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PHYSICS PROGRAMMES AND GOALS OF LARGE TOKAMAK EXPERIMENTS

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This report describes physics programmes and goals of four large Tokamak experiments (LTX); i.e., JET, TFTR, JT-60 and T-15. Physics problems foreseen in LTX are reviewed in the light of presently available results from Tokamaks in operation. Programmes and objectives of LTX are described by each project. Their various aspects are then reviewed. Possible collaborative programmes among the projects are briefly discussed.

The report is based on the discussion held at the physics session of the fourth IAEA Technical Committee Meeting on LTX (Tokyo, April 1980). It is compiled from contributions by session chairmen of the meeting and appropriate persons of each project.

Keywords: Tokamak, Plasma, Experimental Programme, International Cooperation,

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Report on the Physics Session of Fourth IAEA Technical Committee Meeting on Large Tokamak Experiments, Tokyo, 14-18 April, 1980).

大型トカマク実験の計画と目標

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IAEA主催第4回大型トカマク装置技術委員会（1980年4月，東京）の，物理セッションにおいて，四大トカマク（JET，TFTR，JT－60，T－15）の実験の計画と目標に関して行われた議論をまとめた。内容は，大型トカマク実験の物理的考察，各プロジェクトの計画，プロジェクト間の協力からなり，執筆者は上記会議における座長ないし各プロジェクトの代表者である。

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1. INTRODUCTION (H. Kishimoto, JT-60)

The 4th IAEA Technical Committee Meeting discussed the status and plans of the large Tokamak experiments (LTX) now being constructed[1]. Physics considerations and experimental programs of the respective project were introduced in physics sessions. The latest results of medium-size Tokamak research; ISX-B, PLT, PDX, DOUBLET-III, and T-10, were reported and, in addition, proposals for the future Tokamak research programmes were presented. The present situations and roles of the large Tokamak programmes became clear from these presentations and discussions.

The four large Tokamaks, TFTR(USA), JET(EC), T-15 (USSR), and JT-60(Japan), are now under construction after the completion of principal design and are expected to enter the experiment successively within a few years. In the meanwhile, the Tokamak experiments in small-size and medium-size devices have proceeded successfully in the recent few years. The encouraging results with the increased parameters of plasmas have been obtained in these devices. Such progress of Tokamak research makes it certainly promising to produce reactor-grade plasmas in the LTX.

The enhanced activities of the next-generation Tokamak (ETR) design study have given a considerable impact to the LTX programmes. INTOR is the typical one, which has been undertaken by several countries such as USA, USSR, Japan, and EC under a sponsorship of IAEA. The INTOR workshop held in Vienna picked up the research and development items required for the design, construction, and operation of INTOR. The roles of the LTX programs should now be reexamined from a view point of the ETR design.

Every project proposed in the meeting up-grade programmes to extend the performance beyond the initial one. Modification programmes of the LTX would be discussed futhermore to meet the future reguirements of fusion research.

It seems doubtless that any LTX project cannot cover the whole area of Tokamak fusion research toward a reactor. The programmatic collaboration among the four projects might be important and was discussed in the LTX meeting.

Section 2 gives physics consideration for large Tokamak experiments based upon results from devices presently in operation. Section 3 consists of two parts. The first part reviews LTX programmes under the headings of the four large Tokamaks, while the second reviews them from different view-point, under the headings of main objectives, up-grade programmes, data base assessment and experimental schedules. Section 4 briefly discusses collaborative programmes among the four projects.

2. PHYSICS CONSIDERATIONS FOR LTX (A. Gibson, JET)

The meeting was privileged to hear new results presented from some of the largest Tokamaks now in operation (Doublet III, PLT, PDX and T-10). The results from these experiments are directly relevant to the problems and uncertainties which confront us when we attempt to predict the plasma behaviour in the large Tokamaks in construction (JFT, JT-60, TFTR and T-15).

In the session on physics considerations some of the more important of these problems were reviewed in the light of presently available information under the following headings.

(i) Impurity Control

Present experiments obtain tolerable to good impurity control by using extensive wall conditioning procedures and by providing a low Z environment for the plasma. These methods will certainly be used in the new large Tokamaks but may prove inadequate with high temperature, long duration plasmas. Simulation calculations with a variety of models suggest that impurity generation (particularly by sputtering) and impurity influx can be effectively restricted by controlling the plasma edge conditions in various ways, such as by inducing a cool plasma mantle to form. There is no guarantee that this technique will prove as effective with plasma as in the simulation, but there are indications that similar processes to those modelled do appear in present experiments and can be used to reduce the impurity influx. Workers from JAERI have operated Doublet III in a mode where a magnetic separatrix forms the boundary of a D-shaped plasma in the upper half of the device.

The lines of force outside the separatrix then intersect the vessel in a band near the inside mid-plane and act as a magnetic limiter. Initial results indicate a reduction of impurity level in the magnetic limiter configuration, and this is encouraging for JT-60 which has as a main aim the exploration of impurity reduction by magnetic limiter.

One of the wall conditioning methods used in present tokamaks is gettering with titanium. This is undesirable in Tokamaks intended to use tritium[2]. Alternative getter materials such as chromium could be tried but it is interesting to note that: while PLT requires gettering to reach the highest temperatures and while the maximum attainable density in Doublet III is increased by 50% with gettering; ISX-B, T-10 and TFR do not getter and obtain densities similar, in terms of B/R, to those in other experiments.

(ii) Density Limits; Heating Methods

The first requirement on the density in a large tokamak is that it should be possible to produce an ohmic discharge with high enough density to act as a target plasma for neutral injection at the required energy of 60 to 80 keV/nucleon. The second requirement is that with additional heating it should be possible to reach high enough densities to approach ingition (say $\bar{n} \sim 2 \times 10^{20} m^{-3}$ or $\bar{n}a \sim 2.5 \times 10^{20} m^{-2}$ in JET). Experience with present Tokamaks suggest that it will be possible to meet the first requirement in the new Tokamaks, possibly with some staging of injector switch on. On the second requirement there were encouraging results from ISX-B which showed that the "density clamping" effect previously reported, could be overcome by obtaining pure, high current discharges. With such discharges the density and β value increase continuously with injected power, β^* values of 3.5% are obtained although the density is only ~20% higher than in the best discharges with ohmic heating alone.

