GA-A14614
VOLUME I
UC-20d

GAC – ANL
TNS SCOPING STUDIES

STATUS REPORT FOR FY-77
OCTOBER 1, 1976 – SEPTEMBER 30, 1977

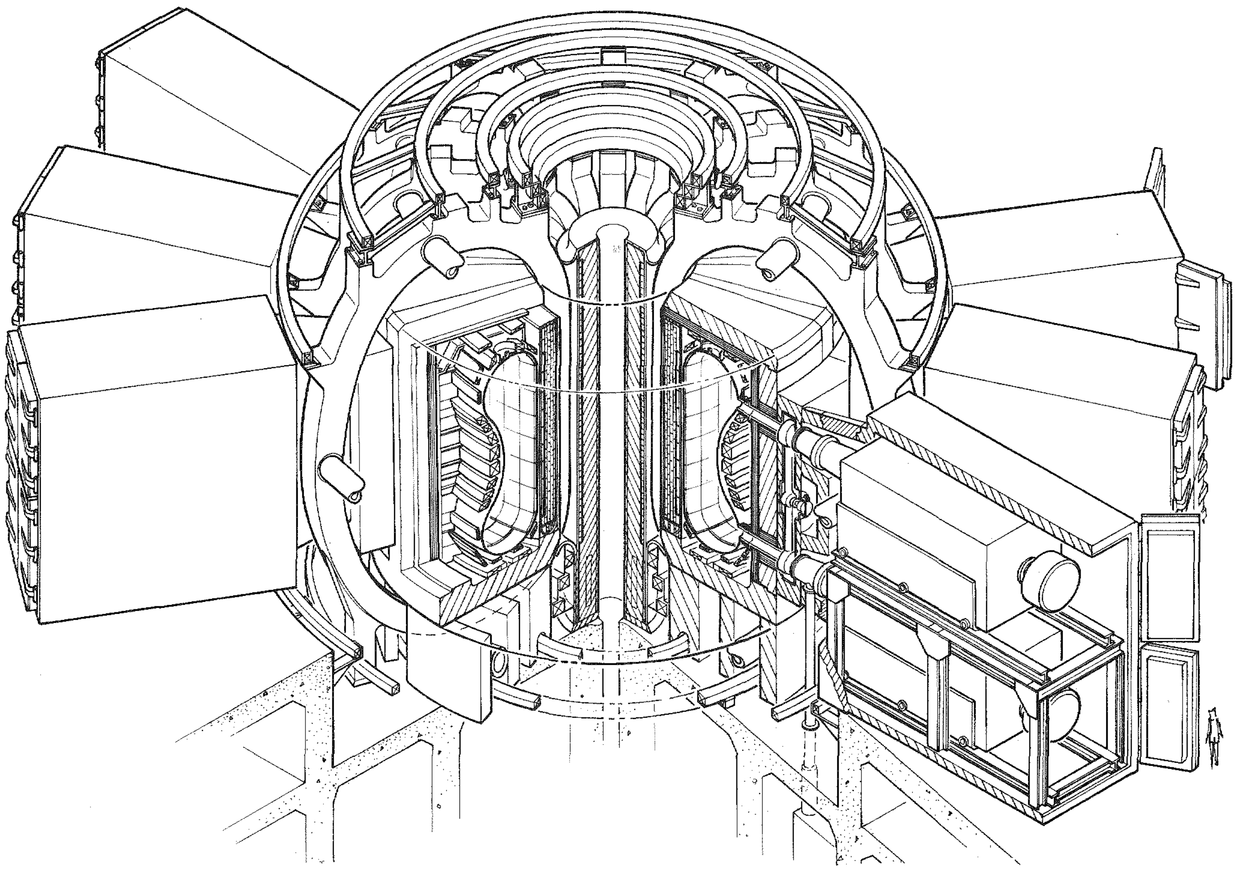
VOLUME I
SUMMARY

by
PROJECT STAFF

Prepared under
Contract EY-76-C-03-0167
Project Agreement No. 38
for the San Francisco Operations Office
Department of Energy

GENERAL ATOMIC PROJECT 3235
DATE PUBLISHED: JANUARY 1978

GENERAL ATOMIC COMPANY



ABSTRACT

This volume, the first of a total of eleven, summarizes the major results of a one-year effort in a continuing study of the TNS (The Next Step) program being funded by the Department of Energy. In this study, five candidate reactors were identified, ranging from a copper toroidal field coil ignition test reactor (ITR) with a 10 second burn and a 0.1 duty factor to a superconducting TF coil ignition test reactor which can be upgraded to an experimental power reactor. Study emphasis was placed on a 3.8 m ITR with superconducting TF coils designed for a 30 second burn and a duty factor of 0.1. The results indicate that such a reactor is technically feasible and that it can be built in about 6-1/2 years after completion of a 3-year program of conceptual design and initial preliminary design. The cost of this particular facility is approximately $475 million (1977 dollars) without escalation and R&D costs.

