for example the core calculation, to the design basis accident and safety analysis calculations. So we intend to go in this way. As the project stands now we only have the budget for a very preliminary project study instead of a real project. I think that when we go into the real project we will think very carefully about what you just said.

Question from Hans-Joachim Roegler of Siemens :

Although the fuel design was already put under question by Prof. Böning, I want to add one question : In case you really take the rod-type element, why do you use boxes around it instead of an open elements? What was the reason for the boxes?

A : We have discussed this question a lot in our institute. We did not find out that this open channel design had very strong advantages, because from our thermal hydraulic calculations we did not find that there is much of a difference for the minimal departure for nuclear boiling ratio. The structural people suggested a box - actually we are not thinking about a box, we are thinking about a channel flow tube for this core, so that would be better from a structural point of view. But this also should be considered in more detail when we go to a real design.

Question from Christian Desandre, Consultant :

Considering the cylindrical shape surrounded by the beryllium, I'm surprised that first you have a beryllium reflector and then afterwards a heavy water reflector. Have you made any calculations with the beryllium reflector and without it, replacing the beryllium by a void or another material? I am wondering if by using the beryllium reflector you are not decreasing the flux into the heavy water reflector?

A : This still very preliminary study actually is a typical study that you have not in the other research reactors. You are saying that for a circle one - there is a possibility that you don't need a beryllium one outside the fuel bundle. That will be one possible consideration for the optimized design.

Question from Johannes Wolters of Jülich Research Center :

What is the power density in your concept? Is it higher than in a pressurized water reactor and if so, what is the temperature in the center of the fuel rods?

A : The power density is almost exactly as in the pressurized water reactor. It is a little higher than one hundred kilowatts per liter. For the center of fuel temperature I do not have an exact number. Dr. Yang, do you have an exact number?

Dr. Yang : 1400 degrees.

Question from Albert Lee of AECL :

I'd like to make a comment about your physics calculations. If you have a fuel temperature of around 1400 °C, you have to be very careful about comparing your calculations from your CASMO-VENTURE results with the results from MCNP because, unless you've modified the cross-section library for MCNP, the default library that comes with the MCNP has uranium temperatures that are room temperature and has uranium temperatures that are up in the range used at Los Alamos for verifying thermonuclear devices. And it doesn't have temperatures for uranium that's in the 600 to 200 degree range. There will be an absolute reactivity shift as well as a reactivity coefficient shift between the two sets of calculations if you're not careful.

A : Yes, thank you. We'll look into that.

CONSTRUCTION OF THE HTTR AND ITS IRRADIATION PROGRAM
(Masahiro ISHIHARA)

Question from Christian Desandre, Consultant :

I would like to know if the type of fuel element is the same as in Fort St. Vrain and if you use coated particles with three coatings, one of silica carbide between two pyrocarbons?

A : Yes, it's exactly the same.

Q : And are you not afraid that operating at 950°, because at Fort St. Vrain it was operated at 850°. Are you not faced with moisture problems and corrosion problems? Because you know if there is a small amount of water in the circuits, corrosion increases very much with the temperature.

A : The composition is the same but the size is different from the hot source brightness and we also added a test with our coated particles.

Question from Jean-Jacques Verdeau of Technicatome :

Could you tell us some information about the materials you used for the internals of the core, the pressure vessel and the thermal insulation?

A : The carbon material is used because it has low thermal conductivity. It's very spare thermal resistance. So, the bottom of the part has carbon components for thermal insulation.

Q : You use carbon. But is the pressure in carbon steel?

A : Chrome molybdenum steels. One coat of molybdenum steel.

Question from Klaus Böning of TU München :

What was the reason for placing the HTTR reactor underground? Because of airplane crashes?

A : Because, in Japan, as you know, we have many many earthquakes. This reactor has graphite blocks which are piled up like this, so it is an advantage to have this reactor underground for earthquake motions. That is the main reason.

Question from Johannes Wolters of Jülich Research Center :

Is the pressure vessel cooling system capable of removing the afterheat in an emergency situation? And I have a comment on the question from Mr. Desandre. I believe that in Fort St. Vrain they has steam-driven blowers and there was sometimes steam ingress into the system, and that caused the corrosion problem in the system. Here they have electrically-driven blowers, I think.

A : You mean the cooling system on the pressure vessel? We have two independent groups of cooling systems for the pressure vessel. Also, in the safety analysis, if the cooling system is not working, but the temperature of the component is not too high, around 100°C, if this cooling system is not working it is not a significant problem for the safety demand.

Question from Kir Konoplev of PNPI :

There is one thing I do not understand. What is the reason for the irradiation high temperature superconductor? Why such a high temperature? Is it for superconductor technology or especially for high-temperature reactor technology?

A : It is set by the researchers that at high temperatures, irradiation promotes the characteristics of the superconductors. Because the damage produced by irradiation is annealing and results in an effect in the characteristics that is higher.

Question from Jean-Luc Minguet of Technicatome :

What are the funding for this project and their origin ?

A : This project originated with the government.

REACTEUR JULES HOROWITZ : A NEW MATERIAL TESTING REACTOR PROJECT (F. Merchie)

Question from Albert Lee of AECL :

Francis, I won't ask you how much it costs for the reactor because I think I know the answer. What I'm curious about is the preliminary safety analysis report. In Canada, when we produce a preliminary safety analysis report for our regulator, we have, technically, between 60 and 75% of all the engineering of all of the nuclear systems complete and we have between 40 and 50% of the conventional balance of plant systems design complete. What stage of the engineering will you be at when the preliminary safety analysis is complete?

A : The preliminary safety report is produced at the end of what we call the feasibility studies. So, the studies are less advanced than in your case. We need that to obtain the authorization decree. But after that, we have submit a provisional safety report and then the final safety report - there are three stages. About the cost : OK, it is difficult to answer, but we this concept now, the estimation is between 300 and 400 million dollars for the reactor itself, not including all the experiments that are associated with the reactor. I mean the central loop, for example.

Question from Hans-Joachim Roegler of Siemens :

Your evaluation of the lifetime of the existing European Material Test Reactors differs drastically from the evaluation Colin West made for the HFIR reactor. It assumes that it can easily be handled up to 2030. Now, since the techniques of corrosion and embrittlement would be the same on both sides of the Atlantic, would you assume that this is just a question of licensing requirements which come additionally to all power plants in relation to the United States?

A : Well, there are several aspects : the technical aspects are of course important, but the political aspects are also important. For example, some of those material testing reactors are located in countries where there are no more nuclear programs, such as in Norway. So it is very good news for us if such a country decides to replace the reactor after it reaches the end of its life. But, as you know, we need at least 10 or 15 years to build a new reactor nowadays because of safety requirements. So, we have to take time in advance and that is the reason why we have decided to go ahead with this project.

Question from Horst Hassel of Jecta Consulting :

Is it correct understood that this reactor will replace the two reactors, SILOE and OSIRIS. SILOE will be shut down at the end of next, and you are planning to shut down OSIRIS in 2005.

A : Something like that. This is another aspect because, as I told you at the beginning, the long-term strategic plan of CEA is to set up R&D facilities for the next decade. This is a political decision. Of course, we know that the OSIRIS reactor will still be able to produce irradiation at the beginning of the next century, but the CEA has to reduce its size in the future and operating two big nuclear centers will be too expensive.

Q : Will the reactor also have a capacity for moly 99 production?

A : The reactor will be totally dedicated to material and fuel irradiations but, as a backup for other reactors, there will be provision for producing some radioisotopes in the reflector. But is not the main objective of the reactor.

Question from Klaus Böning of TU München :

You said that you expect the "creation authorization" by the end of next year.

A :No, no! We will submit all the documents. We know that it takes approximately 18 months or two years to obtain the governmental decree.

Q : My question was that I got the impression that the decision to construct the R.J.H. reactor has already been made, that the money is there.. Is this correct?

A : The decision has been made inside CEA, it has been approved by our Ministry of Research but as with your project - you know the situation well! - we have different obstacles to overcome.

Question from Christian Desandre, Consultant :

R.J.H. is dedicated to technological irradiation. FRM II is dedicated to basic research. Don't you think that there is a need on the European level for a reactor dedicated to radioisotope production in the near future?

A : I know, and of course this is not the same type of reactor. We know that some people are considering this idea, I mean the producer of radioisotopes in Europe. It could be that in the future a project of this type may come, but it will be very interesting to see the Canadian experience by building two Maples for radioisotope production. I think that the Europeans will be very much influenced by the Canadian decision. I think it's still open.

IGORR 5

SESSION 3

COLD NEUTRON SOURCES

PAPERS

PROPOSAL

Experiments with cold sources for neutron physics analysis.

Konoplev K.A., Kudrjashov V.A., Porsev G.D., Potapov I.A., Trunov V.A.
Vasilev G.Ja. and Zakharov A.S.

Cold neutron sources (CNS) on the research reactors provide the considerable volume of investigation. Advance computing programs on CNS and experiments in this range - is one of the priority tasks of future development of investigations using the reactor engineering.

It is suggested to carry out the experiments with CNS using hydrogen and deuterium moderators, it's mixtures, experiments with changes of steam fraction and moderator temperature up to solid phase. Source shape also will be changed. The task of all this benchmark experiments is to perform cold neutron physics analysis and verification of computing programs.

It is possible to test the computing programs of customer by experiment results on the reactor PIK critical facilities with cold neutron spectra investigation and carrying out the experiments for special customers request.

In the project it is planned to use existing critical facility "Physical model of the PIK reactor" and cryogenic helium facility with help of personnel's having experience on creation and exploitation of cold and ultracold sources at ≈20°K.

The aim of the work

1. Verification of computing programs and cross-section libraries using for design cold neutron sources (CNS). Benchmark experiments convenient for computing programs.
2. Definition of some technological and structural parameters influence on neutron outlet from the experimental channel in the range of wave length 1-10Å. The range expansion up to 15Å is depended on CNS design.
3. Development of cold neutron source design in order to rise the neutron outlet, efficiency and experiments safety.

1. Introduction

Experiments at the critical facility are flexible and accessible during the investigations, well known initial data allow to ensure truth comparison of experimental data and calculation results. Another approach to experiment can be carried up for some details of design that is difficult to account in computing programs.

2. *Critical facility review*

- "Physical model of the PIK reactor" is a full-scale copy of PIK reactor which is under construction.

critical assembly power	100W
diameter of the heavy water reflector vessel	2400 mm
external diameter of the active core vessel	420 mm
moderator	H_2O
active core volume	50 l
active core height	500 mm

Critical facility is equipped by set of the experimental channels, which can be removed for time of the experiment to provide convenient "clean" conditions for the experimental and calculated comparison. Critical facility at 100W give thermal neutron flux in the level 10^9 n/cm^2 s. Calculated spectra is presented in fig. 3-5. Neutron flux in the reflector is given in fig. 6

3. *Review of the PIK reactor cold neutron source*

It is an example of one of the variant for investigations(fig. 1 and 2).

Source: Al. sphere Ø 380 mm, two types of the performance: without and with the inner cavities, which allowed to outlet neutrons from the center of moderator.

Protective zirconium jacket with wall thickness 6 mm.

Calculated γ,n power releases: 10^{-2} W at 100 W power.

The main heat influx 20 W will be defined by thermal radiation.

Moderator D2 (T=20 °K).

The connection of source and jacket with condenser is carried out with Zr - stainless steel, Al - stainless steel adapters.

Equipment for the experiments (partly exist):

Heat exchanger - condenser;
Cryogenic pipes;
Receiver (D2);
Vacuum system;
Nitrogen system (liquid and gaseous nitrogen).

4. *Time-flight analyzer review*

Resolution of TOF - device $\Delta\lambda = 0{,}25$Å .

Ranges of the measurements on λ, 0,5 - 15Å.
Fermi - chopper characteristics:

path-length	2,67 m
amount of the apertures	4
rotation frequency	1500 resolution per minute
width of the apertures	5 mm (0,25Å)
middle radius of the disk by apertures	191 mm

Counters types CHM - 17 (^{3}He)
natural background ≈ 0,001 - 0,003 counts per second.
forecast intensity in the channel at 7-8 Å 0,3 counts per second after CNS filling by liquid D2.
Distance from CNS to chopper 2 m.
The scheme of analyzer arrangement is given on fig. 7.

5. *Cryogenics supply*

Variant 1.

The development on base of the existing equipment ХГУ 4000/20 with cold helium connection to condenser by cryogenic pipes is supposed. At present time helium refrigerating facility ХГУ 4000/20 is used for the experiments service on WWR-M reactor, but it's capacity will be enough for the additional service of the experiments at the critical facility.

ХГУ 4000/20 main characteristics:

- refrigerating capacity 4000 W
- working gas - high purity He
- working pressure in the cycle - 2.8 MPa
- He volume 600 m^3
- power 452 kW.

The merit of this variant is in saving time for the project realization by using of ready cryogenic pipes, shortage is due to long cryogenic pipes and the project cost increasing.

Variant 2.

Using cryogenic facility КГУ 600/15 from PIK reactor with it's location near critical facility is possible.

КГУ 600/15 characteristics:

- capacity at 15° K 600W.
- He pressure not more 2.5 MPa
- compressor capacity not less 360 m^3/hour
- flow of liquid nitrogen 32 kg/hour
- He volume 12 m^3

- power 80 kW

- This variant is less costly. Shortage is connected with the increasing of project realization time.

Total cost of the project will be decreased on 60-70 thousand USD due to reduction of cryogenics pipes length.

The scheme of cryogenic supplies in fig. 8.

6. *Calculations methods.*

For the spectrum and flux in the cold neutron source evaluation the coupling of discrete ordinates (Sn) and the Monte-Carlo (MC) methods across surface sources are applied.

In the calculations by Sn-method the angular fluxes are calculated along the direction with the given cosines and weights when the source is prepared for MC-method. The angular flux is rearranged in the polar and azimuths angles, in the space intervals at the chosen surfaces (from 1 to 6) and in the energy groups. Then normalization is carried out and the necessary functions of probability density is received.

The calculated neutron spectra in the heavy water reflector inside of the cold neutron source and in the outlet of the experimental channel N3 are presented on fig. 3-5.

7. *Peculiarities of experiments realization*

Small power of critical assembly creates the problems with sufficient statistics. This problems are expected to solve due to selection of detector with small background level and creations of effective neutron shielding.

Preliminary investigations show the possibility of the background reduction in the physical hall conditions and the possibility of using time of flight method for the investigations in the region near 8Å. Background near natural can be done by liquidation of all experimental channels, except one, which is used in the experiment and by increasing of neutron shielding thick. As a reserve for the statistics improvement the possibility of power increasing is considered.

For researches it is proposed to define the testing designs which are the most appropriate for the testing of the applied calculated model and/or applied design.

CNS can be disposed in vertical channel with beam through the horizontal one and in horizontal that has diameter 400 mm. As an one of the possible CNS it may be consider the design which is developed for the PIK reactor.

8. *Stages* .

1. The investigation of initial neutron spectrum without cold neutron source by time of flight method. Refinement of energy boundaries for region of possible investigations.
CNS spectra calculations without and with using of low temperature moderator. Development the expansion of neutron wave length range more than 10 Å.
2. Experiments with concrete CNS mock-up without using low temperature moderator. Refinement of initial data, which determined. CNS working conditions : radiation energy release, fast and thermal neutron fluxes. Calculations and experiments comparison.
3. CNS investigation with cryogenic equipment connection.

9. *Tentative cost of the project.*

	variant 1	variant 2
analyzer with neutron shielding	15 000 $	15 000 $
cryogenic equipment		
cryogenic pipes	100 000 $	10 000 $
cetera	60 000 $	80 000 $
design	2 000 $	2 000 $
experiments (exploitation expenses and personnel's salary)	140 000 $	140 000 $
calculations (partially renovation of the software's)	10 000 $	10 000 $
Total:	327 000 $	257 000 S

Expenses distribution by stages :

		variant 1	variant 2
stage 1	0,6 year	50 000 $	50 000 $
stage 2	0,4 year	50 000 $	50 000 $
stage 3 (beginning after 1 stage)	1,4 year	227 000 $	157 000 $

For then investigation of second and subsequent CNS designs the cost will decrease due to using of the equipment which is indicated in first two lines.

10. *Proposal for collaboration* .

For the project realization it is necessary a financial support of CNS consumers.

PNPI offers on experimental results for the verification of programs for physical and technical calculations of CNS, implementation of calculations for design of CNS. Participation in experiments is welcome.

PNPI is eager in the wide exchange of information about the experiment results, methods of calculations and software's in CNS neutron physics analyses.

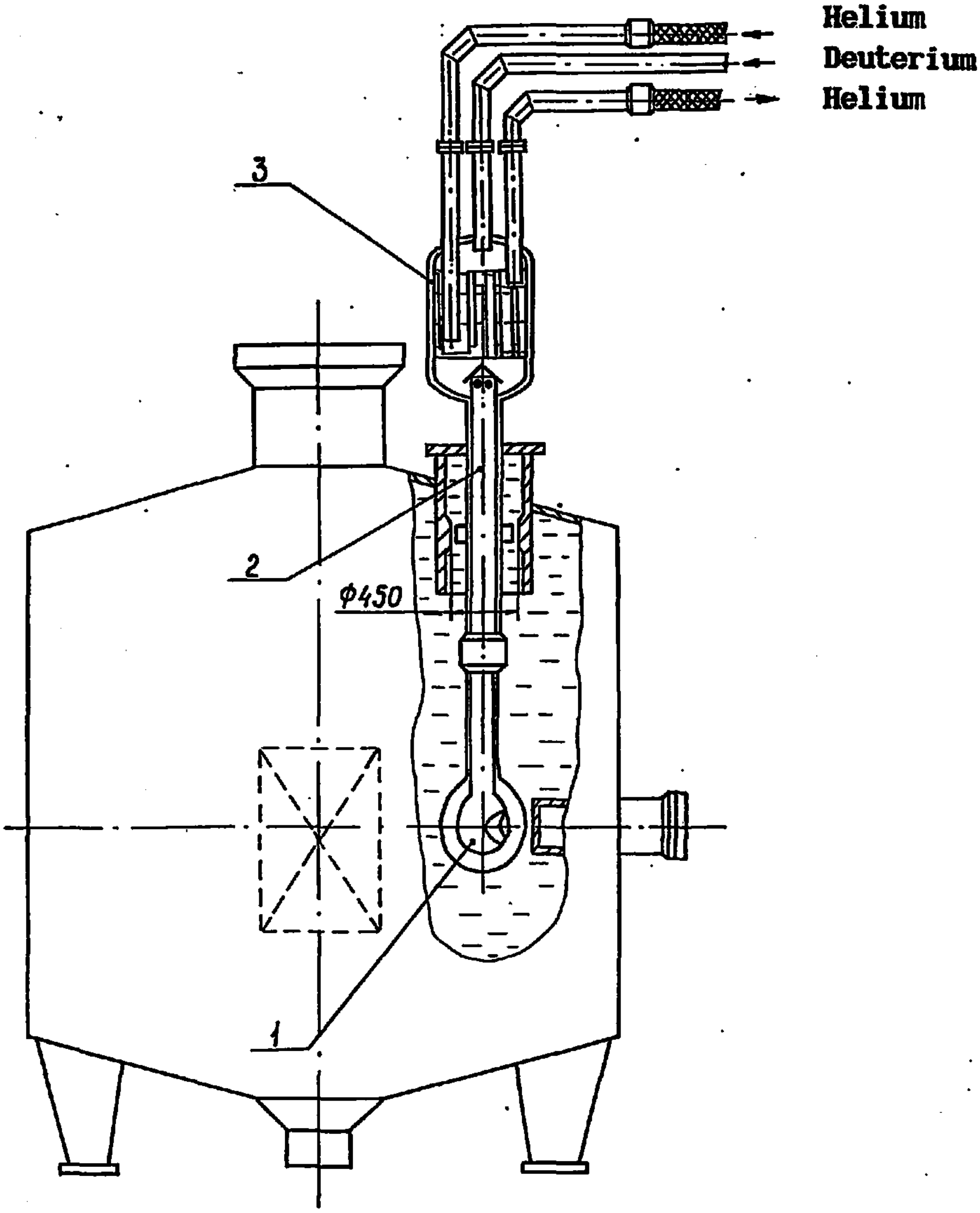


Fig.1. Layout of CNS at the reactor facility
1- CNS
2- Cryogenic pipe
3- Heat exchanger-condenser

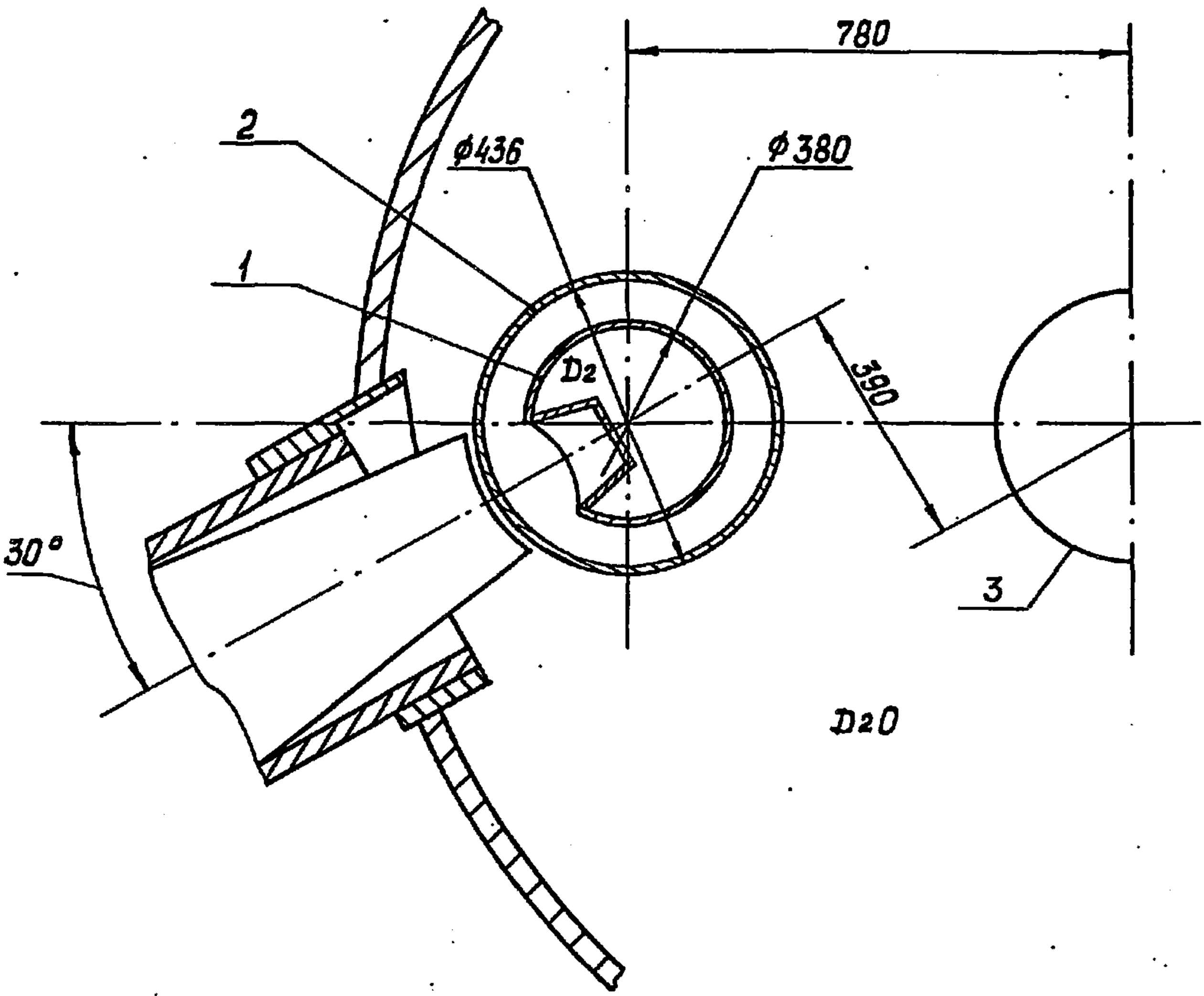


Fig.2. Layout of CNS in the heavy water reflector
1- CNS chumber
2- Protective zirconium jacket
3- Core

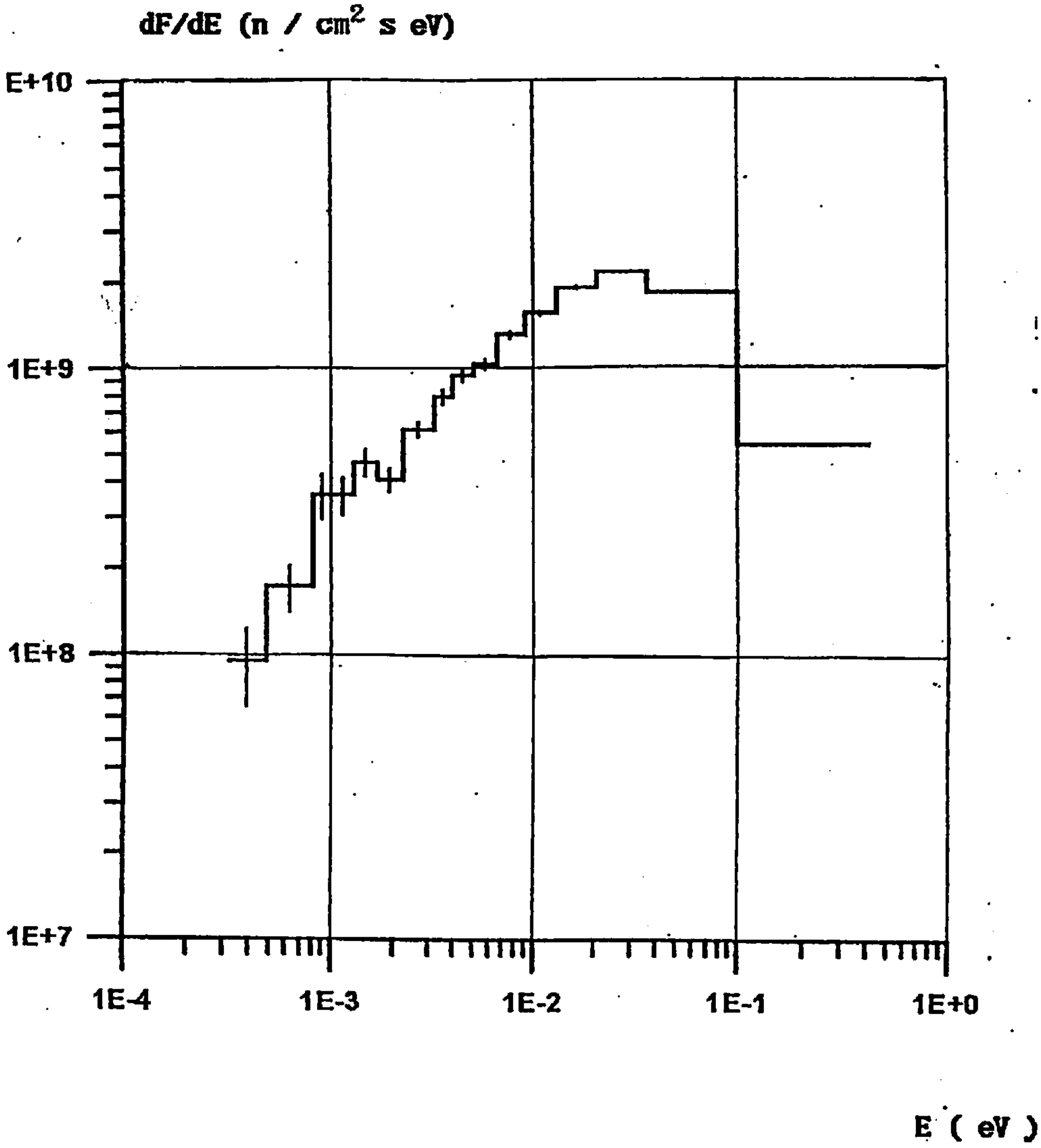


Fig.3. Neutron spectrum incide heavy water reflector at place of cold neutron source (P=100 W)

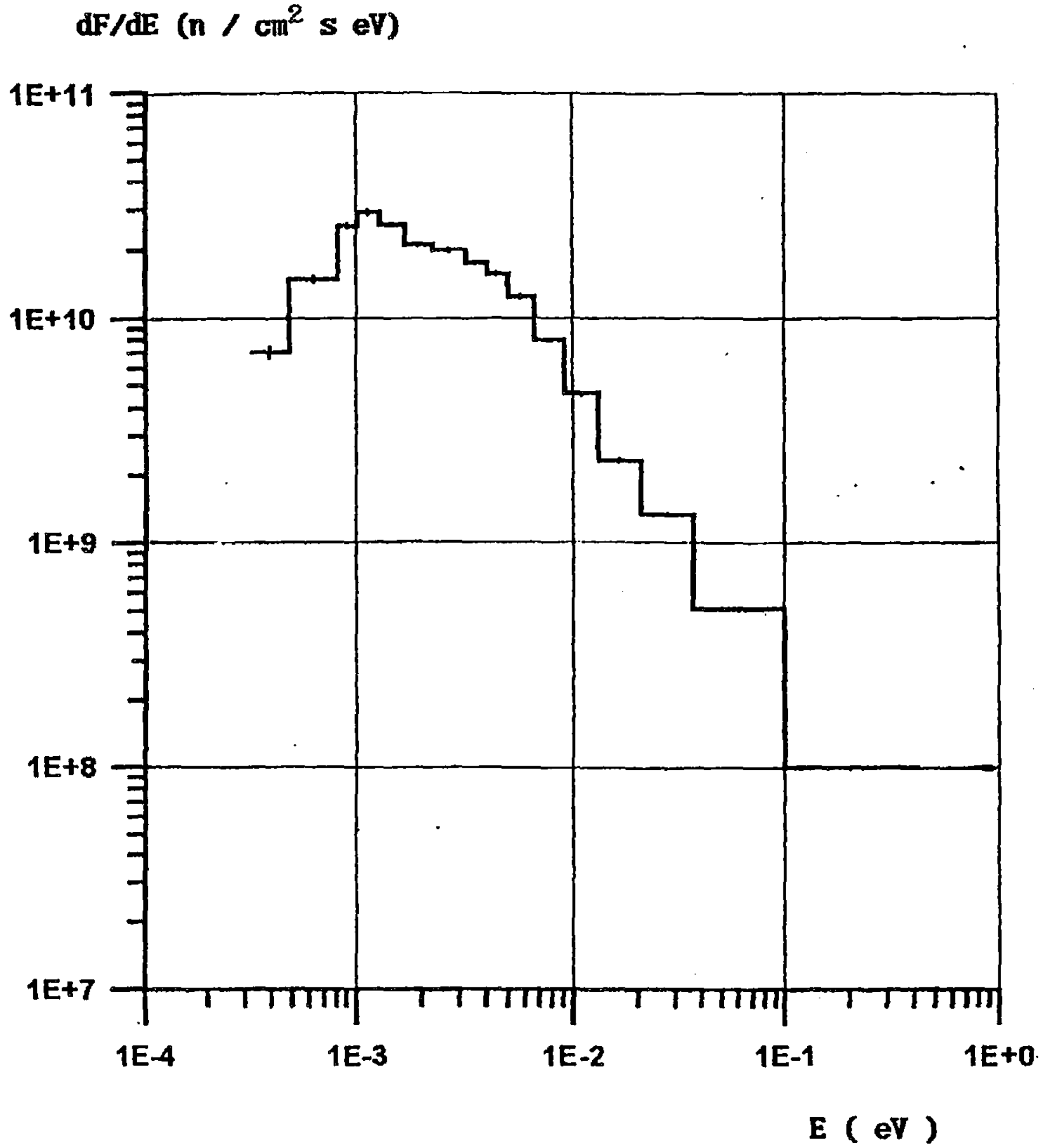


Fig.4. Neutron spectrum incide cold neutron source
(P=100 W)

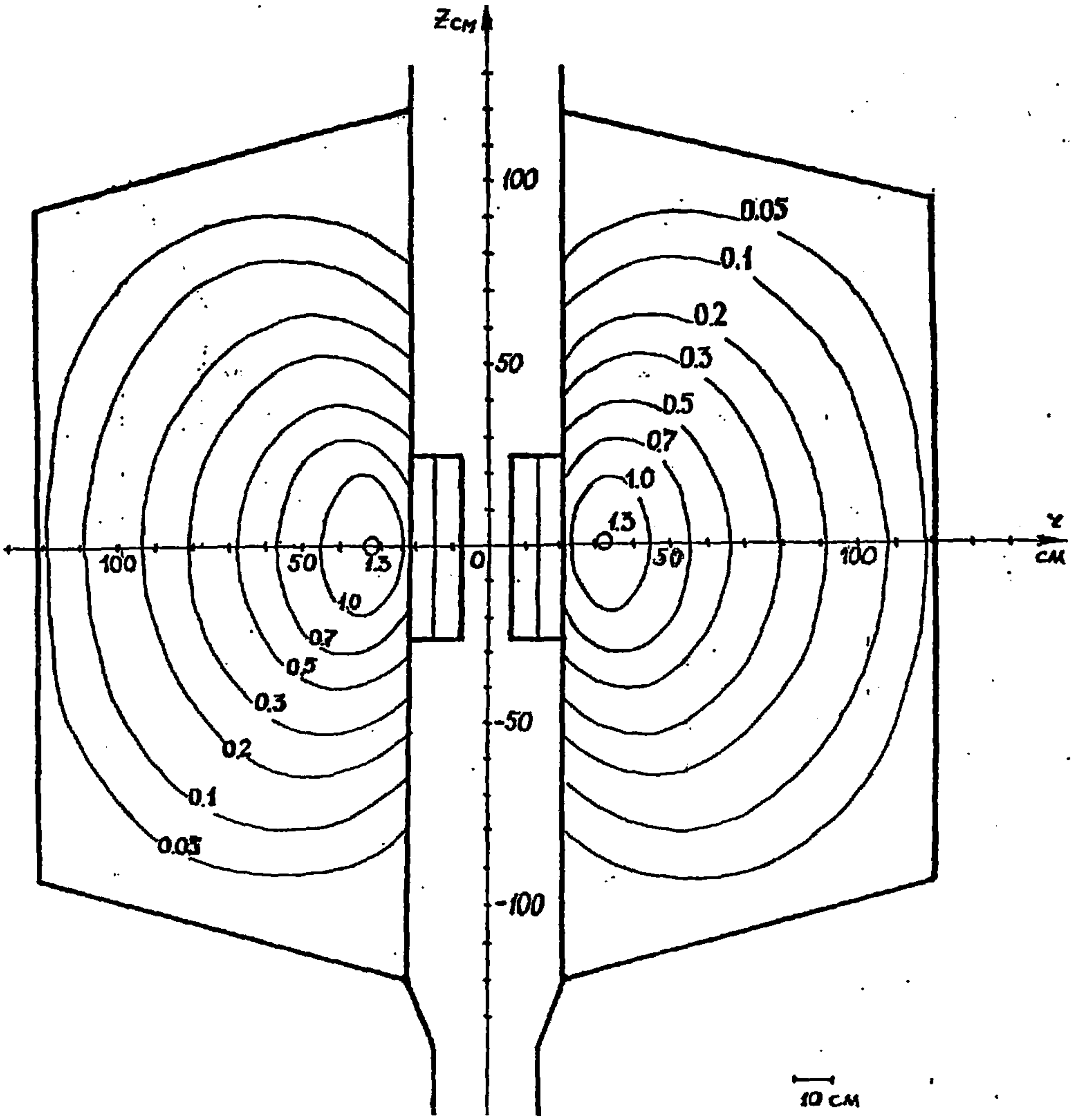


Fig.6. Thermal neutron flux distribution, 10^9 n/cm^{2}s
(P=100W)

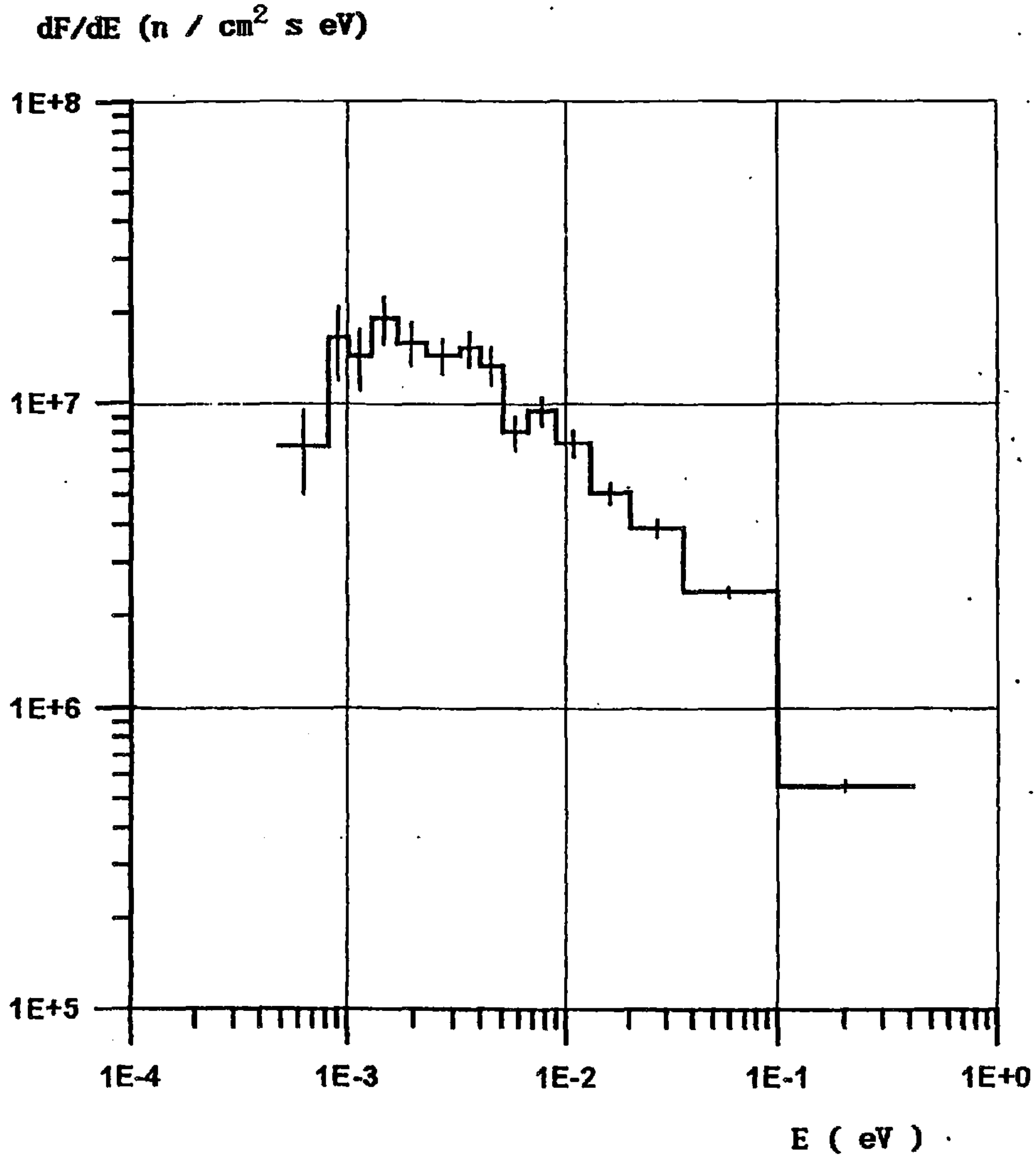


Fig.5. Neutron spectrum at exit of HEC-3
(95 cm from channel bottom,P=100 W)

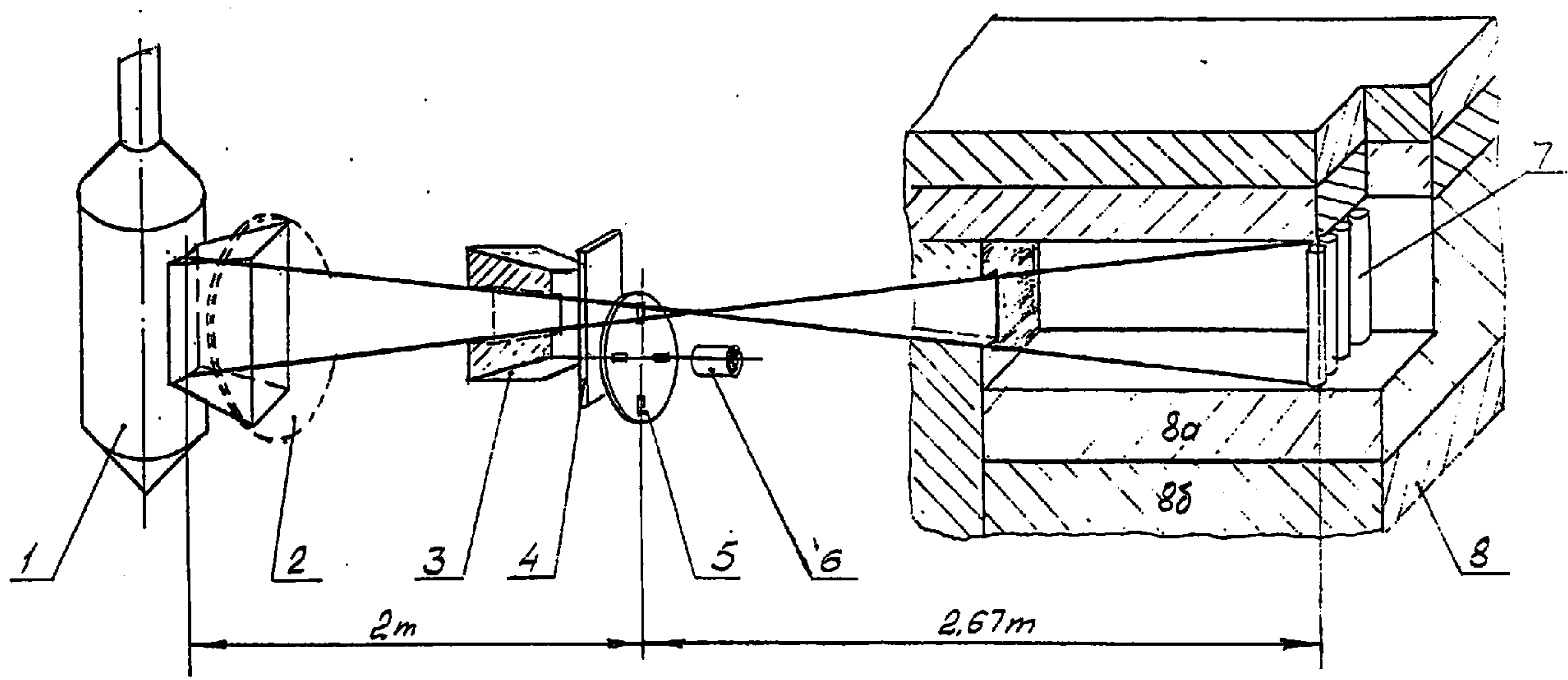


Fig.7. Neutron time- flight installation

1- CNS	5- Chopper
2- Experimental channel	6- Motor
3- Neutron collimator	7- ^{3}He-detectors
4- Chancheable filter (Pb or Be)	8- Neutron shielding

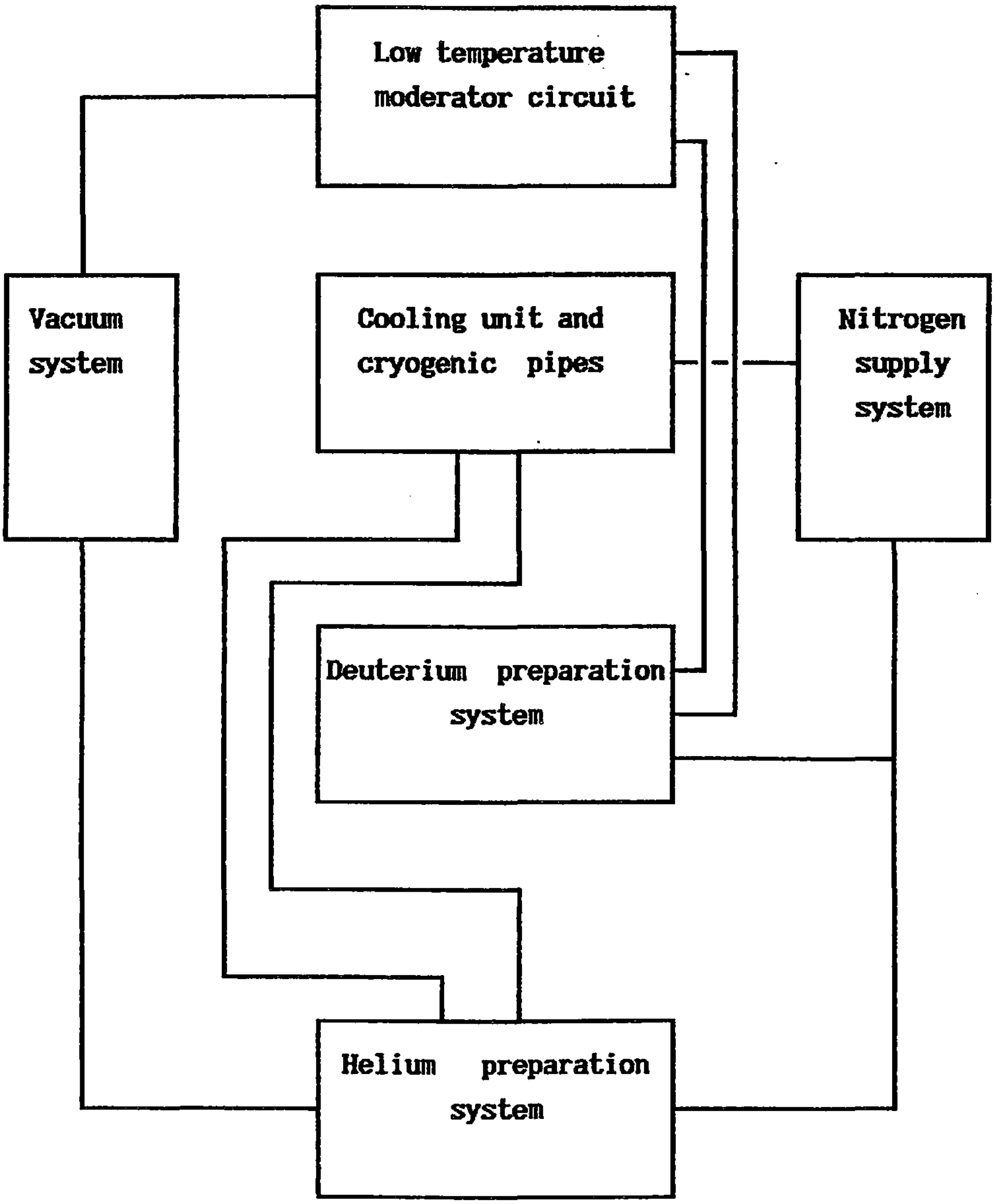


Fig.8. Diagram of cryogenic supply

Paper to be presented by K. GOBRECHT at the IGORR V Conference Nov. 4 - 6, 1996 in Aix-en-Provence :

THE COLD NEUTRON SOURCE AND OTHER IN-PILE EXPERIMENTAL FACILITIES OF THE NEW RESEARCH REACTOR FRM-II IN GARCHING

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Abstract :

The new high flux research reactor of the Technical University Munich will have 13 beam tubes and several in-pile irradiation facilities : 5 thermal beam tubes, one for medical applications (converter for fast neutrons), one for a fission product accelerator, one for a positron source, one viewing the hot neutron source (HNS) and 4 on the cold neutron source (CNS), including one for very cold and ultra-cold neutrons. The cold neutron beam-tubes will house all together 10 neutron guides, 6 of which will feed about 12 high performance neutron spectrometers in a modern neutron guide hall. Among other irradiation facilities a silicon doping loop and two rabbits will be installed. We introduce briefly these facilities and report in more detail on the CNS. The highest possible flux of low energy neutrons will be obtained by placing the CNS very close to the reactor core in the thermal neutron flux maximum. This can only be achieved if the nearly 5000 W developed by the nuclear heating in the secondary moderator at 25 K are evacuated by liquid deuterium in a thermal siphon. A 1:1 model of the thermal siphon is studied around room temperature with freon. The CNS will go into operation in the year 2000. The basic design, as well as heat load and cold neutron flux calculations will be presented.

