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IGORR-IV

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

May 23–25, 1995
Gatlinburg, Tennessee

Compiled by
K. F. Rosenbalm
Oak Ridge National Laboratory

MANAGED BY
LOCKHEED MARTIN ENERGY SYSTEMS, INC.
FOR THE UNITED STATES
DEPARTMENT OF ENERGY

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Prepared by Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831,
operated by Lockheed Martin Energy Systems, Inc.,
for the U.S. Department of Energy under contract DE-AC05-84OR21400

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CONTENTS

International Group on Research Reactors

Charter

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

IGORR Organizing Committee

PREFACE

Overview

The fourth meeting of the International Group on Research Reactors (IGORR-IV) was held in Gatlinburg, Tennessee, on May 23—25, 1995. Attendance was good (55 registered participants from 28 organizations in 13 countries), which compares well with the previous meetings. Twenty-nine papers were presented in five sessions over the two-day meeting, and written versions of the papers or hard copies of the viewgraphs used are published in these Proceedings.

The meeting was a great success with much valuable information shared, and I think we are all looking forward to the next one.

K. F. Rosenbalm
Technical Program Coordinator
International Group on Research Reactors

AGENDA

Tuesday, May 23

Evening Registration		6:30-8:00 p.m.

Wednesday, May 24

	Speakers	Session times
Final Registration and Coffee		8:00-8:30 a.m.
Welcome, Opening, Agenda	C.D. West	8:30 a.m.
I. Operating Research Reactors and Facility Upgrades		**Colin West**
ATR	J.M.Ryskamp	8:45 a.m.
DR3	K. H. Nielsen	9:10 a.m.
FRJ-II	J. Wolters	9:35 a.m.
BREAK		10:00 a.m.
HANARO (commissioning status)	J.-S. Wu	10:15 a.m.
HANARO (initial criticality and reactor physics experimental program)	C.-S. Lee	
HFIR	G.F. Flanagan	11:00 a.m.
HFR	M. Bieth	11:25 a.m.
HIFAR	A. Murray	11:50 a.m.
LUNCH		12:15 p.m.
NIST	R.E. Williams	1:30 p.m.
ORPHEE Upgrades*	B. Farnoux	1:55 p.m.

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* B. Farnoux' presentation also covered ILL.

Wednesday, May 24 (cont'd)

II. Research Reactors in Design and Construction		**Bob Lidstone**
FRM-II	K. Böning	2:20 p.m.
IRF	R.F. Lidstone/ A.G. Lee	2:45 p.m.
BREAK		3:10 p.m.
PIK	K. Konoplev	3:25 p.m.
Pulsed TRIGA neutron source	W.L. Whittemore	3:50 p.m.
III. ANS Closeout Activities	C.D. West	4:15 p.m.
IV. Results of Survey on Accident and Safety Analysis Codes and Benchmarks or Other Verification & Validation Methods	A. G. Lee	4:30 p.m.
ADJOURN		5:00 p.m.
Evening Reception		6:00-7:30 p.m.

Thursday, May 25

Coffee/muffins		8:00-8:30 a.m.	

V. Research, Development, and Analysis Results — Colin West

Thermal Hydraulic Results and Calculations

ORNL Results — flow blockage tests and analyses* — flow excursion data — fuel plate stability and deflection	G.L. Yoder (ORNL)	8:30 a.m.
CHF Correlation Scheme Proposed for Research Reactors Using Plate-Type Fuel	M. Kaminaga (JAERI)	9:00 a.m.
Thermal Hydraulic Model Tests of the NIST Liquid Hydrogen Cold Source	R.E. Williams (NIST)	9:20 a.m.
Modeling/Analyses of LD_2/D_2O Reactions	R.P.Taleyarkhan (ORNL)	9:40 a.m.

U_3Si_2 Fuel Performance and Fabrication

U_3Si_2 Irradiation Performance and Thermal Conductivity Calculations*	J.L. Snelgrove (ANL)	9:50 a.m.
BREAK		10:10 a.m.
Status of development and main features of plates and dummy elements fabrication for FRM-II	J. P. Durand/ G. Harbonnier (CERCA)	10:30 a.m.
Status of development on fabrication and double grading of U_3Si_2 fuel plates	B. Pace (B&W)	10:50 a.m.

*These are results that were requested at IGORR-III.

Thursday, May 25 (cont'd)

V. Research, Development, and Analysis Results (cont'd)

Structural Materials Performance

Al 6061 Irradiation Tests and Results	D. J. Alexander (ORNL)	11:10 a.m.

Neutronics

Coupled MCNP Neutronics/Burnup and Depletion Code	C. A. Wemple (INEL)	11:30 a.m.
Graphical User Interface Simplifying MCNP Use and Burnup Calculation Capabilities	B. R. Lewis (Atom Analysis)	11:50 a.m.
LUNCH		12:10 p.m.

Severe Accident Program Results

Chemical and other energy release and fission product release from core melt events*	R.P. Taleyarkhan (ORNL)	1:30 p.m.
Fission Product Release from Molten Research Reactor Core	H.-J. Didier (Siemens)	1:50 p.m.
JAERI/ORNL tests and analyses on transient heating of U_3Si_2 miniplates in the NSRR	T. Fuketa/ R.P.Taleyarkhan (JAERI/ORNL)	2:10 p.m.

VI. Workshop on R&D Needs — Klaus Böning

New R&D needs to be addressed	2:30 p.m.

VII. IGORR-IV Group Photo — 3:00 p.m.

*These are results that were requested at IGORR-III.

Thursday, May 25 (cont'd)

VIII. Business Session	**Colin West**
Discussion of IGORR-V Election of IGORR Chairman	3:15 p.m.
Closing	3:30 p.m.
ADJOURN	3:45 p.m.

XA04C1674

Potential for New Societal Contributions from the Advanced Test Reactor

John M.Ryskamp, Julie E. Conner,
Fred W. Ingram
(INEL, EG&G Idaho)

TABLE I

ATR Experiment and Isotope Production Fluxes at 250-MW Core Power

Number	Diameter (cm)	Name	Thermal Flux ($cm^{-2}s^{-1}$) (2200 m/s)	Fast Flux ($cm^{-2}m^{-1}$) (>1 MeV)
8	2.22	B	5.75×10^{14}	1.85×10^{14}
4	3.81	B	2.48×10^{14}	0.37×10^{14}
14	1.59	H	4.28×10^{14}	3.83×10^{14}
8	1.59	A	4.82×10^{14}	3.83×10^{14}
4	1.59	A	4.64×10^{14}	5.25×10^{14}
4	12.7	I	3.89×10^{13}	3.05×10^{12}
16	8.26	I	7.71×10^{13}	3.01×10^{12}
4	3.81	I	$1.89 \times 10^{1\cancel{4}3}$	$0.07 \times 10^{1\cancel{4}3}$
7	6.05	FT[a]	8.80×10^{14}	1.90×10^{14}
2	10.16	FT[a]	8.80×10^{14}	1.90×10^{14}

[a]Flux traps (nominal flux unperturbed, water-filled loop).

Potential for New Societal Contributions from the Advanced Test Reactor, *John M. Ryskamp, Julie E. Conner, Fred W. Ingram (INEL, EG&G Idaho)*

The mission of the Advanced Test Reactor (ATR) at the Idaho National Engineering Laboratory is to study the effects of intense radiation on materials and fuels and to produce radioisotopes for the U.S. Department of Energy (DOE) for government and commercial applications. Because of reductions in defense spending, four of the nine loop test spaces will become available in 1994. The purpose of this paper is to explore the potential benefits to society from these available neutrons.

The ATR is a 250-MW(thermal) light water reactor with highly enriched uranium in plate-type fuel. Forty fuel elements are arranged in a serpentine pattern, as shown in Fig. 1. The ATR uses a combination of hafnium control drums and shim rods to adjust power and hold flux distortion to a minimum. The different quadrants of the ATR can be operated at significantly different power levels to meet a variety of mission requirements. Irradiation positions are available at various locations throughout the core and beryllium reflector.

Table I summarizes the flux levels at various ATR reflector and loop positions. These fluxes are maintained with a relatively constant axial flux profile throughout cycles that last 35 to 42 days. These neutrons can be used for testing and irradiation programs that support commercial reactor license extension, advanced fuel development, materials effects studies, failure cause/effect studies, coolant chemistry evaluations, prototype testing programs (such as for space), isotope production, and basic research.

Radioisotope production falls into three categories: medical, industrial, and research. An approximate breakdown of radioisotopes produced by DOE includes medical (65%), research (20%), and industrial (15%), with a total product listing of ~1100 variations of isotopes of differing forms and enrichments. Serious shortages of radioisotopes and lack of existing inventories threaten U.S. industrial competitiveness and even more seriously threaten the lives of seriously ill patients requiring treatment from nuclear medicine. Table II shows the current and proposed isotope production at ATR along with the applications that benefit society.

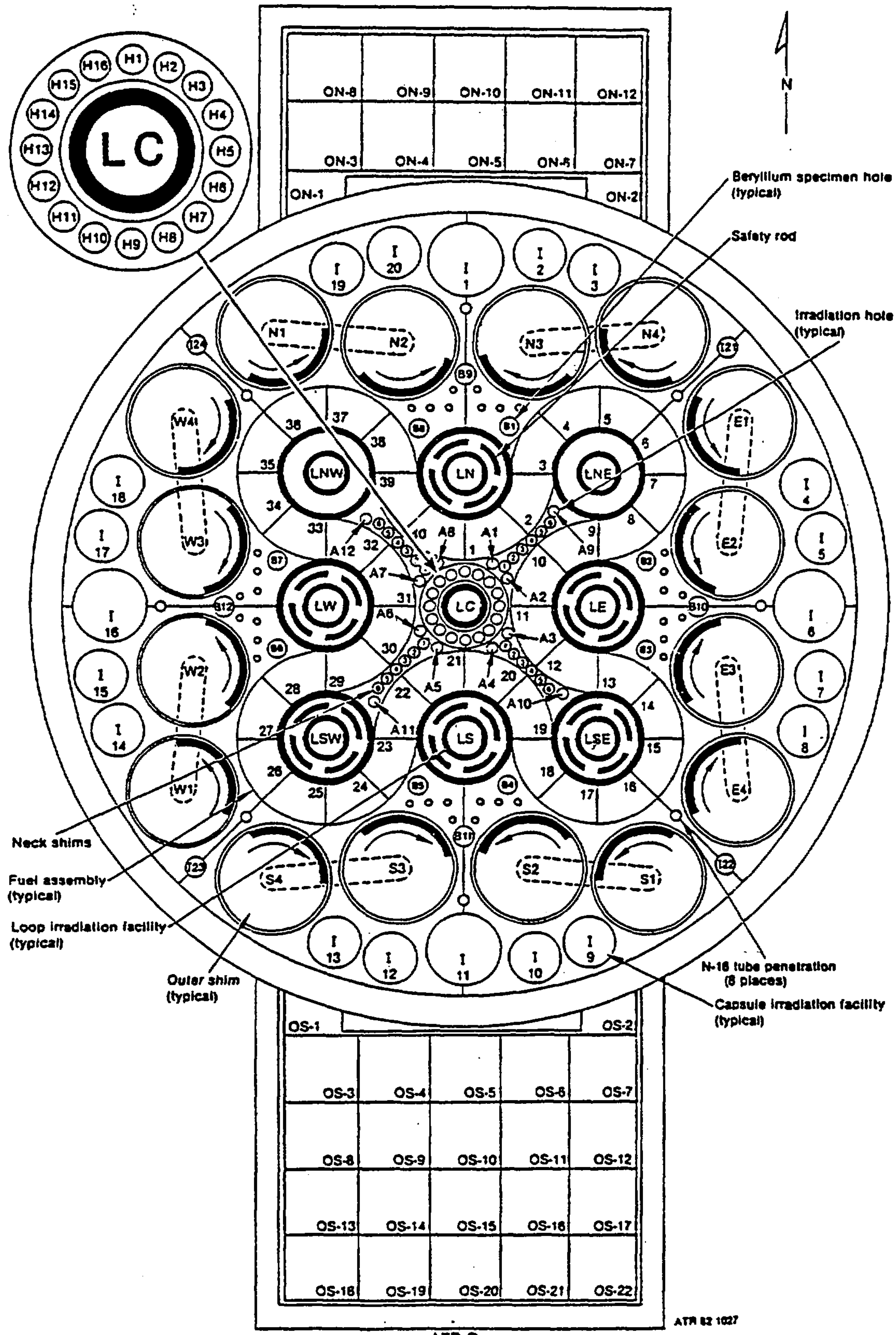
Plutonium-238 is used to support the development of radioisotope thermal-electric generators (RTGs) for deep-space exploration missions. These ^{238}Pu RTGs have supported National Aeronautics and Space Administration (NASA) missions including *Apollo*, *Voyager*, and *Galileo*. Current NASA

TABLE II

Current and Proposed ATR Isotope Production

Isotope	Current Production (Ci/yr)	% of World Need
Ir-192	1.5M	50
Co-60	1.0M	10
Ni-63	250	50

Isotope	Proposed Production (Ci/yr)	Application
Strontium-89	3	Bone Cancer, Pain Relief
Ytterbium-169	120	Cancer Diagnostics
Sulphur-35	600	Bone Disease
Phosphorus-32	5	Leukemia Treatment
Selenium-75	70	Brain Imaging
Iodine-125	500	Diagnostic Imaging
Yttrium-90	25	Liver Cancer
Cadmium-109	0.5	Pediatric Imaging and Environmental Research



ATR Core
Cross-sectional Diagram

Fig. 1. Cross-sectional diagram of the ATR.

capabilities of ATR in support of ^{238}Pu production. Results of the study indicated that production of up to 13 kg ^{238}Pu/yr in the ATR is feasible.[1]

All nine of the experimental loop facilities are currently composed of pressurized water systems that are independent of the reactor primary coolant. This provides the opportunity for advanced fuel development testing at high neutron flux levels in various flow temperature, pressure, velocity, and chemistry conditions. For example, several types of advanced plutonium-based fuels are under consideration for disposition of weapons-grade plutonium in reactors. Some of these fuel forms will require extensive development programs. The ATR is capable of playing a key role in the development and licensing of these and other fuel types.

In summary, the ATR is a unique, high-power test reactor capable of supporting the current DOE mission and producing radioisotopes. Space available for radioisotope production and fuels or materials testing will increase by 44% in 1994, improving DOE's ability to support national needs in health care, industry, and research.

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1. B. G. SCHNITZLER, "INEL Advanced Test Reactor Plutonium-238 Production Feasibility Assessment," *Proc. Space Nuclear Power and Propulsion Conf.*, Albuquerque, New Mexico, 1993, Vol. 1, p. 143.

XA04C1675

IGORR-IV

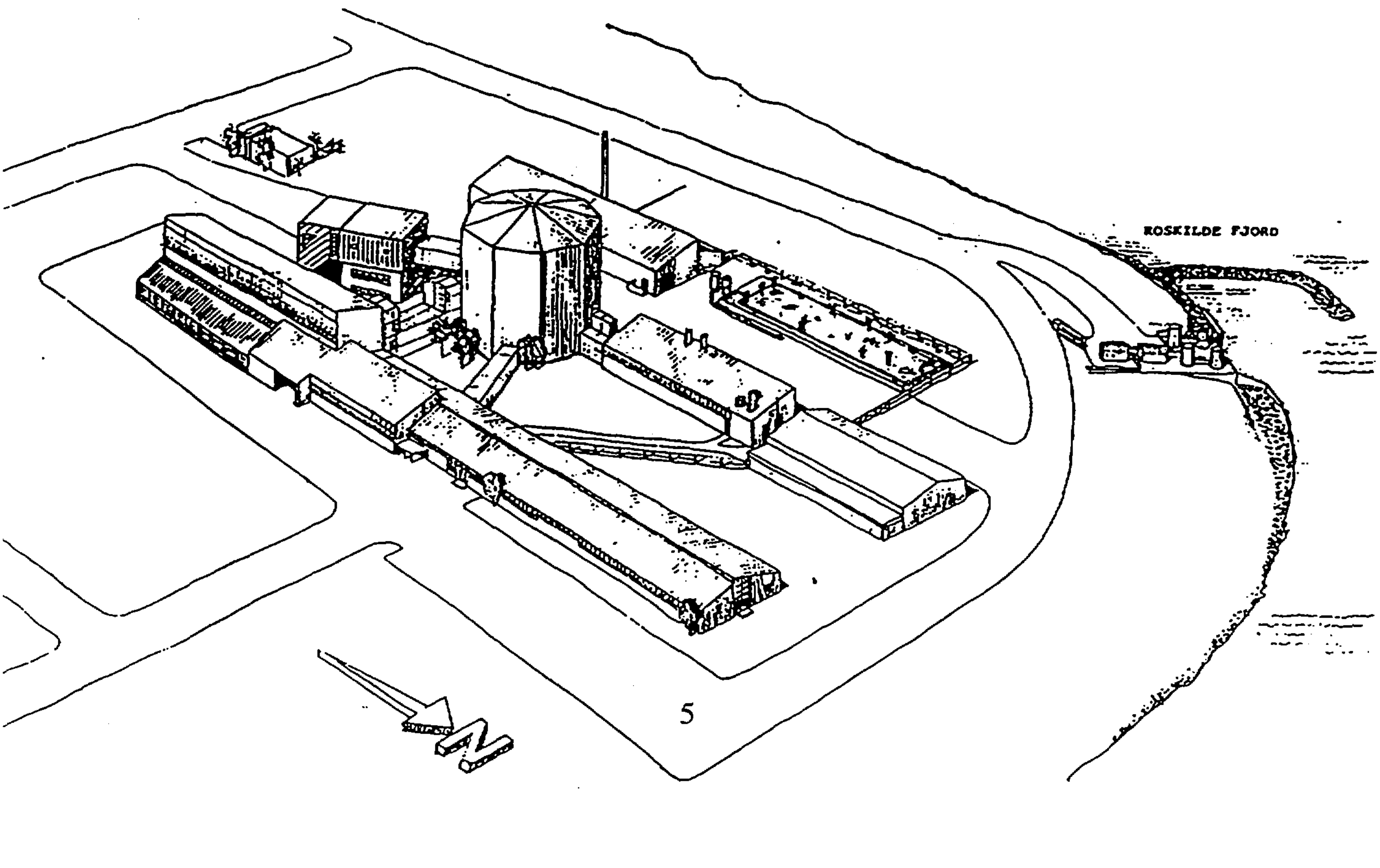
4th Meeting of the International Group on Research Reactors

May 24-25, 1995

Park Vista Hotel, Gatlinburg, Tennessee, USA

Absorber Rigs - a Better Concept than Burnable Poison.

Kirsten Hjerrild Nielsen
Reactor DR 3
Risø National Laboratory
DK-4000 Roskilde, Denmark



Introduction.

The reactor, DR 3, at Risø National Laboratory in Denmark is a 10 MW heavy water cooled and moderated research reactor. DR 3 is of a design similar to the British "PLUTO" type. DR 3 reached criticallity for the first time January 15, 1960, and its regular operation at power began in November 1960.

The reactor is controlled by seven coarse control arms (CCA's), which move like signal arms between the rows of fuel elements. In addition, there are one fine control rod (FCR) and two safety rods (SR's) situated in corners of the core.

The reactor core consists of 26 fuel elements, each one containing four concentric Aluminium-clad fuel tubes, which are arranged to provide a 5 cm. centre hole for experiments. In 1988 the conversion from HEU (High Enriched Uranium) to LEU (Low Enriched Uranium: U_3Si_2/Al) started. Since December 1990 DR 3 has run on a full LEU core.

The operation cycle is 4 weeks of which 23½ days is continious operation and 4½ days is shut down. Operational statistics have been extremely good, with a utilization exceeding 80%.

DR 3 is originally built as a Material Testing Reactor, but today it is used as a Multipurpose Research Reactor. With a cold neutron source, six three-axis spectrometers and a small angle neutron scatter instrument DR 3 is appointed as a Large European Beam Facility and these neutron beam instruments are intensively used by researchers from Risø and from the other EEC-countries. The main production activities are Silicon doping, isotope production and activating analysis.

Irradiation Facilities.

DR 3 has a large production of Silicon irradiation - almost 30 tons in 1994. There is one 5-in. horizontal facility and another similar is just about to be built. Vertically we have five 3-in. facilities in the Graphite, two 4-in. heavy-water cooled facilities and one 5-in. light-water cooled facility.

To improve the conditions for the Silicon production and the experiments it would be desirable to be able to control the flux distribution in the core radially as well as axially.

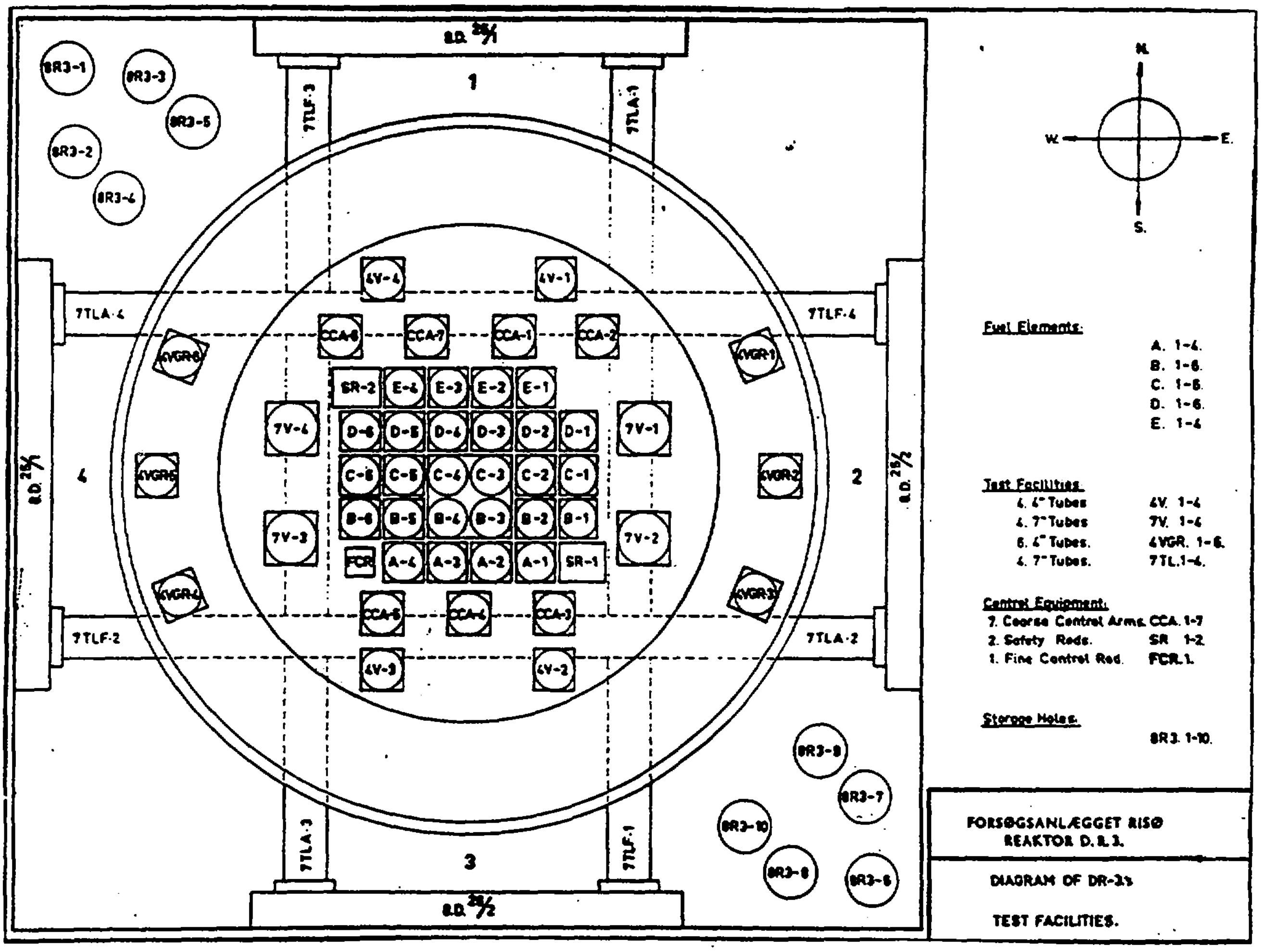


Figure 1: Test Facilities at DR 3.

During a normal reactor run the position of the CCA's will vary from about 17° to about 28° on account of burnup. These variations affected the flux distribution considerably, not only between the upper and the lower half of the core, but radially as well. Changes of up to 80% had been seen for the neutron flux in the tops of the

A-row and E-row elements. This variation was especially causing problems with uniformity of Silicon-irradiation. Often parts of the reactor period could not be used for irradiation due to this effect.

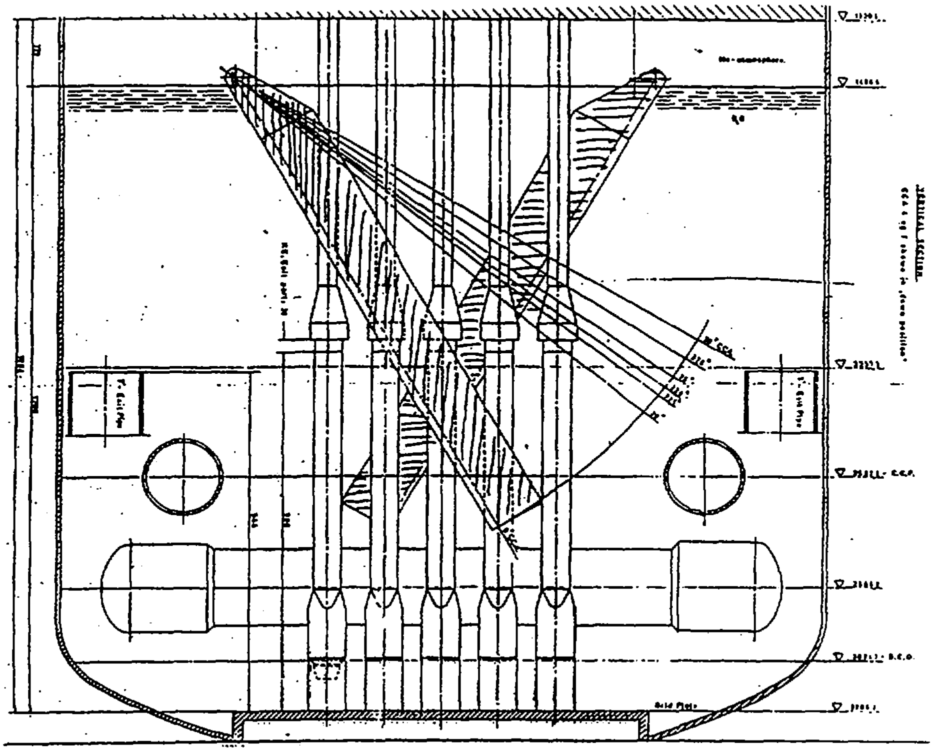


Figure 2: Various positions of the CCA's.

Disadvantages regarding burnable poison.

Burnable poison is a more common way of controlling flux distribution in the core. But it has some disadvantages compared to absorber rigs:

- Burnable poison is a once through method, and it has a limited effect - only one running period.

- Burnable poison will reduce the thermal flux inside the new fuel elements where it is already low.

- Using burnable poison will not allow control of the radial fluxdistribution without interferring with the refueling-program.

- Burnable poison makes it impossible to carry out measurements of reactivity changes caused by fuel element replacements and burnup.

<u>Controlling the flux distribution by means of absorber rigs.</u>

To improve the research facilities in DR 3 we have constructed some movable absorber rigs positioned in all those fuel elements that have no experiments installed. Thereby we obtain a constant CCA angle during a reactor run and thus a steady flux distribution in the core. In addition, we get the possibility of raising the flux exactly where it is needed.

Using movable absorber rigs opens up to new ways of controlling the fluxdistribution in the core - radially as well as axially.

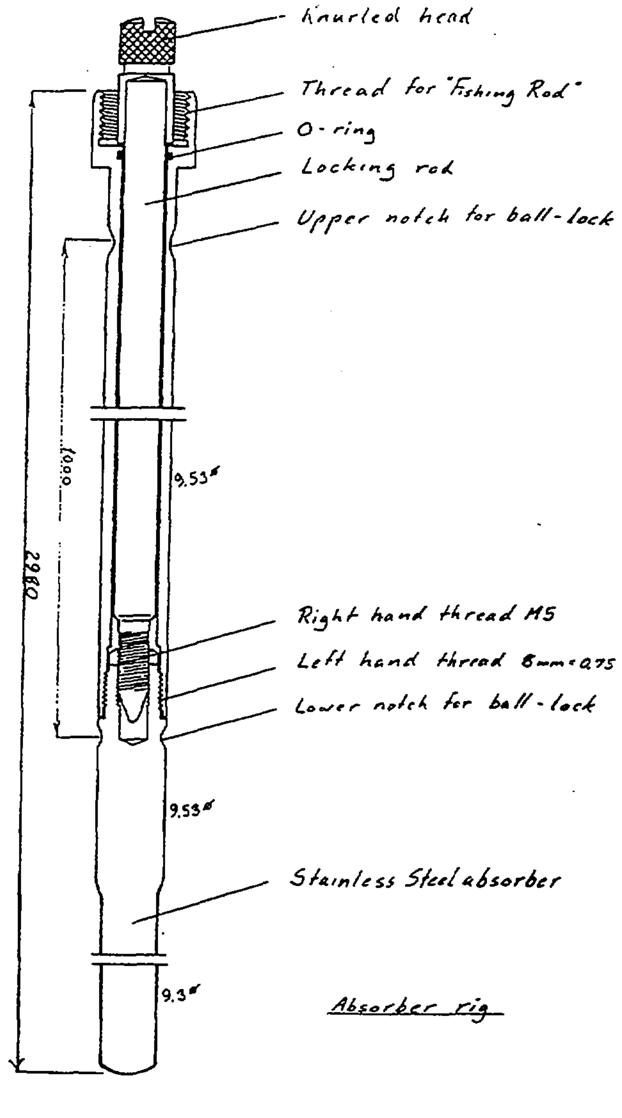
The axial effect of movable absorber rigs is to enable us to maintain an almost constant, high CCA-angle throughout the reactor cycle (Fig. 3). A CCA angle of 22° has been chosen as the most ideal position for the CCA's concerning Silicon irradiation. At 22° the CCA-group turn out of the top of the A- and E-row, and consequently the neutronflux is not reduced so much any more by the presence of the CCA's between the fuel elements.

The radial effect is gained because it becomes possible to lower the flux in the center positions and raising it in the outer fuel elements. Another possibility is to straighten a wry flux distribution.

In case an experimenter wants it, the absorber rigs gives us the opportunity of making a fluxtrap in an area of the core by positioning the absorber rigs in the opposite side of the core.

Description of the absorber rig.

The whole arrangement with the absorber rig has been made to replace a normal flux scan rig, a condition that has been decisive for the design. Consequently that leaves only very little space - less than 1 cm.



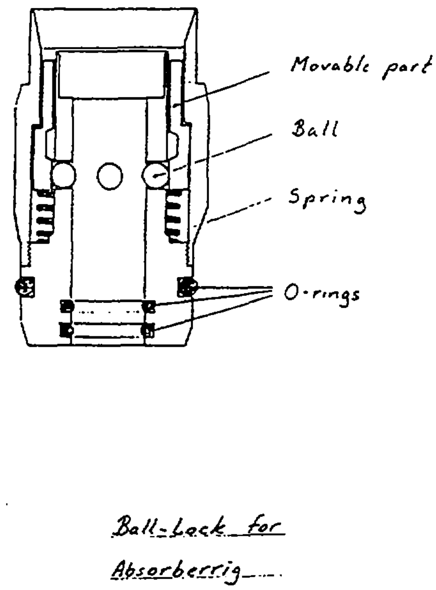


Figure 3: The absorber rig and the ball lock.

An absorber rig is made up of 3 parts (Fig. 3):

- An upper part, made entirely of stainless steel, which is removable.

- An under part with a stainless steel absorber rod which can be parked in two positions: Either in the core ("down"-position) or above the core ("up"-position).

- A ball-lock to keep the absorber in position.

The connection of the under-part to the upper part is effected by means of threads in the two parts. The inner locking rod is used to make a lock that will prevent the two parts from loosening caused by vibrations. For this purpose the main joint has left-hand thread whereas the locking rod has a normal right-hand thread.

When the two threads are tightened against each other they form an effective lock, because a tendency for one joint to come loose is counterbalanced by a tendency for the other joint to become tighter, and vice versa. An O-ring in the top together with a double O-ring in the ball-lock provides friction to the locking rod, which is desirable, when the parts are to be unscrewed. The O-rings also makes the arrangement Helium-tight. The parts of the absorber rig will provide shielding upwards in the "up"-position as well as the "down"-position.

<u>The choise of absorber material.</u>

In the original concept the absorber itself was an 8 mm. Titanium rod. The reasons for this choice were firstly that it had the right magnitude of absorption, secondly that neutronirradiated Titanium has a reasonably short halflife so that it should be possible to handle an irradiated absorber rig after a reasonable cooling time.

However, after measuring an irradiated prototype rig, it turned out that it contained rather longlived isotopes so that handling would be out of the question anyhow. With no longer any reason for using the very expensive Titanium we looked for a simpler solution, and calculations showed that a 9.3 mm. stainless steel rod would have the same absorption.

Handling the absorber rigs.

When the absorber is to be retracted from the core, the ball-lock is activated and the whole absorber rig is lifted by means of a "fishing rod". It is lifted until the ball-lock engages the lower locking notch (on the under part). A "click" indicates, that the absorber rig has been locked in position. It is only possible to operate one absorber rig at a time, because there is only one piece of tool (fishing rod).

When the lower part thus is locked securely in retracted position, first the locking rod and then the upper part will be unscrewed and removed. The absorber is now lifted 1 m. and is entirely out of the core. At the same time nothing is protuding upwards from the fuel element plug which would otherwise interfere with the shielding plugs.

The handling of the absorber rigs is done manual. Experiments have shown that there is no need for a motordrive for this purpose. It is possibel to avoid reactor trip with a retracting speed of 5 cm/s, but it will cause minor effect changes, as the FCR is not capable of compensating at that speed. In stead a retracting speed of around 2 - 2.5 cm/s is suitable and it is no problem keeping that speed using handpower.

Operation.

In two of the fuel elements we have experiments placed permanently. In addition there are four movable irradition rigs at the moment, which are running during some periods. That leaves us at least 20 positions, in which we are able to place absorber rigs.

A schedule is made for each reactor period showing the needed absorber rigs in the concerned cycle and the sequence in which the absorber rigs are retracted during the period. Fig. 4 shows the order of retraction of the absorber rigs to get a smooth flux distribution.

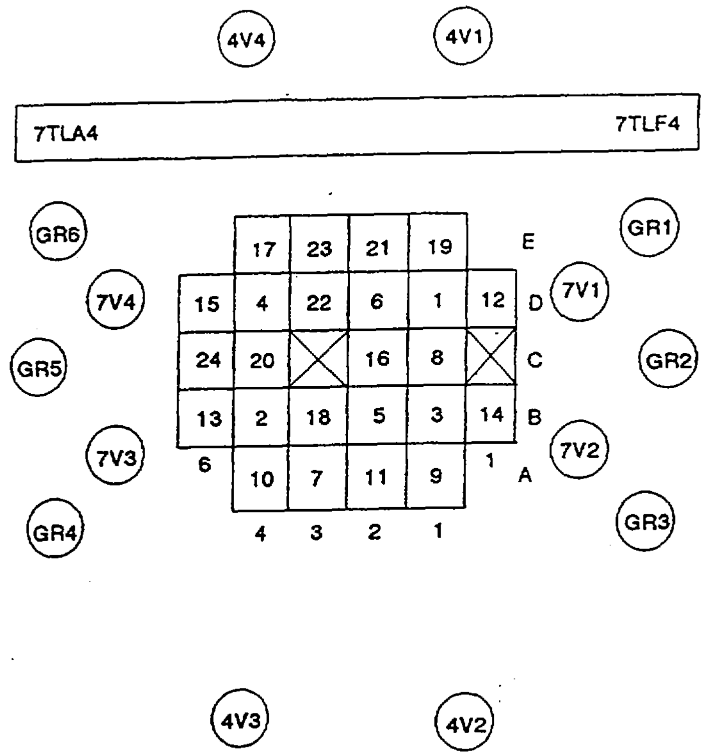


Figure 4: The order of retracting the absorber rigs.

The lowest number of absorber rigs needed is 17, but it depends on the fuelling and the CCA start-up-angle. The effect of an absorber rig depends on the core position and vary from 0.07 (B6) to 0.25 (C3).

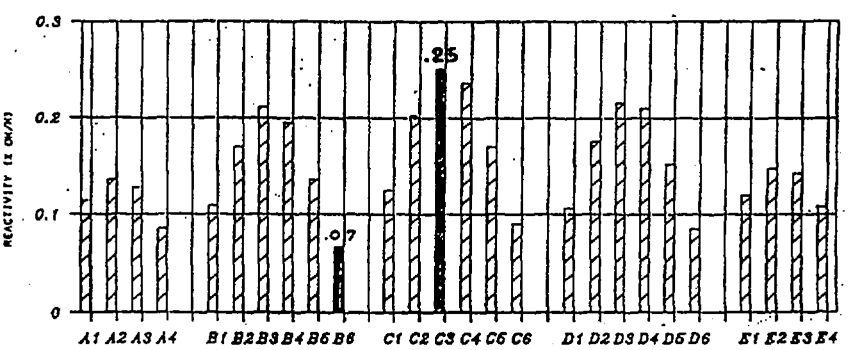


Figure 5: Reactivity worth of absorber rigs in different core positions.

Fig. 5 shows the reactivity worth of the absorber rigs in the different corepositions. During a running period the reactivity worth of burnup is around 2.8 % dk/k. 24 absorber rigs placed in the core will absorb 3.53 % dk/k, which is more than enough to compensate for the burnup.

At the beginning of the reactor period the needed absorber rigs will be in their "down" positions. When the Xenon equilibrium has been established after 30-40 hours, the CCA angle will have exceeded 22°. When the FCR reaches an upper limit of 40 cm. it is time to retract the first absorber rig. During this retraction the FCR will compensate and it should end close to its lower operating limit (6 cm.) with the absorber rig in the "up"-position. (The reactivity worth of the FCR in this 40 - 6 cm. interval is 0.35 % dk/k which is more than the worth of an absorber rig in a center position). Likewise the rest of the absorber rigs are to be retracted one by one, so the CCA angle is kept at approximately 22° throughout the reactor run.

Fig. 6 and fig. 7 show the reactivity loss by increasing burn-up with and without absorber rigs.

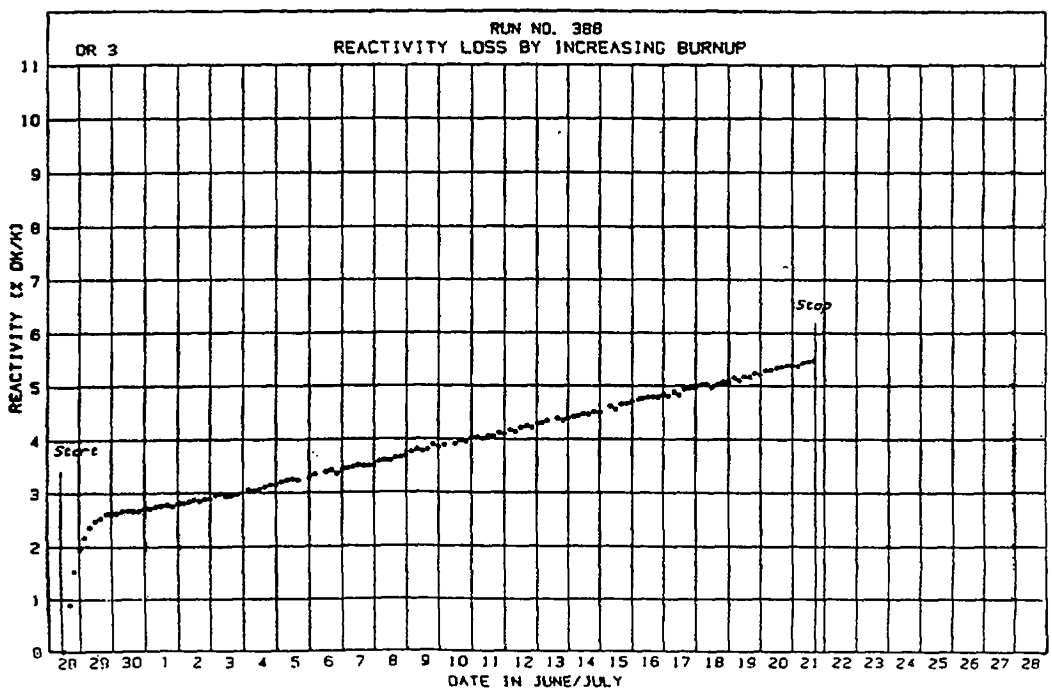


Figure 6: Reactivity loss by increasing burn-up. Without absorber rigs.

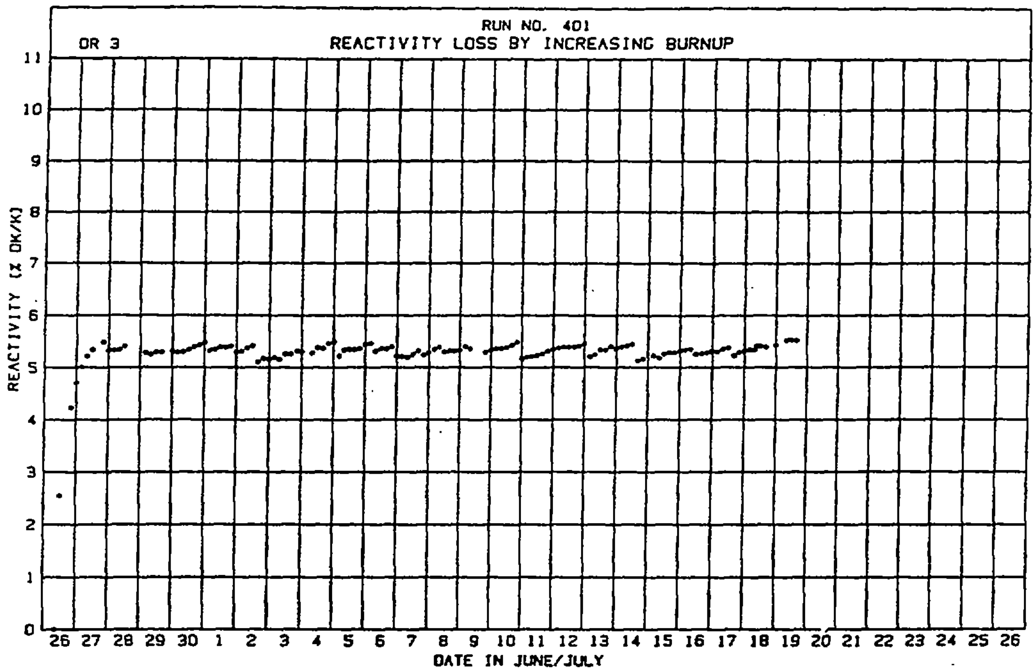


Figure 7: Reactivity loss by increasing burn-up. With absorber rigs.

<u>Operations Experiences.</u>

During the last four years we have used movable absorber rigs succesfully at DR 3. We have made some minor modifications, but basically they have performed very well. Of course there were some difficulties when we took the absorber rigs into use.

During the mechanical test of the absorber rig a minor problem occured. Descenting the absorber, it "caught" an edge in the flux scan rig. But when we had to make a number of new flux scan rigs anyway, this konus was just made bigger. Thereby this problem had been taken care of.

At the first test of the absorber rig at full power it was ascertained that the absorber was hanging in the O-rings in stead of being locked in position by the ball lock. By falling down the absorber would cause a reactor trip as well as a Helium leakage in the topvoid. A potential unpleasant situation but with no safety significance. The situation occured because of lack of instruction of the operator and his lack of familiarty with the construction. Now the operation procedure has been changed and extended. The "fishing rod" has been marked to indicate where the locking positions are. This event has not happened since.

Handling of absorber rigs during a reactor trip.

During an undesired reactor shut down the risk of getting a Xenon poisoning is larger with the absorber rigs in the core. The procedure is then to retract the absorber rigs from the "down" position. Though it depends on how many absorber rigs is left in the core and how long the trip is estimated to last. Moving all absorber rigs takes around half an hour.

There will always be at least 2 hours from reactor trip until the latest startup time before Xenon poisoning occurs. If the trip is estimated to be of short duration i. e. caused by a power failure or so, the remaining absorber rigs stay in the core during the trip. If the trip is caused by a more severe failure, which requires repair of some components, it is possible to gain some time by retracting the remaining absorber rigs to the "up"-position.

According to the moment of the operation cycle there is more or less time to gain. A full running period is 561 hours. In the tables below (Tab. 1 and Tab. 2) the signification of the absorber rigs in the core during reactor trip is shown.

100 hours 15 absorbers	Absorber rigs in the core	Absorber rigs out of the core	Gain
Before Xenon poisening	2.0 hours	3.5 hours	1.5 hours
During Xenon poisening	29.0 hours	23.0 hours	5.0 hours

Table 1: Trip after 100 hours of running. 15 absorber rigs in the core.

400 hours 5 absorbers	Absorber rigs in the core	Absorber rigs out of the core	Gain
Before Xenon poisening	2.0 hours	2.5 hours	0.5 hours
During Xenon poisening	28.5 hours	27.0 hours	1.5 hours

Table 2: Trip after 400 hours of running. 5 absorber rigs in the core.

Safety analysis.

- Break of the absorber rod or the upper part: The absorber rod will fall down to the bottom of the thimble of the flux scan rig. The reactivity worth will not change if the break occurs in the "down"-position. But if the break occurs in the "up"-position, the reactor will shut down for two reasons: Because of adding absorber material to the core, as the absorber rod falls down and trip will occur due to a negativ period.

- Exceeding pressure in the reactor tank: During normal operation the absorber rigs are locked in the "down"-position by the ball lock. It has been ascertained that even if the ball locks fail, the rigs are sufficiently heavy that even a pressure rise, caused by a transient, to the maximum allowable pressure for the reactor vessel, 0.6 bar, can not cause the rigs to be shot out of the core, which would aggravate the transient.

- Distortion of the flux distribution in the core: It is not possible to provoke this, as the total reactivity worth of absorber rigs in all 26 core positions is 3.8 % dk/k, which corresponds to normal operation.

- Momentary retraction of one absorber rig: The maximal permissible step-change in reactivity is 0.5 % dk/k. This operational limit is established on the basis of the ability of the core to resist transients without any risk of melting a fuel plate. The maximum reactivity worth of one absorber rig is 0.25 % dk/k, which is only half of the permitted change, so this is not a problem.

As a conclusion of the safety analysis we can say that the absorber rigs are not able to cause any risk for the reactor safety even in case of a glaring blunder.

Healthphysical aspects.

At the same time as testing the absorber rigs, radiation measurements have been done with the absorber in both positions. The levels were found to be quite normal. The rig provides satisfactorily shielding under all circumstances. The two O-rings make the absorber rig totally Helium-tight.

Handling of the removable upper part can be done without special precautions as it is only insignificant activated.

The absorber rod made of stainless steel becomes very activated due to the contents of Cobalt. However that is not a problem as the absorber rigs are handled inside the fuel elements.

Decommissioning.

The lifetime of the absorber rods is very long. The stainless steel contains 73 % of Iron which form 64 % of the relative absorbtion effect. Iron has the longest lifetime: 124 years. The weighed burn-up time for one absorber rod is thus around 100 years. Consequently there will be no need for making new absorber rigs.

So far we have taken two absorber rigs out of operation. They are stocked but we hope to be able to fix them. At the moment they are stored in a storage hole in the external storage block. When time comes for decommissioning, we plan to cut the absorber rigs into small pieces and put these into a container for permanent storage.

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STATUS OF THE RESEARCH REACTOR FRJ-2 AT THE RESEARCH CENTER JÜLICH, GERMANY

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Abstract:

After an outage of more than four years, the FRJ-2 resumed operation on March 10, 1995. The outage was caused in fall 1990 by corrosion detected at the tank pipes within the stuffing boxes. The stuffing boxes were replaced by a new sealing design which uses steel bellows to compensate the different thermal expansion of the reactor tank and its containment. The repair time had to be used by order of the supervisory authority to upgrade the safety concept of the plant so that it meets modern German regulations and standards. The improvements made are related mainly to the protection of the plant against earthquakes and fire. In addition, an emergency handbook had to be issued.

1. Introduction.

The FRJ-2 at the Research Center Jülich (KFA) is the most powerful research reactor in Germany. It was constructed in the years 1958 to 1962 and commissioned in 1962. In 1967, its power was increased from the originally designed power of 10 MW to 15 MW, followed by a second power increase to 25 MW in 1972. In the aftermath of the Chernobyl disaster, a thorough safety review of the plant was carried out by a German engineering company by order of the supervisory authority of the state North Rhine-Westphalia. The results were issued in the middle of 1988 /1/ revealing 13 deficits compared with modern German safety rules and standards. But it was explicitly stated that the findings were not of the order which required immediate action. In addition, an emergency handbook was required in accordance with the practice at German nuclear power stations.

Some of the required safety improvements could be realized very easily, but others needed much greater efforts in planning, evaluation and realization. Among these were the measures concerning earthquake and fire protection. The issue of an emergency handbook also proved to be very time consuming since the necessary knowledge first had to be generated.

Two years later, in November 1990, a longer period of outage was necessary in order to repair damage detected at the stuffing boxes which seal the gap between the reactor tank pipes and their containment steel tubes outside the biological shield /2/. Most effort was focused on this repair in order to resume operation as quickly as possible. But the longer the outage lasted the more it became obvious that the authority would also require most of the detected deficits to be remedied before it would agree to the reactor restart. Finally, almost all of them had to be remedied prolonging the outage to more than four years until 10th March 1995 when the reactor became critical again and it has operated at full power since 24th April.

2. Repair of the Sealing between the Tank Pipes and their Containment Steel Tubes.

The FRJ-2 is a DIDO-class tank-type research reactor cooled and moderated by heavy water. The reactor aluminum tank is housed in a massive reactor block standing on four steel pillars (Fig. 1). The space between the pillars accommodates the primary circuit. Seven aluminum pipes welded to the bottom of the reactor tank and penetrating the biological shield beneath the tank in the vertical direction connect the tank to the primary circuit.

The reactor tank is surrounded by a graphite reflector enclosed within a double-walled steel tank. This steel tank, together with its liner tubes for the various penetrations, constitutes the containment for the graphite-helium system and also for the reactor tank. All penetrations are sealed at the outside of the biological shield so that a leak in the reactor tank and its pipes would only fill the aluminum/steel tank interspace, which is designed such that the water circulation would not be interrupted.

The sealing between the tank pipes and the liner tubes of the steel tank was provided by stuffing boxes as is shown in Fig. 2. The reason for this design was that the fixed point of the tank lies at the top flange so that the thermal expansion of the tank and its pipes requires a free movement of the tank pipes at the sealing.

In the past, leaks in two smaller pipes resulted in the ingress of heavy water into the graphite reflector and thus also into the gaps between the liner tubes of the steel tank and the reactor tank aluminum pipes. Of course, after the events had been detected the water was drained from the graphite reflector, but the moisture left in the bottom part resulted in corrosion of the steel tubes. The gaps were filled with corrosion products which became very solid during the course of time and hindered the movement of the tank pipes so that unacceptable stresses were caused in the bottom of the reactor tank.

The problem became obvious when the movement of the tank pipes against the steel tubes was measured and was found to be inadequate for all pipes. After having realized that the problem required a fundamental solution, it was decided to replace the stuffing-box design by a new design using steel bellows and O-ring gaskets (Fig. 2). The lower part of each unit of steel and aluminum tube was cut in two steps using a compass saw and a milling cutter device. The radioactive corrosion products were removed from the gaps and new flanges were welded to the steel tubes and new nozzles to the aluminum pipes. Tritium release during welding was almost totally avoided by heating the aluminum material used before welding.
The new design represents an improvement of the plant since all the aluminum pipework is now enclosed by stainless steel and the sliding seal has been replaced by a static one.

3. Upgrading.

3.1 Earthquake Analysis and Improvements.

A systematic approach was chosen for the required earthquake analysis and improvements. The components and systems were grouped into three classes in accordance with their significance for the safety of the plant. Class I contains all components and systems needed for safe shutdown, afterheat removal and enclosure of radioactive substances. For them, it had to be shown that in the case of a design-basis (safety) earthquake stresses and strains remain within the linear elastic range and that the stability and safety function are maintained.

Very important class I components of the FRJ-2 are the reactor block with the primary circuit beneath and the reactor hall, an airtight cylindrical steel shell with a steel dome. Steel pillars are welded to the cylindrical shell supporting the track of the polar crane and the experimental floor which surrounds (Fig. 3) the reactor block. The experimental floor only rests on the pillar cantilevers but is not attached to them. The first dynamic analysis for the reactor shell with its interior revealed a relative movement of the reactor block of 70 mm compared with the basement in the case of a design-basis earthquake. Although the strain of the block pillars remained mainly in the elastic range it was decided to couple the block, via the experimental floor, to the pillars of the shell by buffers in order to reduce the block movement. This would have meant that the shell had to be surrounded by a steel collar at the height of the experimental floor. The license had already been applied for, when a strong earthquake occurred on 13th April 1992 with its epicenter about 50 km from the research center. The evaluation of this earthquake resulted in an alteration of the acceleration-response spectrum for the design-basis earthquake at the site. When the new spectrum was applied the deflection of the reactor block was reduced to 35 mm. This was deemed acceptable so that no modification with respect to the connection of the experimental floor to the shell and its pillars was necessary.

The strain and stress analysis of the primary circuit on the basis of the calculated block movement resulted in minor modifications of the supports of some components. The main modifications are related to the flanges. In order to guaranty a definitive distance between the flanges, each pair of flanges was equipped with an individually adapted ring as a spacer. This was necessary to prevent the gaskets lying in series with the flanges from being squeezed, which had resulted in leakages in the past. In order to make sure that the flanges would not twist against each other in case of the design-basis earthquake, it proved to be necessary to replace the screws by new screws of greater strength and to tighten them with a torque wrench. This had to be done in the presence of a TÜV expert.

The earthquake analysis for the secondary circuit were restricted to the pipework inside the reactor hall taking credit from the fact that an auxiliary cooling system is available for after heat removal. The latter feeds water from the fire extinguishing system onto the shell side of the heat exchangers and from there into the waste water sump from where it is pumped into the waste water tanks outside the reactor hall. Of course, for all components of the auxiliary cooling system it had to be shown that they meet the earthquake requirements. This meant that a lot of components had to be replaced. Although the system already existed, a special license was necessary for this modification.

For buildings not belonging to class I components but accommodating the latter, it had to be proven that their collapse would not impair the function of those components, which is in general very difficult to prove, or else that their stability is maintained. The second approach was taken in case of the FRJ-2, which meant that a lot of brick walls had to be reinforced by steel frames and supports.

3.2 Fire Protection.

The measures for the improvement of fire protection also required a special license which was granted in autumn 1994. The measures were mainly related to the electrical part of the plant and they included the coating of cables, the insulation of cable trays, the separation of electrical busses and many other improvements. Fortunately, some of them were not required to be realized before restarting of the reactor. The latter include the replacement of wooden air ducts in the reactor hall by steel ducts, the installation of a new fire alarm facility and the installation of a CO_2 fire extinguishing facility for the relay room. The supervisory authority expects that most of the outstanding measures will be realized within 100 days of operation.

4. Emergency Handbook.

The required emergency handbook, which describes the regulations and procedures in the case of beyond-design-basis accidents, consists of three parts. Part I with general information such as the definition of terms and the criteria for the different degrees of alarm and part II with the organizational emergency regulations were put into force before restarting the reactor. Part III with the accident management measures has been discussed with the supervisory authority and the technical expert involved. It will be put into force soon.

The accident management procedures are based on the current technical status of the plant. However, in order to facilitate some procedures, modifications will be necessary and will be carried out in the near future. For instance, in the case of a large primary coolant leak with a late failure of both emergency cooling pumps, light water can be injected into the reactor tank. However, the reactor hall is filled up within days to a level at which the design pressure of 3 m water column is exceeded due to the flow rate needed for the emergency cooling of the core. Therefore, an interruption of emergency cooling and a recirculation of the water by a tank lorry is planned after one day of light water injection. Of course, this can be achieved much more easily and without the necessity of interrupting core cooling by a recirculation pump which must be installed in the reactor hall. Additionally, it proved helpful to install remote control of the light water injection system so that it can be operated from the shielded emergency control room. It is then possible to operate it in the case of a core meltdown accident when the level of direct radiation from the reactor hall is very high. The advantage of this would be that the molten core would be cooled and that radiologically significant fission products would be washed out of the atmosphere.

When investigating the efficiency of possible accident management procedures it was discovered that the loss of forced secondary cooling would probably not result in the loss of afterheat removal from the primary circuit. Natural convection in a closed loop, including a small part of the total secondary system, seems to remove enough heat to the environment so that the coolant temperature in the reactor tank would not exceed the boiling point, as can be seen from Fig. 4. This will be demonstrated by a reactor experiment using the primary cooling pumps as the heat source which will be of the order of the decay heat after about one hour.

5. Outlook.

Upgrading is a permanent task for the thirty-year old plant. It is necessary in order to obtain and maintain reasonable availability. Most components are as old as the plant and it is practically impossible to get spare parts for them. Thus, it is necessary to replace them by new components as has already occurred in the case of the emergency cooling pumps.

In the past, one weak point of the plant proved to be the main primary coolant pumps. Many outages were caused by the failure of one pump. Therefore, it was decided a year ago to replace the pumps by new ones. In this connection, it has been suggested that one of the new pumps should be supplied with electrical power from a machine composition equipped with a fly wheel in order to maintain forced flow for about 100 seconds in the case of a loss of offsite power. After that time, natural convection cooling of the core is adequate so that the safety of the plant no longer depends on starting a shutdown pump .

The results of an aging evaluation program did not reveal significant aging/damaging effects on nonreplaceable parts of FRJ-2. Taking into account the various modifications carried out or to be carried out in the near future in order to meet modern German safety standards it is realistic to assume that FRJ-2 can be operated at least for about 5 to 7 more years. This assumption is supported and confirmed by the final conclusion of the TÜV which performed the aging evaluation analysis. It reads /3/: Due to the results of the inspections and tests carried out for FRJ-2 aging and operation-related damage critical from a safety engineering point of view are not expected for a period of 10 years.

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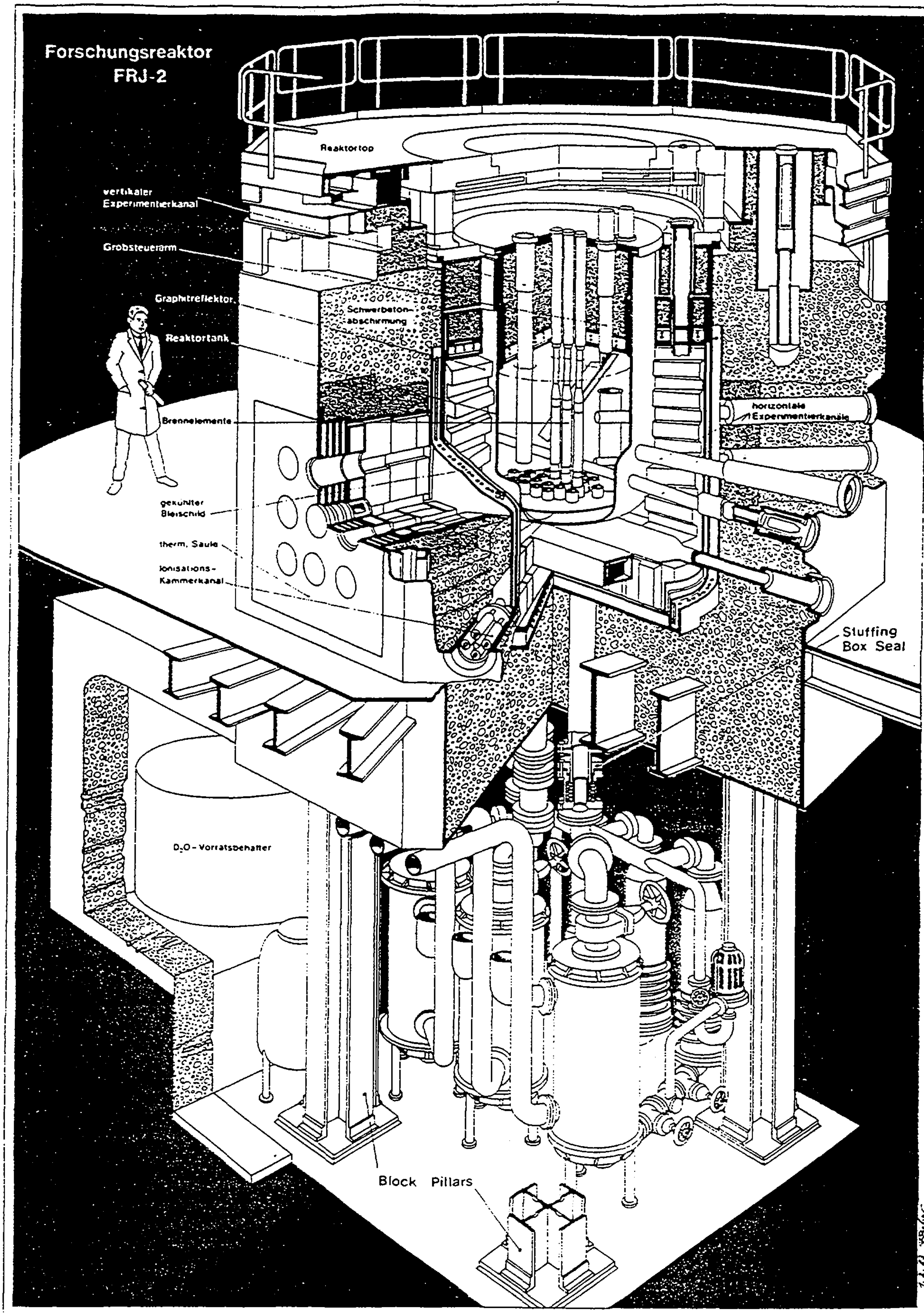

Fig. 1: Perspective View of the Reactor Block and the Primary Circuit of the FRJ-2

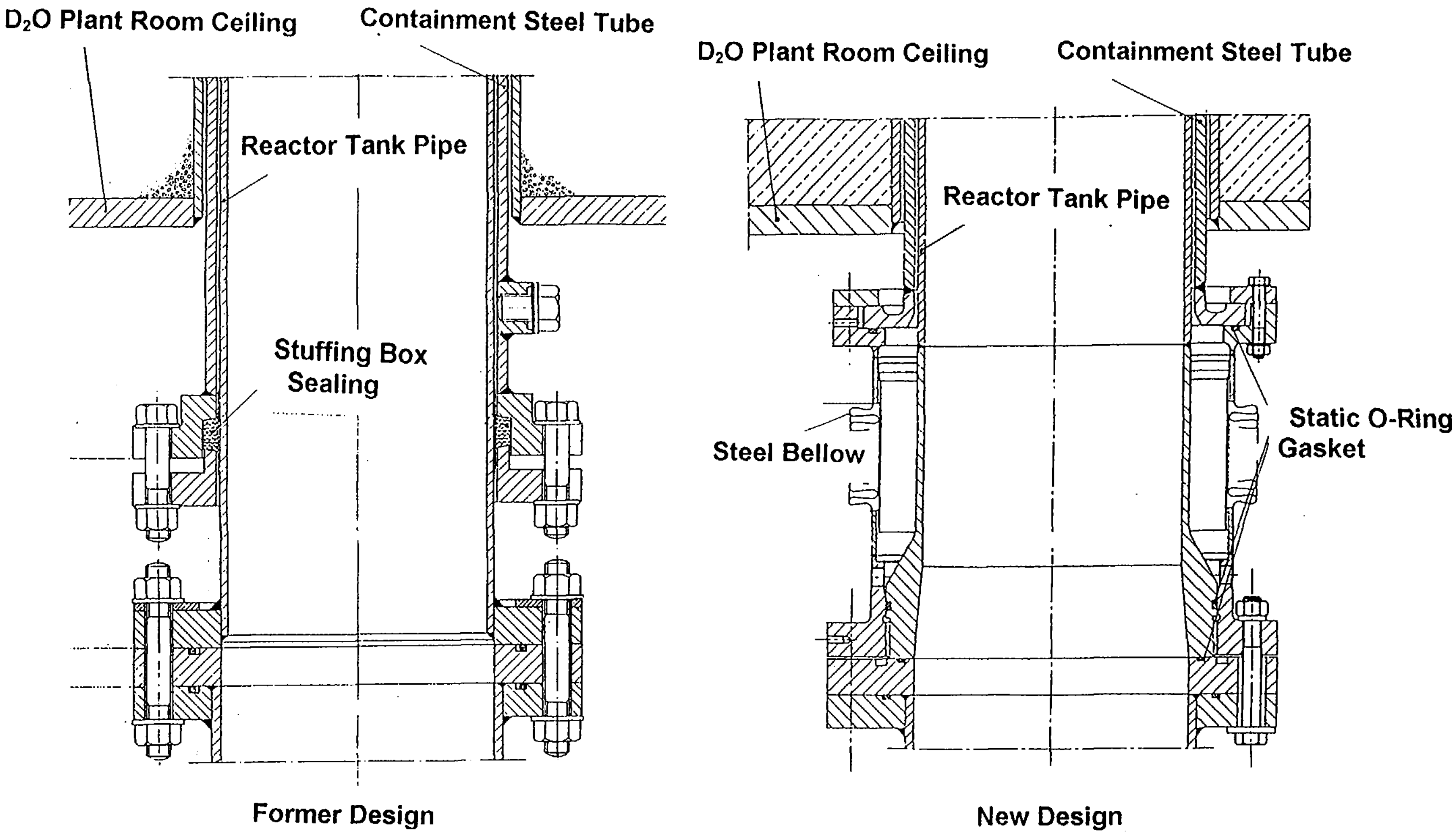

Fig. 2: Sealing between the Reactor Tank Pipe and its Containment Steel Tube at the Outside of the Biological Shield

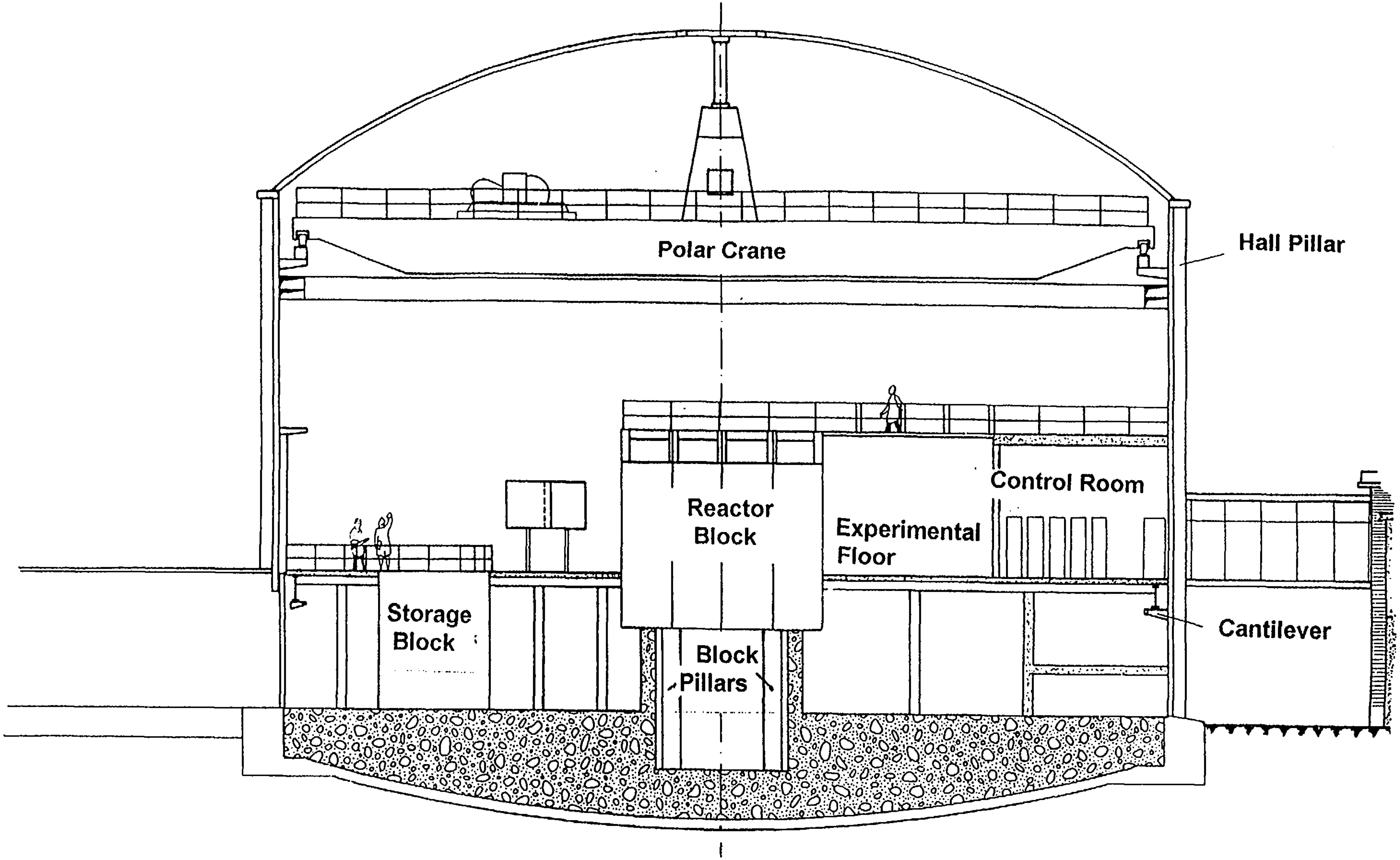

Fig. 3: Sectional View of the Reactor Hall and its Interior

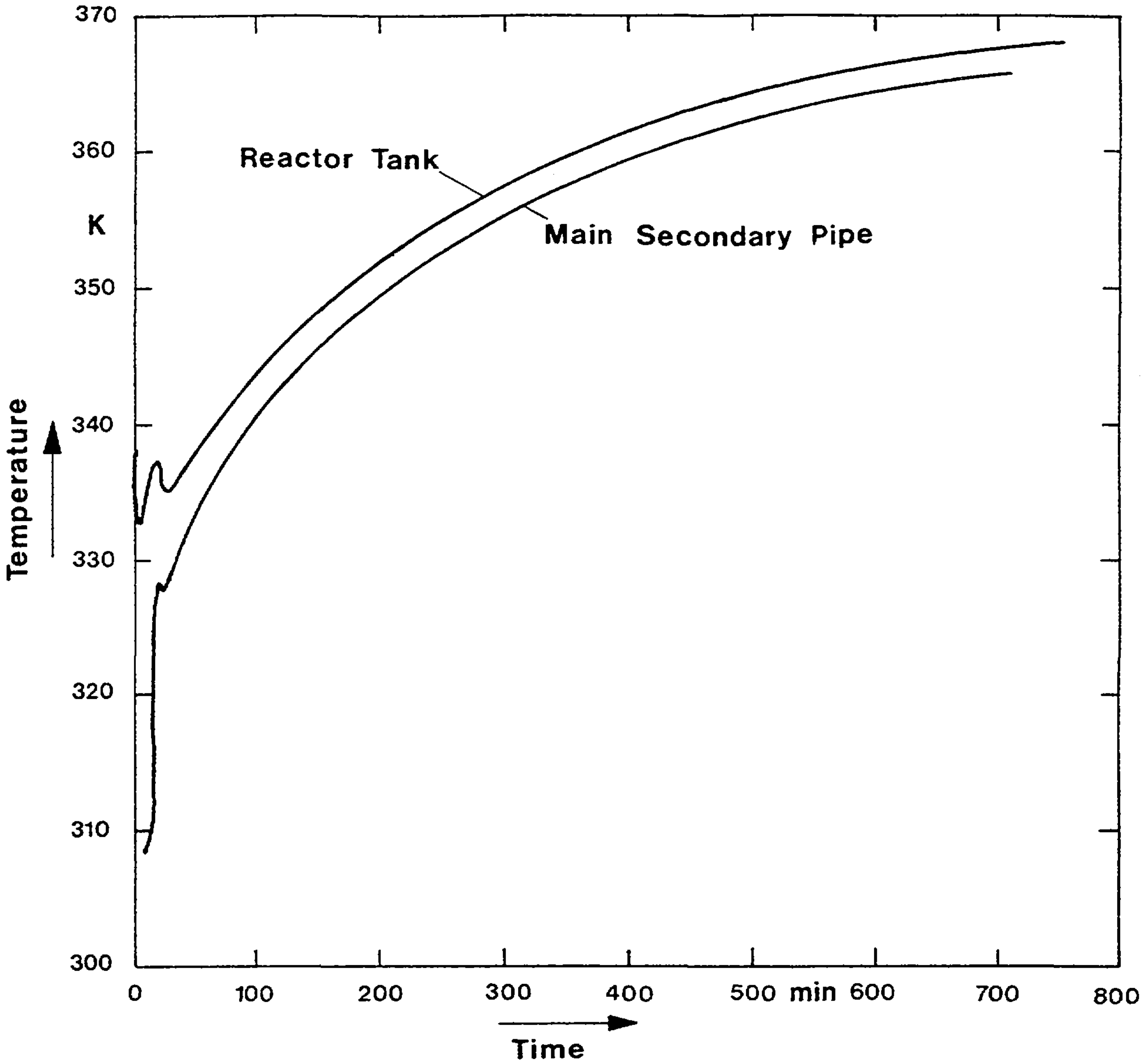

Fig 4: Time Dependent Temperature of the Water in the Reactor Tank and in the Secondary Manifold in Front of the Primary-Secondary Heat Exchanger in the Case of Secondary Cooling by Natural Convection

COMMISSIONING STATUS OF HANARO, KOREA MULTI-PURPOSE RESEARCH REACTOR

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ABSTRACT

The 30MW Korea Multipurpose Research Reactor, HANARO, located at KAERI site has achieved its initial criticality on February 8, 1995. Non-nuclear commissioning was started in the middle of 1993, and all the system performance tests were successfully finished at the end of last year, 1994. The fuel loading for initial core was completed. Nowadays, many activities of nuclear commissioning are ongoing under zero power condition.

Besides on the reactor commissioning, some experimental facilities are in progress on design, fabrication and purchasing.

This report describes the outlines of the major activities and experiences of system commissioning and presents the future plan in HANARO reactor.

Introduction

At present, there are two research reactors in KOREA, TRIGA MARK-II 250KW and TRIGA MARK-III 2MW which were constructed in 1962 and 1972 respectively.

But the utilization of these reactors have been limited in coincidence with the recent trend of nuclear research and development because not only the neutron flux level is relatively low but also the facilities are old to be retired soon.

To meet the recent various requirements for example, nuclear basic research, production of radioisotope for medical and industrial applications, development and research of nuclear materials a new powerful research reactor project, HANARO, was started 10 years ago.

The major milestones are followings :

1987. 12. Construction permit acquired from MOST (Ministry of Seience & Technology)
1989. 3. Ground breaking
1993. 2. Reactor installation
1993. 7. Started system commissioning
1994. 12. Finished system integration test
1995. 2. Fuel loading & first criticality

In the one and half years from the beginning of the system commissioning in July, 1993 to fuel loding of this year KAERI engineers had led the system performance tests. Partially, some manufacturers and construction people had participated in field works.

HANARO has about 40 systems for reactor operation. Each system has been verified its function through construction acceptance test, system performance test and system integration test.

Summary of Design

HANARO is a light-water cooled and heavy-water reflected open-pool type reactor. The reactor is located at the bottom of a concrete shielded, stainless steel lined pool of dimineralized water.

The core heat is removed by two loops of primary cooling system. When the power is less then 50% of full power the reactor can be operated by one loop with one pump and one heat exchanger. If the forced primary cooling function is not available the core residual heat is removed by a natural circulation through the primary loops with a gravity dump of secondary cooling water from yard basin or by a natural convection of pool water through the flap valves located inside pool.

The reflector system contains 4,700 liters heavy water and has two circulation pumps (one is standby). The system continuously circulates the D_2O in the reflector tank to dissipate up to 2.5MW at full power condition. A 6″ rupture disc is installed at the suction side of pump to prevent the overpressure to the inner shell of the reflector tank.

The reactor is composed of inner and outer core. The inner core consists of 23 hexagonal flow tubes and 8 cylindrical flow tubes. The outer core has 8 cylindrical flow tubes embedded in the reflector tank. In normal operation, 36-element fuel assemblies are loaded in 20 hexagonal flow tubes while 18-element fuel assemblies are loaded in 12 cylindrical flow tubes.

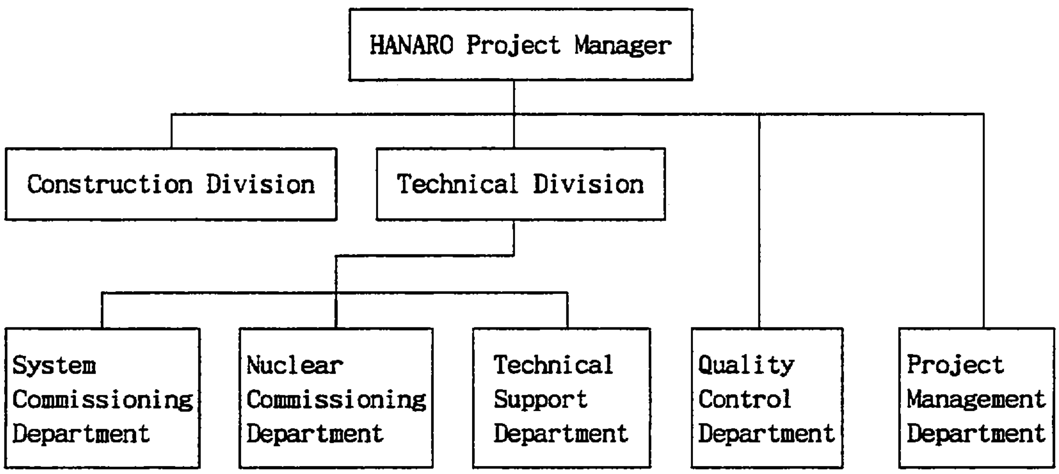
The fuel of HANARO is low enriched (20 w/o U-235) U_3 Si-Al alloy. The active length of the fuel element is 700mm.

The reactor is controlled by 4 shutoff and 4 control absorbers, The material of the absober is hafnium shaped hollow cylinder. The driving concept between shutoff and control absorbers is basically different. The shutoff absorber element is actuated by a hydraulic pressure, while the control absorber element is moved by a stepping motor turning a ball screw.

The secondary cooling system consists of a cooling tower, 50% three pumps and heat exchangers. The thermal capability of cooling tower is 33.5MW.

Commissioning

1) Organization



Most commissioning engineers have involved in the system design of HANARO and prepared lots of commissioning documents.

2) Schedule

Four(4) degrees of different schedules have been used for the management of the overall commissioning programs.

Level 1 : Project Master Schedule
Identifies important project milestones

Level 2 : Start-up Summary Schedule
Identifies primary critical paths and main commissioning activities at the system level.

Level 3 : 6 Month Schedule
Indentifies detail work plans for coming six(6) month with the progress of the work including interfaces between all systems.

Level 4 : 3 Weeks Schedule
Identifies daily activities for three(3) weeks based on the the Level 3 schedule.

The table 1 shows the actual commissioning schedule. The completion of the system commissioning was delayed by six(6) months compared with the original plan due to the unexpected interferences in the field.

3) Test Procedures (Documentation)

Commissioning documentations including commissioning plans, start-up manuals, and commissioning procedures etc, shown in the table 2 had been started two(2) years before the beginning of the system commissioning. These commissioning documents were prepared based on the Safety Analysis Report, Design Drawings and Documents, Technical Specifications and Manufacturer's information.

Parts of the commissioning procedures have been revised during commissioning to incorporate the field interferences and these changes served as a good reference to the complement of the various Operating Procedures. The commissioning Procedures were practically used as a training manual for the commissioning personnel and as a guidelines for all the testing activities.

4) Major Activities

- Utility Supplies

In order to ensure the safe and reliable electric power supply, the following function tests have been carried out according to the appropriate test procedures.

· Standby power transfer tests
· Full load tests
· Independence and separation tests
· Protection and logic tests

Especially, the Reactor Protection System (RPS) with three(3) redundant channels requiring separate Class 1E grade of power was subjected to the severe test criteria based on the International Codes and Standards.

Service water and demineralized water have been supplied from the main production facilities of KAERI, through the common underground gallery.

- Flushing

It was the flushing work for process pipings and equipments that a lot of time and manpower have been used up prior to the start of the system performance test. It is important to concentrate more in a system base flushing rather than a component base flushing and to check the cleanliness of the pipings repeatedly.

It was a good chance to experience that any installation defects and faults like system leakage or unreasonable vibration could be detected and repaired during this period. Careful consideration must be given to the quality control of the flushing work.

- Instrumentation & Control

I&C work requires a sufficient time to work and a skilled technical manpower to achieve the complete verification of the integrated system performances from the field instruments to the control logic and display. System-level function tests have been carried out after the confirmation of the calibration and setpoint adjustments of each component. The followings are the special test activities performed during I&C commissioning work.

- Real-time dynamic function test of the reactor power control algorithm
- Response time testing for trip parameters of safety system
- Examination of cause and solution for Electro-Magnetic Interference(EMI) noises

- Design Data Confirmation

It has been verified, as shown in the table 3 that commissioning test results are compatible with the design data calculated by design and analysis groups.

- **Integrated Pre-operation Test**

As a final commissioning program, integrated pre-operation was performed to verify the integrated reactor operating performances by checking the system-to-system interlocks with all process system running. This test program is the last commissioning stage to decide the appropriateness of the initial fuel loading. The major test activities are as follows :

- Operation transient function tests
- Interface logic checks
- Failure mode tests

Failure mode tests including loss of electric power, loss of instrument air and control computer failure are to demonstrate that the system transient behaviors with displays and alarm functions after each failure conform to the design specifications.

Future Plan

Nuclear physics tests have been in progress as increasing the number of fuel loading step by step since the initial criticality on Feb. 8, 1995. At present, the fuel loading for initial core was completed. The zero-power test is scheduled to be finished by August, 1995 and then power increasing will start and it will take about 9 months to reach full power.

Nuclear commissioning of power increasing stage include the various programs to confirm the core physics characteristics and heat removal capability and natural convection coolability which could not be carried out during cold function test.

Planned experimental facilities around HANARO are listed below

- In-core irradiation test facilities
- Radioisotope production facilities
- Neutron beam application facilities
- Neutron doping silicone production facilities
- Neutron activation analysis facilities

The project schedule indicated that the basic components of these experimental facilities would be equipped until the end of 1996.

Coclusion

Since there was not a verified reference reactor similar to HANARO, a lot of design changes and trial and error have been experienced during design and commissioning stage. The commissioning data is being accumulated through the system pre-operation test, zero power and power increase tests. All these experiences and data will be useful for reactor operation, in-service inspection, maintenance and upgrade work.

Throughout the system commissioning test programs, all system performances were verified and approved by regulatory body then finally the initial criticality was successfully accomplished as intended. It can be said that all these achievements made a remarkable contribution to the development of a new research reactor.

Conclusively, HANARO will be a good reference of new research reactor. The design improvements and R & D activities experienced in HANARO will be of help to the development of research reactor technology.

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Table 1. Commissioning Schedule

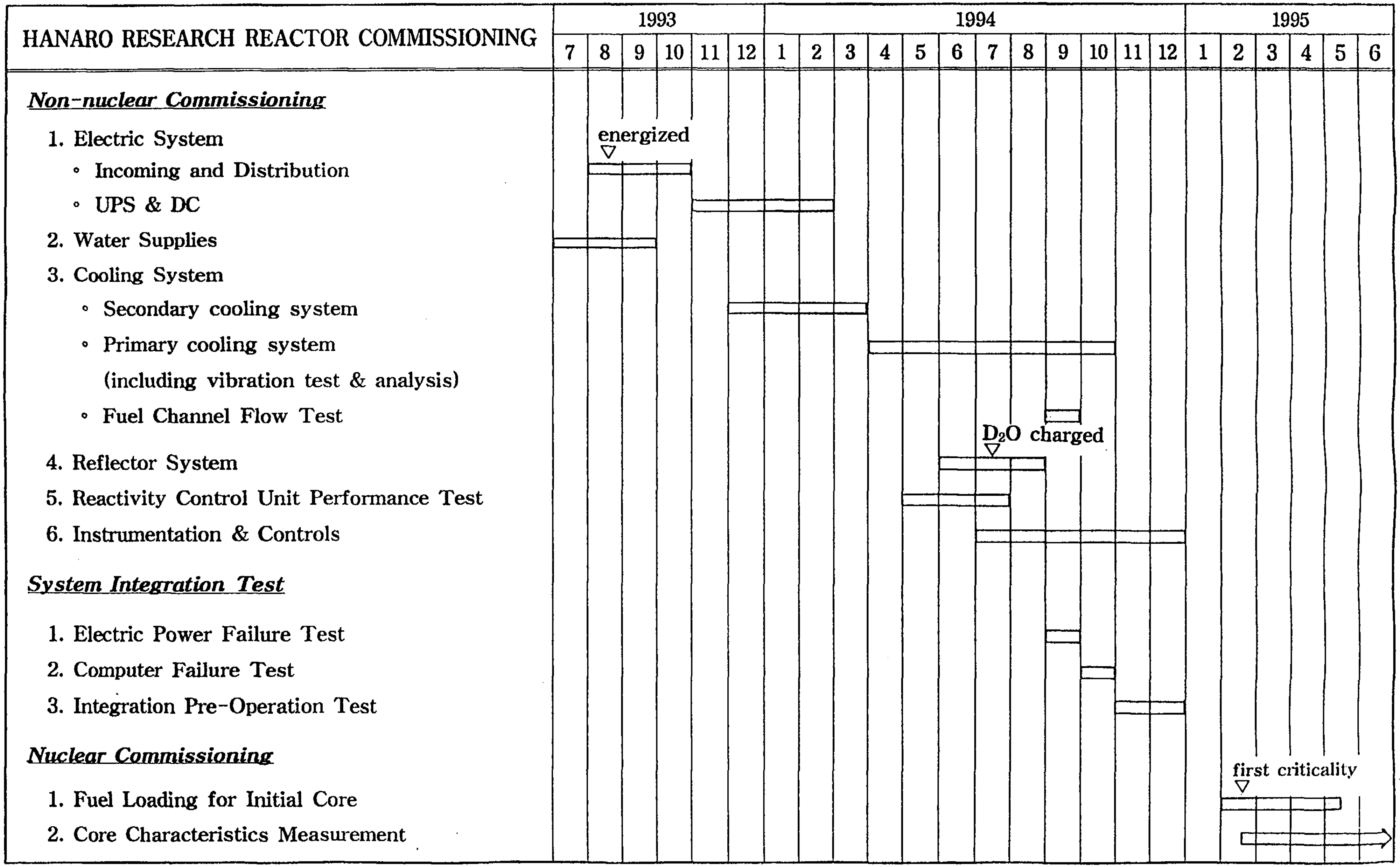

Table 2. Commissioning Procedures

HANARO

Items	Q'y
1. Construction Acceptance Tests	57
- Mechanical Eq.	12
- Electrical Eq.	19
- Instrument	26
2. System Performance Tests	40
- Mechenical System	8
- Process System	14
- I & C System	9
- Electric System	9
3. Integrated System Tests	7
- System Failure	3
- Integrated Pre-Operation	1
- Core Flow Measurement	3
4. Reactor Performance Tests	30
Total	**134**

Table 3. Test Results of System Commissioning

HANARO

Major Parametes	Design Value	Commissioning Results
1. Primary Cooling System		
- TPTH* flow	703 kg/s	750 kg/s
- OPOH* flow	454 kg/s	460 kg/s
- TPTH Pressure	316 KPa (g)	360 KPa (g)
- OPOH Pressure	200 KPa (g)	215 KPa (g)
2. Secondary Cooling System		
- 2 pumps flow	915 kg/s	930 kg/s
3. Reflector Cooling System		
- Flow	43.3 ℓ/s	47.5 ℓ/s
- Pressure	433 KPa (g)	440 KPa (g)
4. Reactivity Control Unit		
- Shutoff Rod Drop time	< 1.13 sec	1.027~1.076 sec
- Control Rod Drop time	< 5 sec	0.53~0.65 sec
5. Response Time of Reactor Protection I&C		
- Response time of flow & pressure in primary cooling sys.	< 300 msec	242~266 msec
- Neutron power measurement	< 100 msec	54~84 msec
- Lograte measurement	< 1.5 sec	67 msec

* TPTH : Two Pumps Two Heat Exchangers
* OPOH : One Pump One Heat Exchanger

The Initial Criticality and Nuclear Commissioning Test Program at HANARO

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ABSTRACT

The construction of the Korea Multipurpose Research Reactor - HANARO of 30MW, developed by Korea Atomic Energy Research Institute, was completed at the beginning of this year. The first fuel loading began on February 2, 1995, and initial criticality was achieved on February 8, when the core had four 18-element assemblies and thirteen 36-element assemblies. The critical control rod position was 600.8 mm which represents excess reactivity of 0.71 $. Currently the nuclear commissioning test is on going under the zero power range.

This paper describes the initial criticality approach of the HANARO, and its nuclear commissioning test program.

1. Reactor Description

The HANARO is an open-tank in pool type research reactor cooled by upward forced convection of light water. The fuel is a rod type made of co-extruded U_3Si-Al with finned aluminum cladding. The fuel has a nominal composition of 61.4 w/o U_3Si and 38.6 w/o Al and its enrichment is 19.75 w/o ^{235}U in total uranium. The fuel elements are clustered to form a fuel assembly. Two types of fuel assemblies are used in the HANARO : 18-element and 36-element assemblies. The 36-element fuel assembly is the primary fuel assembly. It is loaded into the hexagonal flow tube in the main core region of the reactor. The 18-element assemblies are placed in the circular flow tubes.

The reactor structure is made up of five main components as shown in Fig. 1 : the inlet plenum supporting the reactor tank and distributing inlet coolant, the lower grid plate holding the fuel assemblies and experimental facilities, the reactor tank, the outlet chimney mixing coolant passed through individual flow tubes and bypass flow, and the flow tube channels.

As shown in Fig. 2, the core consists of three parts : the inner core, the outer core and the reflector in the radial direction. The compact inner core, inside the inner shell, looks like a honeycomb, which is composed of 23 hexagonal flow tubes and 8 circular ones. The light water flows inside and outside of the flow tube to remove the generated heat. A suitable fuel assembly can be loaded in each flow tube. Three sites(CT, IR1 and IR2) of 23 hexagonal flow tubes are reserved for material irradiation tests requiring high fast neutron flux. These site shall hold experimental device on duty or aluminum dummy assemblies off duty.

There are eight neutron absorber tubes - four CARs(Control Absorber Rods) and four SORs(Shut Off Rods). During the normal operation condition, CARs regulate neutron power and SORs are at the fully withdrawn state and shut down the reactor power rapidly in case that any reactor trip signal is initiated. While CAR is operated by stepping motor, SOR is operated by hydraulic power.

The inner core is surrounded by the inner shell of zircaloy which separates the inner core and reflector. The inner diameter of zircaloy reflector tank is 2.0 m and its height is 1.2 m. In the reflector adjacent to the inner shell, there are eight circular flow tubes called as the outer core. D_2O is used as the reflector material to maximize the region where the high thermal flux is available. There are 25 vertical holes and 7 horizontal beam tubes in the reflector tank.

2. Initial Criticality

2.1 The Neutron Detection System for Nuclear Commissioning Test

Although there are six fission chambers at the outside of the reflector tank for the power monitoring, they could not be used for initial criticality approach due to their very weak responses.

The neutron detection system for the nuclear commissioning tests was composed of six neutron detectors as shown in Fig. 3, which consist of two fission chambers, two BF_3 counters and two CICs(Compensation Ion Chambers), respectively. Two fission chambers and two BF_3 counters were used to measure neutron count rate in the course of the fuel loading. Two CICs provide signals for reactivity meter. These detectors were placed in the irradiation holes in the reflector tank as shown in Fig. 2.

2.2 Approaching to Initial Criticality

The first fuel loading into the core began on February 2, 1995. The fuel was sequentially loaded from center to outside of the core as shown in Fig. 4. At first, four 18-element fuel assemblies were placed into the control absorber sites and this condition was used as the initial state for inverse multiplication measurement. The count rates were measured under the four reactor conditions to guarantee criticality safety during fuel loading:

(1) Condition 1 : All SORs down, all CARs down
(2) Condition 2 : All SORs up, all CARs down
(3) Condition 3 : All SORs up, all CARs 350 mm (half) up
(4) Condition 4 : All SORs up, all CARs up

The count rates of fission chambers were about 0.6 cps and those of BF_3 counters were about 708 and 1320 cps, respectively under the Condition 4 of initial state. Fig. 5 shows 1/M curves vs. the loading of 36-element fuel assemblies plotted for the four reactor conditions. From the 1/M curve, we can deduce that the minimum number of 36-element assemblies for initial criticality will be thirteen.

After loading 13th 36-element assembly, the count rate was measured when all SORs were fully up and four CARs are being withdrawn step by step to search critical CAR position. Fig. 6 shows 1/M curve vs. CAR position. This curve predicts that the critical CAR position will be near 600 mm.

At last, the HANARO achieved initial criticality on February 8, 1995 with the following core configuration:

(1) All SORs fully out
(2) All CARs 600.8 mm out
(3) Four 18-element fuel assemblies, thirteen 36-element fuel assemblies in the core
(4) Uranium mass in the core : 33.46 kg
(5) Pool water temperature : 16.2 ℃
(6) Excess reactivity : 0.71 $

2.3 Comparison of Analysis and Experimental Results

The multiplication factor of initial critical core in the condition of all rod out, was 1.005 from the experiment. The core physics calculations using design codes, WIMS-VENTURE and MCNP, predicted criticality. The MCNP predicted k_{eff} of 1.0098 with ENDF/B-IV and 1.01802 with ENDF/B-V, WIMS-VENTURE predicted as 1.02239. Though it had been expected that WIMS-VENTURE model used for the HANARO design would overestimate k_{eff}, the the overestimation by MCNP with ENDF/B-V is much far from our expectation. Thereafter, when fuel assemblies are added, the tentative comparisons show that differences between calculations and experiment are consistent.

3. Nuclear Commissioning Test Program

There are four objectives for the reactor physics experiments.

a. Design verification
b. The production of reactor characteristic data for operation and utilization
c. Establishment of the procedures for routine reactor physics experiments
d. Training of the operators

3.1 Nuclear Commissioning Tests at the Zero Power Range

After initial criticality, additional fuel assemblies has been loaded to construct the first operational core which has eight 18-element fuel assemblies and sixteen 36-element fuel assemblies. The CAR worth, fuel reactivity worth, shutdown margin, excess reactivity has been measured to verify design data whenever a fuel was added in the core. Currently, the construction of first operational core has been completed and hereafter, the following nuclear commissioning tests at the zero power range will be performed by September this year:

a. SOR and CAR reactivity worth measurement
b. Noise analysis to measure the kinetics parameters and fission power
c. γ-flux distribution measurement
d. Void coefficient measurement
e. Reactivity worth measurement for irradiation samples
f. Thermal flux distribution measurement and power calibration
g. Fast neutron flux distribution measurement
h. Assemblywise power distribution measurement
i. Coolant temperature coefficient measurement
j. Transfer function measurement in the zero power range

3.2 Reactor Commissioning Tests at The Power Range

Following tests will be carried out at various power level up to 15MW which is the full power of the first cycle:

a. Power defect measurement
b. Verification of heat removal capability and thermal power calibration
c. Transfer function measurement
d. Verification of heat removal capability for the case of the Loss-of-Offsite Electric Power
e. Xe and Sm reactivity worth measurement
f. Measurement of long term operational characteristics

4. Future Plan

Near the end of this year, The HANARO will meet the end of the first cycle. Some physics experiments will be performed to verify design data at the end of cycle and the second cycle core will be constructed by adding two 36-element fuel assemblies and two 18-element ones as per the fuel management plan. Its cycle length is to be 60 days at 24 MW. The third cycle is the first full loaded core, which is made by adding two 36-element fuel assemblies and two 18-element assemblies more. After the third cycle, HANARO will be operate routinely at the design rated power 30MW. Up to the 10th cycle which is expected to reach equilibrium core state, we will make rather extensive physics experiments to tune our core management tools.

