Forschungszentrum Karlsruhe
Technik und Umwelt
Wissenschaftliche Berichte
FZKA 5993

Proceedings of the IEA-Technical Workshop on the IFMIF Test Facilities Karlsruhe, Germany July 7 – 9, 1997

IEA-Implementing Agreement for a Program of Research and Development on Fusion Materials

Compiled by
A. Möslang, R. Lindau
Institut für Materialforschung
Projekt Kernfusion
Association Forschungszentrum Karlsruhe/Euratom

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Contents

1. Summary

IFMIF-CDE Technical Workshop on Test Facilities
Karlsruhe, Germany, July 7-9, 1997

This meeting was the first Test Facility Workshop of the Conceptual Design Evaluation (CDE) phase within the International Fusion Materials Irradiation Facility (IFMIF) project. The main objectives were (i) to review the baseline design concept of the former Conceptual Design Activity (CDA) period, (ii) to present the recent progress made in the design and neutronics fields and to discuss the consequences, (iii) to establish and harmonize major R&D activities following international guidelines specified for the present CDE phase.

Because the activities of the users group will be subject of a related meeting on November 1, 1997 in Sendai Japan, this workshop was mainly focused on design and neutronics work.

1.1 Component design and development (tasks 1, 2 and 6):

Almost all devices components and structures of the Test Facilities can be designed and fabricated with today's technology using the experience and the standards of already existing accelerator and neutron irradiation facilities and therefore do not need urgent development efforts during the present CDE phase. However, because the majority of the users recommended the option to eliminate NaK cooling entirely, helium gas cooled alternatives for the high and medium flux regions have become major tasks.

At the beginning of the CDE phase, the CDA reference design of the high flux test module has been evaluated and modified in a first step according to recent flux/volume calculations of the neutronics group. Although a feasibility study based on initial thermohydraulic calculations has shown, that even for a peak heat deposition of about 40 W/cm^2 specimen temperatures down to 300°C can be achieved and controlled, the present design should be experimentally verified before it is suggested as the reference design for all temperatures from 300 -1000°C. A final decision on the replacement of the NaK concept by the He-gas concept can not be made before a high flux helium-cooled test module has been designed in detail, fabricated and tested. In order to experimentally verify the thermal-hydraulics, the fabricability and the remote handling of the He-gas concept for the high flux test region, a subsized test apparatus equipped with instrumented specimen capsules and rigs has been suggested during this workshop. Initial design studies done by FZK have shown, that the nuclear heating can be well simulated by this test apparatus using commercially available active ohmic heating elements integrated in the specimens and specimen capsules.

Because the CDA reference design considered in the medium flux position one VTA equipped with one test module, only one type of test could be performed in this flux regime at the same time. In order to increase the efficiency of the available irradiation volume and the flexibility in the test matrixes, two design concepts have been developed in the early stage of the CDE phase by FZK and JAERI that allows the simultaneous irradiation of two independent test modules in this flux region. The

JAERI concept has further shown that the somewhat low specimen package density of the Tritium breeding test modules developed during the CDA phase can be significantly increased without loosing the flexibility of adjusting 4 independent irradiation temperatures. However, a matter of concern might be the relative high number of individual ducts in the VTA2 necessary to supply the two independent test modules for in-situ experiments (creep-fatigue and tritium breeding tests). Because the IFMIF neutron spectrum is dominated by MeV neutrons and has a much smaller population at lower energies compared with fusion rector neutrons, the tritium production in the medium flux region of IFMIF would be governed by ^{7}Li(n,pt)α and not by ^{6}Li(n,t)α as in fusion reactors. It became obvious that the tritium production and the calculation of the related heat deposition in this test module needs further consideration during the CDE phase.

Finally the status of the universal robot and remote handling systems has been reviewed. Because these systems are of major importance for the reliability of nearly all systems and devices in the test cell including the Li-Target, further design studies during the CDE phase were suggested.

1.2 Neutronics (tasks 10-14):

A deuteron-Lithium reaction model has been implemented to characterise the neutron flux and engineering responses of the IFMIF neutron source. The uncertainty in the total neutron yield from this source contributes directly to uncertainties in the engineering responses and, therefore, the available high flux test volume. An improvement of the model is underway, including additional reaction components, with the primary goal of reducing the uncertainty in the total neutron yield. Another approach for improving the neutron source description would be made possible by the evaluation of D-Li cross-section data. This necessary activity was proposed for the CDE, but is still pending. Similarly, the evaluation of gamma-ray and by-product (e.g. Tritium, Be-7) generation in the target is still required.

Neutron transport cross-sections and engineering response data were evaluated within the FZK-INPE collaboration for Fe-56, Cr-52, V-51, O-16, Na-23, K-39, Si-28, C-12. Processed data files for these nuclides are available for use with MCNP. Similar evaluations were performed at JAERI for Na-23, Al -27, K-39, Ti-47, Ti-48, Ti-49, Ti-50, V-51, Cr-50, Cr-52, Cr-53, Cr-54, Mn-55, Fe-54, Fe-56, Fe-57, Fe-58, Cu-65, and Y-89, and the nuclear data have also been processed for use with MCNP. In addition, some evaluations for these and other IFMIF-relevant nuclides have been made available by Los Alamos National Laboratory. Nevertheless, further required evaluations, such as H-1, will be performed during CDE by FZK-INPE. An initial version of an intermediate energy activation cross-section file is being produced. All data have been and will continue to be produced in the standard ENDF-6 format.

MCNP calculations were made for all relevant nuclear responses throughout the test cell for NaK and He cooled high flux test modules on the basis of the newly available nuclear data and D-Li neutron source term. Detailed 3-D models including the target, the high, medium and low-flux test assemblies, and the complete test cell have been developed and applied in MCNP calculations. Calculations have verified that simplifications can be introduced to significantly reduce calculational requirements, but full calculations are necessary for specific design purposes.

The damage calculations for the high flux test region demonstrate that significant additional volume can be utilised by changes in the High Flux Test Module design, currently limited to 500 ml. The calculated neutron flux gradients are larger than previously assumed, and therefore, are an important consideration in the design of specimen orientation and loading patterns. Estimations show that additional high flux volume may be made available by reducing the distance between the lithium free-surface and the test rigs. Possible reductions could be found in the lithium, backwall and test module wall thickness and the backwall/module gap.

Comparisons of the damage calculations to fusion reactor analyses show that the IFMIF irradiation environment provides a suitable simulation of the fusion irradiation conditions in terms of the gas to DPA production ratios.

In addition to these results, a better characterization of the material damage is needed. This includes the investigation of PKA-spectra for elastic and non-elastic interactions and their impact on the calculations of the damage characteristics of IFMIF. A comparison of the IFMIF, ITER, and DEMO damage characteristics is necessary to demonstrate the adequacy of IFMIF for fusion irradiation simulations.

A short review of all shielding and activation analysis carried out at ENEA over the CDA phase has been presented, along with some recent improvements. Particularly, it has been shown that the present Test Facilities design allows to respect maximum allowable dose rates in accessible areas, either during operation and at shut-down. New results were also presented on: (i) HEBT and Radiation Isolation Area activation due to neutron backstreaming from test cell; and (ii) impact of neutron source uncertainties on activation and dose responses including comparisons between JAERI and ANL neutron source.

The sensitivity tests performed using a new set of 14 neutron activation reactions and SAND-II unfolding code show that the IFMIF flux energy spectrum can be measured within 10% (1-25 MeV) and 30% (25-50 MeV) provided that the present cross section uncertainties are reduced. For beam stability control miniaturized fission chambers are an attractive option as on-line monitors. However, recent results from neutronic calculations (task 10) predict a neutron flux in the test cell (10^{15} n/cm^2s) higher than available commercial types are designed for. The possibility of extending their operation to IFMIF environment has to be investigated.

1.3 Major goals for the post CDA phase:

1) Component design and development:
   - Validation of CDA design and specification of a subsized test apparatus for the helium cooled high flux test module
   - Calculation of thermal hydraulics and temperature distribution in specimens, capsules and rigs
   - Detailed design of a subsized test apparatus equipped with rigs and instrumented specimen capsules
   - Fabricability of specimens, specimen capsules and test module
   - Handling tests to demonstrate assembling and disassembling as well as initial thermal-hydraulic tests
   - Design of improved reference high flux test module

- Validation of present design and improvement of the test module for in-situ tritium release tests on breeders in the medium flux position
- Feasibility of VTA2 concept taking into account all ducts and helium gas tubes
- Design of improved reference VTA2 equipped with test modules for creep-fatigue and tritium release tests
- Evaluation and improvement of present design of the universal robot system and the remote handling equipment

2) Neutronics:
- Improvement of the D-Li nuclear model to reduce the neutron yield uncertainty.
- Evaluation and preparation of a D-Li cross-section data file.
- Assessment of γ-ray and by-products generated from the D-Li reaction.
- Evaluation and processing of further neutron cross-section data (E>20 MeV) as needed for the neutronics design calculations.
- Preparation of a first version of an Activation Cross-Section File (E>20 MeV).
- Re-evaluation of irradiation parameters for the high flux test region taking into account the recent results of the detailed neutronics calculations.
- Improved assessments for the medium flux region with breeder material samples.
- Qualification of the IFMIF displacement damage characteristics in the High Flux Test Module in comparison with ITER and DEMO irradiation conditions.
- Re-evaluation of the preliminary assessment of shielding and activation calculations as new nuclear data above 20 MeV will be available.
- Assessment of deuteron activation because of deuteron losses in the HEBT. However, to this purpose nuclear data for deuterons are still missing
- Accurate measurements of relevant reactions (e.g. ^{59}Co(n,xn) and ^{197}Au(n,xn))
- Qualification of chosen dosimetry package on existing high energy accelerator facilities, and use of cross section covariances for further sensitivity analysis tests.

2. Agenda

IFMIF-CDE Technical Workshop on Test Facilities
Karlsruhe, Germany, July 7-9, 1997

Monday, July 7

08:30	Pick up from hotel	
09:30	Welcome address	H.-D. Röhrig, K. Ehrlich
	Short review of preliminary CDE tasks	A. Möslang
10:30	Session on component design	
	- He-cooled high flux test module (task 1)	A. Möslang, G. Schmitz
	- Tritium release test module (task 2)	K. Noda, A. Möslang
	- Universal Robot system (task 6)	C. Antonucci
	- Short status of tasks 3-5 and 7-9	A. Möslang
12:30	Lunch	
14:00	Session on neutronics (tasks 10-14)	
	- Source term of D-Li reaction	P.H. Wilson
	- 20-50 MeV data evaluation	U. v. Möllendorff K. Noda
	- Shielding and activation analysis	S. Monti
	- On-line gamma monitors	B. Esposito
	- Re-evaluate irrad. Parameters	
	DPA-volume relations	E. Daum
	3D model MCNP calculations	M. Sokcic-Kostic
	Comparison to fusion reactors	U. Fischer
16:30	Excursion	
20:00	Dinner	

Tuesday, July 8

08:30	Pick up from hotel
09:00	Session on neutronics (tasks 10-14); continued

11:00	Session on He cooled high flux test module - design improvements based on recent engineering and neutronics calculations - agreement on improved reference design and materials - development of an out-of beam test module equipped with various instrumented rigs and specimen capsules - agreement on outline and specifications of that test module
12.30	Lunch
14:00	Session on medium flux test modules - discussion of recent design improvements and implications of the detailed MCNP neutronics calculations - agreement on improved reference design
15:30	Session on tasks reviews - Reviewing CDA tasks, main goals of CDE phase - responsibilities - Update present CDE tasks
17:00	Departure to hotel

Wednesday, July 9

08:30	Pick up from hotel
09:00	Session on tasks reviews (continued) - agreement on revised CDE strategy and tasks
11:00	Preparation of a summary
12.30	Lunch
14:00	Session on ICFRM-8 - discussion on submitted IFMIF papers - agreement on presented reference design and neutronics data - sharing of activities
16:00	Adjourn

3. Participants

IFMIF-CDE Technical Workshop on Test Facilities
Karlsruhe, Germany, July 7-9 1997

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4. Test Facilities/Users tasks for post CDA phase

4.1 General development requirements

In spite of the fact that the bulk of devices specified for the Test Facilities can be designed and fabricated with today's technology, various development efforts are necessary to establish a facility which combines overall structural integrity, high reliability, feasible and tested remote maintenance operations, and advanced safety standards. The development program identified for the Test Facilities has been established basically at the end of the Conceptual Design Activity (CDA) phase and can be subdivided into the three categories (i) development and engineering design of prototypical components, (ii) fabrication and testing of key devices that cannot rely on existing experience, and (iii) neutronics tasks:

Hardware Fabrication and Testing

- He cooled High Flux module full-scale prototype development and testing (Fabricability, remote handling and encapsulation, thermal-hydraulics, in-situ instrumentation and data acquisition
- Medium flux prototype development for in-situ breeding and creep-fatigue tests (SMA sealing under irradiation/thermal environments, demonstration of feasibility of one full-scale tritium release sub-test module, fabricability, thermal-hydraulics and temperature control)
- VTA Prototype Development (Fabrication of a less than full-scale VTA prototype, remote handling, demonstration of precise positioning, vacuum sealing)

Neutronics Tasks

- Nuclear data evaluation and data library preparation (20-50 MeV)
- Source Term of the D-Li reaction (re-evaluation of relevant irradiation parameters to support needs of the Users Group, neutron spectra, dpa- and gas production rates)
- Shielding and activation analyses (Thickness of walls for Access Cell, Hot Cells etc., gamma and total heat generation in test modules, decay heat evaluation for NaK and helium gas coolant)
- On-line neutron and gamma detectors, and data base (E_n < 50 MeV) for Dosimetry.

Meanwhile relevant user specific tasks have been added. Although the user tasks are basically not related to design requirements, they represent important activities in the present phase because of their outstanding relevance for the overall IFMIF performance. The development program that has been specified is structured into 9 component design tasks, 4 neutronics tasks, and 3 user tasks. Because the activities of the users group will be subject of a related meeting on November 1, 1997 in Sendai Japan, this workshop, held at FZK July 7-9 1997, was mainly focused on design and neutronics work.

It has been agreed at the beginning of 1997, that instead of a broad based engineering evaluation phase a smaller reviewing phase, named Conceptual Design Evaluation (CDE) and covering the period until end of 1998 should follow the CDA phase. This is why only a fraction of the necessary tasks originally specified for an engineering oriented evaluation phase can be handled during this CDE phase. The table gives a task overview and indicates whether there will be activities during the CDE phase.

	Component Design & Development	**Priority**[1]	**Tasks during CDE (1997-98)**
1	He cooled high-flux test module[2]	High	Yes
2	Tritium release test module[2]	High	Yes
3	Creep-fatigue test module	Medium	
4	VTA handling mock-up	Medium	
5	VIT system mock-up	Low	
6	Universal Robot System	Medium	Yes
7	Integral test cell design	Low	
8	Remote handling PIE	Medium	
9	Tritium Laboratory[3]	Medium	
	Neutronics		
10	Source term of D-Li reaction	High	Yes
11	20-50 MeV data evaluation	High	Yes
12	Shielding and activation analysis[3]	Medium	Yes
13	On-line gamma monitors	Medium	Yes
	User group		
14	Re-evaluate irrad. parameters	High	Yes
15	Evaluation of Li-target backwall	High	Yes
16	Small Specimen Test Technology	High	Yes

1) Priority for CDE phase, according to the Subcommittee Meeting, Brussels, Jan 30-31 1997.
2) Additional input from companies expected
3) If a general task like „Safety assessment of IFMIF" would be established, it might require input from here

In the following the tasks are specified in more detail, even if there will be no activities during the CDE phase, that is, in the period 1997-1998. Finally, it should be emphasized that the user specific tasks strongly interact with tasks 1 and 2 (specimen geometry and test matrix) as well as with tasks 10-12 (evaluation of irradiation parameters).

4.2 Task Action Sheets

Task 1 Component Design

Title: Helium-Cooled High Flux Module

Description and justification:

Providing a test module for specimen temperatures up to 1000 °C is one of the most challenging design requirements for the IFMIF Test Facilities. The successful development of a He-cooled test module would provide for any material a broad temperature window as well as flexible and very safe operating conditions. It is necessary to demonstrate (1) fabricability of specimen encapsulation, helium gas pipes, active ohmic heating elements, and rigs, (2) remote handling of all assembly and disassembly procedures, (3) thermal hydraulics under different loading conditions, (4) structural integrity, as well as (5) proper function of the instrumentation equipment. All these requirements have to be demonstrated under relevant conditions on a full scale high flux test module assembled with an adequate number of completely instrumented rigs.

Contributors:

Organization	Principal Investigator
FZK	A. Moeslang, G. Schmitz
ORNL	J. Haines

Main responsibility: FZK

Milestones:

1) Validation of present design	July 1997
2) Prototype development and testing of subsized test module	
- Define test apparatus	Oct. 1997
- Calculate thermal hydraulic and temperature distributions in specimens, capsules and rigs	Sept. 1997
- Detailed design of capsules and actively heated specimens	Nov. 1997
- Detailed design of instrumented rigs and dummy rigs	Nov. 1997
- Detailed design of simple test module with flow paths	Nov. 1997
- Fabricate test specimens	Dec. 1997
- Fabricate capsules and rigs	March 1998
- Fabricate test module	June 1998
- Handling and initial thermal-hydraulic tests	Dec 1998
3) Design of reference test module taking into account new neutronics results	Dec 1998
4) Documentation	Dec 1998

Task 2 Component Design

Title: Tritium release test module

Description and justification:

The availability of suitable tritium breeder materials that has been tested and developed under real loading conditions is a major requirement for a fusion reactor. Tritium release experiments using several different types of breeding material candidates are planned for the medium flux region of the Test Cell. To function properly, the temperature of the tritium-release test specimens must be controlled at a prescribed value, while the temperature of the structural material should be maintained as low as possible (ambient temperature) to avoid tritium permeation and large degradation of mechanical integrity of the structural materials due to irradiation. To enable this and to perform the tritium gas release experiment, pipes, six or more in number, with a diameter of about 1 to 2 mm will be attached to a capsule. The capsules are further complicated by the need to easily connect and disconnect these pipes for re-encapsulation. One full scale sub-test module includes four kinds of capsules for disk, pellet, and pebble shaped specimens and compatibility test specimens. The design of the sub-test module is carried out based on evaluation of the thermal behavior of the design concept and consideration of degradation of capsules/sub-test module due to the irradiation and of swelling of the irradiated breeder materials.

Contributors:

Organization	Principal Investigator
JAERI	S. Jitsukawa, K. Noda, S. Konishi,...
ENEA	C. Antonucci
FZK	A. Möslang

Main responsibility: JAERI

Milestones:

- Validation of present design	Sept 1997
- Design study of specimen temperature control	Dec. 1997
- Design study for easy attachment/de-attachment to VTA	Dec. 1997
- Design of one full scale sub-test module	July 1998
- Feasibility of VTA2 concept with two test modules	July.1998
- Design integration Concept to combine VTA2 with 2 test modules (Creep-fatigue and Tritium release test module)	Dec. 1998

Task 3 Component Design

Title: Prototypic in-situ creep fatigue testing device development

Description and justification:

No work planned during CDE phase; a sufficiently advanced design exists already

Task 4 Component Design

Title: VTA handling mock-up

Description and justification

To function properly the Vertical Test Assemblies (VTAs) must (1) be positioned to ensure that the test modules are accurately aligned (~ ± 1 mm) relative to the neutron source, (2) be adequately sealed to ensure that a pressure of 0.1 Pa can be maintained in the Test Cell, (3) be easily removed and replaced with the remote handling equipment available in the Access Cell, and (4) provide adequate shielding for the Access Cell. Satisfying the fourth requirement leads to a VTA configuration that includes about 2 tons of concrete shielding arranged in a stair-step fashion. Satisfying the other three requirements with this heavy and awkward configuration requires that a VTA handling and development effort be undertaken to demonstrate the feasibility of the concept and to develop handling approaches.

A less than full-scale mock-up of the High Flux VTA (VTA-1) will be fabricated and tested. The mock-up will include a small vacuum chamber and vacuum pumping system. The connection between the VTA and the small vacuum chamber will be configured to simulate the Test Cell/VTA vacuum seal interface. Handling and alignment tests will be performed to demonstrate and develop VTA removal, replacement, and alignment operations and to identify improvements in the VTA design that facilitate these operations. Vacuum leak tests will also be performed to demonstrate that the sealing concept performs adequately .

Contributors:

Organization	Principal Investigator
ORNL	J. Haines
TBD	

Milestones:

- Validation of present design
- Design VTA and seal configuration mock-up, including positioning, vacuum sealing and remote handling
- Fabrication of Mock-up
- Handling and alignment tests
- Documentation

No work planned during 1997

Task 5 Component Design

Title: Vertical Irradiation tube (VIT) system prototype development

Description and justification:

No work planned during CDE phase

Task 6 Component Design

Title: Universal robot system

Description and justification

The Universal Robot (UR) System is the main device in the Access Cell used for:

- routine VTA removal and reloading operations in the Test Cells;
- removal and exact insertion of shield plugs and removable Test Cell covers;
- other maintenance operations in the Access Cell and the Test Cells, including maintenance of the Li-target as well as other components located in the Test Cell, during normal and off-normal operations.

The Universal Robot System is required to handle very heavy loads (up to 25 tons) with high precision (±1 mm), and will be equipped with various multiple-purpose modular articulated robot systems. A Vision System able to replace human vision and acquire and process images will be provided as part of the UR System. Considering the shapes, loads and precision required, the ability of the proposed remote handling equipment and the associated Vision System has to be demonstrated as part of the CDE development efforts. This system is of outstanding importance for the reliability of the Test Cell. The experimental activities will validate the Vision System concept and verify the feasibility of all main remote handling procedures. In particular, the activities to be done should include the following two items:

(i) analysis of RH procedures, computer simulations, and, on such basis, design of a suitable RH test program (mainly 1997),

(ii) conceptual evaluation of the most critical features like Vision System, bag/gripper and shutter (mainly 1998).

Contributors:

Organization	Principal Investigator
ENEA	C. Antonucci + other Persons
TBD	

Milestones:

- Evaluation of present design	July 1997
- Identification and simulation of critical RH operation	Dec. 1997
- Specification and design of remote handling test program	Dec. 1997
- Feasibility and fabricability study	Oct. 1998
- Specification and design of bag/gripper and shutter	Dec. 1998
- Documentation	Dec. 1998

Task 7 Component Design

Title: Test Cell design concept development

Description and justification:

No work planned during CDE phase

Task 8 Component Design

Title: Remote handling equipment for PIE cells

Preliminary description and justification

In the interest of minimizing radiation exposures for IFMIF personnel during post-irradiation examination, it is worthwhile to examine possible improvements in conventional hot cell/hot laboratory techniques for handling the irradiated specimens. This is particularly true for procedures such as the preparation of TEM specimens which usually requires close-contact handling of the irradiated specimen by personnel in a shielded glove box. Furthermore, it would be useful to improve the precision for in-cell handling operations of the IFMIF miniaturized specimens compared to present-day hot cell standards. Miniaturized specimens from fission reactor experiments handled in conventional hot cell facilities are occasionally damaged due to the lack of precision of the hot cell equipment. In addition, installation of precision specimen handling equipment in the hot cells that can be computer-controlled would increase the throughput speed for specimen testing. For these reasons, it is concluded that the remotely operated precision equipment used to handle irradiated specimens in IFMIF needs to be developed. In related areas, remote loading of the large number of specimens in the specimen packets and remote welding to seal these packets should be studied to optimize the experimental techniques. Techniques utilized in previous hot cell encapsulation of miniaturized specimens and capsule welding (e.g., the HFIR-MFE-200J and -400J capsules at ORNL,) should be incorporated into the IFMIF procedures, as appropriate.

A design concept for remotely operated tweezers equipped with CCD cameras for the handling of miniaturized specimens including the TEM specimens will be developed; a prototypic remotely operated tweezers and micromanipulator for handling miniaturized specimens irradiated in mixed spectrum reactors is under development. Support equipment for welding and the laser welding machine that is planned to be used for sealing specimen packets will be designed and fabricated. Tests of the prototypic rigs will be performed using the micromanipulator.

Contributors:

Organization	Principal Investigator
JAERI	S. Jitsukawa, K. Noda
ORNL	S. Zinkle

Milestones:

- Validation of present design
- Design of the 2D position recognition system
- Design of remotely operated tweezers for PIE
- Development of the 2D position recognition system
- Development of remotely operated tweezers for PIE

No work planned during 1997

Task 9 Component Design

Title: Engineering design for tritium Laboratory and Tritium Processing

Description and justification:

No work planned during 1997

Task 10 Neutronics

Title: Source term of D-Li reaction

Description and justification:

The D-Li neutron source term is of primary importance for calculating the neutron flux, the spectrum and the nuclear responses in the IFMIF Test Cell. Currently there is an uncertainty band of 20 % in the total neutron yield that is directly transmitted into the neutronic responses calculated for the test module. The uncertainty has to be reduced to a tolerable level for the Engineering Design Evaluation phase of IFMIF. In addition to neutrons, the D-Li reaction generates gamma radiation and other by-products like tritium that accumulate during operation in the lithium system and must be removed. Currently, the necessary cross-section data are lacking to assess the generation of these products.

To assess and reduce the uncertainties in the neutron yield of the D-Li reaction state-of-the-art theoretical tools will be applied to analyze the D-Li interaction processes. An improved nuclear model will be developed to better simulate the D-Li reaction and the parameters of the nuclear model will be adjusted to available experimental data. New results are expected from a thick lithium target experiment planned to be performed at the Karlsruhe cyclotron. Finally, an evaluated set of D-Li cross section data will be produced for deuteron energies from 0 to 50 MeV.

To assess the g-ray and by-products generated from the D-Li reaction, a data file will be prepared that is suitable for γ-ray transport. Production cross-section data for by-products of the D-Li reaction (e. g. Be-9, tritium) will be evaluated.

Contributors:

Organization	Principal Investigator
FZK	U. Fischer, P. Wilson
ANL	I. Gomes
JAERI	Y. Oyama ,

Main responsibility: FZK

Milestones:

- Uncertainty assessment of D-Li neutron yield	June 1997
- Improved D-Li nuclear model	June 1998
- D-Li cross-section data set and file preparation	Dec. 1998
- Assessment of γ-ray and by-products	May 1998
- Reduce yield uncertainty	Dec. 1998

Task 11 Neutronics

Title: 20-50 MeV data evaluation

Description and justification:

In the IFMIF Test Cell there will be a considerable amount of neutrons produced above 20 MeV. Neutron nuclear data files for energies above 20 MeV are lacking. They need to be developed and processed for use with the calculational tools available for IFMIF design analyses. This includes neutron cross section data for collided spectrum calculations and important nuclear responses like dpa, nuclear heating, gas production , and others.

During the IFMIF CDA phase, data evaluation activities were launched to extend the energy range of the nuclear data files to energies up to 50 MeV. So far, data files for only a few nuclides have been prepared, both as basic data files in the ENDF-6 format and as processed files for use with the MCNP Monte Carlo transport code.

It is planned to continue this activity during the CDE phase with the aim of producing a full IFMIF nuclear data file including all the nuclides and data types of interest to IFMIF neutronics analyses. During the CDE phase, the evaluation will focus on the most relevant nuclides for IFMIF. This includes the processing of the evaluated nuclear data into a pointwise MCNP file and a multi-group cross-section set for discrete ordinate codes. The data evaluation and processing effort will have to be continued after the CDE phase to meet the final goal of making available a full data library for IFMIF.

Contributors:

Organization	Principal Investigator
FZK	U. Fischer, U. v. Möllendorff
JAERI	Y. Oyama

Main responsibility: FZK

Milestones:

- Data evaluation for major nuclides	Jan. 97 - Dec. 98
- Processing of data files	Jan. 97 - Dec. 98
- Evaluation of Tritium production and heat generation in breeder materials (relevance of IFMIF spectrum)	July 1998

Task 12 Neutronics

Title: Shielding and activation analysis

Description and justification:

During the CDE phase, more accurate nuclear data and D-Li neutron source will become available that allows more reliable neutronics calculations. As a consequence, a re-evaluation of the preliminary assessment of shielding and activation calculations will be necessary. This task is also needed to maintain the computational tools and the results in consonance with the newly released input data and the IFMIF layout developments.

Detailed 3d-models of the Test Cell Facilities will be simulated and applied in the MCNP shielding calculations as soon as more complete data libraries including 20-50 MeV neutron energies become available.

It was underlined that one of the most important issue should be the assessment of deuteron activation because of deuteron losses in the HEBT. However, to this purpose nuclear data for deuterons are still missing.

Contributors:

Organization	Principal Investigator
ENEA	S. Monti, L. Casalini
US	TBD

Main responsibility: ENEA

Milestones:

- Final report Dec. 1998

Task 13 Neutronics

Title: Activation Foils and Real-Time Neutron/Gamma Detectors

Description and justification:

The importance of the task is to provide reliable methods to measure the gamma/neutron field in the test cell and test assembly.

a) Activation Foils. This is a dosimetric method largely used in fission reactors. Its application to IFMIF requires to improve the available dosimetry database for high energy neutrons. A review of dosimetric cross sections with their uncertainties (as available in published literature or existing databases) for each one of the selected high priority reactions (see IFMIF CDA Final Report) will be performed. The reactions and energy ranges for which cross sections measurements are necessary will be pointed out. It is important to note that a relatively low number of well-measured dosimetry cross sections can guarantee an accurate measurement of the actual neutron field characteristics (fluence and energy spectrum) in the hostile environment of test cell/test assembly (high temperature and radiation level).

b) Real-Time Neutron/Gamma Detectors. The option of using real-time detectors for spatial and time-resolved flux monitoring will be examined by means of a survey of the present status of in-core reactor monitoring instrumentation (microfission chambers and self-powered detectors) in EU/US nuclear sites. An assessment of these detectors for their application to IFMIF (where sensitivity to neutrons up to 50 MeV is required) will be provided.

Contributors:

Organization	Principal Investigator
ENEA	B. Esposito, S. Monti

Milestones:

October 1997: Review of available data for high priority cross sections; list of cross sections to be accurately measured.

December 1998: Assessment of applicability of existing real-time in-core reactor neutron/gamma monitoring systems to IFMIF.

Task14 User group

Title: Re-evaluation of irradiation parameters

Description and justification:

During the CDE phase, a more accurate nuclear data will become available that allows more reliable neutronics calculations. As a consequence, a re-evaluation of the preliminary assessment of nuclear responses for the irradiation modules will be necessary. This task is also needed as the design develops and more detailed description of the test modules and the whole facility are available. Also an assessment of the tritium production and inventory in the facility is to be performed.

MCNP based calculations will be performed for the dpa, gas production, nuclear heat deposition, and neutron/gamma flux distributions in the test cell on the basis of the newly available nuclear data and the D-Li neutron source term. Detailed 3d-models of the target, the test modules and the test cell itself will be developed and applied in the MCNP neutronics calculations .
In addition to the DPA calculations, a better characterisation of the material damage is needed. This includes the investigation of PKA-spectra for elastic and non-elastic interactions and their impact on the calculations of the damage characteristics of IFMIF. A comparison of the IFMIF, ITER, and DEMO damage characteristics is necessary to demonstrate the adequacy of IFMIF for fusion irradiation simulations.

Contributors:

Organization	Principal Investigator
FZK	E. Daum, K. Ehrlich, U. Fischer, P. Wilson, M. Sokcic-Kostic ....
US	NN
JAERI	Y. Oyama

Main responsibility: FZK

Milestones:

- Processing of elastic and non-elastic PKA-spectra for ^{56}Fe (Aug. 97)
- Evaluation and qualification of damage characteristics for IFMIF, ITER and DEMO (Dec. 97)
- Extended damage qualification for other important elements (e.g. Cr, V) (Dec. 98)
- Development of test matrices for the helium gas cooled high flux test module (Dec 98)

Task 15 **User Group and/or Li-Target**

Title: **Evaluation of Radiation resistance of Li-Target backwall materials**

Description and justification:

The solid Li-Target backwall has to withstand more than 55 dpa/fpy, it accumulates at the same time -depending at the alloy - about 500 appm Helium and more than 2000 appm hydrogen, and is the device with the highest neutron and gamma wall load of the overall facility. The structural integrity of that backwall governs besides safety aspects the availability of the IFMIF test facilities and is therefore a matter of concern.

Specialists of radiation damage of structural materials will be nominated as members of a working group to evaluate the lifetime of the candidate materials for the Li target backwall.

Contributors:

Organization	Principal Investigator
ORNL	S. Zinkle,....
FZJ	H. Ullmaier, P. Jung
FZK	E. Diegele, A. Möslang,
JAERI	S. Jitsukawa, K. Noda, K. Shiba
PSI	F. Paschould
ENEA	G. Benamati

Milestones:

- Activity planning	June 1997
- Collection of materials data currently available	Oct. 1997
- Establishment of materials database for evaluation of the lifetime	Dec 1997
- Stress-strain analyses of the Li-backwall geometries under real loading conditions.	June 1998
- Suitable experiments to the low temperature irradiation, He and H embrittlement of potential materials (e.g. f/m steel, V-4Cr-4Ti)	Dec. 1998
- Assessment of the structural integrity including the lifetime	Dec. 1998

Task 16 User Group

Title: Small Specimen Test Technology

Description and justification:

Further activities in that field are essential in order to fully utilize the available irradiation volume and to qualify all miniaturized specimens. It is assumed that also in future the SSTT activities will be mainly organized within the frame of IEA-workshops and also be supported within the respective materials programs.

Contributors:

Organization	Principal Investigator
Univ. of California, Santa Barbara	G.E. Lucas, G.R. Odette
ORNL	S. Zinkle,...
KFA	P. Jung, H. Ullmaier
FZK	J. Aktaa, A. Moeslang,
JAERI	A. Hishinuma, S. Jitsukawa, K. Noda,....
Japanese Universities	TBD

Milestones:

Work is ongoing and coordinated mainly within SSTT community.

Major milestone: ICRFM-8, Sendai Oct. 1997

IFMIF-CDE Test Facility Workshop, Karlsruhe July 7-9 1997

Design Updates for Test Cell and Helium Gas Cooled Test Modules

R. Lindau and A. Möslang

The reference design done during the CDA phase for major components of the Test Cell is presently evaluated mainly

(i) to implement additional requirements of the users community,

(ii) to take into account recent flux/volume and heat deposition calculations from the neutronics group,

(iii) to re-calculate in more detail thermal-hydraulics parameters. and

(iv) to prove fabricability, remote handling operations and instrumentation.

Test Cell

Various design changes have been made. These updates are integrated in the figures 1-5 that shows the Integral Test arrangement. The most important improvement is the Vertical Test Assembly in the medium flux position (VTA2) equipped with two different in-situ test modules.

Helium gas cooled high flux test module

Major development requirement: Elimination of the NaK cooled high flux test module by helium gas cooled one.

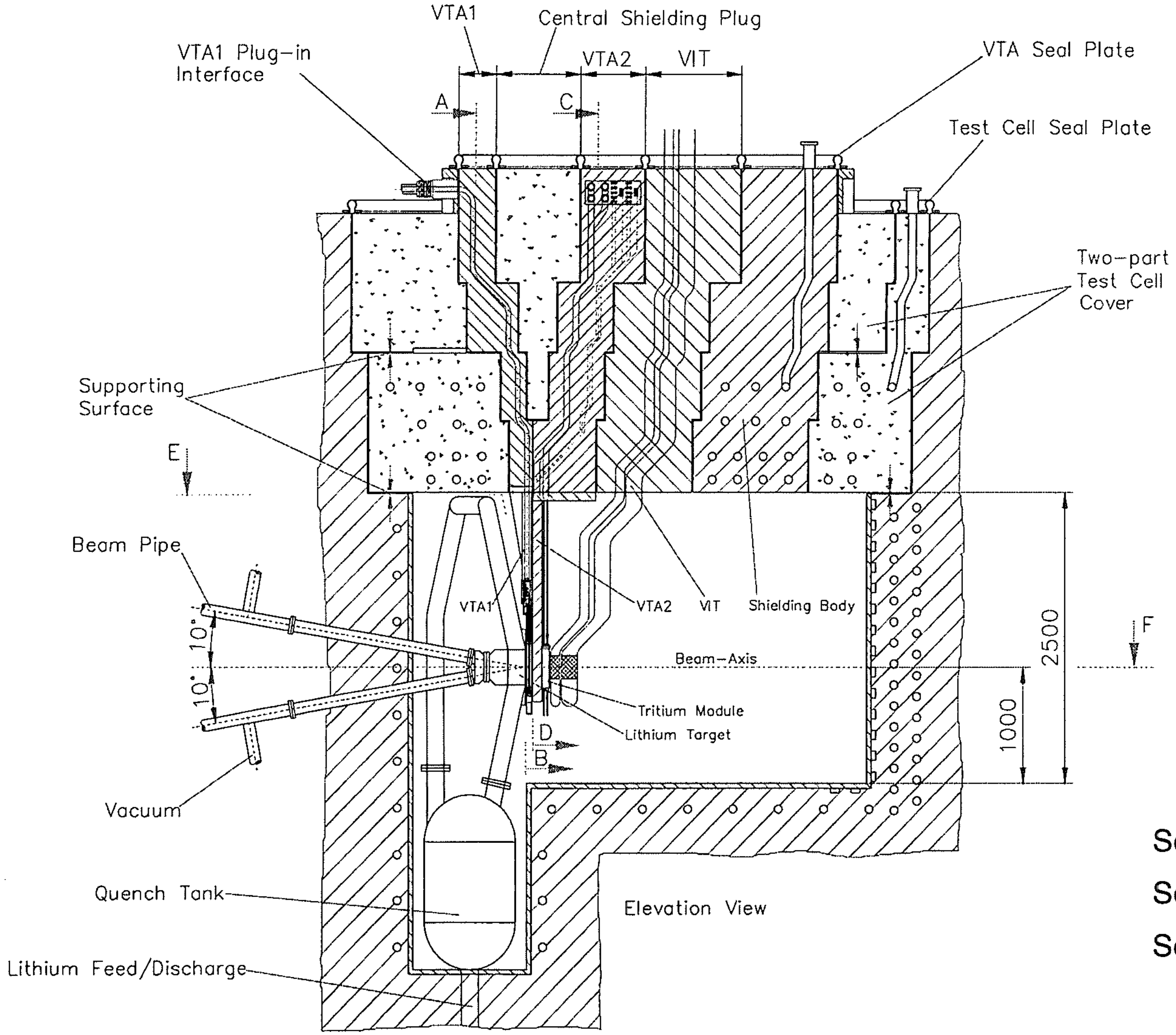
Since the end of the CDA phase, the initial design of the helium gas cooled high flux test module was improved significantly. The gas coolant ducts are now integrated part of the test apparatus to increase the coolant efficiency and the overall mechanical stability. Another design guideline was to minimize the volume and thus the heat deposition of the structural materials. Figs 6-10 show the present status of the reference design. Initial design studies of a sub-sized test apparatus are described in the next section.

In order to achieve a high mechanical integrity of all individual components, it is foreseen presently to use as structural material (e.g. specimen capsules, rigs, test module) the same type as for the specimens itself. Because the high flux test module is arranged immediately behind the Li-Target, the neutron and gamma flux on its front side is practically comparable with that one of the backplate. Also the irradiation temperatures are in the same range (below about 300 °C) Therefore, for both components many material questions can be basically treated in the same manner.

Helium gas cooled medium flux test modules

The reference design considers one VTA equipped with only one test module in the medium-flux region. In order to increase the efficiency of the available irradiation volume and the flexibility in the test matrixes, it would be desirable to simultaneously irradiate two independent test modules in this flux region between 1 and 20 dpa/fpy.

Therefore, it was verified both by FZK and JAERI in the early stage of the CDE phase whether VTA2 can be simultaneously equipped with two test modules.



Section E-F: Fig. 3

Section A-B: Fig. 6

Section C-D: Fig. 12

Fig. 1: Elevation view of the complete test cell arrangement (all dimensions in mm). In this version the test cell walls are cooled by gas coolant passages inside the concrete.

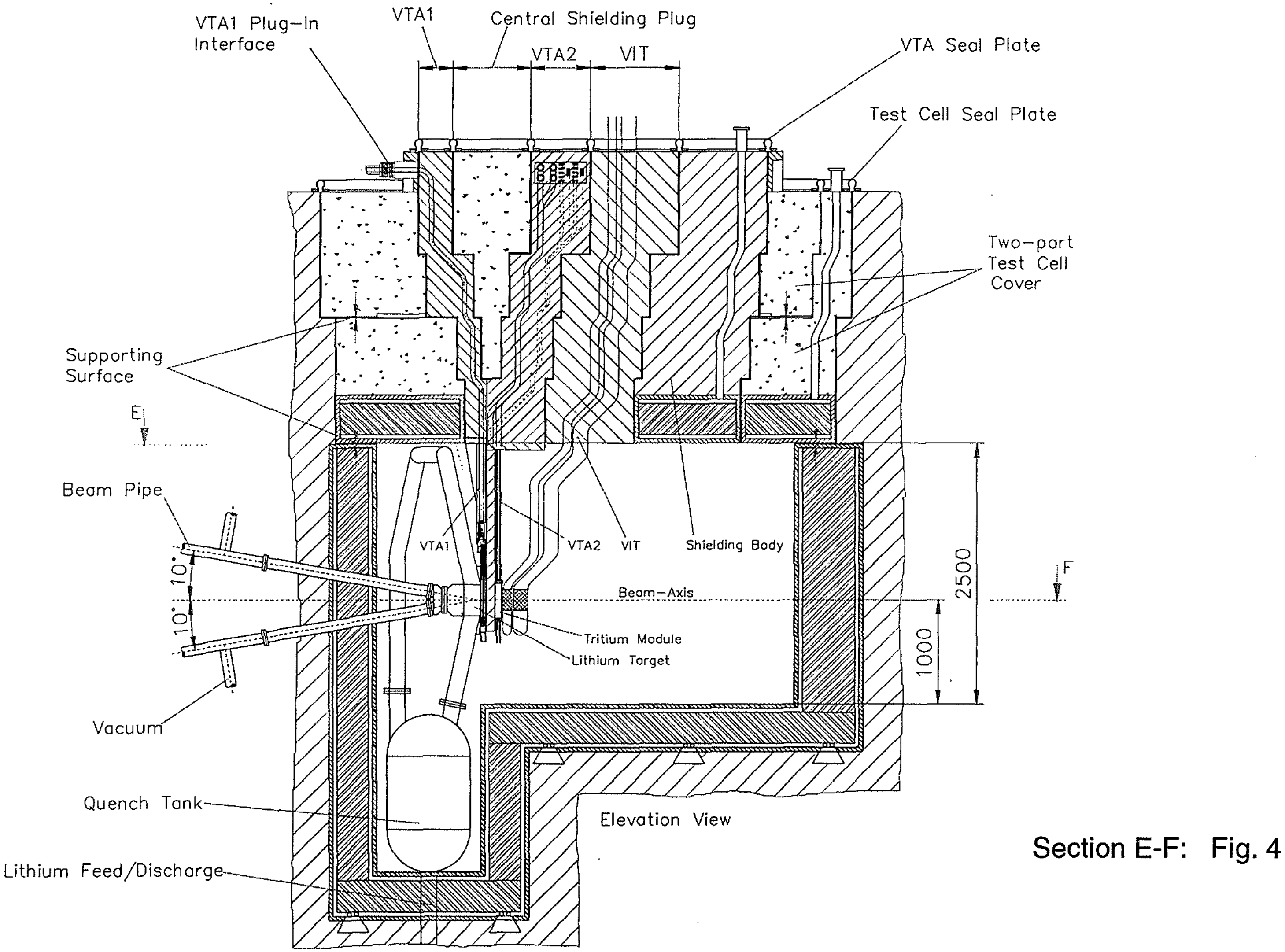


Fig. 2: Elevation view of the complete test cell arrangement (all dimensions in mm). Design version of the test cell with a stainless steel heat shield.

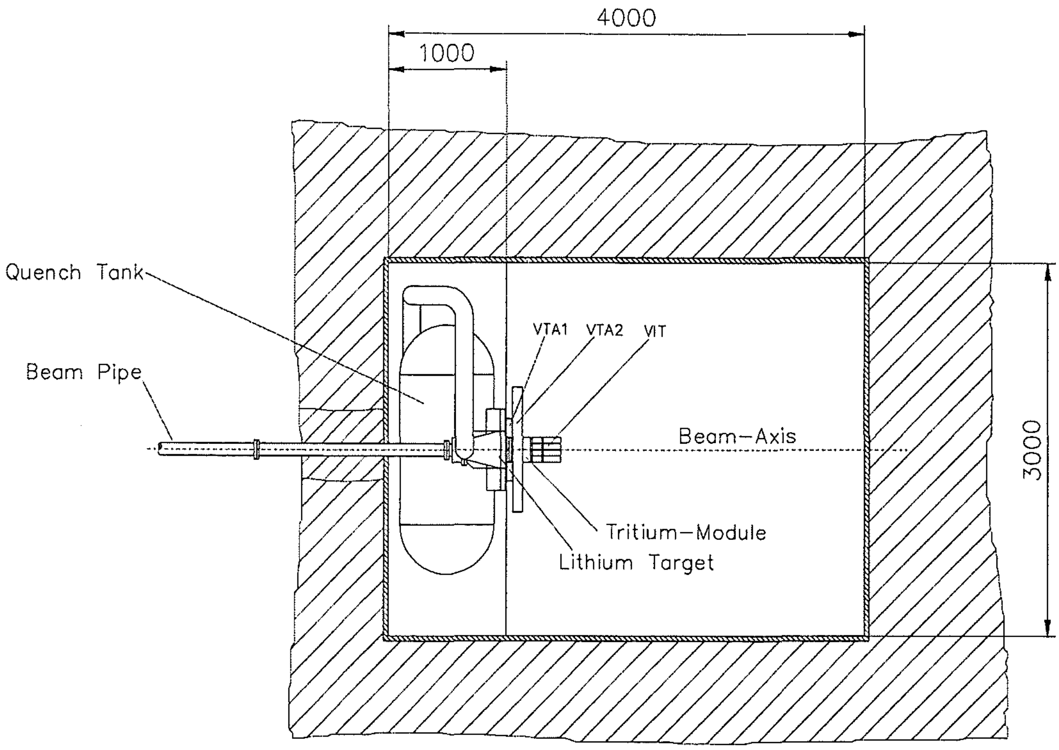


Fig. 3: Plan view of the test cell with lifted removable cover; gas cooled concrete walls (all dimensions in mm).

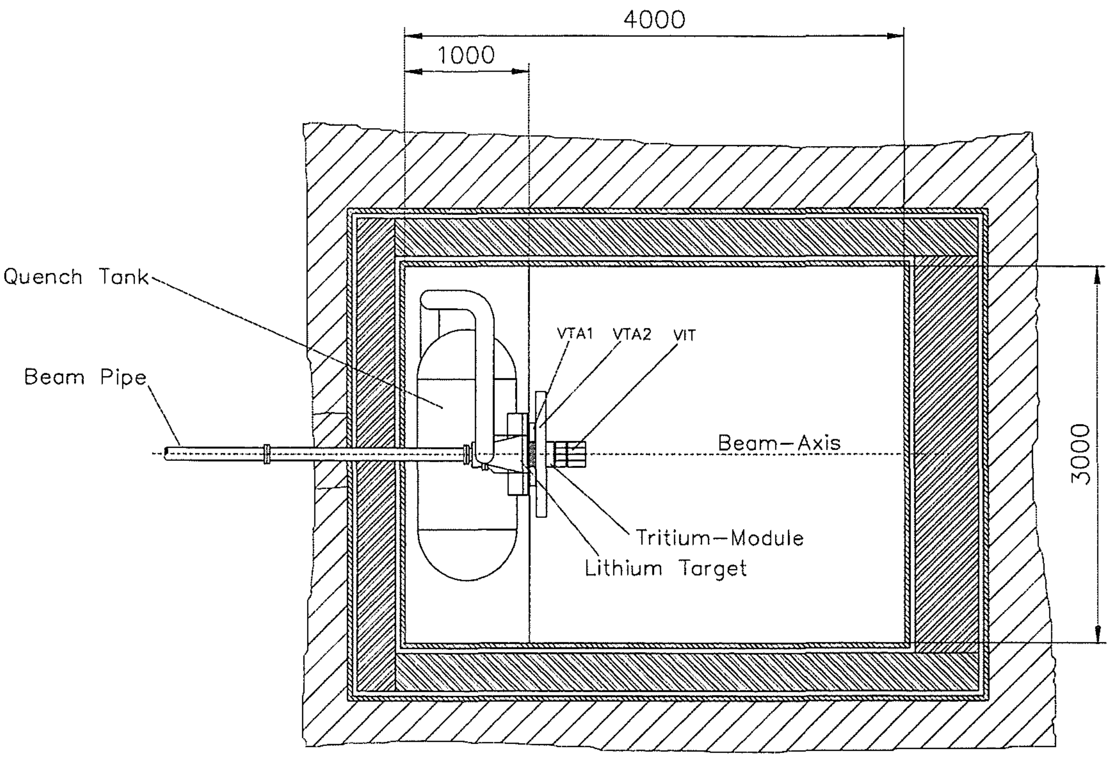


Fig. 4: Plan view of the test cell with lifted removable cover; stainless steel heat shield (all dimensions in mm).

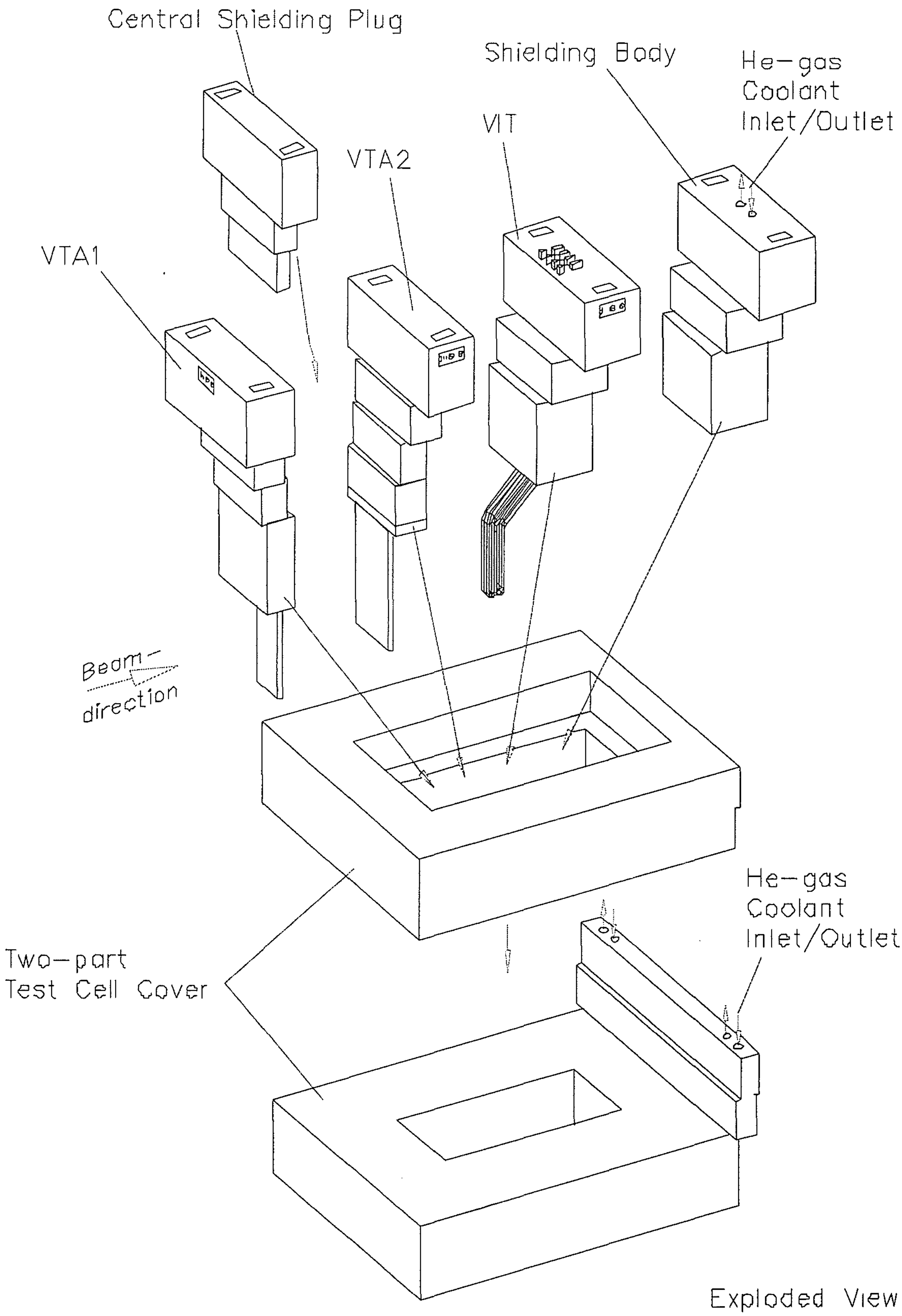


Fig.5: Explosion view of the test cell design concept.

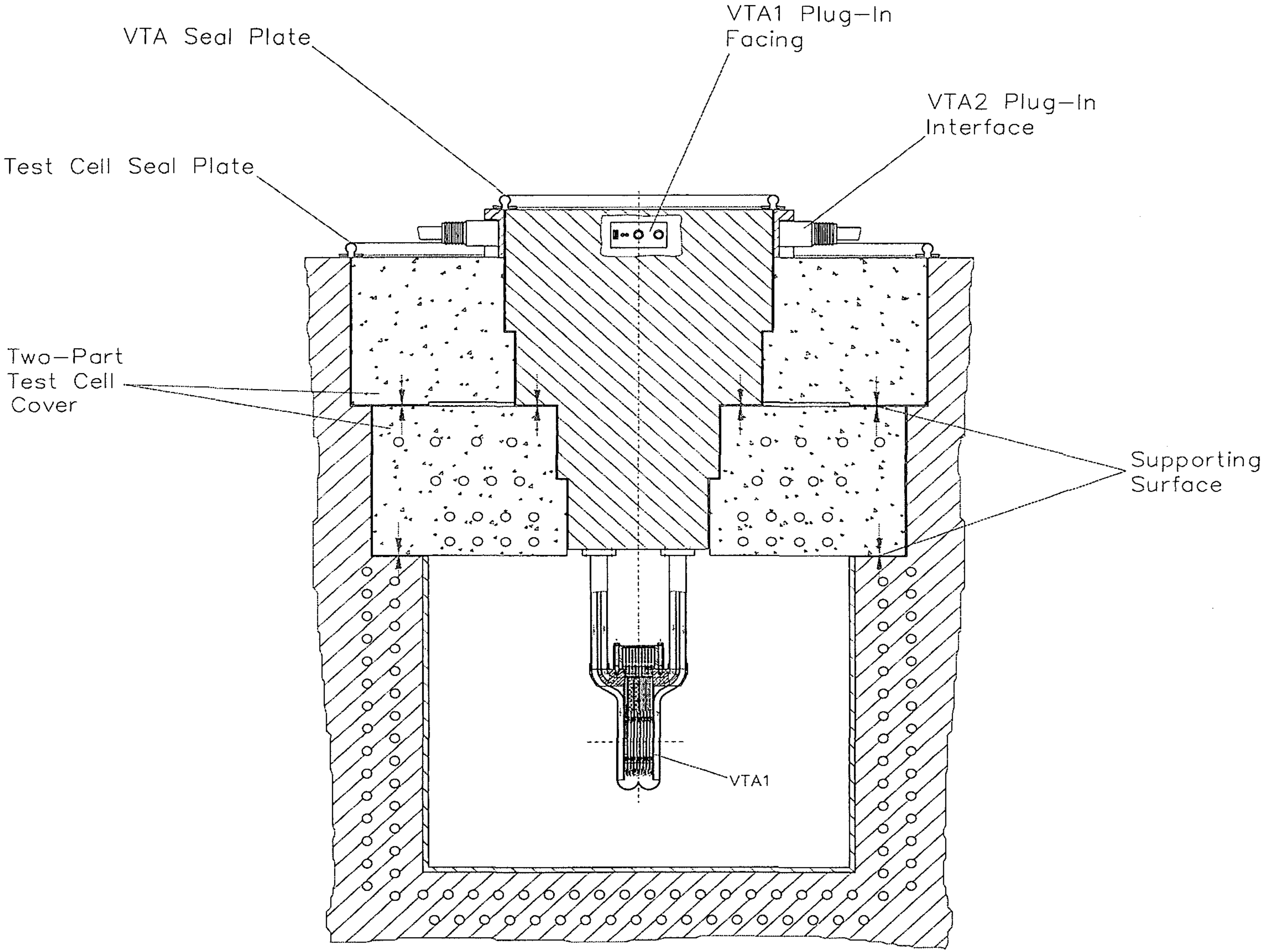


Fig. 6: Improved design configuration for VTA1 with assembled high flux test module; front view.

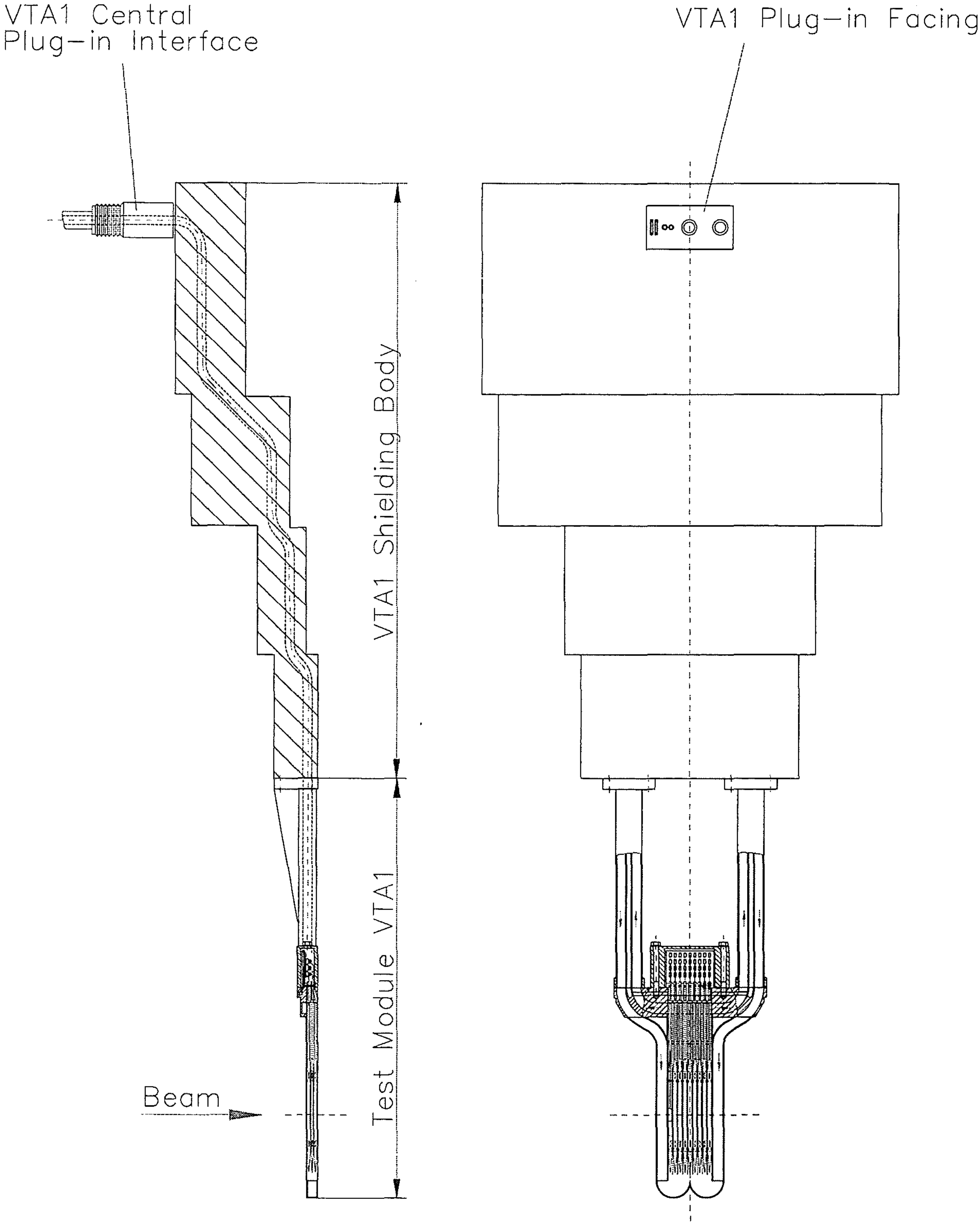
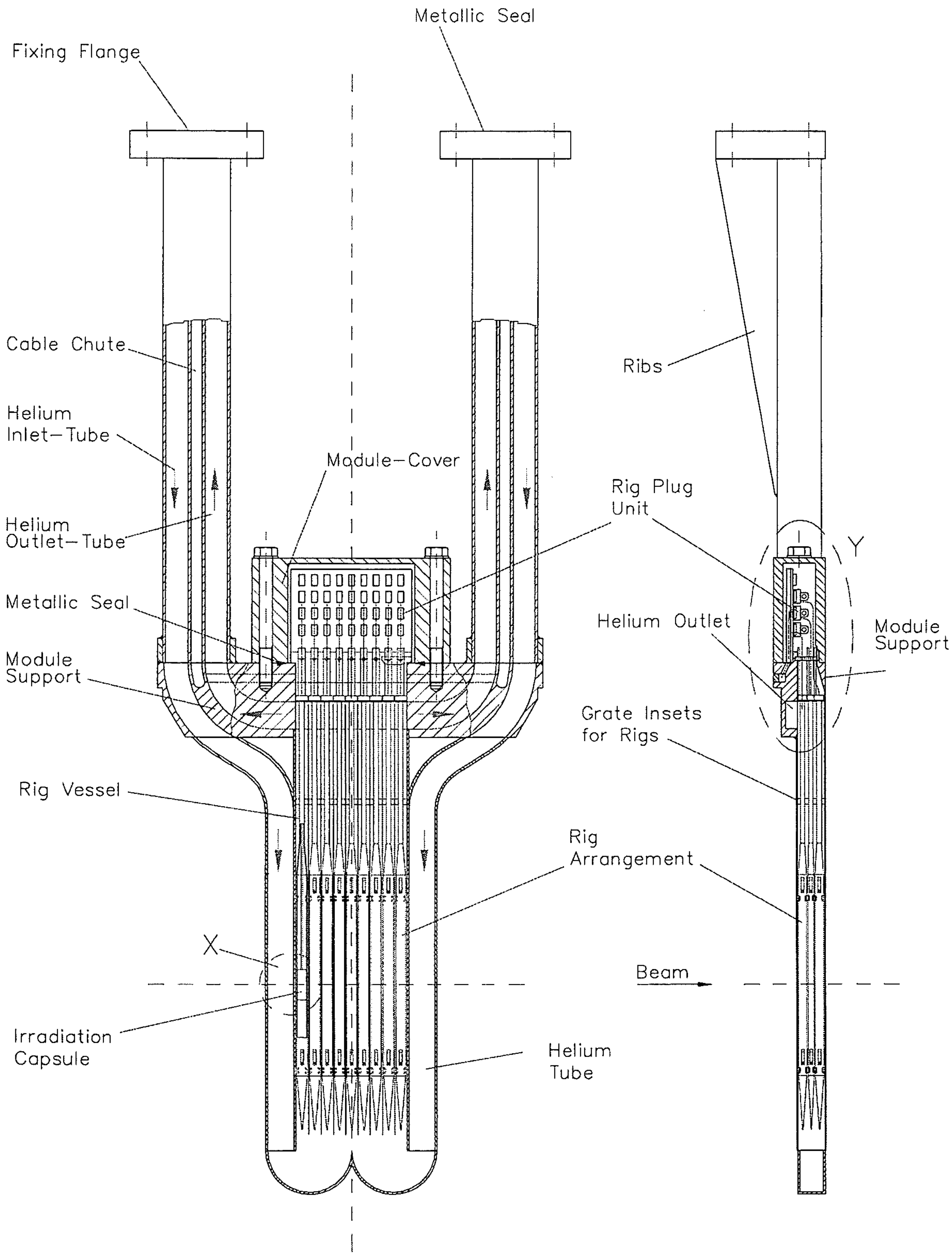


Fig. 7: Design configuration for the VTA1 (helium gas cooled version) with assembled test module; elevation view.



Detail X: Fig. 9
Detail Y: Fig. 10

Fig. 8: Design configuration for the helium gas cooled high flux test module; front view.

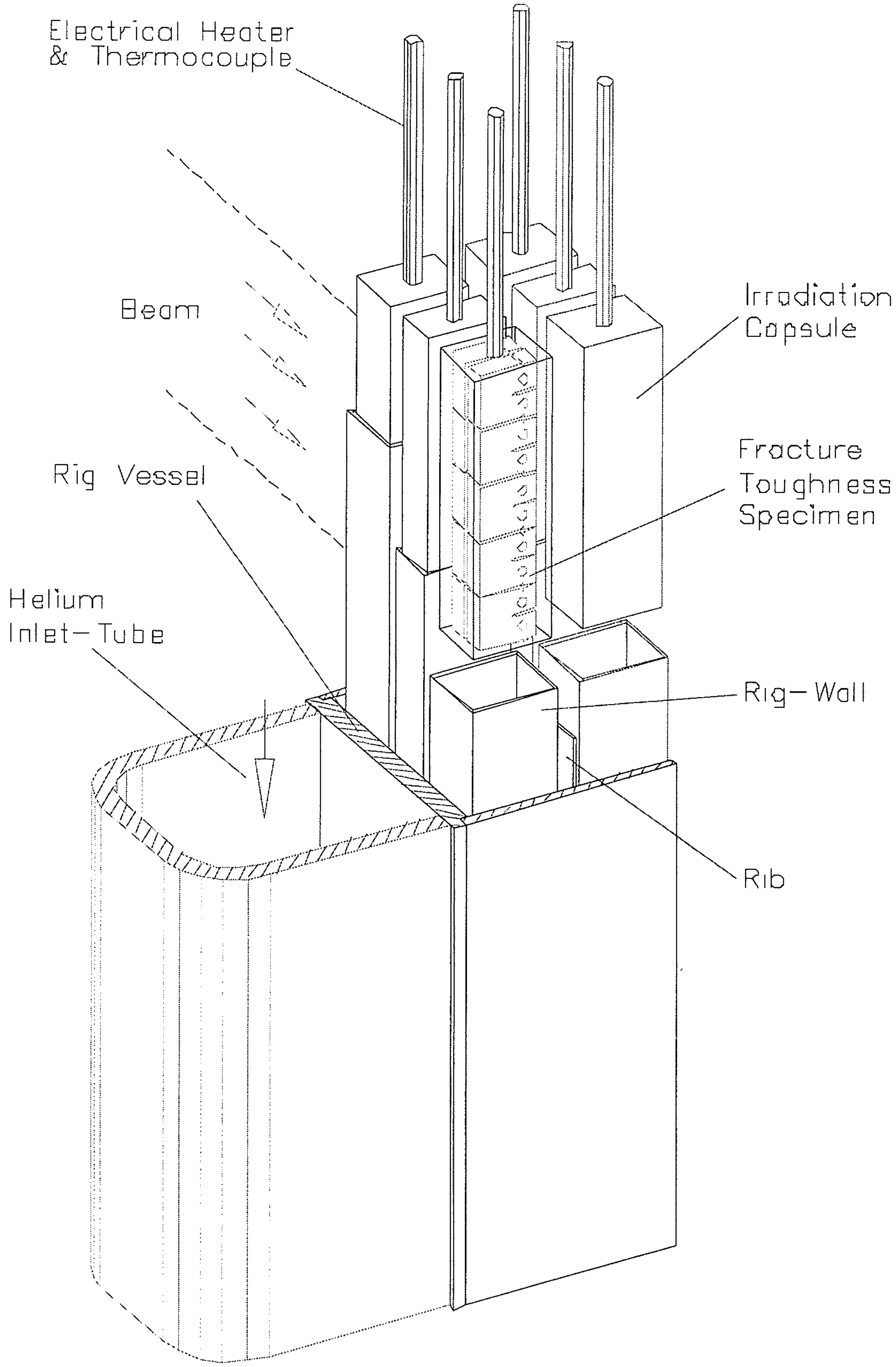


Fig. 9: Helium cooled high flux test module with vertical rigs that accommodate the encapsulated specimens.