1. Introduction

A modern research reactor is a source of neutrons for both neutron scattering experiments in condensed matter research and for production of radioactive materials for many practical and medical applications. For most efficient use the thermal neutron field in the moderator around the fuel element has to be either modified by „spectral shifters" or used by special irradiation devices. For use of the neutrons outside the reactor pool beam tubes and neutron guides have to be installed in an optimum geometrical arrangements with respect to a minimum effect on the core reactivity. All these devices are installed in the immediate neighbourhood of the fuel element and are therefore called „in-pile-facilities".

The FRM-II (Forschungsreaktor München-II) near Garching is a 20 MW reactor with H_2O cooling, D_2O moderation and a relatively small fuel element with highly enriched uranium (HEU 93 % enrichment). With such an optimised design an „undisturbed" thermal neutron flux of 8×10^{14} n/cm²s, comparable to the characteristics of the best research reactors in the world, can be obtained with low power and low amount of radioactive waste. The FRM-II should become operational in the year 2002 and will then be the strongest neutron source in Germany and one of the most attractive in Europe.

The FRM-II will be equipped with a cold neutron source (CNS) /1/, a hot neutron source (HNS) /1/, a „converter" /2/ for fast fission neutrons, 10 horizontal, two inclined and one vertical beam tubes and a series of special irradiation devices, which will be presented in the following chapters. The fact of a high flux-to-power-ratio allows us to locate the cold neutron source quite near the thermal flux maximum. We therefrom expect intensities in the cold neutron beams comparable to those of the high-flux reactor at the ILL in Grenoble (typically 5×10^9 $cm^{-2}s^{-1}$) /4/ and we therefore want to present the design of the large cold neutron source in more detail in the present paper.

2. Experimental Devices in the Moderator Tank

The experimental devices in the moderator tank (2.5 m diameter and 2.5 m high) are organised in a way, that beam tubes are mostly in the horizontal plain and that secondary sources (CNS, HNS, converter) and irradiation devices are vertical with operation from the upper reactor floor. A horizontal cut through the reactor pool is shown in Fig.1. Beam tubes #1, #2 and #4 are looking onto the cold source. In front of beam tubes #5 and #7 an optional vertical cold source can be installed. The big beam tube #5 is designed for future installation of a super-thermal source for ultra-cold neutrons, where voluminous shielding of a liquid helium vessel against radiation from the core have to be installed in the beam tube. Beam tube #9 looks onto the hot source. Beam tube #6 is a double-ended hole, dedicated to experiments with charged fission products. These are produced in a U-235 target (about 1 gram) in the middle of the tube and can be extracted and accelerated outside the reactor hall for heavy ion reactions. Beam tube #1 will take up the principal system of guide tubes going into the neutron guide hall. Beam tube #10 is oriented to the converter, from which a beam of unmoderated fission neutrons can be extracted for medical tumour therapy and for non-destructive material testing by computer tomography with fast neutrons. The large diameter of all beam holes allows for the extraction of two beams with a maximum angle of 8° in between or the installation of additional guide tubes or a horizontal cold source. A vertical neutron guide will be installed into the D_2 tube of the cold source and deliver very cold neutrons into an ultra cold neutron turbine. The two inclined beam tubes are in discussion to be used for neutron scattering, production of a positron beam and an ultra-fast pneumatic irradiation device.

The hot source is a cylindrical graphite block 300 mm in diameter and 300 mm high, which is heated up to about 2200 °C by nuclear radiation from the core, thermally insulated by carbon felt and contained in a zircaloy vessel. The energy range of the neutrons from the HNS extends from about 50 meV to 1 eV. The thermal-to-fast neutron converter is a 2 mm thick U-235 plate of 142 mm x 220 mm in a thermal flux of about 1 x 10^{14} $cm^{-2}s^{-1}$ providing a fast and intermediate (energy > 1 keV) neutron beam of 20 x 28 cm^2 area outside the reactor with an intensity of 2 x 10^9 $cm^{-2}s^{-1}$.

Several irradiation devices will be installed in the FRM-II. For very homogeneous phosphor doping of semiconductor silicon two vertical tubes for maximum 4" and 8" diameter crystals are installed in a thermal neutron flux of about 1 x 10^{14} $cm^{-2}s^{-1}$. Three pneumatic and one hydraulic rabbit systems for irradiation times between seconds and weeks in thermal fluxes between 5 x 10^{12} $cm^{-2}s^{-1}$ and 4 x 10^{14} $cm^{-2}s^{-1}$ of samples between 1 cm^3 and 12 cm^3 cover most of the demands of isotope production and activation analysis.

3. The Cold Neutron Source

The FRM-II will be equipped with two CNS, a vertical one right from the beginning, the other one later, and may be horizontal. The vertical CNS will work with boiling **deuterium** (D_2) at 25 K and 150 kPa, or with a **mixture** of deuterium and hydrogen (<10%) at the same pressure. The typical lay-out is shown in Fig.2. The moderator fluid serves itself as the heat transfer medium, taking the heat away from the moderator volume to a heat exchanger in a two-phase flow driven by natural convection. The cold moderator vessel is a 300 mm diam. cylinder with elliptical bottoms, made of aluminium. It contains - for neutron-optical reasons - a displacer, which fills itself with deuterium vapour and forms a re-entrant hole in the liquid. The liquid deuterium content is about 16 litres. The axis of the vessel is 400 mm away from the reactor core axis, and the closest point of the vessel lies only 110 mm away from the core. The heat load from nuclear radiation emanating from the core onto the vessel and from neutron activation is estimated to 5 kW (250 W per MW of reactor power), leading to a D_2-evaporation rate of nearly 20 g/s.

More **characteristic data** of the planned CNS can be found in Tab. 1.

In order to limit the heat load on the plumbing, a single, "heat-pipe"-like transfer tube links the moderator vessel to the heat-exchanger condenser 4 m above. The tube is inclined by about 10°. The condenser is cooled by a 6 kW Brayton cycle helium refrigerator with a 400 kW screw compressor. The cold box is equipped with heat exchangers for a second CNS of about 3 kW at 24 K and the plumbing for an extra turbo-expander. Together with the second CNS a second screw compressor will be installed in the compressor building.

4. The High Temperature Model

The **single tube heat-pipe** is a novelty in CNS design : the liquid deuterium flows downward along the wall of a tube from the condenser to the moderator cell, where it vaporises and streams upwards as vapour in the free section of the same tube. In order to optimise the flow conditions we built a 1:1 scale model working above room temperature with freon R11, a fluid for which the relevant hydromechanical properties can be adjusted to closely match those of D_2 (similarity criteria). The experiments take place at the same density ratio $\rho_{liquid}/\rho_{vapour}$ and at the same Wallis-number, which is the relevant scaling number for our case. It is known that the counter current flow in a pipe has its limitations (CCFL = counter-current-flow limitation), also called **flooding** in the literature /3/. In the CNS single tube heat pipe the flooding takes place above a critical heat load, which depends on the geometry of the CNS. Our experiments have so far not reached the CCFL, which means that the geometrical form of the thermal siphon is satisfactory.

5. The Gas Handling

The deuterium is completely sealed in a double-walled stainless steel circuit. When it is not liquid, it will be stored as a **metal-deuteride** in two tanks containing about 400 kg of a special hydride-forming alloy (e.g. $LaCo_3Ni_2$). Then the residual pressure in the vessel and in the plumbing is <10 Pa. For the condensation of the D_2 into the CNS the tanks have to be heated to desorb the gas from the metal-deuteride. In the neutron flux deuterium becomes radioactive by neutron capture (deuterium + neutron => tritium, and tritium => $^3He + \beta^-$). After several years of operation the storage tanks will be used to ship the radioactive metal deuteride to a reprocessing plant with-out any risk of contamination.

The out-of-pile parts of the D_2 circuit are protected from D_2-air explosion risk by an **inert gas liner** (nitrogen) pressurised to slightly above atmospheric pressure.

6. The Cold Neutron Flux

Monte-Carlo-simulations have been made with **MCNP** (version 4A) with cross-sections for LD_2 and LH_2 supplied by **IKE** Stuttgart (**Bernnat** et al. /5/). The ^{27}Al cross-section has been modified to include the gamma rays produced in the decay of Al-28 after n-capture. The simulations give the spectral distribution of the neutron flux, the brilliance of the CNS, and the heat load on the CNS. The calculations show that the liquid D_2 vessel, as designed, is too small to give the optimum moderation for wavelengths above 4 Å. By adding a few percent of **hydrogen** to the D_2 one can adapt the neutron mean free path to the small vessel dimensions (**Ageron** /1/). A neutron flux increase of 30 % from 6 to 20 Å is expected (Fig.4).

8. References

/1/ Ageron, P. "Special Neutron Sources" in Symp. on **Neutron Scattering in the Nineties** IAEA-Jülich 1985

/2/ Wagner, F.M., et al. in **Nuclear Science and Engineering, 110,** 32-37 (1992)

/3/ Böning, K. "FRM-II, Germany's new neutron source" in **nuclear engineering** dec. 1994

/4/ Gobrecht, K. "The ILL Cold Sources" in Proc. Int. **Workshop on Cold Neutron Sources,** Los Alamos 1990 (Conference Report LA-12146-C)

/5/ Bernnat, W. et al. "Evaluation and Validation of Neutron Cross-Sections for Liquid Hydrogen and Deuterium..." same **Workshop** as in /4/

Tab.1

FRM-II Cold Neutron Source : essential characteristic data

	FRM-II	ILL vertical CNS (when different)	units
Nominal reactor power	20	57	MW
Integral neutron flux in CNS	$7 \cdot 10^{14}$		$cm^{-2}s^{-1}$
Diameter of the moderator vessel	300	360	mm
Mean wall thickness	1,75		mm
Volume of the moderator vessel	20	24	litre
Mass of D_2 in the moderator vessel	2,1	3	kg
Distance from core (axis to axis)	400	760	mm
Temperature of the cold moderator	25		K
Pressure in the cold moderator	150		kPa
Pressure in the warm D_2-system	~0	300	kPa
Expected refrigeration power	5.5		kW
Hydride forming time (for 95 % D_2)	6	n.a.	min
Volume of the buffer	10	18	m^3
Number of tubes in the thermal siphon	1	3	
Material of the in-pile vacuum thimble	Zircaloy		
Mean wall thickness	4	6	mm
Vertical beam tubes for VCN/UCN	1		
Horizontal beam tubes	3	1	
Horizontal cold guides or collimators in-pile	10	5	

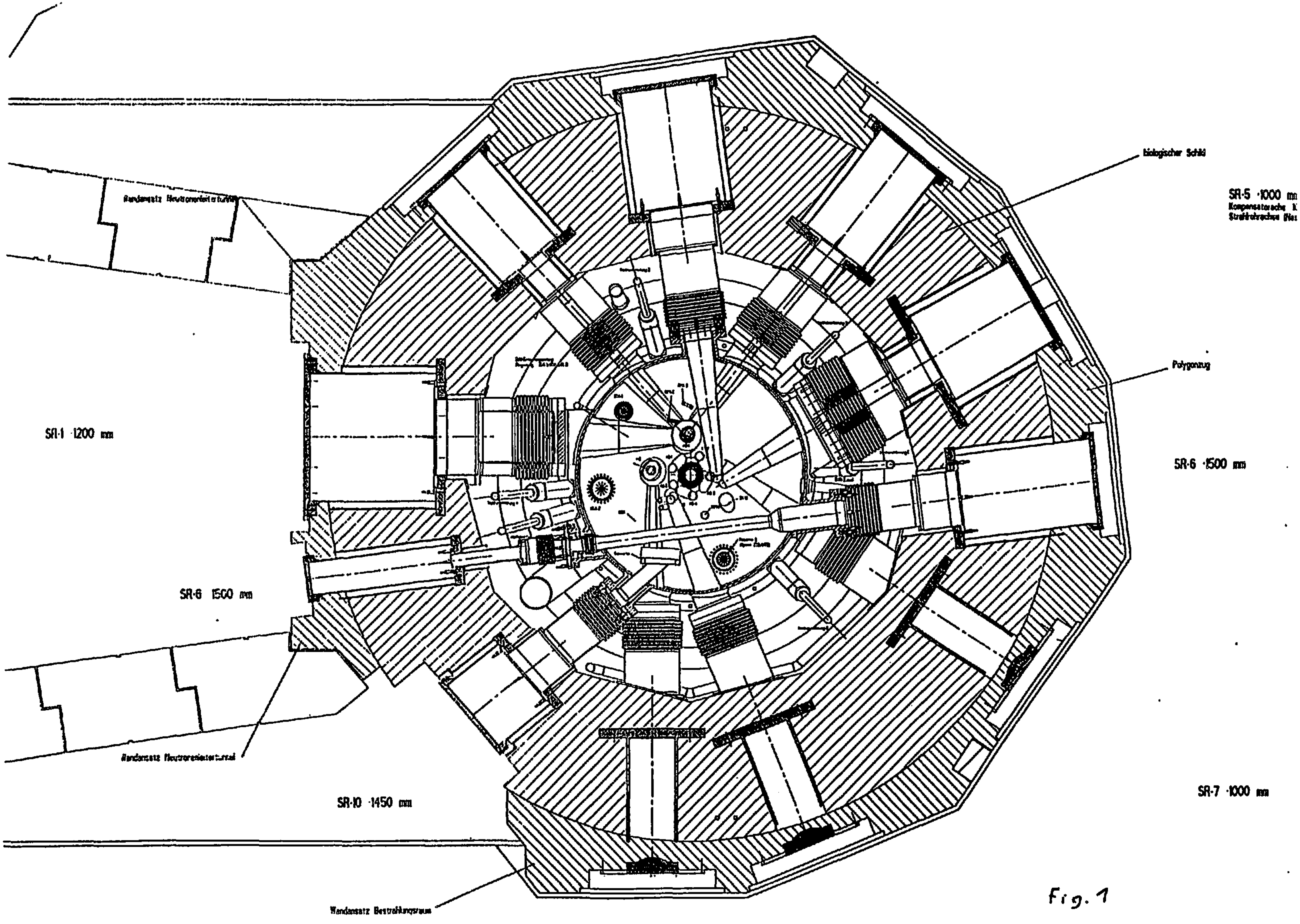

biologischer Schild
SR-5 ·1000 mm
Polygonzug
SR-1 ·1200 mm
SR-6 ·1500 mm
SR-6 1500 mm
SR-10 ·1450 mm
SR-7 ·1000 mm
Wandansatz Bestrahlungsraum

Fig. 1

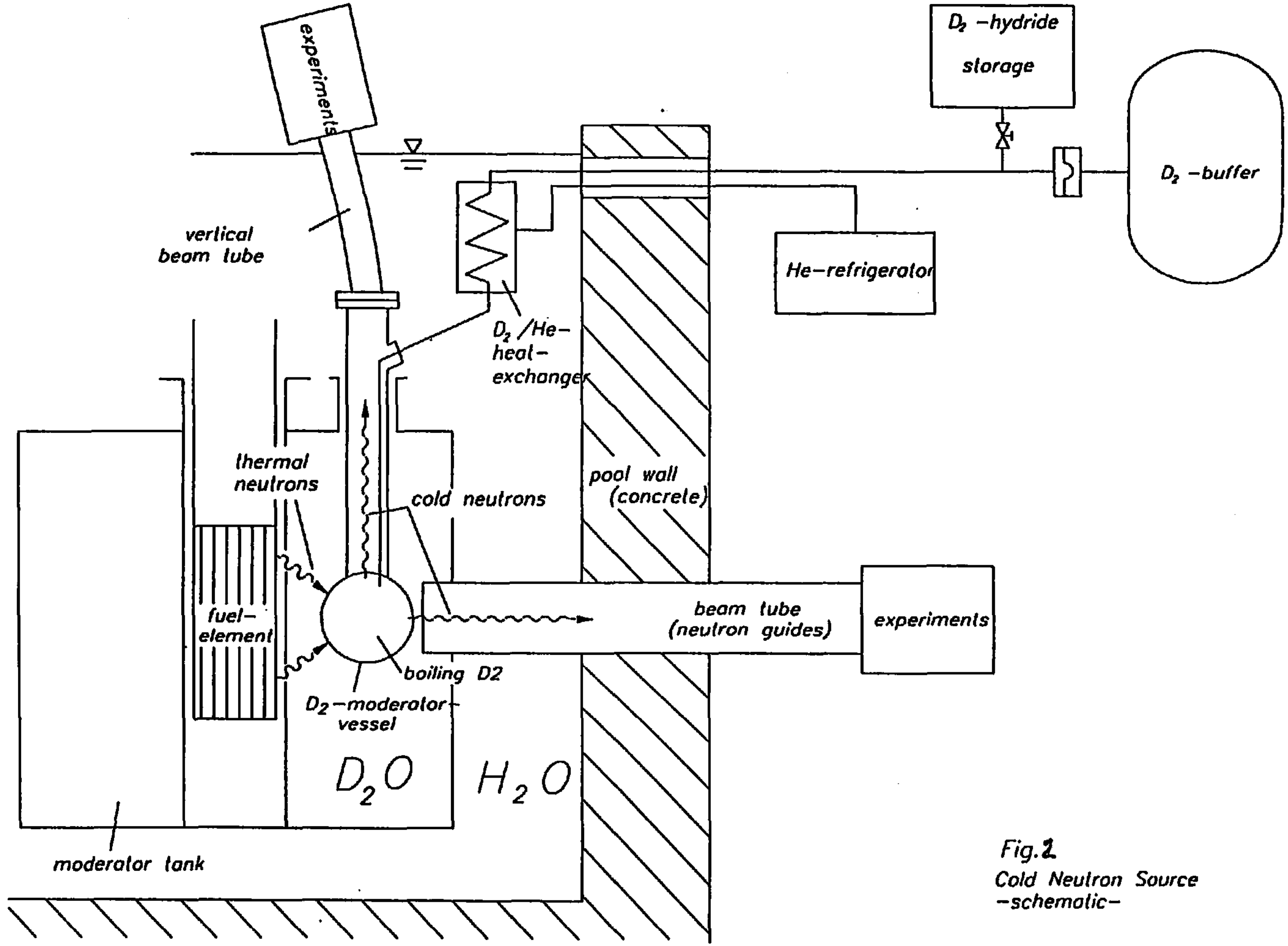


Fig. 1
Cold Neutron Source
-schematic-

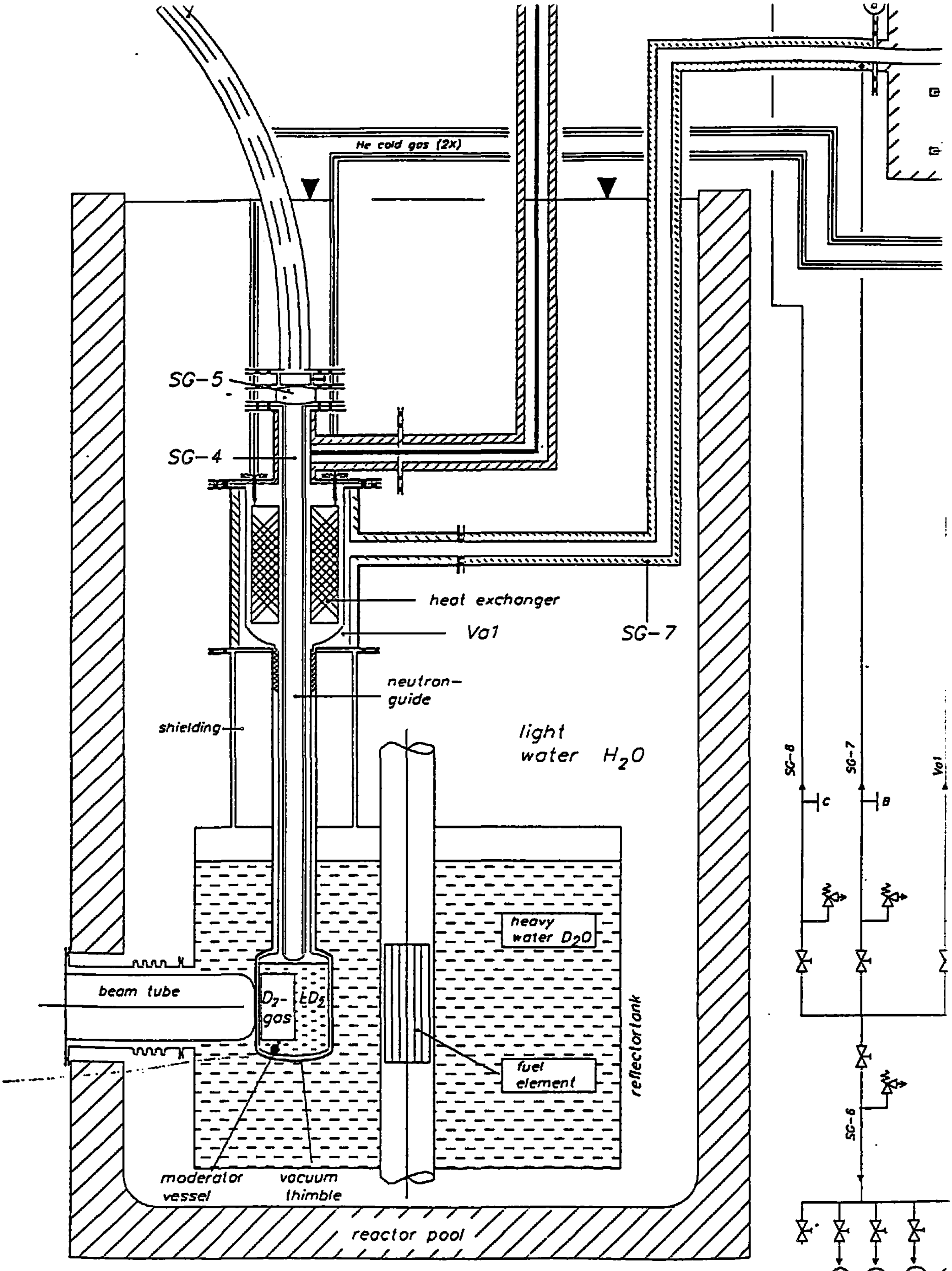
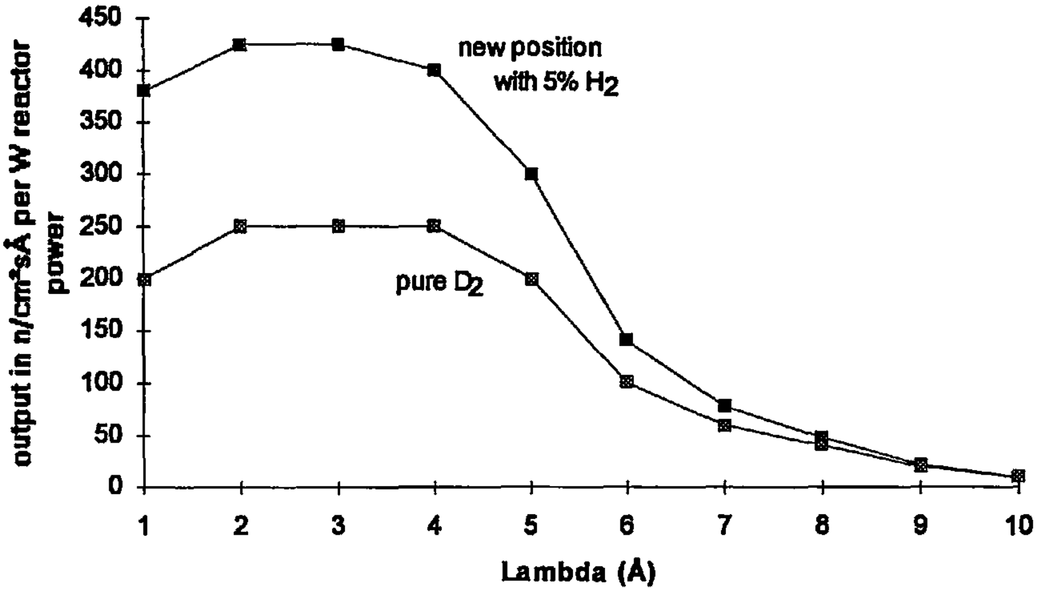

He cold gas (2X)
SG-5
SG-4
heat exchanger
Va1
SG-7
neutron-guide
shielding
light water H_2O
heavy water D_2O
beam tube
D_2-gas
LD_2
reflectortank
fuel element
moderator vessel
vacuum thimble
reactor pool
SG-8
SG-7
Va1
C
B
SG-6
1
2
3

Fig. 3

Fig. 4


output in n/cm²sÅ per W reactor power
450
400
350
300
250
200
150
100
50
0
1
2
3
4
5
6
7
8
9
10
Lambda (Å)
new position with 5% H_2
pure D_2

POST IRRADIATION EXAMINATION OF Z6 NCT 25 STAINLESS STEEL FROM SF2 COLD SOURCE CELL OF THE REACTOR ORPHEE

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ABSTRACT

At the time of the design, by the end of the seventies, the ORPHEE reactor was a second generation source type, devoted only to neutron beam use for fundamental and applied research. For this purpose, it has been equipped with a set of up-to-date experimental such as hot and cold sources and cold neutron guides.

In the original design of the reactor, the choice of liquid hydrogen for the moderator of cold sources has led two use two « flat » cells in order to illuminate the beam tubes looking at the cold sources called SF1 and SF2.

The cells were made of stainless steel in order to offer the possibility of staying in the reactor, under neutrons irradiation, without cooling by the moderator.

As the behavior of the stainless steel cells was bad known, a surveillance programm of the cells has been proposed in 1981. This programm foresaw an examination of the cells. These examinations have been made in 1987 (the results have been reported in a previous IGORR meeting) and 1991. The SF2 cell has been taken out of the reactor in july 1990 and the examinations have started in 1991.

The post irradiation examination includes helium measurements, (on each face of the cell) and correlation with the neutron flux estimation.

Tensile tests have been performed following a programm of 20 samples at various temperature. Comparaison with the result of SF1 cell has been made.

1. INTRODUCTION

The ORPHEE reactor is a medium flux neutron source for fundamental and applied research using neutron beams. In the design, special features of a modern reactor like the High Flux Reactor in Grenoble were used : pool type reactor block of compact size, tangential beam tubes, hod and cold sources, neutron guides[1].

The reactor has 9 horizontal beam tubes tangential to the core carrying 20 neutron beams (two beams per standard channel and four for a double channel). The front ends of each tubes are in the reflector close to the core. Three tubes are directed at two « cold sources » called SF1 and SF2 (see Figure 1).

The moderator selected for the cold sources was hydrogen. The two identical sources are located in the reflector tank, 400 mm away from the core centerline. The active part of the sources is a flask shaped cell (205 mm high, 130 mm wide, 50 mm thick at center) made of 0.8 mm stainless steel. The cell hangs from a concentric supply line (with the liquid in the inner tube). The flask-shaped cell is located within an evacuated tube (150 mm O.D., 5 mm wall thickness) made of aluminium alloy.

The cells were made of stainless steel in order to offer the possibility of staying in the reactor without cooling by the moderator. In such a case, the cell was filled with helium and the external tube of aluminium too. The design shows that the temperature of stainless steel grows up to 425°C.

As the behavior of the stainless steel cells was bad known, specially with the risk of embrittlement by hydrogen and helium formation, a surveillance programm of the cells has been proposed in 1981 when the reactor started.

This programm foresaw an examination of the cells when the quantity of helium due to (n,α) reactions would reach 1100 ppm. These examinations have been made in 1987 and the results have been reported previously. (The main results are recalled in the §2)[2].

The results have permitted to extend the life of the second cold cell (SF2) up to 1500 ppm of helium in the stainless steel. The SF2 cell has been taken out of the reactor in july 1990 and the examination have started in 1991.

2. RESULTS OF THE FIRST POST IRRADIATION EXAMINATION (1987)

Three series of 5 tensile test samples have been cut up in the cell by electro erosion. Helium measurements have shown quantity of helium quietly different on each face of the cell :

- 907 ppm for the front face (looking at the core) and
- 543 ppm for the back face (looking at the beam tube).

That difference has been explained by the disturbance in the flux of neutrons due to the beam tube 8F looking at the back face of the cell and the normal gradient of the flux in the reactor tank.

In the same range, the swelling of alloy was respectively 0,66 % and 0,41 %.
For the tensile tests, the yield strenght (at 0,2 %) remains the same that the original alloy for the samples heat treated at 425°C during 10 days.

Elongation decreases from 15 % to 9 %.

3. POST IRRADIATION EXAMINATION OF THE SF2 CELL (1992)

The stainless steel used for the flat cell was Z6 NCT 25 stainless steel from CREUSOT LOIRE.

His theoretical composition is :

- 25 % Ni
- 15 % Cr
- 1,5 % Mo
- 2 % Ti
- ~ 56,5 % iron.

His mechanical properties are :

- R > 915 MPa
- $E_{0,2}$ > 600 MPa
- A > 15 %

As shown on Figure 1 the cold cell SF2 illuminates two beam tubes (4F and 9F) placed symmetrically on each side of the flat cell.

3.1 Post irradiation examination programm

a) The post examination includes helium measurements (on each face of the cell). Tensile tests have been performed following a programm of 20 samples at various temperature : 12 samples have been tested after heat treatment at 425°C during 240 h and 8 samples without heat treatement.

The tensile tests have been performed at - 170°C, 25°C, 425°C and 650°C. They have been realised with a constant speed of $3.10^{-4}\ s^{-1}$.

The detailed programm is given below.

ORPHEE COLD CELL SF2

TENSILE TEST

N° Test piece	Heat Treatment	Temperature Test	Elongation Speed
4F B5 9F B1	no no	- 170°C - 170°C	3.10^{-4}/s 3.10^{-4}/s
4F B4 9F B2	no no	25°C 25°C	3.10^{-4}/s 3.10^{-4}/s
4F B3 9F B3	no no	425°C 425°C	3.10^{-4}/s 3.10^{-4}/s
4F B2 9F B4	no no	650°C 650°C	3.10^{-4}/s
4F H1 9F H1	425°C, 240 h 425°C, 240 h	- 170°C - 170°C	3.10^{-4}/s 3.10^{-4}/s
4F H2 4F M2 9F H2 9F M3	425°C, 240 h 425°C, 240 h 425°C, 240 h 425°C, 240 h	25°C 25°C 25°C 25°C	3.10^{-4}/s 3.10^{-4}/s 3.10^{-4}/s 3.10^{-4}/s
4F H3 4F M4 9F H3 9F M2	425°C, 240 h 425°C, 240 h 425°C, 240 h 425°C, 240 h	425°C 425°C 425°C 425°C	3.10^{-4}/s 3.10^{-4}/s 3.10^{-4}/s 3.10^{-4}/s
4F H4 9F H4	650°C, 24 h 650°C, 24 h	650°C 650°C	3.10^{-4}/s 3.10^{-4}/s

3.2 SF2 Tensile tests

a) Non heat treated samples

Figures 2 to 4 show the results of tensile tests for the 4F and 9F samples without heat treatment. Measurements have been made at - 170°C, 25°C, 425°C and 650°C, for yield strenght (0,2 %), ultimate strenght and total elongation.

b) Heat treated samples

Figures 5 to 7 show the results of tensile tests for the 4F and 9F samples after heat treatment.

c) Discussion

Except for the temperature of - 170°C, the results show a good accordance between the two faces of the cell. Influence of heat treatment : on the face 4F, the yield and ultimate strenght decrease after heat treatment specially at low temperature. In the same time, total elongation increases, and remains at 5 % for the non heat treated sample at 25°C.

On the face 9F, the yield strenght decreases after heat treatment which have no effect on the ultimate strenght. Total elongation increases after heat treatment and remains at 5 % for the non heat treated sample at 25°C.

3.3 Comparison between SF1 and SF2

Figures 8 to 10 show the comparison between SF1 and SF2, for the non heat treated samples. Comparison is made for yield strenght, tensile strenght and total elongation. For SF2 the values used for the curve are obtained from the average of 4F and 9F values.

For SF1 the values used for the curve are those of the face looking at the beam tube 8F.

Discussion

The yield and ultimate strenght are equivalent except for the measure of yield strenght at 25°C. The material keeps a good ductility at low temperature (- 170°C). At 25°C, the total elongation decreases for SF2 (6 %).

4. CONCLUSION

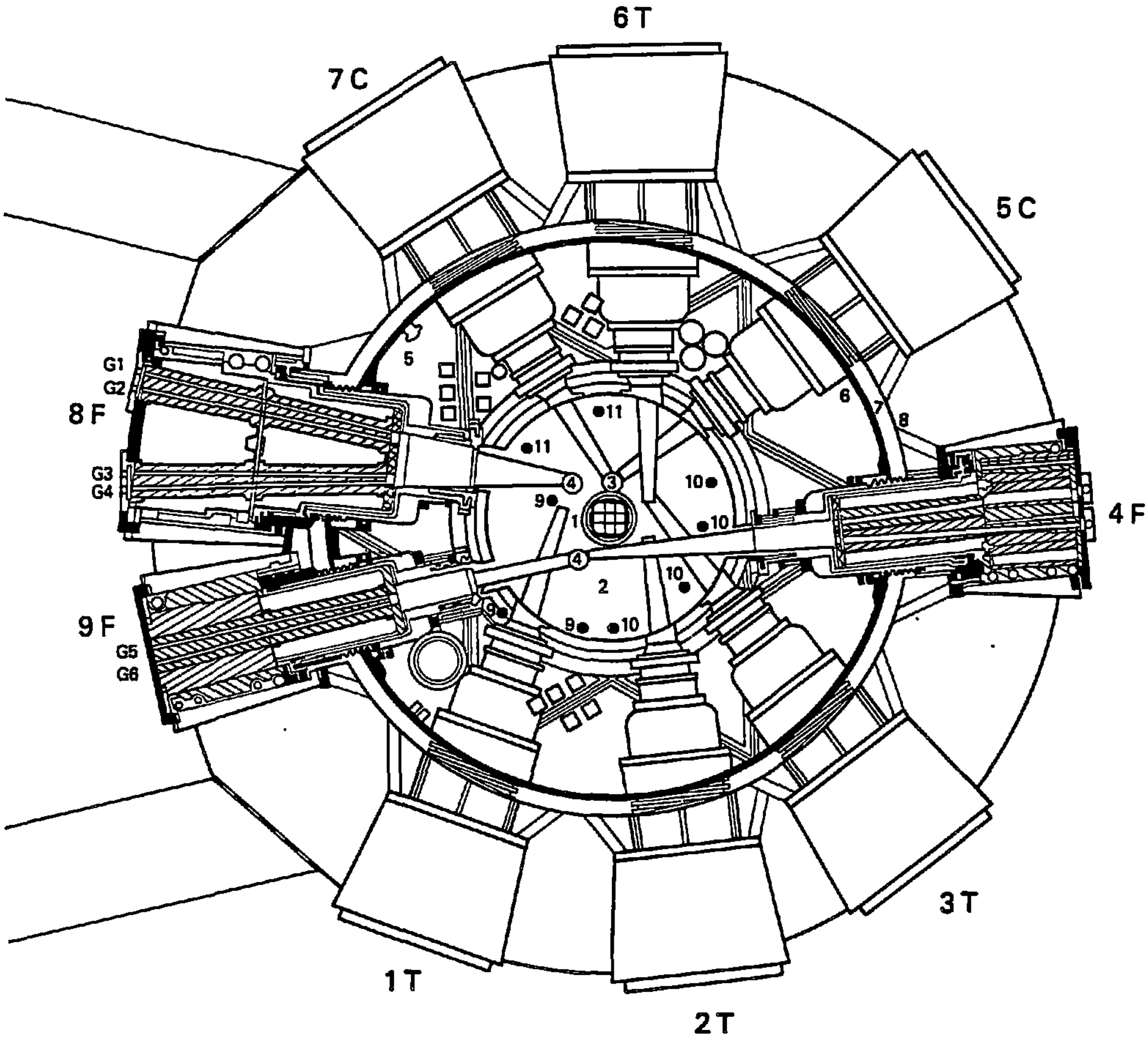
The post examination programm developped for the SF2 cell of the reactor ORPHEE after 1668 EFPD has confirmed the results obtained for SF1 after 1000 EFPD. The material keeps good mechanical properties, for all the working conditions. After heat treatment (425°C, 240 h) the embrittlement at 25°C seems to be reduced.

REFERENCES

[1] P. BREANT, « *The ORPHEE Reactor, Current Status and Proposed Enhancement of Experimental Capabilities* », IGORR I, Knoxwille (USA), Feb.28-March 2, 1990

[2] P. BREANT, B. FARNOUX, J. VERDIER, « *Cold Sources of the ORPHEE Reactor* », International Workshop and Cold Neutron Sources, Los Alamos (USA), March 5-8, 1990.