The copper TF-coil ITR can be somewhat smaller (3 m major radius), but would cost slightly more if a 30 second burn is required. If the burn can be reduced to, say, 10 seconds, the cost would be slightly less.

Both the helium cooled and the water cooled upgradable ignition test reactors would cost substantially more in their final configurations than the reference ITR, but overall program costs could be drastically reduced by eliminating the need for a separate experimental power reactor. A substantial savings in overall program schedule would also be achieved by eliminating the EPR.

Alternatively, it may be possible to eliminate the EPR by building a prototype experimental power reactor (PEPR), which would have the features of an EPR but would provide for heat rejection rather than power generation. This machine costs only moderately more (approximately 35 percent) than the reference ITR.

FOREWORD

This is Volume I of an eleven-volume report constituting the TNS Scoping Studies performed during the 1977 fiscal year. The complete volume list is as follows:

CONTENTS

FIGURES

TABLE

1.1. INTRODUCTION

In 1974, a two-year study of a fusion experimental power reactor (EPR) was initiated at General Atomic. This study, which was funded by the Electric Power Research Institute, resulted in the design definition of a tokamak power reactor which produced a nominal output of net power (Ref. 1.1-1). The results of the study also indicated that an EPR would involve a substantial commitment (approximately $700 million plus escalation and R&D), and that in order to have the reactor operational in the mid-1980s - the target date for the study - a decision to proceed would have to be made immediately. It was concluded that the risks were too great to embark on such an ambitious program without further experimental information and an examination of alternatives.

Accordingly, in 1976 the TNS (The Next Step) program was started at General Atomic under the sponsorship of the Energy Research and Development Administration. During the first phase of this program, objectives were examined, design concepts were explored, and initial parameters were established for two candidate TNS reactors. The two candidates identified were a 3.8 m ignition test reactor (ITR) and a 4.2 m upgradable ignition reactor (UITR), the latter incorporating a 25 cm stainless steel blanket so that a power conversion system could be added after an initial experimental program (Ref. 1.1-2).

The second phase of this program, presented in this 11-volume report, involved a conceptual design and an examination of the physics considerations of the two basic candidate reactors. In addition, three other candidates were identified and briefly examined. These were a water-cooled UITR, a copper coil ITR, and a prototype experimental power reactor.

The TNS study was carried out as a joint effort by General Atomic Company and Argonne National Laboratory, supported by subcontractors for specialized studies. General Atomic carried out program planning studies and generated overall reactor designs (Vol. II), and, in addition, performed studies in reactor physics (Vol. III) and reactor engineering (Vol. IV). Argonne National Laboratory's efforts were directed to the study of neutral beam and radio frequency heating schemes, vacuum pumping and tritium systems, a superconducting ohmic heating coil, power supplies, and shield penetrations (Vol. V). McDonnell-Douglas performed a conceptual design study of the plasma chamber (Vol. VI). Battelle Columbus carried out maintainability studies (Vol. VIII), while Aerojet Manufacturing Company developed conceptual designs for the equipment required for the remote maintenance system (Vol. VII). Nuclear Services Corporation made an initial safety assessment, examined regulatory considerations, and established initial design criteria and site selection criteria (Vol. IX). The Ralph M. Parsons Company carried out a facility design study, established a cost estimate for the BOP, and examined the program planning and quality assurance aspects for the TNS program. Under separate funding, Los Alamos Scientific Laboratory studied the ohmic heating (E-coil) power supply (Vol. XI).

The results of these studies form a comprehensive basis for evaluating the various alternative concepts available for the TNS program. Technical feasibility of the various concepts is indicated and area costs which are valid for comparison purposes are established. The alternative concepts must yet be evaluated in terms of program objectives, alternative development scenarios, and cost and schedule implications.

REFERENCES

1.1-1. "Experimental Fusion Power Reactor Conceptual Design Study," General Atomic Report GA-A14000, July 1976.

1.1-2. "TNS Scoping Studies - Interim Status Report," General Atomic Report GA-A14412, May 1977.