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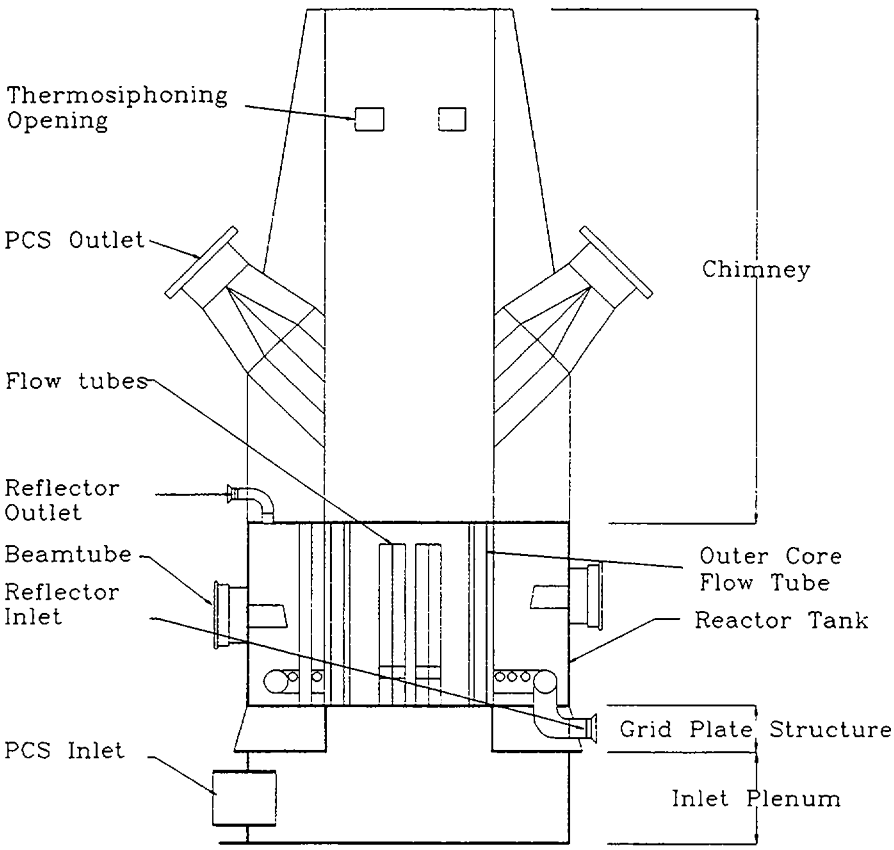
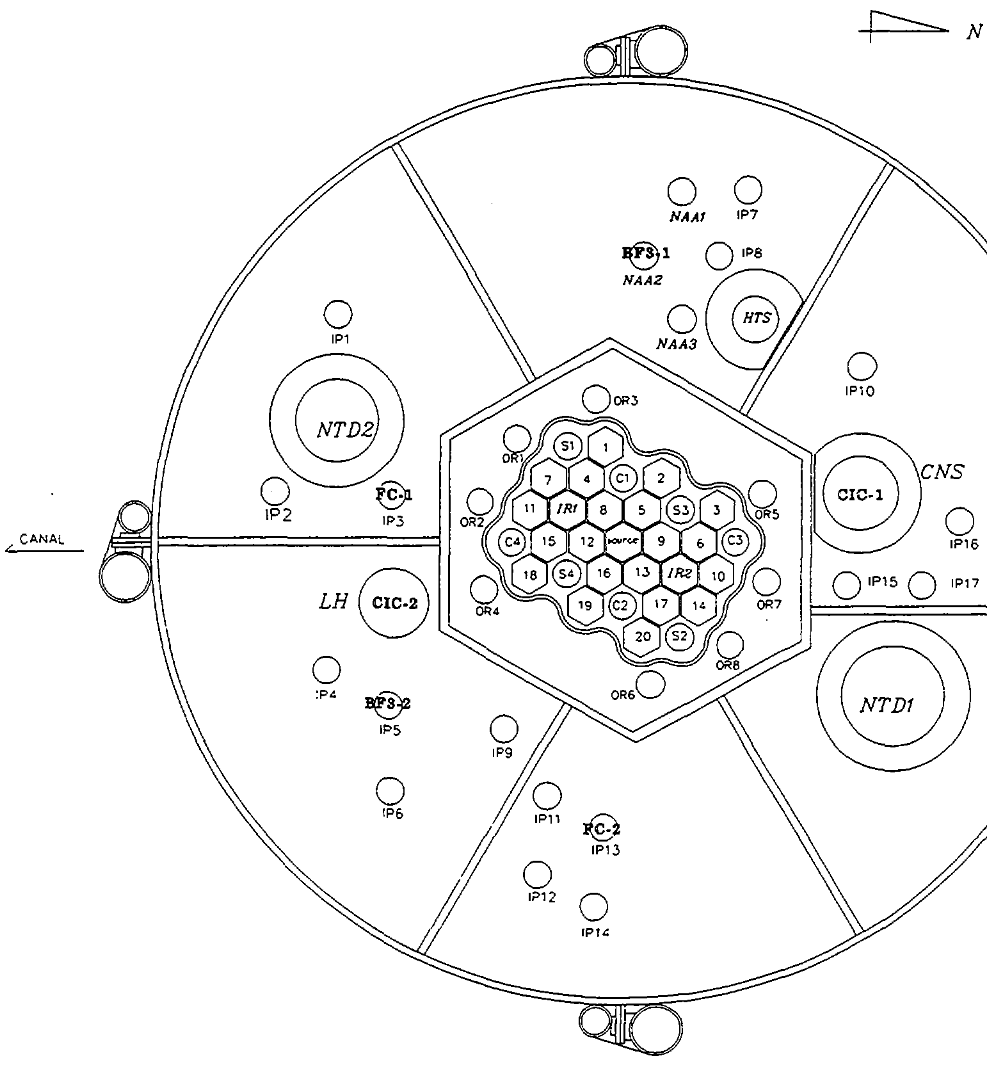


Fig. 1 Vertical View of the HANARO Reactor Assembly



1 - 20	:	36-element fuel assembly sites
S1 - S4	:	SOR sites
C1 - C4	:	CAR sites
OR1 - OR8	:	Outer core sites
IP1 - IP17	:	Isotope production hole
NAA1 - NAA3	:	Neutron activation analysis hole
HTS	:	Large NAA hole
LH	:	Hole to install the fuel test loop
NTD1 - NTD2	:	Neutron transmutation doping hole
CNS	:	Cold neutron source housing hole

Fig. 2 Plan View of the HANARO Core

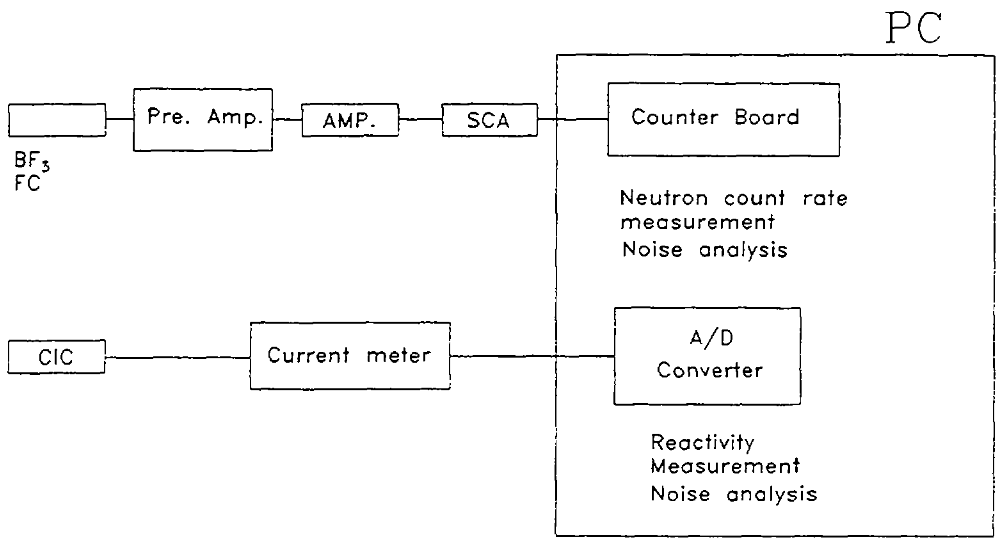


Fig. 3 The Schematic Diagram of the Neutron Detection System

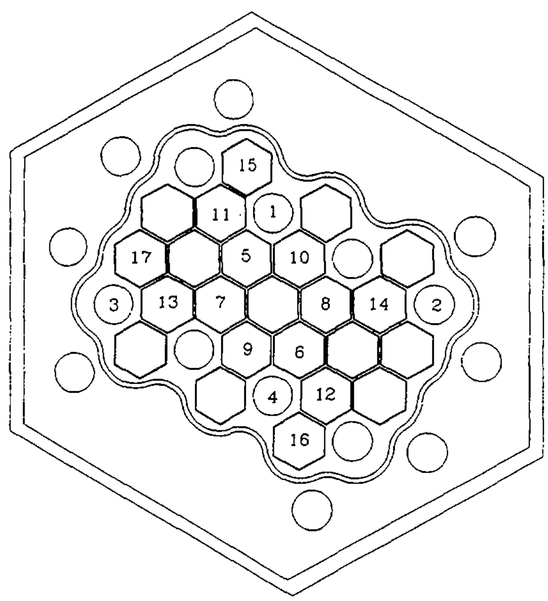


Fig. 4 The Fuel Loading Sequence of the Initial Critical Core

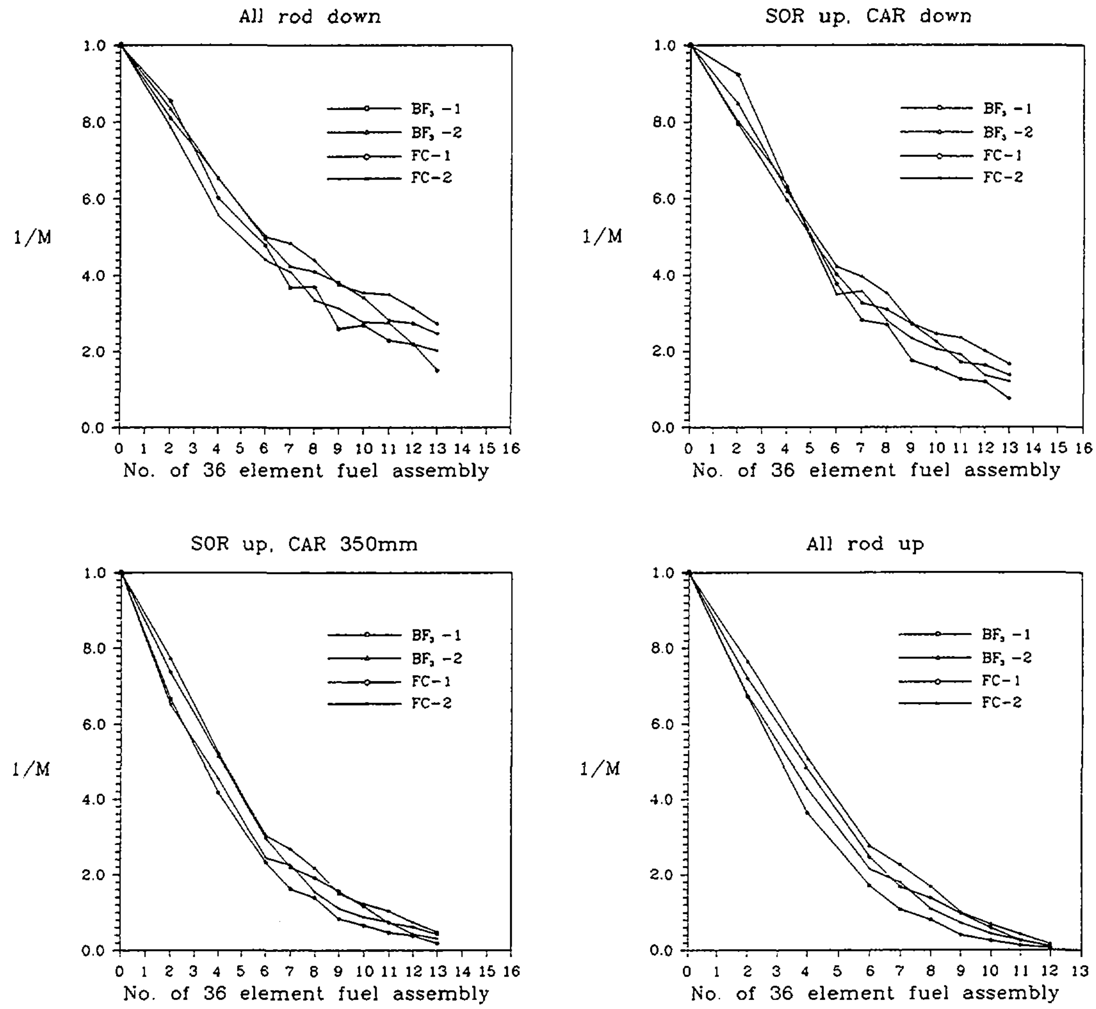


Fig. 5 1/M Curves vs. No. of 36-element Fuel Assembly Loaded

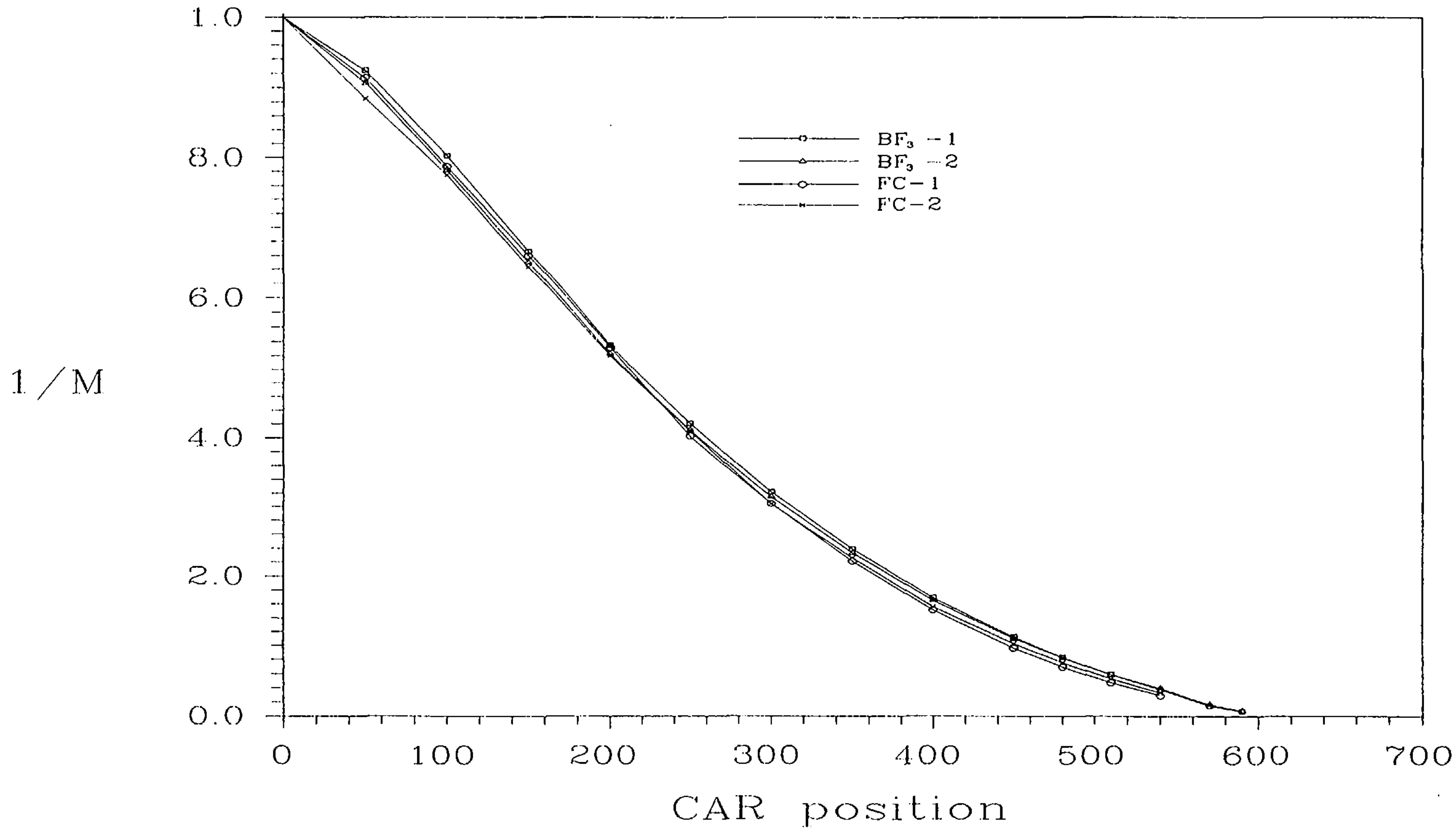


Fig. 6 1/M Curves vs. CAR Position

XA04C1679

Description of the High Flux Isotope Reactor and Future Upgrades

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Oak Ridge National Laboratory

General Description

Description of High Flux Isotope Reactor (HFIR)

The HFIR is a light-water-cooled, beryllium (Be) reflected, 85-MW research reactor. It is a flux-trap design giving a maximum flux of over 2×10^{15} n/cm^2-s in the flux trap located in the middle of the annular core (Fig. 1). The reactor went critical in 1966. In 1986 it was shut down due to vessel embrittlement concerns and restarted in 1989 with power reduced from 100 MW to 85 MW.

The reactor core is surrounded by a Be reflector ~1-ft (30 cm) thick. The core and Be reflector are located in a reactor vessel 8 ft (2.4 m) in diameter and 19 ft (5.8 m) high.

The vessel is pressurized to 468 psig (3.2 mPa). The light water flows downward through the reactor at ~16,000 gal/min (1009 l/sec). The inlet temperature is 120°F (48.9°C) and outlet at 155°F (68.3°C) (Fig. 2).

Control is achieved using four safety/shim plates surrounding a central cylinder. These are raised/lowered respectively to balance the leakage to the reflector. The plate and cylinder are sandwiched between the outer fuel element and the Be reflector (Fig. 3). The control plates are made up of 22 in. (55.9 cm) Eu_2O_3 (black section) followed by 5 in. (12.7 cm) of tantalum (gray section) followed by aluminum. Any of the five control plates can shut the reactor down. They system is fast acting with plate insertion accomplished in less than 0.5 sec.

The pressure is maintained using a feed and bleed process, and the system is "water solid." The pressure is controlled by use of one of two nine-stage centrifugal pumps [capable of producing over 1000 psia (6.8 mPa)] and a system of let-down values.

Heat removal is accomplished by three 600-hp (447.4 kW) primary pumps, resulting in a downward flow rate through the core of 13,000 gal/min (820 l/sec). A fourth pump is kept in standby and can be put on line during reactor operation. The heat is transferred to the secondary heat removal system by means of three primary heat exchangers (fourth in standby). The secondary coolant water dumps heat to the atmosphere using an induced-draft cooling tower [capacity of 375 million Btu/hr (109 MW)].

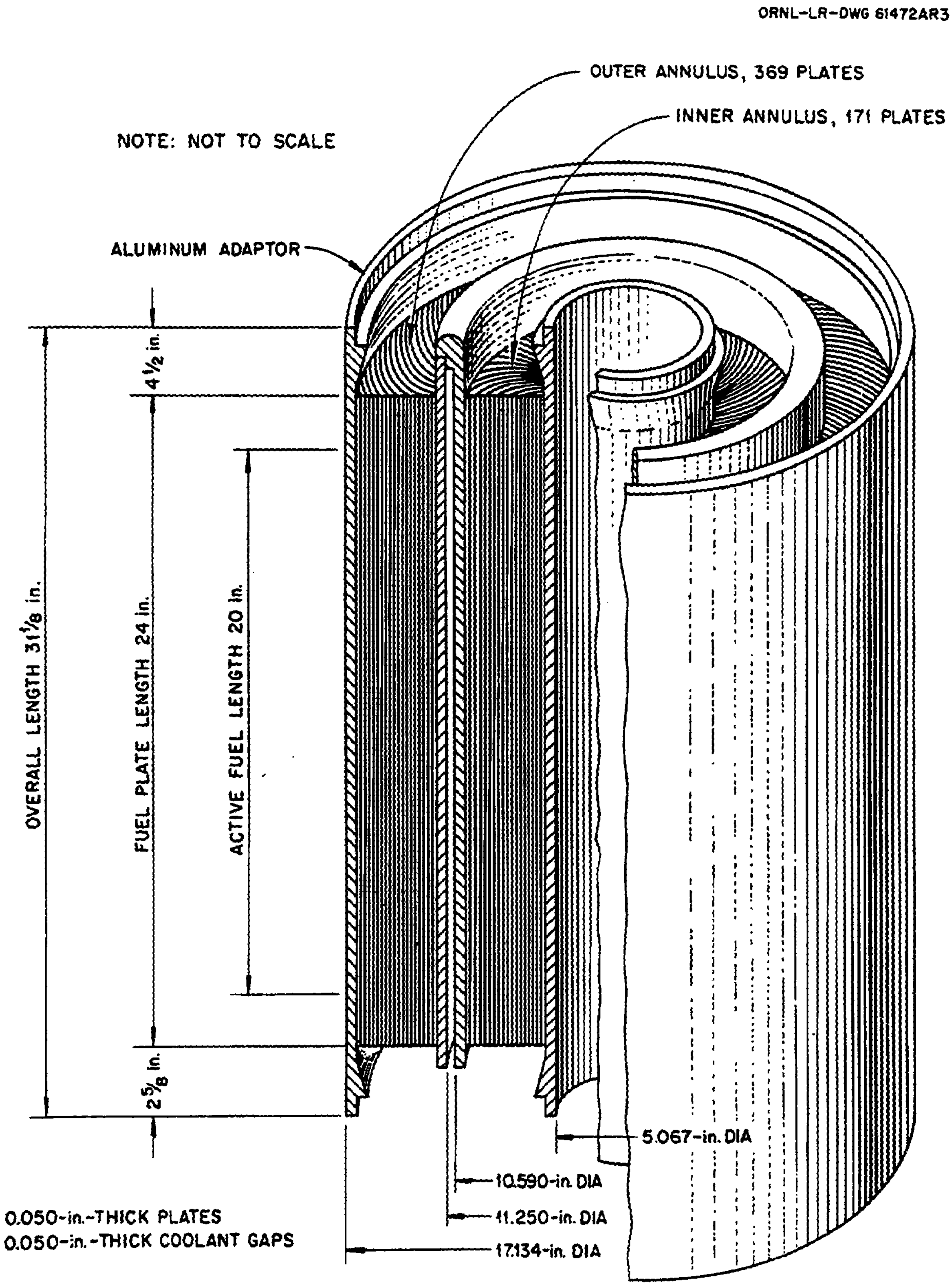

Fig. 1. HFIR Fuel element.

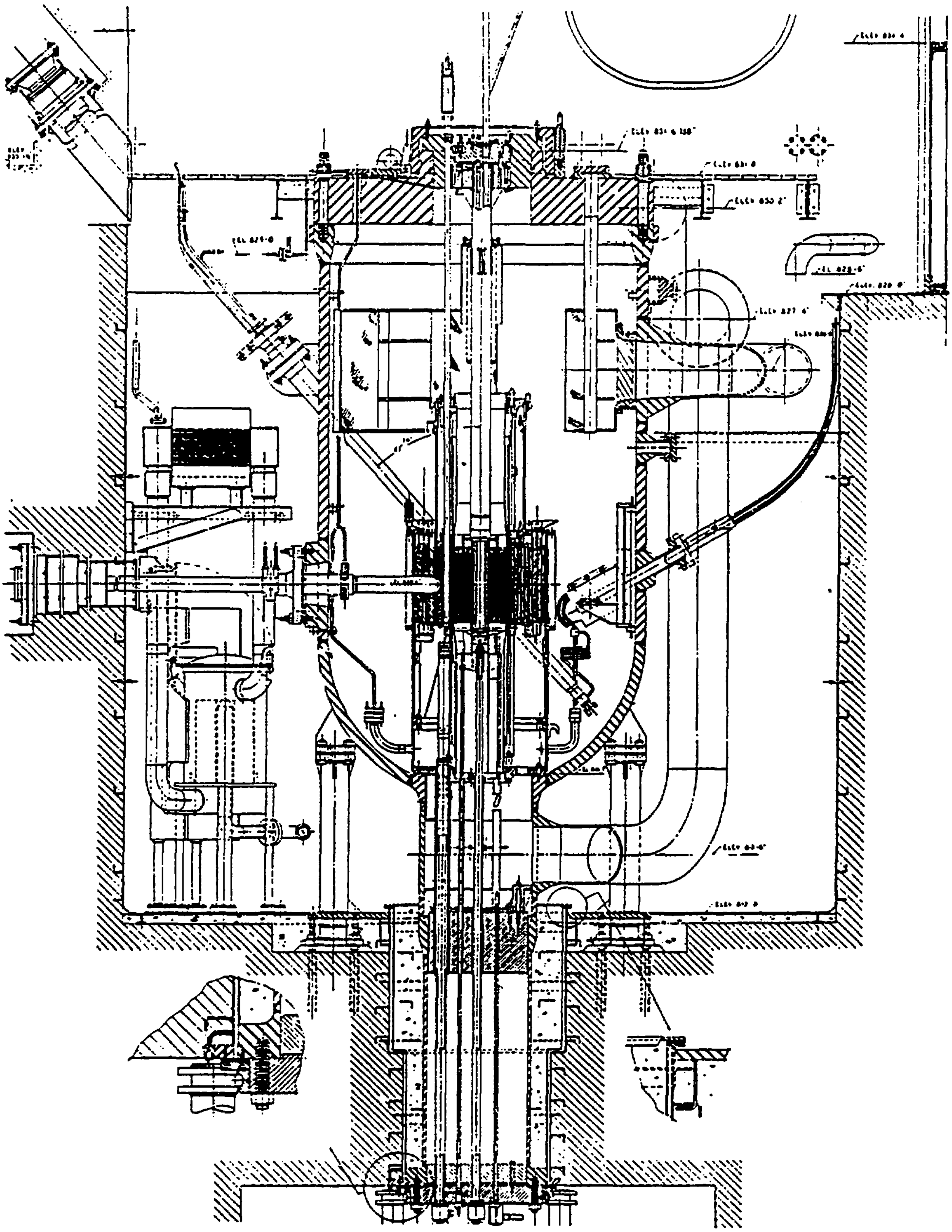

Fig. 2. Centerline view of reactor.

ORNL-LR-DWG-65776

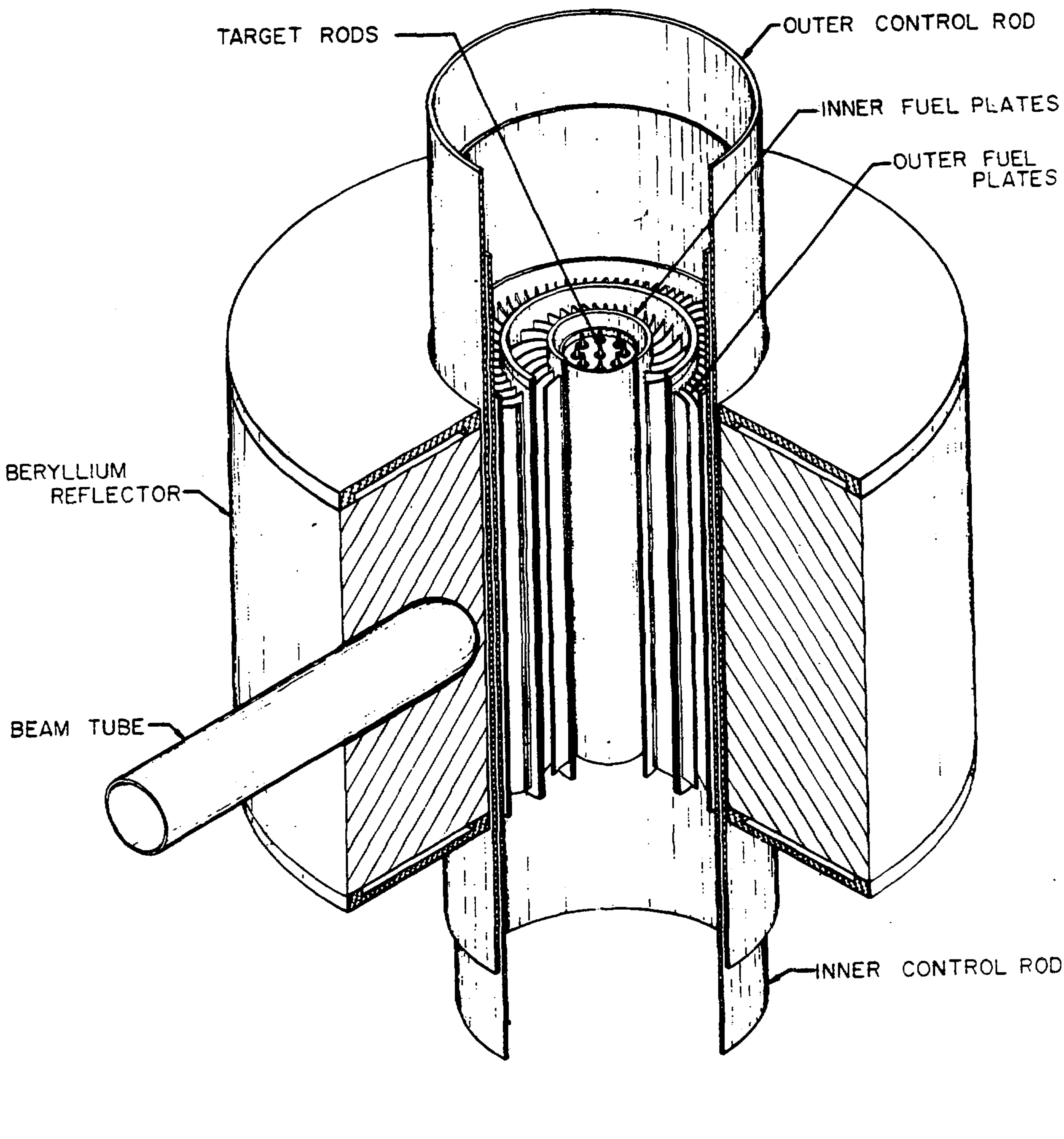

Fig. 3. Schematic of High Flux Isotope Reactor core.

Decay heat removal is generally accomplished using the normal primary/secondary cooling systems. If off-site power is lost to the primary pumps, decay heat can be removed using a DC pony motor system attached to the main coolant pump drives. Each pony-motor-driven pump is capable of supplying 1300 gal/min (82 l/sec) of flow through the core.

It has been demonstrated that the secondary heat removal system is not needed to remove decay heat. The primary piping and reactor pool have adequate heat capacity to prevent reaching saturation conditions.

The power to the DC pony motors is supplied by a dedicated battery system for each of the four pony motors, backed up by one of two on-site diesels that in turn is backed by one of two portable diesels (auxiliary electric power generators) stored off-site.

The primary and secondary heat removal systems are schematically shown in Fig. 4.

The reactor building is 128 ft × 160 ft × 86 ft in height. The building constitutes a dynamic confinement system with one of three exhaust fans continuously pulling air through the building through a series of high-efficiency and carbon filters and exhausts the air up the 250-ft (76.2 m) stack at a rate of ~28,000 cfm (13.2 m^3/sec).

The fuel is a U_3O_8/Al ceremet clad in aluminum [fuel ~30 mils (7.6410^{-2} cm) with 10 mils (2.54×10^{-2} cm) of cladding]. The active core length is 20 in (50.8 cm). Each 50-mil (0.127 cm) plate is separated by a 50-mil (0.127 cm) cooling channel. There are two concentric elements in each core assembly, the inner having 171 plates, the outer 369 plates. The two concentric cylinders have a diameter of 17.5 in (44 cm) and surround a 5-in. (13 cm) diameter flux trap target region. The fuel contains about 9.4 kg of 93-percent enriched uranium (Fig. 1).

An average fuel cycle is 26 or 27 days (at 85 MW) and can be shorter depending on the experimental loadings and target loadings in the core.

The spent fuel is stored in one of two spent fuel pools adjacent to the reactor pool. The storage capacity of these pools is being increased to 210 core assemblies (currently have 65 assemblies stored) due to the shut down of the Savannah River Reprocessing Plant.

Experimental Capabilities

The HFIR has six major experimental missions (Fig. 5).

1. Production of transuranic isotopes [primarily californium-252 (^{252}Cf)] in the target region.

2. Production of industrial isotopes in various Be reflector locations (42 locations).

3. Production of medical/industrial isotopes in a hydraulic rabbit (allows capability of insertion and removal during operation) in the target region.

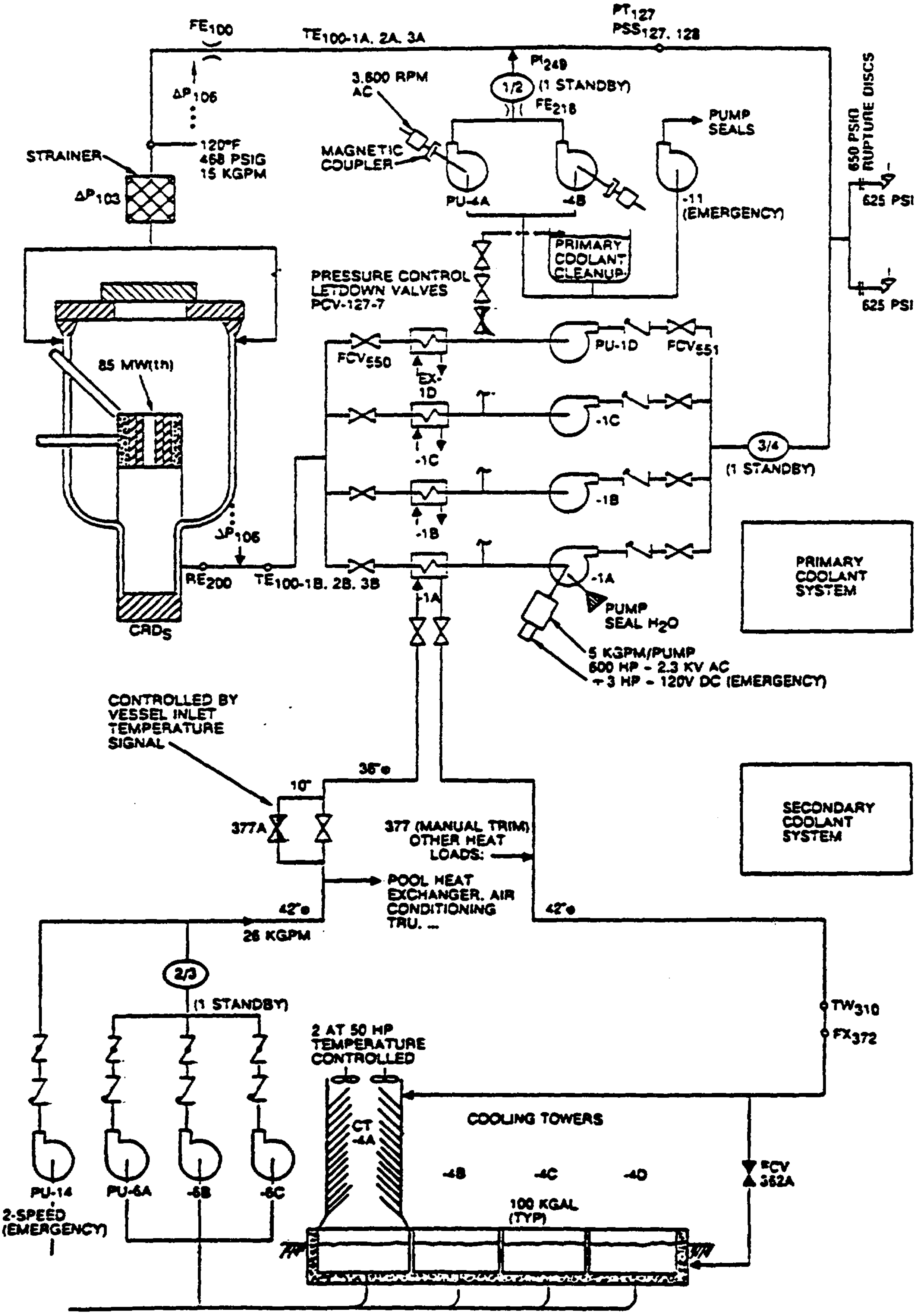

Fig. 4. Simplified HFIR process flow diagram.

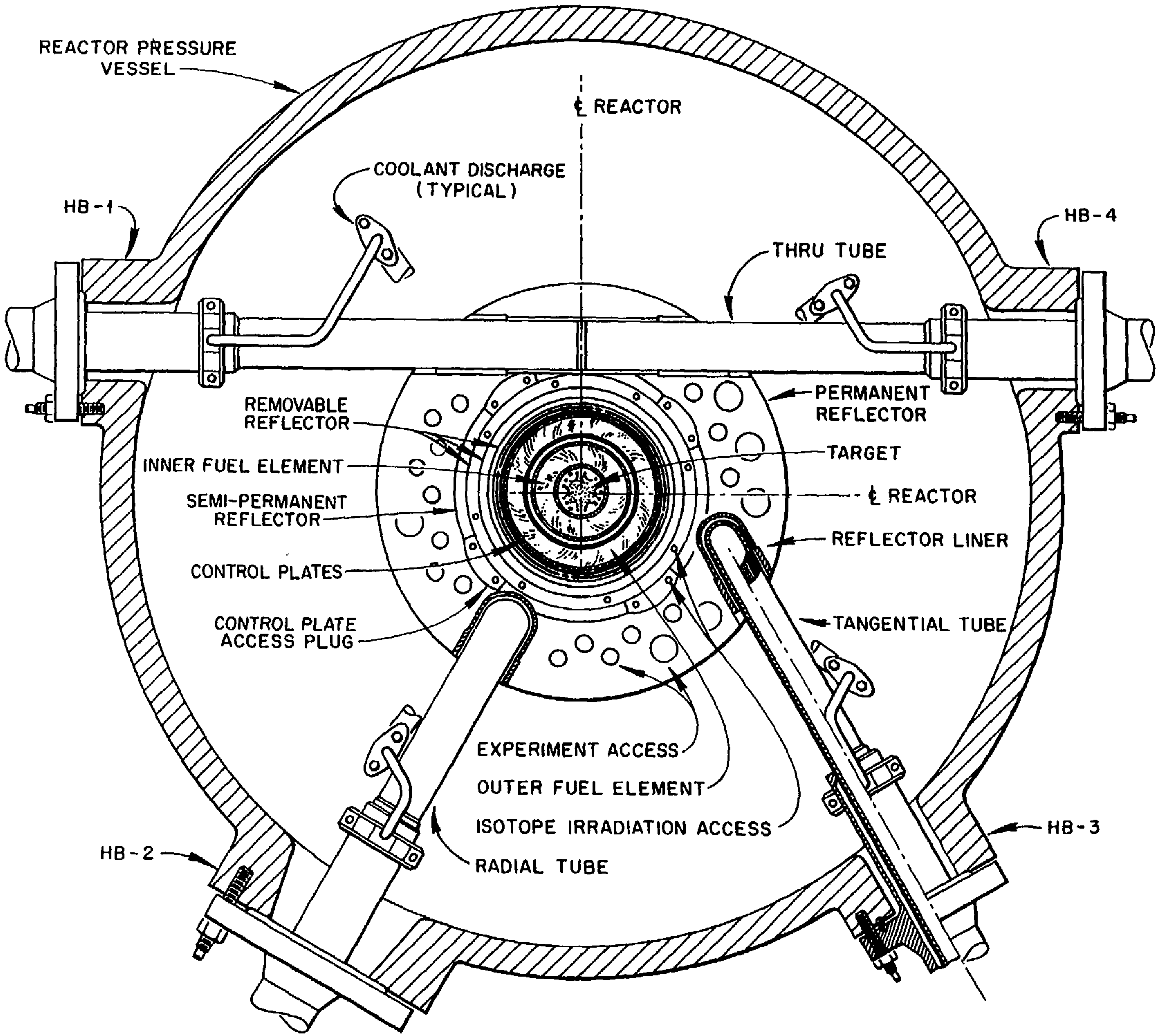

Fig. 5. Cross section of HFIR at core centerline.

4. Four horizontal beam tubes (three tangential, one radial, located at the core centerline) for materials research and four engineering facilities (slant tubes) immediately outside the Be reflector region.

5. Two pneumatic tubes, one in the reflector region and one in use, in the engineering facilities, which are used for neutron activation analysis.

6. Materials irradiation in the 42 penetrations in the Be reflector.

HFIR was designed and built to produce research and commercial quantities of transuranic isotopes and is the Western World's only supplier of significant quantities of ^{252}Cf, ^{253}Es, ^{259}BK, and ^{257}Fm. The principal transuranic isotope is ^{252}Cf, which is separated and processed by the Radiation Engineering Development Center (REDC) located adjacent to the HFIR facility. Californium-252 is used for medical, defense, industrial, and research purposes. To date HFIR and REDC have produced the following quantities of each:

	To date	**Current annual rate**
^{244}Cm	2.2 Kg	50 g
^{245}Bk	0.7 g	50 mg
^{252}Cf	7 g	0.5 g
^{253}Es	30 mg	2 mg
^{255}Fm	16 pg	1 pg

Future Upgrades at the HFIR Facility

HFIR has an ongoing aging management program. It consists of an active In-service Inspection Program (ISI) and a preventative/predictive maintenance program. Several major improvements to the facility have occurred over the last few years. These improvements include replacement of all four heat exchangers, rebuilding the four primary coolant pumps, upgrades to the electrical supply systems to the plant and improved irradiation facilities in the reflector region closest to the core. There was also a major seismic analysis and upgrade, a Probabilistic Risk Assessment (PRA) was produced along with a new Safety Analysis Report (SAR), including re-analysis of all accident analyses using state-of-the-art codes.

Near-term plans include replacement of the primary pressurizer pumps, refurbishment of the cooling tower, replacement of the diesel generators, and upgrade of the Instrumentation and Controls (I&C) systems in the facility.

Until this year, the HFIR was scheduled to be replaced with the Advanced Neutron Source (ANS). However, now that DOE has dropped its support for the ANS, a revised look at the long-term prospect for HFIR has been initiated by a multi-disciplined team chaired by C. D. West (head of ANS project) and composed of users and Research Reactors Division staff. The HFIR

Futures Group has prepared three major classes of upgrades for the facility with the goal of improved availability, predictability, and experimental capability.

The three groupings are:

1. those improvements that can be made with existing funds by rearranging resources,

2. those improvements that can be accomplished in the short term and whose costs are greater than existing funds but less than \$5M, and

3. long-term improvement that will require significant planning, funds, and possible lengthy outages.

Examples of the types of improvements in category 1 are a return to 100-MW operation, installation of a new gamma irradiation facility, redesign of the reflector to enhance materials irradiation and production of ^{238}Pu, upgrades to enhance availability, relocation of current scattering instrumentation, increase in neutron activation analysis capability.

Examples of the items being considered in category 2 would include the addition of an hydrogen-cooled cold source in use of the beam tubes, an expanded neutron scattering guide hall, longer fuel cycles, addition of more hydraulic rabbit facilities for isotope production.

Items in category 3 include the replacement of the Be reflector with a D_2O reflector and large cold source, addition of a hot cell accessible to the reactor pool, and a positron source.

The upgrade suggestions have been presented to Laboratory management for consideration and prioritization.

It is anticipated that several of these upgrades will be supported in the upcoming years since there are only a few neutron sources available for scientific research in the United States.

4th Meeting of the International Group of Research Reactors (IGORR-IV)
Gatlinburg, TN, USA, May 24-25, 1995

SURVEILLANCE PROGRAMME AND UPGRADING OF THE HIGH FLUX REACTOR PETTEN

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ABSTRACT

The High Flux Reactor (HFR) at Petten (The Netherlands), a 45 MW light water cooled and moderated research reactor in operation during more than 30 years, has been kept up to date by replacing ageing components. In 1984, the HFR was shut down for replacement of the aluminium reactor vessel which had been irradiated during more than 20 years.

The demonstration that the new vessel contains no critical defect requires knowledge of the material properties of the aluminium alloy Al 5154 with and without neutron irradiation and of the likely defect presence through the periodic in-service inspections.

An irradiation damage surveillance programme has been started in 1985 for the new vessel material to provide information on fracture mechanics properties.

After the vessel replacement, the existing process of continuous upgrading and replacement of ageing components was accelerated. A stepwise upgrade of the control room is presently under realization.

1. INTRODUCTION

The High Flux Reactor (HFR) at Petten, The Netherlands, is one of the few high power multi-purpose research reactors still in operation in Europe. It is a 45 MW water-cooled and moderated tank-in-pool reactor of the Oak Ridge Reactor (ORR) design [1]. The reactor is the property of the Commission of the European Communities (CEC) and belongs to the Institute for Advanced Materials (IAM) of the Joint Research Centre (JRC) of the CEC.

The HFR programme is predominantly funded by the governments of The Netherlands and Germany [2]. In continuous and successful operation during more than 30 years, the HFR has been kept up-to-date by replacing ageing components. In this respect, the HFR was shut down in 1984 for replacement of the aluminium reactor vessel which had been irradiated during more than 20 years [3].

The design for the new aluminium 5154 alloy reactor vessel has been based on the fact that the reactor can operate provided there is evidence that the vessel contains no critical defects. Knowledge of materials properties and likely defect presence and size is therefore requested for the assessment of the HFR vessel integrity.

2. VESSEL MATERIAL TESTING PROGRAMME

The changes of materials properties with neutron exposure are obviously of particular importance for the HFR aluminium vessel.

The loss of ductility of irradiated aluminium alloys is caused by :

- hardening by displacement damage to a minor extent,
- precipitation hardening for the major part.

Displacement damage is realized through the interaction between the atoms in the crystal lattice and fast neutrons by collision and is shown by formation of lattice defects like interstitial atoms, vacancies and eventually dislocation arrays which harden the metallic matrix and reduce the material ductility.

However, precipitation hardening by finely dispersed Mg_2Si is mainly responsible for the loss of ductility of irradiated aluminium alloys. Mg_2Si originates from a reaction of Mg in the alloy with thermal-neutron-transmutation-produced Si.

Data on the mechanical properties of the aluminium alloy Al 5154 with and without neutron irradiation are necessary for the safety analysis of the HFR vessel at the beginning of the service life (BOL) and at the end of the design life time (EOL).

The construction material of the old HFR vessel is a modified aluminium-magnesium alloy with 3.5 - 3.9% Mg content, close to the specification for type 5154 aluminium alloy.

For the replacement vessel, mainly in order to benefit from the experience provided with the old vessel, 5154 aluminium alloy was chosen with a restriction on Mg content to 3.5% in view to improve ductility at high neutron doses.

In order to confirm the material properties of the replacement vessel, various test programmes were carried out. In addition, the behaviour of the new vessel material under irradiation has been investigated in advance with the data provided by the old vessel material.

Fatigue, fracture mechanics (crack growth and fracture toughness) and tensile properties have been obtained from several experimental testing programmes with materials of the new and the old HFR vessel [4][5].

In particular, at the request of the Dutch licensing authority, low-cycle fatigue testing has been carried out on non-irradiated specimens from stock material of the new HFR vessel. The data obtained confirm that operation of the HFR is fully justified with the specified design parameters.

3. SURVEILLANCE PROGRAMME

An irradiation damage surveillance programme called SURP has been set up in 1985 at the restart of the HFR for the new vessel material to provide information on fracture mechanics properties. The objective is to survey the degradation of specific material properties in the most critically affected areas of the core box, which are located in the West and the South walls. This will be realized through selective mechanical testing of representative material specimens which are taken from selected irradiation positions periodically during the operational life of the HFR.

The material for the fabrication of the specimens has been taken from the forged cut-offs left from the core box manufacture.

Two capsules containing 15 compact tension (CT) specimens with a thickness of 25 mm and 10 tensile specimens are being irradiated respectively in an in-core and a pool side facility (PSF) position.

The in-core position has been chosen for the proper simulation of the neutron flux conditions in the West wall central area while the PSF position reflects the irradiation of the critical area of the South wall due to a typical thermal to fast flux ratio.

The capsules are of the reloadable type to allow for periodic removal of specimens and flux monitors. Neutron monitor sets are replaced every 3 years. The irradiated specimens are scheduled to be tested at different intervals of the reactor vessel life. The first testing campaign is planned end of 1995 with the removal of three CT and two tensile specimens from each capsule after an irradiation time of 10 years. Their thermal neutron exposure is reaching about 4.10^{26} n/m^2 in 1995.

4. IN-SERVICE INSPECTIONS OF THE HFR VESSEL

The design of the new HFR vessel takes into account the ability to inspect in-service [3]. The welds are positioned in accessible areas and the internal surface of the core box is machined flat.

In 1984, during its installation in the HFR containment building, the replacement vessel was subject to a Pre-Service Inspection (PSI) [6]. The periodical in-service inspections are performed on the basis of a 3 years interval, the first one being in 1988 [6], the second in 1991 and the third one in 1994.

4.1. Purpose and extent of the reactor vessel inspection

The reactor vessel is an all welded aluminium structure about 5 m high which includes the core box, the core box support structure and the cooling water inlet plenum (see fig. 1). The core box is a rectangular shaped construction fabricated from 50 mm thick forged material joined by electron beam welding. The other vessel areas use 40 mm thick plate material with MIG welds. The goal of the in-service inspection of the HFR vessel is to demonstrate that all parts assuring the core geometry and the control rod alignment will not fail in service. Ultrasonic examination is then required for a selection of welds and for the most highly exposed core box walls. An eddy current testing is also conducted on core box walls and corners for surface and near surface defects. The ISI also includes dimensional checks for the control rod alignment, visual inspection using t.v.- cameras for the general condition of the vessel and any surface flaw and penetrant testing of primary coolant inlet.

4.2. Inspection equipment

The ultrasonic inspection is carried out using the water immersion technique by means of a cylindrical mast suspended from a support frame placed on the reactor vessel top flange. The scanning mechanism mounted on this mast is remotely controlled from the pool adjacent floor. Specific probe arms are attached to this mechanism in order to achieve the requested beam angles in the test material of 45, 60, 70 and 0 degrees. The inspection is conducted in compliance with the ASME XI code requirements using the 6dB drop sizing technique and calibration blocks with side drilled holes. Ultrasonic signal treatment is realized through a data processing system, allowing the real time data display from the top and the side view of the inspected part together with data storage and evaluation of data off-line. The scanning system is also used for the eddy current inspection to check by means of absolute type probes the core box walls and corners. Calibration is carried out with an aluminium calibration block displaying various EDM notches, the reporting level being established from a 20 mm long by 3 mm deep notch.

4.3. Inspection results

Comparing the indications found during the 1994 ultrasonic inspection with those of 1991, 1988 and 1984, it can be established that a very good correlation exists between the inspections. The maximum amplitudes of the reportable indications are generally within $\pm$ 2dB with no significant change in length. No additional reportable indications and no size increases from the defects - all from the manufacturing stage - were found.

No significant defect indications were detected from the visual and dimensional inspections and from the liquid penetrant testing. The eddy current testing did not detect in 1984, 1988, 1991 and 1994 any reportable indications. The next ISI has to be carried out during summer 1997.

5. RECENT HFR UPGRADES

By implementing the policy of preventive maintenance, refurbishing and upgrading, after the vessel replacement, other major components of the reactor have been replaced, such as the primary and pool heat exchangers, the beryllium reflector elements and the guaranteed power supply. The major HFR upgrades are summarized in Table 1. The most recent projects are developed below.

5.1. Renewal of ion drain and storage tanks

Corrosion damage led to the replacement of the tanks of the ion drain system. In 1992 and 1993, the new ion drain and storage tanks have been installed, instrumented and put into operation.

5.2. Renewal of HFR Mains Power Distribution Cabinet

Rearrangements of the electrical power installations at the HFR have necessitated adaptation of the mains distribution cabinet. Furthermore, spare parts of the existing cabinet were no longer available, endangering future reliability. Several cable routings have been modified for fire protection reasons.

The new distribution cabinet was installed during the 1993 spring stop. All mains supply and user cables were renewed. Electrical supply to essential parts of the installation during the replacement was provided by temporary provisions from the site diesel-driven emergency supply station.

5.3. Renovation of the secondary cooling water outlet

In 1993, it became necessary to carry out the renovation of the secondary cooling water outlet pipeline. The work was performed during the summer maintenance stop.

The original concrete piping was internally relined with synthetic interconnected pipe sections over the last 200 meters from the outlet. These sections were pushed in from the dune side in seaward direction. For this operation the outlet pipe was cut on purpose at the foot of the dunes and closed afterwards.

The remaining space between the outside of the relining piping and the inside of the original concrete pipe was filled with a special fluid pumped in under pressure thereby replacing the present water. This fluid, a suspension in water with concrete, marl, clay and plaster, starts hardening after 6 to 7 hours. The outlet stop valve was renewed and relocated 75 meters inland to compensate for dune displacement over past years. The actual outlet diffuser grid was completely renewed.

The breakwater, in which the last part of the piping is situated, was repaired and its stop was covered with concrete on request of the authorities. This concrete coverage should also prevent future cracking of the outlet piping.

5.4. Renewal of the Chlorine Injection System of the secondary cooling

To avoid algae growth in the piping and heat exchanger systems of the HFR, chlorine was injected. On request of the Dutch Labour Inspection authorities the use and storage of chlorine has to be avoided, so alternatives had to be investigated.

Sodium hypochlorite has been chosen for safe handling and storage and for improved environmental effects. An investigation of the dose of sodium hypochlorite needed to avoid algae growth has been carried out.

Building adaptation for and installation of the new sodium hypochlorite injection system comprising the tank, the dosage pump and associated valves and instrumentation was completed in 1993.

5.5. Control room upgrading

Re-configuration and upgrading of the HFR control room functions and equipment became necessary in the beginning of the nineties in order to replace outdated equipment and to introduce modern ergonomic principles in the fields of display and easier access to reactor and experimental data.

The technical requirements and lay-out for a modernized control room have been drawn up. Due to the estimated price for the upgrading and to prevent loss of operation time during the reconstruction it was decided to carry out the upgrading in a stepwise approach.

The main objectives for the new control room are to permit the operators to monitor the performance of the reactor and experiments, to optimize the performance during normal conditions and to control the installation during planned and unplanned operational conditions.

To facilitate monitoring of the experiments the computers and terminals of the experiment data handling system DACOS had to be incorporated near the control room within the physical protected zone. This sub-project has been completed in 1993.

Due to the lack of spare parts and since the old system needed more and more maintenance the distribution board has to be renewed as a first step. The outdated 25 years old annunciator system has then to be replaced. The annunciator system itself will be modernized and a Data Acquisition System (DAS) will be introduced. After testing, the installation of the new annunciator system is planned for the summer stop of 1995.

After this installation, the specialized annunciator system for the interlocking of the experiments will then be renewed. This is planned for the summer of 1996. Presently the interlock is placed in the control desk and has to be replaced due to ageing of the components and a lack of spare parts. The new interlock will be situated in a separate cabinet in the basement of the control room.

The Data Acquisition System (DAS) will be an additional information system for the operators of the HFR. All information from an annunciator signal will be presented on the DAS. It is also planned to present the required operator actions on the DAS in case of an alarm. The DAS will also be a supplement to the present instrumentation and warning systems which will give the operators additional information to optimize the operational parameters. In the future several operational parameters will be available for the experimentators. Based on technical specifications the hardware has been ordered and delivered. A pilot system showing the possibilities has been developed and will be placed in the computer room in the near future. After the review of the operator experience with DAS the final version will be specified and finally built.

Also in the framework of the control room upgrading the main part of the nuclear instrumentation has been renewed: replacement of start-up channels, renewal of off-gas monitor and N-16 equipment, introduction of a second set of safety channels working in a two out of three mode and replacement of the old set of safety channels (scheduled for the summer of 1995), replacement of cladding rupture monitor by a two out of three system, and replacement of low and high activity gas monitor system.

Finally, with the renewal of the control desk, which will be redesigned and adapted to the latest ergonomical principles, planned in 1997, the HFR will be equipped with a new control room and will remain an up-to-date reactor ready for its mission for the next 10 years.

6. CONCLUSIONS

Routine in-service inspections are being performed on the High Flux Reactor vessel at Petten in accordance with the operating licence requirements. The results from the periodic inspections carried out in 1984, 1988, 1991 and 1994 have shown that there has been no change in the overall reactor vessel integrity since its installation in 1984.

The irradiation damage surveillance programme comprises specimens from the new vessel material. Several samples are planned to be removed from irradiation in 1995 and then subsequently tested.

Information on likely defect sizes together with knowledge of the material properties including the effects of irradiation on the 5154 aluminium alloy allows to demonstrate that there is no critical defect and then to assess the integrity of the HFR vessel.

After more than 30 years of operation, the HFR Petten, one of the most powerful multi-purpose research reactors in Europe, can still be regarded as a modern and up-to-date research tool, as a result of continuous preventive upgrading, refurbishment of outdated components and modernization. This is reflected by the high availability of the reactor of on average more than 250 days per year, even reaching 280 days in the recent past years, and by the efficient utilization of the diverse irradiation positions.

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TABLE 1 - HFR PETTEN, HISTORY, MAIN UPGRADINGS AND RENOVATIONS

1958-1961	Design and construction
1961	First criticality of HFR
1962	Transfer from RCN to EURATOM - Maximum power 20 MW
1966	Power increase to 30 MW
1970	Power increase to 45 MW
1972	Introduction of burnable poison
1974-1977	Feasibility study for replacement of reactor vessel
1978	Decision to replace reactor vessel
1980-1981	Design of the new reactor vessel
1982-1983	New reactor vessel manufacture
Nov. 83-feb. 85	Reactor vessel replacement
1987	Replacement of the primary heat exchangers (3 x 22 MW)
1988	Replacement of the start-up channels
1988	Guaranteed power supply replacement
1988	Extension of the HFR building complex
1988	Upgrading of the central data acquisition system DACOS
1989	Replacement of the pool heat exchangers (2.6 MW)
1989-1990	Replacement of the Beryllium reflector elements
1990	Installation of the Second Reactor Power Protection System
1991	Renovation of the high construction hall and the secondary pump building
1992-1993	Ion Exchanger drain and storage system replacement
1992-1993	Renewal of the Chlorine Injection System of the secondary cooling
1993	Renovation of the secondary cooling water outlet
1993	Renewal of HFR Mains Power Distribution cabinet
1995	Renewal of original Overpower Protection System
1993-1997	HFR Control Room upgrading

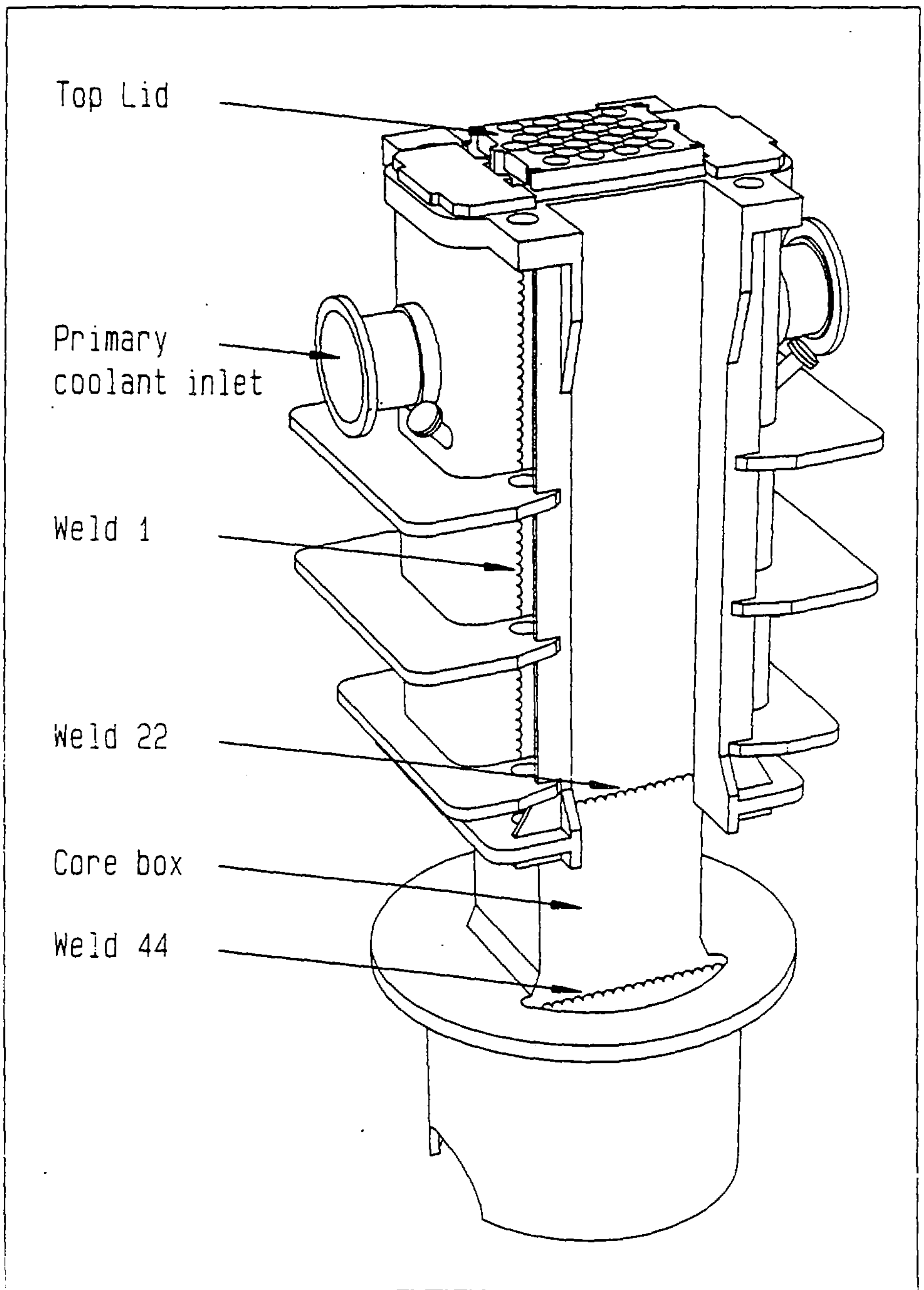

Fig. 1 : HFR Reactor Vessel

Neutrons Down-Under:

Australia's Research Reactor Review

Presented by Allan Murray[#] to the 4th Meeting of the
International Group on Research Reactors
May 24-25, 1995, Tennessee, USA

Notes

The Australian Nuclear Science and Technology Organisation - ANSTO - was formed in April 1987 to replace the Australian Atomic Energy Commission.

ANSTO's MISSION is to ensure that its research, technology transfer, commercial and training activities in nuclear science and associated technologies will advance Australia's innovation, competitiveness and environmental and health management.

ANSTO will also maintain and further develop its scientific and technological resources and will continue to operate as a national center for science and technology to advance Australia's national and international nuclear policies and interests.

ANSTO has science and technology programs in

- Advanced Materials
- Applications of Nuclear Physics
- Biomedicine and Health
- Environmental Science
- Engineering
- Nuclear Technology
- Radiopharmaceuticals.

ANSTO NUCLEAR TECHNOLOGY PROGRAM

- The Nuclear Technology Program maintains and operates a major neutron source, HIFAR, as a national irradiation and beam facility. HIFAR is a tank type reactor with heavy water moderator and primary coolant within an aluminium tank surrounded by a graphite reflector.
- Thermal neutron flux of 1x1014 n cm-2 s-1
- During 1994, HIFAR operated at an average power of 10 MW(th) for 7,424 hours (83.4% of available time).

[#] currently Counsellor (Nuclear), Embassy of Australia, Washington DC

HIFAR NEUTRON BEAM INSTRUMENTATION

- 2 single crystal diffractometers (medium and high resolution)
- 2 powder diffractometers (medium and high resolution)
- a triple axis spectrometer
- a neutron polarisation instrument (LONGPOL)
- small angle scattering instrument (AUSANS)
- 1,100 person-days of research utilization per year, about two-thirds from projects from Universities

HIFAR IRRADIATION FACILITIES

- 81 irradiation facilities in total: 51 vertical, 30 horizontal
- Isotope production: 1,000 irradiations in in-core rigs and 2,400 irradiations in self-service facilities each year
- About 10 kCi of molybdenum-99 produced for generators each year
- NTD silicon irradiation: 556 irradiations; about 15 tonne Si per year
- NAA of mineral samples: about 350 irradiations

AUSTRALIAN RESEARCH REACTOR REVIEW

Commenced in September 1992, the Review had the following Terms of Reference:

- Whether, on review of the benefits and costs for scientific, commercial, industrial and national interest reasons, Australia has a need for a new reactor.
- A review of the present reactor, HIFAR, to include: an assessment of national and commercial benefits and costs of operations, its likely remaining useful life and its eventual closure and decommissioning.
- If Australia has a need for a new nuclear research reactor, the Review will consider: possible locations for a new reactor, its environmental impact at alternative locations, recommend a preferred location, and evaluate matters associated with regulation of the facility , and organisational arrangements for reactor-based research.

RESEARCH REACTOR REVIEW: SUBMISSIONS

ANSTO's submissions:

A: Research Reactors: Local and International Experience
B: The Need for a New Research Reactor in Australia
C: The Proposed New Research Reactor for Australia
D: HIFAR: The High Flux Australian Reactor
E: Regulation and Siting Issues for a New Research Reactor
F: Evaluation of the New Research Reactor

Submissions by others included issues of: safety concerns and risks of accidents; routine and accidental radiation releases; health effects; siting and regulation

RESEARCH REACTOR REVIEW: FINDINGS (August 1993)

- Research reactor neutrons are a unique and broadly applicable scientific tool for leading edge investigations in many disciplines.
- HIFAR is an older model reactor not competitive in design, neutron flux or instrumentation with modern reactors.
- HIFAR is efficient and safe, within its design and instrumentation limits. HIFAR operates safely by an adequate margin.
- No evidence presented to establish a realistic outer limit on the remaining life of HIFAR. A PRA is required to assess the remaining life and upgrade possibilities.
- Scientific accomplishments with HIFAR, training, commercial activities and medical radioisotope services were positively viewed.
- ANSTO's proposed specifications for a new reactor would be a suitable basis for the design.
- No conclusion that a reactor would be the best future choice for a neutron source - possibility exists for a spallation source and a small reactor for radioisotopes.
- A multi-purpose reactor cannot be financially self-supporting.

RESEARCH REACTOR REVIEW: RECOMMENDATIONS

- keep HIFAR going
- commission a PRA to ascertain HIFAR's remaining life and refurbishment possibilities
- identify and establish a HLW repository
- accept that neither HIFAR nor a new reactor can be completely commercial
- any decision on a new neutron source must rest primarily on benefits to science and Australia's national interest
- make a decision on a new neutron source in about five years' time (1998).

RESEARCH REACTOR REVIEW: CONDITIONS

Conditions for a positive decision to be made on a new reactor include:

- a HLW site has been identified
- no evidence that spallation technology can economically offer as much or more than a new reactor
- no practical initiation of a cyclotron anywhere worldwide to produce technetium-99m
- good evidence of strong and diverse applications of neutron scattering capability in Australian science
- that the national interest remains a high priority.

Extract from the Report of the Research Reactor Review, August 1993, based upon ANSTO's preferred specifications as submitted to the Review

Table 4.2: Design Proposals for a New Reactor

Requirement	*Specification*	*Reason*	*Comparison with HIFAR*
Flux n $cm^{-2}sec^{-1}$	$3x10^{14}$	Minimum for world-class beam research	10^{14}
	Compact core	Maximises flux with minimum power	
	Heavy water reflector	Thermal source	Heavy water moderator Graphite reflector
	Large reflector region	Minimises flux leakage	
	Cold source Hot source	To allow research not now possible in Australia	HIFAR has neither cold nor hot sources
Beam Facilities	8 beams: 2 cold, 5 thermal, 1 hot Preferably extra 2 beams for thermal and hot neutrons		9 beams, all thermal neutrons 9 experimental positions
	Guide hall	Experimental space Scientists separate from operators	No guide hall
	Tangential beam ports	Superior quality beam	Radial beams - inferior quality
	Optimised biological shield	Maximum radiation protection	
Fuel	Low enriched uranium	Non-proliferation concerns	High enriched uranium

XA04C1682

UPGRADE AND MODERNIZATION OF THE NBSR

Robert E. Williams
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Gaithersburg, Maryland 20899

INTRODUCTION

The NBSR, a 20-MW research reactor operated by the National Institute of Standards and Technology, has become the leading US laboratory in neutron research. About 1000 scientists from 200 industries, government and foreign laboratories, and universities conducted experiments at the NBSR in 1993[1]. Since 1990, when the first instruments in the Cold Neutron Research Facility (CNRF) became available, the number of research participants has doubled. A major program of modernization and facility upgrade was initiated in order to meet this growing demand, and to assure safe and reliable reactor operations for 30 additional years.

A scheduled shutdown, begun in late May 1994, is nearing completion at this writing (May 1995). To upgrade the CNRF, the D_2O cold neutron source has been replaced with a liquid hydrogen cold source, and the remaining four neutron guides are being installed. In addition, the primary and secondary cooling systems have been modernized with the installation of plate heat exchangers, and the refueling system has been completely refurbished. Periodic maintenance is also being done. The entire D_2O primary coolant inventory is about to be replaced, and new shim arms have been installed.

COLD NEUTRON RESEARCH FACILITY

The liquid hydrogen cold source, and the CNRF, was described in the IGORR-3 Proceedings, and elsewhere,[2,3] so only a brief review will be given here. Figure 1 shows the main components of the LH_2 source. A naturally circulating thermosiphon will provide the moderator chamber a steady supply of liquid hydrogen through the center-most of four concentric tubes. A two-phase mixture of vapor and liquid will rise through the adjacent tube to the condenser, located two meters above the chamber, on the reactor face. A new 3.5-kW refrigerator provides helium at 14 K to the condenser. The anticipated heat load, however, is only 1000 W, so about 2 grams of liquid hydrogen will circulate in the thermosiphon. Thermal hydraulic tests[4] conducted at NIST-Boulder on a full scale glass mockup of the moderator chamber, demonstrated that at least 2200 W can be removed via steady, two-phase flow, with a vapor fraction of less than 20%. A 2-m^3 ballast tank is open to the chamber, so in the event of a refrigerator failure, the hydrogen will expand into the tank, where it is stored at 400-500 kPa when the system is shutdown. All components are surrounded by helium containment jackets, providing at least two barriers between hydrogen and oxygen. The location of the cryostat assembly in the reactor is shown in Figure 2.

The cryostat assembly, the hydrogen condenser, and the ballast tank have been installed, subject to rigorous quality control and testing to assure reliability and safety. The LH_2 source will increase the yield of cold neutrons ($E \leq 5$ meV, $\lambda \geq 0.4$ nm) by at least a factor of four. Coupled with the return to full power (the reactor has been operating at 15 MW for two years), and guide improvements, the flux in the guide hall will be six times higher in 1995 than it was before the shutdown.

Installation of four additional guides, will enable the completion of the remaining cold neutron instruments, bringing the CNRF total to 15 instruments on 8 guides. New in-pile pieces for CTE and CT (see Figure 2) will have ^{58}Ni walls, and supermirrors for their top and bottom surfaces. A curved guide on CTW will be used inside the reactor building.

PRIMARY HEAT EXCHANGERS

Three new plate heat exchangers have replaced the two shell-and-tube exchangers, in service since 1975. NBSR is the first application of this size and type of heat exchanger for a nuclear reactor primary system. They have been operated reliably in other applications for tens of millions of hours. Because the reactor primary is D_2O, several extraordinary quality control measures were taken during fabrication. Each plate is one-third thicker than usual, and subject to nuclear quality assurance. Type 316 stainless steel was used to fabricate the plates, all of which were subject to dye-penetration testing. Pairs of plates were laser-welded into cassettes rather than relying on gaskets to seal D_2O volumes. Mass spectroscopy He leak tests of each cassette demonstrated that there were no leaks at the limit of 10^{-9} stp cc/sec.

Installation of the new heat exchangers necessitated major changes in both the primary and secondary piping. The presence of smaller flow channels requires that the secondary must be kept free of foreign material introduced in the cooling towers (bugs), so new filters and backwash capabilities were added. The new heat exchangers are much smaller (and less expensive) and more efficient than the old ones. Each is rated at 12 MW in normal operation; only two are needed at full power. The rebuilt system is expected to meet the cooling requirements of the NBSR for another 30 years.

REFUELING SYSTEM

After the core was off-loaded in June, operations began maintenance of the refueling tools. They discovered that the boral layer attached to the bottom of the refueling plug was interfering with the tools. Inspection of the plug, using a borescope, revealed swelling and sagging of the boral. It was decided to lift the 10-ton plug from the top of the reactor vessel, remove the boral, and completely refurbish the fuel handling tools penetrating the plug. NBSR fuel is handled remotely to keep the D_2O primary isolated. The plug had been in place since its installation prior to the first criticality in 1967. Galvanic corrosion between Graphitar bearings on the tool shafts and the aluminum in the boral was the cause of the corrosion.

CONCLUSION

By July 1995, the NBSR should again be operating at 20 MW, with greatly enhanced experimental capabilities for the CNRF. Future projects include modernizing the thermal neutron instruments, and relicensing the NBSR to 2024, in order to meet the needs of the scientific community for decades to come.

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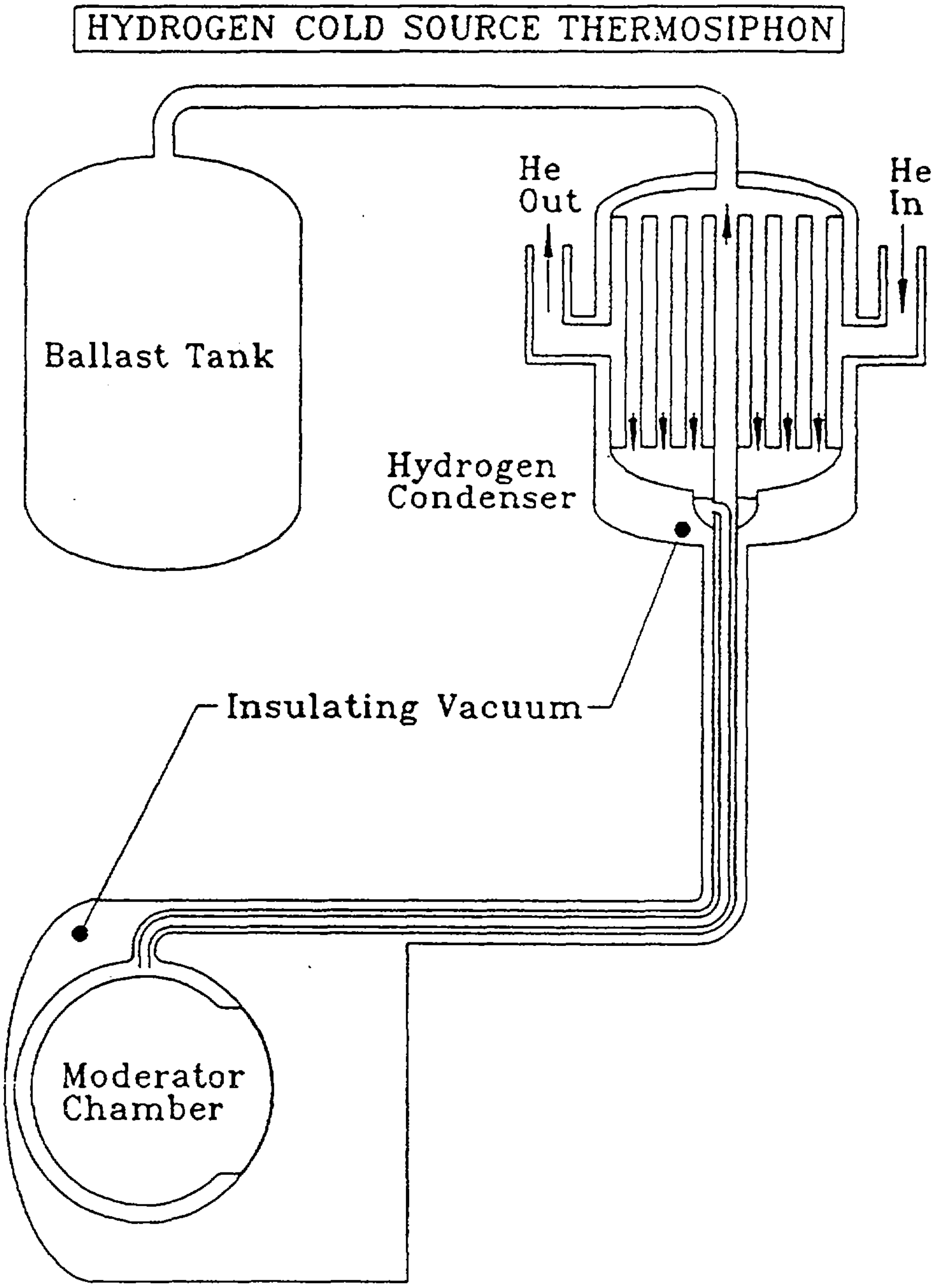

Fig. 1. Schematic of the liquid hydrogen cold neutron source. Each component is completely surrounded by a helium containment, not shown.

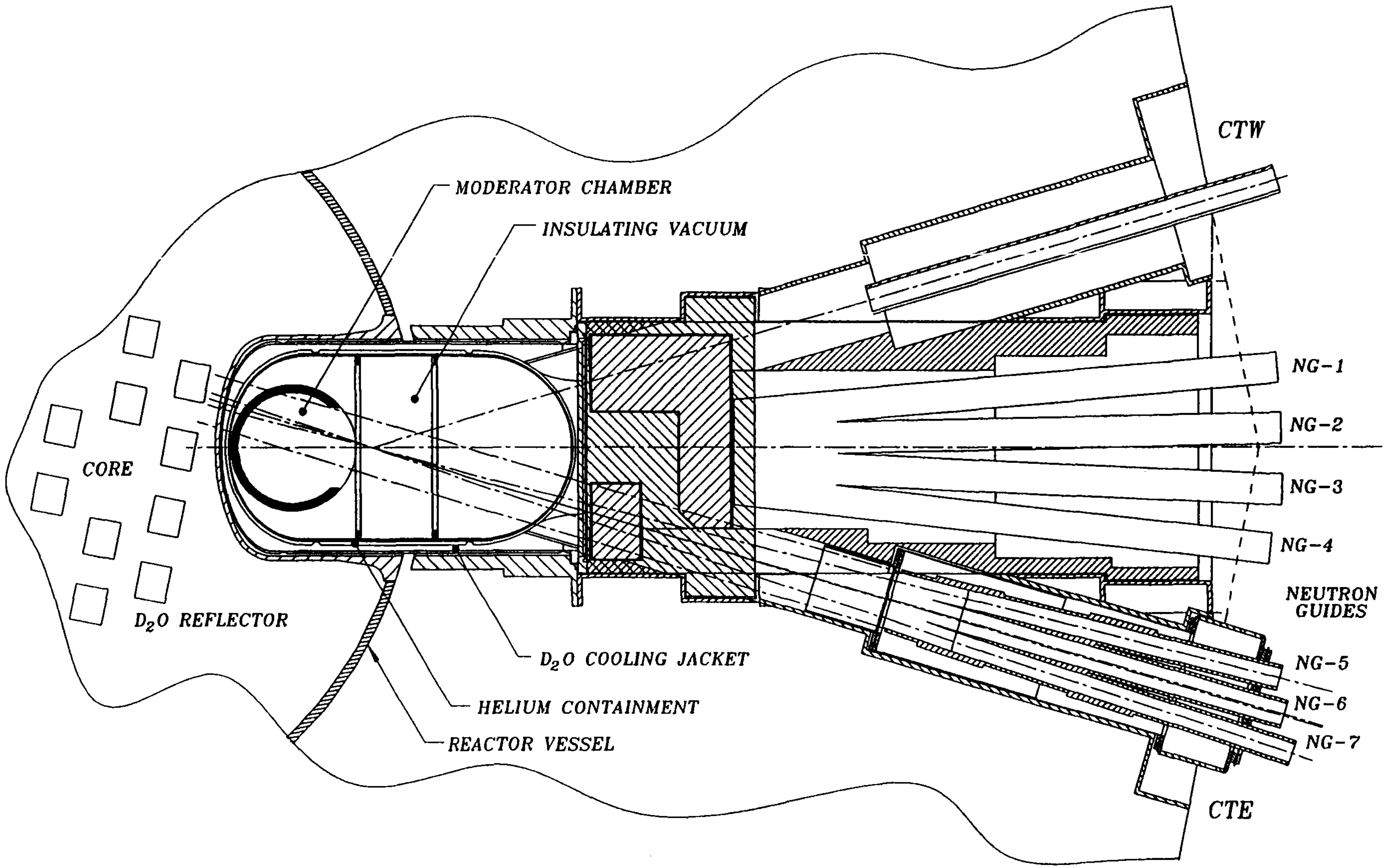

Fig. 2. Plan view of the hydrogen cold source in the cryogenic beam port of the NBSR.

NBSR

National Institute of Standards & Technology

Gaithersburg, Maryland USA

Licensed Power: 20 MW

HEU Fuel, D_2O Moderator

Peak Flux: 4×10^{14} $n/cm^2/sec$

Fuel Cycle: 35 Days

11 Thermal Neutron Beams

8 Cold Neutron Guides

Irradiation Facilities

NBSR History

1967	Critical December 7
1969	10 MW
1985	20 MW
1987	D_2O Cold Neutron Source
1990	Guide Hall Instruments NG-5,6,7
1993	10 Cold Neutron Instruments on NG-3,5,6,7 and CTW
1994	Shutdown May 26 for LH_2 CNS and Modernization
1995	Restart
2004	Relicense for 20 Years

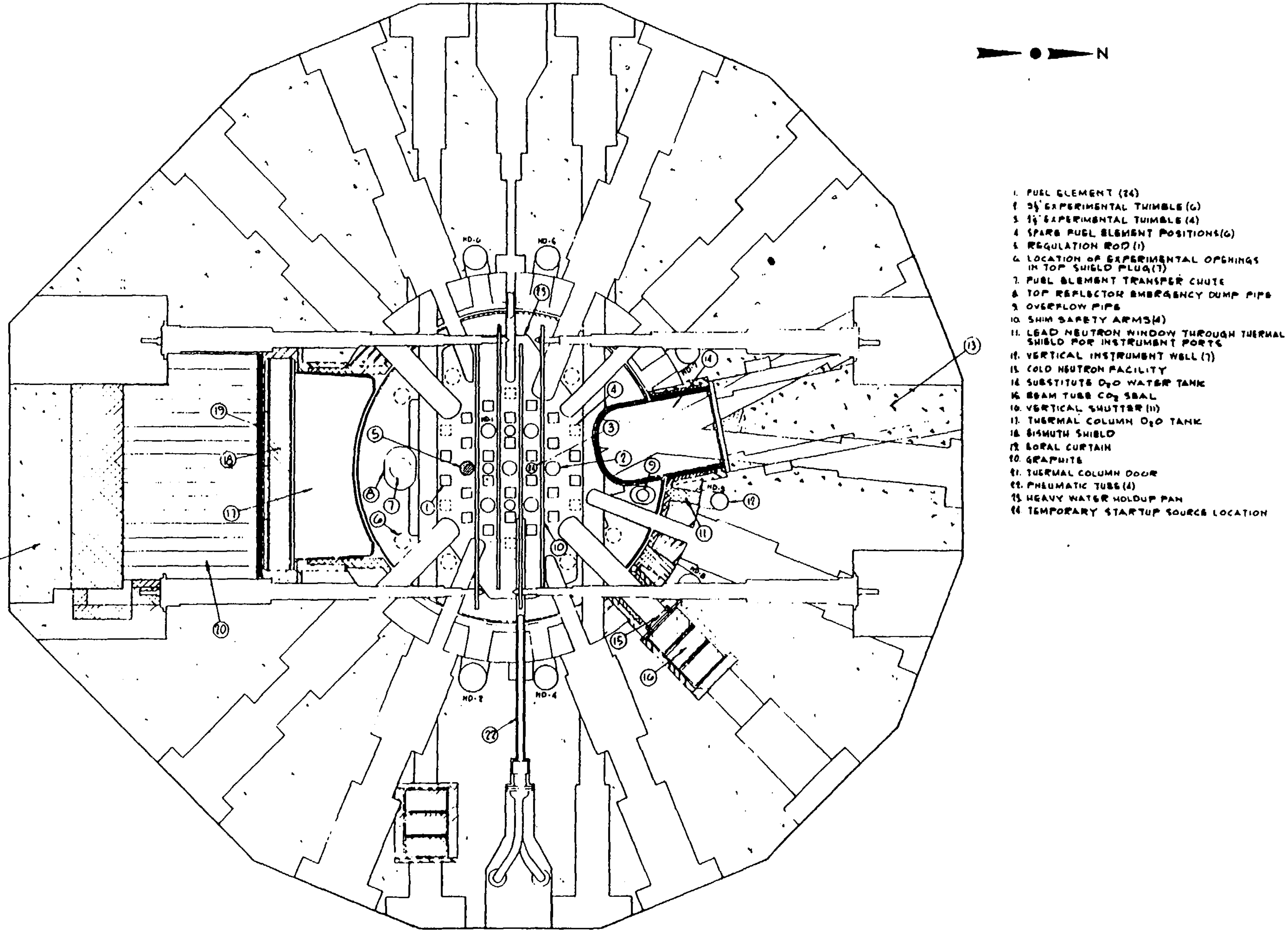

Figure 4.2 Reactor plan view.

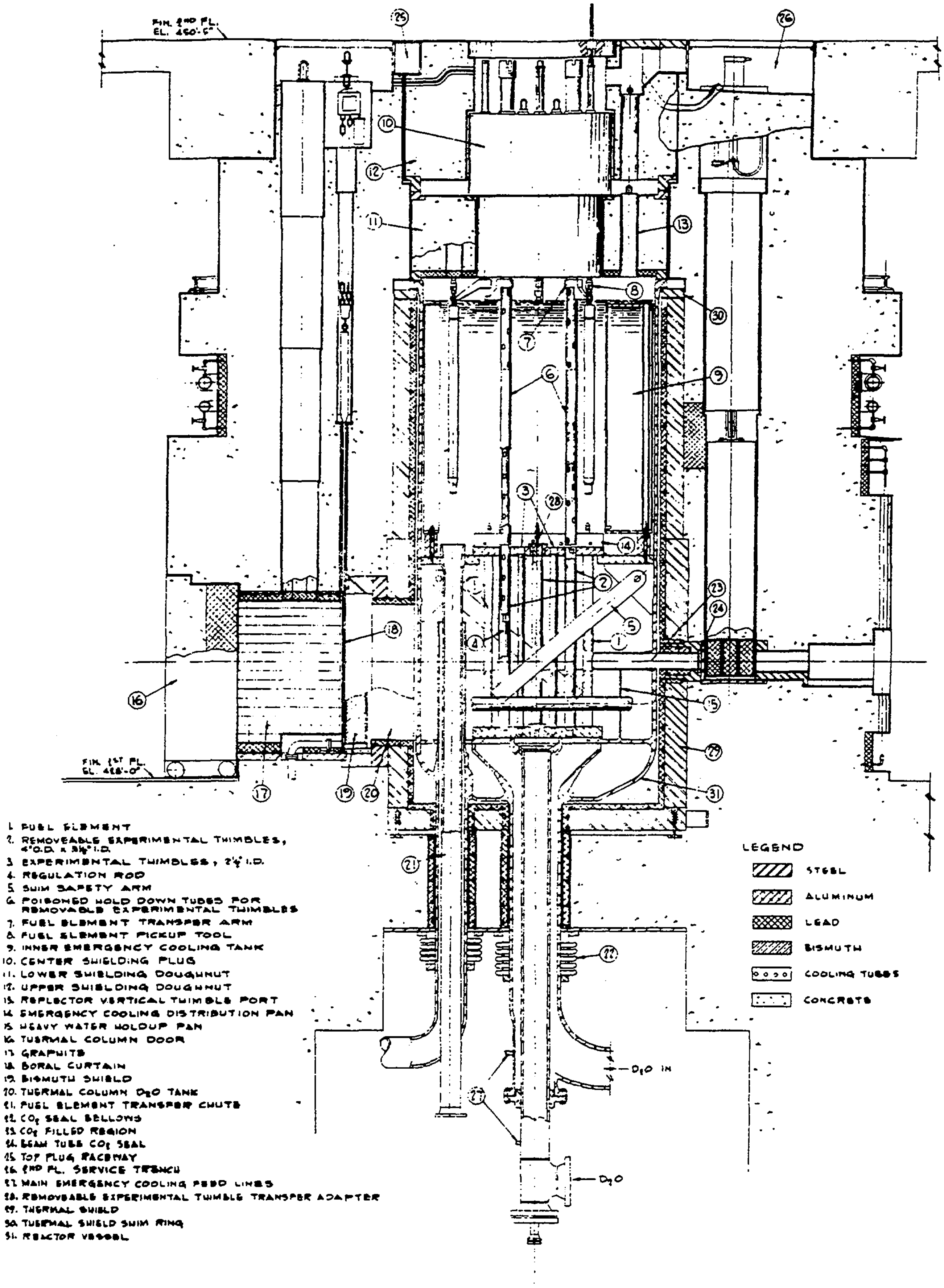

Figure 4.1 Reactor elevation.

Major Shutdown Activities

1. Primary Heat Exchangers, Piping
2. Refurbish Refueling System
3. Replace D_2O Cold Source with LH_2
4. Install 4 Additional Neutron Guides
5. Replace D_2O Coolant Inventory
6. Install New Shim Arms
7. Rad Waste Disposal
8. Lots of Other Stuff

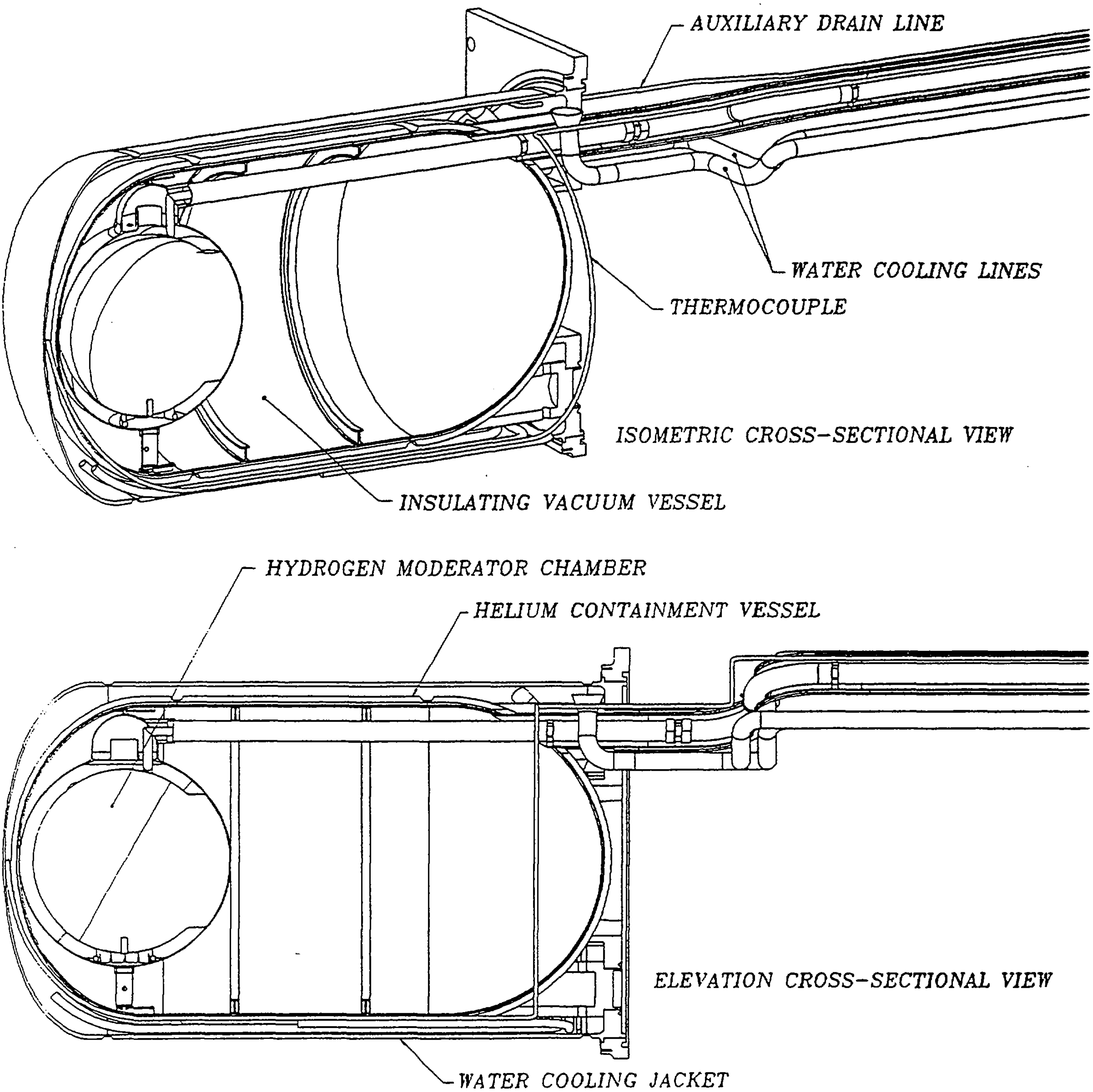
AUXILIARY DRAIN LINE
WATER COOLING LINES
THERMOCOUPLE
ISOMETRIC CROSS-SECTIONAL VIEW
INSULATING VACUUM VESSEL
HYDROGEN MODERATOR CHAMBER
HELIUM CONTAINMENT VESSEL
ELEVATION CROSS-SECTIONAL VIEW
WATER COOLING JACKET

Figure 3

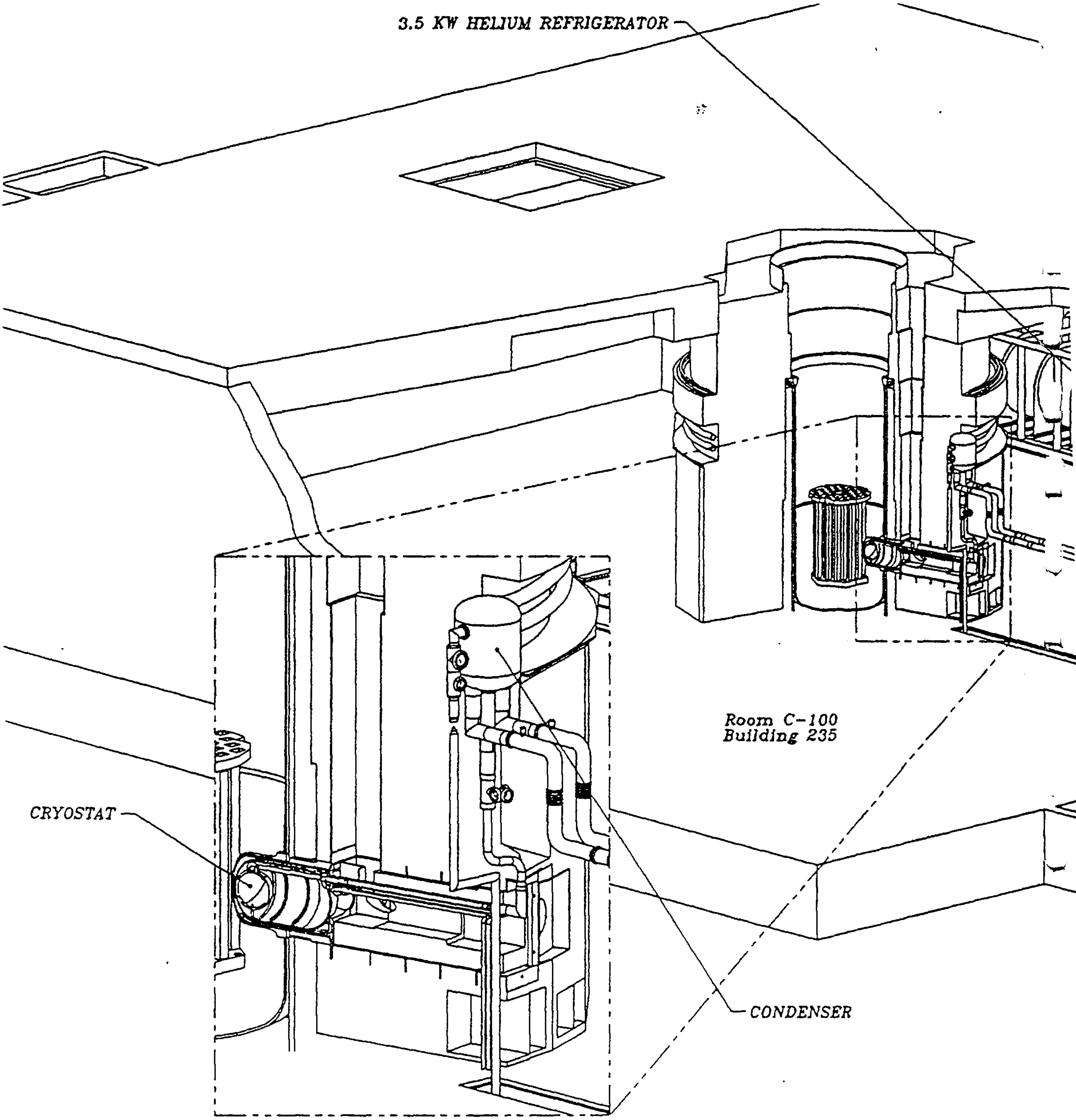
3.5 KW HELIUM REFRIGERATOR
CRYOSTAT
Room C-100
Building 235
CONDENSER

Figure 2

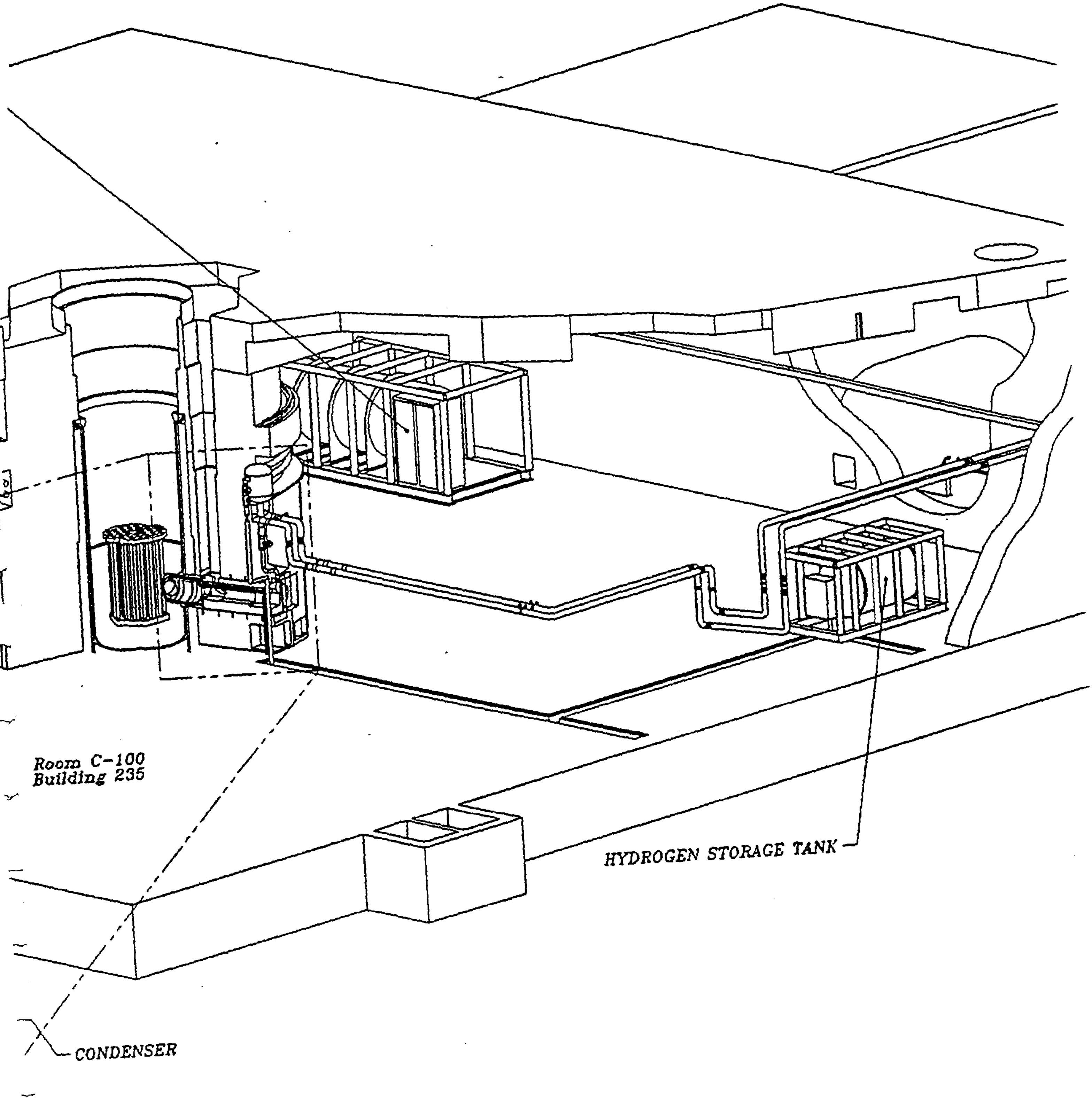

Figure 2

XA04C1683

ORPHEE REACTOR. UPGRADE OF THE INSTALLATION.

B. FARNOUX [a] and M. MAZIERE [b]

a) DSM, ORME DES MERISIERS, CEN SACLAY F 91191 GIF SUR YVETTE

b) Service d'exploitation ORPHEE, CEN SACLAY F 91191 GIF SUR YVETTE

ABSTRACT

Designed by the end of the seventies, the ORPHEE Reactor is equipped with two hydrogen cold sources, one hot source and six cold neutron guides. The neutron beams are extracted by nine beam ports and used in two experimental halls, the reactor hall and the neutron guide hall. After fourteen years of use, a modernisation programme is in progress. One step concerns the neutron guides, another one the cold sources with the modification of the cell geometry in order to increase the cold neutron flux. This operation requires the use a new cryogenerator to ensure liquefaction capabilities for the new cells. It is also scheduled to replace the Zircaloy core housing in order to avoid difficulties linked to the expansion under irradiation.

1. INTRODUCTION.

After fourteen years of continuous run of the ORPHEE reactor, about 70 per cent of the experimental facilities are using cold neutron beams (see figure 2) and the demand for such neutron beams is increasing. In order to fulfil such an increase several actions have been started [1] and are in progress.

One immediate step is a modernisation programme for a more efficient use of the neutron beams. It consists in the improvement of the present neutron guides. This is the result of the co-operative development with industry of the metallic super mirrors. A medium term step is the creation of new cold neutron guides. Preliminary study has been done and is the so-called ORPHEE PLUS project described in a previous paper [2]. On line with the modernisation programme and

11.05.95

induced by this last project, is the change of the shape of the cold source cell and the cryogenerator. On a longer time scale the possible production of super-mirrors with larger critical angles offer the possibility of creating efficient thermal neutron guides.

At short time another important operation on the reactor will be the change of the Zircaloy core housing. With these improvements the ORPHEE reactor and its experimental set-up will remain a modern neutron source for the next decades.

2. *PRESENT NEUTRON GUIDES UPGRADE.*

An extensive study of production of metallic multilayers to get reliable and efficient super mirrors has been developed at the Laboratoire Leon Brillouin these past years in close collaboration with the industry [3]. It results now in an industrial production of very good reflectors with a critical angle two times greater than that of the natural nickel (for that reason these mirrors are called "super mirrors $2\theta_c$") at an acceptable price. This new possibility is now extensively use at the ORPHEE reactor first to improve the present neutron guides and second to create new beams by using benders :

- *Present guides.* In the original design all the neutron guides were curved in order to act as a filter. A curved neutron guide is characterised by a "critical wavelength" below which the transmission is very low. It is now clear that a too large value of this wavelength is not necessary and could be an handicap for many experiments due to the limitation of the wavelength range. Two of the six ORPHEE reactor guides have a strong curvature, G1 and G2 with a characteristic wavelength of respectively 6 Å and 4 Å. During the 1994 winter shut-down, the natural nickel mirrors of the curved part of the G2 guide have been replaced by $2\theta_c$ super mirrors. The result is a decrease of characteristic wavelength from 4 Å to 2 Å and an increase of the transmitted flux in the low wavelength range (figure 4). The next step will be the same operation for the G1 guide.