ORPHEE REACTOR



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 1

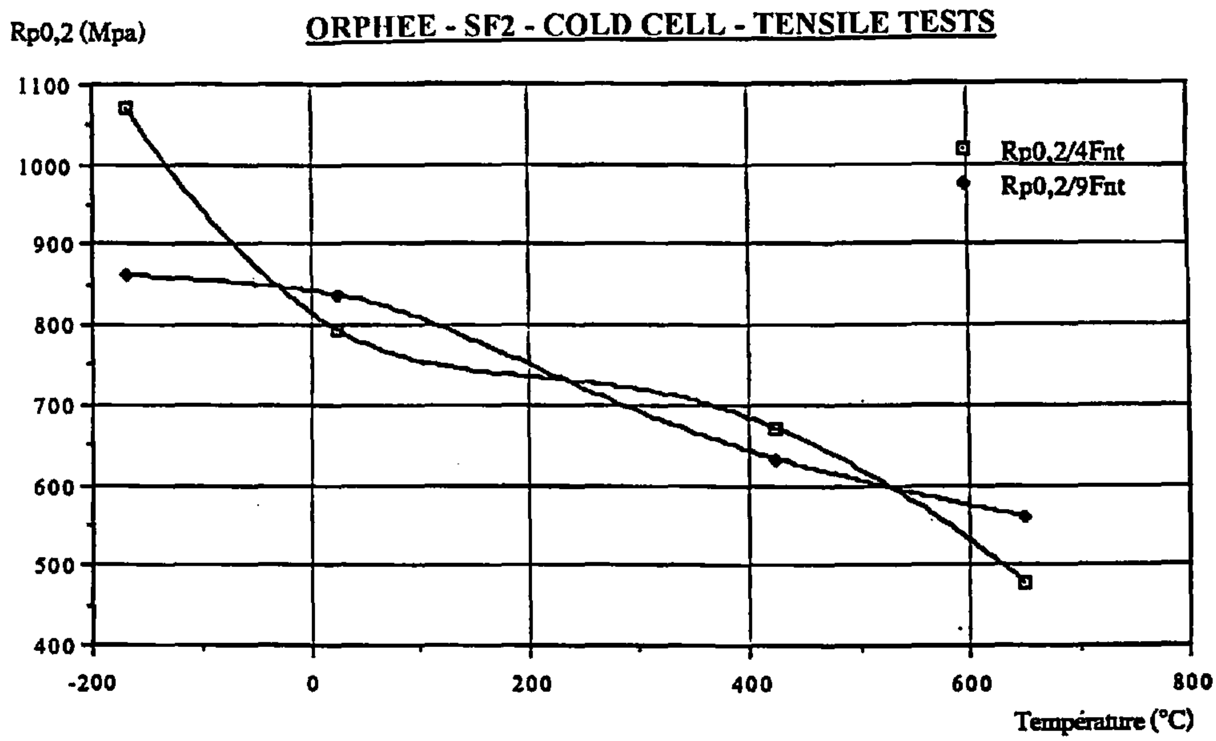


Figure 2 : Yield strenght evolution. Comparison between face 4F and 9F (non heat treated samples).

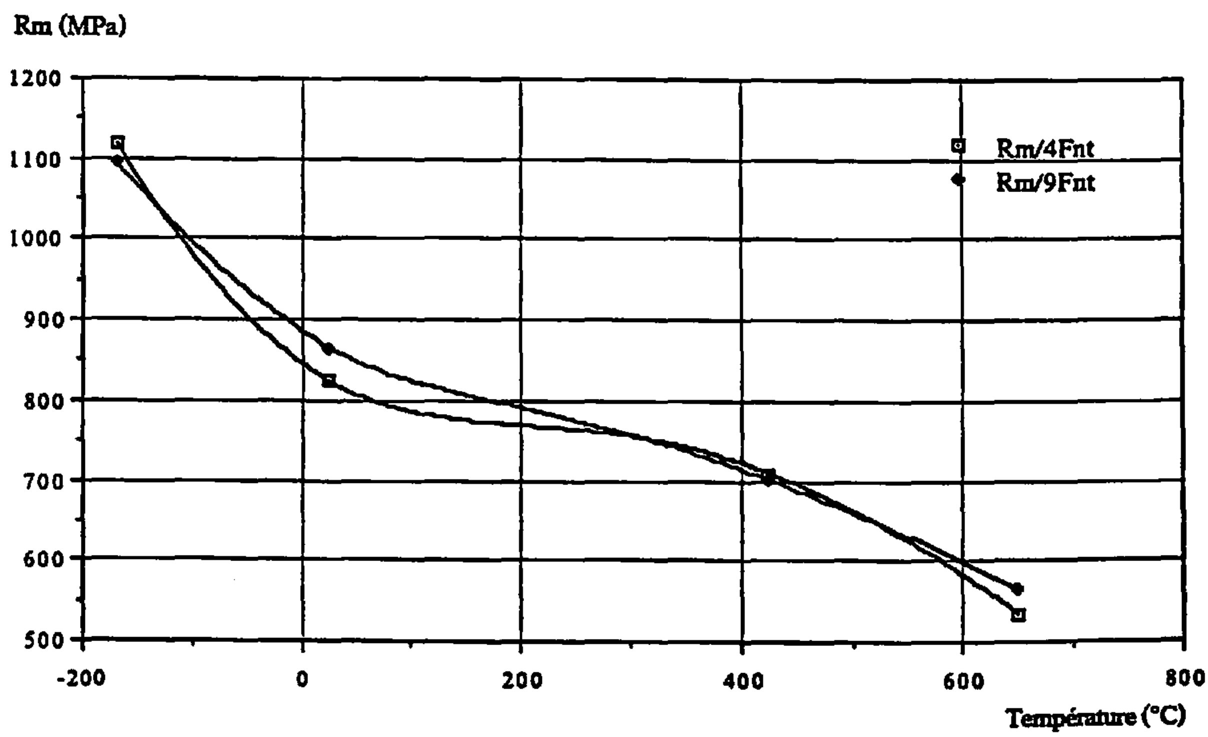


Figure 3 : Ultimate strenght evolution. Comparison between face 4F and 9F (non heat treated samples).

ORPHEE - SF2 - COLD CELL - TENSILE TESTS

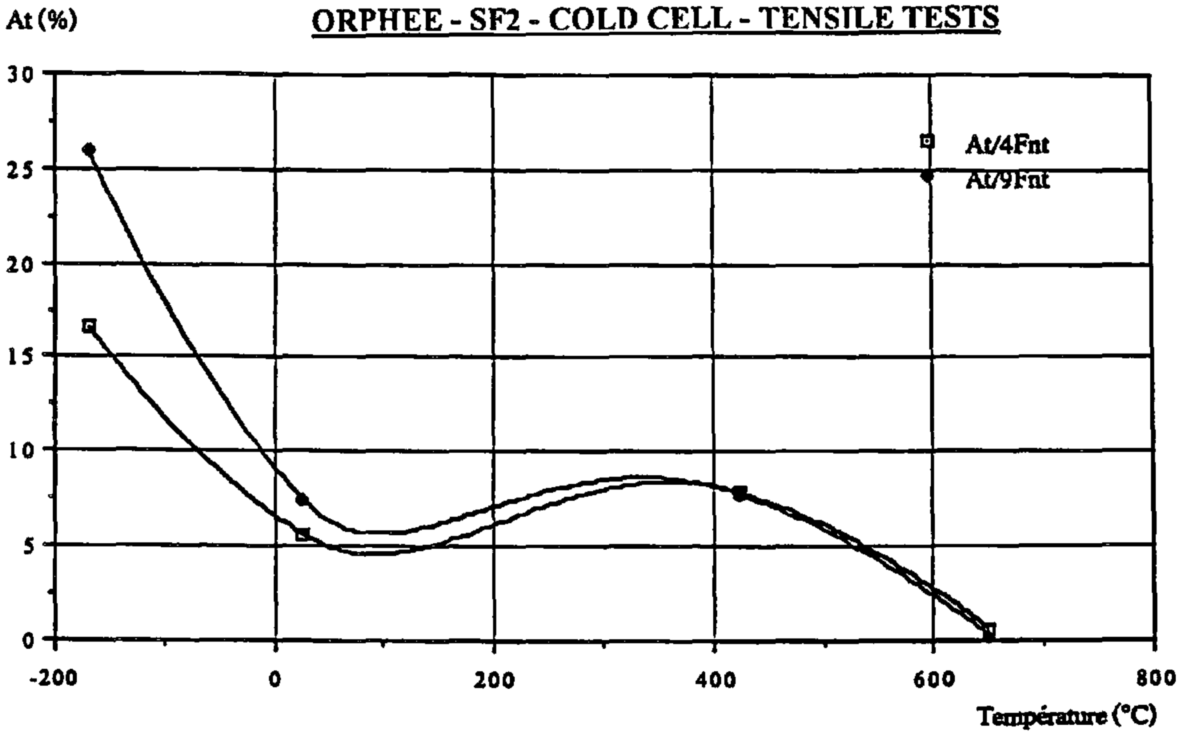


Figure 4 : Total elongation. Comparison between face 4F and 9F (non heat treated samples).

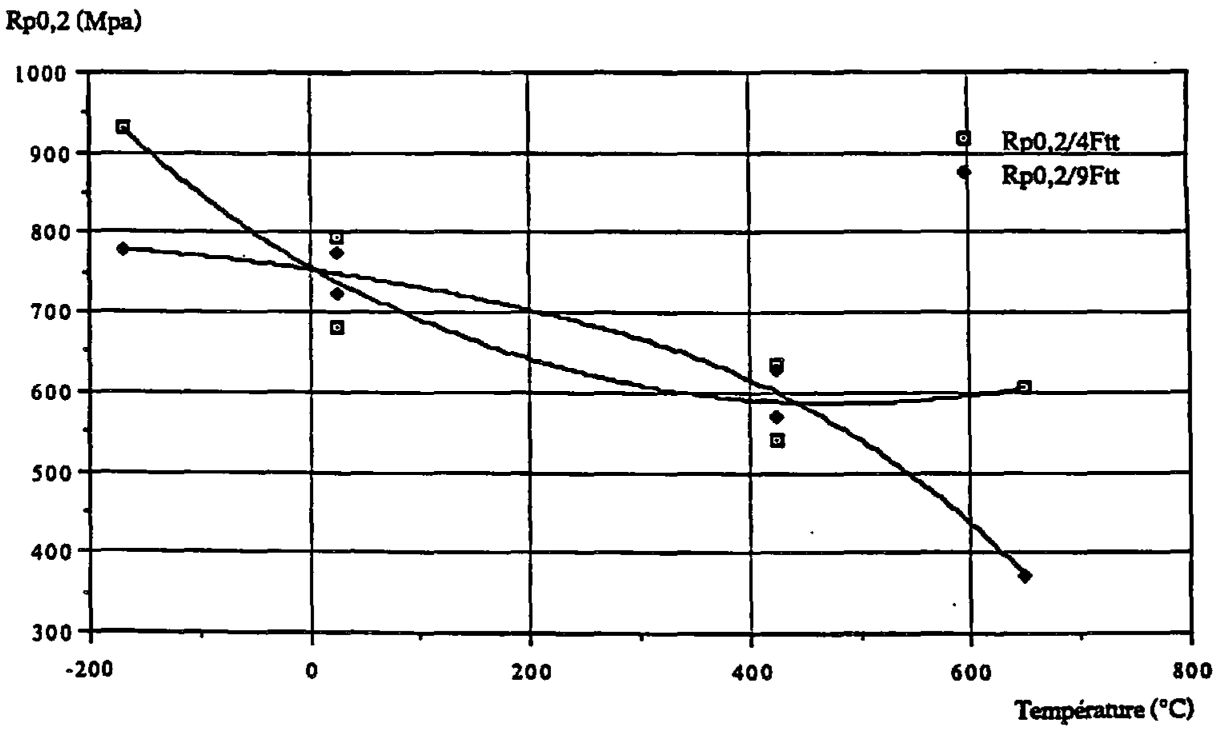


Figure 5 : Yield strenght evolution. Comparison between face 4F and 9F (heat treated samples).

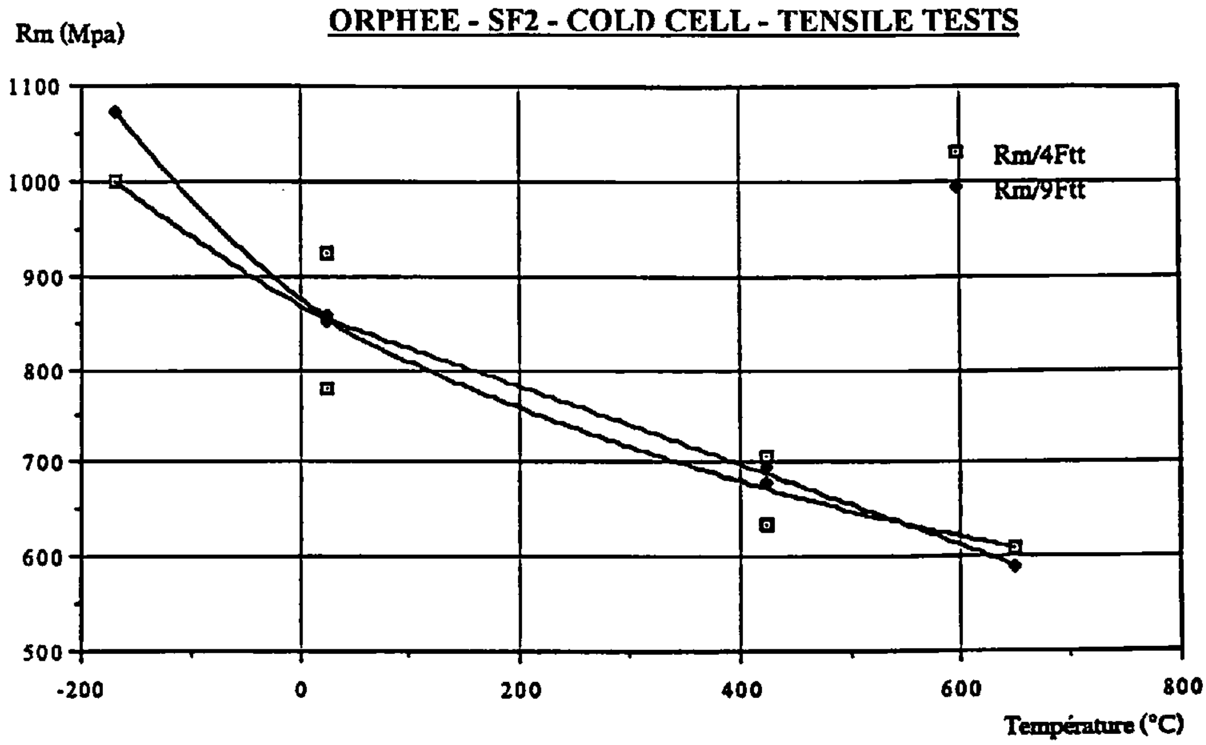


Figure 6 : Ultimate strenght. Comparison between face 4F and 9F (heat treated samples).

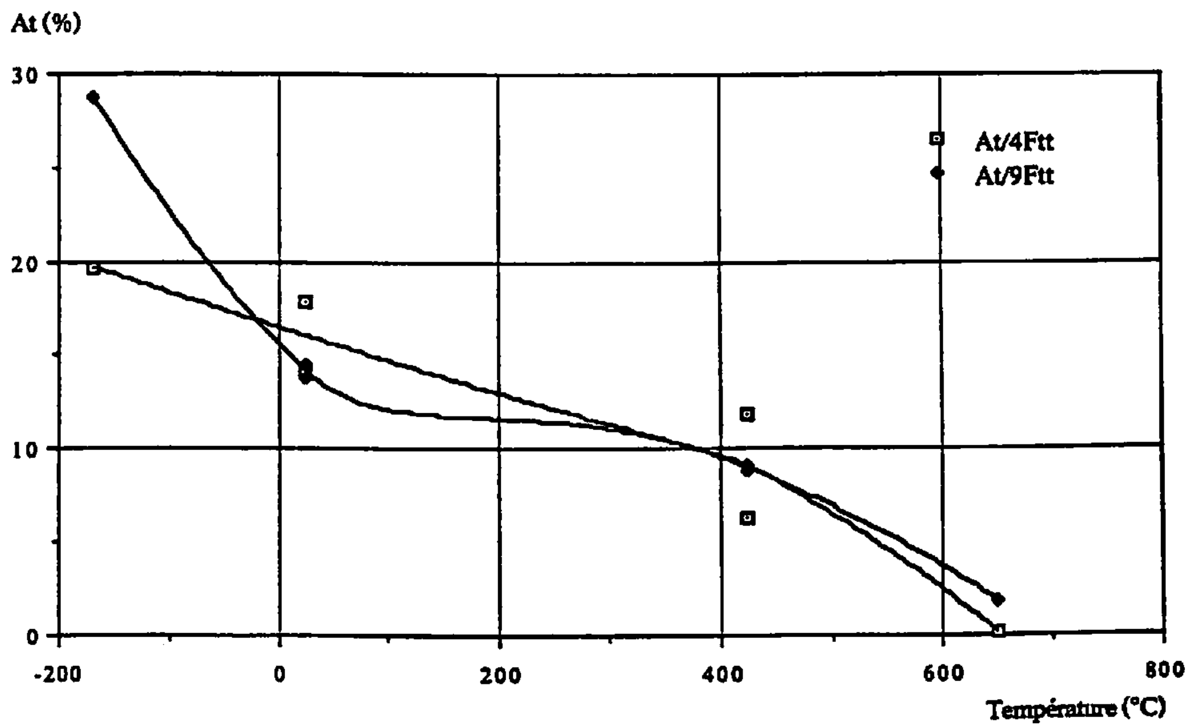


Figure 7 : Total elongation. Comparison between face 4F and 9F (heat treated samples).

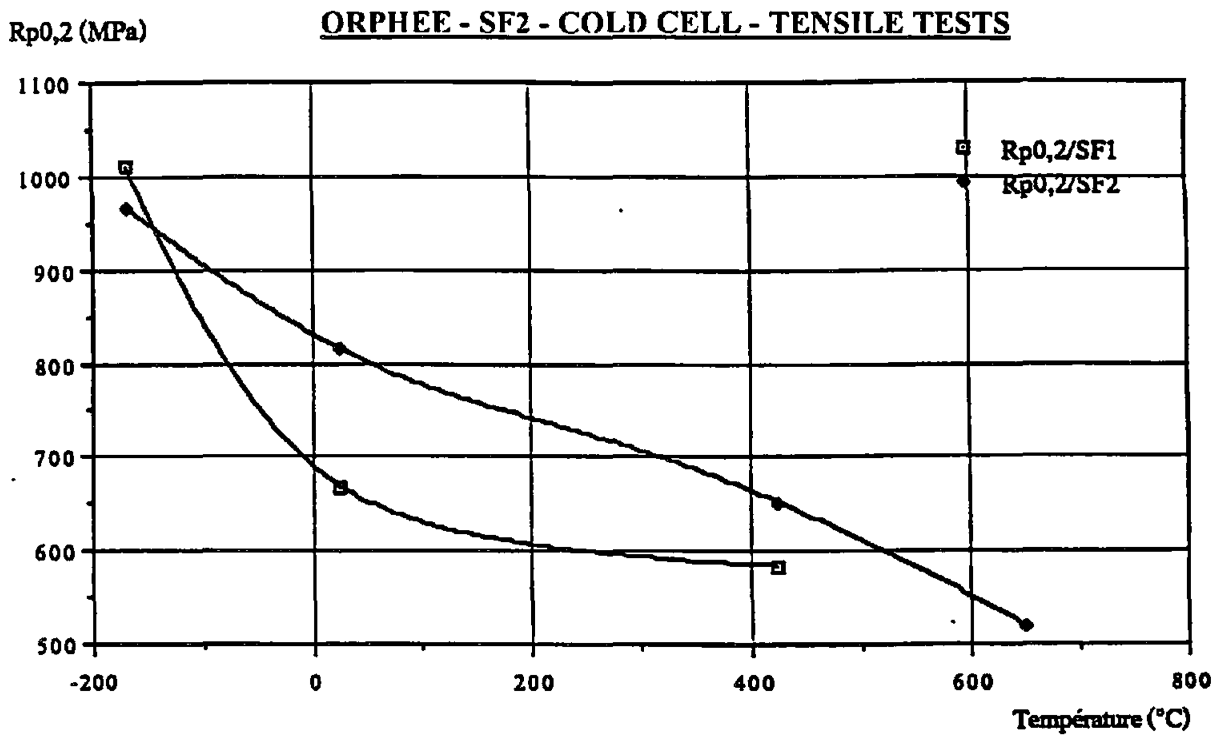


Figure 8 : Yield strenght evolution. Comparison between SF1 and SF2 (non heat treated samples).

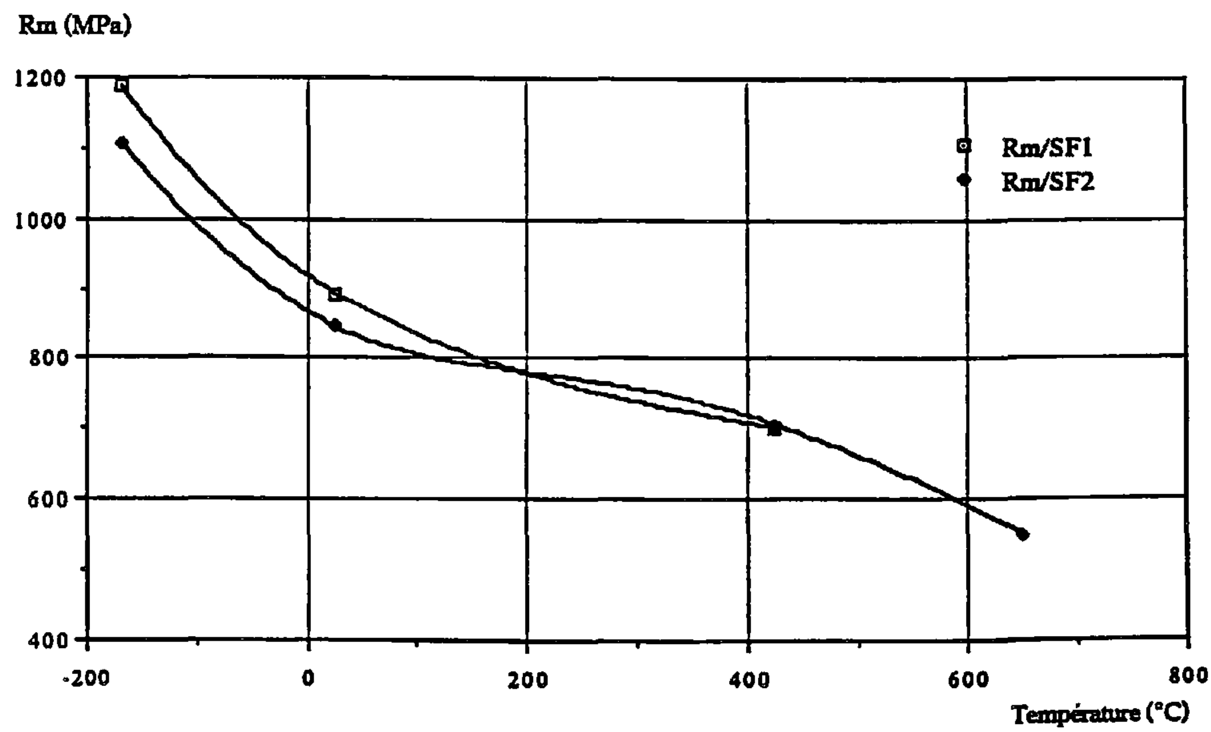


Figure 9 : Ultimate strenght evolution. Comparison between SF1 and SF2 (non heat treated samples).

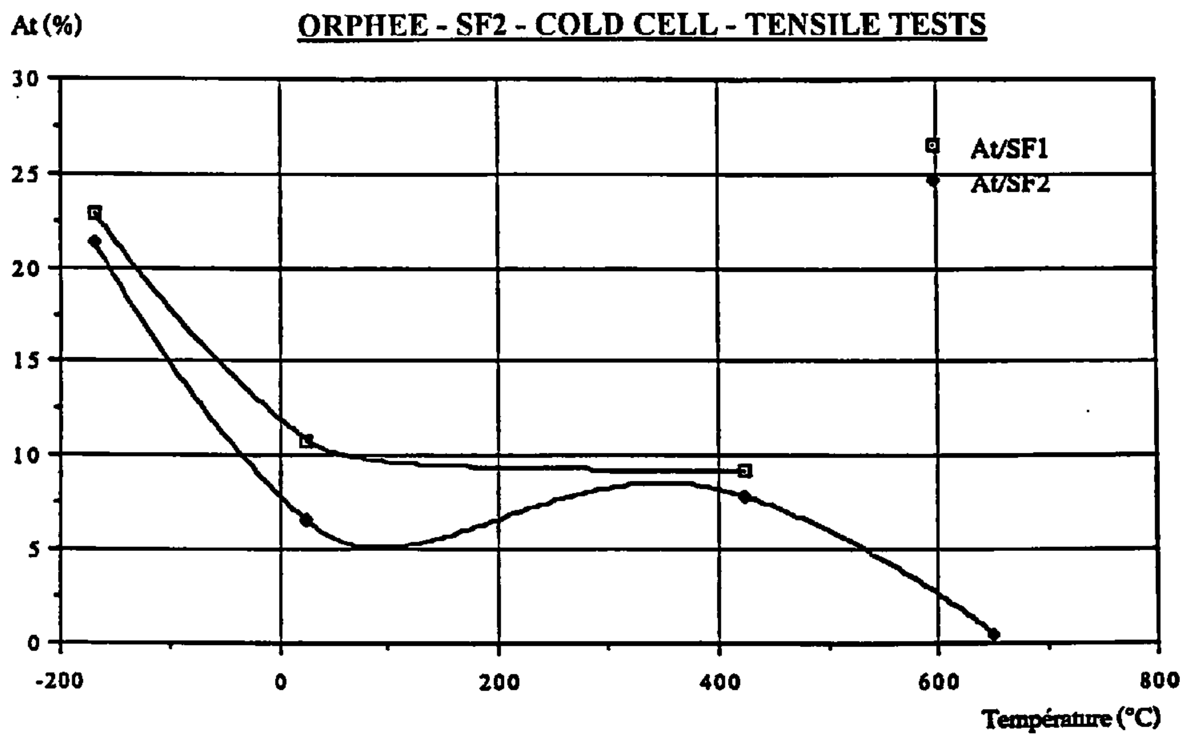


Figure 10 : Total elongation. Comparison between SF1 and SF2 (non heat treated samples).

SUMMARY OF HFIR COLD SOURCE PROJECT

presented to the 5th meeting of the
International Group on Research Reactors

D. L. Selby

November 1996

The U.S. Department of Energy (DOE) Has Initiated A Project To Add A Low Temperature (20K) Hydrogen Cold Neutron Source To The High Flux Isotope Reactor (HFIR) Facility At ORNL

- Objective is to provide a major upgrade to the HFIR neutron research capabilities

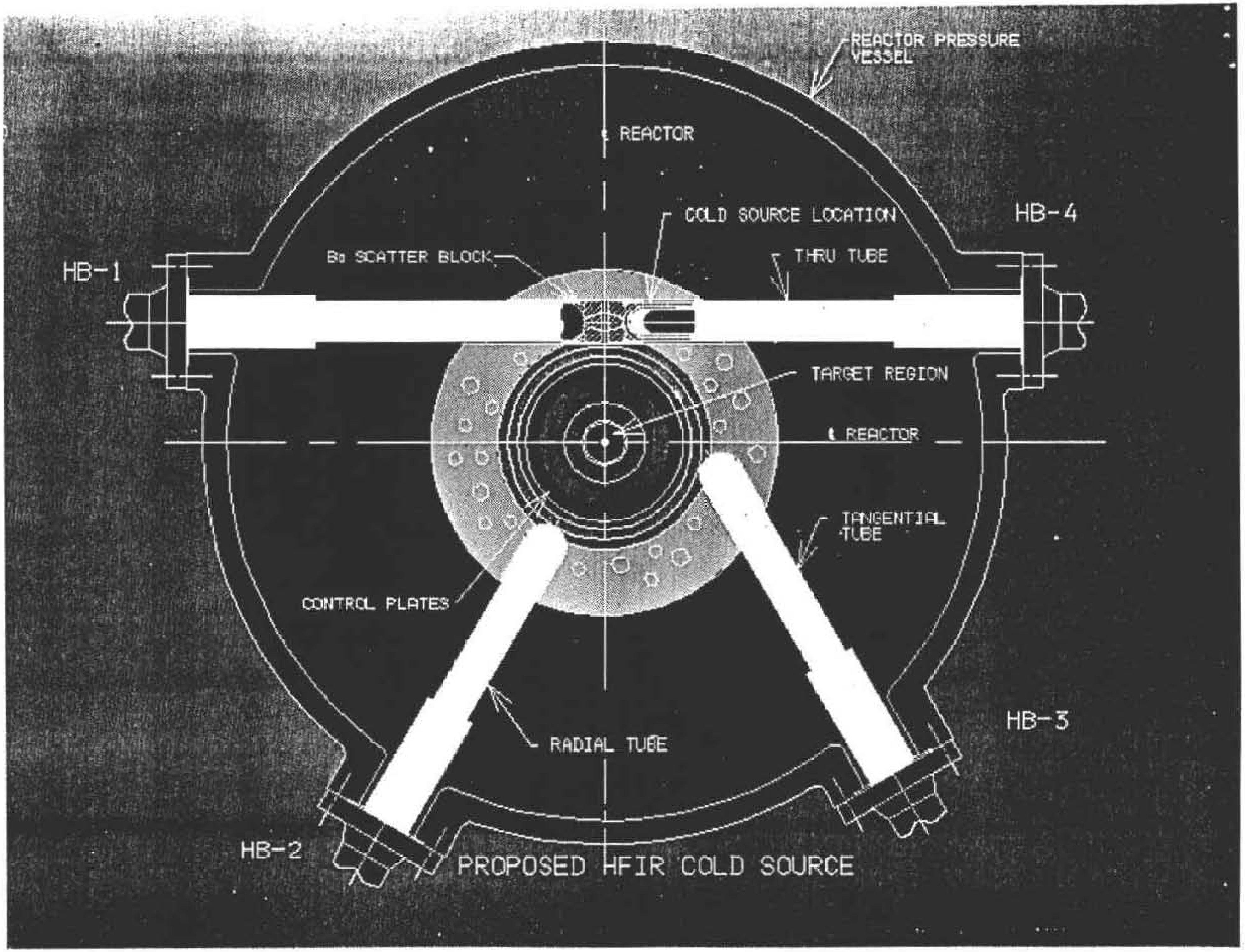
    - Brightness would be comparable to best in the world and significantly better than presently available anywhere within the United States.

- Presently transitioning from the Conceptual Design of the cold source to Detailed Design

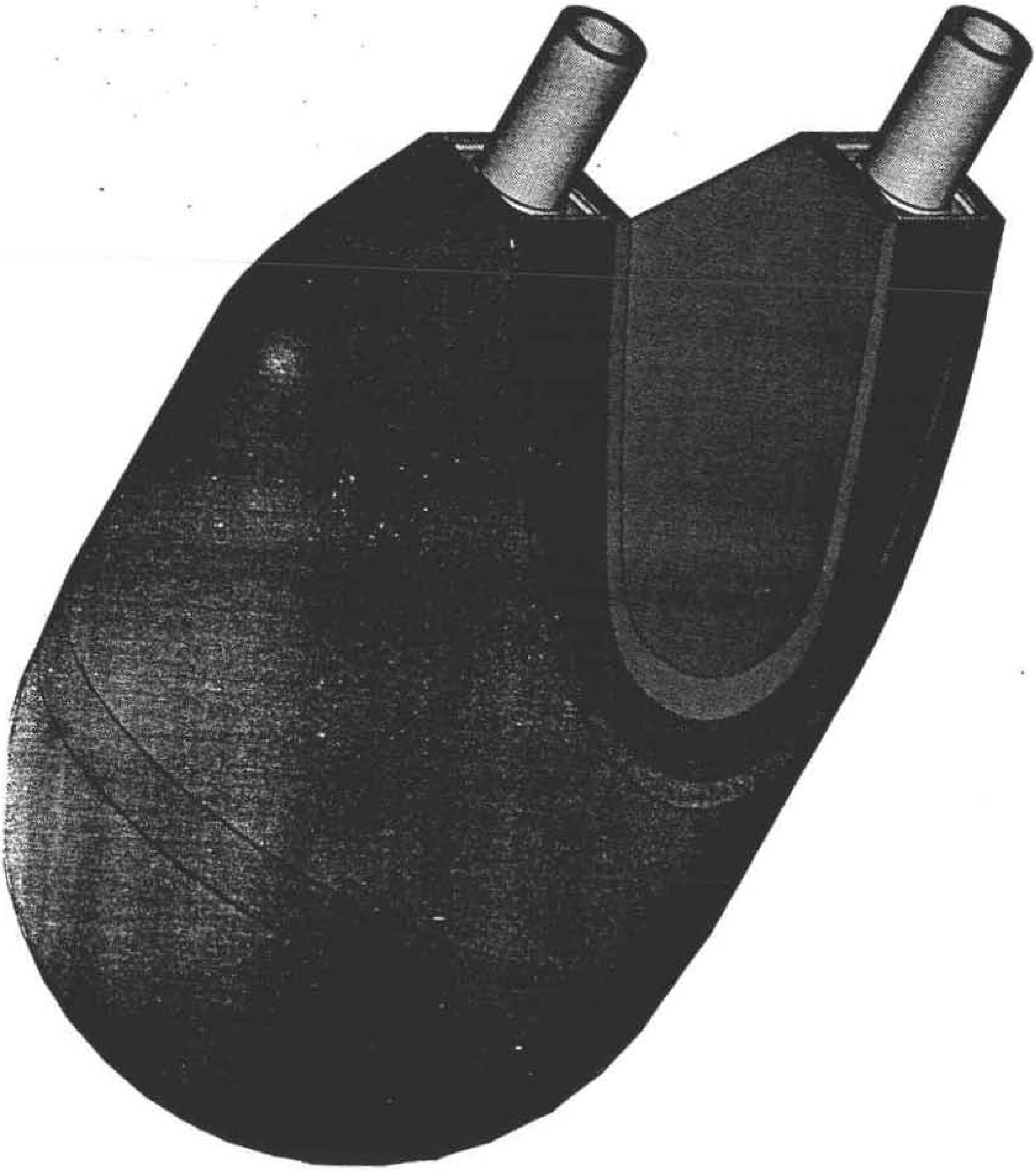
- Cold source is scheduled to be installed in the HFIR facility during an extended (6 month) planned reactor shutdown for beryllium reflector changeout in 1999.

KEY COLD SOURCE PARAMETERS

- Hydrogen moderated system operating at approximately 20-25K
- Located near the tip of the HB-4 beam tube
- Estimated heat load to be removed at cryogenic temperatures by refrigerator is approximately 2.3 KW
- Hydrogen is pumped around the circuit to accommodate the high heat load and the long horizontal run along the HB-4 beam tube
- Estimated cost between 9 and 10 $M



PROPOSED HFIR COLD SOURCE

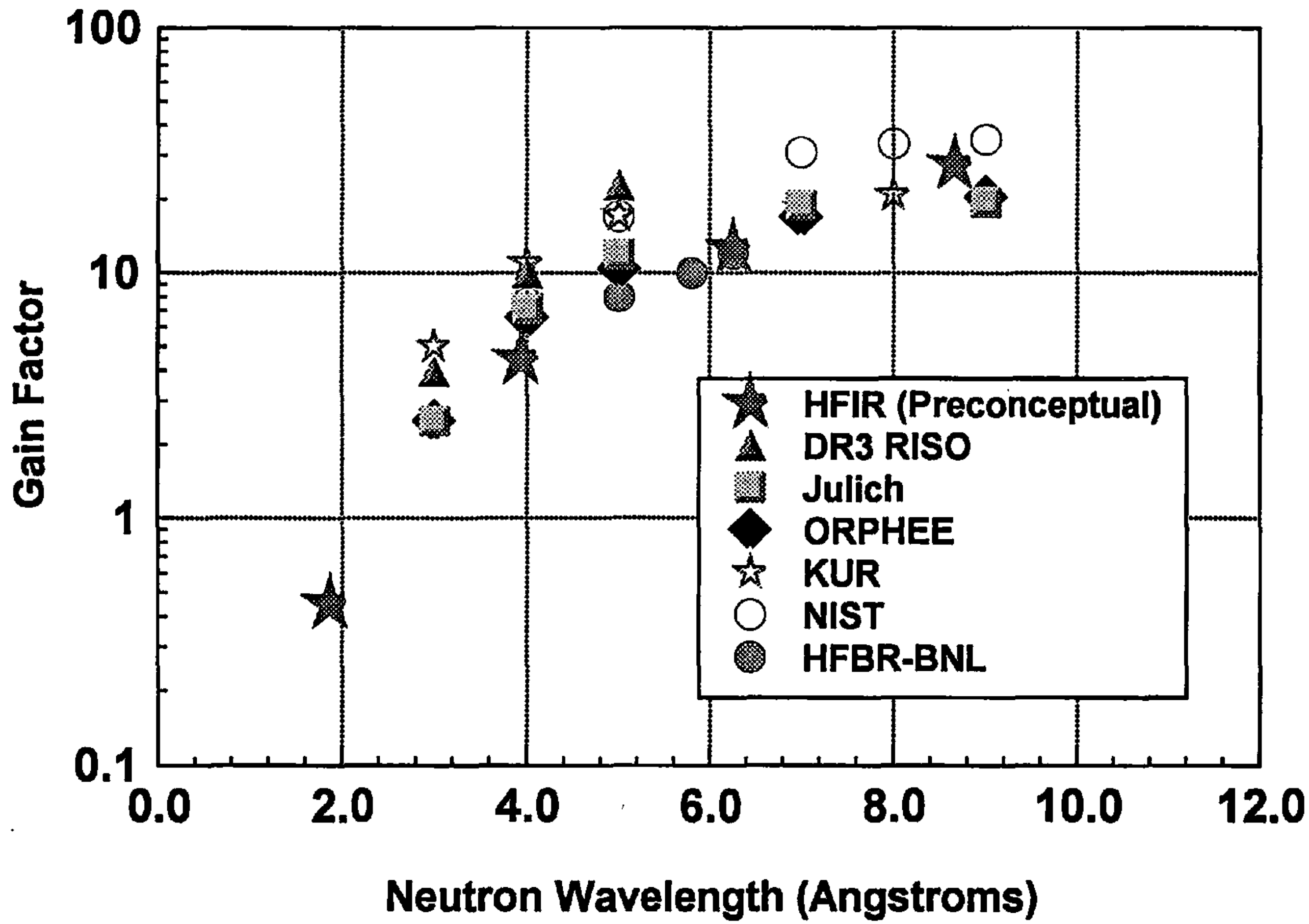


HFIR COLD SOURCE PERFORMANCE

- The HFIR cold source would be located in a perturbed neutron flux field of ~7×10^{14} cm^{-2}- s^{-1} and produce a gain factor of approximately 20 for 8 Å neutrons

- The cold neutron brightness (cm^{-2} - s^{-1} - $Å^{-1}$ - $steradian^{-1}$) down the beam tube is presently calculated to be 1.5×10^{12} for 5 to 7.5 Å neutrons and 5.3×10^{11} for 7.5 to 10 Å neutrons

Comparison of Calculated Gain Factors for the Proposed HFIR LH2 Cold Source with Measured Gain Factors at Other LH2 Cold Source Facilities



A TWO STAGE TESTING PROGRAM IS PLANNED (1)

- The objective of the tests are threefold:
  - To validate correct operation of the system design and demonstrate its stability under all normal circumstances and fault situations that can be practically replicated
  - To provide benchmarking data for computer models
  - To allow the control and instrumentation systems to be developed including the philosophy of operational interlocks and safety shutdown systems

A TWO STAGE TESTING PROGRAM IS PLANNED (2)

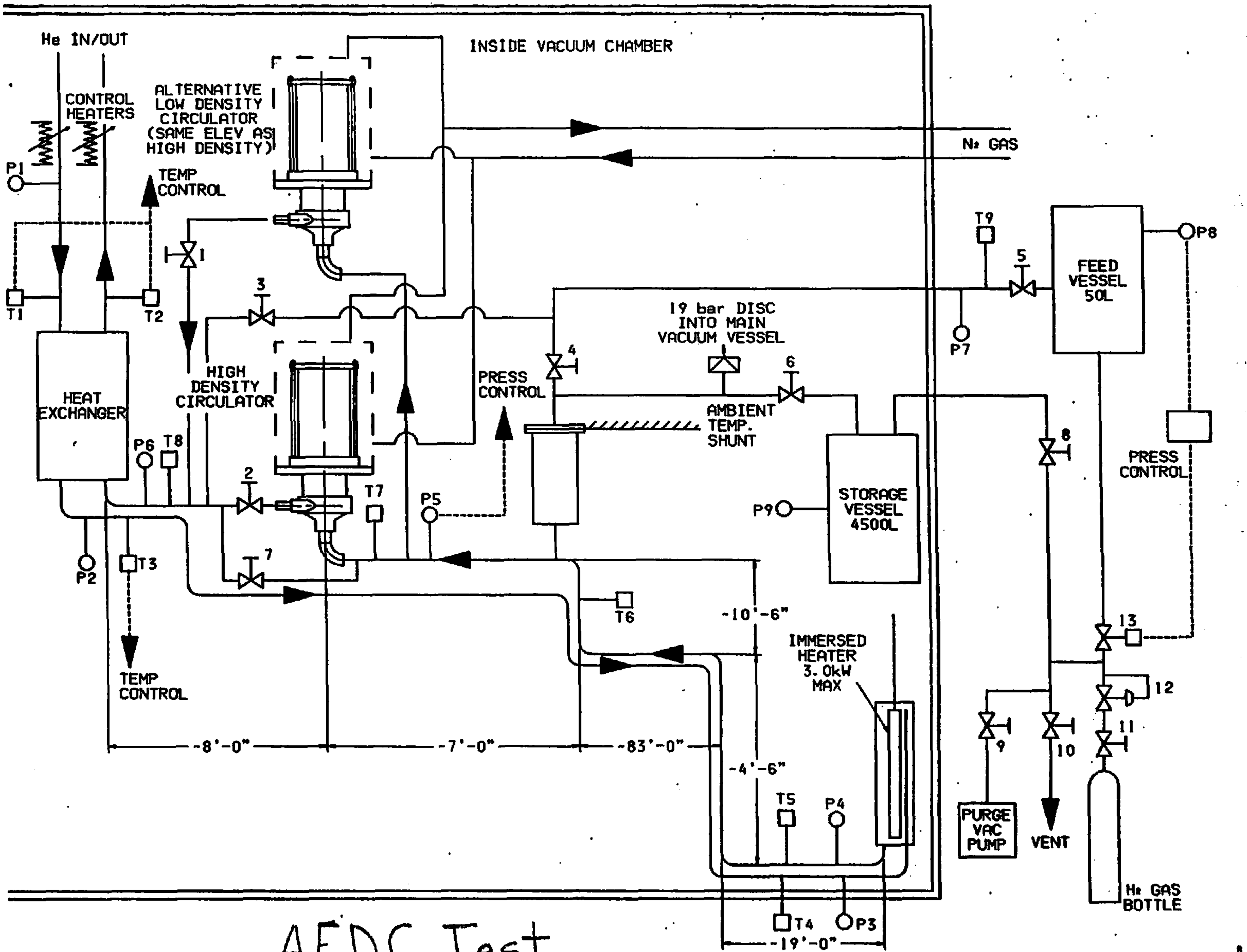
- Phase 1 of the test program is expected to begin this winter and will test the loop concept performance under various conditions with 7 types of tests

  1. Demonstrate stable cooldown operation
  2. Evaluate stability of the system under control and quantify the intrinsic heat load
  3. Evaluate stability of the system under simulated reactor start-up and rise to full power heating level
  4. Evaluate response to simulated reactor scram
  5. Evaluate ability to handle the loss of circulation at steady state operation without loss of heat load
  6. Evaluate ability to withstand the loss of refrigeration cooling
  7. Demonstrate ability to warm up the system to ambient temperature without surging or instability

A TWO STAGE TESTING PROGRAM IS PLANNED (3)

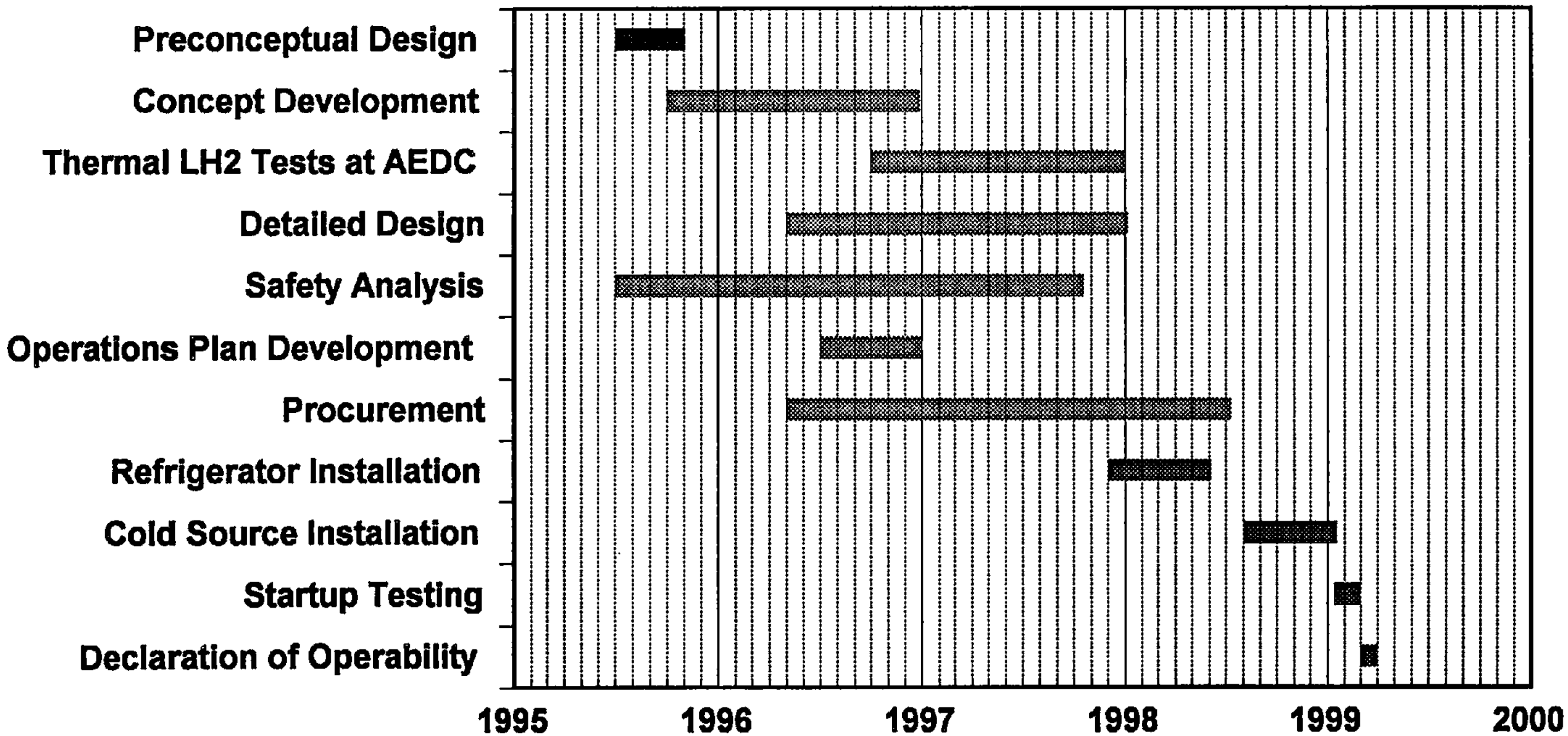
- Phase 2 would provide final testing of the prototype design and would include hot spot evaluation on the moderator assembly
    - Not expected to start until fall of 1997

- Tests will be performed by Arnold Engineering Design Center at the Arnold Air Force Base at Tullahoma, Tennessee
    - Their large vacuum chambers allow the construction of the loop without the expensive triple wall arrangement normally used to ensure hydrogen safety.
    - An existing 2.5 to 3 kW refrigerator at the site will allow the tests to be performed at full prototype heat load conditions


He IN/OUT
CONTROL HEATERS
ALTERNATIVE LOW DENSITY CIRCULATOR (SAME ELEV AS HIGH DENSITY)
INSIDE VACUUM CHAMBER
N2 GAS
P1
TEMP CONTROL
T1
T2
1
3
HEAT EXCHANGER
HIGH DENSITY CIRCULATOR
PRESS CONTROL
4
19 bar DISC INTO MAIN VACUUM VESSEL
6
AMBIENT TEMP. SHUNT
T9
5
FEED VESSEL 50L
P8
P7
PRESS CONTROL
8
STORAGE VESSEL 4500L
P9
P6
T8
2
T7
P5
P2
T3
7
T6
-10'-6"
IMMERSED HEATER 3.0kW MAX
13
12
11
9
10
TEMP CONTROL
-8'-0"
-7'-0"
-83'-0"
-4'-6"
T5
P4
PURGE VAC PUMP
VENT
He GAS BOTTLE
T4
P3
-19'-0"

AEDC Test

● Reference HFIR Cold Source Schedule



PELLETIZED METHANE
COLD SOURCE CONCEPT

presented to the 5th meeting of the
International Group on Research Reactors

Work performed by A. T. Lucas
Material presented by D. L. Selby

November 1996

SOLID METHANE MODERATORS

- Solid methane appears to offer a substantial gain in cold neutron brightness over LH_2

- Radiation increases rate of polymerization of methane, an exothermic process. Heat can rapidly cascade through a solid block moderator with damaging results ("burping effect").