1.2. REACTOR CONCEPTS

Five reactor configurations with varying degrees of sophistication with regard to the incorporation of reactor-like technologies were examined. The five reactor candidates identified are:

Ignition Test Reactor (ITR)
Upgradable Ignition Test Reactor (UITR)
UITR - Pressurized Water Cooled (UITR W-C)
ITR - Copper TF Coil (CC-ITR)
Prototype Experimental Power Reactor (PEPR)

The major features, design parameters, and comparative costs of these reactors are summarized in Table 1.2-1, and a comparison of relative device sizes is shown in Fig. 1.2-1. Of the five devices studied, the Ignition Test Reactor (ITR) and the Upgradable Ignition Test Reactor (UITR) were studied and designed in greatest detail. The other devices were considered for the purpose of an evaluation of cost and schedule implications of the various approaches relative to an assessment of the program objectives and related risks. All devices have the doublet plasma cross section and are designed for D-T burning.

The ITR device was designed to function primarily as a physics test device without particular attention to technology development. Certain advanced or reactor-like technologies are included, but only when they are the most straightforward and cost effective solutions to the problems at hand. For example, the 3.8 m ITR device utilizes superconducting TF-coils as a consequence of an anticipated extended burn and the associated power handling requirements. The copper TF-coil version of the ITR was sized to determine the minimum cost reactor which might be realized using a scaled-up Doublet III approach.

TABLE 1.2-1
CHARACTERISTICS
CANDIDATE REACTORS

	ITR	UITR	UITR Water-Cooled	Copper Coil ITR	PEPR
Features					
Plasma configuration	Doublet	Doublet	Doublet	Doublet	Doublet
Plasma chamber	Inconel 625 Continuous Shell	Inconel 625 Continuous Shell	Inconel 625 Continuous Shell	Inconel 625 Continuous Shell	Inconel 625 Continuous Shell
Field-shaping coils	Copper	Copper	Copper	Copper	Copper
Toroidal field coil	NbTi	NbTi	NbTi	Copper	NbTi
Induction coil	NbTi	NbTi	NbTi	Copper	NbTi
Blanket	None	25 cm 316 SS	25 cm 316 SS	None	25 cm 316 SS
Design Parameters					
Major radius, m	3.8	4.2	4.5	3.0	4.0
Minor radius, plasma, m	1.1	1.2	1.3	1.0	1.15
Toroidal field,					
Max. design, tesla	10.0	10.0	10.0	10.0	10.0
Plasma current, MA	11.3	11.7	12.8	13.1	11.6
Plasma total beta	0.10	0.10	0.10	0.07	0.10
		Initial/Ultimate			
Burn time, sec	20	30/90	30/90	30	90
Duty factor	0.10	0.10/0.78	0.10/0.78	0.10	0.75
Gross electric power, MW(e)	--	103	106	--	--
Plant net electric MW(e)	--	15	20	--	--
Relative Cost, $\$10^6$	475	750	820	500	650