- *New benders*. Few years ago, in order to increase the end guide positions, two benders have been installed using an old multichannel technique with Ni58 coating. Replacement of this old bender by a multi channel super mirror bender leads to an increase in the transmitted flux. In a first step this has been done for the G3 bender (figure 5). More recently a third bender using a super mirror coating has been installed on the G5 guide [1].

3. NEW COLD SOURCES.

a) Old design.

- In the original design of the reactor, the choice of liquid Hydrogen for the moderator [4] has led to use two small cells in order to feed enough cold neutron beams. Due to the geometry of the beam tubes looking at the cold sources (figure 1) it was decided to design a "flat" cell in order to fully illuminate the corresponding beam tubes. The cell size, made of stainless steel, is 205 mm height, 125 mm large and 50 mm thick.

- Boiling hydrogen enclosed in the two circuits of the cold source n°1 (beam tube 8F) and n°2 (beam tubes 9F and 4F) is re condensed by frigorific exchange with an helium refrigerator according a BRAYTON cycle. This cryogenerator delivers gaseous helium at 17°K by mean of two turbines (one on each circuit). That helium exchanges its frigories with gaseous hydrogen by the way of two condensers located in the pool of the reactor. Hydrogen circulation between the condenser and the cell is made by natural convection in a closed circuit under vacuum completely immersed in the pool.

- The thermal power produced by nuclear radiations in each "flat" cell during reactor operating is about 700 Watts. The cryogenerator designed in 1979 and built by AIR LIQUIDE was able to deliver 1400 Watts at 20°K to cool and liquefy hydrogen [5].

b) New design.

- The best suitable shape for the cell designed for ORPHEE PLUS, is a cylinder of 130 mm in diameter [6]. To reduce the neutron absorption this cylinder must be hollow (see figure 6). The thickness of the liquid Hydrogen layer was determined by calculations based on results of experimental studies done for the High Flux Reactor of Grenoble [7]. A value of 15 mm was determined for the best neutron gain below 7Å, the wavelength range used by the majority of the experimental set-up.

- The size of the new cell increases the Hydrogen volume and the thermal power balance of the old cryogenic plant was too low. It was then decided to design a new cryogenerator. The frame of the project was :
  - to keep unchanged the Hydrogen circuit,
  - to design a new cryogenerator able to deliver 1850 Watts at 17°K,
  - to modify as less as possible the existing installation, re-using all parts that can be, considering their design size,
  - to change all the parts whose technology was out of date and to replace them to permit an elaborated automatism.

- A new design study has been set up to verify the evolution capacity of each existing part of the old system. The conclusion of that study has shown :
  - it was necessary to replace the helium compressor and the helium turbine to reach the 1850 Watts,
  - the hydrogen condenser was able to cool and liquefy the hydrogen for the new annular cells,
  - the intermediate helium exchanger (cold box) was limited to 1700 Watts.
  - An increase of helium mass flow in the helium pipes between the compressors and the cold box do not bring significant increase of pressure drop.

- Considering these conclusions it was decided to study a new cryogenerator for a power of 1850 watts. The main characteristics are the following :
  - complete automatic system,
  - automatic start up after failure of electric power, air instrument, or water cooling system,
  - no regeneration of helium before restart,
  - possibility of outside control and telediagnostic,
  - operation through high feasibility components.

c) The new cryogenerator.

The different units of the new cryogenerator built by AIR LIQUIDE are the following :

- the compressor unit,
- the oil removal unit,
- the helium pressure control unit,
- the "cold-box" including heat exchangers, the absorber and only one expansion turbine,

- the helium temperature regulation unit,
- the control command unit.

- *Compressor unit.* The compressor unit is set with two screw compressors settled on the same frame including the bulk oil separator.

  The main characteristics of the screw compressors are :
  - admission pressure : 3.7 bar,
  - delivery pressure : 17.4 bar,
  - number of stage : 1,
  - mass flow rate : 60 g/s (for each compressor),
  - electric power : 125 kW,
  - water cooling flow rate : 40 m^3/h.

  The bulk oil separator is placed just after the compressors. The oil is send back to the compressor's bearings and casing through water cooling exchanger and filter.

- *The oil removal unit* (see figure 7*).* Leaving the bulk oil separator, pressurised helium (25 ppm of oil) goes to the oil removal unit. Aerosol of oil are kept into a serial of three filters. Small particles of oil are agglomerated along very small diameter fibbers. The drops of oil fall down by gravity and are collected at the lower part of the filter and back to the compressor. The very last quantity of oil vapour is absorbed on an active coal bed.

- *The helium pressure control unit.* This unit is settled near the compressor unit. The regulation valves ensure high pressure and low pressure regulation, specially when the compressors are starting.

- *The "cold-box"* (see figure 8). This box contains components running at low temperature; they must be thermally isolated, and are placed in a large cylinder under vacuum. The two heat exchangers (serial) are plate and fin type heat exchangers. They ensure temperature decrease of helium from 300°K to 23°K with a very low pressure drop (0.07 bar).

  The turbine with gas bearings is built by AIR LIQUIDE (see figure 9). High pressure helium at 17 bar and 20°K is expanded into the distributor. The expansion of the gas goes on in the wheel and the diffuser. The wheel drives the shaft supported by the gas bearings (pivoting system) which transfers a torque to the wheel of the brake compressor. The energy absorbed by the brake compressor is turned into heat by helium compression and drained out by a cooling water exchanger.

The helium flow cooled (15°K) and expanded (3.7 bar) goes to the hydrogen condensers.

- *The helium temperature regulation unit.* The hydrogen pressure in the cells is regulated by the temperature of cold helium in the condensers. As the helium temperature of gas going out of the turbine is constant, the gas is warmed by electrical resistance just before the condensers. The electrical power delivered to the helium flow represents the available frigorific reserve of the cryogenerator.

  Large variations of the reactor power during test period (from 14 MW to 4 MW at 1.2 MW/minute), has shown quick answer of the regulation and low variations of hydrogen pressure.

- *The control command unit.* The control command is ensured by a process controller supplied with electronics gauges, relative humidity control and vacuum gauges. The interfaces cards ensure connections between the central unit and the process. They receive directly signals from the gauges and send back orders to the components (valves, electric resistance, ...).

  The process informations are available on a display screen, with a keyboard. The cryogenerator is piloted from that keyboard. The regulation parameters values can be also changed from that system.

d) Normal operation.

The operator can choose among different ways of operation and gives simple :
- starting order,
- starting vacuum,
- forbidding of the turbine running (used during test periods),
- cleaning the cold box (the air staying in the cold box is evacuated and replaced by helium),
- cold box regeneration (the air staying in the absorbers and water set down in the exchanger are evacuated).

In case of emergency or utilities failure, the whole system is protected. All the valves set in the safe position.

If the electric power failure is less than 20 seconds, the helium turbine is fed up with gas coming from a safety capacity. High pressure is kept at 17 bar and low pressure helium is discharged to the atmosphere. When electric power is back, the first compressor starts ; the second one three seconds later and the valves recover their normal position. The turbine does not stop and the cryogenerator runs normally.

The cryogenerator re-vamping at the ORPHEE reactor is the first step of the modernisation programme of this reactor. It allows the increase the cryogenic power delivered to the cold sources and to get a modern device. Setting up the cryogenerator needed two months of reactor shutdown including cryogenic test and neutron tests. The next step of that programme will be the replacement of the flat cell of the two cold sources by new annular cells. This step is foreseen during the 1995 summer shut-down.

4. FUTURE PLANS.

- *New cold neutron guide hall.* This project, called ORPHEE-PLUS, consists in extracting new cold neutron beams by neutron guides exiting in a new neutron guide hall and increasing by about 30 per cent the number of beam positions. It was largely described in the previous IGORR meetings [1]. Technical designs are in progress.

  After a preliminary design, it has been decided to choose two straight neutron guides starting from the end of the beam tube up to external face of the reactor building. The crossing of the wall of this building will be done through safety valves. The beam section will be large, 160 mm height and 50 mm large, in order to easily split each in two secondary guides in the new guide hall (figure 12) leading to four guide ends. The coating will be a super mirror with $2\theta_c$.The biological shielding in the reactor hall is under calculation and will be made by concrete blocks.

  The design of the collimator is now completed. It has a converging part looking at the cold source (figure 10) and the guide separation is done in the in-pile collimator in order to minimise the shielding. In this part the support of the mirror will be made with an aluminium alloy finely polished and then coated with Nickel 58. Experiments are in progress to determine if it is possible to use super mirror coating in this area due to the high radiation level [3]. This collimator will be enclosed inside the plug set in the beam hole tube (figure 11) looking at the cold source n°2.

As previously presented [1], this project will lead to a large modification of the experimental set-up in the reactor hall. Seven of the present eleven instruments have to be moved. At that occasion large modifications of the beam tubes will be done.

The first one is to create a new cold neutron beam tube inside the reactor hall by extending the tube T1 up to the cold source N°1 set in front of the tube F8 (see figure1). Because this new beam tube is nearly at right angle with the beam tube F8, the cold cell must be cylindrical in order to fully illuminate the two beam tubes. The second modification concern the size of the in-pile collimators for thermal neutron beams. They will be increased from 40 x 80 mm^2 to 50 x 120 mm^2. This modification will results in a large increase on available flux on sample.

- *Thermal neutron guide*. The present development of the super mirror coating with large critical angle will allows to produce efficient thermal neutron guides. At present time coating with $2\theta_c$ super mirror is extensively used but is not really enough for a good thermal neutron guide. Experiment are in progress in order to determine the parameters for an industrial process. When good super mirrors will be available, the project is to extract thermal neutron beam from the beam hole 6T (see figure 13) and to use it in an external guide hall.

5. REPLACEMENT OF THE ZIRCALOY CORE HOUSING.

The core housing of the ORPHEE reactor serves the following purpose :

- containing fuel elements and ensuring their correct position,
- channelling water cooling flow through the core,
- keeping water lightness between primary light water and heavy water in the reactor tank.

Because of its position in the reactor, the Zircaloy of the core housing is placed under a huge neutronic irradiation. To get a good knowledge of Zircaloy behaviour under neutronic irradiation, a specific device has been settled in the middle of the reactor core, inside the beryllium reflector block. That device has permitted to follow the evolution of Zircaloy properties with anticipation as the fast neutron flux in the middle of the core is 20 % higher than in the most irradiated area of the core housing.

The different results obtained from 1986 and their extrapolation following a reasonable evolution process for more important doses, have shown that the maximal available growth of the core housing will be reached before the end of the lifetime foreseen for the reactor. That maximal value of the length has been calculated to keep normal stresses in the materials (nut and bolt, bending stresses of the reactor tank head). The maximal over length is 0.79 mm for the 1000 mm of the most irradiated area of the core housing. That over length could be reached during the year 1998 and it has been decided to replace the core housing at the end of the year 1996.

Replacing the core housing is an important operation whose duration has been estimated up to three months. The main phases are :

- emptying and drying the reactor tank from heavy water, then filing it with light water,
- dismantling of the whole mechanism of the control rods,
- taking out the positioning grid, the core grid and the convergent tube,
- dismantling of the upper bolts of the core housing, then of the lower bolts (fixation of the lower flange on the bottom of the reactor tank),
- carrying out the core housing to the storage pool,
- settling the new core housing in the reactor tank,
- leakage test,
- emptying and drying the reactor tank from light water.

The design of the new core housing has been modified to take account the growth of Zircaloy under irradiation. A dilatation compensator made of two concentric bellows with leak detection between the two bellows have been put at the upper part of the core housing (see figure 15).

6 CONCLUSION.

The modernisation programme of the ORPHEE reactor and the experimental devices defined some years ago is in progress. The new fact is the success in the industrial process of the super mirror coating leading to an extensive use for the replacement of old Nickel coating. This is done in places where it will bring a very efficient use of neutron beams : in curved part in order to lower the characteristic wavelength, in bender in order to increase the transmitted flux. The creation of new cold neutron guides of the ORPHEE PLUS project has been slowed during the shut down of the Grenoble High Flux Reactor. It is now in the design stage and the decision to start the construction is linked to the collaboration with foreign laboratories.

Together with the restart of the HFR, the present refurbishment of the experimental facilities of the ORPHEE reactor and the possibility to increase the cold neutron beams potential will give to the European scientific community the opportunity to access neutron beam facility for the next decades.

FIGURE CAPTIONS.

Figure 1.	Horizontal section of the reactor. The nine beam tubes are referenced by the nature of the neutron beam: T thermal, F cold, C hot.
Figure 2.	Instrument layout in the two experimental halls: reactor hall and neutron guide hall. 30 instruments are installed around the reactor, 11 in the reactor hall and 19 in the neutron guide hall. 21 are using cold neutron beams (i.e. 70 %). A neutronography installation set at the end of the G4 guide is used for industrial tests.
Figure 3.	Measured reflectivity of natural Nickel mirror (critical angle θ_c) and super mirror with a critical angle $2\theta_c$.
Figure 4.	Measured neutron flux for the new G2 guide.
Figure 5.	A) Sketch of the G3 and G3 bis guides. W is the width of the elementary channel. B) Measured neutron flux for the G3 bender with Nickel 58 and with super mirror.
Figure 6.	Schematic view of the annular cold cell.
Figure 7.	Compression system of the cryogenerator.
Figure 8.	General diagram of the cryogenerator.
Figure 9.	Cryogenic expansion turbine.
Figure 10.	Design of the beam tube for the two guides of ORPHEE PLUS.
Figure 11.	Design of the beam tube for the two cold neutron guides.
Figure 12.	New neutron guide hall and the splitting of the two guides.
Figure 13.	Neutron guide halls proposed for ORPHEE: ① present hall, ② new cold neutron guide hall, ③ proposed thermal neutron guide hall.
Figure 14.	Vertical section of the ORPHEE reactor showing the position of the zircalloy core tube.
Figure 15.	New zircalloy core tube with the compensator bellow.

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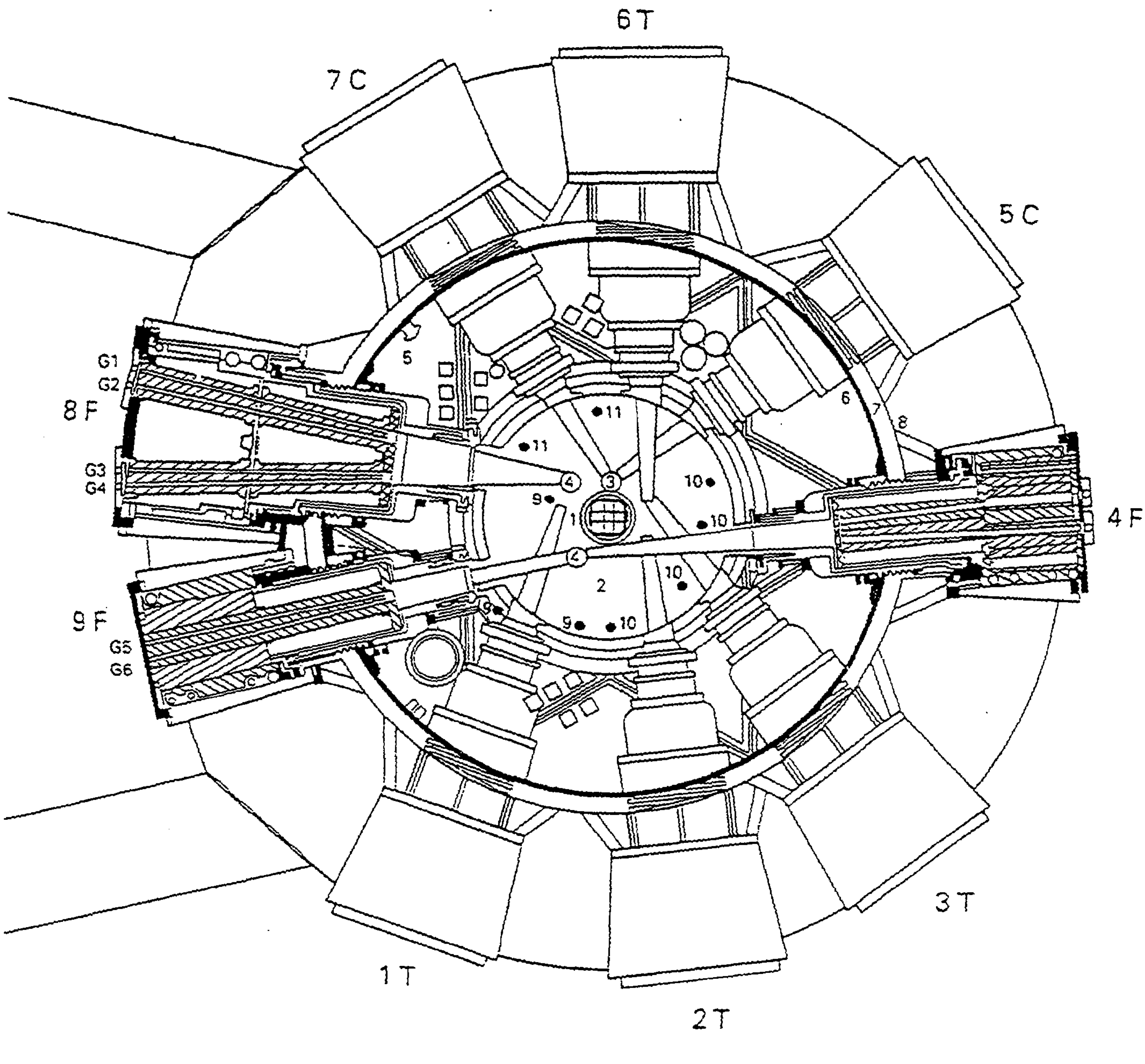


1. core
2. heavy water reflector
3. high-temperature source
4. low-temperature source
5. pool
6. pool inner wall
7. annular space
8. pool outer wall
9. radio-isotope production channel
10. shuttle tube
11. vertical irradiation channel

Figure 1.

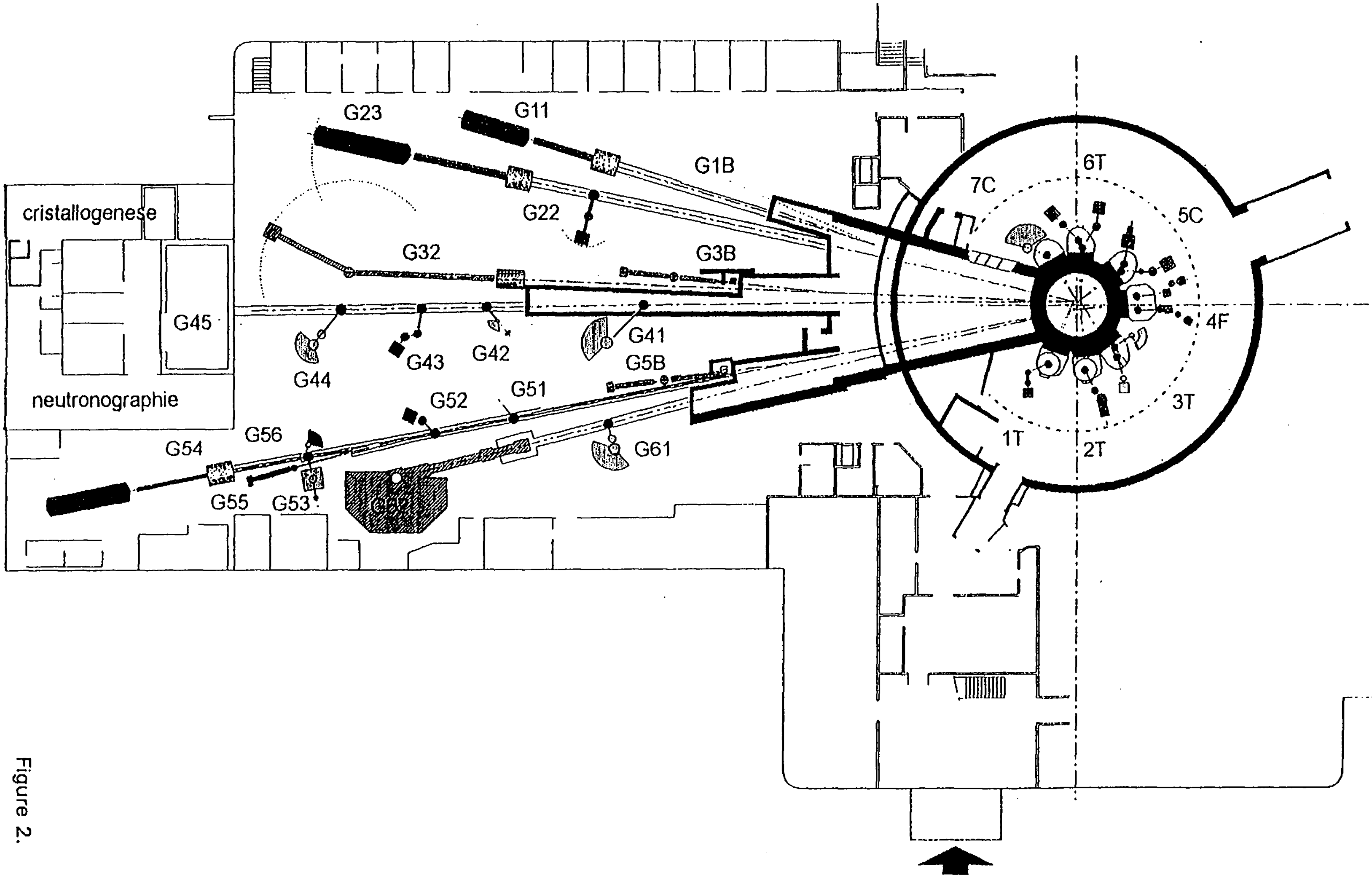
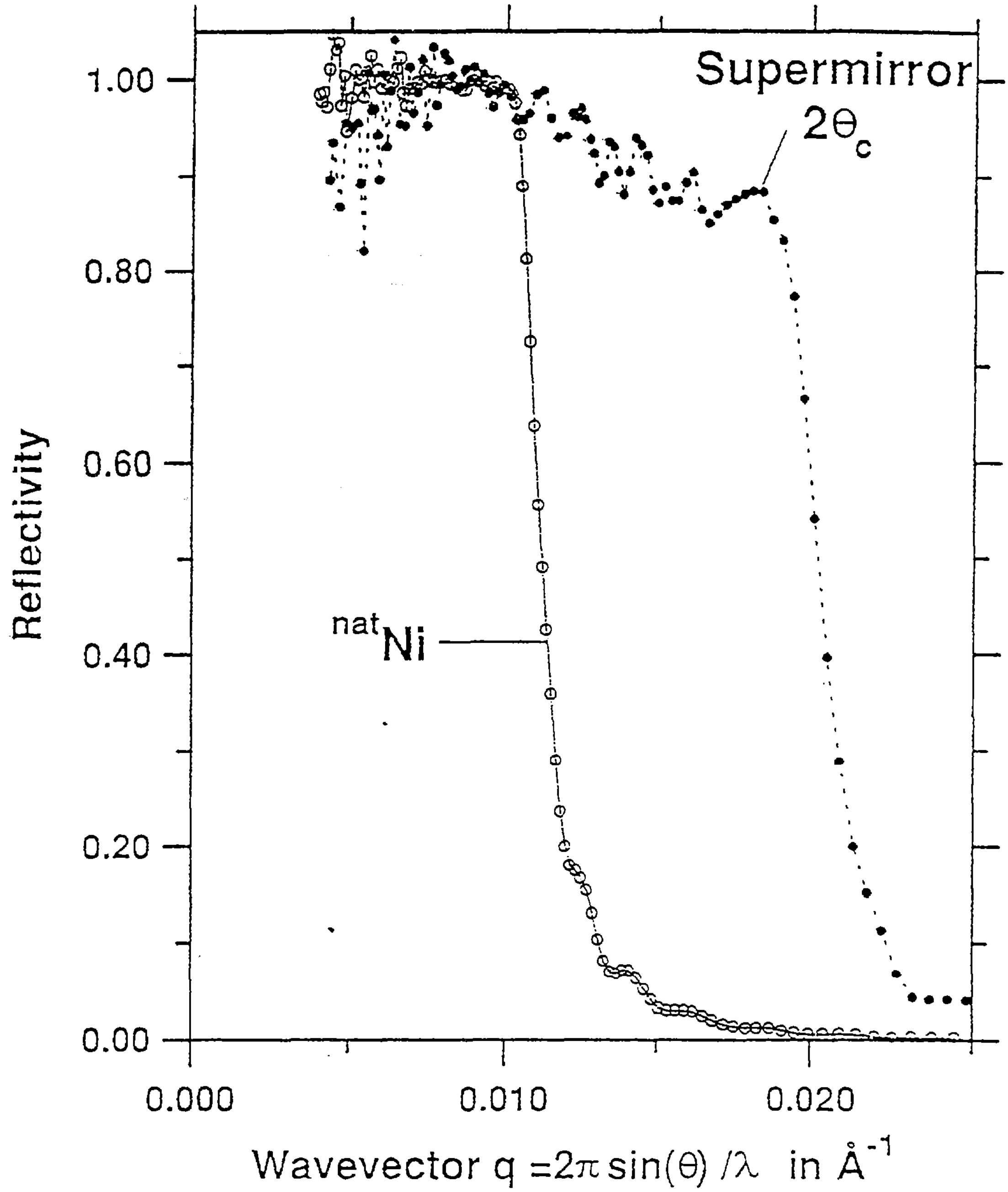
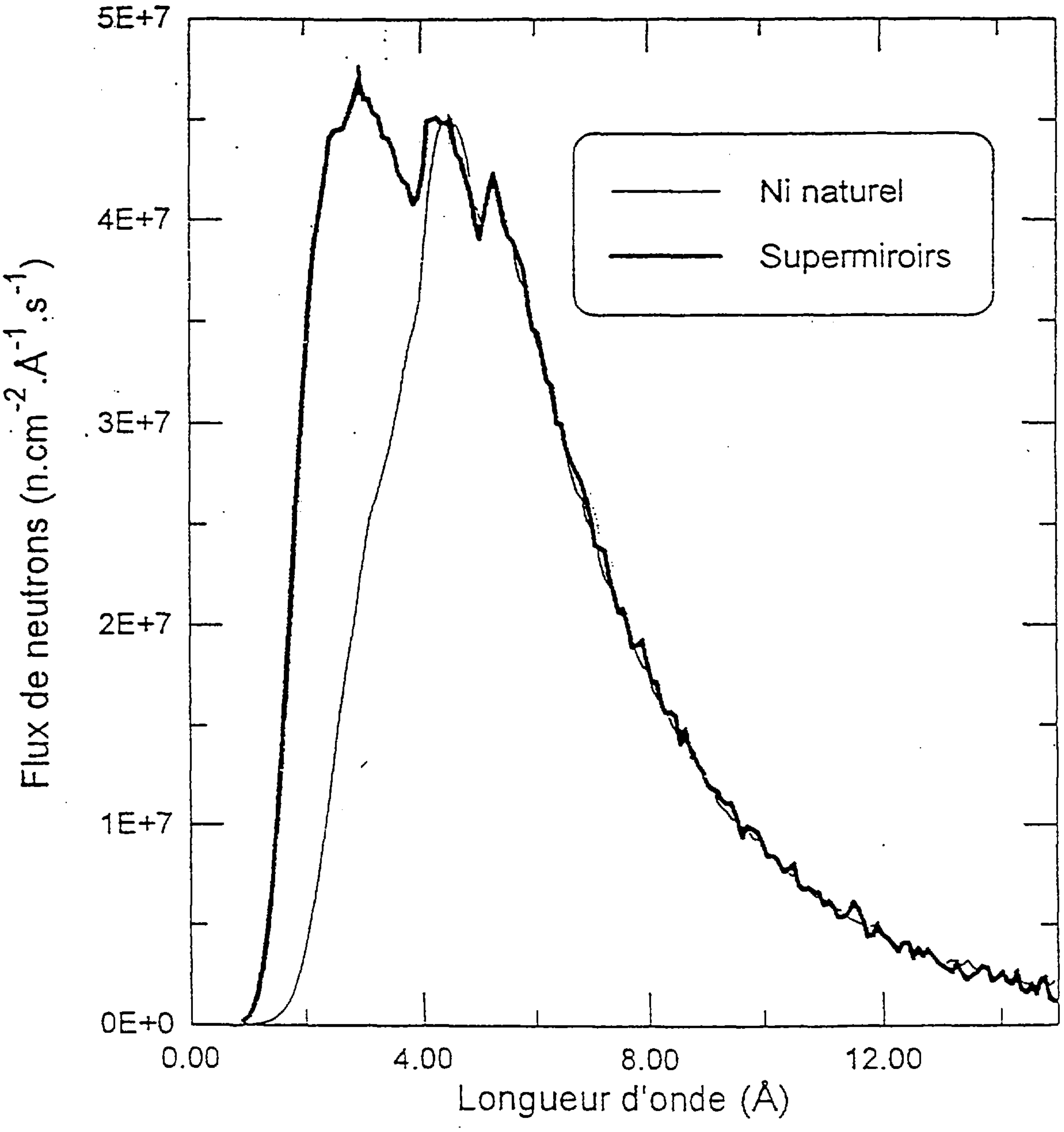

G23
G11
G1B
6T
7C
5C
cristallogenese
G22
G32
G3B
G45
G41
4F
G44
G43
G42
G5B
neutronographie
G52
G51
3T
G56
G54
G61
1T
2T
G55
G53
G62

Figure 2.



Supermirror
$2\theta_c$
^{nat}Ni
Reflectivity
1.00
0.80
0.60
0.40
0.20
0.00
0.000
0.010
0.020
Wavevector $q = 2\pi \sin(\theta) / \lambda$ in Å^{-1}

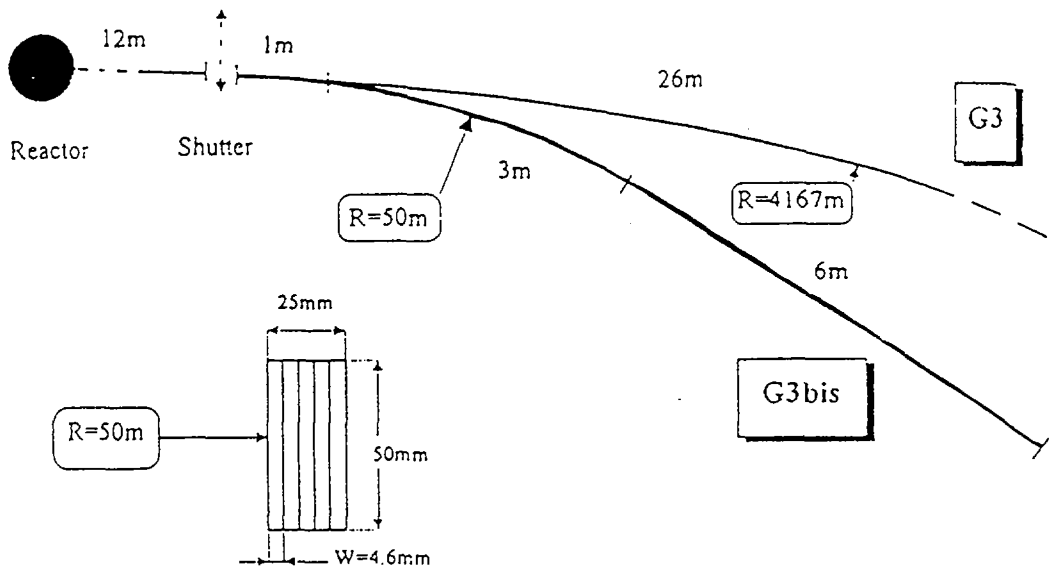
Figure 3.



Measured neutron flux for the new G2 guide.

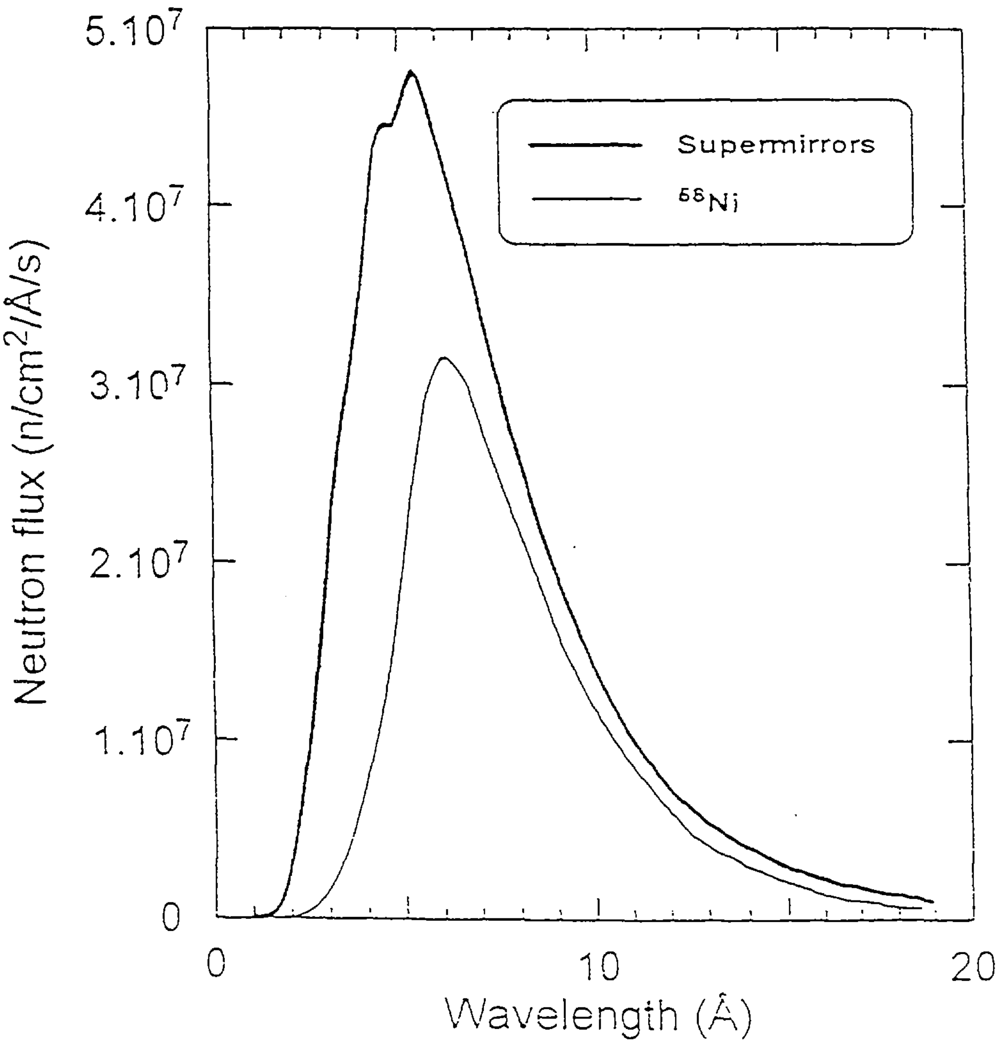
Figure 4.

A



12m
1m
26m
G3
Reactor
Shutter
3m
R=50m
R=4167m
6m
25mm
G3bis
R=50m
50mm
W=4.6mm

B



Supermirrors
58Ni
Neutron flux (n/cm2/Å/s)
5.10^7
4.10^7
3.10^7
2.10^7
1.10^7
0
0
10
20
Wavelength (Å)

Figure 5.

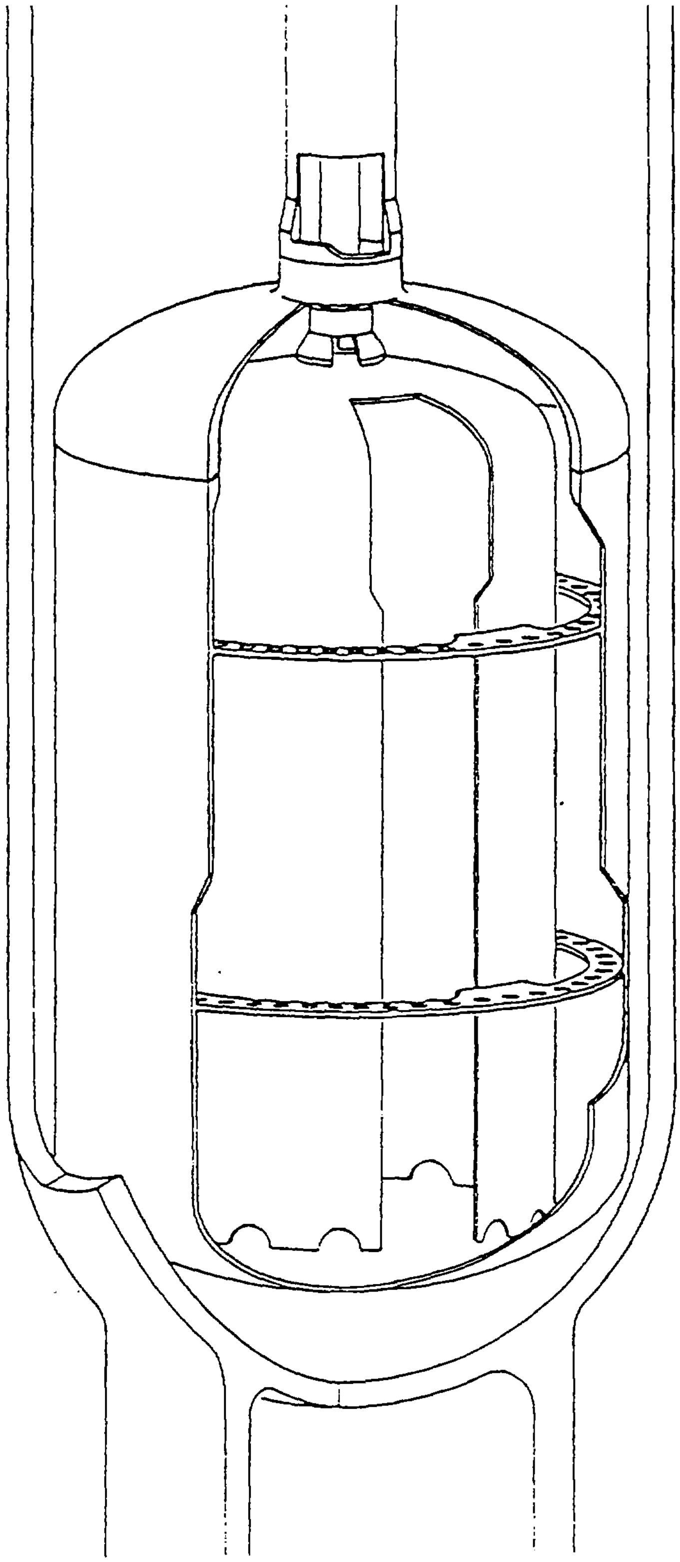


Figure 6.

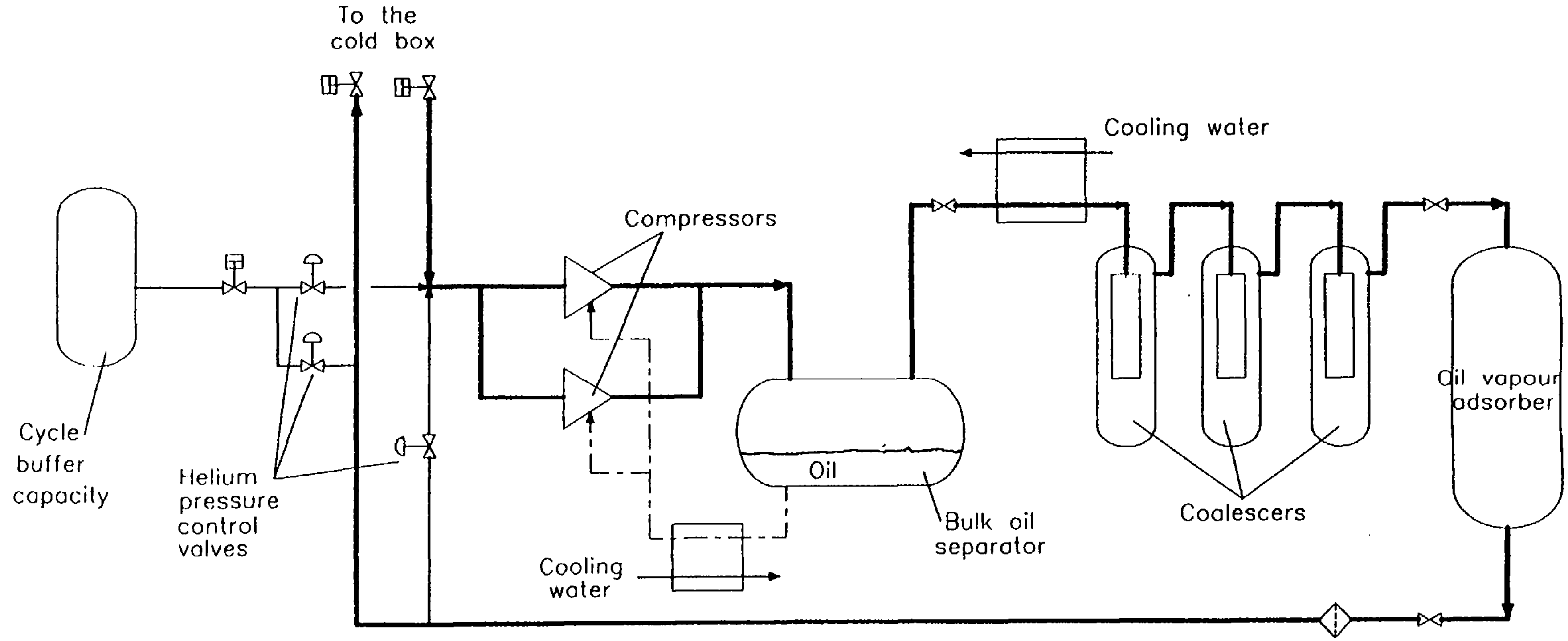

To the cold box
Cooling water
Compressors
Cycle buffer capacity
Helium pressure control valves
Oil
Bulk oil separator
Coalescers
Oil vapour adsorber
Cooling water

Figure 7.

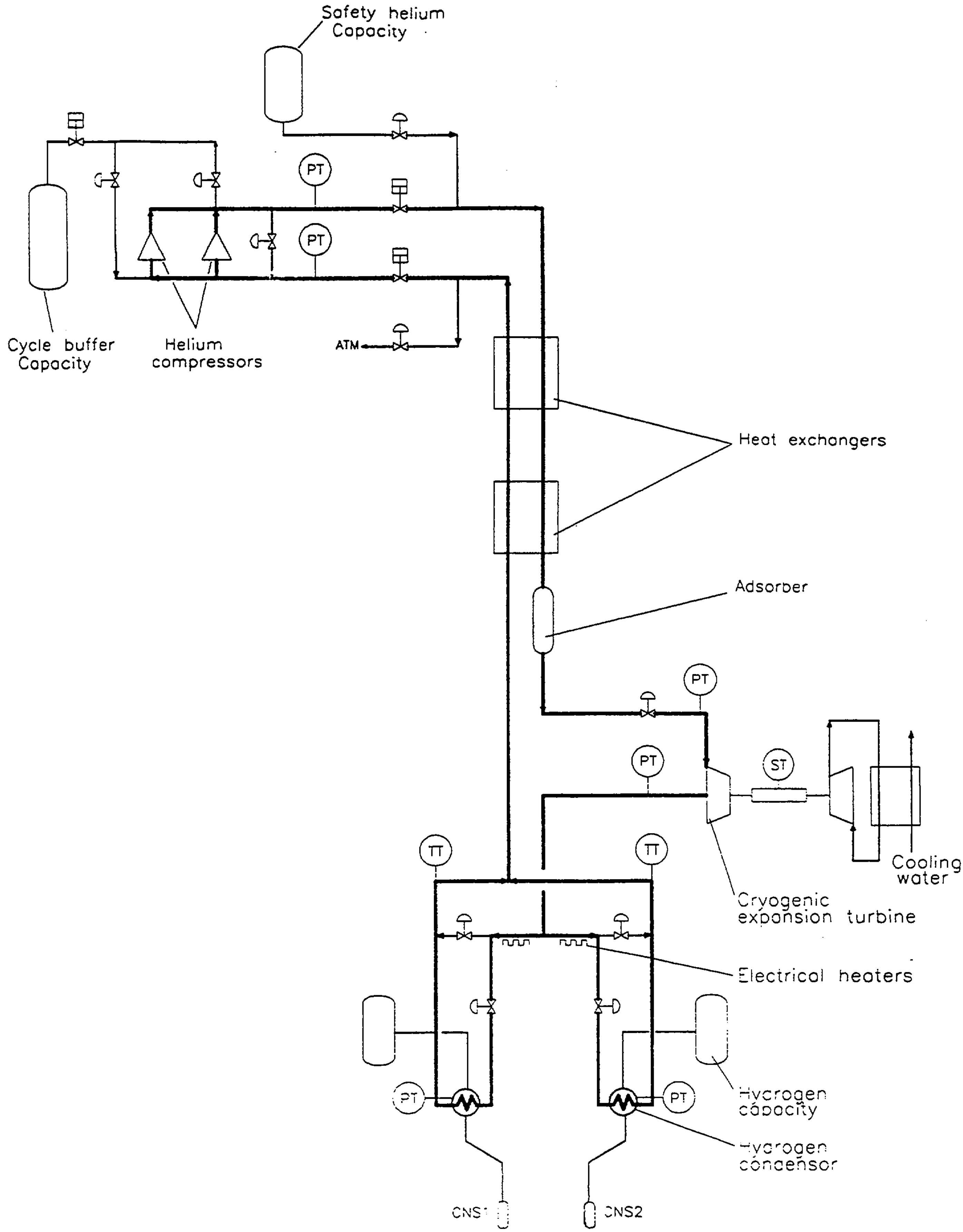

Safety helium
Capacity
PT
PT
Cycle buffer
Capacity
Helium
compressors
ATM
Heat exchangers
Adsorber
PT
PT
ST
TT
TT
Cooling
water
Cryogenic
expansion turbine
Electrical heaters
PT
PT
Hydrogen
capacity
Hydrogen
condensor
CNS1
CNS2

Figure 8.

CRYOGENIC EXPANSION TURBINE

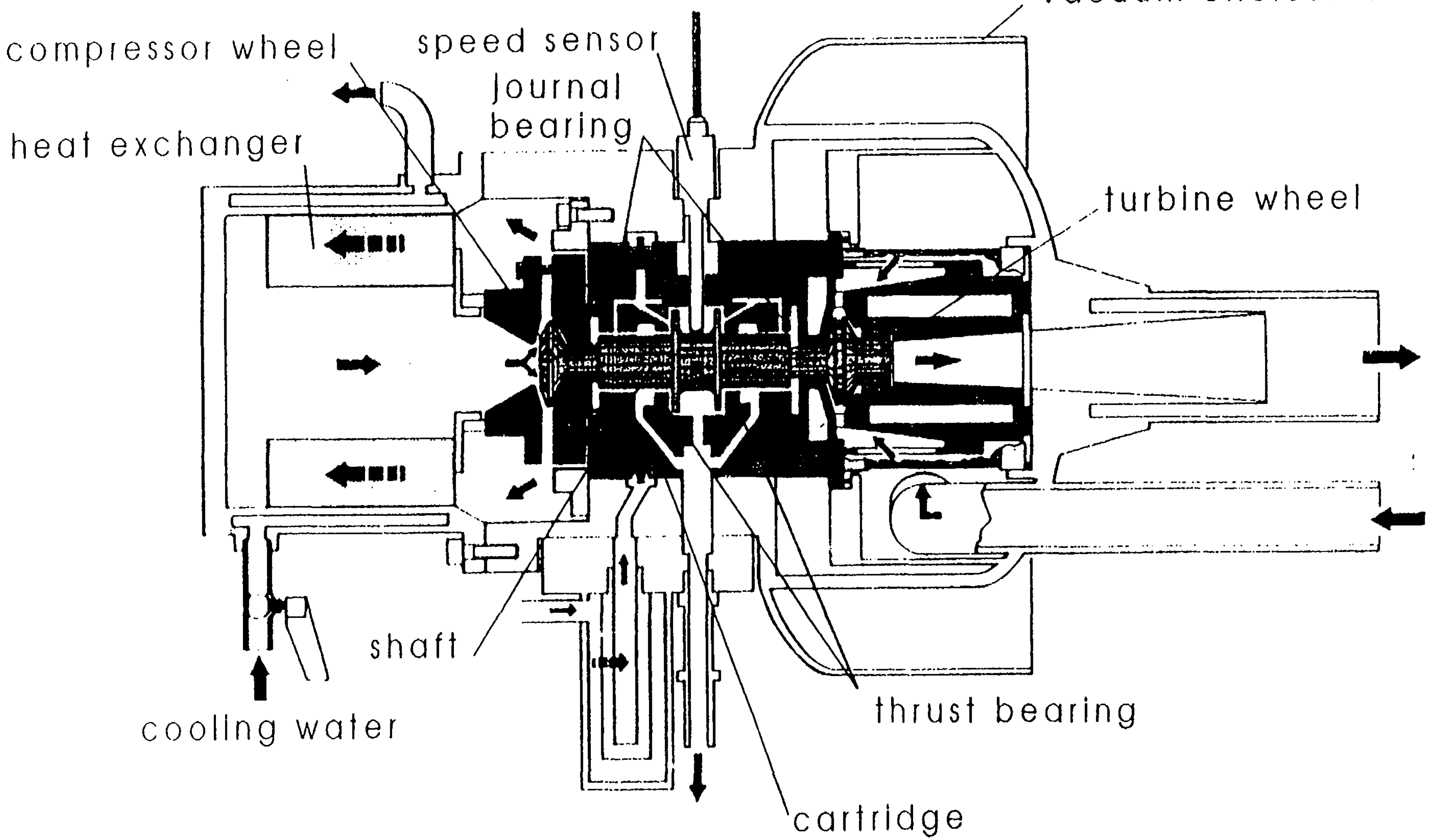


Figure 9.

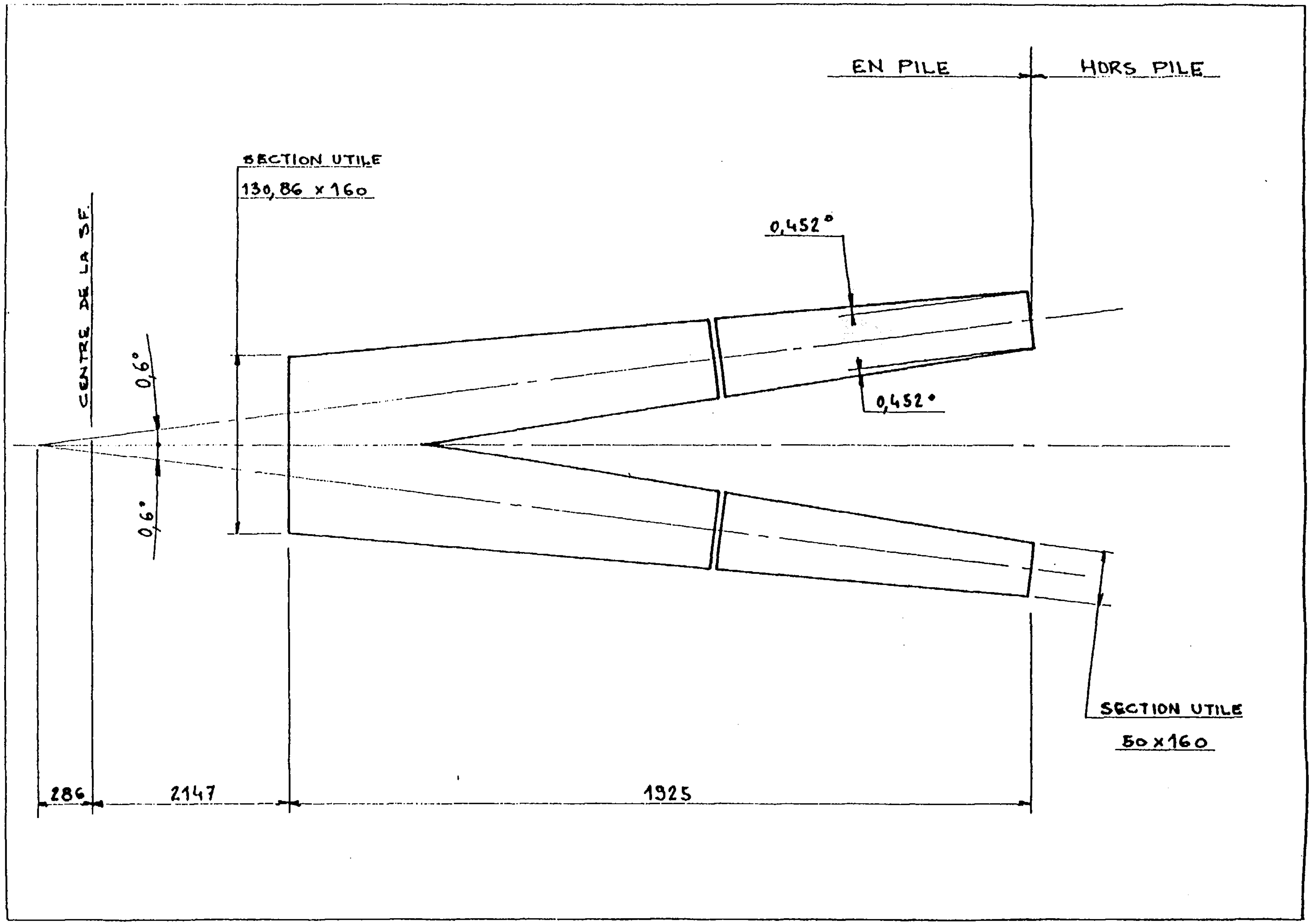

EN PILE
HORS PILE
SECTION UTILE
130,86 x 160
CENTRE DE LA SF
0,452°
0,452°
0,6°
0,6°
SECTION UTILE
50 x 160
286
2147
1925

Figure 10

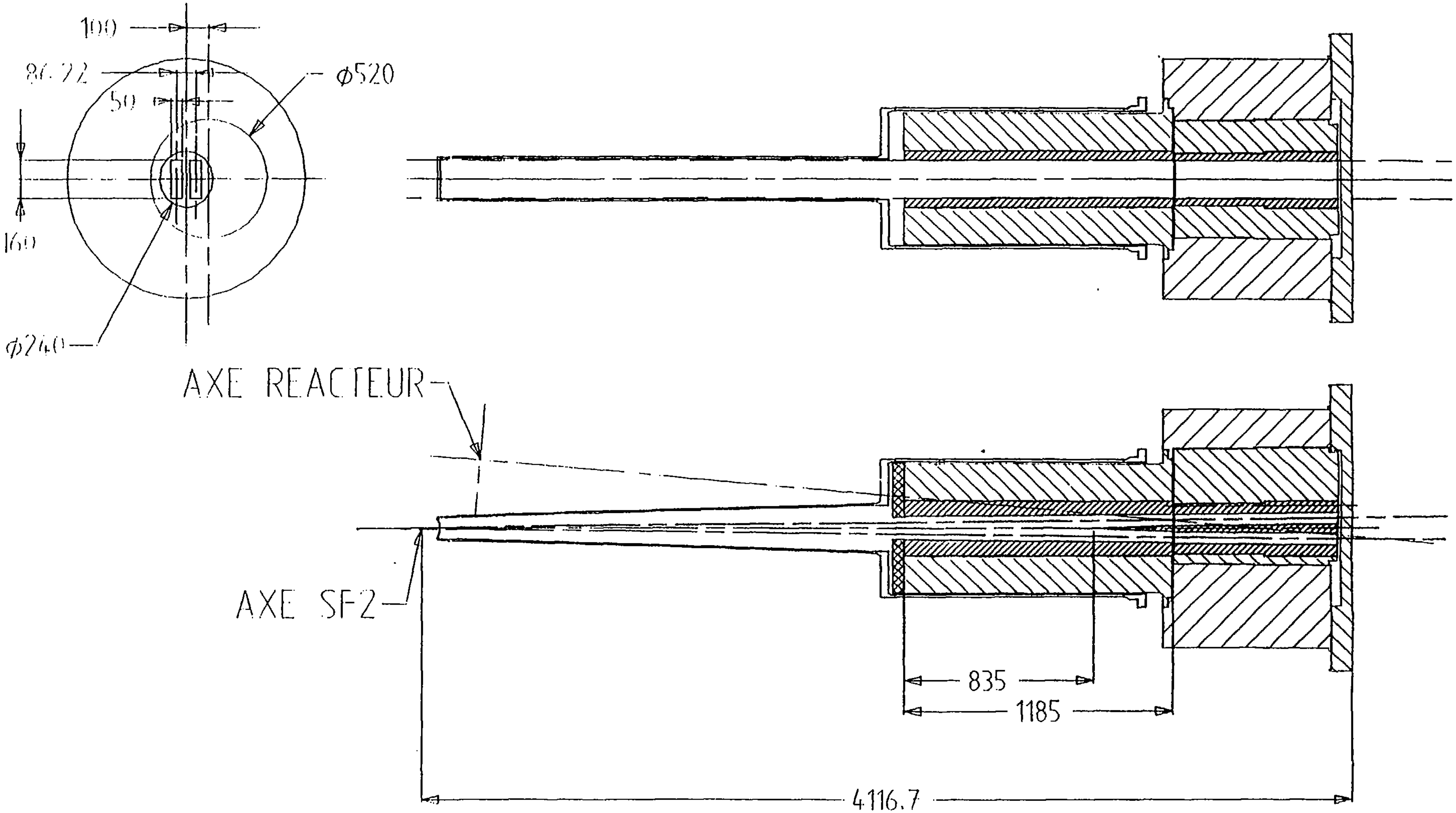


Figure 11

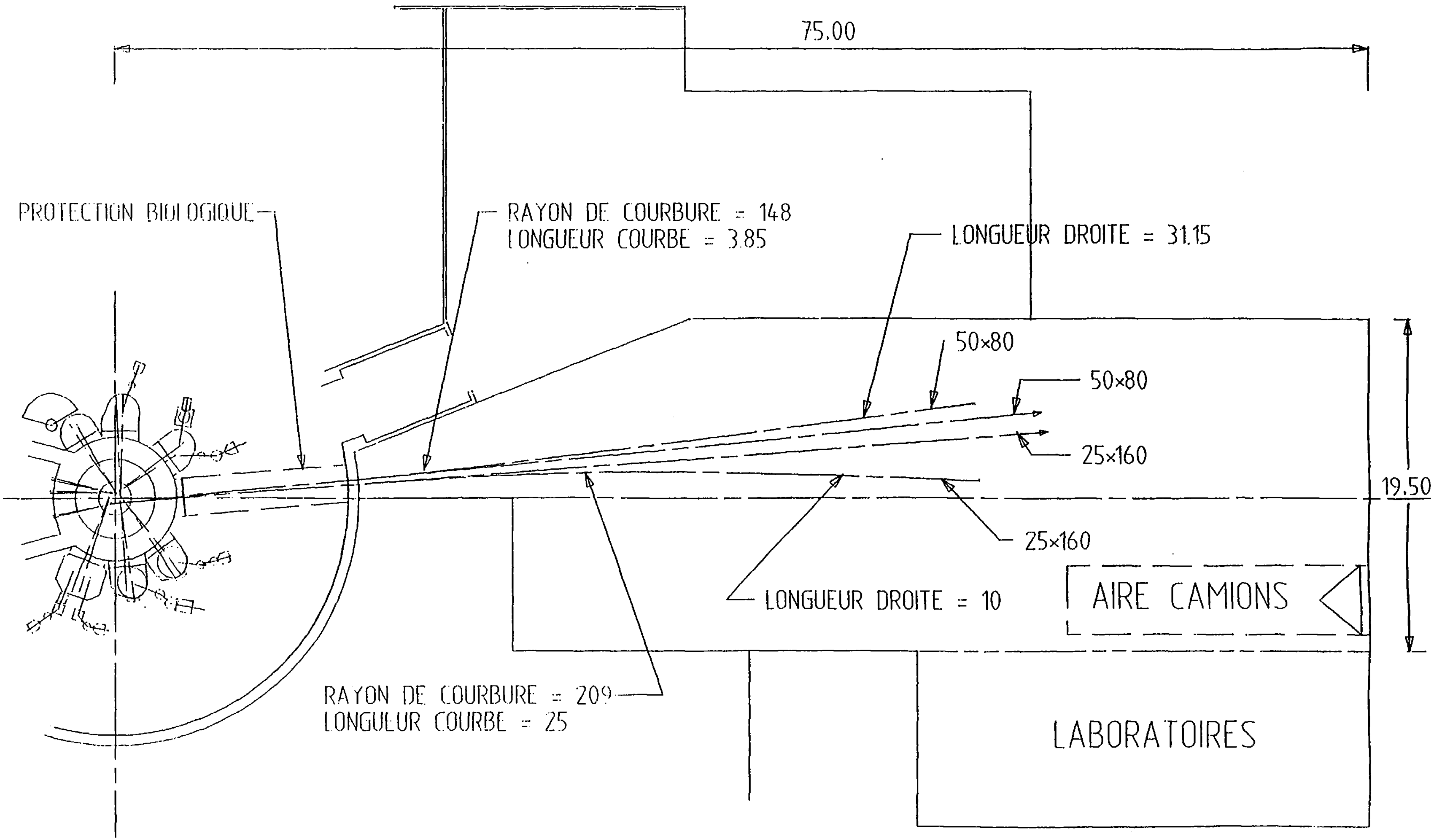

75.00
PROTECTION BIOLOGIQUE
RAYON DE COURBURE = 148
LONGUEUR COURBE = 3.85
LONGUEUR DROITE = 31.15
50×80
50×80
25×160
25×160
19.50
LONGUEUR DROITE = 10
AIRE CAMIONS
RAYON DE COURBURE = 209
LONGULUR COURBE = 25
LABORATOIRES

Figure 12.

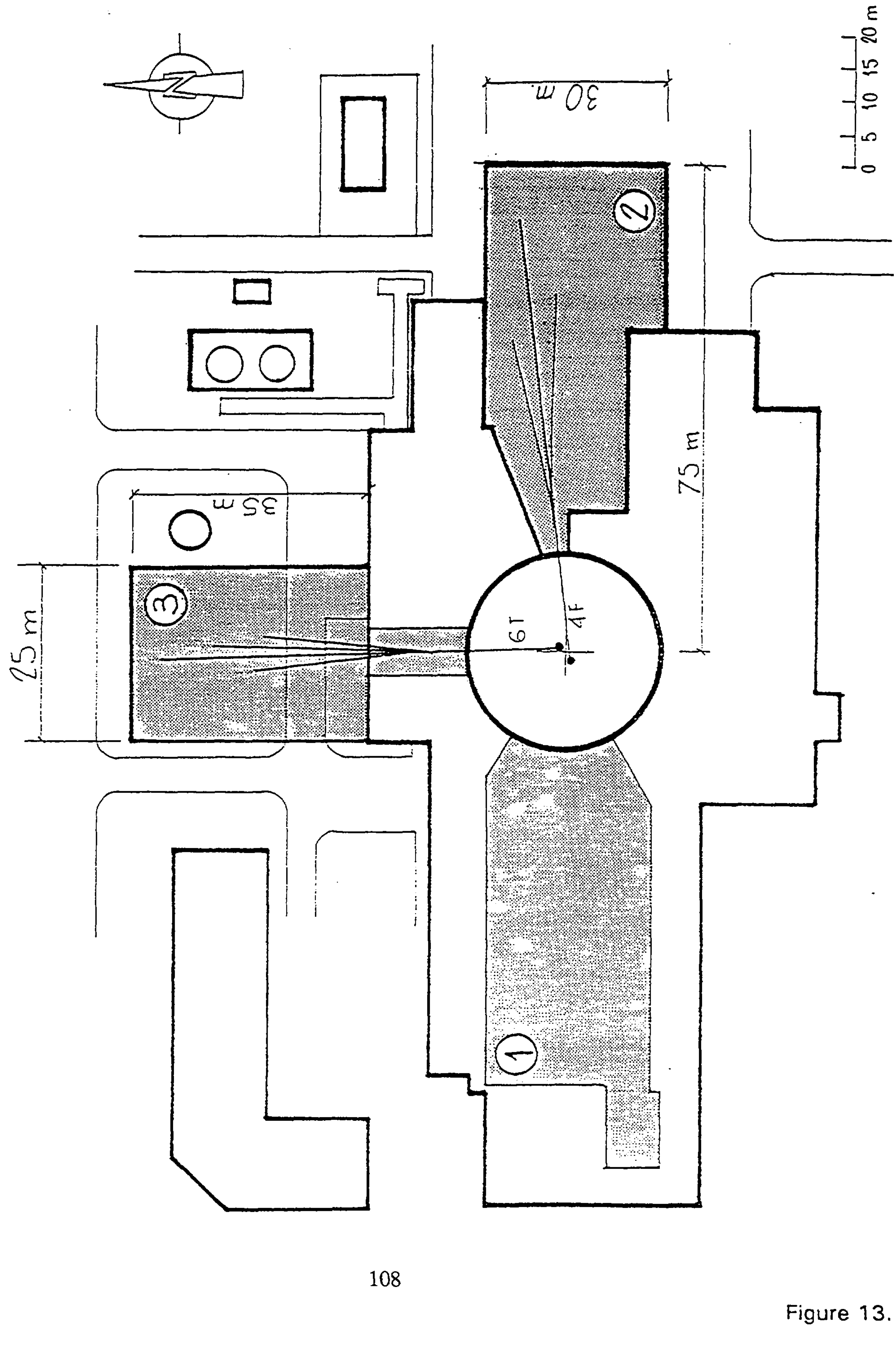
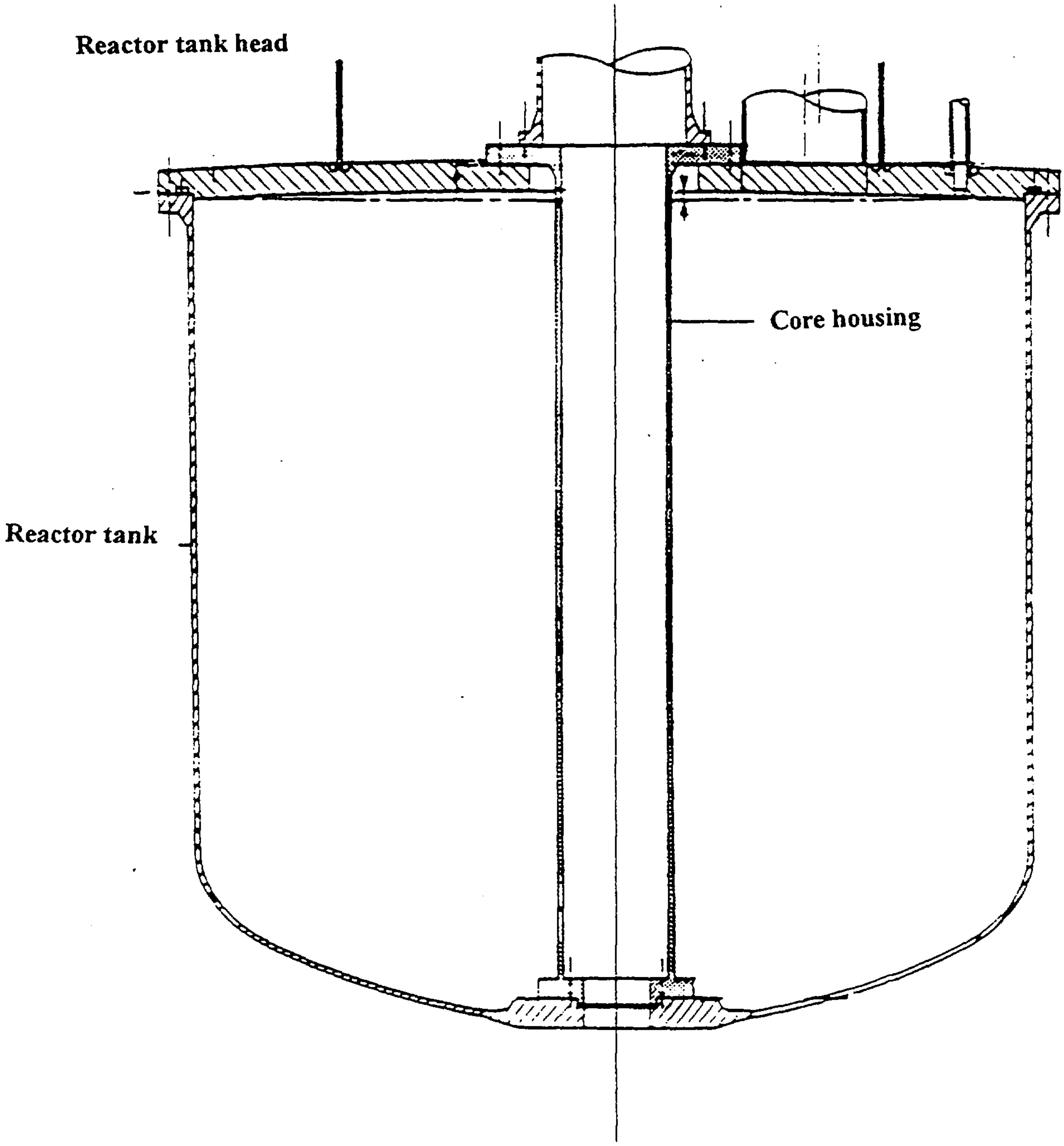


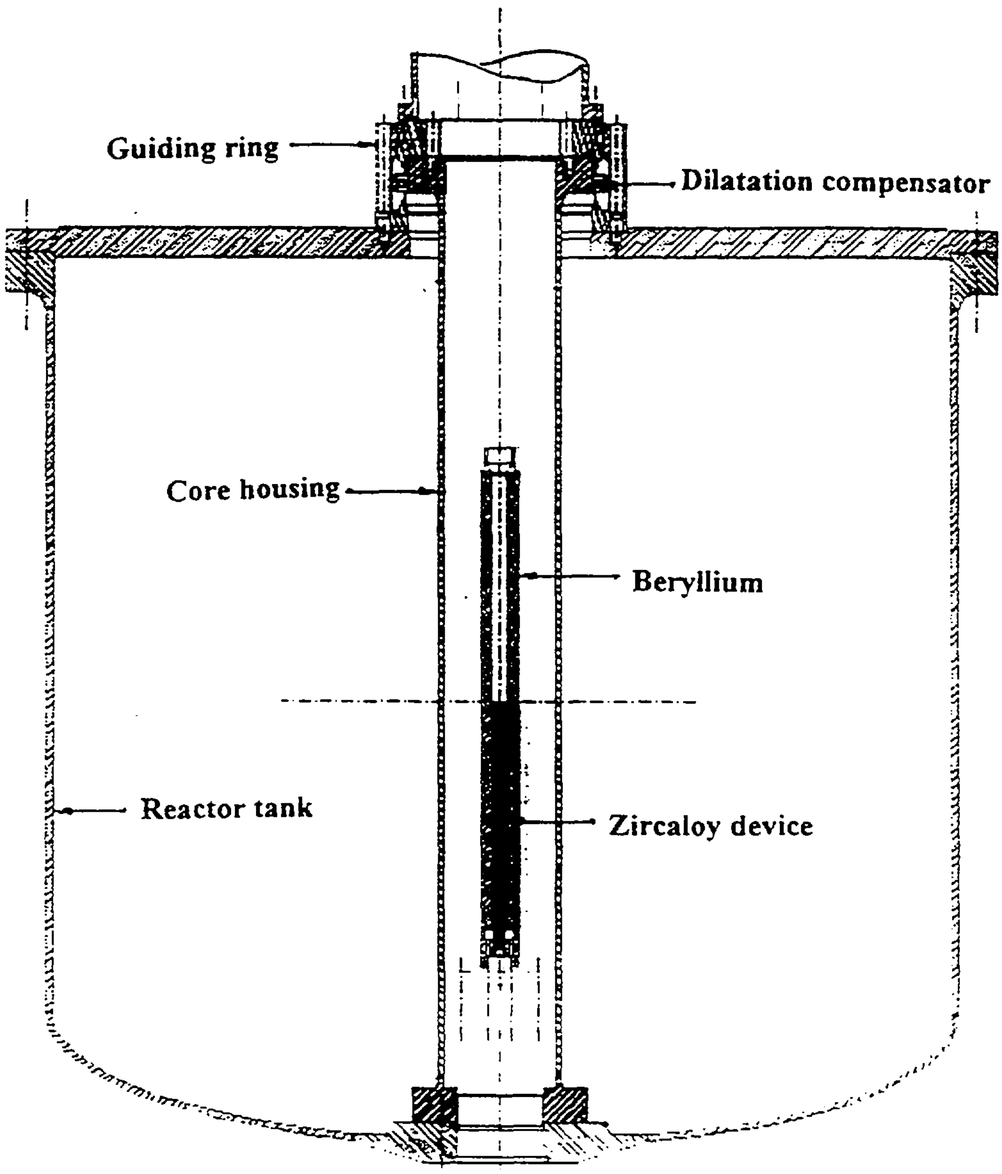
Figure 13.



Vertical section of the ORPHEE reactor tank

Initial assembly of the core housing

Figure 14.



Vertical section of the ORPHEE reactor tank
New assembly of the core housing with dilatation compensator

Figure 15.

XA04C1684

4th Meeting of the International Group on Research Reactors (IGORR-IV), Gatlinburg, Tennessee, USA, May 24-25,1995

STATUS OF THE FRM-II PROJECT

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ABSTRACT

The concept planning work for the new 20 MW high flux neutron source FRM-II to be built at Garching near Munich has been completed. The Siemens company has been selected to be the general contractor for the facility, whereas the Technical University of Munich as the future operator of the facility also acts as the responsible designer and supplier of the experimental installations. - The design concept has now successfully been examined by the responsible authorities and their experts.This will allow to start construction of the reactor building at about the end of 1995 when the first licensing step will be granted. Design work to obtain the next licensing award is underway. This second step will cover the erection of all the other buildings and the installation of all systems and components and is expected for late 1996. Cold startup is scheduled in 2000 and full power operation in 2001.

INTRODUCTION

The existing 4 MW swimming pool type research reactor FRM, which is in operation since 1957 on the research campus at Garching near München, will be replaced by a new high performance multipurpose neutron source named FRM-II (Forschungsreaktor München II, i.e. Research Reactor Munich II). The design of the 20 MW FRM-II pool reactor with moderator tank has been optimized for beam tube applications of thermal neutrons including the generation of slow, hot or fast neutrons for special fields of research, but it also will contain several state-of-the-art irradiation facilities.

FEATURES OF THE FRM-II DESIGN

The design concept of the FRM-II facility has already been shown on the previous IGORR Meetings /1,2,3/. The essential design feature is a very compact reactor core consisting of a single fuel element which is cooled by light water and surrounded by a large heavy water moderator tank. Fig. 1 shows a photograph of a dummy FRM-II fuel element which is being fabricated by the CERCA company /4/. Because of the small volume of this fuel element the use of highly enriched uranium in combination with high density silicide fuel is imperative. Since this compact core is highly undermoderated, about 50 % of the fast fission neutrons immediately leak out into the moderator tank where they are slowed down to yield a high flux level and a pure spectrum of thermal neutrons in a large usable volume. The ratio of flux to power is particularly high for this "compact core" reactor, the unperturbed thermal flux maximum in the D_2O tank being $8 \cdot 10^{14}$ $cm^{-2}s^{-1}$ at 20 MW power. The cycle length will be about 50 full power days.

Fig. 2 shows a vertical cross section of the reactor building. The reactor pool is placed in the center of the building. Horizontal and inclined beam tubes (not shown) lead into the adjacent experimental hall to provide neutrons for about 17 experiments. Further to the rear end of the reactor pool follows the storage pool which is used for the decay of spent fuel elements and other radioactive equipment. The reactor hall on top is mainly used for reactor and fuelling services, but it also provides access to various irradiation facilities located in the reactor pool or moderator tank. Siderooms on either side of the reactor hall will accomodate components belonging to the experimental installations, the reactor ventilation and reactor safety systems. The basement rooms will be used for cleaning systems, intermediate waste storage, light and heavy water handling and other auxiliary systems. On the right side of the reactor building the combined entrance and energy supply building becomes visible whereas the neutron guide hall which allows to perform more experiments is located in the background.

With respect to earlier designs the outer wall thickness of the reactor building has been enlarged and the form of the roof has been modified in order to provide full protection against a hypothetical air plane crash accident. One also can notice small gaps and floating connections between the pool shielding walls and the adjacent floor structures. This feature is to isolate the pool from the shocks of the postulated airplane crash, thus guaranteeing the integrity of the pool. Therefore even under such highly unlikely circumstances the fuel element would still be covered with water.

The FRM-II will be equipped with a number of integrated experimental facilities:

10 horizontal beam tubes partially in combination with neutron guides
2 inclined beam tubes
1 cold neutron source (liquid deuterium) with vertical guide tube
1 hot neutron source (graphite)
1 converter facility (uranium) for fast neutron production
2 silicon doping devices (nuclear transmutation)
2 pneumatic systems for sample irradiations
1 high flux pneumatic system for sample irradiations
1 capsule facility for longterm irradiations
1 positron source.

STATUS OF THE PROJECT

The public hearing act based on the final version of the Nuclear Safety Report took place during five days of May 1994. Questions from citizens and institutions concerning safety-related aspects of the project were answered by the licensing authority assisted by the Technical University of Munich (TUM) and the Siemens company.

In June 1994 a general supplier contract was signed with Siemens, covering all detailed engineering and necessary license-related activities, complete construction of the facility, cold and nuclear startup as well as a fifty days full-power test run of the reactor. This contract will become valid once the first nuclear licensing permit has been awarded by the authority, which is expected for November 1995. The TUM, however, as the future operator of the facility has to plan, supply and obtain licenses for all the experimental installations as mentioned above.

To obtain this first partial license (i.e. approval of concept and start of construction) in November 1995 all necessary informations were prepared and given to the technical surveillance company "TÜV Bayern Sachsen", the independent assessor as nominated by the licensing authority, the Bavarian Ministry of Environment. The TÜV recently completed the evaluation and presented the draft of its assessment to the authority. Now another positive statement is expected from the Reactor Safety Commission (RSK), the assessor of the German Federal Ministry of Environment.

Based on this status a report has been prepared for the German Scientific Council which gives recommandations on the application of funds for science-related projects in the university sector. A positive vote of this council would finally confirm the funding of the whole project FRM-II.

Parallel to the nuclear licensing there are further licensing procedures in progress, such as for hydrological and environmental aspects. Also for those procedures public hearings have to be performed.

The TUM and Siemens are now preparing all the detailed informations necessary for the construction of the remainder of new buildings and for the supply and installation of all components of the facility.Those papers will again be examined by the TÜV and other independent experts of the authorities and are expected to lead to the award of a second licensing step towards the end of 1996 which covers all work before the insertion of the first nuclear fuel element.

During the course of the year 1995 preparational work has to be carried out on the campus of the existing research reactor FRM in order to clear for structural interfaces between the FRM and its adjacent facilities and the site requirements of the new FRM-II.

The cold startup of the plant is scheduled for 2000 and its first criticality for 2001.

ACKNOWLEDGEMENTS

This paper represents a summarizing report on a project which many colleagues and co-workers from various institutions have contributed to. These include numerous members of the Faculty of Physics E21 of the Technical University of Munich as well as of the Siemens AG (previously Interatom GmbH) company.

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/4/ G . Harbonnier and J.P. Durand, CERCA; see contribution to this meeting (IGORR-4).

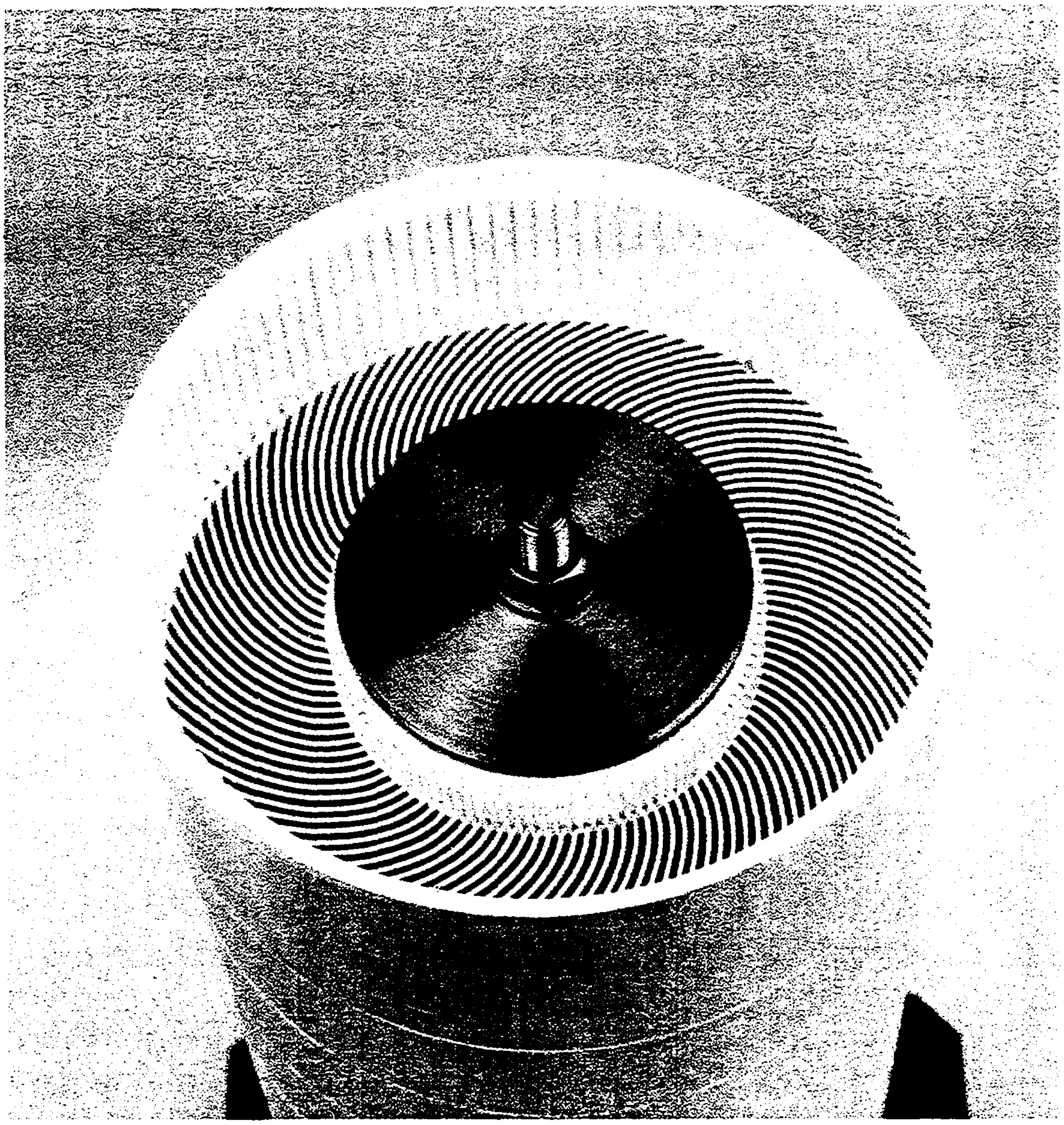

Fig. 1: A dummy FRM-II fuel element presently under construction at the CERCA company. The outer and inner diameters of the two core tubes are 243 and 118 mm, respectively. The 113 fuel plates have involute curvature, the axial dimension of the fuel zone being 700 mm and its volume 17.6 liters.

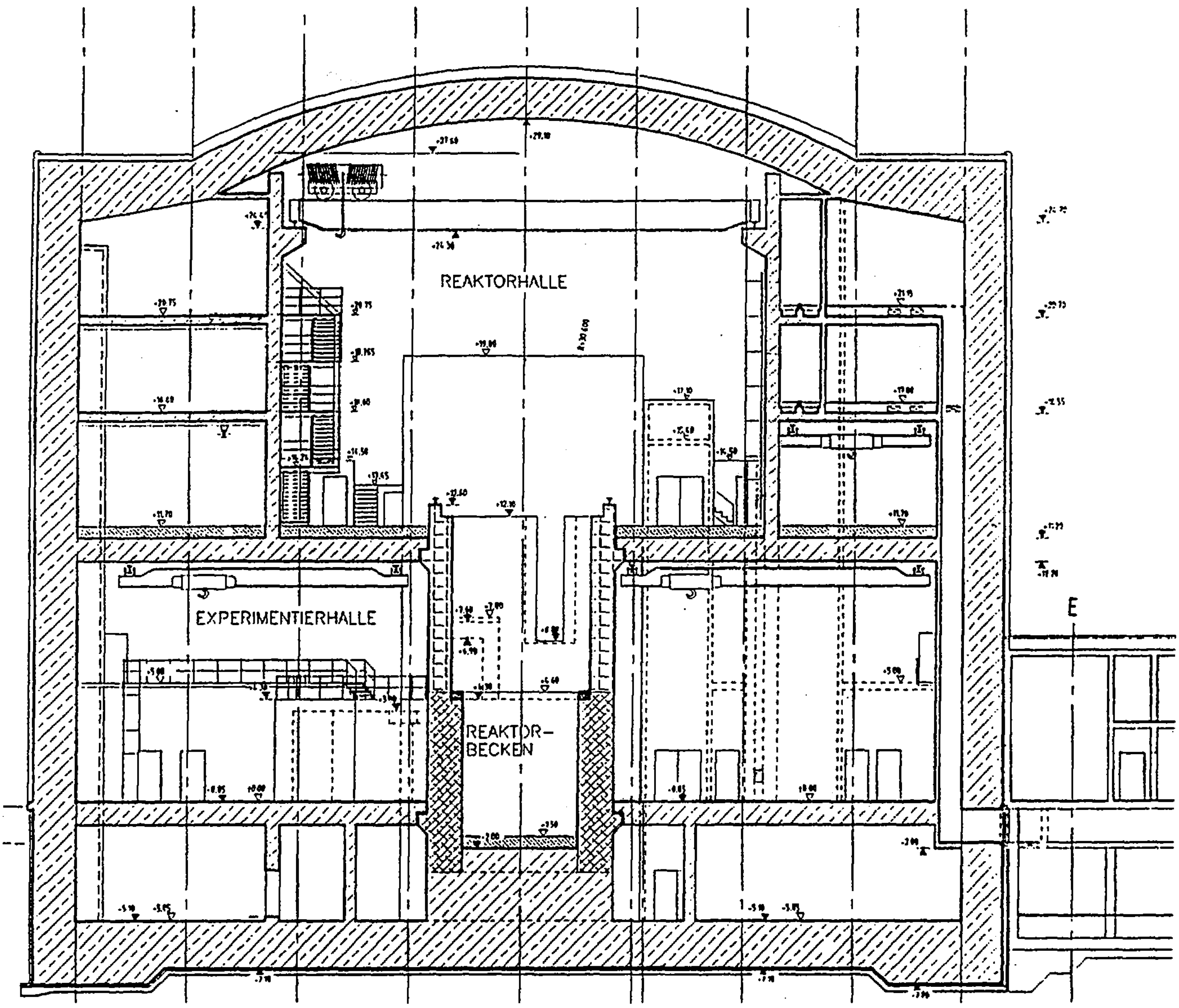

Fig. 2: Cross sectional view of the FRM-II reactor building. The thickness of the outer walls and the form of the roof have been modified to withstand a postulated airplane crash. The pool in the center has been mechanically isolated from the building structures by gaps and floating supports to minimize crash shocks.

DEVELOPMENT OF THE NEW CANADIAN IRRADIATION-RESEARCH FACILITY

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ABSTRACT

To replace the aging NRU reactor, AECL has developed the concept for a dual-purpose national Irradiation Research Facility (IRF) that tests fuel and materials for CANDU® (Canada Deuterium Uranium) reactors and performs materials research using extracted neutron beams. The IRF includes a MAPLE reactor in a containment building, experimental facilities, and support facilities. The reactor concept was developed to provide a realistic environment for irradiating up to nine natural- or enriched-uranium CANDU bundles at powers up to 1 MW_{e}, to generate fast-neutron fluxes up to 1.4×10^{18} $n{\cdot}m^{-2}{\cdot}s^{-1}$ in materials-damage and corrosion specimens, and to match the thermal-neutron fluxes available in NRU for a set of eight thermal beam tubes plus two cold sources equipped with neutron guides.

1. INTRODUCTION

Since 1957, the NRU reactor at the Chalk River Laboratories of Atomic Energy of Canada Limited (AECL) has been the cornerstone for the development of CANDU® (Canada Deuterium Uranium) reactor technology and the national source of neutrons for basic and applied materials research. However, NRU is unlikely to operate much beyond the year 2000. Accordingly, as reported at previous IGORR meetings [1,2], AECL has been conducting an ongoing assessment of the future requirements for irradiation facilities to support continued development of the CANDU reactor and developing a new Materials-Testing Reactor (MTR) concept to meet these requirements. Furthermore, the Committee on Materials Research Facilities (CMRF), which was sponsored by the NSERC (Natural Sciences and Engineering Research Council), has recognized that the development of new materials is a key component of a competitive economy and recommended that Canada give priority to funding and building a new Canadian neutron beam facility for materials research [3]. AECL has responded to national priorities by ensuring that the emerging MTR concept is a fully dual purpose Irradiation-Research Facility (IRF) that provides:

<u>Irradiation facilities for CANDU fuel and materials testing</u>: Advanced CANDU concepts include more passive safety systems, shorter construction times, easier more reliable operation and maintenance, higher load factors, longer plant lifetimes, and a variety of advanced fuel cycles (including thorium fuels and the burning of actinide wastes and spent fuel from light-water reactors). Therefore, the IRF must be able to test new reactor fuels and materials under representative CANDU reactor conditions.

<u>A national facility for materials research</u>: Neutrons provided by NRX and NRU have facilitated world-class materials research using neutrons (which culminated in the 1994 Nobel Prize for physics given to B.N. Brockhouse for his work on determining the excitation properties (phonons and magnons) of materials, and developing inelastic scattering techniques and the triple-axis spectrometer). As an indigenous source of neutrons, the IRF is essential for continuing neutron-based materials science in Canada and thus for gaining reciprocal access for Canadian scientists to foreign materials-research facilities.

2. EXPERIMENTAL REQUIREMENTS FOR THE IRF

To form a consensus on experimental requirements, AECL established a user review committee with representation from all CANDU and basic R&D programs. The following detailed requirements reflect the dual nature of the IRF.

2.1 Experimental Requirements to Support CANDU R&D Programs

Three major CANDU research and development programs require irradiation facilities:

- **Fuel and fuel-cycle technology** which investigates the behaviour of existing and future fuel designs both under normal operating conditions and at extreme limits;
- **Fuel channel technology** which involves long-term research to study end-of-life behaviour in fuel-channel materials, to improve the fundamental understanding of in-reactor material behaviour, to further develop predictive models for materials behaviour, and to develop improved materials and components;
- **Reactor safety research** which will improve the understanding of fuel and fuel-channel behaviour in the event of loss-of-coolant accidents and under severe-fuel-damage conditions by providing data that validates computer codes and models used for safety assessments and characterizes fission-product release, transport and deposition.

The corresponding CANDU experimental requirements are presented in Tables 1 and 2. Reactor-safety-research requirements are included with those of the fuel technology program under multi-element partial bundles.

TABLE 1: EXPERIMENTAL REQUIREMENTS FOR THE FUEL TECHNOLOGY PROGRAM

CAPABILITY	EXPERIMENTAL REQUIREMENTS
CANDU Bundles	**Capacity:** four or more bundles **Flux length:** 1.0-1.5 m per test section **Coolant pressure:** 10 MPa **Coolant temperature:** 300°C **Linear element ratings:** 50-70 kW/m **Power:** up to 1 000 kW per bundle
Multi-element Partial Bundles	**Capacity:** several test sections with 1 to 8 fuel elements per test section **Flux length:** 1.0-1.5 m per test section **Coolant pressure:** 10 MPa **Coolant temperature:** 300°C **Linear element ratings:** 50-70 kW/m
Diagnostic Capability	**Type:** in-pool neutron radiography of irradiated fuel **Thermal-neutron flux:** $\sim 0.7 \times 10^{18}$ $n \cdot m^{-2} \cdot s^{-1}$ **Type:** neutron radiography of unirradiated fuel **Thermal-neutron flux:** $\sim 2 \times 10^{18}$ $n \cdot m^{-2} \cdot s^{-1}$

TABLE 2: EXPERIMENTAL REQUIREMENTS FOR THE FUEL CHANNEL TECHNOLOGY PROGRAM

CAPABILITY	EXPERIMENTAL REQUIREMENTS
Full-diameter CANDU Fuel Channel Sections	**Capacity:** four or more m of test section **Flux length:** 1.0-1.5 m per test section **Coolant pressure:** 10 MPa **Coolant temperature:** 300°C **Fast Flux:** 0.3 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$
Deformation and Fracture	**Medium fast flux:** 0.7 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$ **Capacity:** 4-5 capsules **High fast flux:** 1.8 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$ **Capacity:** 2 capsules **Ultra-high fast flux:** 3.0 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$ **Capacity:** 2-3 capsules
Corrosion Testing	**Type:** a stainless-steel loop with standard CANDU coolant chemistry, a stainless-steel loop with variable coolant chemistry, and a recirculating gas loop
Diagnostic Capability	**Spectrometers:** one for residual strain determination, one for texture determination and one for chemical phase and annealing studies

2.2 Requirements of the Neutron Beam Research Community

The Canadian neutron beam-research community has described its requirements in several CINS (Canadian Institute of Neutron Scattering) publications [4,5,6] and in the interim report [3] by the NSERC-sponsored CMRF. Although perturbed thermal-neutron fluxes of 6 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$ at the noses of the beam tubes are desirable, lower fluxes of ~2 x 10^{18} $n \cdot m^{-2} \cdot s^{-1}$ (i.e., comparable to the peak thermal-neutron flux at the beam tubes of NRU) are acceptable. The minimum complement of beam tubes includes: seven thermal-neutron beam tubes, one cold-neutron beam tube looking directly at the cold-neutron source plus a cold-neutron guide fan leading to an adjacent guide hall, one thermal-neutron beam tube for neutron radiography, and one in-pool short beam tube for neutron radiography. To allow for future growth, provision should be made for a second cold-neutron source with neutron guides leading to a second guide hall.

3. DESCRIPTION OF THE IRF

Although the IRF concept was developed to be non-site specific, suitable locations are available at AECL's Chalk River Laboratories (CRL) and Whiteshell Laboratories (WL). A representative location at CRL was used to develop the reference cost estimate of 500 million (1994) Canadian dollars. Although certain new support facilities (e.g., process water supply and heating) are included, the estimate assumes an existing nuclear infrastructure (e.g., hot cells, cafeteria, waste treatment centre, waste storage, etc.) at the IRF site.

3.1 The IRF Complex

The IRF complex consists of the reactor containment building, an adjacent guide hall, a mock-up building, and adjoining buildings for operations, administration, and utilities.

The reactor containment building, which houses the reactor and service pools and equipment rooms for the reactor process systems, is a cylindrical reinforced concrete structure (40 m diameter by 40 m high) with a hemispherical dome. It is designed to provide shielding for fission-product releases and to withstand any internal pressurization arising from postulated severe accidents. The lowest level of the building provides an array of instrument stations clustered around the beam tubes as well as access to the horizontal fuel test sections. Neutron guides extend from the outer face of the reactor pool through the building wall to the guide hall. Air locks provide normal access to the building at the reactor-operations and beam-research-facility levels. Removable wall panels allow the infrequent movement of large components to and from the building.

The guide hall houses the cold- and thermal-neutron guides outside the containment building plus various instrument stations. Laboratories, offices and other supporting beam-research facilities are adjacent to the guide hall. The operations building contains control rooms for the reactor and the experimental process facilities plus offices and laboratories and a hot cell for handling experimental equipment. The utilities building contains the electrical power distribution switchgear, compressors, refrigeration units, a de-ionized water plant, the ventilation systems and other utilities needed for reactor operation and the various experimental facilities. Located between the operations and utilities buildings is a shipping and receiving area with a pool for storing spent fuel and other radioactive components and for assembling and inspecting experimental equipment. The administration building contains offices, conference rooms, lecture rooms, and a small self-serve cafeteria. Adjacent to the administration building is a separate building that houses a full-scale mock-up of the reactor and the reactor pool; it will be used for testing reactor components, developing tools and procedures, and training operations and maintenance staff.

3.2 The IRF Reactor Assembly

TABLE 3: MAJOR FEATURES OF THE IRF

FEATURE	DESCRIPTION
Reactor Type	open tank-in-pool reactor assembly with split core
Fuel	19.7 wt% ^{235}U as U_3Si_2-Al 18-, 36-, and 58-rod assemblies with 0.7 m fuelled length
Coolant	H_2O
Reflector	D_2O
Reactor Regulation System	hollow-cylindrical hafnium absorbers inserted into the core from above
Shutdown System 1	de-energize electromagnets to drop hafnium absorbers (overriding regulation system)
Shutdown System 2	D_2O dump from fuelled portion of reactor vessel
Total Power (MW_t)	40 (nominal)

The IRF reactor assembly (major features listed in Table 3) employs a split-core concept that places three horizontal CANDU-fuel-channel test loops between two cores each of which consists of an 18-site MAPLE-type core segment plus two fast-neutron (FN) sites. The reactor assembly is located in a 15.6-m deep, light-water-filled pool and consists of four main components (shown in Figures 1 and 2):

- **Inlet plenum:** The inlet plenum supports the core assembly and reactor vessel and provides H_2O from the primary cooling system.

- **Reactor vessel and Core assembly**: The reactor vessel, which rests on the inlet plenum, is a complex D_2O-filled tank with two corrugated inner walls that define the H_2O-cooled MAPLE-type core regions. Each core segment contains 18 flow tubes that accommodate 16 fuel bundles and two irradiation sites; the flow tubes are mounted into a grid plate that directs cooling water from the inlet plenum past the fuel and the irradiation rigs The vessel has horizontal penetrations for three fuel-bundle test sections and ten beam tubes, plus vertical penetrations for the four FN sites, two vertical U-shaped fuel test sections, and tubes for irradiation assemblies (e.g., hydraulic and pneumatic rabbits). As Figure 1 shows, the reactor vessel extends down and around the upper portion of the inlet plenum When the reactor is operating, the upper (core) portion of the vessel is filled with D_2O, which is held in place by differential pressure of helium gas relative to the lower portion. The D_2O may be rapidly drained into the lower portion (to shut down the reactor) by equalizing the helium pressure in the upper and lower portions of the vessel.

- **Chimney:** The chimney which rests on top of the reactor vessel collects the water exiting the two core segments and directs it through two outlet nozzles back to the primary cooling system/ The chimney is open at the top to the pool which provides a natural-circulation path for cooling the fuel when the reactor and primary cooling system are both shut down. The chimney also supports and locates the eight control/shut-off absorbers.

The fuel elements contain a core of U_3Si_2 dispersed in aluminum and surrounded by a co-extruded aluminum sheath. The LEU (Low-Enrichment Uranium) is enriched to 19.75 wt% ^{235}U. The fuel element design is based on the LEU fuel developed for use in NRU, and is very similar to that supplied for use in HANARO. There are three types of driver fuel bundles:

- **36-element driver bundle:** The main type of driver fuel bundle contains a hexagonal array of 36 elements. The elements are held by a top and bottom end plate mounted on a central support shaft which is locked into a receptacle in the bottom of the flow tube. There are 24 of these 36-element bundles in the core.

- **18-element driver bundle:** Fuel bundles for the eight reactivity-control sites contains 18 elements. The flow tubes in these positions have circular exteriors to accommodate the circular sections of the hafnium control absorbers that surround them.

- **58-element FN bundle:** The four FN sites, located just outside the 18-site core segments, each contain 58 elements in two rings surrounding a central irradiation thimble.

3.3 Reactor Process Systems

The **Primary cooling system** cools the driver fuel and the FN fuel with two circuits that supply H_2O to the two nozzles of the inlet plenum. After passing over the fuel, the water is drawn from the two outlets of the chimney and discharged through the heat exchangers and back to the inlet plenum. To ensure that core coolant is not discharged directly to the pool through the open top of the chimney, a fraction of the coolant entering the inlet plenum is discharged to the pool which results in a flow down the chimney that confines the core-exit flow to the lower part of the chimney. Each circuit has a single plate-type heat exchanger two parallel pumps equipped with high inertia rotors to provide a slow pump rundown. Each inlet pipe contains a venturi-type flow diode that limits the reverse flow through the circuit in the event of an inlet piping failure and ensures adequate cooling flow to the core. At low decay powers, heat is removed by natural circulation path out the top of the chimney through the inlet-plenum flow diodes.