- Frozen methane pellets may offer mobility and allow the moderator to be constantly replaced before significant radiation damage occurs. If damage does occur, cascade effect is greatly reduced.

- Liquid hydrogen or helium passing through the pellet bed provides cooling of the moderator.

GENERAL REQUIREMENTS FOR A METHANE PELLET SYSTEM

- Pellet production on demand
- Pellet transport system capable of continuous feeding
- Coolant system with ability to separate the pellets from the coolant

TESTS PERFORMED TO DATE AT ORNL (1)

- Pellet production system operated as follows:

  - Methane liquefied by liquid nitrogen

  - Liquid methane expelled through a nozzle by electronically driven diaphragm pump

    - forms a stream of uniform droplets

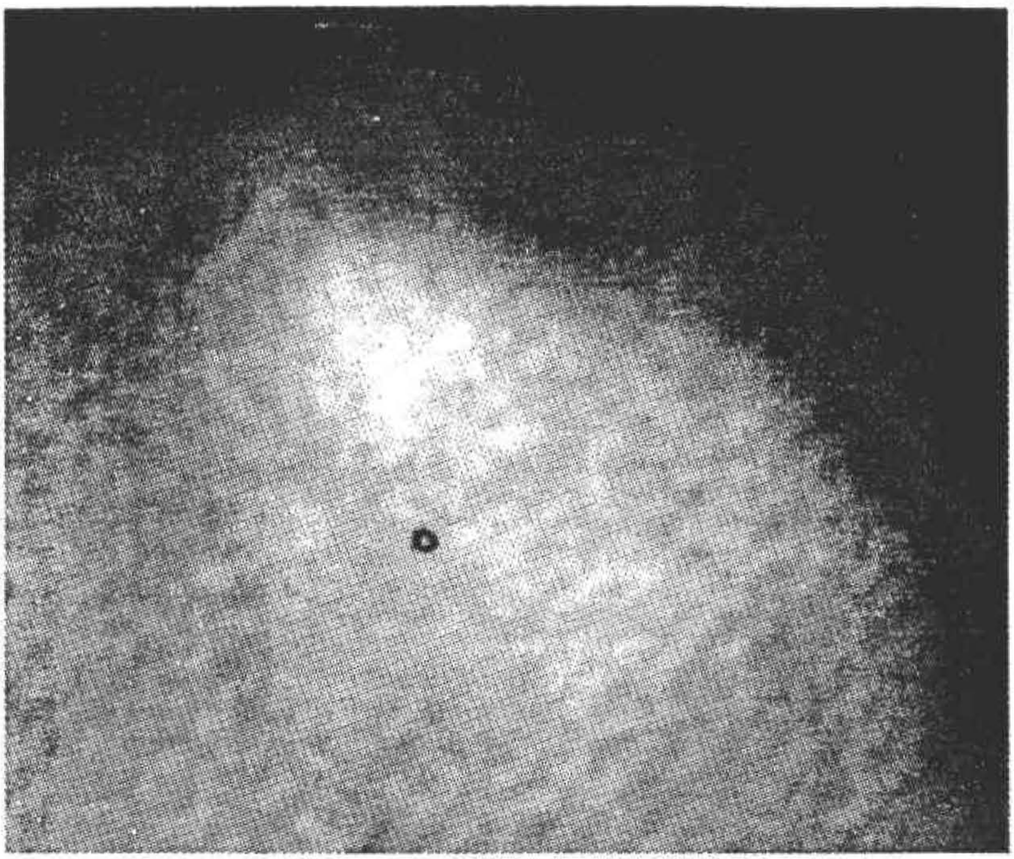
  - Droplets allowed to fall through a room temperature gas column (8m) cooling by evaporation

  - Pellets are then collected in a liquid hydrogen pool at the bottom

TESTS PERFORMED TO DATE AT ORNL (2)

- Preliminary results
  - Pellets could be formed repeatably (good result)
  - cooling was not complete and pellets did not completely freeze in column (not so good result)
  - frozen methane appeared to be granular and dry and could be fractured
  - no solubility of liquid methane in hydrogen was found in mass spec. analysis of the hydrogen (good result)

Methane Drop
for
Advanced Cold Neutron Source



- Diameter: 1 mm, variable repetition rate: 1-100 Hz
- Falling vertically to bath of liquid hydrogen

NEXT PHASE OF DEVELOPMENT

- Improve column cooling or try alternative approach
- Complete investigation of methane pellet behavior in liquid hydrogen and helium
- Demonstrate a practical pellet transport system
- Examine the effects of irradiation on pellets and pellet bed
- However, we have exhausted the present funding source (ANS project) and need to find money for the next phase of development.

Performance of the Liquid Hydrogen Cold Source

J. Michael Rowe, Paul Kopetka, and Robert E. Williams
National Institute of Standards and Technology
Gaithersburg, Maryland USA

ABSTRACT

At the time of the IGORR-V Meeting, the NIST research reactor was shutdown for a number of modifications, including the replacement of the D_2O cold neutron source with a liquid hydrogen cold source. On September 27, 1995, the LH_2 source was operated for the first time, increasing the flux of cold neutrons by a factor of six over the D_2O source. Measurements of the cold source performance are compared with the Monte Carlo simulations used in its design. In its first year the source has been reliable, very stable, and easy to operate.

INTRODUCTION

A description of the liquid hydrogen source was presented at the previous IGORR Meeting [1], so only a brief summary will be given here. The success of the LH_2 cold source at the Orphée Reactor in Saclay, a thin cylindrical annulus [2], inspired us to investigate a similar possibility for our source. Monte Carlo simulations indicated that a nearly closed spherical annulus, with an exit hole just large enough to illuminate the guides, would be the best geometry for a hydrogen source in the NBSR cryogenic beam port.

The source is a 2-cm thick spherical annulus, 32-cm OD, with a 20-cm diameter exit hole facing eight neutron guides. Only 5 liters of LH_2 is needed to fill the annulus. Heat deposited in the chamber is removed by a thermosiphon, in which the liquid flows by gravity to the moderator chamber from a condenser located on the reactor face, and vapor rises back to the condenser by natural circulation; no moving parts are required in the hydrogen system.

The cold source is currently operated at about 110 kPa (just over one atmosphere), so the moderator temperature is 20.4 K. Its refrigerator is programmed to maintain a steady hydrogen pressure by adjusting the flow of cold helium to the condenser as necessary. Thus, the liquid inventory remains almost constant from 0 to 20 MW, and the system responds to scrams and follows power changes automatically. An excess of liquid hydrogen in the system produces two-phase return flow to the condenser, which is isothermal and very stable. If the refrigerator is shutdown, the hydrogen expands into the 2-m^3 ballast tank, to its initial pressure of 430 kPa.

PERFORMANCE

A reactor rundown signal is generated if either the pressure exceeds 175 kPa, a point at which the moderator chamber is still full of liquid hydrogen, or, if the D_2O cryostat assembly coolant flow is too low. There have been about 35 refrigerator trips, nearly all caused by brief power interruptions. The refrigerator is programmed to restart automatically when power is restored (usually without intervention and within 2 minutes), and, as a result, only two days of reactor operation have been lost due to cold source trouble; its availability was 99% in its first year.

The gain in cold neutron flux with respect to the D_2O source, and the gain with respect to the source when shutdown and warm, were measured at the SANS instrument on NG-7. Compared to the D_2O source, the gain varies from 2.5 to 6 between 4 Å and 10 Å. The gain of the source cold-to-warm, is a measure of absolute source performance, allowing a comparison with other facilities. The gain varies from 7, at 4 Å, to 50, at 20 Å, making this hydrogen source one of the best ever built. The gains are normalized per MW, and do not include the increase in reactor power from 15 to 20 MW. Using gold foils, the capture flux was measured to be 1.5-2 x 10^9 n/cm^2/s at sample positions in the guides, at points 35-50 m from the source. The measured gains agree well with Monte Carlo calculations using the MCNP code [3], assuming that the LH_2 is 65% ortho-hydrogen.

To benchmark the gain calculations, it is necessary to know the ortho-para ratio in the LH_2. A molecule of para-hydrogen has atoms with anti-parallel nuclear spins, J=0,2,.., while the spins are parallel, J=1,3,.., in ortho-hydrogen. At room temperature, normal hydrogen, n-H_2, is 75% ortho, but at 20 K, it slowly converts to 99% p-H_2, at a rate of 0.0114 hr^{-1}. In the 1960's, NASA researchers reported that the presence of neutron and gamma radiation can speed this conversion to para by orders of magnitude [4,5]. Since the neutron scattering cross section of hydrogen is much larger for o-H_2 than p-H_2, the yield of cold neutrons from the NIST LH_2 moderator is calculated to be twice as high for 65% o-LH2 than for 100% p-LH_2. To ensure the highest possible ortho content in the source, we installed a recirculating pump between the condenser and the ballast tank, and a room-temperature catalyst, so that the pump could deliver about 0.1 g/s n-H_2 to the source.

In the hours following a startup of the cold source with fresh, presumably normal LH_2, there is no loss of intensity due to conversion of ortho to para. When the recirculation pump was turned on, there was no detectable increase in the cold neutron flux. These observations led us to conclude that either the LH_2 was somehow maintained at a nearly normal ortho content, or that the conversion to para is extremely fast. Comparison a with time-of-flight measurement on NG-1 (see the figures following the text) clearly shows that the shape of the TOF spectrum is much closer to the calculated spectrum with 65% ortho than that of 100% para. Although we have not reconciled the conflict with the NASA studies, we believe the source is operating with at least 65% ortho hydrogen.

The nuclear heat load in the moderator chamber has been recomputed with MCNP, using Al and ^{235}U cross sections modified to include delayed gamma rays from ^{28}Al and fission products, respectively [6,7]. A total heat load of 980 ± 30 W was calculated for the moderator chamber, which includes 340 g of LH_2 (290 W), and 2140 g of Al (690 W). From an indirect measurement of refrigerator performance, the heat load associated with full-power operation of the cold source is 800 ± 40 W. Based on other benchmarks of the MCNP reactor model, better agreement was expected. Similarly, the heat deposited in the entire cryostat assembly was predicted to be 22 ± 2 kW, but flow and temperature measurements of the D_2O cooling the assembly indicate that 32 ± 2 kW are removed at 20 MW. This large underestimate is only partly explained by the fact that the cryostat assembly is not thermally isolated from its warm surroundings.

CONCLUSION

The liquid hydrogen source installed in the NIST research reactor has achieved or exceeded all design goals, and is proving to be highly reliable (availability > 99 %). The performance of the novel spherical annulus design has been verified, and shown to provide good gains to long wavelengths. Detailed comparison of the performance (cold neutron production and heating) to calculations provides a rigorous test of the codes and models used. These comparisons show that the heating rate calculated for the moderator and chamber are 25 % too high. Interpretation of the neutron spectra are dependent upon the assumed ortho/para hydrogen ratios - assumption of an ortho hydrogen fraction between 65 and 75 % provides similar accuracy for neutron intensities, and excellent agreement for spectrum shape, while assumption of a 100% para hydrogen fraction are in complete disagreement with the data. Improvements to the design to enhance the source performance are now being planned for installation at a future date.

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6. Williams, R. E., Rowe, J. M., and Blau, M., "Benchmark of the Nuclear Heat Deposition in the NIST Liquid Hydrogen Cold Source", presented at the Ninth Symposium on Reactor Dosimetry, September 2-6, 1996, Prague, Czech Republic (in press).

7. Gallmeier, F. X. and Gehin, J. C., "MCNP--Cross Section Modifications for Reactor Physics Calculations," Private Communication (1996), to be published.

Figure Captions

Figure 1. Schematic of the NIST liquid hydrogen cold source.

Figure 2. Plan view of the cryogenic beam port showing the relative positions of the cold source, fuel, and the in-pile sections of the neutron guides.

Figure 3. Gains for the NIST liquid hydrogen cold source as a function of wavelength, measured at the NG-7 SANS.

Figure 4. Cold neutron spectrum: measured vs. calculations.

Figure 5. Calculated nuclear heat load.

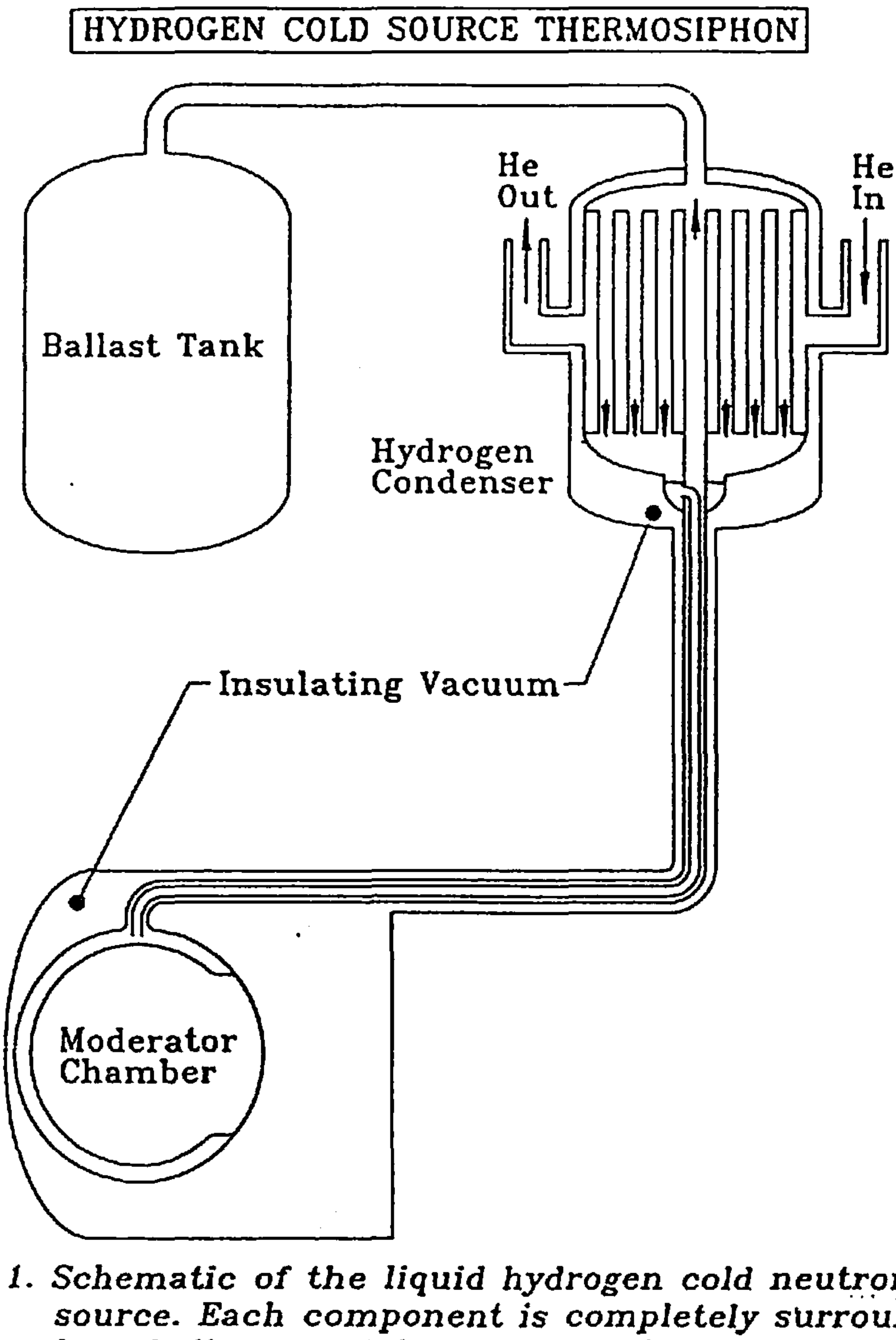
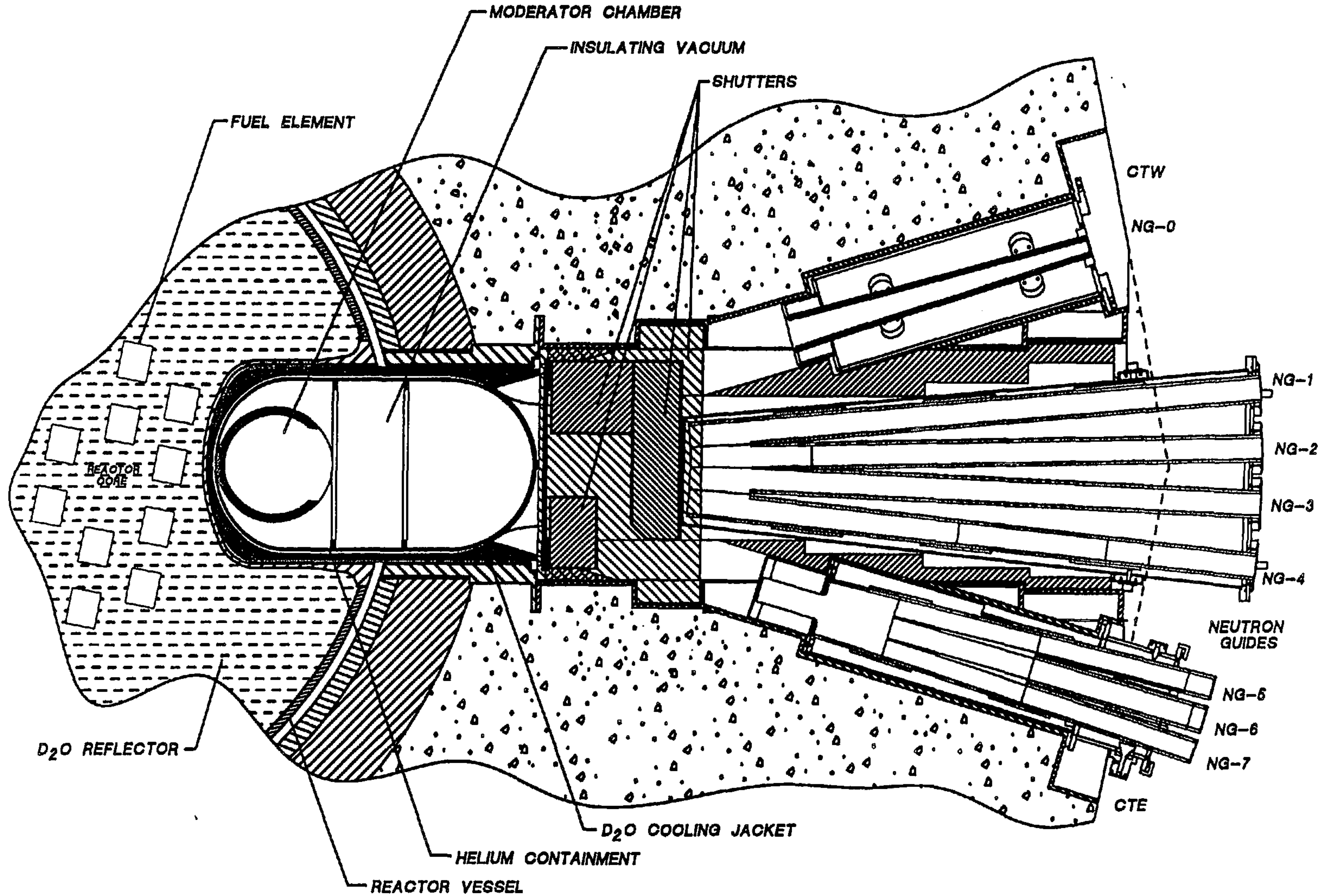
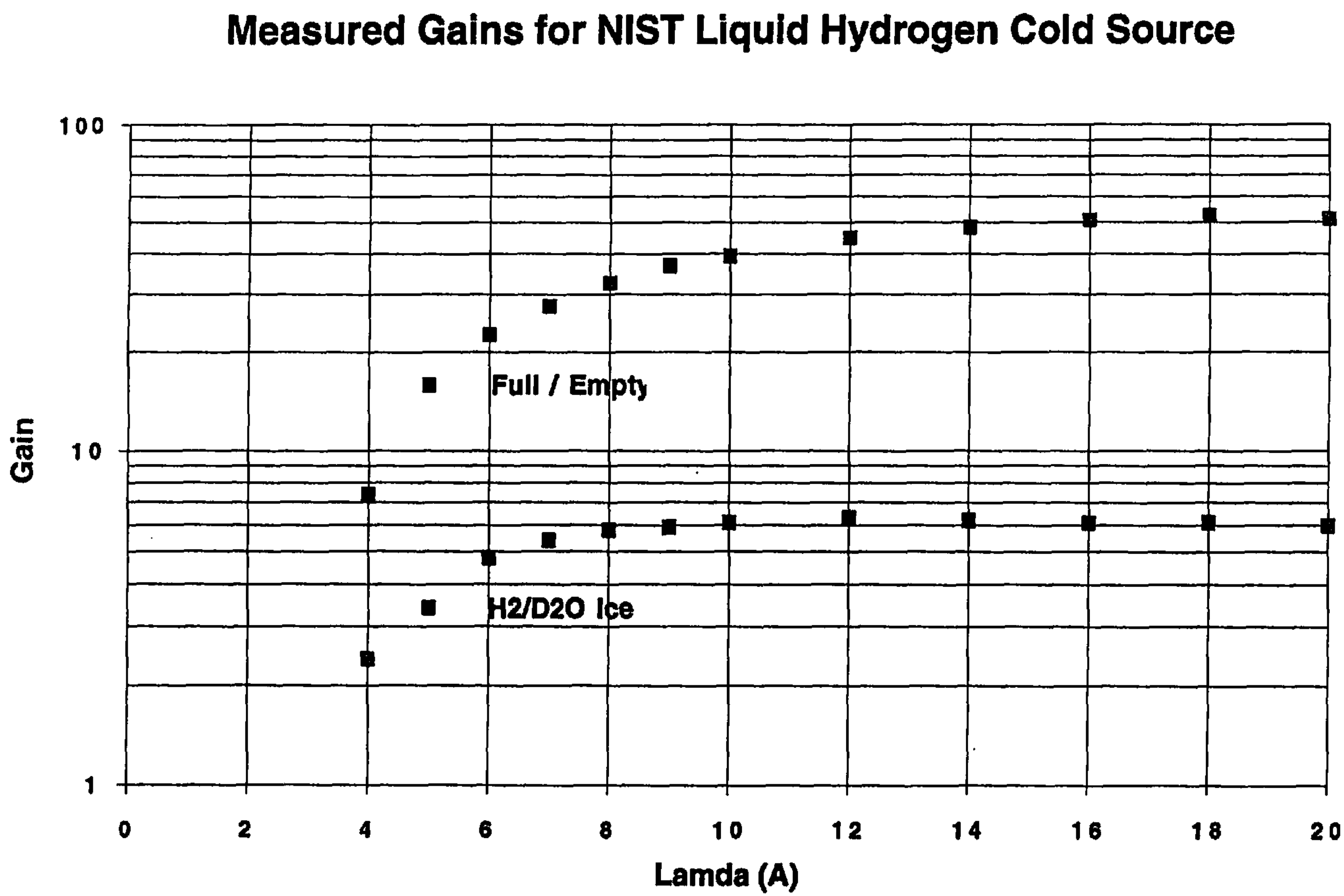


Fig. 1. Schematic of the liquid hydrogen cold neutron source. Each component is completely surrounded by a helium containment, not shown.



NIST HYDROGEN COLD SOURCE



Measured Gains for NIST Liquid Hydrogen Cold Source
100
10
1
Gain
Full / Empty
H2/D2O Ice
0
2
4
6
8
10
12
14
16
18
20
Lamda (A)

Cold Neutron Spectrum — Measured vs. calculations

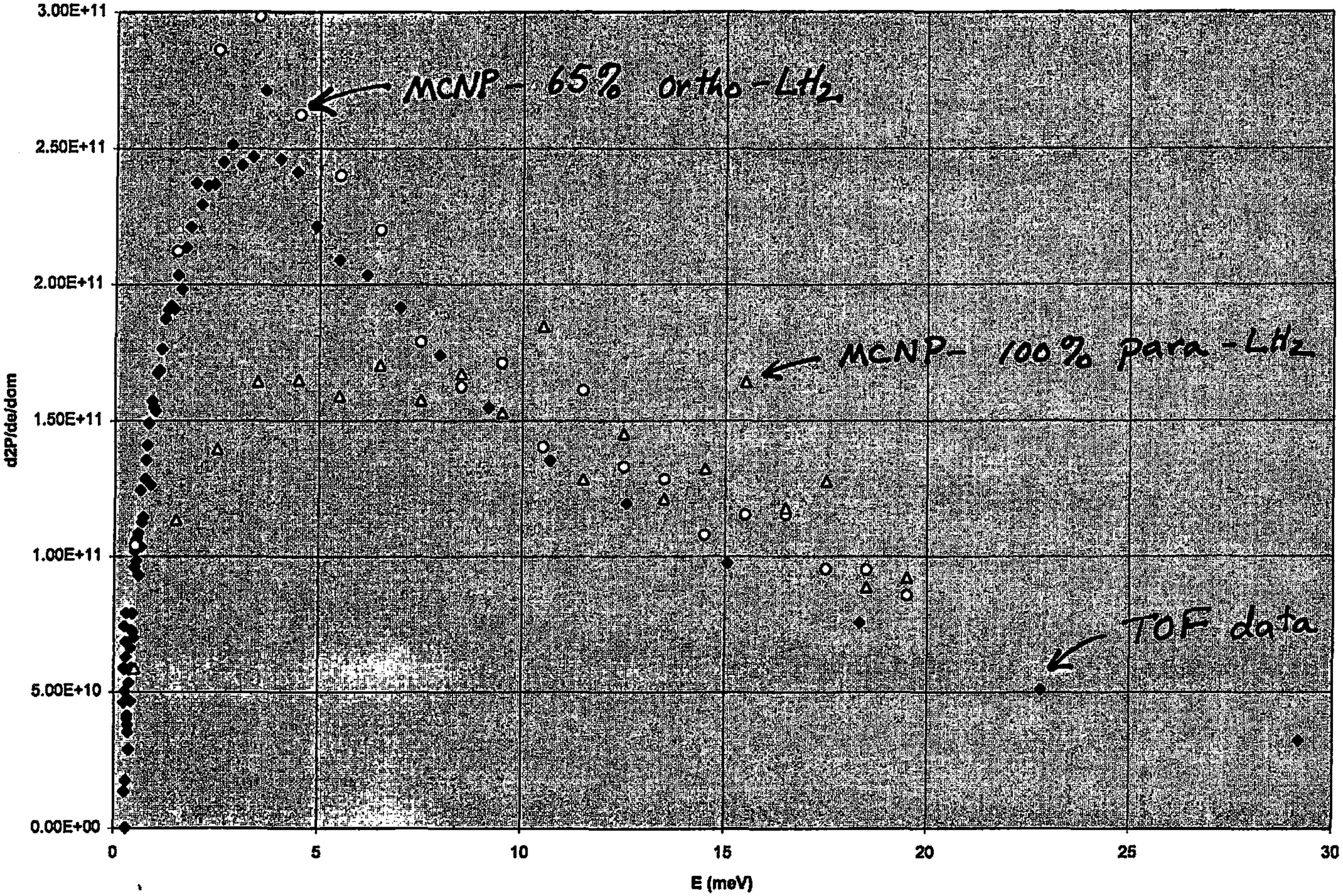


Table 1. Cold Source Heat Load Results (Watts)

Heat Source	Liquid Hydrogen (337 g)	Aluminum (2140 g)
Direct Neutron	123 ± 8	2.0 ± .1
Prompt Fission and Capture Gamma Rays	124 ± 6	405 ± 16
Delayed Fission Product Gamma Rays	30 ± 5	106 ± 14
^{28}Al Gamma Ray	12 ± 1	41 ± 2
Beta Particles	-	140 ± 11
Subtotal	289 ± 11	694 ± 24
TOTAL HEAT LOAD: 980 ± 30		

Design Review in Concept Phase of CNS at HANARO

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Abstract

The Korea Atomic Energy Research Institute(KAERI) has successfully completed the construction of HANARO research reactor, which is pool type of 30 MW thermal power, and achieved its first criticality in February 1995. The HANARO(Hi-flux Advanced Neutron Application ReactOr) is currently under the second cycle operation after successful nuclear tests at zero and low power levels. The reactor has been designed to have the compact core to enhance flux levels and to have the spacious reflector to accommodate a lot of irradiation hole. From the beginning stage of HANARO design, CNS has been considered for basic research and development of applied technology such as material and polymer science. The cold source of hydrogen & deuterium mixture will be incorporated in the design at the concept phase. This presentation introduces the design in concept phase of CNS at HANARO, which will be started from November, 1996 and will last for 6 months.

1. Introduction

The experimental facilities to be installed at HANARO consist of irradiation facilities for RI production, NAA, NTD, Fuel test loop etc. and beam experimental facilities such as neutron beam tubes and neutron radiography etc.. The 32 irradiation holes and 7 beam tubes are positioned for easy access of experimental facilities as listed in Table 1.

Among various kinds of utilization facilities of HANARO, Cold Neutron Source(CNS) facility is expected to carry out important roles in various fields as a means of widening the realm of fundamental research by contribution of developing the state-of-the-art technology in the polymer science, biology, colloidal chemistry, metallurgy, and condensed matter physics.

Table 1 Experimental Provisions at HANARO

Location	Name	Shape	No	Size(cm)	Purpose
Inner Core	CT	Hexagonal	1	7.44	Capsule Irradiation
	IR	Hexagonal	2	7.44	Capsule Irradiation
Outer	OR	Cylindrical	4	6.0	RI Production
Reflector	CNS	Cylindrical	1	16.0	Cold Neutron Source
	NTD	Cylindrical	2	22.0/18.0	Silicon Doping
	LH	Cylindrical	1	15.0	Fuel Test Loop
	HTS	Cylindrical	1	16.0	RI Production
	NAA	Cylindrical	3	6.0	Activation Analysis
	IP	Cylindrical	17	6.0	RI Production
Beam Tube	ST	Rectangular	4	7 x 14	Spectrometer
	CN	Rectangular	1	7 x 15	Cold Neutron Beam Exp.
	NR	Cylindrical	1	10	Neutron Radiography
	IR	Cylindrical	1	10	Irradiation Tests

For the installation of CNS at HANARO, some technical review and analysis have been taken in the concept phase to reveal physical interfaces and problems. Even though CNS installation has been considered from the beginning stage of HANARO design since 1985, only a vertical hole for placing cold source and a rectangular type horizontal beam tube for neutron guides have been secured in the reflector tank. The plan view of HANARO reactor core with various in-reactor experimental provisions is drawn in Fig 1. In the conceptual design of CNS at HANARO, calculation will be performed to determine the best size and hydrogen-deuterium composition of the source. After geometry and composition will be chosen, the heat load will be calculated. The main points of the conceptual design include the followings:

- Heat release
- Design of cold neutron source
- Design of ultra-cold neutron source
- Liquid hydrogen loop
- Safety consideration
- Neutron guide system
- General scheme

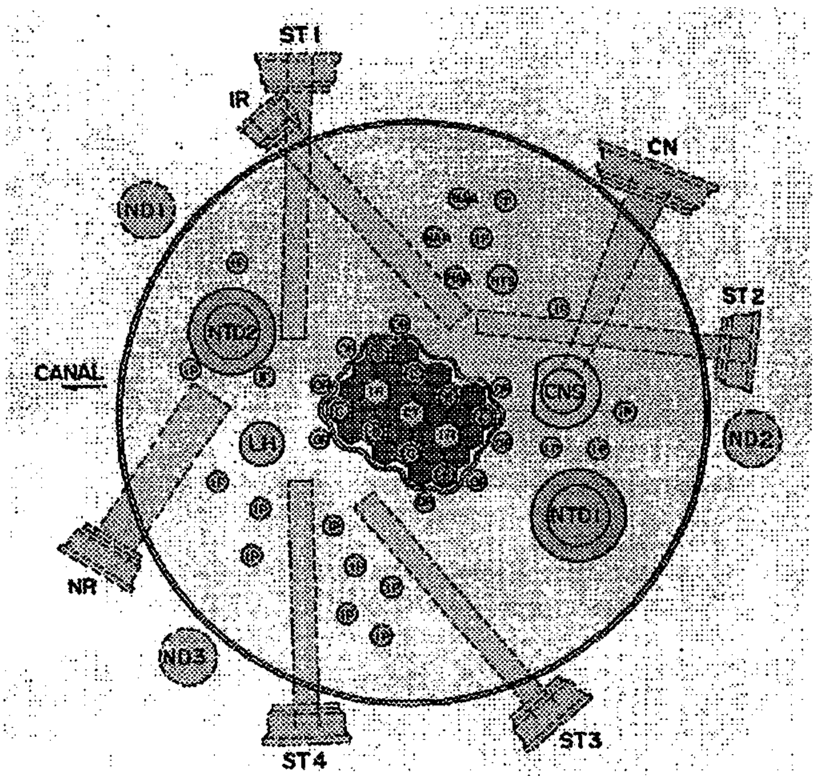

Fig 1. The plan view of HANARO reactor

2. Design Review at Concept Phase

2.1 Basic Data

HANARO is the open-tank-in-pool type which has benefit of free access to pool top and large inventory of water as heat sink. The reactor pool has dimensions 4 m diameter and 13.4 m height. The reflector tank of 2 m diameter and 1.2 m height contains heavy water and accommodates vertical holes and horizontal tubes. The cold neutron source will be placed into the vertical hole in the reflector tank. The diameter of the vertical hole is 16 cm, its length is 120 cm. The horizontal tube is used for extracting cold neutrons. It is connected with the vertical tube. The horizontal tube nose dimension at contacting point to vertical hole is 6 x 15 cm. The neutron and gamma fluxes are shown in Table 2 and Table 3 respectfully.

Table 2 Neutron Fluxes at HANARO

Energy Level	Neutron Fluxes(n/cm^2·sec)
Fast E>0.82 MeV	1.3×10^{12}
Thermal E<0.625eV	1.7×10^{14}
Thermal Flux at Tube Nose	9.7×10^{13}

Table 3 Gamma Fluxes at HANARO

Energy Level(eV)	Gamma fluxes(γ /cm^2·sec)
1.0×10^{-5} to 1.0×10^{5}	5.818×10^{13}
1.0 x 105 to 5.1×10^{5}	4.452×10^{13}
5.1×10^{5} to 6.0×10^{5}	5.610×10^{11}
6.0×10^{5} to 1.3×10^{6}	3.993×10^{12}
1.3×10^{6} to 1.3×10^{6}	5.287×10^{12}
3.0×10^{6} to 7.5×10^{6}	1.286×10^{12}
7.5×10^{6} to 1.4×10^{7}	4.544×10^{10}

2.2 Design requirement

The design criteria for HANARO CNS is maximum increase and stable supply of cold neutron flux, and safety with regard to personnel and the reactor. High technology for cryogenics and gas explosion and reactor safety engineering is required in design, construction and operation for safety of CNS system and the reactor. Especially, the measure to control the hazards related to the hydrogen use is very important in design. The design requirements are summarized in the table 4.

Table 4 Design Requirements for HANARO CNS

Item	Requirement
Gain factor	More than 20 for 5 Å
Cold neutron flux	More than 5×10^8 n/cm^2·sec at the end of cold neutron guide tube
Life time consideration	30 years
Applications	• Crystallographic studies • Neutron non-destructive evaluation • Small angle neutron scattering • Neutron reflectometry • Neutron inelastic for chemical analysis • Neutron and nuclear physics
Safety consideration	• Explosion due to hydrogen-air contact • The effect on reactor by installation of CNS(Reactivity, Pressure etc.)
Installation consideration	• To be installed in the existing vertical hole(φ 16 cm) • Interfaces with the existing facilities in the reactor pool • Space limit for installation near reactor pool
Other	• Design of ultra-cold neutron facility

2.3 Heat release

The heat release in CNS includes heating due to neutrons(slowing down and absorption) and prompt γ-rays and β-radiation because of capture of thermal neutrons. Heat release for CNS is not calculated in detail yet, but some rough estimation can be done. This is important because it is necessary for choosing an efficient way for the heat removal. The estimations are shown in Table 5. This data will be defined more accurately again in the near future.

Table 5 Heat release data

	Fast and epi-thermal	γ-rays	β-rays	Total
Specific nuclear heating in hydrogen	1.3 W/g	0.4 W/g	-	1.7 W/g

Specific nuclear heating in deuterium	0.2 W/g	0.2 W/g	-	0.4 W/g
Specific nuclear heating in zircalloy	-	0.2 W/g	-	0.2 W/g
Specific nuclear heating in Al	-	0.2 W/g	0.3 W/g	0.5 W/g

2.4 Source Chamber

It would be reasonable to use the maximal volume of the moderator and to optimize its efficiency by varying the concentration of hydrogen in deuterium. The source chamber will be made of cylindrical type having 140 mm diameter and 210 mm height, and the available volume will be about 3 liters. The cylindrical chamber with elliptic bottoms and with thickness of Al walls of 2 mm has the weight about 650 g. Taking into account the weight of chamber tubes in the field of irradiation, the total weight is estimated to be about 1.1 kg and the heat load due to the nuclear heat release in the material is about 550 W. The nuclear heat release in hydrogen is estimated 450 W or in deuterium 240 W. It is expected that optimal concentration of the hydrogen in deuterium-hydrogen mixture is about 50 % or less. Therefore the more realistic estimation of the heat load for the deuterium-hydrogen mixture is about 350 W. Thus rough estimation of the total heat load for the CNS is about 900 W.

2.5 Heat Removal

The way for heat removal for HANARO CNS is a subcooled liquid hydrogen thermosiphon. No bubbles, self-regulation, safety are the benefit features of this way. There is a natural circulation of the subcooled liquid hydrogen between the chamber and the heat exchanger. The minimum helium temperature is higher than hydrogen freezing point. The cold helium removes the heat from liquid in the heat exchanger. A counter-flow heat exchanger is the preferable one. The thermosiphon loop is placed entirely in a vacuum containment. The loop is filled by liquid moderator(hydrogen-deuterium mixture) without vapor. The chamber contains about 3 liters of moderator and is made of aluminum. The chamber weight is about 1 kg. The total radiation heat load is expected to be about 1.0 kW.