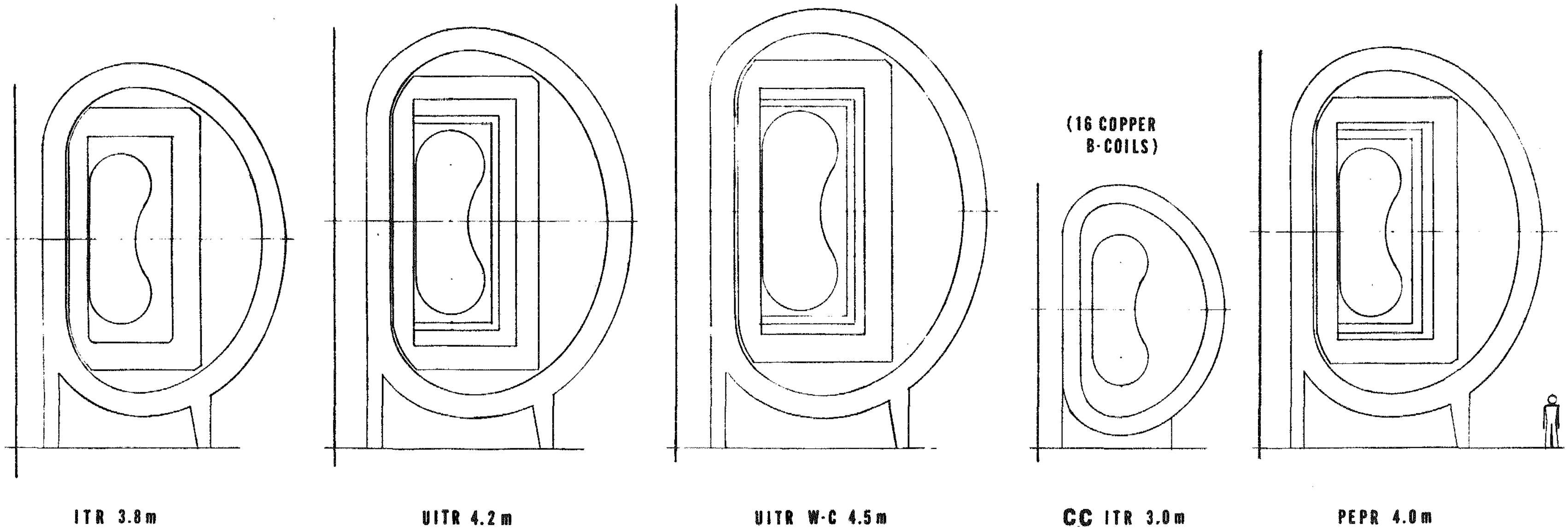


Fig. 1.2-1. Comparative designs — TNS candidate reactors

The upgradable machine operates as an ITR initially and later can be upgraded into a net power device by scaling up power supplies and refrigeration systems and adding power generating and helium coolant pumping equipment. The size is increased from a plasma major radius of 3.8 m for the ITR to 4.2 m for the UITR. The basic UITR is designed with a helium-cooled blanket which permits higher coolant temperatures and therefore results in a smaller and less costly reactor. As an alternative, a pressurized water-cooled UITR was also examined. The final design considered was a Prototype Experimental Power Reactor (PEPR), which is similar to an EPR but produces only high temperature coolant rather than electrical power.

With regard to the cost entry in Table 1.2-1, it is noted that the basic superconducting ITR and the copper TF-coil ITR appear to be the least expensive approaches for the demonstration of the physics objectives of the TNS device.

The UITR costs approximately 60% more than the basic ITR. This system, however, could eliminate the need for building a separate EPR, thereby reducing the overall program cost with an acceleration in the program schedule. To first order, the ultimate cost of the upgraded UITR should be about the same as the EPR. On that basis, and assuming that another program may include both an ITR and an EPR, the program savings would equal the cost of the ITR. After the initial ITR mode experimental program, the facility could be operational as an EPR in about one year. The UITR with a water-cooled blanket costs significantly more than the basic helium-cooled UITR. Any marginal advantages in using water cooling as compared to helium cooling do not appear to justify the higher cost penalty.

The PEPR represents an intermediate step between an ITR and an EPR. As indicated in Table 1.2-1, the cost of the PEPR is substantially higher than the basic ITR and would have to be justified on the basis of ability of the PEPR to be operated at a higher duty factor and to produce higher coolant outlet temperatures. An aggressive program involving the construction of a PEPR, however, may make it possible to eliminate the EPR and make a step directly from the PEPR to a demonstration plant. Such an approach should result in substantial overall cost savings and schedule acceleration as compared to the ITR/EPR/DEMO scenario.

1.3. REACTOR PHYSICS

The physics considerations are presented in terms of the discharge cycle, which is divided into three major phases: 1) the pre-discharge and plasma initiation phase, 2) the approach to ignition, and 3) the burn phase. These three chronological phases also involve quite distinct physics considerations. It is, in fact, the physics involved which to a large extent defines the machine operations and subsequently the design and integration of its systems and/or components.

1.3.1. PRE-DISCHARGE AND PLASMA INITIATION PHASE

The physics of the plasma initiation and early ohmic heating phases of the discharge are periods which are poorly understood theoretically and poorly diagnosed experimentally. That it is an area of considerable concern is evidenced by the fact that greater constraints are placed on starting up large, dense plasmas because the ratio of impurity radiation to ohmic power is an increasing function of both the major radius and the plasma density. Areas impacted by the dynamics of "start-up" include:

1. E-Coil Circuit Design. It is imperative to ascertain the required voltage ramp for achieving proper plasma evolution. Further, since the resistive volt seconds are expended most rapidly during the low temperature phase of the discharge, start-up considerations have an important effect on the overall volt second requirements.

2. Impurity Control. The impurity concentration is a critical parameter during the start-up phase. It is necessary to burn through the highly radiative stage, characterized by temperatures of order 20 eV, of the low Z impurities. Thus, upper bounds exist for these impurity levels. This impacts wall preparation and vacuum pumping requirements and the need for impurity control early in the discharge.