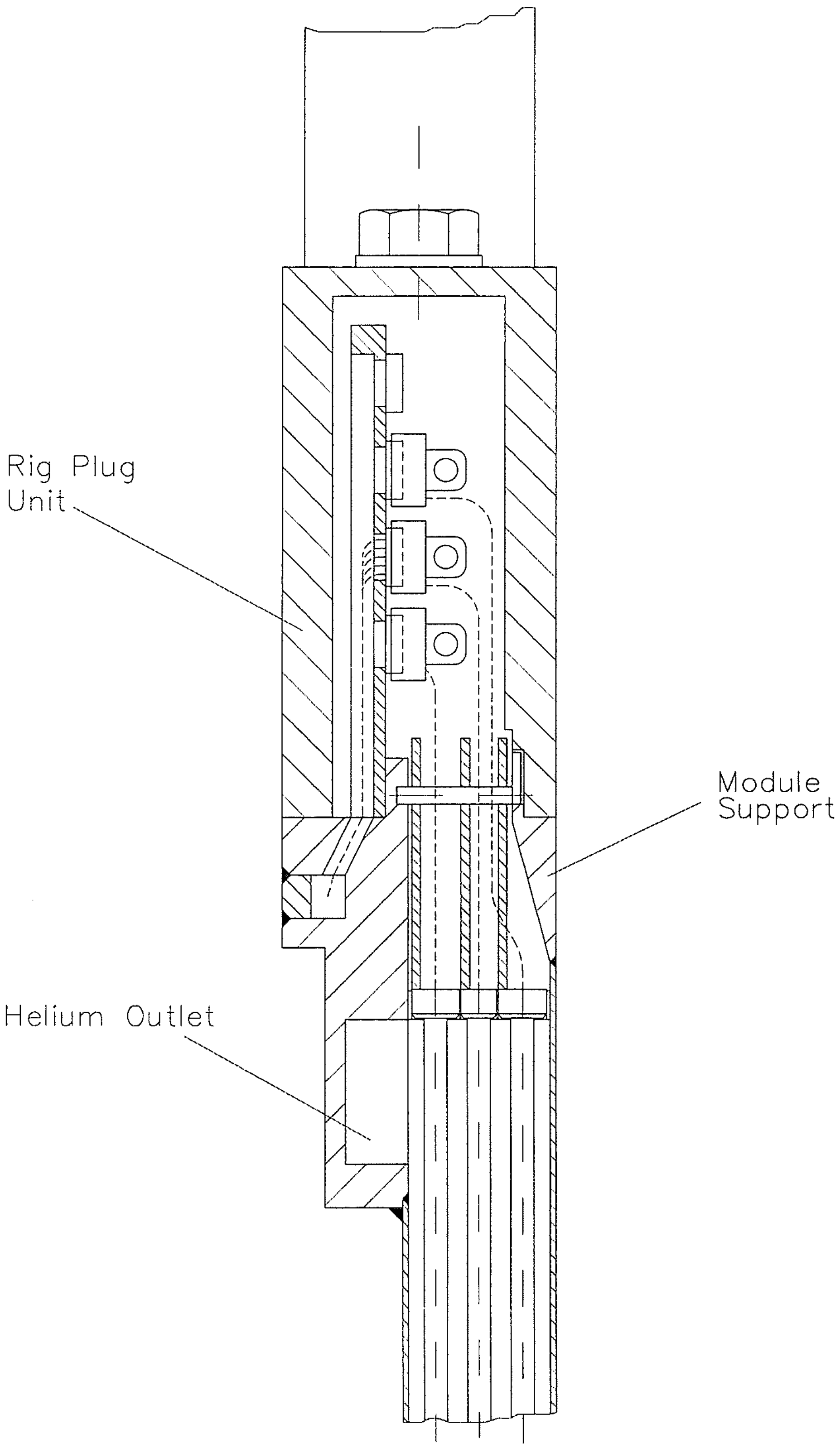


Fig. 10: Elevation view of the upper part of the helium gas cooled high flux test module.

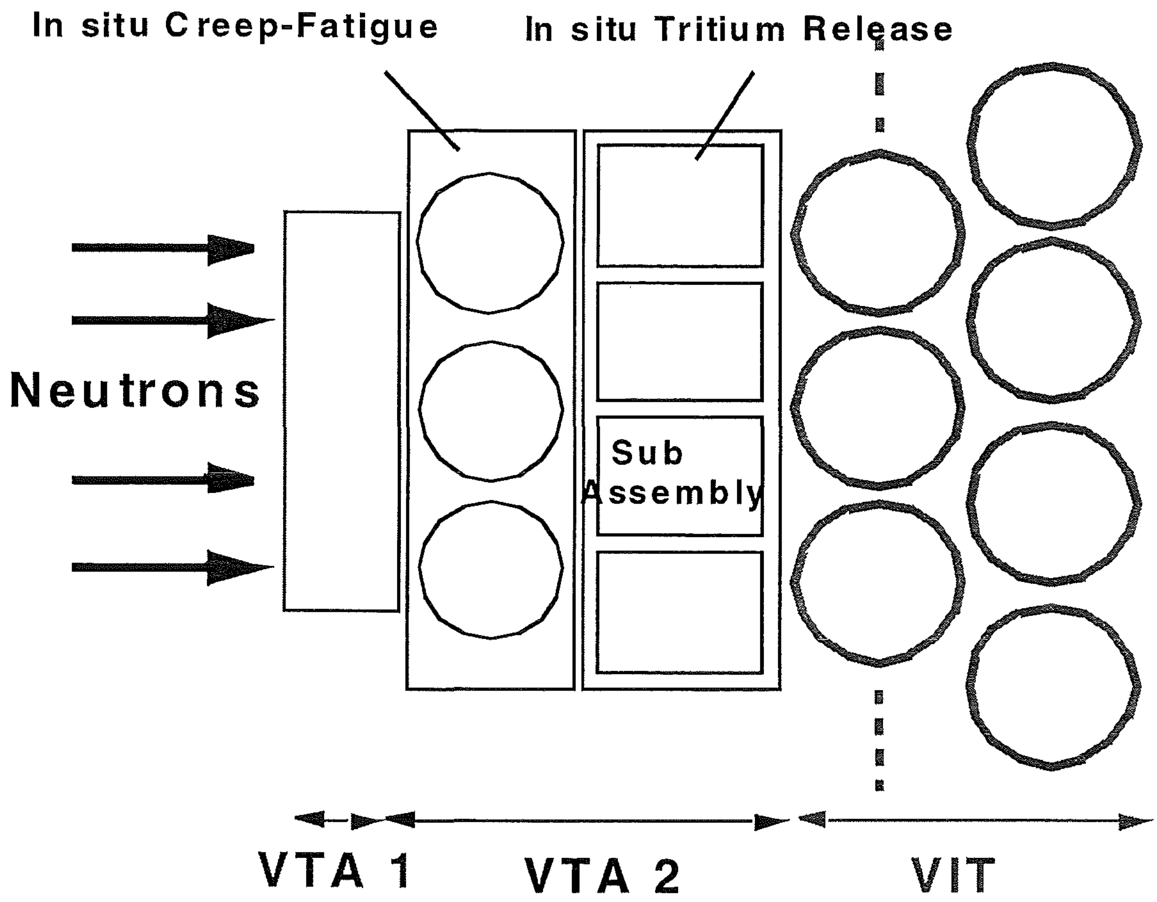


Fig. 11: Updated reference concept of the test module arrangement; VTA2 is assembled with two independent test modules for different types of in-situ experiments.

FZK design:
- Based on CDA reference layout
- Small specimen package density (disadvantage)
- Individual specimen temperatures (advantage)

Figs. 12-16 show the advanced design of VTA2 assembled with two test modules.

JAERI design:
- Completely improved design (see dedicated section)
- High specimen package density (advantage)
- Temperature defined by canisters (limitation ?)

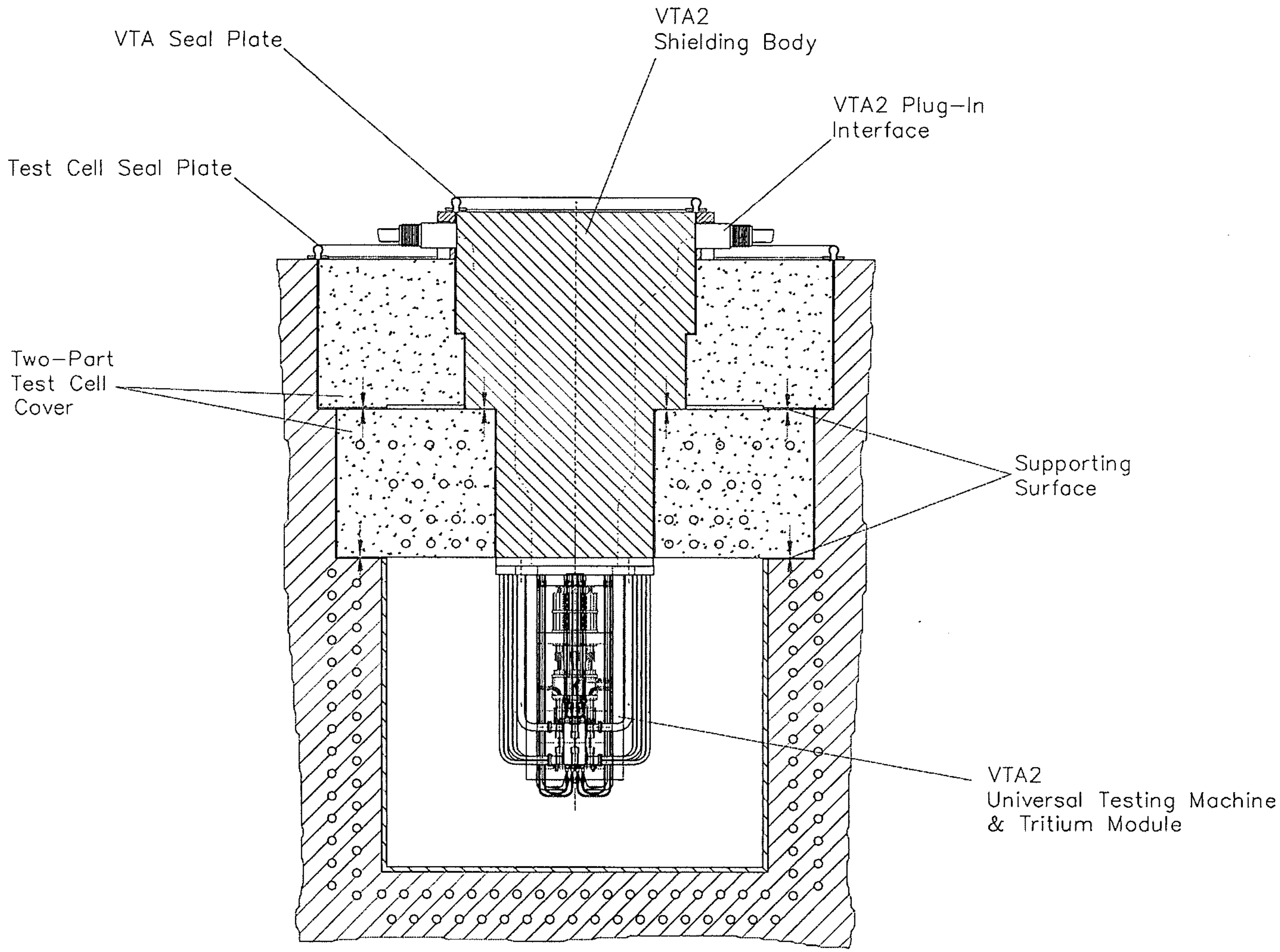


Fig. 12: Front view of the VTA2 with the creep-fatigue test module integrated in the test cell; version with gas cooled concrete.

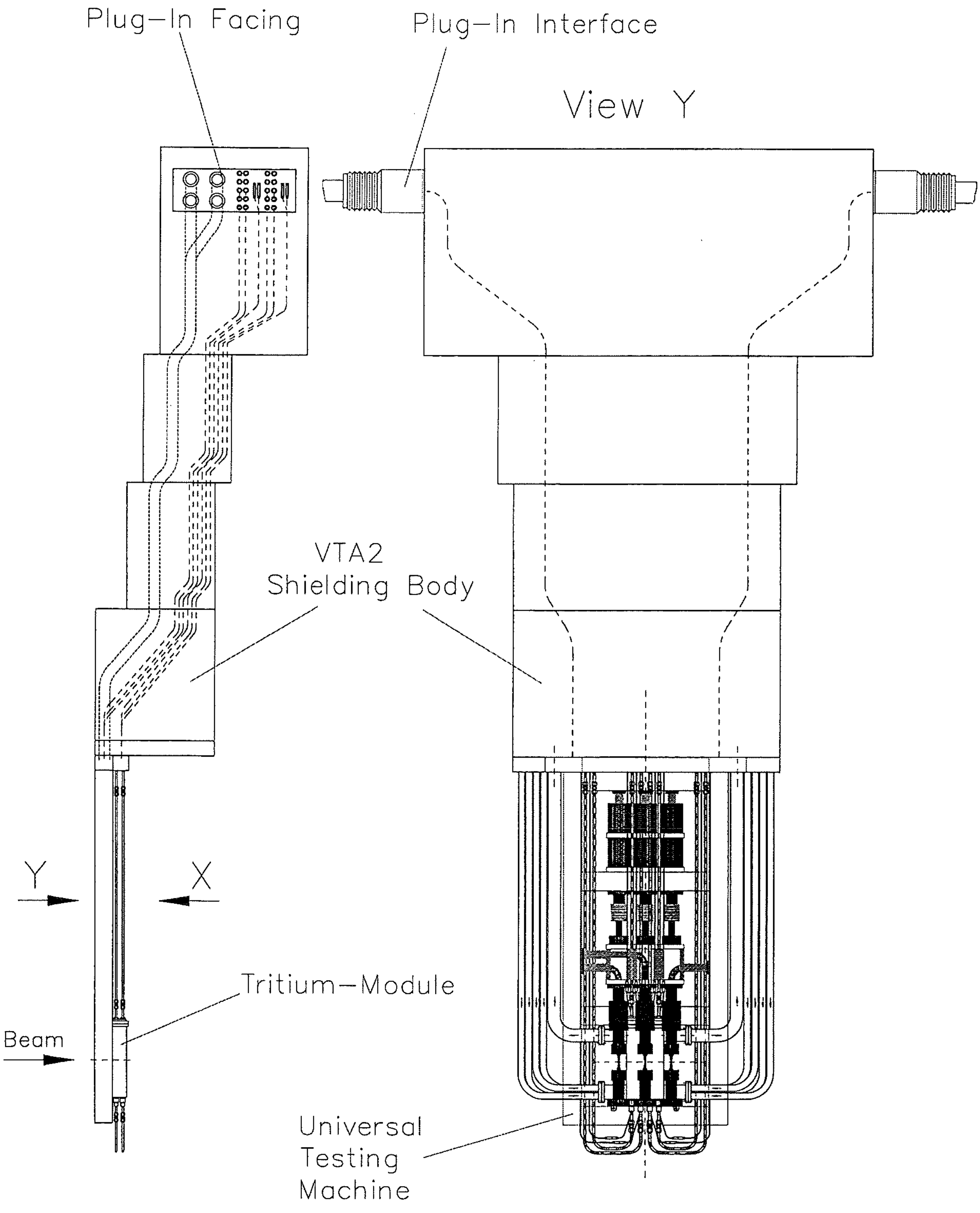


Fig. 13: VTA2 equipped with a test module for in-situ creep fatigue tests and another one for tritium release tests. The elevation (left side) and front views (right side) are shown.

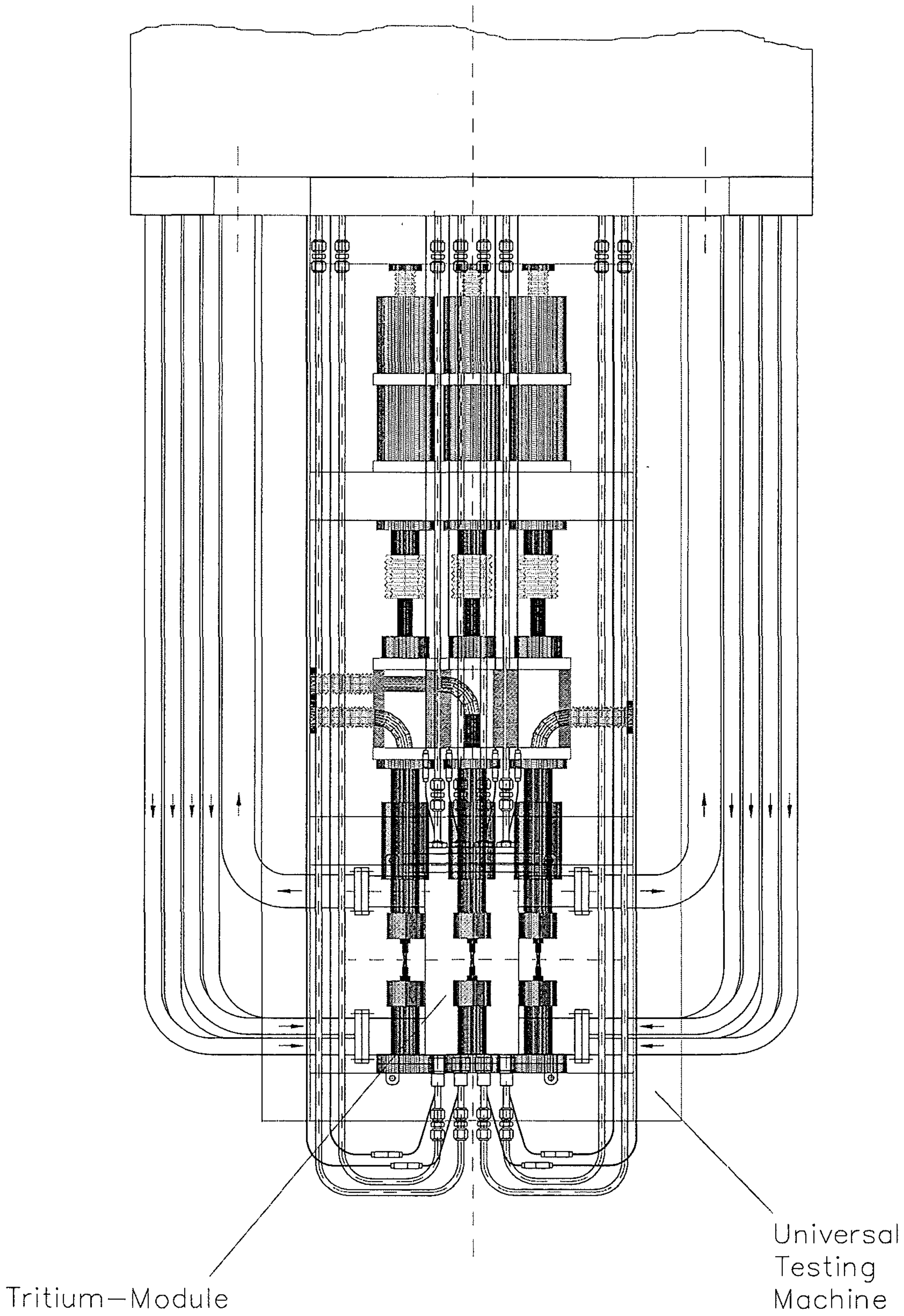


Fig. 14: Back view (upstream direction) showing the test module for tritium release experiments (foreground) and the test module for in-situ creep-fatigue tests (background).

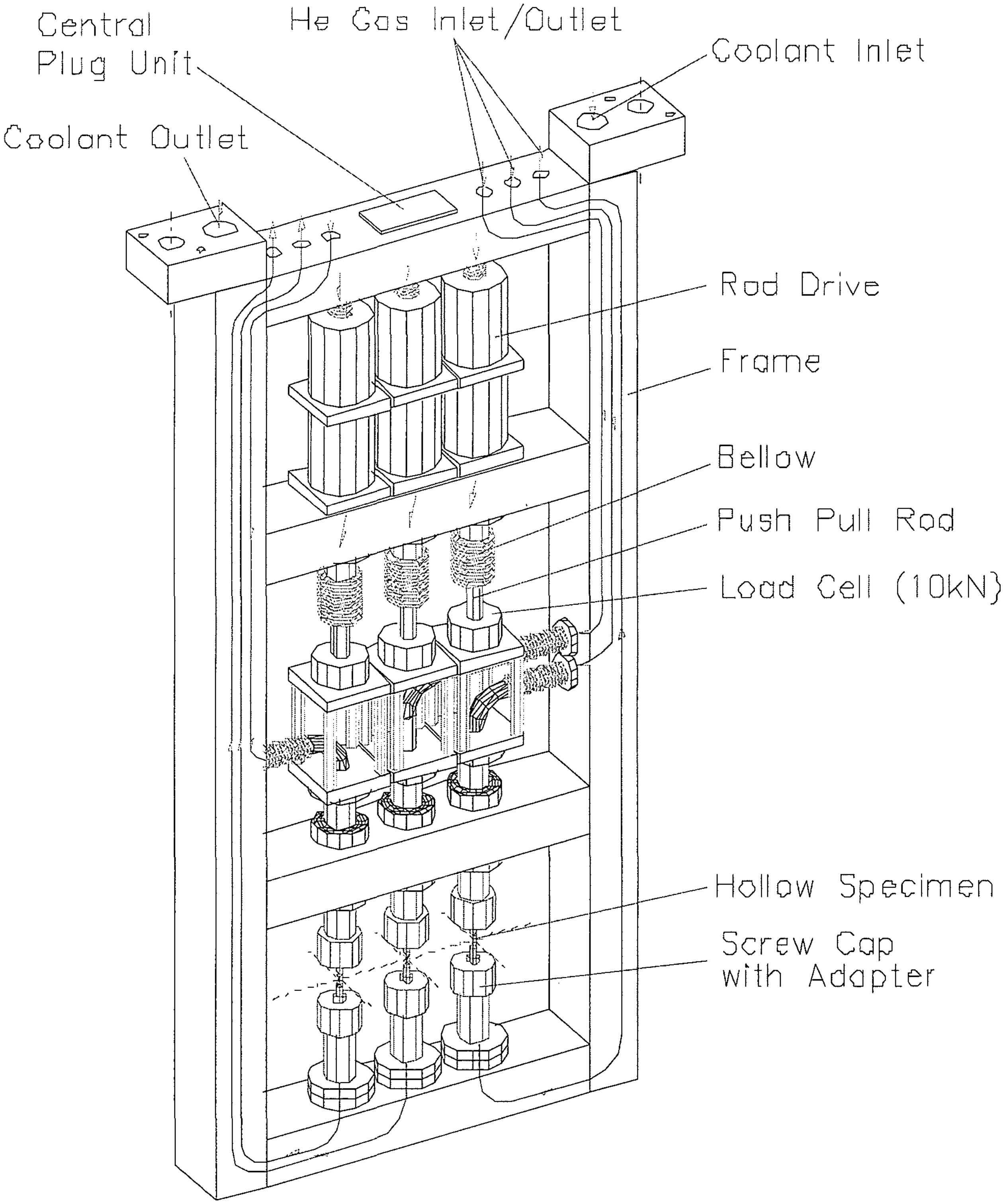


Fig. 15: Test module for in-situ creep-fatigue experiments on three independent specimens.

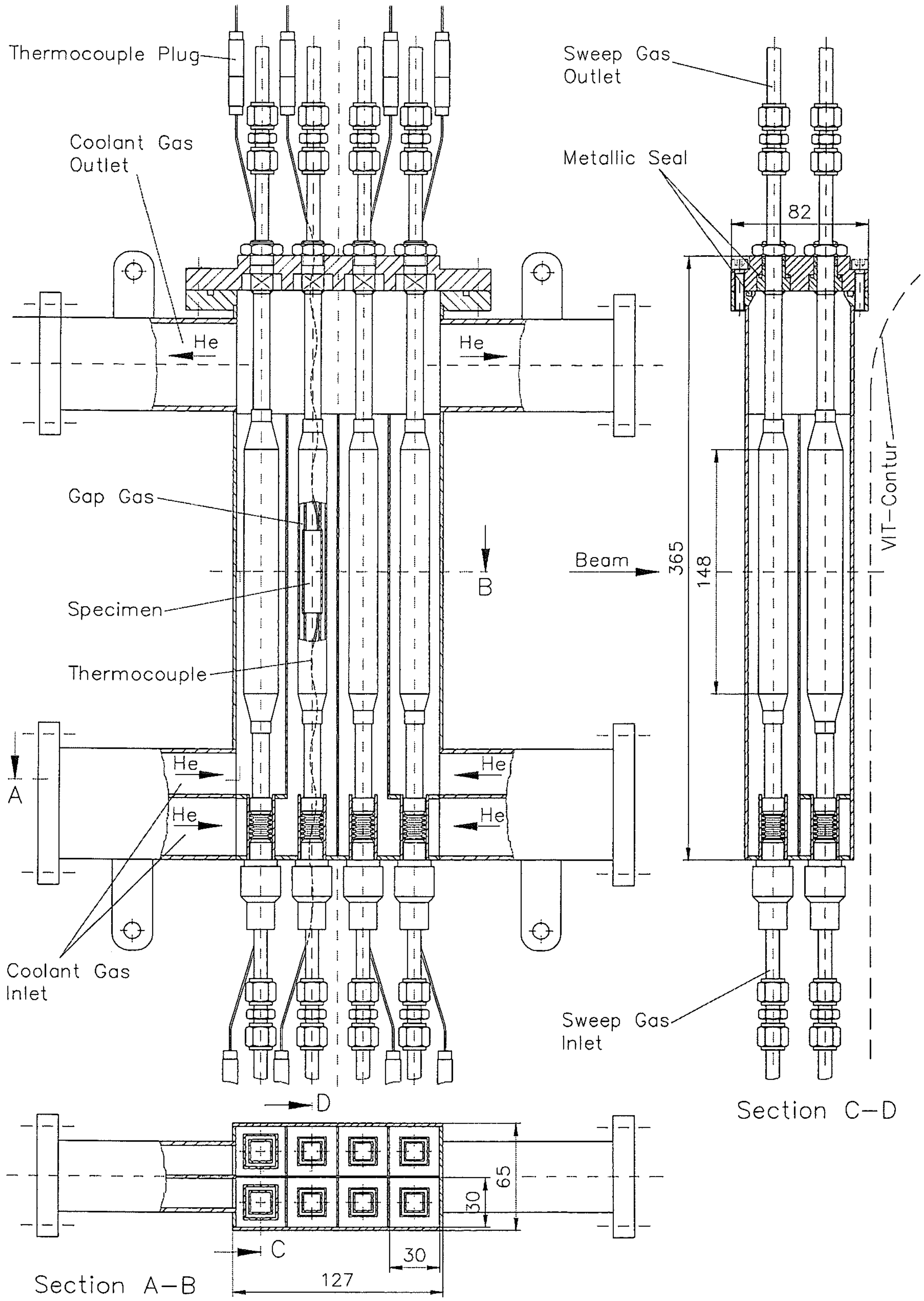


Fig. 16: Test module for in-situ tritium release experiments on ceramic specimens

IFMIF-CDE Test Facility Workshop, Karlsruhe July 7-9 1997

Helium Cooled High Flux Test Module
Task 1

G. Schmitz
Institut für Reaktorsicherheit

A. Möslang
Institut für Materialforschung I

Providing a test module for specimen temperatures up to about 1000 °C is one of the most challenging design requirements for the IFMIF Test Facilities. The successful development of a He-cooled test module would provide for any material a broad temperature window as well as flexible and very safe operating conditions.

During the early phase of the CDE phase the hydraulic calculation of the helium pressure drop (PD) in the helium cooled high flux Vertical Test Assembly (VTA1) as specified in section 2.4 of the IFMIF CDA final report was carried out. Because this calculation has given a too high PD of greater than 0.8 bar in that reference design, form and dimension of the helium pipes were optimized and changed from circular form with a diameter of 30 mm to a rectangular form of about 50 mm side length. Also the shape of the rigs in helium flow direction was changed resulting in a lower hydraulic resistance. These updates of the reference design are shown in fig. 1. The helium is now streaming from the bottom of the rigs upwards, it flows along both sides of the test modules (better cooling efficiency) and the rigs are opened on both sides to guarantee similar helium pressure inside and outside the rig walls. Another important feature is that all rigs can be disassembled and re-assembled individually after the module cover is opened and the electrical signals are disconnected from the rig plug unit.

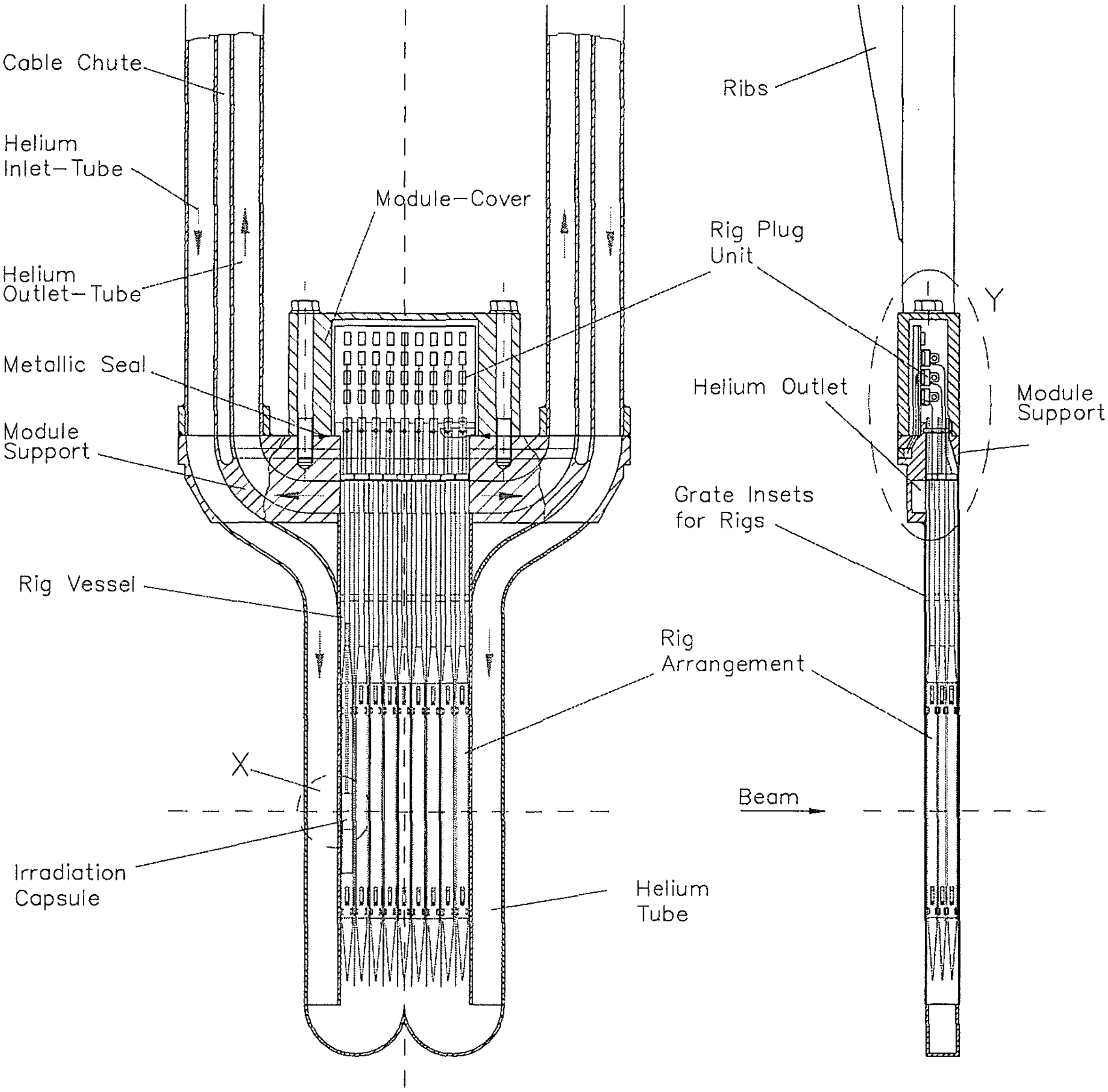
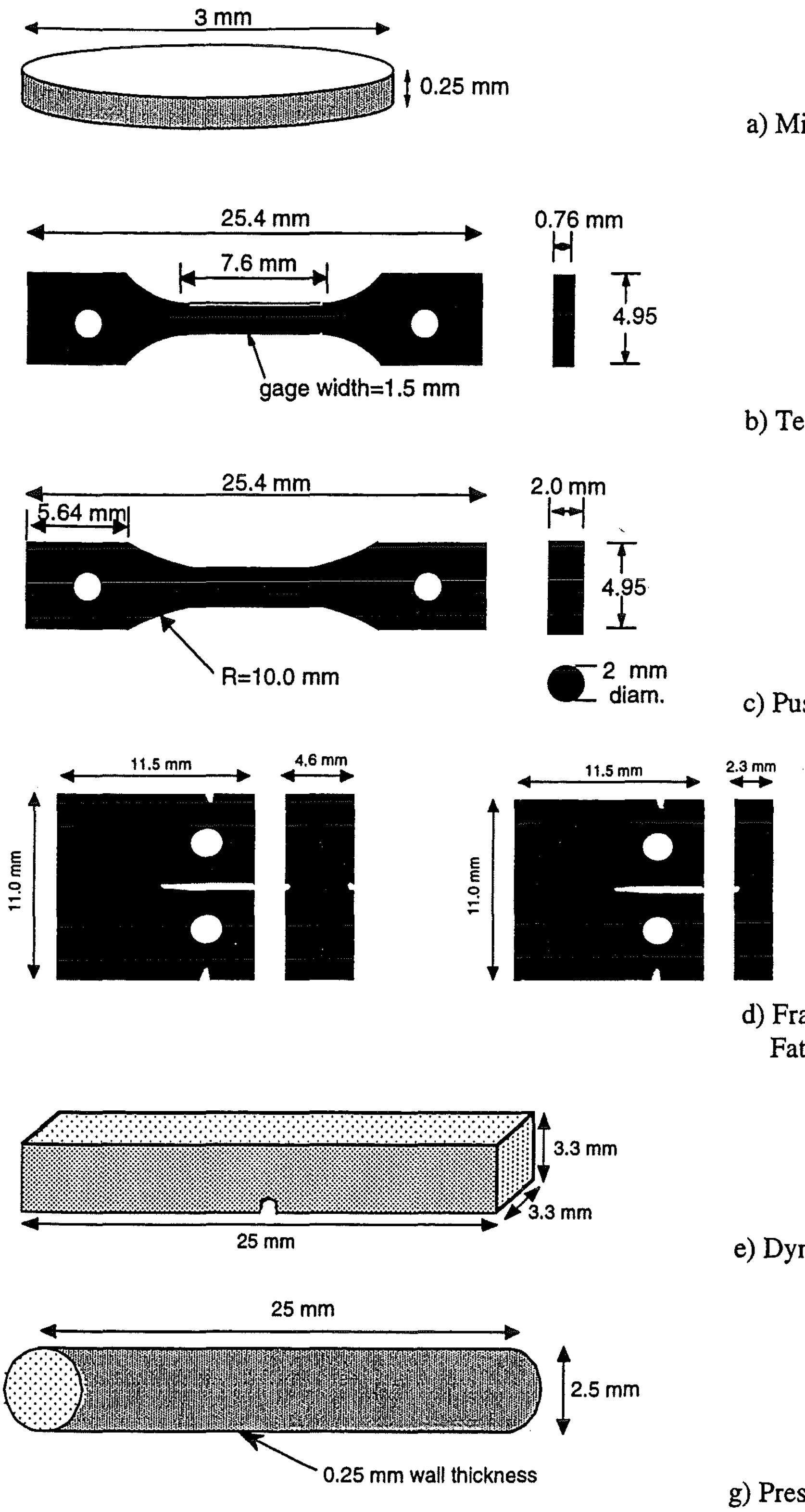


Fig. 1: Front an elevation view of the updated Design configuration for the helium gas cooled high flux test module.

The following figures shows the reference specimens used (from final CDA report), the presently proposed encapsulations, the temperature distribution during irradiation in atypical rig, a cross section of the reference test module, and an elevation view of a rig assembled with 10 pressurized tube specimens. Table 1 lists the main input parameter used in the recent hydraulic calculation, and table 2 shows the results. Obviously, a total PD of only 0.2 bar has been achieved.
In order to test fabricability, remote handling, heat removal etc, a detailed layout for a sub-sized test apparatus with 9 rigs is in progress.



a) Microstructure (TEM disk)

b) Tensile properties

c) Push-pull fatigue

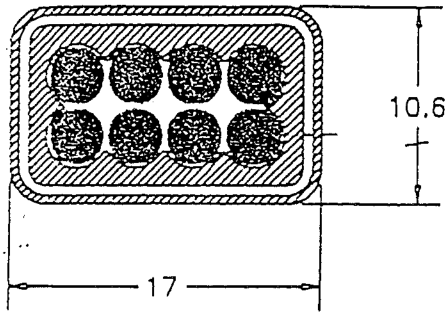
d) Fracture toughness,
Fatigue crack growth

e) Dynamic/static fracture toughness

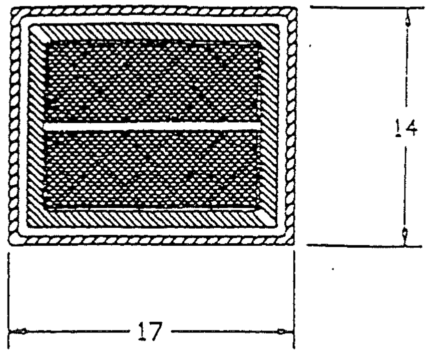
g) Pressurized creep tube

Fig. 2: Reference specimen geometries for the high flux region.

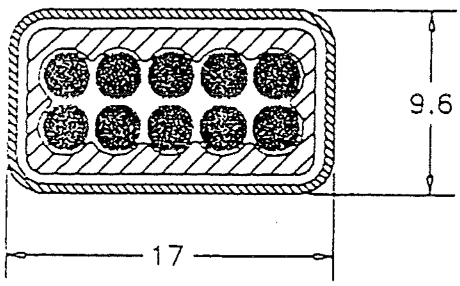
TEM TRANSMISSION
ELECTRON MICROSCOPY DISK



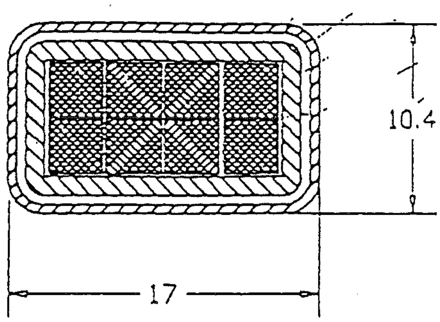
018T DISK COMPACT TENSION
FRACTURE TOUGHNESS



PRESSURISED TUBE



CHARPY/BEND BAR/FTS



SF-1 PUSH-PULL FATIGUE

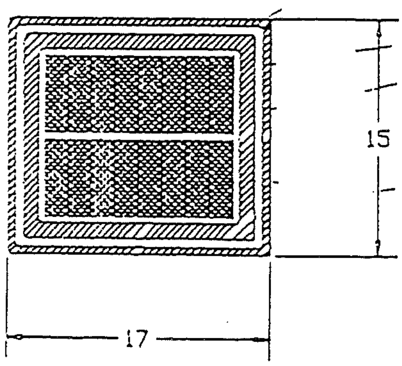


Fig. 3: Cross Section Views of proposed Encapsulations

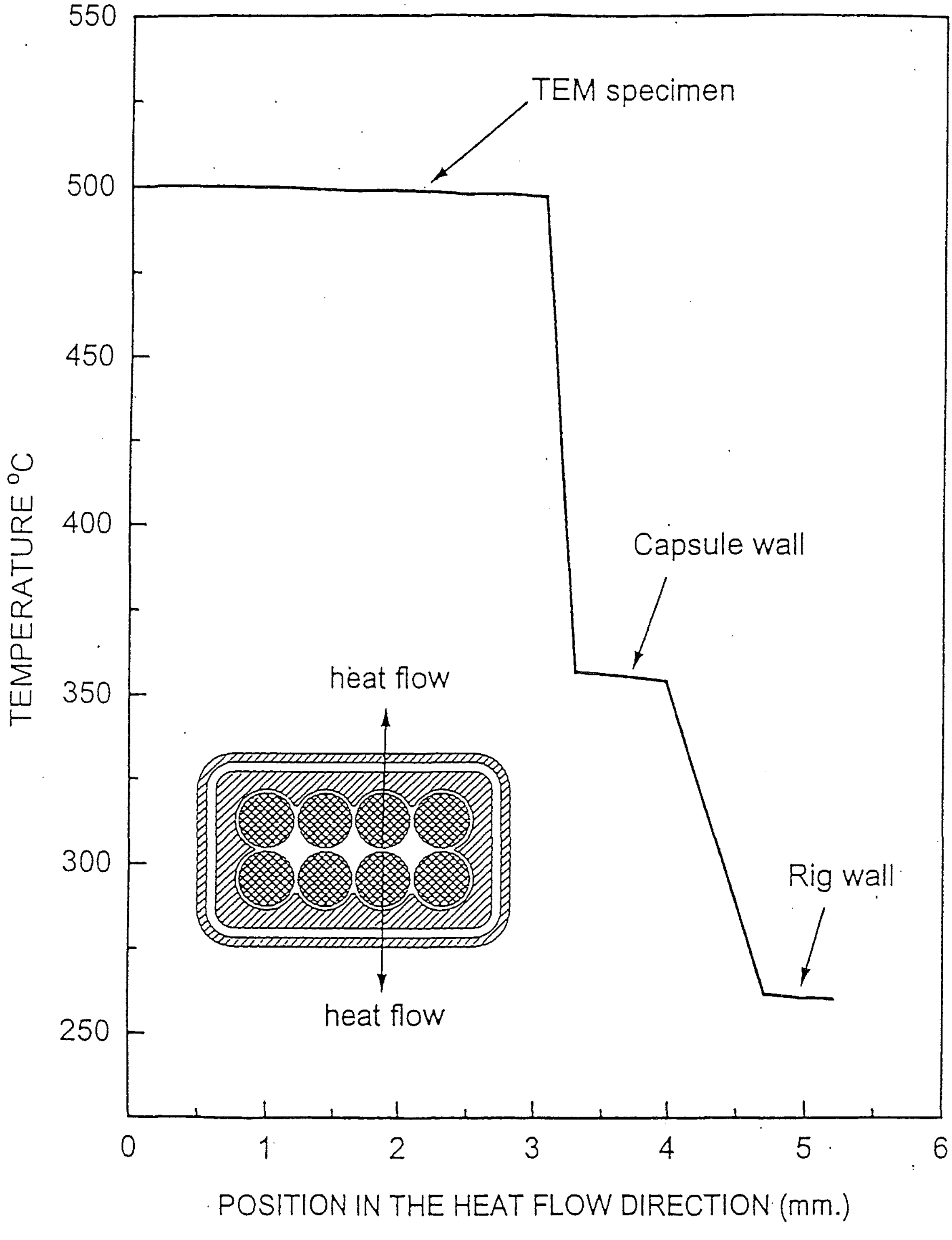


Fig. 4: Temperature Distribution over TEM Disk Specimen Rig

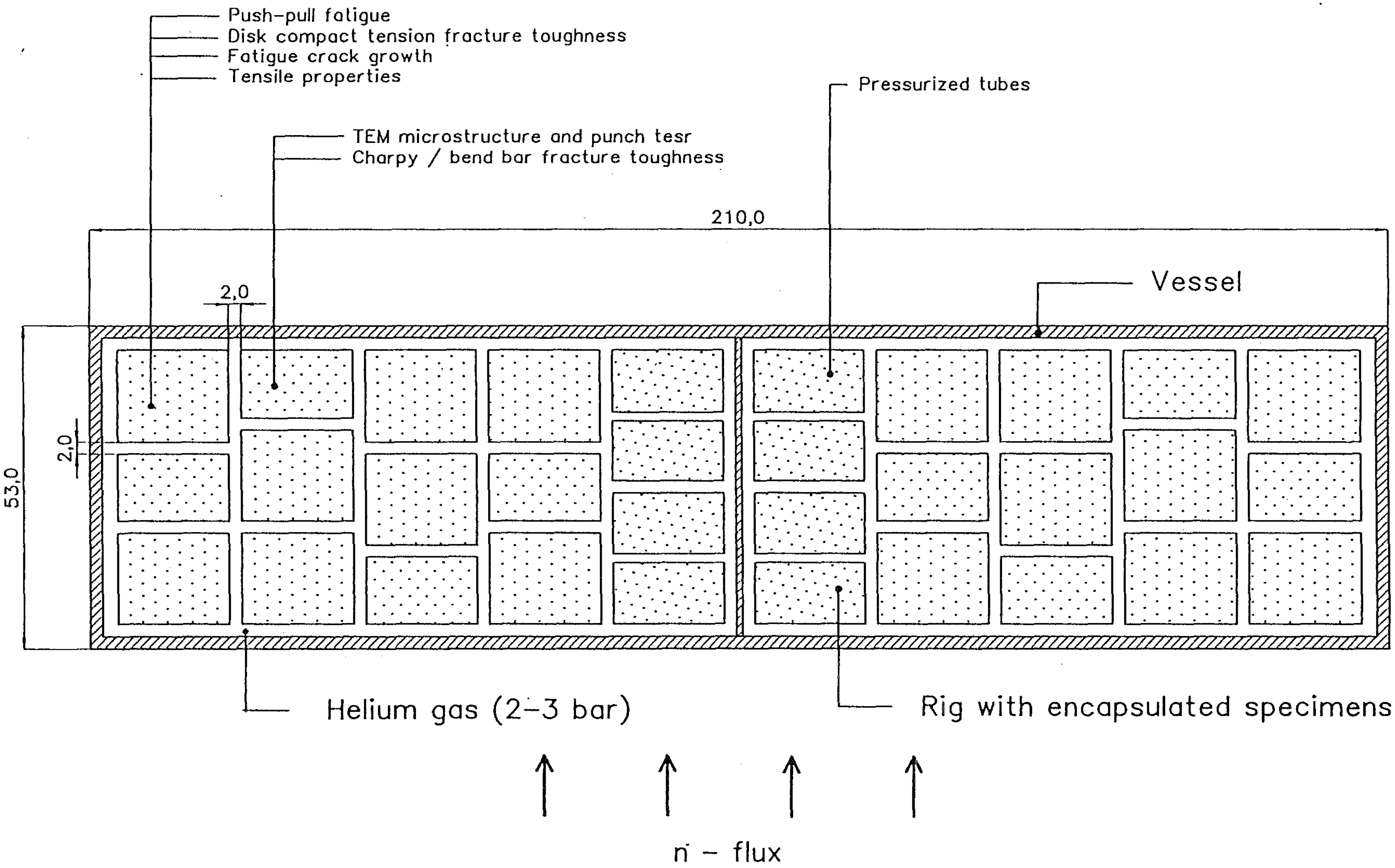


Fig. 5: Top View of Helium Gas Cooled High Flux Test Module

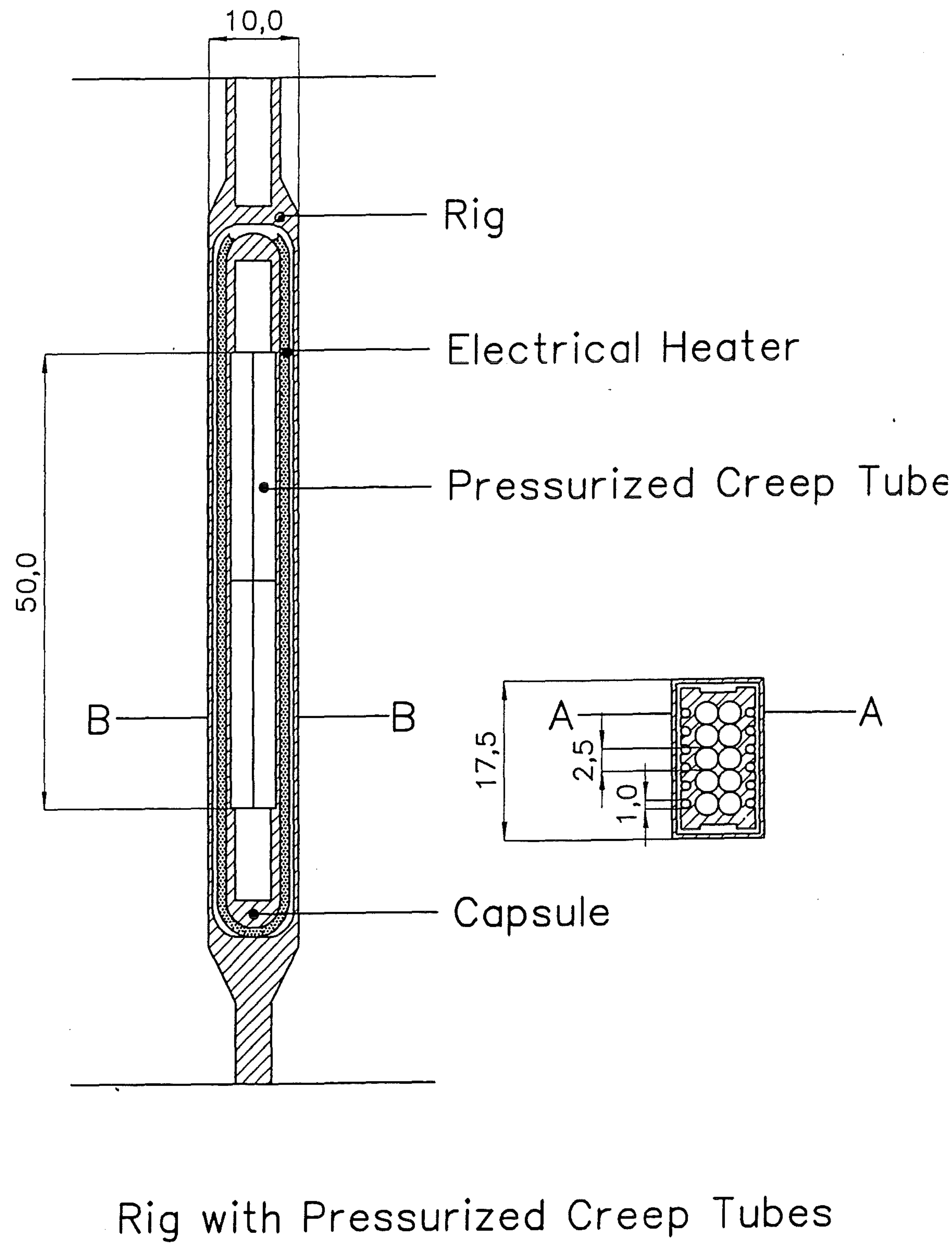


Fig. 6: Rig with Pressurized Creep Tubes

Specific nuclear heating (SS)	5	W/g
Irradiated volume	500	cm^3
Filled Volume	340	cm^3
SS-density (500 °C)	7.7	g/cm^3
Total nuclear heating	13	kW
He pressure (inlet)	2.5	bar
He temperature (inlet)	50	°C
He temperature (outlet)	62	°C
He specific heat capacity	5.21	J/(g*K)
He mass flow	210	g/s

Tabl. 1: Vertical Test Assembly 1 (He-Version),
Technical Data (Extract)

Position Region	Pressure [bar]	Pressuredrop [bar]
Inlet	2.50	
		0.05
Outlet Inletpipe	2.45	
		0.01
Inlet Lower Grid	2.44	
		< 0.01
Outlet Lower Grid	2.44	
Lower Part of Rigs		< 0.01
	2.44	
Region of Specimens		0.07
	2.37	
Upper Parts of Rigs		~ 0.01
Outlet Region of Rigs	2.36	
		~ 0.01
Inlet Outletpipe	2.35	
		~ 0.05
Outlet	2.30	

Tabl. 2: Calculated Pressures and Pressuredrops in VTA 1 (He-version)

Vertical Test Assembly 2 (VTA 2) Tritium Release Test Module -Improvement of Conceptual Design-

Presented by K. Noda
Japan Atomic Energy Research Institute

IFMIF-CDE Test Facility Workshop
Karlsruhe
July 7-9, 1997

Forschungszentrum Karlsruhe
Institute of Materials Research 1
Building 681, Room 214

Department of Materials Science and Engineering

Outline of Tritium Release Test Module

VTA 2 installed in the medium flux region is equipped with Tritium release test module and in-situ creep-fatigue test module.
Measurements of tritium release and compatibility tests during irradiation, followed by PIE, are performed with tritium release test module for ceramic breeders test matrices provided in users' requirements.

Test Matrices of Ceramic Breeders

Specimen	Diam. (mm)	Thick. (mm)	Temp. (C)	Test items	Dose(dpa)
Disk	10	2	400/600/800	In-situ tritium release Neutron radiography	10, 20, 30
Pellet	10	10	Temp. grad.	Mechanical Integrity Dimensional change	
Pebble	1		Temp. grad.	Microstructure change Tritium/He retention Fracture strength, etc.	
Breeder	5	1.5	400/600/800	Compatibility	10, 20, 30
Structural	5	0.5	400/600/800		

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Improvement of Conceptual Design of Tritium Release Test Module

(1) Irradiation capsule structures were changed for better temperature control. Inner capsules containing specimens are surrounded by the gas gap for temperature control.

(2) Number of tubes for gap gas of irradiation capsules was substantially decreased due to the above-mentioned change of capsule structure. This leads increase of fabricability and reliability of the capsules.

(3) Configuration of specimens to beam line was changed. Flat surface of the disk specimens is perpendicular to the beam line. This configuration leads decrease of neutron flux difference in the disc specimens.

(4) Structure of both end of the capsule was changed to avoid steep temperature gradient of the region welded with tubes. This contributes improvement of irradiation durability of the capsules.

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Improvement of Conceptual Design of Tritium Release Test Module

(5) Many irradiation capsules were contained in the sub-test modules in tight configuration with small coolant gas paths. The small paths among the capsules are made by coiling spacer (wire) of 1 mm in diameter spirally. This configuration allows to irradiate specimens of the required test matrices for two different neutron fluence levels simultaneously.

(6) All specimens loaded in the sub-test modules can be set within the medium flux region behind in-situ creep/fatigue test module.

(7) Temperature of structure materials of sub-test modules and outer capsules can be kept at about 100 C during irradiation tests by He coolant gas. This leads minimizing irradiation degradation of the sub-test modules and irradiation capsules.

(8) Low activation ferritic steel will be used for most part of the sub-test modules and irradiation capsules. This contributes to lowering radioactivity of the VTA-2 and to decrease of radioactive waste from IFMIF.

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Configuration of Tritium Release Test Module

Tritium release test module consists of 4 sub-test modules;

-3 sub-test modules for tritium release tests of ceramic breeder disk specimens and for compatibility tests at 3 levels of irradiation temperatures (i.e., 400, 600, 800 C)

-A sub-test module for tritium release tests of ceramic breeder pellet and pebble specimens.

Dimensions of sub-test modules for disk specimen tritium release tests and compatibility tests and for pellets and pebbles specimens tritium release tests are 33X58X273 mm^3 and 95 X64X273 mm^3, respectively.

Size of tritium release test module is about 220 (width) X 75 (thickness) X 270 (height) mm^3 and can be set within medium flux region behind in-situ creep-fatigue test module. The test module allows simultaneous irradiation of specimens in test matrices proposed in users' requirement for two different neutron fluence levels.

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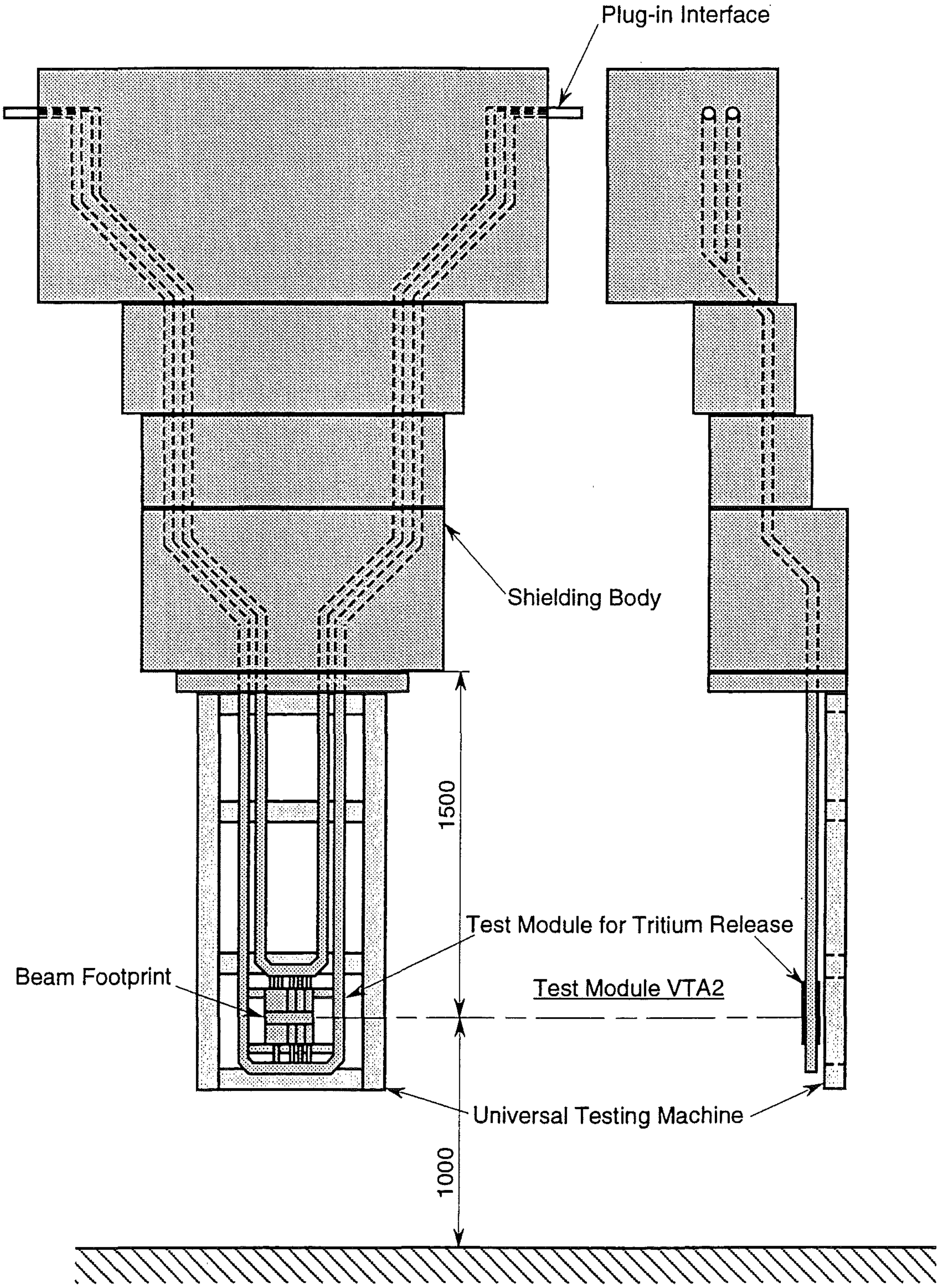
Plug-in Interface
Shielding Body
1500
Test Module for Tritium Release
Beam Footprint
Test Module VTA2
Universal Testing Machine
1000

Sub-test Module for Disk Specimen Tritium Release and Compatibility Tests

* 3 sub-test moldules for irrad. tests at 3 level of temperatures, e.g. , 400, 600, 800 C.

* 12 capsules for disk specimen tritium release test and 2 capsules for compatibility test can be contained in each sub-test module.

* 28 He sweep gas tubes, 28 He gap gas tubes and 14 T.C. (thermocouples) are equipped.

Sub-test Module for Pellet and Pebble Specimen Tritium Release Tests

* 12 capsules for pellet specimen tritium release test and 12 capsules for pebble specimen tritium release test can be contained in each sub-test module.

* 48 He sweep gas tubes, 48 He gap gas tubes and 24 T.C. (thermocouples) are equipped.

* Temperature of inner capsule surface (surface of pellet specimen and pebble bed) will be controlled to be about 400 C.

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Common Features of Sub-test Modules

* Sub-test modules have box structure of low activation ferritic steel (0.5 mm) with ribs.

* Spacers (wire) of 1 mm diam. are coiled around each capsule to maintain coolant gas path.

* Sub-test module has inlet and outlet pipes for coolant gas, gap gas tubes/sweep gas and T.C. (thermocouple).

* Temperature of sub-test module structure is maintained at about 100 C to avoid severe irradiation degradation.

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Common Features of Irradiation Capsules

* Capsules consist of inner and outer capsules of box/cylindrical structure with gas gap between them.

* He gas sweep and He gap gas tubes are connected to inner and outer capsule, respectively.

* Temperatures of specimens are measured with thermocouple.

* Most of parts of capsule are made of low activation ferritic steel.

* Parts welded with tubes are separated from higher flux region by increasing capsule length to avoid severe irradiation degradation.

* Temperature of outer capsule is maintained at about 100 C for irradiation durability of capsule.

*Specimens are loaded in the center part (length; 5cm) corresponding the medium flux region.

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Capsules for Disk Specimen Tritium Release Test

* 5 disk specimens of a kind of ceramic breeder materials are loaded in each capsule.

* Irradiation temperature of specimens can be controlled by changing flow rate of gap gas. Specimens will be irradiated at temperatures in the range 300-900 C (e.g., 400, 600, 800 C).

Capsules for Compatibility Test

* 12 disk specimens of 6 kinds of ceramic breeder materials and 12 disk specimens of 2 kinds of structural materials can be loaded in each capsule.

* Irradiation temperature of specimens can be controlled to be a temperature in the range 300-900 C (e.g., 400, 600, 800 C) by the method same as that of disk specimen tritium release test capsule.

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Capsules for Pellet Specimen Tritium Release Test

* 2 pellets specimens of a kind of ceramic breeder materials can be loaded in each capsule.

* Irradiation temperature of the specimens will be controlled by changing flow rate of gap gas , and the surface of temperature of the pellet specimens is controlled to be about 400 C.

Capsules for Pebble Specimen Tritium Release Test

* Pebble specimens of a kind of ceramic breeder materials in pebble bed size (1cm in diam., 1 cm height) can be loaded in each capsule.

*Irradiation temperature of inner capsule surface will be controlled to be about 400 C by the method same as that of pellet specimen tritium release test capsule.

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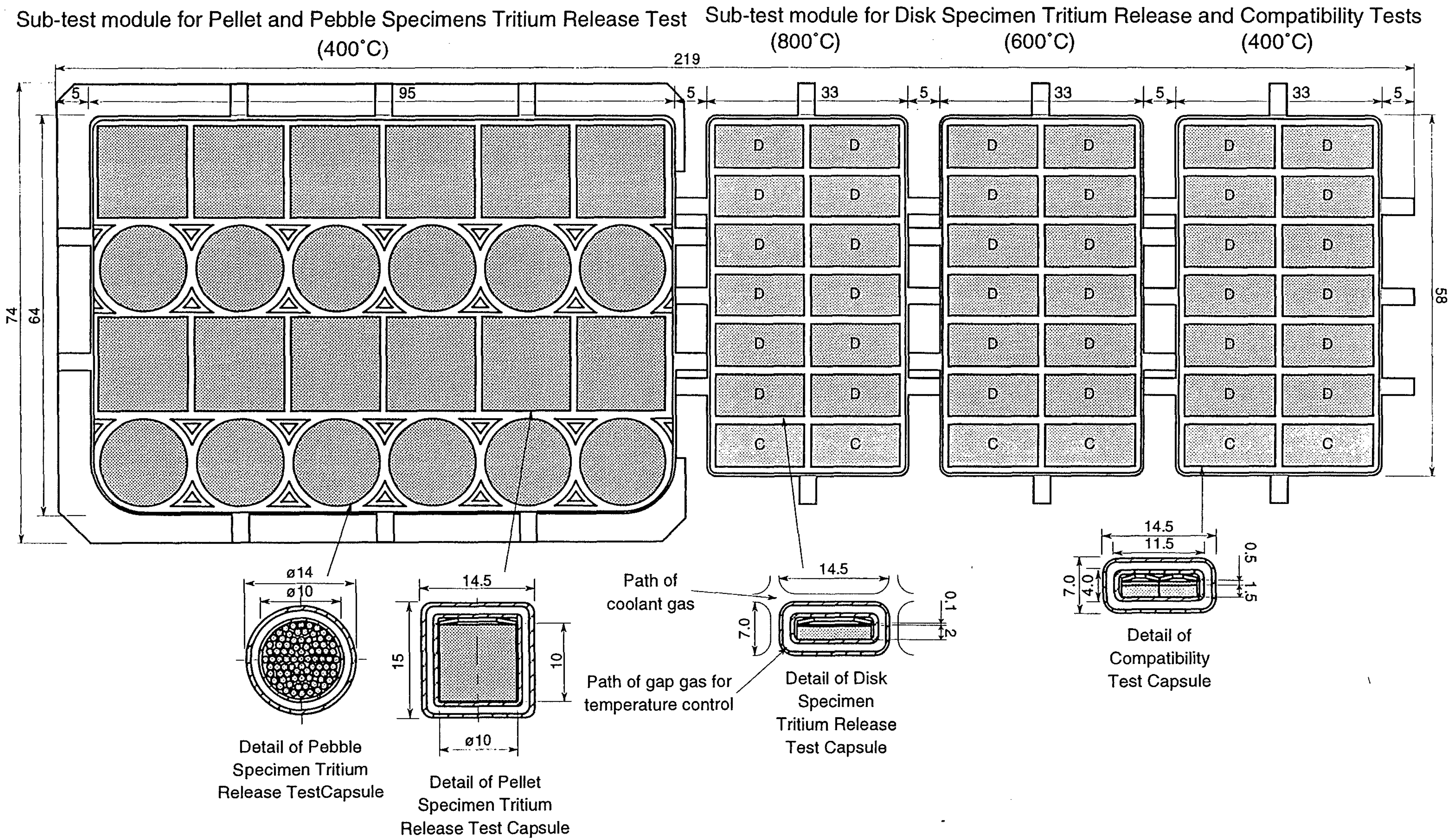
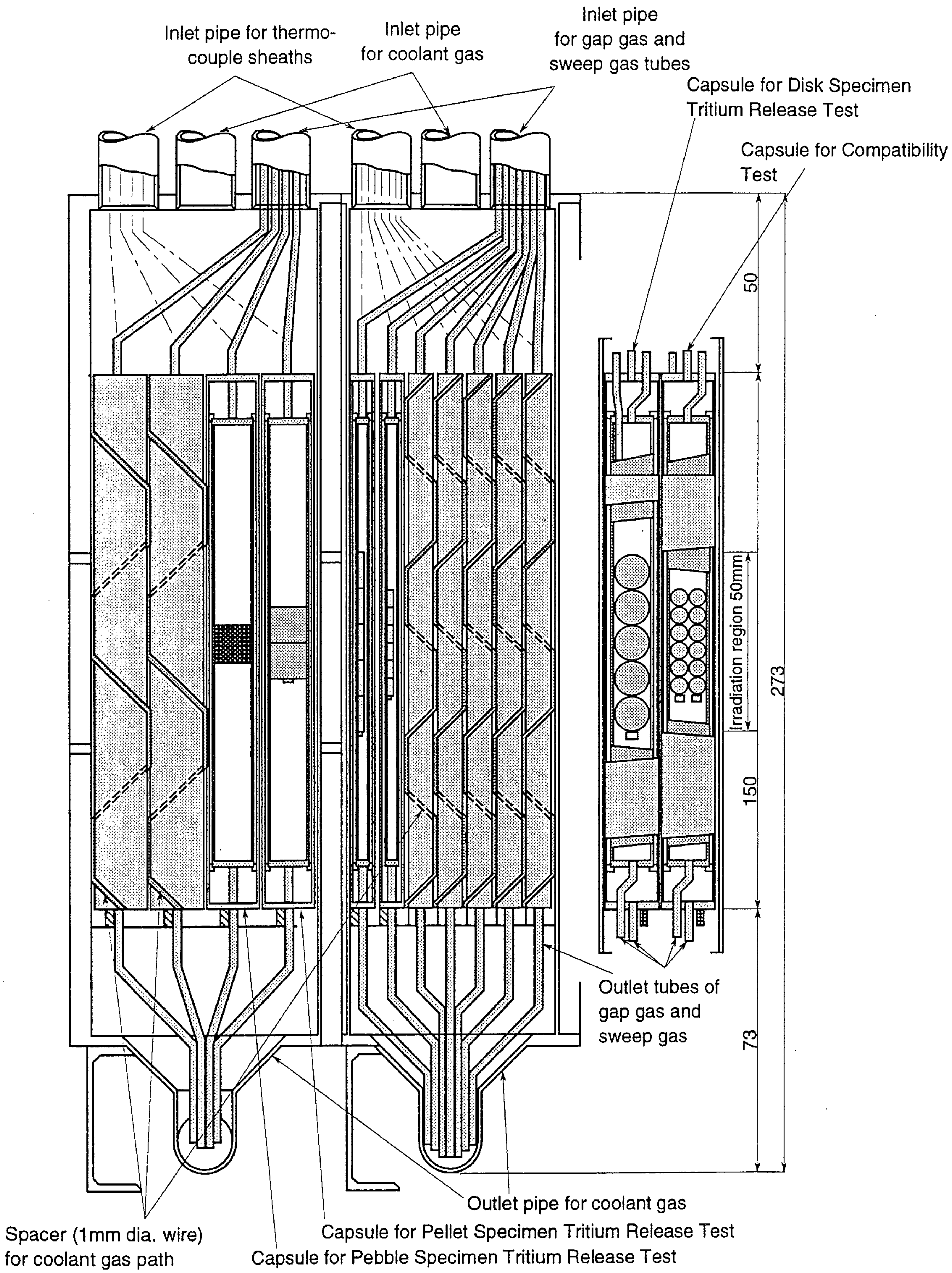

Sub-test module for Pellet and Pebble Specimens Tritium Release Test
(400˚C)
Sub-test module for Disk Specimen Tritium Release and Compatibility Tests
(800˚C)
(600˚C)
(400˚C)
219
5
95
33
74
64
58
D
C
ø14
ø10
14.5
15
10
Path of coolant gas
Path of gap gas for temperature control
7.0
0.1
2
11.5
4.0
0.5
1.5
Detail of Pebble Specimen Tritium Release TestCapsule
Detail of Pellet Specimen Tritium Release Test Capsule
Detail of Disk Specimen Tritium Release Test Capsule
Detail of Compatibility Test Capsule


Inlet pipe for thermo-couple sheaths
Inlet pipe for coolant gas
Inlet pipe for gap gas and sweep gas tubes
Capsule for Disk Specimen Tritium Release Test
Capsule for Compatibility Test
50
Irradiation region 50mm
273
150
73
Outlet tubes of gap gas and sweep gas
Outlet pipe for coolant gas
Capsule for Pellet Specimen Tritium Release Test
Capsule for Pebble Specimen Tritium Release Test
Spacer (1mm dia. wire) for coolant gas path

Temperature Control of Sub-test Module

Temperature of structural materials of sub-test modules is maintained at about 100 C by He coolant flowing in the module.