A less promising result was reported from PLT where it was found that the efficiency of neutral injection heating falls off markedly for densities greater than that at which the beam penetration length becomes less than the plasma radius. This result appears to be more restrictive than suggested by many simulation codes and may well be associated with the observation that the thermal transport coefficient in PLT increases quite strongly with radius.

These difficulties emphasise once again the necessity of developing an alternative to neutral injection heating. In this respect results of ion cyclotron (I.C.F) minority species heating on TFR and PLT have established this technique as a convincing heating method, approximately comparable in status to that of neutral injection heating three years ago. Temperatures of 2 keV are obtained in PLT with 0.5 MW of r.f. power coupled to the plasma, impurity production remains a problem.

It is expected that multimegawatt I.C.F. systems will operate on TFR and PLT in 1981 and 6 MW of neutral beam heating is to be applied to PDX by early 1981. Some variability in the efficiency of neutral beam heating has been observed in PLT, this may be associated with uncontrolled profile variation and may be an indication of similar problems ahead for the new large Tokamaks.

(iii) Disruption Control and Discharge Termination

No effective method of disruption control has been demonstrated and present Tokamaks choose to operate, as far as possible, in regimes where disruption does not occur. A number of disruptions have been encountered at high currents in Doublet III ($\lesssim$ 1 MA D-shaped and $\lesssim$ 2 MA doublet shaped) without severe damage. The time-scale for the current decrease in these disruptions tends to be relatively long (~10 msec) and the highly segmented limiter design of Doublet III may be effective in reducing damage.

PDX, PLT and T-10 appear readily able to terminate the discharge in ~0.1 sec without provoking disruption, merely by removing the drive voltage.

(iv) Shape Control

The JET device is intended to operate with D-shaped plasma elongations of up to 1.6 and it is expected that this will give increases in maximum attainable energy replacement time and β value. Consequently it was encouraging to hear that D-shaped plasmas have been stably maintained in ISX-B and in Doublet III. In Doublet III outer surface elongations of 1.6 have been maintained for long times at current levels as high as 1 MA. There is however, a tendency for the current density to concentrate on axis giving circular surfaces in this region.

(v) Low q Operation

The T-10 experiment has demonstrated stable discharges with limiter q values as low as 1.9 and thermal transport similar to or less than that given by "Alcator" scaling. Argon impurity and hydrogen containment times are equal at about twice the energy replacement time of 25 msec. The discharges are obtained by increasing the plasma-wall separation to about 10 cms, using a large limiter. These results open up the possibility of increasing either the current or the planned plasma-wall distance in the new large Tokamaks.

(vi) Discharge Initiation

The existing large Tokamaks have all found relatively simple ways to build up the plasma current, and this leads to the expectation that the current will be readily established in the new Tokamaks. The T-10 discharge was said to permit rates of rise in the range 10^6 to 10^7 A/sec depending on the wall condition. There were indications from Doublet III that empirically profiling the rate of rise to induce mild fluctuations could increase the maximum attainable plasma density.

(vii) Fuelling

Gas puffing techniques optimised by trial and error have proved successful in existing experiments and should extend to the new Tokamaks. Controlling the D-T ratio in the face of deuterium injection may prove more difficult.

Summary

Many problems remain to be solved before JET, JT-60, TFTR and T-15 can be operated to achieve their objective of demonstrating the physical feasibility of a Tokamak fusion reactor. However, the results which have come from existing experiments have many favourable implications for the new Tokamaks, and if it proves possible to continue to limit the impurity influx as the temperature and time increase, and if enough effective heating can be provided, the prospects look very good indeed.

3. OBJECTIVES AND EXPERIMENTAL PROGRAMMES

3.1 Programme of Each Project

3.1.1 JET (A. Gibson)

(1) Objectives

The objectives of JET were set out in[3] and are as follows.

The essential objective of JET is to obtain and study a plasma with conditions and dimensions approaching those needed in a thermonuclear reactor. These studies will be aimed at defining the parameters, the size and the working conditions of a Tokamak reactor. The realisation of this objective involves four main areas of work:

i) the scaling of plasma behaviour as parameters approach the reactor range,

ii) the control of the plasma-wall interaction in these conditions,

iii) the study of plasma heating, and

iv) the study of α-particle production, confinement and consequent plasma heating.

The important physics features of this fourth main area of work, as given in[4], are:

i) to study the production and behaviour of α-particles and their interaction with the plasma, and

ii) to continue the investigation of plasma behaviour in a system where the profiles are controlled by the balance between fusion power generation and losses due to thermal condition and radiation.

(2) Scientific Programme Aims

The overriding aim of the programme of work on JET must be to achieve as rapidly as possible a plasma with conditions and dimensions approaching those needed in a fusion reactor. This is a joint applied physics and engineering task so that the JET scientific programme is not primarily aimed at basic or fundamental physics. There are of course many aspects of Tokamak containment which are, as yet, not understood, and some progress in this understanding will be necessary in order to achieve reactor-like parameters in JET. However, experiments to determine the detailed nature of these phenomena should be undertaken first in present devices. On JET it is intended to carry out only such physics experiments as in our judgement enable us both to proceed most rapidly to the tritium-burning phase, and to obtain the design information for the post-JET machines such as INTOR.

Four phases are foreseen in the operation of JET and these are summarized with the aims of each phase in table 1. The timing of these phases is almost entirely determined and paced by the schedule for the provision of additional heating power, and dates shown are to be regarded as the earliest possible, given very favourable progress with the construction and exploitation programmes.

3.1.2 TFTR (R. Little)

The major objectives of TFTR are:

i) to generate and confine reactor-like plasmas ($T \sim 5$-10 keV, $n \sim 10^{14}$ cm^{-3}, $n\tau \sim 10^{13}$-10^{14} cm^{-3} sec.)

ii) to study the physics of burning plasmas and the engineering aspects of a D-T Tokamak with reactor-level plasmas (fusion power ~1 W/cm^2).

The present major milestones are:

i)	first ohmic discharge	December 1981
ii)	start 3 neutral beam operation	June 1982
iii)	full D-T diagnostics available	December 1982
iv)	start 4 neutral beam operation	June 1983

The experimental programme is still in an evolutionary state, but some broad outlines are given below.