2.6 Review of interferences with the existing facility

For installation of CNS at HANARO, some technical review and analysis should be taken to reveal physical interfaces and problems. Vacuum system, helium containment and number of the diverging cold neutron guides from the nozzle of reflector wall had not been considered technically when the reactor structures and embedded parts were being designed and constructed. Really, removing the light water from the installation hole in the reflector tank to improve the gain becomes an issue because the wall of the reflector tank is a little bent due to heat for welding. Therefore, we consider extra helium layer between the reflector tank wall and vacuum containment as shown in Fig 2..

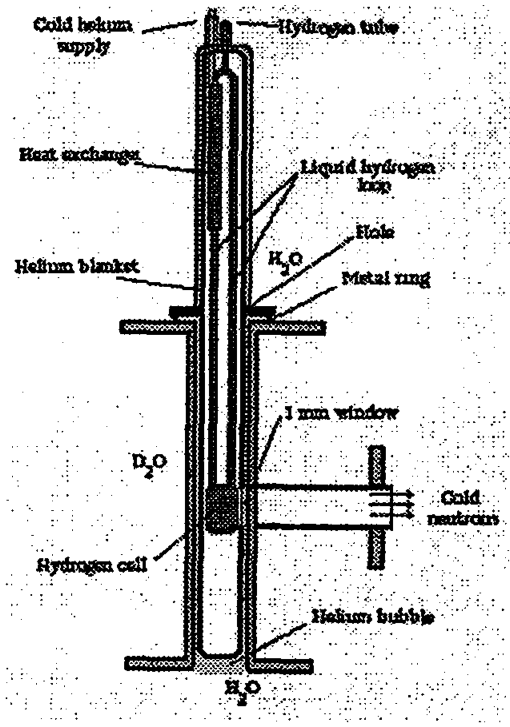

Fig 2. Installation of the cold neutron source

This helium environment can function as the helium safety blanket at the same time. The pressure of helium in the blanket is equal to hydrostatic pressure in the pool at the depth 11.3 m. The plug at the bottom of CNS hole is not necessary, and the water is taken out of the hole by means of the helium bubble.

3. Safety

The CNS facility might be hazardous and demands special approaches for safety though it can be used for various experiments of neutron physics. It is necessary to apply technological and organization approaches in the design to escape dangerous influence on people, reactor and surrounding atmosphere. CNS safety is considered in view of nuclear activity, radiation and hydrogen explosion.

• Cryogenic system failure

In case of cryogenic system failure, all liquid vaporizes and the moderator cell may be overheated. The vacuum containment will be filled by helium automatically to transfer heat from the cell to the containment wall.

• Moderator cell rupture

In case of cell rupture a vacuum disappears in the containment, and cryogenic cooling loses. Moderator vaporizes. The final pressure in the moderator tank will be less than the pressure in normal warm conditions. The moderator should be evacuated after shut-down.

• Failure of helium blanket around vacuum source containment

In this case, only cold neutron flux may decrease due to incoming of water.

• Vacuum failure

In case of vacuum failure, the moderator vaporizes. The vacuum containment should be filled by helium additionally.

• Reactor shut-down

Reactor shut-down does not cause any trouble. Moderator will be in the loop at a lower(a little higher than freezing temperature).

4. Neutron Guide System

The task of the neutron guide system is to transport neutrons produced in the cold neutron source to experimental devices which is intended to perform scientific investigations with neutrons. Transportation of neutrons is realized on the base of the mirror reflection of cold neutrons from the walls of the neutron guide. The curved neutron guide channels can be used to cut off fast neutrons and gamma radiation because their total reflection coefficients are negligible in comparison with that of cold neutrons. The scheme of the neutron guide system is presented in Fig 3. The neutron guides are channels with rectangular cross section, prepared by using mirrors with a glass substrate and ^{58}Ni reflecting coating. The internal cross section is of 40 mm width and 140 mm height. There are two main neutron guide. The possibility of installation of the third channel will be reviewed. CNG in the figure is the converging guides made of Ni/Ti super-mirror. Separation of the neutron beams formed with these channels in the horizontal plane is defined by an appropriate angle between the channels. The value of the angle $2\alpha_1 = 120 \times 10^{-3}$ radians is restricted by dimensions of the channel tube containing the neutron guides as shown in Fig 4. Characteristic wave lengths for different neutron guides, and cross sections of the guides are shown in the Table 6.

Table 6 Characteristic wave length

Neutron Guide	Characteristic λ (Å)	Cross section a x h mm	Coating
1 NG	6	20 x 60	^{58}Ni
2 NG	2.5	40 x 60	^{58}Ni
3 NG	3.5	20 x 60	^{58}Ni
4 NG	8	40 x 60	^{58}Ni
NG		40 x 140	^{58}Ni
CNG			Super Ni/Ti

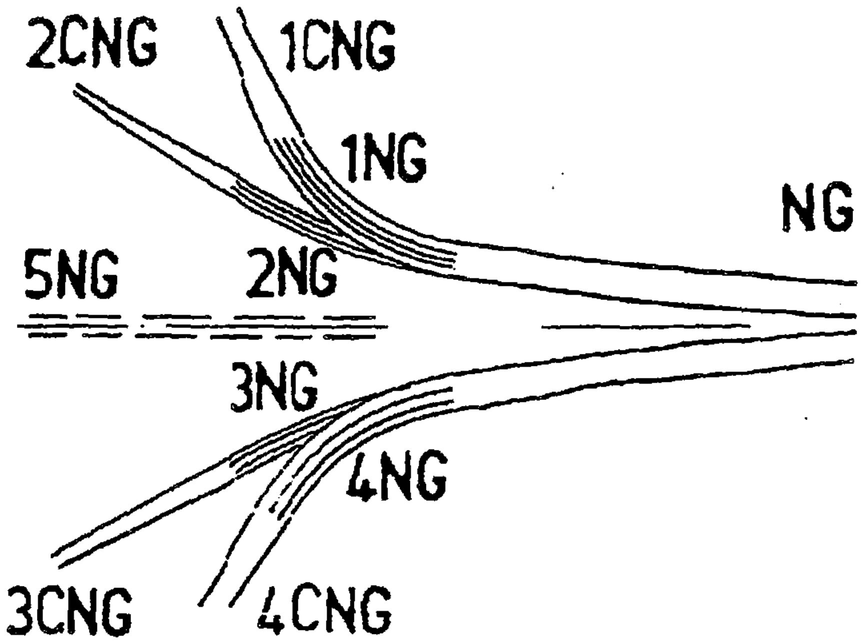

Fig 3. The scheme of the neutron guide system

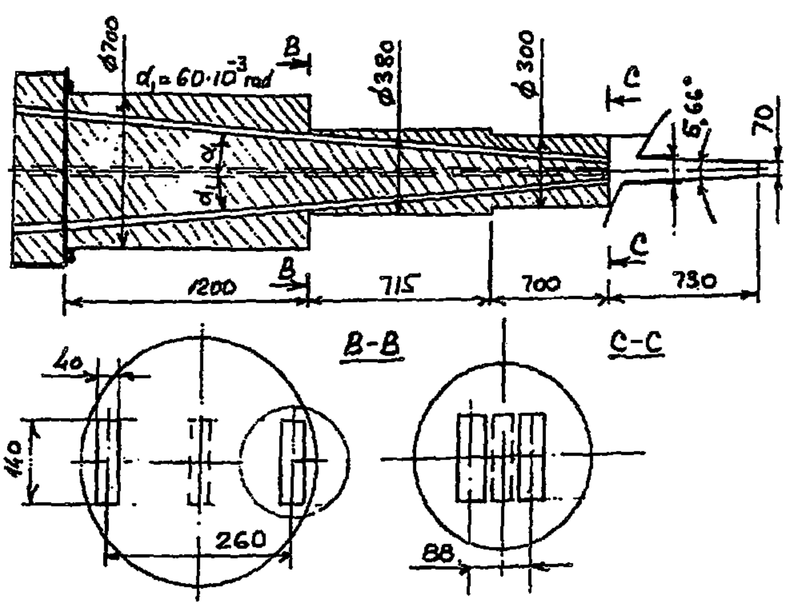

Fig 4. Neutron guide installation in beam tube

5. General scheme

The general scheme for cold neutron source at HANARO reactor is shown in Fig 5. The main parts of the scheme are liquid hydrogen-deuterium loop, hydrogen-deuterium supply system, vacuum system, helium refrigerator. CNS loop has a connection with hydrogen-deuterium supply tank of volume 3 m^3. The hydrogen system is closed and surrounded by helium blanket. Helium pressure in blanket is 0.2 bar, which can be controlled. The helium refrigerator maintains the moderator at liquid state in the loop a nominal reactor power. The refrigerator includes cold box, helium compressor and helium receiver for containing necessary volume of helium.

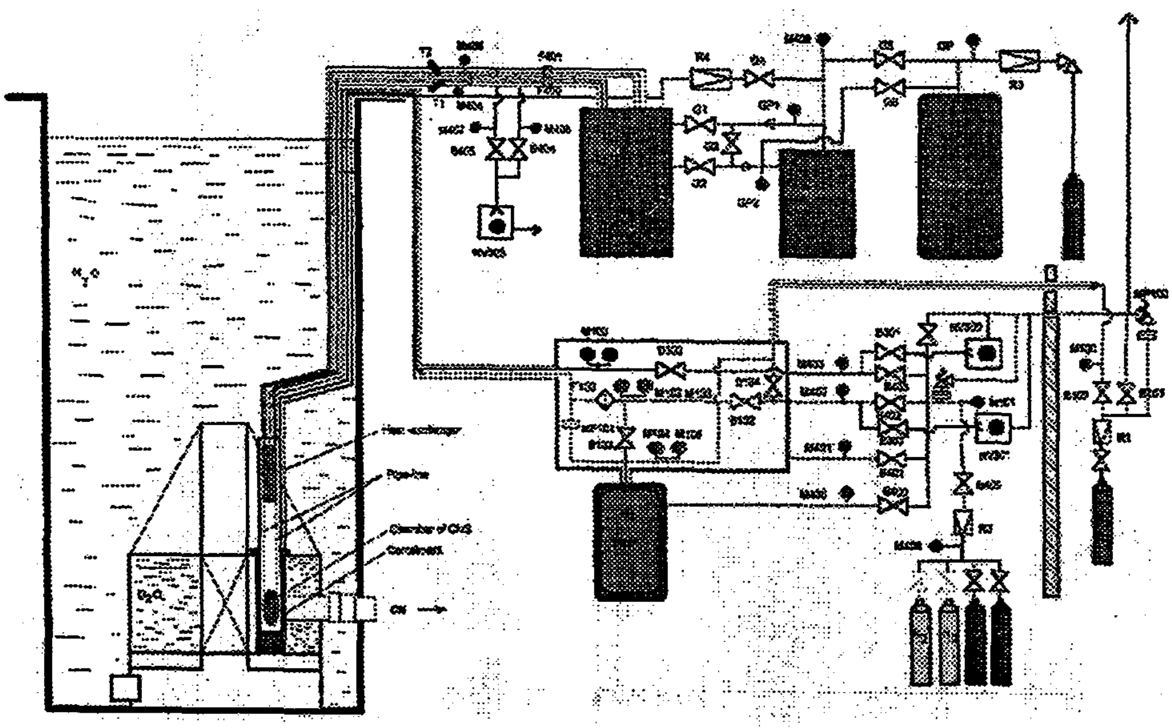

Fig 5. The general scheme for CNS at HANARO

6. Experimental instruments

The cold neutron experimental instruments at HANARO will be utilized for studies in the broad category as follows:

- Crystallography
- Chemical Physics of materials
- Magnetism and superconductivity
- Surface and interfacial studies
- Macro-molecular and microstructure studies
- Residual stress, Texture and radiography
- Analytical chemistry
- Neutron metrology and dosimetry
- Irradiation

For this purpose, about 10 experimental facilities will be installed in the CN experimental hall. The experimental facilities to be installed will comprise a 30m-SANS, a 8m-SANS, a reflectometer, 3 triple axis spectrometer, 2 time of flights, a backscattering spectrometer, a SPINS spectrometer, a prompt-gamma activation analysis spectrometer and 2 facilities for fundamental neutron physics.

7. Conclusion

Cold neutrons has been used extensively for the study of the structure and dynamics of materials in some advanced countries during the last decades or so. The cold neutron source at HANARO will be designed from the end of 1996 and the installation will be completed by the end of 2001. The research activity using cold neutron in Korea stays behind the other advanced countries. With the completion of the CNS facility, the study of the material science, solid physics, chemistry and biology etc. will be going on more vigorously centering around the HANARO. CNS facility is becoming important to develop the state-of-the-art technology by itself in many fields with the recent remarkable growth of the Korean industry. Because potential cold neutron users are recently increasing more and more in Korea, the installation of cold neutron source is essential to provide neutron beams for more users.

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COLD NEUTRON CROSS SECTIONS HISTORICAL REVIEW

presented to the 5th meeting of the
International Group on Research Reactors

D. L. Selby

November 1996

COLD NEUTRON CROSS SECTIONS
HISTORICAL REVIEW (1)

- First major slow neutron scattering data for molecular hydrogen and deuterium was published by Young and Koppel in 1964.

  - Compared with measured data these neutron scattering cross sections were good down to about 3 meV.

  - Model was based on gas model.
    - overpredicted hydrogen with respect to measured values by almost a factor of 2 at energies below 2 meV and overestimated gain factor by almost 50%.
    - overpredicted deuterium with respect to measured values by as much as a factor of 8 at very low energies (< 1 meV).

  - Cross sections have been used to develop Jülich, ISIS, and FRG-1 cold sources.

COLD NEUTRON CROSS SECTIONS HISTORICAL REVIEW (2)

- In 1977, while at ILL, Utsuro modified the Young and Koppel model for deuterium to account for the liquid state at the lower temperatures.

  - Greatly improved the agreement between measured and theoretical deuterium scattering cross sections, but still overpredicted with respect to measured values by almost a factor of 2 at very low energies.

  - The Utsuro model was used to perform the neutronic design for the upgrade of the vertical cold source at ILL and has also been used by the Research Reactor Institute at Kyoto University.

COLD NEUTRON CROSS SECTIONS HISTORICAL REVIEW (3)

- In 1988 Bernnat at IKE used a somewhat different approach to account for the liquid state to develop a scattering cross section model for ortho- and para- hydrogen and deuterium.
  - Agreement with measured data was excellent for hydrogen.
  - Although measured deuterium data was still overpredicted by about 25%, it was an improvement over previous models.
  - This model has been used for the ANS project, the FRM-2 cold source design, and the NIST cold source upgrade.

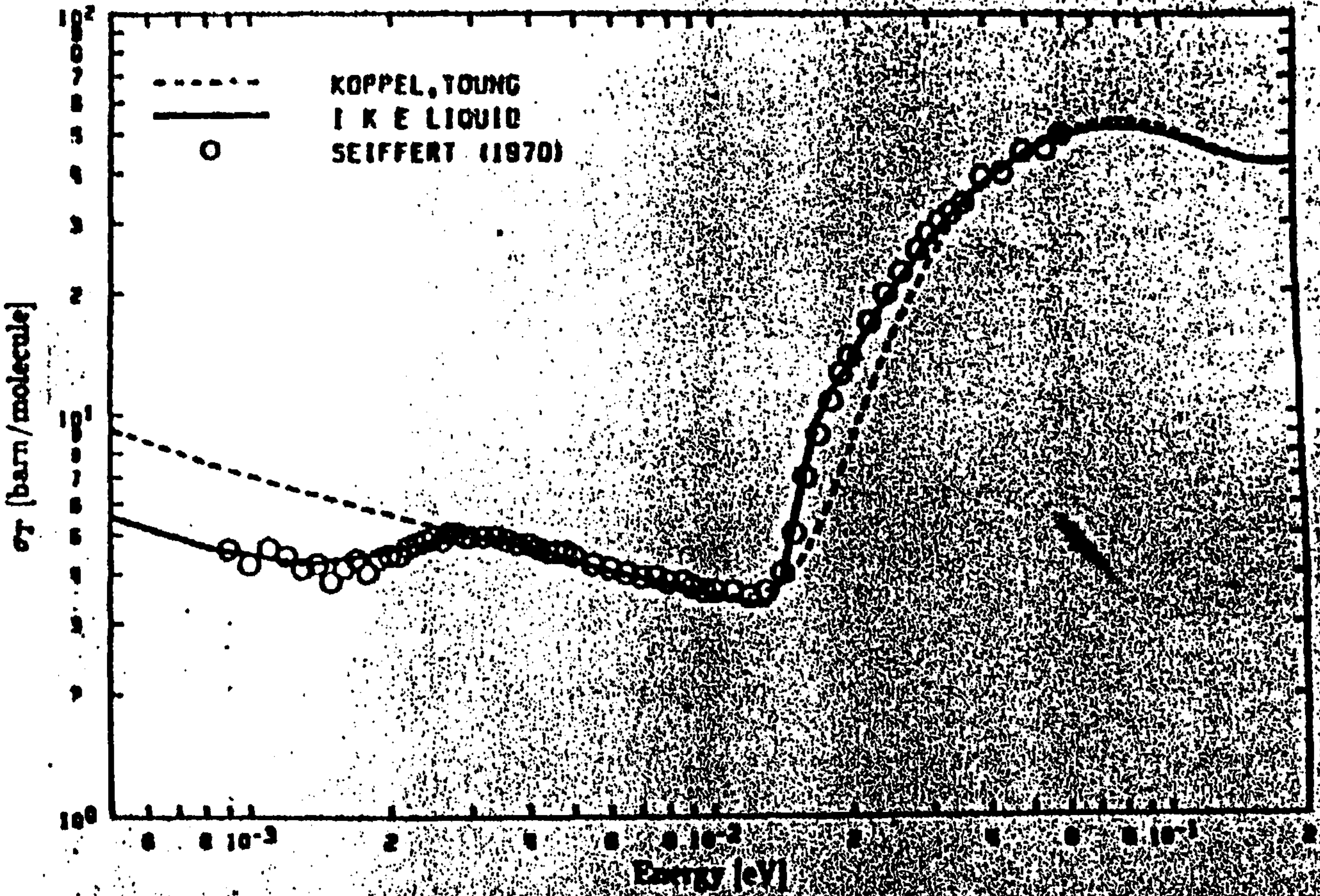

Fig. 4: Total neutron cross-sections for parahydrogen at T = 14 K

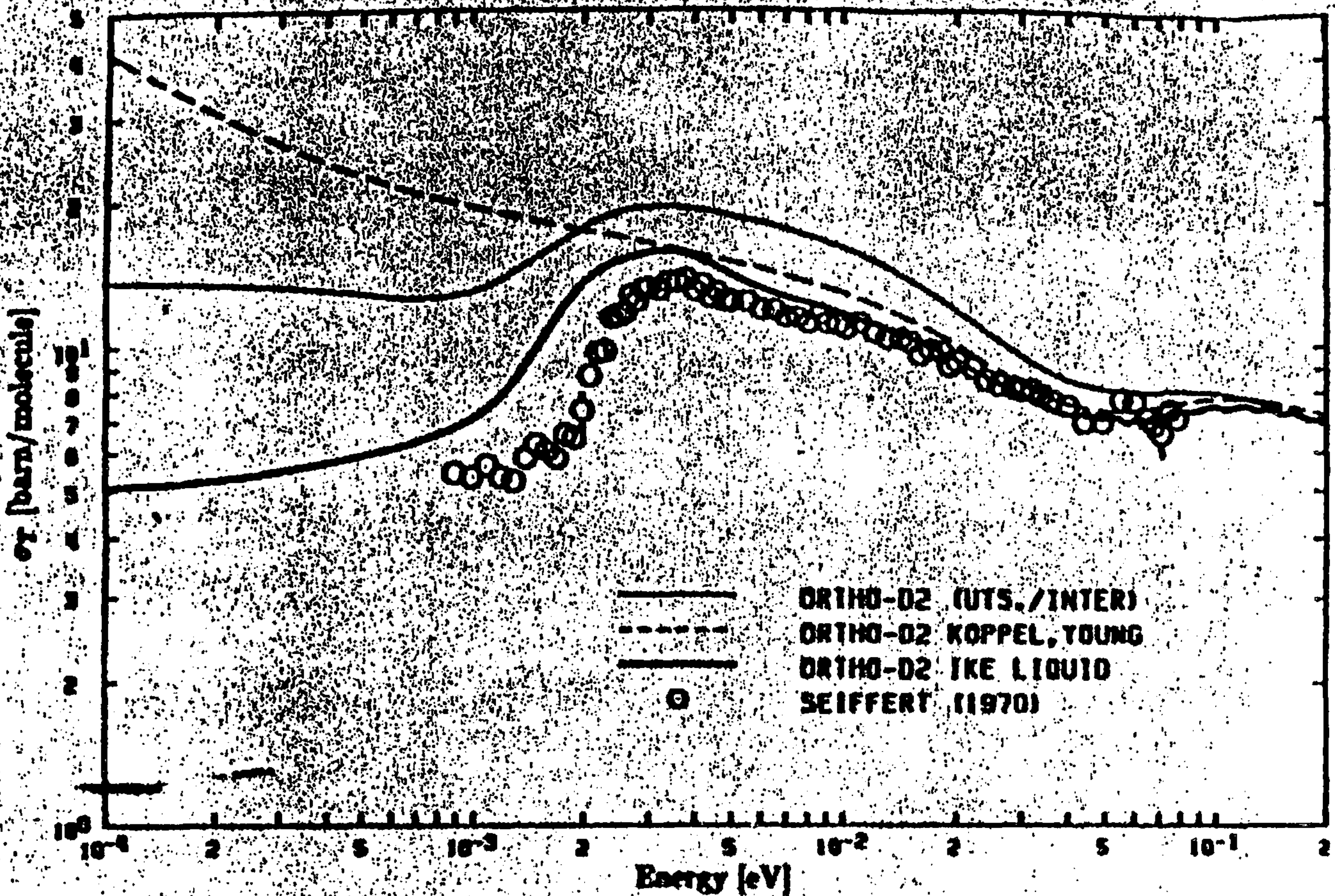

Fig. 6: Total neutron cross-sections for deuterium at T = 19 K

The GKSS CNS: THE POSSIBILITY OF NATURAL CONVECTION OF THE GASEOUS HYDROGEN MODERATOR

W.Knop, W.Krull

Abstract

The CNS is operated with hydrogen beyond the critical point (p=13.2bar and T=14.2K) where hydrogen is still gaseous. The operation values of the CNS are T~25K and p=15 bar.The supercritical gaseous hydrogen is circulated by a blower (forced circulation system) at 1 l/s through the moderator chamber and the H_2/He- heat exchanger.

However, during reactor operation we observed that the CNS remains under normal operation condition (T<35K), even if there is a failure of the H_2-blower.

This surprising observation is due to the fact, that a stable natural convection is cooling the CNS during reactor operation. A detailed study confirms the stability of the natural convection. Hence, in the future we will operate the CNS not only by forced circulation but also by natural convection.

Introduction

The FRG-1 research reactor, in operation since 1958 at 5 MW power, is upgraded and refurbished many times to follow the changing demands on safe operation and the today need for scientific research. This requires during the lifetime of the reactor many measures to follow these demands. Within the last years many additional activities have been made to overcome the ageing of the experiments, to change the experimental facilities and to increase the neutron flux and adapt the neutron spectrum to ensure good scientific utilization of the research reactor for the next 15 to 20 years.

For the kind of neutron scattering experiments at research reactors besides the area of research the neutron flux as well as the neutron spectrum is of importance. Neutron flux and neutron spectrum should be made available to users in the most optimal way to allow the effective use of the experiments. This is based on an optimal beam tube coupling to the thermal neutron flux maxima in the reflector. The thermal neutron flux can be described by a Maxwell distribution with a flux maximum at a neutron wavelength around 1.8 Å, corresponding to a moderator temperature of T=40°C (see Fig. 1).

In contrast to that, long wavelength neutrons (≥ 5 Å) are necessary for the study of larger Structure (d ≥ 5 Å) for example biological macro-molecules, polymers, voids, bubbles, creeks and clusters in the material research. As shown in fig.1 the intensity of thermal long-wavelength neutrons (≥ 5 Å) is only a fraction compared to the thermal neutron maximum. With such low fluxes, effective experiments are not possible.

For this reason, a CNS was installed in one beamtube for further slowing down of the thermal neutons, which yields a shift of the flux maximum to longer wavelength (cold neu trons).

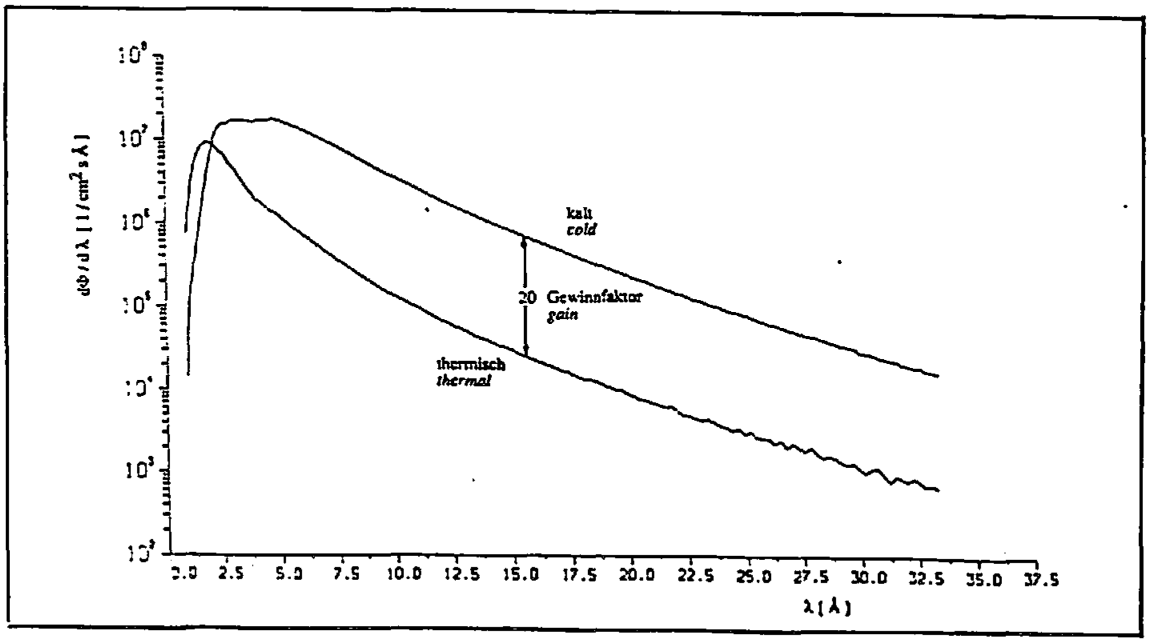

Fig. 1 Opposition of thermal and cold neutron spectrum

Description of the CNS

The CNS is operated with hydrogen beyond the critical point (p = 13.2 bar and T = 14.2K). The operation values of the CNS are T ~ 25 K and p ~ 15 bar. The supercritical gaseous hydrogen is circulated by a blower with a flow of ~1 l/s through the moderator chamber and the H_2/He heat exchanger. The nuclear heating and heat loss of about 1150 W is discharged by a helium refrigerator (see Fig. 2). Compressed helium (16 bar) is cooled by expansion in a turbine to a low pressure level, whereby the helium is cooled down to 19 K.

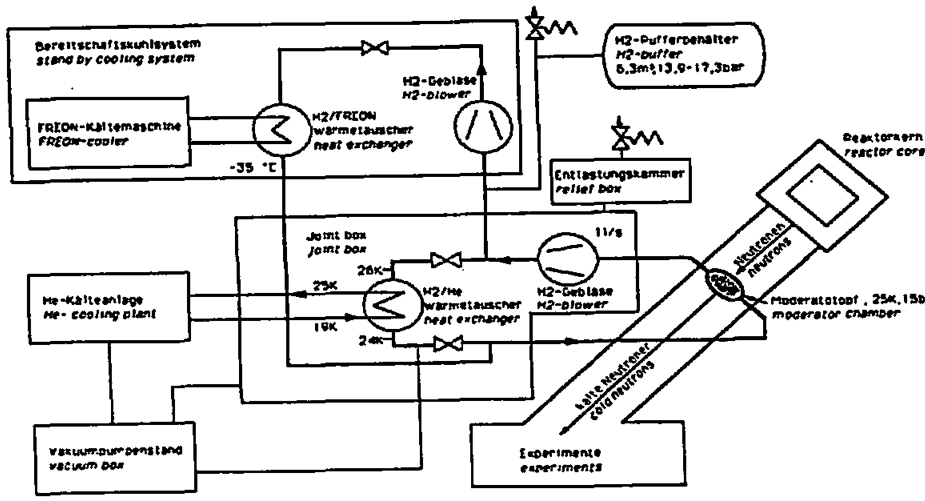

Fig.2 Simplified flow diagram of the CNS

In case the He-cooling plant is not available during operation of the FRG-1 the heat is continued to be discharged by a stand-by freon cooling system (T ~ - 35 °C and p ~ 17 bar) to allow the continuous operation of the reactor.

A H_2- buffer tank is provided in order to ensure that the hydrogen of the H_2 circuit (operating pressure 13.9 - 17.3 bar) is maintained at a pressure level about 15 bar in every operational status. With the aid of this buffer tank, fluctuation in pressure which occurs in the system as a result of change in density from warm (300 K) to cold (25 K) or the changes in the outside temperature (0.056 bar/°C) are maintained within the stated range. The buf fer tank is supplied with hydrogen with a purity level of 99.9996% from gas cylinders.

Natural convection

The maximal cooling power of the CNS is 2000 W. On the contrary, the actual entire absorbed heat power is around 1150 W and is given by following equation:

$$Q_{entire} = Q_{nuclear} + Q_{tube} + Q_{blower} = 1150\ \mathrm{W}$$

with

$Q_{nuclear}$: Nuclear heating of the moderator chamber at 5 MW ~ 900 W

Q_{tube} : Heat absorbing power of the tubing ~ 200 W

Q_{blower} : The heat from the blower ~ 50 W

A frequent failure of the CNS was caused by the breakdown of the blower as a result of bearing damage. In this case, the power from the blower increases from 50 W to 1500 W, that means, the CNS warms up. Therefore a new overcurrent switch was installed, to avoid this enormous heating from the blower, which is now negligible. During such provident switch off, we observed that the CNS remains under normal operation condition, which means a moderator temperature below 35 K (see Fig. 3). An explanation for this surprising phenomena is, that a stable natural convection is cooling the moderator chamber immediately after H2-blower failure. Typical values for the forced circulation (blower operation) and natural convection are opposed in table 1.

Table 1

Heat exchanger	Forced circulation	Natural convection
inlet temp.	~ 27K	34 K
outlet temp.	~ 25 K	21.5 K
H_2 - flow	~ 1 l/s	~ 0.15 l/s

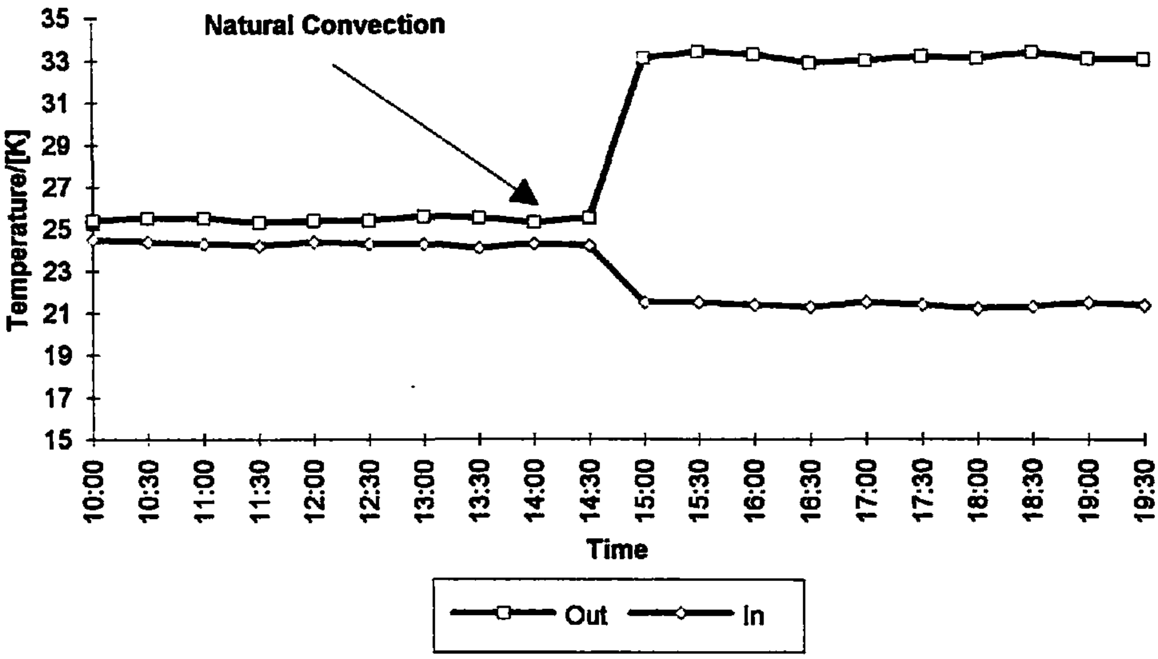

Fig. 3 Moderator inlet and outlet temperatures

The heat exchanger temperature difference $\Delta T = T(in) - T(out) = 12.5$ K increased in the case of natural convection, compared to $\Delta T = 2$ K for the forced circulation. At the same time the H_2 - flow decreased from 1 l/s to 0.15 l/s (see table 1). These values are sufficient to build up a stable natural convection for the cooling of the moderator chamber.The observed value of $\Delta T = 12.5$ K is in good agreement with the calculated temperature. Figure 4 shows the results of the temperature increasement of the heat exchanger inlet temperature for different heat absorbed power.

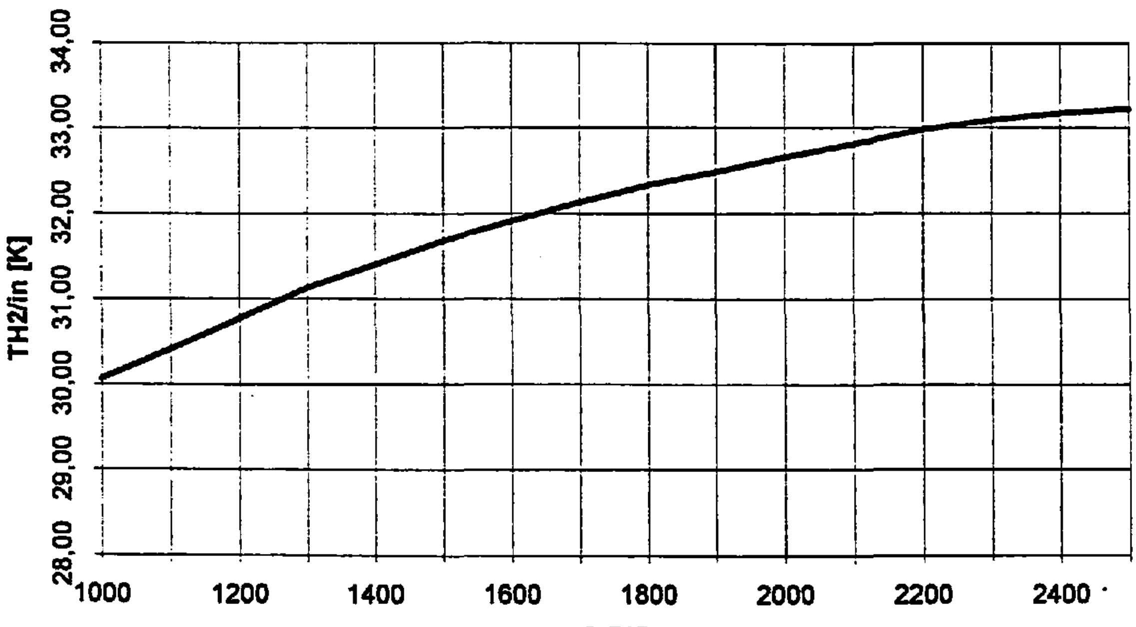

Fig. 4 Calculated heat exchanger temperatures for various heat absorbed power (T_{out}=21.5K, p=15bar)

In the case of natural convection, the driving pressure difference for the H_2 - circulation is given by:

$$\Delta p = \Delta \text{rho} * g * \Delta h$$

The difference of the altitudes between moderator chamber (source of heat) and heat exchanger is $\Delta h = 4.3$ m. Just so a great difference yields from $\Delta T = 12.5$ K of the hydrogen density to Δ rho = 38 kg/m^3 (for T_{in}=34 K --> rho = 33 kg/m^3 and for T_{out} =21.5 K --> rho = 71 kg/m^3). This value is much larger compared to the density difference of the enforced circulation with Δ rho = 5 kg/m^3.

The value of the large difference in altitude and density are the reason of the stability of the natural convection. Furthermore, the peak in the heat capacity at T = 34 K aggravates the increasement of the hydrogen temperature above 35 K, because the hydrogen absorbs a lot of heat (see fig. 5).

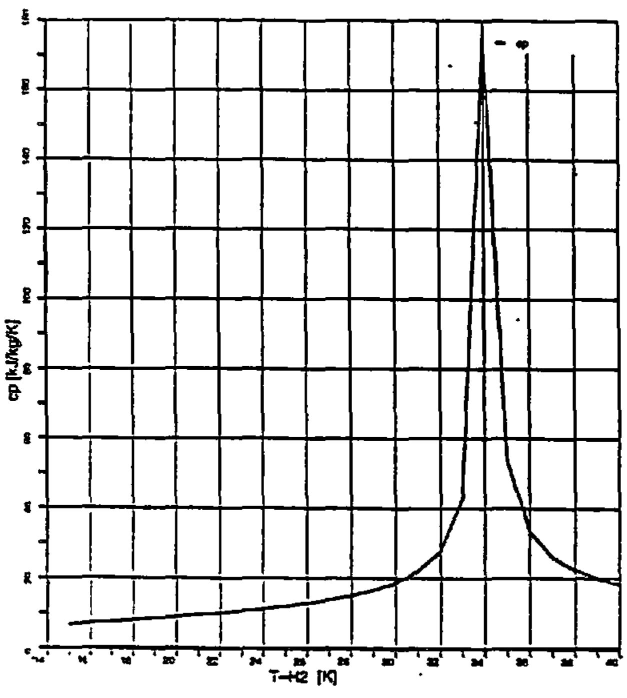

Fig. 5 Hydrogen heat capacity at p=15 bar

On the basis of this stable natural convection we will operate the CNS not only by forced circulation but also by natural convection in the future. This increases the availability of the CNS or keeps it on the high level as in the past years. This is of importance, because around 75 % of the neutron scattering instruments are installed on the cold neutron guide (see fig.6). This reflects the fact of the significance of cold neutrons for sophisticated experiments.

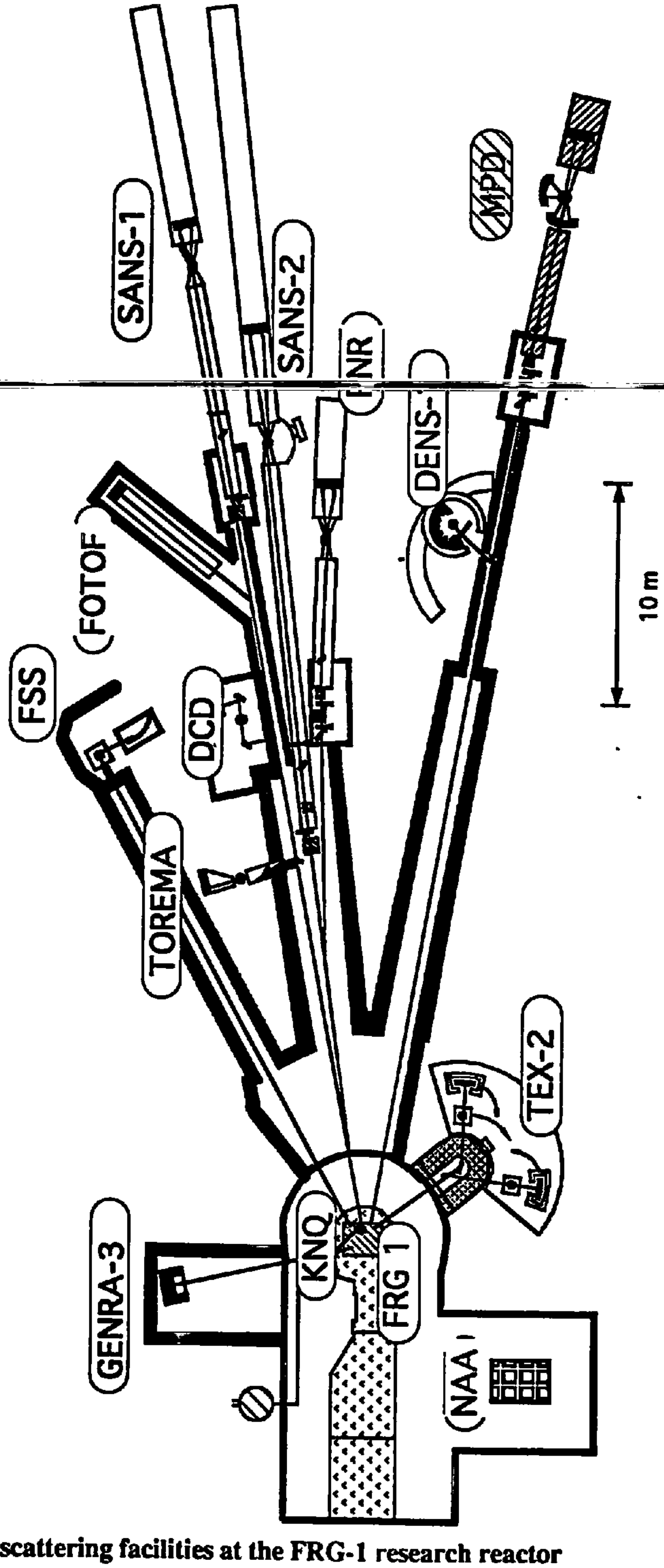

Fig. 6 Neutron scattering facilities at the FRG-1 research reactor

CHAIRMAN : HJ. ROEGLER

SESSION 3

EXPERIMENTS WITH COLD SOURCES FOR NEUTRON PHYSICS ANALYSIS
(Kir Konoplev)

Question from Bernard Farnoux of CEA :

What is the real power of the mock-up?

A :100 watts.

Q : 100 watts? And do you have facilities to monitor the content of ortho and para-hydrogen?

A : Yes.

Q : So you can measure exactly?

A : Yes, we can measure it in our existing cold source very often.

Q : So you can measure on line the quantity of ortho and para-hydrogen.

A : Yes, we can measure this. We have good staff that is experienced with this technique.

Question from Hans-Joachim Roegler of Siemens :

Q : The introduction said that you were aware of that and had an interest in that, but Mr. Selby didn't come up to now with any experiments up to now - is that foreseen for the future?