[Calculation conditions]

Structural material: Low activation ferritic steel

Thickness: 0.5 mm (vessel), 5mmX2.5mm (rib)

Nuclear heat rate of structure materials:7.5W/cm^3

Temp. of He coolant gas: 20 C (Inlet), 100 C (Outlet)

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Mass of He Coolant Gas in the Path for Coolant Gas

Parameters	Unit	Disk Specimen Tritium Release and Compatibility Tests	Pellet and Pebble Tritium Release Tests	notes
Vol. of Capsule Wall in the Irradiation Region	cm^3	23.9	53.47	
Heat Generation per Unit Volume	W/cm^3	7.5	7.5	
Total Heat Generation	W	179.25	401.05	
Mass of Coolant He Gas	kg/h	1.55	3.48	
Volume Flow Rate (100°C)	m^3/h	10.58	23.68	20°C at Inlet
Cross Section of Flow Path	cm^2	2.85	7.48	
Flow Rate in the Path	m/s	10.31	8.79	

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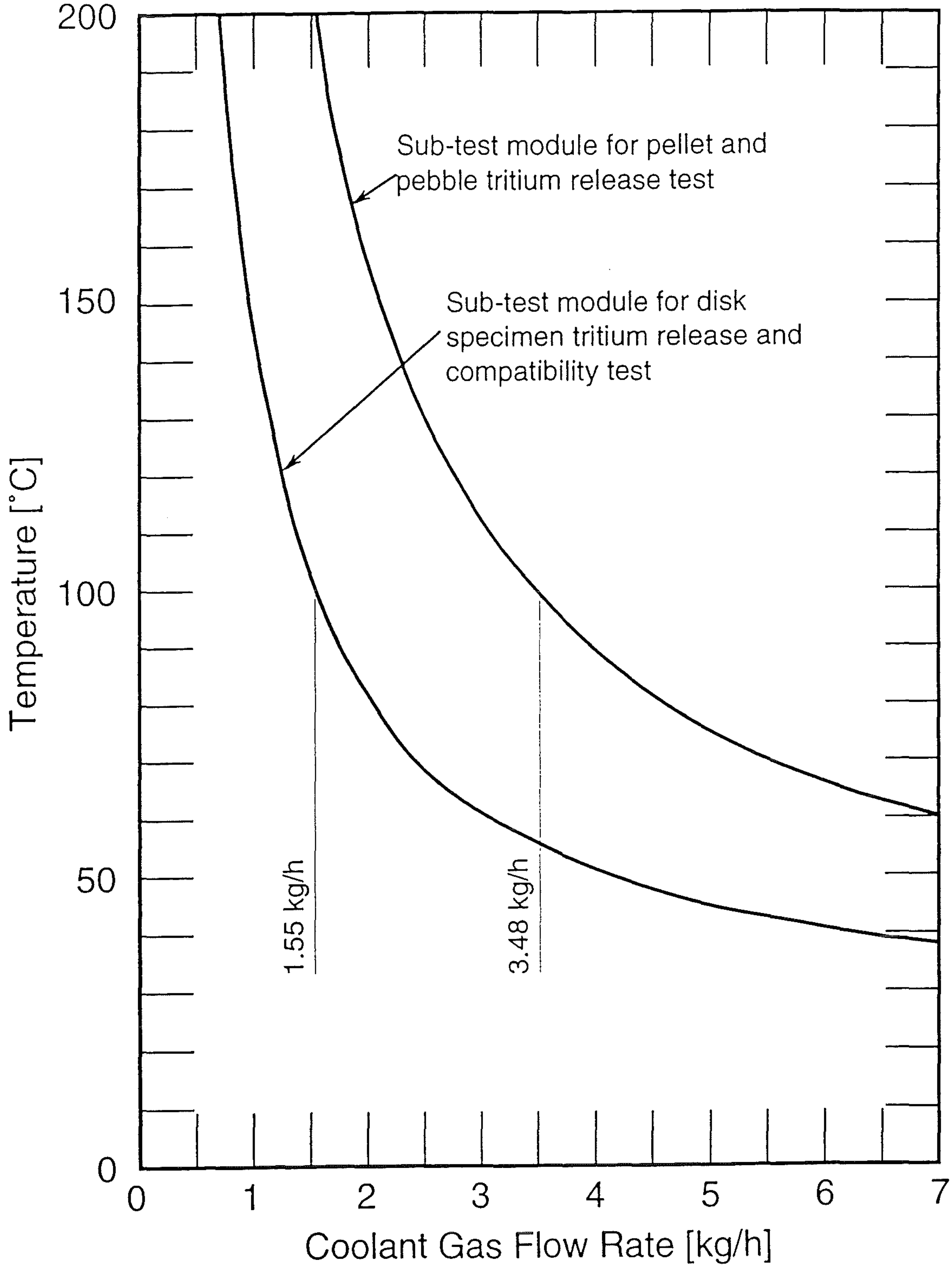
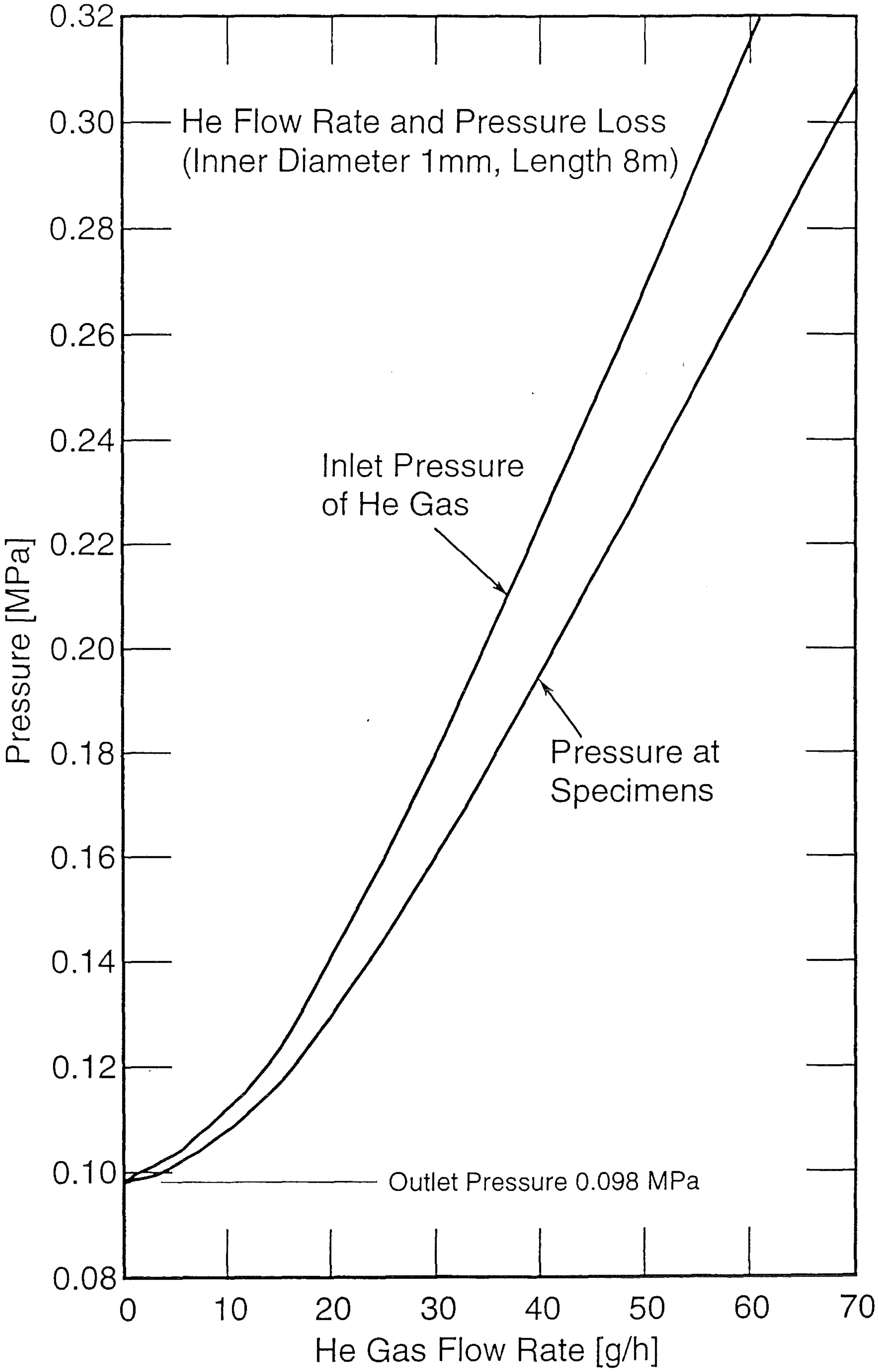

200
150
100
50
0
Temperature [°C]
0
1
2
3
4
5
6
7
Coolant Gas Flow Rate [kg/h]
Sub-test module for pellet and pebble tritium release test
Sub-test module for disk specimen tritium release and compatibility test
1.55 kg/h
3.48 kg/h


He Flow Rate and Pressure Loss
(Inner Diameter 1mm, Length 8m)
Inlet Pressure
of He Gas
Pressure at
Specimens
Outlet Pressure 0.098 MPa
Pressure [MPa]
0.32
0.30
0.28
0.26
0.24
0.22
0.20
0.18
0.16
0.14
0.12
0.10
0.08
0
10
20
30
40
50
60
70
He Gas Flow Rate [g/h]

Temperature Control of Irradiation Capsule

Temperature of outer capsule is maintained at about 100 C by He coolant flowing in the sub-test module.

Temperatures of inner capsule and specimen are controlled by changing He gap gas flow rate.
*Disk specimen tritium release capsules: 3 temperature levels (400, 600, 800 C)
*Pellet/pebble specimen tritium release capsule: 400C (specimen/pebble bed surface)

When one beam stops (i.e., 1/2 neutron flux), the specimen temperature can be maintained by decreasing gap gas flow rate.

[Calculation conditions]
Specimens (disk, pellet, pebble):Li_2O, Li_2TiO_3, Li_2ZrO_3, Li_4SiO_4, $LiAlO_2$, etc.
Structural materials: Low activation ferritic steel (thickness: 0.5 mm)
Nuclear heat rate of structure materials and specimens:7.5W/cm^3,
(75W/cm^3 for specimens)
Temp. of He coolant gas: 20 C (Inlet)

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Heat Generation and He Mass Flow in the Gap Gas Path

Heat generation (W/cm^3)	Temp. of Specimen	Parameters	Unit	Disk Specimen Tritium Release Test	Compatibility Test	Pebble Specimen Tritium Release Test	Pellet Specimen Tritium Release Test
		Vol. of Specimens	cm^3	0.785	0.393	0.472	1.571
		Vol. of Capsule Walls	cm^3	0.725	0.725	0.825	1.125
		Cross Section of Coolant Path	mm^2	24.75	24.75	26.66	36.06
7.5 W/cm^3		Heat Generation of Specimens	W	5.891	2.945	3.338	11.783
		Heat Generaton of Capsule Walls	W	5.438	5.438	6.185	8.438
		Total Heat Generation	W	11.329	8.383	9.523	20.221
	400°C	Mass Flow of Gap Gas	g/h	20.66	15.29	17.37	36.88
		Flow Rate in the Path	m/s	2.85	2.11	1.28	3.49
	800°C	Mass Flow of Gap Gas	g/h	10.07	7.45	8.46	17.97
		Flow Rate in the Path	m/s	2.22	1.64	1.73	2.71
75 W/cm^3		Heat Generation of Specimens	W	58.91	29.45	33.38	117.83
		Total Heat Generation	W	64.35	34.89	39.57	126.27
	400°C	Mass Flow of Gap Gas	g/h	117.4	63.64	71.18	230.32
		Flow Rate in the Path	m/s	16.2	8.78	9.24	21.81
	800°C	Mass Flow of Gap Gas	g/h	57.19	31.01	35.17	112.23
		Flow Rate in the Path	m/s	12.58	6.82	7.18	16.94

Heat generation from Capsule Wall is assumed to be 7.5 w/cm^3.

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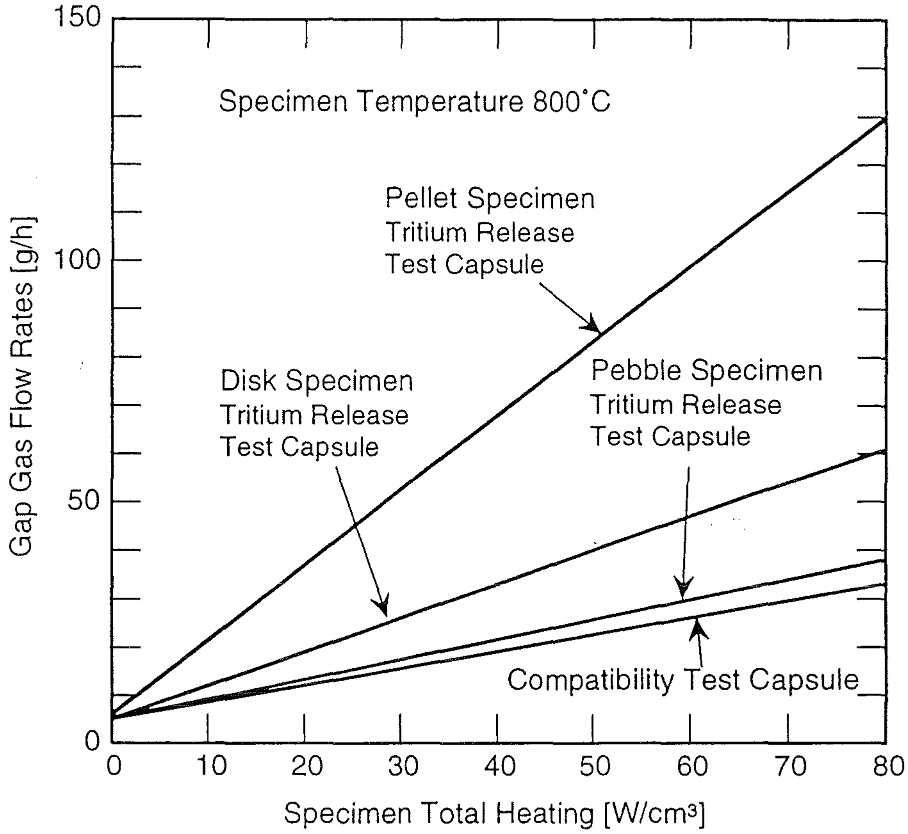
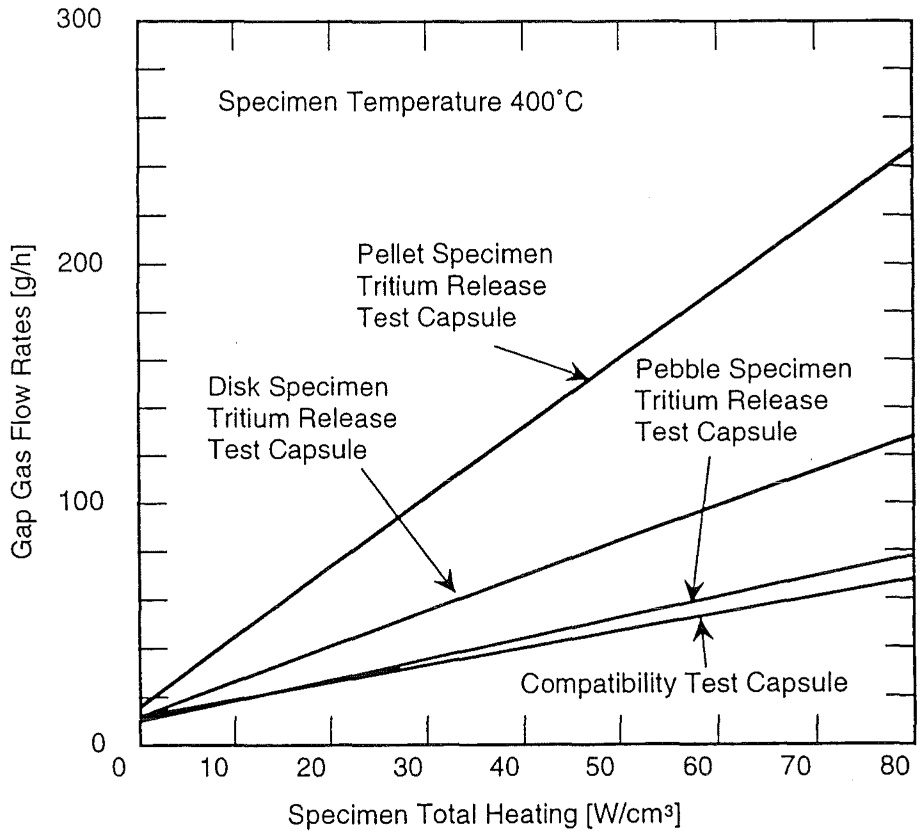

Specimen Temperature 800˚C
Pellet Specimen
Tritium Release
Test Capsule
Disk Specimen
Tritium Release
Test Capsule
Pebble Specimen
Tritium Release
Test Capsule
Compatibility Test Capsule
Gap Gas Flow Rates [g/h]
150
100
50
0
0
10
20
30
40
50
60
70
80
Specimen Total Heating [W/cm³]


Specimen Temperature 400˚C
Pellet Specimen
Tritium Release
Test Capsule
Disk Specimen
Tritium Release
Test Capsule
Pebble Specimen
Tritium Release
Test Capsule
Compatibility Test Capsule
Gap Gas Flow Rates [g/h]
300
200
100
0
0
10
20
30
40
50
60
70
80
Specimen Total Heating [W/cm³]

Summary

Improvement of conceptual design of tritium release test module was carried out after Feb., 1997.

(1) Irradiation capsule design was changed for better temperature control of specimens by using gas gap surrounding inner capsule.

(2) Sub-test module design was changed to contain many irradiation capsules with small He coolant paths. This allows to irradiate specimens of the required test matrices for two different neutron fluence levels simultaneously.

(3) Change of sub-test module and irradiation capsule designs can lead improvement of fablicability, reliability and irradiation durability of tritium release test module.

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Universal Robot System (URS)

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The concept of a remote controled system to perform remote handling & maintenance operations in the Test Cells and in the Access Cell is confirmed. The basic features of the system are reminded:

- four degrees of freedom plus the capability for slight tilting;

- capability of lifting and transporting up to 30 t with precise position control (1 mm);

- capability to transport and deploy various multiple-purpose articulated robot systems for all foreseen remote handling operations.

These features shall be conferred to a gantry Cartesian system, whose axonometric view is given in fig.1. This gantry machine supports a trolley with 4-axis movements (x-y-z axis + rotational w axis around the z vertical direction), which houses a telescopic gripper capable of handling both components of the Test Cells (target backwall, target assembly, quench tank) and of the Test Cell Cover (VTA's, shield plugs, removable cover).

Recent activity on URS has been devoted mainly to check the feasibility of the solution proposed, through preliminary studies of layout, definition of the equipment needed for RH operations, and definition of the related. For this last point, use has been made of computer analysis for simulation of the operations of the equipment and of the flow of the components inside the cells.

The following 14 figures represent a review on the present status of the design activities.

1) **High precision crane on its rails (Universal robot)**

2) **Removable Gripper/bag tool**

3) **Transfer device**

4) **Access to the Test Cell**

5) **Plug modules on a temporary storage**

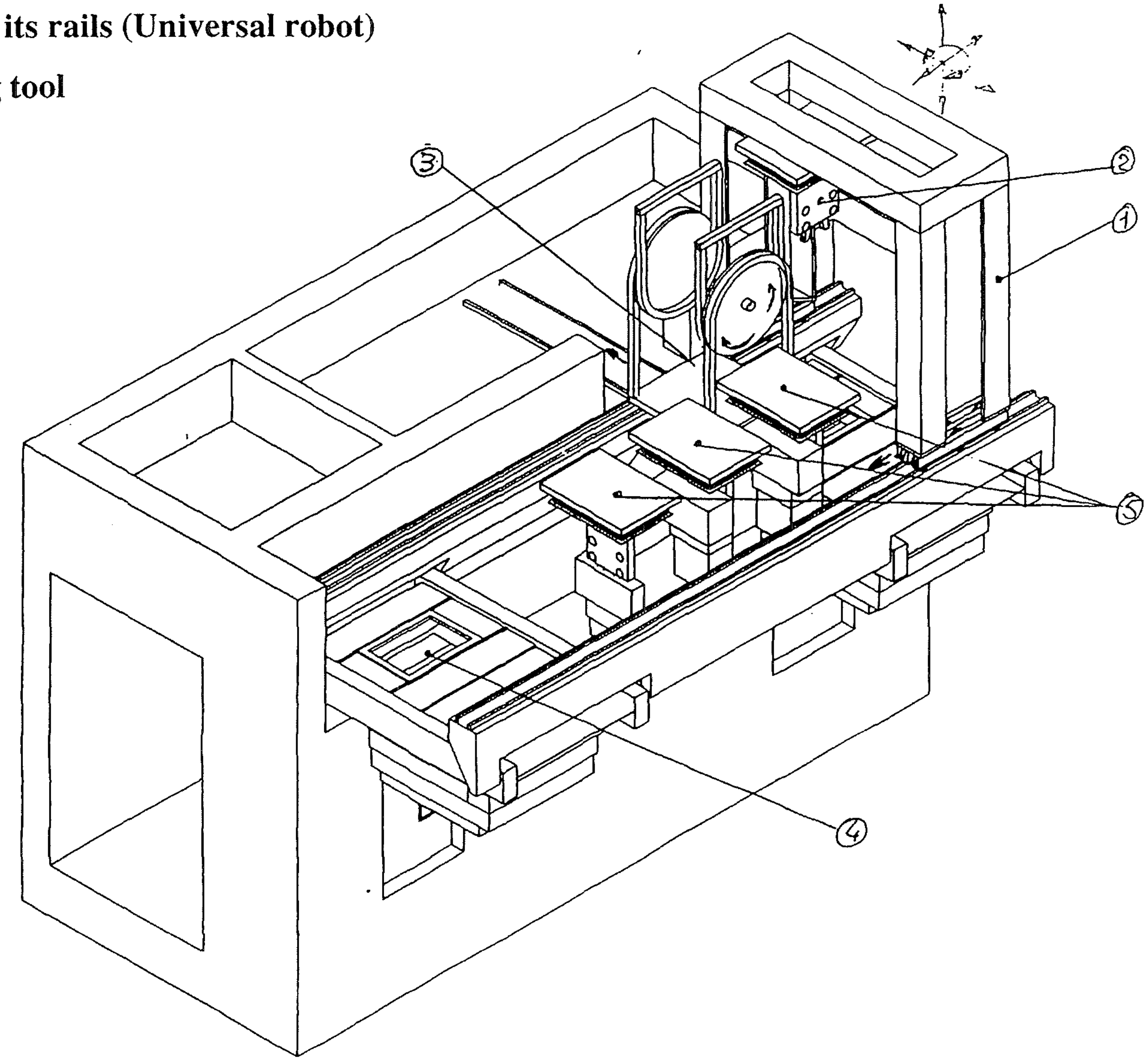


Fig. 1: Dedicated universal robot system (example) for the access cell.

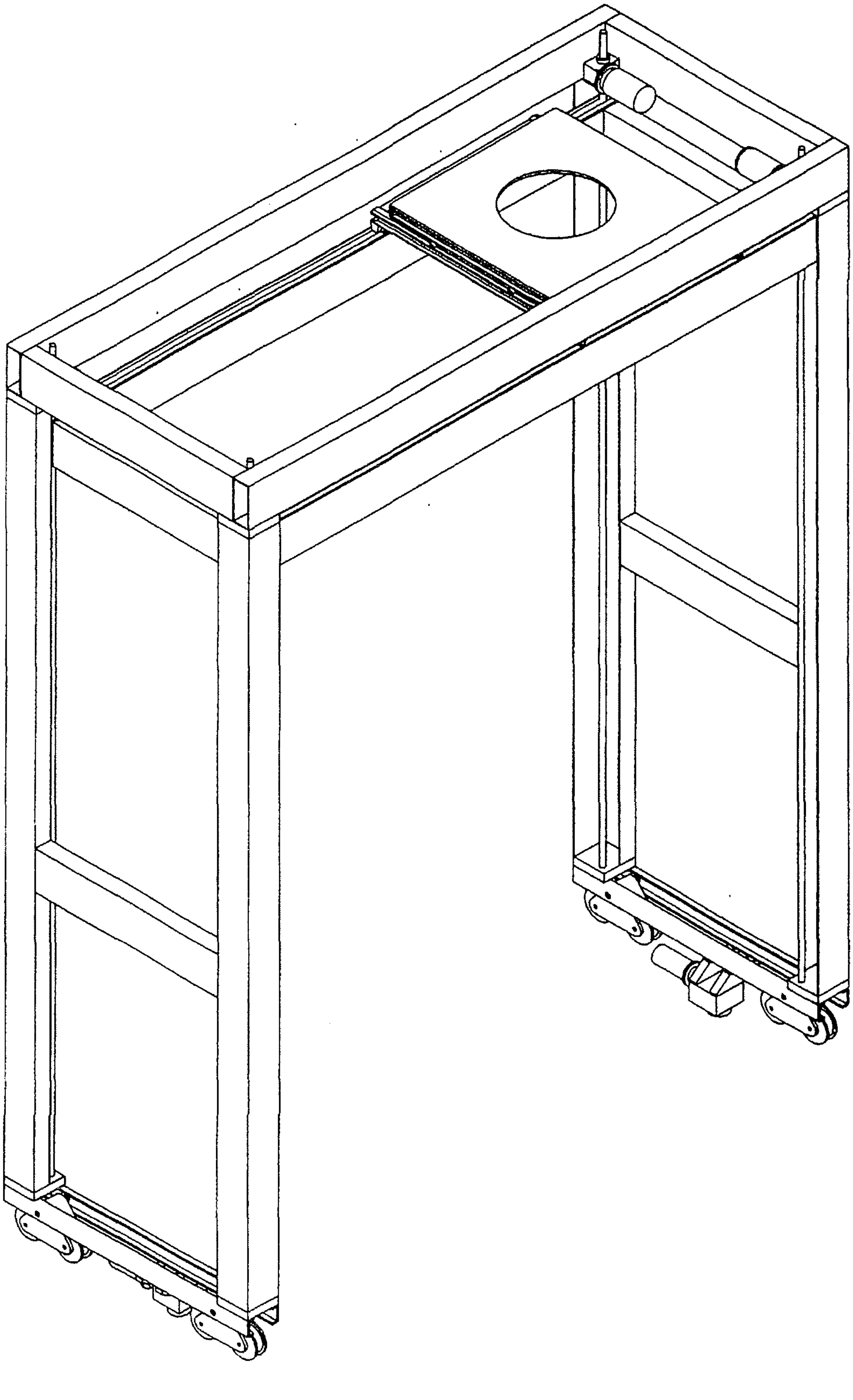


Fig. 2: Axonometric view of the Universal robot system (without handling tools).

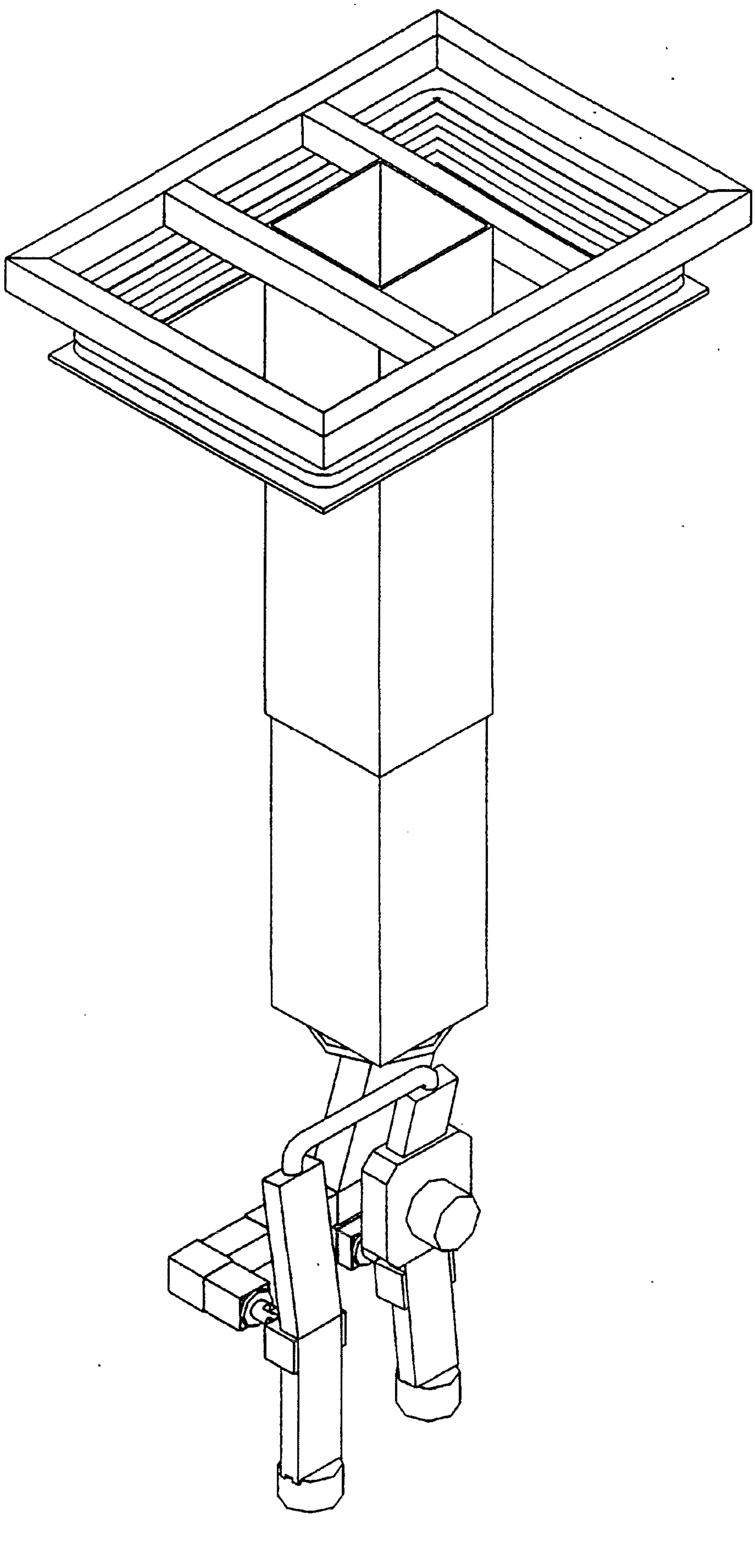


Fig. 3: telescopic tool of the Universal robot system.

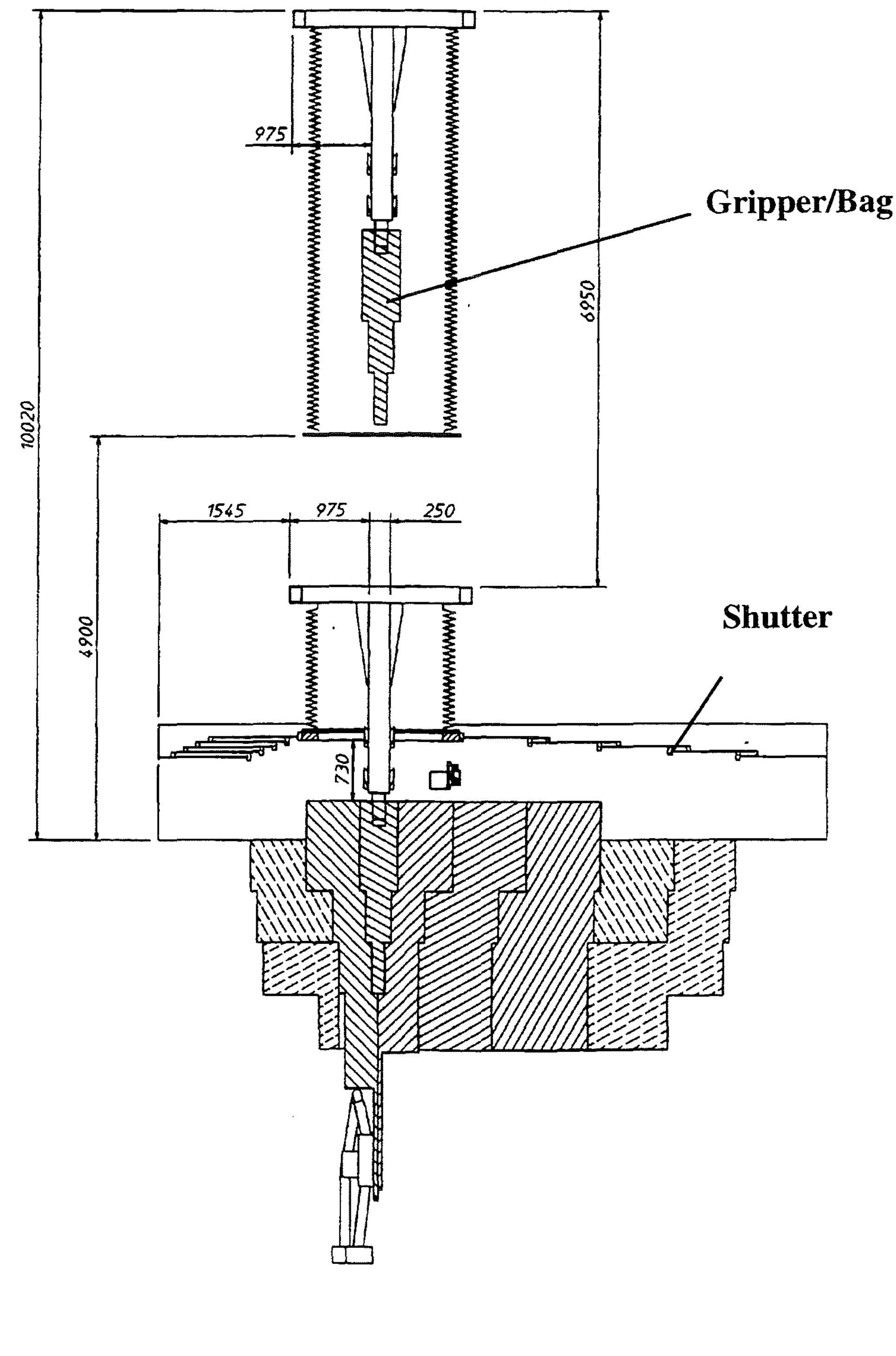


Fig. 4: Remote handling procedure; Gripper/bag gripping a shield plug.

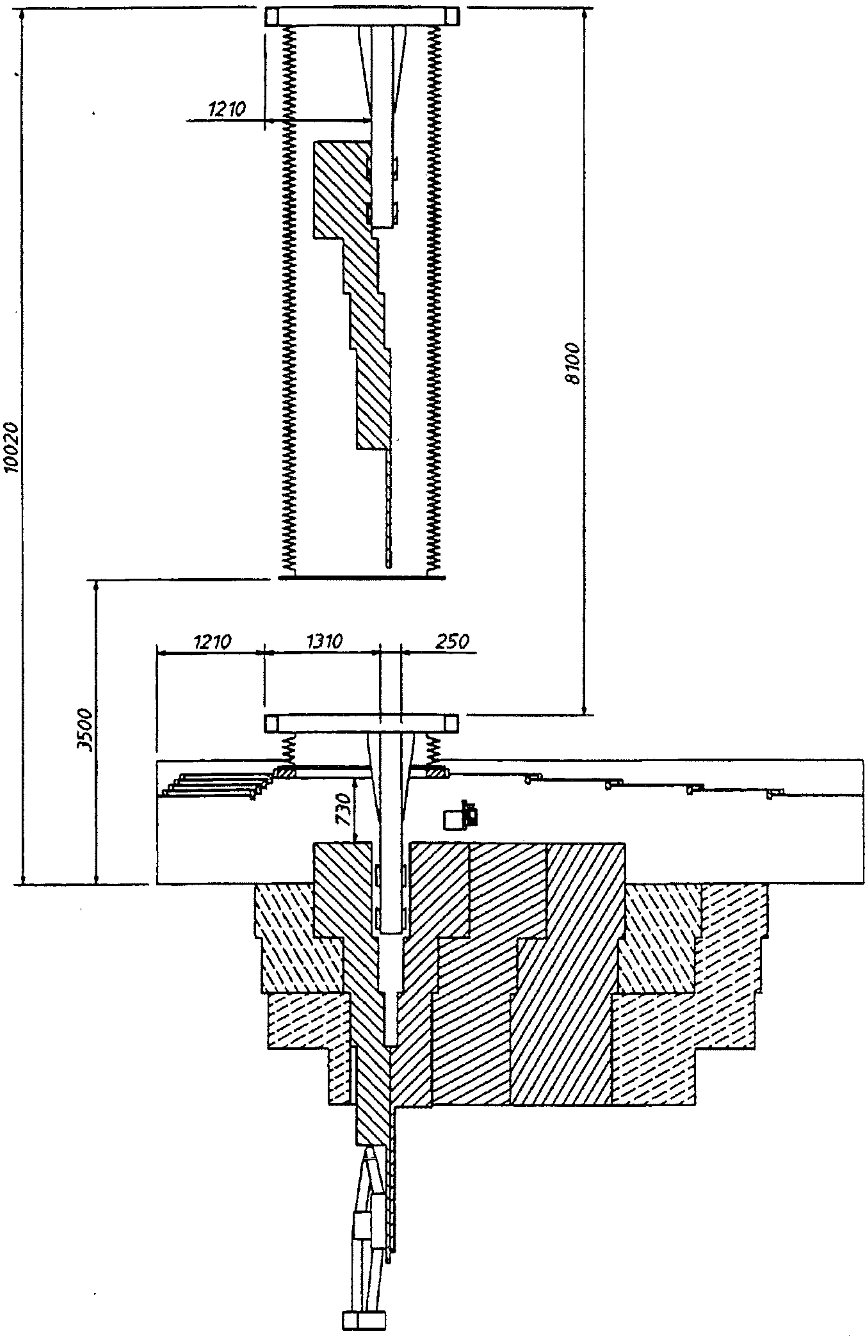


Fig. 5: Remote handling procedure; Gripper/bag gripping the high flux test module.

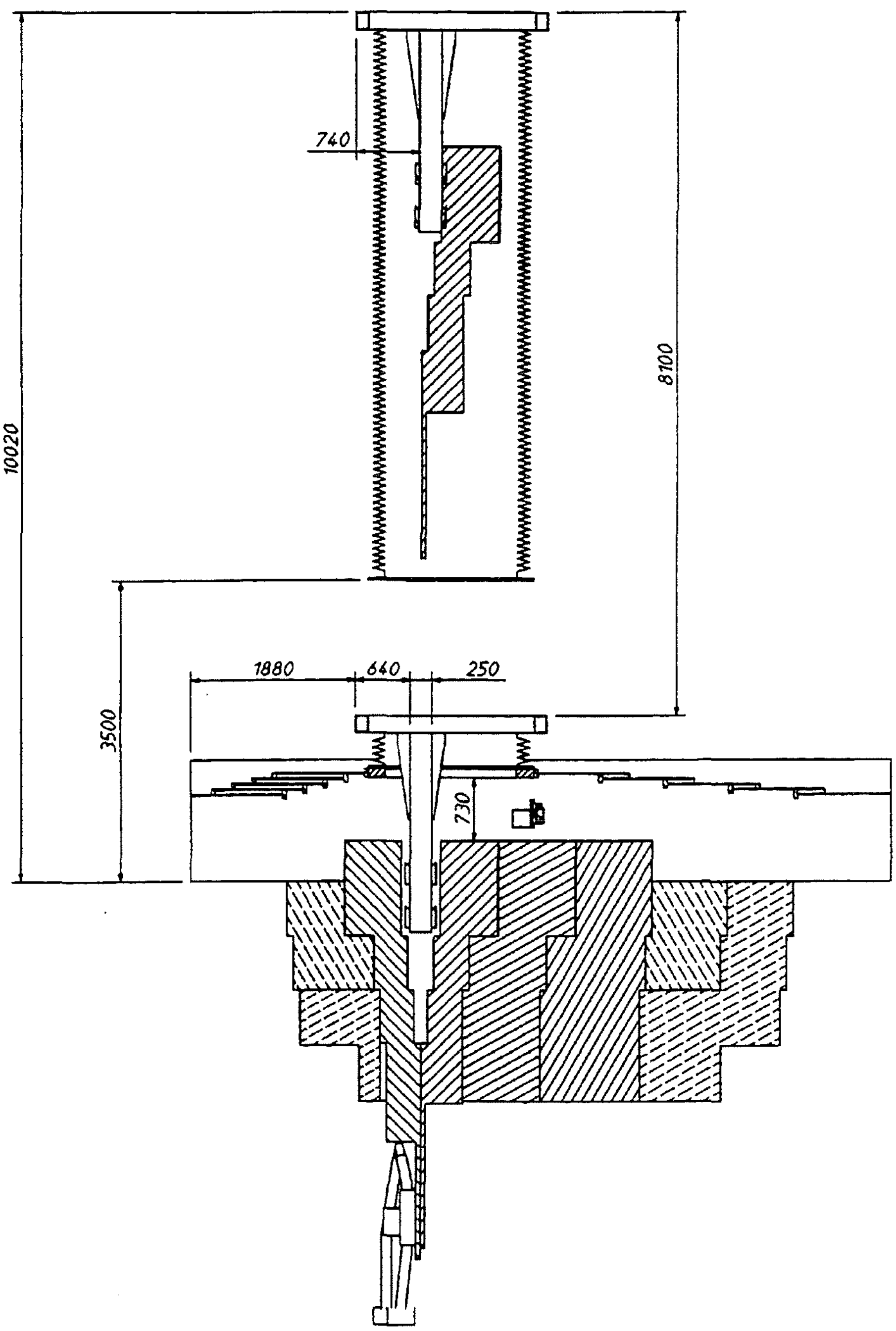


Fig. 6: Remote handling procedure; Gripper/bag gripping the VTA2.

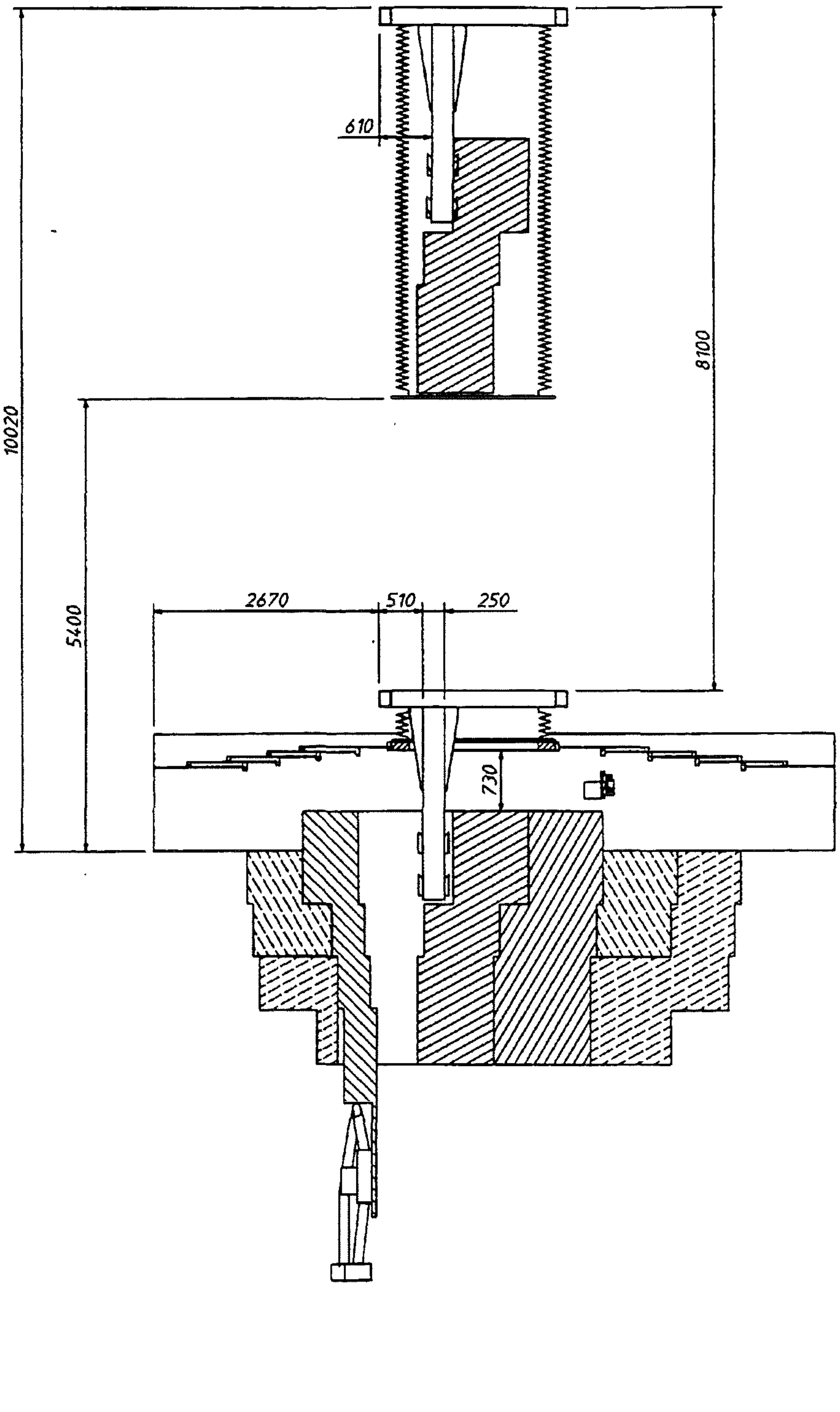


Fig. 7: Remote handling procedure; Gripper/bag gripping the Vertical Irradiation Tube (VIT) system.

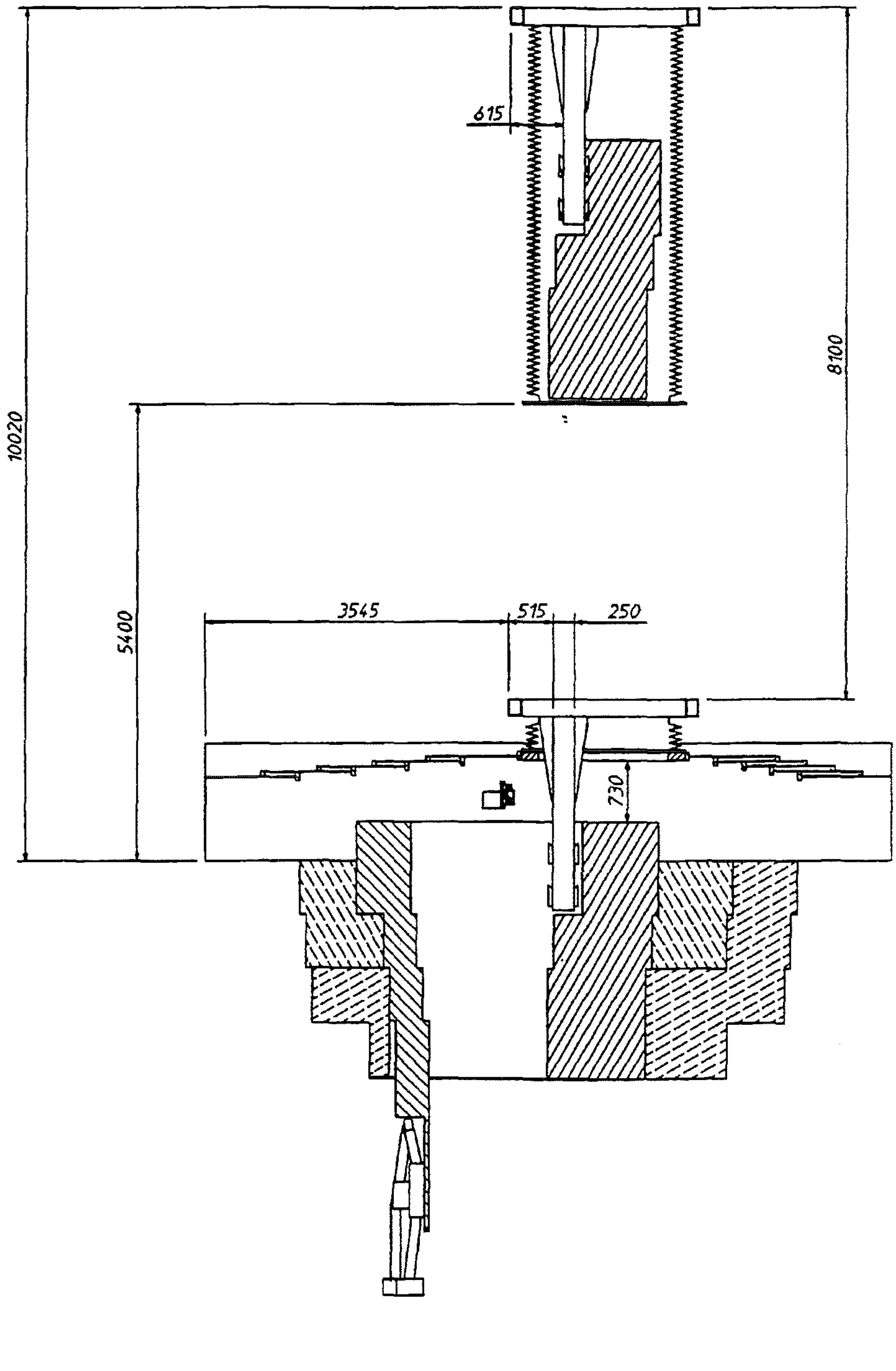


Fig. 8: Remote handling procedure; Gripper/bag gripping the test cell removable cover.

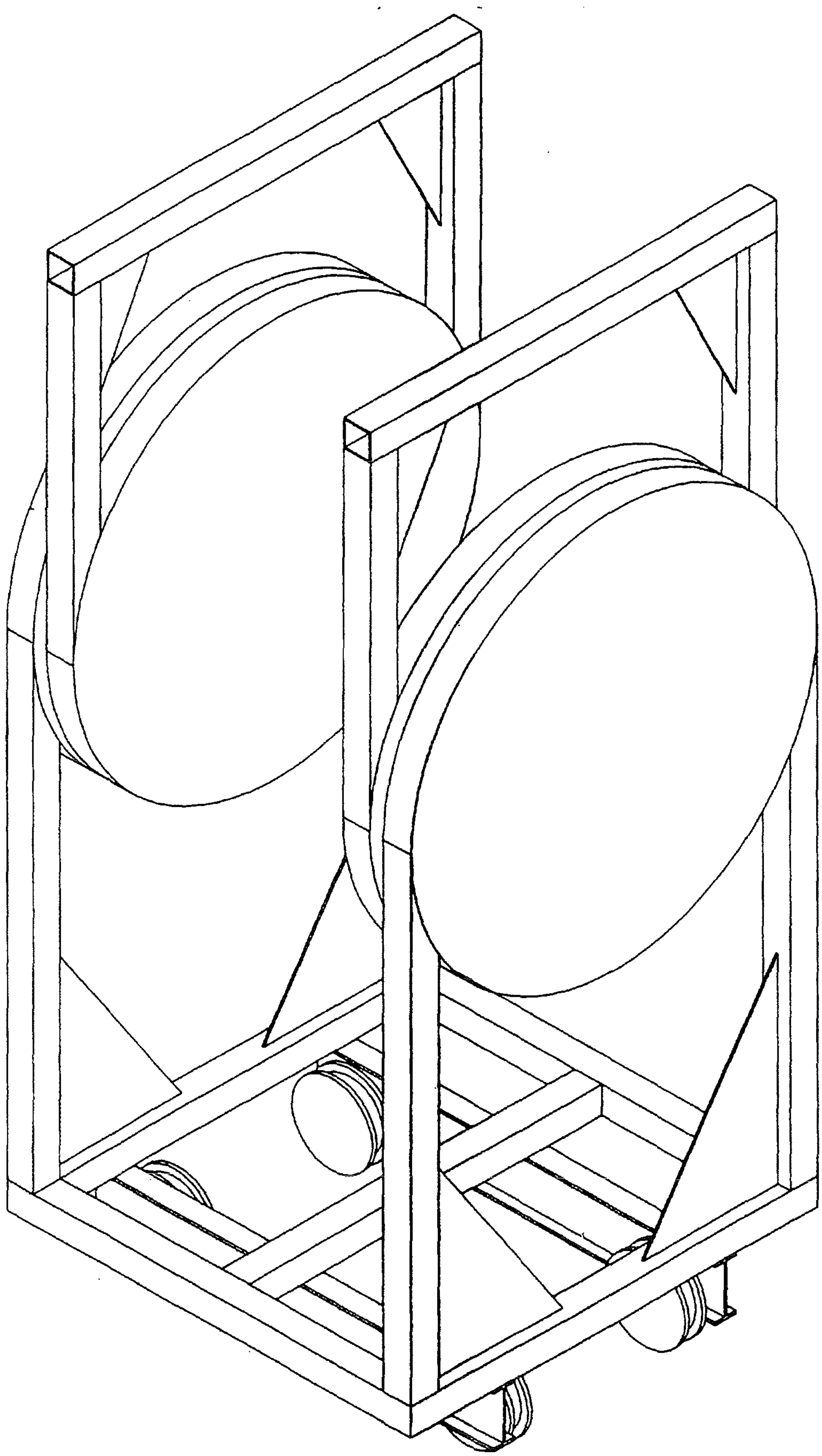


Fig. 9: Transfer device to move VTAs between Service Cell and Access Cell.

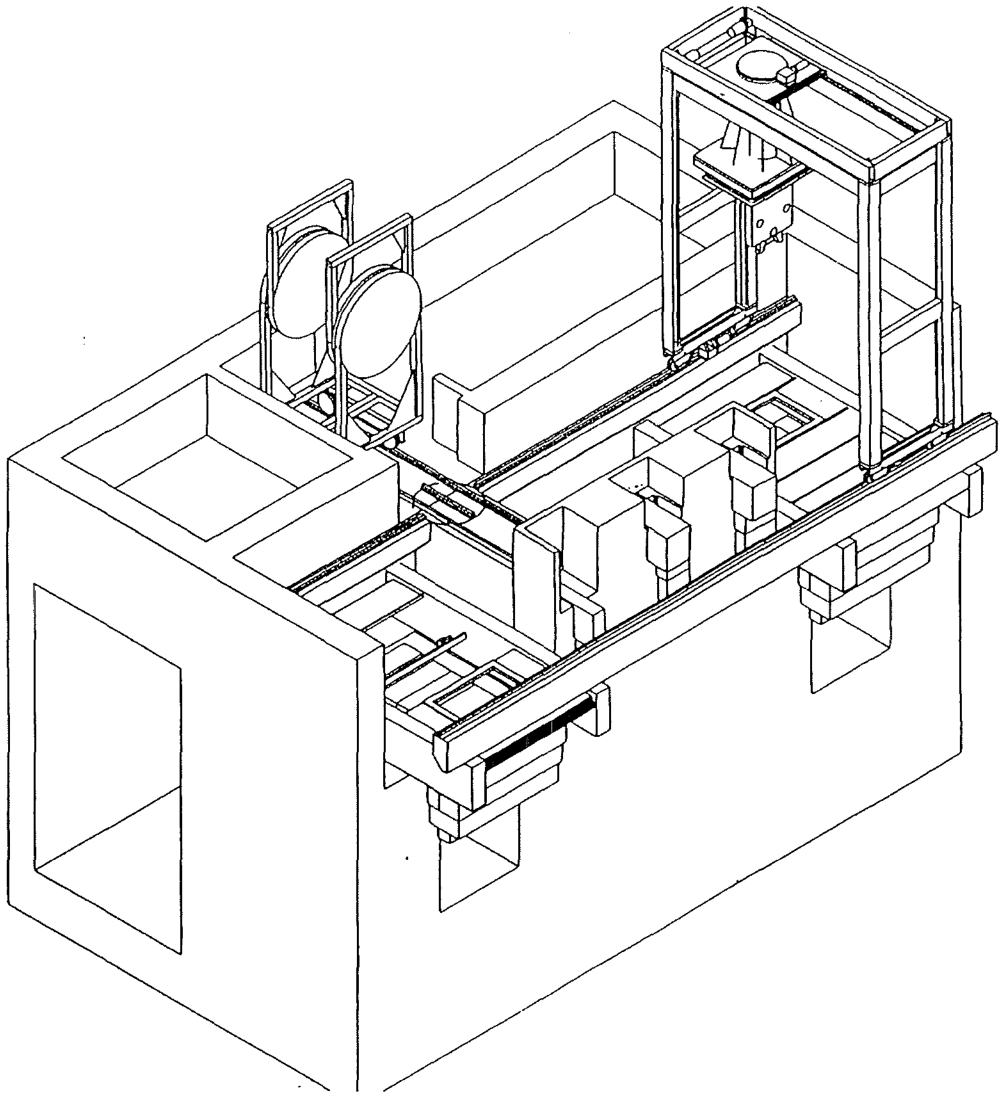


Fig. 10: Alternative configuration to fig. 1 of the remote handling devices.

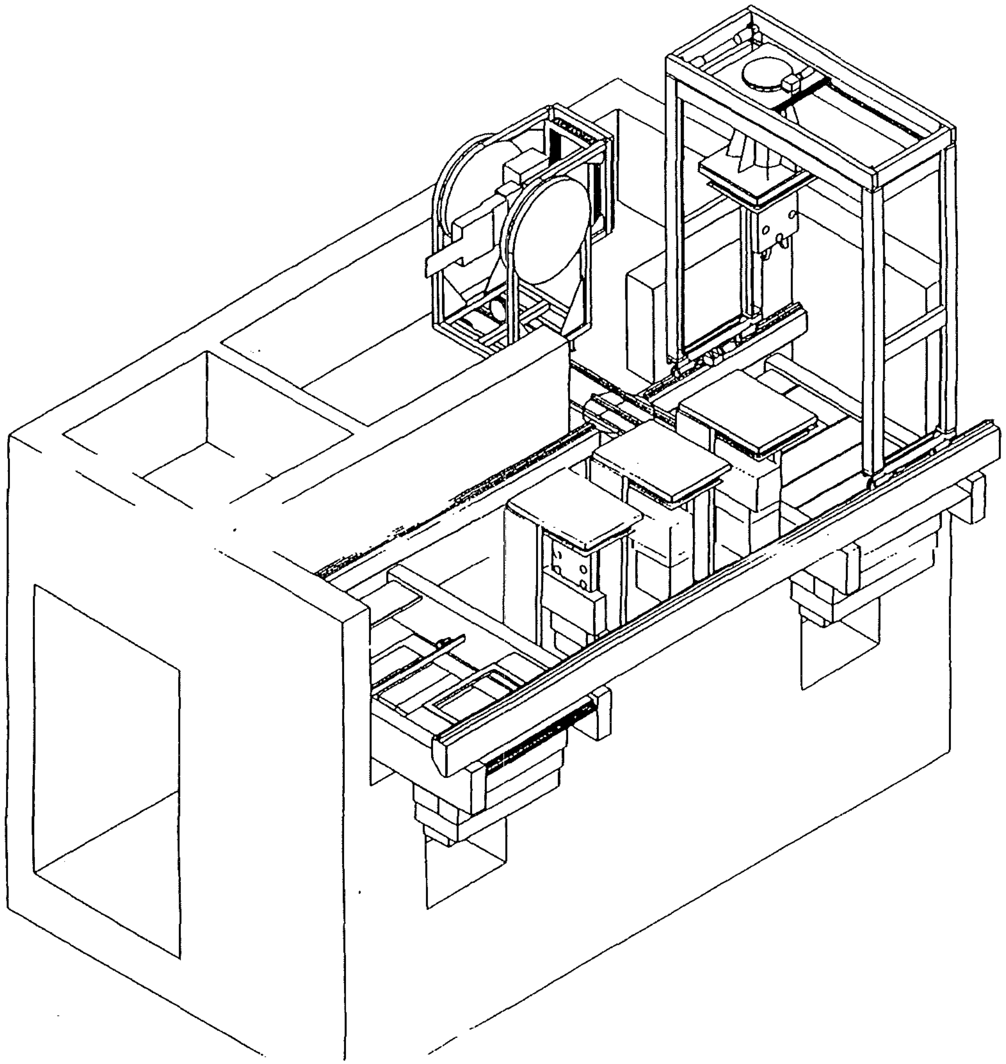


Fig. 11: Transfer device moving a Vertical Test Assembly between Access Cell and Service Cell.