The first six months (December 1981 to June 1982) will be devoted to magnetic measurements and glow discharge cleaning, followed by initial ohmic heating experiments. These experiments will investigate Taylor discharge cleaning, limiter studies, plasma handling and control and diagnostic check-out. Following installation of the first three neutral beams (June 1982) transport studies, impurity control and plasma handling will continue. In addition compression and plasma rotation studies will be performed. These studies will be performed with hydrogen and deuterium plasmas. It is expected that these studies will continue until June 1983, when the fourth neutral beam will be installed. Depending on the results achieved, the neutral beam orientation may be changed from tangential injection to near perpendicular. At this time it is expected that the radiation shielding will be completed around the Tokamak, and limiters and protective armor may be modified.

During the second half of 1983 and first half of 1984 it is hoped to achieve the baseline performance parameters with D-D operation. At this time there may be a rearrangement of priorities to re-orient the neutral beams to concentrate on high-current discharges. It is expected that D-T operation will commence at some time in 1984.

3.1.3 JT-60 (Y. Shimomura)

(1) Objectives

The objective of JT-60 is the study of a reactor grade plasma including plasma-machine interface. In other words, technology to generate and maintain a reactor grade plasma has to be researched and it is necessary to show how we can obtain and control a reactor core with realistic methods. This research programmes, as presently considered, consists of many experimental items as shown below.

1) Preliminary Experiment

In the first phase of this experiment, plasmas only with ohmic heating are studied. A suitable target plasma for heating has to be obtained with and without the divertor. Discharges, however, will not be intensively studied in a wide range of plasma parameters because installation and/or test of heating devices, diagnostics and the real time control system will be more important. A $Q^* = 1$ plasma will be easily obtained with 20 ~ 30 MW heating because the plasma will have been obtained in the other large Tokamak devices. The main objective in this phase is to confirm the results in the other projects and to prepare the following experiments.

2) Basic Experiment

In this experiment, the standard operation with 30 MW and 10 sec heating is investigated with following items.

(i) Build-up phase of $Q^* \approx 1$ plasmas

Clean and stable current raising by using the poloidal divertor, control coils, dynamic limiter, gas puff and/or heating devices is investigated. Heating experiment with 20 MW NBI and 10 MW RF with and without the poloidal divertor will be done. 14 injectors and 4 sets of RF devices are independently operated. It will be necessary to control the scrape-off layer plasma as well as the main plasma.

(ii) Confinement characteristics of reactor grade plasmas

Detailed parameter study with and without the poloidal divertor is made.

(iii) Steady state operation

Control of a long pulse plasmas by the real time control system is studied. Not only the plasma position and shape but also the major plasma parameters are controlled by employing various actuators with and without the poloidal divertor.

(iv) High β operation

The maximum β value will be obtained and high β plasmas are maintained during 5 ~ 10 sec. Build-up process will be intensively investigated as well as characteristics of the high β plasma with and without the poloidal divertor. Very-low-q operations are also investigated.

(v) First wall and impurity experiment

In order to obtain first wall data for a reactor, various kinds of wall materials including low-Z, medium-Z and high-Z materials are tested as the first wall. The first wall material to be used in a reactor will be intensively investigated with various wall temperature up to 500 °C. Impurity control by cooling the scrape-off layer plasma is investigated as well as by using the poloidal divertor.

(vi) Plasma operation to mitigate engineering difficulty

The following items will be studied: Disruption control; slow current raise and shut down; slow heating; reducing the heat flux density by using swinging divertor; edge cooling and/or vibrating the plasma column; conventional wall material; impurity control without the poloidal divertor; systematic study of stress of coils, vacuum chamber and so on; easy cleaning method; simulation of easy ash exhaust system such as the simple poloidal divertor and/or mechanical divertor; high efficient RF heating; very-low-q operation free from disruption.

(vii) High efficiency heating

The maximum RF power into the torus is 23 MW. If the scrape-off layer plasma can be controlled, the heating efficiency can be increased.

(viii) Development of a control system for a reactor core

It is necessary to develop a reliable real time control system, especially simple sensors for controlling plasma parameters including temperature, density, β, impurity, heat flux and heating. This problem is also planned to be studied.

3) Advanced experiment

It is considered to raise the heating power in the advanced experiments. After increasing NBI power up to 30 MW and RF heating efficiency, the total heating power will be 40 ~ 50 MW. A $Q^* \approx 3$ plasma will be obtained, which is very similar to the reactor core plasma. The each experiment listed in 2) will be also done with this condition. Extremely long pulse operation will be also studied if it is required.

4) Modification

When the design of the next generation reactor is established, JT-60 is planned to be modified so as to complement its programme. Among options considered, one is to conduct simulation or exploratory experiments in hydrogen plasmas to support the programme (Proto ETR experiment).

(3) Experimental Programmes and Schedule

Experimental programmes and schedule of JT-60 are briefly summarized in fig.1 and related programmes are also shown in fig.2. The most serious problem of the JT-60 project is the time delay. In order to overcome this situation, high efficient operation is required as well as through preparation. To realize the high efficient operation, it is necessary to obtain reliable heating devices as well as the reliable main machine including the control and diagnostic system, and it is also necessary to operate the machine during 24 hours a day. Concerning the preparation, training of many excellent staffs is very important as well as preparation of excellent numerical codes. These are being intensively pushed by all the members of The Fusion Research Center at JAERI cooperated by other institutes, universities and industries.

3.1.4 T-15 (B.B. Kadomtsev, V.S. Strelkov and V.D. Shafranov)

(1) Programme of investigations

A great number of physical plasma-technological and engineering investigations will be carried out on T-15. These investigations are of great interest for the reactor. In fig.3 are shown the schedule of experiments after physical start-up of the installation.

According to this schedule, after the adjustment, the 1st stage, the detailed examination will be carried out in pure ohmic heating. The goal of this experiments is to check scaling in large-scale plasma, to specify our data of such a plasma and to define the area of stable operation.

But the main thing is that in order to obtain the clear operation of creation of pure enough high-temperature plasma, the initial stage and the stage of current increase should be optimized. On the 3rd stage of auxiliarly heating a wide range of experiments are proposed to be carried out. These investigations involves study of equilibrium, stability, transport processes in high temperature plasma, optimization of auxiliary heating plasma regime, study of plasma-wall interaction process and engineering physical investigation.

The more detailed report of investigations is given below in table 2.