3. Chamber Filling Pressure. The optimal filling pressure is determined by considerations such as runaway electron production, impurity content, and voltage limitations of the E-coil system. Since a maximum filling pressure exists for a given E-coil design, the question of fueling to achieve high density must be addressed.

4. Preionization Requirements. The general question of preionization must also be considered in the context of the dynamics of the breakdown phase. If preionization is desired, the power and energy requirements need to be determined.

5. Runaway Electron Production. Initiation of the discharge requires high applied one-turn voltages, which can produce copious quantities of high energy runaway electrons. These electrons can be deleterious to the mechanical integrity of the device in addition to degrading the plasma properties. The runaway population should be minimized by proper choice of filling pressure, voltage ramp, and preionization.

6. Auxiliary Heating. If it is impossible, or unacceptable, to ohmically heat through the radiation barrier, then it may be necessary to consider the need for auxiliary heating early in the discharge. This poses special problems because the plasma at start-up is in a totally different parameter regime than that usually considered during an auxiliary heating stage.

7. Design and Assembly Tolerances. Start-up dynamics are greatly affected by the non-toroidal components of the magnetic field. Such fields are produced by misalignment of coils and eddy currents produced by the E-coil flux swing. Specification of the maximum allowable "error fields" affects placement of components and implies tolerances that must be maintained in the assembly of the device.

In order to address these difficult questions quantitatively, a code was developed which deals with the atomic and molecular physics of hydrogen together with a thorough treatment of the atomic physics of the impurities.

An example of the results of exercising this code is shown in Fig. 1.3-1, which displays the plasma temperature history for a typical case. Two important features of start-up emerge: the charge exchange plateau and the impurity radiation plateau. Early in the course of the discharge, the H_2 molecules dissociate, leaving a large number of atomic hydrogen atoms and, because the electrons are still cold, these neutrals can penetrate the plasma. Thus charge exchange events can result in a large energy sink. Either the density must be low enough or electric field seen by the plasma must be high enough to assure that the discharge continues to evolve. After passing through this phase, the plasma then heats until impurity radiation becomes significant. The 2% oxygen impurity in this sample case was enough to balance the ohmic power when the temperature reaches 20 eV. At this point in time, the impurities are far from coronal equilibrium and the resulting radiative losses are characteristically larger than the coronal equilibrium values. A quasi-steady-state situation persists until such time as the ohmic power again exceeds the losses and the temperature again rises. A typical temperature history is exhibited in Fig. 1.3-1.