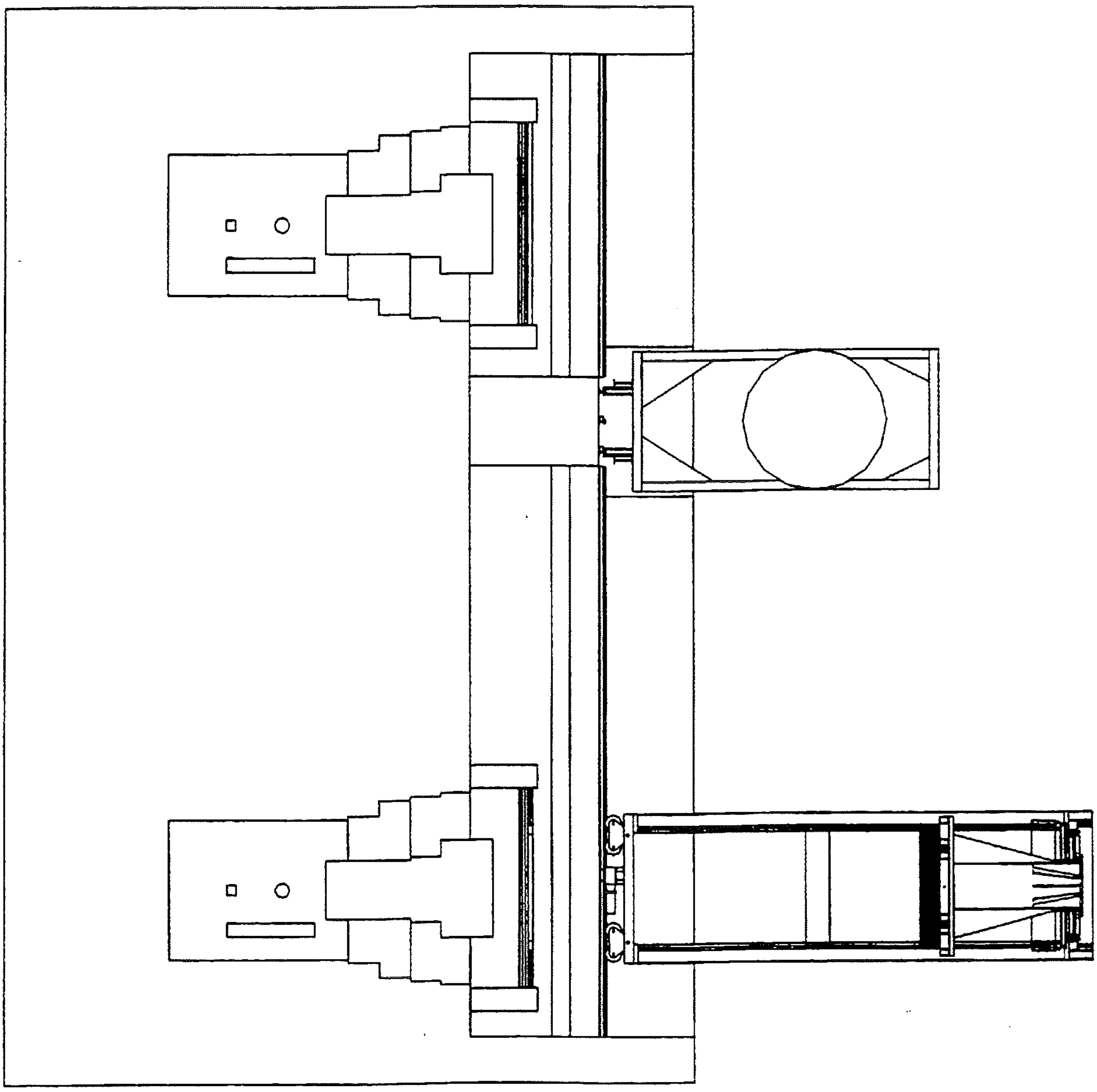


Figure not mentioned in FZK workshop proceedings

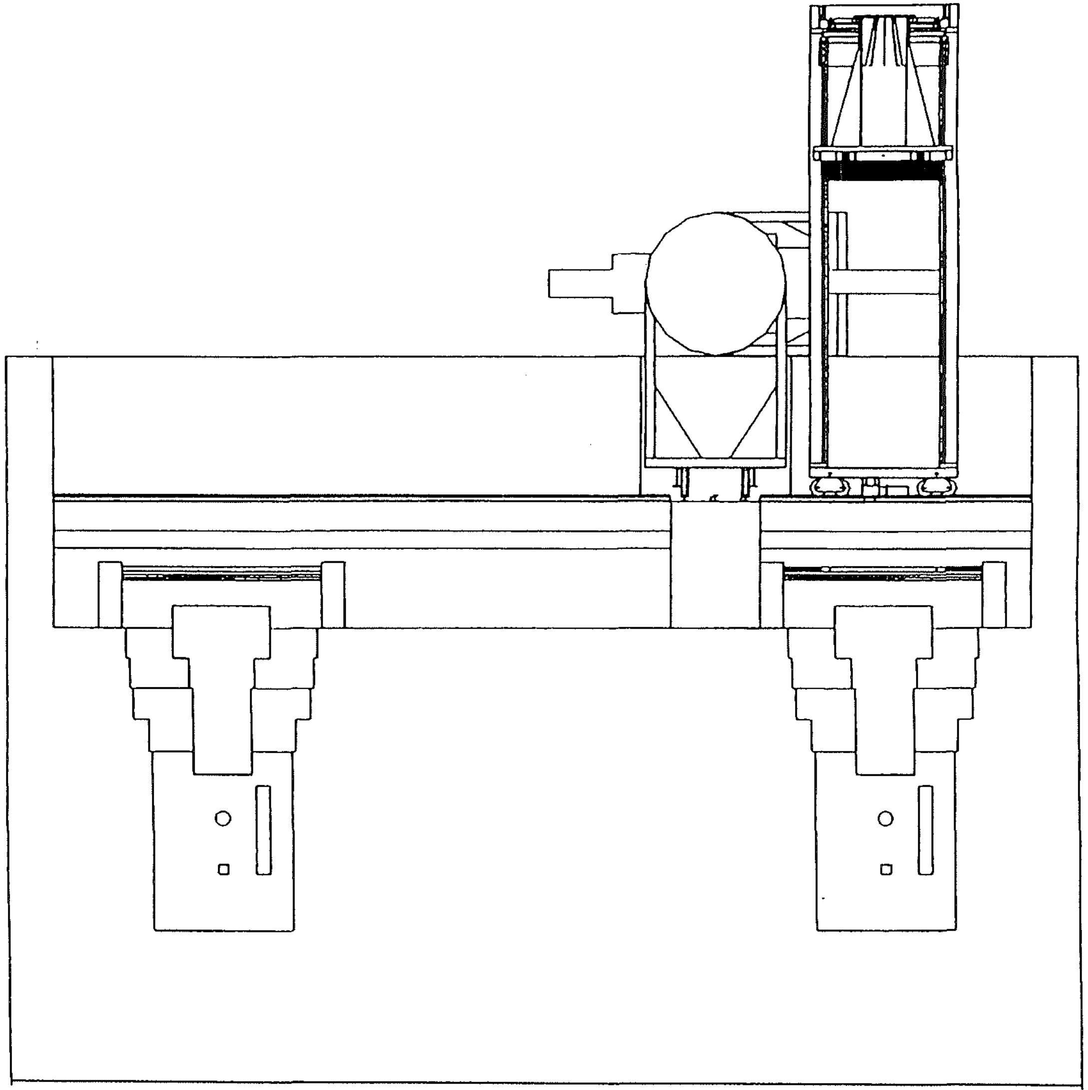


Fig. 12: Elevation view with the universal robot system, the transfer device (gripping a VTA) and the two test cells

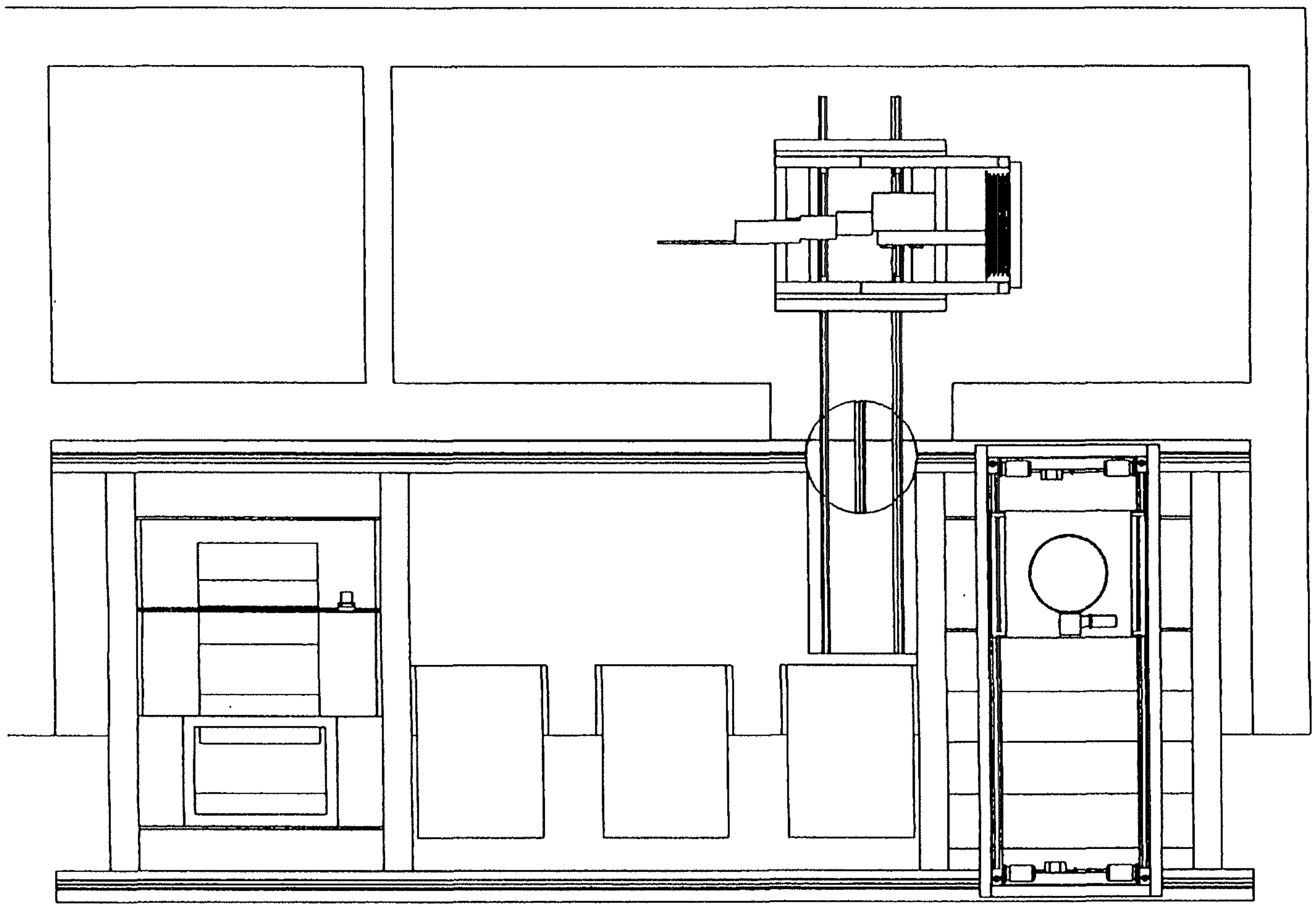


Fig. 13: Top view with the universal robot system and the transfer device (moving a VTA).

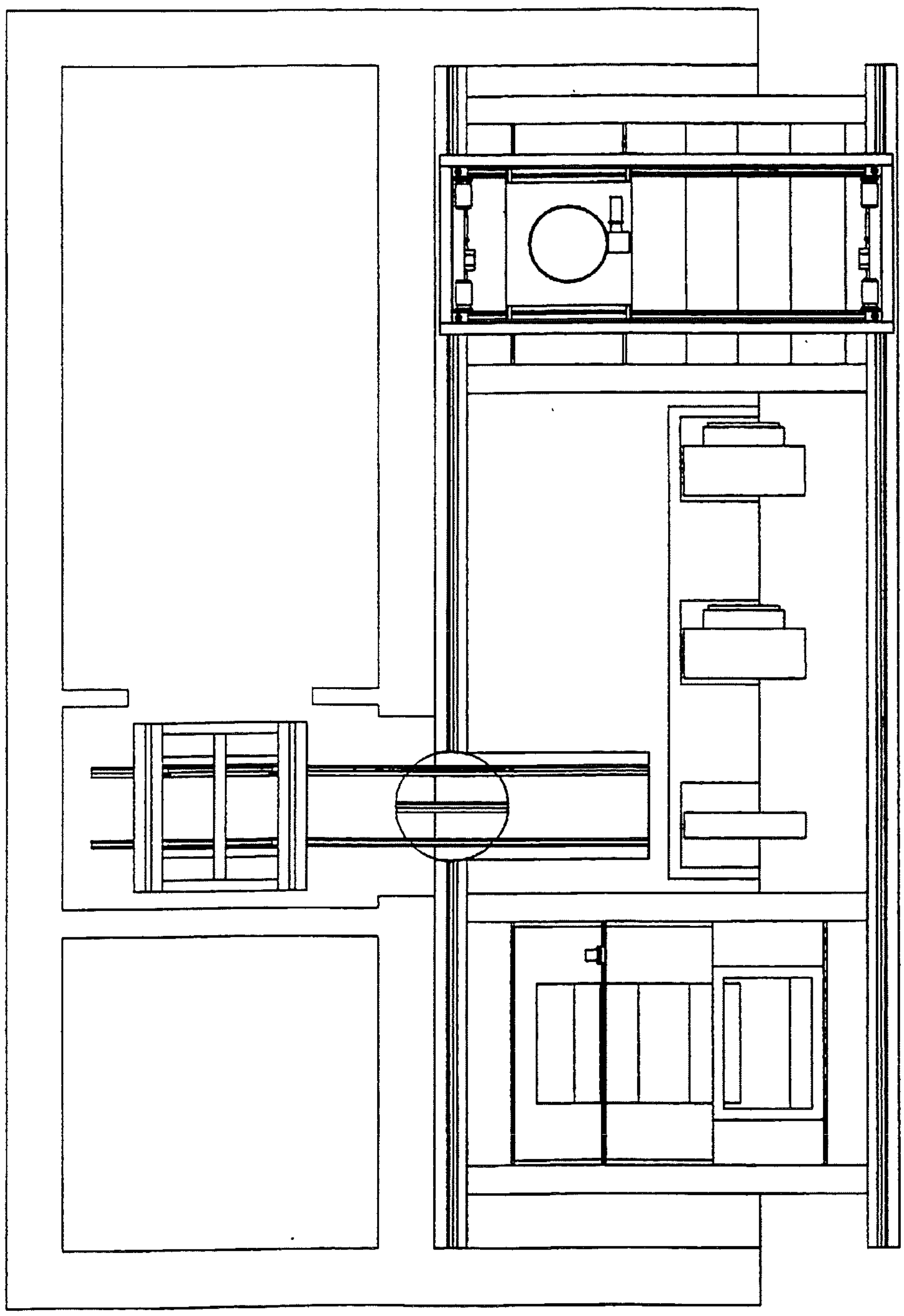


Figure not shown in the CDE Test facility workshop proceedings.

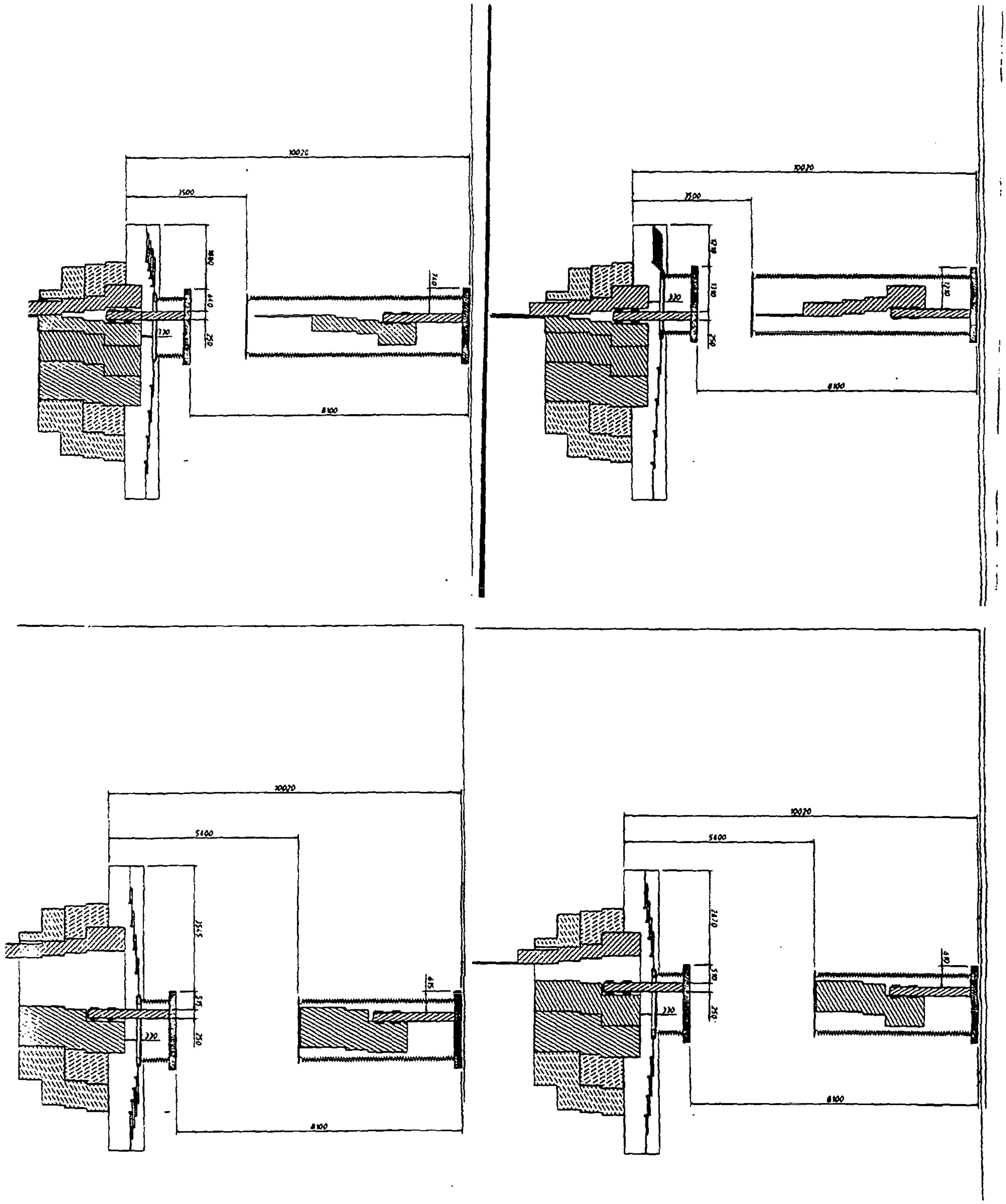


Figure not shown in the CDE test facility workshop proceedings

TA-SUPP

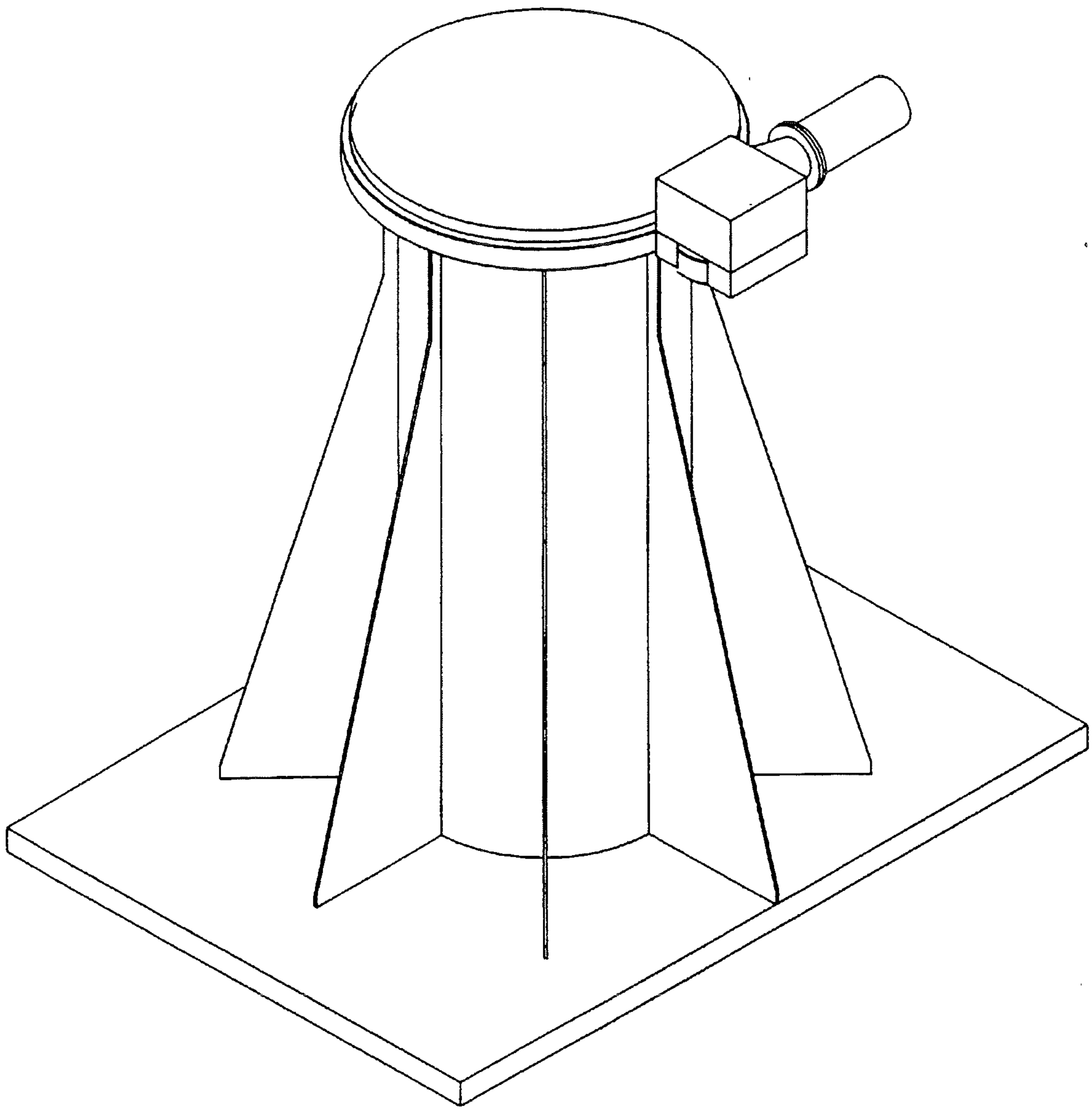


Figure not shown in the workshop proceedings.

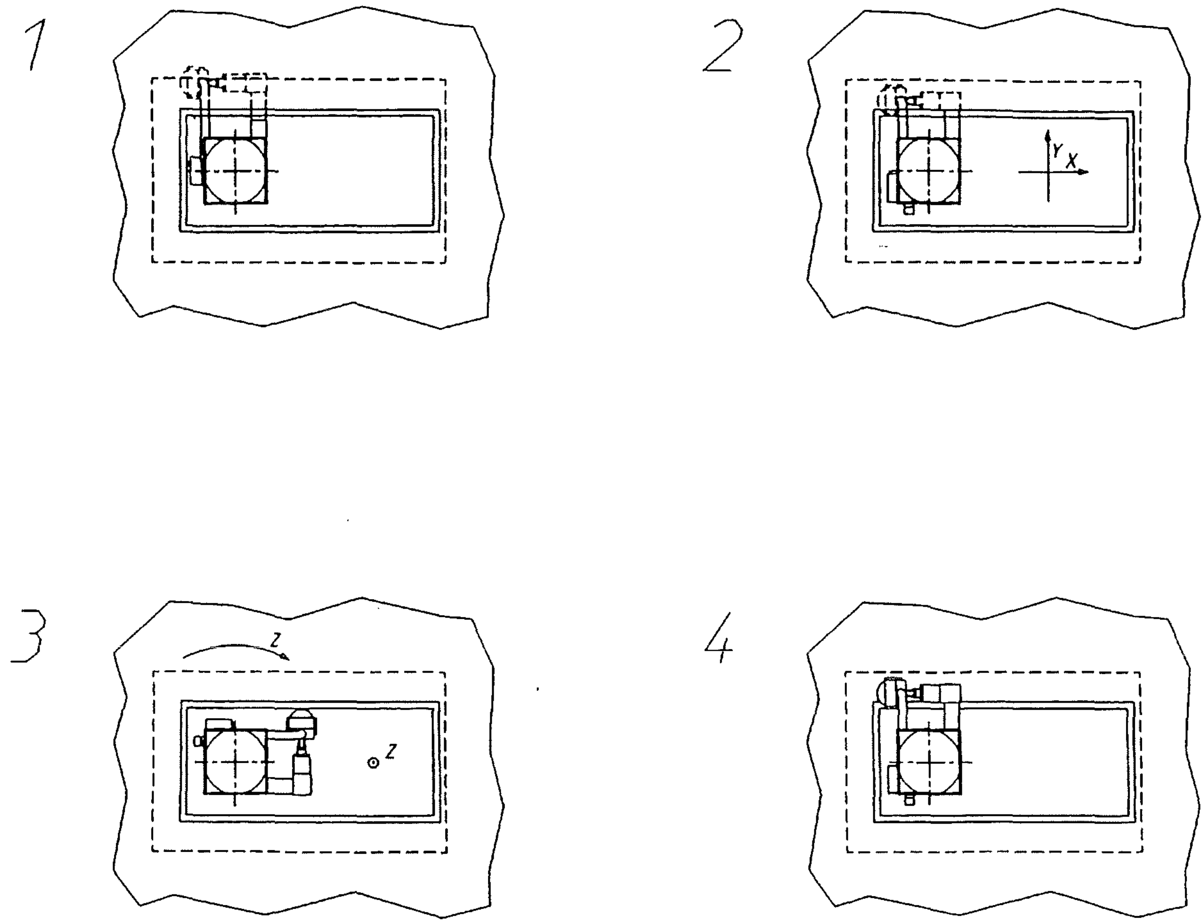


Fig. 14: Steps of operation for the removal of the Li-target

Forschungszentrum Karlsruhe
Technik und Umwelt

Institut für Neutronenphysik und
Reaktortechnik

Task 10: Source Term for D-Li Reaction

Development of Monte Carlo Source Term and Characterisation of IFMIF neutron source.

Paul P.H. Wilson
Forschungszentrum Karlsruhe, INR
Monday, July 7, 1997

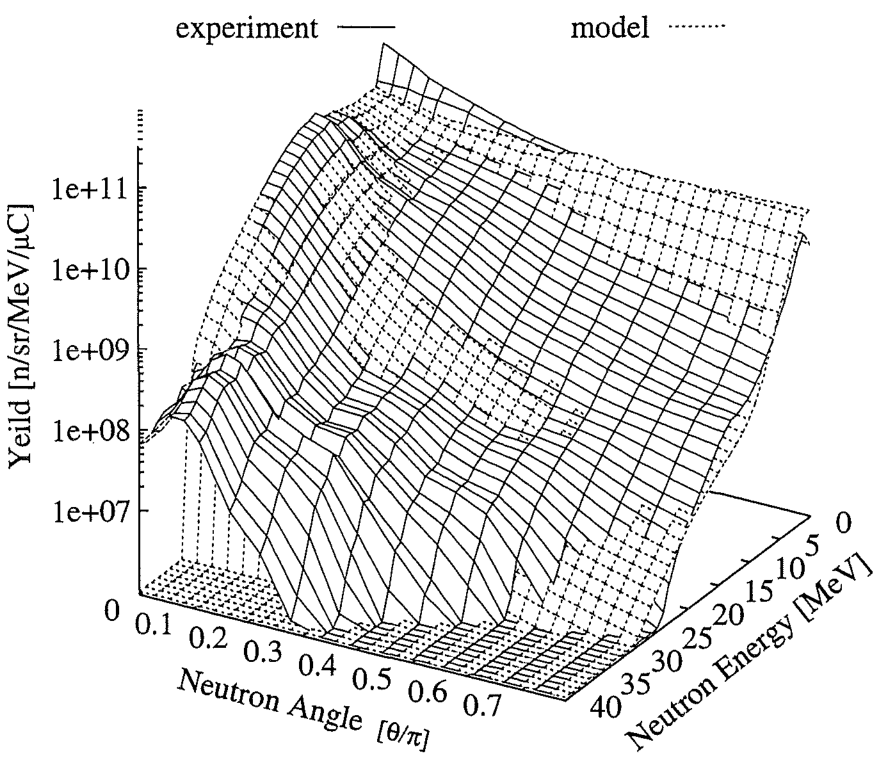
Li(d,n) Reaction Model

Successes

- Comparison with experimental data from Sugimoto (JAERI)
- "Stripping" and "Compound" reaction components
- Fitted to forward-direction and total neutron yield

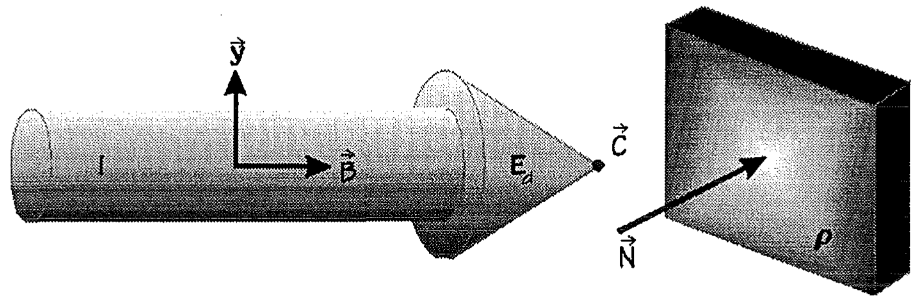
Problems and Improvements

- High energy/exothermic region
- Low energy neutron decay
- More reaction components
- No change in results expected - reduced uncertainty

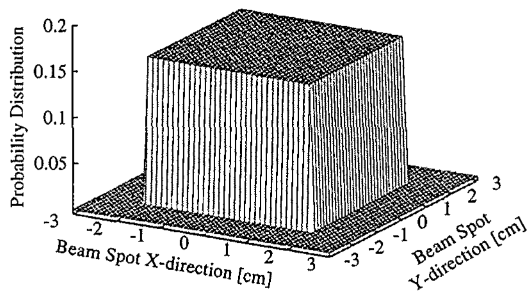


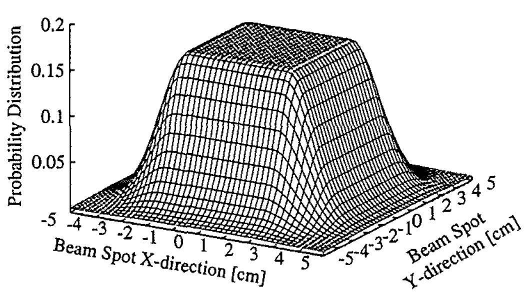
M^{c}DeLi Source Module

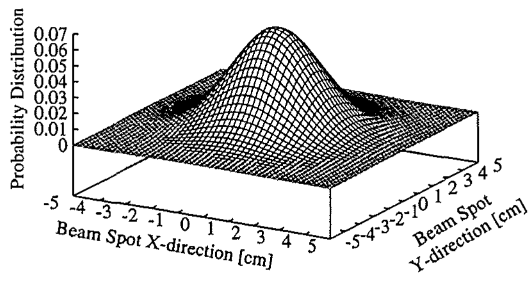
- high flexibility due to many input parameters

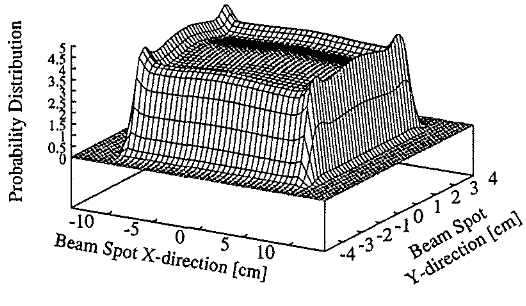


- variable beam profile
  (the following examples can be represented)









Comments on following figures

General

- X = beam direction
- Y = horizontal direction
- All from reference 1: E. Daum, et al., "Neutronics of the High Flux Test Region of the International Fusion Materials Irradiation Facility (IFMIF)," FZKA 5868, June 1997.

Figure 7

Notice flux gradients are at least 15 %/cm.

Figure 9

Notice DPA gradients are also at least 15 %/cm.

Figure 12

He/DPA production ratio is between 10 and 12 appm/DPA throughout most of high flux region.

Figure 15

H/DPA production ratio is between 40 and 45 appm/DPA throughout most of high flux region.

Figures 16 and 17

Notice lower average heating rate with more widely distributed non-uniform beam profile.

Figure 18

IFMIF wall load does not correspond to fusion wall load.

Figure 19

Notice available volume outside CDA reference design for High Flux Test Module.

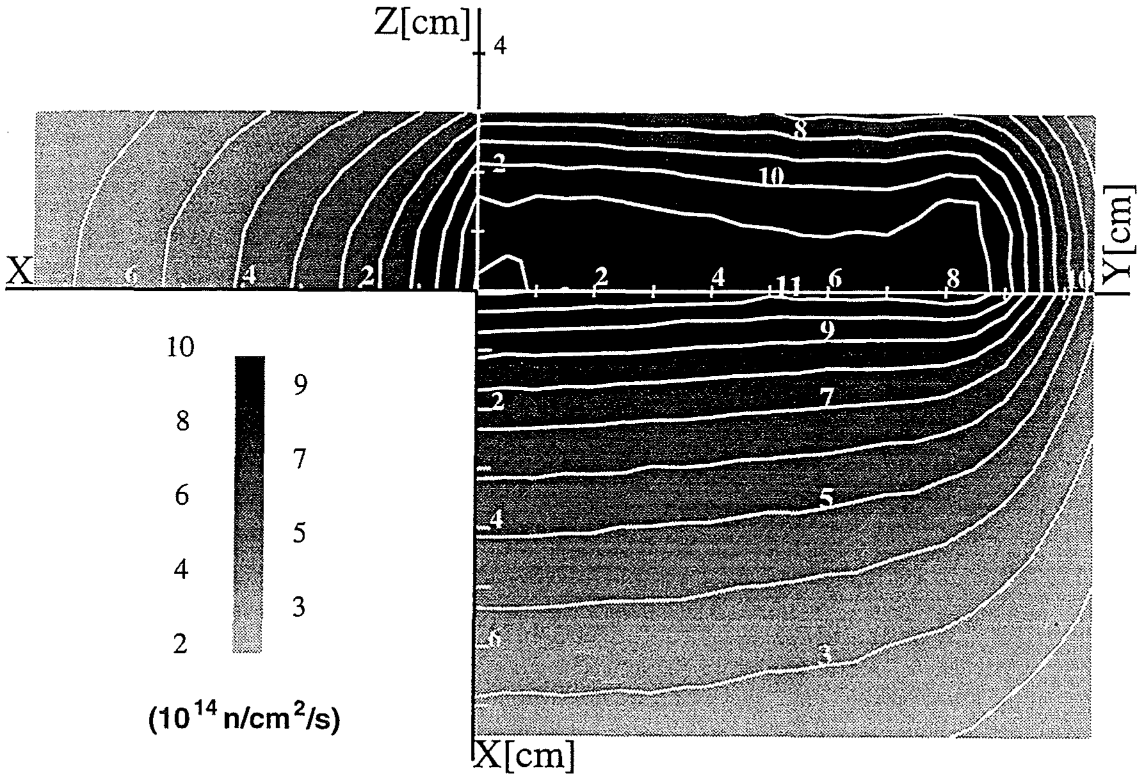


Figure 4. Neutron Flux
(Reference 1: Figure 5-83)

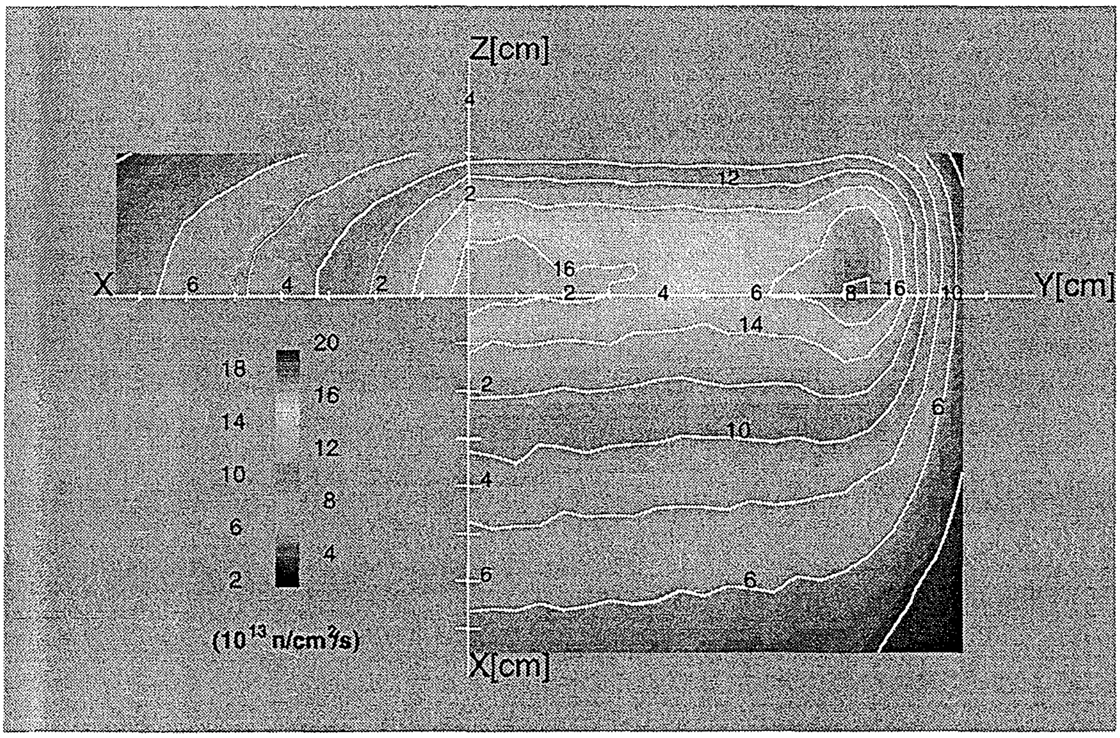


Figure 5. High Energy (> 14.6 MeV) Neutron Flux
(Reference 1: Figure 5-84)

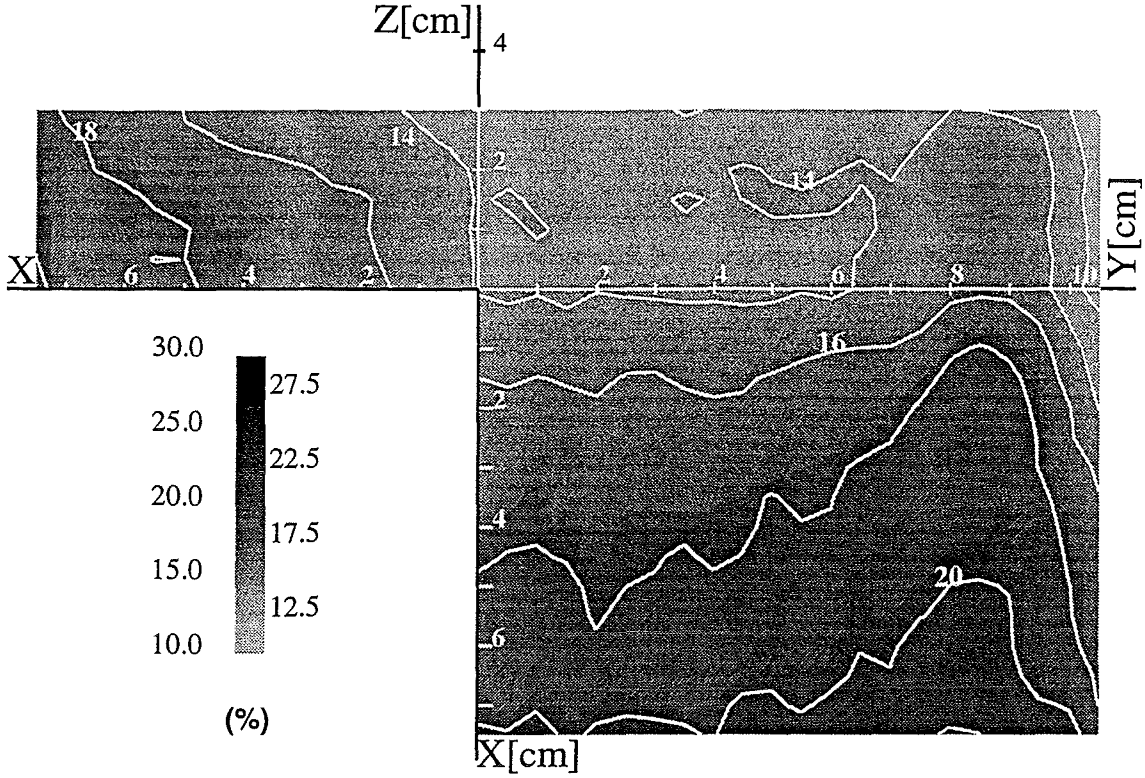


Figure 6. High Energy (>14.6 MeV) Neutron Flux Fraction

(Reference 1: Figure 5-85)

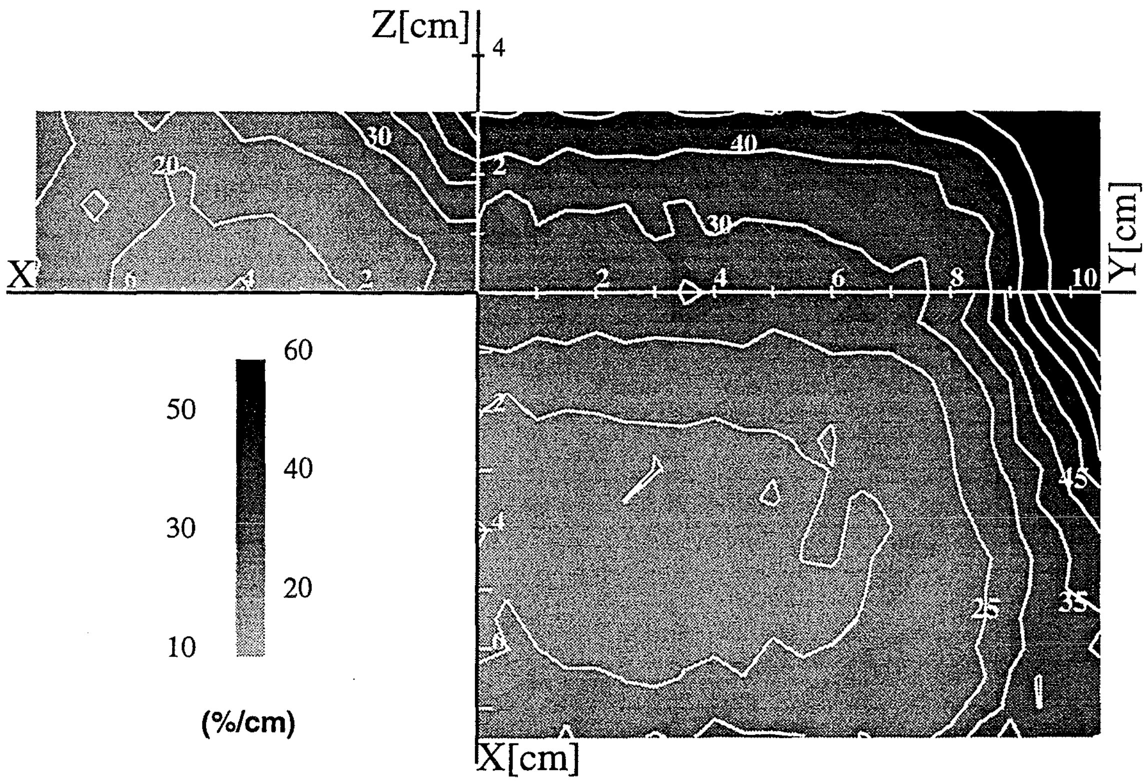


Figure 7. Neutron Flux Gradients

(Reference 1: Figure 5-86)

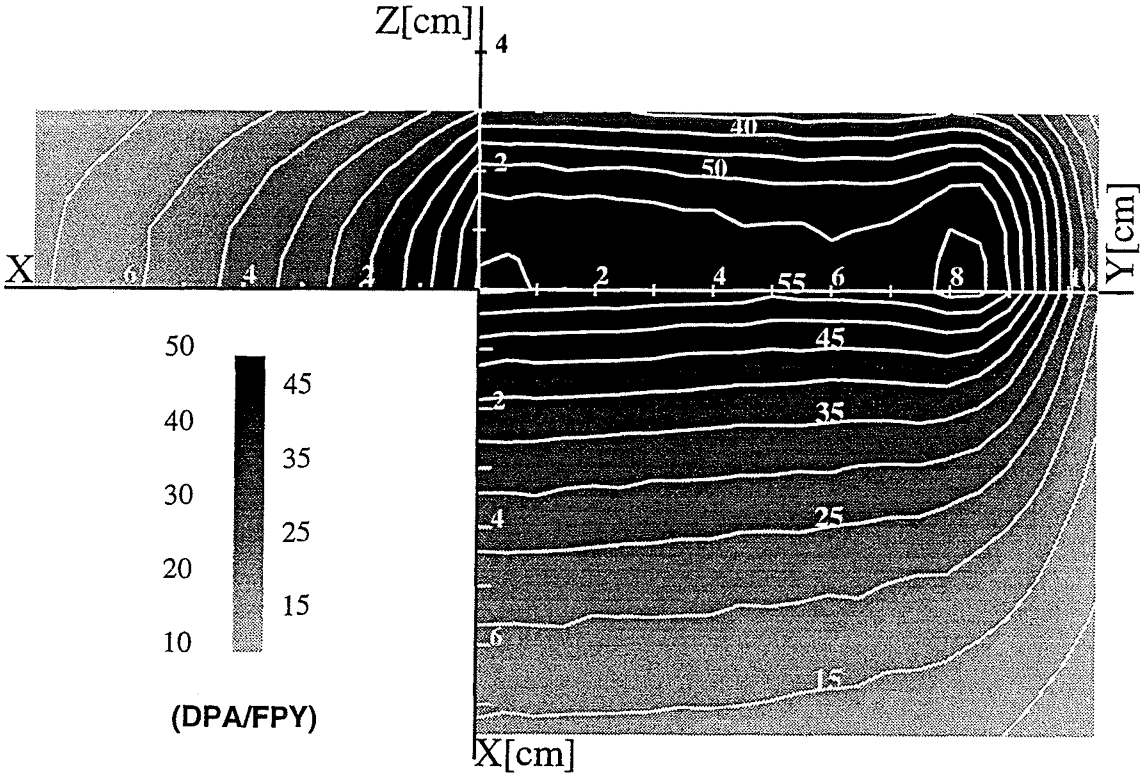


Figure 8. Damage Rate

(Reference 1: Figure 5-93)

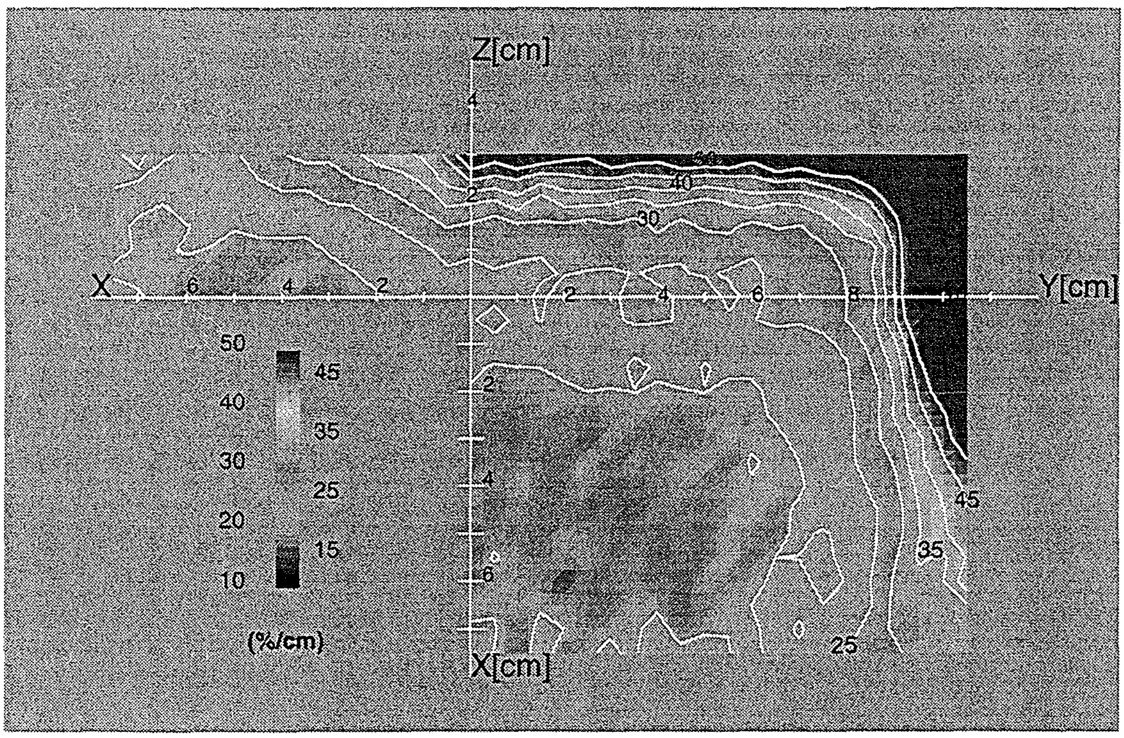


Figure 9. Damage Production Gradients

(Reference 1: Figure 5-94)

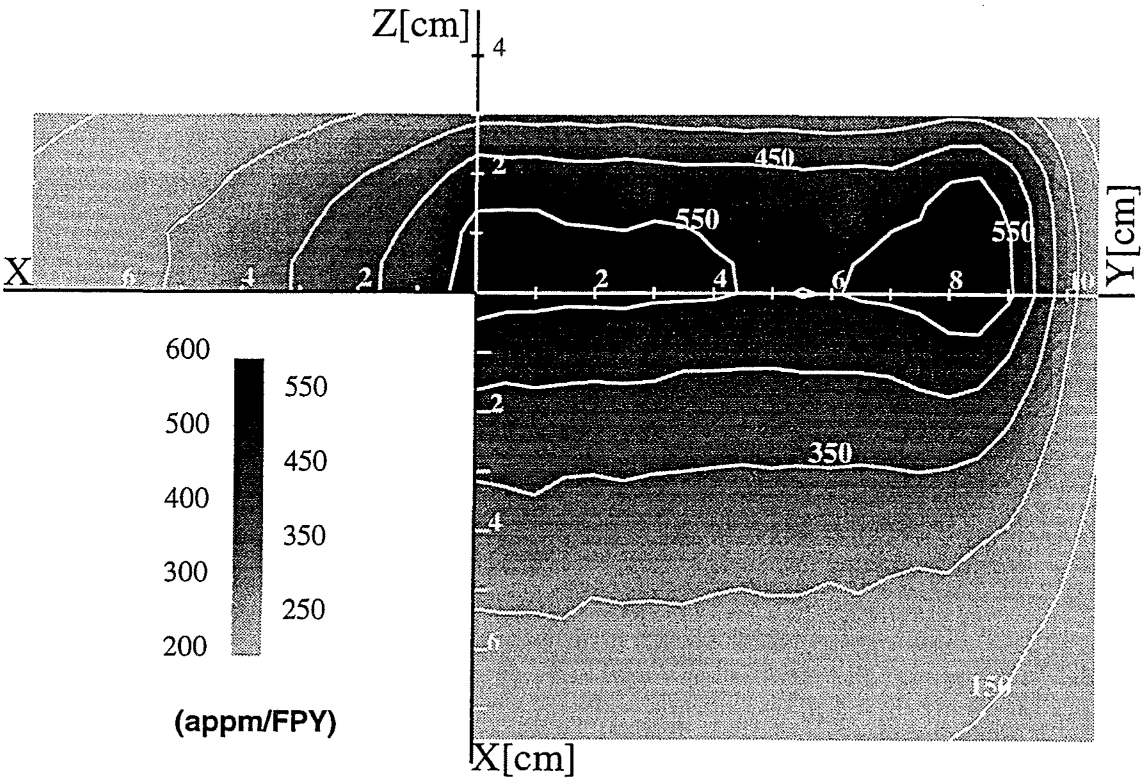


Figure 10. Helium Production Rate
(Reference 1: Figure 5-90)

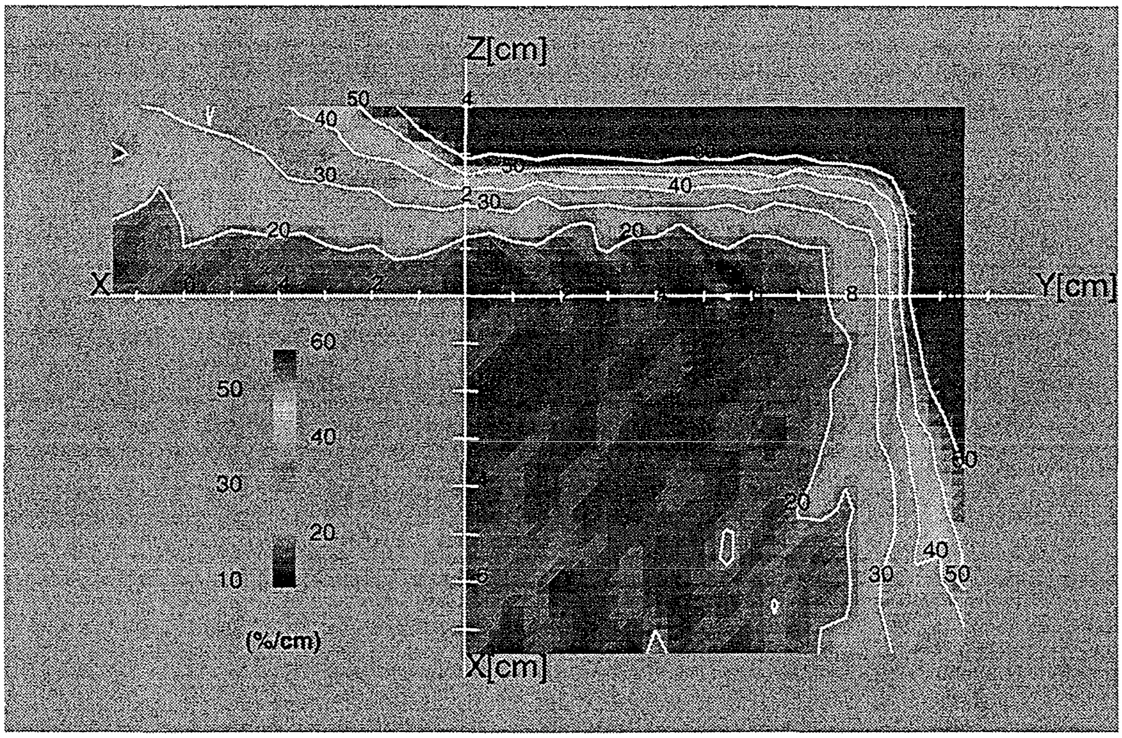


Figure 11. Helium Production Gradients
(Reference 1: Figure 5-91)

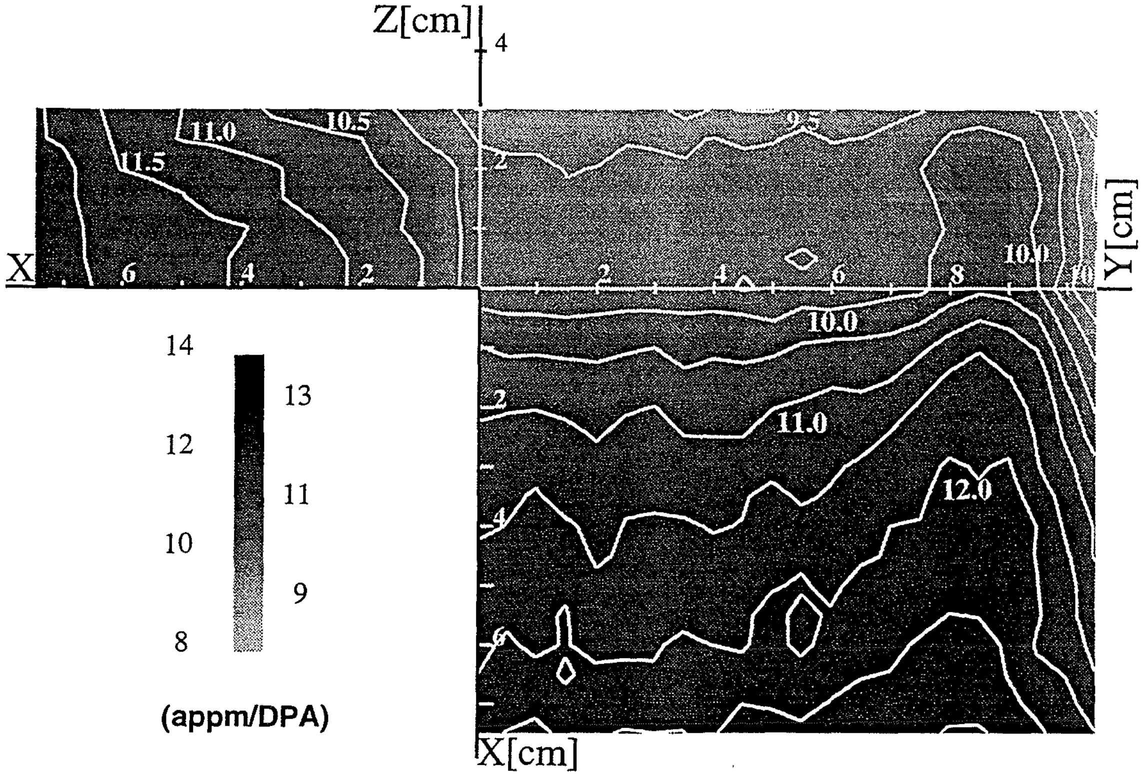


Figure 12. Helium to Damage Production Ratio

(Reference 1: Figure 5-92)

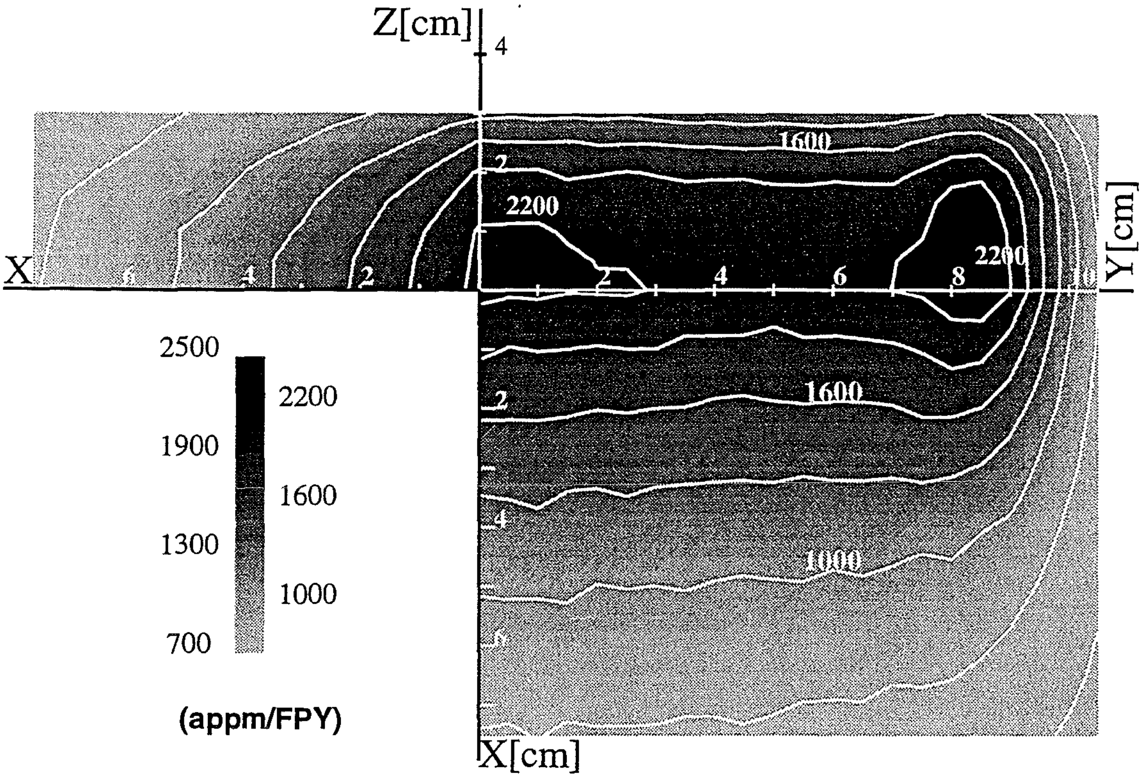


Figure 13. Hydrogen Production Rate

(Reference 1: Figure 5-87)

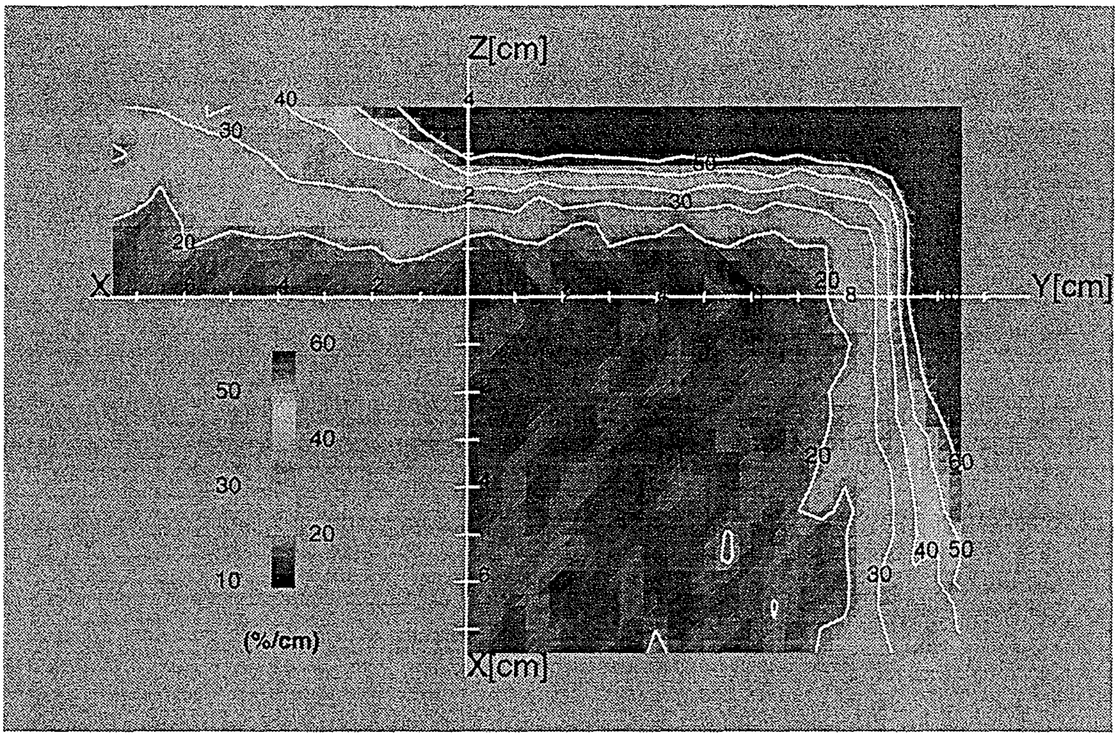


Figure 14. Hydrogen Production Gradients

(Reference 1: Figure 5-88)

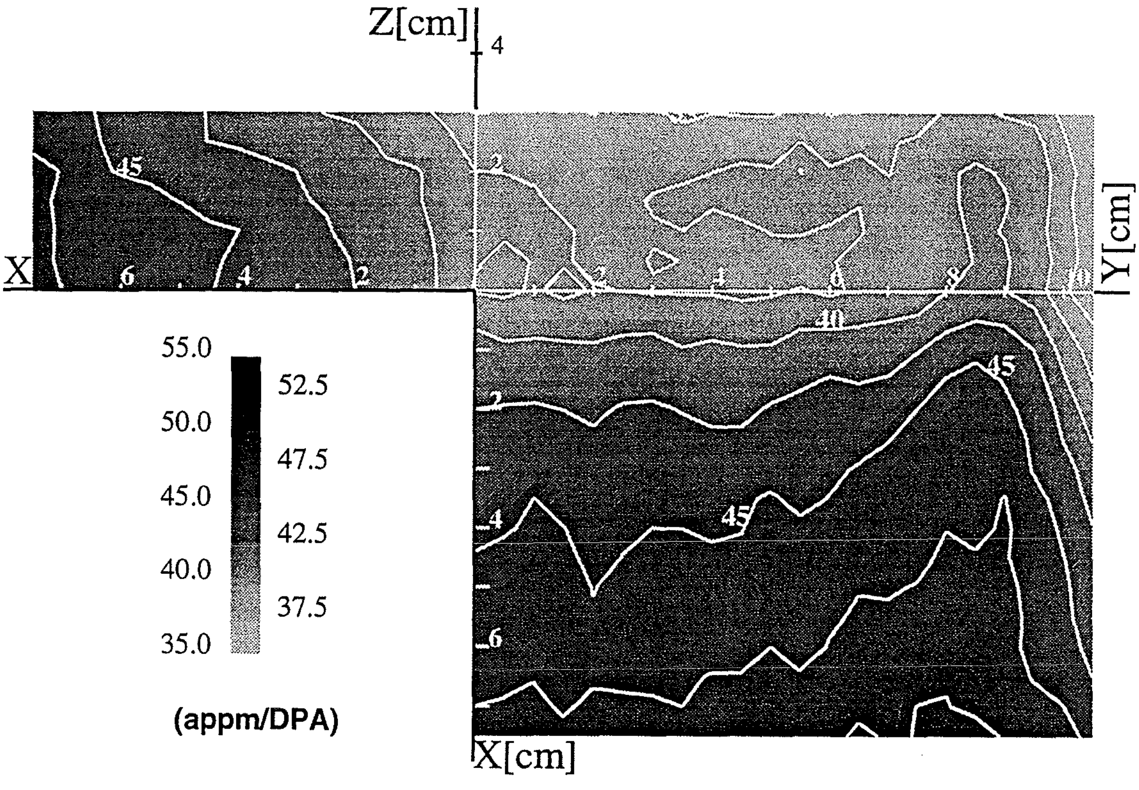


Figure 15. Hydrogen to Damage Production Ratio

(Reference 1: Figure 5-89)

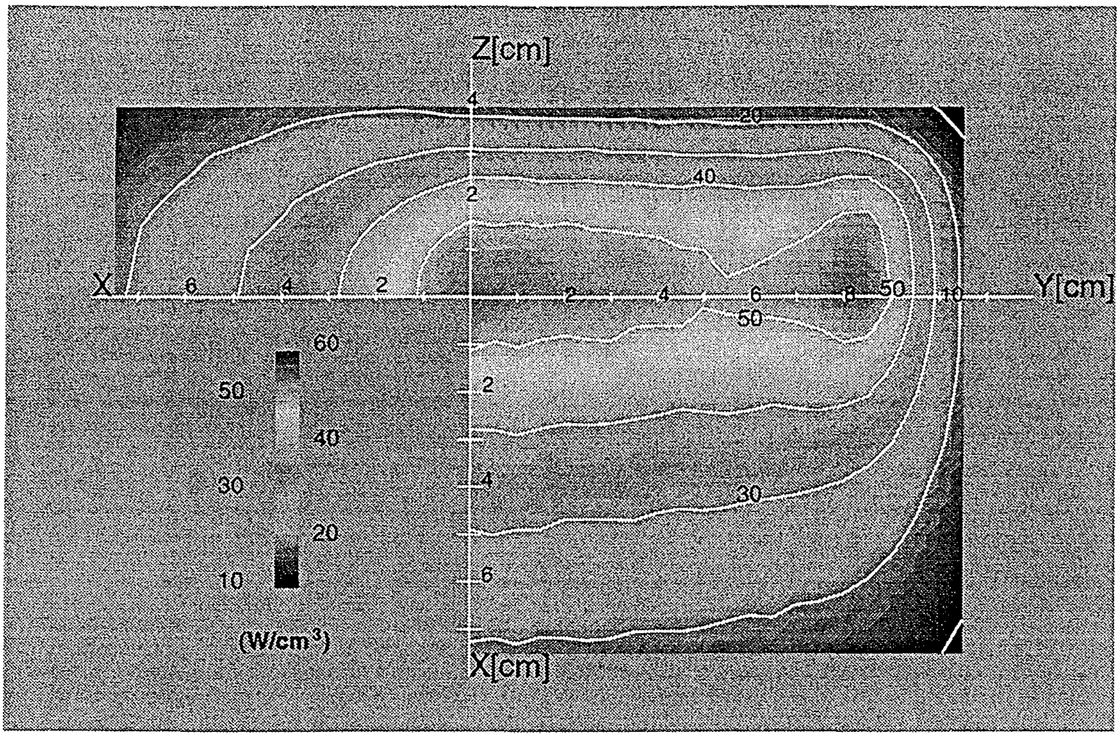


Figure 16. Nuclear Heating [non-uniform beam profile]
(Reference 1: Figure 5-96)

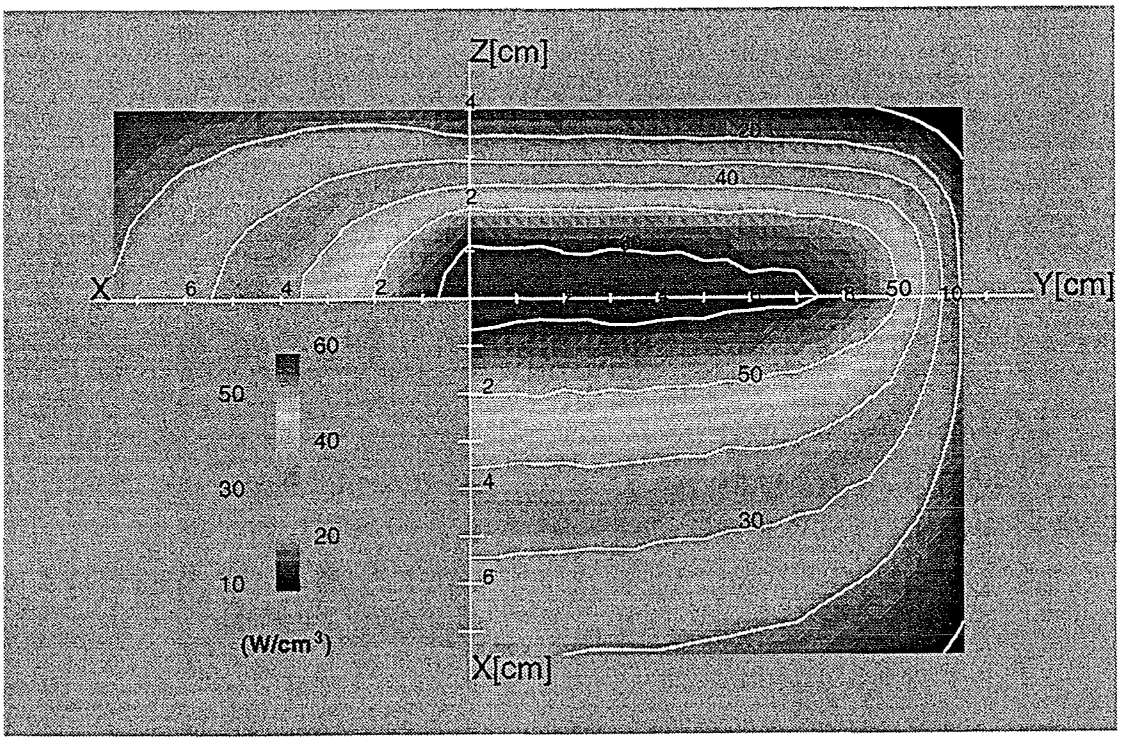


Figure 17. Nuclear Heating [uniform beam profile]
Compare with Figure 16.
(Reference 1: Figure 5-81)

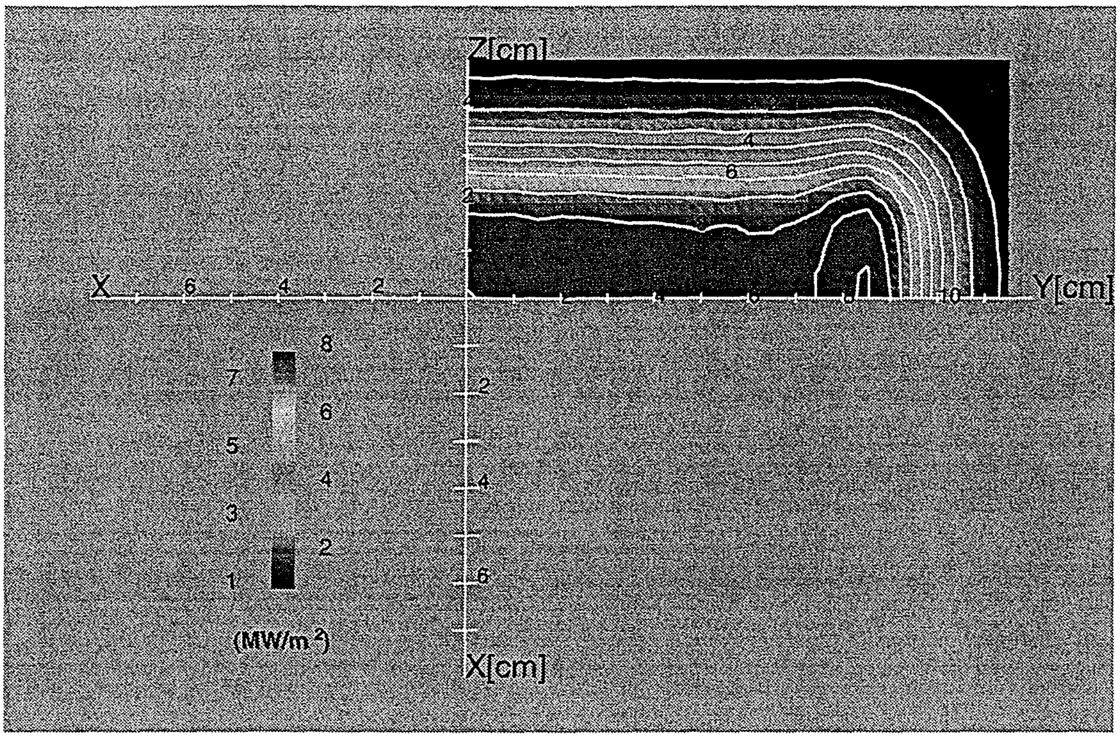


Figure 18. Neutron Wall Load on Backplate

(Reference 1: Figure 5-97)

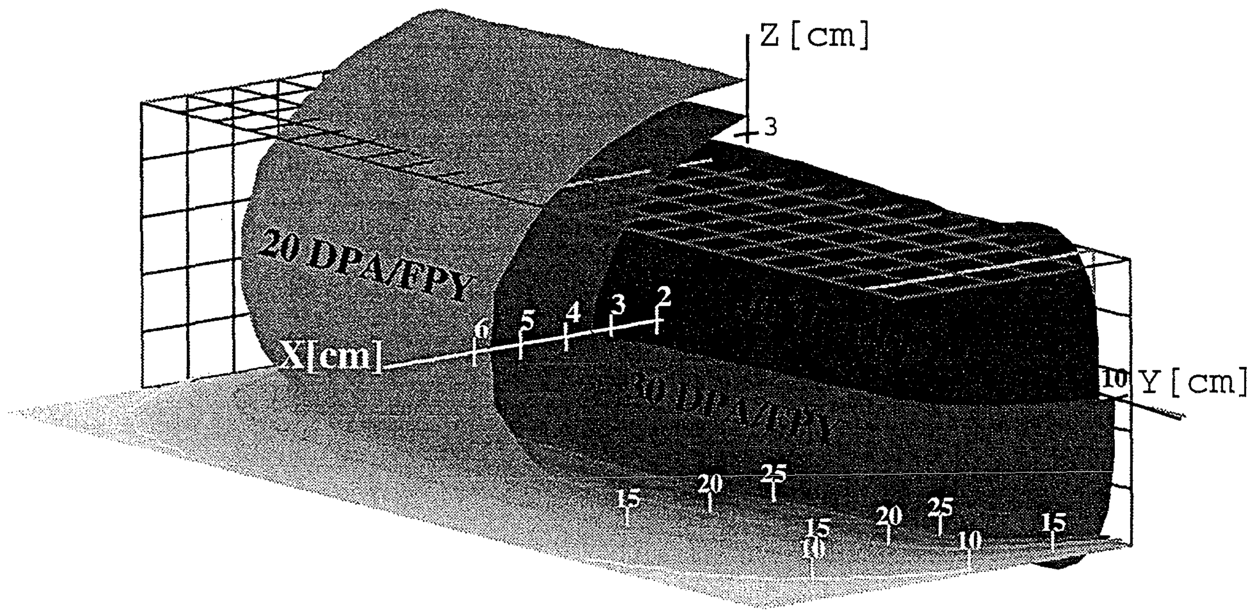


Figure 19. Constant Damage Rate Isosurfaces

Grid shows CDA High Flux Test Module dimensions.