The **Reflector cooling system** provides circulation of D_2O through the upper portion of the reactor vessel over a dump-ring weir to the lower portion, and thence through a cooling circuit that removes the heat deposited in the D_2O and the reactor vessel components. A catalytic recombiner is provided to compensate for the formation of D_2 and O_2 gas that occurs as the D_2O passes through the reactor vessel.

3.4 Reactor Control and Safety Systems

The reactor control system consists of:

- **Control rod and drive:** Eight control rods (four in each of the two core segments) are used to adjust the reactor power. Each control rod consists of a hollow cylindrical hafnium metal absorber attached to a shaft that extends up through the reactor chimney to a drive unit mounted on a support structure located at the top of the pool. The drive unit consists of a stepping motor driven by the control computer, an electromagnet that magnetically couples the connecting shaft to the motor drive, and a hydraulic damper section that arrests the fall of the rod at the end of its travel when the electromagnet is de-energized.

- **Reactor control computer system:** The Digital Control System (DCS) provides computerized control and monitoring for all reactor systems and experimental facilities. Two main components, the Digital Control Computer (DCC) for control functions and data acquisition, and the Plant Display System (PDS), which is the interface between the Reactor Operator and the DCC, make up the DCS. The DCS uses fault-tolerant, dual-redundant, computer hardware to automate operator activities, where possible, during normal operation. The PDS generates the operator displays, processes operator commands and transmits these control commands to the DCC, provides alarm annunciation and acknowledgement, and controls the data logging necessary to support historical data/trend displays and event logs. The display information will also be routed to a remote monitoring and shutdown facility.

Two diverse, independent safety systems are provided:

- **Shutdown system 1:** Shutdown system 1 (SS1) rapidly shuts the reactor down by interrupting the power to the electromagnets of the eight control rods and allowing the absorbers to fall into the core. The parameters monitored include those of the reactor and also those of the various irradiation facilities whose failure without reactor shutdown could result in fuel damage and the release of fission products. All sensing instruments are triplicated.

- **Shutdown system 2:** Shutdown system 2 (SS2) achieves shutdown by rapidly draining the D_2O out of the top part of the reactor vessel that surrounds the core. During normal operation, the D_2O is held up by the pressure of helium gas in the lower part of the vessel. Different types of detectors from SS1 are used, where practical, to avoid failure of both safety systems due to a common failure.

3.7 Experimental Facilities

Table 4 lists the IRF's experimental facilities. The **Horizontal Fuel-Test Facilities** comprise three in-reactor horizontal test sections stacked vertically between the two core segments plus their respective out-reactor process systems. Each test section consists of a calandria tube and a pressure tube that accepts up to three CANDU fuel bundles. Provision is also made for controlling test-section power by adjusting the level of either ^{10}B or ^{3}He poison in a thin annulus outside the pressure tube. Coolant supply piping connects the end fittings to process systems located in shielded rooms where an arrangement of pumps, heat exchangers, tanks and other equipment provides representative CANDU fuel-channel conditions and removes the heat generated by test fuel bundles.

There are two vertical multielement fuel-test facilities. Coolant flows down an inlet leg and then upwards past the fuel assembly located in the outlet leg of a U-shaped test section. A process system located in a shielded room on the reactor process systems level provides the desired temperature, pressure, flow and chemistry conditions in the test section and removes the heat generated by the test assembly.

TABLE 4: SUMMARY OF EXPERIMENTAL FACILITIES

TYPE OF FACILITY	DESCRIPTION
Horizontal Fuel-Test Facilities	3 test sections with several CANDU bundles/section Bottom test section replaceable with a high-integrity test section for BTF tests 3 test loops, 1 per test section
Vertical Fuel-Test Facilities	2 test loops, 1 per test section for multi-element partial bundles
BTF Loop	1 BTF loop system to connect to bottom horizontal test section
Materials-Irradiation Facilities	4 in-core sites with 3 or 4 inserts/site 4 FN-sites with 4 inserts/site or 1 corrosion loop/site
Hot Cells	1 three-compartment cell 1 handling cell for horizontal test sections
Service Irradiation Facilities	10 vertical tubes including: 2 hydraulic rabbit systems 1 pneumatic rabbit system
Neutron Beam Research Facilities	10 beam tubes, 2 of these for cold neutron sources 1 liquid hydrogen cold neutron source 5 cold-neutron guides 2 thermal-neutron guides

The bottom horizontal test section can be configured such that after the reactor is shut down the test section can be depressurized or blown down into the BTF (**Blowdown Test Facility**) loop. The transport of fission products from the fuel can be monitored by gamma spectrometers mounted on the downstream piping and in the BTF loop room at the reactor process systems level. The loop provides for steam and helium cooling of the assembly after the test. The initial water and steam discharged from the test assembly and the post-accident coolant will be directed to a large blowdown tank.

Each core segment has two locations for **Materials-Irradiation Facilities** in which material specimens are irradiated at high fast-neutron flux conditions. The devices (compatible with the QUATTRO rigs employed at the HFR-Petten facility) will consist of four cylindrical inserts in which material specimens are mounted.

In the four FN-sites, the fast-neutron flux is locally increased by surrounding a central irradiation thimble with a 58-element fuel bundle. Different types of devices may be installed in the thimble, including the QUATTRO-type rigs, and assemblies of cylinders containing material specimens that are cooled by water or gas, whose chemistry and temperature can be varied to examine the effect on the corrosion properties of the material specimens. The irradiation effect on the coolant chemistry, such as radiolysis, can also be examined in these devices. Two D_2O loops and one gas corrosion loop will be provided to supply D_2O or gas to the corrosion specimens.

A hot cell on the main floor level is divided into three compartments. The general purpose compartment will contain equipment for assembly and disassembly of the irradiation facilities, such as the QUATTRO-type rigs and the vertical fuel-test assemblies. The other two compartments will be used for sample examination. The rabbit transfer system will be contained in one of these compartments. Rabbit capsules may be transferred from the cell to a laboratory located outside the Reactor Containment Building in the operations area. A hot cell is also located on the beam research facility level, for installing and removing assemblies from the horizontal test sections (i.e., fuel bundles, pressure tubes and BTF assemblies).

Two vertical tubes provided in the reactor vessel will contain assemblies into which material specimens can be installed and removed using a hydraulic rabbit. A pneumatic rabbit will also be provided. Other vertical tubes are provided for general purpose irradiations and future facilities.

Various experimental facilities are provided for the national and international neutron beam research community. There are eight thermal-neutron beam tubes in the reactor vessel to deliver beams of neutrons to instrument stations outside of the reactor. Two of these beam tubes will accommodate thermal-neutron guides to transmit neutrons to instrument stations in the Guide Hall. Two additional beam tubes can accommodate liquid hydrogen cold neutron sources, although only one cold neutron source is included in the initial installation. The second beam tube can be used as a thermal-neutron beam tube until a second cold neutron source is installed. A liquid hydrogen cold neutron source and a set of cold-neutron guides will allow cold neutrons to be delivered to instrument stations in the Guide Hall. It is planned to move three existing triple-axis spectrometers, a high-resolution powder diffractometer (one-half of DUALSPEC), and a polarized neutron triple-axis spectrometer (the other half of DUALSPEC) from NRU to the new facility. These instruments will be attached to the thermal-neutron beam tubes. Six new instrument stations for use on the cold-neutron guides will be supplied as part of the IRF project. The specifications for these instruments will be developed in consultation with CINS.

4. PERFORMANCE ASSESSMENT

The neutronic performance of the reactor and its experimental facilities analyzed using the physics computer codes WIMS-AECL/3DDT [7,8] and MCNP [9]. A total power of 40 MW_t was assumed to be produced by the two core segments (~33 MW_t) and the four FN-sites (~7 MW_t). The design power, and hence the available flux levels, will be optimized during formal design phases of the IRF project.

4.1 Concept Development to Meet CANDU Bundle Testing Requirements

The experimental requirement that has most strongly governed the evolution of the IRF concept has been the capability for irradiating at least four (and preferably eight or more) CANDU bundles under realistic power-reactor conditions. The primary requirement is to operate natural CANDU bundles under the very uniform high-power conditions listed in the second column of Table 5. Additionally, it must be feasible to irradiate the same bundles at lower power and element ratings with a substantial axial power gradient that represents end-bundle conditions, but without introducing power tilts or changing the relative element-to-outer-element powers.

In the NRU facility (as shown in Table 5), it has been necessary to enrich prototype CANDU-6 fuel bundles to ~1.7% to attain a power history that conservatively represents a natural UO_2 bundle that operates at 1000 kW when fresh; the somewhat higher initial power for the enriched test bundle compensates for the slight initial power rise of a natural UO_2 bundle at low burnup. As the NRU loops are vertical, it has been normal practice to replace the centre fuel element with a support shaft that holds up to six bundles per test section. Although NRU's large size enables very uniform irradiation conditions, the inner elements run somewhat cooler at the outer elements somewhat hotter than desired as a consequence of the use of enrichment.

Early MAPLE-MTR studies [2] that considered the vertical irradiation of single CANDU bundles in the heavy-water tank that surrounds a MAPLE-X10-type core showed that power levels of interest (800-1000 kW) could be obtained with 5% enrichment (column 4 of Table 5); however, the relative element powers are quite atypical and the power tilt across the bundle (in the direction away from the core) is relatively large. The use of four FN fuel assemblies interspersed with four vertical CANDU test sections results in a reduced need for enrichment (to ~2%) and improvements in the power tilt (column 5 of Table 5), but the element power distribution remains atypical of natural CANDU bundles.

By placing the CANDU bundle-test sections (horizontally) in the centre of the IRF core, the capability of uniformly irradiating several natural UO_2 bundles at up to 1000 kW is attained with the required element power distribution, especially if D_2O is employed as the coolant (column 6 of Table 5). These IRF results were obtained using MCNP with allowance for fission-product decay heating.

TABLE 5: MEETING CANDU BUNDLE IRRADIATION REQUIREMENTS

Parameter	CANDU*	NRU	Early MAPLE-MTR		IRF
			no FN Sites	with FN Sites	
Number of Fuel Elements	37	36	37	36	37
Enrichment (wt% ^{235}U)	0.7	1.7	5.0	2.0	0.7
Bundle Power (kW)	1000	1046	840	890	~1000
Average Element Rating (kW/m)	56	61	47	52	56
Ave. Outer Element Rating (kW/m)	63	74	60	65	65 63*
Peak Outer Element Rating (kW/m)	64	78	87	78	71 69*
Centre vs. Outer Element Power	0.69	-	0.28	-	0.63 0.69*
Inner vs. Outer Element Power	0.72	0.56	0.35	0.50	0.67 0.72*
Middle vs. Outer Element Power	0.81	0.68	0.50	0.63	0.77 0.81*
Maximum vs. Average Element Power	1.13	1.22	1.40	1.26	1.16 1.13*
Axial Gradient (max./ave.)	1.01	1.02	1.10	1.09	1.07
Axial Gradient (min./ave.)	0.99	0.97	0.80	0.80	0.90
Outer Element Power Tilt (max./ave.)	~1.0	1.01	1.19	1.10	1.02
Outer Element Power Tilt (min./ave.)	~1.0	0.99	0.82	0.90	0.98

* Heavy-Water coolant

The power output predicted by WIMS-AECL/3DDT for the horizontal test sections with two natural uranium CANDU bundles per test section and H_2O coolant is about 600 kW per bundle in the bottom test section, 900 kW per bundle in the middle test section, and 650 kW per bundle in the top test section. Using D_2O coolant in place of H_2O coolant increases the power per bundle by about 5%. With three bundles per test

2 wt% ^{235}U, the power output from each test section would approximately double. To avoid derating the reactor during the irradiation of enriched CANDU bundles, studies have been performed that establish the feasibility of controlling test section power using either ^{3}He in the gas gap between the pressure tubes and the calandria tubes or soluble ^{10}B in a D_2O annulus outside the calandria tubes.

4.2 Performance of Other Experimental Facilities

The average linear element rating for a seven-element assembly and H_2O coolant, estimated using WIMS-AECL/3DDT is 37 kW/m at natural uranium enrichment and 76 kW/m if enrichment is 2 wt% ^{235}U. The extrapolated flux length was calculated to be 1.2 m.

The performance of the in-core materials-irradiation devices was estimated by modelling representative QUATTRO assemblies in two sites in each core segment. The calculations were performed with WIMS-AECL and 3DDT. The fast-neutron ($E > 1$ MeV) flux exceeds 1.3 x 10^{18} $n{\cdot}m^{-2}{\cdot}s^{-1}$ (i.e., 8.8 displacements per atom per year at 90% reactor availability) for a 150 mm length of zirconium alloy, and is greater than 1.0 x 10^{18} $n{\cdot}m^{-2}{\cdot}s^{-1}$ (i.e., 6.7 displacements per atom per year at 90% reactor availability) for a 450 mm length of zirconium alloy. For comparison, the pressure tube material in a typical CANDU reactor experiences one to two displacements per atom per year. Hence, the irradiation assemblies will permit accelerated aging studies to be conducted on pressure tube materials.

The performance of FN-irradiation devices was also estimated by modelling representative QUATTRO rigs. The fast-neutron ($E > 1$ MeV) flux exceeds 0.42 x 10^{18} $n{\cdot}m^{-2}{\cdot}s^{-1}$ (i.e., 3.0 displacements per atom per year at 90% reactor availability) for a 94 mm length of zirconium alloy, and is greater than 0.33 x 10^{18} $n{\cdot}m^{-2}{\cdot}s^{-1}$ (i.e., 2.3 displacements per atom per year at 90% reactor availability) for a 460 mm length of zirconium alloy.

The peak unperturbed thermal-neutron flux is estimated to exceed 4 x 10^{18} $n{\cdot}m^{-2}{\cdot}s^{-1}$ in the D_2O region outside of the core segments. Table 6 lists the unperturbed and perturbed thermal-neutron fluxes at the entrances to the beam tubes, as calculated with WIMS-AECL/3DDT and MCNP. For the unperturbed case, the beam tubes were replaced with D_2O. In Table 6, the entries for BT9/10 in the cold source vessel refer to modelling the beam tubes with a cold neutron source vessel installed but without including the liquid hydrogen in the vessel. There is good agreement between the WIMS-AECL/3DDT and MCNP results.

5. SUMMARY

AECL has responded to the need to replace the NRU reactor by developing the concept for a national dual-purpose IRF to test CANDU fuels and materials and to perform materials research using neutrons. The proposed IRF would meet the Canadian nuclear industry's needs with various CANDU-specific experimental facilities, and would satisfy a very broad spectrum of national and international academic and industrial research requirements with a national neutron-beam research facility.

TABLE 6: UNPERTURBED AND PERTURBED THERMAL-NEUTRON FLUXES FOR THE BEAM TUBES

	WIMS-AECL/3DDT		MCNP	
	Unperturbed ($\times 10^{18}$ $n \cdot m^{-2} \cdot s^{-1}$)	Perturbed ($\times 10^{18}$ $n \cdot m^{-2} \cdot s^{-1}$)	Unperturbed ($\times 10^{18}$ $n \cdot m^{-2} \cdot s^{-1}$)	Perturbed ($\times 10^{18}$ $n \cdot m^{-2} \cdot s^{-1}$)
BT1/2	3.3	2.9	3.2	2.4
BT3/4	3.8	3.0	3.6	2.5
BT5/6	2.9	2.2	3.1	1.8
BT7/8	2.7	1.9	3.0	1.8
BT9/10 (at the entrance)	3.0	2.3		
BT9/10 (in cold source vessel)		2.2		1.8

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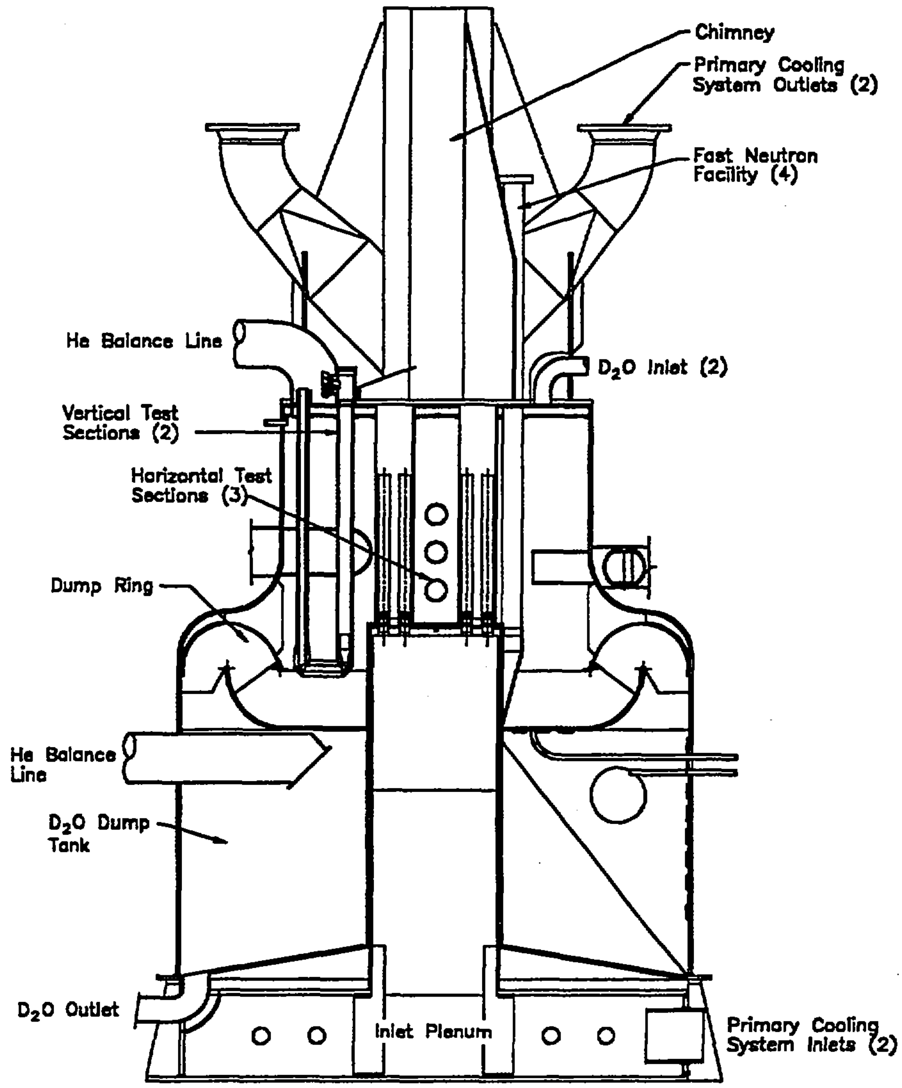

FIGURE 1: IRF REACTOR ASSEMBLY VERTICAL SECTION

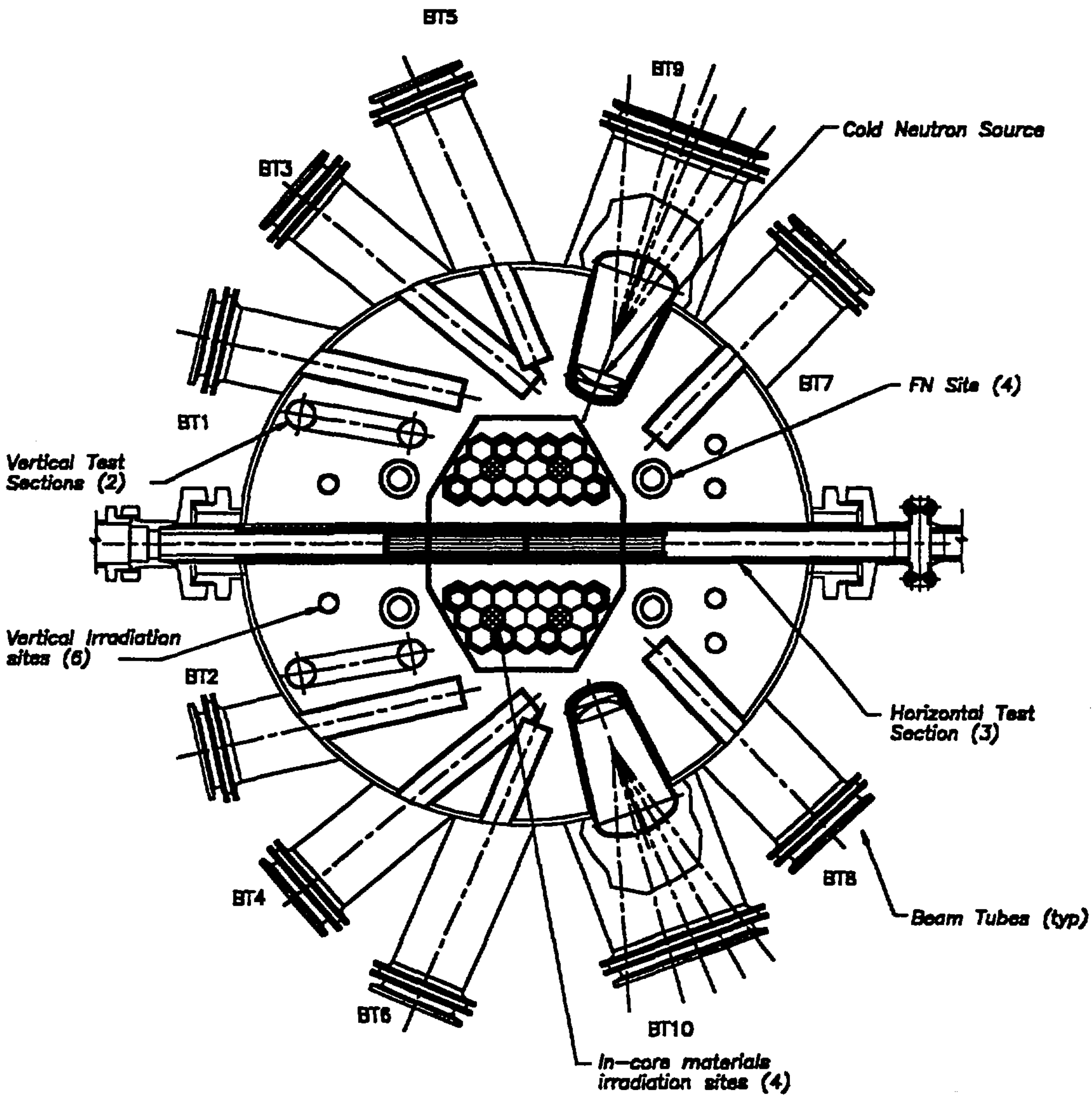

FIGURE 2: IRF REACTOR PLAN VIEW

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IGORR-IV Meeting

The state of the PIK reactor construction

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1. Parameters and destination.

1.1 The PIK reactor complex [1] was designed as the national centre of the USSR for neutron research on the c-w reactor. Now, it is considered as the Russian Centre and, possibly, as the International Centre. At present, only four middle-class research reactors with the flux of the order 10^{14}n/cm^{2}s operate in Russia, on which the work with beams is carried out. These are the reactor WWR-M in Gatchina commissioned in 1959, the reactor WWR-S in Obninsk (1964), the reactor IWW-2M in Ekaterinburg (1963), and the reactor IR-8 in Moscow.

The pulsed reactor IBR-2 at JINR (Dubna) is the only modern source in Russia giving about 10^{16}n/cm^{2}s in a pulse but its mean flux is about 10^{13}n/cm^{2}s.

Principle concepts of the PIK reactor project were stated late in the 60's but its construction was started in 1976. By the year 1986 the initial project was realised by approximately 70% but then, (after Chernobyl accident) the construction was essentially frozen to adjust the project to the revised nuclear safety regulations. The revised project was approved only in 1990 when the country was on the threshold of serious economic problems.

The PIK reactor is a source of neutrons(core volume ~ 50l) placed in the heavy water reflector. The fuel - uranium-235 (90% enrichment) of total weight 27 kg. Light water is used as moderator and coolant.

Design parameters:

- thermal power - 100 MW
- thermal neutron flux
  in the reflector - $1.2\text{x}10^{15}$n/cm^{2}s
  in the central vertical beam tube - 5.10^{15}n/cm^{2}s.
- a number of horizontal beam tubes -10
- diameter of beam tubes - 10 cm with the possibility of their replacement with beam tubes up to 25 cm in diameter.
- a number of inclined beam tubes - 6

- a number of vertical tubes
  for irradiation of samples - 6

The reactor will be equipped with sources of hot, cold (2 pieces), and ultracold neutrons to obtain beams in different parts of energy spectrum.

The low temperature circuit will make it possible to irradiate samples at helium temperatures.

The branched system of neutron guides (4 for cold neutrons and 4 - for thermal neutrons) of total length ~ 300m allows to transport beams into pure (backgroundless) conditions of neutron guide room adjacent with the reactor building. The total number of positions on beams for arrangement of experimental installations - 50 [1]

The reactor has three series cooling circuits. Emergency core cooling systems in LOCA are duplicated and in emergency power supply system is triplicate.

The PIK reactor has no single common containment but four separate systems: for pipelines and units of the first circuit, for heavy water reflector, for operating hall, and for experimental beam tubes hall. Requirements for retention of radioactive products, in accordance with expected consequences of initial events, are imposed on each of them. Naturally, in case of accident, fission product yield is the highest and the most severe requirements are imposed on containment in which the first circuit is placed. The containment of the operating hall which walls serve as exterior walls of the building is constructed as double with an air gap across all the area.

2. The state of construction and prospects

The vast majority of buildings and construction (27) of the complex have been built (fig.) though finishing work have been completed not in all of them.

At present we are lacking of:

- building for a reserve diesel station and for a simulator which were called for during inspection of the project (the foundation has been laid);
- building and a group of satellite objects of the plant for isotopic purification of reflector heavy water (actually they will be needed within three years after the start-up when tritium will be accumulated);
- building for special fire depot;
- shower water purification system.

All these objects are not included into the starting-up complex, except the first one, and may be completed later.

All the problems of power supply are practically solved:

- energy supply of the first category of safety is performed by three already installed independent high voltage lines (power substation is in the stage of construction);

- water is supplied from available artesian wells with a special system for purification and softening;

- heat supply is performed from central boiler operating on the territory of the institute (heat is supplied to all the construction blocks now).

The construction of the double-fenced, instrumented security barrier around the territory of the complex has been completed. It is planned to be put into service in full in 1995.

1995 it is planned to complete the work on reinforcement of ceilings over the main and operating halls (shield containment) that will make possible the erection of the reactor itself as well as systems of the first circuit ("pure zone").

The equipment of the intermediate circuit as well as that of the power supply block is practically mounted in full.

Today the lack of sufficient financing actually prevent mounting work at other objects of the complex.

The critical assembly - a full-scale model of the PIK reactor operating at the power up to 100 W was created and put into operation. This assembly enables to check experimentally, under actual conditions, neutron - physical parameters of the PIK reactor and to optimise them.

At this moment, constructional readiness of the complex is evaluated as approximately 80%, completeness with equipment - as 65 - 70%.

The assessment of uncompleted work $30 million (without the plant for isotopic purification of heavy water). This estimate is approximately 3 times higher that made in 1992. It coincides in the digit 30 but in the year 1992 it included all the objects of the complex and partially, the social sphere. Since that time inflation of the rouble, increase in estimated cost in the field of building, and change of exchange rate have led actually to 3 times increase of the cost of construction in dollars. The assessment of potentialities of industrial enterprises producing equipment as well as those of erection organisations carrying out work at the complex shows that in case of guaranteed and regular financing the work at the starting up complex may be completed within 2,5 - 3 years.

Attempts to internationalise the project, i.e. attraction of financial resources of western partners at the level of governments of a number of leading countries of the West, which were made in 1992-1993 (a special resort of Gaydar E.T. acting chairman of the Government, to leaders of a number of countries) with obligations of the Russian party to organise the International Neutron Research Centre on the basis of the PIK reactor, did not give any results, in spite of the positive, in a whole, conclusion of international experts.

At the present time the project is financed on the lowest level. This resources are sufficient to complete reconstruction of the reactor block, which primarily consists in completion of construction of the containment for operating hall as well as in completion of security barrier of the territory according to high enriched uranium storage regulations. A part of these resources is also intended for design work connected with replacement of the reactor control system with modern element base.

The heavy water purification and processing plant [2] is, probably, the most attractive for possible commercial investors.

At PNPI the original procedure of such reprocessing was developed and tested on experimental plants. The detailed design of large - scale plant has been developed. Its realisation requires capital investments at the level of $10 million. Plant capacity ~ 40 t/year of high conditioned D^2O.

Prior the reactor start-up and over the first three years of its operation, all these capacities and then, approximately a half of them may be used for servicing outside customers.

Estimated time of invested expenses justification will constitute 3 years , taking into account the construction, raw and operation expenses,

3. Discussion of the project with "Goskompriroda"

Some years ago, new requirements on authorisation of projects, prior their construction, with nature protection body were introduced. These requirements include also the discussion with the public. Although the construction of the PIK reactor was started many years ago, it was decided to apply this procedure to the PIK reactor reconstruction project.

The discussion with the public was turned aside to nuclear power development or to non - acceptance of nuclear physics as science. In the positive part it was adopted that PNPI had increased cultural potential of the town Gatchina, that a great contribution had been made to development of engineering infrastructure of the town. The creation of town sewage treatment plant as the first stage of the construction of the PIK reactor was especially pointed out, as well as the contribution to development of social sphere of the town, i.e. living houses, a school, kindergartens, sport complex, and so on. At this point it is appropriate to draw attention to a considerable warming - up in attitudes of the public to the PIK reactor and to nuclear power , as a whole. Late in the 80's and early in the 90's it was extremely difficult to protect NPP in public discussions and it caused a protest in itself, but now, the discussion is much businesslike in character. It may be explained by the fact that the population made certain decision of thoughtlessly of slogans on shut down of NPP.

In the work with experts on the technical part of the project calculations of dose commitments at design and hypothetical accidents of the reactor were discussed most seriously. Calculations were made by independent organisations and using different programs. Experts claimed to consider the full melting of the core though, according to design basis events, such melting had not taken place. The full melting of the core implied more rigid requirements for tightness of containment of the reactor, which were adopted as additions to existing measures on localisation and mitigation of accidents.

The demand on creation of the system for purification of shower waters from the reactor area, prior their discharge to a river, was put forward. It should be noted that it will be the first similar system in the town of Gatchina.

Of requirements of experts which influenced the project volume it should be noted the development of local automated radiation monitoring system of 20 posts situated on the territory of PNPI as well as in the exclusion area and supervision zone and also, full-scope replica training simulation.

The rest requirements only slightly effect the project and are, generally, referred to operation conditions of the reactor.

4. R & D

Specific investigations were carried out in connection with the opportunity of increase of reactor parameters and further study of safety problems.

4.1. The possibility to increase the reactor working cycle.

The introduction of the absorbent to heavy water circulating in the gap between two walls of the core tank when reactor is about subcritical releases a part of reactivity on control rods to increase a cycle duration. According to calculation confirmed on a critical assembly it is possible to increase the working cycle approximately by 4 days. In case of this procedure no alteration in the reactor design and core occur.

The use of roads with burnable absorber instead of dummy one in fuel assemblies enables also to prolong the operation by 6 days. The work on validation of the design model is being continued on the basis of available experimental data.

4.2.

The work on substitution of materials for fuel assembly shroud of stainless steel on zirconium is continued. Such substitution will make it possible the increase of burn up in spend fuel elements by approximately 5% . New fuel assemblies should stand all the test cycle of their safety use to get authorisation.

4.3. Substantial prospects of improvement of parameters of the PIK reactor are connected with replacement of steel core tank with aluminium alloy tank. In so doing, the increase of the thermal neutron flux in the reflector is possible as well as the increase of the reactor service life without replacement of its tank. The work goes rather slowly due to the need in data on radiation damage stability of aluminium alloys up to fluences 2.10^{22} n/cm^{2}s that demands sizeable funds.

4.4. Development and manufacturing of the in-service inspection system for equipment were stopped due to limited funds but available diagnostic devices were used to inspect the equipment which was bought and already mounted. This work is carried out in parallel with surveillance tests of existing equipment in conformity with new requirements for reliability of separate units. Additional calculations and inspection allow to come to decision as to measures on preserve, on necessity of reinforcement or replacement of one or another unit. Thus, of 17 units examined in 1992, 16 were certified by the state safety commission as suitable for future operation and one unit needs additional reinforcement.

Acknowledgement.
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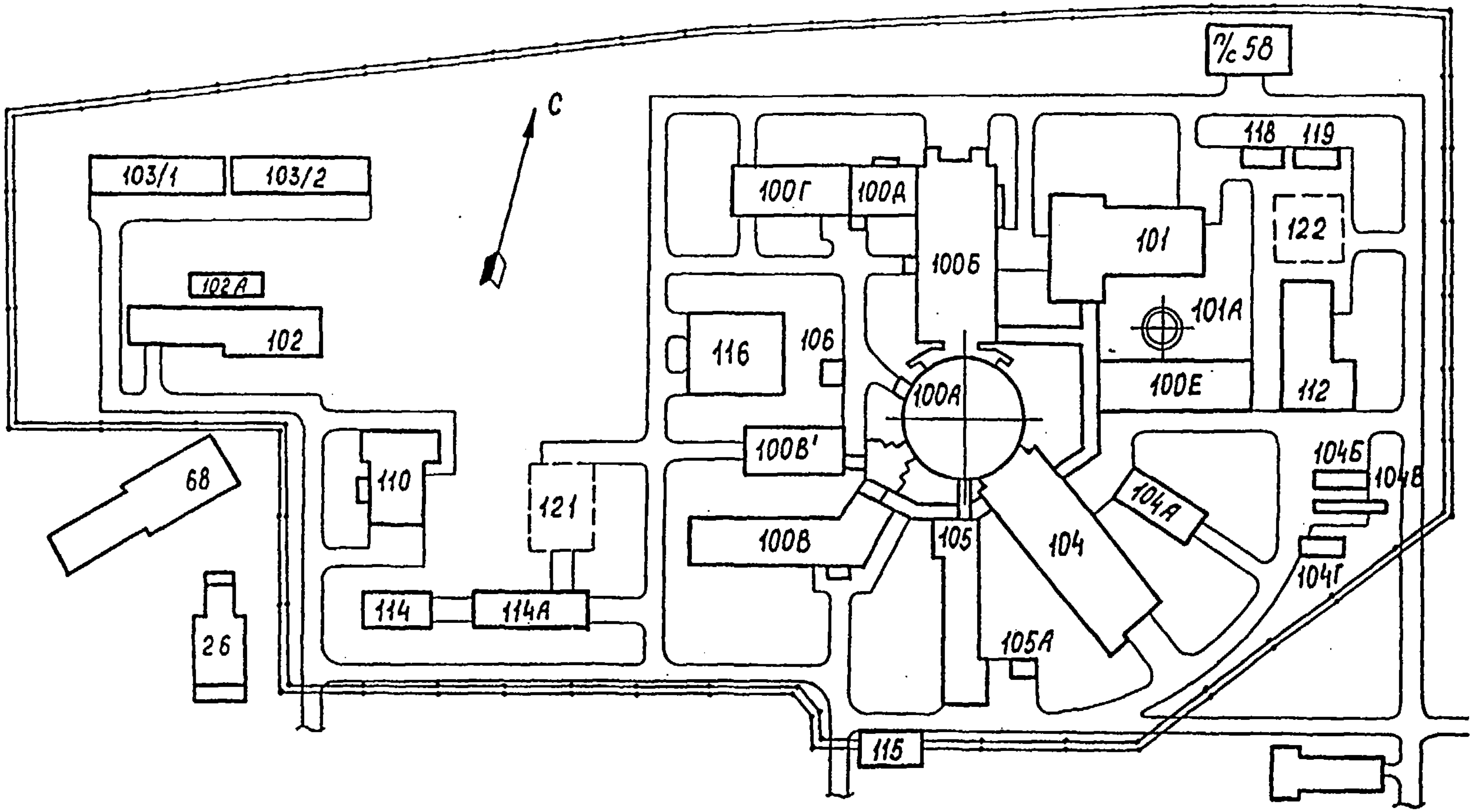

Fig.1. General layout of PIK complex Notations

—fence,

— buildings under construction,

— buildings in design stage.

100A — reactor and physical research laboratories
100B — primary circuit pump station and hot cells
100B — personnel sanitary entrance and ventilation system
100Г — intermediate circuit pump station
100Д — nuclear power unit
100E — cryogenic station
101 — ventilation center
101a — ventilation chimney
102 — circulating water supply pump station
102A — cold water tank
10[illegible] — co[illegible]owers

104 — neutron guide halls and laboratories
104A — technical block
104Б — circulating water supply pump station
104B — cold water tank
104Г — cooling tower
105 — physical research laboratories
105A — storehouse
106 — carbon dioxide
110 — compressor station
112 — isotopic purification plant

118 — nitrogen shop
114 — storehouse
114a — storehouse
115 — guardians office
116 — emergency diesel power plant
58 — power substation
28 — chemical water purification plant
68 — emergency tanks
121 — simulator
122 — liquid radioactive waste state

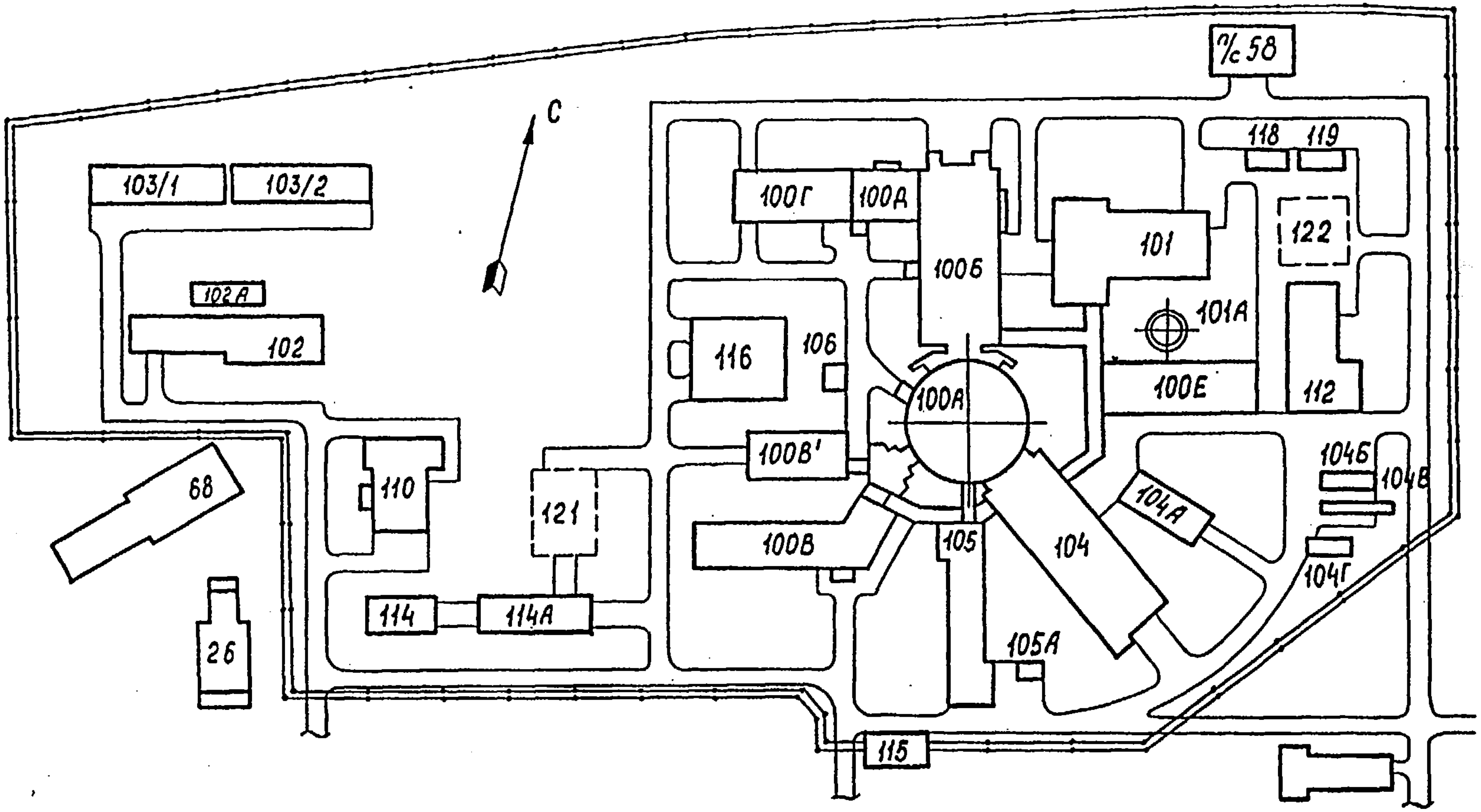

Fig.1. General layout of PIK complex Notations

—fence,

— buildings under construction,

— buildings in design stage.

100A — reactor and physical research laboratories
100Б — primary circuit pump station and hot cells
100В — personnel sanitary entrance and ventilation system
100Г — intermediate circuit pump station
100Д — nuclear power unit
100E — cryogenic station
101 — ventilation center
101a — ventilation chimney
102 — circulating water supply pump station
102A — cold water tank
103 1/2 — cooling towers
104 — neutron guide halls and laboratories
104A — technical block
104Б — circulating water supply pump station
104В — cold water tank
104Г — cooling tower
105 — physical research laboratories
105A — storehouse
106 — carbon dioxide
110 — compressor station
112 — isotopic purification plant
118 — nitrogen shop
114 — storehouse
114a — storehouse
115 — guardians office
116 — emergency diesel power plant
58 — power substation
28 — chemical water purification plant
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A CONTINUOUSLY PULSED TRIGA REACTOR: AN INTENSE SOURCE FOR NEUTRON SCATTERING EXPERIMENTS

by

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ABSTRACT

A variety of configurations of TRIGA reactor cores has been used to demonstrate various pulsing applications, including pulses repeated much more frequently than the more normal six pulses per hour. Because of the renewed interest in sources for neutron beams with intensities ranging up to 10^{16} n/cm.s, the continuously pulsed TRIGA reactor has been reexamined. The TRIGA reactor fuel (U-ZrH_x) has been demonstrated to be very robust especially for pulsing applications. To produce 25 to 50 pulses per second, additional features are available which also have been demonstrated individually. These include the availability of small diameter fuel rods capable of producing high power but with manageable peak fuel temperatures and control features adequately efficient to provide the required rapidly changing reactivity. The TRIGA fuel has been subjected to very great burnup (>65%) with no problem of fuel stability and with all the safety features of TRIGA fuel intact. It appears that an average power of 10 to 16 MW will be sufficient to provide the desired intensities of neutrons. The availability of these additional features makes the rapidly pulsed TRIGA reactor an important candidate for thermal neutron scattering experiments.

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1. INTRODUCTION

Because of the demise of the Advanced Neutron Source (ANS) project and the continued interest in constructing even larger spallation sources for neutron scattering research, it appears useful to reexamine the concept of a continuously pulsed TRIGA reactor. Kursted and Miley[1] reported early work (1971) on a rapidly pulsed TRIGA reactor which they pulsed at rates up to 3 per minute. Considerable experimental research has been devoted to efforts to pulse a TRIGA reactor at rates up to at least 10 per minute and perhaps 30 per minute.

The successful development of a TRIGA reactor core specifically designed to produce peak power pulses of 100 MW or more at rates of up to 50 per second can provide a thermal neutron source intensity of up to 10^{16} n/cm^2·s that is competitive with multiple pulsed fast reactors and the pulsed spallation sources. The relative simplicity of the pulsed TRIGA system contrasts strongly with the complexity of the typical BeV proton synchrocyclotron used with the various spallation sources for neutron production.

2. TECHNICAL DEVELOPMENTS THAT SUPPORT A MULTIPLE PULSING TRIGA

2.1 Small Diameter TRIGA Fuel for Higher Power Operation

Although the present high power pulsed TRIGA reactors[3] use a large diameter fuel (38 mm), the smaller diameter fuel developed originally for higher power steady state power levels (5-15 MW) has also been successfully tested in a long series of applications. The smaller diameter fuel (9.4 mm) has been used in the high power steady state TRIGA reactor in Romania (14 MW) with great success[4] continuously since 1980 with a fuel burnup of 13000 MWD for the core. The local burnup in certain of these fuel rods reached 83%. The smaller diameter TRIGA fuel using the Low Enriched Uranium (LEU) successfully completed all the RERTR fuel tests[5] for LEU fuel with fuel rod burnup reaching 65% for some rods. To demonstrate the robust nature of the small diameter TRIGA LEU fuel, several of the 9.4 mm diameter TRIGA LEU fuel rods were tested in the large pulsing reactor at General Atomics[6] and underwent a long series (~300) of high power pulses plus a very large number of power cycles (zero power to full steady state power) without any evidence for fuel distress.

The remarkably successful development of the smaller diameter TRIGA fuel rod in the LEU format offers considerable advantages for the proposed rapidly pulsed TRIGA requiring up to 50 pulses per second. Considerably lower peak fuel temperatures will result for the same average power levels compared to use of the standard 38 mm diameter fuel rods assumed in the early calculations[7] for the continuously pulsed TRIGA reactor. Since the TRIGA fuel now proposed for the multiple pulsing reactor will use the fully qualified LEU fuel, all requirements for non-proliferation will be satisfied. Furthermore, the proposed TRIGA fuel with its erbium burnable poison will provide an exceptionally long core life, a considerable advantage for neutron beam experiments since the operating cycle can be extended to several months if desired.

2.2. New Type of Pulsing Mechanism

The use of standard control rods in single or banked configuration with stepping motor driving mechanism is well established for steady state operation. For pulsing TRIGA reactors even as rapidly as 10 per minute, the standard pneumatically operated pulsing mechanism is sufficient. For the production of pulses at rates up to 50 per second, the use of a rotating wheel that alternately introduces poison and fuel or poison and non-poison can be considered. This would be similar in some degree to that used successfully with the IBR-2 fast reactor facility in Dubna, Russia.

A new type of pulsing mechanism for fission reactors that does not involve mechanical motion has been proposed by Bowman[8,9]. This system of reactor control depends on the demonstrated property of ^3He polarization by laser light. The reduction in reactivity control results from application of laser light to ^3He inside a control rod. Since the very large 1/v absorption cross section of ^3He is associated with only one of the two spin states that can be formed when S-wave neutrons are absorbed, the absorption cross section decreases by a factor of two for fully polarized ^3He. When the laser light is turned off, the polarization of ^3He decays thus increasing the resulting reactivity effects. Unfortunately, the normal half life for polarization decay in this system would be long, about 12 hours. However, application of a small pulsed magnetic field produced by means of a coil surrounding the control rod can depolarize the ^3He is less than a millisecond. Thus the increase in reactivity insertion can be very rapid for this system and could be of significant value in the rapidly pulsed TRIGA system.

If the ^3He reactivity control system could be developed to produce a rapid enough polarization, this control system would contribute significantly to the safety of the overall system since the need for the rotating wheel would disappear.

2.3 Improved Instrumentation for Neutron Scattering Measurements with a Pulsed Neutron Source

The traditional methods to perform thermal neutron elastic and inelastic scattering measurements are based on either some form of neutron diffractometer or time-of-flight techniques. Both of these general approaches are used with steady state neutron sources such as the high power reactor (ILL) at Grenoble, France. For the time-of-flight applications, it has usually been necessary to use a beam chopper to provide the initial narrow pulse widths required for high resolution scattering measurements.

In more recent times, a high energy accelerator ($\sim$1 BeV protons) has been used with a spallation target to produce the short pulses of thermal neutrons required for neutron scattering measurements. For many applications it has also been necessary to use *"poisoned"* moderators to achieve the required narrow pulses especially with cold neutron sources. Such a poisoning technique does indeed achieve the required narrow source pulses of neutrons but at the cost of reduced neutron source intensities.

In the mid-1960s, Whittemore and associates developed the high efficiency Fourier Chopper[10] and obtained a U.S. patent[11]. At about the same time, Dr. Hiismaki and associates in Finland developed a modified approach based on a similar Fourier Chopper concept. In a practical manifestation of the Finnish approach, the so-called Reverse Time-of-Flight (RTOF) technique has been applied to scattering studies using a steady state reactor source[12]. The Geesthacht and Finnish group have close cooperation[13] with the Russian group in Leningrad and with the IBR-2 group at Dubna. A great advantage of the RTOF method with the Fourier Chopper is that it can be used without an additional phased chopper for the broad pulses produced by IBR-2 and possibly by the multiple pulsed TRIGA system. Elimination of the additional chopper greatly enhances the efficiency with which the pulsed neutron sources are used since source neutrons in the entire broad pulse (or at least most of the broad pulse) are accepted for the RTOF technique.

3. THE CONTINUOUSLY PULSED TRIGA REACTOR FOR NEUTRON BEAM EXPERIMENTS

3.1 Principle of Continuously Pulsed TRIGA Reactor.

A modification of the pulsing mechanism traditionally used with pulsed TRIGA reactors is suggested which will permit the reactor to be pulsed repetitively many times per second to a peak power level in the 40 to 400-megawatt range with the power between pulses of the order 0.5 megawatt. All of the inherent safety features associated with the large negative prompt coefficient of reactivity are retained for the proposed reactor.

It is not sufficient simply to pulse the reactor to a power level of the order 100 MW by adding a step insertion of positive reactivity and then relying on the prompt negative temperature coefficient to limit the peak power since the resulting pulse will be quite wide ($>$ 0.010 sec) with a *"tail"* which contains a significant amount of unwanted energy and of limited repetition rate. Rather, it is proposed to shape the pulse through the successive insertions of positive and negative reactivity. The programmed insertion of negative reactivity will have the beneficial effect of strongly perturbing the reactor flux and will result in a substantial reduction of the trailing edge and tail of each pulse. Taking advantage of this feature, one can add a larger amount of positive reactivity than actually needed to produce the desired peak power, relying on the clipping to limit the peak as well as the tail of the pulse. The advantage of adding as much positive reactivity as possible in this manner is to decrease the reactor period and thus reduce the overall pulse width. It is also essential to reduce as much as possible the tail of the pulse (due fundamentally to delayed neutrons) as well as the width of the pulse in order to reduce the average power. The insertion of positive and negative reactivity, as proposed here, serves both these purposes.

For slow neutron beam research using time-of-flight techniques, the low power of the reactor between pulses will contribute to improved signal-to-noise ratio compared with the use of a reactor operating at a steady-state power sufficient to produce a source flux of 10^{15} to 10^{16} n/cm^2·s. Further improvement of the signal-to-noise ratio is possible with additional gain of neutron beam intensity through the use of a D_2O reflector tank and tangential beam tubes.

The pulse width at half-height will probably be on the order of 3 to 6 milliseconds. This is considerably wider than desired for neutron beam research where pulse widths of the order 5 to 10 μsec are desired. However, the broad source pulses can be efficiently used for the RTOF technique based on the Fourier Chopper, as discussed in Section 2.3 above. Alternately, Fermi choppers can be used to provide the desired narrow pulses. The pulsed thermal reactor considered here also provides an excellent opportunity to produce *"cold"* neutrons in a cryogenic source since the cold source can be of optimum size. The cold source can thus be designed for maximum intensity of *"cold"* neutrons and not be limited in size or *"poisoned"* as usual with cryogenic sources used with spallation sources.

3.2 Description of Pulsed Reactor System

The proposed reactor system is expected to use the 9.4-mm diameter fuel rods containing TRIGA LEU UZrH-Er fuel. The uranium loading is expected to be 30 wt-% or 45 wt-%, depending on the core lifetime desired. The average power for the core will be on the order of 10 MW. Forced cooling with light water would be used. Similar thermal hydraulic considerations have been successfully applied to the Romanian TRIGA 14-MW reactor[4] and the standard 10 MW TRIGA LEU reactor[14]. The pulsing mechanism would be a sequenced insertion of positive and negative reactivity. A rough conceptual sketch of the core and one possible pulsing mechanism consisting of one or two wheels is shown in Figure 1. The core and pulsing mechanism will be located in a usual TRIGA shield with possibly D_2O surrounding at least part of the core. Presumably the wheel(s) will spin in a water-free environment, though this detail has not yet been settled.

Within the general framework discussed above, a number of parameters for the pulsed system can be varied to satisfy various needs. Among these parameters will be notably

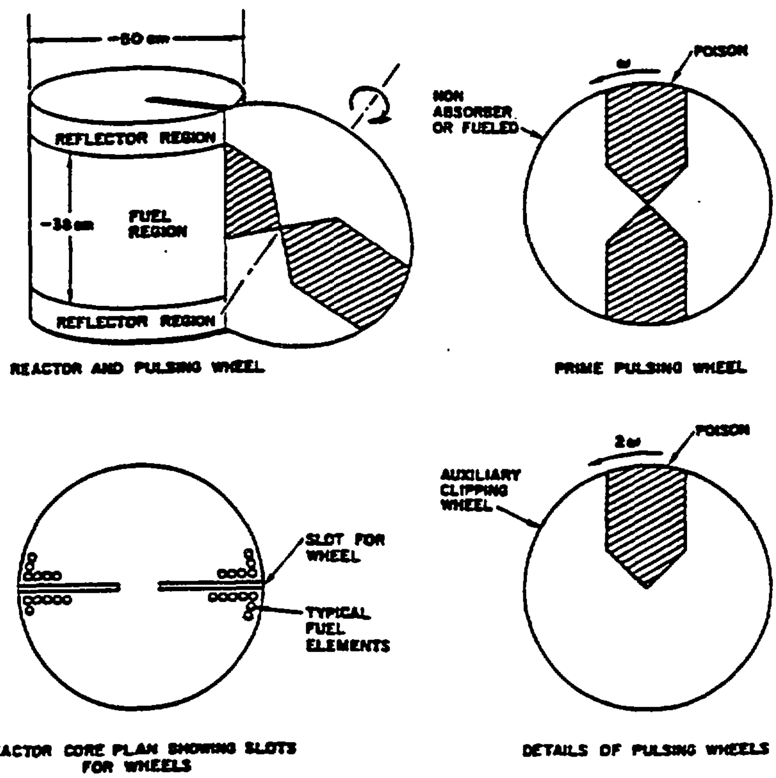

Figure 1: *A sketch showing the relation of a pulsing wheel and core. The arrangement of poison and non-poison is also shown for the prime pulsing wheel and a possible auxiliary "clipping" wheel.*

average power and pulse frequency. Although other selections have been made, the early study was aimed at providing a suitable system with an average power in the range of 4 to 7 megawatts with a pulse rate of 50 per second. Table 1 is a summary of the results of two different sets of calculations made with the General Atomics kinetics code BLOOST[15]. More recent evaluations have considered fewer pulses per second but with much larger peak power levels.

Table 1

Summary of Operational Characteristics for an Unoptimized System

	Uniform Pulsing and Clipping	Accelerated Clipping
Pulses per Second	50	50
Peak Power (MW)	40	29
Average Power (MW)	7	4
Minimum Power between Pulses	0.6	0.5
Pulse Width at half power (sec)	3×10^{-3}	2×10^{-3}
Energy in Pulse, MW-sec	0.14	0.080
Peak Thermal Leakage Flux in reflector (n/cm^2-sec)	0.5×10^{15}	0.36×10^{15}
Prompt Neutron Lifetime, μsec	12	12
Prompt Negative Temperature Coefficient ($\delta k/k°C$)	6×10^{-5}	6×10^{-5}
Average Fuel Temperature in 38 mm (9.4 mm) diam. fuel rods (°C)	365 (300)	355 (275)
Temperature Variation in Fuel During Pulse (°C)	~1	~1

3.3 **Reactivity Sequence**

The schedule for reactivity (positive and negative) insertions needed to provide the results of Table 1 is shown in Figure 2. The reactor is brought to a steady-state power with the slow insertion of \$3.00 excess reactivity (above cold critical). For one schedule of pulsing (shown by the solid curve of Figure 2 and corresponding to the uniform rate of insertion of +\$5.50 reactivity (above zero reactivity) followed by the uniform rate of insertion of -\$11.50 reactivity) a peak power of 40 megawatts is produced. If the insertion of negative reactivity is accelerated as shown by the dotted curve in Figure 2, the narrower pulse results in a 43 percent reduction in average power while the peak power is reduced by only 28 percent. Figure 3 shows the actual time dependence of the equilibrium pulses associated with these two sequences of reactivity insertions. The average core fuel temperature is about 300°C for the smaller diameter fuel proposed for this system. Since the smaller diameter fuel rods can accommodate larger peak power levels and somewhat increased average power still with acceptable fuel temperatures, a slight delay in the insertion of the -\$11.50 can increase the peak pulsed power levels to 100 MW or more. For larger peak powers (above ~100 MW), the average power would be increased to about 10 to 15 MW.

It may be noted that a calculation of the reactivity worth of the poison section of a rotating wheel has been made; its worth was calculated to be \$13.90, well in excess of the \$11.50 discussed above. A satisfactory design can be based on alternate sections of poison and non-poison. In this case, the reactor would have \$5.50 reactivity inserted with regular control rods. The rotation of the wheel would then produce the reactivity sequence shown by the solid curve in Fig. 2. If it is desired to clip the trailing edge of each pulse by accelerating the insertion of negative reactivity, this could be accomplished by the use of a second wheel with poison as shown in Fig. 1 and rotated at twice the speed.

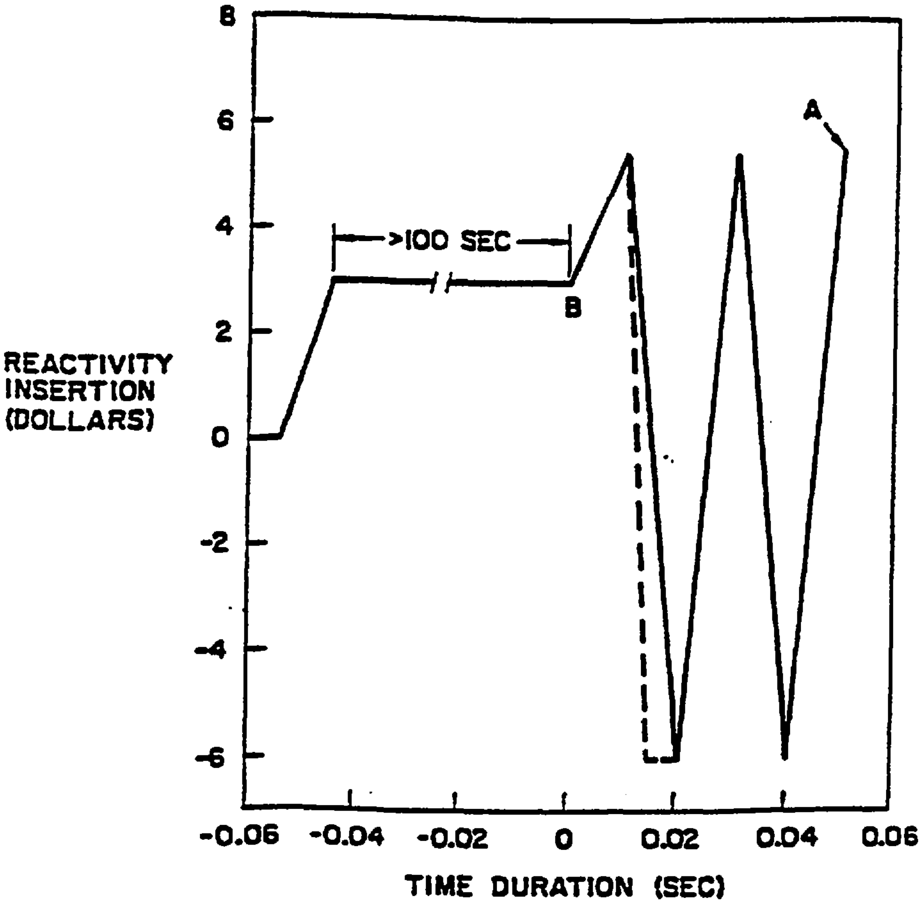

Figure 2: *A sketch showing the time dependence of reactivity insertions. Normally the oscillatory behavior is continued indefinitely. For accident analysis, the insertion is assumed to be halted at point A.*

An additional benefit accrues from the postulated use of a wheel which provides two pulses for each revolution. For 50 pulses per second, its rotational speed would be 25 rps or 1500 rpm, a relatively low speed. There should be no difficulty in obtaining the necessary mechanical strength of the rotating members which might have a diameter as large as 60 cm. Of course, an auxiliary wheel of about the same size used as described above would rotate at twice this speed.

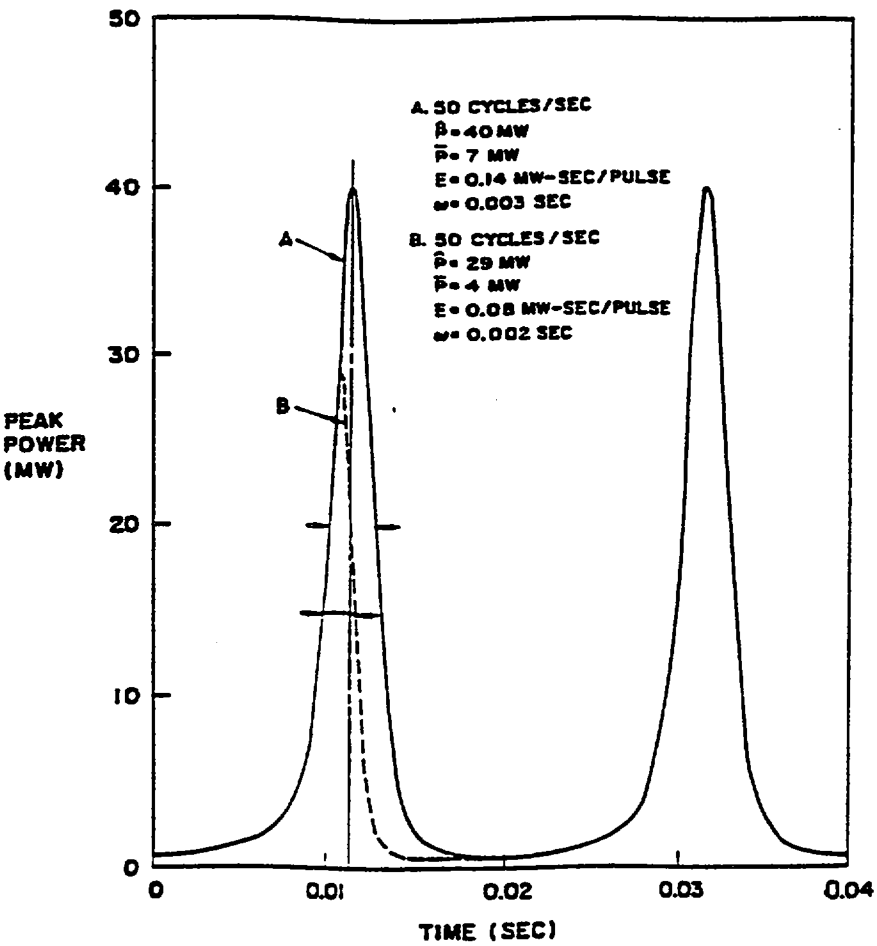

Figure 3: *The time dependence of the equilibrium pulses for a uniform rate of inserting positive and negative reactivity. The dotted curve results if the insertion of negative reactivity is accelerated.*

The potential for vibration and the hazards associated with a rotating wheel could be eliminated if the polarized ^3He control system with adequate reactivity and frequency response could be developed. At this time, the adequacy of a ^3He system has yet to be demonstrated for the proposed multiple pulsed TRIGA system.

Variations in the timing between the positive and negative reactivity sequence can increase the magnitude of the peak power. Decreasing the number of pulses per second and using the features provided by the second rotating wheel (as discussed above) can limit the average power level to about 10 MW. An average power level of about 10 to 15 MW is easy to cool and is well demonstrated for the TRIGA case[4]. The data in Table 2 give an idea of higher performance levels of the continuously pulsed TRIGA where the average power level is constrained to about 16 MW.

Table 2

Summary of Estimated Performance Levels for a Continuously Pulsed TRIGA Reactor

$\hat{P}$ (MW)	$\bar{P}$ (MW)	Flux at Core End of Beam Ports (n/cm^2.s)	Pulses per second
29	4	5.5×10^{14}	50
40	7	7.6×10^{14}	50
140	10	2.7×10^{15}	35
400	16	7.6×10^{15}	24

3.4 TRIGA LEU Characteristics.

The heat removal rate summarized in Table 1 and coolant flow characteristics for this TRIGA reactor are very similar to those for the Romanian 14 MW TRIGA reactor[4]. The Romanian reactor has demonstrated the robustness of the 9.4-mm diameter fuel. In operation for more than 13 years with more than 13000 MWD not one fuel rod released any fission products. This operation included two unplanned reactivity excursions[4], also with no fuel rod damage. Since the multiple pulsed TRIGA reactor as proposed herein will have essentially steady state characteristics (the fuel temperature varies only about 1°C during pulsing), the performance of this fuel for multiple pulsing is expected to provide the same excellent, long term performance.

The core life for the multiple pulsed TRIGA reactor will depend on the fuel type selected; that is, 30 wt-% or 45 wt-% uranium loading. Depending on the fuel type selected and the quantity of erbium burnable poison used, the core life can equal or exceed 5000 MWD before additional fuel will be required.

While considering temperature effects, it is worthwhile noting that the temperature coefficient does not provide the shutdown mechanism on a time scale suitable for the rapid pulsing rate. Before the temperature rise becomes significant for this purpose, a shutdown mechanism is provided by the mechanical addition of negative reactivity. However, the prompt negative temperature coefficient is still available as an inherent safety feature. The prompt neutron lifetime is calculated to be 12 μsec and the prompt negative temperature coefficient is calculated to be 6×10^{-5} $\Delta k/k/°C$. These are characteristics computed for a core composed of the smaller diameter, more highly loaded fuel rods.

3.5 Safety Considerations

Safety of the proposed system has been partially demonstrated by the performance of standard pulsed TRIGA reactors in which single reactivity insertions of +\$5.00 have been safely made in a cold core. The resulting peak energy release of ≅1.0 megawatt-sec per fuel element caused no indication of fuel element distress. In additional pulsing tests of fuel elements, the calculated peak temperature in some elements have safely reached as high as 1170°C with a hydrogen-to-zirconium ratio of about 1.65. Thus, a fuel-moderator element having a lower hydrogen-to-zirconium ratio of 1.58 to 1.60 can safely reach peak temperatures of ~1250°C. Additional demonstration of the safety of the core during pulsing is provided by the two unplanned reactivity excursions[4] in the 14 MW Romanian core with the 9.4-mm diameter fuel rods.

Since calculations have agreed well with the observed reactor parameters for the pulsed standard TRIGA, it is reasonable to place confidence in the same calculational techniques to compute the pulsing characteristics of the proposed system for a maximum credible reactivity insertion. The worst accident for the various systems of pulsing proposed above occurs when \$5.50 positive reactivity is inserted in the warm core (point A in Figure 2) and no subsequent negative reactivity is provided. Such a case would occur if the rotating wheel were to stop rather abruptly or to fly out of the core. A transient analysis indicates that the peak power and energy released in such an event would be 11,300 megawatt and 52 megawatt-sec, respectively, with a resulting momentary peak temperature in the hottest fuel element of about 1250°C, which falls quickly to a lower temperature especially since the regular control rods would drop into the core to terminate the described event. With a hydrogen-to-zirconium ratio of 1.58-1.6 for the proposed system, the equilibrium pressure corresponding to a peak temperature of 1250°C for the maximum credible reactivity insertion is no higher than already experienced in deliberate pulse tests. For comparison purposes, it may be further noted that the NSRR TRIGA reactor routinely produces peak power pulses of more than 20,000 MW with total energy release of more than 100 MW-sec[3].

Appropriate design parameters, including the hydrogen-to-zirconium ratio and the heat transfer characteristics of the core, can thus be selected to assure complete safety for the reactor system in this maximum contingency.

4. SUMMARY

Calculations have verified that a thermal reactor of the TRIGA type using U-ZrH fuel elements and light water coolant can be operated to give continuously many pulses per second. Peak pulses in the range 50 to more than 100 megawatts and peak thermal neutron leakage fluxes of nearly $\sim 10^{16}$ n/cm^2·s can be achieved. Pulsing can be accomplished with a wheel with relative ease since there is no requirement on precise mechanical location of the wheel within the core slot. A more simple pulsing system may be possible with a polarized ^{3}He control rod if adequate reactivity and frequency of polarization and depolarization can be developed. It is envisioned that the proposed reactor system would be constructed within a normal, above-ground concrete shield and would appear quite similar to a normal 5-10 MW

TRIGA reactor. To achieve the best signal-to-noise ratio for some neutron experiments as discussed earlier, a heavy water reflector would replace at least a part of the light water core reflector and be used to feed the core-end of tangential beam tubes. In addition, a cryogenic source of "cold" neutrons could be incorporated as desired for experiments. The reverse time-of-flight neutron scattering analysis system based on the Fourier chopper would be able to make singularly efficient use of the few millisecond duration pulses from the rapidly pulsed TRIGA reactor. Using the rapidly pulsed TRIGA reactor concept, it becomes possible to produce a competitive and intense source of neutrons for neutron beam research purposes, based on a reactor which possesses important elements of inherent safety and economy. It should be noted that specified frequency of pulses and peak power levels were chosen for illustrative purposes. A wide variety of alternative operating conditions can be chosen to meet specific user requirements.

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STATUS OF ANS CLOSEOUT ACTIVITIES

Presented to IGORR-IV Meeting

May 24, 1995

C. D. West

ANS CLOSEOUT ACTIVITIES (1)

- All design work was stopped (except where finishing the design was more efficient than documenting its incomplete status)

- Activities that will generate information useful to other DOE programs are being completed or being brought to a logical stopping point

  - completion of some R&D test series (where partial information would be of little or no use, but the whole series would be valuable)

  - documentation and publication of scientific and engineering data (to make useful results available)

  - complete development of some electronic tools for information management (needed to complete the documentation and archiving efficiently)

ANS CLOSEOUT ACTIVITIES (2)

- Information is being packaged for storage and potential use in the future
  - special efforts are being made to preserve easy access to the information that would be valuable to other DOE reactor or neutron source projects

COMPLETION OF ACTIVITIES "THAT MAKE SENSE" (1)

- Fuel fabrication and testing
  - statistical analysis of fuel plate defects
  - homogeneity studies
  - complete examination of specimens from HANS-3 irradiation capsules
  - evaluate techniques for centering graded fuel meat within the plate

- Al plate corrosion experiments
  - complete tests at low pH (<5.0)

COMPLETION OF ACTIVITIES "THAT MAKE SENSE" (2)

- Thermal hydraulics
  - — tests with wide-span plates
  - — measure effect of alternate flow blockage shapes
- Aluminum structural materials
  - — testing of specimens from HANSAL-T2 and -T3 irradiation
  - — complete aluminum irradiation creep tests
- Neutron beam handling
  - — test modular neutron beam polarization analyzer

COMPLETION OF ACTIVITIES "THAT MAKE SENSE" (3)

- Nearly completed software for
  - officially releasing documentation
  - storage and retrieval of information
  - managing technical interfaces
  - managing the design integration process

PACKAGING OF INFORMATION (1)

- Backup studies, calculations, etc. and incomplete material felt to be potentially useful for future projects

  - about 100 boxes of paper (about 1/2 million pages or 20 file cabinets) will be stored in the Federal Records Center in Atlanta, Georgia

  - A magnetic tape (about 2 Gbytes) for electronic files will be stored at an appropriate location to be determined

  - A listing of the material available and its completion status is being prepared

PACKAGING OF INFORMATION (2)

- Officially released research, development, and design documentation will be available on a set of three or four CD-ROM discs

  — documentation represents over $120 M of effort

  — includes over 500 journal articles, conference papers, design descriptions, and other documents (over 30,000 pages); over 200 engineering drawings, and several videos

  — software for reading the information is included on the discs

  — an index and some word search capability will be included

  — a listing of the backup material in the Federal Records Center will be included

  — test discs have been sent overseas and found to be readable

ANS CLOSEOUT STATUS

- All work is expected to be complete by September 1995

- Approximately 30% of backup and incomplete material is ready for shipment to the Federal Records Center

- Over 17,000 pages of CD-ROM material (that was already in electronic form) is ready now

  — test disks were sent to some people outside the U.S. (Bernard Farnoux reported that he was able to read the disk without problems)

  — the rest is being optically scanned right now

  — distribution arrangements are not yet finalized

    - all IGORR members will be informed

RESULTS OF A SURVEY ON ACCIDENT AND SAFETY ANALYSIS CODES, BENCHMARKS, VERIFICATION AND VALIDATION METHODS

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1. INTRODUCTION

During the "Workshop on R&D Needs" at the 3rd Meeting of the International Group on Research Reactors (IGORR-III), the participants agreed that it would be useful to compile a survey of the computer codes and nuclear data libraries used in accident and safety analyses for research reactors and the methods various organizations use to verify and validate their codes and libraries. The following organizations submitted information for this survey:

Atomic Energy of Canada Limited (AECL, Canada),
China Institute of Atomic Energy (CIAE, Peoples Republic of China),
Japan Atomic Energy Research Institute (JAERI, Japan),
Oak Ridge National Laboratories (ORNL, USA), and
Siemens (Germany).

2. DEFINITION OF BENCHMARK, VERIFICATION AND VALIDATION

In their submissions the various organizations refer to "benchmark" methods and calculations, "validation" work, and "verification" for computer codes and libraries. The authors of this survey have attempted to compile a consistent survey by applying a consistent definition to those terms:

Verification: confirms that the intended equations, initial conditions, and boundary conditions are correctly programmed and perform as intended.

Validation: confirms, via comparison to available measurements, that the equations as programmed capture reality with a sufficient degree of fidelity.

Benchmark: a standard problem set with known or mutually agreed upon results used to verify a computer code, or a standard set of measured data used to validate a given application of the computer code.

3. NATIONAL STANDARDS

Several organizations submitted information about their national standards for software quality assurance and examples of how those standards are implemented for specific research reactor projects:

Canada: AECL is currently implementing a software quality assurance (SQA) program based on the requirements set out in the Canadian Standards Association (CSA) N286.7-94 standard [1]. This standard covers the development of new software, the use of existing software, and the modification of existing software, where such software is used in support of safety related nuclear systems. The term software includes the encoding of correlations and mathematical methods and the data input to the models. Each specific nuclear project is required to develop and implement a project-specific quality assurance plan that encompasses all activities within the project. In addition to providing procedures for a SQA program, all design calculations or calculations to provide input to design are executed in accordance with design procedures based on CSA N286.2-86 [2]. As a specific example, the Research-Reactor Technology Branch (RTB) in AECL has implemented a SQA program [3] for the computer codes, data libraries and input models used to analyze research reactor concepts such as the proposed Irradiation Research Facility [4].

China: The submission [5] from the CIAE stated that China issued the Nuclear Industry Standards, EJ/T617-91, "A Guide to Verification and Validation for Computer Software Codes in Nuclear Industry Science and Engineering," in 1991. This standard is equivalent to the American National Standards, ANSI/ANS 10.4-1987. Implementation of a SQA program for verification and validation of computer codes is at an early stage.

Germany: The submission [6] from Siemens did not mention any specific standard for SQA. However, two computer software systems for performing nuclear design calculations, MARS and RSYST, are described. The MARS system was developed at Siemens/INTERATOM, whereas the RSYST system was developed at IKE-Stuttgart and at the Computer Centre at the University of Stuttgart. The MARS system and two versions, RSYST-I and RSYST-III, of the RSYST system have been used for the nuclear design of the FRM-II. Two different code systems were used to provide a broad verification of the nuclear design of the FRM-II.

Japan: The Japanese did not indicate any specific standard for SQA [7] is in use. However, JAERI uses a standard neutronic code system, SRAC (Standard Thermal Reactor Nuclear design code system) [8], for any type of thermal reactors.

USA: At ORNL, the ANS (Advanced Neutron Source) Project has implemented a SQA program [9] based principally on the requirements of Supplements 3S-1 and 11S-2 of the NQA-1 standard [10] and of Part 2.7 of the NQA-2 standard [11]. In addition, the ANS Project is committed to being judged licensable under the standards applied by the US Nuclear Regulatory Commission (NRC).

4. METHODS AND CODES USED IN ACCIDENT AND SAFETY ANALYSIS

4.1 Computer Codes and Methods for Static Neutron Physics Calculations

Each organization participating in the survey provided information on their computer codes for performing:

- Cell calculations: These codes are used to perform spectral calculations in the cells and to produce condensed few-group constants, macroscopic absorption and fission cross sections, and macroscopic reaction rates for use in the core calculations. Two calculational methods are generally used, discrete ordinates transport theory and the collision probability form of the transport equation.

- Core calculations: Three calculational methods are generally used, diffusion theory, discrete ordinates transport theory and Monte Carlo theory, to solve the Boltzmann transport equation. The CIAE also use the nodal method to calculate criticality, flux and power distributions, and reactivity coefficients.

The computer codes and methods are listed in Table 1. The RSYST code system [6] contains a sequence of modules for microscopic library compilation, macroscopic constant generation, performing spectral cell calculations. This has been represented in Table 1 by referring to RSYST rather than including the names of the specific modules. The same has been done for the AMPX/SCALE [12,13] code system. The WIMS-AECL [14] and WIMS-D4 codes use both discrete ordinates transport theory and the collision probability form of the transport equation.

Table 1: Summary of Computer Codes and Methods

METHOD	AECL	CIAE	JAERI	ORNL	Siemens
Cell calculation	WIMS-AECL	WIMS-D4		AMPX/ SCALE	RSYST, MONSTRA
- collision probability			PIJ		
- transport		PASC-1	ANISN TWOTRAN [22]		
Core calculation - diffusion	3DDT [15]	CITATION [19] EXTERMI-NATOR-2 [20]	CITATION	VENTURE [23]	DIF1D, DIF2D DIXY
- Monte Carlo	MCNP [16] KENO [13]	MCNP		MCNP, KENO	MORSE-K MOCA
- transport	DANTSYS [17,18]	ANISN [21] DOT3.5	ANISN TWOTRAN	DORT [24]	IANISN, SN1D DOT
- nodal method		PSUI-LEOPARD/ NGMARC			

As shown in Table 2, the key core performance parameters are generally calculated using different methods to provide independent verification of the results. The only exception is in the case of fuel depletion calculations where only diffusion theory is used to estimate the core burnup.

Table 2: Summary of the Codes Used to Calculate key Physics Parameters

Parameter	AECL	CIAE	JAERI	ORNL	Siemens
k-effective	3DDT MCNP	MCNP ANISN DOT3.5 PSUI- LEOPARD/ NGMARC	CITATION TUD TWOTRAN	MCNP KENO DORT	MORSE-K MOCA DOT
reactivity worth	3DDT MCNP DANTSYS	MCNP	ANISN TWOTRAN	MCNP	MORSE-K MOCA DOT
reactivity coefficients	3DDT	CITATION ANISN DOT3.5	CITATION TUD ANISN TWOTRAN	VENTURE DORT	DIF2D
flux and power distributions	3DDT MCNP	CITATION MCNP ANISN DOT3.5	CITATION TUD ANISN TWOTRAN	VENTURE MCNP DORT	DIF2D MORSE-K DOT
fuel depletion	3DDT/ FULMGR	CITATION/ 2DFGD EXTERMI- NATOR-2	CITATION TUD	VENTURE / BURNER	RSYST MARS

Most of the computer codes listed in Tables 1 and 2 have a long history of applications in many projects. Nevertheless, the SQA programs for many recent research reactor projects (e.g., ANS and IRF) require that the computer codes used for design calculations and safety analyses be verified and validated for the specific applications. Verification of the computer codes are addressed as follows:

- AECL relies on benchmark problems, inter-code comparisons and verification reports from code maintainers; the software includes in-house development (e.g., WIMS-AECL) and international sources (e.g., 3DDT, MCNP and DANTSYS),
- CIAE relies on software obtained from international sources (e.g., RSIC, NESC, NEA),
- JAERI has verified SRAC system using international benchmark problems and critical experiments,

- ORNL relies on verification reports from code developers (e.g., ORNL, LANL) for the ANS Project, and

- Siemens relies on inter-code comparisons between the RSYST and MARS systems for FRM-II.

Validation of the computer codes are addressed as follows:

- **AECL**: Code validation relies on comparisons against benchmark problems, inter-code comparisons and comparisons against critical experiments. For the current work on research reactor projects (e.g., IRF) in AECL, RTB has been undertaking a validation program for the set of computer codes routinely used to perform design calculations and safety analyses. A validation report [25] has been produced to compile information pertaining to comparisons of WIMS-AECL predictions against:

  - CANDU-type fuel assemblies in a variety of coolant types (e.g., D_2O, H_2O, D_2O/H_2O mixtures, void and organic) using the ZED-2 critical facility,
  - burnup and isotope depletion data from CANDU fuel bundles discharged from the Bruce and Pickering power reactors, and
  - the R1/100H lattice experiments for H_2O coolant temperature and density effects.

  The WIMS-AECL/3DDT code set has also been validated [25] against the SPERT-1B reactor experiments for k-effective, reactivity coefficients and kinetics parameters, and the TRX experiments for k-effective. A similar report has compiled validation data for MCNP [26]. For example, MCNP has been validated against commissioning data from the SLOWPOKE Demonstration Reactor for k-effective and gamma dose rates [27]. The need for further validation work will depend on the specific requirements of a research reactor project.

- CIAE relies on IAEA benchmarks.

- JAERI relies on IAEA benchmarks (e.g., IAEA 10 MW Benchmark Reactor [28]) and critical experiments and JRR-3 commissioning data. The information from JAERI indicated that the SRAC system had been validated against many experiments (e.g., Tank-type Critical Assembly for light-water reactors, Deuterium Critical Assembly for the Advanced Thermal Reactor for H_2O-cooled and D_2O-moderated reactors, Semi-Homogeneous Experimental facility for 20 wt% enriched uranium in a graphite moderator, JMTRC critical facility for JMTR, TRX experiments and a series of FBR cores). Comparisons have also been made against international benchmarks developed for the RERTR (**R**educed **E**nrichment for **R**esearch and **T**est **R**eactor) program (e.g., LEU inital core for the Ford Nuclear Reactor), and temperature and void coefficient measurements in KUCA.

- ORNL has validated their computer codes for the ANS Project against:

- Los Alamos critical mass data for enriched uranium in bare H_2O- and D_2O-reflected critical experiments, ORNL H_2O-solution critical experiments, and D_2O-moderated, natural uranium ZEEP critically-buckled lattices,
- FOEHN critical experiments [29] to validate predictions from MCNP, VENTURE/BURNER, DORT and KENO,
- ANS critical experiments to supplement validation from the FOEHN experiments, and
- HFIR and ILL operating data to validate the fuel depletion calculations.

- Siemens has commissioning data from RSG-GAS-30 (Indonesia) for validation.

4.2 Nuclear Data Libraries

The survey identified the list of nuclear data libraries listed in Table 3 are being used for physics calculations. For the ANS Project, a dedicated multigroup nuclear library, ANSL-V, was prepared from the ENDF/B-V library, and validated [7]. Verification and validation of the nuclear data libraries are combined with the verification and validation of the codes.

Table 3: Summary of Nuclear Data Libraries and Computer Codes

LIBRARY	AECL	CIAE	JAERI	ORNL	Siemens
CSRL-IV		PASC-1			
ENDF/B-IV	KENO	MCNP	ANISN, PIJ TWOTRAN		MONSTRA CGM
ENDF/B-V	WIMS- AECL MCNP			AMPX/SCALE DORT VENTURE/ BURNER MCNP, KENO	CGM
ENDF/B-VI	WIMS- AECL				
GAM/ THERMOS		PASC-1			
JEF1					CGM
JENDL-2			PIJ, TWOTRAN		
VITAMIN-C		PASC-1			
WINFRITH	WIMS- AECL	WIMS-D4			

4.3 Computer Codes and Methods for Thermalhydraulic and Transient Analysis

The information from the participants in the survey identified the following thermalhydraulics codes in use for accident analyses:

AECL: CATHENA [30] is a two-fluid (6 equation) code used for the dynamic simulation of reactor transients involving thermalhydraulics and kinetics. It was originally developed for the fluid conditions in a CANDU reactor and subsequently modified for use with MAPLE-X10 coolant conditions. Heat transfer correlations for the MAPLE-X10 coolant conditions were obtained from heat transfer experiments using electrically-heated fuel-element simulators in a flow test rig. Work is in progress to extend those heat transfer correlations to cover the expected coolant conditions for the IRF.

CIAE: The information from the CIAE indicated that they are using a code called THAS-PC4, modified from COBRA-IV, to perform steady-state thermalhydraulic analyses for the CARR Project. Thermalhydraulic accident analyses for the CARR Project will be performed using RETRAN-02.

JAERI: JAERI used HEATING-5 [31], EUREKA-2 [32] and THYDE-P [33] for thermalhydraulic analyses for the upgraded JRR-3 reactor. HEATING-5 is designed to solve steady-state and transient heat conduction problems in one-, two-, or three-dimensional Cartesian or cylindrical coordinates. EUREKA-2 provides a coupled thermal, hydraulic and point kinetics capability for evaluating a postulated reactivity initiated transient. THYDE-P is designed to analyze anticipated operational transients and accident conditions in light-water power reactors. JAERI modified the heat transfer correlations and the DNB (departure from nucleate boiling) correlations [34] for the thermalhydraulic design and safety analysis of the upgraded JRR-3.

ORNL: For the ANS Project, RELAP5 [35] is used for the thermalhydraulic design of the cooling systems. RELAP5 has been verified by INEL (Idaho National Engineering Laboratory), and the ANS Project planned to validate it against HFIR, the Thermal-Hydraulic Test Loop [36] and a planned integral test facility. CONQUEST is planned ti be used for reactivity initiated transients. It has been verified against IAEA benchmarks and it was planned to validate CONQUEST against measurements frm the planned ANS critical facility.

5. SUMMARY

This report is a compilation of the information submitted by AECL, CIAE, JAERI, ORNL and Siemens in response to a need identified at the "Workshop on R&D Needs" at the IGORR-III meeting. The survey compiled information on the national standards applied to the SQA programs undertaken by the participnats. Information was assembled for the computer codes and nuclear data libraries used in accident and safety analyses for research reactors and the methods used to verify and validate the codes and libraries. Although the survey was not comprehensive, it provides a basis for exchanging information of common interest to the research reactor community.

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AECL EACL

RESULTS OF A SURVEY ON ACCIDENT AND SAFETY ANALYSIS CODES, BENCHMARKS, VERIFICATION AND VALIDATION METHODS

4th Meeting of the
International Group on Research Reactors
Gatlinburg, Tennessee
1995 May 24-25

BACKGROUND

- Identified as useful at IGORR-III
- C. West sent out invitations for information
- A. Lee agreed to compile information
- 5 responses:
  - AECL
  - CIAE
  - JAERI
  - ORNL
  - Siemens

DEFINITIONS

- **Verification:** confirms that the intended equations, initial conditions, and boundary conditions are correctly programmed and perform as intended.
- **Validation:** confirms, via comparison to available measurements, that the equations as programmed capture reality with a sufficient degree of fidelity.
- **Benchmark:** a standard problem set with known or mutually agreed upon results used to verify a computer code, or a standard set of measured data used to validate a given application of the computer code.

SOFTWARE QA STANDARDS

- **AECL** implementing SQA program for software used in support of safety related nuclear systems
  - CSA N286.7-94 covers development of new software, use of existing software, and modification of existing software
  - project-specific procedures (e.g., IRF)
- **CIAE** implementing SQA program under
  - Nuclear Industry Standards, EJ/T617-91
  - equivalent to ANSI/ANS 10.4-1987
- **JAERI** has no specific standard required by regulators
- **ORNL** specific SQA program for ANS Project
  - Supplements 3S-1 and 11S-2 of NQA-1 and Part 2.7 of NQA-2
- **Siemens** did not provide information

Computer Codes for Cell Calculations

AECL	CIAE	JAERI	ORNL	Siemens
WIMS-AECL	WIMS-D4	PIJ ANISN TWOTRAN	AMPX/SCALE	RSYST MONSTRA

Performs spectral calculations in cells and produces condensed few-group , macroscopic absorption and fission cross sections, and macroscopic reaction rates

Computer Codes for Core Calculations

Parameter	AECL	CIAE	JAERI	ORNL	Siemens
k-effective	3DDT MCNP	MCNP ANISN DOT3.5 PSUI- LEOPARD/ NGMARC	CITATION TUD TWOTRAN	MCNP KENO DORT	MORSE-K MOCA DOT
reactivity worth	3DDT MCNP DANTSYS	MCNP	ANISN TWOTRAN	MCNP	MORSE-K MOCA DOT
reactivity coefficients	3DDT	CITATION ANISN DOT3.5	CITATION TUD ANISN TWOTRAN	VENTURE DORT	DIF2D
flux & power distributions	3DDT MCNP	CITATION MCNP ANISN DOT3.5	CITATION ANISN TWOTRAN	VENTURE MCNP DORT	DIF2D MORSE-K DOT
fuel depletion	3DDT/ FULMGR	CITATION/ 2DFGD EXTERMI- NATOR-2	CITATION TUD	VENTURE/B URNER	RSYST MARS

NUCLEAR DATA LIBRARIES AND CODES

LIBRARY	AECL	CIAE	JAERI	ORNL	Siemens
CSRL-IV		PASC-1			
ENDF/B-IV	KENO	MCNP	ANISN PIJ TWOTRAN		RSYST MONSTRA
ENDF/B-V	WIMS-AECL MCNP KENO			AMPX/ SCALE DORT MCNP KENO	RSYST
ENDF/B-VI	WIMS-AECL				
GAM/ THERMOS		PASC-1			
JEF1					RSYST
JENDL-2			PIJ TWOTRAN		
VITAMIN-C		PASC-1			
WINFRITH	WIMS-AECL	WIMS-D4			

VERIFICATION

- AECL relies on benchmark problems, inter-code comparisons and verification reports from code maintainers; software is a mix of in-house development and international sources (RSIC, LANL)
- CIAE relies on software obtained from international sources (RSIC, NESC, NEA)
- JAERI has verified SRAC system using international benchmark problems and critical experiments
- ORNL relies on verification reports from code developers (ORNL, LANL) for the ANS Project
- Siemens relies on inter-code comparisons between the RSYST and MARS systems for FRM-II

VALIDATION

- AECL relies on benchmark problems and comparisons against critical experiments (FOEHN, ZED-2 lattices, TRX, SPERT-1B)
- CIAE relies on IAEA benchmarks
- JAERI relies on IAEA benchmarks and critical experiments (TCA, DCA, SHE, JMTRC, KUCA, LEU core for Ford Nuclear Reactor) and JRR-3 commissioning data
- ORNL has validated codes for the ANS Project against the FOEHN critical experiment and plans to extend the validation with ANS critical experiments
- Siemens has commissioning data from RSG-GAS-30 (Indonesia) for validation

BENCHMARKS

- FOEHN critical experiment (D_2O, beam tubes, HEU)
- ZED-2 lattices (D_2O, D_2O/H_2O, variety of uranium fuels, void coefficient)
- TRX (H_2O, LEU metal and oxide cores)
- SPERT-1B (H_2O,reactivity coefficient, kinetics parameters)
- IAEA 10 MW Benchmark Reactor (H_2O, HEU)
- TCA (H_2O, oxide cores)
- DCA (D_2O, oxide cores)
- SHE (LEU, graphite moderated)
- JMTRC (H_2O, MEU)
- KUCA (coolant temperature and void coefficients)
- LEU core for Ford Nuclear Reactor (H_2O)
- JRR-3 commissioning data (H_2O coolant, D_2O reflector)
- RSG-GAS-30 commissioning data (H_2O, beryllium reflector)

THERMALHYDRAULICS AND TRANSIENT ANALYSIS CODES

- **AECL:** CATHENA
  - 6-equation, 2-fluid code with coupled point kinetics
  - heat transfer correlations obtained from experiments
- **CIAE:** separate codes for steady-state and transient analyses
  - THAS-PC4 for steady-state analysis
  - RETRAN-02 for transient analysis
- **JAERI:** separate codes for steady-state and transient analyses
  - HEATING-5 for heat conduction problems
  - EUREKA-2 for thermalhydraulic and point kinetic transient analysis
  - THYDE-P for operational transients; heat transfer correlations obtained from experiments
- **ORNL:** separate codes for steady-state and transient analyses
  - RELAP5 for steady-state analysis
  - CONQUEST for reactivity transients
  - heat transfer experiments planned

XA04C1690

Selected Thermal and Hydraulic Experimentation in Support of the Advanced Neutron Source Reactor

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Gatlinburg, Tennessee

Outline

- Thermal Hydraulic Limit Testing
- Fuel Plate Stability Testing
- Flow Blockage Testing

The ANS Reactor Has Unique Thermal-Hydraulic Characteristics in Comparison to Other Research and Commercial Reactors

- Heavy water coolant
- Parallel Rectangular channels (involute)
- Very small channel gap (1.27 mm)
- Very high velocity (25 m/s)
- Very high exit subcooling (110°C)
- Moderately high heat flux (5.9 MW/m^2 average and 12 MW/m^2 maximum)
- High average power density (4.5 MW/L)
- Large L/D (200)

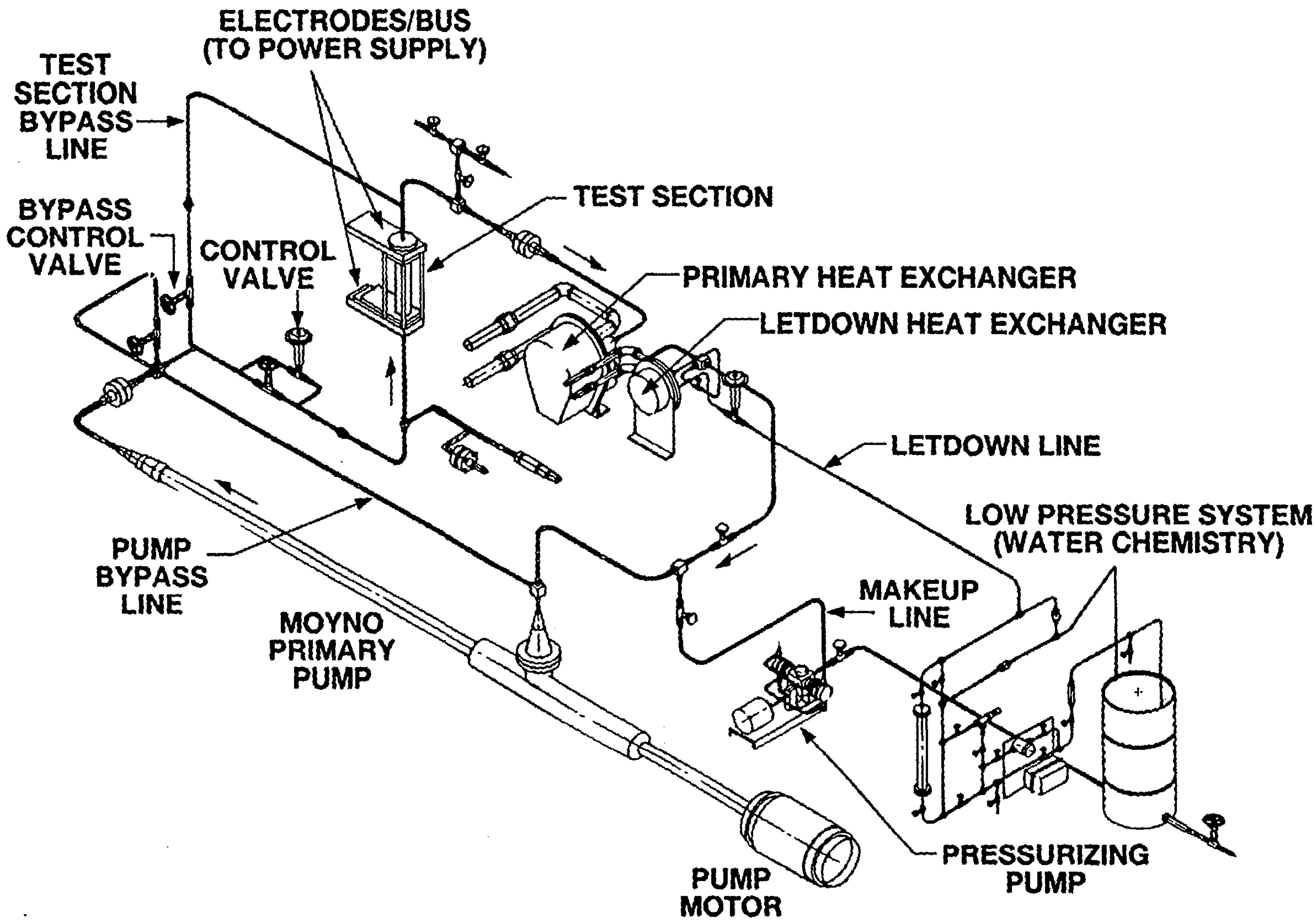
ORNL-DWG 95-2310 ETD

Advanced Neutron Source

Thermal Hydraulic Testing

Objective: To determine experimentally the appropriate core thermal hydraulic limits at ANS conditions.

