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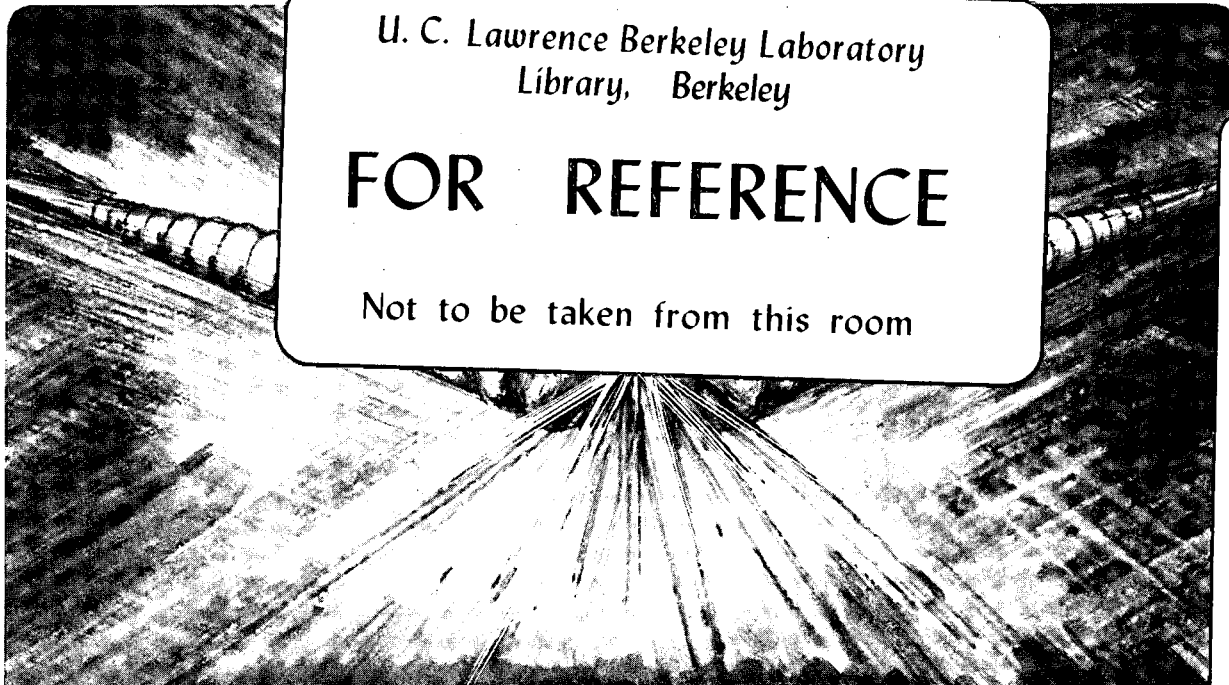# Lawrence Berkeley Laboratory
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ABSTRACT

The International Thermonuclear Experimental Reactor (ITER) has been proposed as the next major step in the development of fusion power [International Thermonuclear Experimental Reactor (ITER), Establishment of ITER: Relevant Documents, International Atomic Energy Agency, Vienna (1988)]. During the Conceptual Design Activity, deuterium neutral beams were chosen for heating, driving current, and controlling the current profile in the central region of the ITER plasma. In order to penetrate to the center of the ITER plasma, an energy of 1.3 MeV is required, an order of magnitude higher than in existing neutral beam systems. The neutral beam system must deliver 75 MW of D^0 to the plasma, with a pulse length that ultimately will reach two weeks. The neutral injection system will consist of nine modules, and will be based on D^- ion sources and novel high current dc accelerators optimized for flexible operation and control of breakdowns. The negative ions can be converted to neutrals in gas or plasma targets, with a resulting system efficiency of 40-50%. The successful development and operation of neutral beam injectors for ITER will be a major step toward the application of neutral beams on power reactors.

I. INTRODUCTION

The goals of the International Thermonuclear Experimental Reactor (ITER) are stated in the Terms of Reference:[1] "The overall objective of the ITER is to demonstrate the scientific and technological feasibility of fusion power. The ITER will accomplish this by demonstrating controlled ignition and extended burn of a deuterium and tritium plasma with steady state as an ultimate objective by demonstrating technologies essential to a reactor in an integrated system, and by performing integrated testing of the high-heat-flux and nuclear components required to utilize fusion power for practical purposes." During the Conceptual Design Activity (CDA), the steady-state goal was reduced to a pulse length of two weeks, which was judged sufficiently long for satisfactory testing of nuclear components such as a tritium breeding blanket. Using only inductive current drive, ITER can sustain pulses of about 400 seconds;[2] for extended burn and testing of reactor grade components, an auxiliary means of driving the current must be provided.