(2) T-15 features and possibilities

T-15 has a number of specific features. These features can stimulate special experimentation with large-scale high-temperature plasma.

i) T-15 has the superconducting magnetic system which needs carrying out of prophylactic works and long phases of cooling-recooling. The work of T-15 is expected to consist of cycles at 1.5 - 2 months each thrice a year. So the diaphragm can be replaced and special material put on plasma-wall boundary through the vessel windows (window dimensions 0.5×1 m^2) during stop intervals.

ii) The pulse duration of 5 sec is chosen due to the suggestion that the loop voltage in ohmic heating will be about 1.5 V. In 19 V/sec of inductor being planned, the smaller loop voltage will give the possibility to increase the pulse duration. Auxiliary heating power being smaller, the increase of heating duration can be obtained.

iii) In acceptable iron core saturation the current can be risen up to 2.3 MA (25 V/sec). It affords an opportunity of investigating of conditions with $q \leq 2$.

iv) Injection-HF heating ratio and electron temperature profile control can be optimized due to the combination of two methods of heating-injection and HF heating.

v) Poloidal field fast coil is inside toroidal field coils in T-15. This fast coil permits obtaining of the field up to 150 Gauss with the velocity 4×10^4 Gauss by steps of 5 Gauss. Sections of this coil assumes joining with $m = 0, 1, 2$. It results in control and use of feedback for correction of equilibrium and mode stabilization $m = 1, 2$.

vi) Gas puffing and pellet injection are expected to be used for experiments in plasma density profile control.

(3) Supporting programme of investigations on smaller Tokamaks

Investigations will be carried out in Tokamaks of smaller dimensions before the start-up of T-15 and, particularly, during the operation of T-15.

Experiments in plasma heating with neutral atom injection and in maximal value β will be made on T-11.

HF heating will be investigated on T-10 (electron and ion cyclotron resonance) and T-M-4 (electron cyclotron resonance).

Experiments in divert programme will be carried out on TO-2, T-12 and T-13 (T-12 modification).

Experiment with the chamber, made of graphite will be carried out on TMG (TM-3 modification).

The information, obtained on the installation of the stellarator type and related to Tokamaks, will be taken into consideration.

3.2 Aspects of Large Tokamak Programmes (H. Kishimoto, JT-60)

(1) Main Objectives

The four LTX projects have common objectives of producing reactor-grade plasmas and investigating their physical and technological aspects relevant to fusion reactor development. Main subjects are energy and particle confinement study, start-up and shut-down scenario, equilibrium and stability control at high beta, high power auxiliary heating, and impurity control relating to plasma-wall interaction. In addition to them, the respective projects have their special features which can be complementary to each other in fusion research progress. TFTR aims at performing the high-yield D-T burning and alpha particle physics. JET, provided with large-size dee-shape configuration, is to demonstrate long-burning D-T plasmas and to pursue the alpha particle study. JET is scheduling a high-power wave heating, either LHRF or ICRF heating scheme. T-15 is characterized by the employment of superconducting toroidal field coils and the combination heating of neutral beam injection and ECR wave. Main features of JT-60 are the impurity control with magnetic limiter, the high-power long-pulse heating operation and the flexibility of wave heating.

(2) Up-Grade Programmes

Every project has its successive up-grade programmes associated with improved performance. Typically they are the heating power increase, plasma current increase, discharge and heating pulse extension, and transfer of working gas from hydrogen or deuterium to deuterium-tritium mixtures.

Major parameters of the four LTX devices are summarized in table 3, including the up-grade characteristics. TFTR has a flexibility modification programme in which the plasma current and the beam injection power will be increased with the extension of pulse length. JET is also to increase the heating power and the heating pulse duration in both the beam injection and wave heating. Full D-T operations in TFTR and JET are scheduled in the final stage of experiment. T-15 intends to strengthen in the second phase the toroidal magnetic field produced by the SC coils in accordance with the increase of the plasma current and the heating power. Heating power increase is also considered in JT-60.

(3) Data Base Assessment

A wide range of research and development is required for approaching a Tokamak fusion reactor. The LTX programmes have essential roles to provide sufficient data bases for the future ETR and to fill the physical and technological gaps between the present-day Tokamaks and the ETR concept such as INTOR. The feasibility of the respective LTX project is evaluated as shown in table 4 in regard to the R & D items for INTOR.

Confinement study of reactor-grade plasmas will be carried out extensively in every LTX. It is expected that fairly conclusive information for both D-T and non-D-T plasmas is provided on how to achieve reactor-relevant plasma parameters.

The control of the level, species, and distribution of impurities is the basic problems in Tokamak fusion research. Surface plasma cooling as well as the employment of suitable first wall material is considered as the impurity control method in the LTX. Detailed investigations will be made for the plasma-wall interaction. It is recognized. However, that provision of some active impurity control technique is unavoidable in the ETR. JT-60 is the only LTX facility which features the divertor (magnetic limiter) and is expected to provide appropriate information on impurity control with and without divertor.

Neutral beam injection with high-energy and high-power beams is the major auxiliary heating method in the LTX to attain the reactor-grade plasmas. Beam technology and beam heating physics will be developed highly in the LTX programs. Wave heating should be important because of its potential for technological advantages in a future Tokamak reactor. First priority is placed on ECRH in T-15 and LHRF in JT-60. Other wave heating schemes such as ICRF and ECRH are also considered in JT-60. JET is to employ either ICRF or LHRF. Sufficient data bases on wave heating might be obtained in these experiments.

Long pulse operation assisted by powerful heating is an important feature in the LTX, ETR, and fusion reactor. JT-60 can be operated with hydrogen for ten seconds and will reveal a lot of problems on long pulse operation; particle balance, impurity accumulation, long duration plasma control, and heat deposition on the first wall, etc.. JET is to demonstrate D-T burning for ten-twenty seconds in the later stage of the experimental program. The typical feature of long burning discharge will appear initially in JET.

D-T burning is planned in TFTR and JET. TFTR aims to achieve a high-Q plasma operation in the up-grade programme. In addition to the tritium handling technology, neutronics, and remote maintenance capability, alpha physics including alpha heating, alpha particle confinement, and alpha particle driven instabilities will be studied in these devices.