Advanced Neutron Source (ANS) Thermal Hydraulic Test Loop (THTL) Was Designed to Operate in "Stiff", "Soft" and "Modified Stiff" Modes

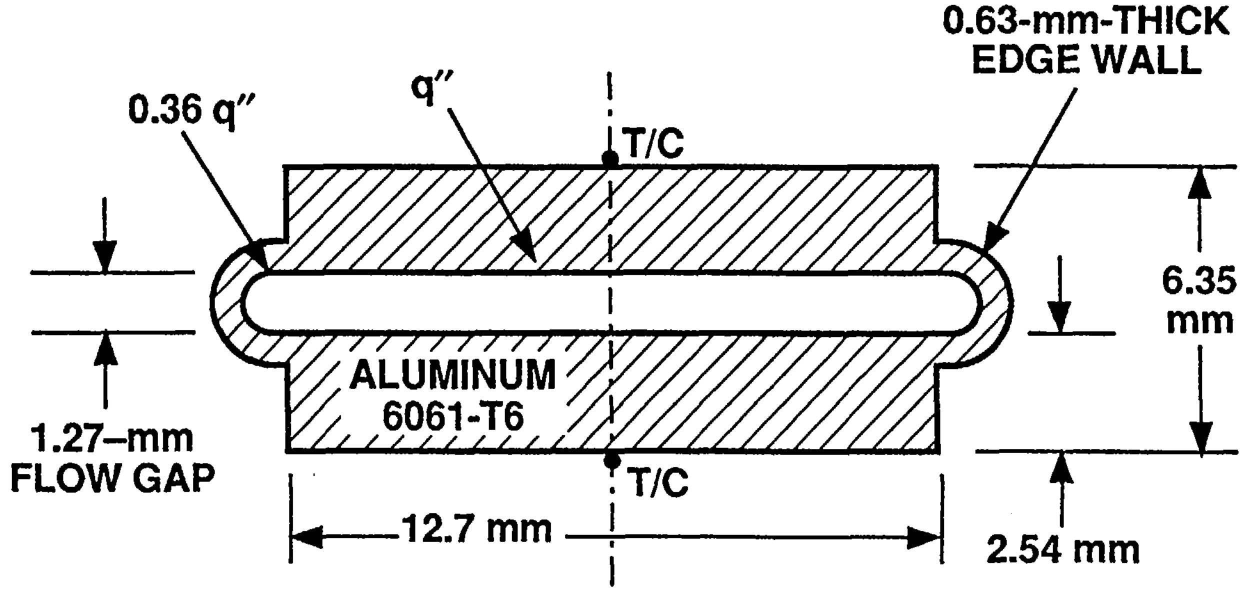


ORNL-DWG 95-2311 ETD

Advanced Neutron Source

Direct Resistance Heating of the Rectangular Cross-Section of the Channel in the Thermal Hydraulic Test Loop Provided Some Challenges

- 1.27–mm x 12.7–mm rectangular channel
- Full length - 507 mm
- Directly heated using dc current



(WIDE CHANNEL: 24.4 mm)

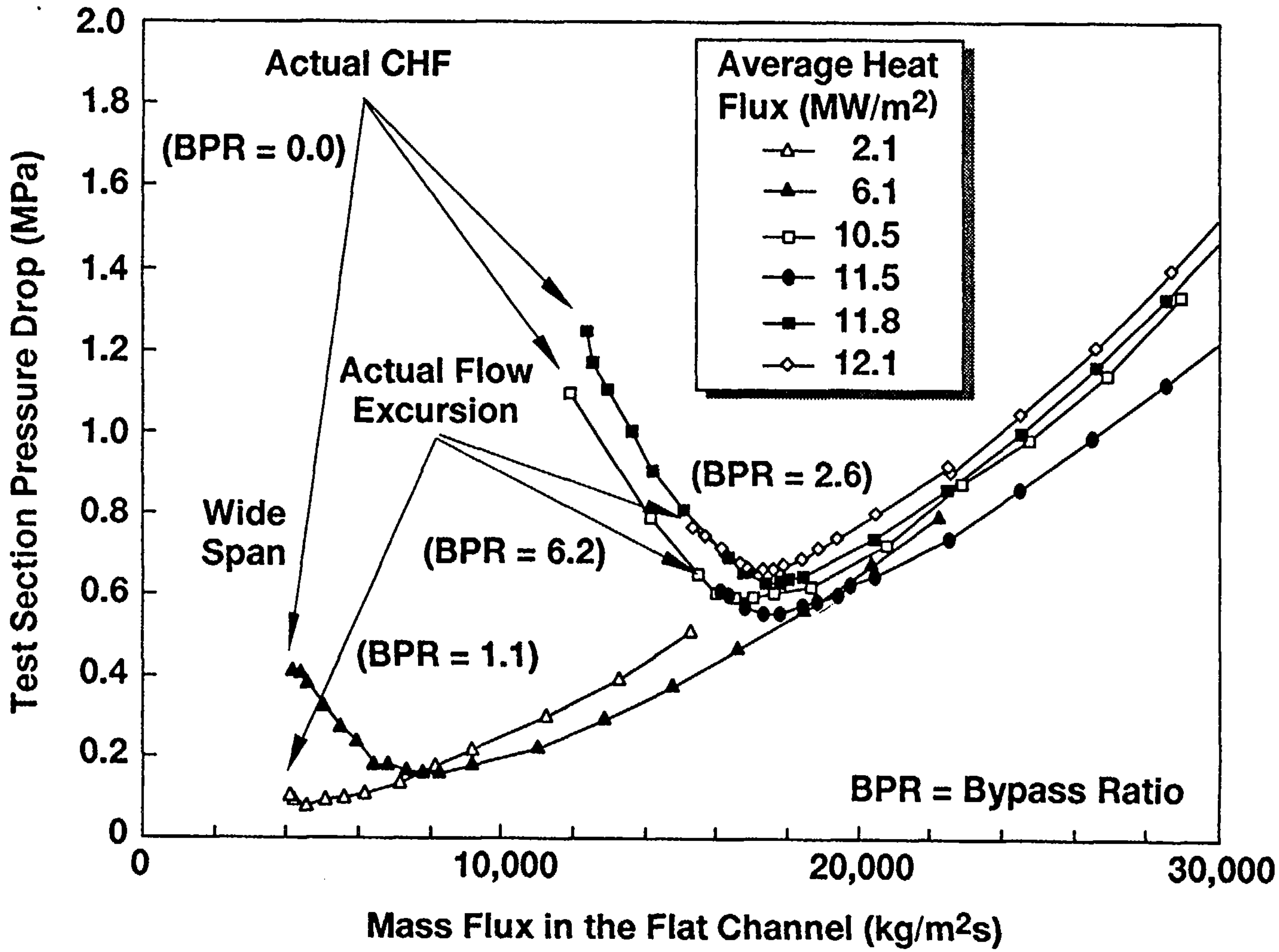
ORNL-DWG 93M-3501A3 ETD

Advanced Neutron Source

Range of Flow Excursion Tests Performed Is Beyond Any Data Range Previously Available

- Coolant: Water
- Inlet coolant temperature: 45 and 40^{o}C
- Exit coolant pressure: 1.7 (and 0.45, 0.17) MPa
- Exit heat flux range: 0.7–18 MW/m2
- Corresponding exit velocity range: 2.8 – 28.4 m/s
- Channel configuration: rectangular, 1.27 X 12.7 X 507 mm, aluminum

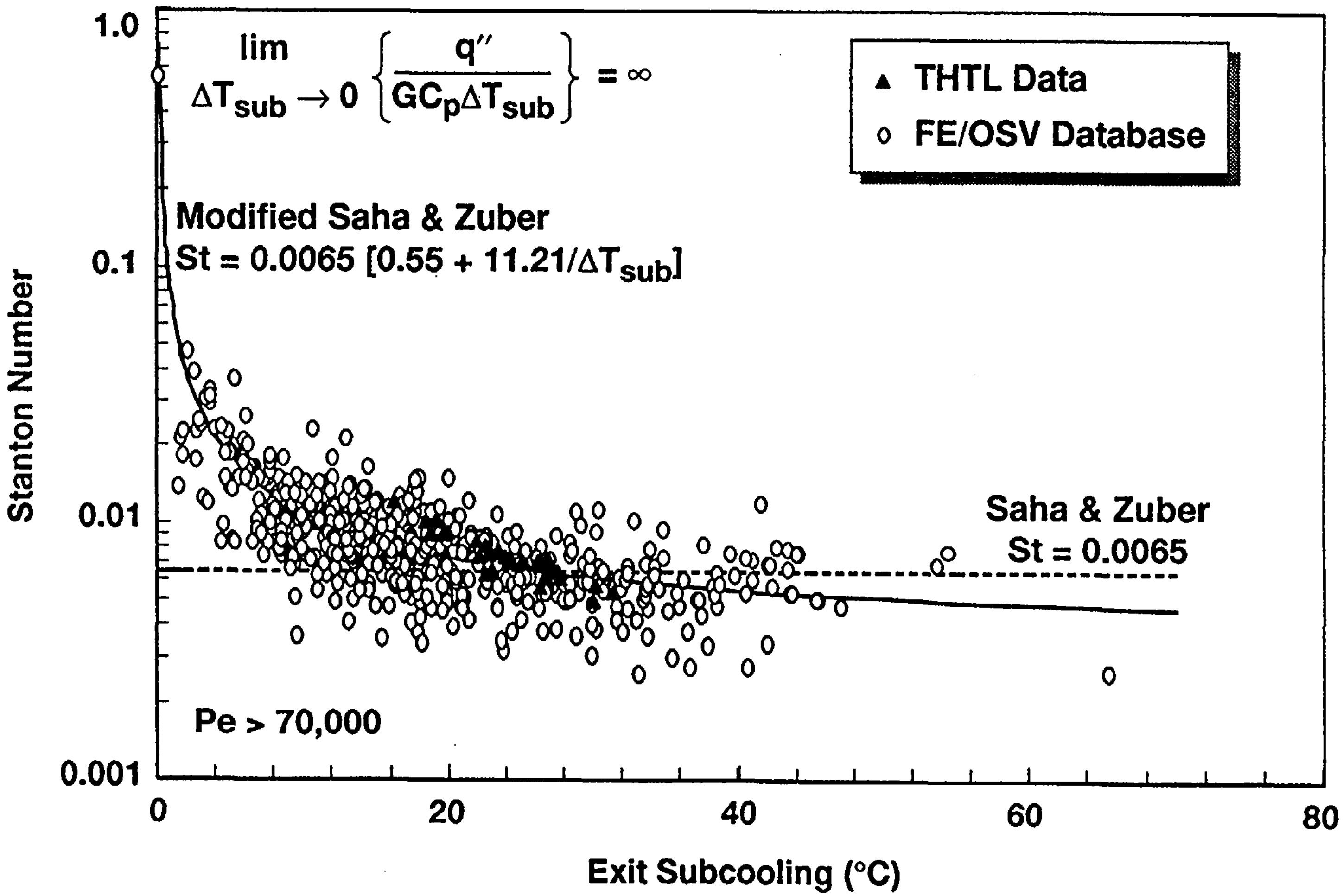
Destructive CHF Tests Performed in a "Stiff" System Showed a 30% Additional Margin in Critical Velocity Compared to the Flow Excursion Velocity (Minimum Pressure Drop). Bypass Flow Ratio (BPR) Does Effect the Point of Destructive Flow Excursion.



ORNL-DWG 95-2452 ETD

Advanced Neutron Source

The Modified St Number Correlation Compares Well With the Data Trends and Is Consistent With Its Definition. The Extreme Data Point at Very Low Subcooling Strongly Supports this Conclusion.



ORNL-DWG 95-2455 ETD

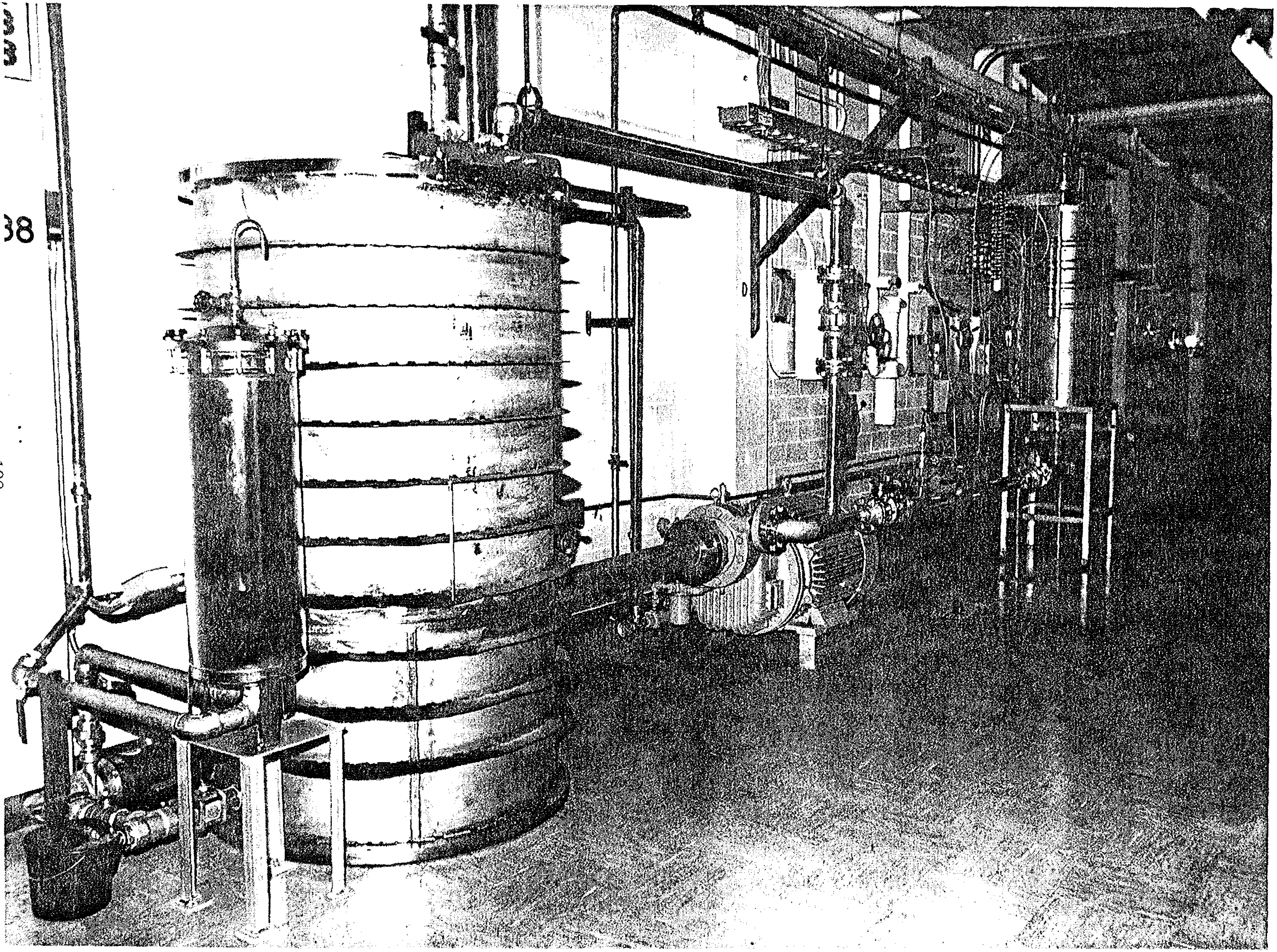
Advanced Neutron Source

Summary of Thermal Hydraulic Limit Testing and Analysis

- FE data has been acquired at ANS typical flow velocities
- An extensive OSV/OFI data base has been developed with a very broad parameter range
- A modification of the Saha-Zuber correlation was proposed to account for reduced subcooling effects
- Closeout activities include continued investigation of wider span test channels
- Some testing for HFIR will be performed to evaluate the effect of reduced channel gap
- Future plans called for additional testing at 3-core conditions, hot spot testing, etc.

Fuel Plate Stability Testing

Objective: To experimentally evaluate the structural response of ANS fuel plates to hydraulic loads.



EXPERIMENTAL FLOW LOOP

EPOXY INVOLUTE PLATE TEST SECTION

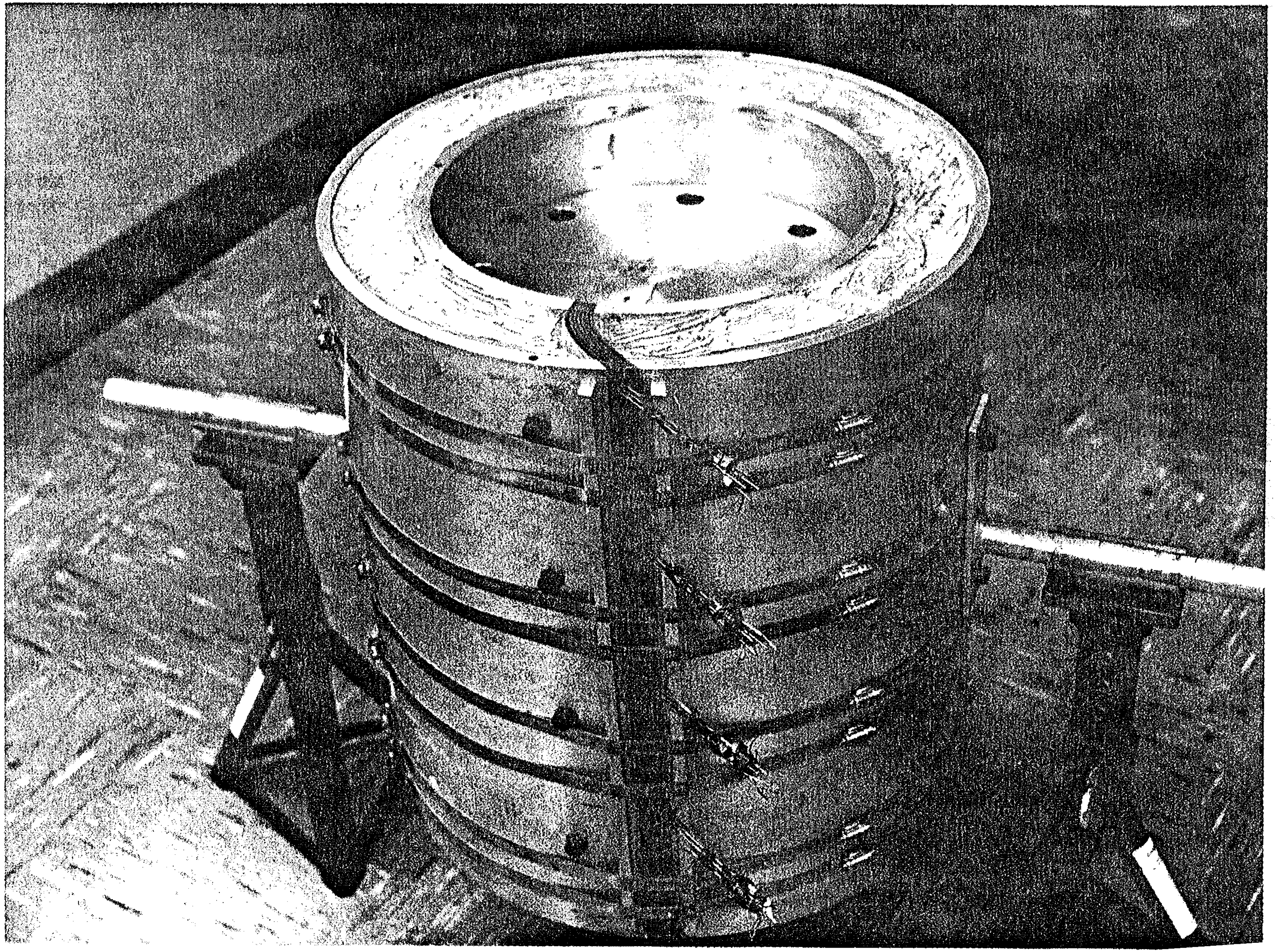
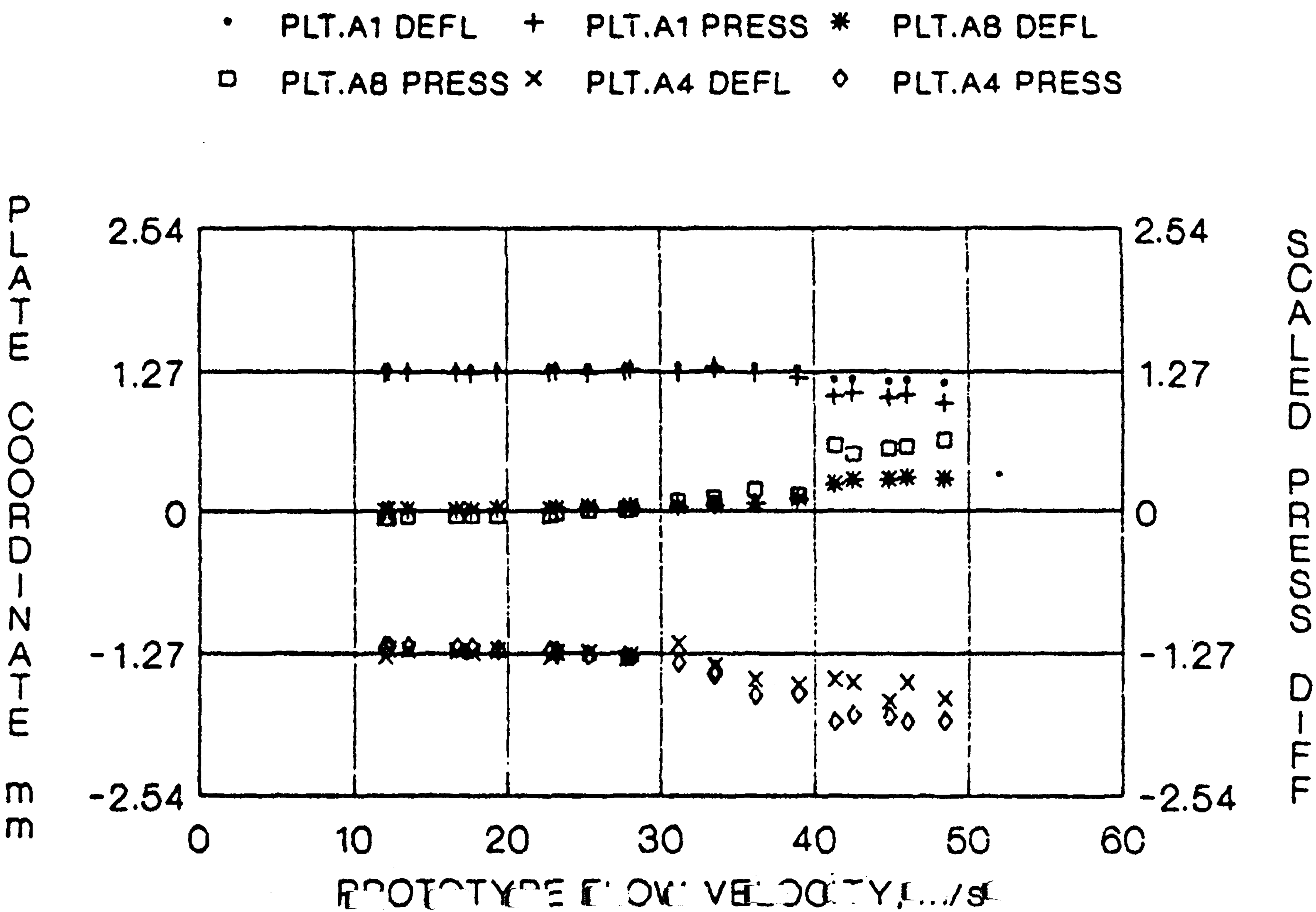


Plate Deflection Was Found to be Proportional to the Pressure Load

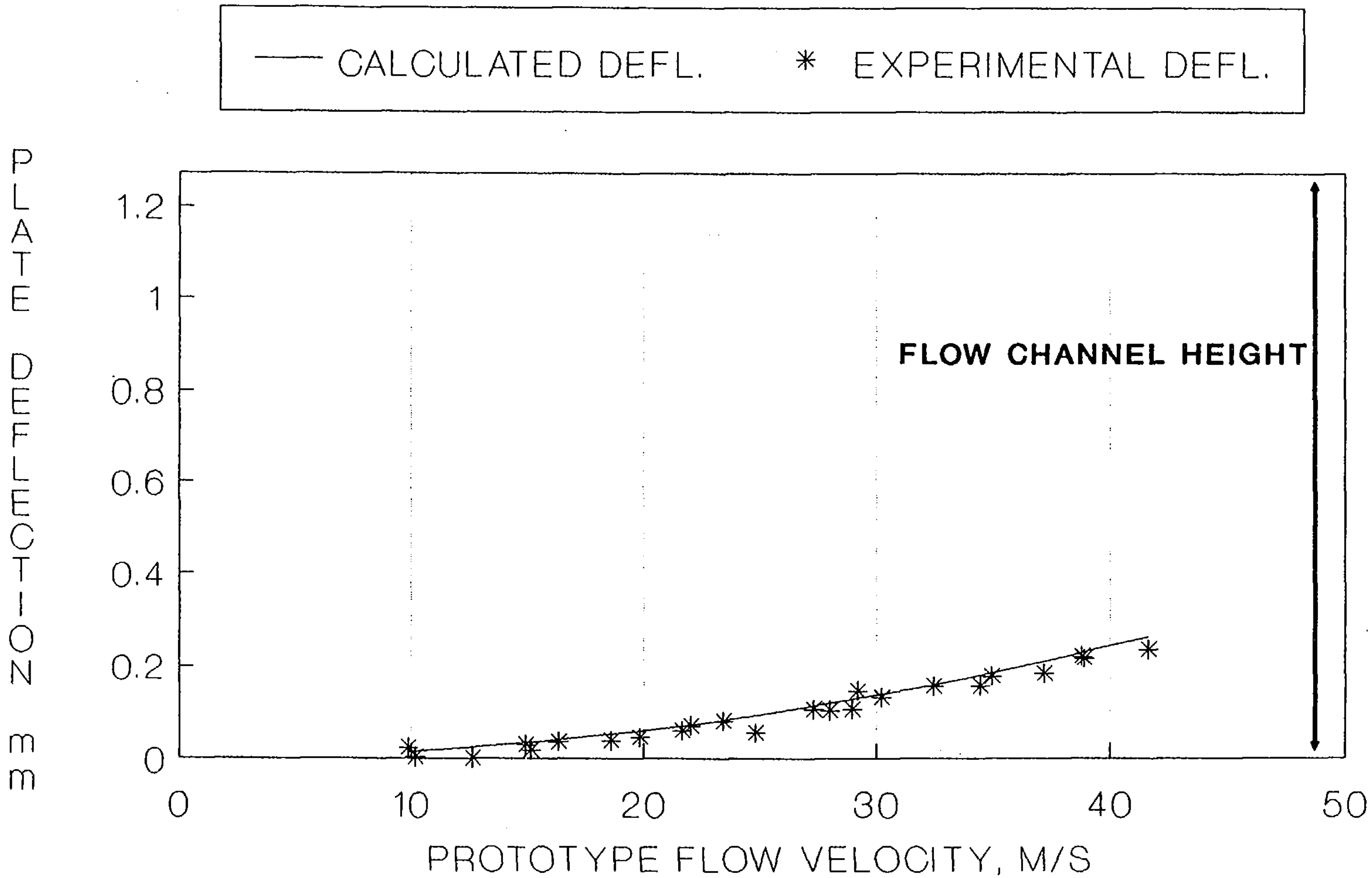


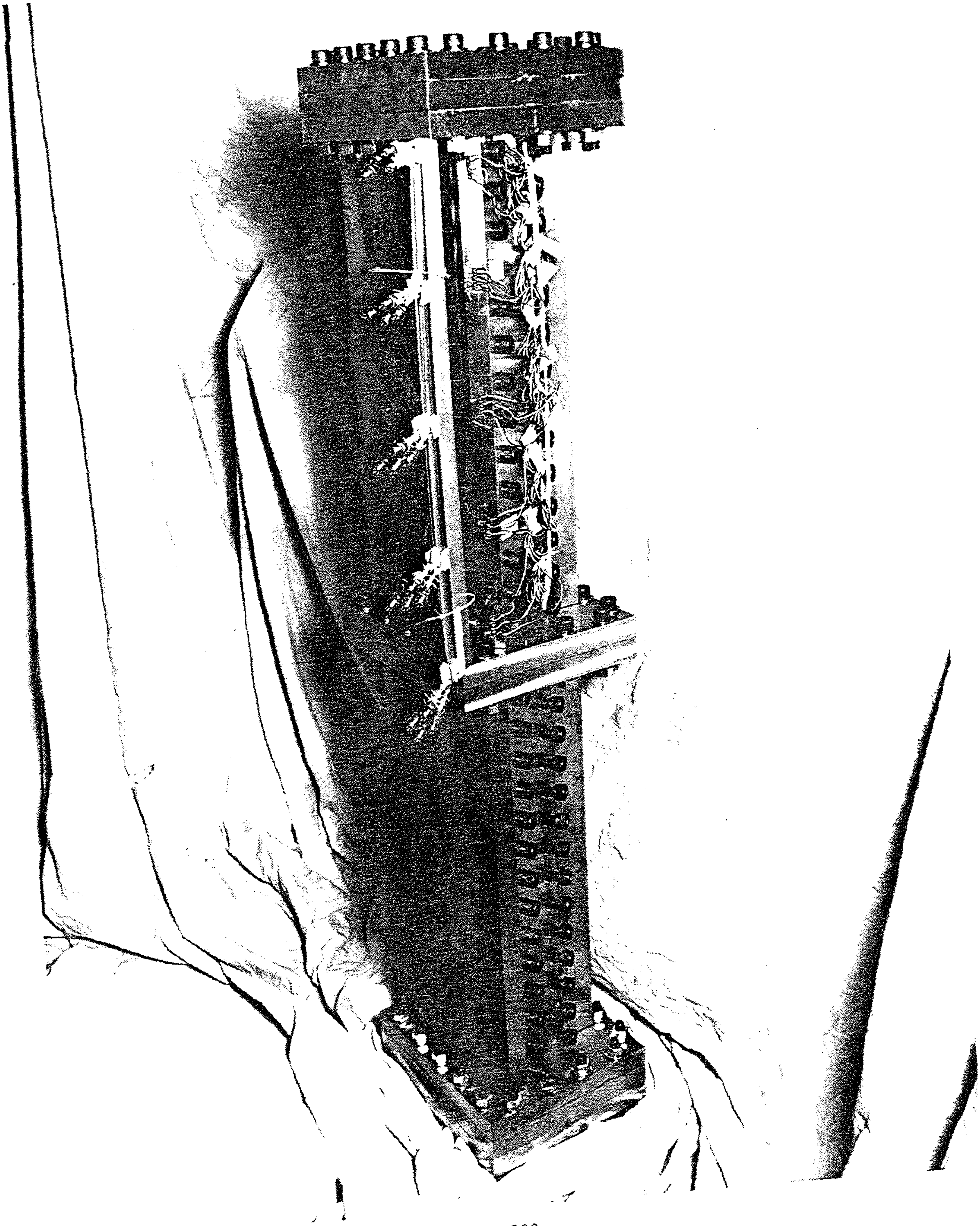
PRESSURE COEFFICIENT AND REYNOLDS' NUMBER RELATED TO YIELD PRESSURE LOAD ON PLATES

$$\frac{\Delta p}{\rho V^2} = 0.040\left(\frac{Vh}{\nu}\right)^{0.177}$$

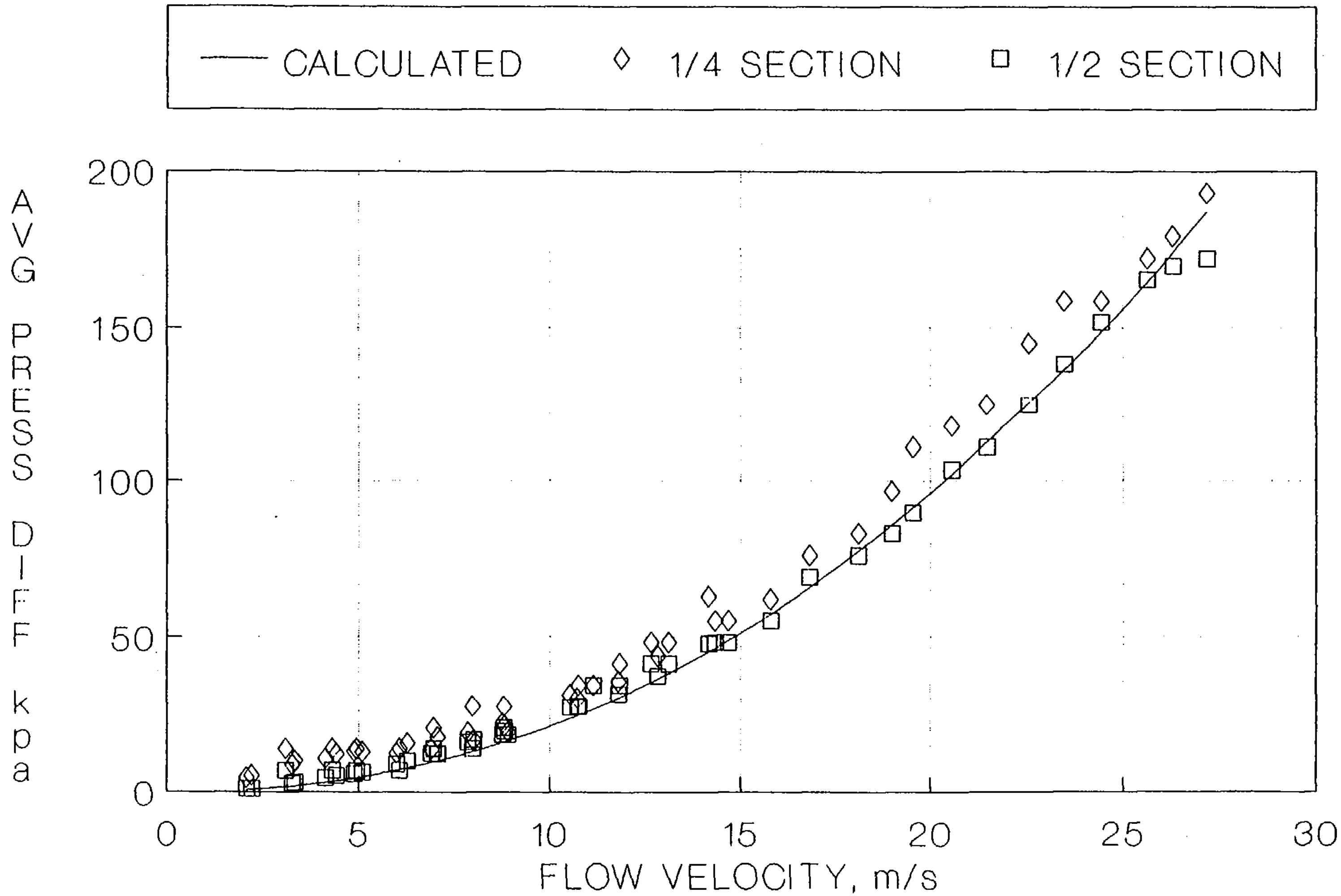
DERIVED FROM EXPERIMENTAL FLOW DATA

LOWER ELEMENT CENTRAL PLATE DEFLECTION
CALCULATED VALUES COMPARED WITH
EXPERIMENTAL VALUES





AVERAGE PRESSURE DIFFERENCE AS A FUNCTION OF FLOW VELOCITY



Summary of Fuel Plate Stability Testing

(1) A Method Has Been Developed to Predict Structural Response of Fuel Plates to Hydraulic Loading

- Prediction of ΔP across plates
- Determine deflection/stress levels using structural analysis

(2) ANS Specific Conclusions:

- No evidence of potential plate collapse in the coolant velocity range from 0-50 m/s
- No evidence of plate flutter with coolant velocities below 33 m/s
- Local stress levels appear to dictate plate limits as opposed to plate deflection

Flow Blockage Testing

Objective: To experimentally determine local thermal and fluid behavior downstream of a core inlet blockage.

Flow Blockage Testing

Coolant - H_20

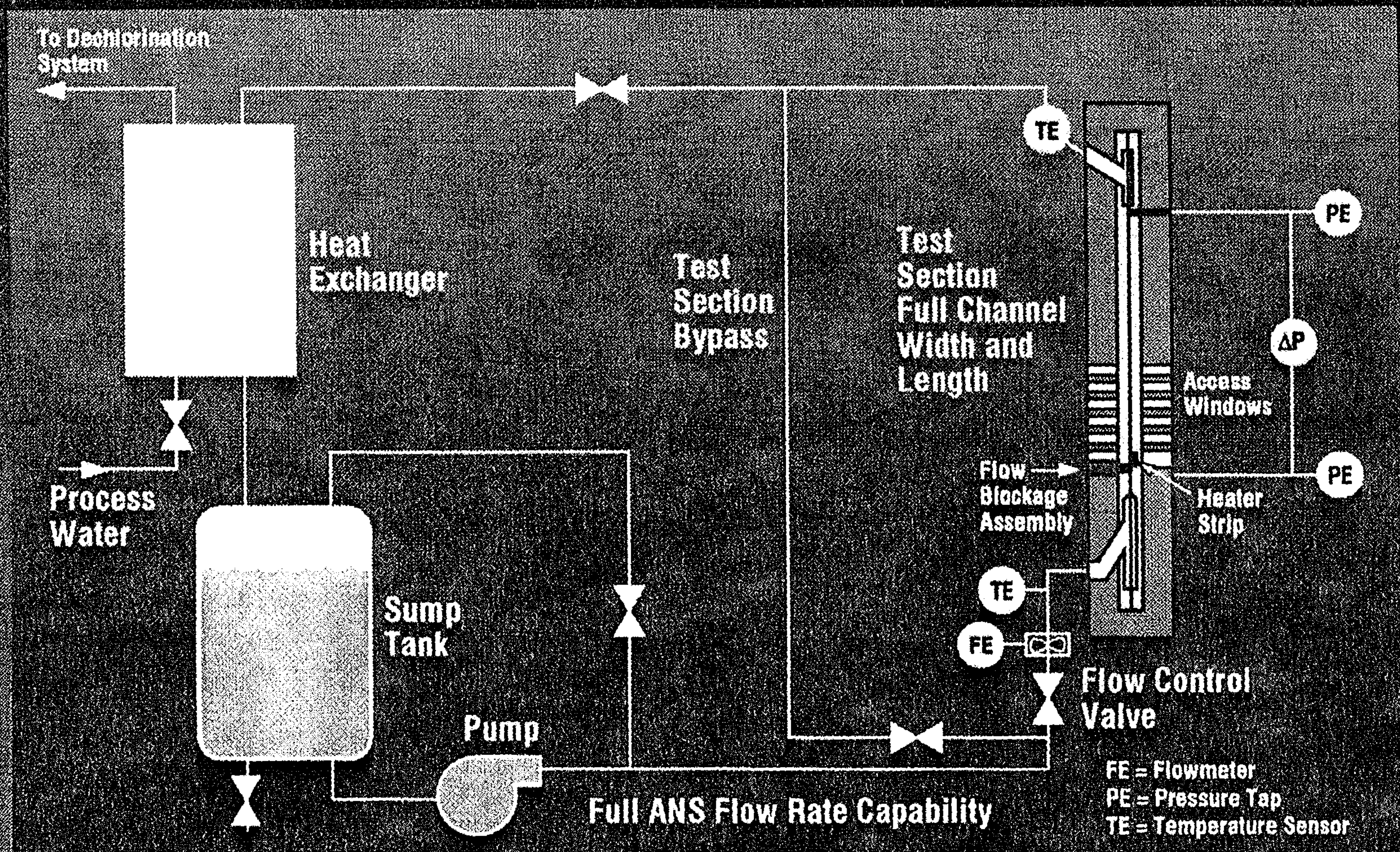
8 — Hydraulic tests using LDV

25 — Thermal tests using TLCs

Flow velocity range : 5 - 25 m/s

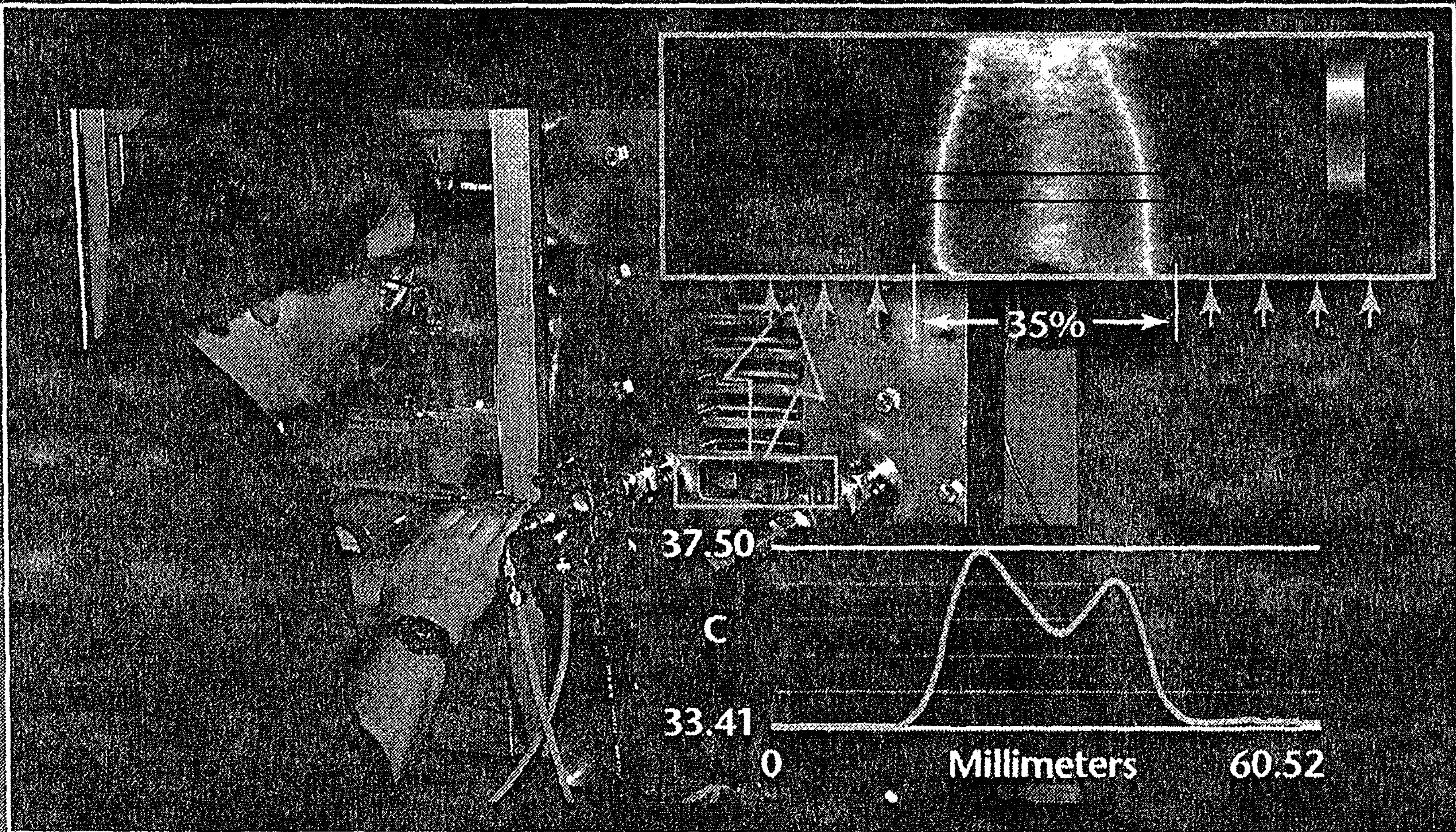
Blockage sizes : Center - 15%, 25%, 35%, 40%
Edge - 10%, 25%

Flow Blockage Test Facility Is Designed to Precisely Match Fuel Channel Hydraulic Conditions



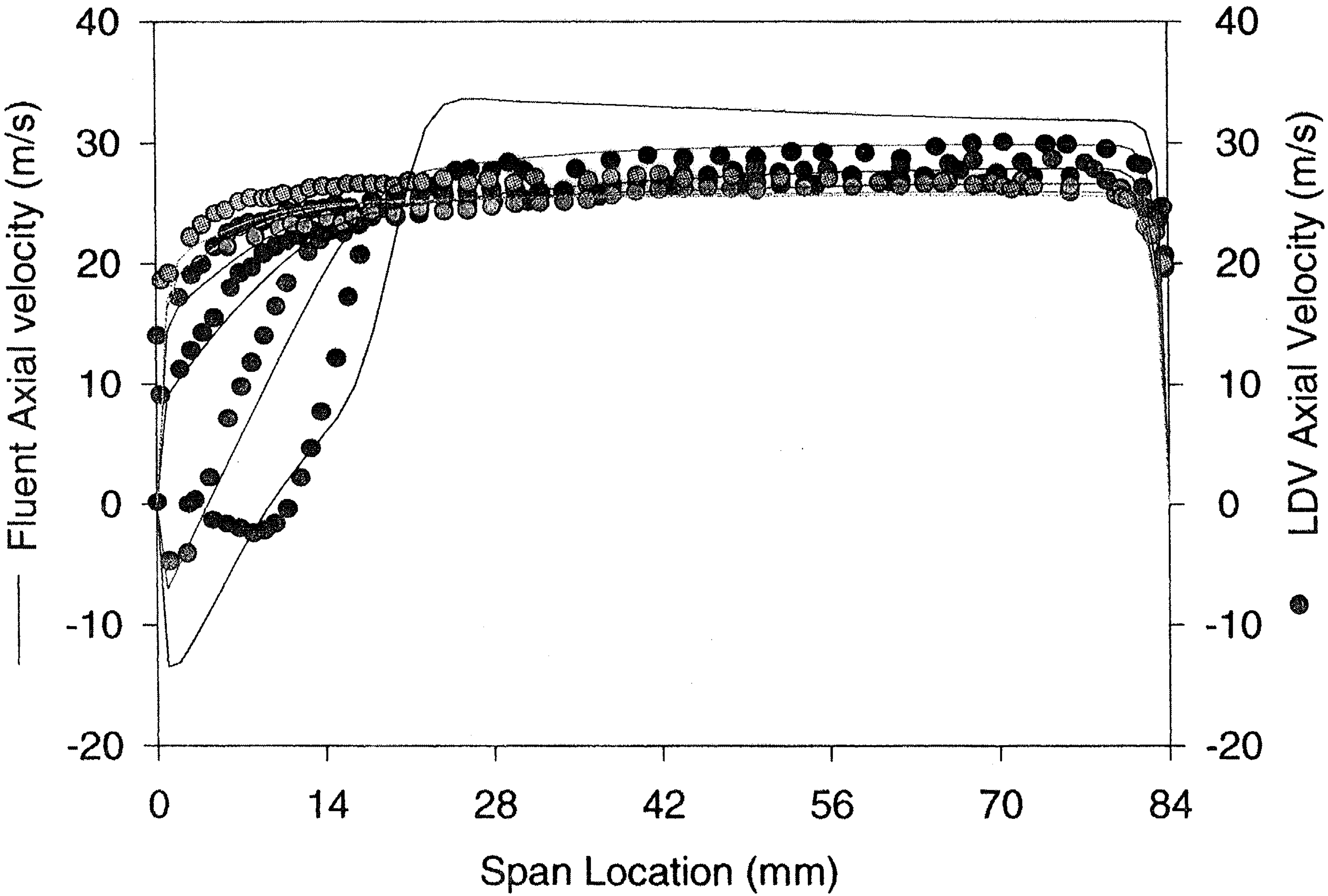
ORNL-DWG 94M-2605C ETD

Thermochromic Crystals and Image Processing Techniques Are Used to Experimentally Determine Channel Wall Heat Transfer



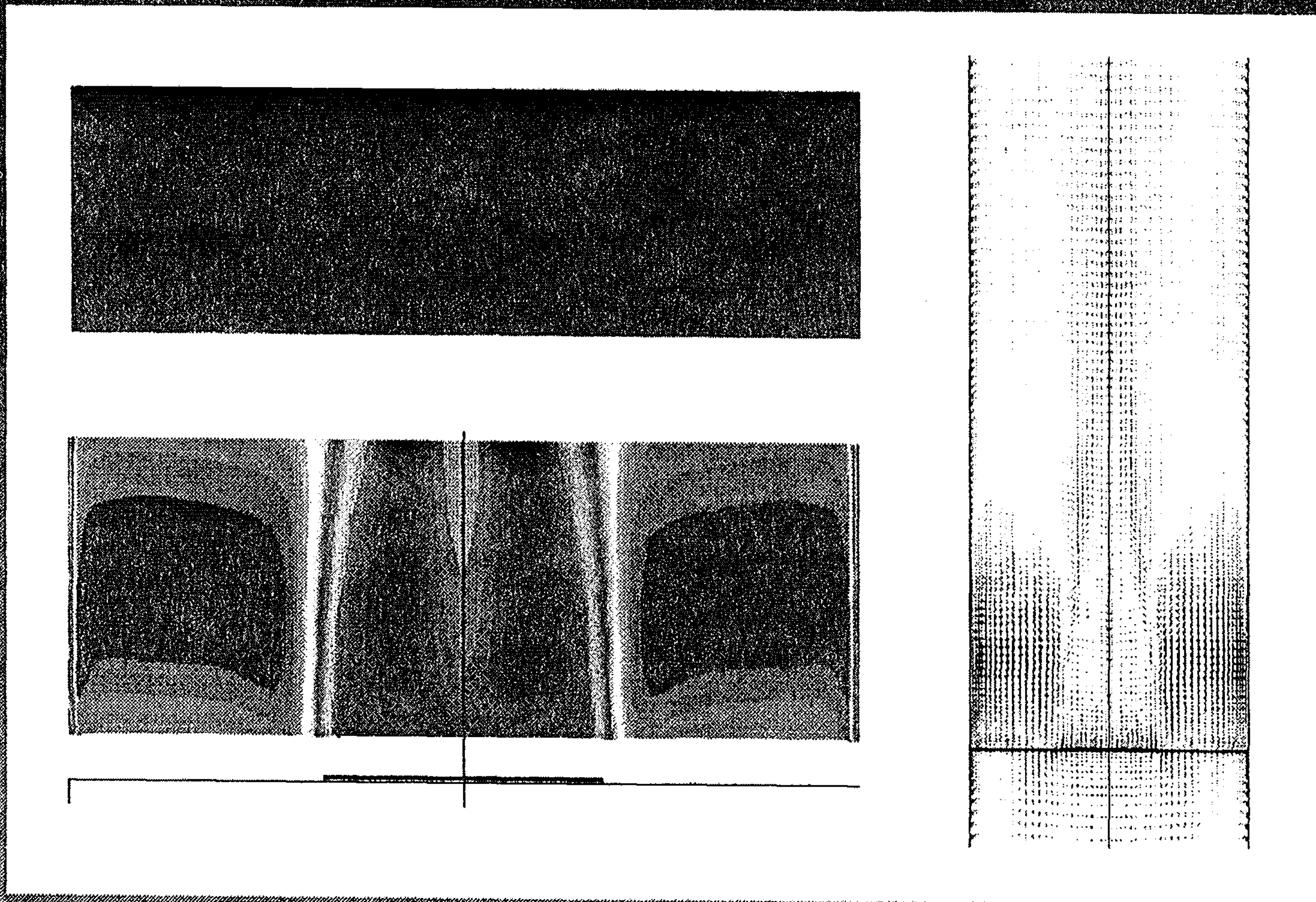
ORNL-DWG 94M-2607C ETD

CFD results are in close agreement with experimental data

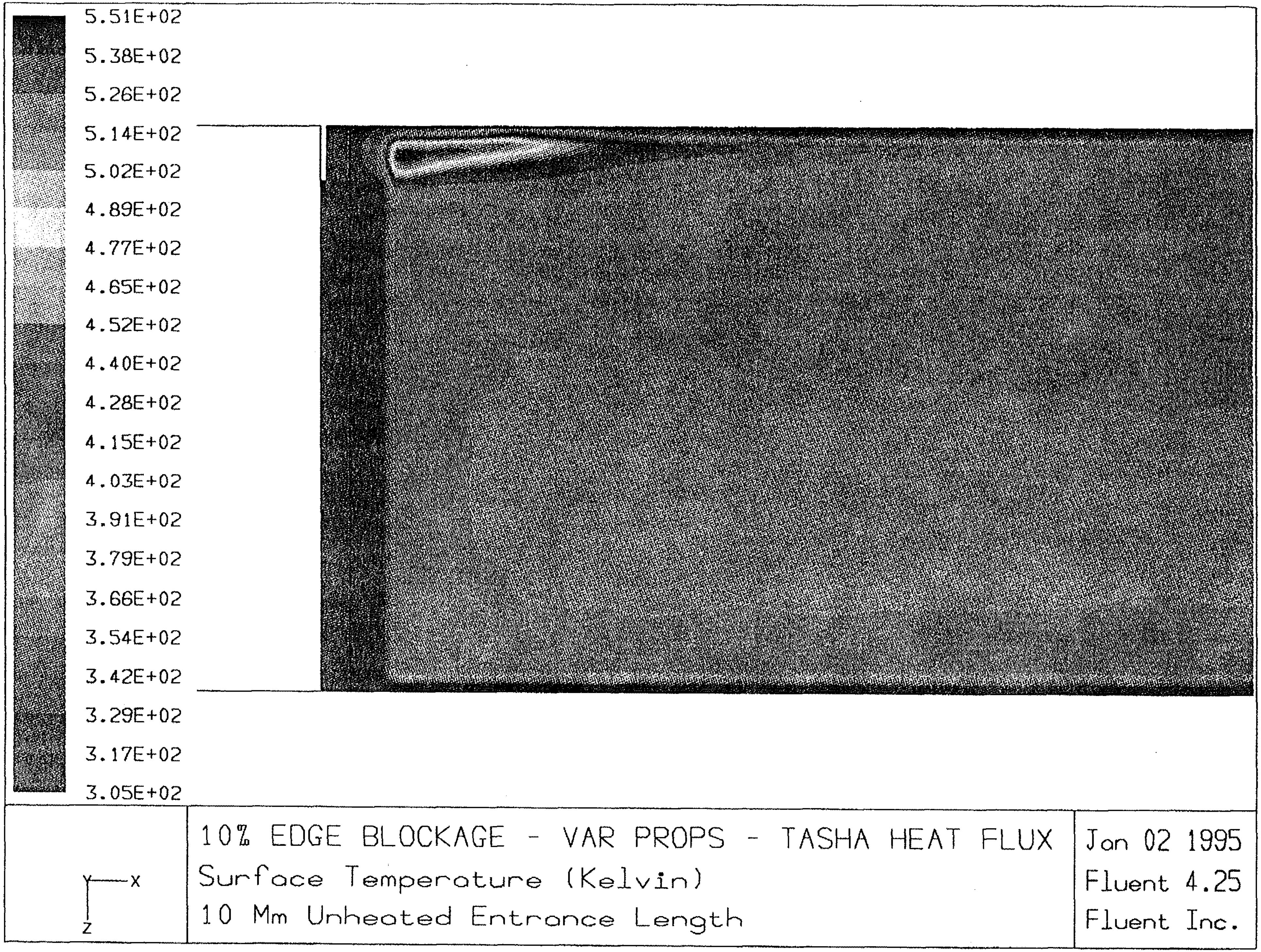


210

Computational Fluid Dynamics Improved Our Understanding of the Experimental Results



ORNL-DWG 94M-2609C ETD


5.51E+02
5.38E+02
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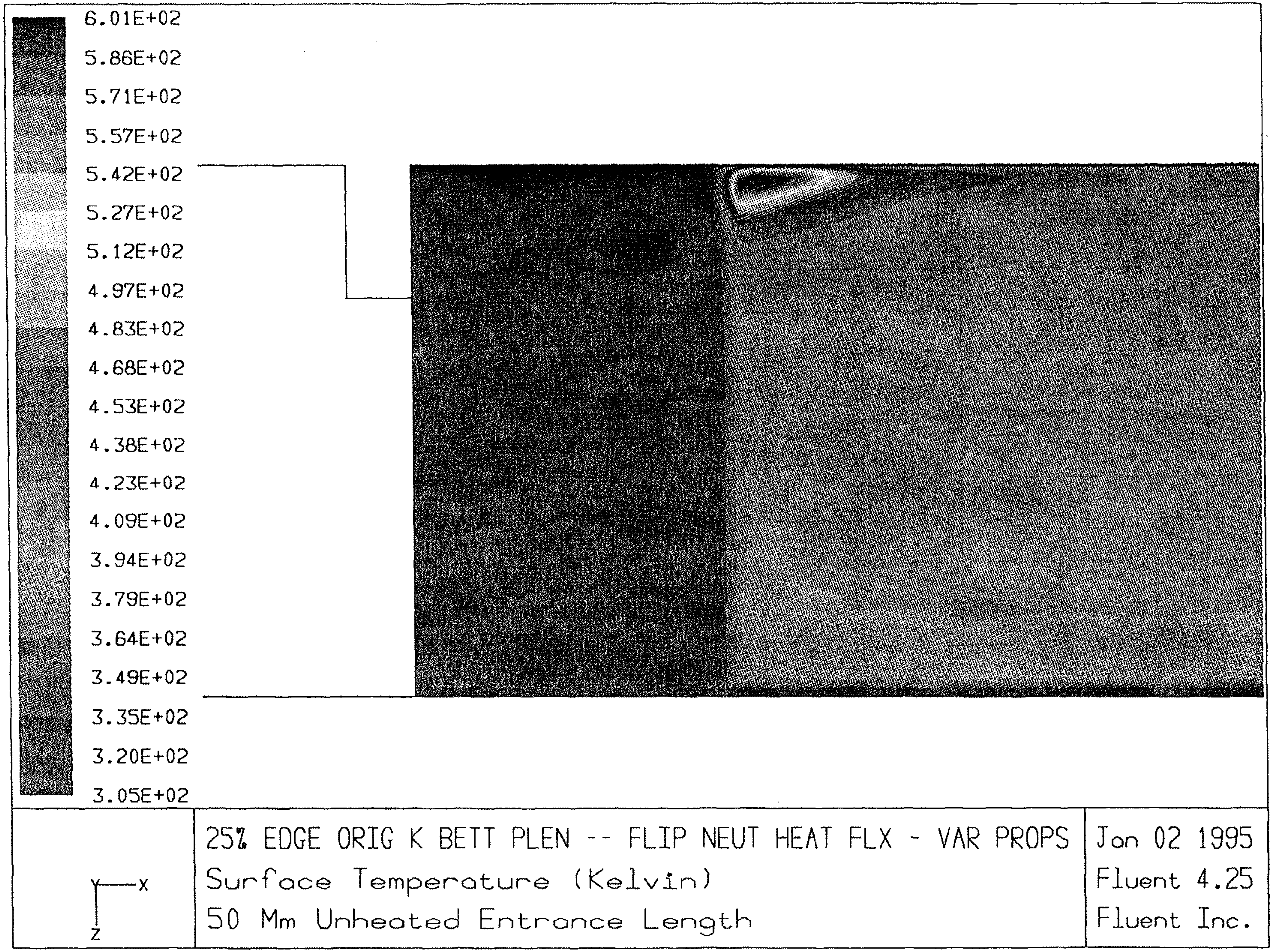
4.52E+02
4.40E+02
4.28E+02
4.15E+02
4.03E+02
3.91E+02
3.79E+02
3.66E+02
3.54E+02
3.42E+02
3.29E+02
3.17E+02
3.05E+02
Y
X
Z
10% EDGE BLOCKAGE - VAR PROPS - TASHA HEAT FLUX
Surface Temperature (Kelvin)
10 Mm Unheated Entrance Length
Jan 02 1995
Fluent 4.25
Fluent Inc.


6.01E+02
5.86E+02
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4.83E+02
4.68E+02
4.53E+02
4.38E+02
4.23E+02
4.09E+02
3.94E+02
3.79E+02
3.64E+02
3.49E+02
3.35E+02
3.20E+02
3.05E+02
Y X Z
25% EDGE ORIG K BETT PLEN -- FLIP NEUT HEAT FLX - VAR PROPS
Surface Temperature (Kelvin)
50 Mm Unheated Entrance Length
Jan 02 1995
Fluent 4.25
Fluent Inc.

Reattachment Lengths
for 1.4 MPa Pressure Drop (mm)

Blockage Size	Blockage Position	Fluent Model		LDV Data
		Near wall	Channel center	
15%	Center	37	34	x < 27
35%	Center	80	80	x < 27
25%	Edge	74	74	60 < x < 90

Summary of Flow Blockage Testing and Analysis

- CFD code has been benchmarked against prototypic ANS flow conditions and geometry
- CFD analysis appears to be conservative with respect to experimental results
- Unheated entrance length was increased to prevent localized boiling downstream of blockage
- Closeout testing will focus on determining the importance of blockage shapes on local cooling
- Next step would have been to evaluate alternate core inlet designs

CHF Correlation Scheme Proposed for Research Reactors using Plate-Type Fuel
- New CHF Correlation under CCFL Condition -

IGORR-IV

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Presented at IGORR-IV, Gatlinburg, TN

May 25 , 1995

Introduction

- The detailed understanding of *Critical Heat Flux* (CHF) for vertical rectangular channels is essential for core thermal-hydraulic design and safety analysis of research reactors using plate-type fuel.
- In research reactors which are cooled by downward core flow, the core flow reversal should take into account for dasign and safety .
- At JAERI, reduced enrichment work for JRR-4 is now in progress. JRR-4 is a swimming pool type research reactor with the maximum power of 3.5 MWt. Because of comparatively small thermal power, it is not nessesary to credit auxiliary or emergency pumps for the decay heat removal after coast-down of main pumps in case of emergency. Therefore, the core flow reversal would occur just after coast-down of maim pumps.
- High power research reactors have auxiliary or emergency pumps for the decay heat removal after coast-down of main pumps in case of operational transients or accident condition such as "*loss of commercial electric power supply*". So that the core flow reversal could occur under decay heat level low enogh and safety margin against CHF is large enough.

Objective of this study

- For the low mass flux region including stagnant flow condition, CHF is closely related to *Counter-Current Flow Limitation* (CCFL). The CHF correlation in this region used so far at JAERI is very conservative one, that is,
  - » The effects of channel inlet subcooling and axial heat flux distribution on CHF in this region have not been taken into account.
- Therefore, the effects of channel inlet subcooling and axial heat flux distribution on CHF were investigated in this study based on the existing CHF experimental data under low mass flux region.
- The CHF at high subcooling and high mass flux were also investigated to the experimental data at flow excursion (FE) as well as CHF.

Previous CHF Correlation Scheme Proposed for Research Reactors using Plate-Type Fuel

IGORR-IV

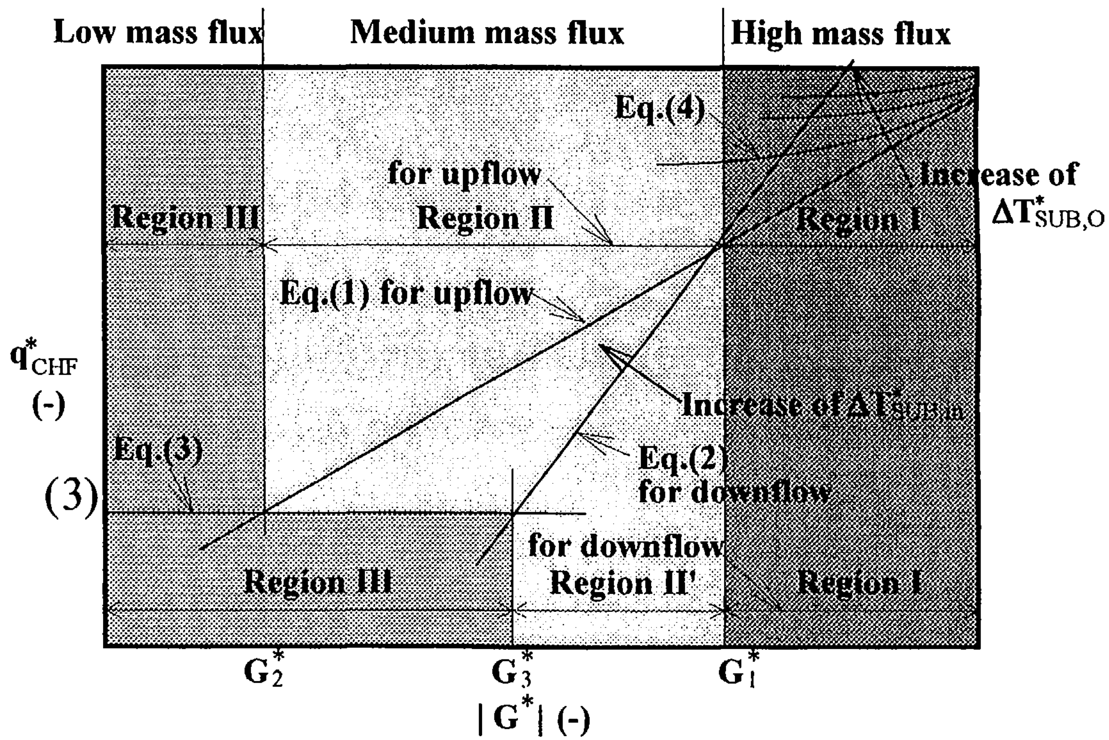
$$q^*_{CHF,4} = 0.005\left|G^*\right|^{0.611}\left(1+\frac{5000}{\left|G^*\right|}\Delta T^*_{SUB,O}\right) \quad (4)$$

$$q^*_{CHF,4} = 0.005\left|G^*\right|^{0.611}\frac{\left(1+\frac{5000}{\left|G^*\right|}\Delta T^*_{SUB,in}\right)}{\left(1+25\left|G^*\right|^{-1.389}\frac{A}{A_H}\right)} \quad (4')$$

$$q^*_{CHF,1} = 0.005\left|G^*\right|^{0.611} \quad (1)$$

$$q^*_{CHF,2} = \frac{A}{A_H}\,\Delta T^*_{SUB,in}\left|G^*\right| \quad (2)$$

$$q^*_{CHF,3} = 0.7\,\frac{A}{A_H}\,\frac{\sqrt{W/\lambda}}{\left\{1+\left(\rho_g/\rho_\ell\right)^{1/4}\right\}^2} \quad (3)$$



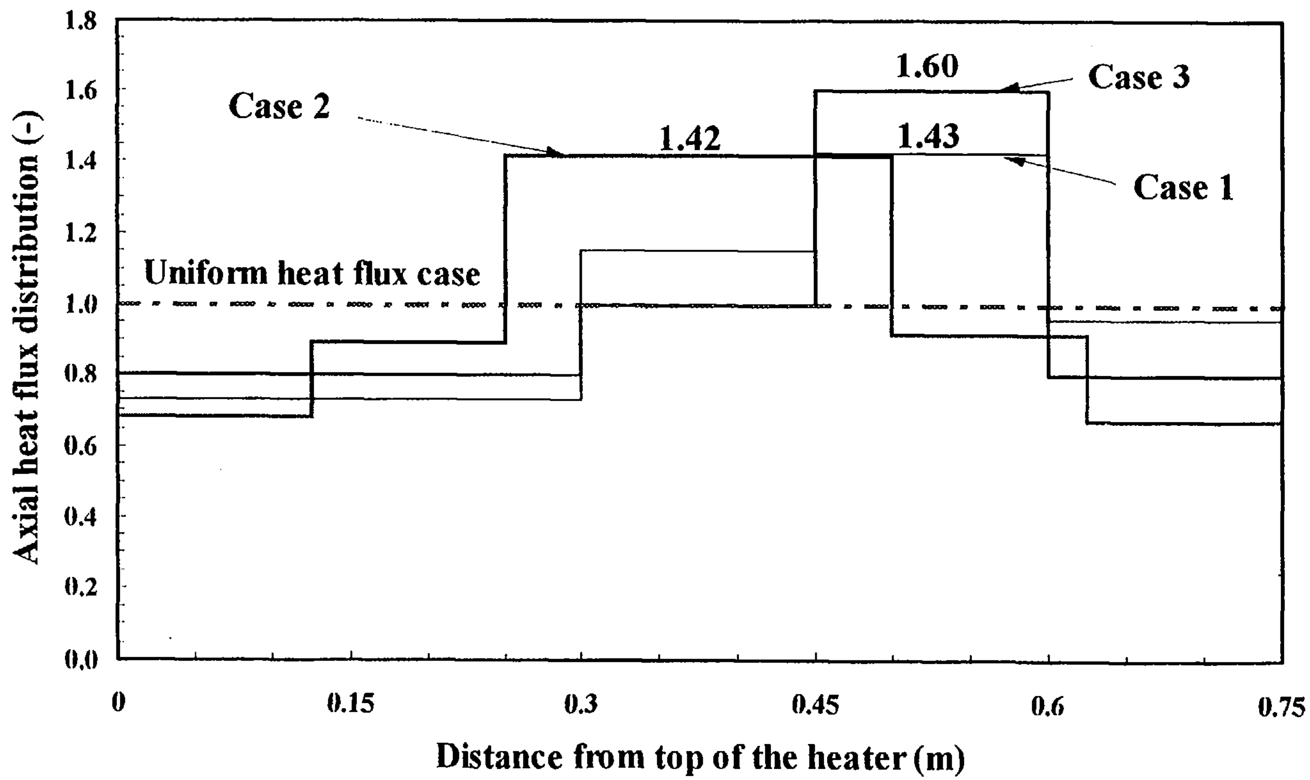
Experimental conditions of existing CHF tests investigated in this study

■ Coolant :	Water
■ Pressure :	Atmospheric pressure
■ Mass flux :	0 to -73 kg/m²sec (Downward) (0: stagnant flow conditions)
■ Inlet subcooling :	0 to 78 K
■ De :	4.3 to 9.1 (mm)
■ L/De :	71 to 174 (-)
■ Axial heat flux distribution:	Non-uniform and Uniform
■ Axial peaking factor:	1.0 to 1.6 (-)
■ Total number of data :	69

Axial heat flux distribution of existing CHF tests investigated in this study

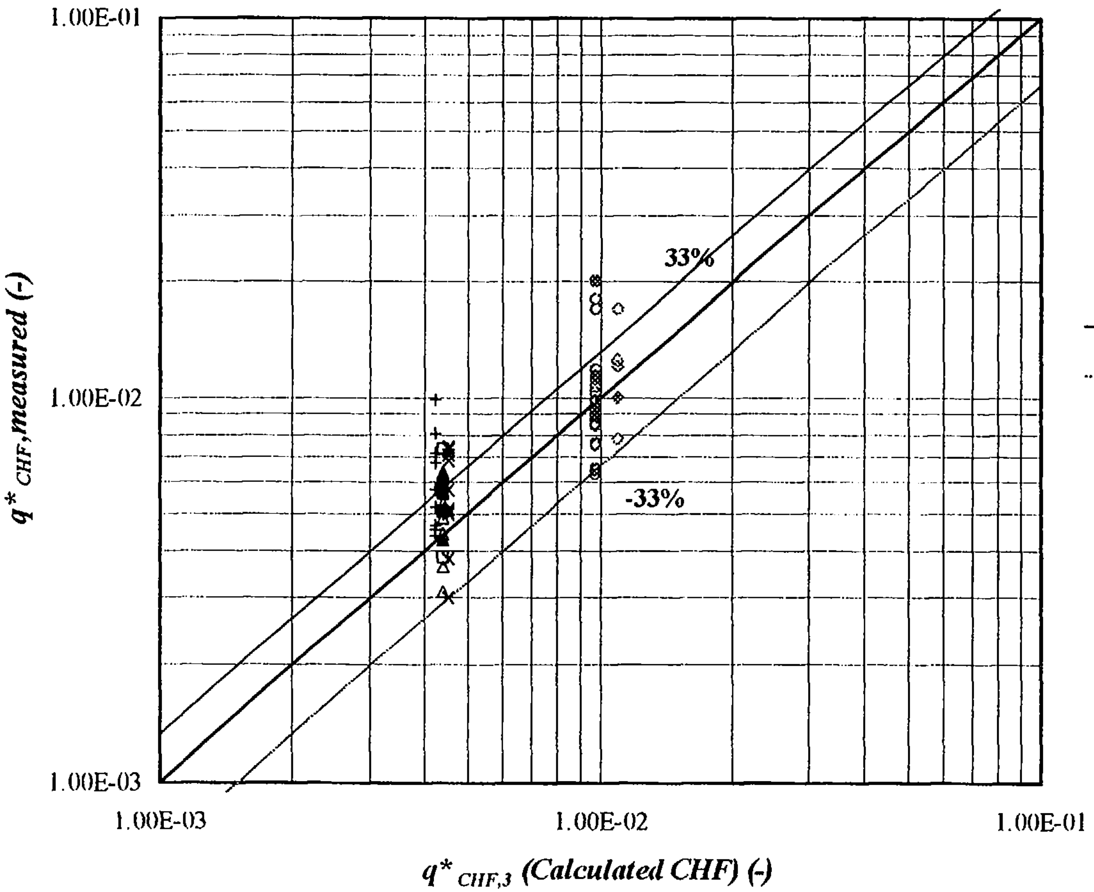
IGORR-IV

Axial heat flux distribution investigated in this study



Comparison between all JAERI experimental results and previous CHF correlation under CCFL condition for rectangular channels

CHF experimental results for both non-uniform and uniform heat flux condition



□ $\Delta Tsub,in$=27-68 °C U Gap=2.25mm, L=750mm
◇ $\Delta Tsub,in$=28-59 °C U Gap=2.8mm, L=375mm
○ $\Delta Tsub,in$= 1-58 °C U Gap=5.0mm, L=750mm
△ $\Delta Tsub,in$=11-77 °C N/U Case 1
× $\Delta Tsub,in$= 2-78 °C N/U Case 2 Gap=2.25mm, L=750mm
+ $\Delta Tsub,in$= 0-74 °C N/U Case 3
— +33 %
······ -33%

U : Uniform heat flux
N/U : Non-uniform heat flux

■ ▲ DNB data obtained under stagnant flow (zero flow) condition

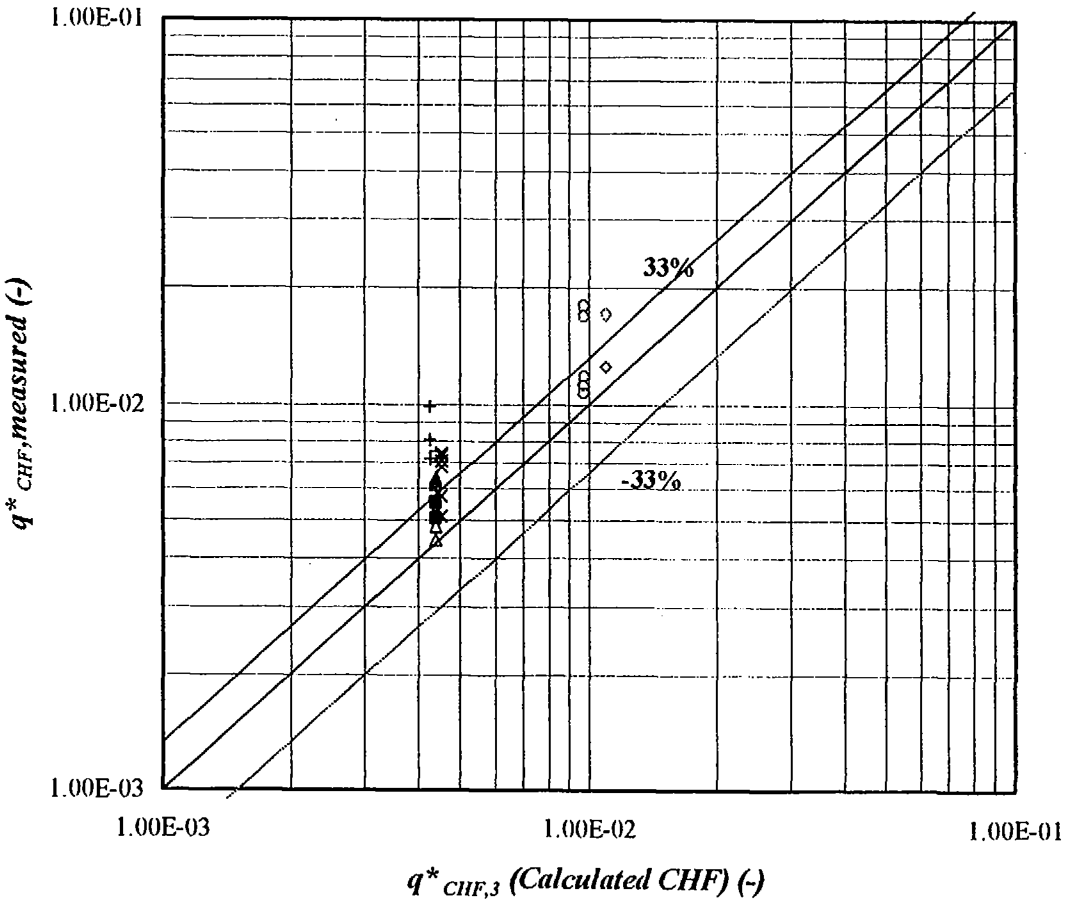
$$q^*_{CHF,3} = 0.7 \frac{A}{A_H} \frac{\sqrt{W/\lambda}}{\left\{1+\left(\rho_g/\rho_l\right)^{1/4}\right\}^2}$$

Comparison between experimental results and previous CHF correlation under CCFL condition for rectangular channels, $\Delta T_{SUB,in} > 30$

(Except the data obtained saturated or near saturated condition at the inlet of channel)

IGORR-IV

CHF experimental results for both non-uniform and uniform heat flux condition



□ $\Delta T_{sub,in}$=30-68 °C U Gap=2.25mm, L=750mm
◇ $\Delta T_{sub,in}$=57-59 °C U Gap=2.8mm, L=375mm
○ $\Delta T_{sub,in}$=41-58 °C U Gap=5.0mm, L=750mm
△ $\Delta T_{sub,in}$=36-77 °C N/U Case 1
× $\Delta T_{sub,in}$=36-78 °C N/U Case 2 Gap=2.25mm, L=750mm
+ $\Delta T_{sub,in}$=70-74 °C N/U Case 3
—— +33 %
······ -33%

U : Uniform heat flux
N/U : Non-uniform heat flux
■ ▲ DNB data obtained under stagnant flow (zero flow) condition

$$q^*_{CHF,3} = 0.7 \frac{A}{A_H} \frac{\sqrt{W/\lambda}}{\left\{1+\left(\rho_g/\rho_\ell\right)^{1/4}\right\}^2}$$

New CHF correlation under CCFL condition

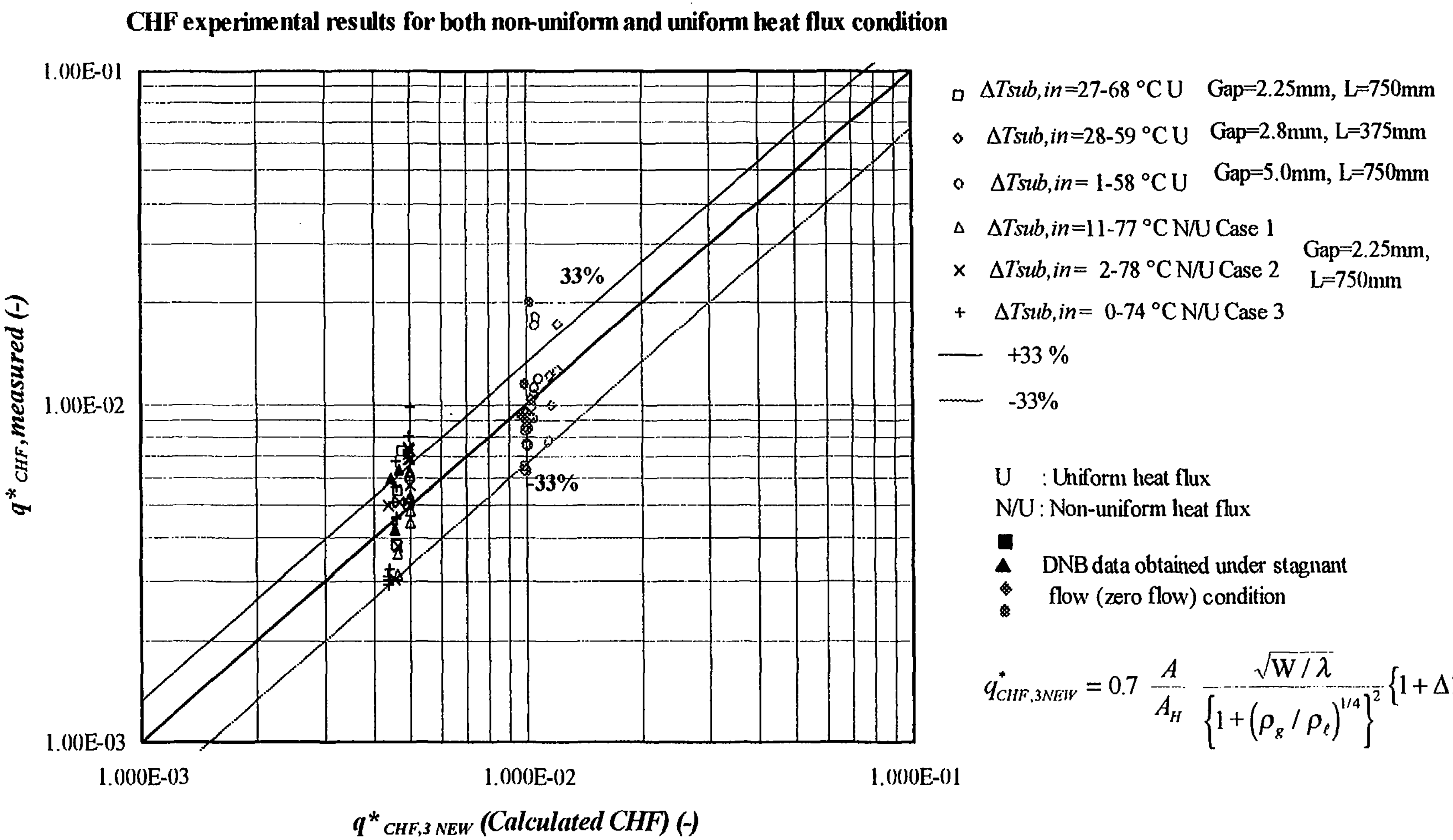
- Based on the investigation results of this study, following correlation is proposed as a preliminary CHF correlation under CCFL condition to take into account the channel inlet subcooling effects to CHF.

$$q^*_{CHF,3NEW} = 0.7\, \frac{A}{A_H}\, \frac{\sqrt{W/\lambda}}{\left[1+\left(\rho_g/\rho_\ell\right)^{1/4}\right]^2} \left\{1+\Delta T^*_{SUB,in}\right\} \quad (5)$$

$$\Delta T^*_{SUB,in} = \frac{C_{p\ell}\Delta T_{SUB,in}}{h_{fg}}.$$

Comparison between all JAERI experimental results and New CHF correlation under CCFL condition for rectangular channels

IGORR-IV

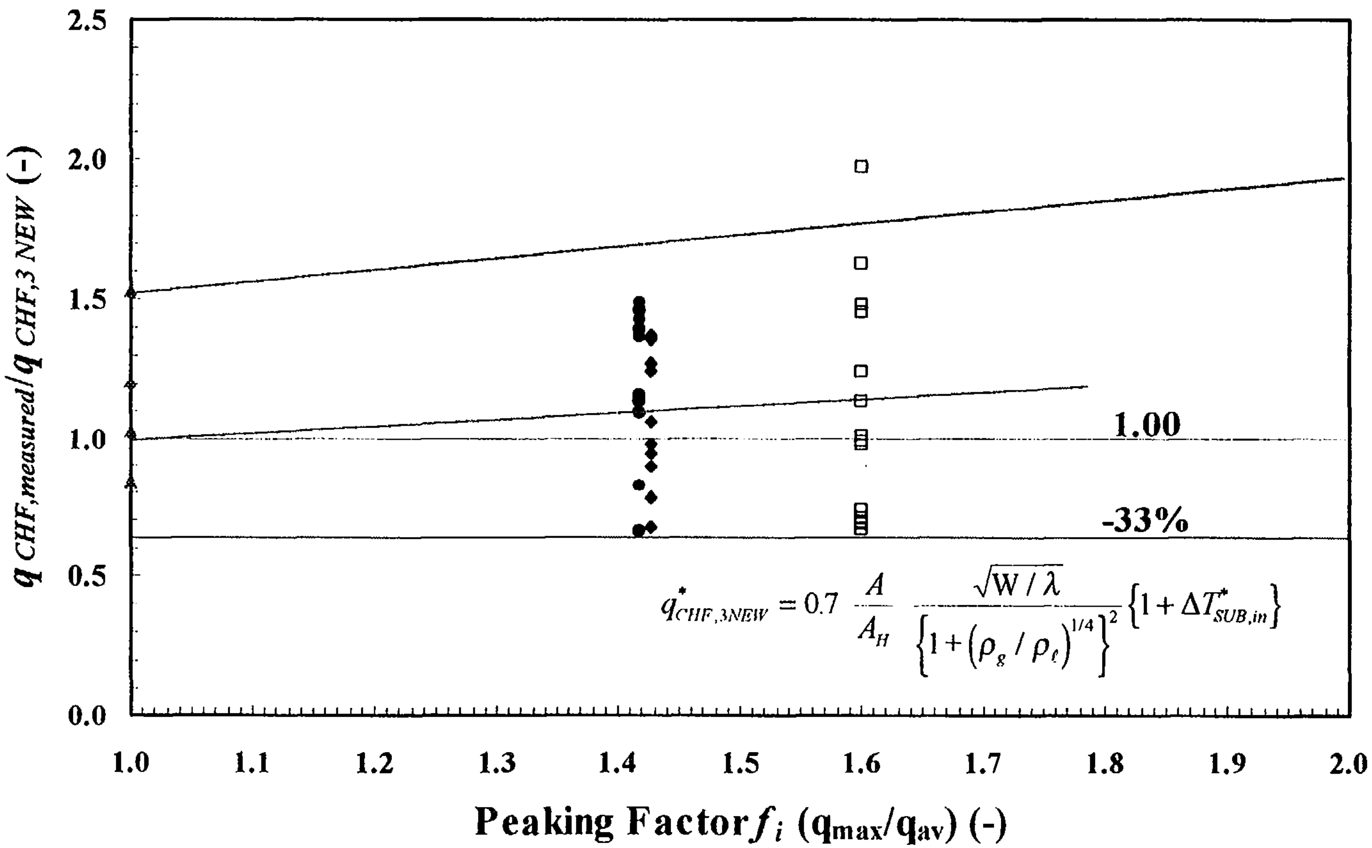


$$q^*_{CHF,3NEW} = 0.7\,\frac{A}{A_H}\,\frac{\sqrt{W/\lambda}}{\left\{1+\left(\rho_g/\rho_\ell\right)^{1/4}\right\}^2}\left\{1+\Delta T^*_{SUB,in}\right\}$$

Effect of axial heat flux distribution to CHF

IGORR-IV

Effects of Axial Peaking Factor on CHF, Gap=2.25mm, L=750mm



Experimental conditions of Flow Excursion (FE) and CHF test performed at ORNL

High mass flux region

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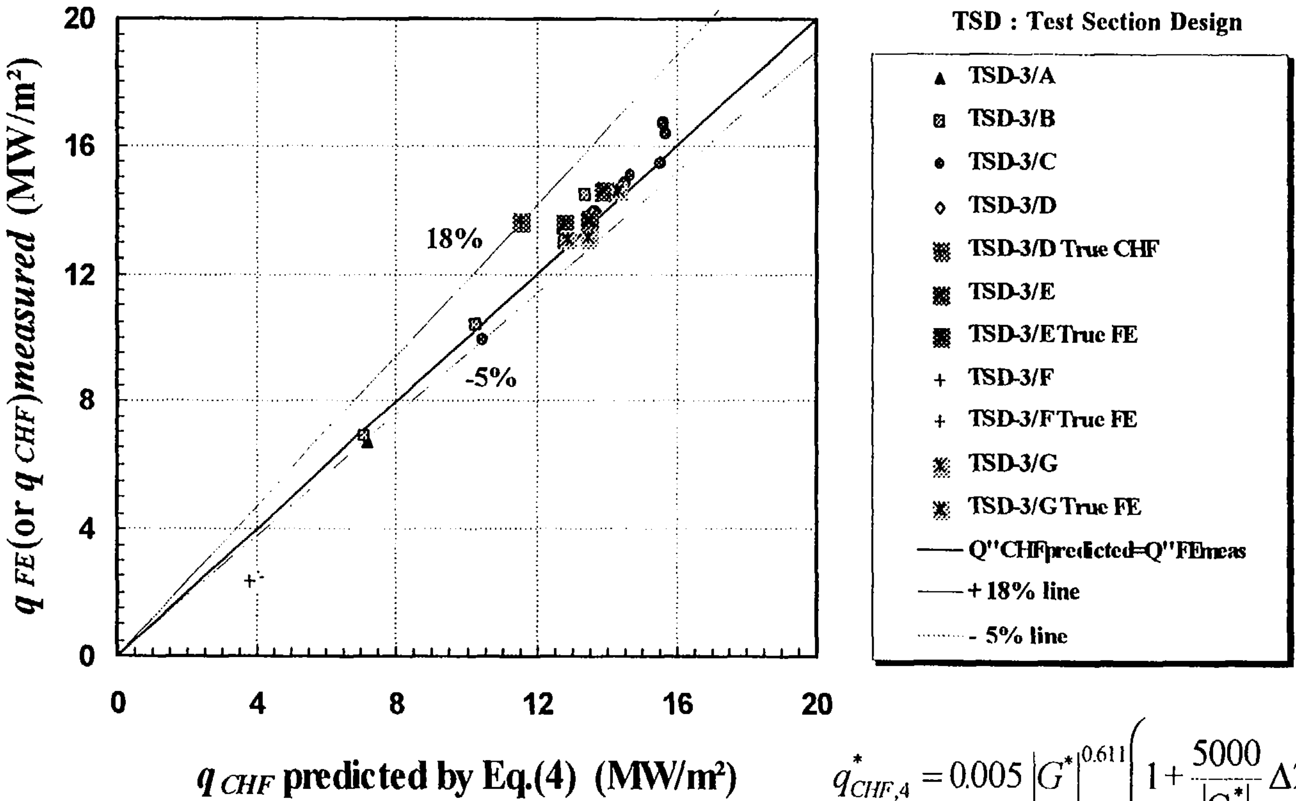
227

- Coolant : Water
- Inlet coolant temperature : 45 °C
- Exit coolant pressure : 1.7 MPa
- Nominal average heat flux range : 6～14 MW/m^2
- Corresponding velocity range : 8～21 m/s
- Channel configuration : Rectangular channel,
  1.27 x 12.7 x 507 mm

Comparison between measured FE heat flux, CHF obtained at ORNL and CHF predicted by Eq.(4)

IGORR-IV



$$q^*_{CHF,4} = 0.005\,|G^*|^{0.611}\left(1 + \frac{5000}{|G^*|}\Delta T^*_{SUB,O}\right) \quad (4)$$

New CHF Correlation Scheme Proposed for Research Reactors using Plate-Type Fuel

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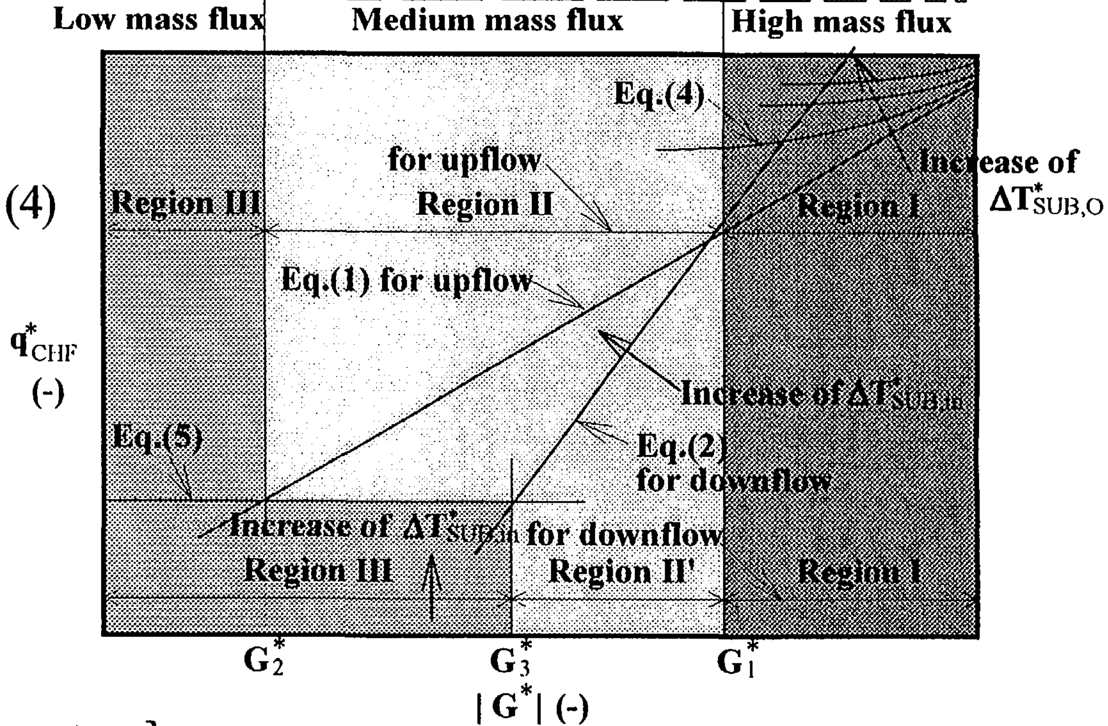
$$q^*_{CHF,4} = 0.005 \left|G^*\right|^{0.611}\left(1+\frac{5000}{\left|G^*\right|}\Delta T^*_{SUB,O}\right) \quad (4)$$

$$q^*_{CHF,1} = 0.005 \left|G^*\right|^{0.611} \quad (1)$$

$$q^*_{CHF,2} = \frac{A}{A_H}\,\Delta T^*_{SUB,in}\left|G^*\right| \quad (2)$$

$$q^*_{CHF,3} = 0.7\,\frac{A}{A_H}\,\frac{\sqrt{W/\lambda}}{\left\{1+\left(\rho_g/\rho_\ell\right)^{1/4}\right\}^2}\left\{1+\Delta T^*_{SUB,in}\right\} \quad (5)$$



229

Region boundaries are identified by the following equations.

$$G_1^* = \left[\frac{0.005}{\frac{A}{A_H} \Delta T^*_{SUB,in}} \right]^{\frac{1}{0.389}} \tag{6}$$

$$G_2^* = \left[140 \frac{A}{A_H} \frac{\sqrt{W/\lambda}}{\left\{1 + \left(\rho_g / \rho_\ell\right)^{1/4}\right\}^2} \left\{1 + \Delta T^*_{SUB,in}\right\} \right]^{\frac{1}{0.611}} \tag{7}$$

$$G_3^* = 0.7 \frac{\sqrt{W/\lambda}}{\left\{1 + \left(\rho_g / \rho_\ell\right)^{1/4}\right\}^2} \left\{1 + \frac{1}{\Delta T^*_{SUB,in}}\right\} \tag{8}$$

Factors effective to CHF in each region

IGORR-IV

Region	Effective factors to CHF	q^*_{CHF}	Note
Region I	$\Delta T^*_{SUB,O}$ and G^*	Eq.(4)	High mass flux, Upflow and Downflow
Region II	G^*	Eq.(1)	Medium mass flux, Upflow
Region II'	$\Delta T^*_{SUB,in}$, G^* and A/A_H	Eq.(2)	Medium mass flux, Downflow
Region III	A/A_H, W and $\Delta T^*_{SUB,in}$	Eq.(5)	Low mass flux, Upflow and Downflow

Conclusions

- The effects of channel inlet subcooling and axial heat flux distribution on CHF under CCFL condition were investigated in this study.
- As the results, Eq.(5) was proposed as a new CHF correlation including the effect of channel inlet subcooling.
- Based on the comparison between Eq.(5) and CHF experimental data obtained non-uniform heat flux condition, this new correlation can be adopted within the range investigated in this study.

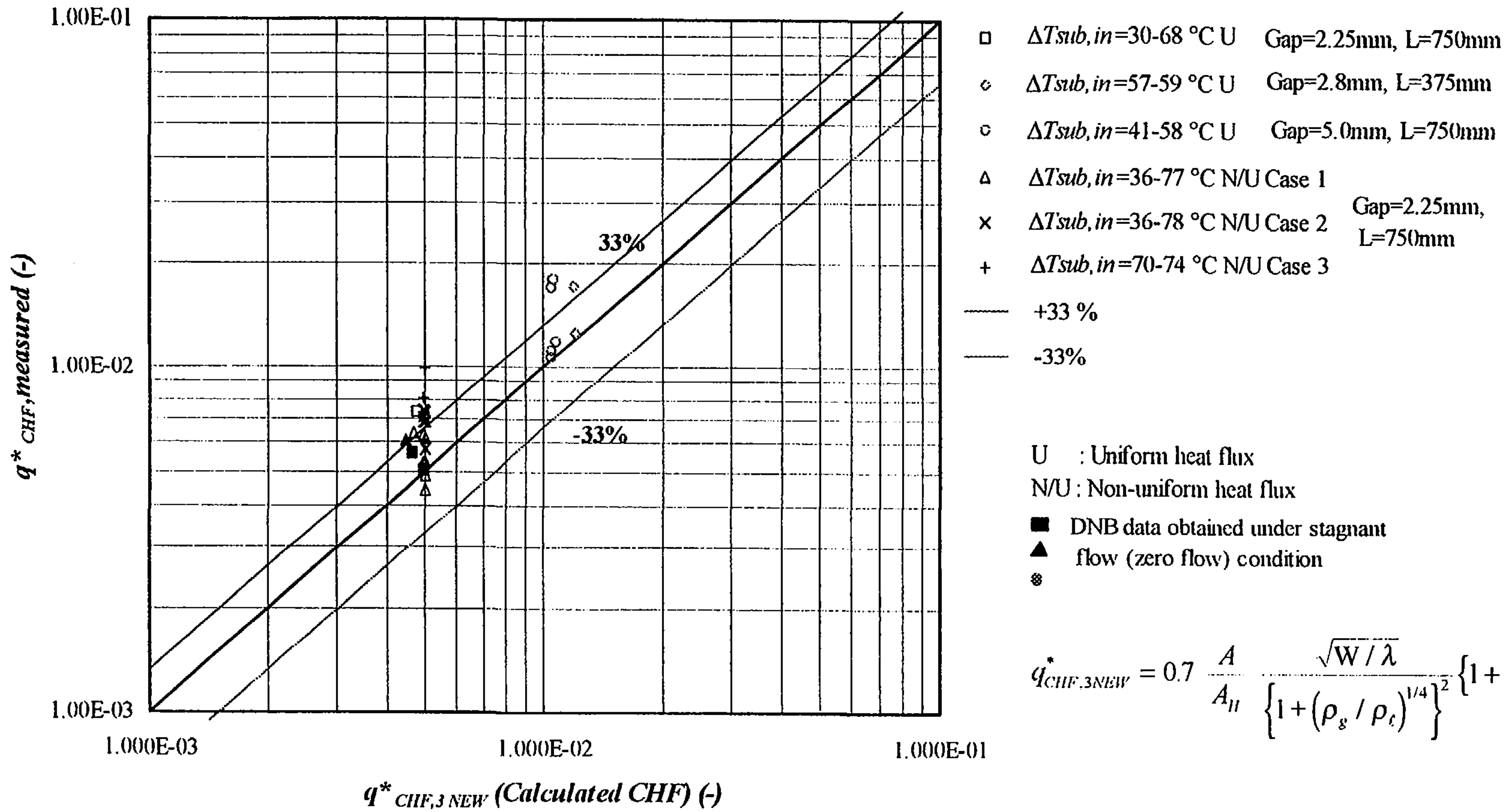
  (Axial peaking factor : 1.0 ~ 1.6)
- For high mass flux region, Eq.(4) was compared with FE and CHF experimental data obtained at ORNL. Eq(4) can be used to identify the thermal limit of research reactors for the condition investigated in this study.
- A new CHF correlation scheme was proposed based on this study.