II. NEUTRAL BEAMS IN THE REFERENCE DESIGN

A study of candidate systems during the ITER CDA phase resulted in the selection of a reference design for the current drive and heating system,[3] with parameters given in Table I.

A. The choice of neutral beams for current drive

There were several reasons for the selection of neutral beams for driving current in the center of the machine. First, the physics of neutral beam current drive is well understood, and has been demonstrated experimentally in several machines.[4] The major remaining question has to do with the possibility that ions traveling faster than the Alfvén speed can couple energy to the plasma via instabilities and be slowed down prematurely or even ejected from the plasma, reducing the current drive efficiency; this question is being investigated.[5] Second, because of the high heat load on the divertor in ITER, it is important that the current drive figure of merit γ, which is a normalized efficiency (amperes/watt of injected power), be as large as possible. The figure of merit,[6] given by $\gamma = <n>IR/P$, where $<n>$ is the volume-averaged plasma density, I is the plasma current, R is the major radius, and P is the injected power, is estimated to be 0.45 for neutral beam current drive in ITER, in units of 10^{20} A/m^{2}W. Other current drive schemes considered had smaller figures of merit:[7] lower hybrid waves (0.35), ion cyclotron waves (0.3), and electron cyclotron waves (0.2). Third, the system efficiency for neutral beams, defined as the ratio of the injected power to the wall-plug power, is adequately high, 0.4 to 0.5, depending largely on the choice of neutralizer.

B. Neutral beam system requirements

The most important requirements of the neutral beam system are given in Table II. A neutral beam power of 75 MW is needed in order to drive the current required in the center of the plasma. The high beam energy, 1.3 MeV, is needed for penetration of the tangentially injected beams into the center of the plasma. In order to control the beam penetration into plasmas of varying densities, the beam energy must be variable; three discrete energy steps are sufficient. The pulse length of two weeks is dictated by the requirement that the pulse length be much longer than the time constants in the breeding blanket, some of which are very sensitive to the temperature of the blanket structure. The ability to generate hollow, flat, or peaked power deposition profiles will permit tailoring of the plasma current distribution during a long pulse (together with lower hybrid current drive in the edge plasma), for optimizing confinement.

C. System configuration

The CDA phase resulted in an agreement on the general configuration of the neutral beam system: there would be 9 modules, each capable of injecting up to 10 MW of D^0 into the tokamak, arranged in three vertical banks of three modules per bank. The layout is shown in Figure 1. Each module is approximately 4 m in diameter and 15 m long.

While agreement was reached on the envelope of a neutral beam module and on the general types of components within the module, at the end of the CDA the four ITER parties, the European Communities (EC), Japan, the Soviet Union (SU), and the United States (US), had not reached agreement on the detailed design of the components. All parties agreed that a module will contain negative ion sources capable of delivering about 16 A of D⁻ ions to the accelerator, and that the accelerator would be electrostatic (i.e., not rf). A deuterium gas neutralizer would be used, and means would be provided both to steer the ion beams to provide control of the power profile in the plasma, and to sweep the remaining ions out of the beam after the neutralizer.

Beams based on negative ions are necessary because the efficiency of converting positive ions to neutrals in a gas target, while adequate for existing 80-140 keV neutral beam systems, is too low (less than 0.1 percent) at 1.3 MeV. It is easy to remove the weakly bound electron from a D⁻ ion in a gas target, with an efficiency of about 60%,[8] or in a sufficiently highly ionized plasma target, with an efficiency of ≥ 80%.[9]

III. PROPOSED DESIGNS FOR AN INJECTOR MODULE

Multiple approaches are possible for every component within the injector module, as well as for the high voltage power supplies. We can expect that selections will be made during the next phase of ITER, the

Engineering Design Activity (EDA), but at this stage, the multiplicity of choices is an advantage, as it increases the probability of successful development of the ITER neutral beam system. The choices of key features of components are shown in Table III.

In the case of the ion source, each party favors a different approach. The EC prefers a "pure" volume-production source. In this type of source, by tailoring the electron temperature profile in the deuterium plasma, one can generate respectable currents of D⁻ ions. The ions appear to be produced by a two-step process:[10] excited D_2 molecules are produced in the body of the plasma, and negative ions are produced by dissociative attachment in the region of low electron temperature. Japan favors a cesiated volume-production source;[11] this type of source has recently produced 10 A of H⁻ ions.[12] The addition of small amounts of cesium into a "pure" volume-production source has been shown to enhance the negative ion output by a factor of 3 to 5,[13] apparently due to the production of additional negative ions on surfaces inside the source. Soviet researchers have selected a hydrogen-cesium volume source.[14] In this source a hollow cathode discharge fed by hydrogen and cesium generates a plasma that may consist mainly of D⁻ and Cs⁺ ions. The US team chose a barium surface-conversion source, in which the negative ions are produced on the surface of a negatively-biased barium electrode imbedded in a deuterium plasma.[15,16]