A : We do only the preliminary measurements of the background and the preliminary stage of the benchmarks that can be done. But now we cannot do it because we do not have enough financial support for it. So we propose that if anybody is looking for this kind of experiment, he's welcome.

POST IRRADIATION EXAMINATION OF Z6 NCT 25 STAINLESS STEEL FROM SF2 COLD SOURCE CELL OF THE REACTOR ORPHEE
(Maurice Mazière)

Question from Chang-Oong Choi of KAERI :

You perform mechanical testing, for tension and elongation tests just for standard steels of the moderators cells. So you don't have similar data computed with the existing mechanical data for the aluminum material?

A : No because it is not necessary because the aluminum material is well known. We did that for this special material because in the beginning it was not well known. All the data on this special material was very scarce and not reliable. For that reason the safety authorities asked us to do some tests on this material. But in the case of aluminum, we are using pure aluminum, that is well known, so there is no problem.

HFIR COLD SOURCE PROJECT (Douglas Selby)

Question from Kir Konoplev of PNPI :

For what reason the methane is better than the hydrogen? What was the density of the hydrogen atoms in the mixture that you try to use.

A : Solid methane has more hydrogen spin states, or I'm not sure what the best phrasing is on that, but it is and has been shown repeatedly - there are measurements at spallation sources in particular where they have seen this type of a difference, and we have performed MCNP analysis and got the same types of results. In fact, with a liquid methane system, that's operating at about 70°K, that's about as far down as you can go with the liquid phase, you get performance that is very good even at that temperature with the methane. The problem with the liquid methane is that it causes all sorts of gunk - I don't know a better word to phrase it - but the polymerization effects generate things that are a problem in flowing the liquid around.

Q : There is more hydrogen in methane than in liquid hydrogen ?

A : I don't know what the figure is.

Q : I think it's not only the hydrogen content, it's also the difference in the internal degrees of vibration and rotational modes.

Yes that's right. The rotational modes are different also. But you can transfer the energy to the neutrons with a thinner thickness much as you can do hydrogen over deuterium. Methane is better than hydrogen from that perspective.

Question from Jean-Luc Minguet of Technicatome :

I have one question and one comment. Do you have a gain factor for this kind of CNS in comparison with hydrogen-one for instance? And my comment is, to have something reliable, we have to make it simple.

A : Yes. I agree certainly with the last point. These kinds of systems, if they were to become usable systems would have to be very simple. And it is very clear that this type of system would not work in a high flux or high power reactor. Particularly in a spallation source, where you have maybe more room to put such a device, we believe that one could make this fairly simple and work. The next stage would be transporting these in the fluid.
In terms of gain factors, it of course depends on the neutron wavelength, but in general what we've seen both at the spallation source at Argon National Laboratory as well in MCNP analysis is that you get somewhere between a factor of 2.7 and 3 increase in gain factor over hydrogen.

Question from Guy Gistau of Air Liquide :

In case of fusion research, they are making solid hydrogen pellets. Maybe you could talk with them - there could be some similarity with your problem.

A : Yes, in fact it's the fusion people that we've had doing this work. It's the fusion division of the laboratory that has been generating the pellets for us.

PERFORMANCE OF THE NIST LIQUID HYDROGEN COLD SOURCE
(Robert Williams)

Question from Johannes Wolters of Jülich Research Center :

Mr. Williams, do you have an idea why not all the ortho is transferred into para-hydrogen?

A : Well, there is a great deal of dissociation of hydrogen molecules and as they recombine, they may recombine at three-to-one ortho. There may also be - it's very complicated. In the liquid, there is a buildup of hydrogen 3 and hydrogen 1 molecules and ions, it's quite complicated how they interact. I'm not entirely sure. Also, our system is open to the ballast tank and the warm hydrogen. There may be migration of normal hydrogen that way. There are pressure fluctuations. Our pressure gauges are at the ballast tanks and they go up and down a kiloPascal.

DESIGN REVIEW IN CONCEPT PHASE OF CNS AT HANARO
(Chang-oong Choi)

Question from Klaus Gobrecht of TU München :

Can you tell me what cross-section library you have used for the deuterium - hydrogen mixtures.

A : I'm sorry I can't give you such details. I'm sorry - I didn't consider this.

Q : You have never thought about it?

A : Just now in Hanaro, I consider how to get the maximum gain factor and deduce the best facility within the limited space. This is my main concern.

Q : How do you avoid demixing, because you have a continuous distillation of the mixture, so the vapor has a different concentration of hydrogen than the liquid. How do you avoid this demixing?

A : Because we didn't have any real experience with that, we had to further discuss with advanced technology experts of countries like France and other countries, so it is possible. This is our concept, just to have good gain factors and to install in limited space. This is my main concern, so how to keep liquid status is something we have to study further.

Question from Der-Jhy Shieh of INER :

How many cold neutron guide tubes are you going to install with this CNS ?

A : Presently two channels are possible to be installed but we are looking to install a third channel. It's a possibility

Question from Kir Konoplev of PNPI :

What is the main reason for 50/50 H_2/D_2?

A : No, no! This is not a fixed figure, but 60 - 40 depending on the study results. This is just to quickly estimate the heat load - to make it simple. This the pre-conception stage. I didn't have a detailed study of engineering work. Still, I'm gathering the necessary details for that.

Question from Bernard Farnoux of CEA :

What is the size of the cell? 14 cm in diameter?

A : The bottom hole inside diameter is 16 cm, so if I consider some containment vessel.

Q : But the cell is full, there is no hollow in the middle, so it's the same as if we had 14 cm of hydrogen?

A : Yes, it is too much.

Q : So you have to put deuterium inside. You have to put deuterium ions, 70% or something like that. You have to reduce the amount of hydrogen.

A : OK

Remark from Hans-Joachim Röegler of Siemens :

May I make one remark? Anyone who is not happy with the answer he has given here to the audience may of course in the proceedings modify his answer to give a better body to the expression. So everybody who has given a contribution has the chance to remodel his answer.

A : Thank you!

COLD NEUTRON CROSS SECTIONS (Douglas Selby)

Question from Albert Lee of AECL :

What generations of cold scattering kernels are being used at ORNL? It is my experience that there have been several cross section libraries and several revisions of cross section processing with revisions of NJOY at LANL that produce different results. Users of MCNP should be aware of the potential for misleading results if the wrong cross section library is used.

Douglas Selby : Bob and I have been having a number of discussions on this in the last few months. I guess one might say they're taking the Monte Carlo approach to developing the cross sections. Sooner or later they'll hit it correctly at Los Alamos. All of those evaluations, all of the MCNP data sets right now as I understand it are based on the IKE data and the data itself is still the same - that has not changed. What has changed is the processing approach that they have used in developing them. The newest version, which is not available in general right now - but Bob has a copy and that's what he did his analysis of, and I now have a copy and we're redoing some analyses based on that - is supposed to have additional expansion terms and a substantial increase in the number of angles that are evaluated in the generation of the actual MCNP cross sections. And so it is the belief that this newest data set that has just been this year, I don't know - when did you first have access to it, Bob (Williams)?

I first had access to it in August, late in August - sort of an informal contact with Gary Russel. But it has not been formally released by Los Alamos.

Right!

Albert, you have been using scattering kernels from Los Alamos, but when did you get them?

Albert Lee : The scattering kernels we got from Los Alamos are about two years old at the moment I'll caution people because we have various intermediate pre-release versions of NJOY from Los Alamos and we have discovered that sometimes the compilation of NJOY is hardware dependent. So we have compiled the same version of NJOY on two different *UNIX* machines and got two different sets of answers when we processed the same test set.

That is not encouraging.

Douglas Selby : No!

Albert Lee : Well, the data that Bob has, he seems to be very happy with now. Is that a fair statement, Bob?

Douglas Selby : Of the three sets that I had, that had the best agreement with the measurements expected.

That's about all I can say about it.

I think the geometry will also play a role here in terms of how much variation you have between the sets. We're getting ready to rerun our geometry with the new set and one of the things I wanted to do with Bob is to compare the differences we got in our geometry versus his geometry to see the impact there. And I don't guess we really have a full answer to your question, Albert, except that I believe the newer is better.

Albert Lee : It's not just raising a question. All I'm trying to do is to raise a warning flag to everybody that does any modeling with the MCNP that one has to be careful about the conclusions one draws depending on the generation of the cold scattering kernels and what year it was generated, because the experience I've had with Los Alamos, and I recognize they're doing the best they can, is that there have been some random errors that have crept into some of the coding in the processing system, so it's just warning everybody to be careful about they use the cross sections and how they interpret the results.

Douglas Selby : The previous evaluation, that I believe is the one that you had, is based on about a 4-year old evaluation and I believe that it predicts about a 40% higher - is that what you'd say, Bob.

Robert Williams : I'd say a bit less - about 30.

Chairman (Hans-Joachim Röegler):

One remark: historical evaluation of the projects are from the '80s or so. The FRG1 and BR2 as far as I remember, were also calculated with KOPPEL and YOUNG model because at that time we were already in discussions with Mr. Donner who worked on that project, so it was not pure KOPPEL and YOUNG model according to my memory at that time, but I tried to figure that out looking at the old papers and send you something about that.

Yes, I would appreciate that. The information that was sent directly to me said straight YOUNG and KOPPEL.

This is not true. I am sure that this is not true because I remember that we had contact through Stuttgart and that said a lot because that didn't fit with the experimental results and we approved that model.

THE GKSS CNS: THE POSSIBILITY OF NATURAL CONVECTION OF THE GASEOUS HYDROGEN MODERATOR (Wolfgang Knop)

Question from Jean-Luc Minguet of Technicatome :

Yes, I just have a question about the gain factor when you operate the loop with free convection.

A : Yes, there was also expression if you have some shift, and I asked all the experiments in the hall which measured at different wavelengths : 5, 10, 8 or 4 angstrom, and there was no shift in the intensity. That means that the distribution is the same.

Question from Klaus Gobrecht of TU München :

Did you observe spontaneous onset of the natural convection immediately before starting the blower or when starting the refrigerator?

A : Without the reactor we need the blower. We have to cool down the CNS with the blower and then when the reactor is in operation we can switch off the blower. Our reactor cycles are six weeks. Then that means we can switch off the blower and wait for the end of the reactor power, and then we get signal of the N16 which automatically switch on the blower. And there it is. Instead of 280 days per year of blower operation, we only need maybe 10 days. And there it is.

Question from Hans-Joachim Röegler of Siemens :

And do you pay Siemens back for that good design?

A : I don't want to comment, because it's still a question why we still have such power supply as before, to 1.5 kilowatts!

Question from Johannes Wolters of Jülich Research Center :

You told me that you intend to increase the power density of the reactor while reducing fuel elements core. Is natural convection still adequate under these conditions?

A : Yes, Siemens has also taken this into account and now we have around 1100 watts and maybe in the future it will be stable up to 2000 watts, more than we expected.

IGORR 5

SESSION 4

WORKSHOP ON CONTAINMENT SURVEY

PAPERS

RESULTS OF A SURVEY ON THE DESIGN BASIS FOR RESEARCH REACTOR CONTAINMENT/CONFINEMENT BUILDINGS

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1. INTRODUCTION

During the "Workshop on R&D Needs" at the 4th Meeting of the International Group on Research Reactors (IGORR-IV), the participants agreed that it would be useful to compile a survey of the design basis for containment and confinement features in research reactor buildings. The following organizations submitted information for this survey:

- CEN/SCK (Belgium): BR-2
- CEA (Commissariat a l'Energie Atomique, France): ORPHEE
- Technicatome (France): SIRIUS-2
- BATAN (National Atomic Energy Agency, Indonesia): RSG-GAS-30
- JAERI (Japan Atomic Energy Research Institute): JRR-2, JRR-3M, JRR-4, JMTR and NSRR
- KAERI (Korea Atomic Energy Research Institute): HANARO
- NIST (National Institute of Science and Technology, USA): NBSR
- ORNL (Oak Ridge National Laboratory, USA: ANS
- BNL (Brookhaven National Laboratory, USA): HFBR
- KFA Julich (Germany): FRJ-2

The information on NSRR was not included in the analysis of the survey responses because the operating power and fission product inventory in the core are comparatively low.

1.1 Design Considerations for Research Reactor Buildings

A reactor building generally encompasses the building's structures, ventilation systems, penetrations and any feature that is important for an engineered safety function. The design of a reactor building takes into account all operational states. The design considerations for a reactor building generally include:

- pressure and temperature loads for normal operating conditions,
- pressure and temperature loads expected during conditions for design basis accidents,
- leakage rates at design pressure and a test schedule to verify the leakage rate,
- extreme loads from internal hazards, e.g., fires, explosions and reactor experiment malfunctions,
- extreme loads from external hazards, e.g., earthquakes, aircraft crashes, tornadoes and tornado missiles,

- extreme loads from design basis accidents, e.g., rapid insertion of excess reactivity or loss of flow, and
- maintaining radiation levels and releases on and off-site consistent with the ALARA principle and below prescribed limits.

Reactor buildings have been designed to provide either a confinement or containment function. The choice of confinement or containment is based on the reactor facility design, operating characteristics, accident scenarios and location. In this paper the terms confinement and containment are defined as follows:

- **confinement**: Confinement systems control the airflow through the reactor building and release the reactor building air in a controlled manner at a location that allows for dilution and diffusion of the radioactive material before it comes in contact with the public. Confinement systems prevent an uncontrolled release to the environment of radioactive effluents resulting from operation by a system of ducts, louvers, blowers, exhaust vents, or stacks.

- **containment**: Containment systems are designed to prevent the rapid, uncontrolled release of radioactive material to the environment. The containment is designed to control the release to the environment of airborne radioactive material released in the reactor building even if the accident is accompanied by a pressure surge or a steam release within the building. The thick walls of the containment may also help mitigate direct radiation exposure during certain accidents. The design bases for the containment include the postulated peak pressures, the duration of the event, the pressure-versus-time envelope, the time during which containment integrity must be maintained while recovery from the event is implemented, limits on leakage or controlled release from the containment to the environment, the quantity of failed fuel, and the quantity and type of released radioactive material.

2. <u>CONTAINMENT/CONFINEMENT INFORMATION</u>

2.1 Survey Questions

The survey on containment and confinement design information solicited the following information:

A) Design basis parameters
- pressure and temperature loads for normal operating conditions
- pressure and temperature loads expected during conditions for design basis accidents
- leakage rates at design pressure
- requirements and frequency of testing done to verify the leakage rate

B) Design basis accidents
- postulated initiating events considered for containment/confinement system (e.g., internal fires or explosions, reactor experiment malfunctions, earthquake, aircraft crash, tornadoes and tornado missiles, insertion of excess reactivity, loss of flow, etc.)

C) Factors included in the design of the containment wall (e.g., shielding, internal or external explosion, tornado missile, earthquake, aircraft crash, etc.)

D) Features provided for personnel and equipment access
- type of personnel access (e.g., dual airlock, simple doors)
- type of access for large and small pieces of equipment

E) Requirements and features for controlling releases of radioactivity

F) Special design features
- hydrogen igniters to prevent hydrogen explosion
- provisions for accidents beyond design basis
- other special design features

2.2 Responses to the survey

The detailed responses to the survey are compiled in the following sections.

2.2.1 Design Basis Parameters

The design basis parameters (i.e., operating temperature and pressure, design basis temperature and pressure, and leakage rates) for the reactor buildings are listed in Table 1.

2.2.2 Design Basis Accidents

The design basis accidents considered in the design of the reactor building are:

- **ANS**: Full spectrum of events postulated but analysis not completed.
- **BR-2**: The design basis accident assumes melting of 40 fuel elements with high burn-up at the end of a cycle of 30 days at 100 MW with the accompanying assumption that all the aluminum in the fuel region can react with water, producing hydrogen which subsequently burns. In addition 50 kg of sodium is assumed to burn in the air of the containment building. This postulated accident results in a maximum internal pressure of 1.0 bar.
- **FRJ-2**: The postulated initiating event is an uncontrolled reactivity insertion of about 3% $\Delta k/k$. It is assumed that 60% of the metal in the core would react with the D_2O forming deuterium gas, which escapes into the containment via a leak in the cover gas system and which completely burns there.
- **HANARO**: The design basis accidents were not provided in the response to the survey.
- **HFBR**: The postulated initiating events include the design basis earthquake, tornadoes and tornado missiles and the accidents in 10CFR100 Siting Criteria.

- **JRR-2**: Two design basis events, a loss of heavy water coolant event and a fuel failure event, are considered.
- **JRR-3M**: A coolant channel blockage event is the only accident considered in the design of the containment building.
- **JMTR**: No design basis events whose initial events are related to failures of the reactor building are considered in the JMTR safety assessment.
- **JRR-4**: A fuel handling accident and a coolant flow blockage event are considered.
- **NBSR**: The design basis accident is a postulated blockage of coolant to one fuel element resulting in the complete melting of the aluminum fuel clad, allowing fission products into the primary coolant system. It is further assumed that the primary boundary also fails, releasing 100% of the noble gases and 50% of the iodine, and the absolute filters function at only 95% efficiency. Using weather conditions specified in the 1973 AEC (Atomic Energy Commission, the predecessor to the Nuclear Regulatory Commission) Regulatory Guide, an individual remaining at the site boundary 24 hours a day for 30 days would receive a gamma ray dose of 0.17 rad and an iodine dose to the thyroid of 1.2 rem.
- **ORPHEE**: A BORAX-type reactivity insertion event with 135 MJ energy release is considered. About 9% of the energy is assumed to be converted to mechanical energy. A part of the mechanical energy is supposed to deform the pool containment and the remainder blows water out of the pool.
- **RSG-GAS-30**: No design basis accidents were provided in the response to the survey.
- **SIRIUS-2**: A BORAX-type reactivity insertion event with 135 MJ energy release is considered. A complete core melt under water is assumed.

The beyond design basis accidents considered in the design of the reactor building are:

- **ANS**: Beyond design basis accidents were planned to be considered as part of the probabilistic risk assessment.
- **BR-2**: A "general emergency plan" with an on-site co-ordination cell for acting as a command centre is used to deal with accidents beyond the design basis.
- **FRJ-2**: None.
- **HANARO**: None.
- **HFBR**: The provisions for beyond design basis accidents are based on security accident analysis and a probabilistic safety assessment.
- **JRR-2**: None.
- **JRR-3M**: A fuel failure accident followed by release of radioactivity is considered to be the accident beyond the design basis.
- **JMTR**: None.
- **JRR-4**: None.
- **NBSR**: None.
- **ORPHEE**: None.
- **RSG-GAS-30**: None.
- **SIRIUS-2**: None.

2.2.3 Factors Included In The Design Of The Containment Wall

Reactor buildings are designed to take into account the effects of extreme loadings and environmental conditions due to accidents, including those arising from internal and external events.

Seismic events often impose significant constraints on the design of reactor buildings. The survey obtained the following responses for seismic considerations in the design of the reactor buildings:

- **ANS**: The design requirements for the containment building were based on the requirements in the US NRC Safe Shutdown Earthquake guideline.
- **BR-2**: A horizontal ground acceleration of 0.1 g is being used in a seismic analysis as part of the BR-2 refurbishment program.
- **FRJ-2**: The horizontal ground acceleration for the design basis earthquake is 0.22 g.
- **HANARO**: The horizontal ground acceleration for the design basis earthquake is 0.2 g.
- **HFBR**: Seismic design requirements were included in the design of the confinement building.
- **JRR-2**: The concrete containment structure is designed to meet the seismic requirements for the Tokai site.
- **JRR-3M**: The concrete containment structure is designed to meet the seismic requirements for the Tokai site.
- **JMTR**: The reactor building is designed to meet the seismic requirements for the Oarai site.
- **JRR-4**: The reactor building is designed to meet the seismic requirements for the Tokai site. The seismic design of the reactor building is being re-examined as part of the JRR-4 modification program.
- **NBSR**: The reactor building is designed to meet normal building code requirements. The reactor site is not in a seismic zone.
- **ORPHEE**: The concrete containment structure is designed to meet the seismic requirements for the Saclay site.
- **RSG-GAS-30**: The reactor building and civil structure is designed to meet the seismic requirements for the Serpong site.
- **SIRIUS-2**: The seismic requirements are site dependent. The design basis is SMHV+1 on the MKS scale.

Shielding requirements are often a significant consideration in specifying the thickness of reactor building walls. The survey obtained the following responses for shielding considerations:

- **ANS**: Not yet evaluated.
- **BR-2**: Shielding was not considered in the design of the building.
- **FRJ-2**: Shielding was not considered in the design of the building.
- **HANARO**: Shielding was not considered in the design of the reactor building.
- **HFBR**: No shielding requirements were provided in the response to the survey.
- **JRR-2**: The concrete structure is designed to meet shielding requirements for the reactor.
- **JRR-3M**: The concrete structure is designed to meet shielding requirements for the reactor.
- **JMTR**: The reactor building is designed to meet shielding requirements for the reactor.

- **JRR-4**: None.
- **NBSR**: None, the shielding requirements are designed into the experimental equipment and the shielding for the reactor vessel.
- **ORPHEE**: No shielding requirements were provided in the response to the survey.
- **RSG-GAS-30**: No shielding requirements were provided in the response to the survey.
- **SIRIUS-2**: No shielding requirements were provided in the response to the survey.

Internal or external explosions are sometimes considered in specifying the structural strength for reactor building walls. The survey obtained the following responses for internal or external explosions:

- **ANS**: Not yet evaluated.
- **BR-2**: None specified in response to the survey.
- **FRJ-2**: External explosions are not considered to be possible due to the distance of potential gas sources from the reactor building.
- **HANARO**: None.
- **HFBR**: The siting requirements in 10CFR100 Siting Criteria accident are considered in the design of the confinement building.
- **JRR-2**: None specified in response to the survey.
- **JRR-3M**: None specified in response to the survey.
- **JMTR**: None specified in response to the survey.
- **JRR-4**: Outside human events and internal missiles are considered in the design of the reactor building walls.
- **NBSR**: None.
- **ORPHEE**: External explosions from a gas pipeline located 300 m from the reactor building are considered.
- **RSG-GAS-30**: Internal and external explosion events are not considered because the probability for these events are considered to be very low.
- **SIRIUS-2**: The requirement for considering internal and external explosion hazards is site and client dependent.

Wind driven (e.g., tornado) missiles are sometimes considered in specifying the structural strength for reactor building walls. The survey obtained the following responses for wind driven missiles:

- **ANS**: The design of the containment structure was not completed. The containment structure would be designed as needed to meet US NRC tornado design requirements.
- **BR-2**: The bottom of the steel shell is anchored to the foundations in order to resist the uplift of the roof caused by internal positive pressure and the overturning effect due to lateral wind forces.
- **FRJ-2**: Not considered.
- **HANARO**: Typhoons with wind velocities of 20 m/s are considered.
- **HFBR**: Tornadoes and wind driven missiles are considered.
- **JRR-2**: None specified in response to the survey.
- **JRR-3M**: None specified in response to the survey.

- **JMTR**: Tornadoes and wind driven missiles were considered in the siting criteria for the Oarai site.
- **JRR-4**: None.
- **NBSR**: The confinement building is designed to meet the normal structural requirements of wind and snow loadings.
- **ORPHEE**: None.
- **RSG-GAS-30**: None.
- **SIRIUS-2**: The need to consider wind driven missiles depends on the local siting criteria for the reactor.

Aircraft crashes are sometimes considered in specifying the structural strength for the roofs of the reactor buildings. The survey obtained the following responses for aircraft crashes:

- **ANS**: Not yet evaluated.
- **BR-2**: None specified in response to the survey.
- **FRJ-2**: Aircraft crashes were not a requirement at the time the reactor was built.
- **HANARO**: None.
- **HFBR**: None.
- **JRR-2**: None specified in response to the survey.
- **JRR-3M**: None specified in response to the survey.
- **JMTR**: Aircraft crashes were considered in the siting criteria for the Oarai site.
- **JRR-4**: None specified in response to the survey.
- **NBSR**: None.
- **ORPHEE**: Aircraft crashes are considered.
- **RSG-GAS-30**: None.
- **SIRIUS-2**: The requirement for considering aircraft crashes depends on the type of aircraft and the results of a probabilistic safety assessment.

2.2.4 Features Provided For Personnel And Equipment Access

- **ANS**: A dual door airlock with a refuge area is included in the design for personnel to ensure meeting the life safety code requirements under all conditions. Equipment hatches, including one with a dual airlock, are included in the design for large equipment access.
- **BR-2**: A dual door airlock is provided for personnel and small equipment. An emergency escape dual door airlock is connected to the main stair well. A large double door airlock for trucks is located at ground level.
- **FRJ-2**: There are two personnel airlocks, one for routine access and the other for emergency escape. An airlock for lorry access is provided for movement of large pieces of equipment. A transfer chute permits transfer of fuel elements, experiments and other radioactive materials from the pool into a storage and dispatching pit in the adjacent service building.
- **HANARO**: A dual door airlock is provided for personnel access. A single airtight truck door is provided for movement of large equipment.
- **HFBR**: There are two dual door personnel airlocks. There are two dual door truck airlocks.
- **JRR-2**: A dual door airlock is provided for personnel access. A truck dual door airlock is provided for movement of large equipment.

- **JRR-3M**: A dual door airlock is provided for personnel and small equipment access. Large-sized airtight and simple doors are provided for large pieces of equipment. Only one door at a time can be opened to preserve the negative pressure in the containment.
- **JMTR**: Airtight doors are provided for personnel and truck accesses.
- **JRR-4**: A personnel door is provided. A loading dock is provided for large equipment access.
- **NBSR**: From the laboratory building, there are two access corridors with two sets of doors, each to facilitate ventilation control. A third door permits access to an elevator at the basement level of the laboratory building. All three are closed and sealed with inflatable gaskets in the event of an emergency. In addition, a manually operated fire exit, and a large truck door, are normally closed and sealed. The truck door must be sealed during operation and refueling.
- **ORPHEE**: A dual door airlock is provided for personnel access. A large dual door airlock is provided for movement of large equipment such as flasks.
- **RSG-GAS-30**: A dual door airlock is provided for personnel access. Large and small equipment are moved from the operations level (elevation 13 m) to the bottom level (elevation 0 m) through an equipment hatch and placed on a truck in the truck bay.
- **SIRIUS-2**: A dual door airlock is provided for personnel access. A large dual door airlock is provided for movement of large equipment.

2.2.5 Requirements And Features For Controlling Releases Of Radioactivity

- **ANS**: The ANS design featured a low leakage primary containment with isolation mode to be initiated on detected radiation release or high radiation levels. The confinement building is maintained at a slight sub-atmospheric pressure by blowers whose effluent is vented to the environment via HEPA and charcoal filters.
- **BR-2**: The requirements are found in the BR-2 Technical Specifications.
- **FRJ-2**: An accident filter system is provided and it consists of two trains of fiber and charcoal filters in series. It can be used either to control the containment pressure or to clean the containment atmosphere. It has a very limited capacity for retaining removable fission products due to the hazard of filter burning. The exhaust system for normal operation has fiber filters and the containment is automatically closed when the radiation level in the filters exceeds the trip limits.
- **HANARO**: The radioactivity is confined in the reactor building by closing the ventilation system when high radioactivity is detected in the ventilation system.
- **HFBR**: The building is exhausted to a 108 m tall stack via charcoal and HEPA filters which have specified minimum iodine removal efficiencies. The confinement is isolated when general area dose rates exceeding 5 rem/h are detected.
- **JRR-2**: Radioactive releases are controlled by an emergency airtight damper based on a water seal system and HEPA filters on the exhaust ventilation flow.
- **JRR-3M**: Gaseous radioactivity is released through a stack after lowering its concentration. An emergency ventilation system, consisting of blowers and filters, is provided to prevent releases of radioactivity to the atmosphere in the case of fuel failure accidents.

- **JMTR**: Radioactive releases are controlled by charcoal filters and an emergency exhaust system. It is not necessary to seal the building because the emergency exhaust system can maintain a negative internal pressure.
- **JRR-4**: Radioactive releases are controlled by HEPA filters and an emergency exhaust system.
- **NBSR**: If a radiological release occurs inside the confinement building automatic emergency actions are initiated to close and seal the doors, to secure the normal ventilation, and to exhaust the building through 99% absolute filters and charcoal filters in the emergency ventilation system.
- **ORPHEE**: Releases of radioactivity are controlled by the ventilation flow out of the building. Gas and dust are monitored for alpha, beta and gamma radiation by collecting on paper filters and monitoring.
- **RSG-GAS-30**: Releases of radioactivity are controlled by the ventilation flow out of the building to the adjacent stack.
- **SIRIUS-2**: Releases of radioactivity are controlled by absolute filters during normal operations and iodine filters on the emergency exhaust ventilation system during emergencies.

2.2.6 Special Design Features

The following additional design features for the reactor buildings were identified in the survey responses:

- **ANS**: A catalytic converter was being designed for the cold source equipment room ventilation system. Hydrogen igniters or similar devices were being considered for two relatively small rooms (i.e., the sub-pile room and the room housing the letdown tank).
- **BR-2**: A "general emergency plan" with an on-site co-ordination cell for acting as a command centre is used to deal with accidents beyond the design basis.
- **FRJ-2**: None.
- **HANARO**: None.
- **HFBR**: None.
- **JRR-2**: None.
- **JRR-3M**: None.
- **JMTR**: None.
- **JRR-4**: None.
- **NBSR**: None.
- **ORPHEE**: None.
- **RSG-GAS-30**: None.
- **SIRIUS-2**: The penetrations (i.e., pipes, air ducts and electrical cables) are grouped in order to direct the majority of the potential leaks to a special compartment located outside of the reactor building. All penetrations are doubly isolated, inside and outside of the reactor building. It should be possible to operate the isolation equipment located outside of the reactor building. An emergency control room is located outside of the reactor building from where the isolation of the reactor building can be controlled.

3. SUMMARY

This survey of the design basis for containment and confinement features in research reactor buildings has been compiled to provide the international research reactor community with an overview of the historical approaches to defining the design requirement. From the responses to the survey it is seen that it is often difficult to clearly associate the terms "containment" and "confinement" with each of the research reactor buildings.

From the responses to the survey, three approaches have been taken to define the requirements for the design of the reactor building:

1. **"Bounding" scenario**: This historical approach relies on defining a "severe" accident, such as the energy released from a fast reactivity insertion event, the so-called Borax type event, to establish the extreme loads on the reactor building. The energy released in this event is assumed to produce a vapour explosion, originating from the rapid passage to the water, of energy stored in the fuel plates. The vapour explosion is assumed to be accompanied by a chemical reaction of molten metal in water. Further details of the BORAX type approach is found in reference 2.

2. **Explicit design for "severe" accidents**: This approach involves choosing a design pressure to adequately cover a high probability severe accident that has been identified through a probabilistic risk assessment. The requirements for the reactor building are then specified to accommodate the design pressure and the consequences for the identified severe accident.

3. **Design for higher probability events**: In this approach the containment structure is designed to meet the structural load requirements for a range of higher probability events (e.g., tornadoes and wind driven missiles, earthquakes, and expected internal pressure capability) and other requirements (e.g., volume of containment to house equipment, leak rates and test pressures). Then the containment structure is post evaluated to determine its ability to accommodate events beyond the design basis.

The choice of an approach to defining the design requirements for the reactor building from the design basis accidents has been strongly influenced by past practice and by what is considered acceptable to the local regulatory authority. The "bounding" scenario approach has been widely used:

- **BR-2**: large scale fuel melt.
- **FRJ-2**: reactivity insertion event but less severe than BORAX.
- **JRR-2**: loss of heavy water coolant event and a fuel failure event.
- **JRR-3M**: single fuel channel blockage event.
- **JRR-4**: fuel handling accident and a fuel channel blockage event.
- **NBSR**: single fuel channel blockage event.
- **ORPHEE**: BORAX-type reactivity insertion event.
- **SIRIUS-2**: BORAX-type reactivity insertion event.

The use of probabilistic safety assessments to identify higher probability events and then establishing a bounding design pressure from the event that would produce the highest pressure is a more recent development. Although the work on the ANS containment design was not completed, this was the general approach taken.

The approach where the containment or confinement structure is designed to meet the structural load requirements for a range of higher probability internal and external events has many similarities to the approach where a bounding design pressure or load is established. The main difference is that additional factors are included in the design requirements for the reactor building. An example where this approach was taken is the reactor building for HFBR.

Whereas the limiting or bounding criteria for establishing the design requirements for reactor buildings can be very diverse, the functional requirements for the reactor buildings are very similar. As can be seen in the design parameters in Table 1, the operating temperatures and pressures for the various reactor buildings are very similar. Also, there is common usage of airlocks or dual air tight doors to preserve the negative pressure within the reactor buildings while allowing personnel and equipment access.

As well there is a common approach to dealing with the release of radioactivity from the reactor buildings through the use of emergency ventilation systems with HEPA, charcoal and/or absolute filters.

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Table 1: Summary of Temperature and Pressure Data for Containment/Confinement Buildings

Reactor and Power	In-Service Date	Normal Operating Temp. (°C)	Normal Operating Pressure (kPa)	Leakage Rate	DBA Temp. (°C)	DBA Pressure (kPa)
ANS 350 MW		20	atm.	< 0.2%/day	< 100	30
BR-2 100 MW	1960			$<2.4\times10^{-3}$%/ day @ 0.3 kPa		100
FRJ-2 23 MW	1962	22	-0.05	< 1%/day @ 0.3 kPa	87	0.26
HANARO 30 MW	1995	20	-0.034	< 600 m^3/h @ 0.25 kPa	20	0.086
HFBR 60 MW	1964	20	-0.05 to -0.1	15%/day @ 14 kPa	45	2.3
JRR-2 10 MW	1960		-0.69	1%/day @ 0.3 kPa		
JRR-3M 20 MW	1989	20	negative	10%/day	20	atm.
JRR-4		20	negative	28800 m^3/h @-2 mm H_2O	20	atm.
JMTR 50 MW	1968	20	-0.08	100%/day	20	98
NBSR 20 MW	1967		-0.25			
ORPHEE 14 MW	1980	20	-0.15	<200 Nm^3/h	35	0.15
RSG-GAS-30 30 MW	1991	26	-0.15	< 2000 m^3/h		
SIRIUS-2 25-35 MW		19-25	-0.08 to -0.1	1 vol/day	34 to 40	15

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RESULTS OF A SURVEY ON THE DESIGN BASIS FOR RESEARCH REACTOR CONTAINMENT/ CONFINEMENT BUILDINGS

A.G. Lee

IGORR 5

1996 November 4-6

Topics

- **Background**
- **Survey Participants**
- **Survey Results**
- **Summary**

Background

- Understanding design basis for research reactor buildings useful to new reactor designs and upgrades for existing reactors
- Interest identified at IGORR-IV "Workshop on R&D Needs"
- Offered to compile the information on containment and confinement features in research reactor buildings
- Survey prepared and sent out in Sept. 1995

Survey Participants

Responses received from:

- CEN/SCK (Belgium): BR-2
- CEA (France): ORPHEE
- Technicatome (France): SIRIUS-2
- BATAN (Indonesia): RSG-GAS-30
- JAERI (Japan): JRR-2, JRR-3M, JRR-4, JMTR and NSRR
- KAERI (Korea): HANARO
- NIST (USA): NBSR
- ORNL (USA): ANS
- BNL (USA): HFBR
- KFA Julich (Germany): FRJ-2

Approaches to Reactor Building Design

- "Bounding" scenario
  - postulate single severe event
  - estimate energy release, fission product release
  - identify challenges to RB
- Explicit design for "severe" accidents
  - pick a design pressure to cover a range of postulated severe accidents
  - estimate energy release, fission product release
  - identify challenges to RB
- Design for higher probability accidents
  - post evaluate containment capability for severe accidents

Design Basis Accidents

- ANS: Full spectrum of events postulated
- BR-2: Melt 40 fuel elements at EOC @ 100 MW, all Al reacts with water to make H_2 which burns, 50 kg Na burns, max internal pressure of 1.0 bar
- FRJ-2: 3% $\Delta k/k$ reactivity insertion, 60% of metal in the core react to make D_2 which burns
- HFBR: DBE, tornadoes, 10CFR100 Siting Criteria
- JRR-2: loss of coolant event and a fuel failure event
- JRR-3M: coolant channel blockage event
- JRR-4: fuel handling accident and coolant flow blockage event
- NBSR: single fuel element coolant channel blockage
- ORPHEE: BORAX reactivity insertion event with 135 MJ energy release, 9% of energy converted to mechanical energy
- SIRIUS-2: BORAX reactivity insertion event with 135 MJ energy release, complete core melt under water

Beyond Design Basis Accidents

- ANS: Events beyond DBA only considered for PRA
- BR-2: A "general emergency plan" with an on-site co-ordination cell for acting as a command centre is used to deal with accidents beyond the design basis.
- HFBR: Beyond DBAs based on security accident analysis and PSA
- JRR-3M: Fuel failure accident followed by release of radioactivity

Seismic Requirements

- Seismic considerations are site dependent
- All reactors in seismic zones include seismic design considerations
- Some (BR-2) re-evaluating seismic capability during refurbishment

Shielding Requirements

- Not a major design consideration
- Only 3 indicated shielding consideration for building design - JRR-2, JRR-3M and JMTR

Internal and External Explosion Hazards

- Not a major consideration
- Site dependent for external explosion hazards - HFBR and ORPHEE
- Most RRs do not have internal sources for explosion hazards - few high temp. & press. systems, limited sources of H_2 or D_2

Tornadoes and Wind Driven Missiles

- **Site dependent**
- **ANS: US NRC tornado design requirements**
- **BR-2: Designed to resist the overturning effect due to lateral wind forces**
- **HANARO: Typhoons with wind velocities of 20 m/s**
- **HFBR: Tornadoes and wind driven missiles**
- **JMTR: Included in siting criteria for the Oarai site**
- **NBSR: Meet normal structural requirements of wind and snow loadings**
- **SIRIUS-2: Depends on the local siting criteria for the reactor**

Aircraft Crashes

- Site dependent
- FRJ-2: Not a requirement at the time the reactor was built
- JMTR: Included in the siting criteria for the Oarai site
- ORPHEE: Aircraft crashes are considered
- SIRIUS-2: Depends on the type of aircraft and the results of a PSA

Airlocks

- Most have dual door personnel airlocks
  - all have negative building pressure
- Many have dual door truck airlocks
- Airtight hatches also used for truck accesses

Features For Controlling Releases Of Radioactivity

- Emergency ventilation systems
- Active ventilation dampers that close when high radioactivity detected
- Filtration system
  - HEPA
  - Charcoal
- Building leak rates and pressure retention capability varies greatly from reactor to reactor

Special Design Features

- ANS: Catalytic H_2 converter for cold source and H_2 igniter for small volume rooms
- BR-2: Emergency control center
- SIRIUS-2: Double isolation on containment penetrations, emergency control center

Summary

- Design basis accidents govern building design
  - coolant flow blockage
  - BORAX
  - other reactivity insertion events
  - most use "Bounding" scenario - historical
  - explicit "design" for severe accident - ANS
  - design for high probability events - HFBR
- Limited consideration for beyond DBAs
- Site conditions govern external hazards considerations (DBE, tornadoes, aircraft crashes, external explosion hazards)
- Common approach to ventilation and filtration considerations and building operating pressure

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FRENCH RESEARCH REACTORS

DESIGN OF REACTOR BUILDING IN ACCORDANCE WITH

SAFETY APPROACH AND IAEA RECOMMENDATIONS

- *Jean-Luc MINGUET* **TECHNICATOME**
- *Pascal ROUSSELLE* **TECHNICATOME**
- *François ARNOULD* **TECHNICATOME**

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[4] IGORR 2 meeting. SIRIUS 2 - A versatile medium power Research Reactor
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[5] Overall description of ORPHEE Reactor - CEA/DRN publication

[6] Overall description of SILOE Reactor - CEA/DRN publication

[7] Overall description of OSIRIS Reactor - CEA/DRN publication

[8] Dimensional stability of low enriched Uranium Silicide plate type fuel for Research Reactors at transcient conditions JAERI.
Journal of Nuclear Science and Technology - March 1992

LIST OF TABLES

LIST OF FIGURES

1 INTRODUCTION

This paper describes the French Engineering approach for the design of buildings housing Research Reactors.

Two kinds of considerations have consequences on the design of the Reactor building :

- operational requirements : an adequate building should provide large enough areas for an easy arrangement of Reactor components and namely for experimental devices. These devices may be very various (experimental loops, neutron beam equipment, spectrometers....) and are evolutive all along the Reactor lifetime, according to research programs. Moreover in a Research Reactor, operators and users are working inside the Reactor building when the facility is on.

- safety requirements : the building is at the same time the third barrier against the uncontrolled release of radioactive materials towards the environment and a physical protection of the Reactor against postulated external events.

Theses two aspects are developed hereafter.

The authors will describe afterwards the related building design options and the resulting main characteristics.

2 OPERATIONAL REQUIREMENTS

The Research Reactors family includes various kind of facilities from critical mock-ups and low power Reactors dedicated to educational programs and training, up to medium and high power Research Reactors involved in irradiation programs, fundamental research or safety studies.

The Reactors of the second type are considered in the following analysis. Table 1 gives the main characteristics of the French Research Reactors operated by CEA and of SIRIUS concepts. Their thermal power is between 14 MW (ORPHEE) to 70 MW (OSIRIS).

All these Reactors are of pool type :

- OSIRIS (70 MW) is dedicated to irradiation activities. The core has to be large enough to allow irradiations in the core and around the core.
- SILOE (35 MW) is a multipurpose Research Reactor with important irradiation programs. The design of the Reactor block, with three neutron beams ports, allowed experimental programs using neutron beams.
- On the other hand, ORPHEE (14 MW) (like RHF/ILL high flux Reactor - 57 MW) is a specialized Reactor producing high flux for fundamental research using neutron beams.
- SIRIUS is a family of multi purpose Reactors, designed mainly on the basis of SILOE reference (power range from 5 to 30 MW).

These Reactors have an open pool housing the core and an auxiliary pool or canal for storage and under water working purposes.

Some of them are equipped with hot cells linked to the auxiliary pool. The figures 1 to 8 give an overview of the OSIRIS, SILOE, ORPHEE and SIRIUS 3 Reactors.

In all these facilities, the Reactor pool, the associated auxiliary canal and the primary circuit are integrated into a unique concrete block, avoiding by passive provisions any risk of dewatering the core.

Taking advantage of the pool type Reactor concept, providing an easy access to the core, the building has to facilitate this access as well to the core as to the Reactor block experimental area (fig. 9) :

- At the pool surface level, the building has to be as free as possible of equipment, giving free areas for future experimental devices, associated instrumentation and control panels.
  SILOE and OSIRIS like many others Research Reactors in the world are housing currently some tens of irradiation devices. Moreover, the experimental irradiation devices may be very various during the Reactor life (like CHOUCA and CYRANO furnaces in SILOE, ISABELLE, IRENE and OPERA loops in OSIRIS, aso...).

When designing a new research facility we have to take into account the possible evolution of the number and type of experimental equipment which may be installed in the future, with enough margin to enable their installation, even if their characteristics are not well defined when freezing the main building dimensions (see OSIRIS Fig 3A and 3B).