(Reference 1: Figure 5-95)

IFMIF-CDE Test Facility Workshop, Karlsruhe, July 7-9, 1997 *U. von Möllendorff*

CDE Task 11:

Evaluated Nuclear Data Files for Neutron Energies 0-50 MeV

U. Fischer, U. von Möllendorff, A.Yu. Konobeyev[1],
Yu.A. Korovin[1],V.P. Lunev[2], P.E. Pereslavtsev[1], A.Yu. Stankovsky[1]

[1] Institute of Nulear Power Engineering, Obninsk, Russian Federation
[2]Institute of Physics and Power Engineering, Obninsk, Russian Federation

Since 1995, a comprehensive program of nuclear data evaluation for IFMIF has been going on as a prerequisite for neutronics analyses of IFMIF.

Specifications:

- Standard ENDF-6 file format, including double-differential emission cross sections ('file-6' data), for processing with NJOY/ACER into working libraries for MCNP

- Data for both
  neutron transport (scattering, absorption, multiplication) and
  engineering responses (displacement damage, gas production, nuclear heating including γ emission)

Method:

- Above 20 MeV: Evaluation using state-of-the-art nuclear reaction theory
- 0-20 MeV: Adoption of JENDL-3 or ENDF/B-VI data

Present Status:
- Complete files for Fe-56, Cr-52, V-51, O-16
- Neutron transport files for Na-23, K-39, Si-28, C-12

Being prepared:
- Comprehensive file for activation calculations (many nuclides)

Planned:
- Transport/responses file for H-1

Envisaged:
- Transport/responses files for Cr-53, Ti-48, Ti-46, Ti-47

Under discussion (possibly LANL files to be used):
- Transport/responses files for Al-27, Ca-40, minor Fe isotopes

New situation:
- Files for 0-150 MeV recently released by LANL
- Emerging new European evaluation activity for intermediate energy range; may result in decision to increase energy range and change some details of representation in our future evaluations

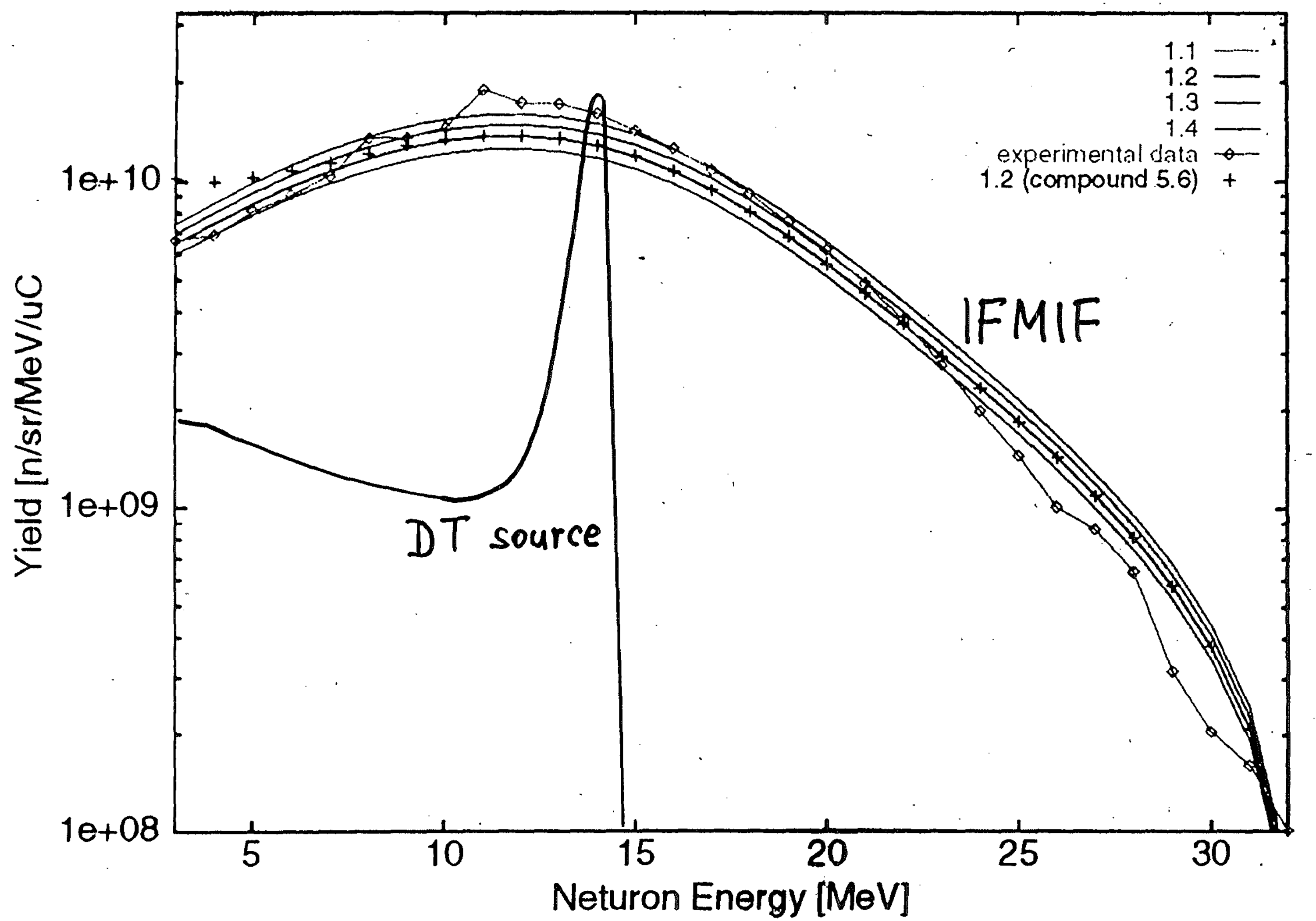


Fig. 1: Comparison between data and model in forward direction to determine magnitude of stripping component

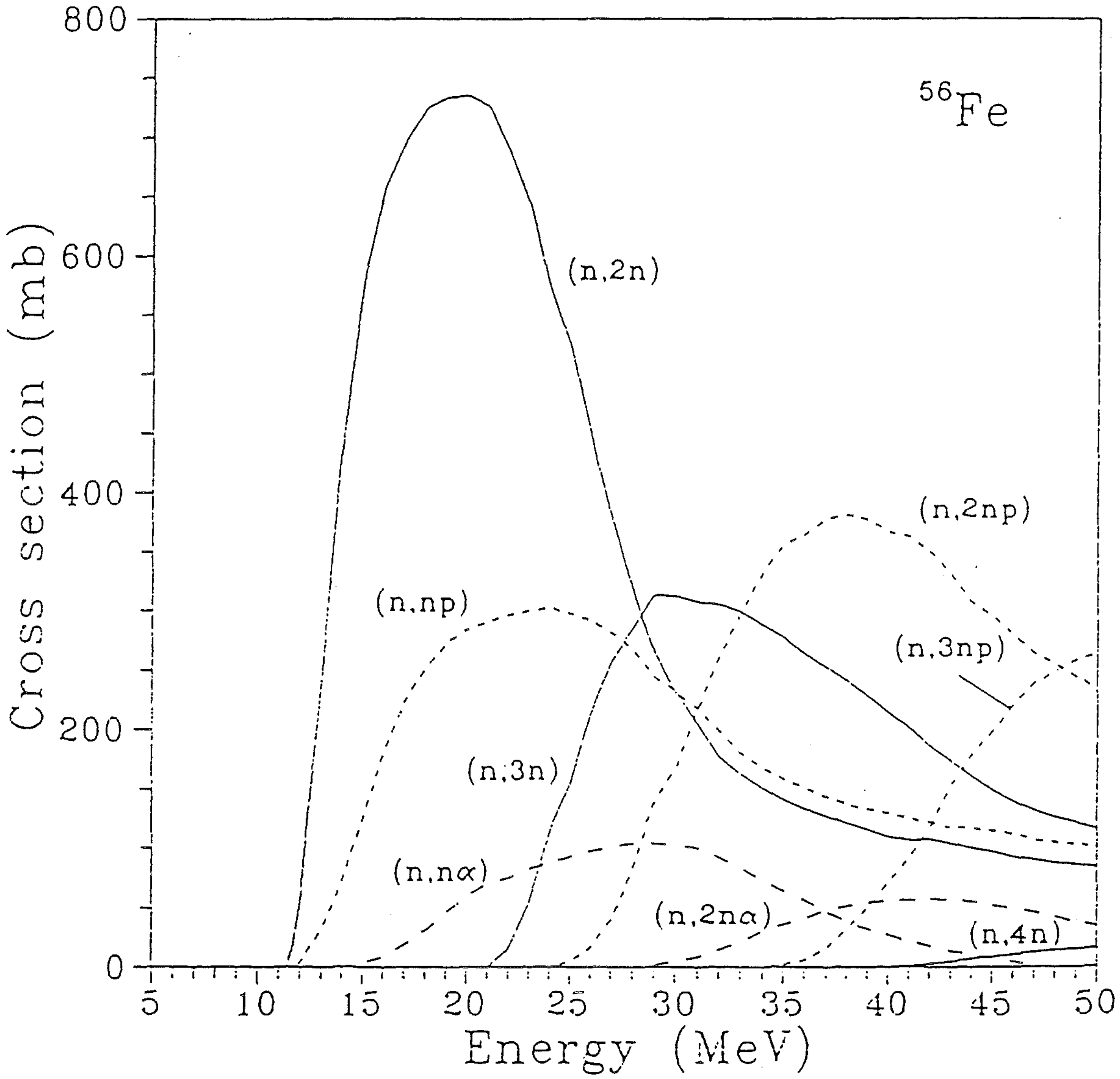


Fig. 2: Cross sections of kinematically possible neutron reactions in Fe-56

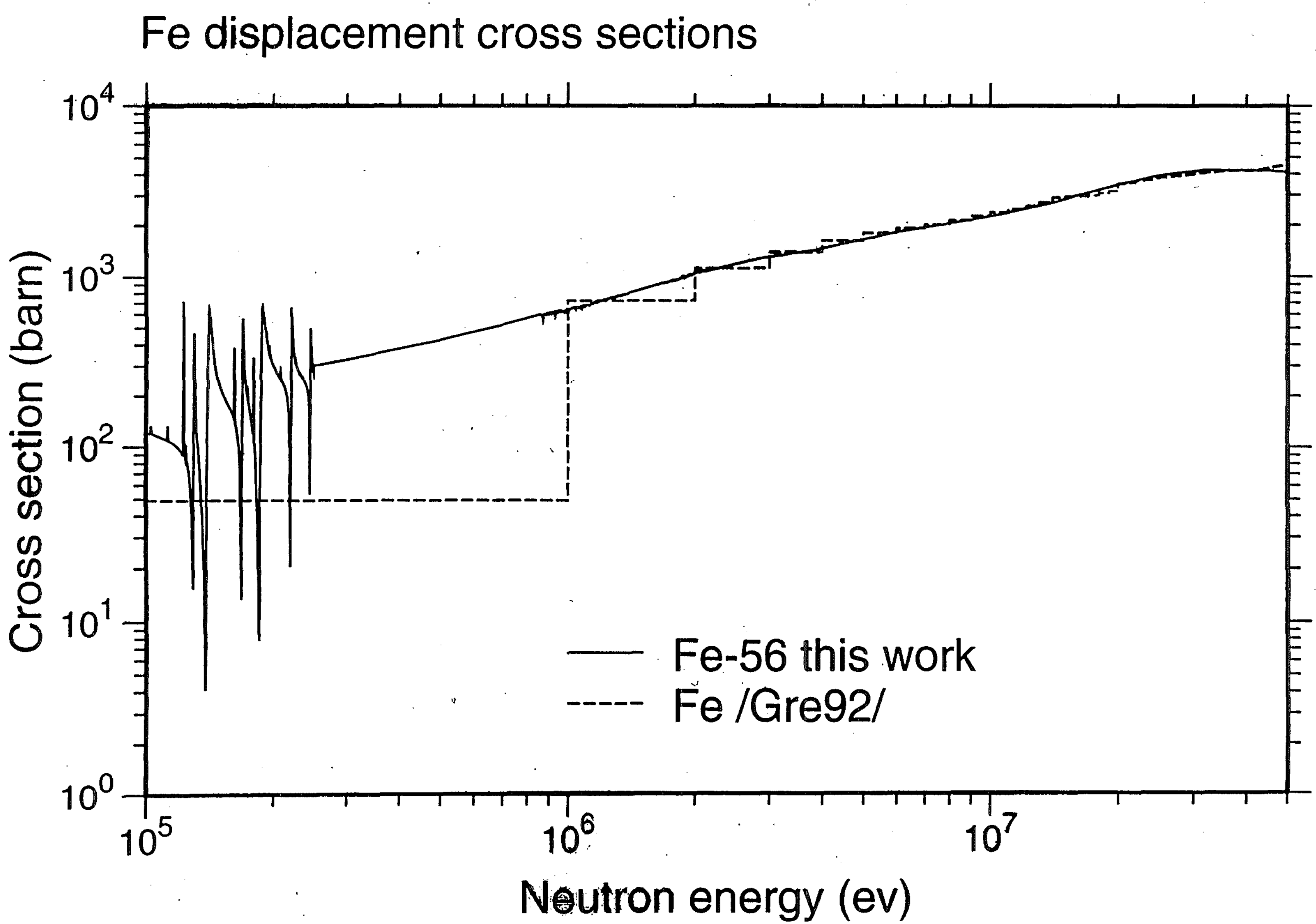


Fig. 3: Fe displacement cross sections

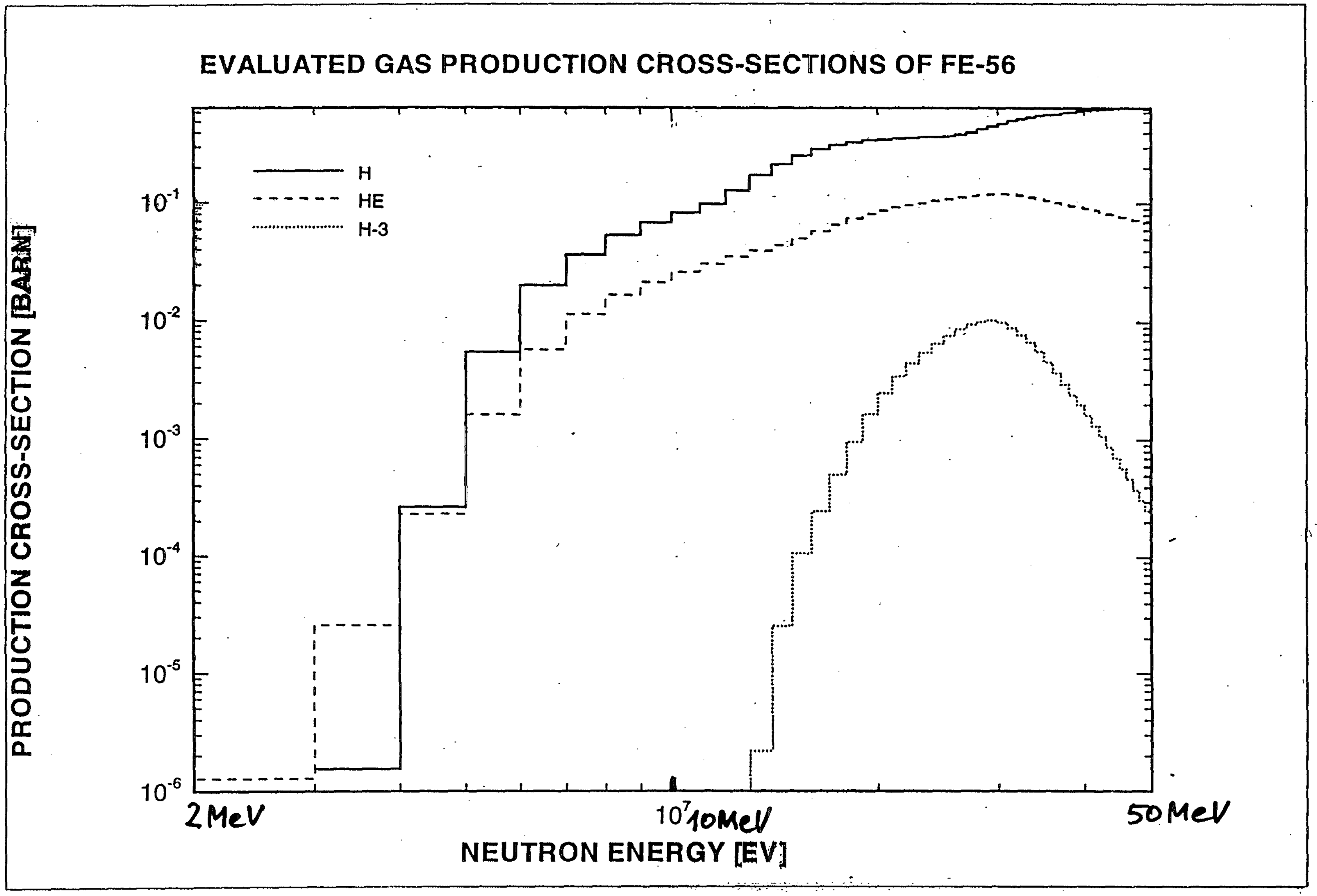


Fig. 4: Evaluated gas production cross sections of Fe-56

Impact of neutron flux fraction above 15 MeV

In the CDA He cooled High Flux Test Module,
the flux above 15 MeV is **15%** of the total flux,

but contributes

- **31%** of the total displacement damage,
- **71%** of the total He production,
- **73%** of the total H production

(cf. presentation by U. Fischer)

Status of JENDL High Energy File for IFMIF

Y. Oyama and T. Fukahori

Japan Atomic Energy Research Institute

- Basic Policy of JENDL High Energy File
- Revision of NJOY94
- MCNP Library
- Release of Library

IFMIF 1997/7/7

Basic policy of JENDL High Energy File

- JENDL-Fusion File or -3.2 is used below 20 MeV, to make the conection point at 20 MeV smooth, the data above 20 MeV is adjusted.

- Cross section data given are 8 cross sections: total, elastic, nonelastic, inelastic, nuclide production, fission, radiative capture, particle production.

- Angle-energy distribution is given in MF=6 format

Revision of NJOY94

- NJOY94.66 was used, but there were troubles by inconsistency with the nuclear data file.

For example,
>Total XS is not equal to sum of partial reactions.
>There is no partial cross section, eg.,
(n,2n),(n,3n), continuum..., but only particle and nuclide production XSs are given.

- NJOY was modified to process the above irregular data

IFMIF 1997/7/7

Status of JENDL-HE file and MCNP Library

- JENDL-HE file
  21 nuclide were edited to the files.

  Na-23, Al-27, K-39, Ti-47, Ti-48, Ti-49, Ti-50, V-51, Cr-50, Cr52, Cr-53, Cr-54, Mn-55, Fe,54, Fe-56, Fe-57, Fe-58, Cu-63, Cu-65 and Y-89

- MCNP Libray
  -Above 21 nuclide have been processed.
  -But there were problems in some nuclide.

IFMIF 1997/7/7

Status of JENDL-HE file and MCNP Library(continued)

- Status of MCNP Library

  Na-23, Mn-55 data were incomplete, so they could not be used.

  The data has still gap at 15 MeV between J-HE and J-FF.

- JAERI Nuclear Data Center will fix these inconsistencies this year.

IFMIF 1997/7/7

Benchmark Test of Fe Data

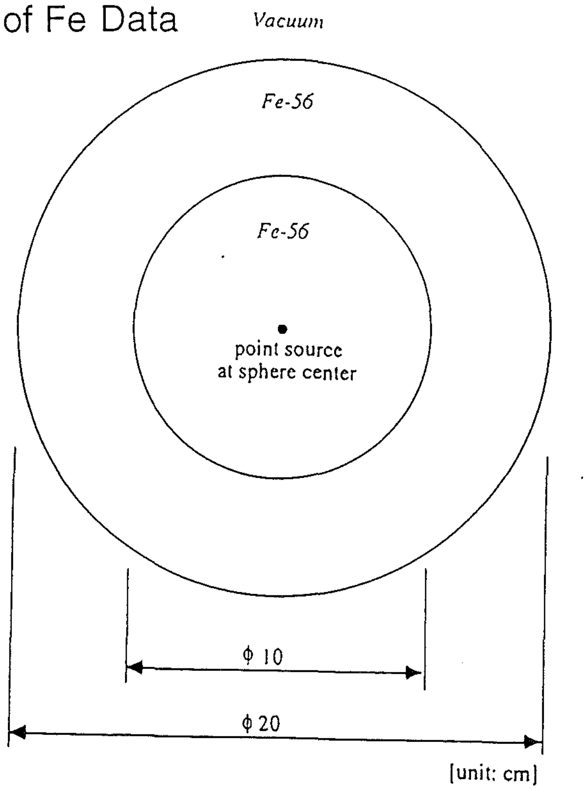


Fig. 3.1 Test calculation model (10 cm-radius sphere) of JENDL high-energy library for the MCNP code.

Benchmark Test of Fe Data (continued)

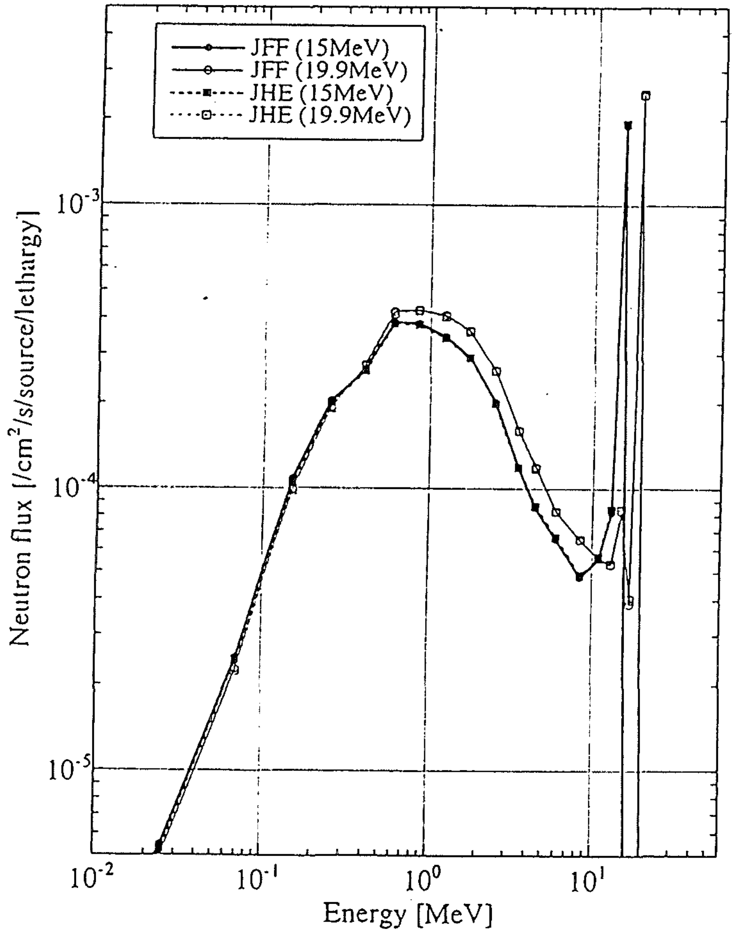


Fig. 3.3 Neutron spectra at radius 10cm in 10cm-radius sphere of ^{56}Fe for checking the JENDL High-Energy File.

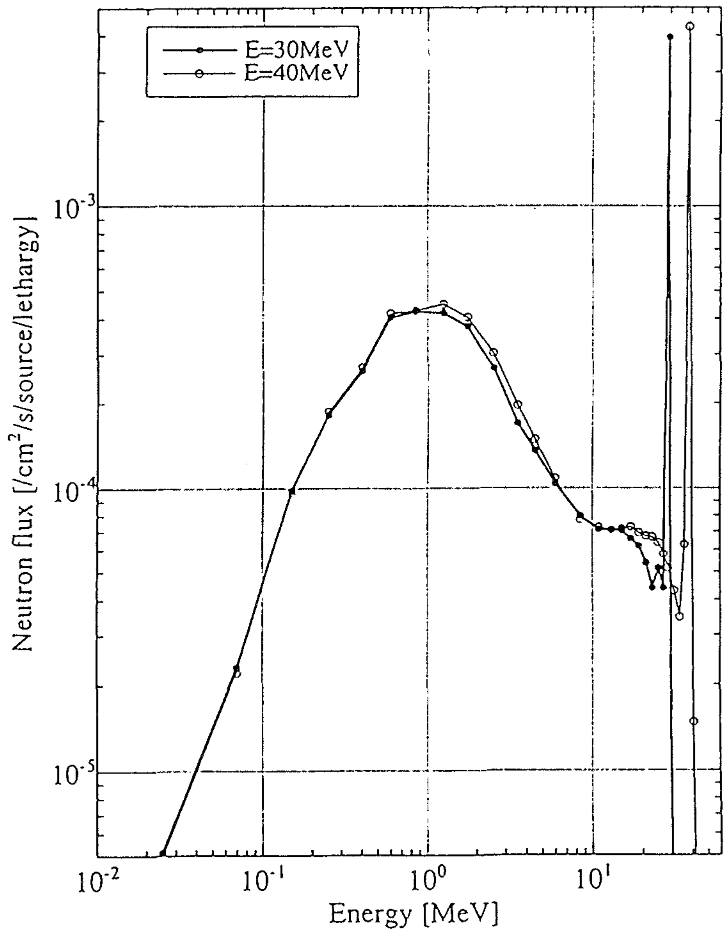


Fig. 3.8 Neutron spectra at radius 10cm in 10cm-radius sphere of ^{56}Fe for checking the JENDL High-Energy File.

SHIELDING AND ACTIVATION ANALYSIS

S. MONTI

IFMIF-CDE Test Facility Workshop - Karlsruhe - July 1997

Main Objectives

- General Shielding Design Criteria
- Computer Model and Nuclear Data Needs
- Shields for Test Facilities during Operation
- Neutron Activation for Test Module and Test Cell
- Dose Rates during maintenance
- Shielding for Access, Service, Test Handling Cells and Hot Laboratories
- Comparison between JAERI and ANL Neutron Sources
- HEBT and Beam Turning Room Activation and Doses due to Neutron Backstreaming.

Calculational Tools

Shielding Calculations performed with MCNP4A_MORSE code.

Two versions of the code available:

- Multigroup (66 neutrons, 22 photon groups) HILO library in the whole energy range
- HILO library above 20 MeV and point-wise MCNP treatment below 20 MeV

Calculational Tools

Neutron induced activation calculations performed with FISPACT code and EAF4.1 cross section library

Only data from thermal to 20 MeV are available

Test Cell Geometry Model

- Test Cell geometry, materials and dimensions as reported in CDA Final Report
- Personnel accessibility during operation only at second floor
- MCNP4A Geometry Model with SABRINA code

Other Input Data

- According to ICRP60: 20mSv per year (10 μSv/hr for 2000 hr/yr: occupancy factor =1)
- Fluence-to-personal dose equivalent conversion factors by Siebert (new ICRU stopping power) with log-log Lagrange interpolation

Test Cell Source Term

- JAERI neutron source for two primary 40 MeV deuteron beams of 125 mA at 10 degrees on Li-target
- ANL (Gomes) D-Li neutron source

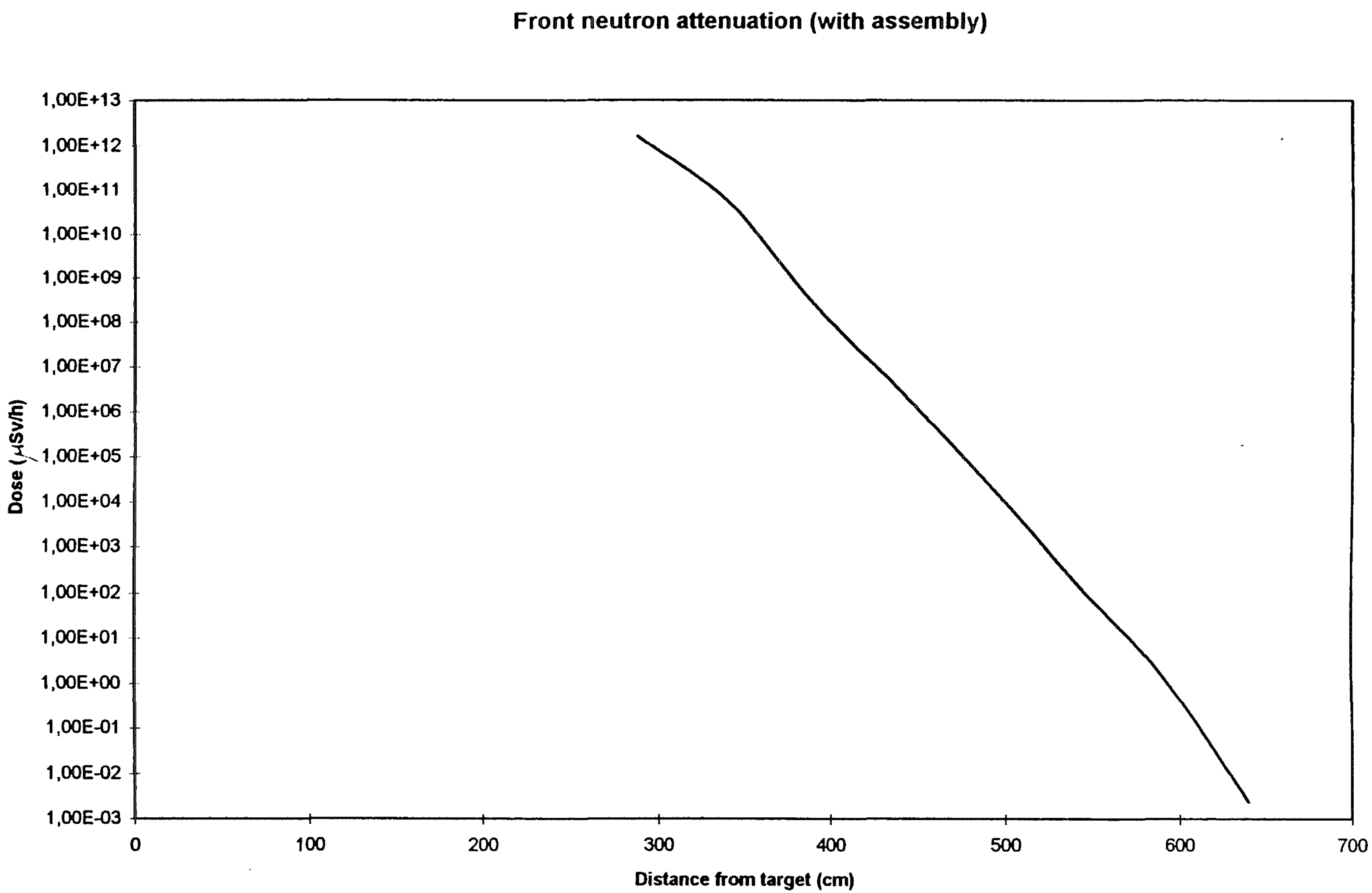


Fig.9 - Test Cell Frontal wall neutron attenuation with Assembly.

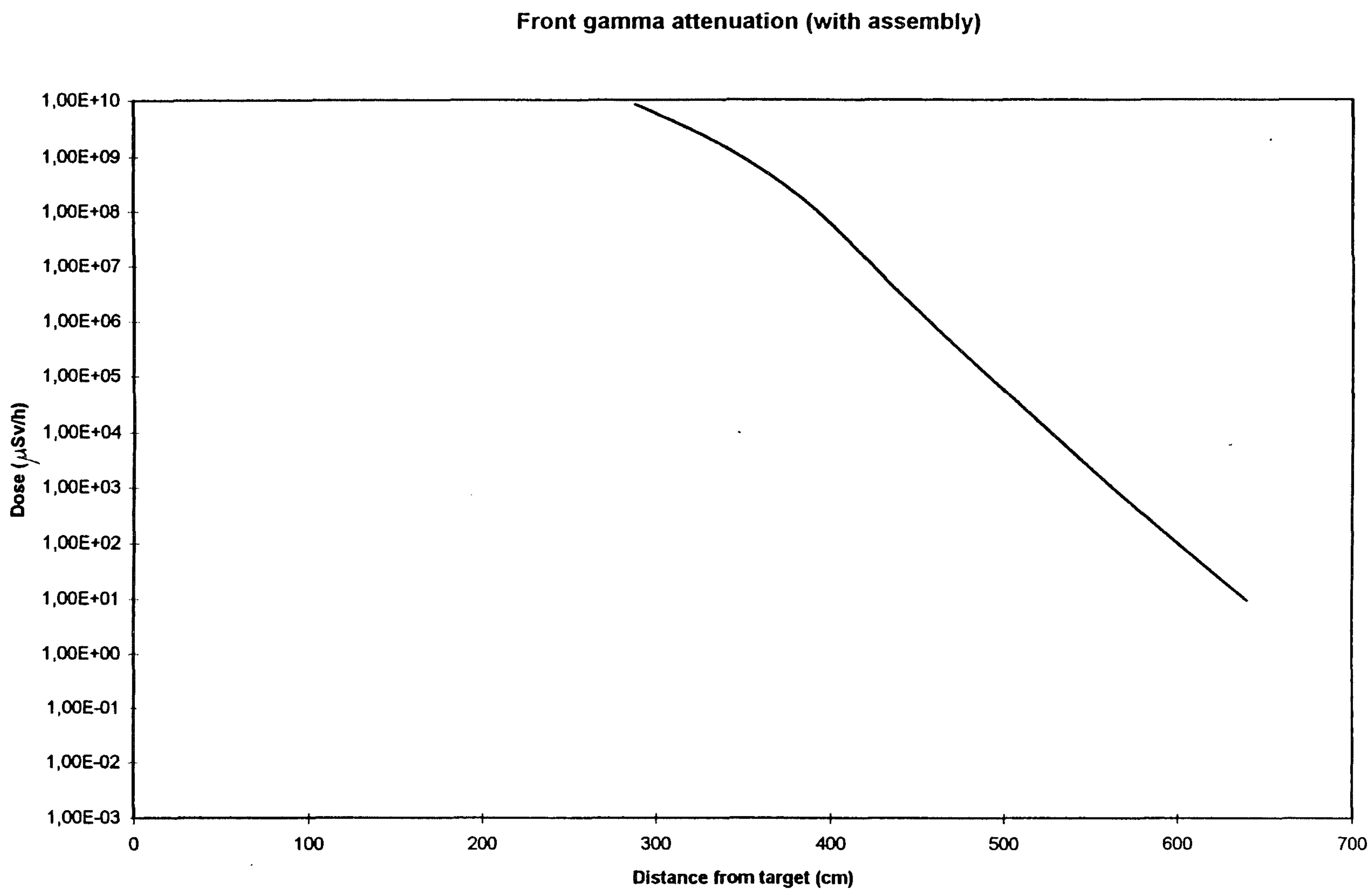


Fig.10 - Test Cell Frontal wall gamma attenuation with Assembly.

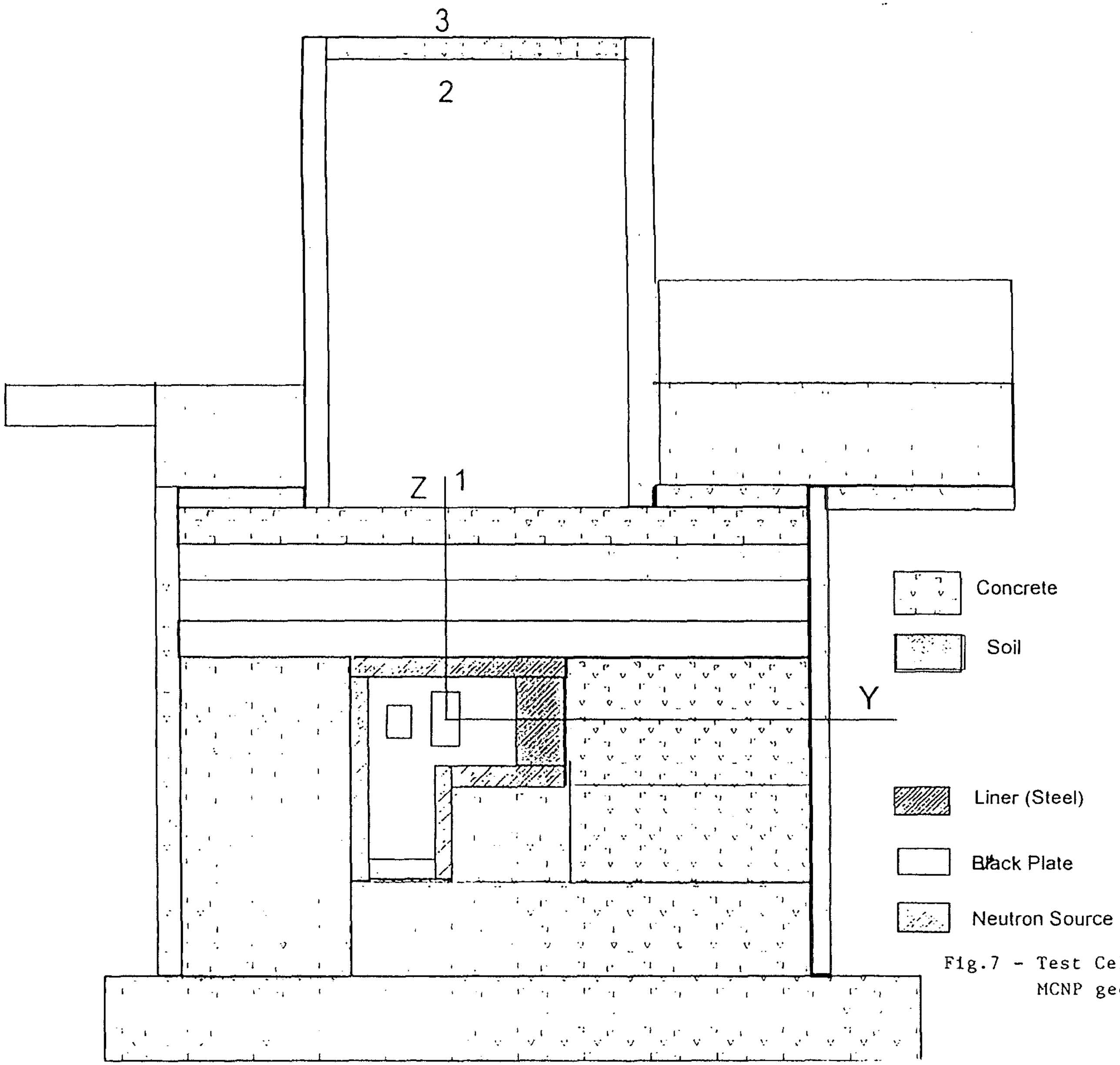


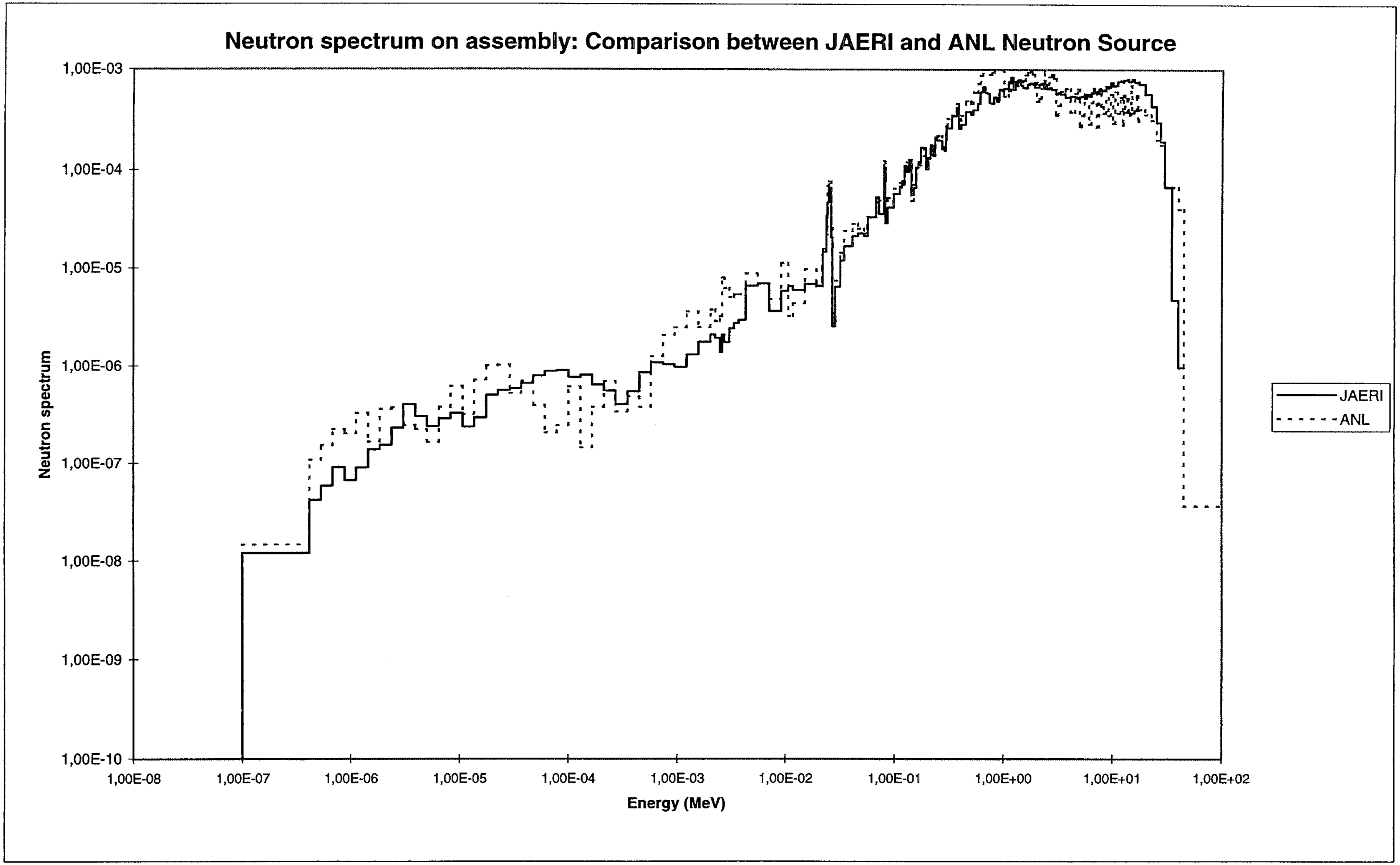
Fig.7 - Test Cell Ceiling wall MCNP geometrical model.

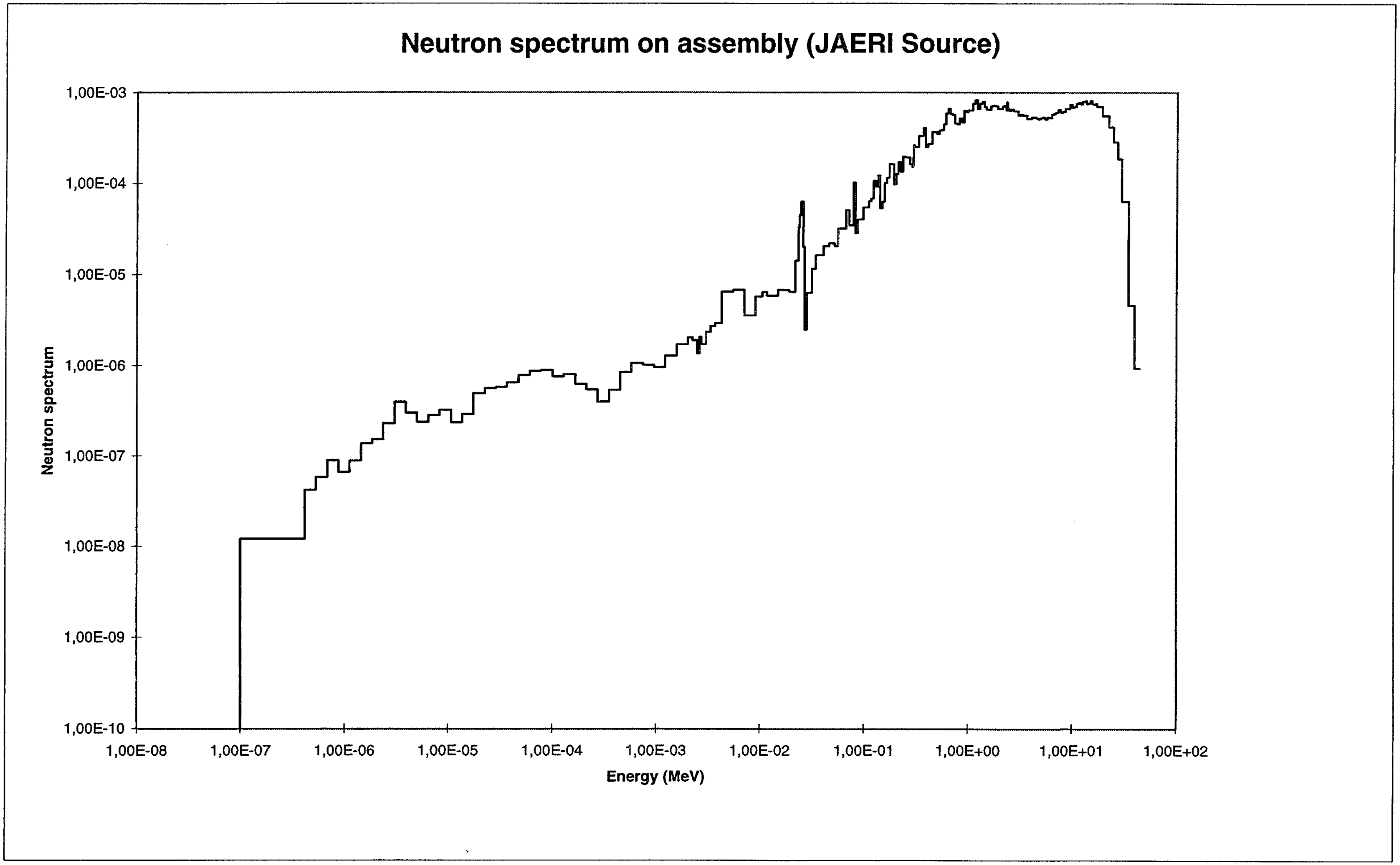
DOSE WITHOUT ASSEMBLY			
POINT CALCULATION	NEUTRON DOSE μSv/h	GAMMA DOSE μSv/h	TOTAL DOSE μSv/h
1	25.9 0.08	4211 0.09	4237
2	1.95 0.20	506 0.14	508
3	-	2.2 0.198	2.2
4	9.0E-3 0.19	26.8 0.17	26.8
5	2.0E-3 0.21	10.5 0.08	10.5
6	1.2E-3 0.10	5.9 0.22	5.9

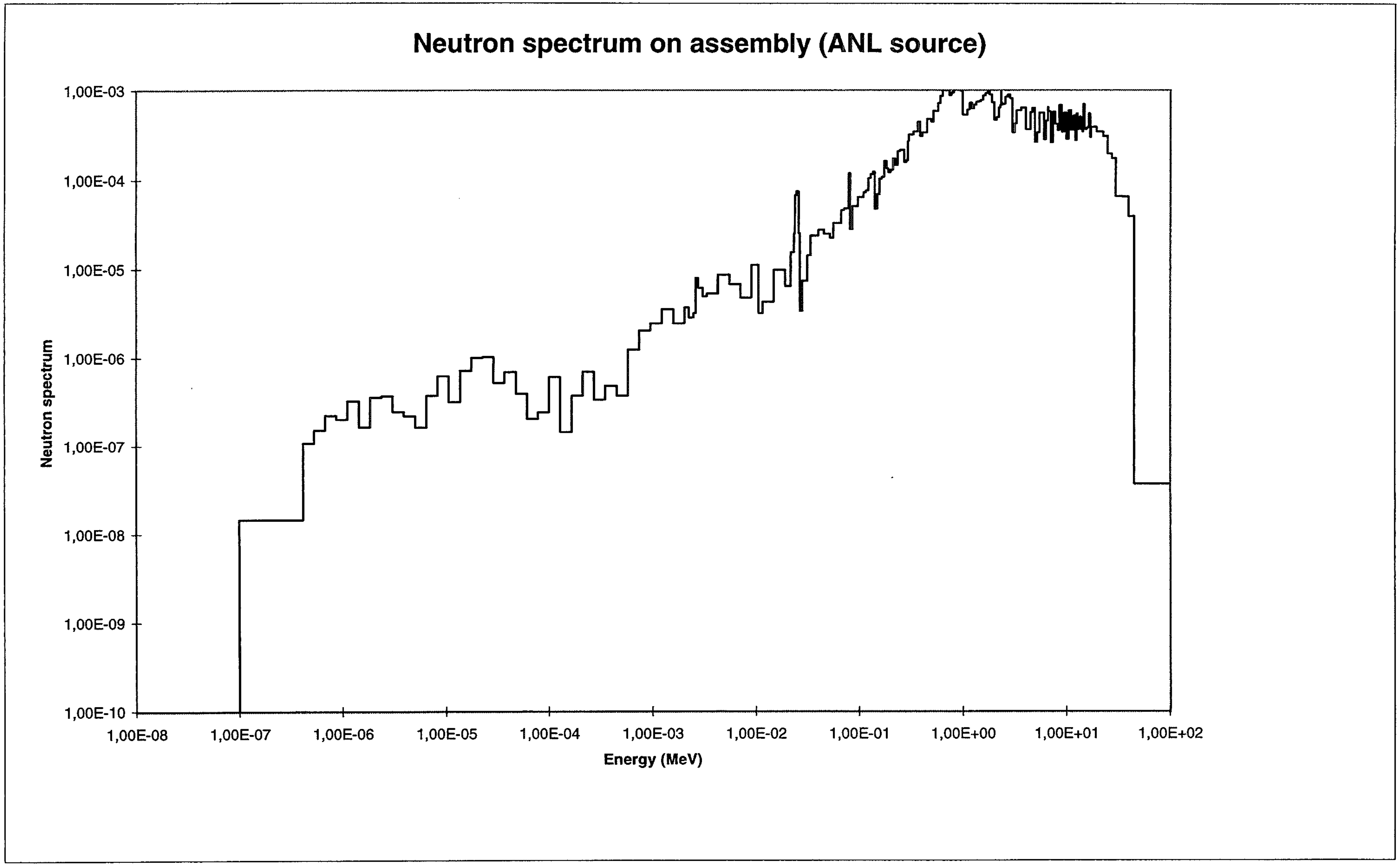
Tab. III - Summary of Test Cell main shielding results (without Assembly).

DOSE WITH ASSEMBLY			
POINT CALCULATION	NEUTRON DOSE μSv/h	GAMMA DOSE μSv/h	TOTAL DOSE μSv/h
1	43.4 0.12	7536 0.07	7579
2	2.0 0.192	507 0.11	509
3	-	1.57 0.18	1.57
4	2.3E-3 0.20	9.8 0.196	9.8
5	2.6E-3 0.17	17.7 0.18	17.7
6	3.6E-3 0.08	14.6 0.21	14.6

Tab. IV - Summary of Test Cell main shielding results (with Assembly).


Neutron spectrum on assembly: Comparison between JAERI and ANL Neutron Source
Neutron spectrum
1,00E-03
1,00E-04
1,00E-05
1,00E-06
1,00E-07
1,00E-08
1,00E-09
1,00E-10
1,00E-08
1,00E-07
1,00E-06
1,00E-05
1,00E-04
1,00E-03
1,00E-02
1,00E-01
1,00E+00
1,00E+01
1,00E+02
Energy (MeV)
JAERI
ANL


Neutron spectrum on assembly (JAERI Source)
1,00E-03
1,00E-04
1,00E-05
1,00E-06
1,00E-07
1,00E-08
1,00E-09
1,00E-10
Neutron spectrum
1,00E-08
1,00E-07
1,00E-06
1,00E-05
1,00E-04
1,00E-03
1,00E-02
1,00E-01
1,00E+00
1,00E+01
1,00E+02
Energy (MeV)


Neutron spectrum on assembly (ANL source)
1,00E-03
1,00E-04
1,00E-05
1,00E-06
1,00E-07
1,00E-08
1,00E-09
1,00E-10
Neutron spectrum
1,00E-08
1,00E-07
1,00E-06
1,00E-05
1,00E-04
1,00E-03
1,00E-02
1,00E-01
1,00E+00
1,00E+01
1,00E+02
Energy (MeV)

Gamma Source (gamma/sec) from 500 cc Test Module Activation vs. Cooling Time for Different Test Module Loadings

Cooling Time	T = 0			T = 1 Day			T = 1 Week		
Energy Group (MeV)	Iron (4 kg) 1 Yr 100% f. p.	SS-316+Nak (1.965 kg) 5 Yrs 100% f. p.	SS-316+Nak (1.965 kg) 20 Yrs 70% f. p.	Iron (4 kg) 1 Yr 100% f. p.	SS-316+NaK (1.965 kg) 5 Yrs 100% f. p.	SS-316+Nak (1.965 kg) 20 Yrs 70% f. p.	Iron (4 kg) 1 Yr 100% f. p.	SS-316+Nak (1.965 kg) 5 Yrs 100% f. p.	SS-316+Nak (1.965 kg) 20 Yrs 70% f. p.
0.0 - 0.01	9.43E+13	1.01E+14	1.02E+14			1.0E+14	1.84E+14	1.0E+14	1.01E+14
0.01 - 0.02	2.81E+09	2.63E+12	2.17E+12			2.02E+12	2.15E+04	2.38E+12	1.92E+12
0.02 - 0.05	1.30E+10	2.81E+10	2.81E+10			1.56E+10	3.59	3.30E+09	3.27E+09
0.05 - 0.1	3.99E+10	2.51E+11	2.52E+11			3.05E+07	4.62	6.04E+06	6.44E+06
0.1 - 0.2	8.41E+11	2.08E+13	1.67E+13			1.63E+13	4.08E+11	1.95E+13	1.54E+13
0.2 - 0.3	1.24E+10	3.95E+10	5.71E+10			9.35E+09	1.56E+06	2.46E+09	2.45E+09
0.3 - 0.4	1.35E+12	3.29E+12	3.28E+12			2.31E+12	9.81E+11	1.99E+12	1.98E+12
0.4 - 0.6	5.09E+11	2.13E+13	2.06E+13			1.06E+13	2.73E+09	1.02E+13	9.55E+12
0.6 - 0.8	1.67E+11	1.18E+12	1.17E+12			6.97E+11	2.46E+09	3.94E+11	3.86E+11
0.8 - 1.0	2.94E+14	7.04E+13	6.57E+13			3.98E+13	4.83E+13	4.23E+13	3.78E+13
1.0 - 1.22	5.00E+12	1.74E+12	2.15E+12			1.54E+12	4.39E+12	1.11E+12	1.51E+12
1.22 - 1.44	4.03E+12	1.36E+13	1.40E+13			2.94E+12	3.30E+12	1.57E+12	1.93E+12
1.44 - 1.66	2.88E+10	2.28E+12	2.28E+12			4.33E+09	5.20E+09	1.79E+08	1.76E+08
1.66 - 2.0	7.08E+13	8.88E+12	8.85E+12			3.89E+11	1.88E+07	1.65E+11	1.55E+11
2.0 - 2.5	3.55E+13	4.22E+12	4.23E+12			1.23E+10	4.36E+05	1.42E+09	1.25E+09
2.5 - 3.0	4.96E+12	5.62E+11	5.65E+11			1.46E+10	6.37E+02	2.15E+09	1.88E+09
3.0 - 4.0	4.29E+11	6.97E+10	6.96E+10			1.79E+09	3.57	1.70E+09	1.47E+09
4.0 - 5.0	1.68E+07	4.47E+08	5.33E+08			7.30E+04	-	8.21E+01	9.32E+01
5.0 - 6.5	1.72E+05	2.09E+08	3.18E+08			-	-	-	-
6.5 - 8.0	-	4.18E+06	1.17E+07			-	-	-	-
8.0 - 10.0	-	2.87E+05	4.13E+05			-	-	-	-
10.0 - 12.0	-	1.41E+03	3.92E+03			-	-	-	-
12.0 - 14.0	-	1.52E+02	4.22E+2			-	-	-	-
Tot. Activity	5.12E+14	2.52E+14	2.44E+14			1.83E+14	2.41E+14	1.80E+14	1.72E+14

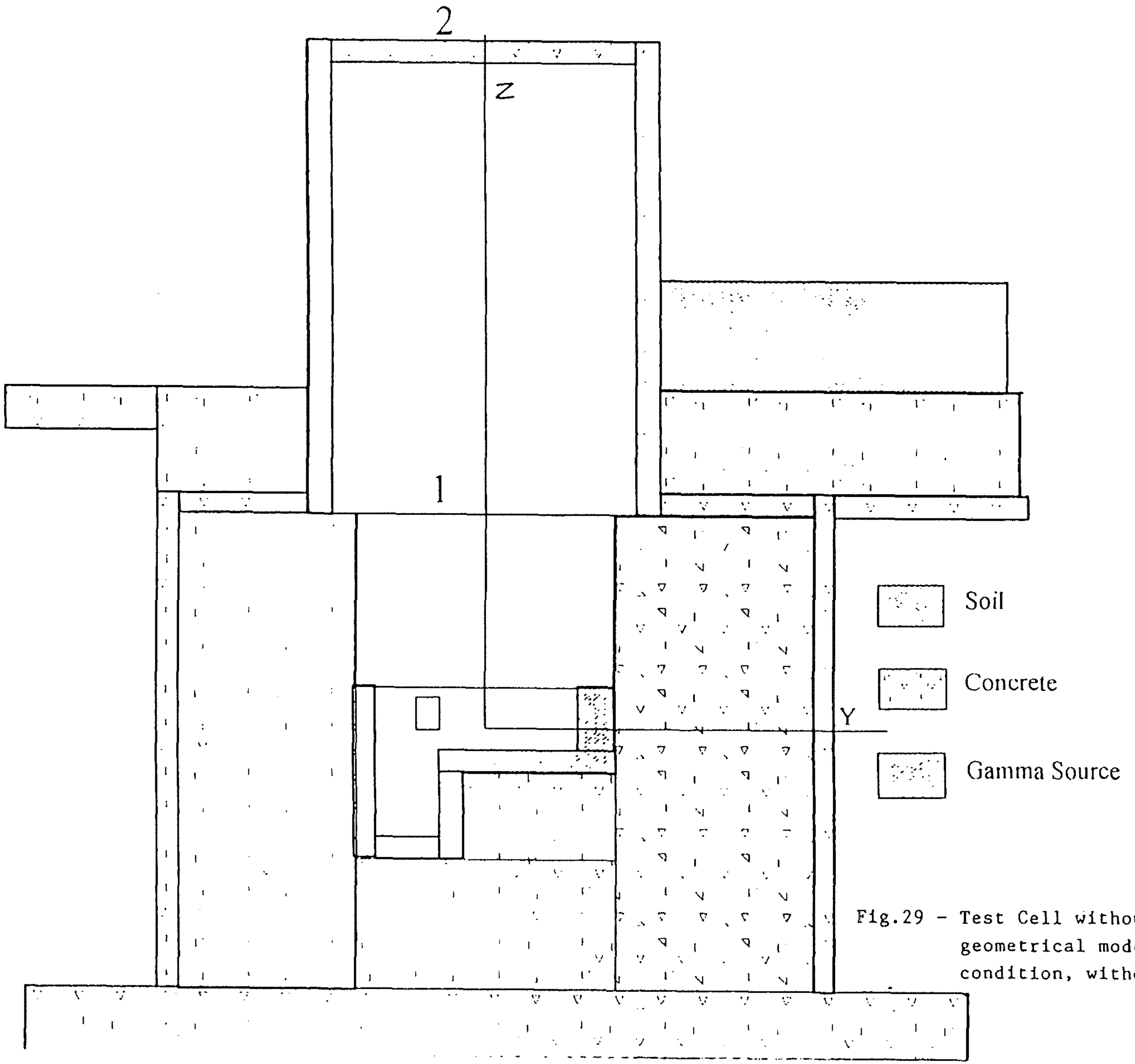


Fig.29 - Test Cell without plug MCNP geometrical model (shut down condition, without plug).