A key to success or failure of the ITER neutral beam system is the accelerator. The accelerator for an ITER neutral beam module must accelerate 16 A of D^- to 1.3 MeV. No one in the world has demonstrated experience at this level. The closest is probably the acceleration of 0.35 A of positive molecular ions to 600 keV in 1967 in the DCX experiment at the Oak Ridge National Laboratory.[17] The performance of this accelerator was limited by x-ray damage to the insulators, which enables us to identify what is probably the most critical element of the design of the ITER accelerator: the control of secondary electrons in the accelerator, which produce x-rays when they strike electrode surfaces. Two types of accelerators have been proposed for the ITER task: the EC, Japan, and the SU propose a more or less conventional electrostatic (ES) accelerator, but with small permanent magnets built into the electrodes to provide magnetic fields for sweeping out secondary ions and electrons produced by collisions within the accelerator structure; the US proposes an accelerator that uses electrostatic quadrupoles (ESQ) both for focusing the beam and for sweeping out secondary charged particles. As is evident from Table III, there is another difference in accelerator design philosophy: the EC and Japan propose ES accelerators with nearly a thousand channels per module, with each channel carrying 15-30 mA of D^-; the SU and the US propose an ES and an ESQ accelerator with few channels (15 or 16), with each channel accelerating about 1 A of D^-.

There are other, more subtle distinctions in the proposed module designs. The EC proposes a plasma neutralizer, which, although adding complexity, offers higher overall system efficiency ($\geq$ 50%) than the simple gas neutralizer preferred by Japan, the SU, and the US. The EC also proposes a grounded source, with the neutralizer at high voltage, as opposed to the conventional approach with the source at high voltage favored by the other three parties.

IV. THE US DESIGN

In order to illustrate some of the design considerations, we will consider in some detail the US design proposed for an ITER neutral beam injector module.[18] This design is shown in Figure 2.

A. Beam production, acceleration, and steering

The heart of the concept is the electrostatic quadrupole accelerator.[19] By using strong-focusing electrostatic quadrupoles, we can separate the problem of focusing the beam from that of accelerating it, thus permitting a solution to the ITER requirement of varying the beam energy over a 2:1 range at full current while maintaining low beam divergence. The accelerator is designed to accelerate 1 A of D$^-$ per channel, and there are 16 parallel channels per module. A single channel ESQ accelerator is shown in Figure 3. The design is modular, with a spatial period corresponding to an energy gain of 250 keV. The average

secondary electron produced by the ion beam gains 64 keV before striking an electrode;[20] it will require an experiment to determine if the x-ray generation is tolerable. The average accelerating gradient is 4 kV/cm, and the maximum average electric fields in the structure are $\leq$ 30 kV/cm. The final minimum beam divergence, based on a beam from a surface-conversion source, is expected to be about 3 mradians (1/e half-width).

While an ESQ accelerator has not yet operated at the level required for ITER, there is a substantial body of operating experience: 1) a similar accelerator operated in Novosibirsk accelerated 80 mA of H^+ to 1.2 MeV for 5 msec pulses,[21] 2) we have operated an ESQ accelerator at LBL to 200 keV, both with 42 mA, 200 msec beams of H^- and with 100 mA, 1 sec beams of He^+,[22] with little beam loss or emittance growth (the He^+ beam tested the ESQ accelerator to its design limit), and 3) we have operated the same structure as a beam transport system at 100 keV, with no acceleration, thus demonstrating the ability to vary the beam energy independently of the current.[23]

In this design, each channel of the ESQ accelerator is fed a 1 A beam of D^- ions from a barium surface-conversion source. The space required for the surface converter electrode dictates the separation distance of the accelerator channels, which is 22 cm. The source plasmas are produced by

1-2 MHz rf discharges; the rf power is transmitted to the sources by coupled, insulating transformers.

The US proposes a high frequency switching type high voltage supply constructed around the high voltage insulator of the accelerator. This high voltage power supply consists of 250 keV transformer-coupled modules operating at 100 kHz that provide the accelerating voltages; a separate set of coupled power supplies provides the focusing voltages for the quadrupoles. In this way, the accelerator is operated by two "knobs," one controlling the beam energy and the other the focusing. The oscillators driving the power supplies are located at ground potential, outside the neutron shielding. This design avoids megavolt transmission lines and large penetrations in the neutron shielding, but at the expense of maintainability and radiation damage. It appears from preliminary calculations that the radiation damage to power supply components will be tolerable, but more detailed calculations remain to be done.