Comparison between experimental results and New CHF correlation under CCFL condition for rectangular channels , $\Delta T_{SUB,in} > 30$

(Except the data obtained saturated or near saturated condition at the inlet of channel)

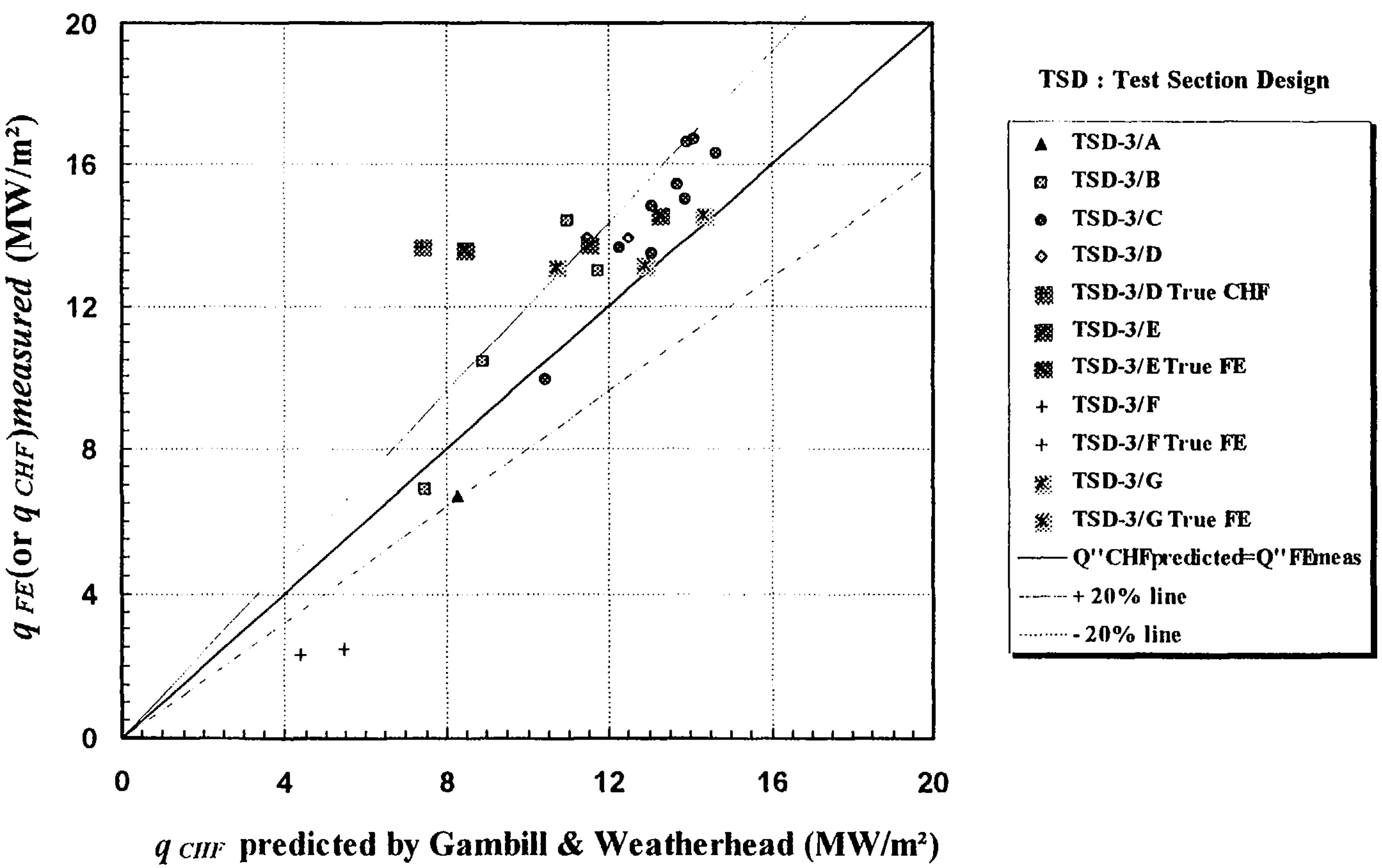
IGORR-IV

CHF experimental results for both non-uniform and uniform heat flux condition



$$q^*_{CHF,3NEW} = 0.7 \frac{A}{A_H} \frac{\sqrt{W/\lambda}}{\left\{1+\left(\rho_g/\rho_\ell\right)^{1/4}\right\}^2}\left\{1+\Delta T^*_{SUB,in}\right\}$$

Comparison between measured FE heat flux (CHF) at ORNL and CHF predicted by Gambill & Weatherhead



XA04C1692

United States Department of Commerce
Technology Administration
National Institute of Standards and Technology

235

NISTIR 5026

Thermal Hydraulic Tests of a Liquid Hydrogen Cold Neutron Source

J.D. Siegwarth
D.A. Olson
M.A. Lewis
J.M. Rowe
R.E. Williams
P. Kopetka

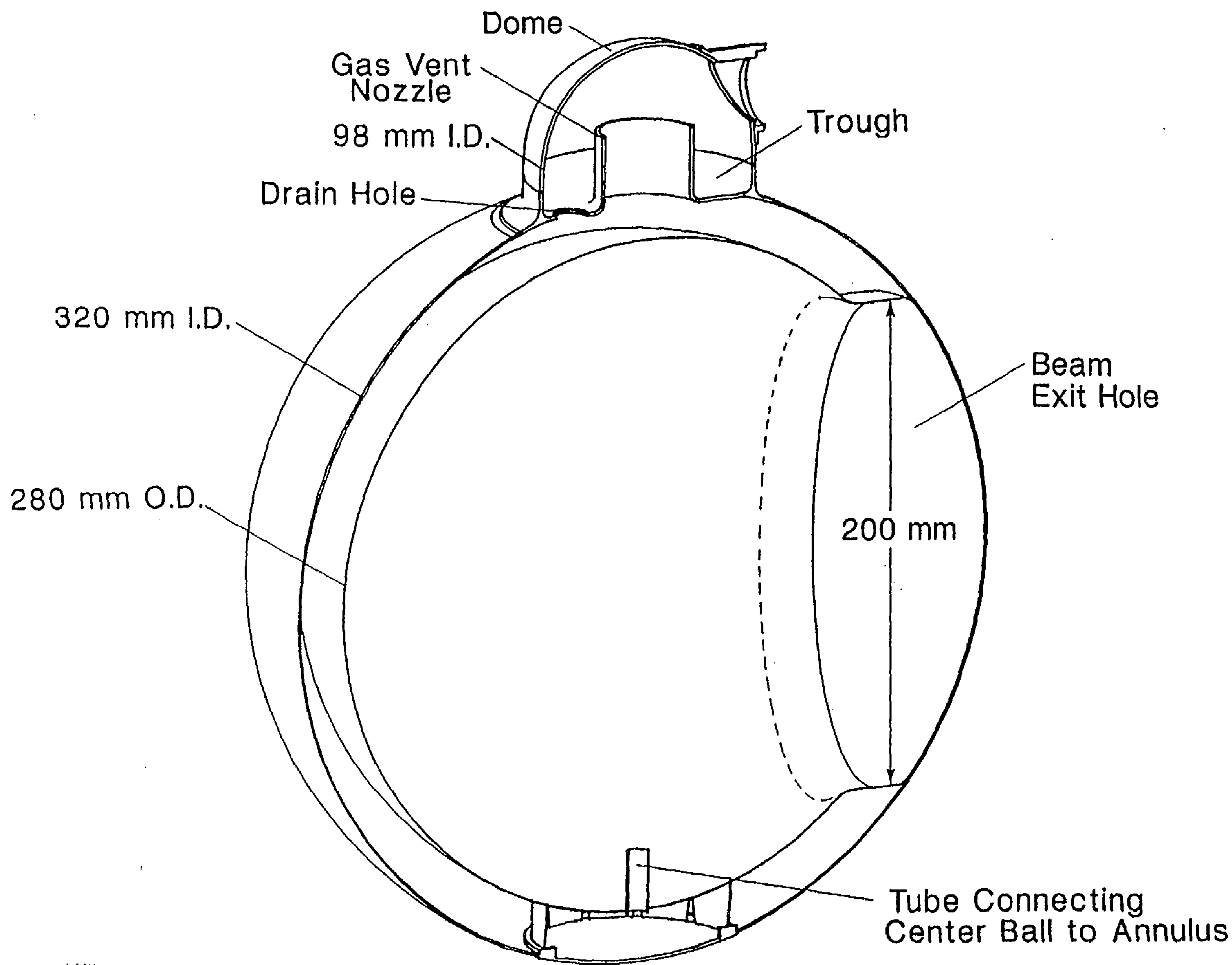


Figure 1. Design of the liquid hydrogen containing neutron moderator chamber for the NBSR.

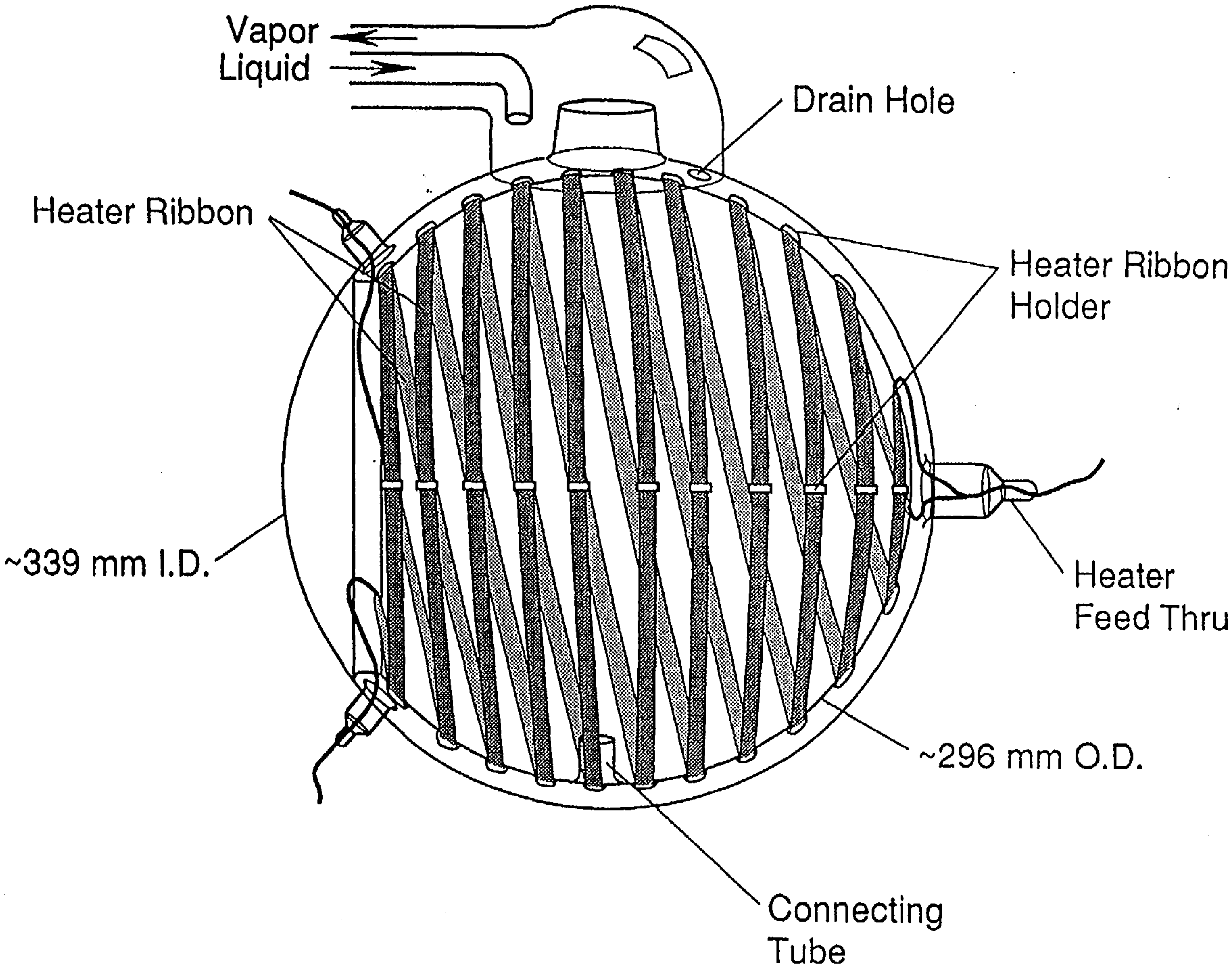


Figure 4. Drawing of the NIST-B electrically heated glass moderator chamber used to test the NBSR chamber.

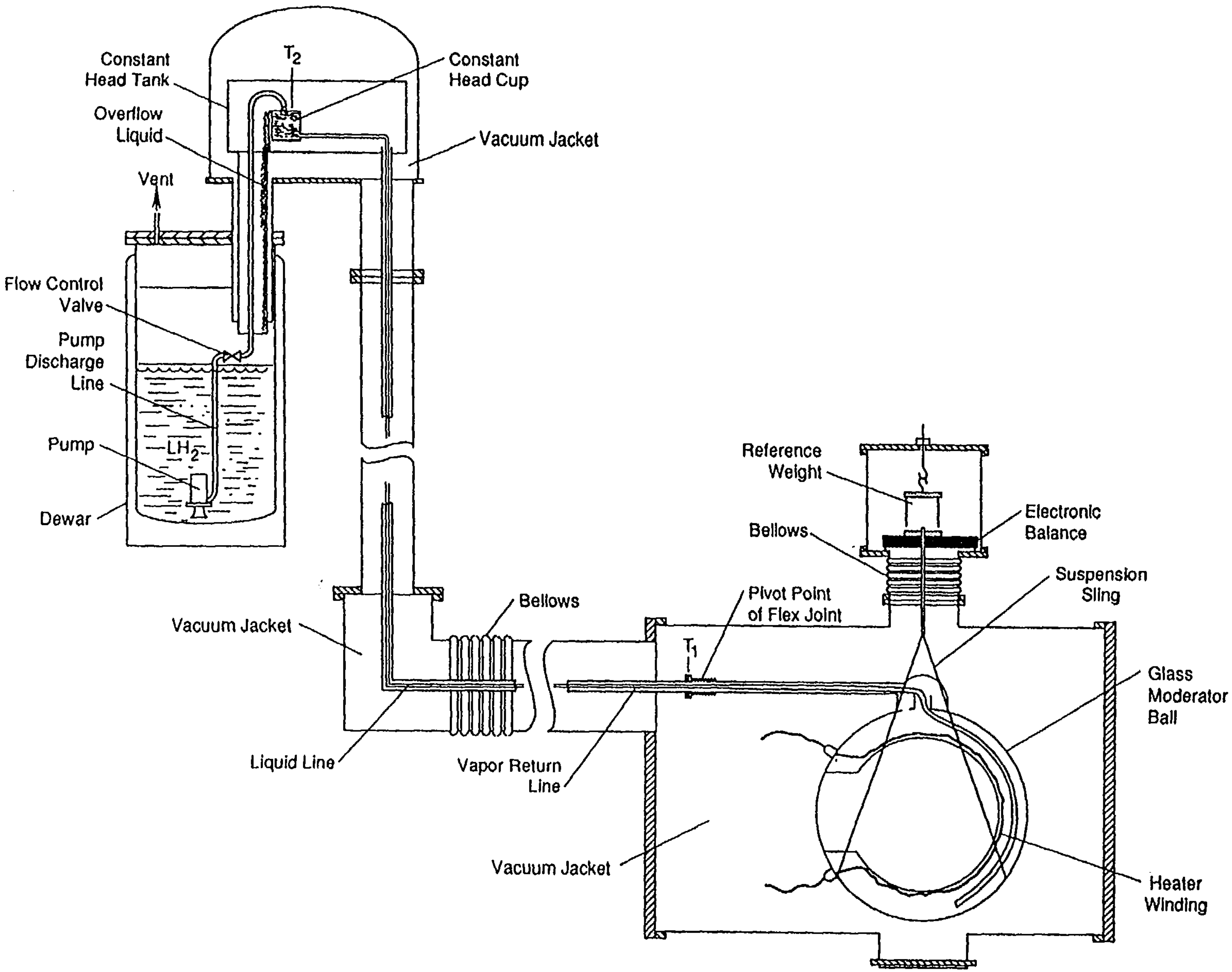


Figure 3. Unscaled diagram of the moderator test system built at NIST-B.

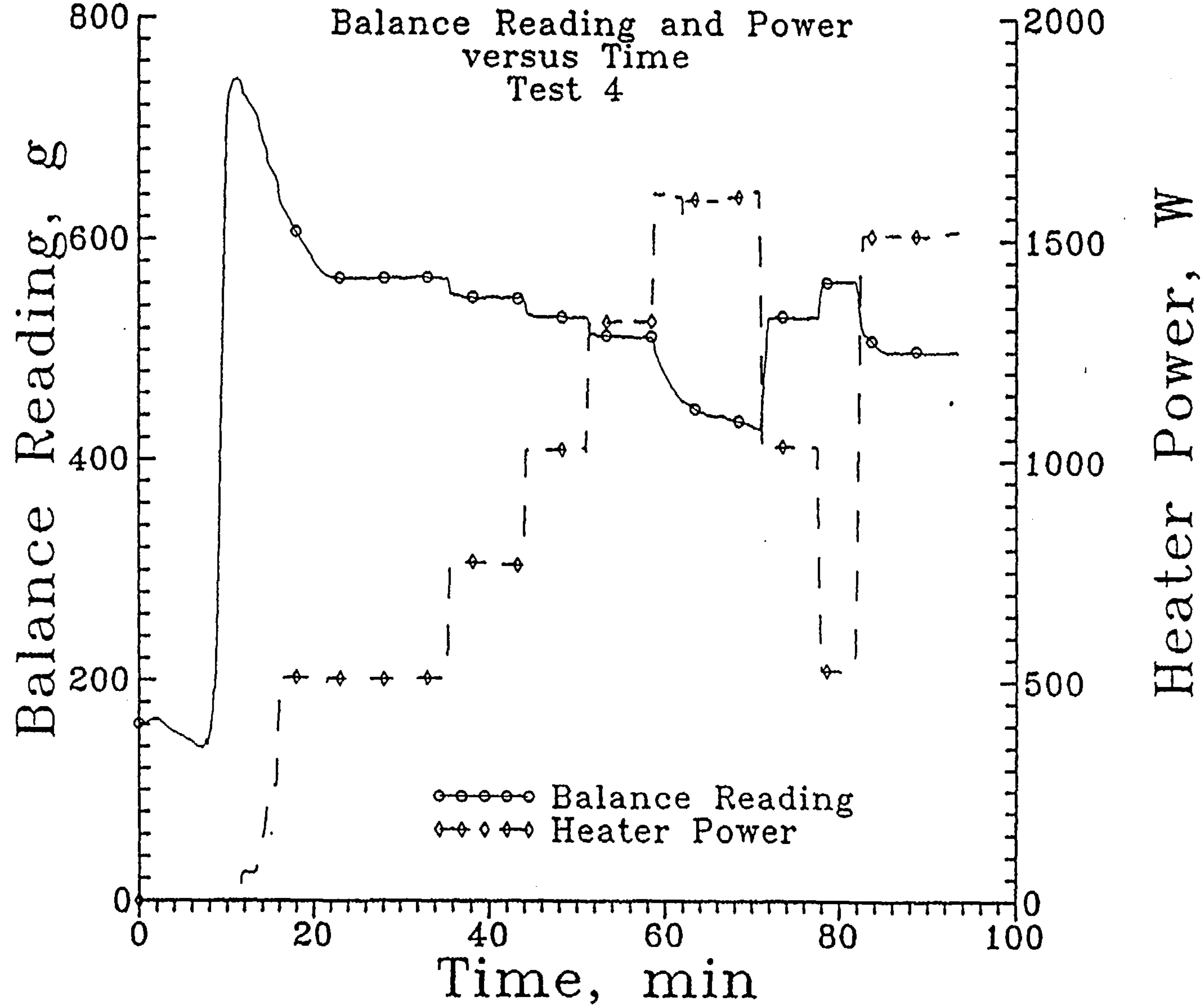
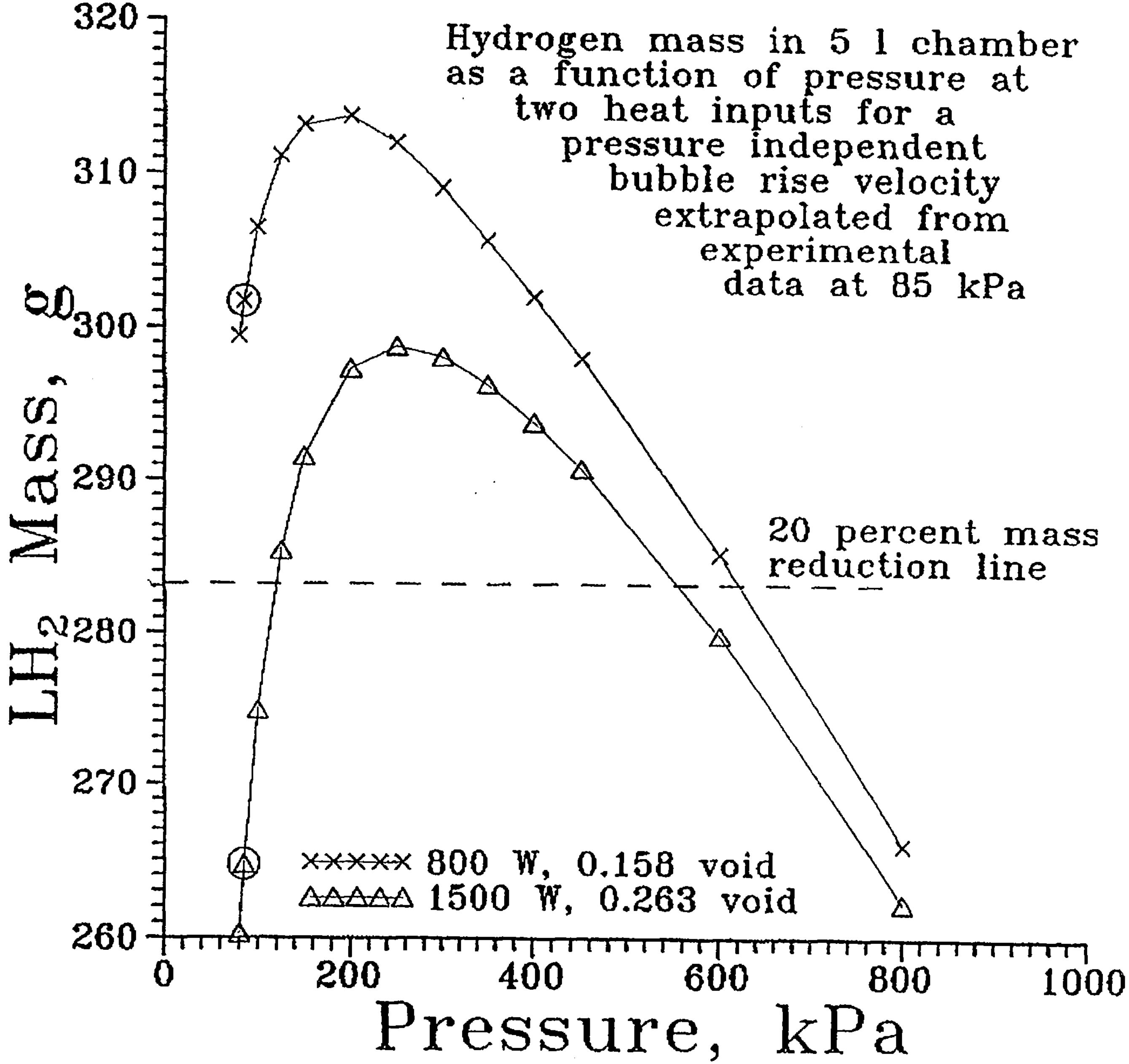
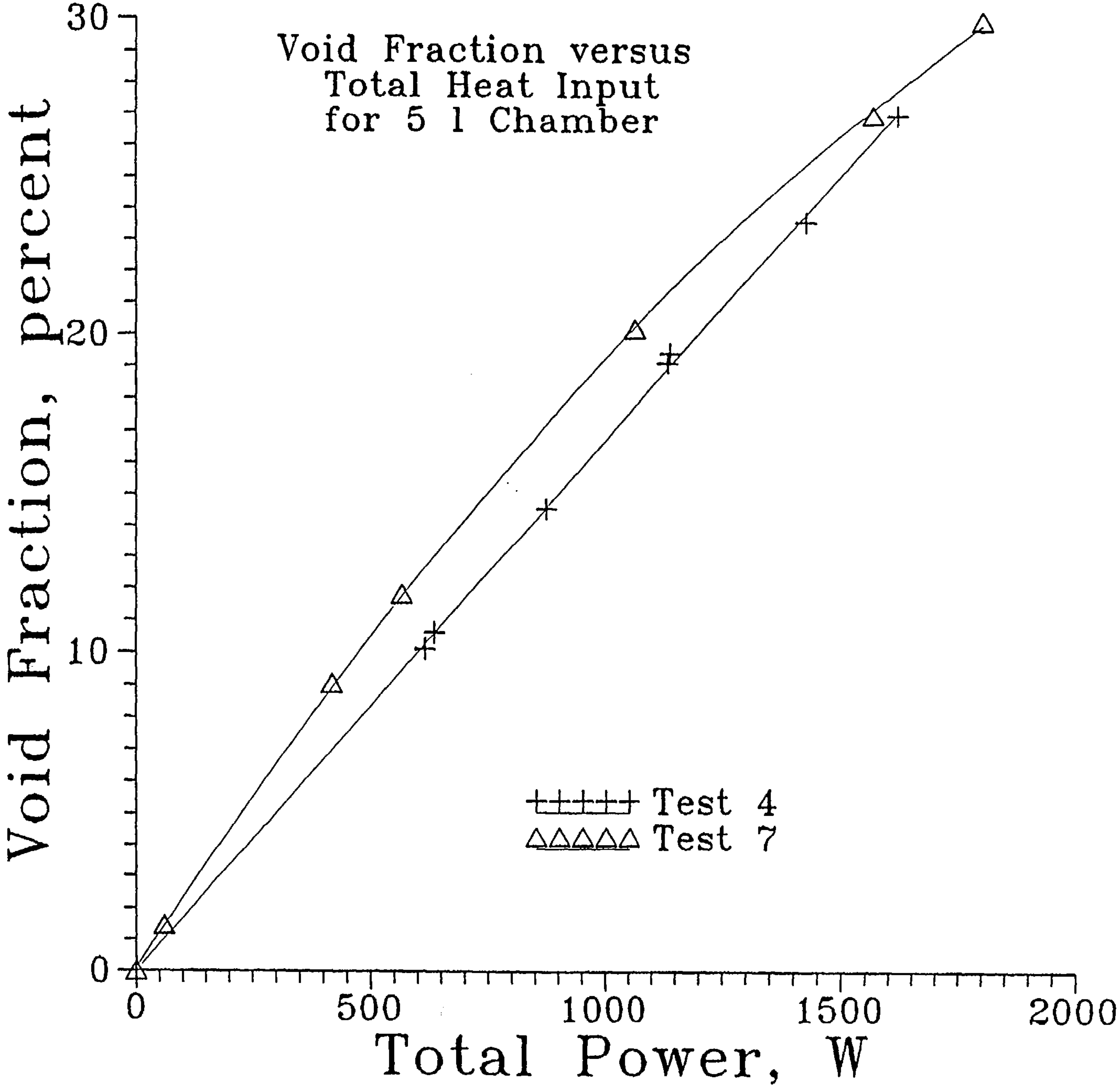


Figure 7. Test 4 showing the balance reading as a function of time at the power levels shown and 85 kPa pressure.



240

TOTAL P.02



Void Fraction versus
Total Heat Input
for 5 l Chamber
Void Fraction, percent
Total Power, W
Test 4
Test 7
0
10
20
30
0
500
1000
1500
2000

Results of Boulder Tests

1. Stable operation possible up to at least 2200 watts with two-phase flow.

2. LH_2 mass quickly reaches new, stable value after heat load change.

3. Void fraction well below 20 % at anticipated power and pressure.

4. Restart of LH_2 flow verified after extending supply line.

5. Visual inspection showed no dryout or unexpected voids.

XA04C1693

Modeling & Analysis of Liquid Deuterium-Water Reactions

by

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May 24-25, 1995

Prepared for Presentation at IGORR-IV, Gatlinburg, TN, USA

This Presentation Will Highlight

- o Overview of LD_2-Water reactions & their connections to research reactors with cold sources
- o Some key features and ingredients of vapor explosions in general
- o Examination of results of 1970 experiment at Grenoble Nuclear Research Center
- o Thermodynamic evaluations of energetics of explosive LD_2-D_2O reactions

**** This presentation will concentrate only upon the technical aspects of LD_2/LH_2 Water reactions; it is not intended to draw/imply safety-related conclusions for research reactors ****

Notes on Vapor Explosions

******* It is well-known from several Freon-water, LNG-water experiments and experiences that such interactions can be explosive under the right circumstances *******

- o Vapor explosions (also referred to as FCIs) occur (if they do so) in 3 stages:

  - Intimate premixing of hot and cold fluids
  - Triggering to initiate film collapse and dispersion --> explosive heat transfer
  - Propagation through mixture ---> pressure buildup and mechanical work

- o An LD_2-Water explosion would fall in the general category of FCIs where water is now the hot fluid

- o Important effects and features to keep in mind are:

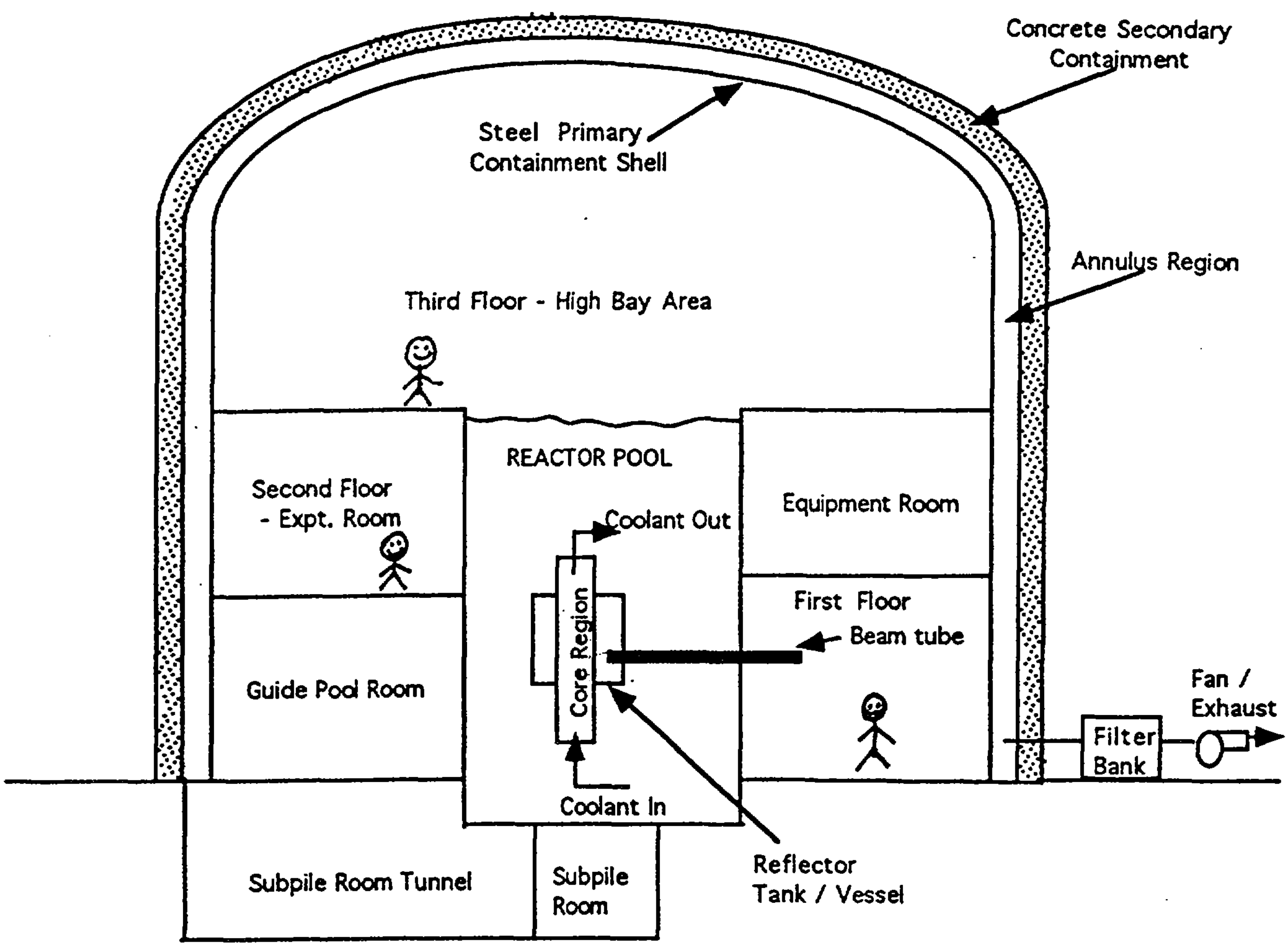
  - Initial contact mode (e.g., injection, stratification, radial egress, etc.)

  - Scale effects (small quantities usually need robust external triggering compared with large scale explosions)

  - Thermodynamic states of hot and cold fluid

  - Geometry of reaction zone (inertial constraint)

ANS CONTAINMENT



Grenoble Experiments

- o Geometry was carefully engineered to represent a scaled-down representation of ILL cold source within the reflector tank

- o Experiment parameters vs ILL reactor cold source

Parameter	Experiment	ILL Reactor
-Cold source fluid	LH_2	LD_2
-Source volume (L)	.025 to 1	38
-Source geometry	double walled (glass)	double walled (aluminum)
-Distance from source to reflector tank (m)	0.4	0.7

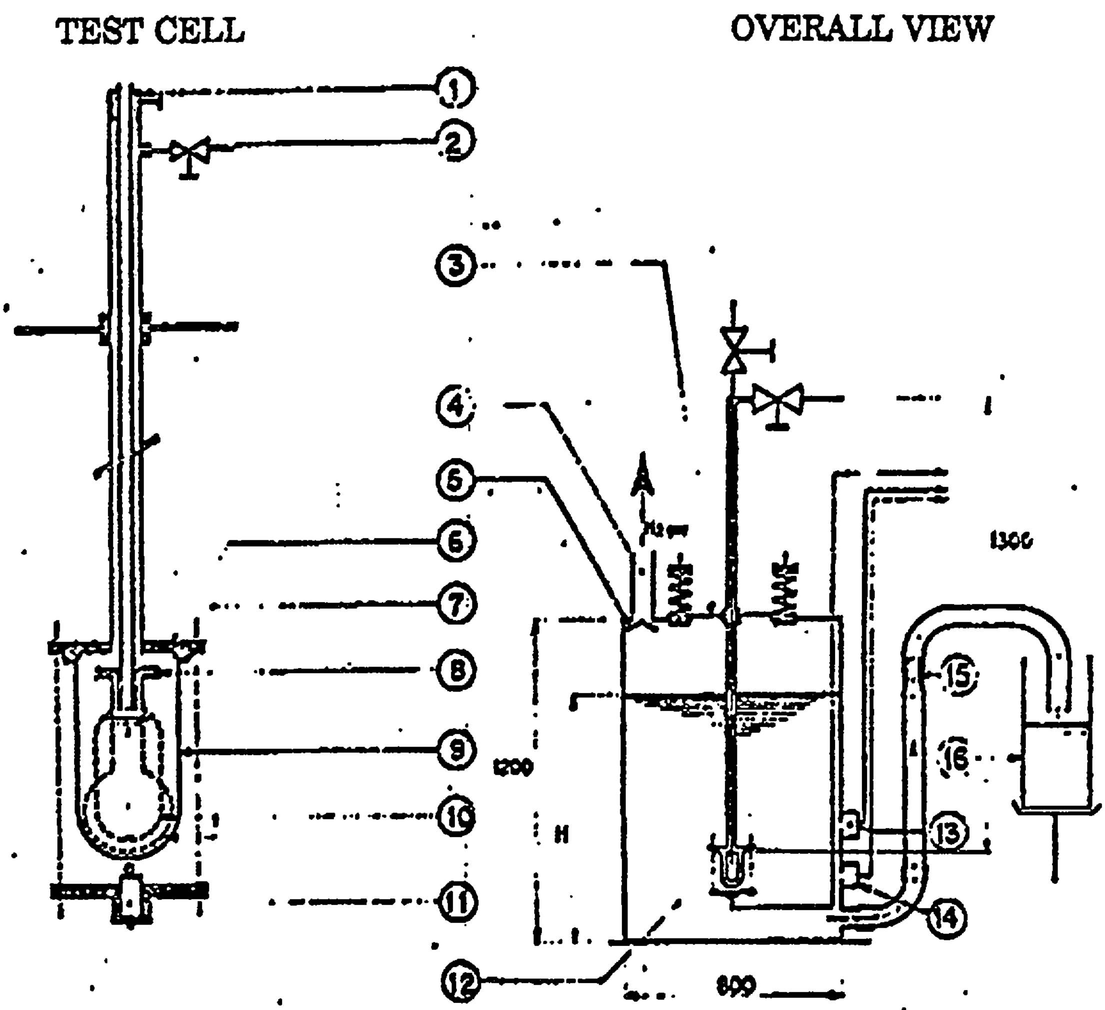
- o Instrumentation

  - Pressure taps at walls (response time ?), visual & camera film (<200 fps)

- o Experiment types

  1) Impact hammer induced double-wall perforation ---> No explosion
  2) Internal pressure buildup-induced forced ejection --> Explosive reaction

ORNL/OLS - 94/52



1.	Direct passage valve	8.	Strap and O ring
2	Vacuum plug	9.	PYREX vacuum bell
3.	Valve	10.	PYREX container of LH2
4.	Stack	11.	Striker
5.	Deflector	12.	Water
6.	Strap	13.	Membrane manometer
7	Rubber gasket	14.	Piezoelectric manometer
		15.	Venturi
		16.	Container

Figure 1. Schematic of Experimental Facility (dimensions in mm)

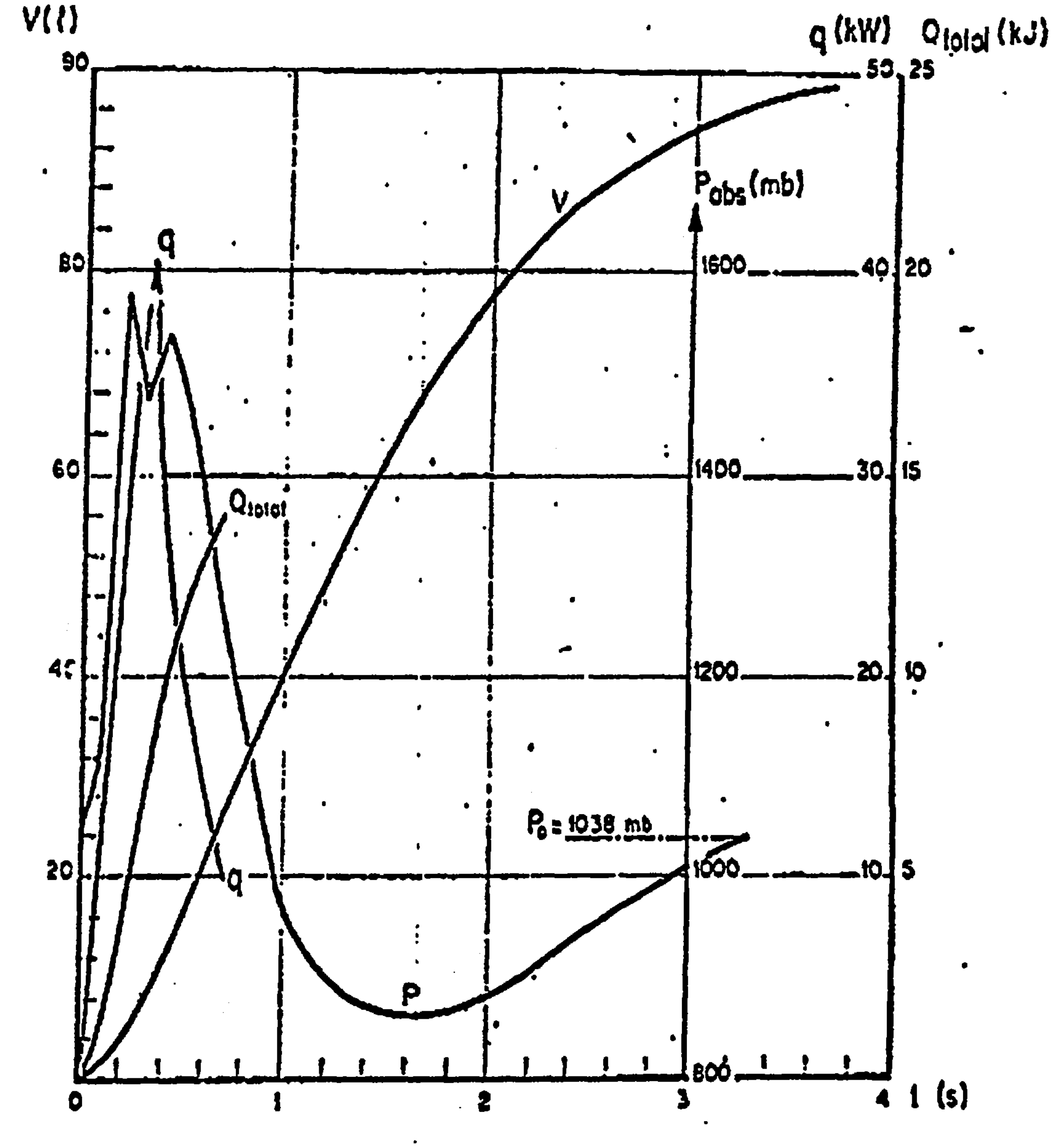


Fig: 4 – V_0 = 236 cm^3 LH_2, sans ciel.

MODE OF CONTACT IS IMPORTANT

- o Several Type 1 experiments were conducted by breaking the walls locally using an impact hammer

  - No explosions occurred, although significant vapor is formed over 1-3 s
  - Localized breakage of walls leads to significant bubbling, and relatively gradual mixing with water through "slits" causing vaporization of LH_2

  ---> *Such a contact mode can not be expected to result in explosions as the principal criterion of premixing with hot fluid is not present; Grenoble experiments clearly demonstrate this aspect.*

- o Type 2 experiment gave rise to explosive interaction between LH_2 & Water

  - Overheating and pressurization to 1.5 MPa by breaking the vacuum led to bursting of walls and forced ejection into the bulk coolant
  - Excellent premixing followed by localized spontaneous triggering is evidently sufficient to cause explosive thermal energy transfer and vaporization of LH_2 **** No data are given on pressure traces, etc. ****

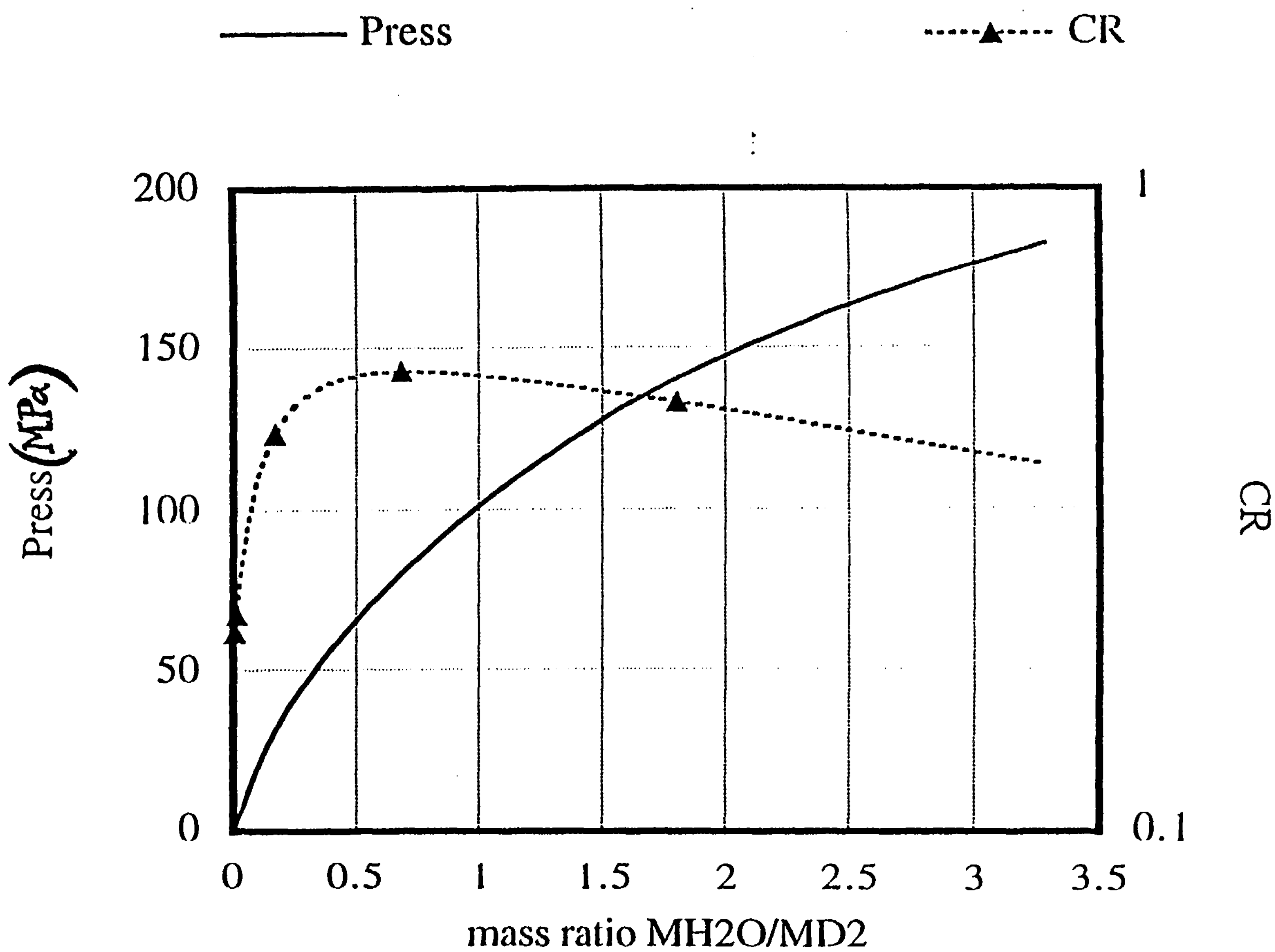
  ---> *Contact modes that force premixing will likely lead to explosions*

Note: 1 ml of LH_2 at 20.3 K = 55 ml of gas at 20.3 K = 850 ml of gas at 293 K

ENERGETICS OF EXPLOSIVE LD_2-WATER REACTIONS

- o MODELING OF ENERGETICS CAN BE DONE MECHANISTICALLY & ALSO USING THERMODYNAMIC MODELS
  - But, mechanistic models for modeling cryogenic fluid-water explosions are not well developed
  - Thermodynamic models of vapor explosions can be used to provide physically bounding estimates (but should be used with caution since perfect mixing is assumed and no directional effects are considered)
- o WE HAVE UTILIZED THERMODYNAMIC MODELS (to evaluate reasonable upper bound estimates of pressurization, and thermal-to-mechanical energy conversion for Advanced Neutron Source beyond design basis accident studies)
  - Hicks-Menzies model: Essentially adiabatic mixing followed by isentropic fuel-coolant expansion
  - Board-Hall model: Essentially simulation of C-J shock front to a given pressure followed by isentropic fuel-coolant expansion

Note: Actual properties of LD_2 were utilized; work is preliminary



Variation of Pressurization (Press) and Conversion Ratio (CR) with Mass Ratio (Hicks-Menzies Approach)

ADVANCES IN THE UNDERSTANDING OF U_3Si_2-Al DISPERSION FUEL IRRADIATION BEHAVIOR

J. L. Snelgrove, G. L. Hofman, J. Rest, and R. C. Birtcher

Argonne National Laboratory

SWELLING BEHAVIOR DURING IRRADIATION

- Around 1987, at the beginning of ANL's participation in the ANS Project, we had found that certain intermetallic compounds, e.g., UAl_x, U_3Si_2, and USi, exhibited very stable swelling behavior to full burnup with LEU, while other compounds, e.g., U_6Fe, U_6Mn, and U_3Si became unstable before full burnup was achieved.
- We concluded that the unstable, or break-away, swelling is associated with irradiation-induced amorphization and that the compounds exhibiting stable swelling remained crystalline during irradiation.

SWELLING BEHAVIOR DURING IRRADIATION (CONT'D)

- Irradiation of MEU and HEU U_3Si_2 showed that the swelling fell below the LEU values. Detailed analysis revealed a two-stage swelling behavior with an apparent fission rate effect.

- The fission gas bubbles in U_3Si_2 were distributed in a very regular pattern to the highest fission densities reached in our tests in the ORR (for ~65% burnup in HEU particles). The bubbles grew in diameter with increasing fission density but did not coalesce.

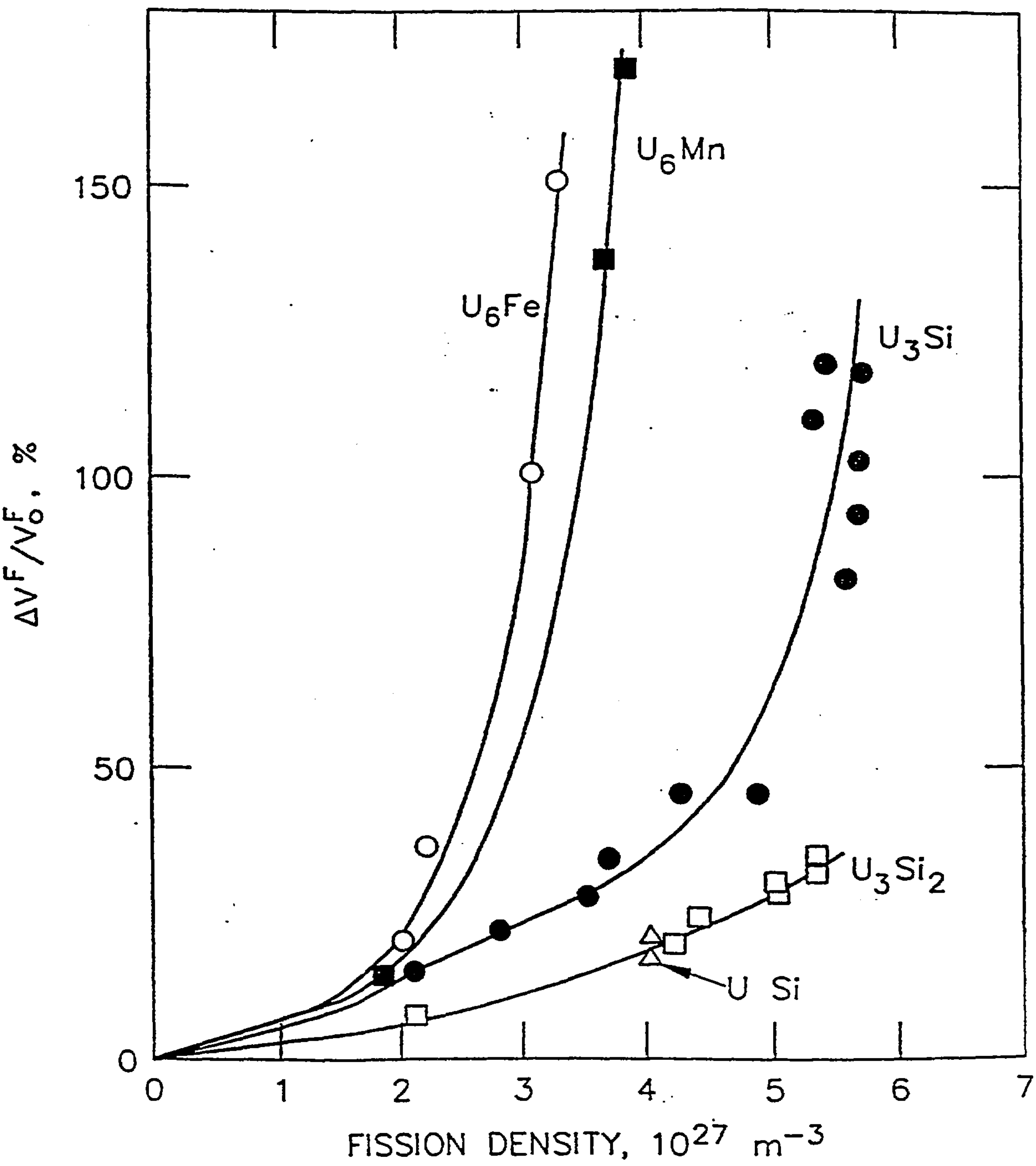
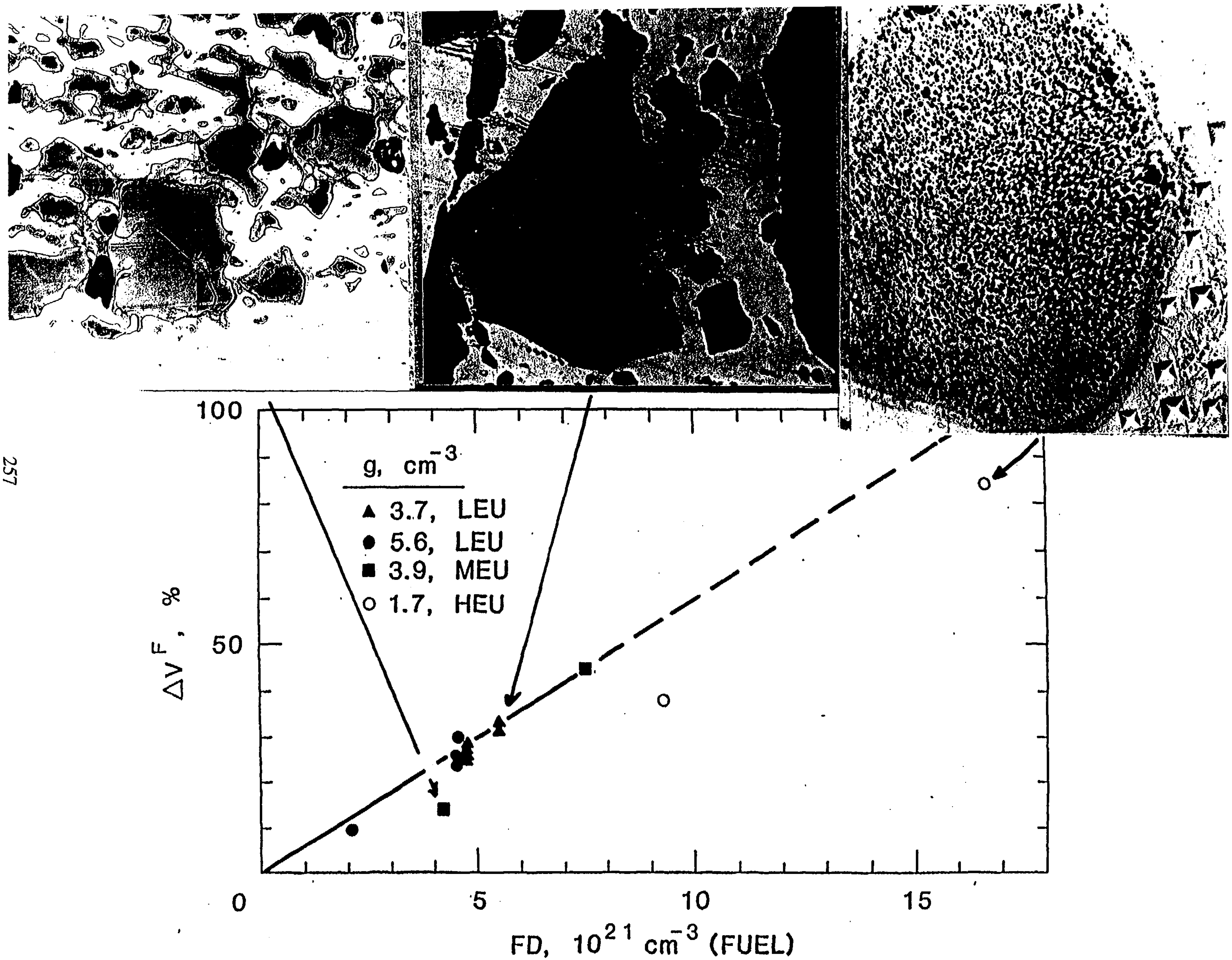
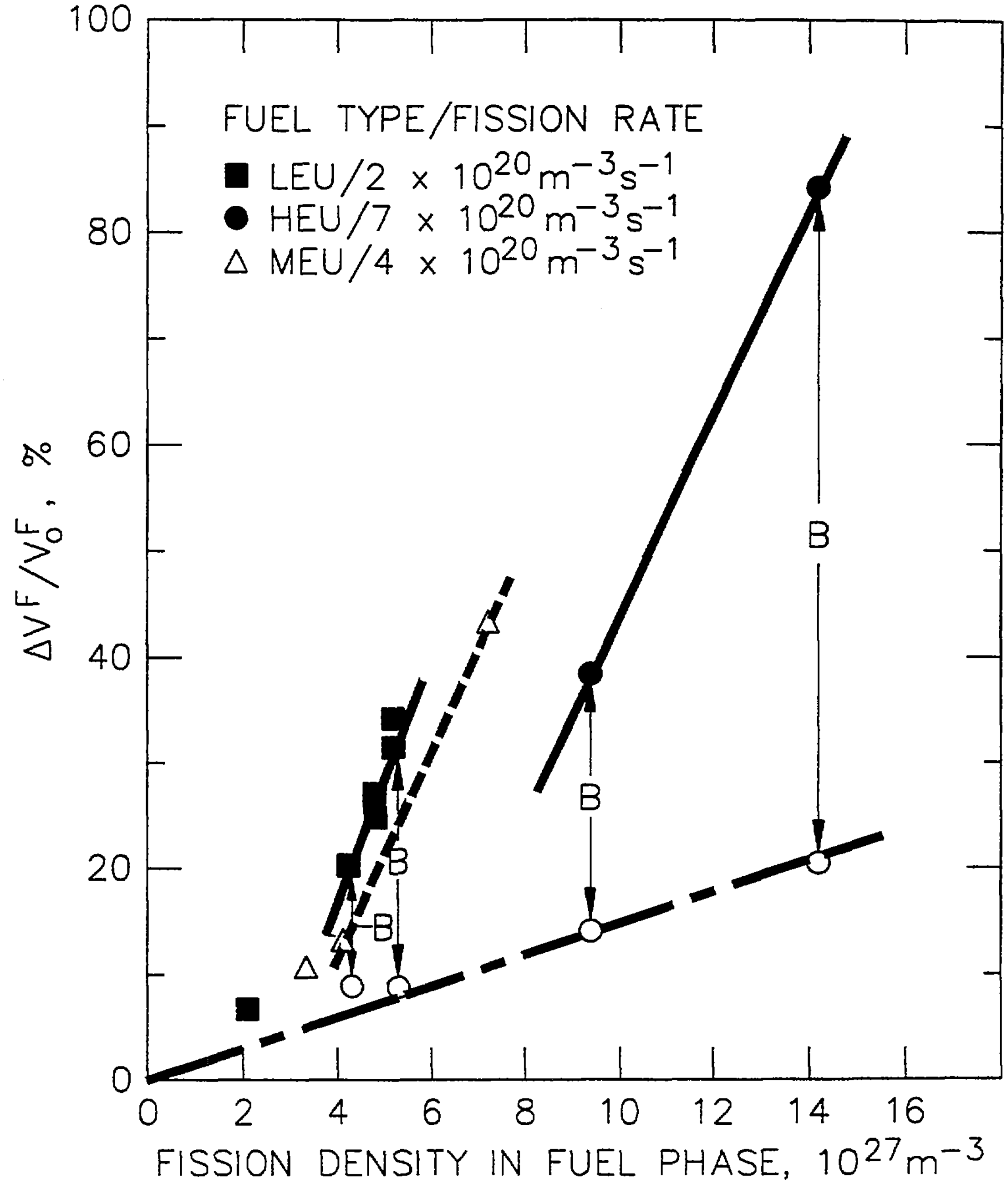
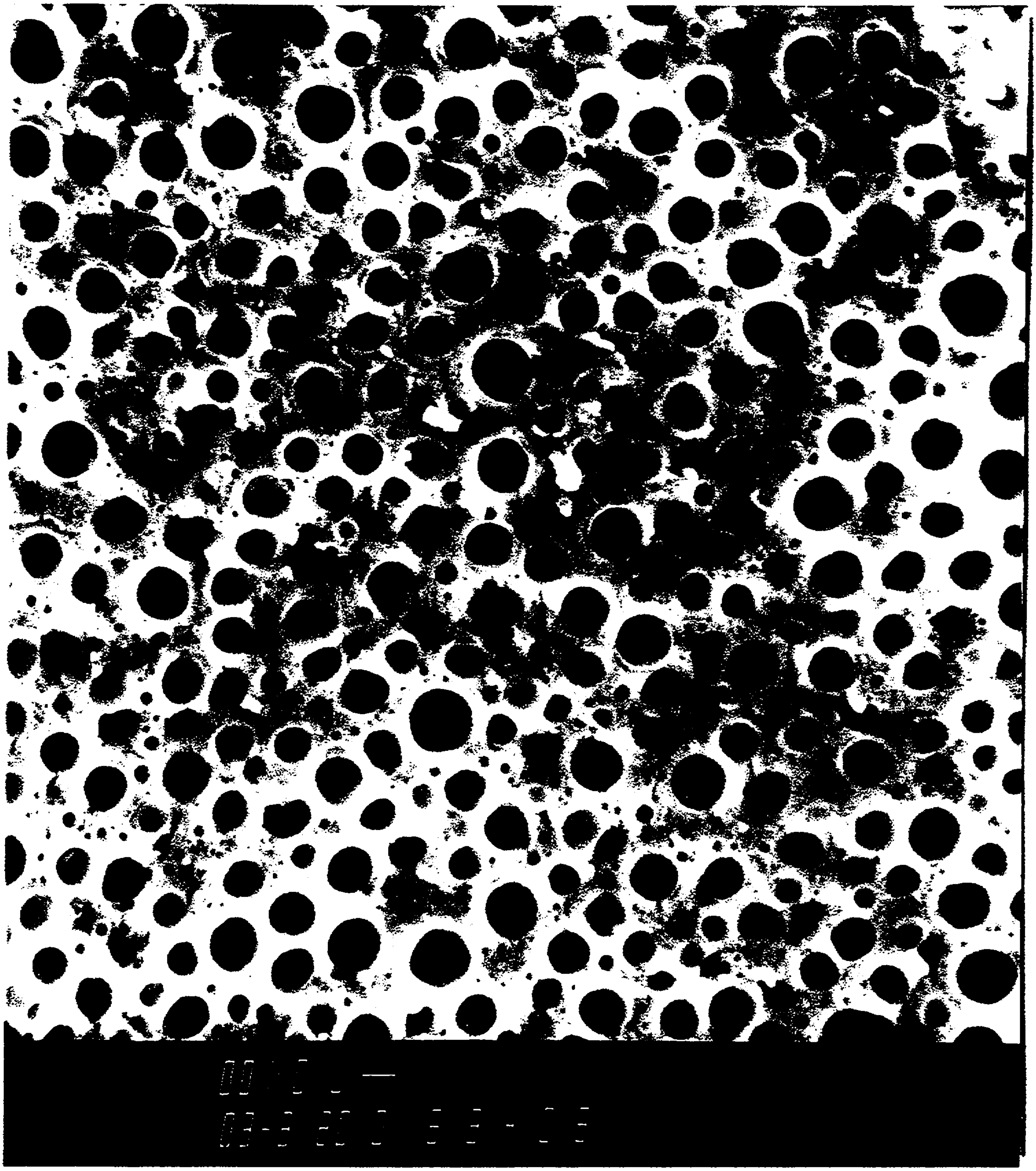


Fig. 1. Swelling of various LEU intermetallic fuel compounds as a function of accumulated fissions in fuel.


g, cm^{-3}
3.7, LEU
5.6, LEU
3.9, MEU
1.7, HEU
100
50
0
5
10
15
ΔV^F, %
FD, 10^{21} cm^{-3} (FUEL)


FUEL TYPE/FISSION RATE
LEU/2 x $10^{20} m^{-3} s^{-1}$
HEU/7 x $10^{20} m^{-3} s^{-1}$
MEU/4 x $10^{20} m^{-3} s^{-1}$
$\Delta V^F/V_o^F$, %
FISSION DENSITY IN FUEL PHASE, $10^{27} m^{-3}$
B
0
20
40
60
80
100
2
4
6
8
10
12
14
16



Bubble morphology in HEU, U_3Si_2@ ~70% Bu, ORR (SEM)

SWELLING BEHAVIOR DURING IRRADIATION (CONT'D)

- Further investigation revealed two problems:
  - Neutron diffraction studies of fuel irradiated in the IPNS at ANL indicated that both U_3Si and U_3Si_2 became amorphous at very low doses.
  - Our swelling model could not reproduce the bubble distribution seen in U_3Si_2.

SWELLING BEHAVIOR DURING IRRADIATION (CONT'D)

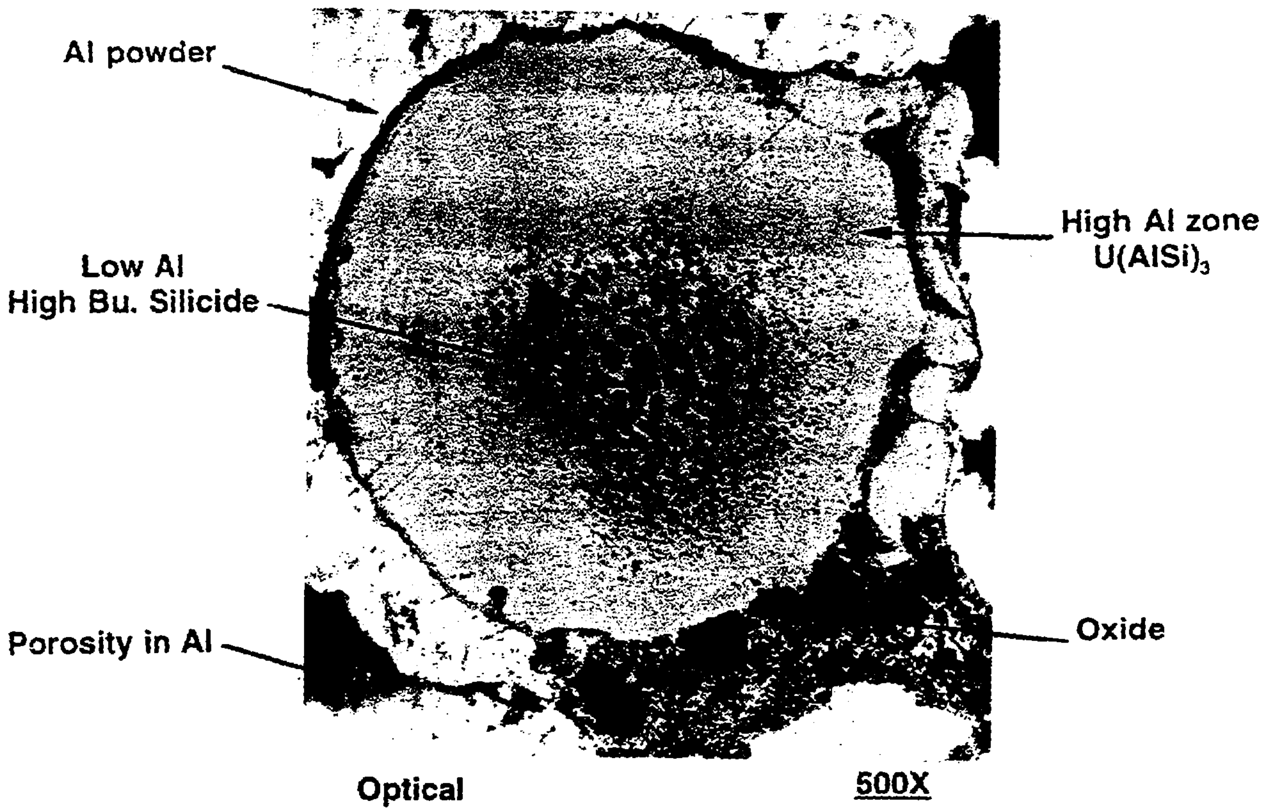
- The solution was to postulate the formation of subgrains in U_3Si_2 at the fission density of the knee of the swelling curve. These could be formed by subdivision of grains of crystalline material or by recrystallization of amorphized material.
  - Subgrains had been found in UO_2 and U_4O_9, with similarities to U_3Si_2 in the fission gas bubble morphology.
  - Ion simulation experiments gave evidence of grain refinement.
  - We have now found evidence of subgrains in U_3Si_2 irradiated in the HFIR.

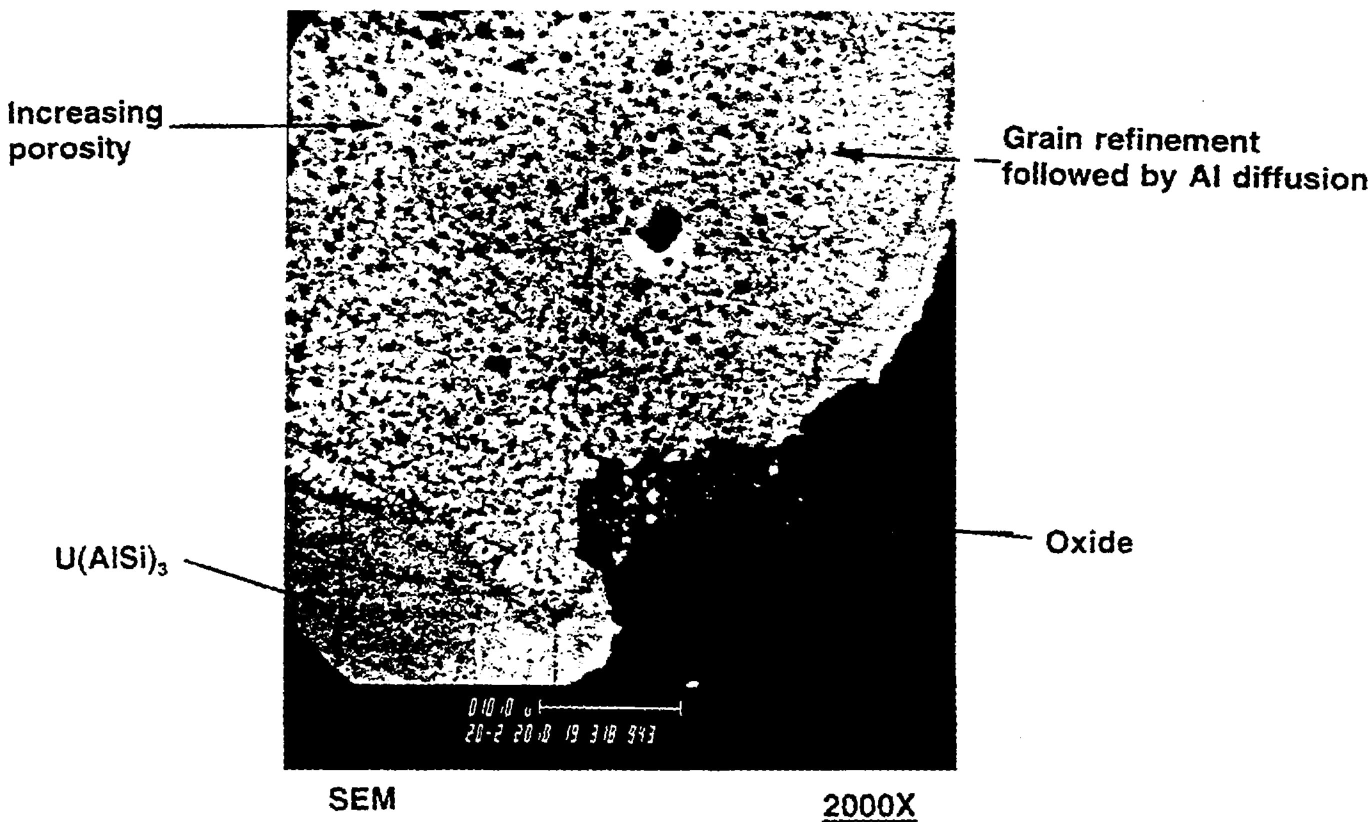
SWELLING BEHAVIOR DURING IRRADIATION (CONT'D)

- ANS irradiation conditions of temperature, fission rate, and fission density were well beyond those in the ORR irradiations.
- Examination of samples irradiated in the HFIR target region has provided further insights into the behavior of U_3Si_2.
- We postulate a multi-stage process:
  - The outer portion of the U_3Si_2 particle reacts with Al to form a compound with a UAl_3-type structure, $U(SiAl)_3$. This material has the high stability characteristic of UAl_3; no fission gas bubbles are seen.

SWELLING BEHAVIOR DURING IRRADIATION (CONT'D)

- Inside this zone subgrains form in the unreacted U_3Si_2 when the fission density reaches the knee of the swelling curve, and typical swelling ensues.
- The center of many fuel particles, however, show bubbles characteristic of the unstable, amorphous materials. We postulate that depletion of the U has led to the formation of an unstable phase in high-burnup HEU fuel particles. We think that this phase may be the peritectoid USi_2, analogous to U_3Si. Break-away swelling is restrained, however, by the outer $U(SiAl)_3$ shell.


Al powder
Low Al
High Bu. Silicide
Porosity in Al
High Al zone
U(AlSi)$_3$
Oxide
Optical
500X


Increasing porosity
Grain refinement followed by Al diffusion
U(AlSi)$_3$
Oxide
SEM
2000X

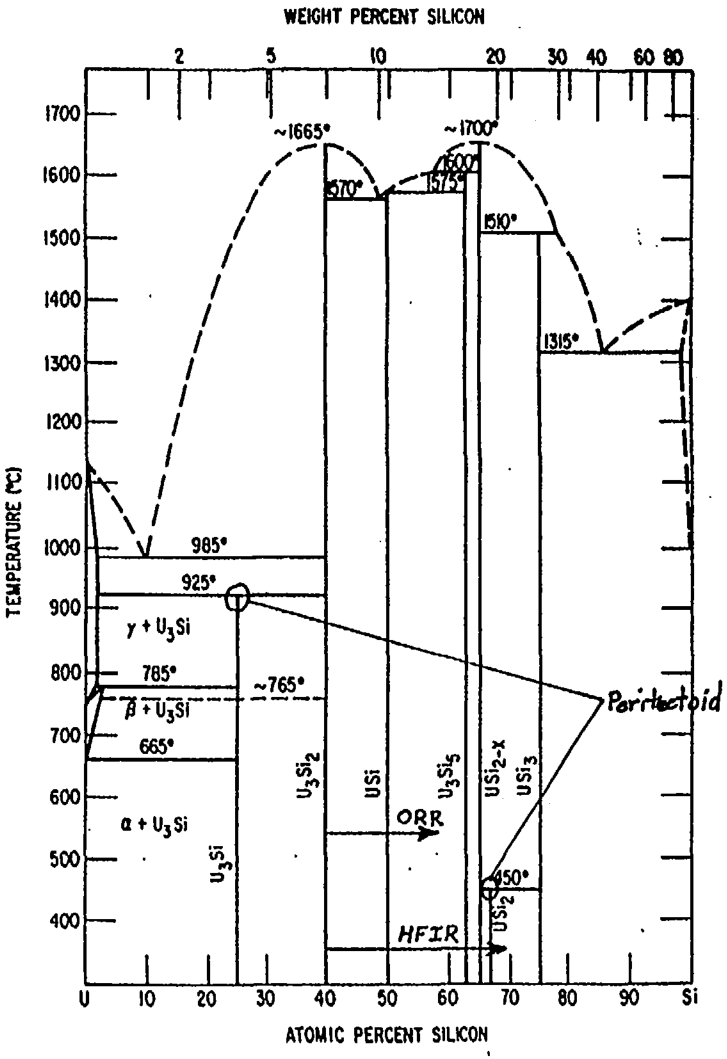
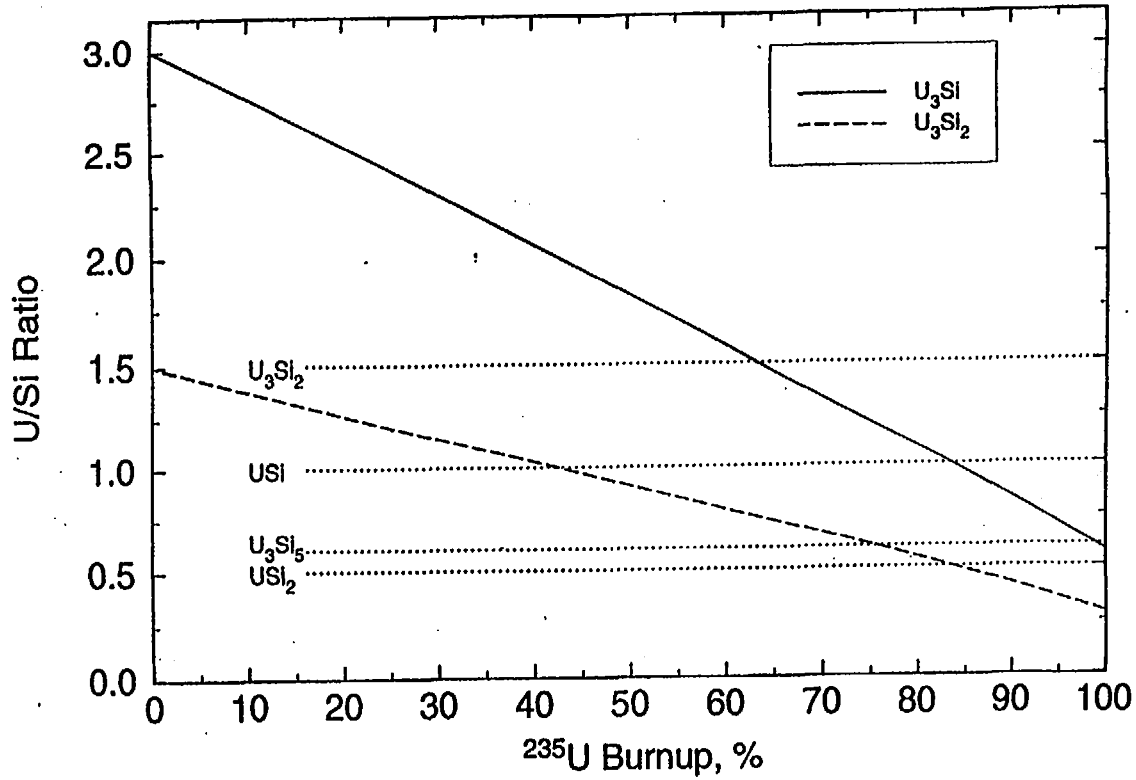
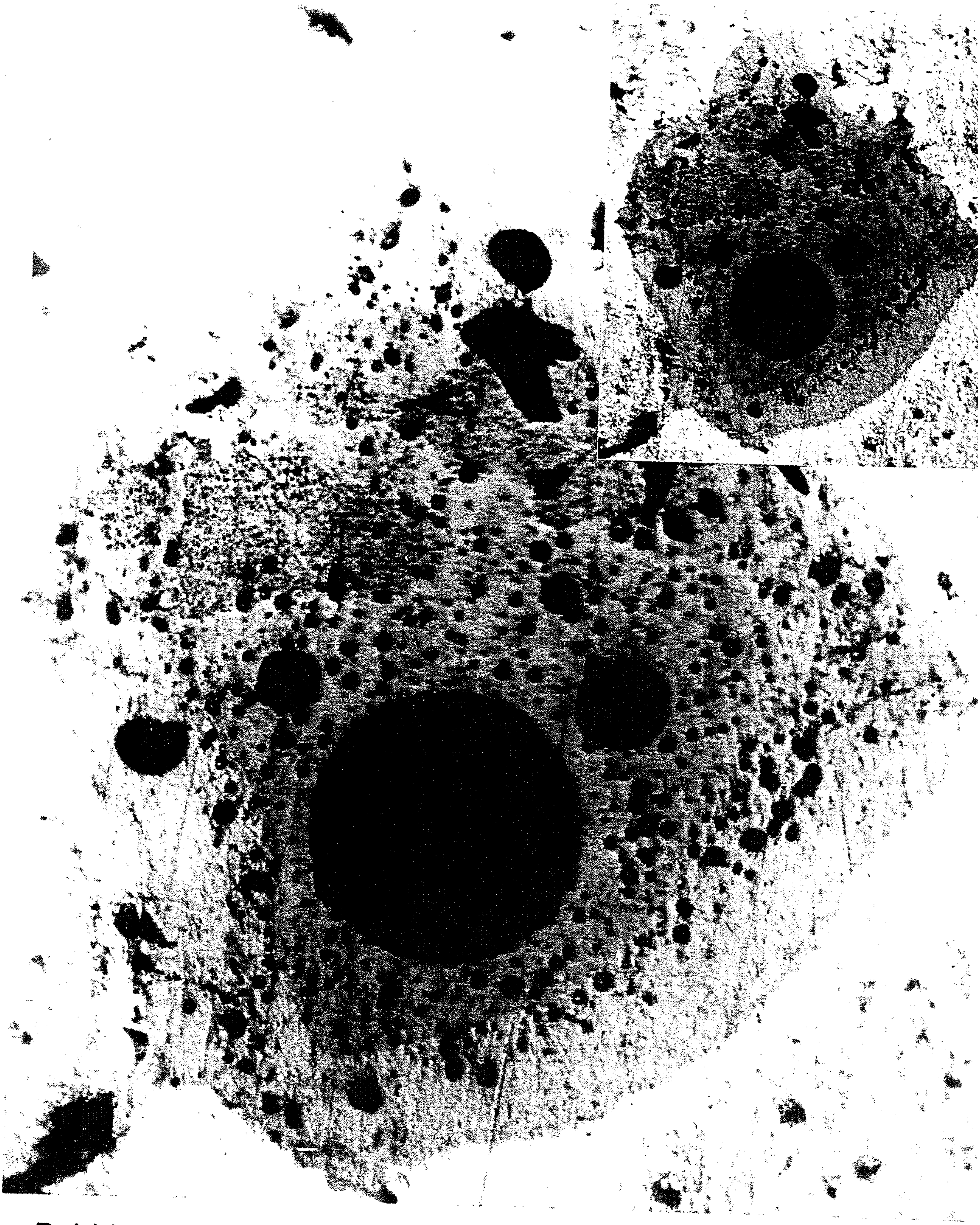


FIG. 4. CHANGE IN U/Si RATIO VS. ^{235}U BURNUP AS 93%-ENRICHED U_3Si AND U_3Si_2 ARE BURNED





Bubble morphology and Al interaction @ ~85% Bu, 250°C, HFIR

THERMAL CONDUCTIVITY CALCULATIONS

THERMAL CONDUCTIVITY MODEL

- Model developed and implemented in the Dispersion Analysis Research Tool (DART) code by J. Rest, ANL.
- Thermal conductivity is a function of fuel and porosity volume fractions, which are calculated based on a mechanistic model of fuel particle swelling.
    - Fuel particle swelling model has evolved. Calculations now based on multiphase model of fuel swelling which fits both ORR and HFIR data.

THERMAL CONDUCTIVITY MODEL (CONTINUED)

Model Basis

- Spherical fuel particles are mixed into matrix aluminum.
- As-fabricated porosity is mixed into fuel meat.
- Irradiation-induced porosity is mixed into fuel particles.
- Aluminum matrix is assumed to be the continuous medium.
- Successive application of classical mixing formula gives equation for thermal conductivity as a function of fuel and porosity fractions.
- Fuel meat volume is assumed to remain constant as fuel particles swell--porosity is first closed then aluminum removed from meat.

THERMAL CONDUCTIVITY MODEL (CONTINUED)

<u>Thermal Conductivity Equation</u>

$$k_e^m = k_{al}\left[z_1 + z_2 F_v^{2/3} + z_3\left(k_e^g / k_{al}\right)F_v^{1/3}\left(1 - F_p^{2/3}\right) + z_4 F_p^{2/3} + z_5\left(F_v F_p\right)^{2/3}\right]$$

where

k_e^m, k_{al}, and k_e^g are the thermal conductivity of the fuel meat, matrix aluminum, and U_3Si_2 particles containing irradiation-induced porosity;

F_v and F_p are the fuel and as-fabricated porosity volume fractions;

and

z_1 - z_5 are geometric constants.

THERMAL CONDUCTIVITY MODEL (CONTINUED)

- The physical limits:
  - $k_e^m = k_{al}$ for $F_p = 0, F_v = 0$
  - $k_{em} = 0$ for $F_p = 1$
  - $k_{em} = k_{al}$ for $F_v = 1, k_{eg} = k_{al}$
  - $k_{em} = 0$ for $F_v = 1, k_{eg} = 0$

  imply that $z_1 = +1,\ z_2 = -1,\ z_3 = +1,\ z_4 = -1$

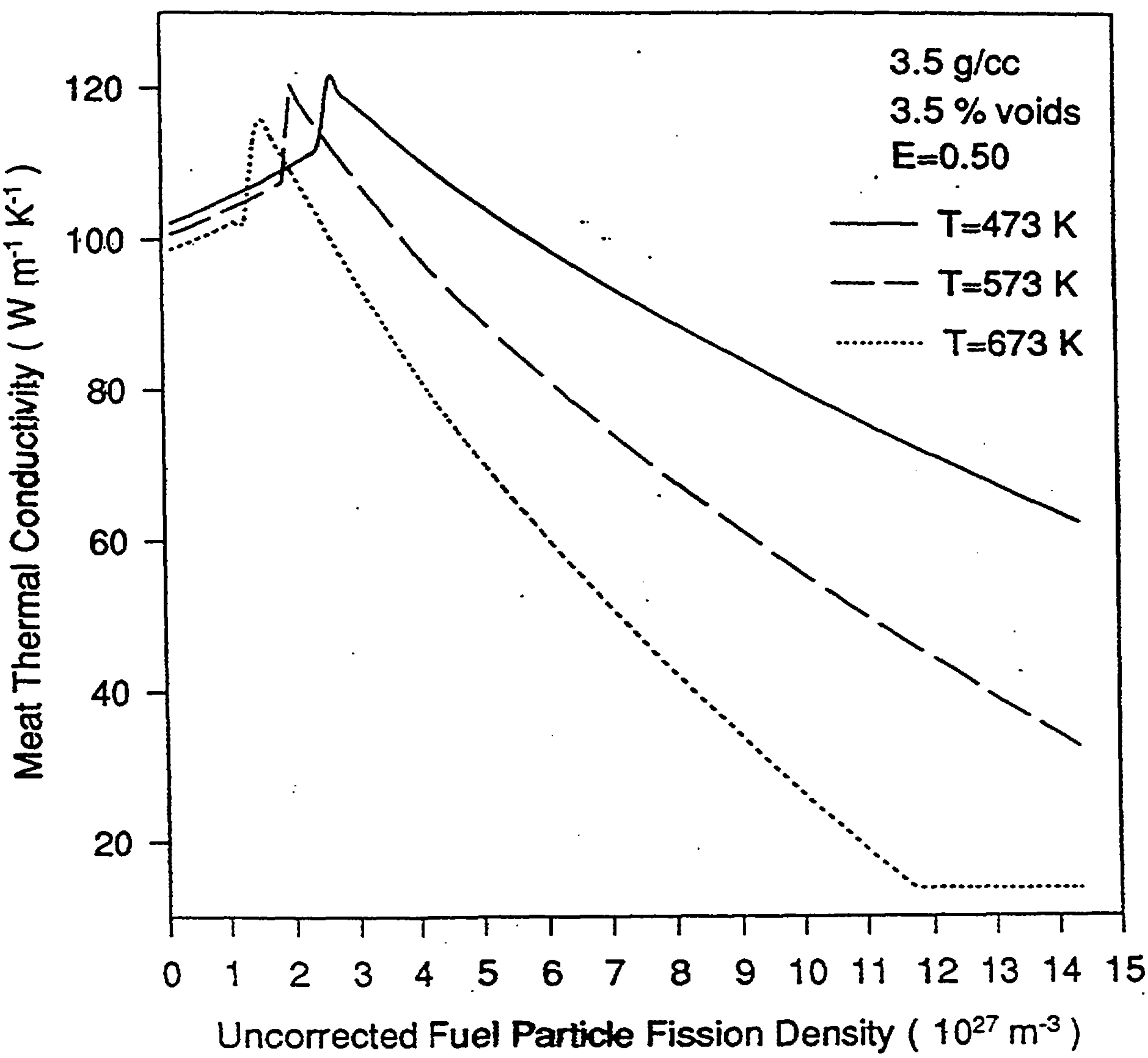
- z_5 is determined by fit to thermal conductivity data for unirradiated fuel to correct for idealized geometry assumption:
  - $z_5 = -0.3275$

THERMAL CONDUCTIVITY MODEL (CONTINUED)

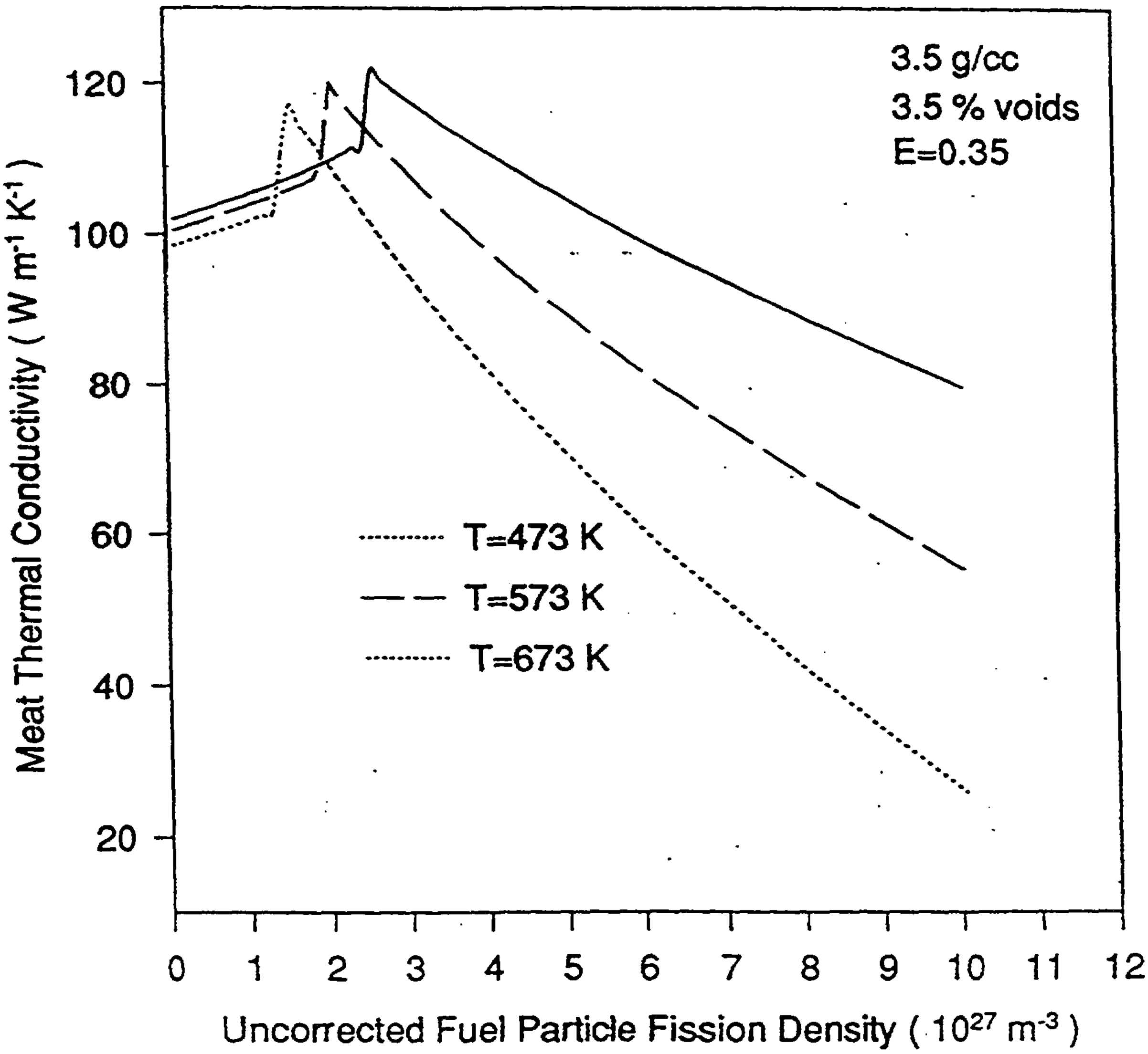
Model Limitations

- Treats only single size of fuel particle.
- Invalid when aluminum matrix is discontinuous, i.e., for less than 35 to 40 vol% aluminum.
- Does not account for any degradation of aluminum thermal conductivity due to irradiation.
- Accuracy of the results depends on the accuracy of the thermal conductivity and fuel swelling models and data. Fuel swelling depends on fission rate, total fissions, and temperature.

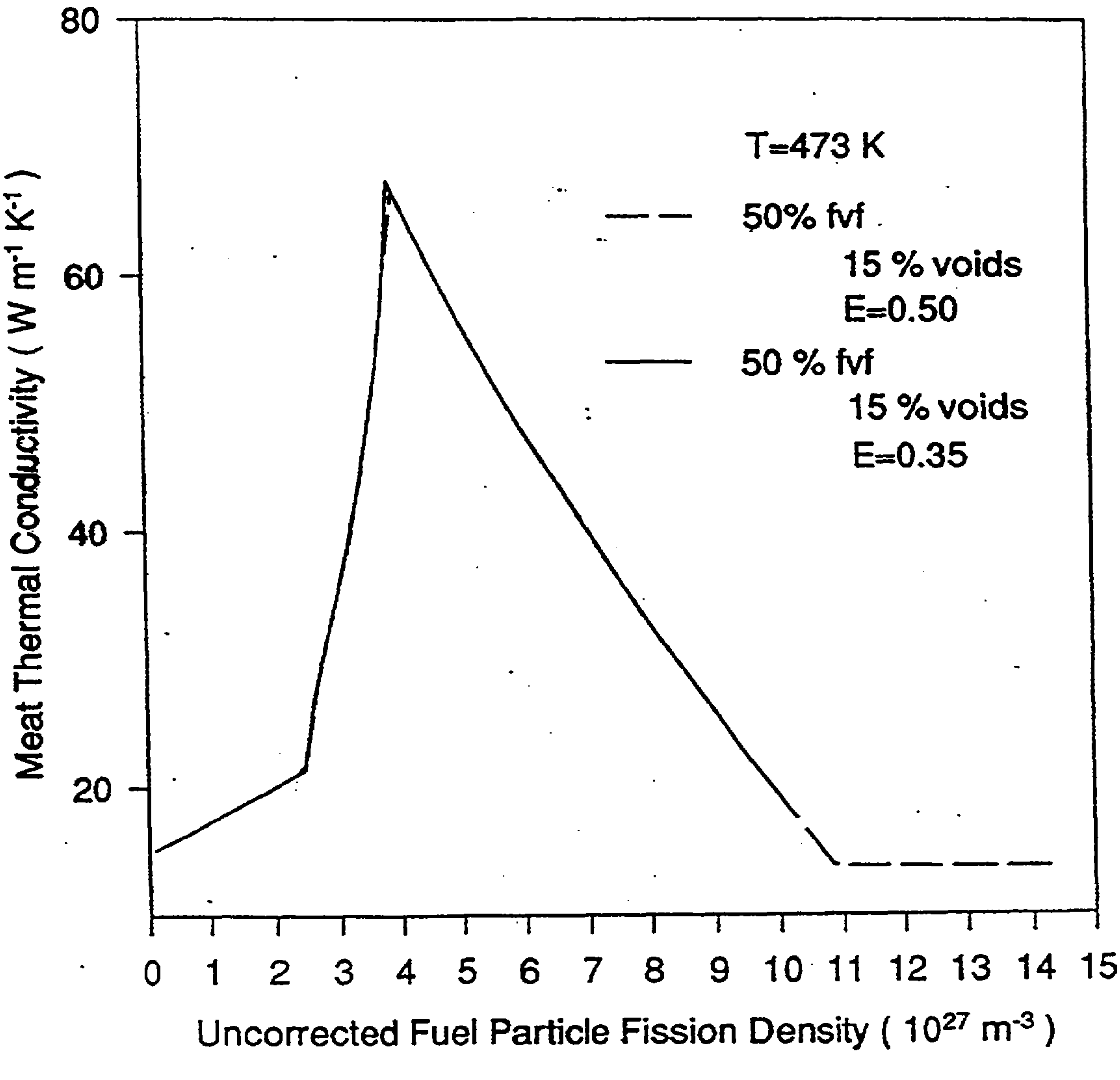
MULTINODE DART CALCULATIONS



MULTINODE DART CALCULATIONS



MULTINODE DART CALCULATIONS



CERCA

XA04C1695

STATUS OF DEVELOPPEMENT
MAIN FEATURES
OF PLATES AND DUMMY ELEMENT FABRICATION
FOR FRM II

C. Ailloud, P. Colomb, J.P. Durand, G. Harbonnier

CERCA
Zone Industrielle Les Berauds
26104 Romans sur Isere France

Presented at the IGORR-IV meeting
May 24-25 1995
Gatlinburg, Tennessee, USA

FRAMATOME
FABRICATION COMBUSTIBLE

CERCA

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ABSTRACT

Dummy element fabrication is part of the developpement of FRM II concept.

CERCA has received order from Technical University of Munich (Germany) to ensure industrial possibility for fabrication and control of such element.

Main points about plates and element fabrication, and some of major difficulties in performing this order are outlined in following report.

CERCA

1 INTRODUCTION

Since the beginning of the FRM II project CERCA, as a major research reactor fuel manufacturer, has been involved in the developpement of the element concept as far as fabrication is concerned.

CERCA has received from Technical University of Munich in december 1993 the order for fabrication of two dummy elements and about 400 fuel plates loaded with depleted uranium. One of these elements has to be made with aluminium plates, the other with depleted uranium plates.

This order was aiming at following goals:

For CERCA:

- set up an industrial process for manufacture of fuel plates including all necessary control and quality steps
- develop the various tools required for plates fabrication and control, element assembly and welding
- prepare of a whole set of paperwork, as the result of prototype fabrication, needed for German licensing and surveying authorities, to get final approval for real element fabrication and reactor start-up

For TUM:

- validate the design of the element in an hydraulic test loop
- allow some tests in the future reactor itself for handling and non active flow-tests

At the 1992 IGORR meeting in Saclay, CERCA had already presented some features of the fuel plates, but this was more the summary of some kind of laboratory work, than the report about an industrial fabrication.

This present paper, will emphasis the status of plates and dummy elements fabrication, considered as the beginning of an industrial stage for such a fuel.

2 SCHEDULE OF FABRICATION

The dummy elements fabrication is planned over a duration total of two years and a half, starting in december 93.

Main milestones for the order are:

- september 95	completion of depleted uranium fuel plates fabrication
- december 95	delivery of the first dummy element (with aluminium plates)
- march 96	delivery of the second dummy element (with depleted uranium plates)

3 FRMII FUEL ELEMENT DESCRIPTION

FRM II reactor is a compact core design and the fuel element is very similar from ILL RHF fuel element altough smaller. In order to get a size comparison, FRM II element could be introduced inside ILL RHF element !

General shape of the element is a cylinder about 300 mm diameter and 1000 mm high.

The top outer nozzle has a cone shape on which the element stays in the reactor, and inside this upper nozzle a grove is used by the handling tools during handling of the element

The bottom nozzle has six nuts to allow guiding the element in the water channel, and the inner upper part has also six nuts which purpose is to guide the control rod assembly inside the fuel element.

113 plates with involute shape are the active part of this element whose total uranium loading is 8.106 kg uranium, with a nominal U235 enrichment of 93% for the real element.

Two combs welded to each plate, one at the inlet and one at the outlet of fuel section ensure that in all conditions the water channels keep their width, and a sieve, drilled with about 5000 holes 1.8 mm in diameter, is located between the inner and outer upper nozzles to prevent foreign material to clogg the channels.

All plates and All structural parts are attached to the tubes by EB welding.

A boron ring is inserted at the lower end of the element, on the bottom nozzle immediatly after the fuel plates, to minimize the neutron flux peak at this place.

The depleted Uranium dummy element is equipped with strain gages and pressure sensing probes which will be used during the hydraulic tests.

4 FRM II FUEL PLATES DESCRIPTION

Due to very high power load in this compact core design, distribution of Uranium loading throughout the plate cannot be constant.

Near the exterior of the element, where thermal and neutron flux are maximum, Uranium loading per surface unit must be lower than in the inner region. To achieve this goal, two approaches are to be considered:

- vary continuously Uranium loading across the plate by varying meat thickness, but with a constant u/cm^3 content.
- vary discontinuously Uranium loading across the plate with a constant meat thickness, and a variable U/cm^3 content.

The first approach had been choosen by the ANS project, but TUM has choosen the simpler second possibility.

Thence FRM II fuel plates cores are composed of two different parts, a large core with a high Uranium loading and a smaller one with a lower Uranium loading.
The smaller core with low U/cm3 content is located near the outside of the element.

The main caracteristics of fuel plates are described in the following table.

	SMALL CORE	LARGE CORE
Cladding material	AlFeNi	
Frame material	AG2NE	
U alloy	U_3Si_2	
Total U content (g)	6.38	60.33
U loading (g/cm^3)	1.5	3

5 FABRICATION OF FUEL PLATES

Fabrication of such fuel plates with two cores of different densities is not so easy as fabrication of normal fuel plates and some know-how must be acquired to achieve a good quality, acceptable for reactor use.

Large core is very similar to those used in usual fabrication, but small core pressing must be done with special tooling due to the very small width of the piece. Care must be taken to avoid inhomogeneity in powder distribution.

Special technics have been developped to insure same groth of the two parts of the core during rolling process, because of the difference in density of components (3 and 1.5 gU/cm^3) which leads to very different mechanical behaviour of the two cores.

Also special preparation of sandwiches is necessary to avoid stray particles and difference in bonding of the components of the plate.

Frame and cover material are well known from ILL RHF fuel plates fabrication, and CERCA has taken advantage of this experience to master the rolling skill.

6 FUEL PLATES CONTROL RESULTS

Special control technics have also been developped for FRMII especially in the field of ultrasonic and Uranium distribution homogeneity measurements.

Sensing parameters and standards have to be adjusted to allow an ultrasonic control of both parts of the plate at the same time. Interface between the two cores is fully checked as well as the ends of the plate in the dog-bone area.

For the homogeneity of Uranium distribution, CERCA has taken advantage of the new fully computerised machine, but nevertheless every plate must be controlled in two sequences, one for the small core and one for the large one with two different standards because of the large difference in Uranium loading.

Figures in appendix present some samples of the overall quality of plates fabrication.

Radiographic examination shows no or very small difference in the length of the two cores. Also homogeneity is excellent and stray particles are absent of extremity of fuel plates.

Micrographic examination shows no overshooting of one core on the other and cladding thickness is very regular all over the plate.

Uranium distribution test is excellent for the large core. The small core shows some variations due to its small size, but these variation stay within the allowable limit of ±12% in the middle portion of the plate.

Ultrasonic results are not presented here because they are merely numerical results and not easily interpreted outside of their environnement, but all plates tested up to now are within the allowable limits of the specification.

7 FUEL ELEMENT FABRICATION

Due to its small size compared to RHF ILL element, FRM II fabrication has necessited special machine development and numerous tooling and machining conception to achieve a reasonably easy assembly and welding on an industrial fabrication scale.

All structural parts, including inner and outer grooved tubes of the fuel section as well as the sieve, have been developed and machined in CERCA plant at Bonneuil sur Marne near Paris.

Fuel element assembly begins with plates insertion in the grooves of the tubes, before they are affixed to these tubes by EB welding.

As the inside diameter of the fuel element is only 104 mm,CERCA, in collaboration with equipement manufacturer, has developed a very small EB gun (100mm diameter).

Welding of plates on outer tube and of all accessories (end fittings, boron ring...) is done with a classical EB gun.