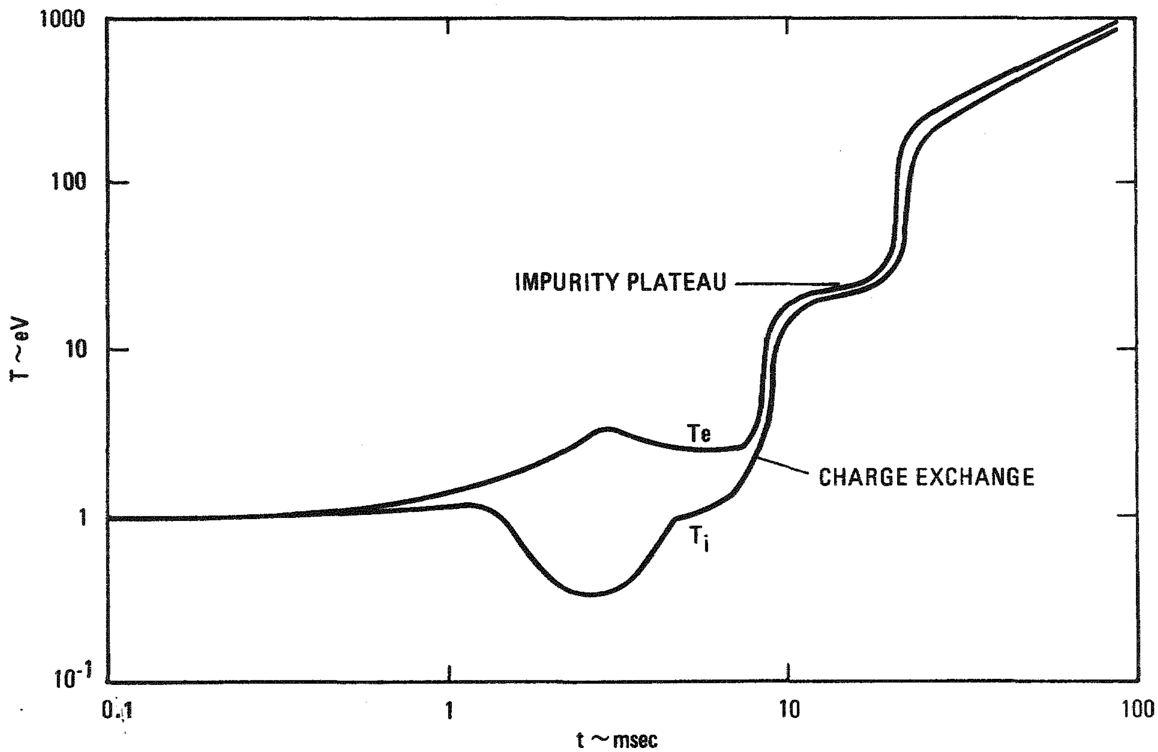


Fig. 1.3-1. Temperature history of the discharge

To translate these start-up considerations into constraints on the systems and components employed, consider the following sequence of events, taken to represent typical start-up scenario:

1. Back bias the E-coil
2. Inject initial gas loading (D + T)
3. Switch E-coil to initiate start-up flux swing
4. Preionization
5. F-coil excitation for plasma breakdown.

The timing sequence of the E-coil swing and F-coil excitation is such that the voltage required to sustain the plasma current after initial breakdown is just that corresponding to the phase of the E-coil swing. From this time on, the E-coil supports the plasma current and the approach to the ignition phase is begun.

Start-up operations involve the E- and F-coil systems directly and the vacuum vessel as a consequence of flux penetration requirements and plasma electron runaway considerations.

During this phase of operation, the E- and F-coils must act independent of each other, yet they are magnetically coupled because of their coaxial arrangement. In order that they may be decoupled, the circuits are connected in parallel and the number of turns of each coil is made equal. This is the incentive for the series-parallel winding configuration developed for the E-coil. If a single series wound E-coil solenoid were used, each of the many F-coils would be required to have a large number of turns and consequently a large number of current joints. The motivation for making the E-coil a superconductor stems from power consumption considerations and the inherent reduction of leakage flux within the TF-coil envelope. Since all E-coils see the same voltage and have a common number of turns, they all experience the same interim flux variation and, being superconductors, adjust their respective currents to exclude the leakage flux.

The design criteria for the E-coil which have evolved subject to the above arguments and sizing considerations are as follows:

- Flux change of 37 volt-sec
- Stray fields < 10 G in the plasma
- Field at the TF coil < 500 G
- Outside diameter = 2.0 m
- $j < 300\ A/cm^2$
- Series turns/parallel section = 20

The coil system, designed by Argonne National Laboratory (Vol. V), is comprised of a solenoid in the poloidal bore formed by the toroidal field coils, and upper, lower, and outboard coils arranged so as to conform to the stray field criteria in the plasma region. NbTi superconductor, cryogenically stabilized by bath cooling, is utilized for all portions of the E-coil except for the outermost coils which are normal conducting and have their own power supplies.

The primary function of the F-coil system during the start-up phase is plasma breakdown. The required one-turn preionized plasma breakdown voltage is determined by the initial gas (D + T) loading. Reduced fill pressures, on the order of $5 \times 10^{19}/m^3$, are prescribed to reduce the required E-field and consequently to reduce the electron runaway population. Low density start-ups will then require density increases to reach ignition conditions.

The estimated F-coil voltage pulse is on the order of 8 kV for a duration of 20 - 30 msec. These conditions then define the required turn-to-turn insulation and ground standoff spacing. Because of the coils proximity to the vacuum vessel and resulting high neutron fluence, copper is used in their construction with the number of turns in each coil being equal to that of the E-coil segment. The inner coils are bath cooled in high purity deionized water which also serves as the insulation medium to ground. The breakdown voltage for 1 megohm-cm water is estimated to be 10 kV/cm which allows a ground standoff of 2 cm with margin for radiation effects. The turn-to-turn button insulators for both the inboard and outboard coils are made of alumina which should be adequate for fluences of 10^{21} - 10^{22} neutrons/cm^2.