Electromagnets are used to steer each 1 A beamlet ± 15 mradians in the vertical direction after acceleration to accomplish control of the power (and current) deposition profile in the plasma.

B. Neutralizer and gas handling system

The neutralizer consists of a set of narrow ducts 5 m long enclosing the beamlets. The ducts are shaped to fit the beamlets snugly, but still

permit beam steering; they function as restrictions to minimize the gas flow and also incorporate neutron shielding, to reduce the flux of neutrons into the region of the accelerator, ion sources, and power supplies to a tolerable level.

Pumping is by cryocondensation pumps that can be regenerated during operation.[24]   A total of 20 m^2 of pumping surface is active at any one time, with 13.75 m^2 being regenerated.

C. Ion dumps

A neutral beam emerging from a gas neutralizer of optimum thickness consists of approximately 60% neutrals, 20% unstripped negative ions, and 20% positive ions. A compact ion dump was chosen, with an integrated magnetic field. The dump consists of two arrays of water-cooled copper louvers arranged parallel to the beam axis. The louvers carry a current, 8 kA per meter of height, which generates a vertical magnetic field between them of about 100 Gauss, enough to deflect the remaining ions in the beam into the same louvers and remove them from the beam.

D. Performance

By deflecting the beams to the extent permitted by the constraints of the duct, the neutral beam power deposition profile in the plasma can be made hollow, flat, or moderately peaked. These calculated profiles are shown in Figure 4.

It will be a difficult task to develop neutral beam systems capable of operating reliably for two weeks at a time. As a start, and as a guide to determining the component reliability required, we have simulated the operation of the ITER neutral beam system proposed by the US.[25] With mean times to failure of 1000 days for cryopumps and beam dumps, and 300 days for other components, and with mean times to repair of 1-5 days (depending on the component), we found that 44% of the time the neutral beam system would still be delivering 75 MW at the end of a two week shot, and that the average beam power at the end of two weeks would be 68 MW. It was assumed that necessary repairs could be effected in the 6 weeks between shots (satisfying the ITER availability requirement during the technology phase of 25%).

V. NEUTRAL BEAM R&D FOR ITER--GOALS, STATUS, AND PLANS

During the CDA phase, a plan was developed for neutral beam research and development for ITER that should be carried out during the next phase, the Engineering Design Activity (EDA).[26] This plan was reviewed by the ITER Steering Committee, U. S. (ISCUS), and also by the U. S. National Review of the ITER Conceptual Design Activity.

A. Goals

Three major milestones were identified by the U. S. National Review; they were quite similar to those identified in the ITER CDA phase, and are

being used to guide the course of neutral beam R&D in the U. S. They are

1. Carry out proof-of-principle demonstrations of both the ES and the ESQ accelerators, at the levels of 1.3 MeV, $\geq$ 1 A of H$^-$, and a pulse length of $\geq$ 2 sec,

2. Operate a suitable (scalable) negative ion source for 2 weeks, and

3. Operate a Scalable Model Beamline at a level of 1.3 MeV, 2 A (D^0), and a pulse length of two weeks.

B. Status of worldwide neutral beam R&D

Neutral beam development for ITER has been coordinated via a series of Specialists' Meetings; a summary of the status of worldwide neutral beam R&D, as reported at recent such meetings, is summarized in Table IV. In this table the results are arbitrarily arranged in order of decreasing current.

We see from the table that there have been several demonstrations of negative ion source outputs of 1-10 A, and that there is a demonstration at a lower current of 24 hour operation. Since an ITER neutral beam module must produce and accelerate about 16 A of negative ions from several sources, and will not have to operate for two week pulses until the year 2010 at the earliest, it appears that the development of suitable ion sources is relatively well in hand. Although much work remains to be

done, there seems to be no fundamental reason that one or more of the concepts being investigated cannot meet the ITER requirement.

It is quite a different matter in the area of accelerator development, however; there is no experience whatsoever at levels approaching those required for ITER. In fact, there is no facility anywhere in the world known to the ITER neutral beam developers where the required proof-of-principle accelerator demonstrations could be carried out. One or more new facilities are required.