Boron ring was first intended to be an Al/B alloy, but no manufacturer was willing to develop such fabrication of special alloy for a very small quantity.
So CERCA has developped a boron ring concept derivated from the boronated side plates technic, taking advantage of the experience from other fabrications (HFR Petten or ORPHEE for instance).

A plate with a Boron/Al core is laminated like a fuel plate, and its geometry is checked by X radiographs and neutron radiography.
This plate is machined, rolled and welded to form a ring and this ring is then inserted at the bottom of the element and welded to the structural part.

Very tight geometrical tolerances on the finished element, and geometry of the seat which is a sphere portion require that the complete element is machined on a precision lathe as final operation.
Thermal deformations during welding are rather important, and it is not possible to achieve final geometry directly from the individually machined structural parts.

8 INSTRUMENTATION OF FRM II FUEL ELEMENT

In the instrumented element used for hydraulic tests, strain gages are attached to one fuel plate, and corresponding leads are routed along the plate and through a small hole drilled in the bottom nozzle before completion of assembly and welding.

Special care must be taken to avoid damage to these leads during welding and machining of the element.

Pressure probes oulets are also foreseen on two separate channel, and at several places on top and bottom nozzle to monitor the pressure drop during hydraulic tests.

9 FUEL ELEMENT CONTROL

Plates to tubes welds are controlled by ultrasonic under water test with a special equipement internally developed and all other welds are controlled by radiographic examination.

After assembly and welding of end fittings, every water channel is measured with strain gages, and the results are plotted.

10 STATUS OF FABRICATION

A dummy element internally used at CERCA to develop manufacturing process has been assembled and welded and is now ready for final machining.

Fuel plates with depleted Uranium have been fabricated. As to now about 150 plates over a total of 400 ordered have been rolled and are under quality control.

The first official dummy element as ordered by TUM is about to be assembled with aluminium plates and will be used to finally set-up all parameters and to check the feasability of strain gages and pressure probes installation and will be delivered during the third trimester of 95.

The depleted Uranium element will be assembled later this year, and will serve, in the factory, the purpose to make final welding agreement by customer and survey authorities, as well as demonstration for the industrial fabrication.

Delivery of this element to TUM will take place as foreseen at the beginning of 1996

11 CONCLUSION

FRM II fuel element has necessited the synthesis of various fields of know-how at CERCA.

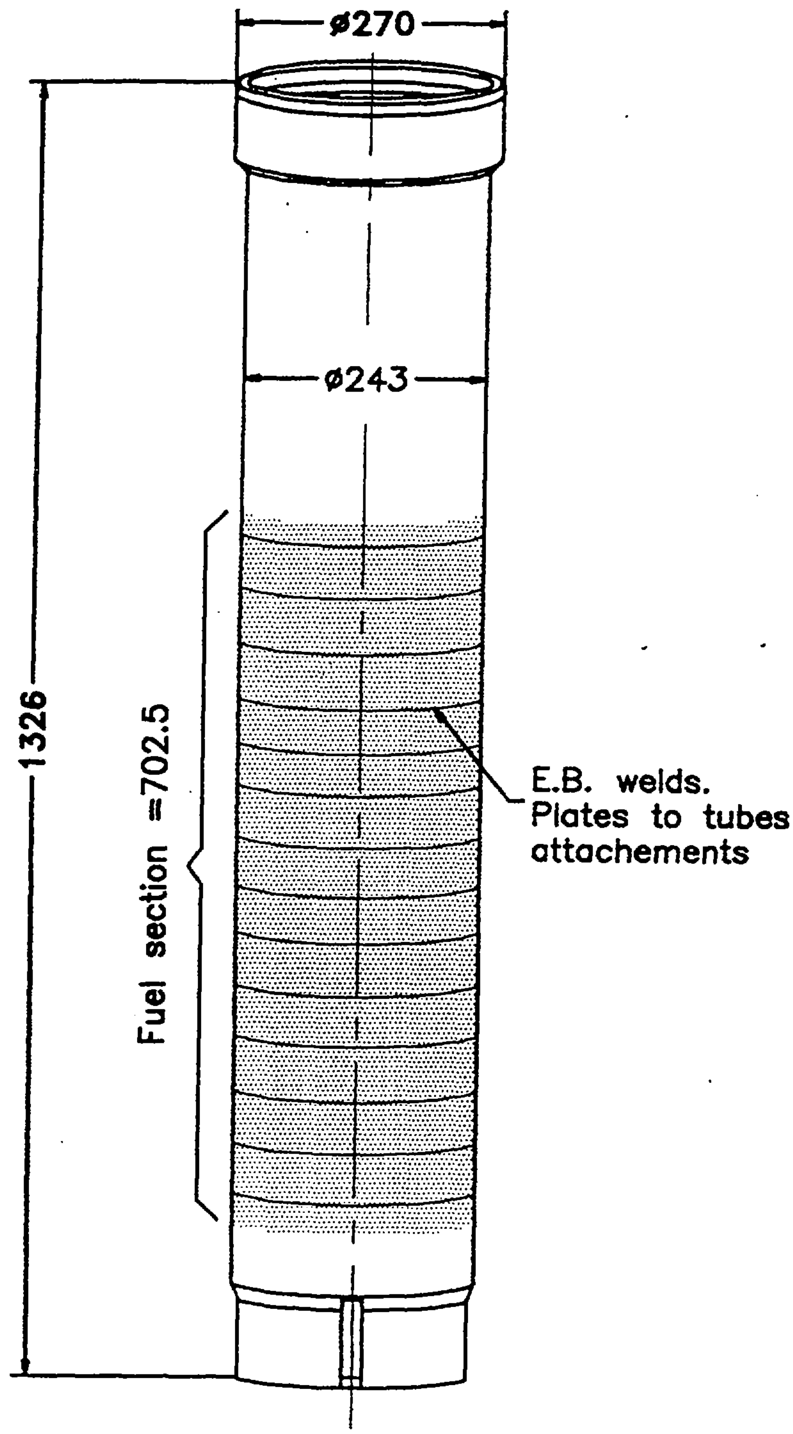
Among these we may summarize mainly:

- manufacture of multi density cores plates
- adaptation of inspection devices
- EB welding

Completion of fabrication is now underway, and the hydraulic tests should begin early next year.

CERCA

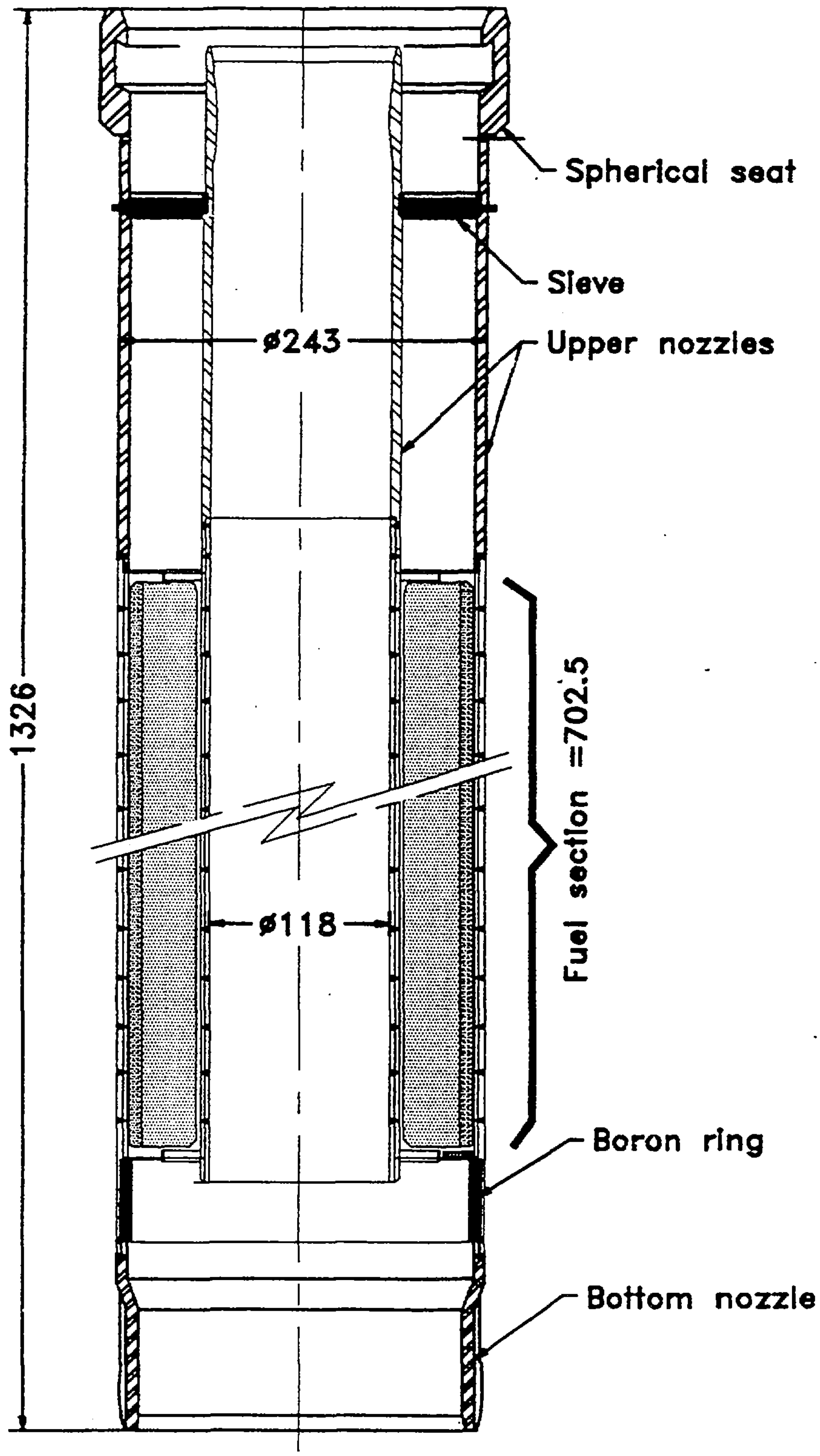
FRM II
External dimensions



FRAMATOME
FABRICATION COMBUSTIBLE

CERCA

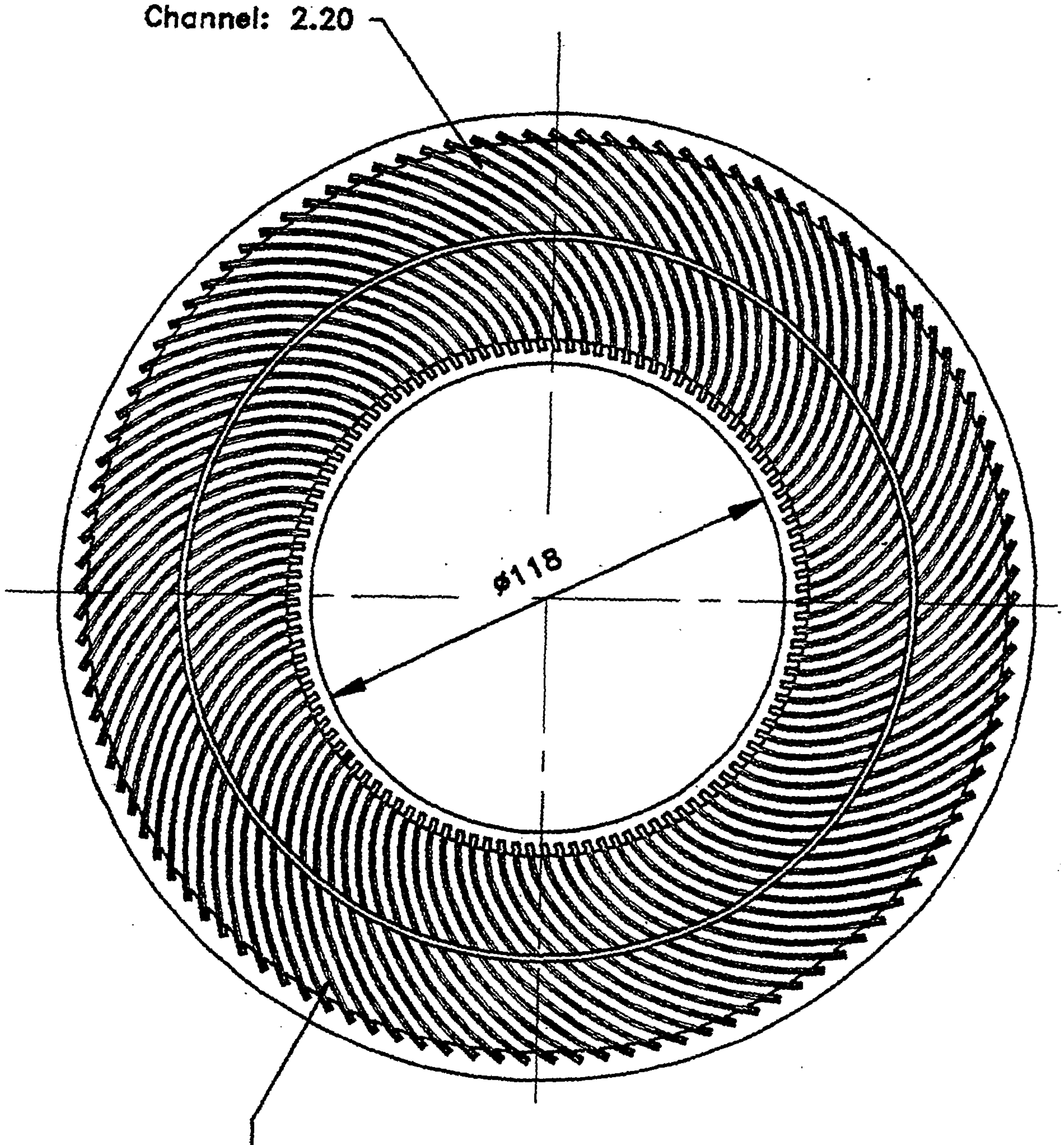
FRM II
Fuel element description



FRAMATOME
FABRICATION COMBUSTIBLE

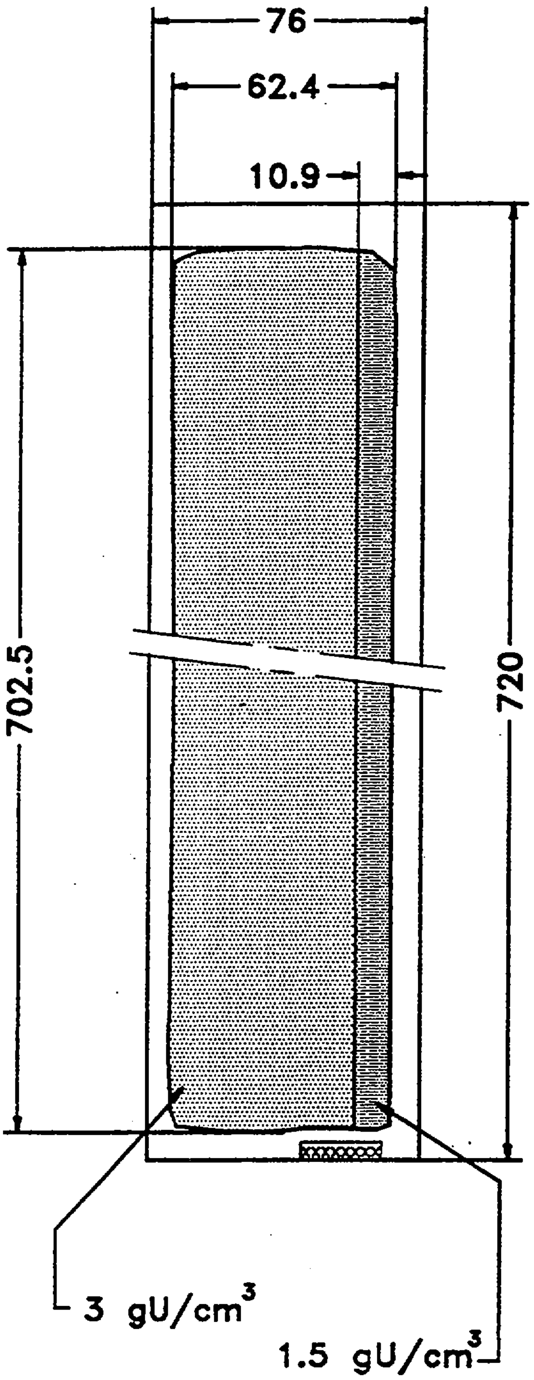
CERCA

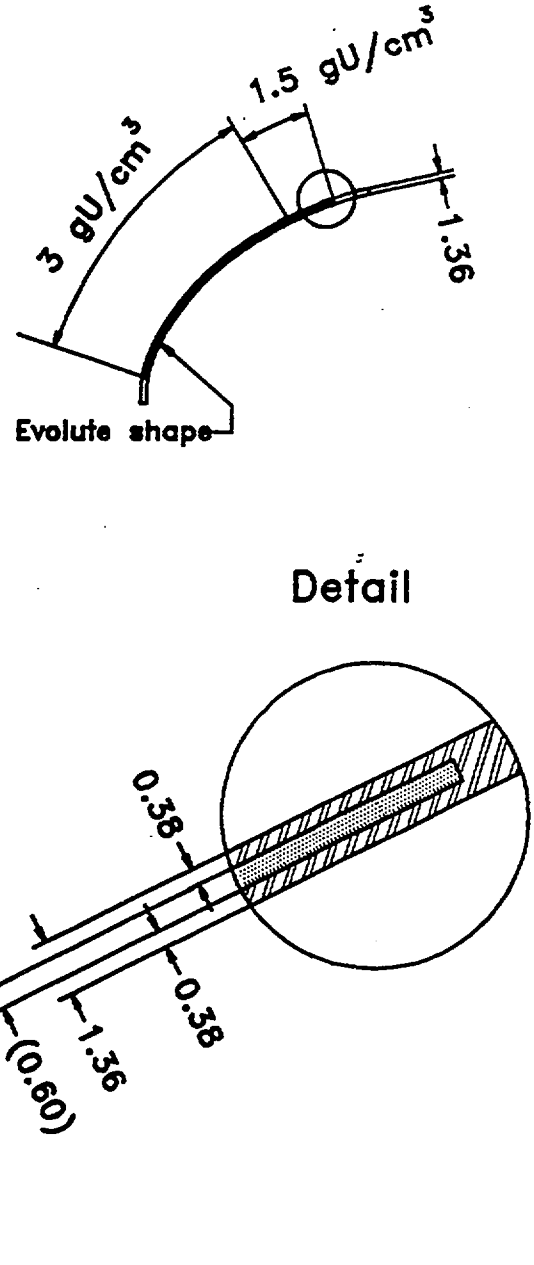
FRM II
FUEL SECTION



FRAMATOME
FABRICATION COMBUSTIBLE

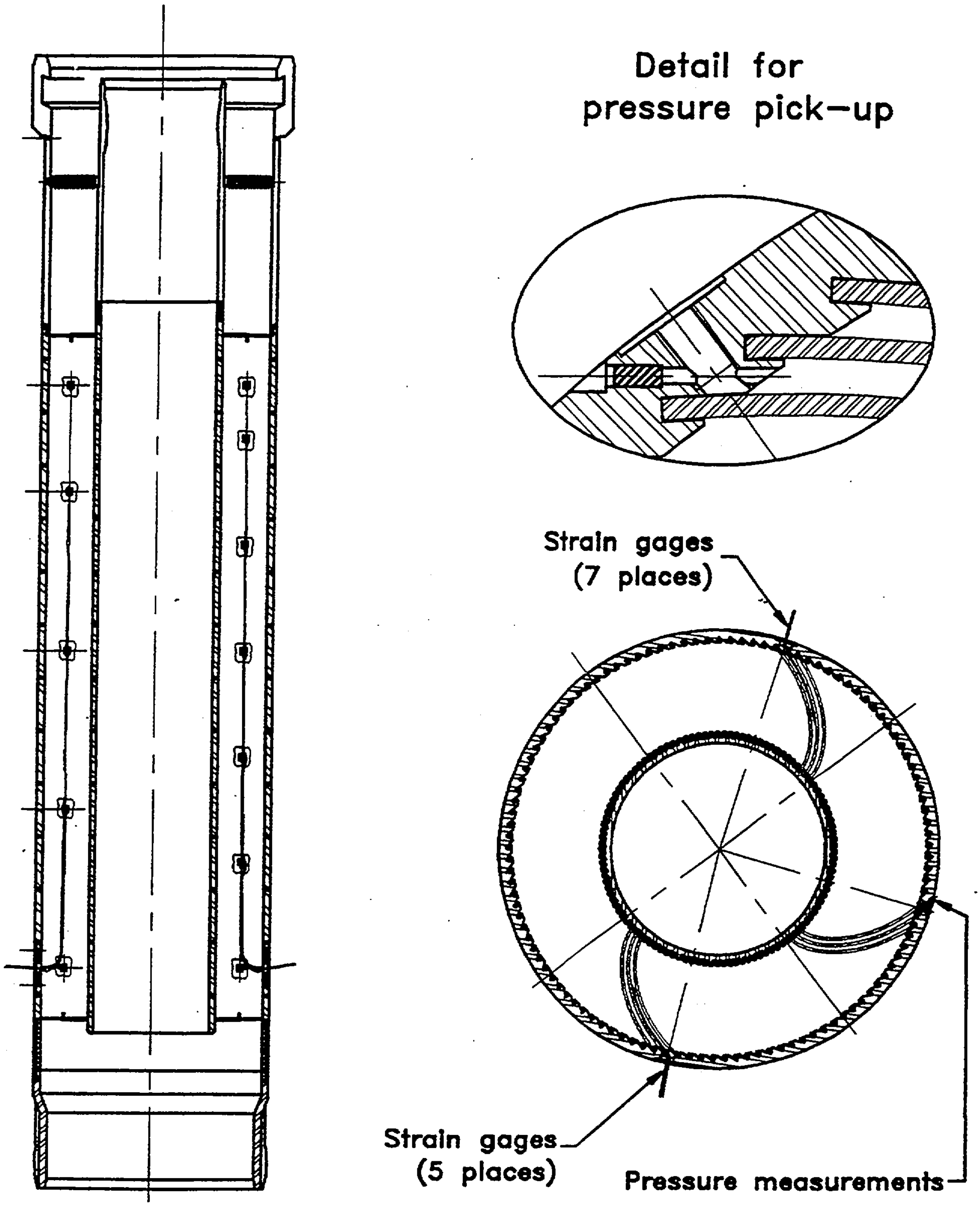
FRM II
Fuel plate





CERCA

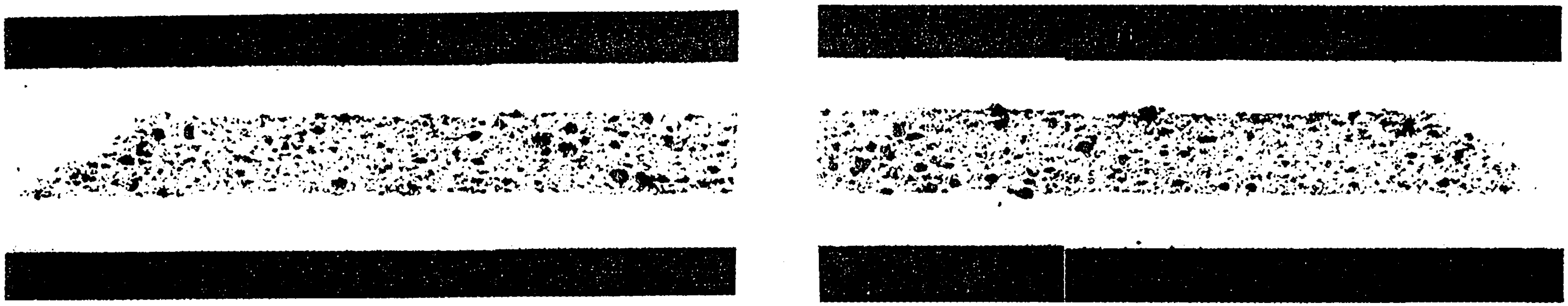
FRM II
Instrumentation



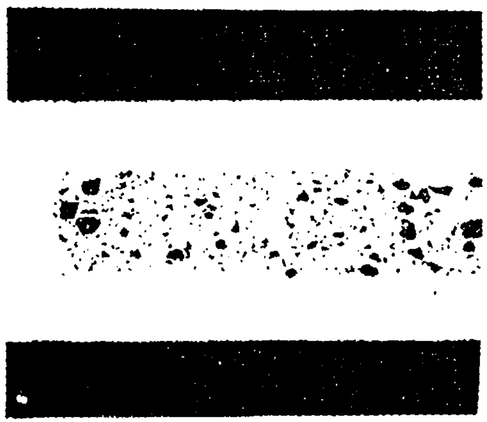
FRAMATOME
FABRICATION COMBUSTIBLE

CERCA

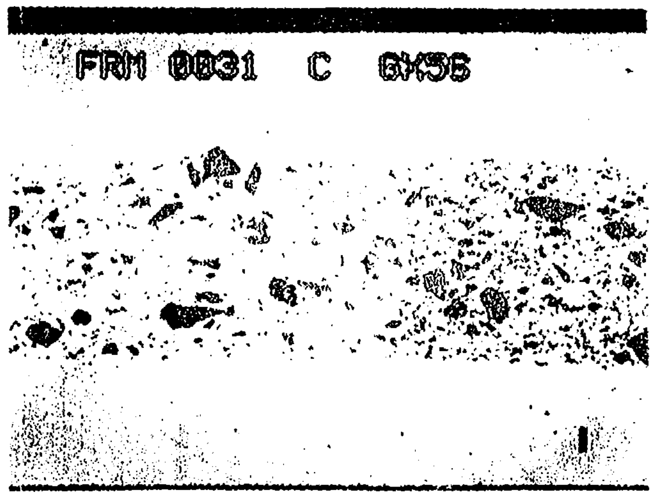
FRM2 U3Si2 FUEL PLATE



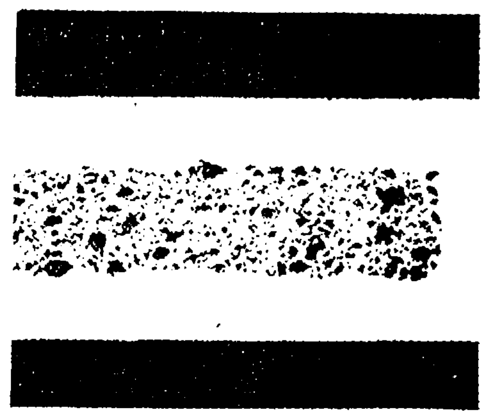
3.0 gUt/cm3 CORE (dog-bone area)



1.5 gUt/cm3 CORE SIDE



1.5 and 3.0 gUt/cm3 CORE INTERFACE



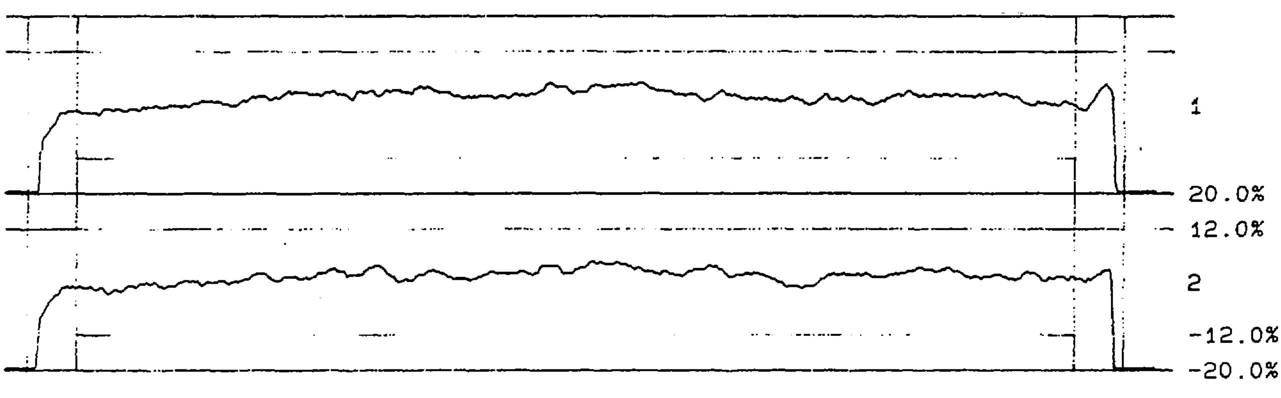
3.0 gUt/cm3 CORE

CERCA

URANIUM DISTRIBUTION HOMOGENEITY

SMALL CORE

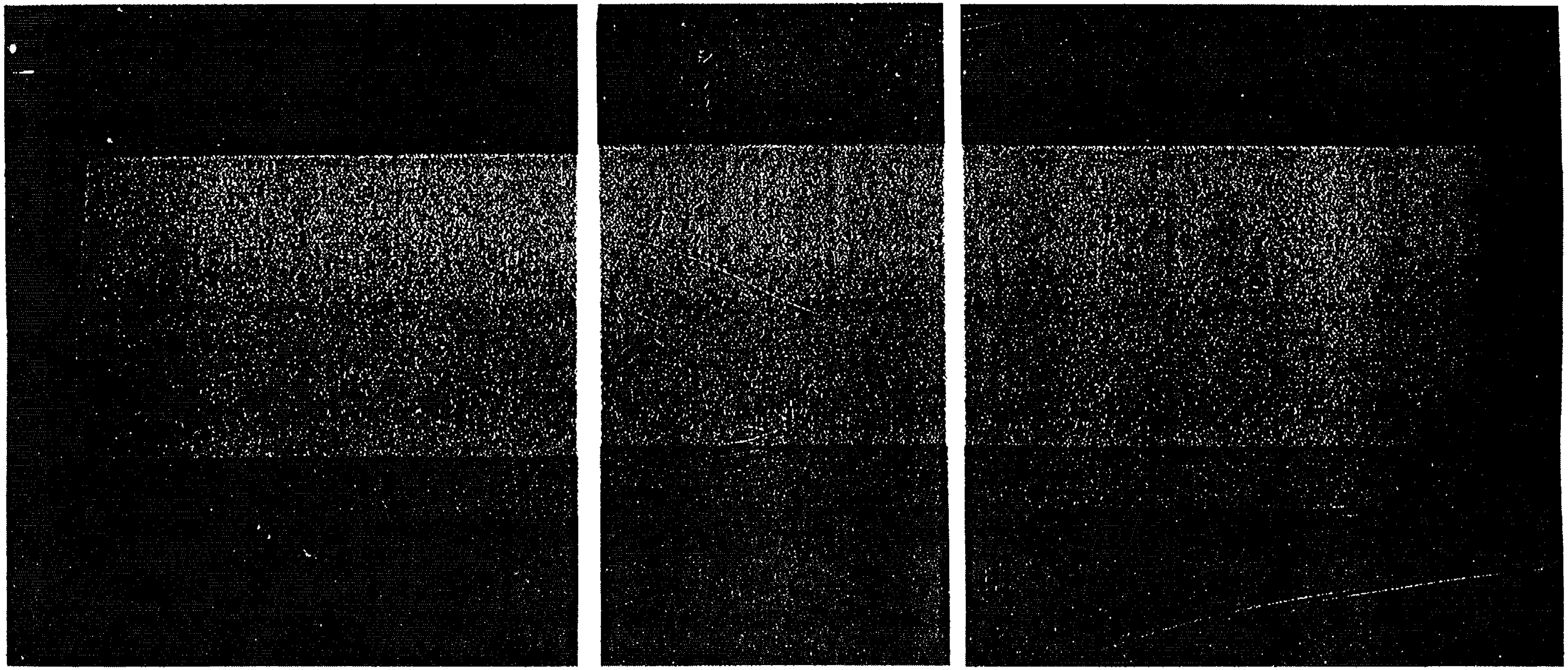
LARGE CORE



FRAMATOME
FABRICATION COMBUSTIBLE

FRAMATOME
FABRICATION COMBUSTIBLE

CERCA



FRM II PLATE X-RAY RADIOGRAPH

DUAL FUEL GRADIENT DEVELOPMENT UPDATE*

Presented at
The 4th Meeting of the International Group on Research Reactors
May 24-25, 1995

by
Brett W. Pace, Research and Development Engineer
Research and Test Reactor Fuel Elements
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Lynchburg, Virginia, USA

ABSTRACT

Development with fuel gradients in Uranium Silicide plates has continued through the past year at Babcock and Wilcox. In that time, dual gradient plates with loadings ranging from 1.3 gU/cc through 4.8 gU/cc have been manufactured. The results from these development fuel plates have been analyzed showing that the dual fuel gradient is possible in fuel loadings up through 4.8 gU/cc. The development will continue with work in maintaining the dual gradient while centering the fuel core within the cladding.

INTRODUCTION

Babcock and Wilcox (B&W) has done extensive work in cooperation with Oak Ridge National Laboratory (ORNL) to achieve a controllable bidirectional fuel gradient in uranium silicide (U_3Si_2) plates. The effort has been focused for the purpose of the Advanced Neutron Source (ANS) which was scheduled to be built at ORNL.

The goals of the development were to create a dual gradient in U_3Si_2 fuel plates to vary the effective fuel loading both down the length of the plate and across the width. To achieve the dual gradients, alternative manufacturing and fuel processing methods have been evaluated.

To date, 40 development plates have been manufactured with dual fuel gradients with varying degrees of success. The fuel loadings of these plates vary from 1.3 gU/cc to 4.8 gU/cc and depleted uranium was used throughout. A fair degree of success was achieved in each lot of plates with the exception of only one of the early lots which did not display the desired fuel gradients. B&W is currently preparing to complete another phase of the fuel gradient development, the results of which should be available this summer.

*Research sponsored by the US Dept. of Energy under contract #DE-AC05-840R21400 through Martin Marietta Energy Systems, Inc.

DUAL GRADIENT FUEL CORES

The fuel gradient which has been chosen for development can be best described by a comparison of the compacts used in the development to the typical HFIR compact profile shown in Figure 1. The HFIR gradient is asymmetrical across the width with no gradient down the length of the plate. The original dual gradient plates used a similar gradient across the width as shown in the width view on Figure 2. The length gradient was designed to be symmetrical from one end to the other.

As in the case of the HFIR compact, the development gradient across the width was swept using a tapered die surface. A flat layer of aluminum powder was swept over the fuel and then compressed. The bottom of the compact, where the lengthwise gradient is located, was addressed in the early stages of development when different methods of ensuring the gradient were tested. Using an aluminum powder filler was one of the original ideas of how to fill the void area left by the length gradient. However, 3 separate powder sweeps were involved and the die tended to gall due to aluminum powder buildup. Other methods tried were machining a contour in the associated cover plate and the use of wedge shaped inserts during packing. Both of these methods proved satisfactory and the wedge method was used for the remainder of the development due to its simplicity.

After the methods for achieving the dual gradients were determined, the loadings were varied to evaluate the response of the gradient under different conditions. In each case the gradients were predictable and in line with the original objectives.

Gradient Specific Effects

The manufacturing of the different gradients produced conditions not normally found in standard fuel plates. Two conditions displaying the most interesting results are fuel smear along the edges and thin edge homogeneity consistency.

Fuel smear occurs after the fuel powder is swept into the die cavity and then is adjusted to accommodate the aluminum powder filler. Friction between the die block and powder pulls small quantities of the powder up against the aluminum powder cap. During hot rolling, the smear is stretched along the length edge such that it does not produce any appreciable change in local homogeneity in the effected areas.

One of the primary concerns of this development has been to understand how the homogeneity in the thin edges and corners will be effected by overall homogeneity effects and by particle size. To further evaluate this area, a comparison of the DE data from the plate center (maximum core region) to the plate ends was made. The study was done using data from the 3.0 gU/cc loaded plates by enlarging photos of the DE section involved and measuring the length of the fuel particles at the intersections of a traverse through the plate from the top to the bottom as shown in Figure 3. The data was converted into a unitized gU/cc loading for each section and up to 10 traverses were used across each section spaced at approximately 0.16mm (scaled) apart. Therefore, the loading was an average across each section up to 1.6mm in width. This study does not give any real information about the true homogeneity of the plate but does allow a microscopic look at the end characteristics

HFIR OUTER COMPACT DESIGN

Single Gradient Contour

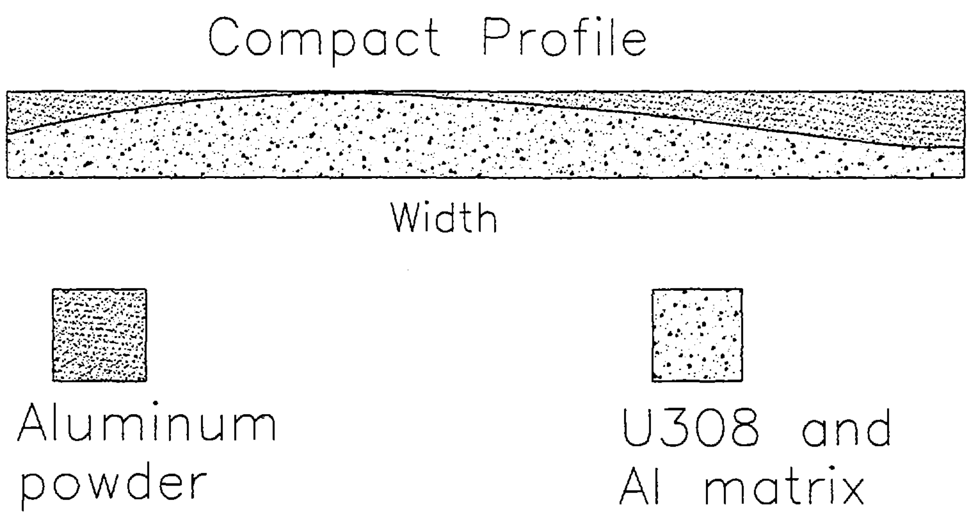


Figure 1

RECENT ANS DEVELOPMENT
COMPACT DESIGN

Dual Gradient Contour

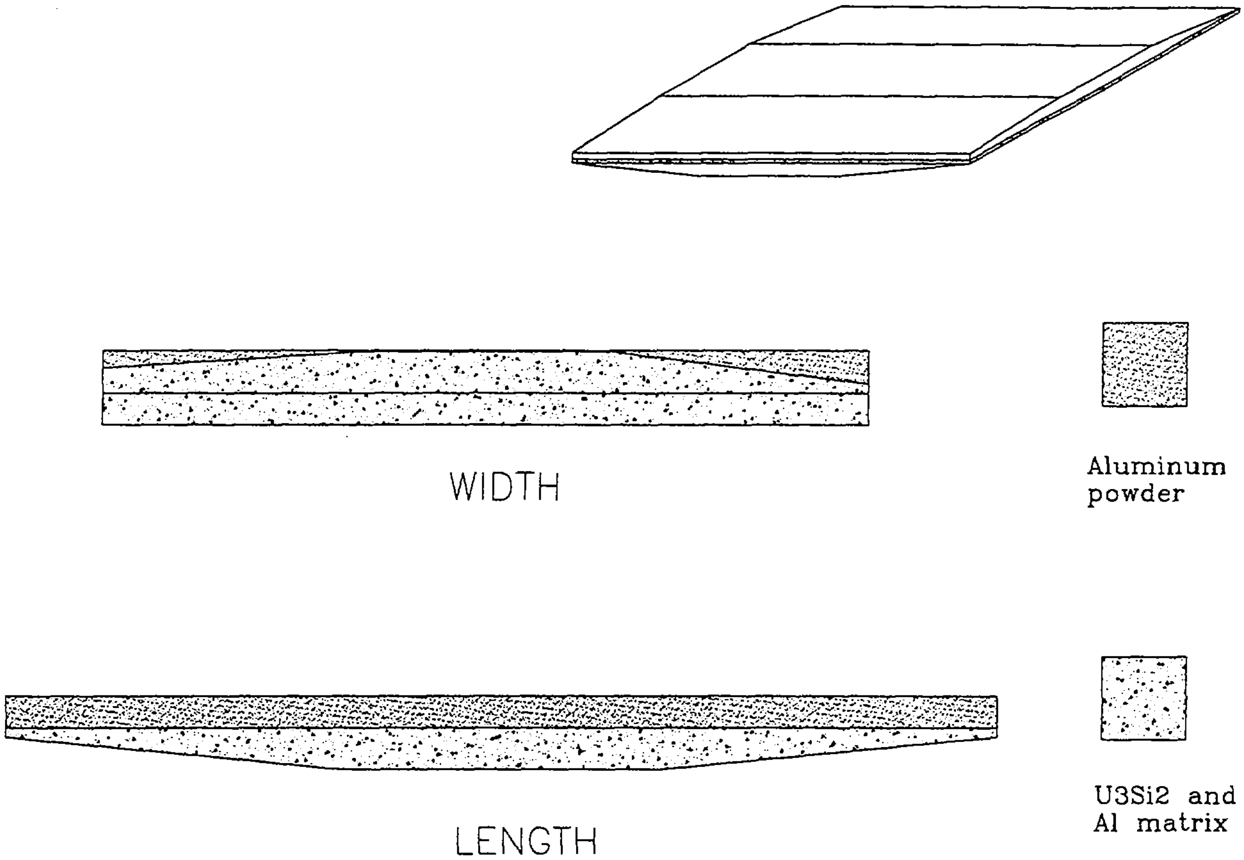
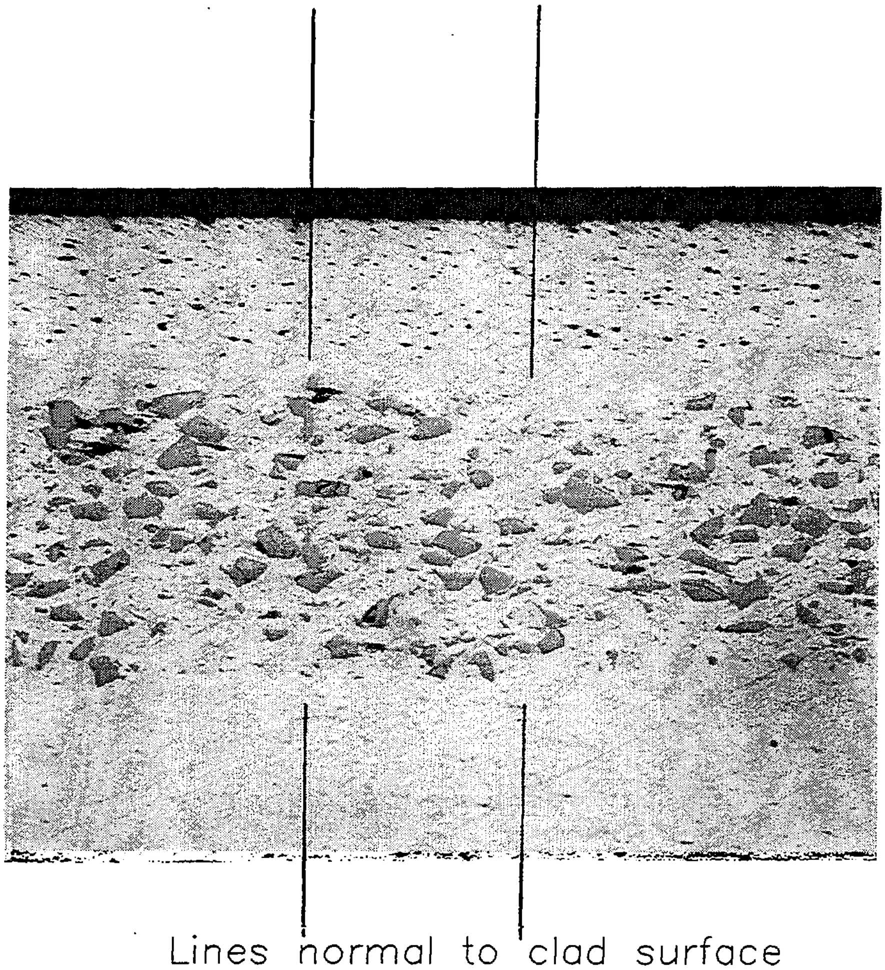


Figure 2

FUEL PLATE SECTION



Linear fuel densities measure along transverse (top to bottom) sections located approximately 0.16mm apart.

Figure 3

Gradient Specific Effects Continued

compared to the center. The average loading in each spot was 3.16 gU/cc in the center and 3.14 gU/cc on the end. The standard deviation in the data was 0.6 gU/cc for the center and 1.2 gU/cc for the edge. By observing the fuel particle section sizes one can estimate that the higher variability of the fuel on the edge of the plate is due to the fact that there are fewer particles and an increased incidence of drawing a line normal to the fuel core which may not intersect any sizeable fuel particles. No homogeneity conditions involving quality issues were found and this study will continue to determine the overall magnitude of the effect on the thin fuel edge.

Homogeneity

One of the most useful tools in evaluating the overall homogeneity of the fuel plates is a digital homogeneity scanner. The most recent higher loaded development plates were all evaluated using this equipment showing that the dual gradient concept is feasible over a range of fuel loadings through 4.8 gU/cc. Data for one of the highest loaded plates, 4.8 gU/cc, is shown graphically in Figure 4. The gradients of all plates were very similar among plates with like loadings showing that the process was repeatable.

Research and testing of the digital homogeneity scanner has been ongoing since the system came online in 1992. As part of the latest fuel development, the sensitivity of the scanner has been tested using different diameters of tungsten wire. The results of these test are currently being evaluated.

NEW FUEL DEVELOPMENT

Dual Gradients

The next step in the development of dual gradients is to center the fuel within the cladding at all locations in the plate. The gradient chosen for further development was a redesign of the previous compact to be symmetrical in all aspects as shown on Figure 5. The compact will not have any aluminum powder filler and no wedges will be used to fill the void areas. Instead, the compact will fit inside two dished cover plates with a very thin frame between them to maintain proper fit up. With this configuration, any fuel smear or core end buildup will be removed and homogeneity should be similar to prior development. The plate loadings will be 2.8, 3.0, and 3.5 gU/cc and the quantities will be 4, 4, and 6 respectively.

Spherical U_3Si_2

Through ORNL and in association with Argonne National Laboratory and the Korean Atomic Energy Research Institute (KAERI), B&W will endeavor to manufacture

HOMOGENEITY OF HIGH LOADED

DUAL FUEL GRADIENT U3Si2 PLATE

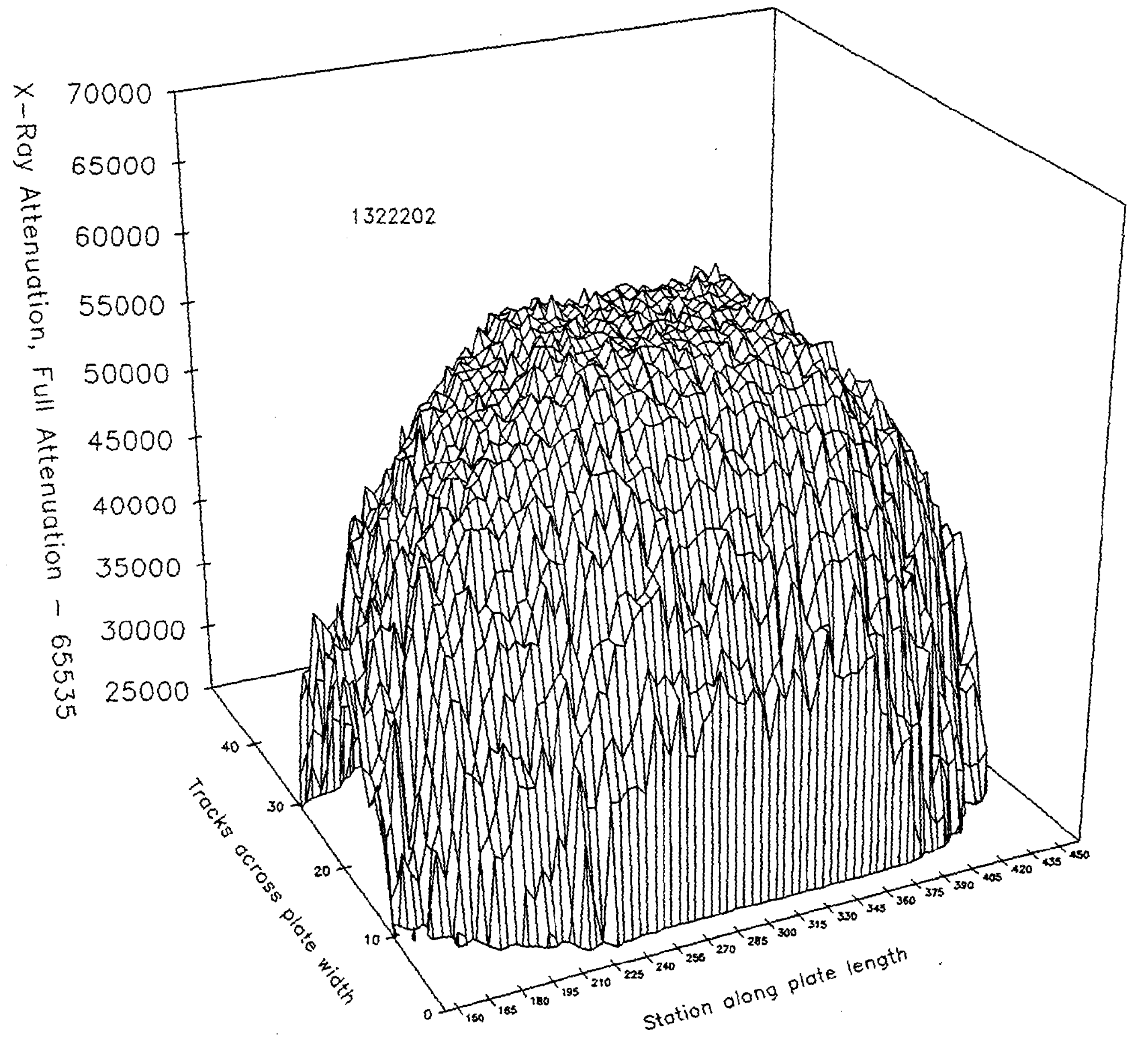


Figure 4

NEW ANS DEVELOPMENT
COMPACT DESIGN

Centered Dual Gradient Contour

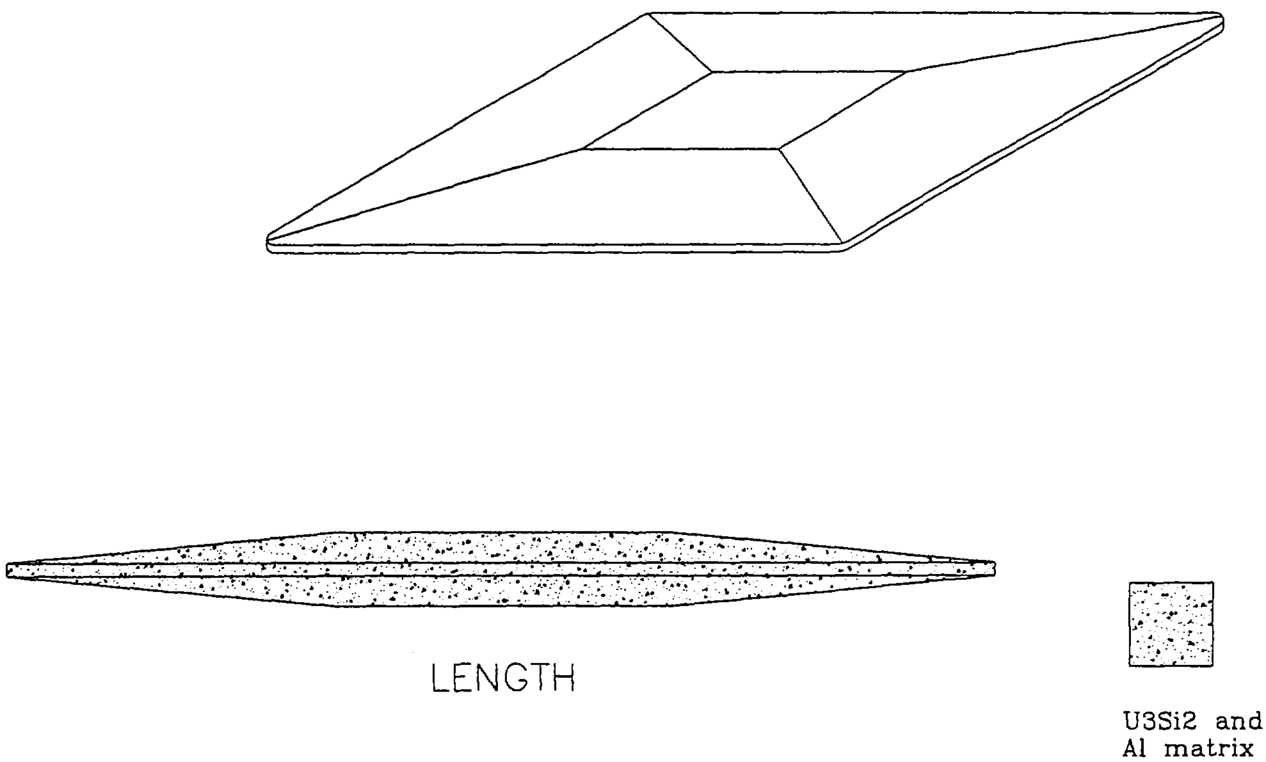


FIGURE 5

fuel plates using spherical fuel manufactured by KAERI. Two sets of three plates each will be manufactured with the original fuel gradient used in development. One set of plates will be loaded at 3.0 gU/cc and the other will be 4.8 gU/cc.

CONCLUSION

The development with dual gradients in U_3Si_2 fuel plates over the last year has increased the understanding of fuel characteristics which will benefit all uranium silicide work at B&W. The ability to produce dual gradients in uranium silicide plates has been proven successful and repeatable without significantly increasing the effort required to produce plates. The fuel core centering development will be an excellent way to finish the development with dual gradients. A final report on all gradient development at B&W will be written in conjunction with ORNL.

THE EFFECTS OF IRRADIATION TO 8 x 10^{26} m^{-2} ON THE MECHANICAL PROPERTIES OF 6061-T651 ALUMINUM

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ornl

IGORR-IV
4th Meeting of the International Group on Research Reactors
May 25, 1995
Gatlinburg, Tennessee

IRRADIATION EFFECTS ON STRUCTURAL MATERIALS

The effects of irradiation on the mechanical properties of candidate structural materials are being examined.

A key to the generation of useful neutron beams is allowing the neutrons produced in the core to escape. Therefore, an aluminum alloy has been selected for the first wall containment adjacent to the core, the Core Pressure Boundary Tube (CPBT).

- 6061-T651 (Al-1.0Mg-0.6Si-0.3Cu-0.2Cr)
- acceptable mechanical properties in unirradiated condition
- low neutron cross-section
- high thermal conductivity for heat removal

This alloy may also be used for the beam tubes and reflector tank.

An irradiation program is underway to determine the effects of irradiation on the mechanical properties of 6061-T651, in particular the fracture toughness. This data will allow the operating lifetime of the CPBT to determined, which will in turn determine its replacement schedule in the ANS.

- irradiations have been conducted to 10^{26} and 8×10^{26} m^{-2} (thermal, < 0.625 eV); these correspond to approximately 1 cycle and 6 months (9 cycles) of ANS operation, respectively
- irradiations are being conducted in the HFIR

- first capsule, designated HANSAL-T1, has been irradiated to 10^{26} m^{-2} (three HFIR cycles) and disassembled
- all specimens have been tested

A second irradiation capsule, HANSAL-T2, has been irradiated in HFIR to 8×10^{26} m^{-2} (21 cycles). This fluence is equivalent to approximately 6 months of operation of the ANS for the CPBT.

The capsule has been disassembled and the specimens have been prepared for testing.

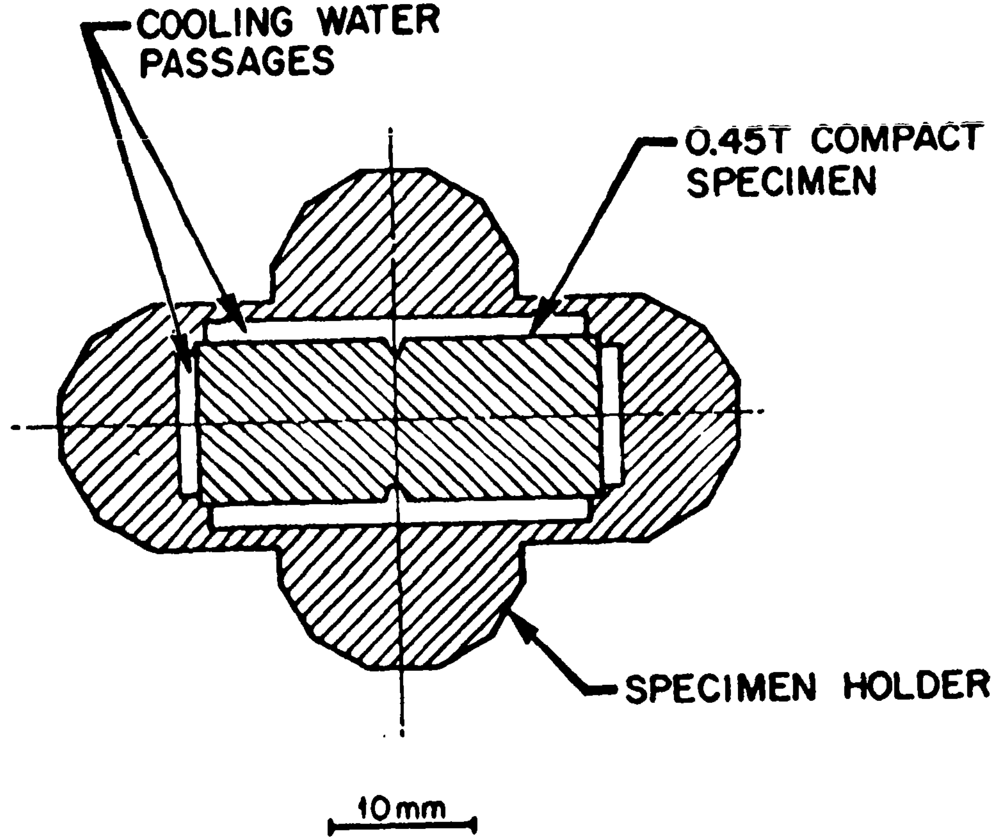
- removal of loading hole and center notch filler pieces from 8 specimens was successful
- the temperature monitor capsules have not been removed from the 8 remaining specimens

The tensile specimens and the first 8 compact specimens have been tested.

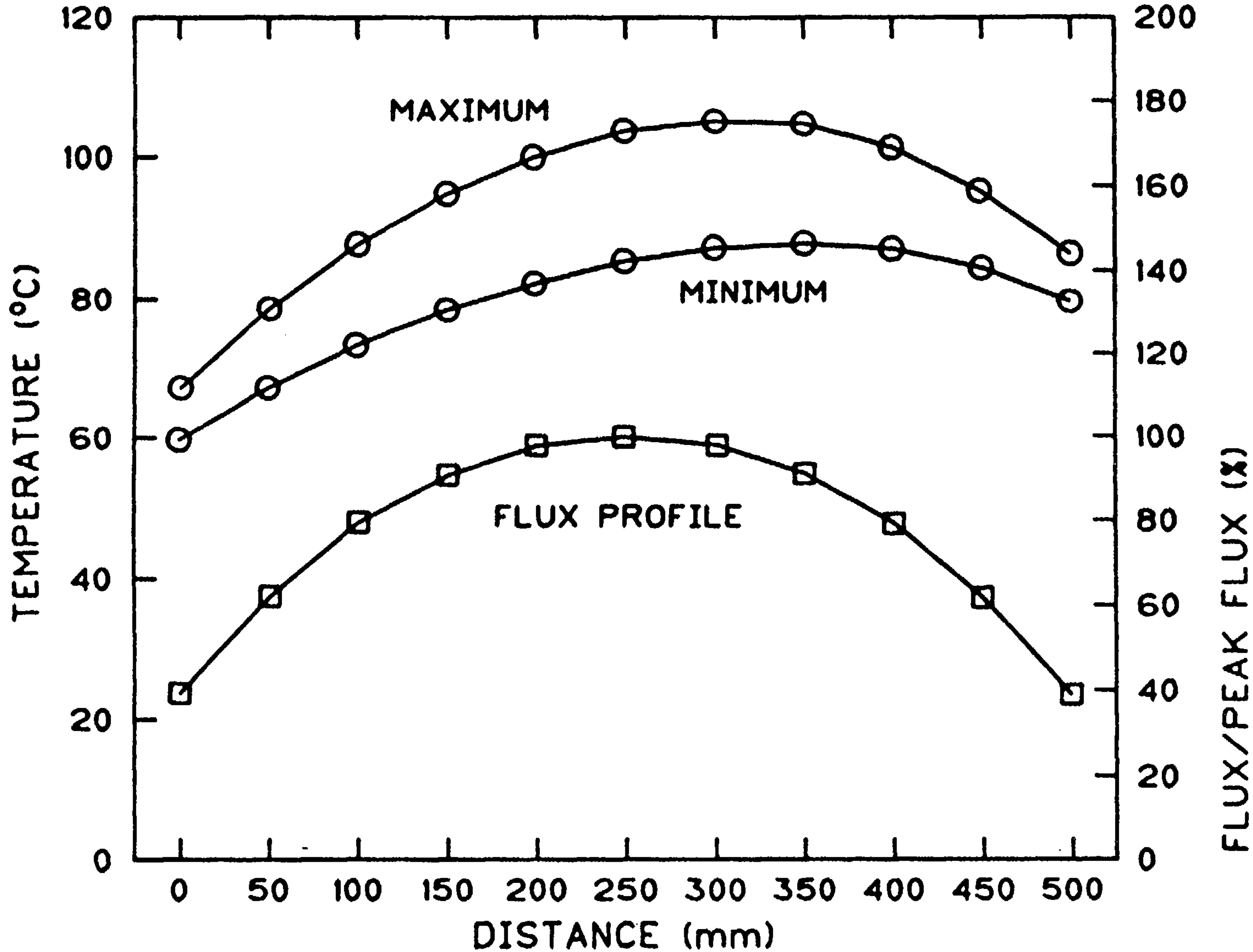
HANSAL-T1 DESIGN

- inserted in HFIR target region
- cluster of four target positions were combined to create a volume large enough for conventional fracture toughness specimens
- 16 compact specimens and 15 flat tensile specimens were arranged in column, with 30 TEM discs and 30 atom probe needles
- irradiation target temperature was 95°C (CPBT operating temperature); specimens were cooled by flowing reactor cooling water, so all openings in specimens were filled to reduce flow perturbations in flow channels
- uninstrumented capsule, but melt wire canisters were inserted in loading holes of some compact specimens, and flux monitors for post-irradiation analysis of flux profile

A new capsule that occupied four target positions was designed.



The temperature and flux varied along the length of the HANSAL-T1 capsule.



TEST TECHNIQUES

Tensile Testing

- flat pin-loaded specimens tested with servohydraulic machine in stroke control
- record load and crosshead displacement for analysis
- tests at 25, 95, and 150°C (clip-on thermocouple)

Fracture Toughness Testing

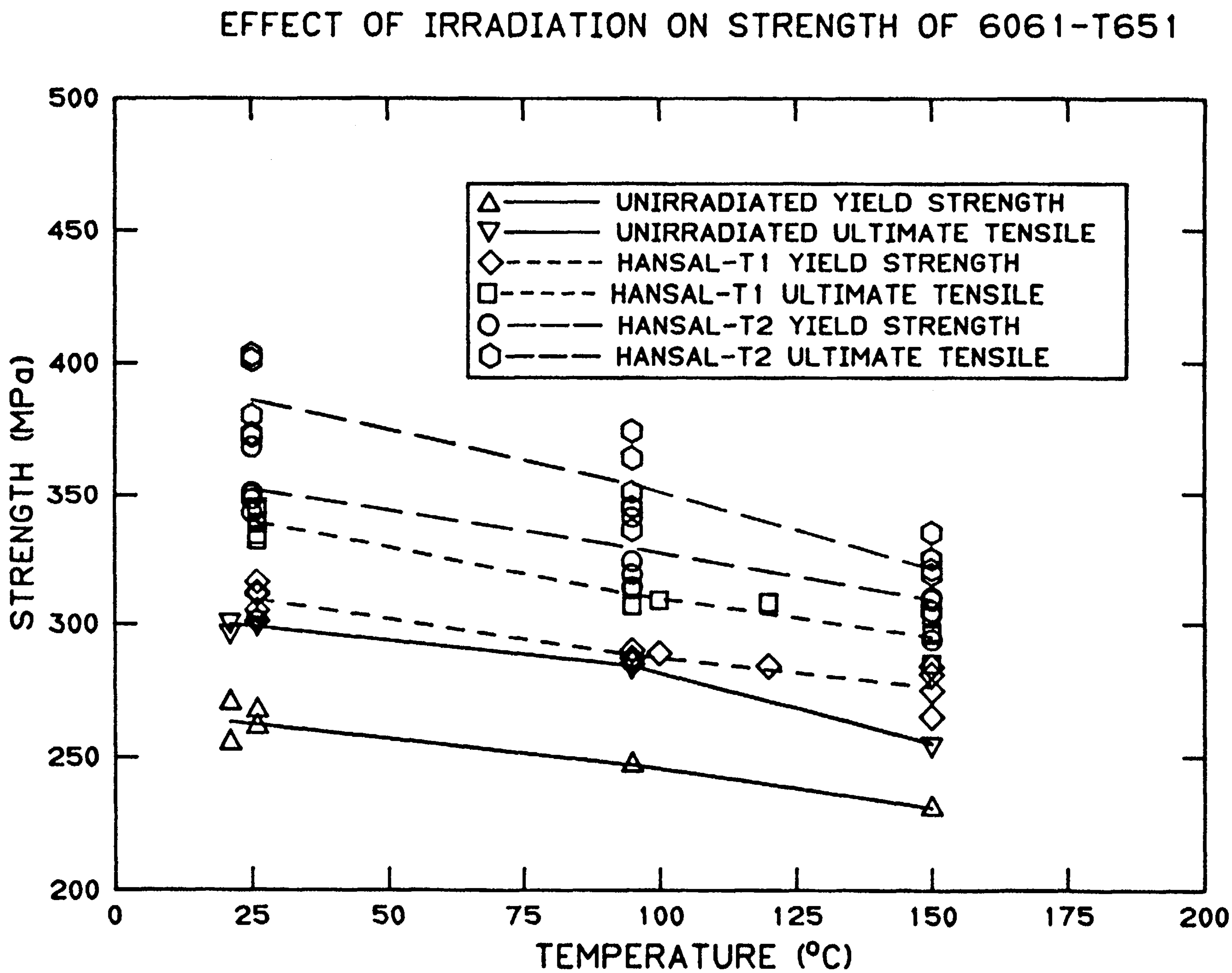
- 0.45 T compact specimens precracked at room temperature and then sidegrooved 20% prior to irradiation
- unloading compliance measured from outboard clip gage on loadline used to monitor crack extension
- tests at 25, 95, and 150°C (clip-on thermocouple)
- final crack extension marked by cyclic loading
- photographs of surface used to measure crack lengths

TENSILE TESTING

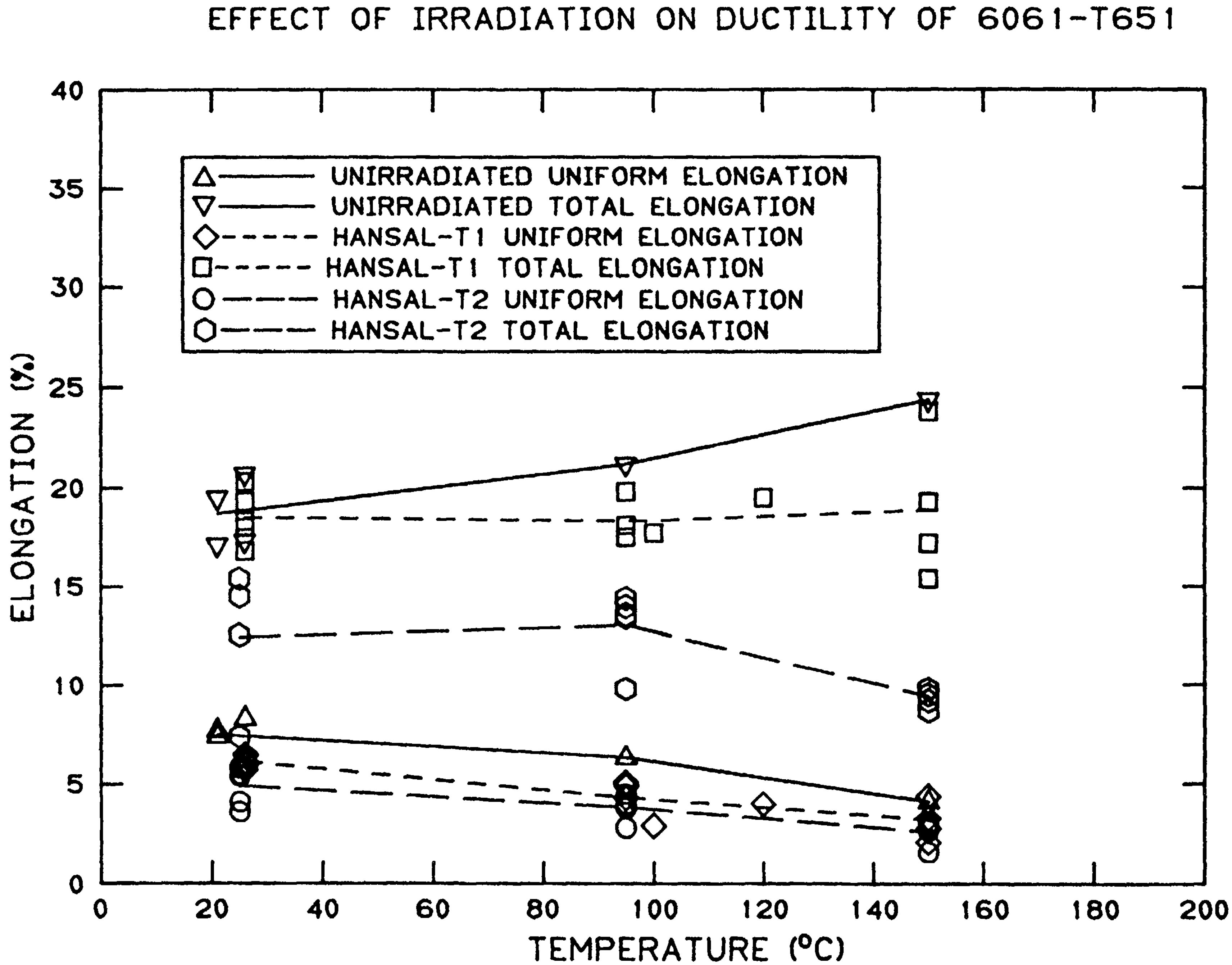
- yield and ultimate strengths were increased significantly by irradiation, as expected

- uniform and total elongation are reduced, as expected, although total elongations are still fairly high after irradiation

- results are similar to literature data



EFFECT OF IRRADIATION ON STRENGTH OF 6061-T651
UNIRRADIATED YIELD STRENGTH
UNIRRADIATED ULTIMATE TENSILE
HANSAL-T1 YIELD STRENGTH
HANSAL-T1 ULTIMATE TENSILE
HANSAL-T2 YIELD STRENGTH
HANSAL-T2 ULTIMATE TENSILE
STRENGTH (MPa)
500
450
400
350
300
250
200
0
25
50
75
100
125
150
175
200
TEMPERATURE (°C)



EFFECT OF IRRADIATION ON DUCTILITY OF 6061-T651
UNIRRADIATED UNIFORM ELONGATION
UNIRRADIATED TOTAL ELONGATION
HANSAL-T1 UNIFORM ELONGATION
HANSAL-T1 TOTAL ELONGATION
HANSAL-T2 UNIFORM ELONGATION
HANSAL-T2 TOTAL ELONGATION
ELONGATION (%)
TEMPERATURE (°C)
0
5
10
15
20
25
30
35
40
0
20
40
60
80
100
120
140
160
180
200

FRACTURE TOUGHNESS TESTING

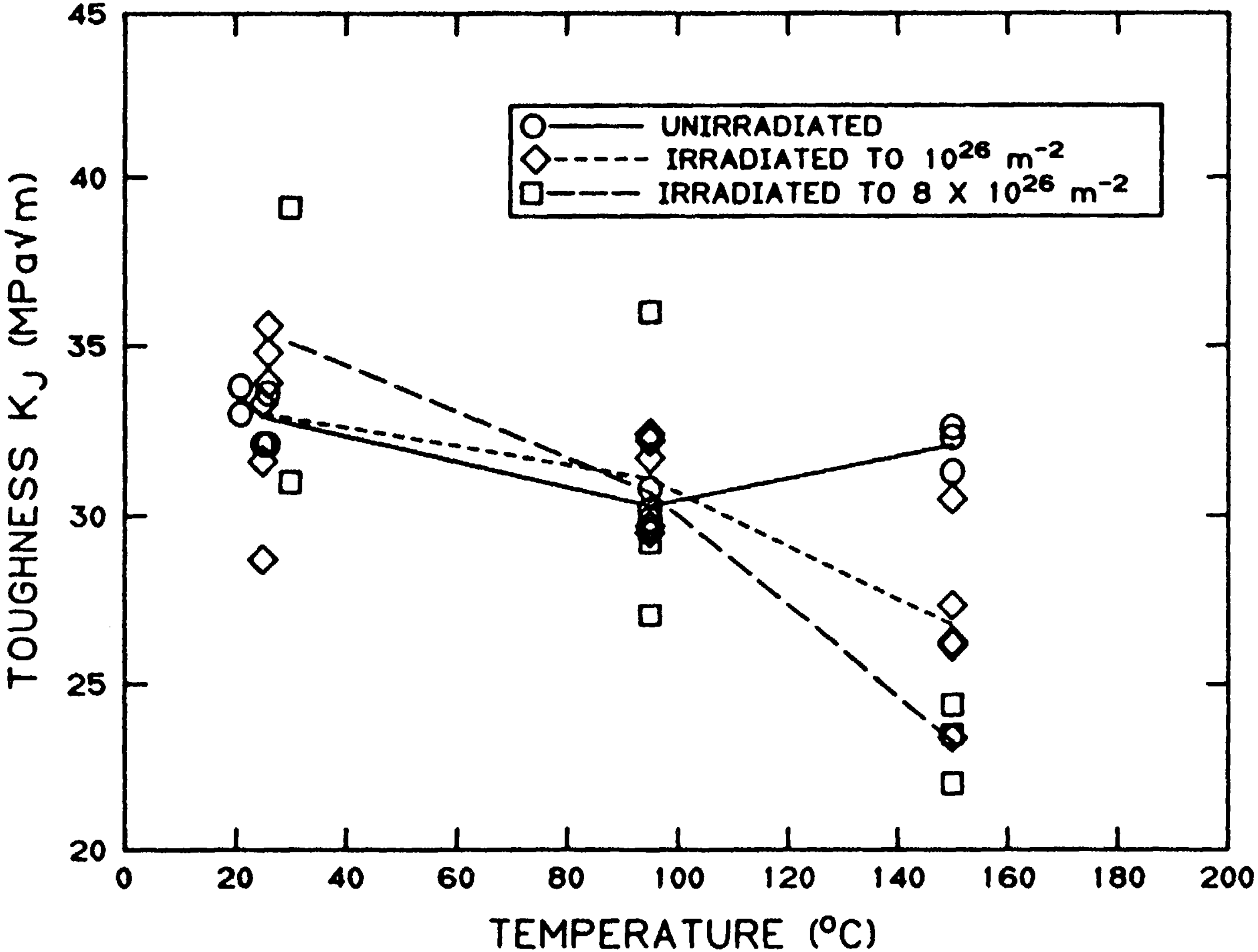
- the fracture toughness at 25 and 95°C is not significantly affected by irradiation, but does decrease at 150°C

- alloy shows tendency for sudden, rapid crack extension

- tearing modulus of unirradiated material is low, and is reduced to extremely low value after irradiation

- poor agreement between measured and predicted final crack lengths

EFFECT OF IRRADIATION ON FRACTURE TOUGHNESS



IRRADIATION EMBRITTLEMENT OF ALUMINUM

- irradiation with thermal neutrons results in the transmutation of Al to Si
- fast neutrons generate point defects in the lattice which enhances the diffusion of Si
  - this allows Si precipitates to form throughout the material
- precipitate formation accounts for the increase in the strength and the decrease in ductility following irradiation

The HANSAL-T3 capsule has been irradiated for 6 cycles and removed from HFIR for cooldown.

- contains fracture toughness and tensile specimens from weldments

Arrangements are being made for the shipping and disassembly of this capsule.

CONCLUSIONS

The first two capsules for the ANS Irradiation Effects program have been successfully irradiated to 10^{26} and 8×10^{26} m^{-2} (thermal), respectively, at a nominal irradiation temperature of 95°C. The testing of the specimens of 6061-T651 aluminum has shown:

1. The yield and ultimate tensile strengths are increased by irradiation.

2. The uniform and total elongations are reduced, but useful ductility remains, even at the higher irradiation level.

3. The fracture toughness at 25 and 95°C is unaffected by irradiation, but at 150°C, it decreases with an increase in irradiation.

4. The tearing modulus of 6061-T651 is low in the unirradiated condition, and is reduced to very low values by irradiation. This alloy also shows a tendency for sudden unstable crack extension.

XA04C1698

318

Idaho
National
Engineering
Laboratory

MOCUP: MCNP-ORIGEN2 Coupling Utility Programs

Contributors

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Presented by: C. A. Wemple

IGORR-IV Meeting
Gatlinburg, TN, USA
May 24-25, 1995

Overview

- Goals and purpose
- Program flow
- Program capabilities
- Applications to test reactors
- Future of MOCUP

C062-WHT-595-02

What is MOCUP?

- System of interface codes wrapped around MCNP Monte Carlo transport code and ORIGEN2.1 depletion and isotopics code

- GUI for user interface w/codes

- Performs depletion in complex, non-lattice geometries for research and test reactors

C062-WHT-595-04

Goals and Purpose

- Provide capability for depletion using Monte Carlo fluxes
  - Complex geometries (i.e., test reactors)
  - Target depletion
  - Isotope production

- Minimal extra user input

- Maximum flexibility

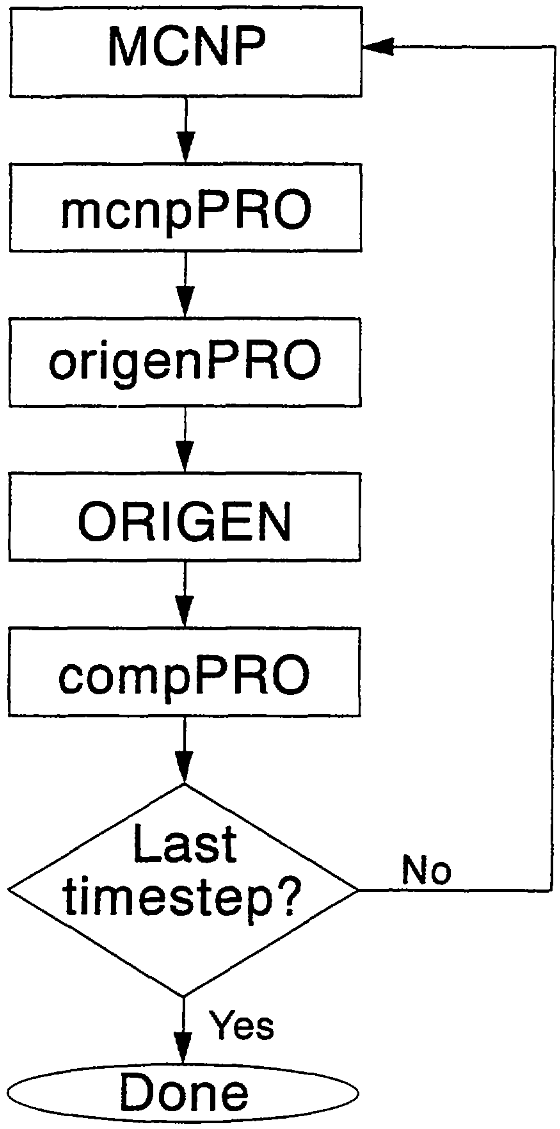
- Ease of use (GUI)

- Easy expansion of code

- Completely external to MCNP and ORIGEN

C062-WHT-595-05

MOCUP Program Flow



Z061-CAW-1193-004b

MOCUP Capabilities

- Fuel depletion
  - Arbitrary geometry
  - Arbitrary number of regions
  - Arbitrary number of nuclides tracked
  - Constant flux or power across timestep

- Target depletion
  - Independent nuclide tracking
  - Arbitrary number of target locations and compositions
  - Constant flux across timestep

- Structural material transmutation

- Control depletion

C062-WHT-595-06

User Supplied Data

- MCNP input with special comments
- ORIGEN2 skeletal input files and execution script
- Normalization factors for flux tallies
- Table of nuclide identifier equivalence (master)
- Computer environment variable settings (for GUI)

Applications of MOCUP (so far . . .)

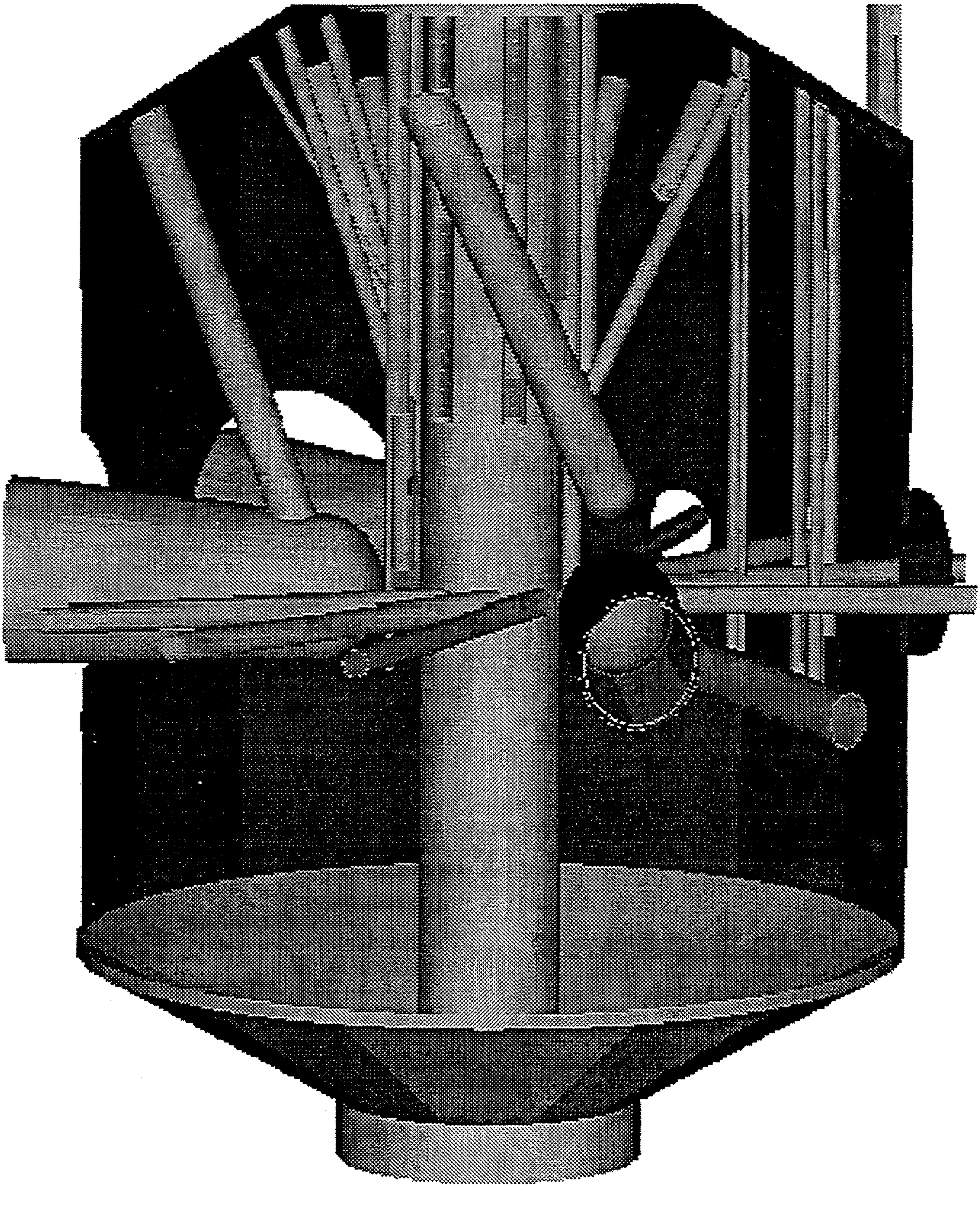
- INEL
  - Advanced Neutron Source Reactor
  - Advanced Test Reactor
  - Pu disposition reactor analysis

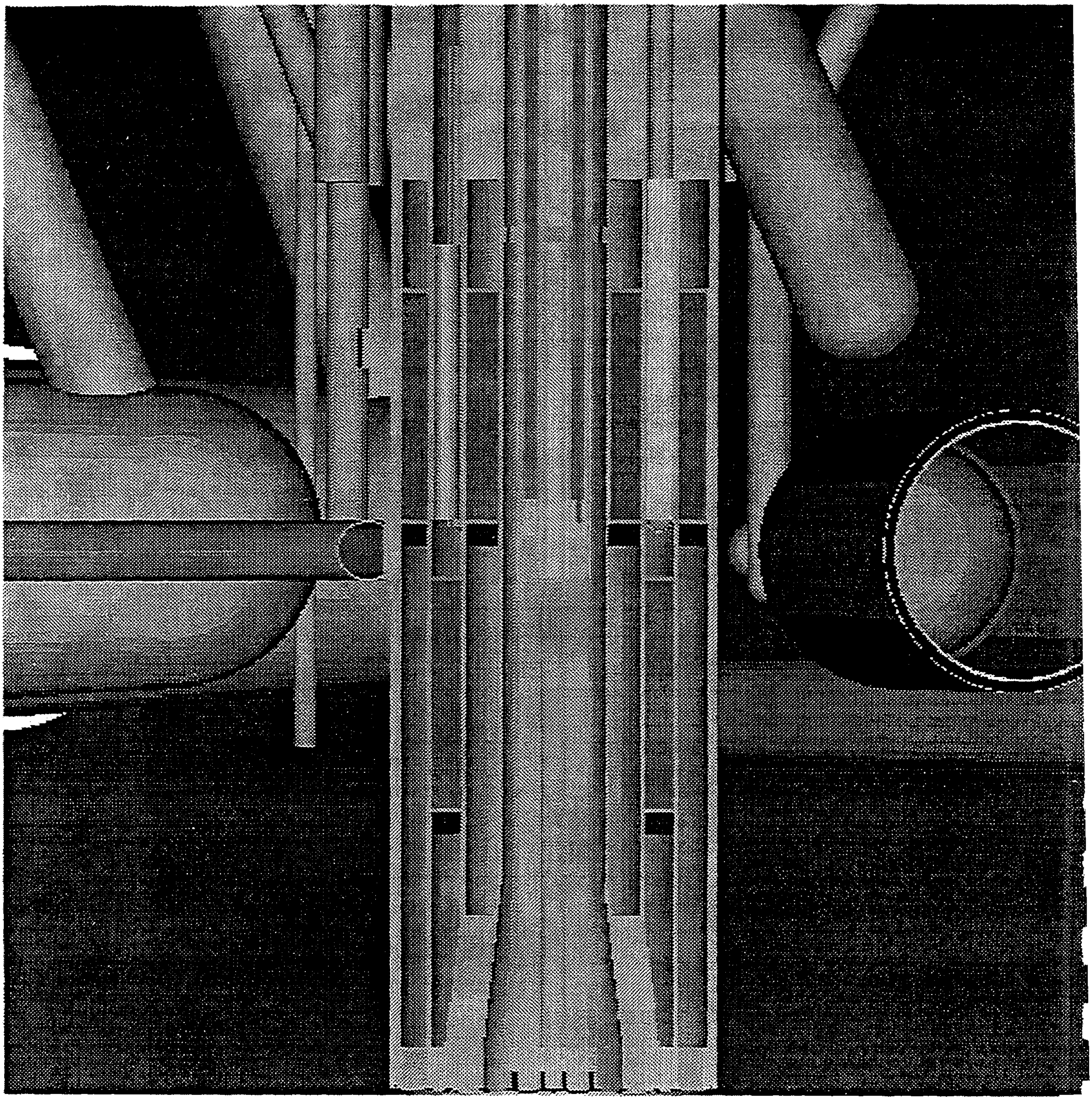
- MIT
  - Pu disposition reactor analysis

C062-WHT-595-08

Advanced Neutron Source Reactor

- Full complexity model
- Up to 210 fuel zones
- Up to 42 boron zones
- 50 fission products tracked
- Actinides from U-234 to Pu-242 tracked
- Up to 6 depletion steps

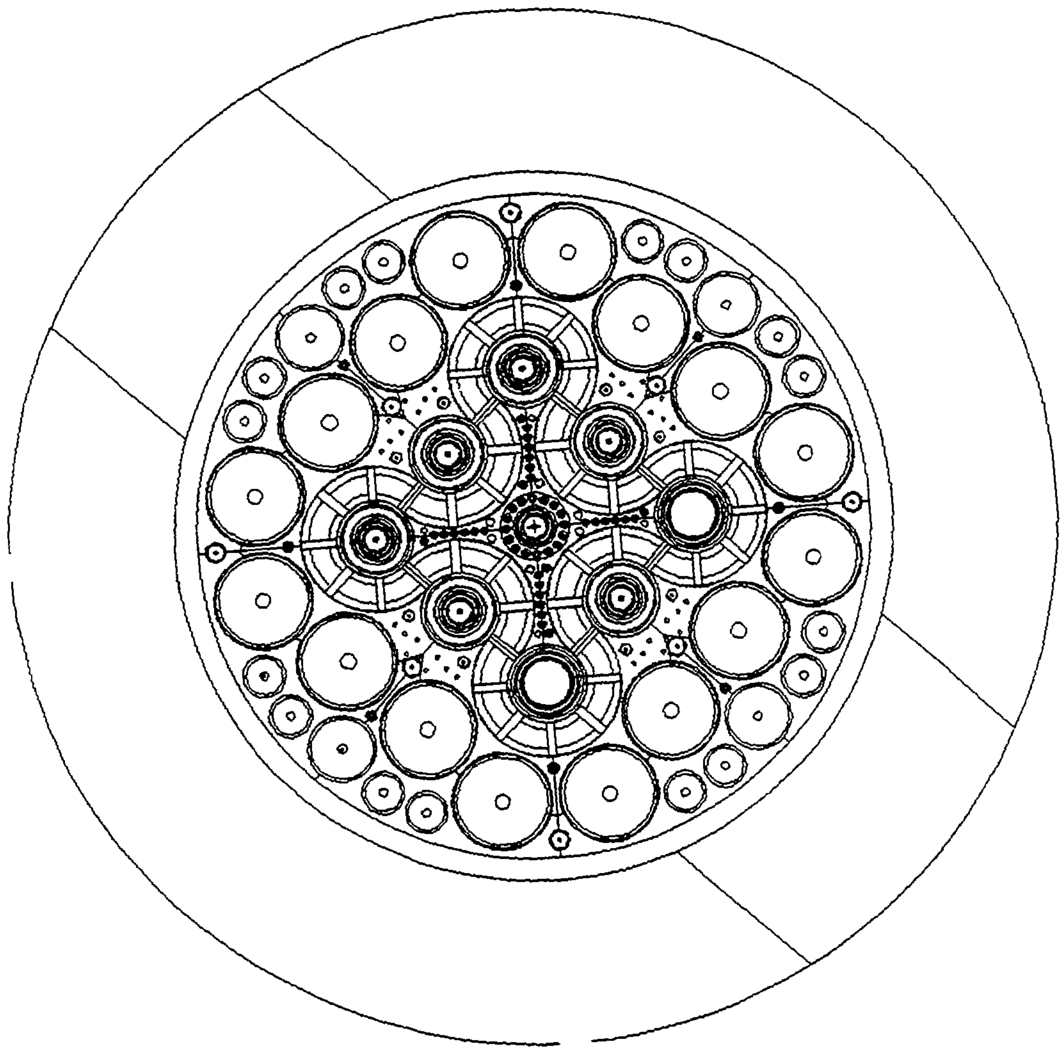




Advanced Test Reactor

- Full complexity model
- Some fuel plates explicitly modeled
- 40 fuel assemblies
- Boron zones
- 50 fission products tracked
- Actinides up to Pu-242 tracked
- Calculations for
  - Pu-238 production with Pu-236 contamination
  - Gd filter depletion in fusion materials test vehicle
  - Isotope production (I, Co, etc.)

C062-WHT-595-10



Pu Disposition Studies (continued)

- MIT work
  - Student MS thesis
  - Peripheral regions of PWR
  - Very high burnups (to 100 MWd/kg)
  - Variety of burnable absorbers
  - Variety of fertile isotopes
  - Variety of lattice parameters
    + F/M ratio
    + Pins per assembly (17 x 17 to 10 x 10)

C062-WHT-595-11b

Pu Disposition Studies

- Work at both INEL and MIT

- INEL work
  - Weapons-grade Pu disposition
  - Non-fertile fuel types
  - LWR-type lattices
  - Variety of burnable absorbers
  - Very high burnups

Conclusions

- MOCUP is an excellent tool for non-lattice reactor analysis
- Unique capability
- Available for "beta testing" now
- General distribution by late '95

Future of MOCUP

- Final stages of development
- Available for distribution - late FY '95
- User manual in editing
- User-defined modifications accepted
- INEL copyright pending

C062-WHT-595-20

Graphical User Interface Simplifies MCNP Use and Provides Burnup Capabilities

presented by Bryan Lewis

Atom Analysis, Inc.

Research Reactor Analysis Program (RRAP)

Overview

- Provides MCNP interface
- Based on Object-Oriented Programming
- Substantial work leading up to RRAP
- Models are very detailed
- Simplifies data management for burnup calculations
- Demo / Results

Looking for candidates interested in development of RRAP versions for other reactor types

RRAP's MCNP Features

- Customized for each facility's reactor geometry
- Allows geometry and/or material changes via a graphical interface
  - move rods around
  - optimize irradiation facilities
- Builds full core MCNP input model (fully commented)
- Retrieves data from output file, then processsses and stores it (typical run may bring in over 1,000 values)
- Provides a variety of calculation results
  - excess reactivity
  - flux spectra in irradiation facilities
  - 3-D burnup in fuel
  - activation levels in samples

[redacted]

RRAP History

- Five years ago, funded by U.S. Air Force for Space Reactor Design Optimization
- In 1992 AAI recieved funding to develop the Detailed Reactor Analysis Code (DRAC) that allowed optimization of four types of thermionic space reactors
- In 1993 AAI started modification of DRAC to make RRAP
- Currently there are three TRIGA facilities successfully using RRAP
- Ready to expand to other Research Reactor types

MCNP Model Details

- Full core models
- Includes all beam port holes in reflector regions
- 3-D description of fuel in the core
- Ability to adjust control rod positions independently
- Typical input file ~2500 lines
- burnup rates are calculated in five axial loactions for each rod
- Tracks fuel inventory for every rod in the reactor and in storage

[redacted]

MCNP Burnup

- MCNP can provide very good reaction rates
- These rates along with a given time step leads to very good burnup data
- Current methods under review by LANL
- MCNP not traditional burnup code due to large amount of manual data manipulation required
- Solved by RRAP

Fission Product & Chemical Energy Releases During Core Melt Events in U-Al Research Reactors

by

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May 24-25, 1995

Prepared for Presentation at IGORR-IV, Gatlinburg, TN, USA

This Presentation Will Highlight

- o Analysis & Modeling of Fission Product Releases From Heated Uranium-Aluminum Reactor Fuels

  - UAl (alloy and dispsersion); U_3O_8-Al, UxSiy-Al Fuels
  - Modeling of Burnup, Transient & Individual Species Releases
  - Correlation Library Development & Statistics

- o Modeling of Aluminum-Water Ignition

  - Modeling aspects
  - Key predictions and comparisons against data

<u>Key papers summarizing above:</u>

1) R. P. Taleyarkhan, "Analysis & Modeling of Fission Product Release from Heated Uranium -Aluminum Plate Type Reactor Fuels," Nuclear Safety Journal, Vol. 33-1, 1992.
2) S. N. Valenti, V. Georgevich, S. H. Kim and R. P. Taleyarkhan, "The Importance of Fragments Size Distribution on Underwater Aluminum Ignition," Proceedings of ANS, San Francisco, CA (11/93).