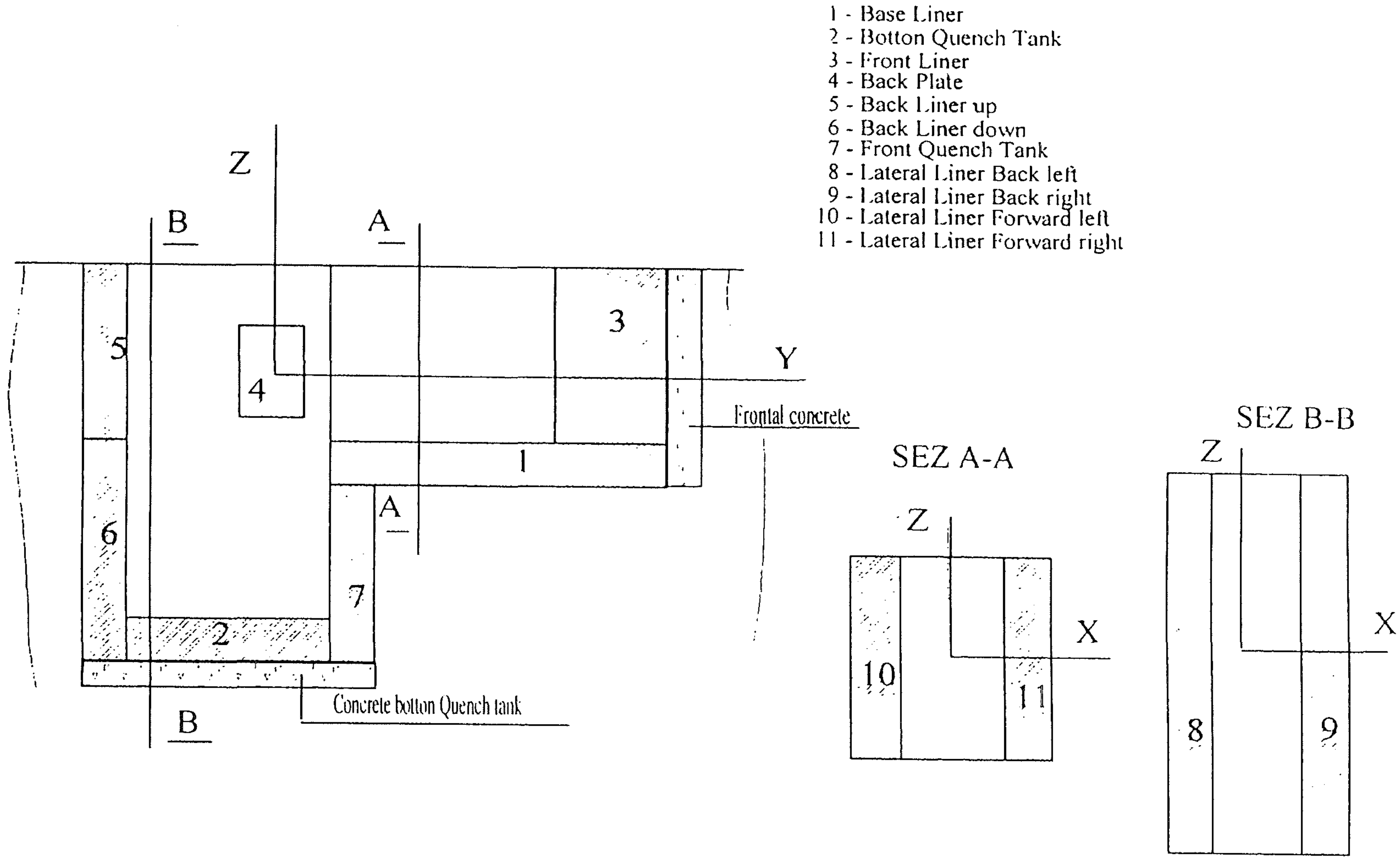
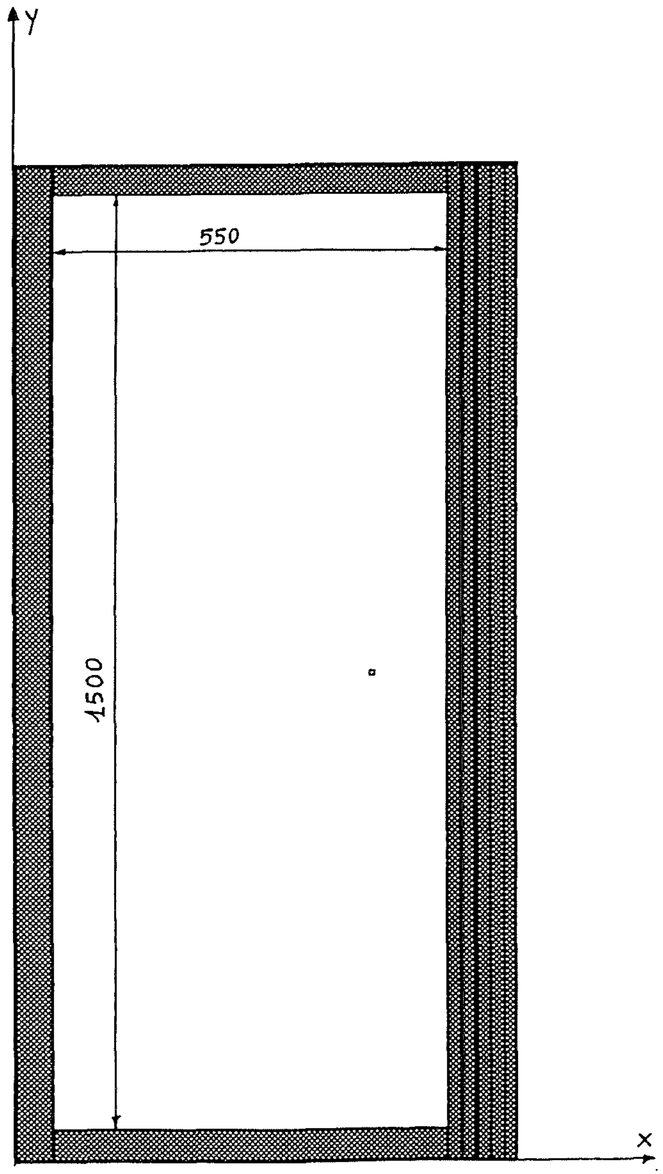


Fig. 30 - SS Liner Activation sources after irradiation

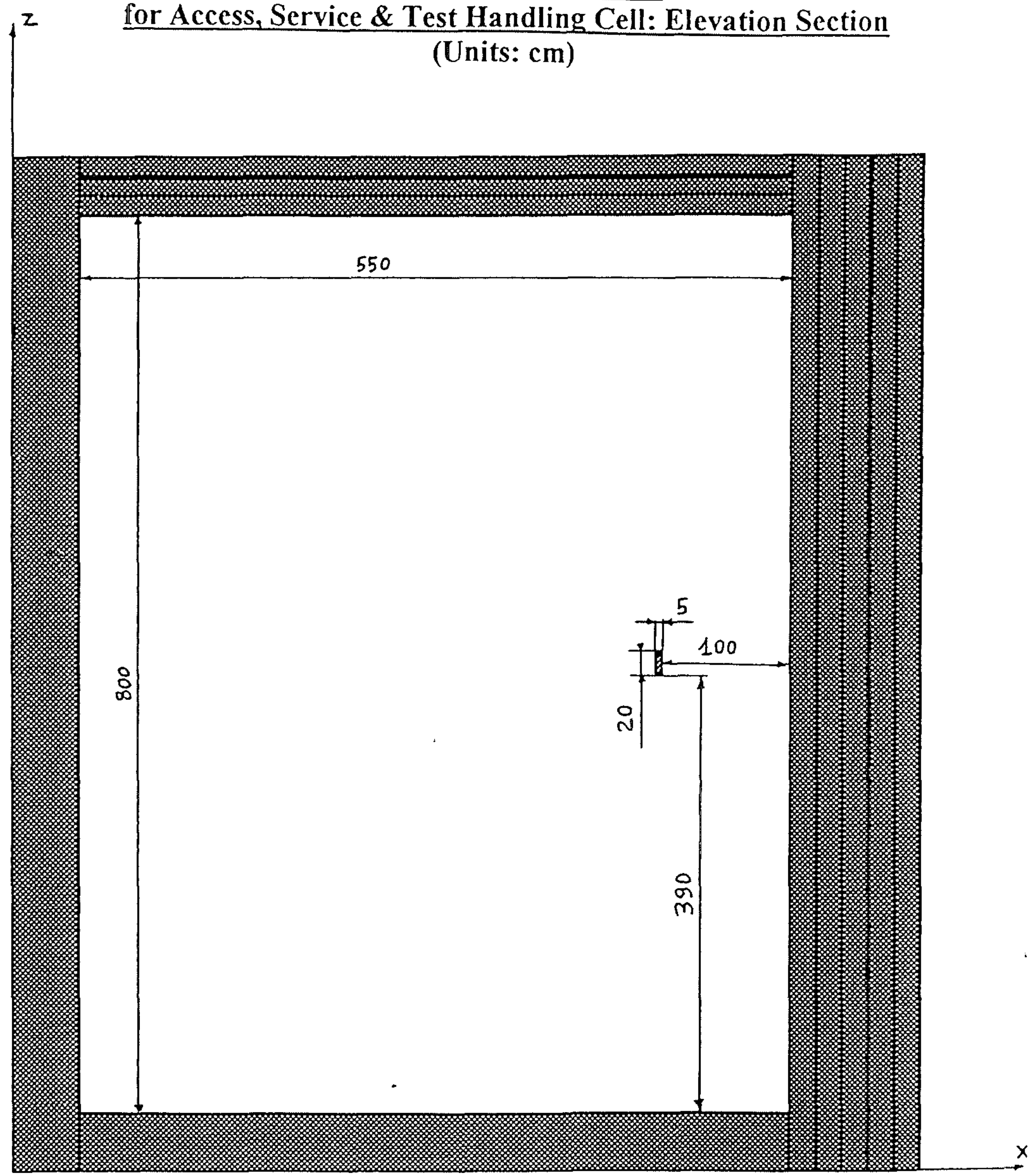
ACTIVATION SOURCE	POINT 1 GAMMA DOSE μSv/s	POINT 2 GAMMA DOSE μSv/s
1 - Base Liner	839.2 (0.018)	8.326 E-4 (0.034)
2 - Botton Quench Tank	125.0 (0.022)	1.720 E-4 (0.038)
3 - Front Fe Liner	952.0 (0.020)	2.953 E-4 (0.061)
4 - Back Plate	93.5 (0.011)	0.210 E-4 (0.052)
5 - Back Liner up	762.5 (0.019)	2.233 E-4 (0.051)
6 - Back Liner down	58.8 (0.033)	1.605 E-5 (0.108)
7 - Front Quench Tank	109.0 (0.031)	1.117 E-4 (0.170)
8 - Lateral Liner Back left	261.0 (0.033)	3.875 E-5 (0.123)
9 - Lateral liner Back right	258.4 (0.031)	4.769 E-5 (0.132)
10 - Lateral Liner Forward left	506.0 (0.026)	1.000 E-4 (0.273)
11 - Lateral Liner Forward right	498.6 (0.033)	2.207 E-4 (0.574)
TOTAL	4464.0 [16.1 Sv/h]	2.080 E-3 [7.49 μSv/h]

Tab. IX - Dose rate in the Access Cell floor level and ceiling level at shut down after 1 hour of cooling time (see figg. 29 and 30).

MCNP Geometry Model for
Access, Service & Test Handling Cell: Plan View
(Units: cm)



MCNP Geometry Model
for Access, Service & Test Handling Cell: Elevation Section
(Units: cm)



ACCESS, SERVICE & HANDLING CELLS SHIELDING CALCULATIONS: *Walls*

Dose Rate (μSv/hr) from Test Module Neutron Activation vs. Cooling Time

Common Assumptions:

- Shielding thickness: 1 m of TSF-5.5 (2.35 g/cc)
- Test Module distance from the wall: 1 m
- Test Module Volume: 500 cc at 3.93 g/cc

COOLING TIME / IRRADIATION TIME	At shut-down (T = 0)	1 Day	1 Week
1 Yr (Iron)			9.85 (±14%)
5 Yrs (ANL)	1.28E+5 (±5%)		8.19E+3 (±4%)
5 Yrs (JAERI)	4.53E+4 (±3%)		1.00E+2 (±2%)
20 Yrs (70% f.p.)	3.83E+4 (±7.8%)	3.73E+2 (±3.5%)	9.85E+1 (±12%)

ACCESS, SERVICE & HANDLING CELLS SHIELDING CALCULATIONS: *Ceiling*

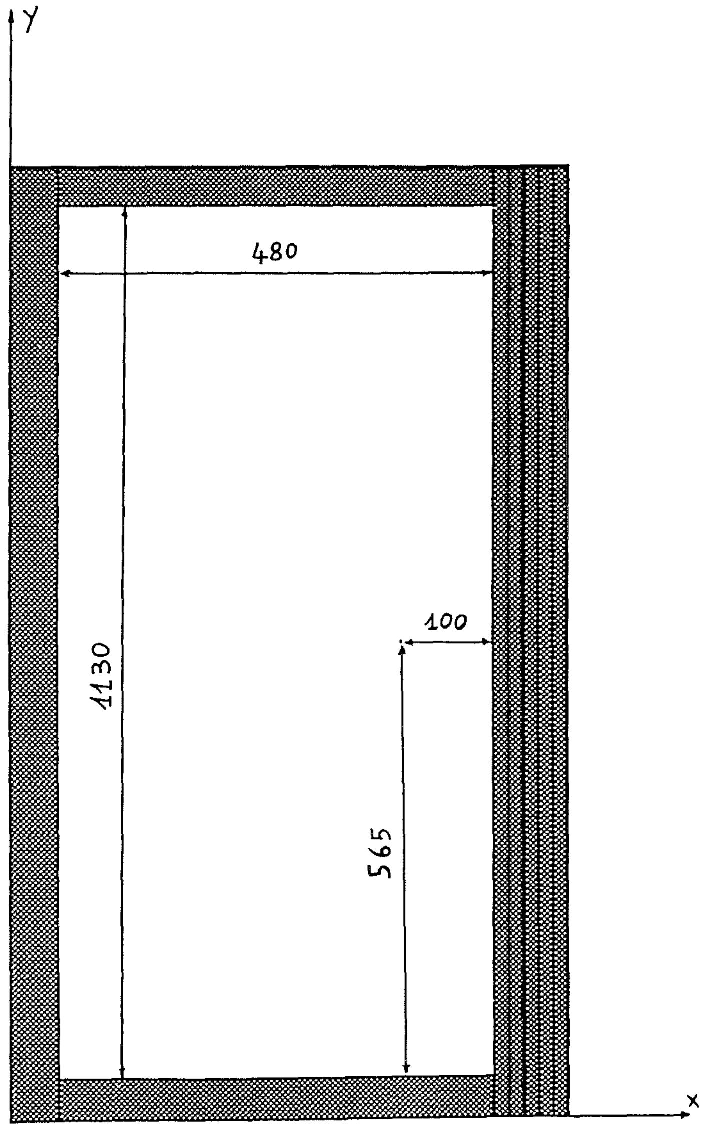
Dose Rate (μSv/hr) from Test Module Neutron Activation vs. Cooling Time

Common Assumptions:

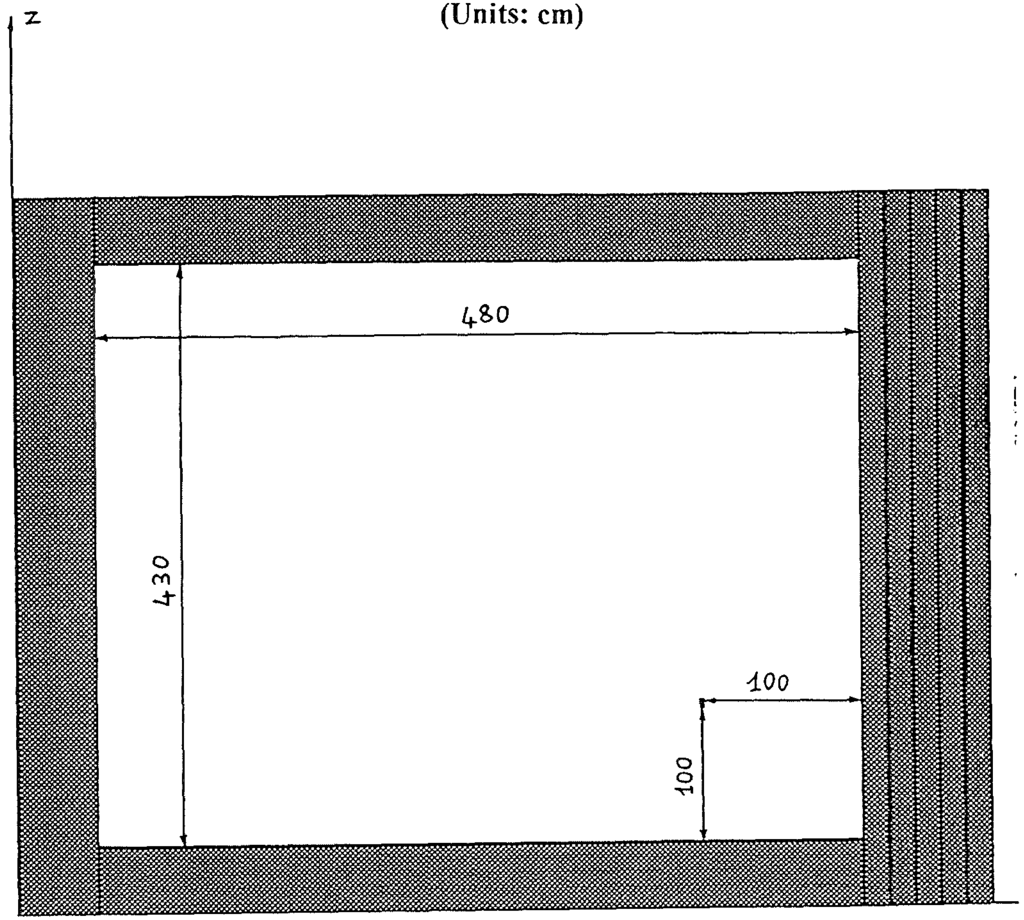
- Shielding thickness: 1 m of TSF-5.5 (2.35 g/cc)
- Test Module distance from the ceiling: 4.0 m
- Test Module Volume: 500 cc at 3.93 g/cc

COOLING TIME / IRRADIATION TIME	At shut-down (T = 0)	1 Day	1 Week
1 Yr (Iron)			
5 Yrs (ANL)	1.16E+4 (±7%)		6.52E+2 (±8%)
5 Yrs (JAERI)	3.90E+3 (±9%)		8.15 (±4%)
20 Yrs (70% f.p.)	3.01E+3 (±22.9%)	2.84E+1 (±13.9%)	~ 8.0

MCNP Geometry Model
for Conventional Hot Cell Laboratory: Plan View
(Units: cm)



MCNP Geometry Model
for Conventional Hot Cell Laboratory: Elevation Section
(Units: cm)



CONVENTIONAL HOT CELL LABORATORY SHIELDING CALCULATIONS: *Walls*

Dose Rate (μSv/hr) from 12-Miniaturized Specimens Neutron Activation vs. Cooling Time for Different Shields

- Source distance from the wall: 1 m
- Specimen Total Volume: 3.3 cc at 7.86 g/cc

Shield / Cooling Time	Concrete TSF-5.5 (cm)				Lead (cm)		
	80	90	92	100	15	16	21
1 Day		1.68E+1 (±11%)	1.21E+1 (±4.7%)		1.13E+2 (±8.1%)		7.29 (±1.5%)
1 Week	9.15 (±4%)			1.70 (±12%)		1.22E+1 (±2.4%)	

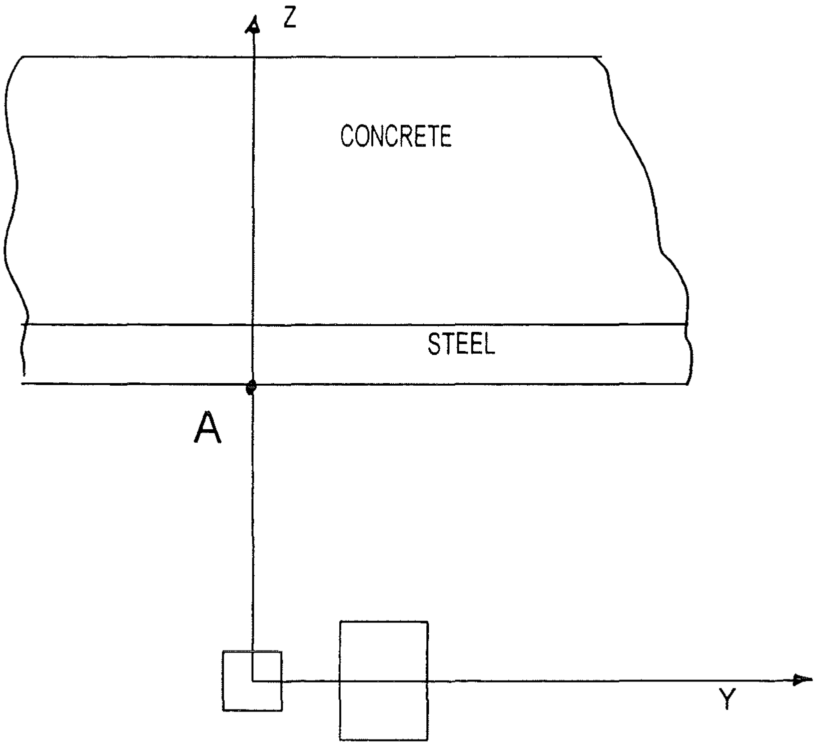
CONVENTIONAL HOT CELL LABORATORY SHIELDING CALCULATIONS: *Ceiling*

Dose Rate (μSv/hr) from 12-Miniaturized Specimens Neutron Activation vs. Cooling Time for Different Shields

- Source distance from the ceiling: 3.3 m
- Specimen Total Volume: 3.3 cc at 7.86 g/cc

Shield / Cooling Time	Concrete TSF-5.5 (cm)			Lead (cm)	
	50	60	73	12	15
1 Day	9.97E+1 (±9.1%)		1.1E+1 (±6.5%)		1.18E+1 (±4.5%)
1 Week		1.15E+1 (±10%)		8.1 (±2.3%)	

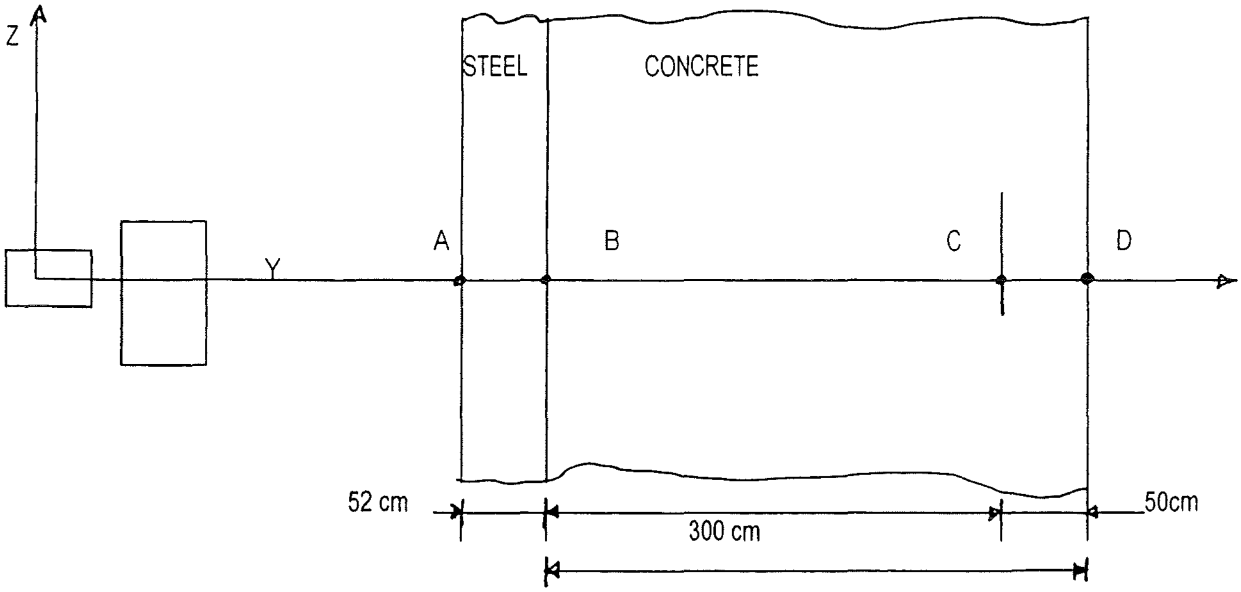
CONFRONTO SUPERIORE



	[redacted] JAERI n- source			[redacted] ANL n - source		
	Neutron Dose µSv/h	Gamma Dose µSv/h	Total Dose µSv/h	Neutron Dose µSv/h	Gamma Dose µSv/h	Total Dose µSv/h
A	1.37E+12 0.05	8.27E+9 0.10	1.38E+12	1.94E+12 0.06	9.79E+9 0.16	1.95E+12

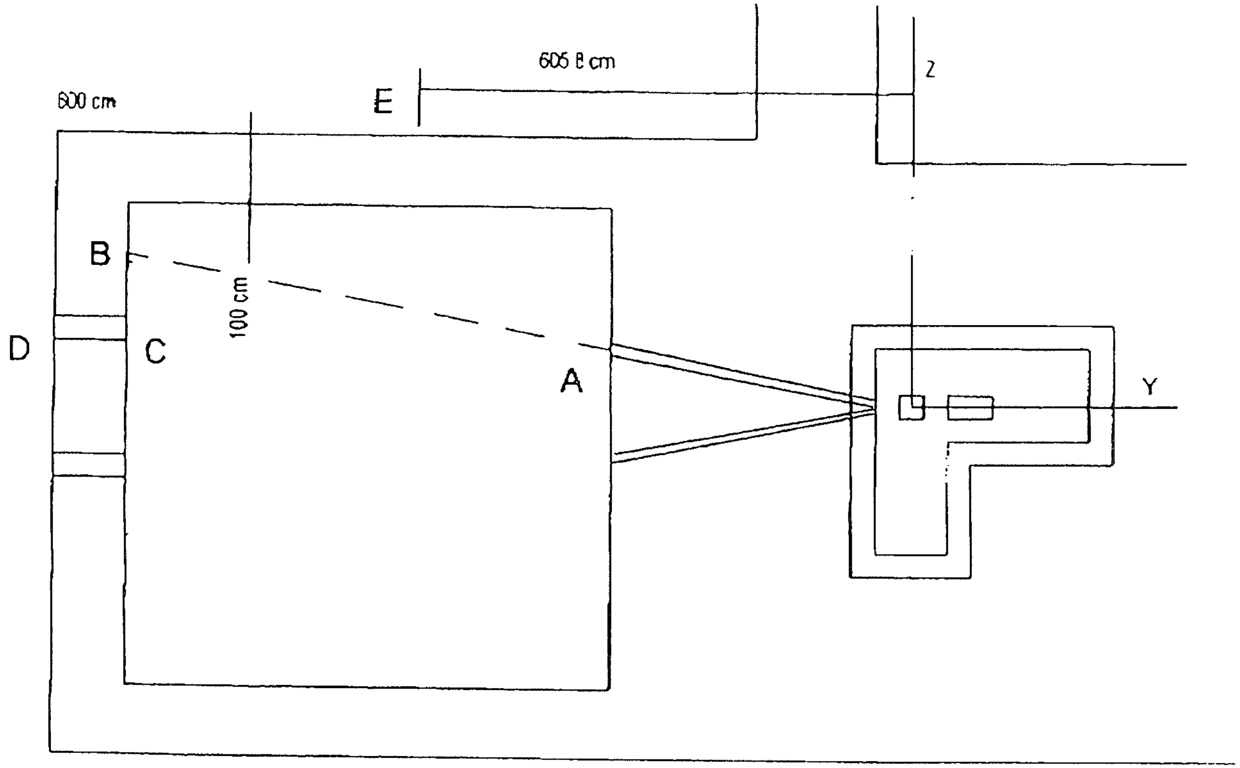
Nella Zona superiore ci sono dosi maggiori a contatto rispetto alla vecchia subroutine

CONFRONTO FRONTALE
100.000 particelle



	[redacted] JAERI			[redacted] ANL		
	Neutron Dose μSv/h	Gamma Dose μSv/h	Total Dose μSv/h	Neutron Dose μSv/h	Gamma Dose μSv/h	Total Dose μSv/h
A	1.61E+12 0.01	8.37E+9 0.01	1.62E+12	1.47E+12 0.01	5.93E+9 0.02	1.48E+12
B	5.49E+10 0.01	1.41E+9 0.02	5.63E+10	3.82E+10 0.03	1.20E+9 0.04	3.94E+10
~~C~~	~~1.55~~ ~~0.28~~	~~187~~ ~~0.11~~	~~188.55~~	~~0.15~~ ~~0.35~~	~~140~~ ~~0.23~~	~~140.15~~
~~D~~	~~1.93E-3~~ ~~0.26~~	~~9.62~~ ~~0.16~~	~~9.62~~	~~1.58E-4~~ ~~0.47~~	~~11.74~~ ~~0.43~~	~~11.74~~

to be improved in statistics



Calculation points	Neutron Dose μSv/h	Gamma Dose μSv/h	Total Dose μSv/h
A	2.11E+10	4.07E+8	2.15E+10
B	2.90E+9	3.74E+7	2.94E+9
C	2.83E+8	2.60E+6	2.86E+8
D	2.45E+6	1.32E+6	3.77E+6
E	17.3	203.5	220.8

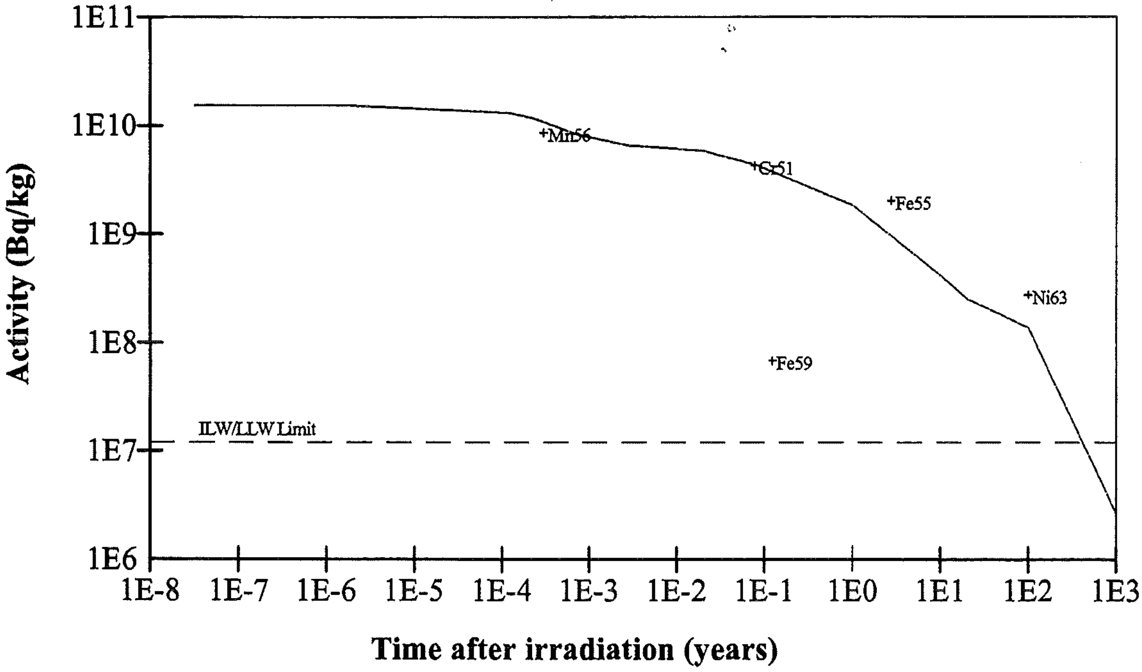
Calculation points location for
penetration radiation streaming

Energy spectral distributions of gamma sources on HEBT for various cooling times

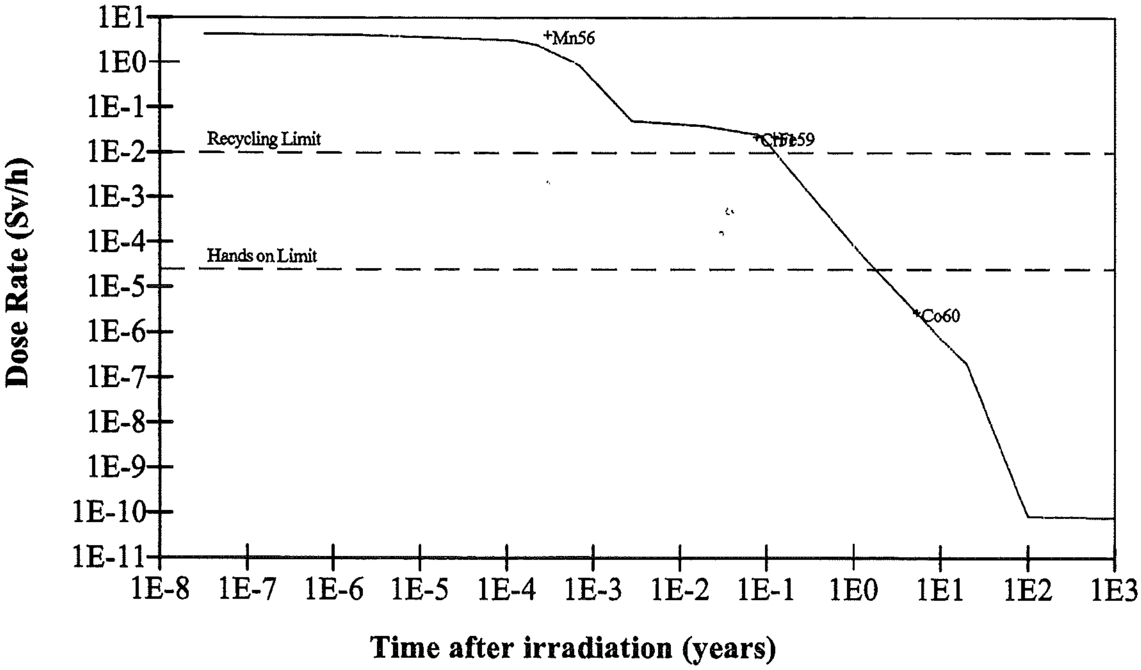
(gamma/cm^{3}s)

Energy (MeV)	T = 0	T = 1s	T= 1m	T = 1h	T = 2h	T = 6h	T=1day	T=1week	T=1month	T=1year	T=10years	T=20years	T=100y	T=1000y
0,0 -0,01	1,27E+07	1,27E+07	1,27E+07	1,27E+07	1,27E+07	1,27E+07	1,25E+07	1,15E+07	8,62E+06	4,14E+06	4,31E+05	4,28E+04	9,33E+03	9,25E+03
0,01-0,02	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
0,02-0,05	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
0,05-0,1	4,86E-01	4,85E-01	4,55E-01	6,77E-10	6,77E-10	6,77E-10	6,77E-10	6,77E-10	6,77E-10	6,77E-10	6,77E-10	6,77E-10	6,77E-10	6,77E-10
0,1 -0,2	2,67E+04	2,67E+04	2,67E+04	2,67E+04	2,66E+04	2,66E+04	2,63E+04	2,39E+04	1,65E+04	8,92E+01	0,00E+00	0,00E+00	0,00E+00	0,00E+00
0,2 -0,3	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
0,3 -0,4	3,03E+06	3,03E+06	3,03E+06	3,01E+06	3,00E+06	2,98E+06	2,92E+06	2,52E+06	1,38E+06	3,21E+02	6,17E-04	1,66E-04	4,48E-09	0,00E+00
0,4 -0,6	9,84E+02	9,84E+02	9,80E+02	7,49E+02	5,70E+02	1,93E+02	3,17E+00	6,04E-03	6,04E-03	6,04E-03	6,04E-03	6,04E-03	6,03E-03	5,98E-03
0,6 -0,8	9,43E+02	9,42E+02	9,18E+02	5,70E+02	4,33E+02	1,44E+02	1,02E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
0,8 -1,0	6,30E+07	6,30E+07	6,27E+07	4,81E+07	3,68E+07	1,26E+07	9,95E+04	2,17E-03	2,16E-03	1,91E-03	5,85E-04	1,57E-04	4,25E-09	0,00E+00
1,0 -1,22	3,80E+05	3,80E+05	3,80E+05	3,61E+05	3,47E+05	3,16E+05	2,97E+05	2,71E+05	1,86E+05	1,04E+03	8,85E+00	2,38E+00	6,43E-05	0,00E+00
1,22-1,44	3,00E+05	3,00E+05	3,00E+05	2,82E+05	2,68E+05	2,39E+05	2,22E+05	2,02E+05	1,39E+05	7,80E+02	8,40E+00	2,26E+00	6,10E-05	0,00E+00
1,44-1,66	7,95E+04	7,95E+04	7,91E+04	6,03E+04	4,59E+04	1,55E+04	3,91E+02	2,58E+02	1,78E+02	9,62E-01	0,00E+00	0,00E+00	0,00E+00	0,00E+00
1,66-2,0	1,84E+07	1,84E+07	1,83E+07	1,41E+07	1,08E+07	3,67E+06	2,91E+04	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
2,0 -2,5	9,12E+06	9,11E+06	9,07E+06	6,97E+06	5,33E+06	1,82E+06	1,44E+04	3,23E-04	3,20E-04	2,84E-04	8,69E-05	2,33E-05	6,31E-10	0,00E+00
2,5 -3,0	1,31E+06	1,31E+06	1,30E+06	1,00E+06	7,64E+05	2,61E+05	2,07E+03	1,42E-06	1,41E-06	1,25E-06	3,82E-07	1,03E-07	2,78E-12	0,00E+00
3,0 -4,0	1,03E+05	1,03E+05	1,03E+05	7,88E+04	6,02E+04	2,06E+04	1,63E+02	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
4,0 -5,0	1,36E-03	1,36E-03	1,19E-03	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
5,0 -6,5	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
6,5 -8,0	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
8,0 -10,0	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
10,0-12,0	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
12,0-14,0	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
14,0--->	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00	0,00E+00
Total	1,08E+08	1,08E+08	1,08E+08	8,67E+07	7,01E+07	3,46E+07	1,62E+07	1,45E+07	1,03E+07	4,14E+06	4,31E+05	4,28E+04	9,33E+03	9,25E+03

Irradiation of HEBT (Point A)



Irradiation of HEBT (Point A)



Dosimetry for IFMIF

B. Esposito - Enea Frascati

Need for test assembly neutron measurements:

- characterization of neutron field (spectrum/flux vs position, effect of spectrum on damage parameters, spectral gradient)
- monitoring for stability control
- validation of neutronic calculations

Redundancy of detectors:

- passive detectors: multi-foil/wire activation
- on-line detectors: miniaturized fission chambers
- others

Severe environment (T < 1000 °C, high radiation level, long irradiation times: ~9 months)

Expected flux in high-flux test module: 0.8×10^{14} n.cm^{-2}.s^{-1} ==> 2.5×10^{21} n.cm^{-2}.y^{-1}

Neutron energy spectrum range: < 50 MeV, with ~20% of total yield above 15 MeV

IFMIF-CDE Test Facility Workshop - Karlsruhe, July 7-9 1997

Fraction of displacement damage in nickel as a function of neutron energy (Be(n,d) source, E_d=40 MeV)

Energy Range (MeV)	Damage (%)
< 2	5
2 - 5	5
5 - 10	15
10 - 15	25
15 - 20	25
20 - 25	15
25 - 30	5
> 30	5

Reference: L.R. Greenwood, *The status of neutron dosimetry and damage analysis for the fusion materials program*, Proceedings of the International Conference on Nuclear Cross Sections for Technology, Knoxville, (1979)

Foil Activation

technique

irradiation of multiple foils/wires => γ-ray spectrometry => neutron spectrum unfolding (e.g. using SNL SAND-II code including analysis of uncertainties in cross sections, activation measurements and input spectra)

output

- time-integrated neutron flux
- neutron energy spectrum
  - wires ==> good spatial resolution, flux gradients
  - foils ==> accurate absolute data

advantages

- virtually no limits on dynamic range of neutron flux detection
- no electronics, no cables, radiation resistant

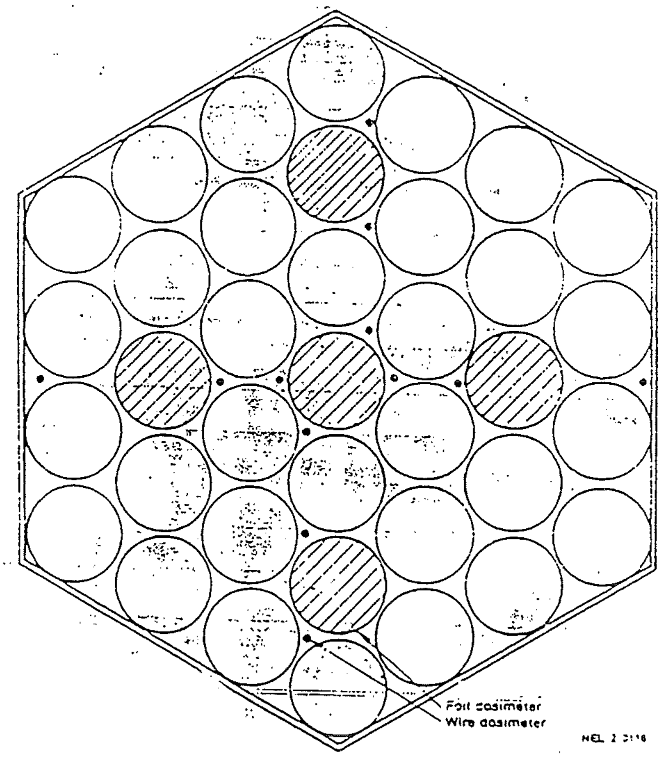
disadvantages

- do not provide on-line measurement

routinely used in

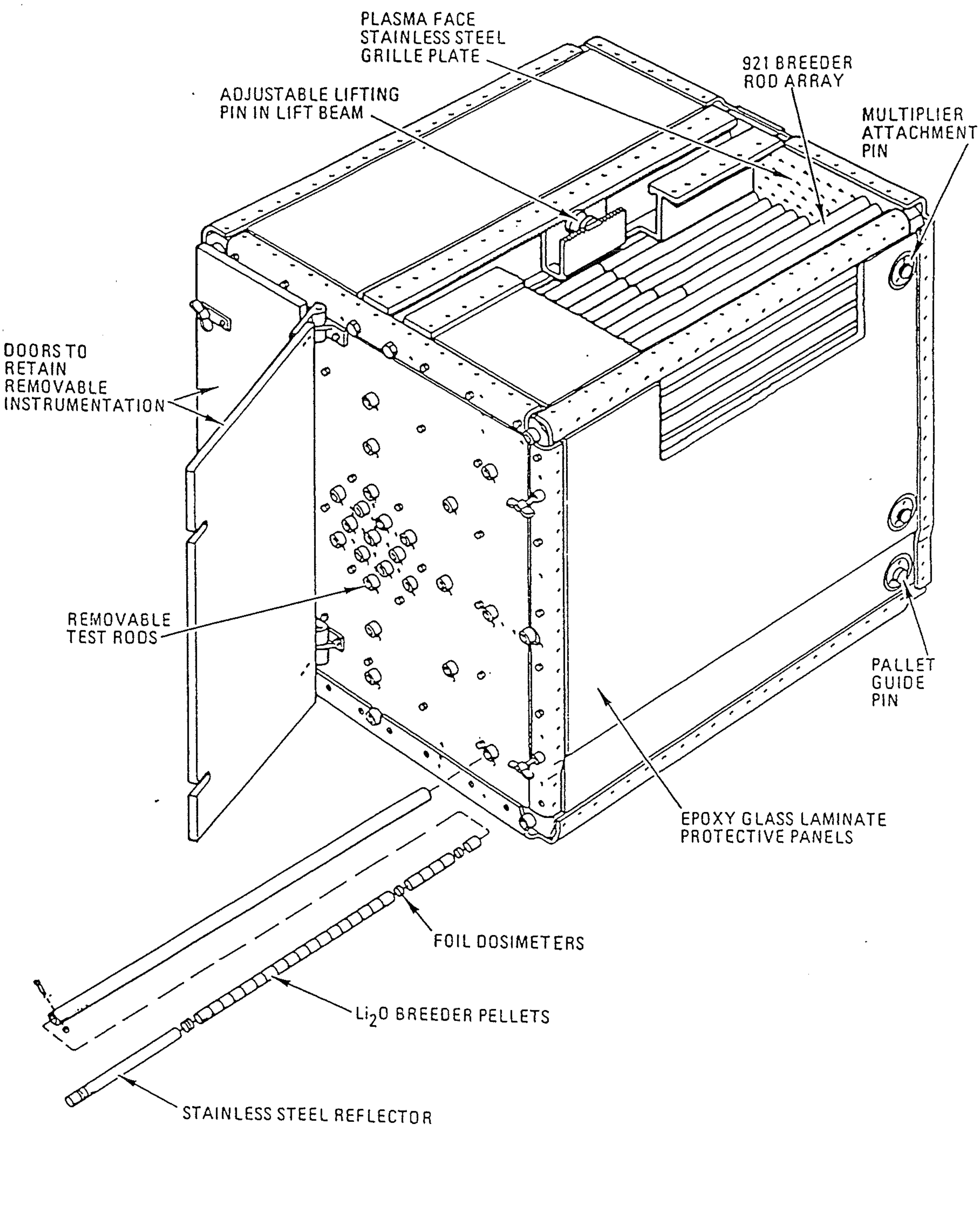
- fast fission reactors ($E_n < 10$ MeV)
- experimental fusion reactors: JET, TFTR ($E_n < 14$ MeV)

IFMIF-CDE Test Facility Workshop - Karlsruhe, July 7-9 1997



Example of Foil and Wire Dosimeter Loading

From: Y.D. Harker et al., Neutron Dosimetry for the Li-Blanket Program, Proc. 4th ASTM-EURATOM Symp. On Reactor Dosimetry (1982)



Example of Wire Dosimeter Loading in Li Blanket Module

From: The LBM Program at the Lotus Facility, J. File et al., Trans ANS, 52, 209 (1986)

problems

cross section dosimetric status poor above 20 MeV

present status

- dosimetry package suitable to IFMIF (see Table) ==>

  14-reaction set + SAND-II spectrum unfolding can provide IFMIF neutron energy spectrum with following uncertainties (assuming zero uncertainties on cross sections):

  10% for 1 MeV < E_n < 25 MeV
  30% for 25 MeV< E_n < 40 MeV

- sensitivity test of the neutron energy spectrum to uncertainties in cross sections data ==>

  20-50% uncertainty on single cross section may cause ~20% errors on final spectrum, while multiple 20% uncertainties may lead to 80% errors

- from earlier analysis using integral cross section testing on Be(d,n) spectra (Greenwood, 1979) it was estimated that flux and spectral accuracies of ~30% should be achievable in the 2-30 MeV range, but most comprehensive measurements and calculations of cross sections for activation were strongly recommended

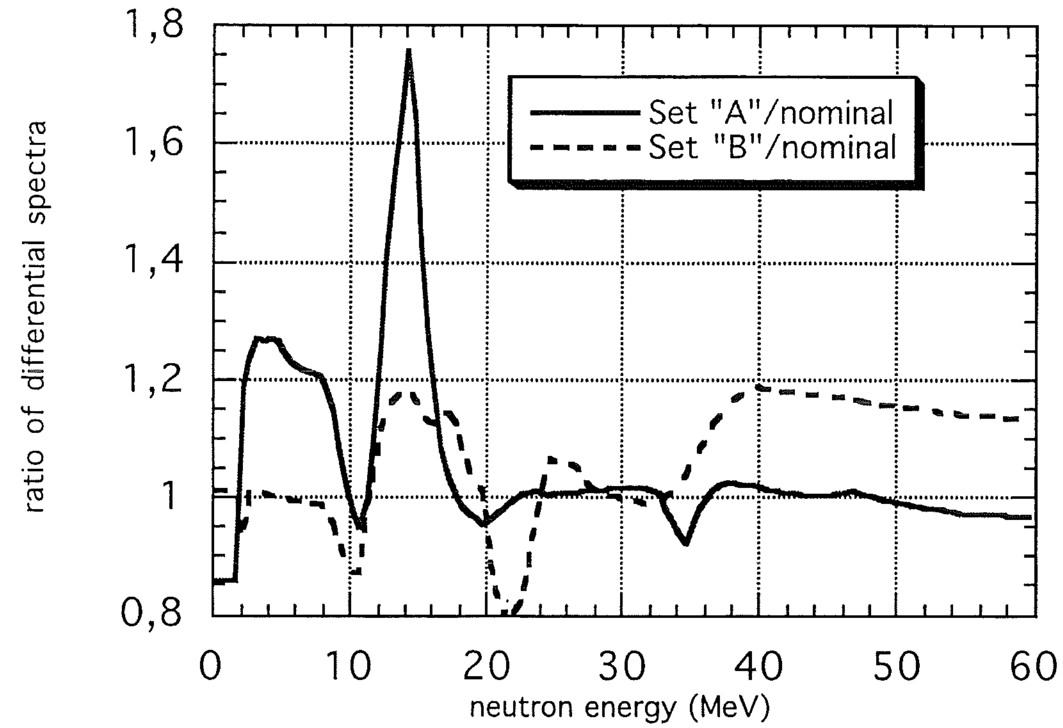
- considerable discrepancies exist between various evaluations (ENDF/B-VI, JENDL, BROND) of relevant dosimetry activation cross sections

IFMIF-CDE Test Facility Workshop - Karlsruhe, July 7-9 1997

Table I: 14-reaction set for test cell neutron dosimetry (1-50 MeV)

Reaction	Q-Value (MeV)	Half-Life	Isotopic Abundance (%)	γ - energy (keV)	γ - abundance (%)	Melting Point (°C)	ENDF/B-VI (MeV)
^{58}Ni(n,p)^{58}Co	0.4020	71.6 d	68.3	810.7	99.44	1453	<20
^{46}Ti(n,p)^{46}Sc	-1.58	83.8 d	8.0	889, 1120	99.98, 99.98	1660	<20
^{93}Nb(n,n')^{93}Nb^m	-2.17	16.1 y	100	16.6	0.092	2468	<20
^{197}Au(n,2n)^{196}Au	-8.06	6.18 d	100	355.7	87.7	1064.4	<30
^{59}Co(n,2n)^{58}Co	-10.45	70.8 d	100	810	99	1495	<19
^{89}Y(n,2n)^{88}Y	-11.47	106.6 d	100	1840	99.3	1523	<19
^{175}Lu(n,3n)^{173}Lu	-14.4	499 d	97.4	270	13	1656	<20
^{197}Au(n,3n)^{195}Au	-14.8	183 d	100	100	11	1064.4	<30
^{93}Nb(n,3n)^{91}Nb^m	-16.7	62 d	100	100	0.56	2468	<20
^{59}Co(n,3n)^{57}Co	-20	271.6 d	100	122	85.6	1495	-
^{90}Zr(n,3n)^{88}Zr	-21.5	83.4 d	51.5	394	97.3	1852	-
^{55}Mn(n,4n)^{52}Mn	-31.8	5.59 d	100	700, 1430	90, 100	1244	-
^{59}Co(n,4n)^{56}Co	~ -32	78.76 d	100	847	99.2	1495	-
^{103}Rh(n,5n) ^{99}Rh^g	~ -37	15 d	100	528	38	1966	-

Fig. 7.3: Effect of Uncertainties on Cross Sections



Set "A"	
Ni58Co	+20% norm
Ti46	+20% norm
Nb93	+20% norm
Au1972	+20% norm
Co592	+20% norm
Y89	+20% norm

Set "B"	
Co592	+20% norm
Au1973	+20% lin thr
Nb933	-20% norm
Co593	-20% norm
Zr90	-20% lin thr
Mn54	+20% norm
Co594	+30% norm
Rh1035	30% norm

Fig. 7.4: Effect of changing Au1973 cross section

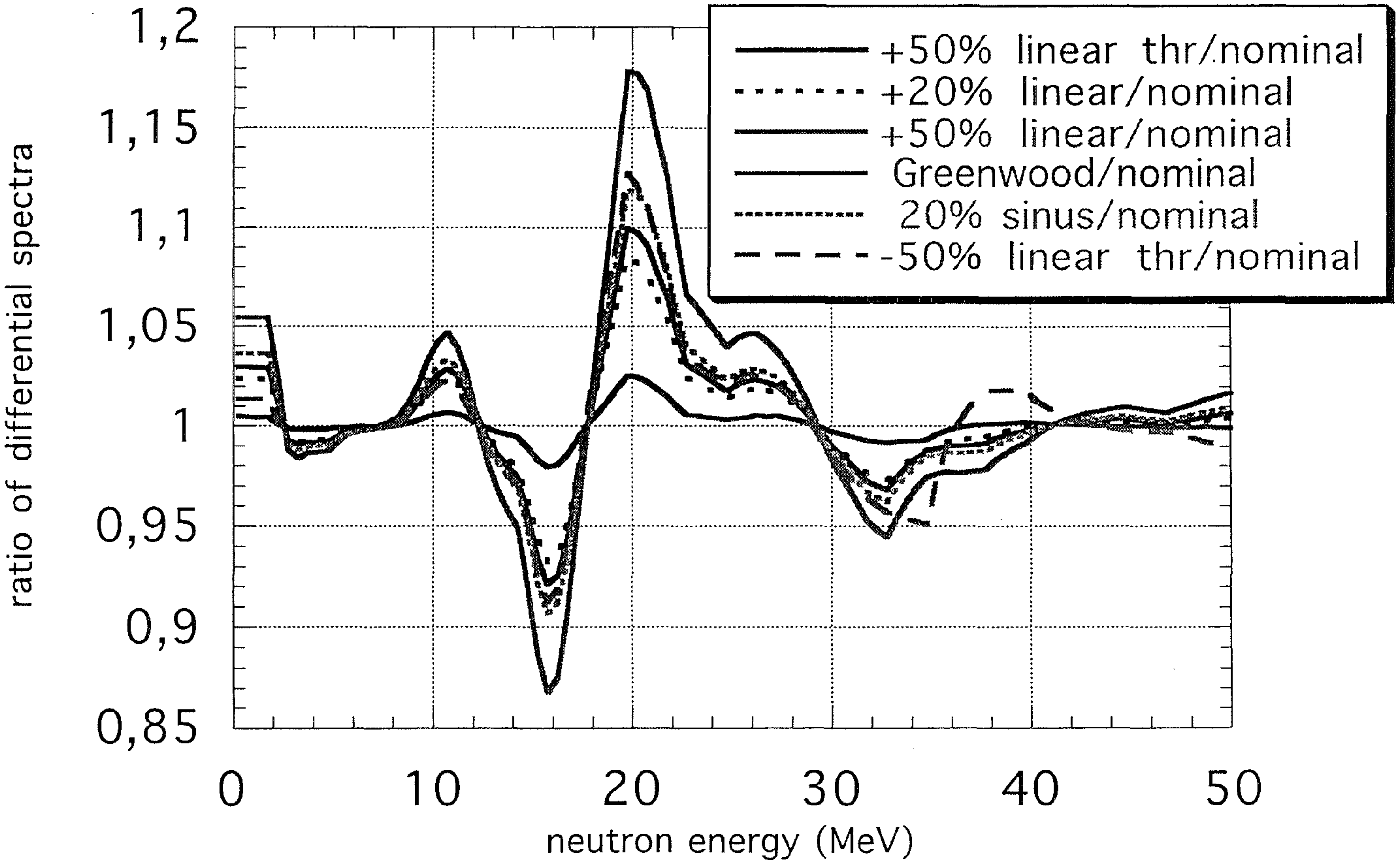


Fig. 7.5: Au1973 cross sections

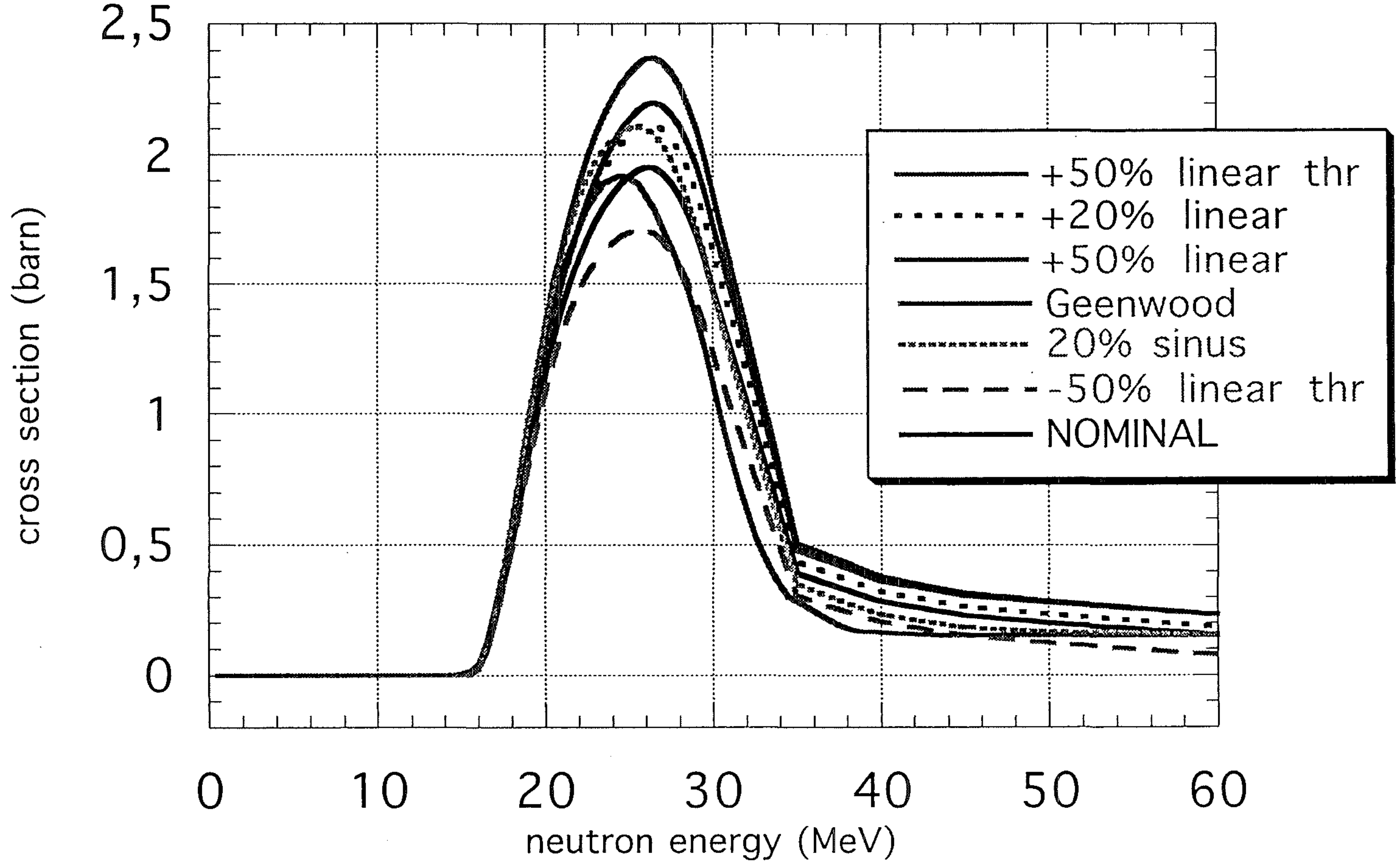
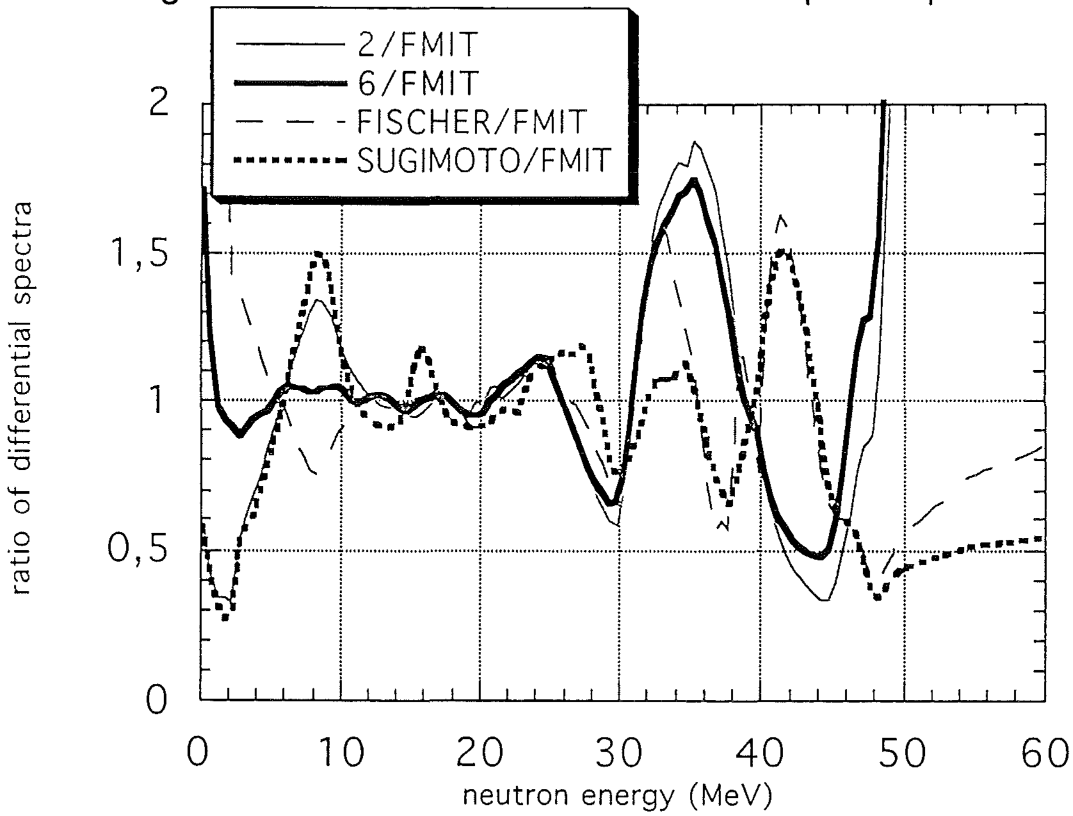


Fig. 7.1: Effect of Different Input Spectra



a set of neutron activation reactions with <u>well-measured cross sections</u> will provide an adequate spatial mapping of the neutron energy spectrum and flux in IFMIF test cell

<u>development</u>

- experimental qualification of selected dosimetry package on existing high-energy accelerator sources, e.g. using Be(d,n) reaction
- use of cross section covariances for improved sensitivity analysis
- measurements of cross sections at least for few relevant reactions (^{59}Co(n,xn), ^{197}Au(n,xn) have high priority)

On-line Detectors

technique

miniaturized ionization chambers ==>

coating: ^{235}U (mainly sensitive to thermal neutrons)
(^{238}U chambers, sensitive to fast neutrons only (> 1 MeV), have very low efficiency and the neutron signal would be probably overwhelmed by γ-rays)

output

spatial and time-resolved neutron flux monitoring (beam stability control)

advantages

^{235}U miniaturized models available commercially (e.g.: Philips CFUR 43) ==>

filling gas: Ar
pressure: 110 kPa
external diameter: 3 mm
length: 45 mm
maximum cable length: 80 m
cable diameter: 1 mm
neutron flux range: 10^{11} - 1.5×10^{14} n.cm^{-2}.s^{-1} (current mode @ 20 °C)
maximum operating temperature < 350 °C

disadvantages

- burn-up of neutron sensitive material ==>

sensitivity of ^{235}U chamber decreases of 50% after integrated fluence of 2×10^{21} n.cm^{-2}, but use of regenerative models ($^{234}U/^{235}U$) may extend range up to 5×10^{21} n.cm^{-2}
(a 1.5 mm diameter regenerative chamber is available from Philips)

- sensitivity to γ-rays if used in current (or fluctuation) mode ==>

maximum γ-ray flux: 10^7 Gy.h^{-1}
pulse mode operation is free from γ-ray contamination, but limited to ~10^{10} n.cm^{-2}.s^{-1}
(may be suitable to lower flux test modules)

routinely used in

fission reactors (in-core and out-of core)

development

- verify possibility of use in IFMIF environment, investigate if modifications are required
- effect of γ-rays, calculation of γ-ray flux necessary for proper design
- analysis of possible additional/alternative on-line monitors: self-powered detectors, γ-ionization chambers (survey of present reactor in-core radiation monitoring instrumentation)

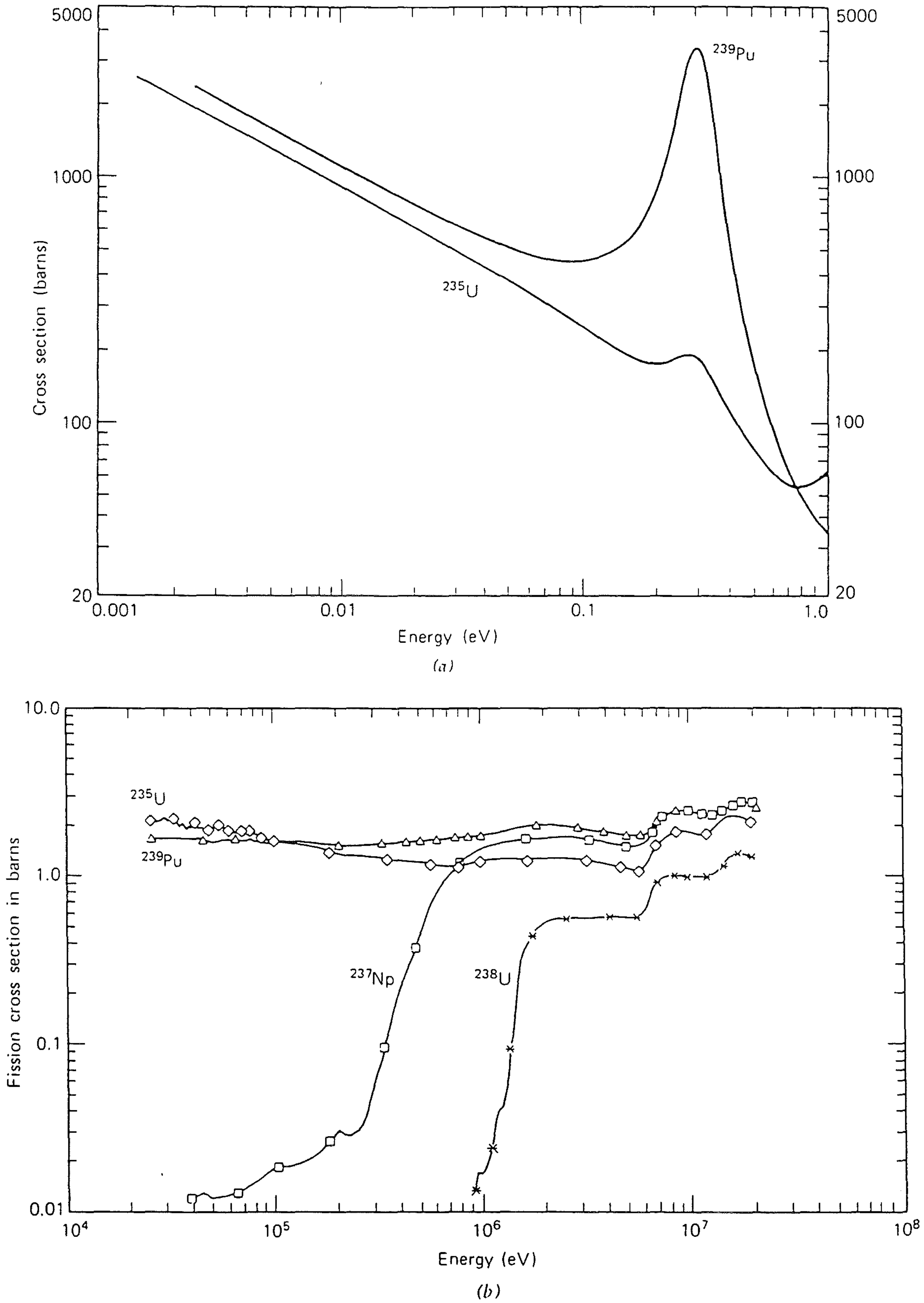
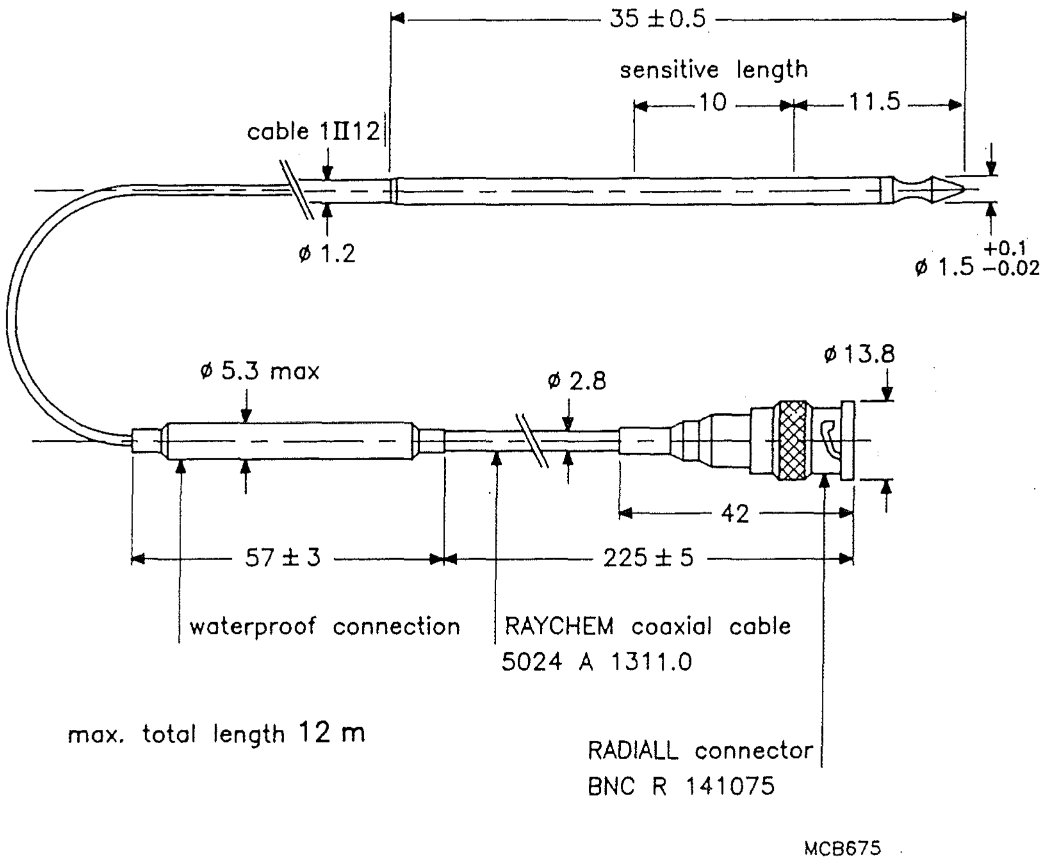


FIGURE 14-2. Fission cross sections of some common target nuclides used in fission chambers. Part (a) includes the slow neutron region where the cross sections shown are relatively large. The fast neutron region is shown in (b). Chambers with ^{237}Np or ^{238}U are used as "threshold detectors" sensitive only to fast neutrons.