C. Plans

At the most recent ITER Neutral Beam Specialists' Meeting, held at the Kurchatov Institute in Moscow in October, 1991, the four parties reviewed their plans for neutral beam development. All four are continuing development of negative ion sources. The Japanese programs at the National Institute for Fusion Science (NIFS) and the Japanese Energy Research Institute (JAERI) are focused on domestic applications, at 250 and 500 keV, respectively. The EC, SU, and US programs have no domestic customers, and so support ITER directly, but do not have strongly funded programs for ITER neutral beam development. All four parties proposed new facilities for carrying out all or part of the proof-of-principle accelerator tests. However, only the SU is actually constructing a new facility, MIN-2, at the Kurchatov Institute. This facility will be able to

satisfy the energy and current goals of a proof-of-principle test of an ES accelerator, but will be capable of a pulse length of only 20 msec initially, with a later upgrade to 0.1 sec. It is not likely that an adequate program will be in place to carry out the required neutral beam development for ITER until all four parties have signed the agreement to permit initiation of the next phase, the EDA, and the goals and mission of the machine have been reexamined as the first step in the EDA.

VI. NEUTRAL BEAMS FOR REACTORS

It seems reasonable to conclude this survey of neutral beam development for ITER with some comments on the next step, the application of neutral beams on fusion power reactors. While neutral beam systems offer a transparently simple and passive interface to the tokamak, namely a duct through the neutron shielding, this same advantage means that the neutral beam system will be exposed to neutrons produced by the fusion reactions in the reactor. The JAERI FER reactor design[27] was the first in which the intrinsically small divergence of high energy neutral beams (a few mradians at energies appropriate for a reactor) was used to advantage: the neutral beam modules were deliberately removed as far as possible from the reactor, in order to reduce the neutron radiation level at their location. Another recent reactor study[28] indicated that the neutral beam system for a 1200 MW (electrical) power reactor

based on ITER would have to operate relative to the ITER neutral beam system at 50% higher energy, 20% higher power (for a net *reduction* in total negative ion current required), and higher resistance to damaging radiation by a factor of 7 to 50, which could be accomplished by locating the neutral beam systems 100 m from the reactor. Such an advanced neutral beam system should operate with a system efficiency of over 70%. Substantially higher reliability or maintainability will also be required, but it is clear that the successful development of neutral beam systems for ITER will constitute a major step toward reactor-level systems.

VII. CONCLUSION

It seems likely that several negative ion source concepts now under investigation can be developed to meet the ITER neutral beam requirements. The key question, however, is that of accelerating amperes of negative ions to 1.3 MeV, under dc conditions. Although paper studies and some experimental evidence imply a reasonable probability of success, this requirement is substantially beyond our present experience. No facility exists to carry out a proof-of-principle accelerator demonstration; several of the ITER partners have proposed suitable facilities, but none is funded at the present time. It will take an aggressive program to develop neutral beams for ITER. When this goal is reached, the next step, in some ways easier, will be the development of

very efficient neutral beam systems for heating and driving current in steady-state fusion power reactors.

ACKNOWLEDGMENTS

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TABLE I. Reference design for ITER current drive and heating systems.

Source	Power (MW)	Function
1.3 MeV neutral beams	75	Heating the plasma Driving current in the plasma center Controlling the current profile Burn control (physics phase)
5 GHz lower hybrid rf	50	Heating the plasma Assisting current ramp-up Driving current in the plasma edge
120 GHz electron cyclotron rf	20	Plasma initiation Disruption control

TABLE II. Requirements of the ITER neutral beam system

| Parameter	Value
Injected power (H^0 or D^0)	75 MW
Energy	1.3 MeV
Energy steps at full current	1.0, 0.75, 0.5 times full energy
Maximum pulse length	2 weeks
Power deposition profiles	Hollow, flat, or peaked

Table III. Features of proposed neutral beam modules.

Party	Ion source	Accelerator	Method of electron control	Neutralizer	Source potential	Power supply
EC	Pure volume	Electrostatic, 960 channels	Magnetic	Plasma	Ground	External
Japan	Cesiated volume	Electrostatic, 720 channels	Magnetic	Gas	High voltage	External
SU	Hydrogen-cesium volume	Electrostatic, 15 channels	Magnetic	Gas	High voltage	External
US	Barium surface-conversion	Electrostatic, 16 channels	Electro-static	Gas	High voltage	Integrated

Table IV. Representative accomplishments in H⁻
production and acceleration.

Current (A)	Energy (keV)	Pulse length (sec)	Institution
10	50	0.1	JAERI
3	3	100	KIAE
1	80	30	LBL
0.25	50	8.6×10^4	JAERI
0.04	300	3	JAERI
0.04	200	0.2	LBL

FIGURE LEGENDS

Figure 1.   Plan view of ITER, showing the current drive and heating systems.   The neutral beam system utilizes three vertical arrays of three injector modules each, for a total of nine.   Each module can inject 10 MW of D^0.

Figure 2.   Schematic design of the proposed U. S. version of an ITER neutral beam module.

Figure 3.   A single channel Electrostatic Quadrupole (ESQ) accelerator. An ITER neutral beam module would use 16 such channels in parallel, each accelerating 1 A of D^- ions to 1.3 MeV. Electrode potentials are shown relative to the ion source.