Table 1. Salient Aspects of Fission Product Release Experimental Programs

Institution (Researchers)	Fuel Type	Burnup (%)	Ambient	Temperature Range (K)	Heating Time (min)	Principal Species Investigated
HEDL (Woodley et al)	UAl_4, U_3O_8	52	Air, Steam Argon	973 - 1373	2.5	Noble Gases, I, Cs, Te
ORNL (Parker et al)	UAl_3	24	Air, Steam Helium	973 - 1373	2, 60	Noble Gases, I, Cs, Te, Ru
(Shibata et al)	Dispersed UAl_x	62	Helium	< 973	30	Noble Gases
JAERI (Saito et al)	Fuels Dispersed in Al U_3Si_2-Al UAl U_3Si_2-U_3Si-Al	23	Air	973 - 1373	60	Noble Gases, I, Cs, Te, Ru

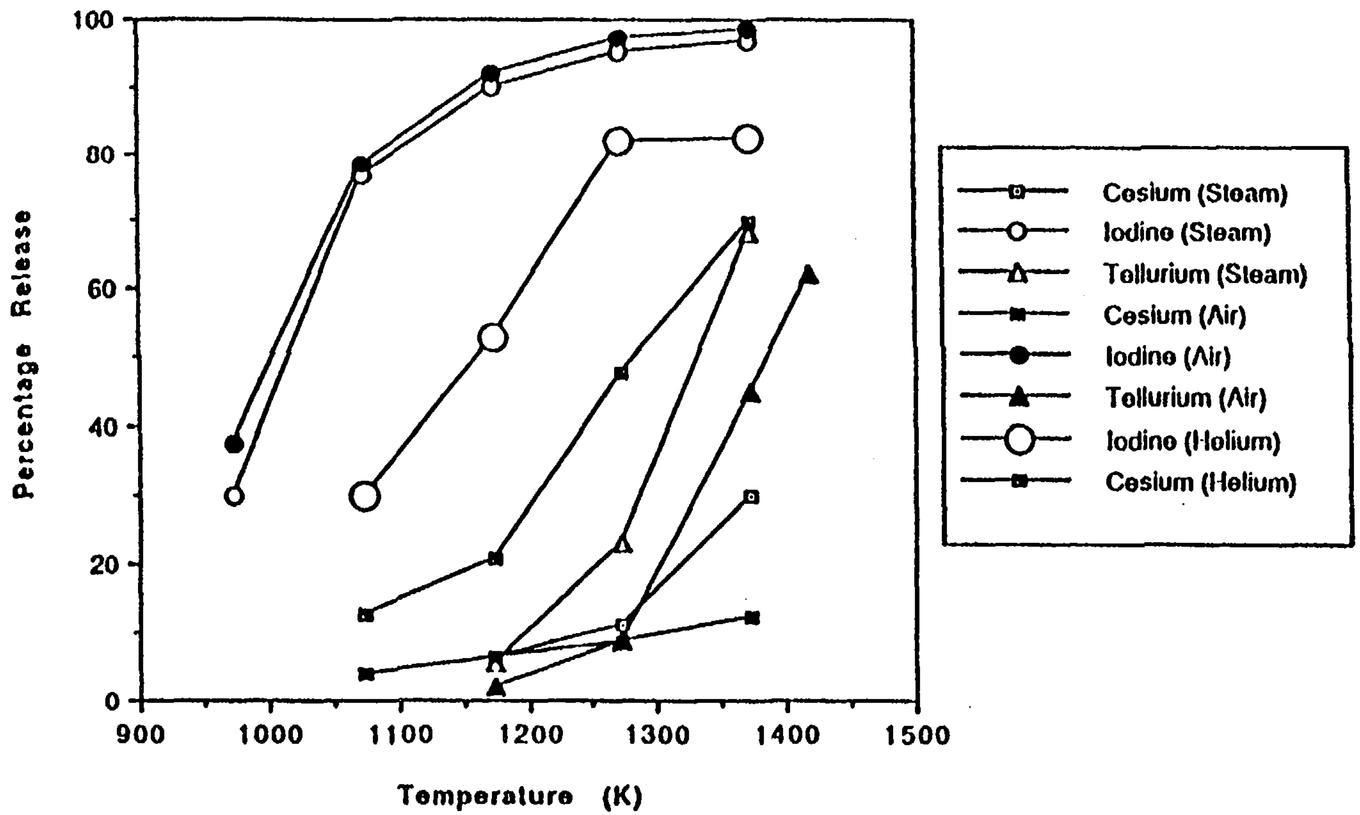


Figure 3. Variation of Volatile Fission Product Releases in Steam, Air, & Helium (ORNL Data)

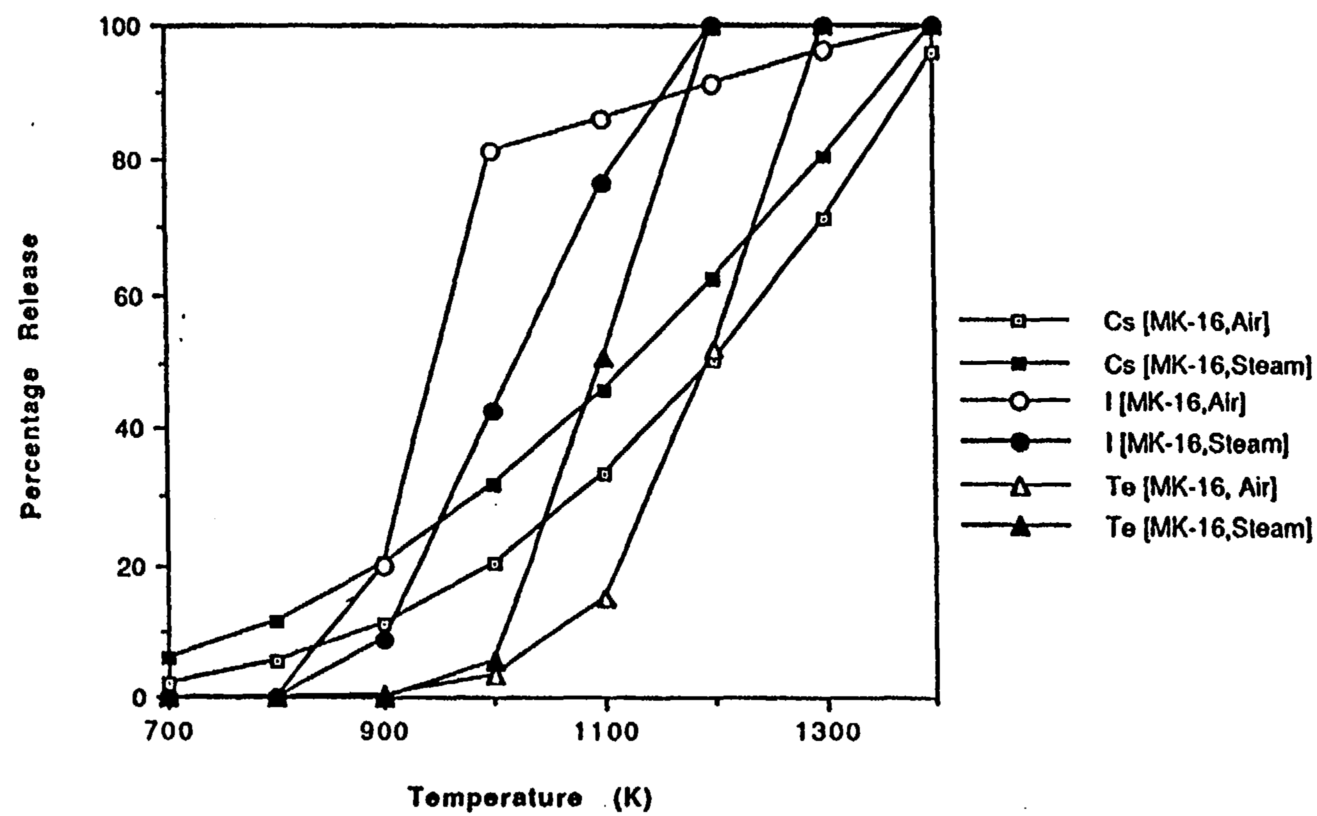


Figure C.16 Variation of Volatile Fission Products in Air & Steam [HEDL Data]

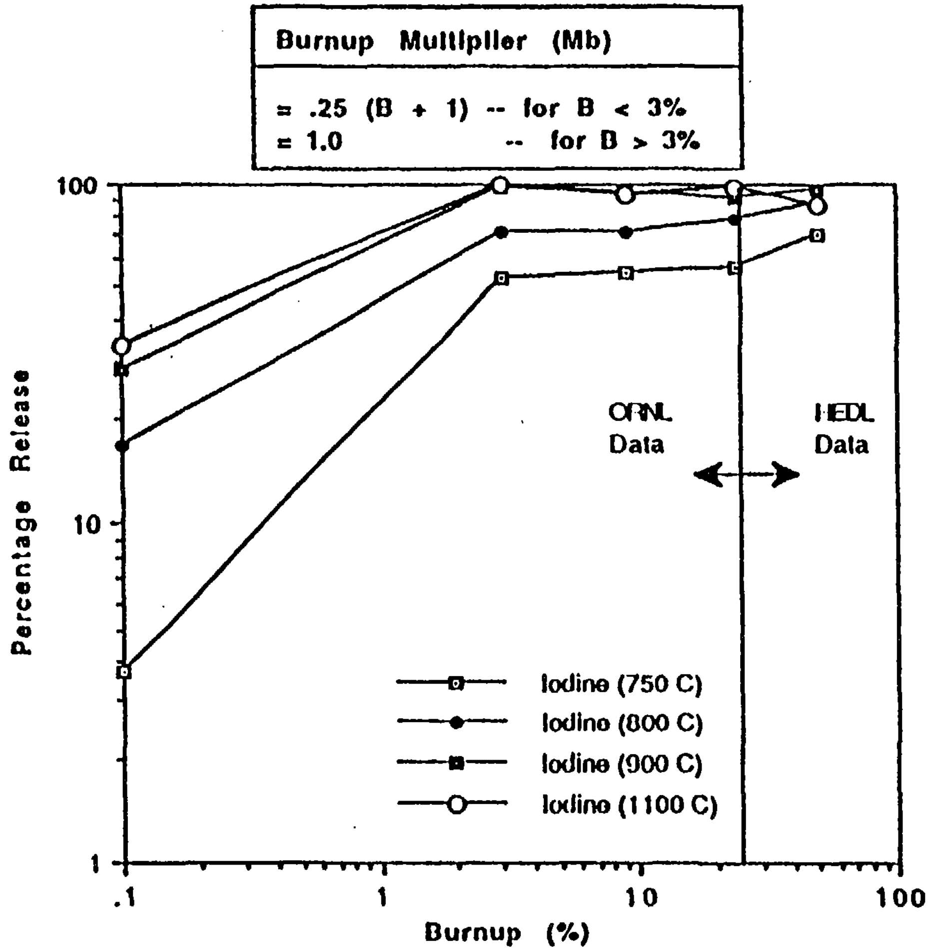


Figure C.13 Variation of Iodine Release From UAl Alloy Fuel With Burnup

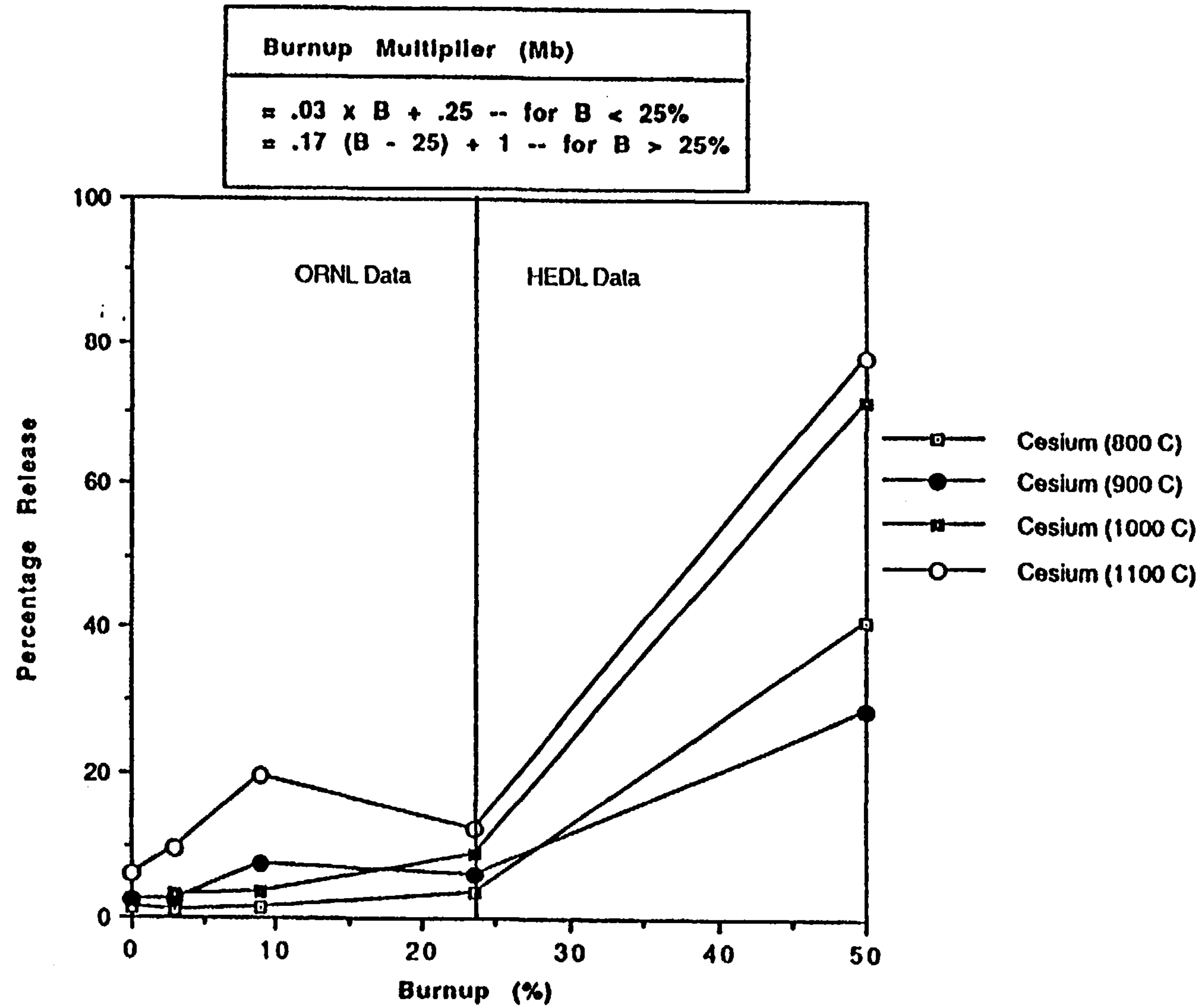


Figure C.12 Variation of Cesium Release From UAl Alloy Fuel With Burnup

TRANSIENT FISSION PRODUCT RELEASE

- NONE OF THE EXPERIMENTAL PROGRAMS STUDIED TRANSIENT EFFECTS

- STUDY OF ORNL DATA FOR UAL ALLOY FUEL INDICATED A UNIQUE & SIGNIFICANT DEPENDENCE OF RELEASE AMOUNTS WITH HEATING TIME

- IMPACT OF USING CONVENTIONAL CORSOR MODELING APPROACH
  - $R(t) = 1 - \exp(-kt)$ -- (rate constant 'k' based upon data taken over a certain time frame
  - Study to Evaluate Potential Inaccuracies in Using CORSOR Approach

TRANSIENT FISSION PRODUCT RELEASE

PRELIMINARY CONCLUSIONS

- Using Conventional CORSOR Approach
  - can give rise to significant over- or under-predictions
  - should be used carefully in codes such as MELCOR, CONTAIN, etc.

- Different Approach is Necessary for Capturing Time Dependence
  - Necessity of Additional Data for Guidance and/or Confirmation

PRACTICAL CONSIDERATIONS

- Can Have Significant Impact On:
  - Evaluation of Delayed Release Effects
  - Evaluation of Core Melt Progression Phenomena (e.g., Structural Ablation)
  - Debris Coolability & Dispersion
  - Molten Core Concrete Interaction

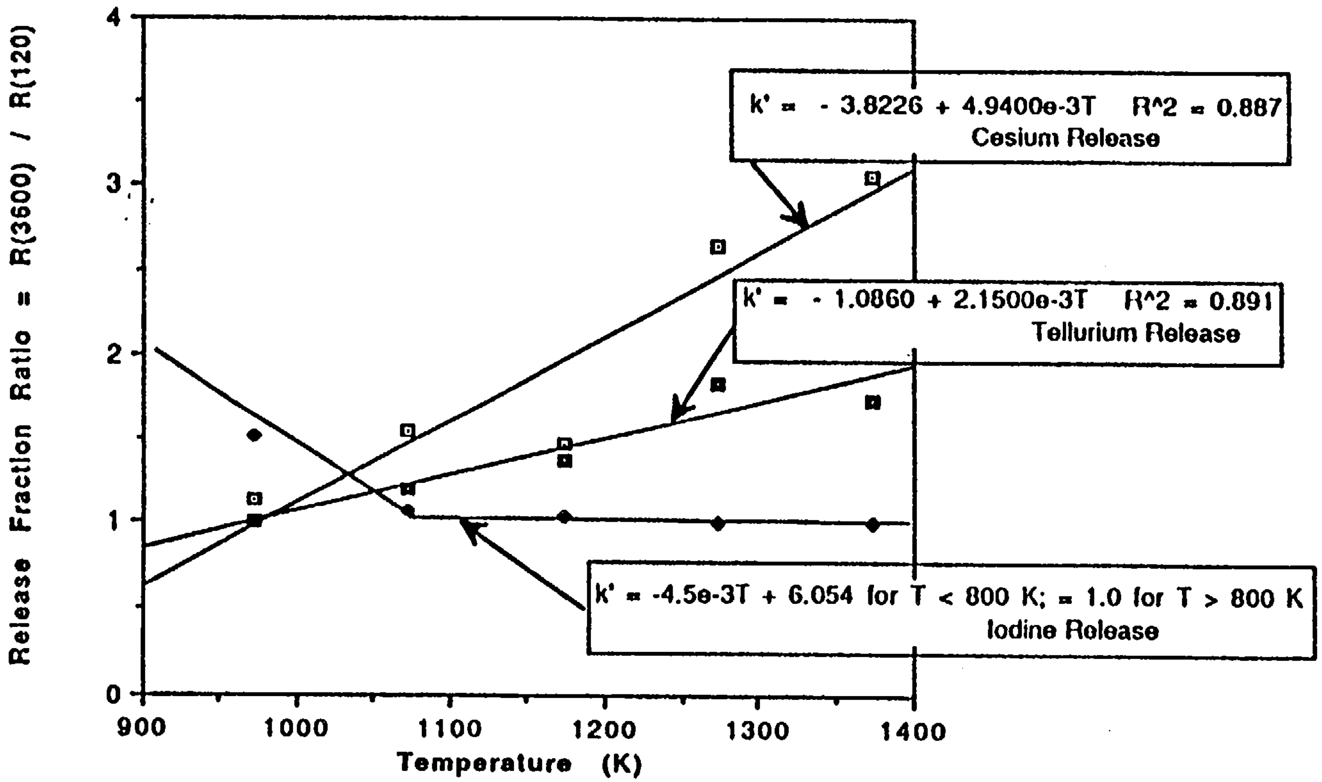


Figure 1. Variation of Ratio of Releases (60 min. to 2 min.) vs Temperature

GENERAL FORM OF CORRELATION

- FOR EACH FUEL TYPE & INDIVIDUAL FISSION PRODUCT SPECIES
  - R(t, T, Bu, Ambient) = f1(Bu, T) x f2(Ambient) x R(t, T)
  - f1(Bu, T) = Burnup Dependent Function
  - f2(Ambient) = Ambient Dependent Function or Multiplier
  - $$R(t,T) = \begin{cases} R(120,T) \text{ -- for } T < 120s \\ R(120,T) + R(120,T) \times [k'(T) - 1.0] \times (t-120)/3480 \text{ -- for } t > 120s \end{cases}$$
  - k'(T) = R(3600,T) / R(120,T)

- ASSUMPTION
  - In the Absence of Prototypical Data Time and Temperature Dependence as Observed for UAl Alloy Fuel in Air Would be the Same for Other UAl Reactor Fuels and Different Ambient Conditions

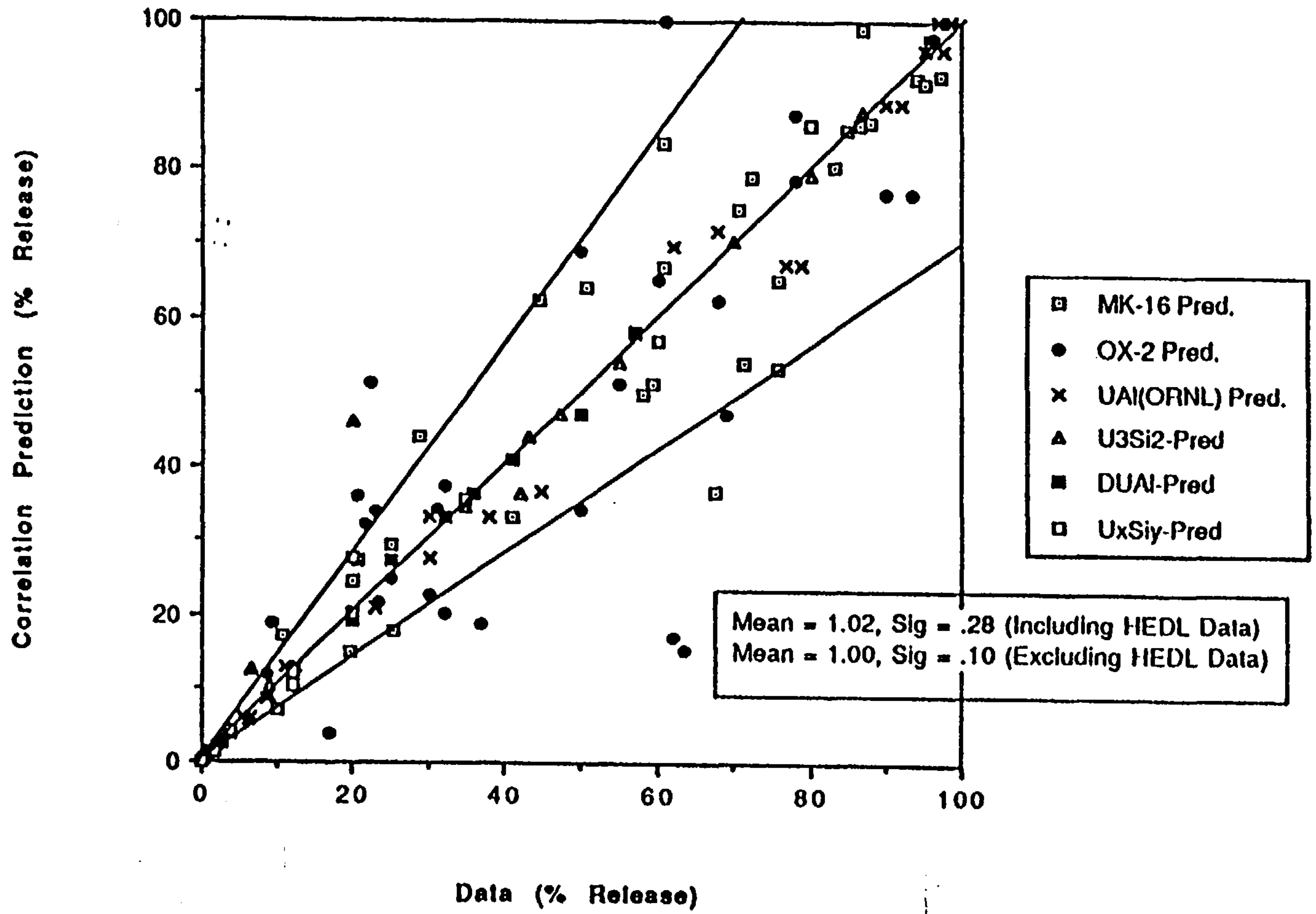


Figure C.74 Variation of Suggested Form (Combination) Correlation Predictions vs. Experimental Data (Overall Statistics)

SUMMARY & CONCLUSION

- **AVAILABLE UAL-FUEL FISSION PRODUCT RELEASE DATA ANALYZED**
  - Extensive Library of Correlations Developed For Predicting Releases Which May Vary With Time, Burnup, Ambient, & Fuel-Type Subject To Certain Assumptions
  - Correlations Developed In Various Forms For U-AL (Dispersed/Alloy), U_3O_8-AL (Dispersed) and Dispersed U_3Si_2-AL, & U_3Si-Al Fuels
  - Overall Statistics Quite Favorable

    (Mean = 1.0/1.04, Sig = .10/.28 - Excluding/Including HEDL Data)

- **UNRESOLVED ISSUES & DATA NEEDS FOR BEST-ESTIMATE ANALYSES OF REACTORS USING U_3Si_2-AL FUEL**

- **COOPERATIVE EFFORTS**
  - JAERI/ORNL Joint Development Program
  - Interactions With Other Programs
  - Integration With Other Safety & Design Issues

ORNL

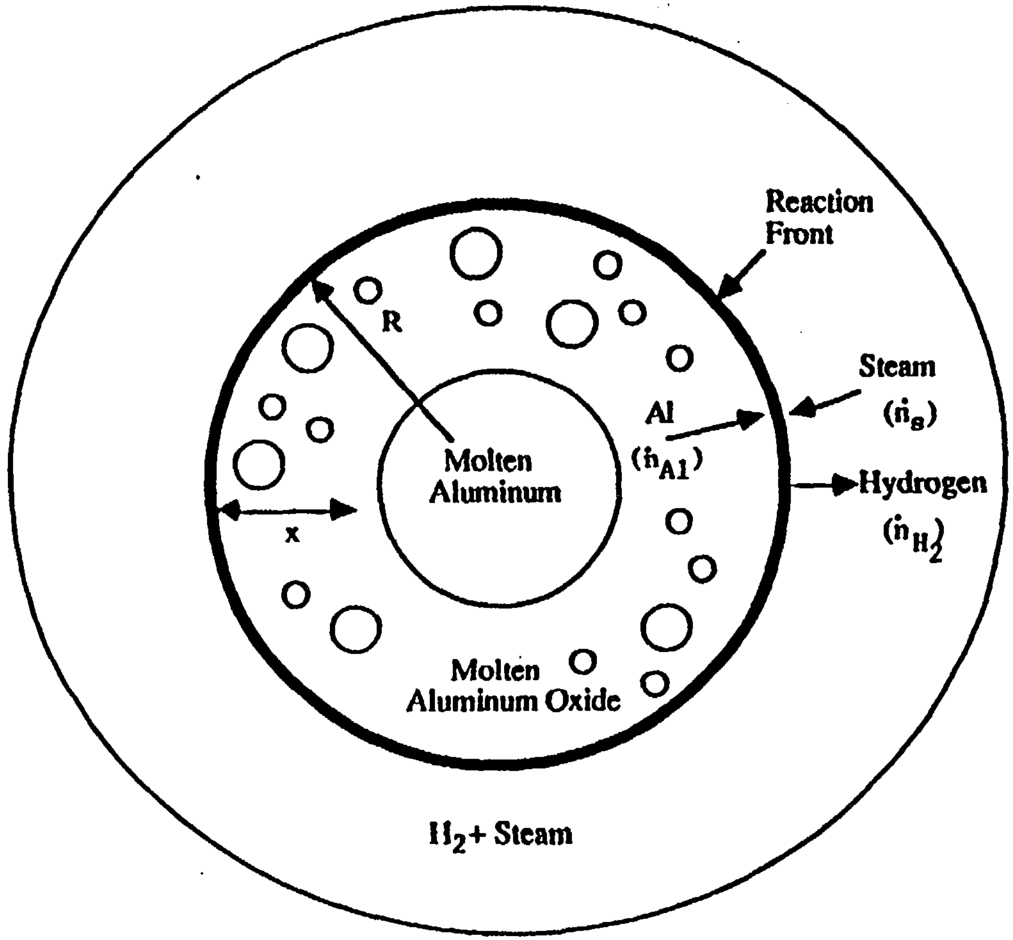


Figure 1. Molten Aluminum Droplet Model.

ORNL MODEL INCLUDES:

- ALUMINUM MASS BALANCE
- ALUMINUM OXIDE MASS BALANCE
- ALUMINUM ENERGY EQUATION
- ALUMINUM OXIDE ENERGY EQUATION
- DROPLET MOMENTUM EQUATION
- 2 CRYSTALLIZATION RATE EQUATIONS
- EXTENT OF REACTION CALCULATION

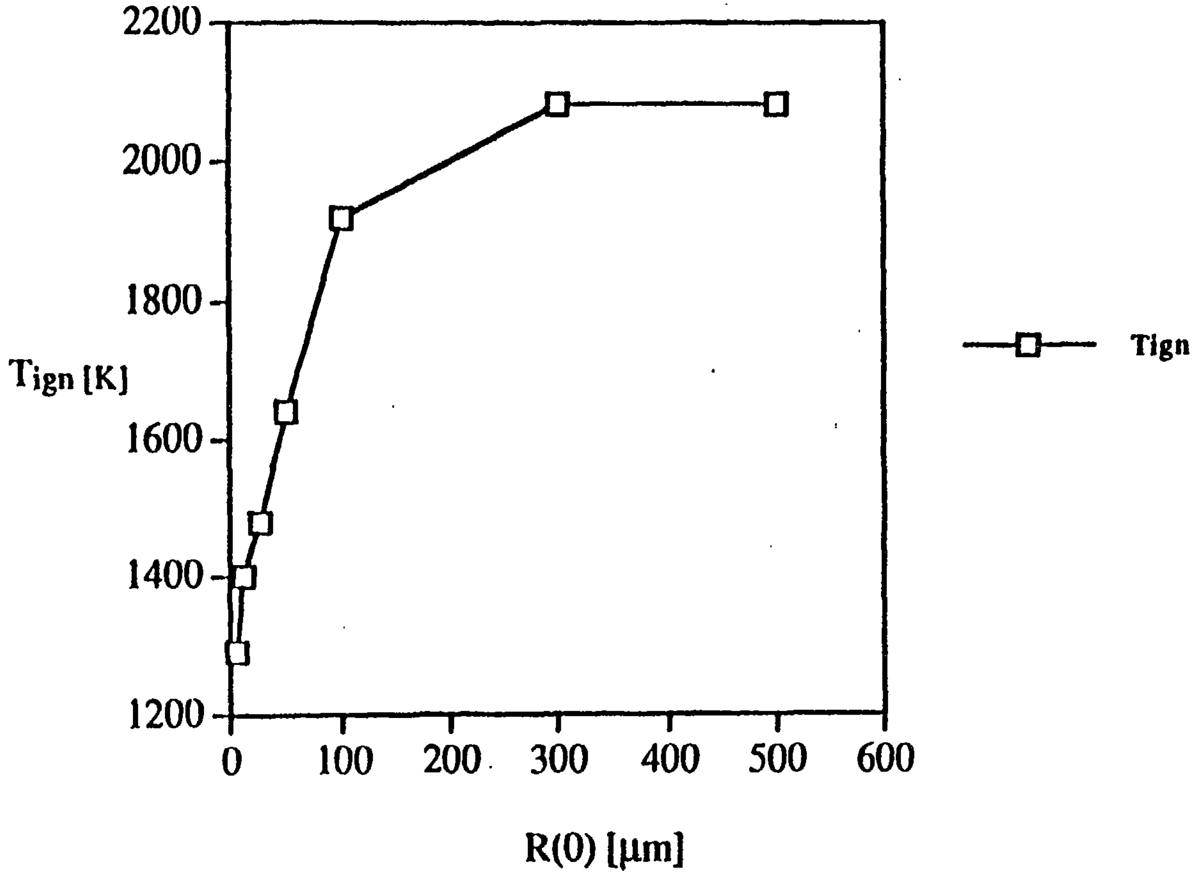


Figure 10. Ignition Curve (P_∞=10 MPa, v_∞=22 m/s).

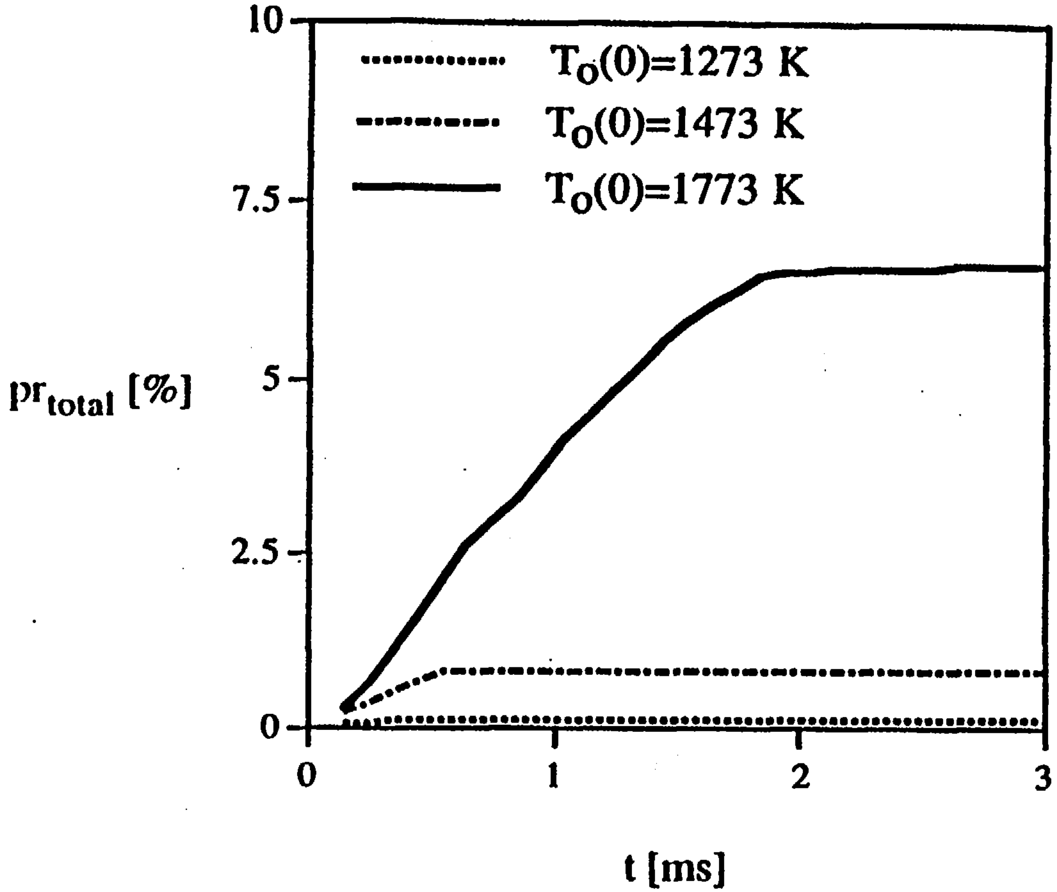


Figure 15. Total Extent of Reaction Predicted for Nelson's Experiments [4]
(P_∞=10 MPa, v_∞=22 m/s,σ_{fs}=0.49 N/m).

NELSON'S DROPLET EXPERIMENT

- SMALL-SCALE ALUMINUM/WATER EXPERIMENTS
- FOR LOW TEMPERATURES: THERMAL-TYPE INTERACTIONS OCCUR WITH BUBBLE COLLAPSE
- FOR THERMAL CASES (1273 K, 1473 K), NELSON ESTIMATED A RELATIVE VELOCITY OF 22 M/S.
- IGNITION-TYPE INTERACTION OCCURRED FOR 1773 K, WHEREIN 3 TO 6 % OF AL PARTICIPATED IN THE EXPLOSION.
- DEBRIS SIZE DISTRIBUTION WAS MEASURED.

CONCLUSIONS

- IMPORTANCE OF CAPTURING FRAGMENT SIZE DISTRIBUTION WAS DEMONSTRATED.
- RESULTS AGREE WITH NELSON'S OBSERVATIONS FOR ONSET OF IGNITION.
- EXTENT OF REACTION PREDICTED AGREES VERY WELL WITH NELSON'S OBSERVATIONS.
- THE NEED TO DEVELOP AN APPROPRIATE FRAGMENTATION MODEL WAS EVIDENT.
- EXTENSION TO LARGE-SCALE EXPLOSIONS REQUIRES FURTHER RESEARCH (ONGOING).

SIEMENS

XA04C1701

IGORR IV MAY 23 - 25, 1995

GALTINBURG, TN, USA

FISSION PRODUCT RELEASE FROM THE MOLTEN RESEARCH REACTOR CORE, FRM-II

H. J. DIDIER SIEMENS AG (KWU)

I. Agenda

1. Background for the investigations
2. radiological design basis accident.
3. activity inventory in the fuellement
4 development of the accident
5 radiation exposer in the environment

5.1 design basis accident
5.2 whole core (beyond design)

II. Main aspects

ad 1

- main assumption for realasation of the FRM II-project:
  minimizing of nuclear risks,
  political and social acceptance
- safety report and independant expert's reports
- accident analyses

ad 2

- radiogical design basis accident = postulation of 15 (out of 113) plates of the core melting under water
- defect on one plate, influence to the neighbouring plates, safety factor 5 mounts to 15 plates = 13 %
- beyond design accident on request of the licensing body: Melting of the whole core under water

(ref. Fig 1 and 2)

ad 3

- calculation of the relevant fission products in the melt (ref Fig 3)

ad 4

- barriers for fission products release

  ° fuel - water
  ° water - reactor hall
  ° reactor hall - environment

- retaining release factors
- fission product release via the ventilation stack (ref Fig 4 and 5)

ad 5.1

- radiation exposer design basis accident, doses for organs and effectiv dose far below limiting value of 50 mSv (ref Fig 6)

ad 5.2

- core melting under water, building tight, low pressure ventilation system with filters in operation

- effective doses at the plant fence approx. 11 mSv for adults respectively approx. 13 m Sv for children.

- effectiv doses without low pressure ventilation system without filters, building closed, release only by leakages:

  approx 12 mSv adults
  approx. 17 mSv children

Result:

protection measures against emergencies not necessary in this cases, if it can be realized, that the core stays in the pool under water for all circumstances (ref fig 7 and 8)

III Figures

1.) reactor building, + 11,70 m

2.) reactor main pool, Storage pool primary, cooling system

3.) fission product inventory

4.) principle flow sheme for fisssion product

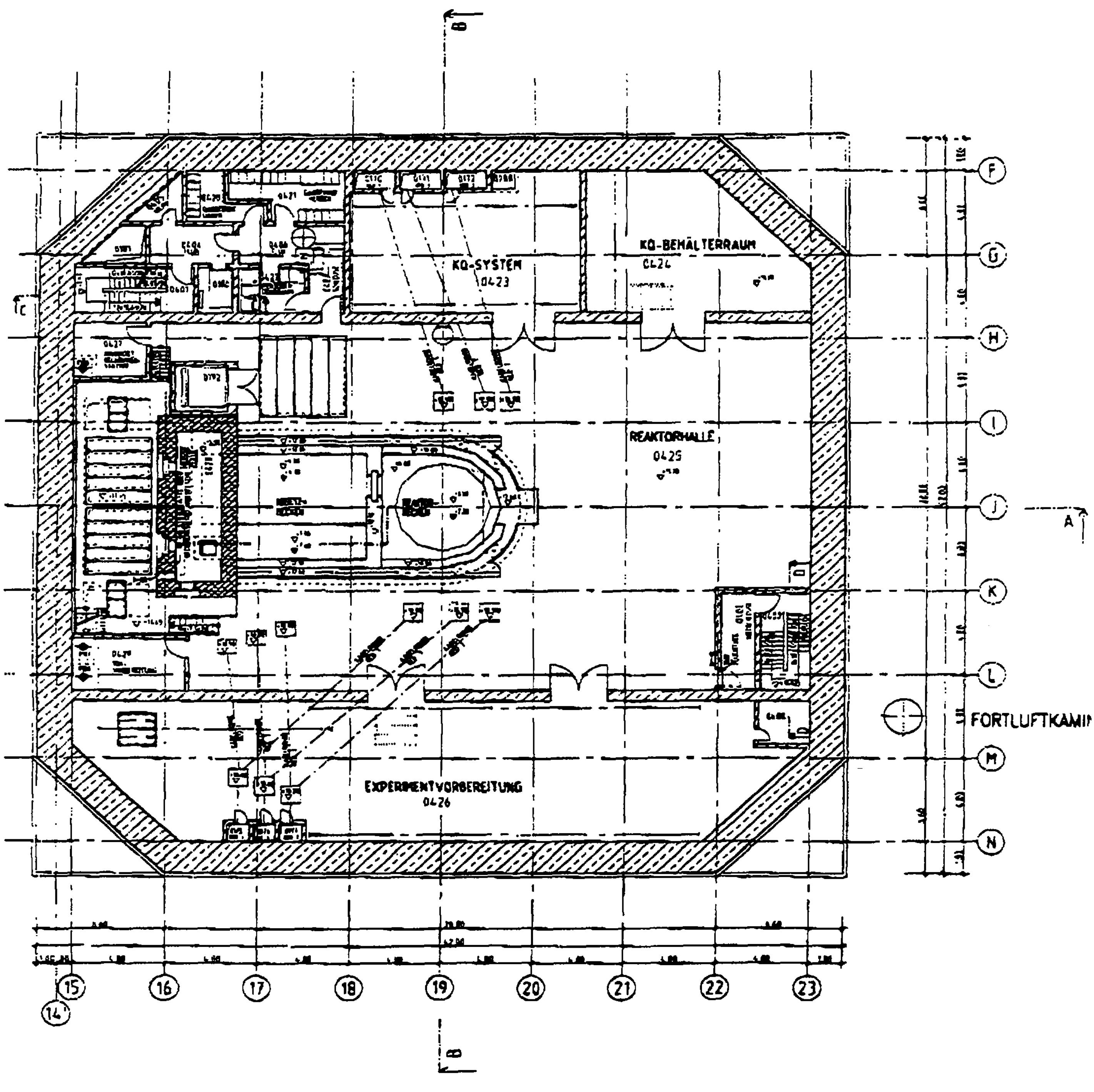
5.) release factors molten core under water

6.) effective doses in mSv for design basis accident

7.) effective dose dependent to the distance, melting of whole core under water, adults

8.) effective dose dependent to the distance , melting of whole core under water, children

UJA - REAKTORGEBAEUDE



New Munich Research Reactor FRM-II
Reactor Building, Horizontal Section, + 11.70 m **Fig. 1**

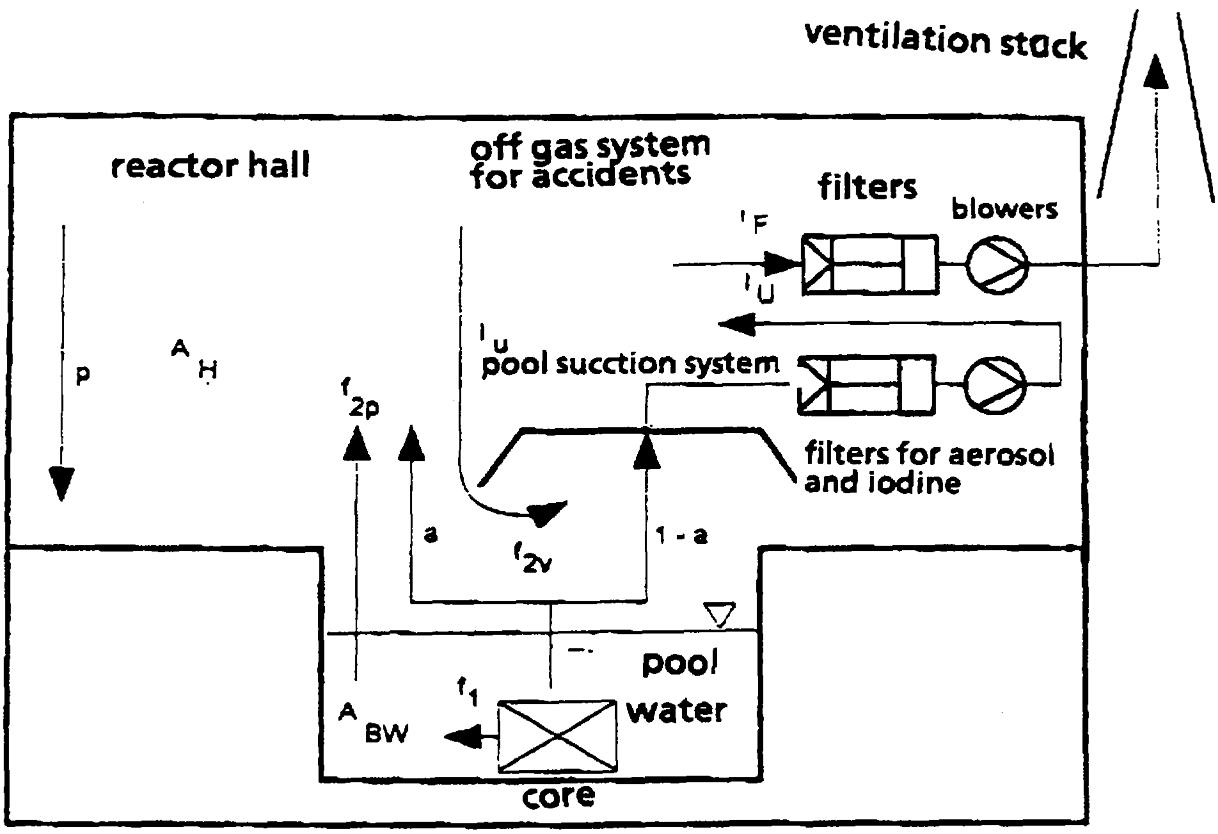
SIEMENS

fission groups	fission	activity inventry whole core
noble gas	Kr 85 Kr 85m Kr 87 Kr 88 Kr 89 Xe 133 Xe 133m Xe 135 Xe 135m Xe 137 Xe 138	1,47 E + 13 *) 7,81 E + 15 1,58 E + 16 2,23 E + 16 2,83 E + 16 4,11 E + 16 1,22 E + 15 3,89 E + 15 6,99 E + 15 3,67 E + 16 3,85 E + 16
Halogene	Br 83 Br 84 Br 85 I 131 I 132 I 133 I 134 I 135	3,30 E + 15 6,25 E + 15 7,73 E + 15 1,76 E + 16 2,69 E + 16 4,14 E + 16 4,66 E + 16 3,85 E + 16
Cäsium/Rubidium	Rb 88 Rb 89 Cs 134 Cs 137 Cs 138	2,25 E + 16 2,93 E + 16 2,51 E + 13 1,20 E + 14 4,14 E + 16
Tellur/Ruthen	Te 131 Te 132 Ru 103 Ru 106	1,59 E + 16 2,65 E + 16 1,14 E + 16 2,33 E + 14
Strontium	Sr 90	1,15 E + 14
Aktiniden	Pu 238 Pu 239 Pu 240 Cm 242	2,98 E + 10 2,47 E + 09 8,44 E + 08 9,99 E + 07

relevant fission product inventory
20 MW, 50 full powerdays

Fig. 3

SIEMENS



description of abbreviations

f_1	= prompt release into pool water
f_{2p}	= prompt release into reactor hall
f_{2v}	= delayed release via evaporation
a	= part of evaporation not taken by the pool suction system
I_F	= off gas rate
I_U	= circulation rate
p	= plate out rate
A_H/A_{BW}	= activity concentration in reactor hall and pool water
DF_U/DF_F	= retain factors of ventilation systems

principle flow sheme for fission product releases

Fig. 4

Release factor, molten core under water

fission groups	release factor melt / water (f1)	retaining factor water / air (f2)	release faction f1 * f2
	1,0	1,0	1,0
Halogene	0,75	$5 \cdot 10^{-4}$	$3{,}8 \cdot 10^{-4}$
Cäsium	0,25	$1 \cdot 10^{-5}$	$2{,}5 \cdot 10^{-6}$
	$1 \cdot 10^{-3}$	$1 \cdot 10^{-5}$	$1{,}0 \cdot 10^{-8}$

Fig. 5

	maximum dose (mSv) in the surrounding				limiting values with regard to § 28, Abs. 3 StrlSchV
	up to 2000 m		more than 2000 m		
	children	adults	children	adults	(mSv)
Bladder	1,41	1,18	0,41	0,34	150
Breast	1,97	1,64	0,57	0,47	150
Upper intestine	1,39	1,16	0,40	0,34	150
Lower intestine	1,32	1,10	0,38	0,32	150
Small intestine	1,35	1,13	0,39	0,33	150
Brain	1,85	1,54	0,54	0,45	150
Skin	2,82	2,56	1,11	1,01	300
Testicles	1,63	1,36	0,47	0,39	50
Bone - Surface	2,23	1,86	0,65	0,54	300
Liver	1,48	1,24	0,43	0,36	150
Lungs	1,65	1,37	0,48	0,40	150
Stomach	1,51	1,26	0,44	0,36	150
Spleen	1,51	1,25	0,44	0,36	150
Suprarenal gland	1,41	1,17	0,41	0,34	150
Kidneys	1,51	1,26	0,44	0,36	150
Ovaries	1,28	1,07	0,37	0,31	50
Pancreas	1,34	1,12	0,39	0,32	150
Red bone marrow	1,47	1,23	0,43	0,36	50
Thyroid	1,93	1,61	0,63	0,48	150
Thymus	1,64	1,37	0,48	0,40	150
Uterus	1,26	1,05	0,36	0,30	50
Effective dose	1,68	1,40	0,49	0,41	50

effective doses for design basis accident **Fig. 6**

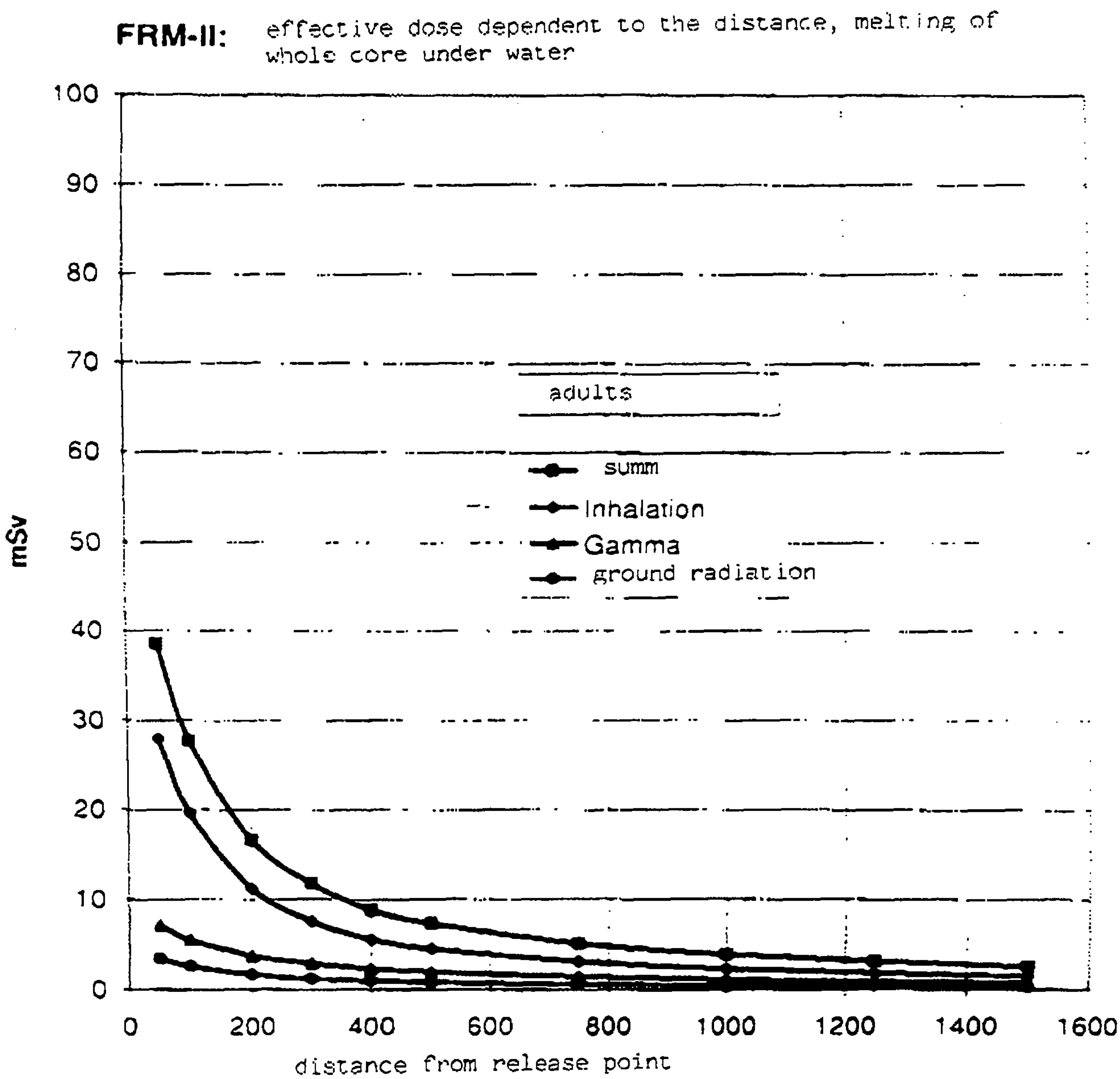

FRM-II: effective dose dependent to the distance, melting of whole core under water
adults
summ
Inhalation
Gamma
ground radiation
mSv
100
90
80
70
60
50
40
30
20
10
0
0
200
400
600
800
1000
1200
1400
1600
distance from release point

Fig. 7

SIEMENS

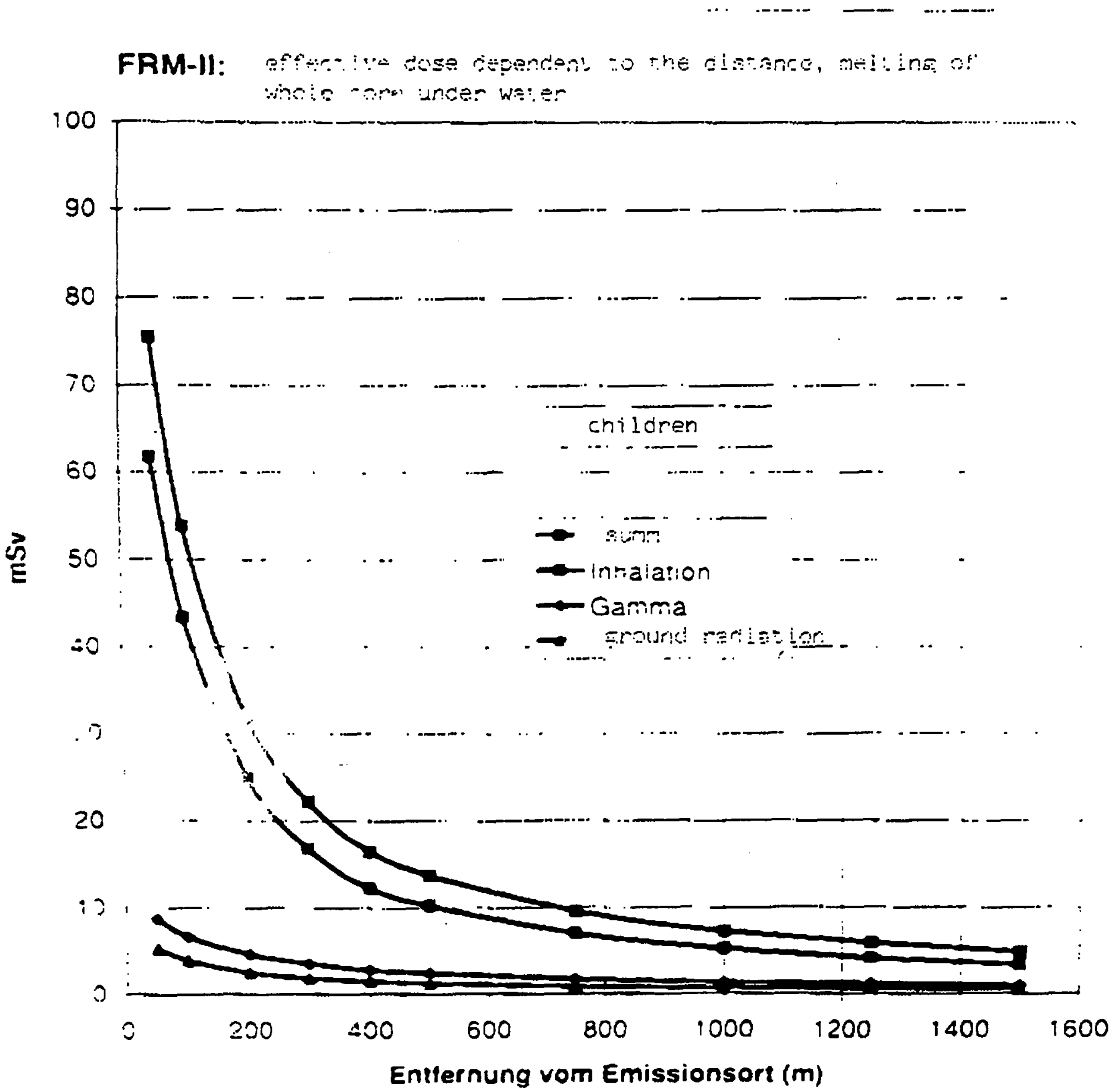


Fig. 8

XA04C1702

JAERI / ORNL Tests & Analyses on Transient Heating of U_3Si_2-Al Miniplates in Nuclear Safety Research Reactor

by

T. Fuketa* and R. P. Taleyarkhan**
(*)- NSRR Laboratory, JAERI, Tokai, Japan
(**) - Engineering Technology Division, ORNL, USA

May 24-25, 1995

Prepared for Presentation at IGORR-IV, Gatlinburg, TN, USA

370

This Presentation Will Highlight

- Overview of JAERI-ORNL collaborative work and how it fits in with ANS Severe Accident Program Plan for FCI issue closure
- NSRR tests (with ANS miniplates) during 1993-94 & beyond
- Modeling and analysis framework for:

  - Thermal-hydraulic response
  - Material breakup and dispersion

- Key results of analyses and experiments

<u>Full Papers to be presented at</u>:

1) 1995 Natl. Heat Transfer Conference, Portland, Oregon, USA
2) Nuclear Reactor Thermal-Hydraulics Conf., NURETH-7, NY, USA

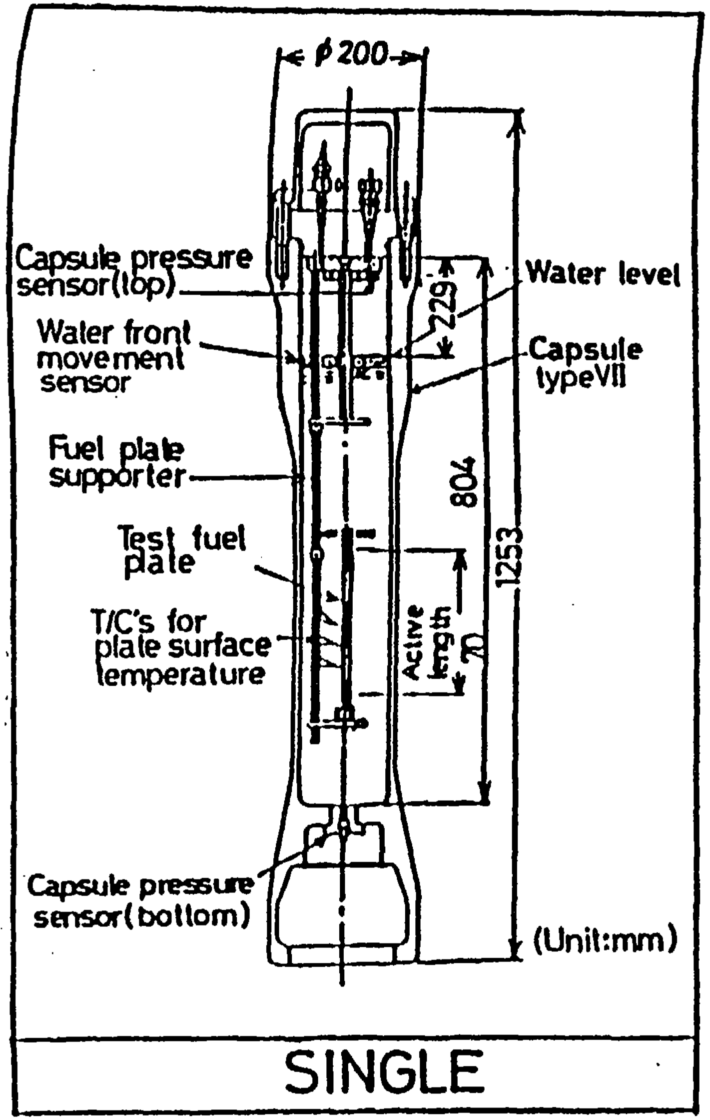


Fig. 2.4 NSRR irradiation capsule type VII prepared for the test of silicide plate-type fuel in experiment 508 series

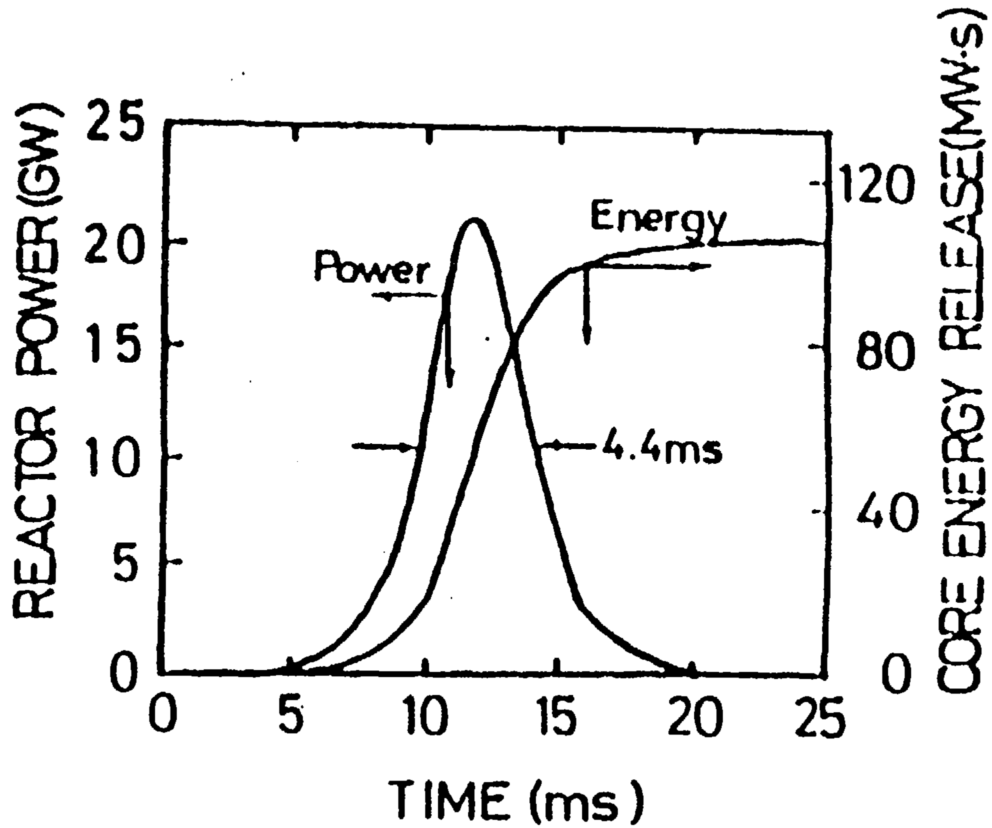


Fig. 2.5 Transient reactor power and core energy release attained in pulsing operation NSRR with $4.67 reactivity insertion

JAERI-ORNL TESTS IN NSRR

o A FEW (KEY) QUESTIONS CONCERNING ISSUE CLOSURE

- What is the likely mechanical behavior of plates during rapid heatup?
- What is the impact of fuel homogeneity on damage thresholds ?
- What are the kinetics of U_3Si_2-Al and Al-H_2O ignition during rapid heatup?
- What are the triggerability and onset requirements for material dispersion ?
- What are the energetics of a resulting (if any) energetic FCI "for ANS fuel" ?

****- How does fuel burnup affect the above outcomes ? ****

o NSRR TESTING & MODELING WORK CAN PROVIDE VALUABLE INFORMATION (esp. for closure of FCI-related issues)

- Plate cracking, bowing and steam explosion onset thresholds
- Impacts of fuel homogeneity and preirradiation on damage thresholds
- Degree and onset of aluminum ignition can be quantified
- Degree of "transient" onset and degree of U_3Si_2-Al exothermic reactions
- Realistic thermal-to-mechanical energy conversion estimates

** This information is obtained with nuclear heating, but, in the absence of external triggers and propagation - aspects to be looked at via out-of-pile testing and appropriate modeling **

Test Summary 1993–1994

Test #	Configuration	Fuel Homogeneity	NSRR Core Energy Release (MJ)	Max. Cladding Surface Temp. (°C)	Post-test Fuel Plate
518-1	Single	Inadequate	27.0	410	Mechanical failure
518-2			40.3	700	Clad melting
518-3		Improved	27.6	N/A	No failure
518-4			21.3	210	No failure
518-5			42.7	600	
518-6	with two dummy plates		27.3	320	
518-7			43.1	610	

OVERVIEW OF MODELING FRAMEWORK

o HEAT TRANSFER (--> Material response, dispersion and ignition)

- 3-D HEATING model developed (heatup, melting and freezing)
- Impact of thermocouples quantified
- Surface boiling heat transfer correlations for transient heatup
- Estimates of voiding and heat transfer in enclosed geometries

o MATERIAL BREAKUP / DISPERSION (--> explosive loads estimation)

- Multidimensional CTH, PCTH and LS-DYNA3D models of fuel plate
- 1-D modeling of important effects (strain induced due to thermal expansion, melting, vaporization and void / fission gas expansion)

o NEUTRONICS (--> Integral thermal energy generated in miniplates)

- 2-D model of NSRR, with 3-D model of miniplates using MCNP
- 1-D deterministic neutronic calculations with XSDRNPM (cross-checking)

*** Future efforts will investigate transient energy deposition profiles, and impact of pre-existing fission products ***

Overview of Neutronic Analysis Results

EVALUATION OF ETA (cal/g/MJ)

- Overall results of MCNP and XSDRNPM evaluations show good agreement with isotopic gamma scan data analyses (** data scatter **)

Fuel Plate	Gamma Scan*	MCNP		XSDRNPM
		Void Thickness(mm)		
		0	4	
ANS	7.71 (9.51)	7.79	7.43	8.1
JMTR	13.98 (17.30)	12.22	13.68	15.4

(*) -- ETA values defined over >> 1s and (1s) of NSRR pulse duration

Notes: 1) Temperature effects (up to 700°C) are less than 1%
2) The above values for ETA represent integral energy deposition

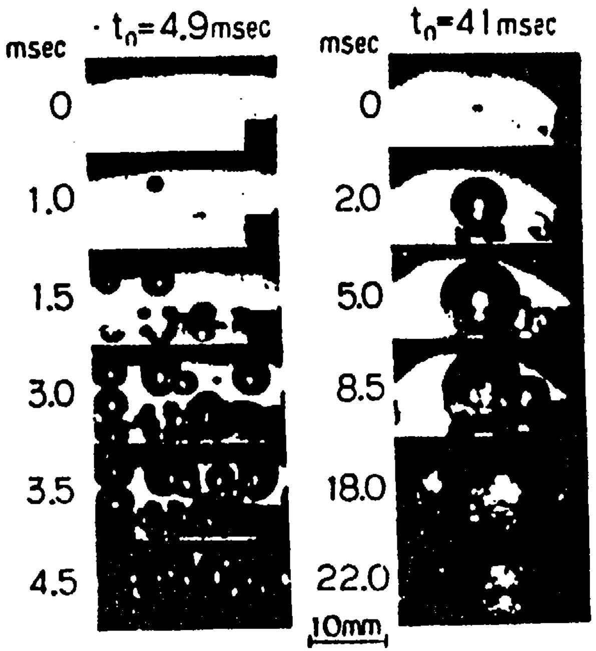


Fig.4 Dependence of bubble pattern on the period in saturated pool boiling.

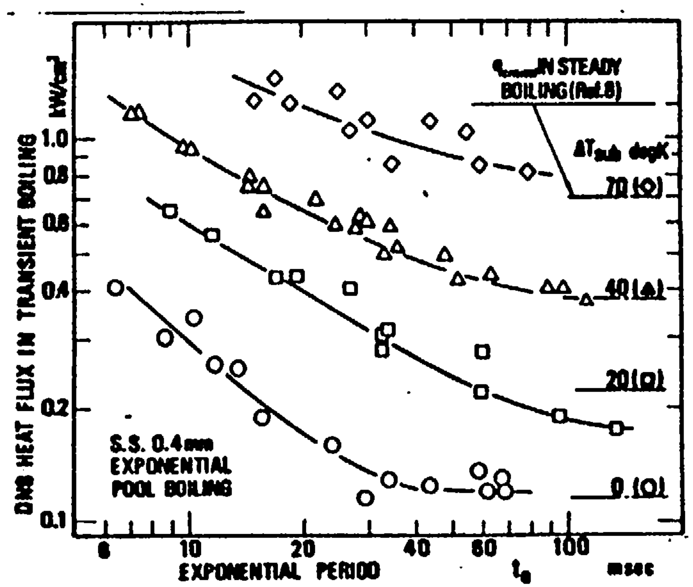
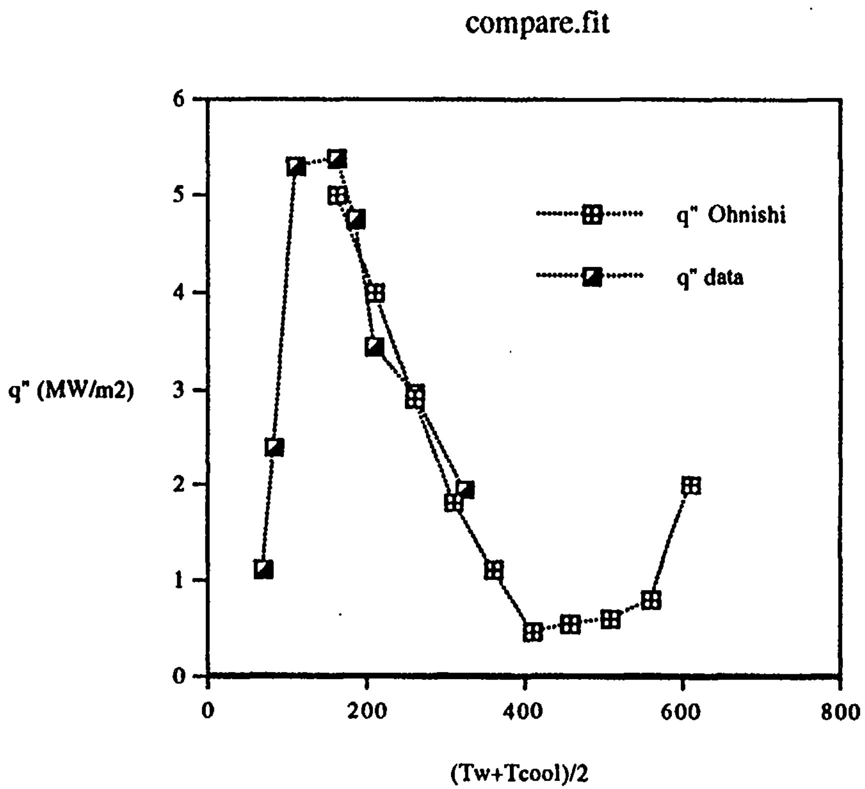
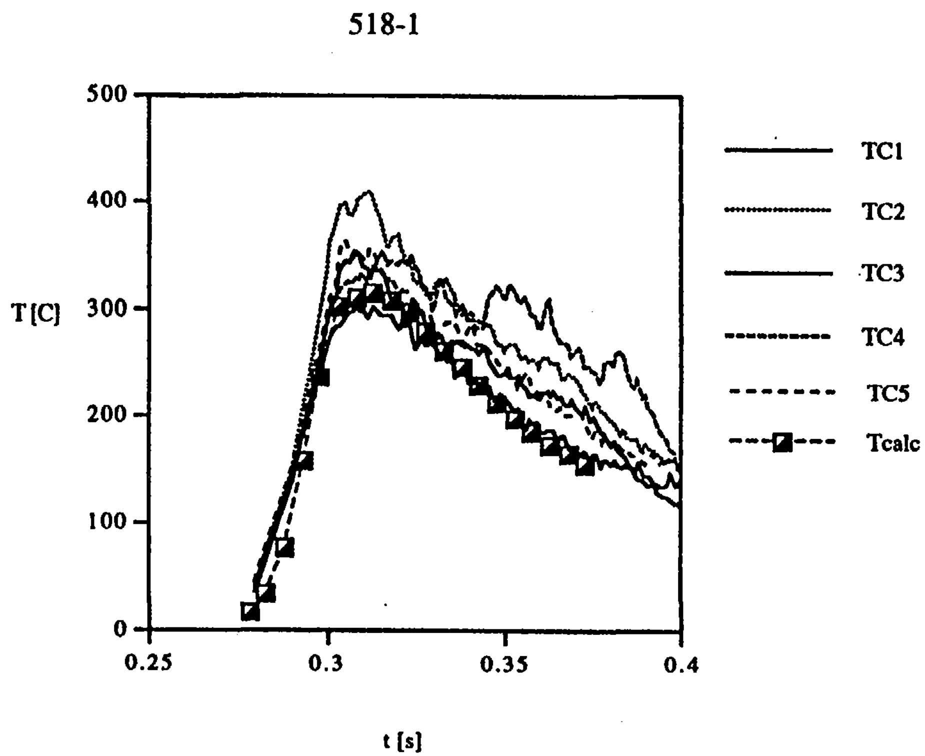


Fig.3 DNB heat flux vs. period.

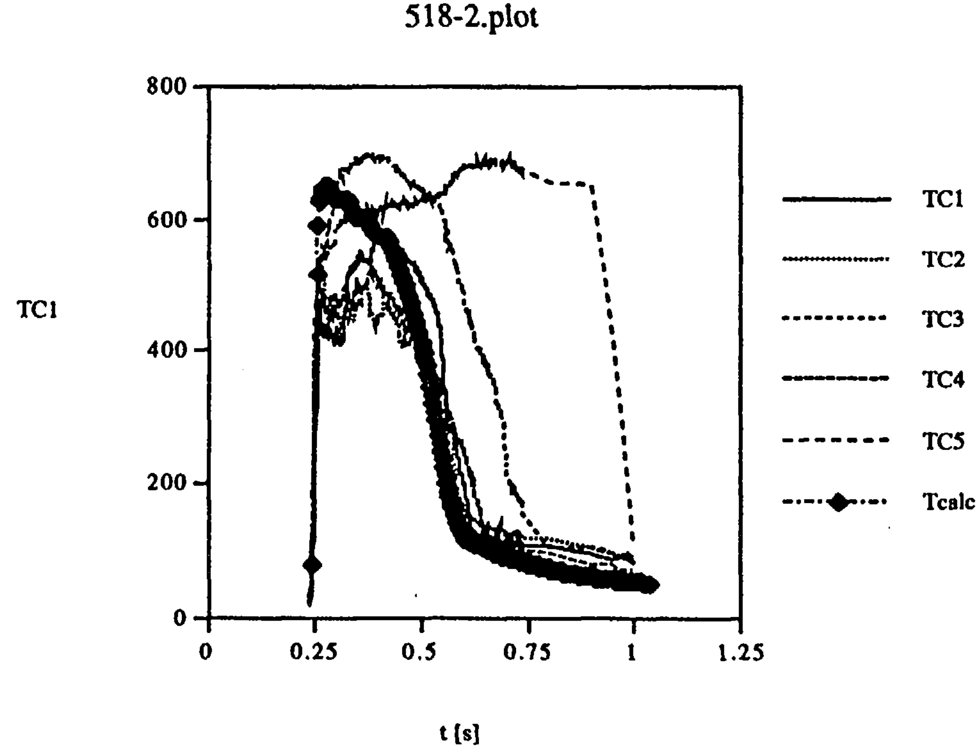
HIGHLY SUBCOOLED STEADY BOILING



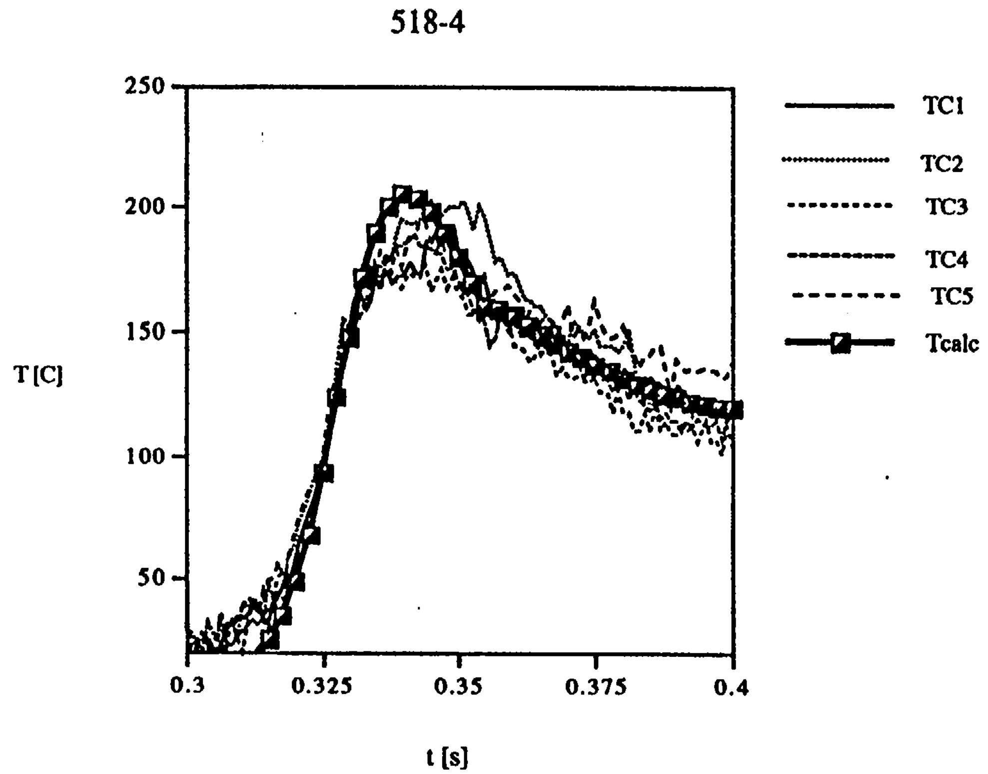
RESULTS OBTAINED FOR TEST #518-1



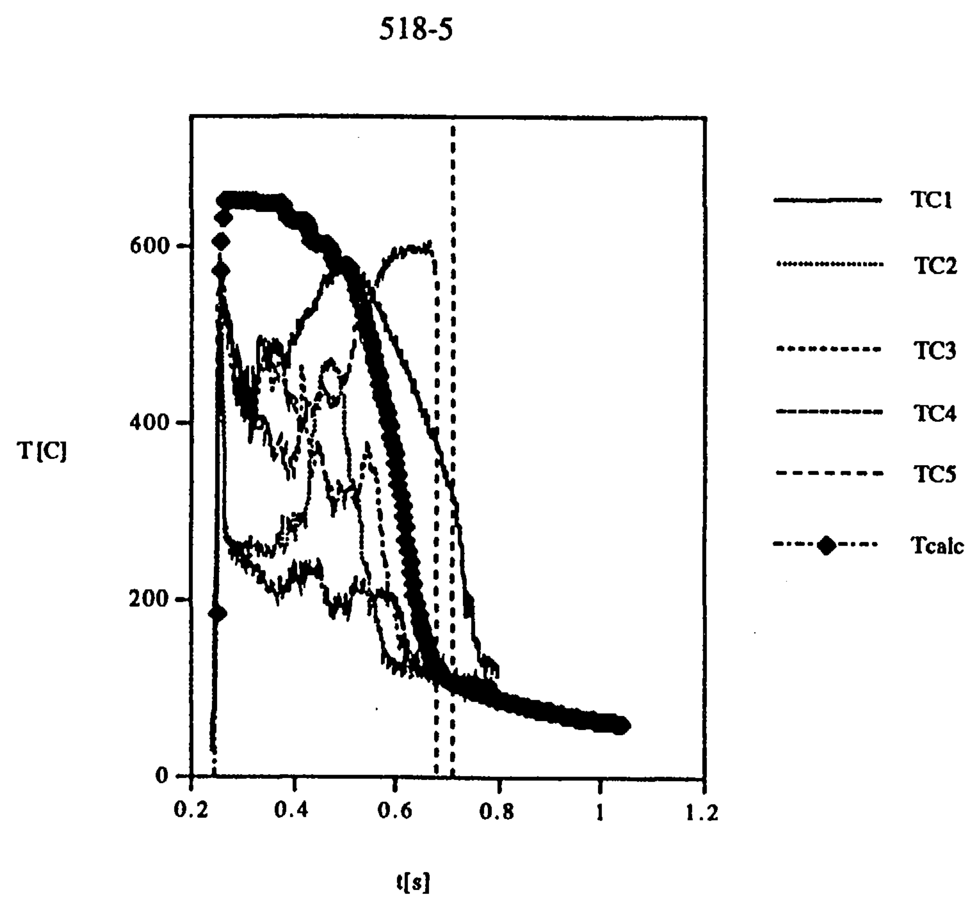
RESULTS FOR CASE 518-2



RESULTS FOR CASE 518-4



RESULTS FOR CASE 518-5



Experimental data base

o TREAT EXPERIMENTS

x <u>SL-1 fuel plate tests</u> (mass composition: 81% Al; 17% U; 2% Ni)

- U-Al alloy fuel; no clad; Dimensions: 5mm x 12.7 mm x 12.7 mm cutouts

x <u>HFIR fuel plate tests</u> (mass composition: 59% Al; 41% U_3O_8)

- Cermet (U_3O_8) fuel; Dimensions: 25.4 mm x 12.7 mm x 1.27 mm cutouts

o NSRR EXPERIMENTS

x <u>JMTR fuel plate tests</u> (mass composition: 23.3% Al; 76.7% U_3Si_2 --> 4.8 g/cc)

- Cermet (U_3Si_2) fuel; Dimensions: 125 mm x 75 mm x 1.27 mm picture frame

x <u>ANS fuel plate tests</u> (mass composition: 63% Al; 37% U_3Si_2 ---> 1.4 g/cc)

- Cermet (U_3Si_2) fuel; Dimensions: 125 mm x 75 mm x 1.27 mm picture frame

Experimental data base (cont)

	TREAT Facility	JAERI Facility
Pulse width	0.3 - 1 sec	0.015 - 0.08 sec

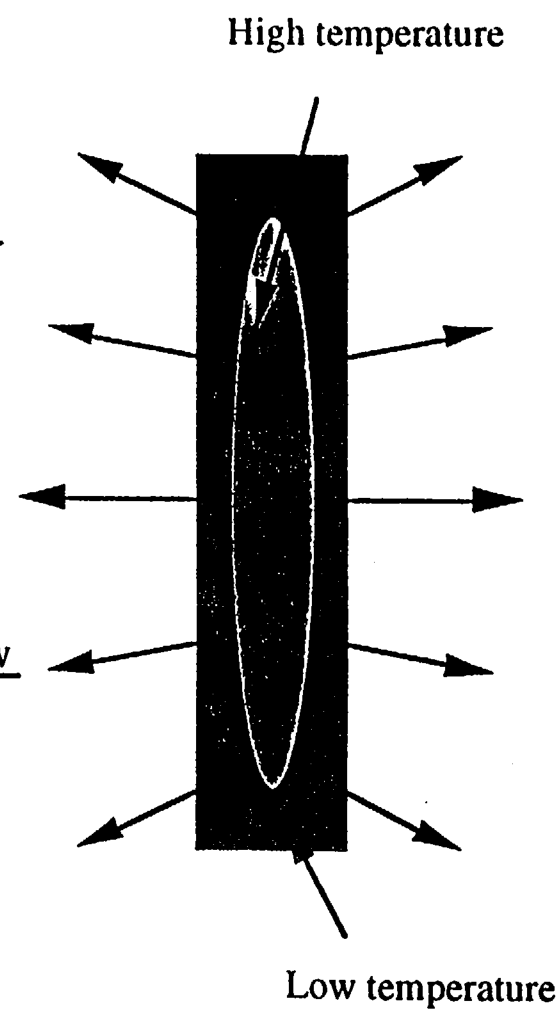
- TREAT Experiments
  - Longer heat up time allows energy to spread
  - Uniform temperature distribution
  - More energy leaves the plate
  - Able to deposit more energy
- JAERI Experiments
  - Shorter exposure time
  - Steeper temperature gradients

- JAERI experiments show that rate of energy deposition affects fuel performance

Dispersion model (cont)
Mechanism for accelerating the fuel plate

<u>Dispersion</u> - break up of fuel into small particles by acceleration induced hydrodynamic breakup

- **Thermal expansion of the solid** $$\alpha_s \frac{dT}{dt}$$
- **Expansion of material while changing phase from solid to liquid** $$\frac{\rho_l - \rho_s}{\rho_s} \frac{dx_{s-l}}{dt}$$
- **Thermal expansion of material while in liquid phase.** $$\alpha_L \frac{dT}{dt}$$
- **Expansion of material while changing phase from liquid to vapor** $$\frac{\rho_v - \rho_l}{\rho_v} \frac{dx_{l-v}}{dt}$$
- **Expansion of gases trapped in the plate (f.p., gas occupying voids)** $$\frac{1}{T} \frac{dT}{dt}$$



Dispersion model

- **Thermally driven expansion process**

- **Transient temperature change rate and phase change rate affects the expansion process**

- **Finite difference heat transfer model with melting and vaporization was coupled with thermally induced expansion models**

$$k\frac{d^2T}{dx^2}+q'''=\rho c_p\frac{dT}{dt} \qquad T\neq T_m \,\&\, T\neq T_v \qquad k\frac{d^2T}{dx^2}+q'''=\rho h_{sf}\frac{dx_{sf}}{dt} \qquad T=T_m$$

$$k\frac{d^2T}{dx^2}+q'''=\rho h_{fg}\frac{dx_{fg}}{dt} \qquad T=T_v$$

- **Strain rate is defined as the following:**

$$SR=(1-f)\left(\alpha_s\frac{dT}{dt}+\frac{\rho_l-\rho_s}{\rho_s}\frac{dx_{s-l}}{dt}+\alpha_L\frac{dT}{dt}+\frac{\rho_v-\rho_l}{\rho_v}\frac{dx_{l-v}}{dt}\right)+f\frac{1}{T}\frac{dT}{dt}$$

Dispersion model: sources of energy

- Heating caused by nuclear fissioning process

  Follows the shape of thermal neutron flux incident on the fuel plate.

  0.5 - 4.5 MJ/kg of fuel plate

  25 - 400 MW/kg of fuel plate

- Exothermal reaction between U3Si2 and aluminum.

  0.3 - 0.35 MJ/kg of U3Si2 depending on fuel volume fraction

  0 - 100 MW/kg depending on the temperature

  At aluminum melting temperature the reaction is slow.

  At U3Si2 melting temperature the reaction in fast.

- Chemical oxidation reaction between metal and water

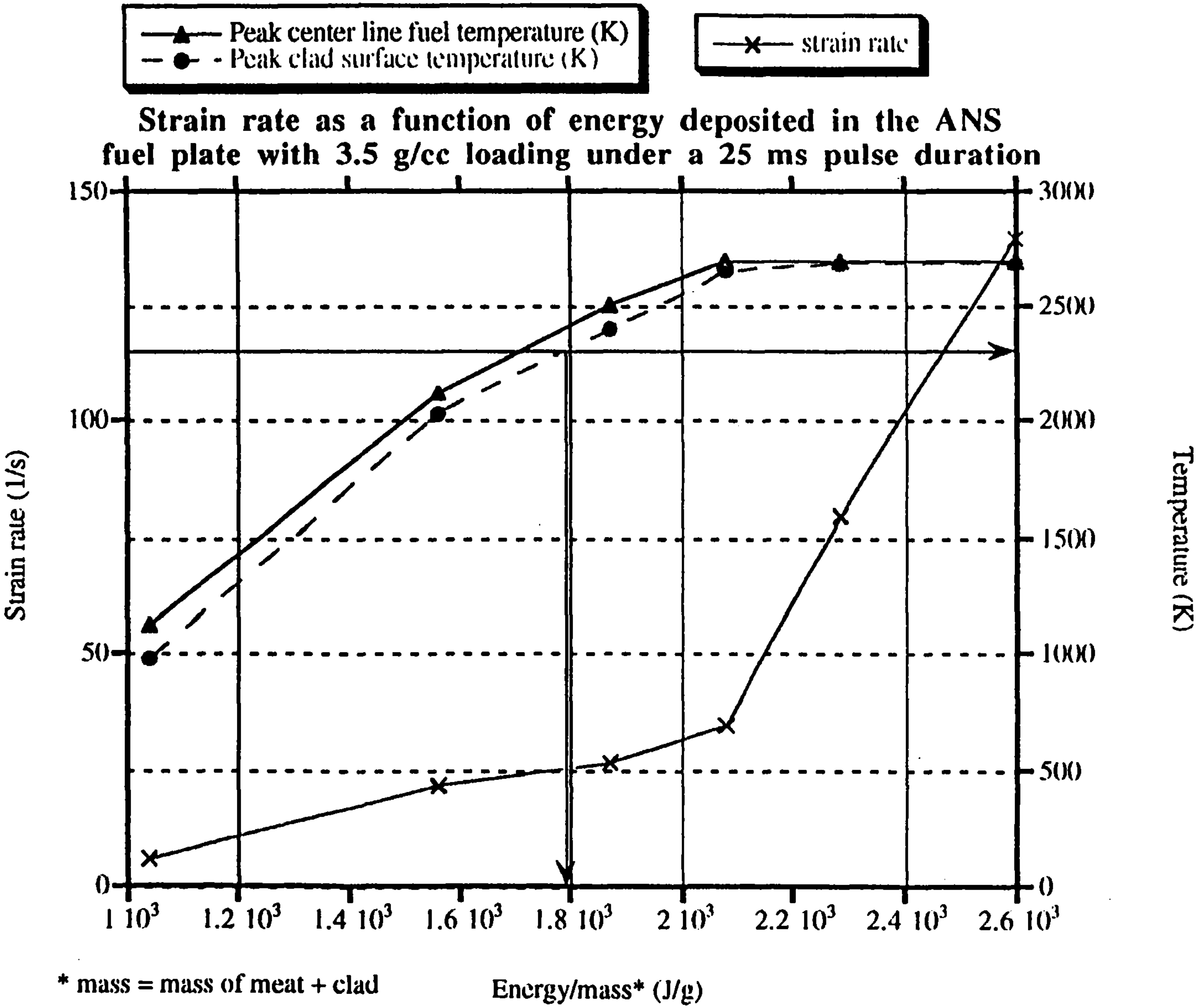
  18 MJ / kg of aluminum (0.18 MW/kg - 180 MW/kg)

  - Significant oxidation was reported for tests where surface temperature exceeded melting temperature of the oxide layer
  - Two mechanisms of aluminum oxidation reaction are :

    Vapor phase burning (vapor Al reacting with steam)

    Diffusion of steam though the liquid/solid oxide layer

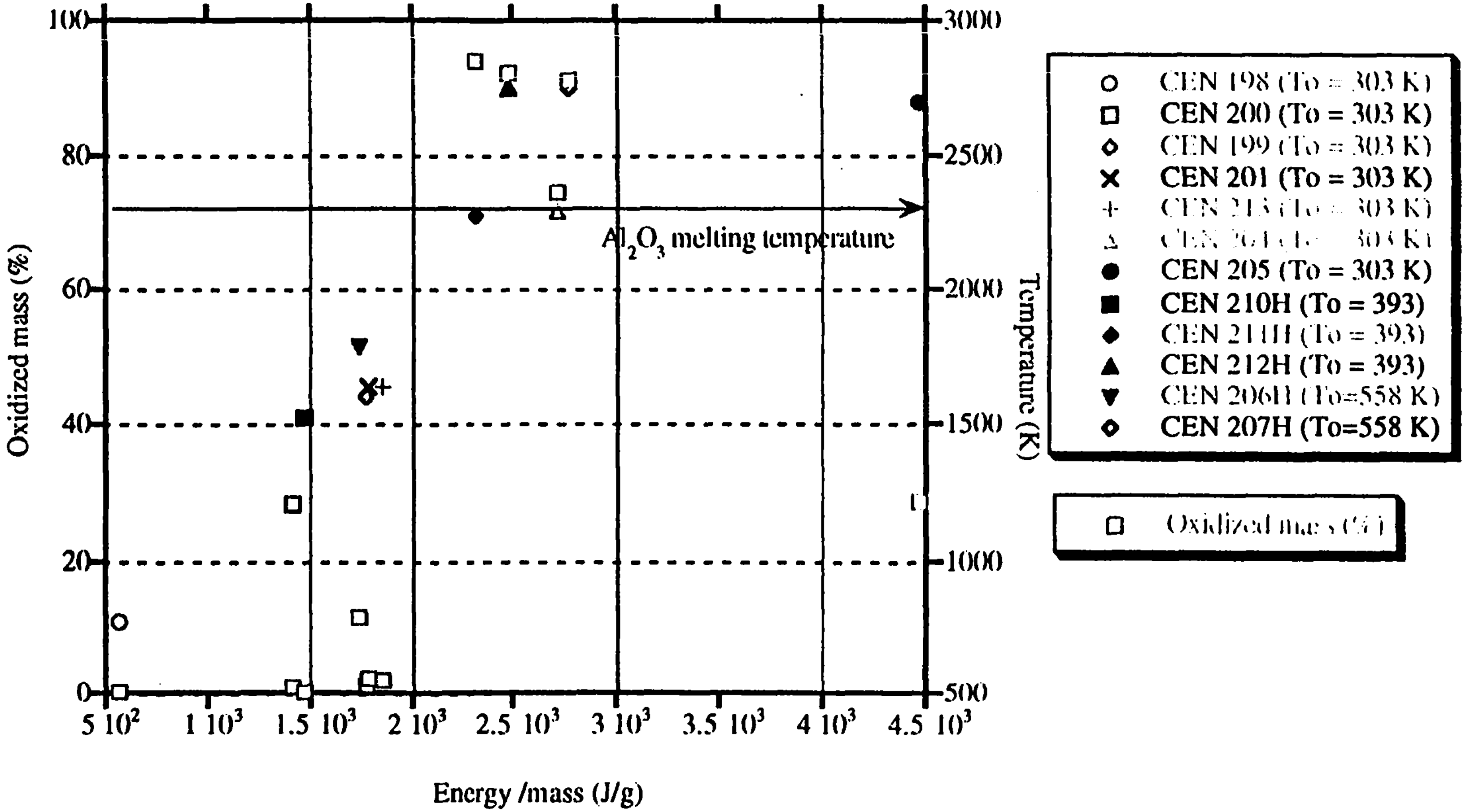
Dispersion model results (cont)



* mass = mass of meat + clad

Dispersion model results (cont)

Fraction of mass that oxidized compared to the peak clad temperatures reached for different energy depositions in HFIR fuel performed in the TREAT facility



SUMMARY & FUTURE WORK

o PRESENT STATUS OF ANS MINIPLATE SHIPMENTS & NSRR TESTS

- Seven NSRR tests have been conducted (heatup to cracking, bowing & melting; impact of TCs and enclosures investigated)
- Five additional plates have been shipped for testing in FY95

o MODELING & ANALYSES WORK AREAS AT ORNL (** for JMTR & ANS plates**)

- Multidimensional neutronic simulations are in good agreement with data and shall be refined in the future for evaluating transient energy deposition
- Heat transfer and material response simulations have given much useful information on impact of TCs, homogeneity, transient energy deposition, and for estimation of conditions necessary for dispersion and oxidation

o FUTURE WORK WILL INVESTIGATE FOLLOWING AREAS (1994 to 1998):

- Determination of threshold to cracking & evaluation of transient energy inserted
- Impact of higher energy deposition levels on thresholds for steam explosions
- Impact of preirradiation on damage thresholds
- Extension of modeling and analysis framework for reactor safety evaluations

**** Planning for future tests is being jointly conducted ****

XA04C1703

IGORR-IV

391

WORKSHOP ON R&D NEEDS

Klaus Böning

WORKSHOP ON R&D NEEDS

Klaus Böning

As in recent IGORR meetings, the goal of this workshop on R&D Needs was to bring together those groups of research reactor operators and designers, who need answers to particular R&D problems they have, and those groups of research reactor operators and designers, who plan to perform or have already performed R&D investigations on problems they have. As could be expected — and as the former IGORR meetings have demonstrated — there is a considerable range of overlap in the interests and needs of various groups in the same R&D topics.

During IGORR-I and IGORR-II, a matrix was set up — see Table 1 — connecting numerous R&D topics (vertical) and groups interested (horizontal). Results have been reported on some of these R&D topics during IGORR-II (italicized texts) while other problems still had to remain open — these have been labeled by 2a—2f in Table 1. Further R&D needs have been identified at IGORR-III: these are shown in Table 2, labeled by 3a—3g.

ORNL DWG 92M-8315R

R&D needs identified at IGORR-I and IGORR-II

	Topics	ANS	BERLIN	BNL	FRM-II	JAERI	JÜLICH	MAPLE	MIT	MURRI	ORPHEE	PETTEN	RISO
	Thermal-hydraulic tests and correlations	●		○	○	●				●			
	Corrosion tests and analytical models	●			○				○	○			
2a	Multidimensional kinetic analysis for small cores	○			●			●					
	Fuel plate fabrication	●			●								
	Fuel plate stability	●			●					○			
	Fuel irradiation	●			○	●	○			●			
2b	Burnable poison irradiation	●			●								
	Structural materials irradiation	●	○	●	○			●		●	●	●	
2c	Neutron guides irradiation	○			○	●							
2d	Cold source materials irradiation	○	○		○						●		●
2e	Cold source LN_2 test	●											
2f	Cold source LH_2-H_2O Reaction (H or D)	○		●	○	●							
	Instrumentation upgrading and digital control system	●		○					●	○	●	●	
	Man-machine interface	○			○				●				

↑ Still open at IGORR-II

● Results needed and work already underway or planned

○ Results needed but work not already planned

Note: Italicized text — results to be reported at IGORR-II

Table 1. (from IGORR-II Proceedings)

Further R&D needs identified at IGORR-III

New Open:	Topic	Notes
3a	Accident and safety analysis codes and benchmarks	IGORR members will be asked to supply and share information on the methods they use
3b	Thermophysical properties of D_2O liquid and vapor	Risoe has prepared a report on this, with 150 references and will publish a heavy water handbook in about six months. IGORR members will be informed.
3c	Chemical and other energy release from core melt events	Need better estimates of steam production for containment and design (Lee/AECL)
3d	Fission product release from a molten MTR core	
3e	Thermal conductivity of irradiated fuel meat	Analytical, or better still experimental, data are needed (West/ANS)
3f	Tests of cryogenic circulators for single-phase forced-convection cold source	
3g	Flow blockage tests	ANS will measure flow and heat transfer effects in the wake of a partial blockage and will use the results to benchmark a computational fluid dynamics model. Results will be reported.

Table 2. (from IGORR-III Proceedings)

WORKSHOP ON R&D NEEDS

Klaus Böning

During the present IGORR-IV Meeting, numerous "Research, Development, and Analysis Results" were reported on as outlined in Chapters IV and V of these Proceedings. Some of these reports were on more detailed work referring to R&D topics which had already been raised and discussed earlier (see Table 1), but some of them also gave answers to R&D topics which were still open at that time, namely numbers 2f, 3a, 3c, 3d, 3e, and 3g. Further, R&D results have become available which have not explicitly been reported on during the meeting, i.e., neutron guide irradiation tests from ORPHEE (no. 2c) and a "Heavy Water Handbook" from Riso (no. 3b). If one also cancels R&D needs nos. 2d and 2e, which were only identified by the ANS project, the termination of which was deeply regretted by the auditory, the only open R&D topics where no answers have been given so far were those with nos. 2a, 2b, and 3f. This conclusion does, of course, not mean that all the answers given were sufficient to really solve the corresponding R&D problems, so it may well be that some of these topics will come up once again in the future.

The IGORR-IV Workshop on R&D Needs, which is the subject of this summarizing paper, provided the opportunity for the participants to report on new R&D needs they have as well as on R&D work which was going on at their home institutions and which could be of interest to other groups. As a result, Tables 3 and 4 show a compilation of these presentations concerning R&D work needed or announced, respectively. Finally, Doug Selby (ORNL) presented a list of topics of R&D investigations which have been performed (at the time being more or less completely) for the ANS project and which could be finalized and the results written down with relatively small effort if there was interest from the IGORR group members. This list is shown in Table 5 where the column on the full right-hand side demonstrates that there was a strong interest in practically all of the R&D topics by relatively large numbers of groups.

In conclusion, the present Workshop on R&D Needs (Session VI of the Agenda) together with the Sessions IV and V on R&D Results have demonstrated once again that these activities indeed serve an important purpose of the IGORR meetings. The goal of bringing together research reactor operators and designers who have R&D needs, and those who produce R&D results, helps to save costs, improve the understanding of the ongoing processes, and so is beneficial to the safe and reliable operation of the facilities.

NEW R&D NEEDS IDENTIFIED AT IGORR-IV

	Affiliation/ Name	Topic	Comment
4a	AECL/Albert Lee	Requirements for the design of containment	Survey IGORR members on design basis for containment (confinement) — design basis events — external hazards — overpressure requirements, etc.
4b	ORNL/Doug Selby	Cold neutron nuclear data	— Sources of data • LD_2 • LH_2 — Benchmarks for cold source physics analysis • LD_2 • LH_2 — Cold source heat loads • methods of analysis • benchmarks
4c	TUM/ Klaus Böning	Thermal-hydraulic data (flow instability) for high cooling water velocities (≈17-18 m/s), but low system pressure (≤10 bars)	The recent data obtained from the ANS test loop is very very valuable, but they were all obtained for relatively high system pressures. The local TUV responsible for the assessment of the FRM-II is not fully convinced that the data obtained do not depend on pressure, although the state-of-the-art theory says they do not.
4d	MURR/ Charlie McKibben	Method to calculate decay time required after full power operation before the core is safe in air.	

Table 3

NEW R&D WORK ANNOUNCED AT IGORR-IV

	Affiliation/ Name	Topic	Organizations interested
4e	ORNL/Doug Selby	Cold neutron beam tube guides size and geometry optimization studies	TUM, AECL

Table 4

POTENTIAL R&D TOPICS FROM ANS CLOSEOUT OF R&D ACTIVITIES

ORNL/Doug Selby

	Topic	Organizations interested
4f	HANS-3 fuel capsule irradiation (in HFIR) evaluation	GA (TRIGA); Siemens; MURR; TUM
4g	Meat fabrication with spherical powder fuel	TUM
4h	Centering of fuel in plate fabrication	MURR; . . .
4i	Effect of flow blockage shape	KAERI; KFA; MURR; TUM; . . .
4j	Reduced ph effects on Al corrosion	All
4k	Final summation of our thermal hydraulic test program	All
4l	Final results from the HANSAL aluminum irradiation tests (include BNL analysis)	All
4m	Aluminum irradiation creep tests	KAERI; CERCA; . . .

Table 5

BUSINESS SESSION

Colin West chaired the Business Session of IGORR-IV. The first topic for discussion was IGORR-V. It was agreed by all that the IGORR organization should continue to hold meetings, and the principle of rotating the IGORR meetings among America, Europe, and Asia was reaffirmed; the next meeting will be held in the Fall of 1996 in Europe. Christian Desandre announced that he and Bernard Farnoux had discussed holding the IGORR-V meeting in France and volunteered to have it either at Grenoble or Cadarache. The IGORR Organizing Committee will be polled on a date and location for the meeting.

The second topic for discussion was the election of the IGORR Chairman. It was unanimously decided that Colin West will remain as Chairman of IGORR and that Kathy Rosenbalm will continue as the Technical Program Coordinator.

K. F. Rosenbalm

International Group on Research Reactors
(IGORR-IV)

May 23-25, 1995
Gatlinburg, TN, USA

Attendees

Argentina

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Allan Murray, Australian Nuclear Science and Technology Organization, c/o Embassy of Australia, 1601 Massachusetts Ave., NW, Washington, D.C. 20036 (Phone: 202-797-3042; Fax: 202-483-5156; e-mail: amurray@capcon.net)

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