35 ±0.5
sensitive length
10
11.5
cable 1II12
ø 1.2
+0.1
ø 1.5 −0.02
ø 13.8
ø 5.3 max
ø 2.8
42
57 ± 3
225 ± 5
waterproof connection
RAYCHEM coaxial cable
5024 A 1311.0
max. total length 12 m
RADIALL connector
BNC R 141075
MCB675

CFUZ 53R

IFMIF CDE Phase

1st Test Facility Group Meeting
July 7, 1997 to July 9, 1997

DPA to volume relation and related topics

(Progress from October 1996 to June 1997, CDE Task 14)

Presented by E. Daum

Outline

1. Comparison of uncollided / collided INS and MCNP calculations
2. Effect of the present HFTM design on the volume
3. Volume gain by improvement of the current design
4. Other possibilities to enhance the volume
5. Placement and orientation of the specimen
6. Qualification of the displacement damage

Summary

The information given in this presentation can mainly be found in the report of the Forschungszentrum Karlsruhe FZKA 5868, published in June 1997. The scope of this presentation is to cover following:

- Comparison of uncollided and collided INS and MCNP calculations
- Show the effect of the present High Flux Test Module (HFTM) design on the available test volume
- Discuss the volume gain by design improvements and other possibilities
- Discuss the effect of positioning and orientation of the specimens on the gradients
- Discuss the qualification of the displacement damage in IFMIF

The main user requirements to be fulfilled in the HFTM are a neutron flux gradient of less than 10 %/cm based on minimum specimen dimensions and a test volume of at least 0.5 liter for > 20 DPA/FPY.

The comparison of the volume calculations shows that the total volume (> 20 DPA/FPY) varies between 450 cm^3 and 730 cm^3 while the volume inside the HFTM varies between 390 cm^3 and 480 cm^3. This variation comes on one hand from the usage of the different codes (INS and MCNP) but mainly it is due to the fact, that the neutron source function has an uncertainty of +/- 20 %. The neutron yield used in INS calculations was 6.7 % and in the MCNP calculations 7.3 %.

The present design of the He-cooled HFTM limits the available test volume to 0.5 liter because of its box shape. As in the 3-dimensional volume plot can be seen the ellipsoidal iso-surface for 20 DPA/FPY covers a not negligible volume fraction which is located above and below the physical boundaries of the He-cooled HFTM. This implies, that additional test volume (max. 200 cm^3) can be gained by improving the present HFTM design. On the other hand, due to the n-yield uncertainty, there is presently still a large uncertainty in the available test volume.

An other possibility of enhancing the high flux volume is the reduction of the distance between the Li-jet free surface (accelerator side) and the specimen in the HFTM (e.g. reduction of Li-jet thickness, reduction of gap between backwall and HFTM, etc.).

One of the user requirements is the n-flux gradient. If comparing the n-flux gradient and the DPA-gradient contour plots, it is obvious, that both gradients behave similarly. Therefore the user requirement for n-flux gradient can also expressed as dpa-gradient. This is very important for the positioning and orientation of the miniaturized specimens. Each specimen has its sensitive region where the user requirements have to be fulfilled. Since the strongest gradients are in downstream direction and are at any location larger than 10 %/cm , the tensile properties specimen for example should be placed in an orientation that dpa-gradient times specimen width will be minimized. This guarantees that over the specimen width in the sensitive region the dpa-gradient will be less than or equal 10 %.

The characterization of the quality of the displacement damage is necessary to demonstrate how suitable IFMIF is for simulation irradiations. One very important quantity in the damage model is the PKA-spectrum. With both available, the PKA-spectra and the n-spectra, a displacement damage characterization can be done which is expressed by the W(T) function. For now, it is expected that in IFMIF a somewhat harder displacement damage appears than in a DEMO first wall.

USER Requirements in the HFTM

- **Neutron flux gradients less than 10 % / cm based on minimum specimen dimensions**
- **0.5 liter volume with at least 20 DPA/FPY**

Comparison of uncollided / collided INS and MCNP calculations

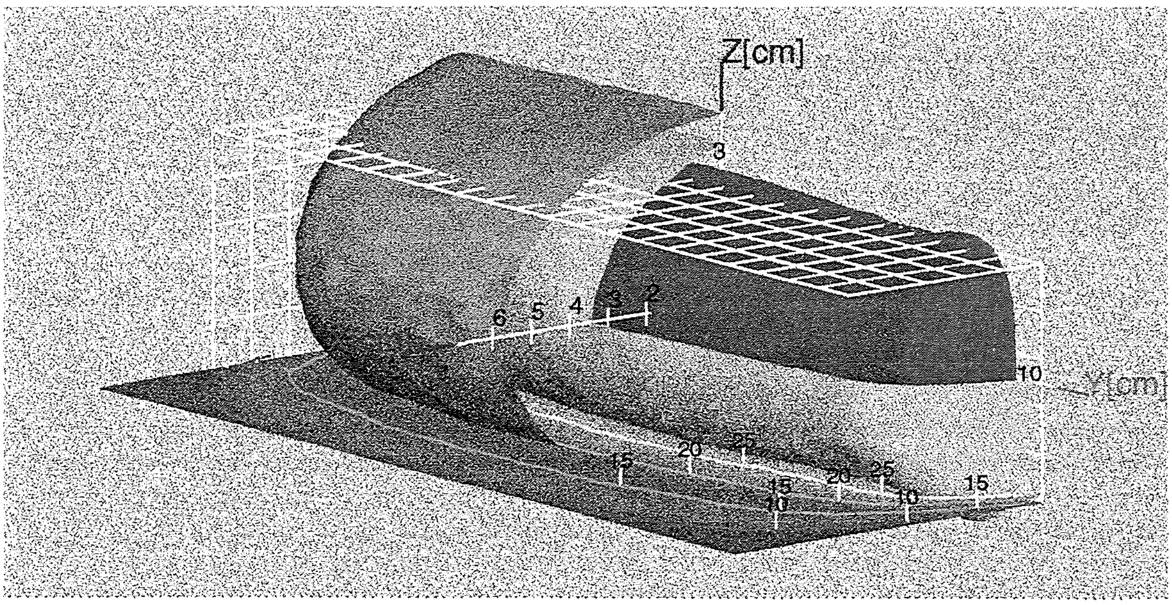
Calculation parameters:

Part	Parameter	Value
Accelerator	E_d	40 MeV
	ΔE_d	not considered
	I	250 mA
	Angle	0°
	Beam profile	Uniform, non-uniform
	Beam footprint	5 x 20 cm^2 (uniform), 7 x 22 cm^2 (non-uniform)
Target	d_{Li}	2.6 cm
	T_{Li}	300 °C, $\rho_{Li} = 0.512$ g/cm^3
	D_{Range}	ZBL-Theory [Zie 85]
	ΔD_{Range}	not considered
	d_B	1.6 mm
	r_B	none
	n-source model	BNL data [Man 91]
Test cell	Calculation code	INS
	Sort of calculation	Uncollided calculations
	Loading parameters	none
	Nuclear response	[Gre 92]
	Transport data	none
	Kind of results	Point values

Volume comparison for > 20 DPA/FPY

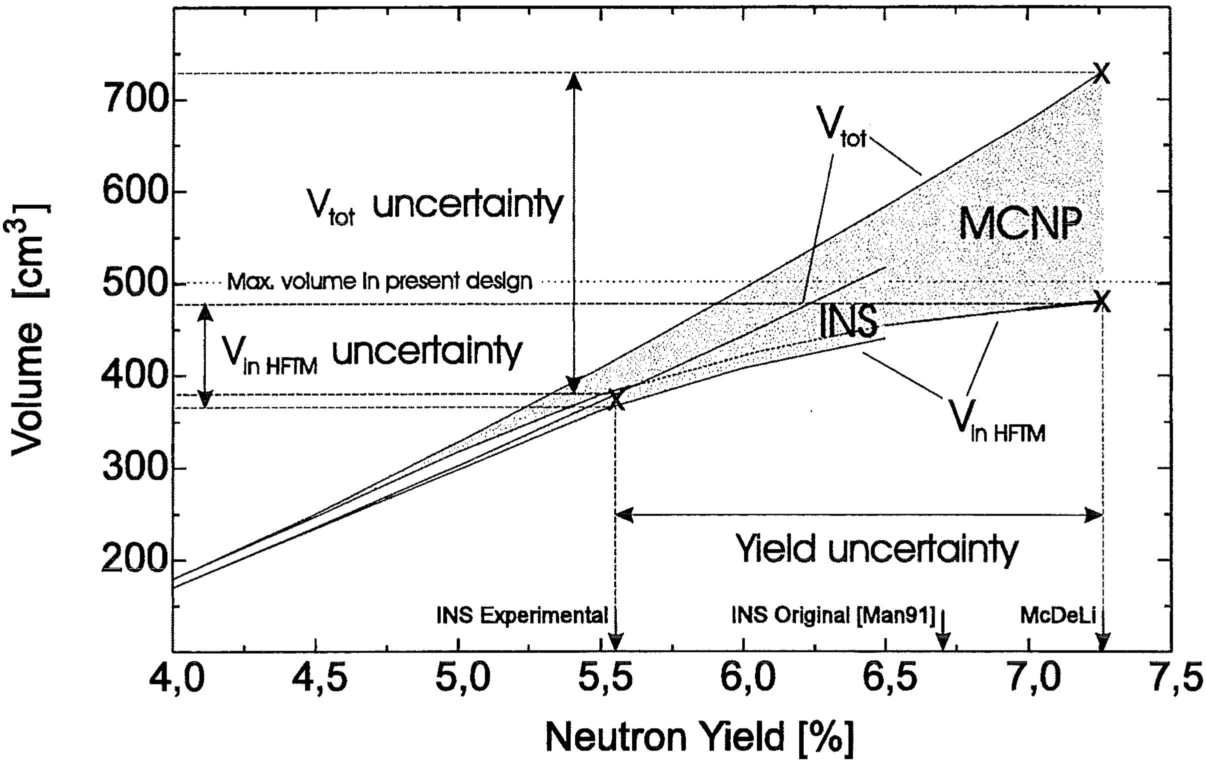
	Total volume [cm^3]	Volume inside HFTM [cm^3]
INS, uniform	466	419
INS, non-uniform	451	389
MCNP, uniform, uncollided	732	478
MCNP, non-uniform, uncollided	716	480
MCNP, uniform, collided	594	460
MCNP, non-uniform, collided	590	452

3-dim volume plot



Displacement damage cutout isosurfaces. Each isosurface characterises the available volume for a DPA/FPY limit. Green = 20 DPA/FPY, yellow = 30 DPA/FPY and red = 40 DPA/FPY. The grid structure shows the physical boundary of the helium cooled high flux test module. (MCNP collided, McDeLi data, non-uniform beam profile.)

Volume to n-yield relation



Irradiation volume for displacement damage > 20 DPA/FPY as a function of the total neutron yield (uncollided calculations MCNP and INS 'experimental').

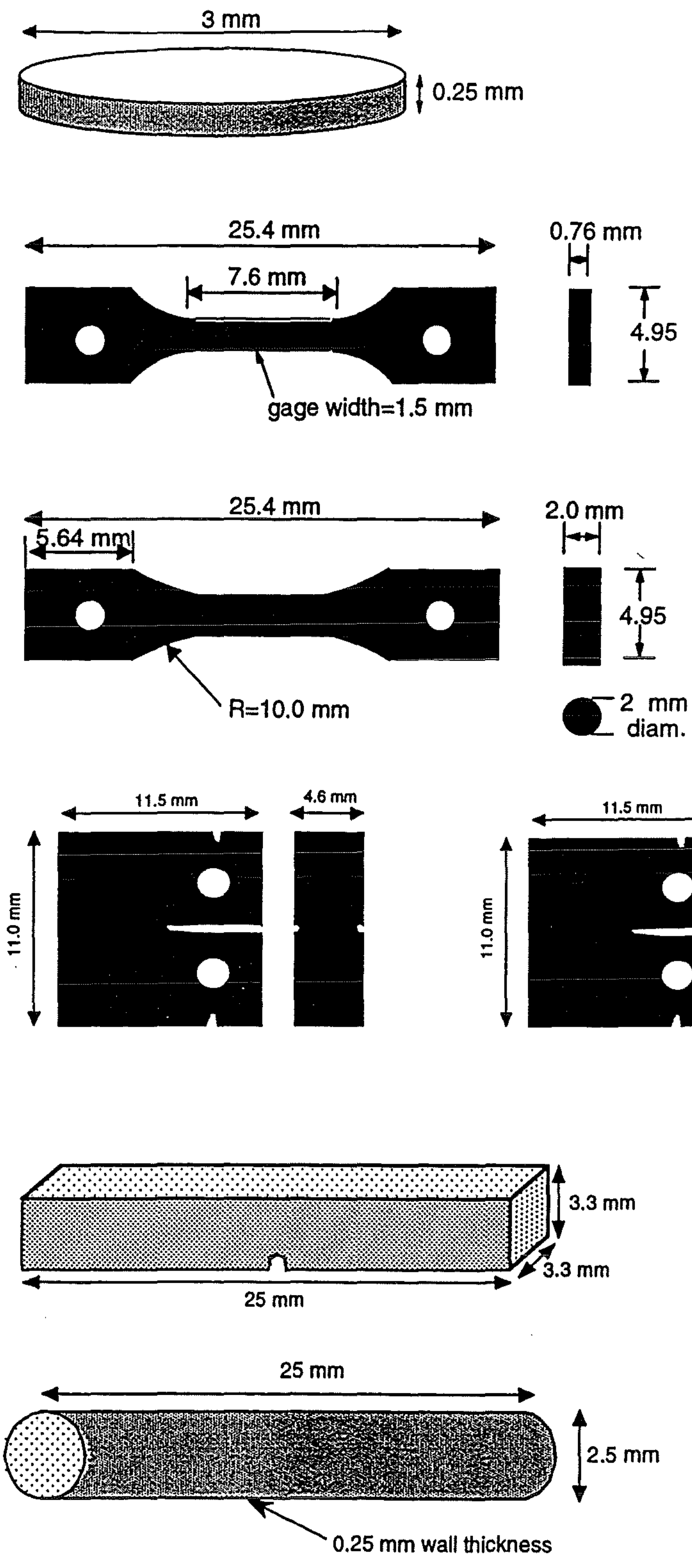
Other possibilities to enhance the volume

Li-Jet thickness	**25 mm for 40 MeV**
	22 mm for 36 MeV
	19 mm for 32 MeV

Range of 40 MeV in Li is 21 mm +/- ?

Volume gain per mm thickness reduction is:

approx. 7 cm^3 per mm at 50 DPA/FPY (25%/mm)
approx. 11 cm^3 per mm at 40 DPA/FPY (10%/mm)
approx. 13 cm^3 per mm at 20 DPA/FPY (2%/mm)



a) Microstructure (TEM disk)

b) Tensile properties

c) Push-pull fatigue

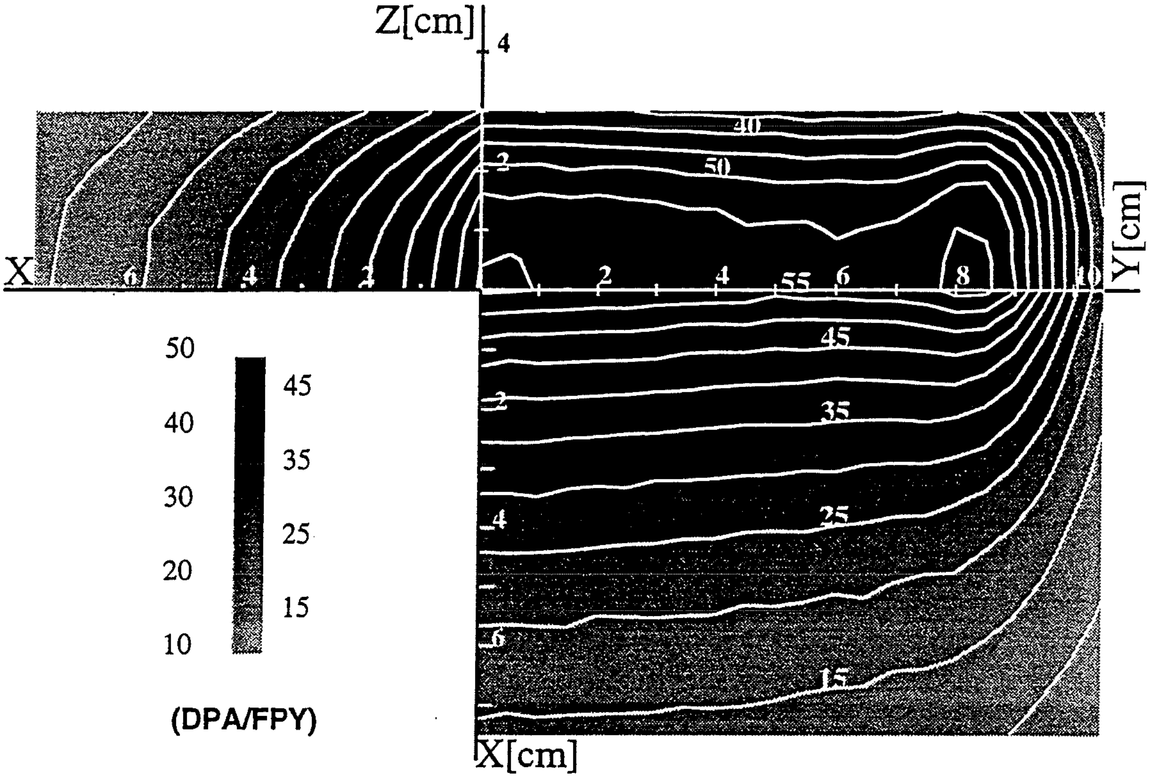
d) Fracture toughness,
Fatigue crack growth

e) Dynamic/static fracture toughness

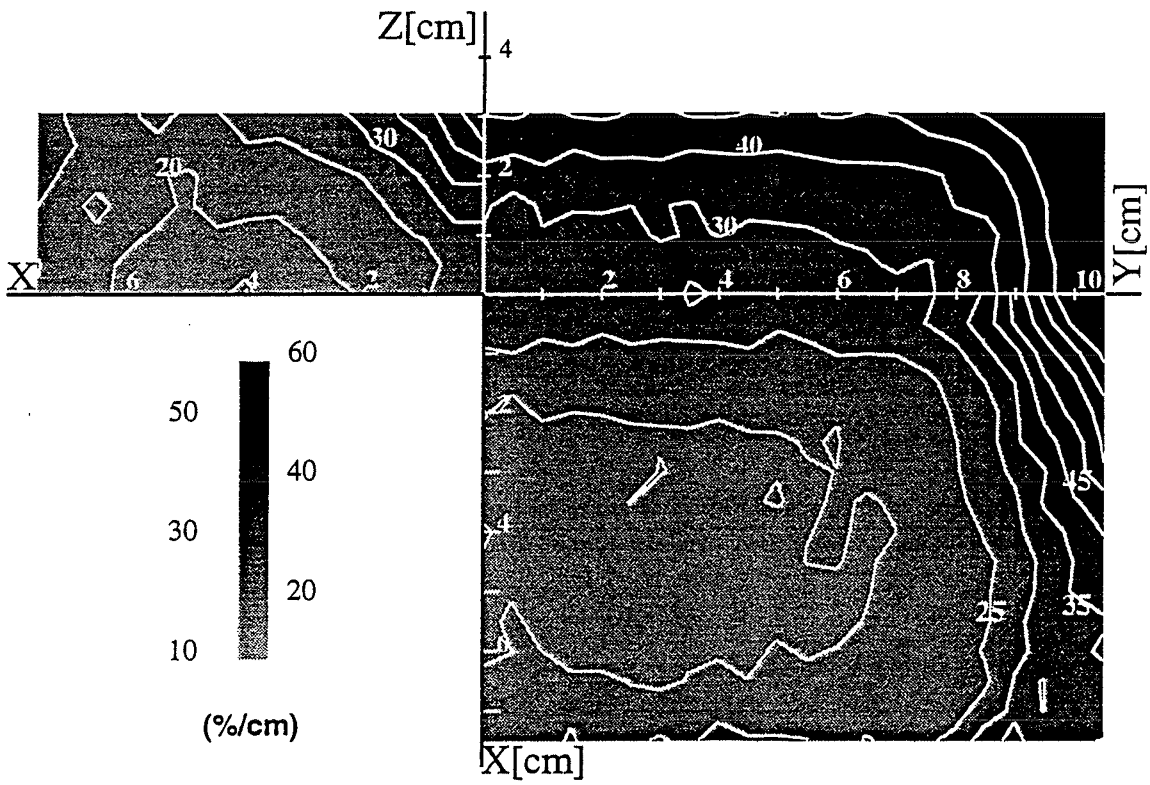
g) Pressurized creep tube

Fig. 2: Reference specimen geometries for the high flux region.

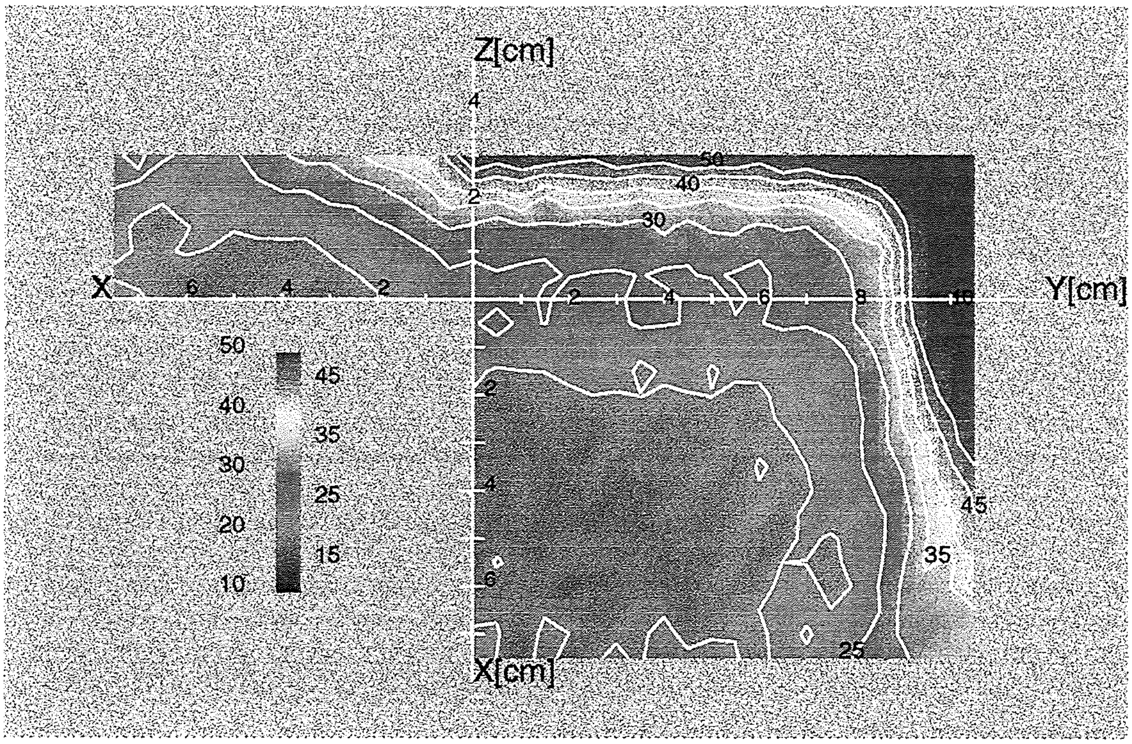
DPA Contours in units of DPA/FPY



Neutron flux gradient contours in units of %/cm



DPA gradient contours in units of %/cm



Displacement Damage from test Cell Users point of view

Quantity is given by **DPA** but **quality** is given by **PKA-spectrum**

Damage Model:

$$DPA = \int \phi \ \sigma_D \ dE \qquad \sigma_D = \int \frac{d\sigma(E, T)}{dT} \nu(T) \ dT$$

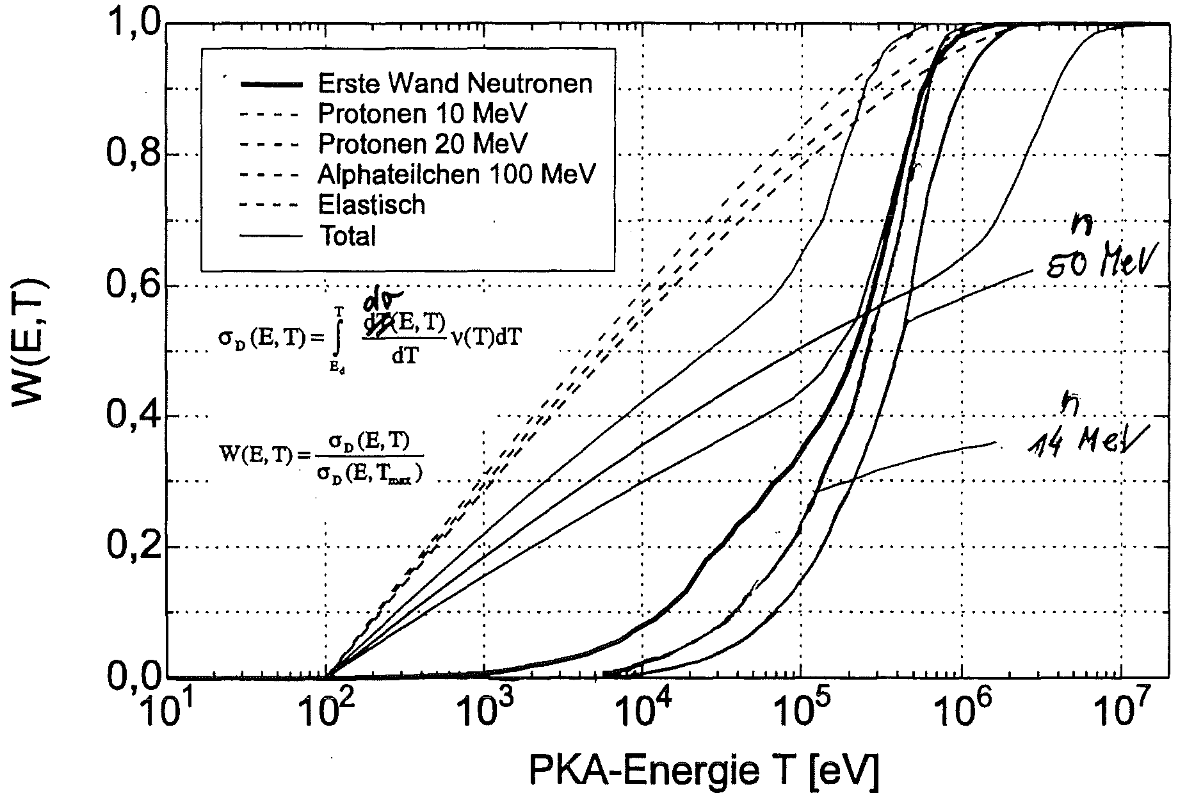
PKA-spectrum — NRT-Model

low T — high T

point defects — cascades

$$\nu(T) = 0.8 \frac{T_{dam}}{2E_d}$$

Forschungszentrum Karlsruhe
Technik und Umwelt



Conclusions

- **Depending on n-yield uncertainty of +/- 20 %, the total volume varies very much.**
  - **How much uncertainty will be tolerable for the HFTM design ?**
- **Li-jet thickness reduction can bring valuable high flux volume.**
  - **Should a memo be sent to the target people ?**
- **The orientation of the speciment can influence irradiation results.**
  - **Would it be better to change the USER requirement from % / cm to % over specimen with ?**
- **The quality assessment of the displacement damage shows a somewhat harder damage for IFMIF than for the 1st wall.**

Forschungszentrum Karlsruhe
Technik und Umwelt

IFMIF-CDE Test Facility Workshop, Karlsruhe July 7-9 1997

IFMIF Neutronics Calculations for Nak and He Cooled High Flux Test Modules

M. Sokcic-Kostic

DTI Dr. Trippe Ing. Ges. mbH

The results of MCNP neutronic calculations for test modules with NaK and He cooled high flux test cells are presented.

Some geometry figures of developed models are shown in Fig. 1-8. The numbers of cells used in Tables 1-10 are shown in Fig. 1,.2 and 3.

The presented calculations are made with medium flux module filled with ceramic breeder material specimens and without VTA2 - tritium module.

In Tables 1-5 the results for neutronic quantities (dpa, neutron flux density, helium and hydrogen production and nuclear heating) for three different calulation options: 1) complete geometry with Fe, 2) test modules only and 3) complete geometry with combination of 92% Fe and 8% Cr are compared for test modules with NaK cooled high flux test cell.

In Tables 6-10 the results for neutronic quantities (dpa, neutron flux density, helium and hydrogen production and nuclear heating) for three different calulation options: 1) complete geometry with Fe, 2) test modules only and 3) complete geometry with combination of 92% Fe and 8% Cr are compared for test modules with He cooled high flux test cell.

These calculations show that in the first approximation, to save computer time the models 2) (test cells only) give sufficient informations. Despite of that, for detailed calculations (design and usage) the complete geometry must be taken into account.

Abbreviations

OSHVNK15	complete geometry with Fe for NaK case	OSHVHE15	complete geometry with Fe for He case
ONKTZ21	test cells only for NaK case	OHETZ21	test cells only for He case
ONKCR15	compete geometry with Fe+Cr for NaK case	OHECR15	compete geometry with Fe+Cr for He case

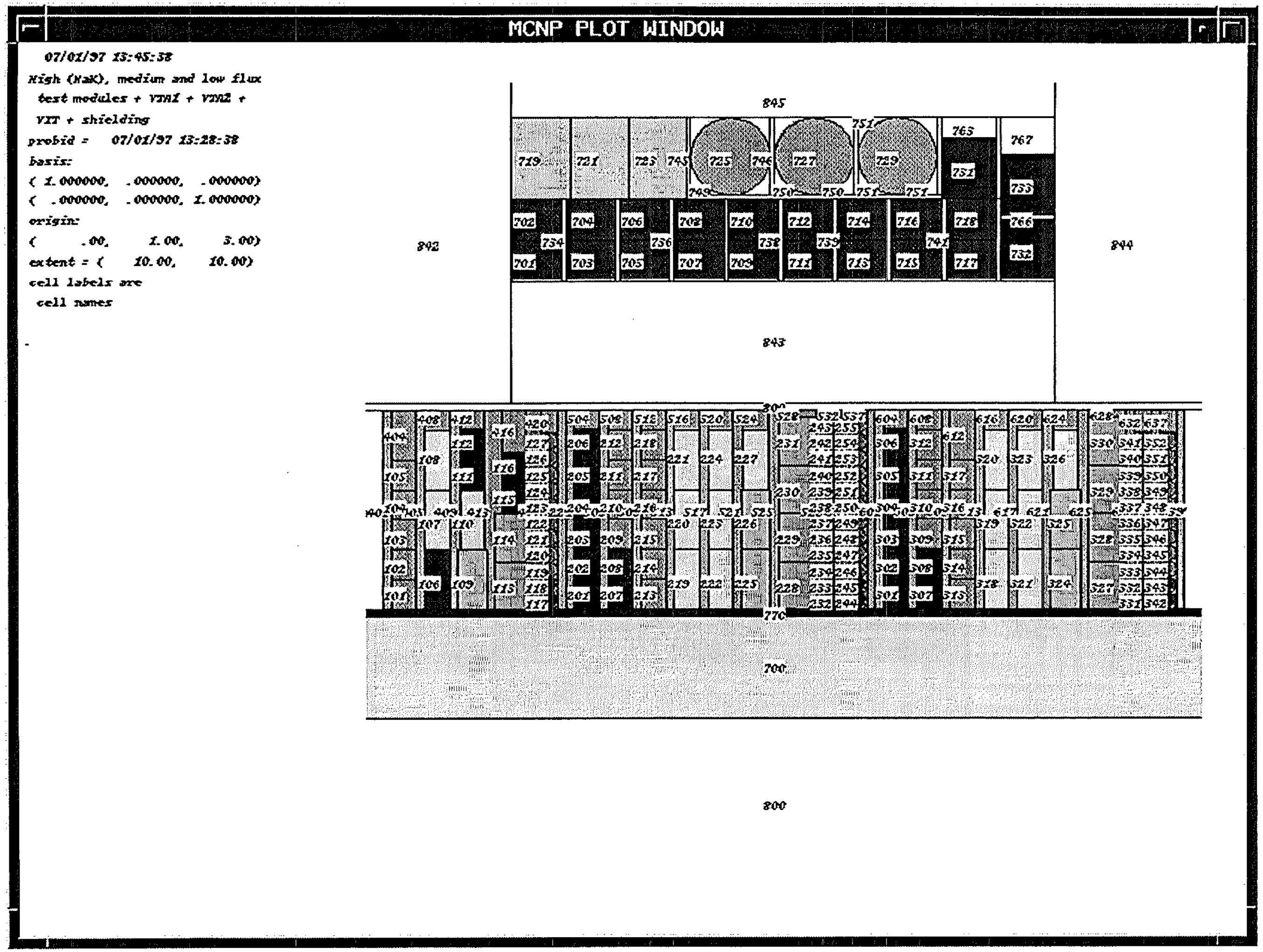


Fig. 1 High and medium test modules for test modules with NaK cooled high flux test cell

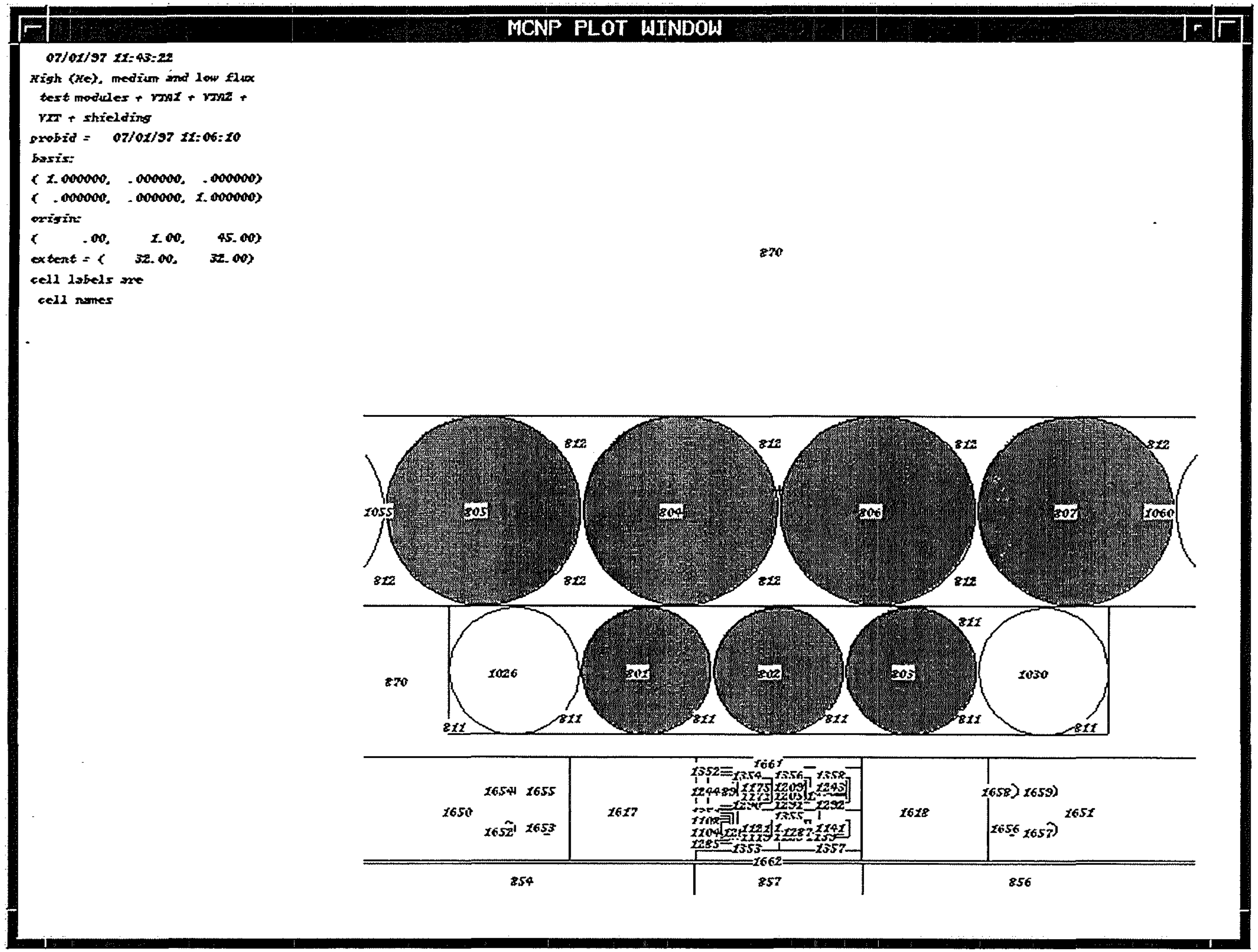


Fig. 2 Low and very low flux test module

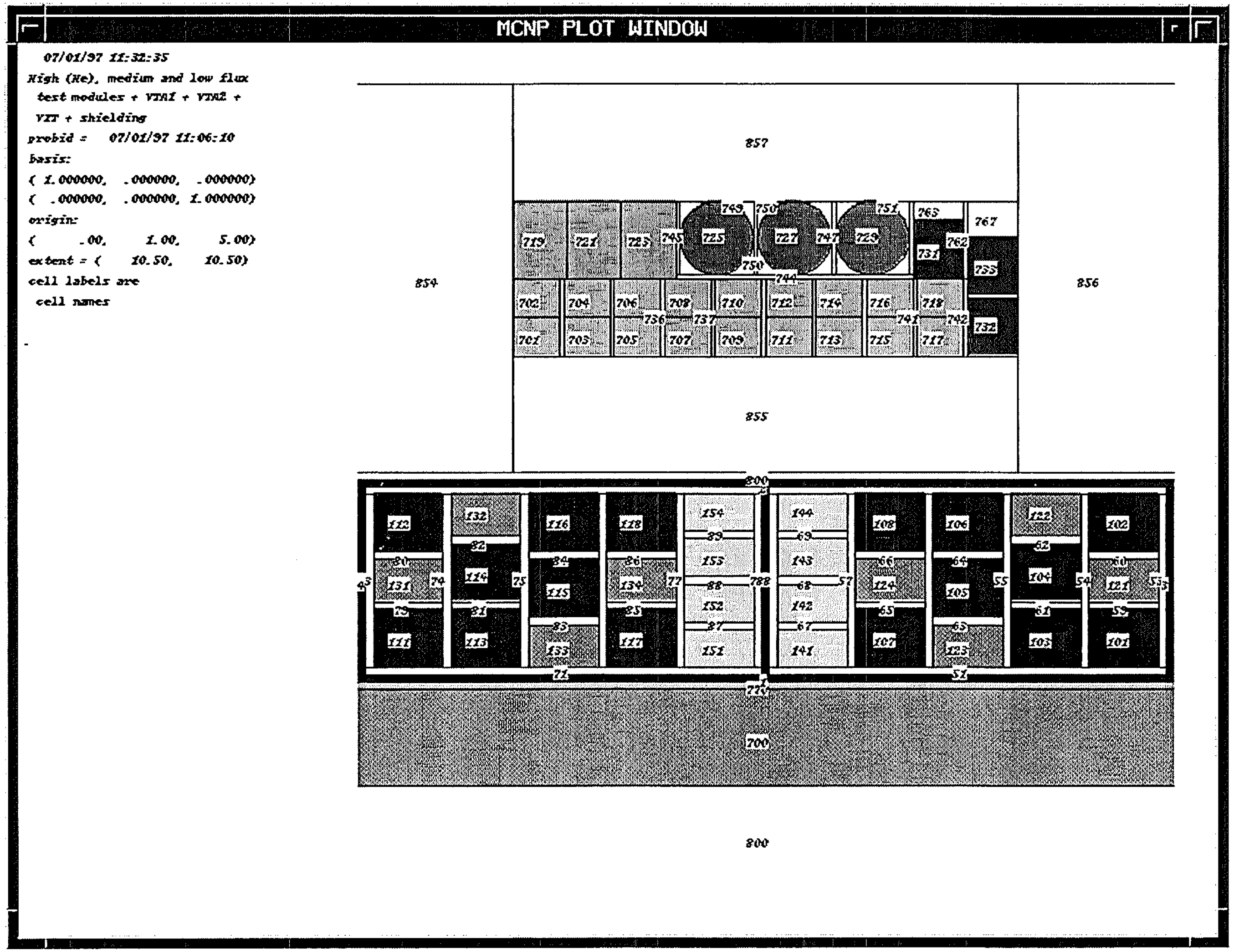


Fig. 3 High and medium flux test modules for test modules with He cooled high flux test cell

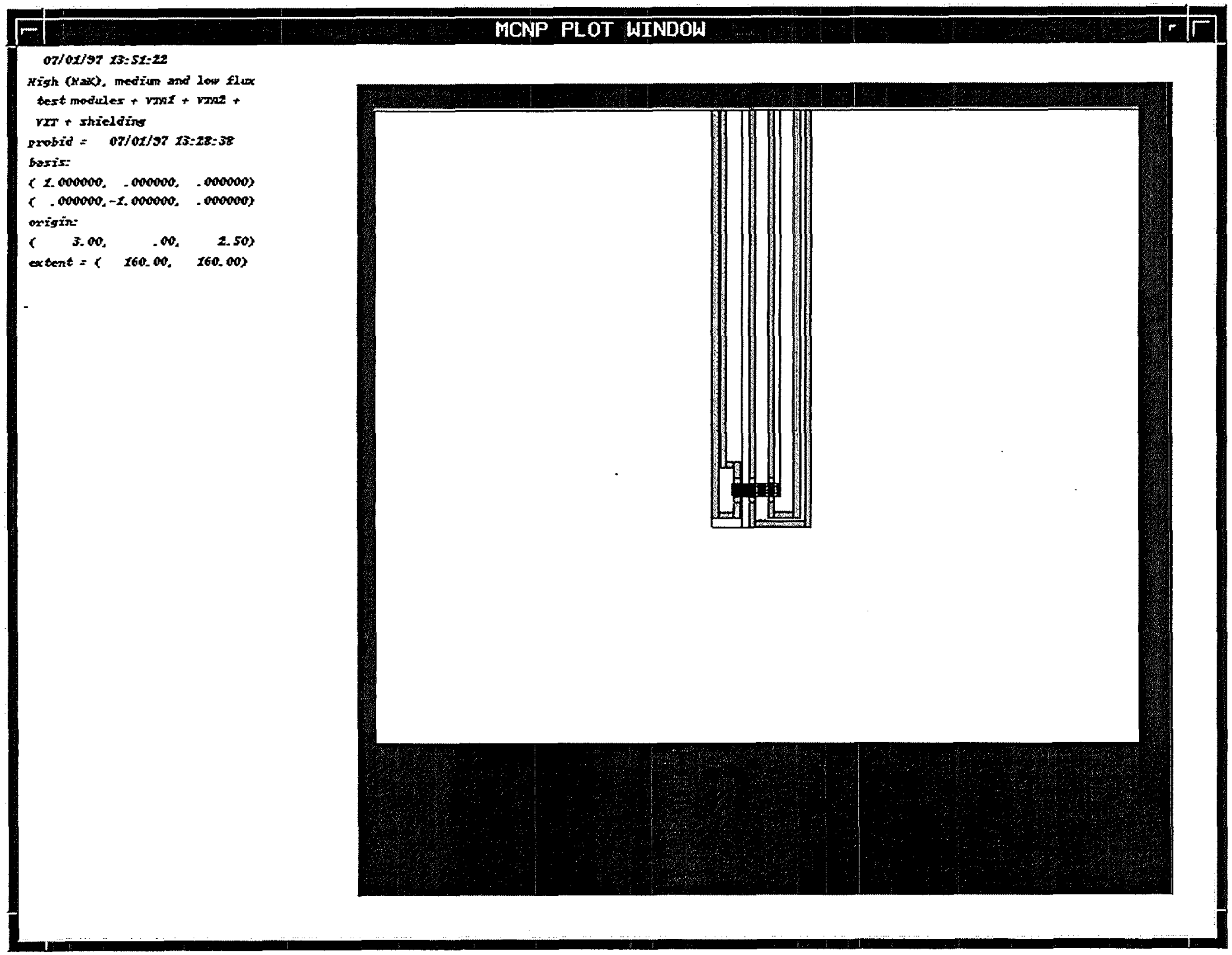


Fig. 4 VTA-1 for test modules with NaK cooled high flux test cell

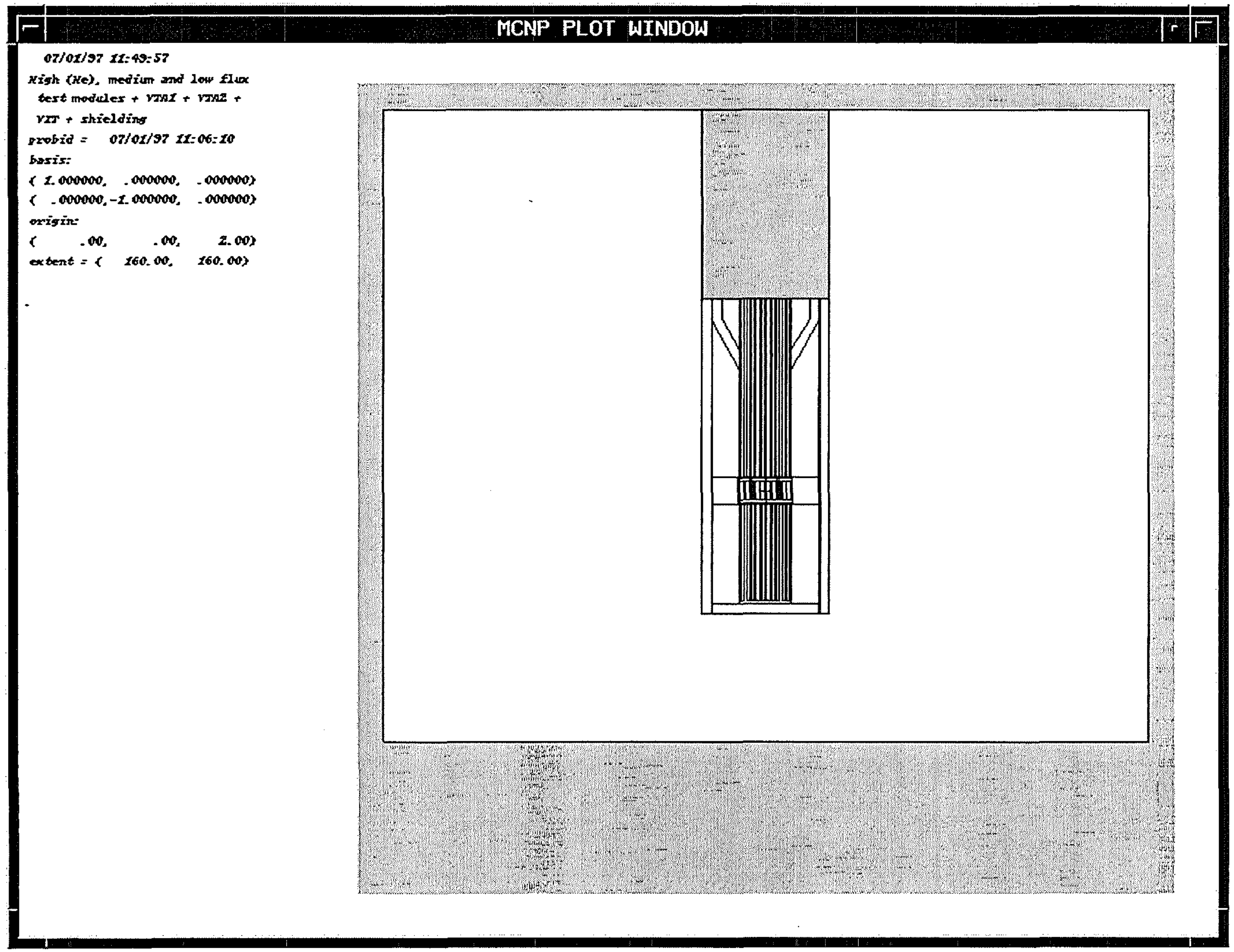


Fig. 5 VTA-1 for test modules with He cooled high flux test cell

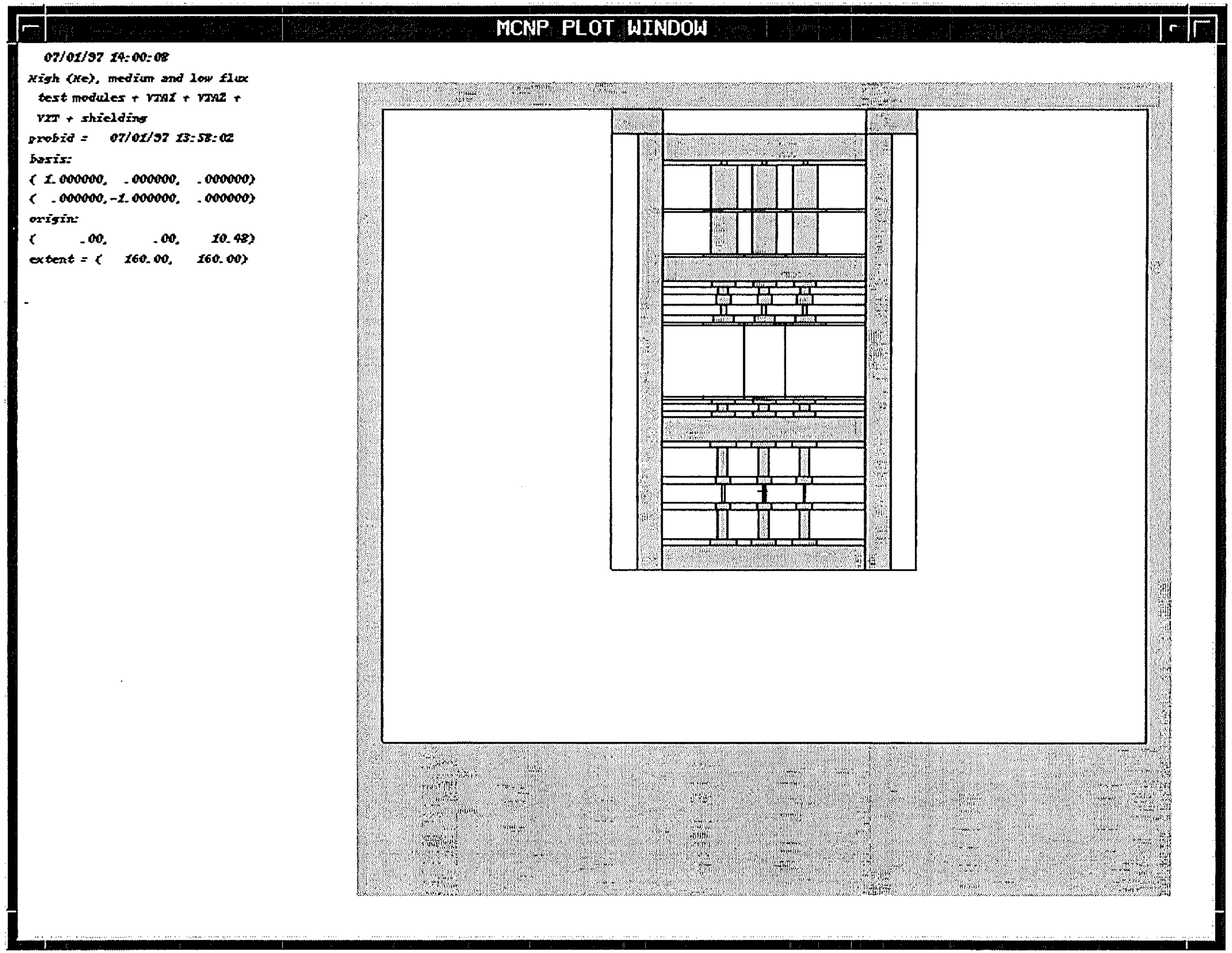


Fig. 6 VTA-2 (universal testing machine) with creep fatigue specimens

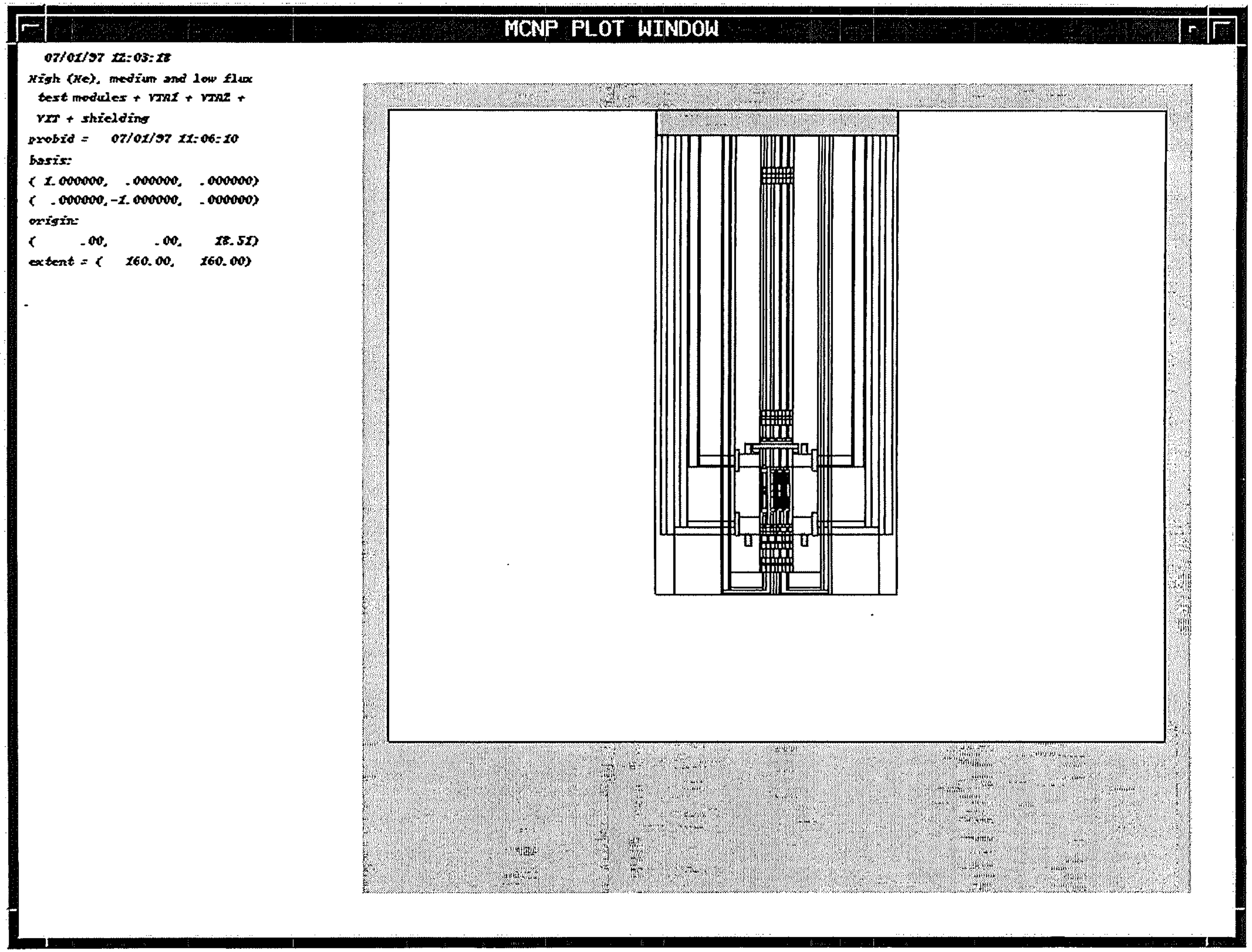


Fig. 7 VTA-2 (tritium module)

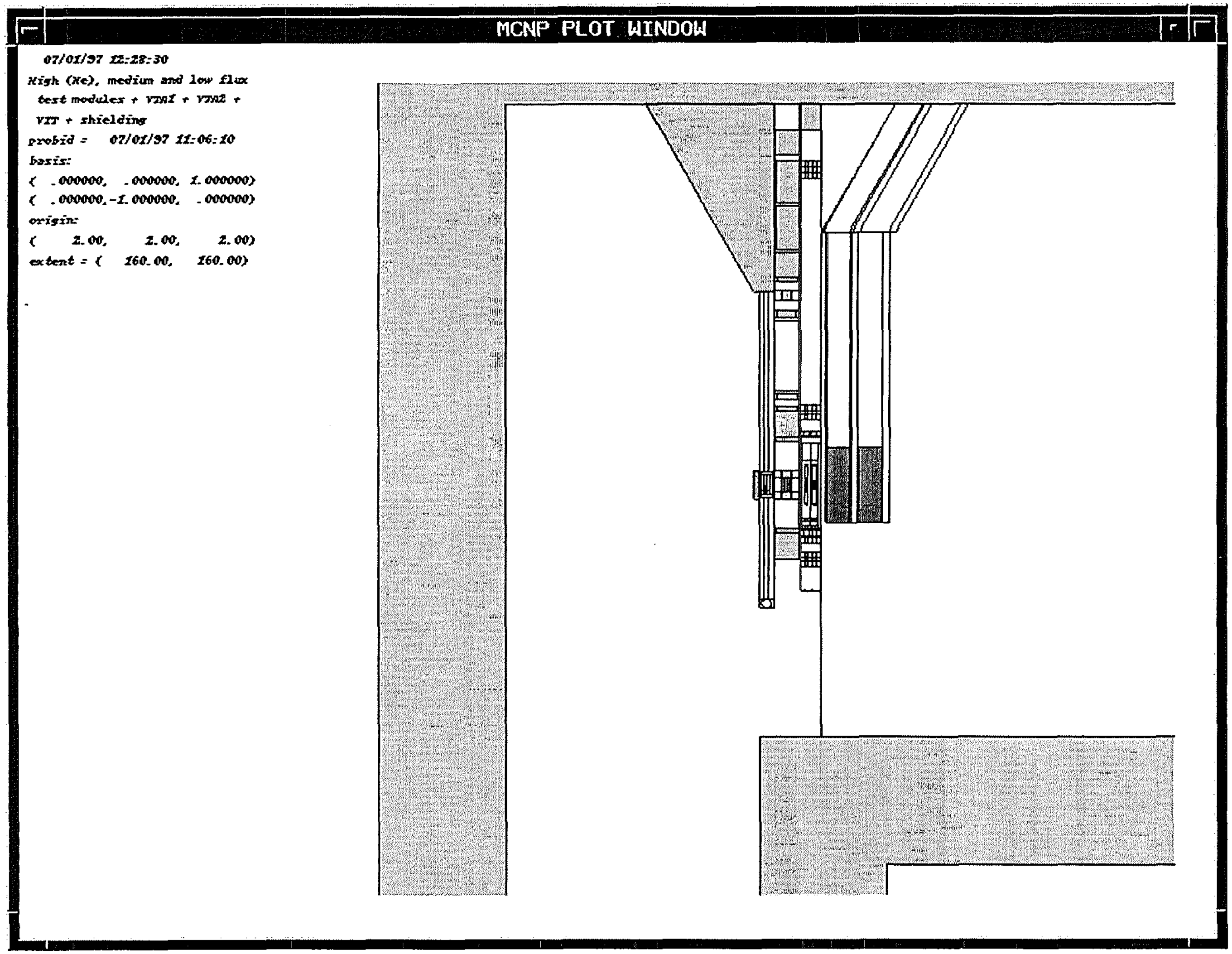


Fig. 8 Test modules with He cooled high flux test cell

Table 1. DPA for test modules with NaK cooled high flux test cell

#	ONKTZ21	ONKCR15	OSHVNK15
101	4,35E+01	4,42E+01	4,42E+01
102	3,37E+01	3,49E+01	3,50E+01
103	2,71E+01	2,88E+01	2,88E+01
104	2,22E+01	2,40E+01	2,40E+01
105	1,84E+01	2,03E+01	2,03E+01
228	4,88E+01	4,94E+01	4,94E+01
229	3,77E+01	3,84E+01	3,84E+01
230	2,96E+01	3,06E+01	3,07E+01
231	2,38E+01	2,51E+01	2,51E+01
342	4,42E+01	4,48E+01	4,47E+01
343	3,85E+01	3,95E+01	3,94E+01
344	3,40E+01	3,54E+01	3,54E+01
345	3,01E+01	3,18E+01	3,18E+01
346	2,69E+01	2,87E+01	2,87E+01
347	2,46E+01	2,65E+01	2,65E+01
348	2,23E+01	2,42E+01	2,43E+01
349	2,00E+01	2,19E+01	2,20E+01
350	1,81E+01	2,01E+01	2,01E+01
351	1,65E+01	1,84E+01	1,85E+01
352	1,43E+01	1,68E+01	1,68E+01
701	8,66E+00	1,27E+01	1,27E+01
702	7,43E+00	1,14E+01	1,14E+01
711	1,22E+01	1,45E+01	1,45E+01
712	1,10E+01	1,34E+01	1,34E+01
719	6,15E+00	9,78E+00	9,79E+00
720	6,25E+00	9,84E+00	9,83E+00
727	9,18E+00	1,15E+01	1,15E+01
728	9,14E+00	1,14E+01	1,14E+01
731	8,59E+00	1,08E+01	1,08E+01
732	8,67E+00	1,26E+01	1,26E+01
733	6,89E+00	1,07E+01	1,07E+01
801	6,61E-01	3,46E+00	3,46E+00
802	3,25E+00	4,90E+00	4,90E+00
803	6,64E-01	3,50E+00	3,50E+00
804	5,60E-01	1,42E+00	1,42E+00
805	4,03E-02	6,43E-01	6,44E-01
806	5,70E-01	1,43E+00	1,43E+00
807	4,00E-02	6,51E-01	6,51E-01

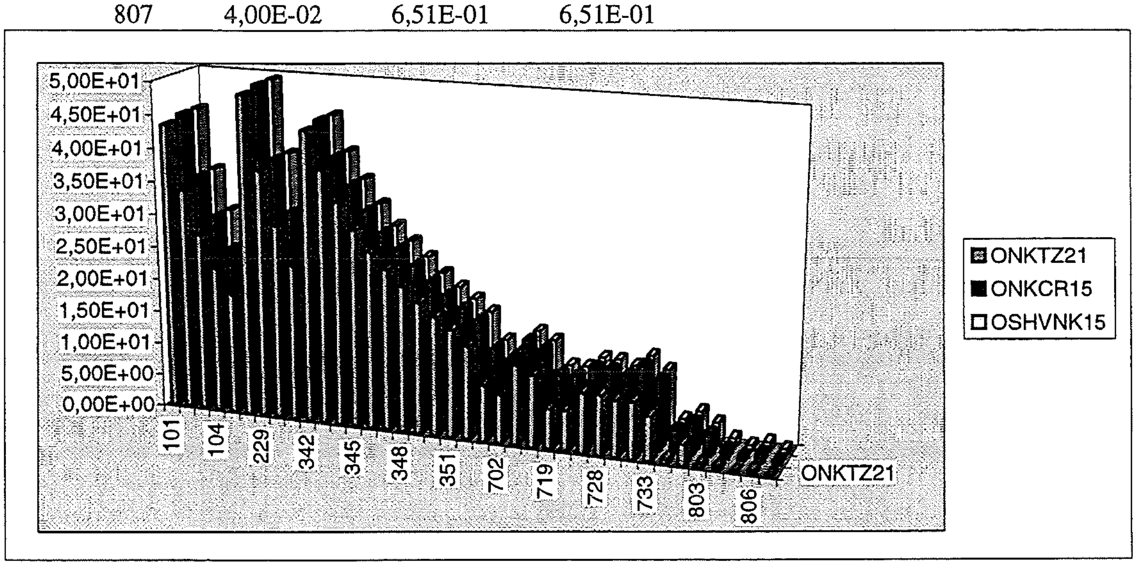


Table 2. Fluxes for test modules with NaK cooled high flux test cell

#	ONKTZ21	OSHVNK15	ONKCR15
101	7,96E+14	8,22E+14	8,24E+14
102	5,87E+14	6,26E+14	6,26E+14
103	4,60E+14	5,04E+14	5,04E+14
104	3,69E+14	4,16E+14	4,16E+14
105	3,00E+14	3,47E+14	3,47E+14
228	9,21E+14	9,45E+14	9,45E+14
229	6,72E+14	7,02E+14	7,02E+14
230	5,04E+14	5,42E+14	5,42E+14
231	3,88E+14	4,30E+14	4,30E+14
342	8,12E+14	8,34E+14	8,37E+14
343	6,82E+14	7,15E+14	7,17E+14
344	5,91E+14	6,32E+14	6,32E+14
345	5,15E+14	5,61E+14	5,62E+14
346	4,53E+14	5,01E+14	5,00E+14
347	4,10E+14	4,58E+14	4,59E+14
348	3,68E+14	4,17E+14	4,15E+14
349	3,27E+14	3,75E+14	3,74E+14
350	2,93E+14	3,40E+14	3,41E+14
351	2,63E+14	3,10E+14	3,10E+14
352	2,36E+14	2,82E+14	2,83E+14
701	1,29E+14	2,06E+14	2,07E+14
702	1,11E+14	1,87E+14	1,87E+14
711	1,84E+14	2,48E+14	2,48E+14
712	1,66E+14	2,32E+14	2,32E+14
719	9,22E+13	1,63E+14	1,63E+14
720	9,34E+13	1,63E+14	1,64E+14
727	1,40E+14	2,03E+14	2,03E+14
728	1,40E+14	2,02E+14	2,02E+14
731	1,26E+14	1,80E+14	1,80E+14
732	1,29E+14	2,04E+14	2,05E+14
733	1,02E+14	1,75E+14	1,75E+14
801	1,34E+13	7,46E+13	7,49E+13
802	5,37E+13	1,00E+14	1,01E+14
803	1,36E+13	7,54E+13	7,56E+13
804	1,10E+13	3,60E+13	3,64E+13
805	1,27E+12	1,89E+13	1,92E+13
806	1,15E+13	3,65E+13	3,68E+13
807	1,30E+12	1,91E+13	1,94E+13