Figure 4.   Calculated power deposition profiles on a plane normal to the beam and coincident with a major radius of the machine.   The neutral beam system can generate peaked, flat, or hollow profiles for controlling the current distribution.
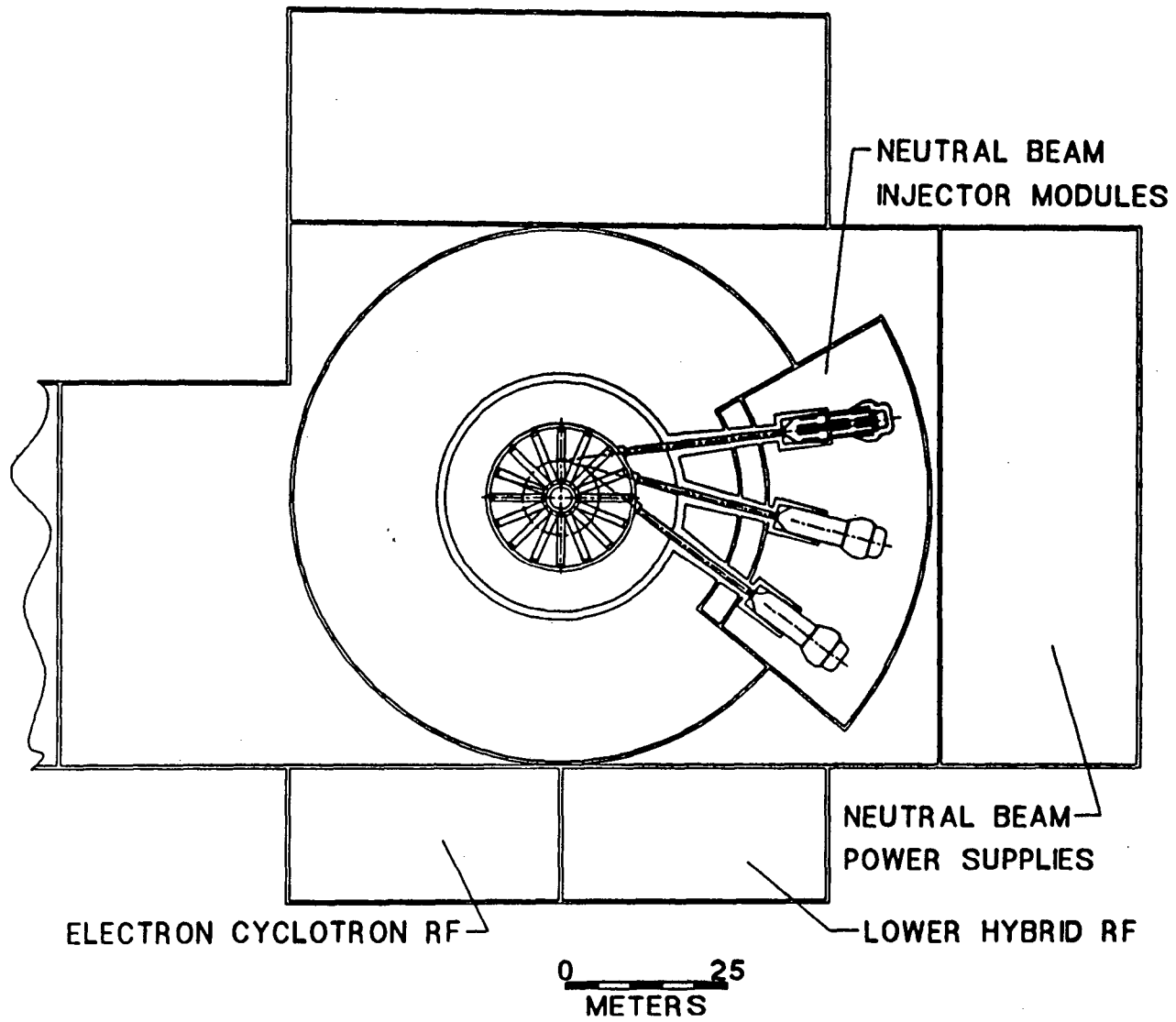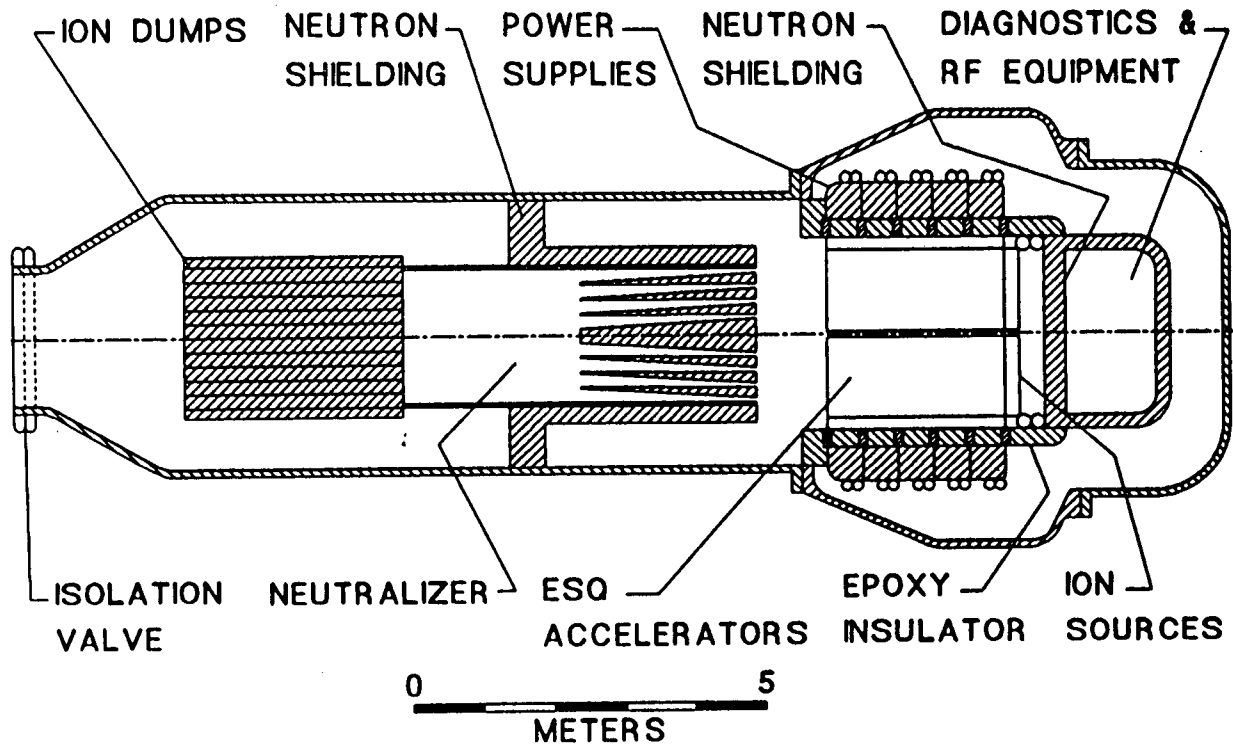
NEUTRAL BEAM
INJECTOR MODULES