Because the F-coils are positioned outside the vacuum tank, certain constraints relative to the vessel toroidal resistance are implied. The requirements for vessel toroidal resistance depends on the volt-second capacity of the OH coil, the allowable error fields during start-up, and the desired response time of the plasma/F-coil circuit. Because the strength of a continuous vessel increases with decreasing resistance, it is desirable for structural reasons to estimate the minimum possible vessel toroidal resistance. This was taken to be that at which the plasma resistance at breakdown, about 5 eV, is no more than twice that of the vessel. Although this induces several MA in the vessel, it is shortlived and therefore the temperature rise is only a few degrees. This vessel resistance criterion is also dependent upon the allowable start-up error fields. Once the average plasma temperature reaches about 50 eV, the vessel behaves nearly as an insulator. Subject to these stipulations and anticipated loading due to a disruptive condition, the vessel wall is fabricated from two Inconel 625 metal skins. The skin thicknesses are controlled by plasma disruptive loads and static buckling considerations.

1.3.2. APPROACH TO IGNITION

From the physics standpoint, the primary objective of TNS is to achieve a state of ignition with burn times of sufficient duration to examine burn dynamics and control in a mode appropriate to high performance fusion reactor operation. The ignition requirements affect all aspects of the design and operation of TNS. The operational sequence describing the approach to ignition might be as follows:

- Continue E-coil flux swing to maximum plasma current
- Continually adjust F-coil current to perform plasma shaping and provide for vertical and horizontal stability

- Initiate auxiliary heating
- Build up the plasma density to achieve the confinement needed for ignition

This phase of operation is considered complete when the auxiliary heating is turned off in response to diagnostic tests for ignition.

The physics constraints on the ignition point are most conveniently displayed on a density-temperature (n-T) plot such as Fig. 1.3-2 in which the ignition region is typically bounded by three curves:

1. An ignition curve defining the lower boundary of the region in which the fusion α-power exceeds the power losses of the plasma
2. A β-limit β_c above which the plasma is magnetohydrodynamically unstable
3. A temperature limit T_L beyond which the auxiliary power P_A is unable to heat the plasma.

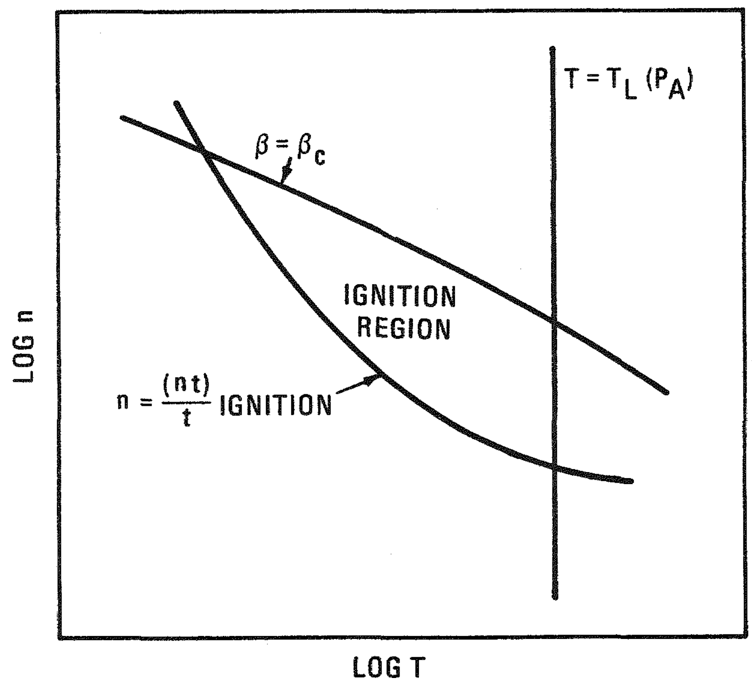


Fig. 1.3-2. Typical ignition region in n-T space

The existence of a non-void ignition window is then reflected in conditions on the plasma size (through the required confinement) and the auxiliary heating power (through the temperature limit) as well as the plasma shape and the toroidal field (though the β-limit). Ignition margin can be measured in terms of the size of the ignition region; increases in plasma size, auxiliary heating power, β_c, and toroidal field all serve to increase the margin for reaching ignition.

Operationally, the plasma size and heating requirements are determined by demanding that ignition be achievable with $\beta < 5\%$ using a magnet that produces a 10 T maximum field under both of two distinct transport models: Alcator scaling and 0.1 trapped ion mode scaling. The choice of 5% as the ignition β-limit is made to allow for the thermal excursion to the higher β values associated with the burn equilibrium point. The choice of 10 T is made on the basis of technological limitations for NbTi coils.