T-15 employs superconducting magnets for the toroidal field coils. The basic SC technology relevant to Tokamak toroidal coil system would be established in T-15. The electromagnetic influence of plasma behavior on the SC toroidal coils will become apparent. Nevertheless the superconducting technology for poloidal coils might not be achieved in the present LTX programs.

From a view point of future ETR programme such as INTOR, further efforts in the LTX are required concerning the following subjects; disruption control, species control, first wall material development, impurity control and ash-exhaust possibility with and without divertor, burn temperature control, current profile control, ripple effect study on plasma transport and suprathermal particle confinement, and the associated technological developments.

(4) Experimental Schedules

According to the present schedule shown in fig.4, TFTR will initiate the discharge operation at the end of 1981, earliest among the four projects, and will enter the full D-T burning experiment on 1984. JET, starting the initial discharge in the beginning of 1983, undergoes successive increase of heating power and will operate the full D-T discharge on 1987. T-15 is intended to initiate the Joule discharge on 1984 and to perform the full power heating experiment from 1987. JT-60 is expected to start the initial discharge on 1984 and to enter the high-power long-pulse experiment on 1985. Further increase of heating power is also considered in JT-60.

The schedule of the large Tokamak devices seems to be critical for contributing the ETR design. Nevertheless, the experimental results coming from the LTX should be essential to confirm the ETR performances.

4. COLLABORATIVE PROGRAMMES (R. Little, TFTR)

The four large Tokamaks JT-60, JET, T-15 and TFTR, have some major common objectives. These can be summarized as production of reactor-grade plasmas, significant auxiliary heating, study of impurity control techniques and plasma/wall interaction, and solution of similar engineering problems.

On the other hand, these Tokamaks have significant complementary features. TFTR will concentrate on D-T operation and has the capability of adiabatic compression. JET has a large D-shape plasma, and D-T operation near ignition. JT-60 has a magnetic limiter and more possibilities for impurity control, while T-15 will have neutral beam and RF heating systems, superconducting coils and good access to the first wall of the vacuum vessel.

These programmatic correlations are not accidental. They are due to evolution caused by exchange of programmatic information.

The majority of discussion at the meeting was focussed on the needs for the next large machine, such as INTOR. There are many areas of concern, but the major ones which need early understanding appeared to be experimental impurity control and plasma profile control information.

Doubts were expressed on the practicality of shared or joint development of equipment in any formal general arrangement without significant changes in the existing bureaucratic procedures.

The following three recommendations were made:

i) The INTOR group has identified areas of R & D concern and is in the process of providing desired schedules for resolution. It would be very useful for the LTX to consider these concerns, and try to correlate their programmes with them.

ii) The LTX groups consider exchange of information and personnel very important, and strongly encourage it.

iii) There is potential for much more collaboration in development of diagnostic techniques (and possibly some identical equipment). Implementation should be performed directly between interested laboratories.

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Table 1 Phases in the operation of JET

	Description	Object	Possible Date	Pα/Ploss
Phase I	Initial Operation and Commissioning with ohmic heating only. Mainly H_2 operation, possibly He and D_2.	1) Establish effective cleaning techniques. 2) Evaluate limiter performance 3) Control of D shaped plasmas 4) Establish limits on I and n 5) Establish fuelling methods 6) Compare performance(T,Te)with transport codes 7) Examine discharge termination methods	1983	
Phase II	Operation with Additional Heating 10 MW H at ~ 80KeV plus possibly a prototype r.f.system for evaluation. Gases mainly H_2 possibly He and D_2 or H_2/D_2 mixtures.	1) Develop effective impurity control techniques 2) Evaluate and optimise heating performance 3) Establish limits on n with Additional Heating 4) Compare performance(T,Te)with transport codes 5) Refine definition of heating power for future phases.	1984	
Phase III	Operation with extended performance Additional Heating: 25MW, either neutral injection of 160 keV D or a combination of r.f. and neutral injection.	1) Establish limits on n,T and Te while limiting the total numbers of neutrons produced 2) Compare behaviour with transport codes 3) Establish whether central region ignition is to be expected in D-T. Decide whether D-T operation is justified.	1985	equivalent conditions to those giving 0.2→1 in D-T
Phase IV	Operation with significant α particle heating in deuterium-tritium gas mixtures.	1) Study approach to ignition 2) Study plasmas with profiles determined by P_α (r) 3) Examination of β limit violation by α -heating.	1986	Initially 0.2 extending later to 1

Table 2 The main investigations on T-15

1. Equilibrium:
   - ellipticity position and control
   - start equilibrium
2. Stability:
   - stability region
   - MHD oscillations and their connection with transport
   - study of possibilities of disruption stabilization by feedback control
   - energy output time in disruption, disruption protection
   - internal disruptive instability - its influence on losses
   - instabilities in heating
   - smallscale fluctuation and their connection with transports
3. Transport phenomenon:
   - heat conductivity of electrons and ions
   - plasma diffusion
   - impurities diffusion: mixing of isotopes, helium diffusion
4. Heating:
   - injection
   - HF heating
   - Optical combination of heating
   - Influence on profiles, profiles control
5. Discharge condition (start-up, stationary current maintenance, shutdown)
   - break down, current rise, temperature increase
   - optimization, control at plateau
   - shutdown: conditions of shutdown without disruption, search of shutdown optimization
6. Plasma-wall interaction
   - physics of outer part of plasma column and transport phenomenon experiments in this part
   - particles of material transport along the chamber, renewable coating
   - physics of surface layers (dust, adhesion characteristics) their change during the long cycle of the work
   - corrosion and erosion of the wall, particularly, in long-continued heating
   - behaviour of different materials while interacting with T-15 plasma
   - active influence on boundary plasma-cooling of periphery, reradiation condition
   - pumping of helium and erosion products from the wall during the pulse without divertor
7. Engineering-physical investigations
   - testing of all installation systems in complex-electromagnetic, cryogenic, vacuum, power supply and automatization systems
   - testing of elements and assemblies of electromagnetic systems, power supply, auxiliary heating system
   - testing and study of superconducting magnetic system characteristics, particularly, study of influence of plasma current disruption
   - accumulation of experience in the operation with large system for extrapolation to reactor conditions