For instance, in ORPHEE Reactor, the Cold Neutron Source cryogenic auxiliaries have been set up on the floor, closely to the pool surface for a short connection to the hydrogen loop (see Fig 7). Pressurised or sodium furnaces and loops for fuel testing have been installed in SILOE and OSIRIS after many years of operation.

- The access to the pool (at the periphery) needs to be as free as possible with a low level of constraints or restricted areas for handling purpose.

- The building height is directly linked to the pool depth and the water height above the core for an adequate possibility of handling with the main Reactor crane.

- At the lower part of the building, access to neutron beam ports (if any) is the main constraint to deal with when choosing the lay out of the Reactor.

  Around the Reactor block, a minimal free distance is necessary for the arrangement of experimental equipment (spectrometers, chromatographs...) between the Reactor block and the Reactor building (6 to 7 m).

  In ORPHEE Reactor, a special care was taken when deciding to locate the beam ports and the core level at the ground floor, thus giving an easy connection to the LLB, and to provide adequate reservations for future neutron guides creation and extension (project ORPHEE PLUS).

The above mentioned operational requirements have direct consequences on the overall building dimensions as shown in Table 3.

Fig. 1, 4 and 6 give an overall view of the Reactor buildings of OSIRIS, SILOE and ORPHEE Reactors.

3 SAFETY REQUIREMENTS

The Reactor building is at the same time the third barrier against an uncontrolled release of radioactive materials towards the environment and a physical protection of the Reactor itself and associated radioactive products against external aggressions.

We summarize hereafter the external hazards and the internal origin events (ie. initiating events) which have to be considered and may have some consequences on the design parameters of the containment building according to the results of safety analysis.

The French safety approach is a deterministic one with complementary probabilistic analysis which provide the necessary data to appraise the actual risk of each initiating event.

3.1 EXTERNAL HAZARDS

In accordance with IAEA recommendations, French safety analysis takes into account the following external hazards (fig. 10) :

- earthquakes,
- climate loadings (rain, snow, wind)
- flooding risks, if any,
- tornadoes, hurricanes, depending on the area risks,
- explosions, (gas,.....),
- fire,
- man induced events, including malevolence,
- aircraft crashes,
- risks induced by neighbouring facilities.

IAEA recommendations are very close with the French approach :

- for the earthquakes hazards, IAEA defines two seismic loadings :
  - the level S1 which is the same as the French "maximal historically credible earthquake (SMHV)", and close to the US "Operating Basis Earthquake (OBE)",

  - the level S2 which corresponds to the French « SMS » (which means increased safety earthquake), or the US "Safe shutdown earthquake (SSE)", but with an increased level of 1 (French practice) or 2 (if the site is not well characterized) in comparison to the level S1.
- when analyzing aircraft crash risks, we are led at least in fact to take into account the planes considered by IAEA like the LEARJET and the CESSNA, as the French practice (unless the site location is near an airport or busy areas), including the missile effect of the Learjet Reactor.
- the extreme weather hazards need to be considered, according to the plant location of the facility and the level of probability of these events.

The above listed loadings lead to the necessity for a mechanical resistant building designed to avoid any risk of creating internal missiles which could fall towards the Reactor pool or the fuel storage channel.

3.2 INTERNAL INITIATING EVENTS

Where external hazards are considered vis-à-vis the protection of the core and other radioactive material, postulated events of internal origin are related to the release of activity outside the containment (third barrier).

In accordance with IAEA recommendations, the following initiating events are considered when designing the third barrier (Reactor building including the raft and the penetrations) :

- reactivity accidents,
- internal fire,
- missiles of internal origin,
- radioactive material leakages.

Our purpose is not to describe here all the safety analysis for a Research Reactor project but to highlight the following points :

- **reactivity accidents**
  IAEA, considering the specificity of Research Reactors such as their easy access to the core, recommends to take into account, with the support of probabilistic analysis, an extended list of possible criticality accidents and to determine for each of them the potential consequences.

But Research Reactors are only a few and of different designs. Consequently reliable and numerous statistical data are not available as for French Nuclear Power Plants.
So French Safety Authorities have, in a deterministic approach selected as Design Basis Accident (DBA) a so called "BORAX type", even if the probability of occurrence of such an accident is very low in a modern Research Reactor.

That DBA is supposed to occur after a rapid and important reactivity insertion into the core leading to its complete melting underwater. On the basis of US experiments with UAl cores, a sudden release of 135 MJ is considered, 9 % of this thermal energy being transformed into mechanical energy, creating dynamic effects.

These dynamics effects have two physical origins :

- a steam explosion due to the rapid transfer of energy from the fuel plates heaten up to the cladding melting point (640°C to 660°C) towards the cooling water,
- the chemical reaction of the molten metal with water followed by the explosion of the hydrogen produced.

The dynamic effects are :

- a pressure wave against the walls and the bottom of the Reactor pool,
- the ejection upwards of a fraction of the pool water which may - depending on the height of the building - reach the roof and produce a water hammer on it.
- a progressive increase of pressure inside the Reactor building up to a rather low level of pressure depending on the free volume of the building (see table 3, less than 150 mbar, in ORPHEE for instance).

Should that kind of accident occur, the second barrier (pool liner and primary collant circuit) and the third barrier (the containment building) have to keep their integrity in order to limit the release of radioactive materials outside the Reactor building below an acceptable level.

So described, this BORAX type accident can be considered as an enveloppe for all other possible reactivity accidents, relatively to external consequences.

This leads to the necessity for a leaktight containment building.

From the French Safety Authorities point of view, one would have to demonstrate that the melting point of the cladding cannot be reached or the accident limited to less consequences, not having to consider the BORAX effect for the building design. Considering new types of fuel elements, experiments performed with LEU silicide fuel ref [8] show a better behaviour of LEU fuel plates at high temperatures in comparison to HEU.MTR aluminium fuel plate elements, but have to be further developped at a larger scale.

However, as long as this cannot be demonstrated for the considered specific fuel, the consequences of the DBA of BORAX type will have to be considered, and lead to potential acceptable consequences, to get the licensing from the French Safety Authorities.

On the other hand, this constraint has a positive effect : whatever the types of core or of experiments then considered, the highest level of safety will be achieved, thus giving an interesting contribution, on the safety point of view, to the versability of the Reactor.

- **Radioactive material leakages :**
  they may proceed from the core itself, from irradiation experiments, from radioactive circuits or waste storages, despite all the care given to the design and the technology.
  We also have to consider that many Research Reactors may have to support during their lifetime major core technology changes (i.e. UAl, UO_2, silicide fuel).
  A leaktight containment is an advantage on a safety point of view, enabling the facility to withstand any unexpected major events without any significant consequence outside.

3.3 OTHER SAFETY RELATED REQUIREMENTS

The containment building is, as seen above, the third barrier and consequently its leak-tightness (controlled leak or tightness) should be monitored including the raft.

This can be achieved by periodic testing or monitoring during normal operation.

So, the design of the building should be done in such a way that :

- leakage through the walls, the roof and around penetrations are minimized, and can be periodically tested,
- anly liquid leakage through the raft can be detected,
- the penetrations of the Reactor building are designed and built with a special care.

4 RESULTING BUILDING DESIGN

What are the consequences of all the above requirements on the design parameters of a new Reactor building ? Of course, the answer is not unique and depends effectively on the project considered as a whole.

Some parameters may have consequences on others. For example, the height (above water) of the building may be sufficient - it is often true - to make negligible or nul the effects on the roof of the water column expelled, as considered in the enveloppe BORAX event.

Though, depending on the location of the facility, seismic conditions and aircraft crash will have preponderant consequences on civil works parameters design.

However, only the complete Preliminary Safety Analysis allow to build the correct hierarchy of events with their consequences.

It is not the place for a detailed parametric analysis, but let us summarize the main features of the French Reactor building concept and the conservative provisions which enhance the safety of Research Reactors versus the release of radioactive products :

- design of structures resistant to the worse external conditions related to the given site,
- selection of a Borax type accident as DBA,
- possibility of monitoring the third barrier including the raft (sump monitoring, periodic testing of pressure drop).

The major part of leaks have their origin through the penetrations (a ratio of approximatively 60 % were measured in ORPHEE). It is the reason why penetrations which are leading outside are routed through a peripherical gallery where the major parts of the leaks coming from the Reactor building are collected. This gallery is equiped with a specific emergency ventilation.

Table 3 gives the main building characteristics of some of the French Research Reactors.

5 CONCLUSION

As a conclusion, let us remind three important considerations :

- on the basis of the here above described safety approach, with BORAX type event as a DBA, it can be stated that French design Research Reactors provide a better control of release and dispersion of radioactive materials outside the Reactor building in case of a major accident, and of course with even more lower consequences for less severe accidents such as partial melting of one single or several fuel elements,

- so a fair comparison between various Research Reactors concepts with different designs and safety approach would enhance the "added safety margin" given to the concepts considering a BORAX type event as a design basis accident, and equipped with a containment building.

- the versatility of such Reactors is improved, on a safety point of view, allowing future adaptations of the core or experimental programs, without any added concerns on the licensing aspects.

TABLE 1

FRENCH RESEARCH REACTORS MAIN CHARACTERISTICS

REACTOR	SILOE	OSIRIS	ORPHEE	SIRIUS 2	SIRIUS 3
Location	Grenoble	Saclay	Saclay	-	-
Type	pool	pool	pool	pool	pool
Utilization	multipurpose	irradiation	fundamental research	multipurpose	multipurpose
First criticality	1963	1966	1981		
Thermal power	15 → 35 MW	50 → 70 MW	14 MW	15/30 MW	5/10 MW
Core :					
type of fuel elements	HEU-UAl	LEU - U_3Si_2/Al	HEU - UAl	LEU - U_3Si_2/Al	LEU - U_3Si_2/Al
Number of fuel elements	40	44	8	33	25
Reactor block					
Neutron beam ports	3	-	9	4 - variable	4 - variable
CNS	-	-	2	1- option	1 - option
Hot sources	-	-	1	1- option	1 - option
Main approx dimensions :					
- outer diameter			Ø 7.5 m	Ø 7.3 m	Ø 6m
(at core level)	4.6 x 4.6 m^2	7.5 x 6.5 m^2	Ø 4.5 m	4.8 x 4.8 m^2	Ø 3.5 m
- pool inner dim	11 m	11 m	15 m	11 m	9.7 m
- pool height					

TABLE 2

IAEA RECOMMENDATIONS LIST

Safety series n° 35.S1	- Code on the Safety of Research Reactors Design
Safety series n° 35.S2	- Code ont the Safety of Research Reactors operation
Safety series n° 35.G1	- Safety assessment of Research Reactors and preparation of the Safety Analysis Report.
Safety series n° 35.G2	- Safety in Utilization and Modification of Research Reactors
Collection TECDOC 348	Earthquake design of nuclear facilities with radioactive limited inventory
Collection TECDOC 403	- Siting of Research Reactors

TABLE 3

FRENCH RESEARCH REACTORS CONTAINMENT CHARACTERISTICS AND DESIGN PARAMETERS

REACTOR	SILOE	OSIRIS	ORPHEE	SIRIUS 2	SIRIUS 3
Design	reinforced concrete	reinforced concrete	reinforced concrete	reinforced concrete	reinforced concrete
Main dimensions :					
int. Diameter	Ø 27 m	Ø 32 m	Ø 28 m	Ø 28 m	Ø 22 m
height	27 m	21 m	29.2 m	29 m	27 m
free volume	14 000 m^3	16 000 m^3	20 000 m^3	15 000 m^3	6 000 m^3
Pressure loadings :					
- Normal operation	-1,5 mb < P < -1mbar	- 0,5 mbar	- 1 mbar	- 1 mbar	- 1 mbar
- Accidental conditions (BORAX) :					
max hall design pressure.[1]	< 68 mbar	<20 mbar	< 150 mbar	< 150 mbar	< 150 mbar
water hammer loading on the roof	none	0.56t/m^2 on 45 m^2	none	5T/M^2 on 60 M^2	
Leakrate control	loss of pressure periodic testing	loss of pressure periodic testing	loss of pressure periodic testing	loss of pressure periodic testing	loss of pressure periodic testing

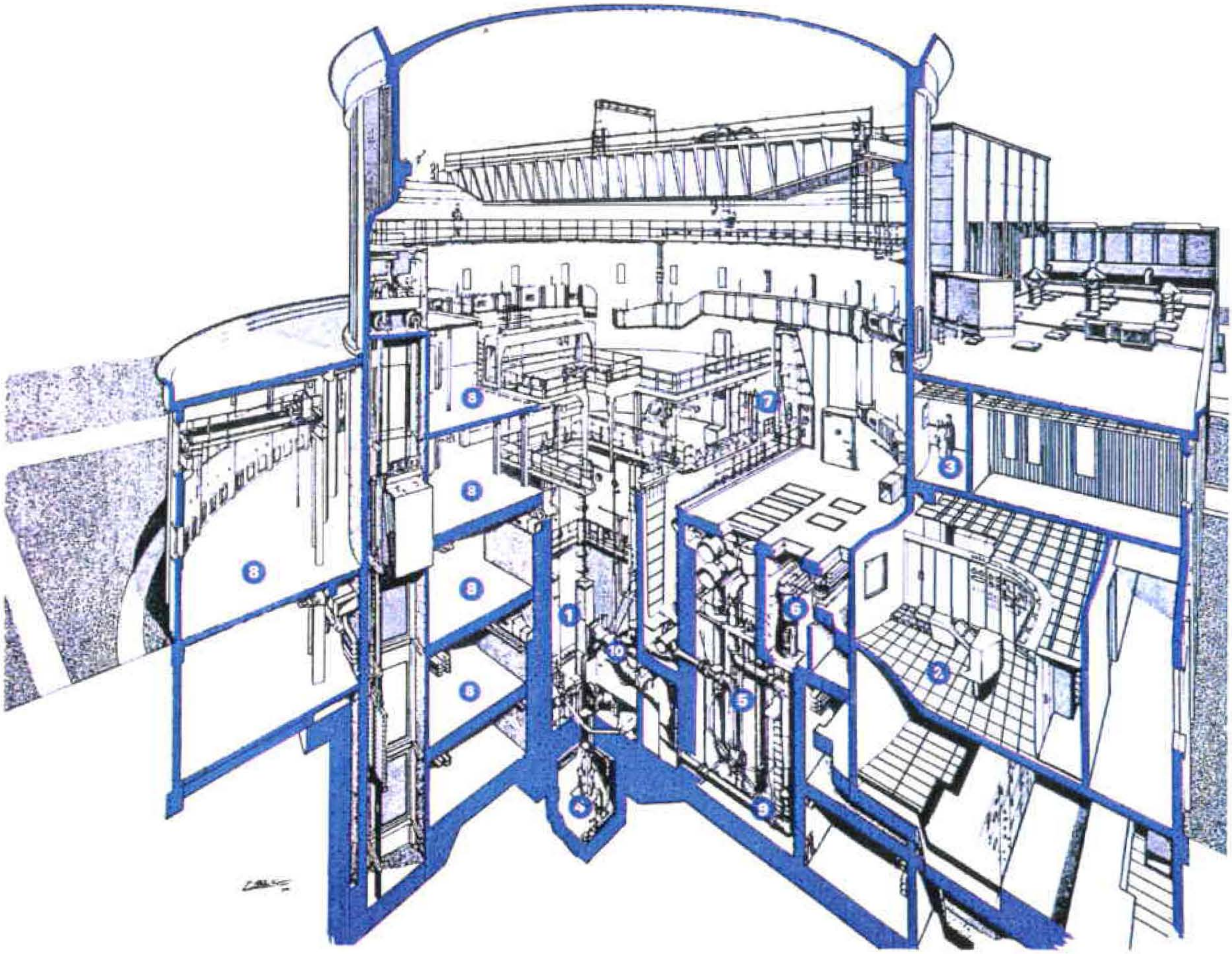
[1] These are design values. Expected calculated values are significantly lower. These values are in most cases naturally achieved by buildings withstanding aircraft crashes and seismic loadings.

FIG 1

OSIRIS REACTOR
photo : MFP

technicatome

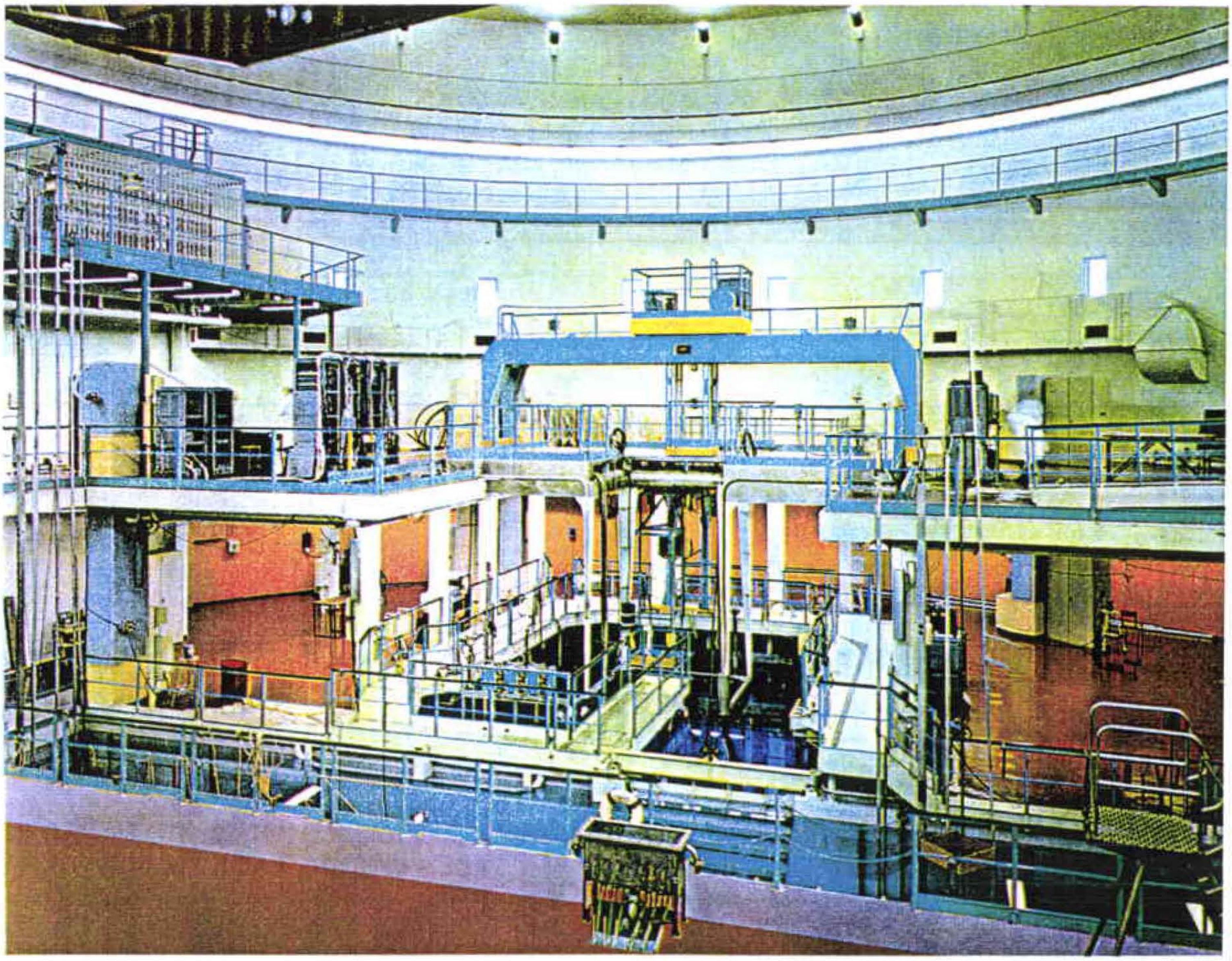
FIG 2



OSIRIS - OVERALL LAYOUT
photo : CEA

technicatome

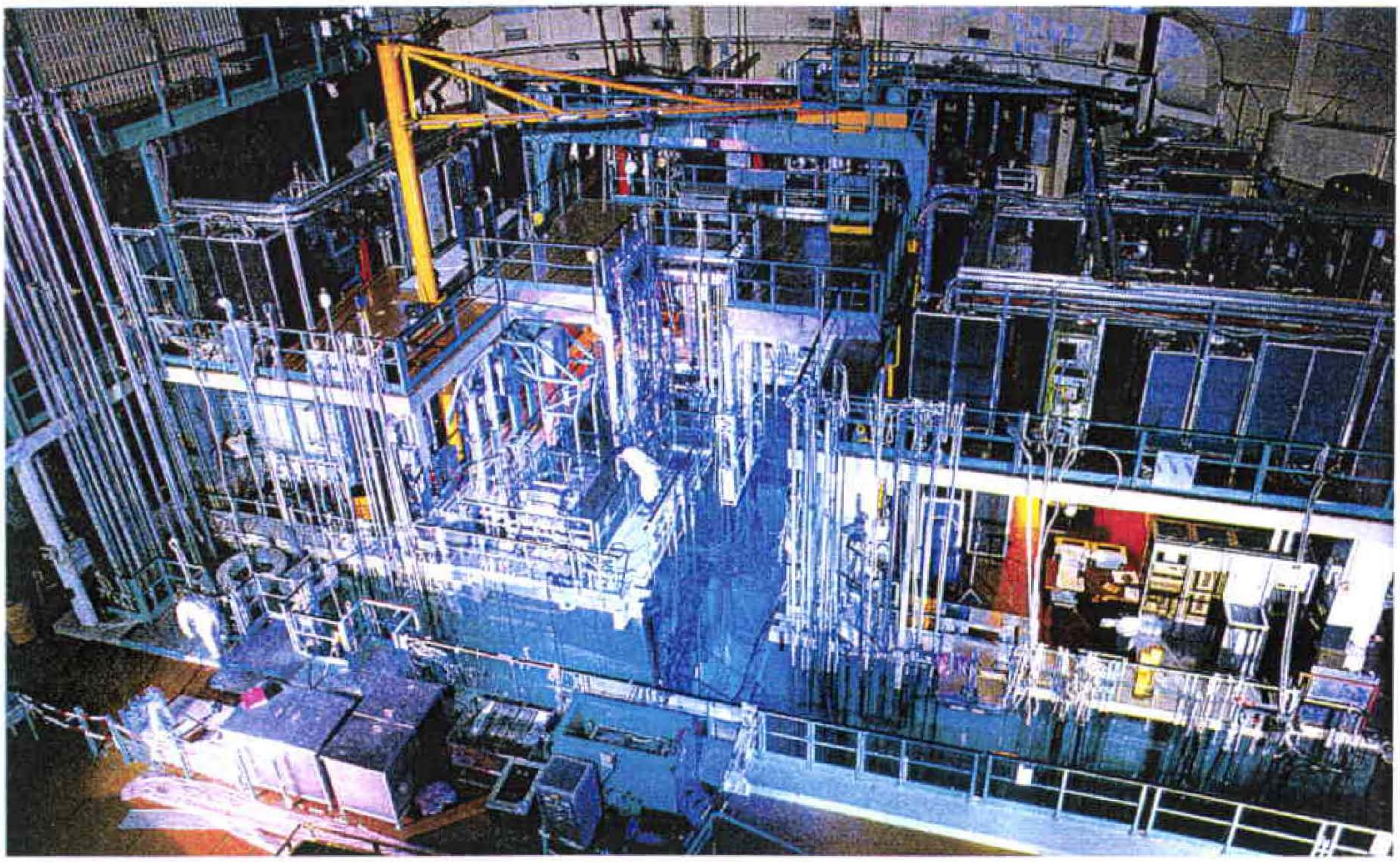
FIG 3a



OSIRIS REACTOR HALL
AT THE BEGINNING OF LIFE
photo : CEA

technicatome

FIG 3b



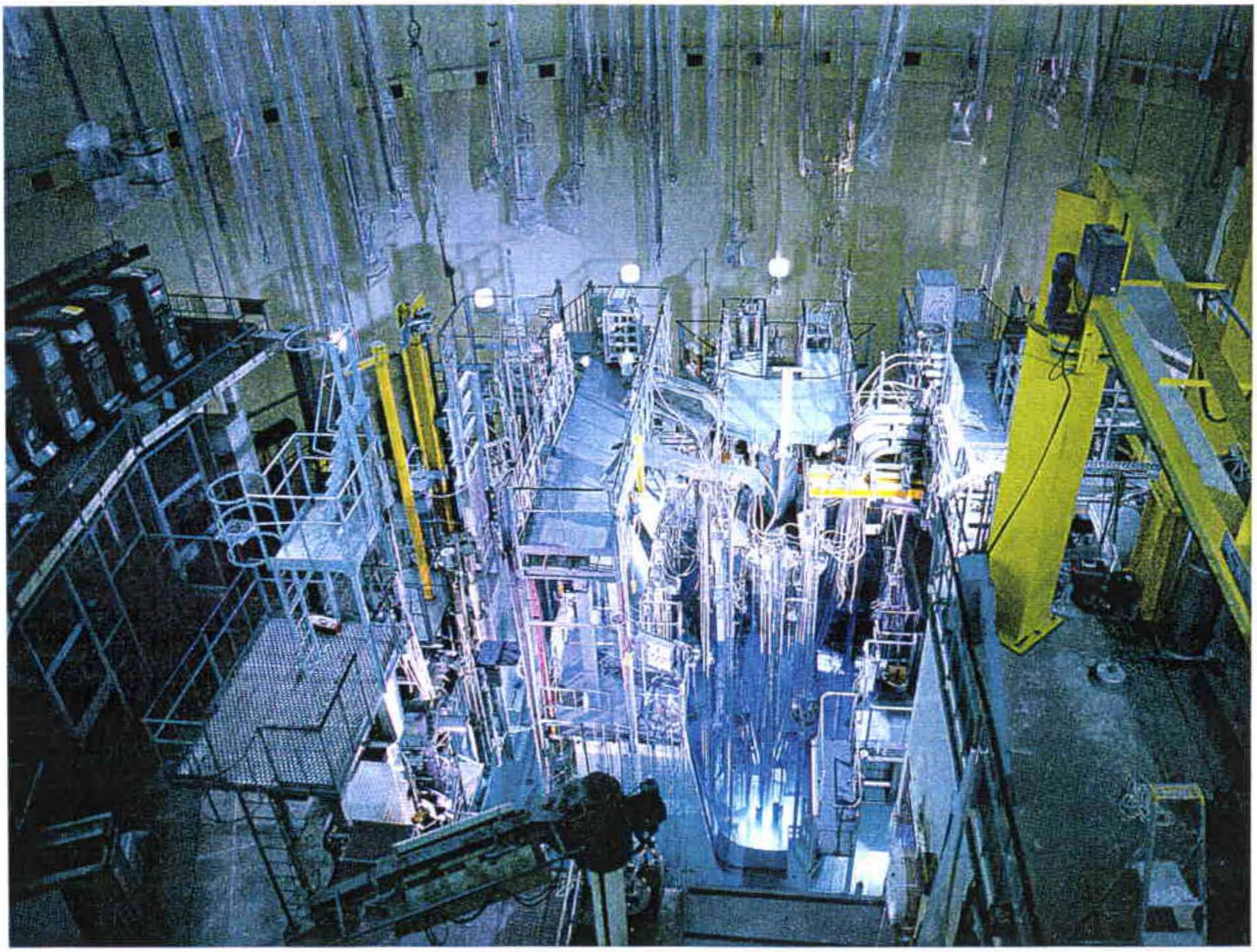
OSIRIS REACTOR HALL
AFTER MANY YEARS OF OPERATION AND EXPERIMENTATIONS
photo : MFP

technicatome

FIG 4

SILOE REACTOR
photo : CEA

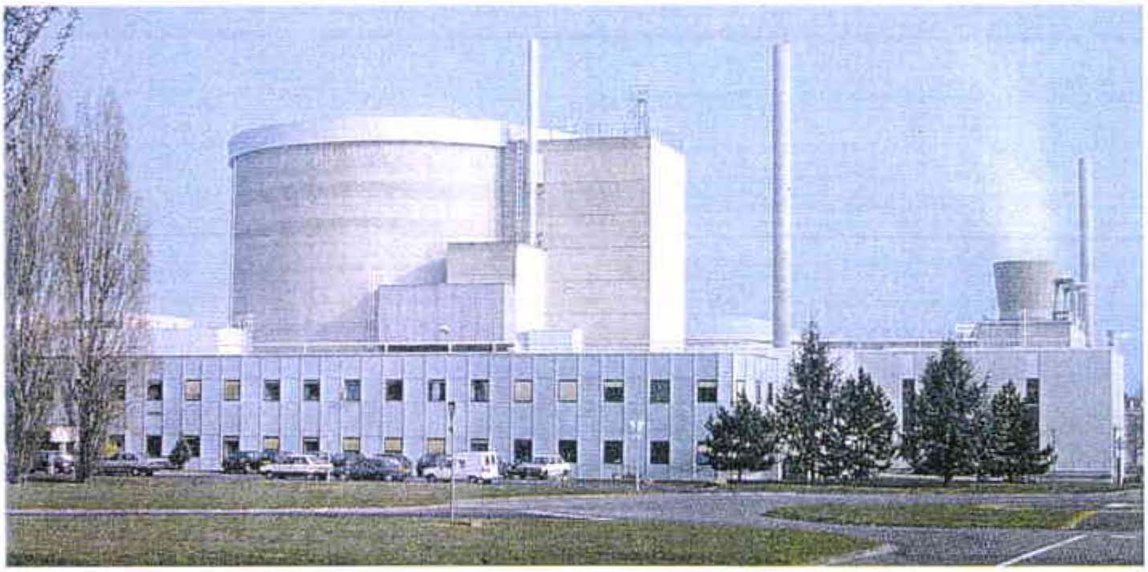
FIG 5



SILOE REACTOR HALL
photo : MFP

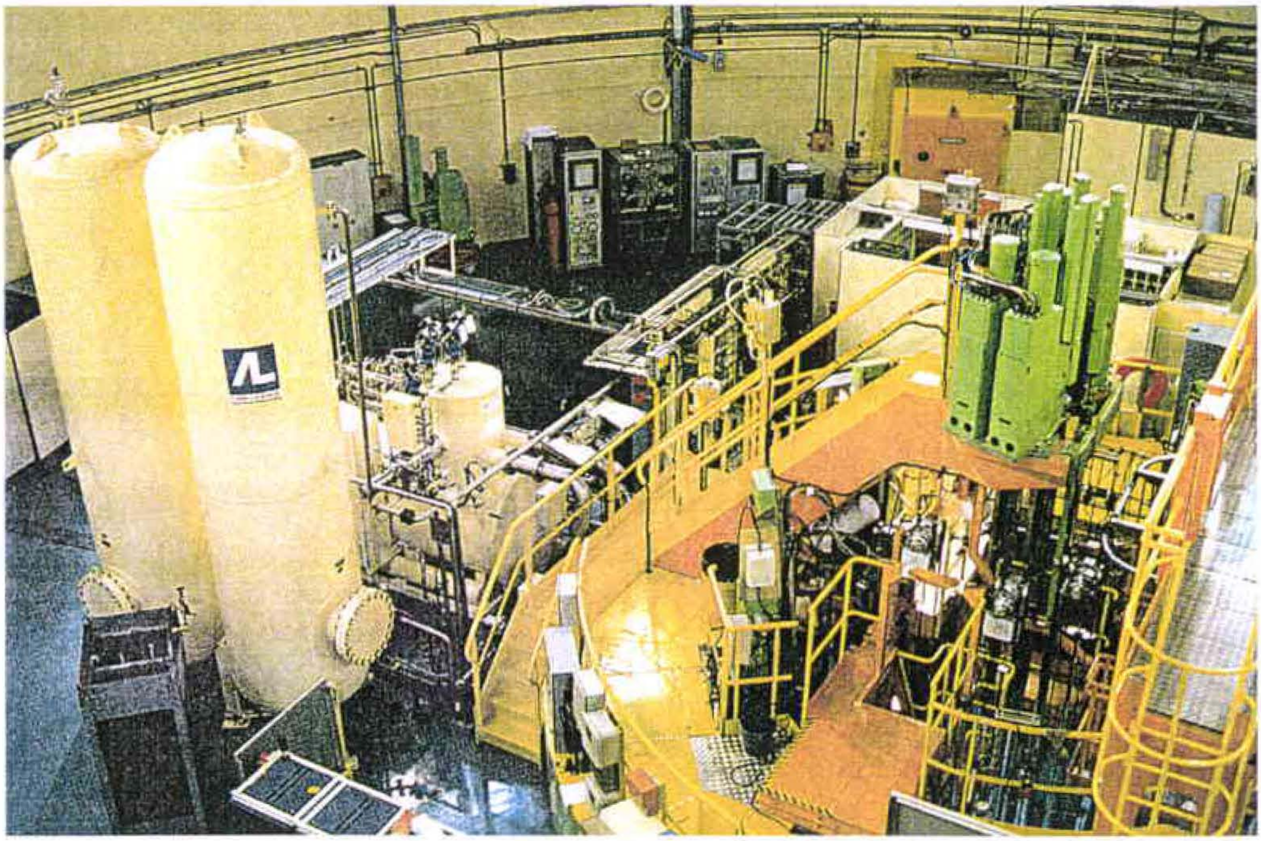
technicatome

FIG 6



ORPHEE REACTOR
photo : CEA

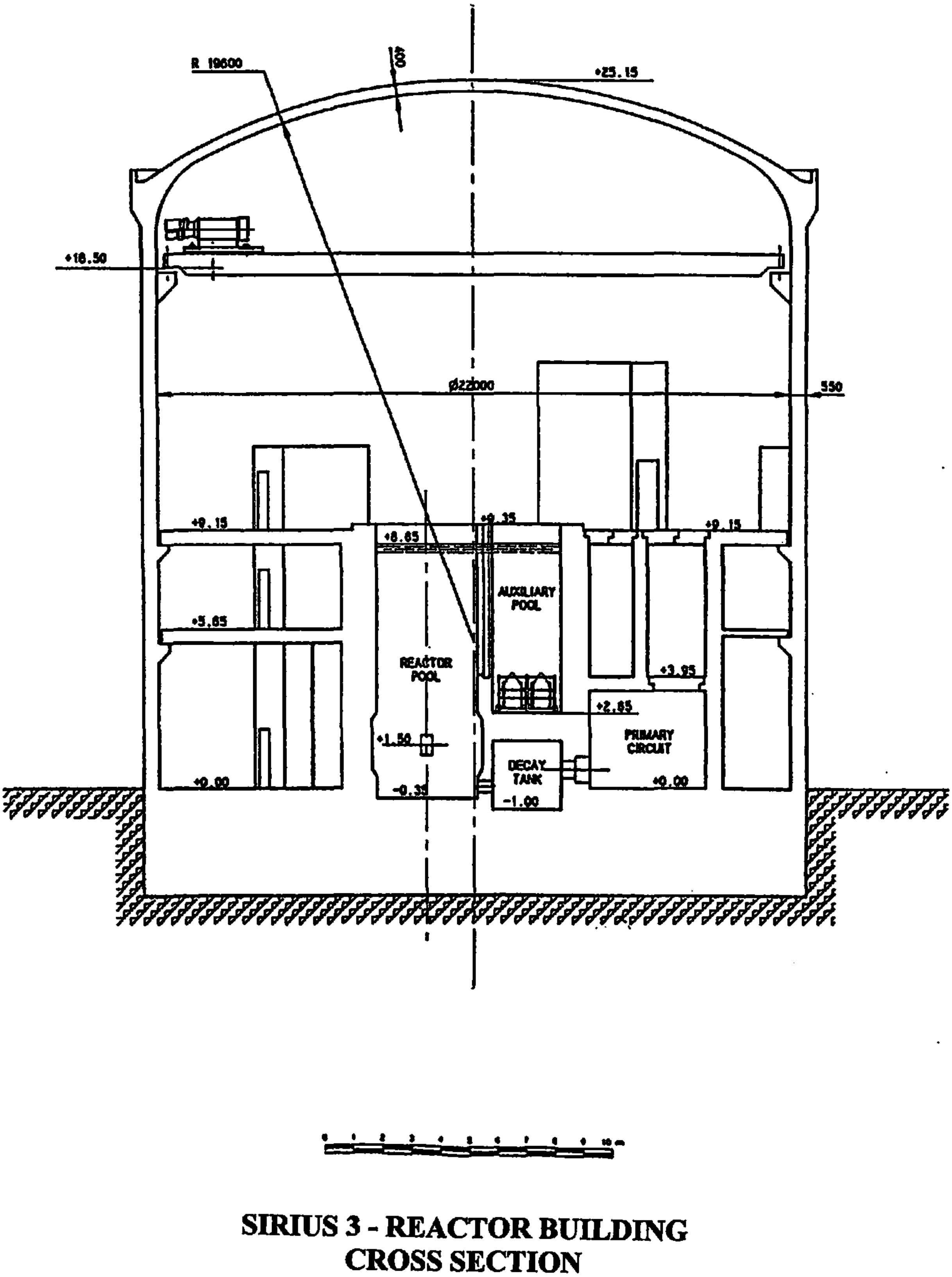
FIG 7



PARTIAL VIEW OF ORPHEE REACTOR HALL
photo : ANPHI

technicatome

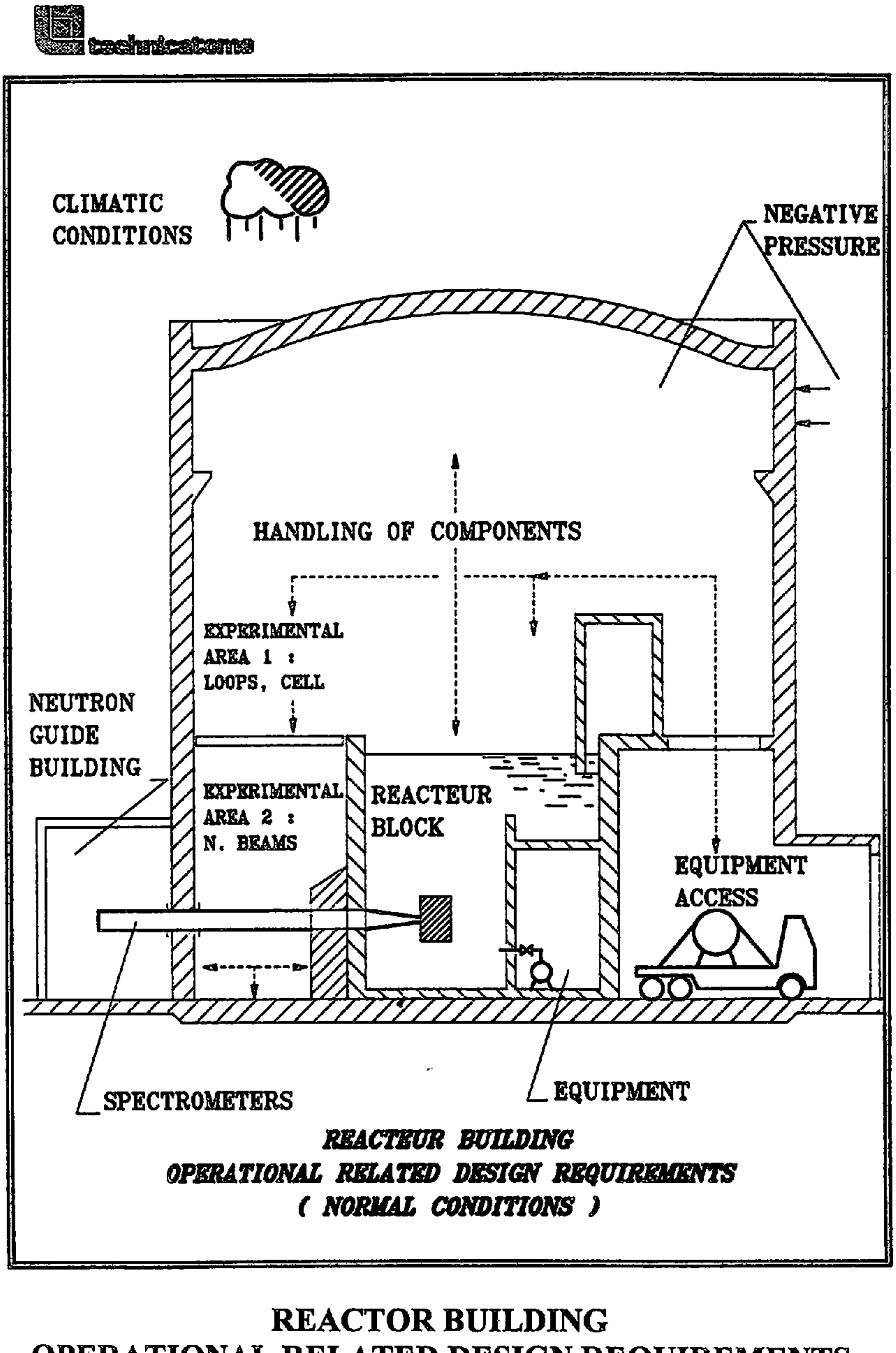
FIG 8



SIRIUS 3 - REACTOR BUILDING CROSS SECTION

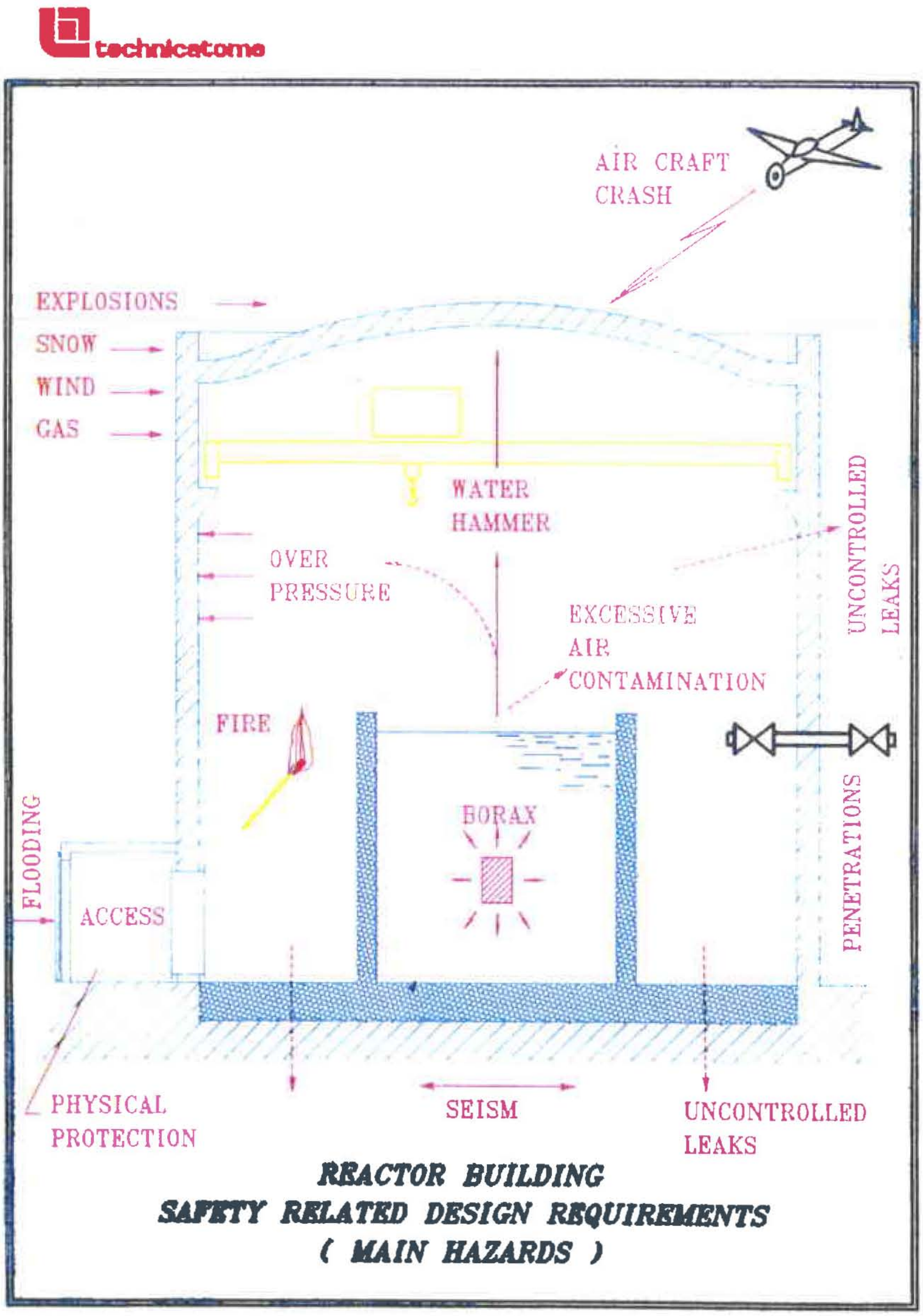
technicatome

FIG 9



REACTOR BUILDING
OPERATIONAL RELATED DESIGN REQUIREMENTS
(NORMAL CONDITIONS)

FIG 10



REACTOR BUILDING
SAFETY RELATED DESIGN REQUIREMENTS
(MAIN HAZARDS)

CHAIRMAN : R. WILLIAMS

SESSION 4

RESULTS OF A SURVEY ON THE DESIGN BASIS FOR RESEARCH REACTOR CONTAINMENT / CONFINEMENT BUILDINGS (Albert Lee)

Question from Klaus Böning of TU München :

And here we should mention that we take into consideration a full core melt under water, which is important, so that there would not be any evacuation of the population outside of the facility fence necessary, so no emigration. But, we have not taken into consideration a severe increase in internal pressure, as for the BR2 , we have to discuss the figures. Another point that we also should designate to be taken into consideration is the airplane crash, from a military or high-velocity jet into the building, which is unique. We have 1.8 meters of thick walls around the building.
OK, that is a brief report. One question : I wonder why you did not consider terrorists?