NEUTRAL BEAM
POWER SUPPLIES

ELECTRON CYCLOTRON RF

LOWER HYBRID RF

0 _____ 25
METERS

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FIGURE 1

FIGURE 2

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FIGURE 3

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FIGURE 4

POWER DENSITY (kW/cm²)

HOLLOW

PEAKED

UNIFORM
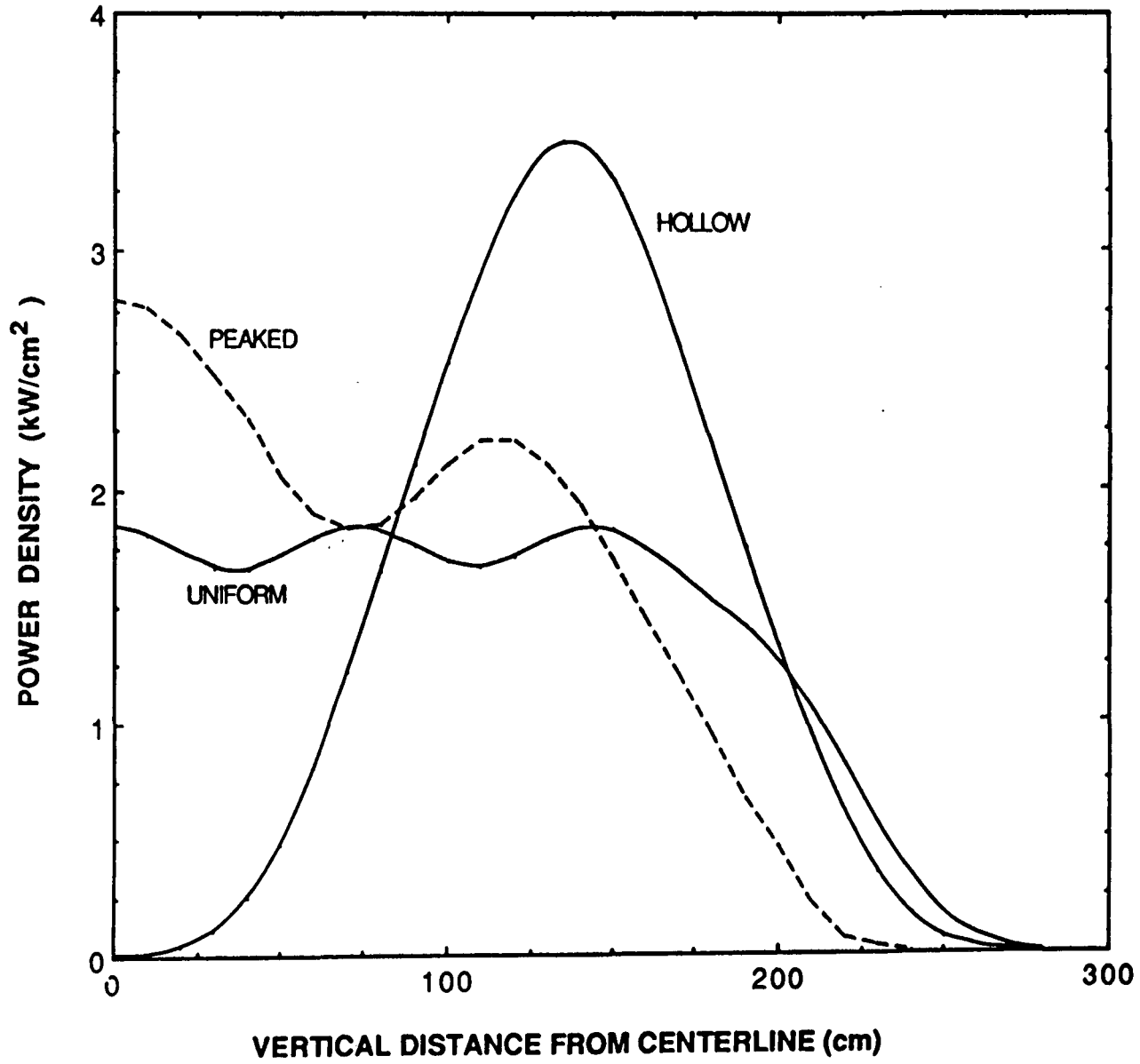
4

3

2

1

0

0   100   200   300

VERTICAL DISTANCE FROM CENTERLINE (cm)
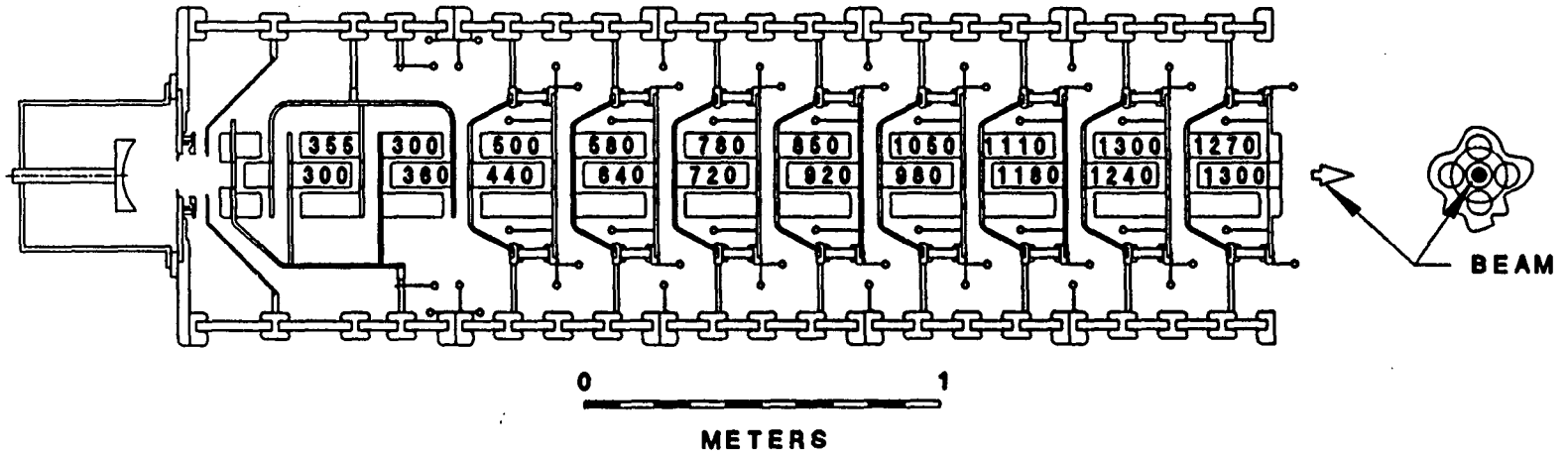
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