Table 3 Main parameters of large Tokamaks

ITEMS \ DEVICE	TFTR	JET	T - 15	JT - 60
COUNTRY	USA	EURATOM	USSR	JAPAN
EXPERIMENT INITIATION	DEC . 1981	JAN. 1983	1984	NOV. 1984
MAJOR RADIUS (m)	2.65	2.96	2.4	3.03
MINOR RADIUS (m)	0.85	1.25 × 2.1	0.7	0.95
CROSS-SECTIONAL SHAPE	CIRCULAR	CIRCULAR/DEE	CIRCULAR	CIRCULAR
TOROIDAL FIELD (T)	5.2	3.5	3.5(PHASE I) → 4.5(PHASE II)	4.5
TF COIL CONDUCTOR	COPPER	COPPER	Nb_3Sn	COPPER
PLASMA CURRENT (MA)	2.5(PHASE I) → 3.0(PHASE II)	4.8	1.4(PHASE I) → 2.0(PHASE II)	2.7
DISCHARGE DURATION(sec)	1.0(PHASE I) → 3.0(PHASE II)	10 ~ 20	5.0	5 ~ 10
WORKING GAS	H/D/D-T MIXTURE	H / D-T MIXTURE	H + SMALL AMOUNT OF D	H + SMALL AMOUNT OF D
NBI POWER (MW)	25(PHASE I) → 32(PHASE II)	10(PHASE I) → 35(PHASE II)	6(PHASE I) → 9(PHASE II)	20(PHASE I) → 30(PHASE II)
DURATION (sec)	0.5(PHASE I) → 1.5(PHASE II)	1.5(PHASE I) → 10(PHASE II)	1.5	10
ENERGY (keV)	120 , D	80,H°(PHASE I) →160,D°(PHASE II)	40 / 80 , H°	75 ~ 100 , H°
RF HEATING SCHEME	(ICRF)	ICRF OR LHRF	ECRH	LHRF,(ICRF,ECRH)
POWER (MW)	——	A FEW(PHASE I) → 10(PHASE II)	4(PHASE I) → 6(PHASE II)	10
$\bar{T}_i$ (keV)	5 ~ 10	5 ~ 10	5 ~ 10	5 ~ 10
$\bar{n}_e$ (cm^{-3})	10^{14}	$1 \sim 2 \times 10^{14}$	$0.5 \sim 1 \times 10^{14}$	10^{14}
$\bar{n}_e \tau_E$ (sec/cm^{-3})	$0.1 \sim 1 \times 10^{14}$	2×10^{14}	$0.3 \sim 0.5 \times 10^{14}$	$0.5 \sim 1 \times 10^{14}$
$\bar{\beta}$ (%)	2.5(PHASE I) → 3.5(PHASE II)	5 ~ 8	1.6(PHASE I) → 3.7(PHASE II)	4.0

Table 4 Feasibility of large Tokamaks to R & D items for INTOR

R & D ITEMS REQUIRED BY INTOR		TFTR	JET	T - 15	JT - 60
ENERGY AND PARTICLE CONFINEMENT		○	○	○	○
BETA LIMIT		○ (3.5%)	○ (5~8%)	○ (3.7%)	○ (4%)
SHAPE AND PROFILE CONTROL		○ (CIRCULAR)	○ (D-SHAPE)	○ (CIRCULAR)	○ (CIRCULAR)
DISRUPTION CONTROL		Δ	Δ	Δ	Δ
IMPURITY CONTROL DIVERTOR PHYSICS AND TECHNOLOGY		Δ (SURFACE COOLING, LOW-Z WALL)	Δ (SURFACE COOLING, LOW-Z WALL)	Δ (SURFACE COOLING, LOW-Z WALL)	○ (DIVERTOR, SURFACE COOLING, LOW-Z WALL)
LONG PULSE CONTROL WITH HEATING		Δ (0.5→1.5s)	○ (1.5→10s)	Δ (1.5s)	○ (10s)
HEATING	NBI	○ (120keV D°)	○ 80keV H°/D° → 160keV D°	○ (80keV H°)	○ (75-100keV H°)
	RF	Δ (ICRF)	○ (ICRF OR LHRF)	○ (ECRH)	○ (LHRH, ICRF, ECRH)
FUELING		○ (GAS PUFF, PELLET)	○ (GAS PUFF)	○ (GAS PUFF)	○ (GAS PUFF, PELLET)
START-UP AND SHUT-DOWN		○	○	○	○
ALPHA PHYSICS		○	○	Δ	Δ
BURN CONTROL		×	×	×	×
SC COIL	TOROIDAL	×	×	○ (Nb_3Sn)	×
	POLOIDAL	×	×	×	×
FIRST WALL MATERIAL		○	○	○	○
TRITIUM HANDLING REMOTE ASSEMBLY AND MAINTENANCE		○	○	×	×