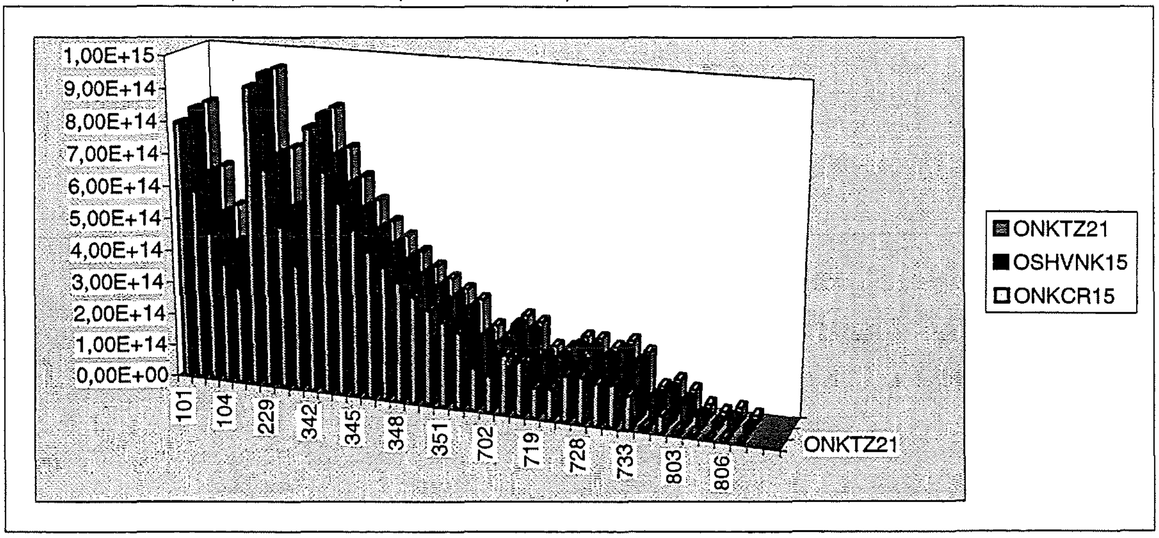


Table 3 He/DPA for test moduules with NaK cooled high flux test cell

#	OSHVNK15	ONKCR15	ONKTZ21
101	10,7	10,7	10,8
102	11,4	11,5	11,7
103	11,9	11,9	12,2
104	12,2	12,2	12,5
105	12,5	12,5	12,9
228	10,3	10,3	10,4
229	11,2	11,3	11,4
230	11,9	11,9	12,3
231	12,5	12,5	13
342	10,6	10,6	10,7
343	11,2	11,1	11,4
344	11,4	11,4	11,7
345	11,8	11,8	12,1
346	12	12	12,3
347	12,2	12,2	12,4
348	12,3	12,4	12,6
349	12,5	12,6	12,9
350	12,7	12,6	13,1
351	12,8	12,8	13,3
352	12,9	12,9	13,8
701	13,6	13,6	14,4
702	13,7	13,6	14,4
711	13,1	13,1	14,2
712	13	13	14,3
719	13,6	13,6	14,4
720	13,6	13,6	14,4
727	12,9	12,9	14,3
728	12,9	12,9	14,2
731	13,5	13,5	14,7
732	13,7	13,7	14,5
733	13,7	13,7	14,5
801	11,4	13,4	11,4
802	12,2	12,2	14,1
803	11,4	11,4	11,3
804	10,7	10,7	13
805	8,21	8,22	6,31
806	10,6	10,5	12,8
807	8,19	8,18	5,97

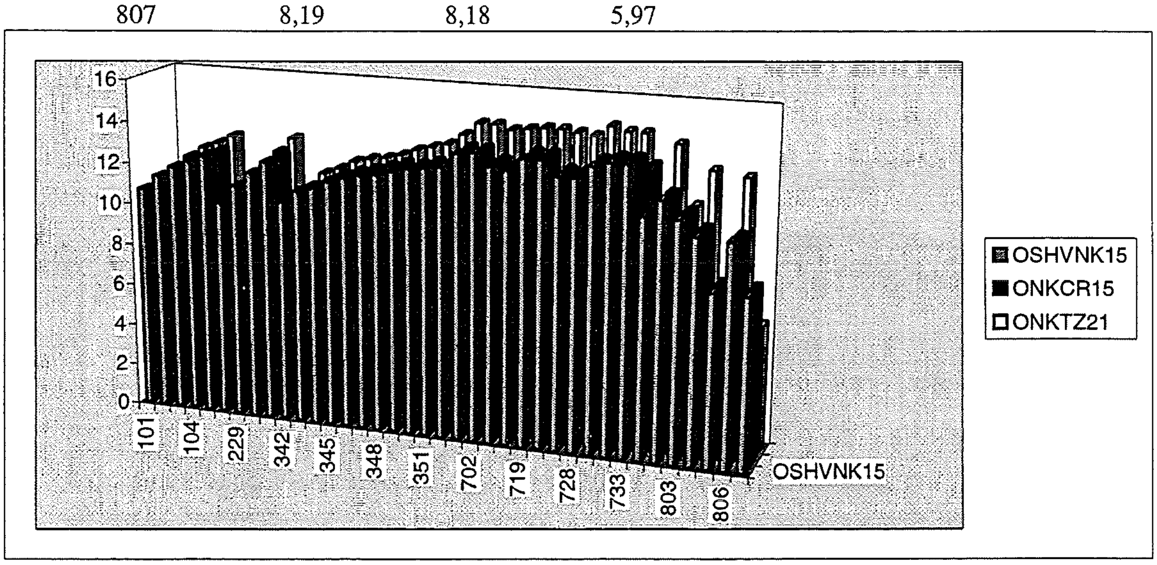


Table 4. H/DPA for test modules with NaK cooled high flux test cell

#	OSHVNK15	ONKCR15	ONKTZ21
101	41,6	41,4	42,1
102	44,6	44,6	45,5
103	46,4	46,4	47,6
104	47,5	47,5	49
105	48,6	48,7	50,5
228	40,2	40,1	40,6
229	43,7	43,7	44,4
230	46,3	46,4	47,7
231	38,7	48,8	50,7
342	41,2	41,1	41,6
343	43,4	43,2	44,1
344	44,4	44,4	45,4
345	45,8	45,8	47
346	46,8	46,8	48
347	47,5	47,5	48,5
348	48,1	48,2	49,3
349	49	48,9	50,4
350	49,3	49	50,9
351	39,9	49,8	51,6
352	50,1	49,9	54
701	53	53	56
702	53,3	53,2	56,1
711	51,1	51,2	55,5
712	50,8	50,8	55,7
719	52,9	53	56,1
720	53,1	53,2	56,4
727	50,4	50,4	55,7
728	50,4	50,4	55,7
731	52,8	52,9	57,4
732	53,4	53,3	56,5
733	53,7	53,7	56,8
801	44,3	44,4	44,5
802	47,5	47,4	55,1
803	44,5	44,5	44,1
804	41,7	41,6	50,6
805	31,8	31,8	24,2
806	41,3	41,2	49,8
807	31,7	31,7	23

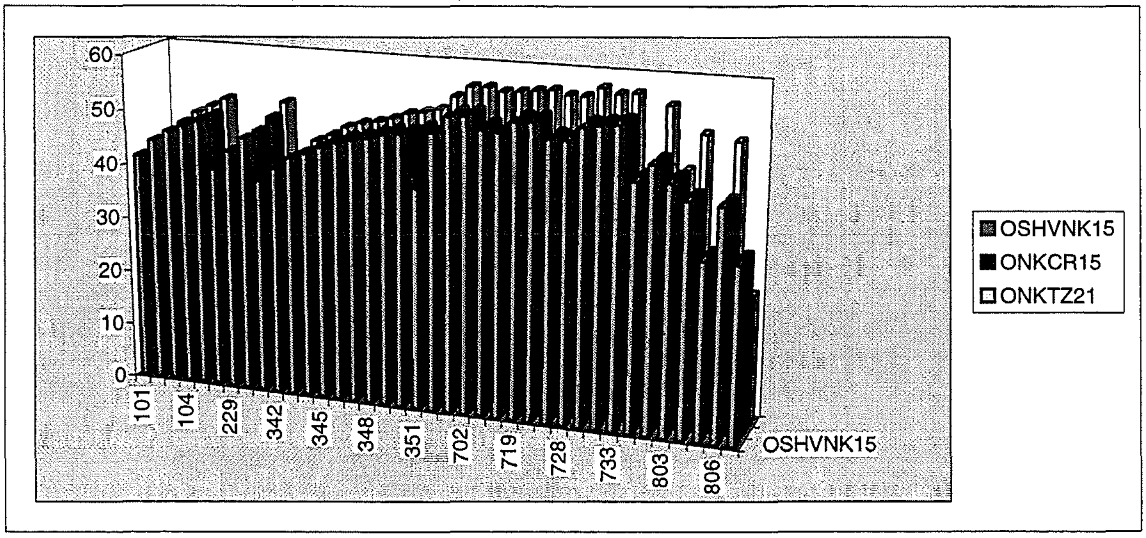


Table 5. Nuclear heating for test modules with NaK cooled high flux test cell

#	ONKTZ21	ONKCR15	OSHVNK15
101	39,3	39,9	39,9
102	33,1	34,1	34,2
103	27,9	29,5	29,4
104	23,5	25	25,1
105	19,9	21,8	21,6
228	43,2	43,8	43,8
229	36,6	37,4	37,4
230	30,7	31,7	31,7
231	25,8	27,1	27,1
342	38,4	39	39,1
343	35,3	36,2	36,2
344	32,5	33,7	33,7
345	30	31,4	31,4
346	27,5	29,1	29,1
347	25,4	27,3	27,3
348	23,5	25,4	25,5
349	21,4	23,3	23,5
350	19,7	21,6	21,7
351	18,1	20	20
352	16,1	18,2	18,3
701	9,94	14,5	14,5
702	8,7	13,4	13,4
711	14	17,2	17,2
712	13	16,2	16,3
719	7,3	11,7	11,8
720	7,44	11,8	11,8
727	11,3	14,5	14,5
728	11,2	14,4	14,5
731	10,2	12,9	12,9
732	10	14,5	14,5
733	8,09	12,6	12,6
801	0,908	4,65	4,66
802	4,57	0,845	6,85
803	0,905	4,72	4,72
804	0,839	2,02	2,02
805	0,0522	0,777	0,775
806	0,86	2,03	2,04
807	0,05	0,784	0,785

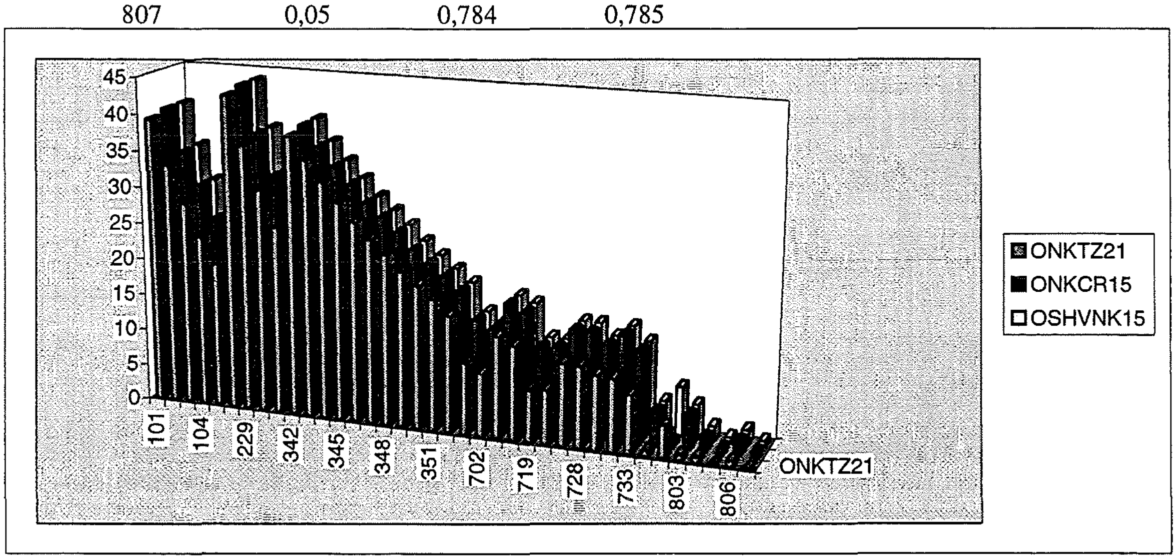


Table 6. DPA for test modules with He cooled high flux test cell

#	OHETZ21	OHECR15	OSHVHE15
141	4,71E+01	4,72E+01	4,73E+01
142	3,65E+01	3,67E+01	3,68E+01
143	2,89E+01	2,92E+01	2,93E+01
144	2,29E+01	2,32E+01	2,33E+01
107	4,35E+01	4,37E+01	4,38E+01
124	3,12E+01	3,15E+01	3,15E+01
108	2;29E+01	2,33E+01	2,33E+01
123	4,42E+01	4,44E+01	4,44E+01
105	3,17E+01	3,15E+01	3,20E+01
106	2,20E+01	2,23E+01	2,24E+01
103	4,25E+01	4,26E+01	4,28E+01
104	2,81E+01	2,83E+01	2,84E+01
122	1,99E+01	2,02E+01	2,03E+01
101	3,40E+01	3,42E+01	3,42E+01
121	2,29E+01	2,31E+01	2,32E+01
102	1,60E+01	1,64E+01	1,64E+01
701	7,45E+00	1,00E+01	1,27E+01
702	6,40E+00	9,03E+00	1,14E+01
711	1,10E+01	1,21E+01	1,21E+01
712	9,87E+00	1,11E+01	1,34E+01
719	5,36E+00	7,72E+00	9,79E+00
720	5,37E+00	7,76E+00	9,83E+00
727	8,14E+00	9,29E+00	1,15E+01
728	8,18E+00	9,37E+00	1,14E+01
731	7,37E+00	8,54E+00	1,08E+01
732	7,38E+00	9,92E+00	1,26E+01
733	5,88E+00	8,39E+00	1,07E+01
801	5,88E-01	2,65E+00	3,46E+00
802	2,81E+00	3,80E+00	4,90E+00
803	5,94E-01	2,65E+00	3,50E+00
804	4,81E-01	1,08E+00	1,42E+00
805	3,65E-02	5,20E-01	6,44E-01
806	4,79E-01	1,09E+00	1,43E+00
807	3,69E-02	5,22E-01	6,51E-01

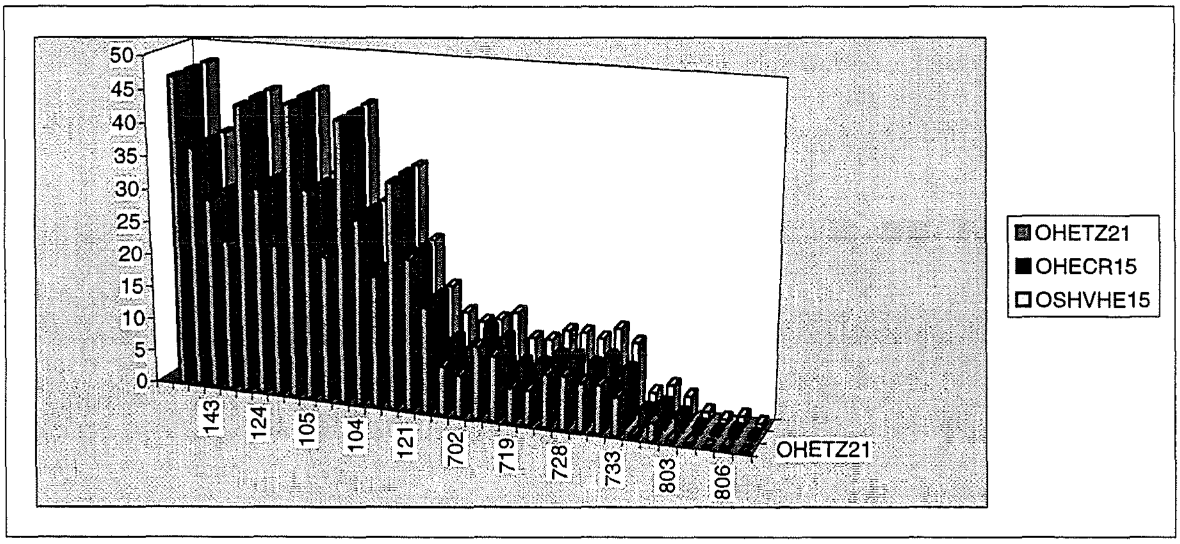


Table 7. Fluxes for test modules with He coole led high flux test cell

#	OHETZ21	OSHVHE15	OHECR15
141	9,50E+14	9,64E+14	9,65E+14
142	7,19E+14	7,35E+14	7,35E+14
143	5,58E+14	5,76E+14	5,77E+14
144	4,31E+14	4,52E+14	4,51E+14
107	8,83E+14	8,97E+14	8,97E+14
124	6,18E+14	6,35E+14	6,36E+14
108	4,41E+14	4,61E+14	4,61E+14
123	8,96E+14	9,09E+14	9,09E+14
105	6,27E+14	6,42E+14	6,43E+14
106	4,21E+14	4,40E+14	4,40E+14
103	8,37E+14	8,50E+14	8,49E+14
104	5,37E+14	5,53E+14	5,52E+14
122	3,67E+14	3,85E+14	3,85E+14
101	6,54E+14	6,67E+14	6,68E+14
121	4,30E+14	4,45E+14	4,45E+14
102	2,94E+14	3,12E+14	3,13E+14
701	1,21E+14	1,78E+14	1,78E+14
702	1,03E+14	1,60E+14	1,60E+14
711	1,79E+14	2,22E+14	2,21E+14
712	1,59E+14	2,05E+14	2,05E+14
719	8,57E+13	1,38E+14	1,38E+14
720	8,57E+13	1,38E+14	1,39E+14
727	1,31E+14	1,75E+14	1,74E+14
728	1,32E+14	1,76E+14	1,76E+14
731	1,15E+14	1,53E+14	1,53E+14
732	1,19E+14	1,74E+14	1,76E+14
733	9,36E+13	1,47E+14	1,48E+14
801	1,24E+13	6,04E+13	6,08E+13
802	4,76E+13	8,15E+13	8,18E+13
803	1,26E+13	6,08E+13	6,11E+13
804	9,65E+12	2,90E+13	2,92E+13
805	1,17E+12	1,60E+13	1,63E+13
806	9,65E+12	2,91E+13	2,94E+13
807	1,18E+12	1,61E+13	1,64E+13

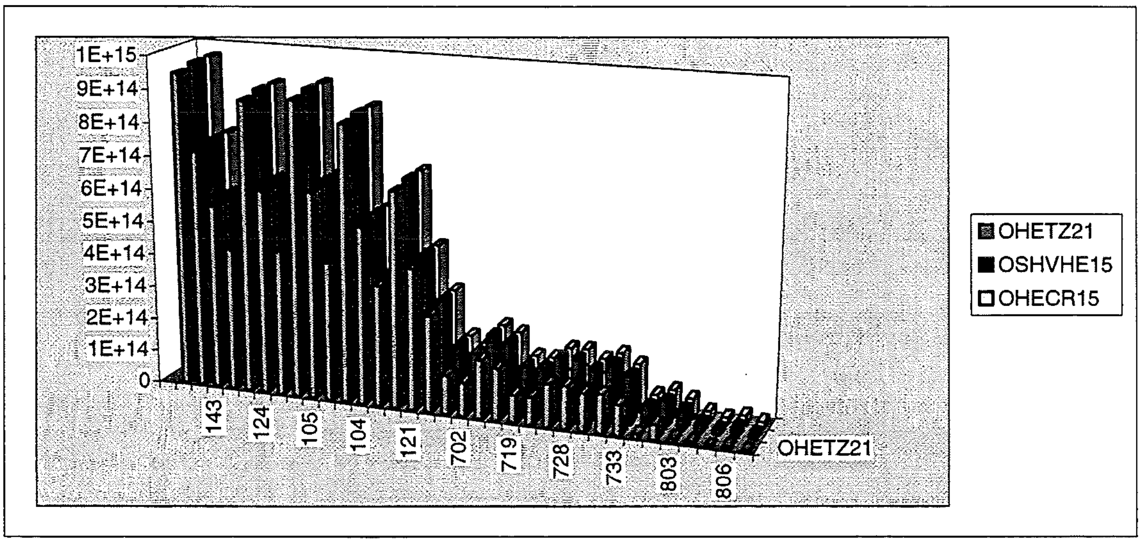


Table 8. He/DPA for test modules with He cooled high flux test cell

#	OHECR15	OSHVHE15	OHETZ21
141	9,7	10	10
142	10,4	10,6	10,7
143	10,8	11,1	11,2
144	11,2	11,5	11,7
107	9,75	10	10
124	10,4	10,7	10,8
108	10,9	11,2	11,4
123	9,64	9,91	9,95
105	10,5	10,7	10,8
106	11,1	11,3	11,5
103	10,1	10,4	10,4
104	11	11,2	11,3
122	11,6	11,8	12,1
101	10,4	10,7	10,8
121	11,1	11,3	11,4
102	11,4	11,7	12
701	12,8	13	13,8
702	12,8	13,1	13,9
711	12,4	12,7	13,6
712	12,4	12,7	13,8
719	12,8	13,2	14
720	12,8	13,1	13,9
727	12,3	12,6	13,9
728	12,3	12,6	13,9
731	12,8	13,1	14,2
732	12,8	13,1	13,7
733	12,9	13,2	13,9
801	10,8	11,1	11,2
802	11,7	12	14
803	10,8	11	11
804	10,1	10,4	12,8
805	7,87	8,11	6,55
806	10,1	10,4	12,7
807	7,88	8,17	6,33

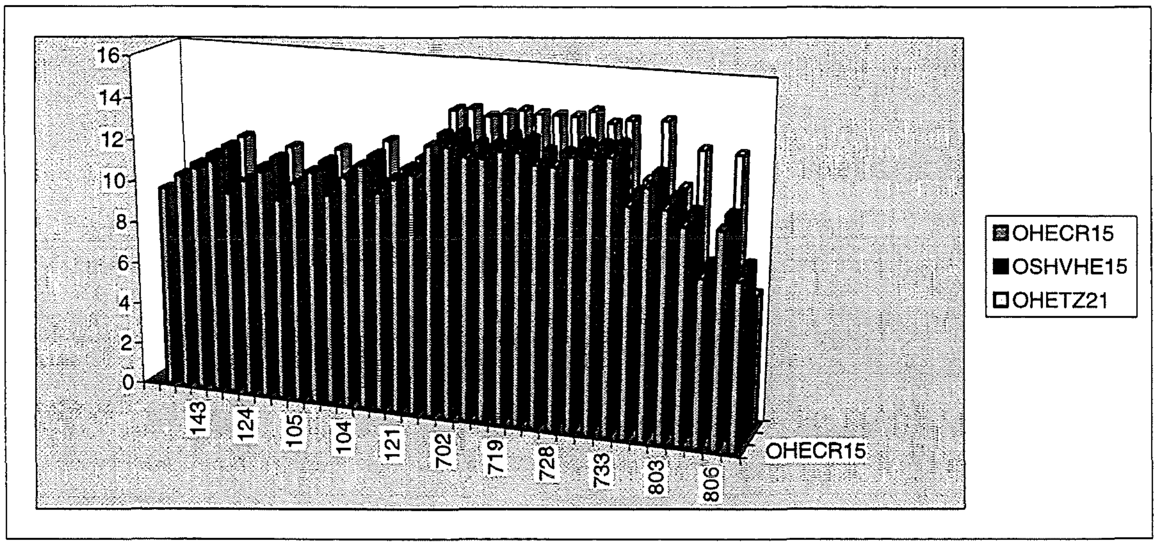


Table 9. H/DPA for test modules with He cooled high flux test cell

#	OSHVHE15	OHECR15	OHETZ21
141	38,6	39,2	38,8
142	41,2	41,8	41,5
143	43,1	43,8	43,7
144	44,7	45,4	45,4
107	38,9	39,4	39,1
124	41,7	42,2	42,1
108	43,8	44,3	44,5
123	38,5	38,9	38,7
105	41,5	42,6	41,9
106	44,1	44,8	44,8
103	40,4	41	40,6
104	43,7	44,3	44,1
122	46,1	46,8	47
101	41,6	42	41,8
121	44,1	44,8	44,6
102	45,7	46,2	46,6
701	50,9	51,7	53,6
702	51,2	52	54
711	49,4	50,2	53,2
712	49,4	50,1	53,8
719	51,3	51,9	54,3
720	51,3	52,1	54,4
727	49,2	50	54,2
728	49,2	50	54,2
731	51,1	52	55,4
732	51,2	51,8	53,7
733	51,7	52,5	54,3
801	43,3	43,9	43,6
802	46,8	47,6	54,9
803	43,1	43,7	43
804	40,5	41	50
805	31,4	31,7	25,4
806	40,5	41	49,5
807	31,7	31,8	24,3

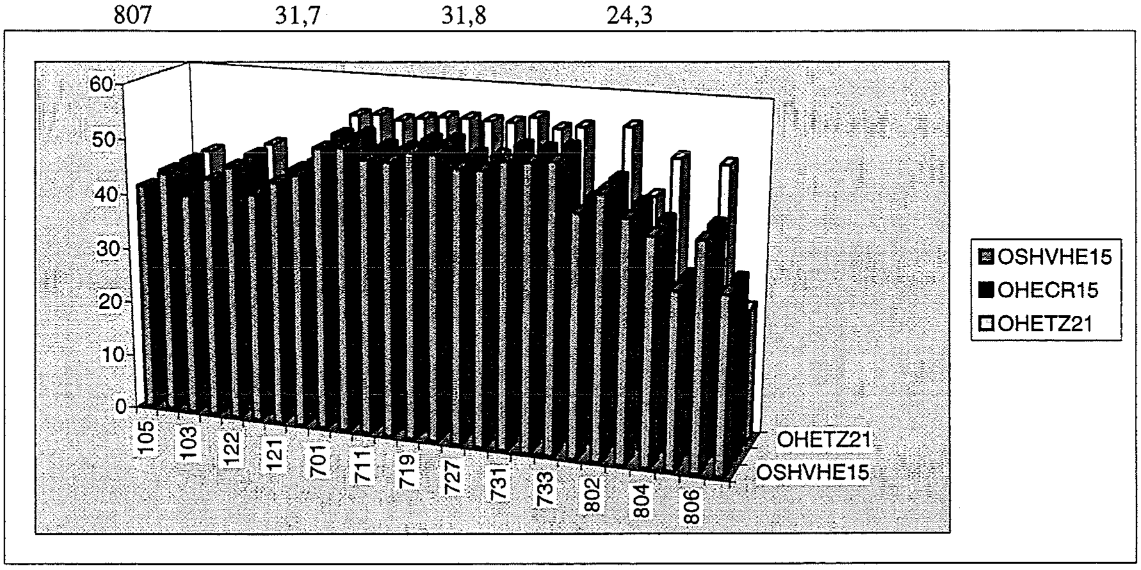
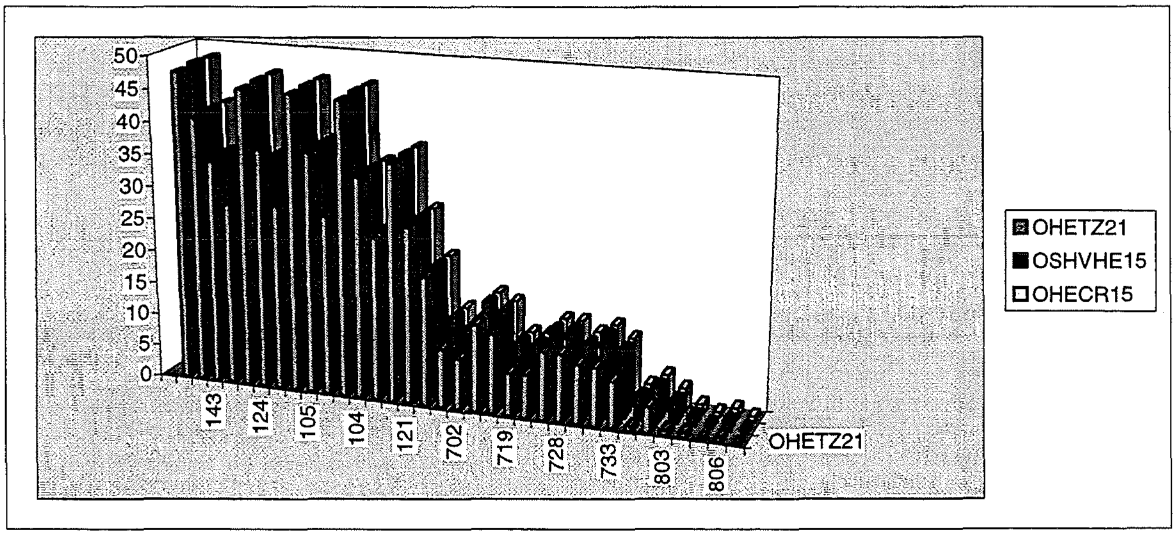


Table 10. Nuclear heating for test modules with He cooled high flux test cell

#	OHETZ21	OSHVHE15	OHECR15
141	47,7	48,2	48,1
142	40,8	41,3	41,3
143	34,3	34,9	34,9
144	27,9	28,5	28,6
107	45,8	46,2	46,3
124	36,7	34,9	37
108	28,4	28,9	29
123	45,6	45,9	45,9
105	37	37,4	37,4
106	27,5	27,9	28
103	45,2	45,6	45,6
104	33,9	34,2	34,2
122	25	25,6	25,6
101	36,4	36,7	36,7
121	27,1	27,5	27,5
102	19,8	20,3	20,4
701	9,1	12,5	12,5
702	7,91	11,3	11,4
711	13,4	15,4	15,4
712	12,2	14,4	14,3
719	6,65	9,8	9,77
720	6,69	9,79	9,88
727	10,4	12,4	12,4
728	10,4	12,5	12,4
731	9,11	10,8	10,9
732	8,96	12,3	12,3
733	7,2	10,5	10,5
801	0,809	3,65	3,65
802	4,01	5,41	5,41
803	0,81	3,63	3,65
804	0,719	1,53	1,53
805	0,0472	0,634	0,637
806	0,707	1,53	1,53
807	0,047	0,642	0,641



Comparison of irradiation conditions in the IFMIF high flux test region and (d,t) - fusion reactors

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Institut für Neutronenphysik und Reaktortechnik

Overview

I. IFMIF vs. fusion reactor
II. 3d Monte Carlo calculations
III. Results for IFMIF & fusion reactors
IV. Conclusions & future work

IFMIF mission

To provide an accelerator-based, deuterium-lithium (d-Li) neutron source to produce high energy neutrons at sufficient energy, intensity and irradiation volume to test samples of candidate materials up to a full lifetime of anticipated use in fusion energy reactors.

IFMIF ⇔ fusion reactor

- Intensity ⇒ neutron flux density, nuclear responses
- Energy ⇒ neutron spectrum
- Full lifetime test ⇒ fluence, neutron wall loading

Comparison of irradiation conditions

- Nuclear engineering responses
  ⇒ dpa, gas production, nuclear heating
- Neutron flux & spectrum
- Neutron wall loading

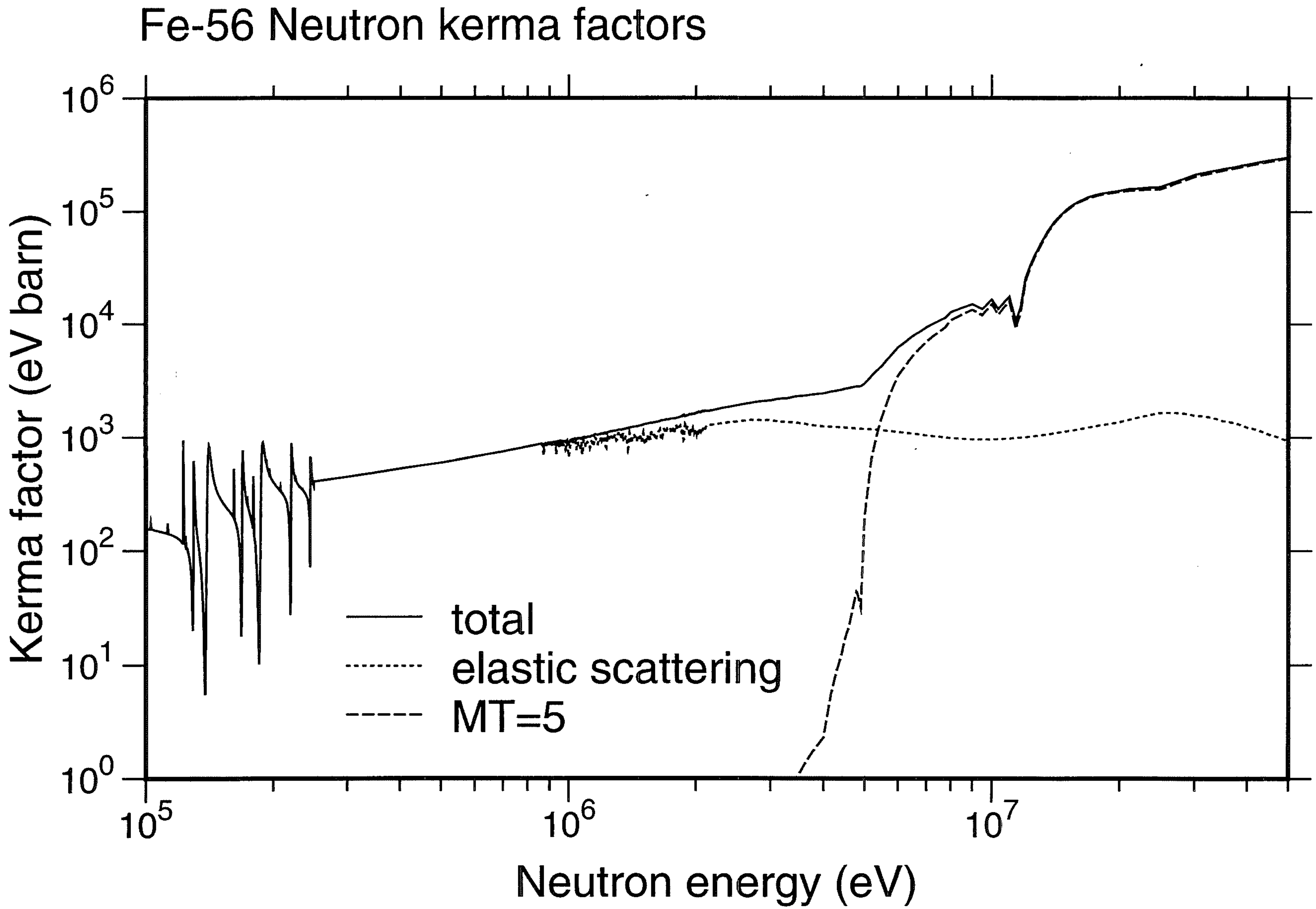
Three-dimensional neutronic calculations

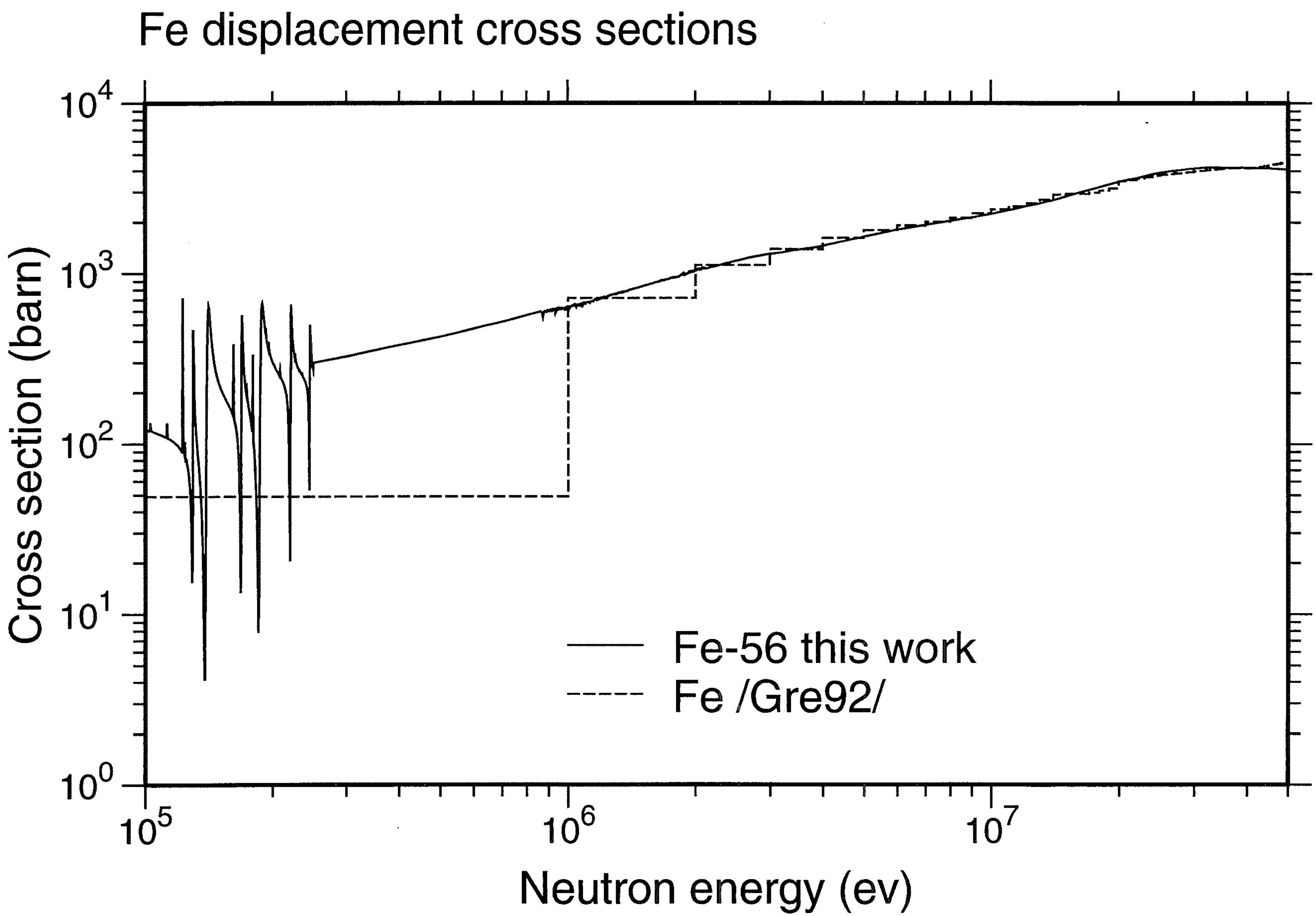
- **MCNP4A Monte Carlo code**

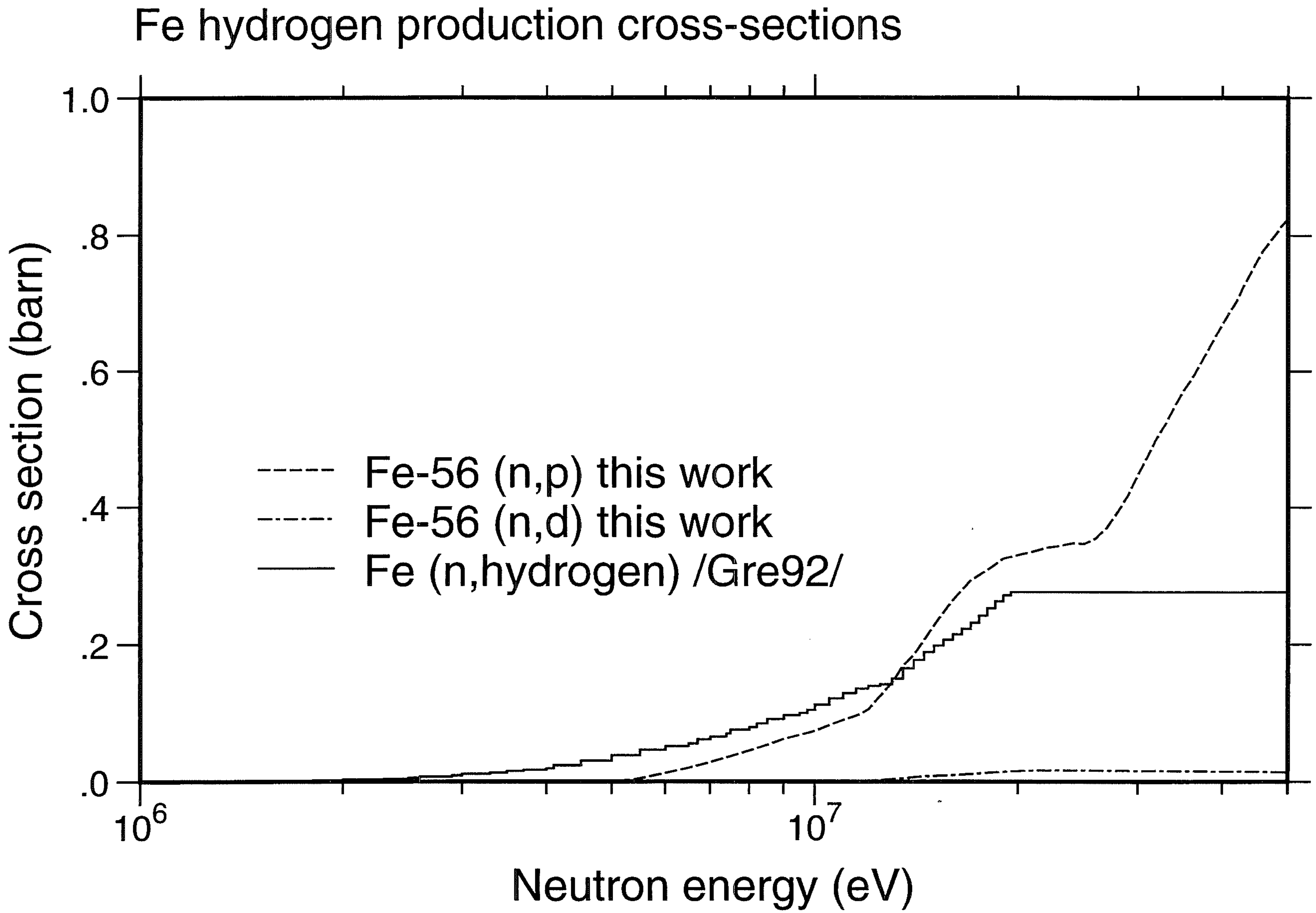
- **ITER & European Demo power reactor**
  - ♦ 3d torus sector model
  - ♦ EFF-1,-2, FENDL-1 nuclear data

- **IFMIF**
  - ♦ Li-target - test module model
  - ♦ Monte Carlo Li(d,n) neutron source function
  - ♦ New high energy cross-section data (FZK-INPE, up to 50 MeV); ^{56}Fe, ^{23}Na, ^{39}K

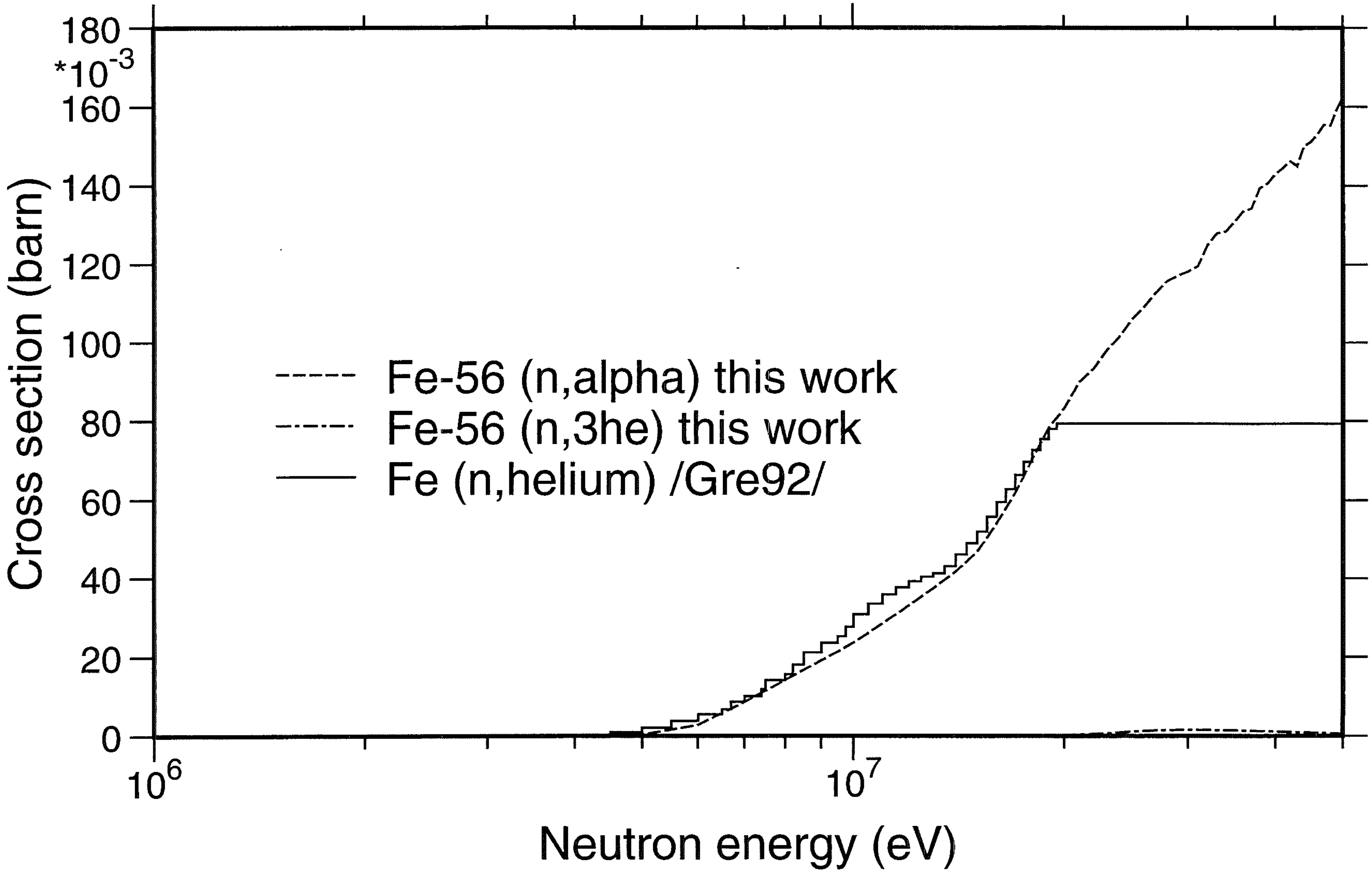
E. Daum, U. Fischer, A. Yu. Konobeyev, Yu. A. Korovin, V .P. Lunev, U. v. Möllendorff, P. E. Pereslavtsev, M. Sokcic- Kostic, A. Yu. Stankovsky, P. P. H. Wilson, D. Woll : ***Neutronics of the High Flux Test Region of the International Fusion Materials Irradiation Facility (IFMIF),*** *Forschungszentrum Karlsruhe, Technik und Umwelt, Wissenschaftliche Berichte FZKA 5868, Januar 1997*


Fe-56 Neutron kerma factors
Kerma factor (eV barn)
10^6
10^5
10^4
10^3
10^2
10^1
10^0
total
elastic scattering
MT=5
10^5
10^6
10^7
Neutron energy (eV)

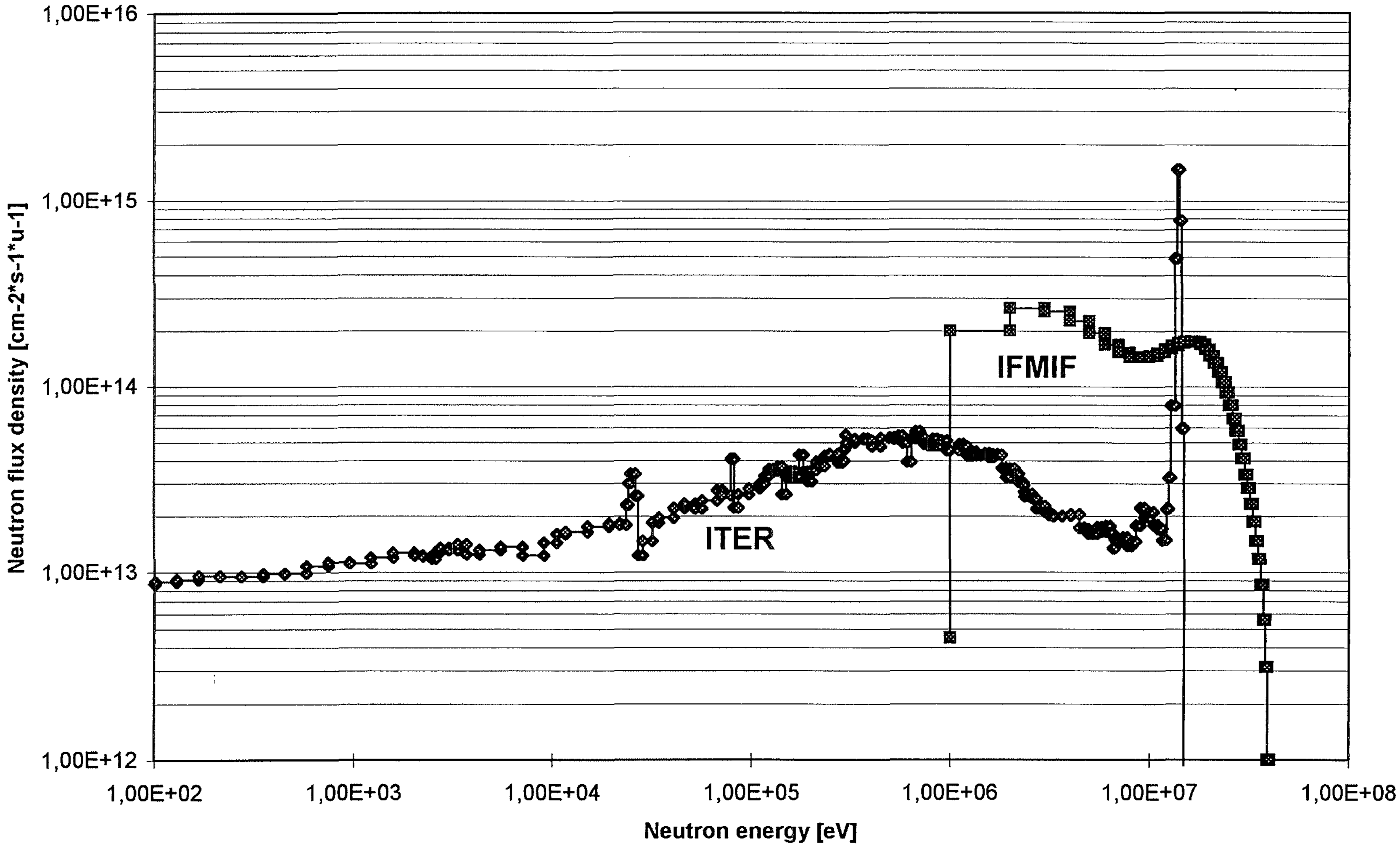

Fe displacement cross sections
Cross section (barn)
Neutron energy (ev)
10^4
10^3
10^2
10^1
10^0
10^5
10^6
10^7
Fe-56 this work
Fe /Gre92/


Fe hydrogen production cross-sections
Cross section (barn)
1.0
.8
.6
.4
.2
.0
10^6
10^7
Neutron energy (eV)
Fe-56 (n,p) this work
Fe-56 (n,d) this work
Fe (n,hydrogen) /Gre92/

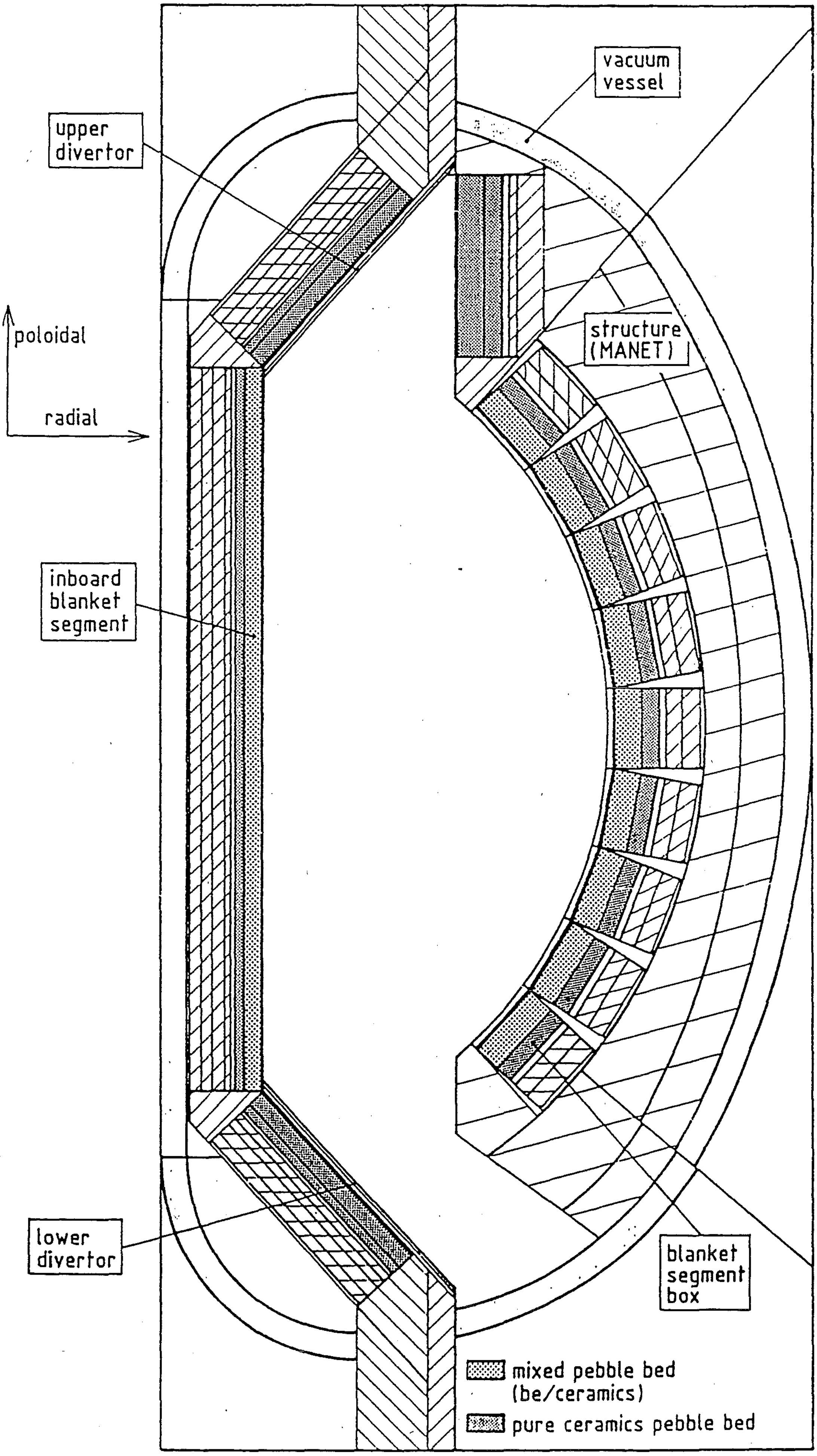
Fe helium production cross-sections

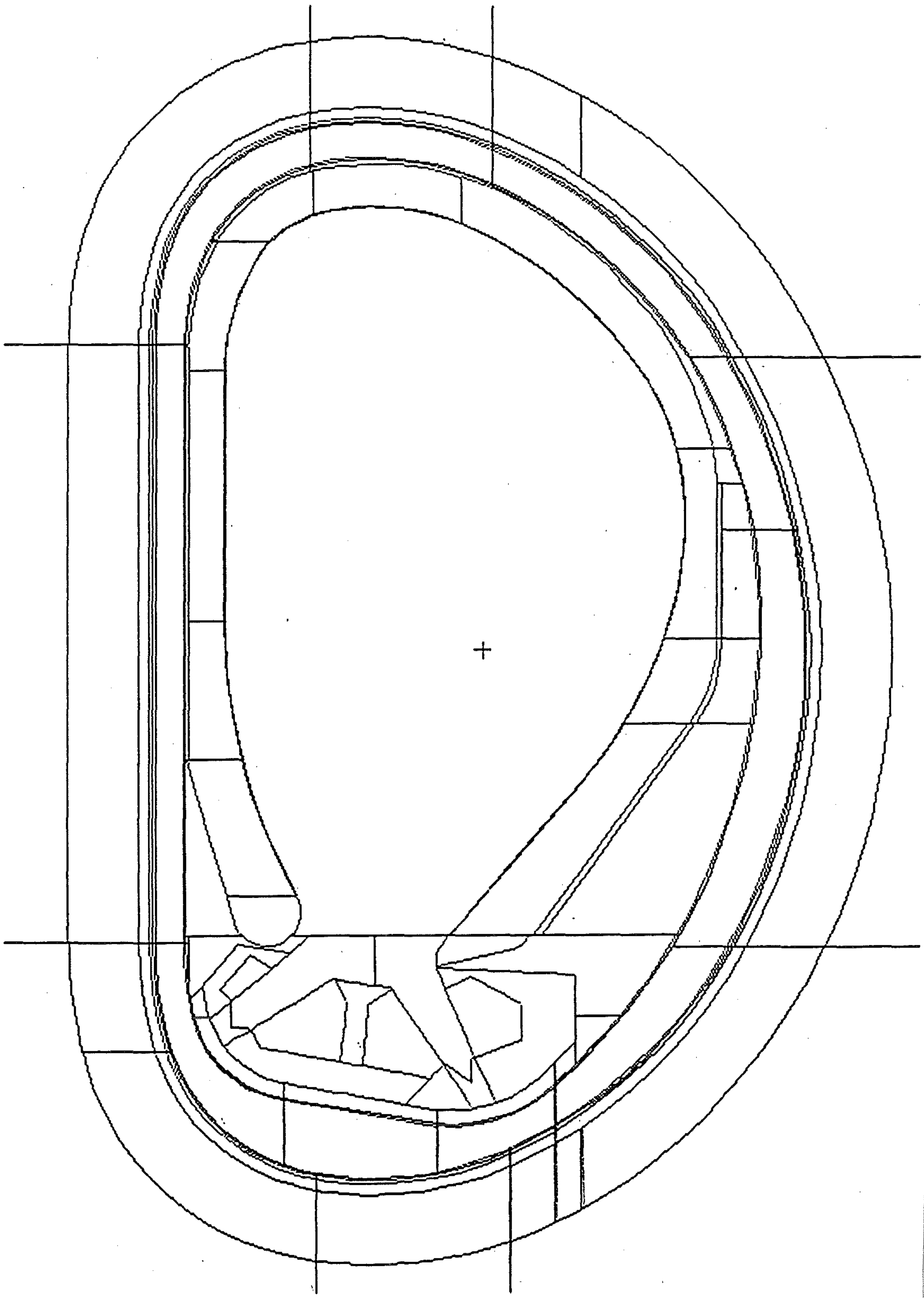


IFMIF vs. ITER: Comparison of first wall spectra

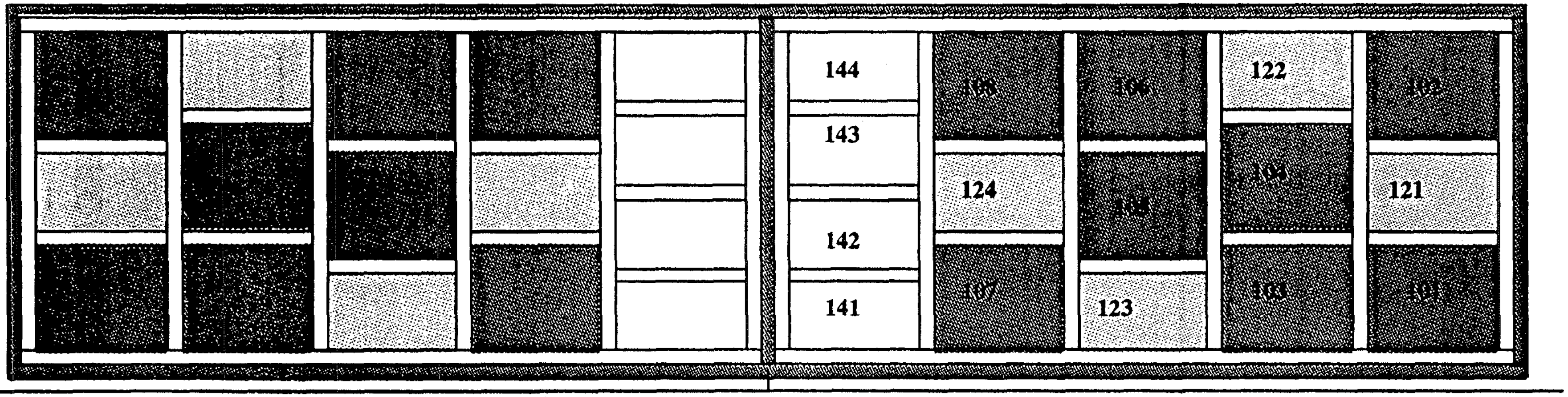


U. Fischer, 07.07.1997


upper divertor
vacuum vessel
poloidal
radial
structure (MANET)
inboard blanket segment
lower divertor
blanket segment box
mixed pebble bed (be/ceramics)
pure ceramics pebble bed



Horizontal cross-section (x-y plane) of the IFMIF helium cooled high flux test module (MCNP model with assigned cell numbers)



Neutron wall loading, first wall flux and current densities

	W_L[MW/m^2]	J_{un}[cm^{-2} s^{-1}]	ϕ_{un}[cm^{-2} s^{-1}]	ϕ_{tot}[cm^{-2} s^{-1}]
ITER	2.0	$8.88 \cdot 10^{13}$	$1.44 \cdot 10^{14}$	$8.00 \cdot 10^{14}$
Demo	2.0	$8.88 \cdot 10^{13}$	$1.15 \cdot 10^{14}$	$7.14 \cdot 10^{14}$
IFMIF *midplane*	8.5	$6.17 \cdot 10^{14}$	$9.63 \cdot 10^{14}$	$1.14 \cdot 10^{15}$
IFMIF *averaged*	6.1	$3.70 \cdot 10^{14}$	$6.32 \cdot 10^{14}$	$7.52 \cdot 10^{14}$

Displacement damage (dpa) and gas production rates

	Displacement damage [dpa]	He-production [appm/FPY]	H-production [appm/ FPY]	He/dpa ratio [appm/dpa]	H/dpa ratio [appm/dpa]
ITER	20	230	891	11.5	45
Demo	17	180	709	10.6	42
IFMIF *midplane*	57	553	2145	9.6	37.4
IFMIF *averaged*	37	339	1320	9.2	35.8

Spectral decomposition [%] of the neutron flux, dpa and gas production rates in the helium cooled test module

	Neutron flux	dpa	He production	H production
E ≤ 0.1 MeV	0.06	-	-	-
0. 1 -1 MeV	14	4	-	-
1 - 5 MeV	45	31	0.2	-
5 - 15 MeV	26	34	29	27
E > 15 MeV	15	31	71	73

Nuclear responses on the IFMIF helium cooled high flux test module (averaged over ± 1 cm around the horizontal mid-plane)

Cell #	dpa per FPY	Neutron flux density [$10^{14} cm^{-2} s^{-1}$]	Helium production per FPY		Hydrogen production per FPY		Nuclear heating [W/cm^3]
			appm	He/dpa	appm	H/dpa	
141	47.3	9.18	492	10.4	1910	40.4	49.3
142	37.0	7.02	410	11.1	1590	43.1	42.3
143	29.4	5.48	340	11.6	1320	45.0	35.6
144	23.3	4.23	281	12.0	1090	46.9	29.0
107	43.8	8.56	457	10.4	1780	40.6	47.4
124	31.7	6.07	353	11.1	1370	43.4	38.0
108	23.4	4.35	274	11.7	1070	45.6	29.3
123	44.5	8.70	459	10.3	1790	40.1	47.0
105	32.1	6.15	357	11.1	1390	43.3	38.3
106	22.4	4.15	266	11.9	1030	46.2	28.4
103	42.8	8.16	462	10.8	1800	42.0	46.6
104	28.7	5.32	333	11.6	1300	45.3	35.0
122	20.3	3.63	251	12.4	978	48.2	26.0
101	34.4	6.42	381	11.1	1480	43.1	37.6
121	23.3	4.27	274	11.8	1070	45.9	28.2
102	20.3	2.94	201	9.9	782	38.5	20.7

Conclusions

- IFMIF requires a higher neutron wall loading than a fusion reactor to achieve the same neutron flux density

- ***But:*** for $I=2\times125$mA, E_D= 40 MeV, neutron wall loading is up to a factor 4 higher as for Demo fusion reactor

- High energy tail of IFMIF neutron spectrum enhances nuclear responses, in particular gas production and dpa

- ***But:*** He/dpa-, H/dpa-ratios are at the same level as in typical fusion reactor spectra

- Nuclear heating in iron of high flux test module dominated by neutron reactions (2/3), while in fusion reactor spectra the opposite is true.

IFMIF well suited to meaningful material tests for fusion reactor development

Future work

- Code system & data to be further developed for IFMIF neutronics

- Neutron source function (neutron yield uncertainty; source neutron spectrum)

  ⇒ Data evaluation for d+Li required

- Data evaluation to be extended

  ⇒ Missing important isotopes & elements

  ⇒ Activation data

- Activation analysis to be included in comparison