A : OK, it's true. That is a question I didn't ask. I consider terrorist events and external humanly-initiated events to be something that is beyond the ability and control of a reactor designer to try and design. I think that has to be covered in the realm of the establishment of security on the site rather than defending it at the reactor building.

Maybe it's not totally true, but one has to look at in terms of site considerations, and I agree that it's much more difficult at Munich than it would be at, say, Serpong in Indonesia. OK? So how one treats it I think, really has to be dealt with within the context of where the reactor is located, and what kind of surrounding site is provided.

I also don't think that the issue of the vulnerability of the reactor building to a terrorist event is something that we really want to publicize with anybody.

Question from Clifford Hickman of Technicatome :

I'd like to talk about aircraft crash risks. You say they are site dependent, which is obviously true. But national regulations - can you say something about that? They are not necessarily site dependent.

A : Well, I think again it varies. In North America, at least, I know that at the Oak Ridge National Laboratory and at the Argone National Engineering Laboratory, and also at Los Alamos, like in Canada at Chalk River and the Whiteshell Laboratories, we have an exclusive air exclusion corridor where aircraft are not permitted to fly. That air exclusion corridor, at least at Chalk River, is 20 kilometers on either side of the site. I can't rule out every event. All it does is establish the probability you have to defend against for an aircraft crash. This is where one always gets into probabilistic assessments on it. I'd never rule out pilot error or a deliberate event like they did at the White House about a year ago when somebody flew a small plane into the back yard of the White House. There is no exclusion corridor around the

White House, but you can't prevent it. You have to look at it and decide what is the most sensible what are the most likely events you have to defend against within your regulatory regime.

Question from Hans-Joachin Roegler of Siemens :

Don't you share my view that, when we start discussing such events like aircraft crashes as potential events that we have to protect our towns against, then it will come up more and more and it will raise the costs more and more. So we should really try to prevent making that an aspect to consider. Although Munich is a bad example, and when reporting on this as the first reactor having this protection GA even claim that their Texas Airbase TRIGA was the first research reactor protected against aircraft crashes. I think we should stop describing this as a positive feature.

A : I agree with you. I don't think it's an issue to raise automatically for several reasons - it does drive up costs, it detracts from the real issue of safety that should be in the design of the building. And one has to keep in mind that there are two issues that appear on the surface to be related but are not always related, and you have to keep this in mind. One should be ensuring, as the operator of a reactor and as the designer of a reactor, that we specify systems to ensure safety. Safety first. The regulator's responsibility is to review your safety case to determine whether or not in their opinion it's licensable. Licensable arguments, licensability, can be based upon judgment criteria that are not always related to safety. They will look at events that could be very highly improbable in your judgment, and the challenge is how far you have to go to defend against very improbable events. One has to try and draw the line somewhere, otherwise costs of these reactor facilities will be totally uncontained and I could easily take the ANS and add some arguments for very improbable events and take the cost of the ANS up to the cost of the superconducting supercollider. But I don't want to be that perverse! One may not, but I could re-evaluate an existing reactor that has operated very safely for close to 30 years and apply inappropriate arguments and convince somebody that it's not licensable today. And that's not the case.

Question from Guy Gistau of Air Liquide :

As I pilot, I can tell you that, in the Rhone Valley, the minimum height above a reactor is 1000 feet, 300 meters. However, above the site at Cadarache it is a restricted area which is not allowed for flights.

A : Just as a closing remark, the building for the Maple 1 reactor is complete. We completed it in 1992. We stopped the project in 1993 for various reasons. We are about to restart that project. But when we completed the building design, the reactor hall that surrounds the top of the pool is constructed out of heavy concrete with about 60% reinforcing steel in it. The walls are 30 cm thick, the roof is 30 cm thick with a 1-inch steel plate on top. OK? The worst case accident we looked at was a single channel flow blockage with internal pressurization of 5 PSI. The building will withstand your average 1500-kg vehicle traveling at 180 km per hour 5 meters above the ground, impacting it.

FRENCH RESEARCH REACTORS
DESIGN OF REACTOR BUILDING IN ACCORDANCE WITH SAFETY APPROACH AND IAEA RECOMMENDATIONS
(Jean-Luc Minguet)

Question from Edgar Koonen of CEN - SEK :

Is there a specific leak rate established, a value of, say, 3×10^{-4} of the air volume ?

A : I don't have the values in mind, but considering the different reactors I mentioned before, we have for each of them specific maximal allowed leak rate is to respect and we have to remain under these specified values. These tests are performed by under-pressure tests, and we observe the loss of pressure during a specified period of time. These values are specified at the issue of the safety analysis.

Q : Is it the result of the borax analysis?

A : It is based on the analysis of the potential consequences outside and these values are linked to the potential acceptable consequences outside the reactor building.

Question from Johannes Wolters of Jülich Research Center :

In Germany, when we speak about design basis accidents, when we consider them, we have to show that we keep below certain planning values concerning the radiation exposure outside the reactor. Is it the same in France and what are the values ?

A : In general terms, it's quite the same : for the accidents considered for the design, the wholebody dose for the public has to remain below acceptable value (0,5 REM). However, for DBA, this point has to be further discussed in the frame of a specific project.

Question from Guy Gistau of Air Liquide :

This is not a comment related to the topic of safety, but could you show the picture of the pool of ORPHEE. As a cryogenist, I would like to show here something which is not necessary to do, or something to be avoided.

OK - I'm looking for it. Is this the view you wanted to see.

Yes. You see here, on the left side, these two big vertical capacities which keep helium. This is equipment which can be put anywhere inside a reactor, but I think not near the pool where the area is quite expensive. You can put that anywhere else in the reactor, but not there.

Question from Jong-Sup Wu of KAERI :

I think the definition of the confinement and containment are different. It is very difficult to verify the allowable leakage rate during operation because it depends on environmental conditions such as wind and external temperature. What kinds of measurement methods are applied to French reactors?

A : Confinement is a building in which, by the ventilation system, you keep a depression in relation to the outside so as to avoid and control any leakage directly from the reactor building to the outside. A leak-tight containment building is a building in which it is possible to maintain radioactive products, even if there is a level of pressure higher than outside. For these containment building, as I said beforre, we perform under pressure tests.

Q : Another question is the method of measurement of the leak rate during operation. In the case of containment, I think it is easy, but for confinement. At HANARO, it is confinement, but in our experience it is very difficult to verify the leak rate during operation. What is the case for you for leak testing in confinement?

A : In fact, we have containments, but for confinements it is possible to observe how the pressure increases when you have a sudden stop of the ventilation.

Comment from Francisco Alcala-Ruiz of IAEA :

I understood that the question was related to buildings which would not be able to withstand any overpressure in that case.

Q : When we tested according to time, it depends on the outside environment, for example the wind, the atmosphere. It is very sensitive. That is my question - how do we verify the exact leak rate during operation ?

Comment from Edgar Koonen of SCK-CEN :

I just want to add some information - when we measure the leak rate, we even have to take into account dilatation of the building in the data. This is a significant correction that we make. If we don't do it, we might even measure a negative leak rate! The building can have quite an important dilatation on top; it's 40 meters high and it's quite hot there.

Right, of course. We have to take into account the possible temperature evolutions during the tests.

Comment from Colin West of ORNL :

Similarly, you have to be careful at what height you measure the pressure. If you have a very tall containment building, the delta P between the basement and the roof is more than the leak rate of delta P you're looking for. Now, if the humidity changes outside more than it changes inside, you have to worry about that, too. That's just a guess!

We still have time left for any other comments you might want to make.

Question from Hans-Jürgen Didier of Siemens :

I only want to make a comment that's a question, too. Isn't it a dogma which you use as a regulation, the Borax. I think it might be very dangerous to have a dogma instead of a regulation, because I don't find any mechanistic reason why you have to handle a Borax event in a research reactor. That was our problem in Germany.

Answer : Jean-Luc Minguet :

I think that it's a rather old story. In the past, when deciding to choose an envelope event, Borax was considered as the worst severe possible accident based on the experience and the studies performed in the US on uranium aluminum fuel. As I said, it could be certainly of a great interest as we start new safety analysis in France for instance for the RJH, to put all these things on the table to see and possibly to define another more relevant approach and to define the new basis for a major severe accident to be considered in the design of a research reactor.

Question from Jean-Jacques Verdeau of Technicatome :

May I add something? For SILOE, we did a refurbishment in 1987 and we had to add a liner to the pool because the pool was not leaktight and in order to prevent the borax we only had this stainless steel vessel around the core. And you shouldn't think that take into account the borax is very expensive. It's not very expensive in fact.

Question from Hans-Joachim Röegler of Siemens :

I say that the French have mainly made a fashion out of that and it's not considered as an accident in full with all the consequences that are potentially there. There have been no experiments to demonstrate that, in all the reactors you have here, borax is really overcome by this small structure from buffering the pressure at the water pool. So I don't think that is a consideration we would have in Germany if we had the same assumption for borax in our licensing.

Comment from Kir Konoplev of PNPI :

I would like to mention that leakage is not a figure that is representative enough for containment. After the international experts of the reactor PIK, we tried to make an investigation of our situation. We compared our situation with the containment of other reactors.

You see, the leakage of the different isotopes of course has a different influence on the radiation on the environment. For example, any containment can help you with this cesium. If it comes into the containment, it will come into the environment.

It is the contrary for the short life isotopes.

So it depends on the leakage and the volume of the containment itself. For our examinations, we melt the core to release all the fission products that exist in the fuel and look to see in the leakage, what is the presence of every isotope, and the dose at the boundary of the site.

Comment from Francis Merchie of CEA :

Just a comment about the role of the confinement or containment building. From a safety point of view, we have to control the three safety functions of the reactor. We have to control the reactivity in normal and abnormal situations, we have to control the heat dissipation and thirdly we have to confine the radioactive products. The building is the third barrier for the confinement and, by applying the defense and death principle, the third barrier must be the last barrier. That means that the first barrier, the cladding, and the second barrier, the primary circuit and the pool, must be inside the third barrier. This is very important and is the reason why in the French reactor design, the primary circuit is included in the building and inside the concrete reactor block.

Coming back to the borax. I think that borax is only a name - the problem is what type of reactivity insertion we have take into account in the design of a reactor. Of course, we know the experience with borax at SPERT, the final experience at BORAX delivered an energy of 135 megajoules. At the SPERT experiment, the energy was higher than 135.

In France, we performed many experiments at CABRI in the '60s, and we came to the conclusion that even with lower insertion of reactivity, I mean for example 2% of delta k/k, the period is very low and the energy liberation is always very important, creating mechanical damage to the core. So, taking the so-called borax accident as a DBA is like a credit in release of energy in the case of reactivity insertion.

We have also performed some simulations using explosive powders in mock-ups of a scale of 1/3 or 1/5 for different swimming pool type reactors, and we saw that there was in fact some mechanical damage to the pool and on the beam tubes. So the beam tubes are very important from that point of view. This is why we have installed what we call safety valves at the outlets of the beam tubes to be sure that in case of any type of leakage on the pool, there will be no leakage of radioactive water outside the concrete block.

The End

IGORR 5

SESSION 5

WORKSHOP ON R & D NEEDS AND RESULTS

WORKSHOP ON R&D NEEDS AND RESULTS

Klaus Böning
Technical University of Munich

1. Introduction

The goal of this R&D workshop has been - as in the IGORR meetings Nr. 1 to 4 - to identify research and development problems which IGORR groups of research reactor operators and designers might have and to stimulate contacts with other IGORR groups who plan to perform or have already performed R&D investigations on problems they have. The former IGORR meetings have demonstrated that there is, indeed, considerable overlap in the needs of various groups in the same R&D topics. So this workshop has developed to something like an open market for the exchange of information between the various groups on

- R&D work needed (without having plans for own work)
- R&D work announced (i.e. work needed and planned)
- R&D work just peformed (so that preliminary or final results could be made available).

2. Old R&D Topics (from IGORR 1 to 4)

In the report on the R&D workshop of the last IGORR meeting (see IGORR-IV Proceedings, page 391 ff) all the R&D topics which have been brought up, but not yet addressed so far, have been identified with an alphanumeric key: the leading number giving the IGORR meeting (Nr. 1- 4) at which this topic has been raised, and the following letter helping to put the various topics in consecutive order.

Many of these topics have been discussed in the R&D workshop of the present IGORR-5 meeting:

Nr. 2a: "Multidimensional kinetic analysis for small cores". Christoph Döderlein, who is now at CEA Cadarache, gave a brief report on what he has done as a Ph.D. student for the FRM-II compact core reactor. He has written a short summary which is included somewhere in these Proceedings.

Nr.3c: "Chemical and other energy release from core melt events". Albert Lee, AECL told the auditory, that Michael Corradini from the University of Wisconsin, USA, has developed a computer code for predicting aluminum fuel to water coolant interactions. The code which is based on experimental data is called TEXAS-III. The documentation is available from M. Corradini.

Nr.4e: "Cold neutron beam tube guides size and geometry optimization studies". Doug Selby, ORNL, reported that work on thermal (not cold) neutron beam tubes has been done at Oak Ridge, but has not been published, whence interested groups should contact him directly.

Nr. 4f: "HANS-3 fuel capsule irradiation (in HFIR) evaluation". Doug Selby, ORNL, gave a summary of these very interesting results; his transparencies are reproduced in Appendix A of this report.

Doug Selby also reported briefly on the following R&D topics for which results have been obtained from the ANS closeout activities:

Nr. 4f: (as above)
Nr. 4g: "Meat fabrication with spherical powder fuel".
Nr. 4h: "Centering of fuel in plate fabrication".
Nr. 4i: "Effect of flow blockage shape".
Nr. 4j: "Reduced ph effects on aluminum corrosion".
Nr. 4k: "Final summation of the ANS thermal hydraulic test program".
Nr. 4l: "Final results from the HANSAL aluminum irradiation tests and examination of BNL data".
Nr. 4m: "Irradiation creep in aluminum".

The references and further information on all these topics can be found in Appendix B of this report.

Finally, Doug also submitted a list of references refering to fuel plate stability studies for the ANS, see Appendix C. This topic had already been brought up in the R&D workshop of IGORR-1 and has been further discussed after the presentation of Jürgen Adamek at this meeting.

Other R&D topics from IGORR-4 were addressed in special contributions to this IGORR-5 meeting: so, concerning Nr. 4a ("Requirements for the design of containment") and Nr. 4b ("Cold neutron nuclear data") separate papers of Albert Lee/AECL and Doug Selby/ORNL can be found in these Proceedings.

If we further drop all those topics which have been discussed at IGORR-4, we end up with the Table 1 of R&D problems from previous IGORR meetings for which answers have not been given yet:

Table 1

R&D needs identified at IGORR-1 to IGORR-4
which have not been addressed so far:

2b	Burnable poison irradiation
3f	Test of cryogenic circulators for single-phase forced-convection cold sources
4c	Thermal-hydraulic data (flow instability) on the heat transfer in fuel elements (up to 500 W/cm^2) for high cooling water velocities (around 17 m/s) but low system pressure (below 10 bars)
4d	Method to calculate the decay time required after full power operation before the core is safe in air.

The cancellation of "old" R&D topics does, of course, not mean that the answers given were sufficient in all cases to really solve the particular R&D problems; so it might well be that some of these items could come up again in the future.

3. New R&D Topics (from IGORR-5)

In the second part of this workshop transparencies were made available to the audience which could be completed by interested participants. These forms covered the three areas "R&D work needed /or announced/ or performed". What followed was a most lively activity of many participants coming up to give their presentations. The various R&D topics are listed in Tables 2 and 3, labelled 5a to 5h according to the order in which they were presented.

Table 2

New R&D work announced at IGORR-5

Nr.	Affilation/Name	Topic
5a	Klaus Böning/TUM	Irradiation tests of highly enriched silicide fuel up to high fission densities in the silicide particles
5b	H.J. Didier/Siemens	Afterheat removal from highly loaded fuel elements cooled by natural convection, including flow reversal
5c	Albert Lee/AECL	Probabilistic safety analysis work to support major research reactor refurbishments
5d	Kir Konoplev/PNPI	Cold neutron source benchmark experiment on an existing critical facility
5e	Edgar Koonen/SCK-CEN	Embrittlement, fracture toughness and fatigue crack growth on irradiated series 5000 and 6000 aluminum alloys
5f	Guy Gistau/Air Liquide	Survey of operating cold neutron sources and their comparison based on indentical evaluation criteria
5g	Jean-Luc Minguet/ Technicatome	Comparison of regulations for research reactors in various participating countries

Table 3

New R&D work just performed

Nr.	Affilation/Name	Topic
5h	Kir Konoplev/PNPI	Detritiation of D_2O in connection with waste detritiation

It might well be that we will hear more about several of these new R&D topics on the next IGORR meeting.

4. Conclusion

This has been the fifth R&D workshop in the fifth IGORR meeting. The active discussions which developed and the numerous presentations which were given demonstrated once again that this "open market" for the exchange of information seems to meet the demands and interests of the IGORR members, indeed.

RESULTS OF ANS U_3Si_2 FUEL STUDIES

presented to the 5th meeting of the
International Group on Research Reactors

D. L. Selby

November 1996

Appendix A

The primary goal of the HANS-3 test was to compare specimens from hot-rolled plates to previous results for powder mixtures

- HANS-1 and -2 tests were cold pressed mixtures of fuel and Al powders

- HANS-3 specimens were from hot-rolled plates with typical fuel/Al contact and interaction zones

- Since the fuel performance model was based on these results, it was important to compare the hot-rolled plate results with the cold pressed mixture specimens

- Most specimens were low volume fraction, HEU, U_3Si_2, U_3Si, U_3O_8, or UAl_x

- Some high volume fraction specimens of LEU and MEU were included to test lower fission rates and burnups

6

Microstructural examination confirmed the fuel performance model for the low volume fraction fuels

- Diffusion of Al into the fuel particles was enhanced by the improved contact obtained from the hot-rolled plate specimens

- Conclusions remain the same as for the earlier tests for U_3Si_2 and the backup fuels

- U_3Si_2 expected to perform well at ANS conditions at temperatures up to 400°C at volume fractions up to 20%

High U_3Si_2 volume fraction MEU and LEU specimens failed at temperatures ~~about~~ above about 250°C

- Specimens were from existing plates at fuel volume fractions of 0.41 to 0.44

- Fission rates and temperatures for the failures were significantly above those of any existing Al-plate-type test reactor

- Failure occurred only after complete depletion of the Al matrix due to particle growth and Al diffusion into the fuel

Appendix B

R&D Topics from ANS Closeout of R&D Activities

Doug Selby/Oak Ridge National Laboratory ORNL

4f: HANS-3 fuel capsule irradiation (in HFIR) evaluation

a) G.L. Copeland, G.L. Hofman, and J.L. Snelgrove, "Postirradiation Examination of HANS-3 Capsule", ORNL/M-4864, Memorandum to D.L. Selby, September 29, 1995.

b) G.L. Hofman, J. Rest, and J.L. Snelgrove, "Aluminum-U_3Si_2 Interdiffusion and its Implications for the Performance of Highly Loaded Fuel Operating at Higher Temperatures and Fission Rates" (A Preliminary Assessment), Paper presented at the RERTR Conference in Seoul, Korea, October 1995.

4g: Meat fabrication with spherical powder fuel

A paper was presented and published as part of the RERTR Conference held in Seoul, Korea, October 1995. General conclusions of this fabrication work were that somewhat better homogeneity was obtained for higher density fuels when the spherical powder fuel was used.

4h: Centering of fuel in plate fabrication

Main report on this issue was never published. The purpose of this work was to examine the feasibility of fabricating the fuel plate so that the fuel meat was always centered in the plate even when the fuel was graded (i.e. the thickness of the fuel varied along the span of the plate). This has significant thermal-hydraulic advantages for very high power density systems, but is not considered important for most existing research reactors. The conclusion of this study was that approximately centered fuel plates could be fabricated, but the process would add significant cost to the fabrication process.

4i: Effect of flow blockage shape

a) T. Stovall et al., "Flow Blockage Analysis for the Advanced Neutron Source Reactor", ORNL-6860, January 1996.

b) J.A. Crabtree, "The Effect of Alternate Inlet Flow Blockage Shapes on Heat Transfer and Flow Behavior in Rectangular Channels", Master's Thesis, the University of Tennessee, Knoxville.

c) D.K. Felde, T.K. Stovall, and J.A. Crabtree, "Experimental Investigation of Flow Blockage Phenomena in the ANS Core", American Nuclear Society 1995 Annual Meeting, Philadelphia, June 25, 1995.

4j: Reduced pH effects on aluminum corrosion

a) S.J. Pawel, D.K. Felde, and R.E. Pawel, "Influence of Coolant pH on Corrosion of 6061 Aluminum Under Reactor Heat Transfer Conditions", ORNL/TM-13083, October 1995.

4k: Final summation of the ANS thermal hydraulic test program

a) G.L. Yoder et al., "Update to Advanced Neutron Source Steady-State Thermal-Hydraulic Report", ORNL/TM-12398/R1, November 1995 (published May 1996).

b) N.C.J. Chen, M.W. Wendel, and G.L. Yoder, "Transition to Natural Circulation With and Without Depressurization for the Advanced Neutron Source Reactor", Proceedings 1994 American Society of Mechanical Engineers (ASME) Winter Annual Meeting, Chicago, Nov. 6-11, 1994.

c) G.E. Giles, "Advanced Neutron Source Reactor Thermal Analysis of Fuel Plate Defects", ORNL/TM-13072, August 1995.

d) M. Siman-Tov, D.K. Felde, J.L. McDuffee, and G.L. Yoder, "Experimental Study of Static Flow Instability in Subcooled Flow Boiling in Parallel Channels", 4th American Society of Mechanical Engineers/Japanese Society of Mechanical Engineers, Maui, Hawaii, January 1995.

e) M. Siman-Tov et al., "Thermal-Hydraulic Correlations and Experimental Database for the Advanced Neutron Source Reactor - Closing Report", ORNL/TM-13081.

f) M. Simon-Tov et al., "FY 1995 Progress Report on the ANS Thermal-Hydraulic Test Loop Operation and Results", ORNL/TM-12972, November 1996.

g) M.W. Wendel, N.C. Chen, and G.L. Yoder, "Updated Pipe Break Analysis for Advanced Neutron Source Reactor Conceptual Design", Proceedings 1994 American Society of Mechanical Engineers (ASME) Winter Annual Meeting, Chicago, Nov. 6-11, 1994.

4l: Final Results from the HANSAL Aluminum Irradiation Tests and Examination of BNL Data

a) K. Farrell, "Assessment of Aluminum Structural Materials for Service Within the ANS Reflector Vessel", ORNL/TM-13049, August 1995.

b) D.J. Alexander is in the process of preparing a final report on the HANSAL aluminum irradiation studies performed for the ANS project. A draft report has been written and it is presently being reviewed. This report is expected to be ORNL/TM-13084 and should be published early in 1997.

4m: Irradiation Creep in Aluminum

a) "Experimental Simulation of Radiation Creep in the ANS Core Pressure Boundary Tube", ORNL/TM-13063, August 1995

ANS Fuel Plate Stability References

Doug Selby/Oak Ridge National Laboratory ORNL

1. W.F. Swinson, R.L. Battiste, and G.T. Yahr, "An Experimental Investigation of the Interaction of Primary and Secondary Stresses in Fuel Plates", PVP-Vol. 338, Pressure Vessels and Piping Codes and Standards, Volume 1, ASME, 1996.

2. W.F. Swinson, R.L. Battiste, L.R. Luttrell, and G.T. Yahr, "An Experimental Investigation of the Structural Response of Reactor Fuel Plates", Experimental Mechanics, Volume 35-Number 3, September 1995.

3. W.F. Swinson, R.L. Battiste, and G.T. Yahr, "Structural Thermal Tests on Advanced Neutron Source Reactor Fuel Plates", ORNL/TM-13062, August 1995.

4. W.F. Swinson, R.L. Battiste, and G.T. Yahr, "Circular Arc Fuel Plate Stability Experiments and Analyses for the Advanced Neutron Source", ORNL/TM-12977, August 1995.

5. W.F. Swinson, L.R. Luttrell, and G.T. Yahr, "An Examination of the Elastic Structural Response of the Advanced Neutron Source Fuel Plates", ORNL/TM-12712, 1994.

6. W.F. Swinson, R.L. Battiste, L.R. Luttrell, and G.T. Yahr, "Follow-up Fuel Plate Stability Experiments and Analyses for the Advanced Neutron Source", ORNL/TM-12629, November 1993.

7. W.F. Swinson et al., "Fuel Plate Stability Experiments and Analyses for the Advanced Neutron Source", Journal Pressure Vessel Technology, American Society of Mechanical Engineers, July 1993.

8. W.F. Swinson et al., "Structural Response of Reactor Fuel Plates to Coolant Flow", Pressure Vessels and Piping, Vol. 258, Flow-Induced Vibration and Fluid-Structure Interaction, 21-33, American Society of Mechanical Engineers, 1993.

9. W.F. Swinson, R.L. Battiste, L.R. Luttrell, and G.T. Yahr, "Fuel Plate Stability Experiments and Analyses for the Advanced Neutron Source", ORNL/TM-12353, May 1993.

10. W.K. Sartory, "Nonlinear Analysis of Hydraulic Buckling Instability of ANS Involute Fuel Plates", ORNL/TM-12319, March 1993.

The above list provides the major ANS reports dealing with plate stability issues.

The Space-Time Kinetics of the FRM-II Reactor

C. Döderlein[1]

The importance of multidimensional or space-time kinetics in large power reactor-cores is a well-known fact. The size of these cores and the weak neutronic coupling of their different parts limit the applicability of the conventional point kinetics in the analysis of localized reactivity insertions (eg. control rod ejection).

Yet, the analysis of the very compact, D_2O-reflected cores of modern research reactors too reveal a space-time kinetics phenomenon. This effect, which has its cause in the small core size and the long lifetime of thermal neutrons in heavy water (D_2O), has been studied at the instance of the FRM-II's compact core KKE7 by means of analytical and Monte Carlo neutronics calculations. Eventually, a new method of kinetics calculations, which covers this phenomenon, has been developped. Transient calculations with this method allowed to quantify the effect's contribution ot the inherent safety of the FRM-II reactor.

1) The physical basis of the reflected compact core kinetics phenomenon

The compact core concept, as it is going to be realized in the FRM-II reactor near Munich (Germany) [1], consists in a small annular-cylindrical core of 0.7 m height and 0.2 m diameter, which is placed in the center of a D_2O-filled moderator tank of 2 m diameter. This arrangement, called "inverse flux trap", leads to the formation of a maximum of the thermal neutron flux outside the core, which is essential for the reactor's vocation as beam tube neutron source. By choosing light water (H_2O) as in-core moderator and coolant, the core size could be further minimized for the sake of performance, thus yielding an unperturbed flux maximum of about $8 \cdot 10^{14}$ cm^{-2} s^{-1} with a reactor power of 20 MW.

Figure 1 shows schematically the average histories of the neutrons in the chain reaction cycle, detailled in energy (vertical dimension, with high energy at the top) and space (horizontal dimension, simplified in core- and moderator tank-region). Of 100 fission neutrons, about 73 escape from the core with high energy into the moderator tank. Whereas about 25 of these neutrons are almost immediately reflected back into the core, the remaining are moderated in the heavy water, whereupon a part (about 18) re-enter the core by diffusion. These moderator tank neutrons, moderated in the D_2O, account for about 30% of the fissions that give rise to the next generation of fission neutrons.

There are hence two distinct moderation environments, with very contrasting neutronic properties: inside the core, with its light water and high enriched uranium, that limit the average prompt neutron lifetime to some 15 microseconds, and the moderator tank, the D_2O of which permits the thermal neutrons to diffuse about two milliseconds before re-entering the core.

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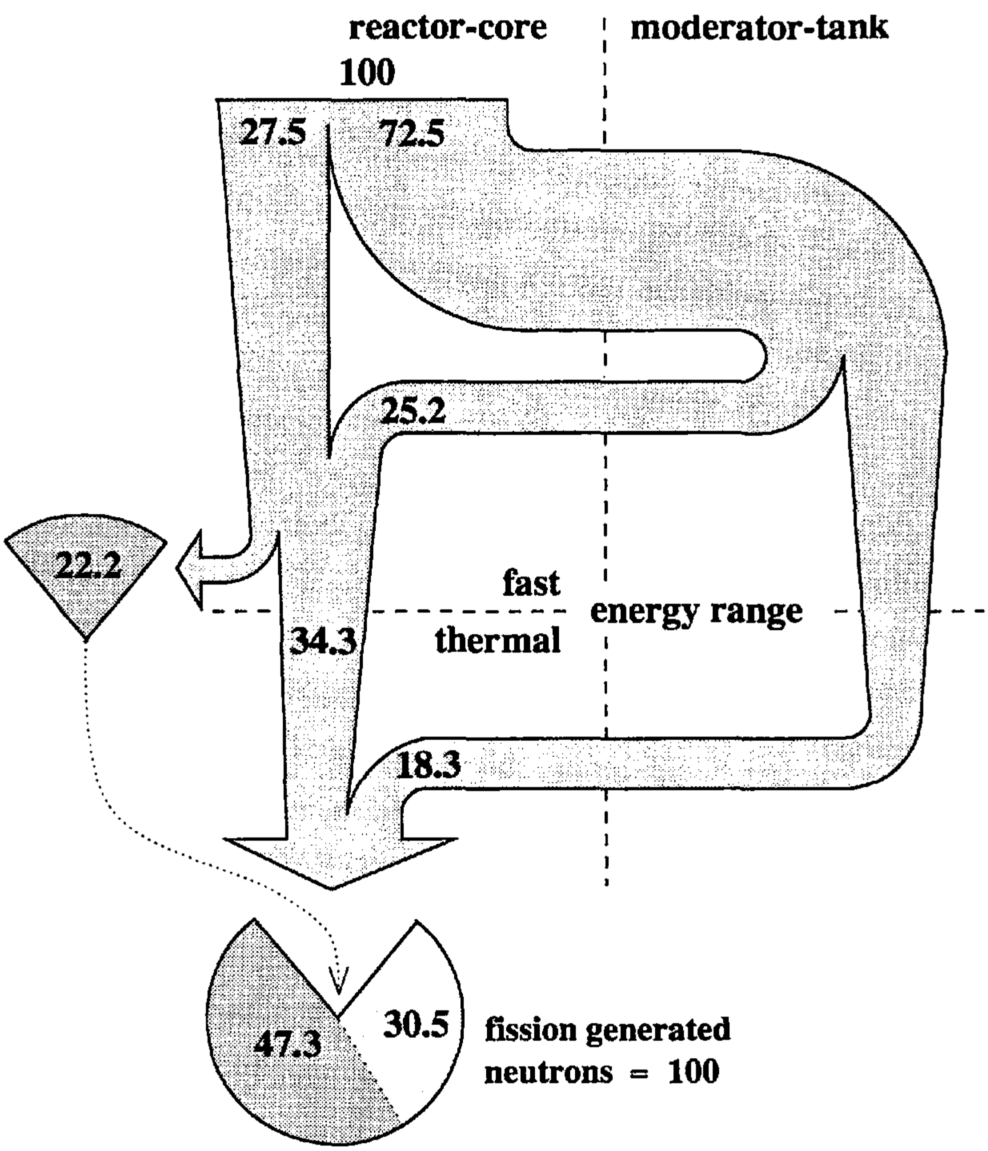

Figure 1: Schematical depiction of the neutron life cycle in the KKE7. Decreasing bar-width symbolizes absorption- and leackage-losses, circle sections represent the contributions to the production of the next generation of fission neutrons; the thermal energy limit is 0.6 eV. The part "reactor-core" comprises the control rod region. Therefore, a part of the 34.3 neutrons moderated in this path, serv for reactor control, i.e. are absorbed in the absorber section of the control rod.

At steady state power, these two paths of moderation are in equilibrium and the "lateness" of the moderator tank neutrons has no effect.

In a power rise, however, the re-flux of thermal neutrons from the moderator tank will follow the time evolution of the neutron population in the core with a certain delay. Given the importance of the moderator tank neutrons for criticality (30% of all fissions!), this delay will have an impact on the reactor's kinetics in form of an additional "inertia", opposed to rapid changes of reactor power.

As this phenomenon is not covered by the conventional point kinetics, a new calculation scheme had to be developped to quantify the significance of this effect for the reactor's kinetic properties.

2) The two points-two groups kinetics model

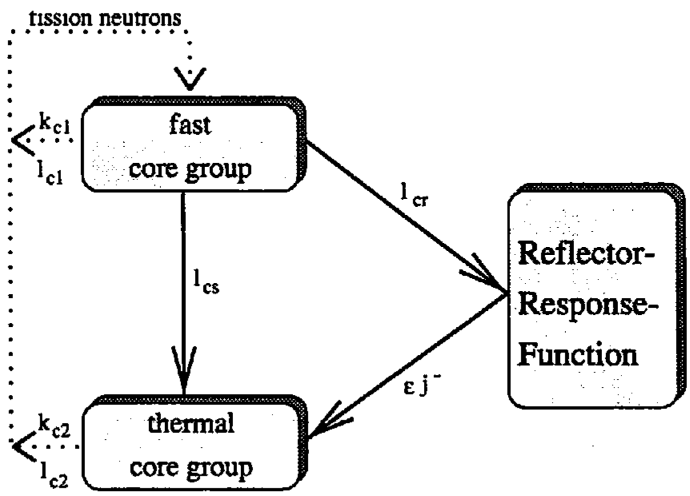

Figure 2: Schematical illustration of the two points-two groups kinetics model with the continous Reflector Response Function $G(\tau)$. The arrows represent the neutronical coupling; the *k*-symbols designate coupling efficiencies, the *l*-symbols stand for transfer delays.

Initially based on a proposal of Difilippo *et.al.* [2], a specialized model of the compact core kinetics has been developped [1] (fig 2). The problem is separated in two spatial regions or "points", namely the core and the moderator tank. The neutron population of the core region is represented in two energy groups, fast and thermal, whereas the kinetic properties of the reflector tank are described by the so-called " Reflector-Response-Function" (RRF).

This time dependent function $G(\tau)$, first proposed by Shinkawa *et.al.* [3] and depicted in figure 3 for the case of the FRM-II, relates the re-flux of thermal neutrons $j^-(t)$ from the tank into the core at time t to the outflow $j^+(t)$ of fast neutrons at an earlier time t-τ:

$$j^-(t) = \int_0^\infty j^+(t-\tau)\cdot G(\tau)\,d\tau \;\cdot$$

In other words: a δ-pulse of neutrons, which leave the core with high energy at the instance t, will return moderated into the core later at time t+τ with an intensity proportional to $G(\tau)$. Moderation and diffusion in the moderator tank stretch the δ-pulse to the form depicted by the RRF.

The first moment G_1 of the normalized response function

$$G_1 = \int_0^\infty \tau \cdot G(\tau)\,d\tau$$

gives the average delay of the thermal moderator tank neutrons, which equals to G_1= $1.8\cdot10^{-3}$ s for the FRM-II.

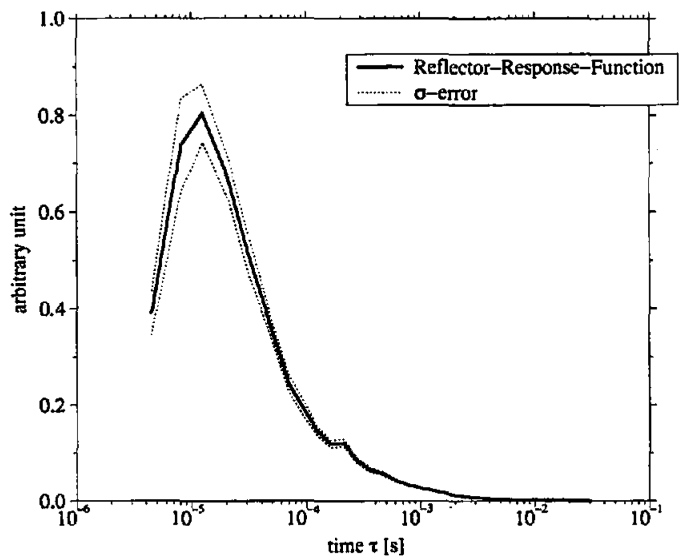

Figure 3: Linear-log plot the the Reflector Response Function $G(\tau)$ of the FRM-II, without experimental installations in the moderator tank. The function has been determined by Monte Carlo calculations and is normalized to

$$\int_0^\infty G(\tau)\, d\tau = 1.$$

The dotted lines designate the statistical 1 σ-error intervall of the Monte Carlo calculation.

The distinct elements of this model (i.e. points and groups) are linked by coupling factors so as to represent the actual flow of neutrons (cf. fig. 1 and 2). The coupling factors consist of coupling efficiencies (k in fig. 2) and time constants (l), which designate an average delay associated with the transfer of neutrons from one point or group to another.

The model can be understood as a variant of the points kinetics method, enhanced by a time-varying core spectrum and a delayed source term[2]. It can be formulated by a system of coupled differential equations, which lend itself easily to numerical integration. The reflector response function $G(\tau)$ is thereby represented by its moments G_n, with an order n up to four. The values of these moments, as well as those of the coupling factors, can be determined by means of Monte Carlo calculations and a statistical analysis of the neutron histories simulated. In order to validate the factors obtained in this way, an analytical model of time-dependent core neutronics in cyclindrical geometry has been developed. Its results, obtained with minimum computational effort, correspond in first and second order to those of the expensive Monte Carlo calculations.

[2] In Difilippo's original proposal, the moderator tank was, like the core, represented by two coupled points. This model, however, proved to be unable to reproduce the rather complex time structure of the real response function. The same objection applies to the model of Ott, who proposed to treat the delayed moderator tank neutrons like the conventional delayed neutrons, that is by families of "pseudo delayed neutrons."

3) Transient calculation of the FRM-II kintic behaviour

The concept of the FRM-II reactor as an university research tool and its location near to an urban area imposed a rigorous safety philosophy from the very beginning of the design process. This philosophy favoured a simple and inherently safe lay-out, as for example an unpressurized primary coolant loop without active control valves. This attitude has led, as well, to an elimination of all effects that might result in an uncontrolled fast injection of reactivity. The assumptions for the reactivity insertion transients, calculated with the two points-two groups model and presented here as an example of its application, were therefore completly hypothetical.

The calculations show that in the case of subprompt reactivity injections[3], the conventional delayed neutrons dominate the excursion. The "inertial" effect of the delayed moderator tank neutrons appears late in the transient, at a point where the active shutdown devices would already have made subcritical the reactor.

For superprompt over-reactivites, however, the space-time kinetic phenomenon is visible from the beginning, manifesting itself in a slowing-down of the power excursion (compared to a simple point kinetics calculation). The use of the new calculation model is therefore recommended for the analysis of this kind of transient.

The case of the theoretical maximum reactivity insertion corresponds to an instantaneous displacement of the central control rod from the highest to the lowest efficiency position. The reactor periode resulting from this unrealistic assumption, as calculated with the two points-two groups model, lies well above the critical value of 4 ms. This critical value is believed to be, for this kind of reactor, a threshold for the BORAX phenomenon. By exceeding this threshold, the compact core KKE7 is physically unable to give rise to a power excursion fast enough to produce the BORAX effect.

The two points-two groups method hence demonstrates the inherent safety of the FRM-II reactor for this extreme and virtually impossible assumption.

References:

[1] Chr. Döderlein: "Dynamics and kinetics of a reflected research reactor core," Dissertation at the Technical University Munich, 1994 (in German)

[2] F. Difilippo, M. Abu-Shehadeh, R. Perez: "Two-Point and Two-Energy Group Kinetics Model of the ANS Reactor," *Trans.Am.Nucl.Soc.*, **59**, 347 (1989)

[3] M. Shinkawa, Y. Yamane, K. Nishina, H. Tamagawa: "Theoretical Analysis fo Coupled-Core Reactors with the Method of the Moderator Region Response Function," *Nucl.Sci.&Eng.*, **67**, 19-33 (1978)

[3] injected reactivity $\rho < 1$ \$

IGORR 5

SESSION 6

BUSINESS MEETING AND CLOSING SESSION

BUSINESS SESSION

Colin West chaired the Business Session of IGORR-V. He resigned as Chairman of IGORR (after six years of service) and conveyed the resignation of the Technical Program Coordinator, Kathy Rosenbalm, who was not present at the meeting. Professor Klaus Böning, a leader of the FRM-II Project, was elected as the new Chairman by unanimous acclamation. A vote of thanks was made to the former officers.

The sixth IGORR meeting, according to the agreed-upon rotation (America, Europe, Asia) should be held in Asia, and the new chairman will investigate the possible host organizations.

Grateful thanks were expressed to the French organizers of IGORR-V and to Technicatome for their support.

Colin D. West

IGORR 5

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IGORR 5

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November 04-06, 1996

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