○ : ADEQUATE Δ : PARTIALLY ADEQUATE × : INADEQUATE

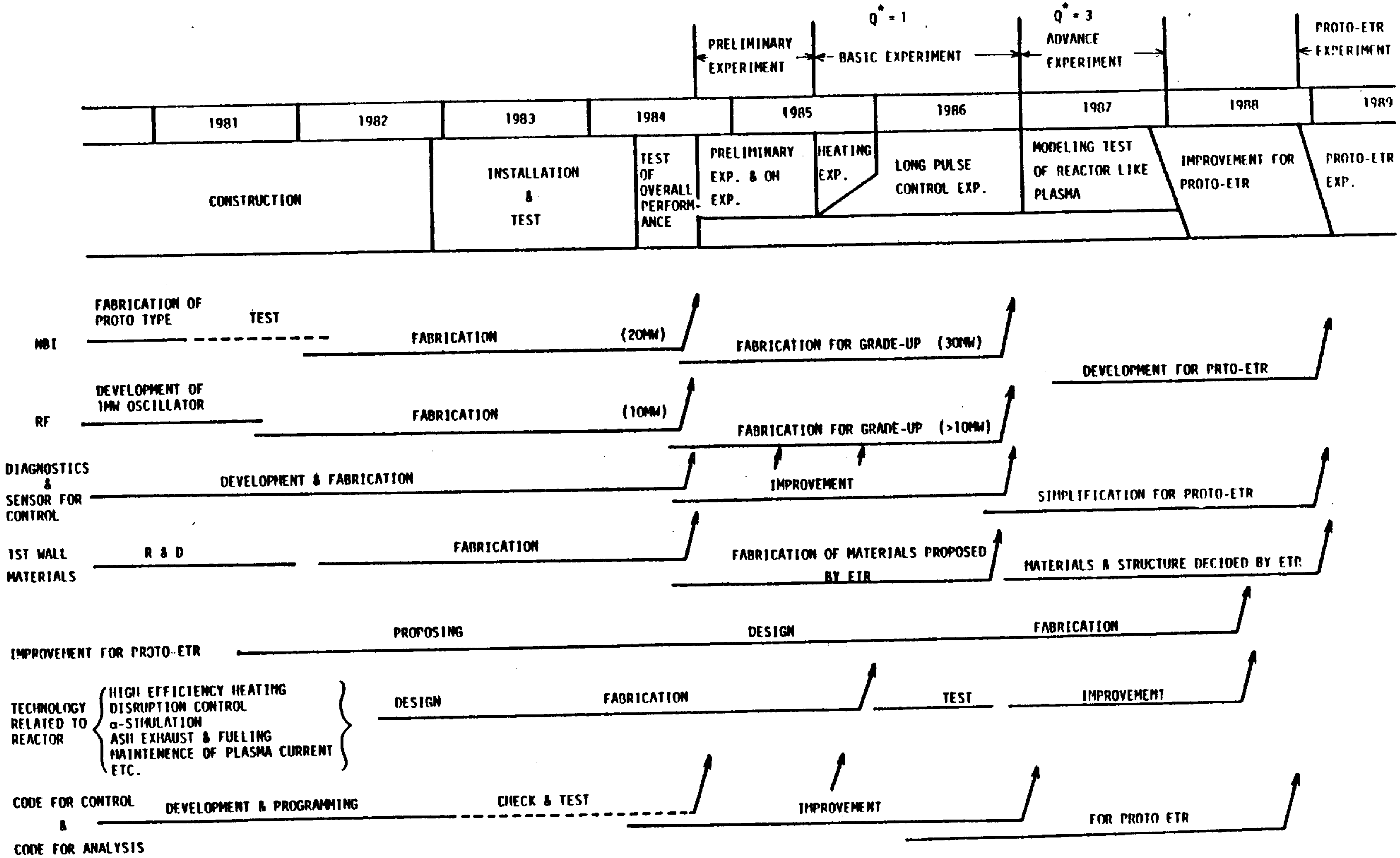

Fig.1 Experimental programmes and schedule of JT-60

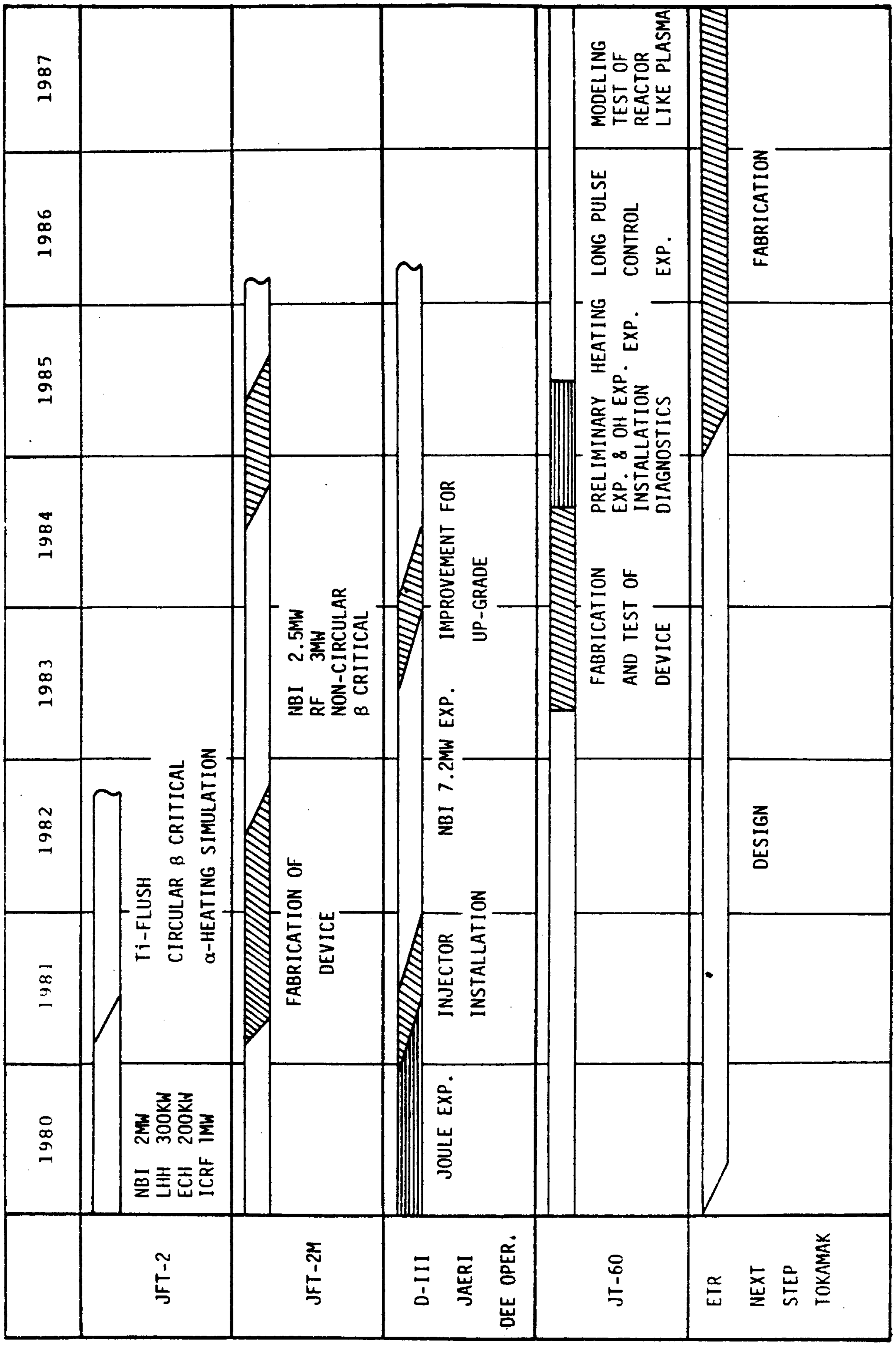

Fig.2 Related experimental programmes of JT-60

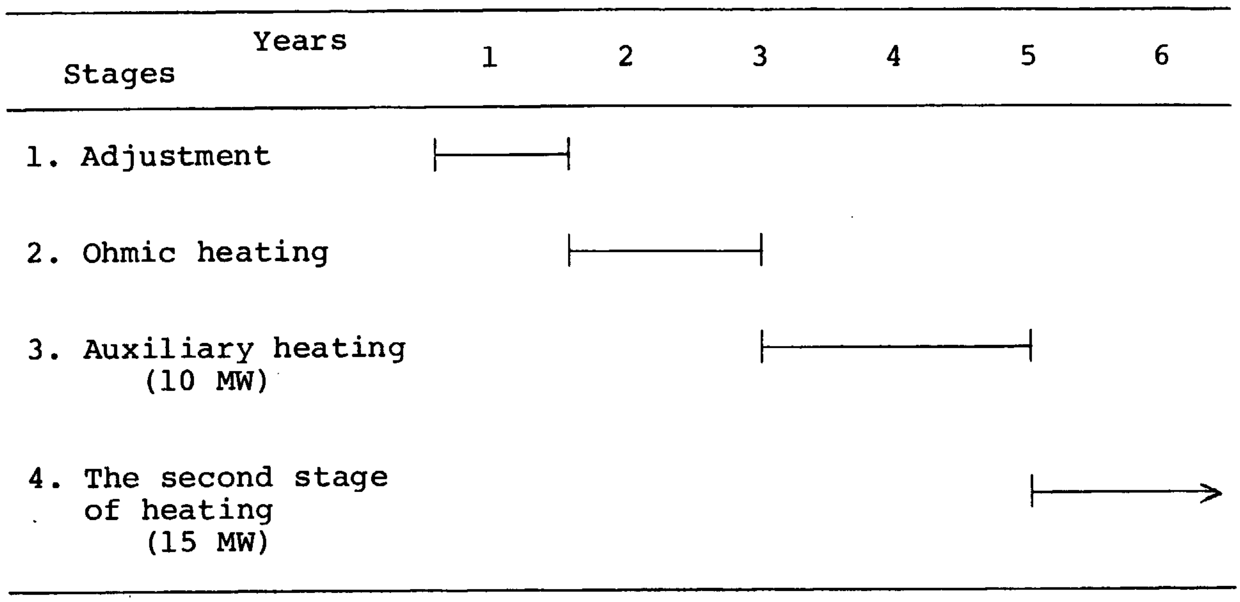

Fig.3 Approximate schedule of T-15 operation after the physical start-up

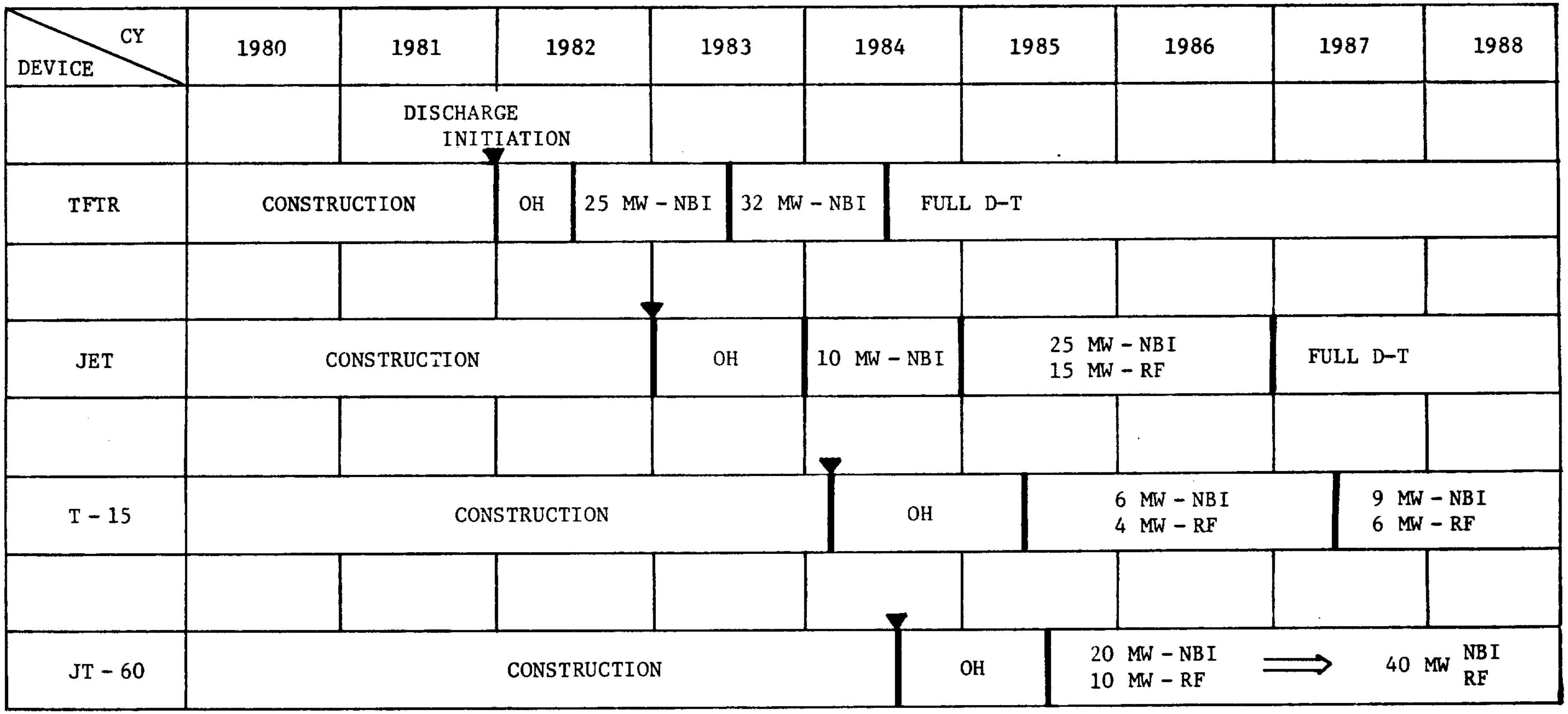

Fig.4 Schedule of large Tokamak experiments