As a model for transport, we have assumed Alcator like scaling normalized to the results from the Doublet IIA device at General Atomic Company. Doublet IIA is a very flexible device with the capability of producing plasma discharges with circular, elliptic and doublet plasma cross sections. All three cross sections have been studied with respect to energy confinement time. The results of these experimental studies are shown in Fig. 1.3-3. Normalized to the $n\tau_E$ value for a circular discharge, the doublet cross section demonstrates an $n\tau$ value nearly eight times higher and the ellipse almost three times higher than the circular plasma cross section (Ref. 1.3-1). When parametrized and translated into sizing conditions on a $B_{max} = 10$ T ignition device, one finds that $a \approx 1.1$ m provides an acceptable ignition margin.

The auxiliary power required to sustain ignition temperatures in hydrogen operation in a transport-dominated regime is given by

$$P = \frac{3nT}{\tau_E} V \propto TR$$

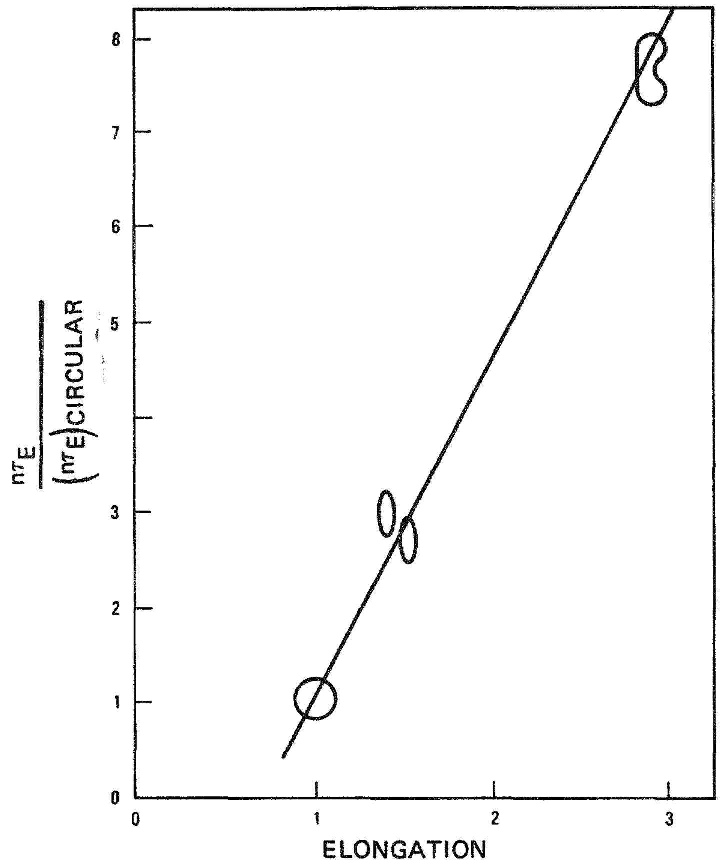


Fig. 1.3-3. Dependence of nτ on elongation:
Doublet IIA results

and is about 80 MW for a ∿ 4 m major radius device. But in D-T operation, alpha power at near ignition temperatures provides a substantial fraction of the heating; it is found that 60 MW of auxiliary heating power is sufficient in the D-T case. These predictions, based on global energy balance arguments, have been verified by means of spatially dependent transport codes; in fact, somewhat smaller estimates for P emerge from such detailed studies.

Although Alcator scaling leads to the seemingly inescapable conclusion that the energy required for adequate penetration by neutral beams of ignition grade plasmas is in excess of 300 keV, in actuality, the situation is not quite so bleak. One can envision optimistic dynamic scenarios in which fueling is judiciously mixed with auxiliary heating and the onset of alpha heating in such a way that one need never penetrate the ignition nτ.

Alternatively, there is hope that clever heating techniques such as ripple injection or heating followed by compression may effectively lower E_b. Another possibility involves the use of direct conversion; in this case, one may operate with comparable efficiencies and use only the full energy component. But, the required energy will very likely remain in excess of 150 keV and the associated R&D program will be substantial.

While many of the systems and much of the physics can simply be scaled up from existing devices to an ignition device, this does not apply to a neutral beam injection system. That the technological problems become far more difficult is seen by computing E_b, the beam energy required for adequate penetraion. Because E_b is proportional to the line density, it can be computed directly in terms of $n\tau$ for the case of Alcator scaling:

$$n\tau \propto (na)^2 \propto E_b^2 \quad .$$

This gives $E_b \approx 150 - 200$ keV for the case of hydrogen beams. For the case of deuterium, the result is 300-400 keV, a frightening technological prospect for positive ion beams; in this energy range, the neutralization efficiency is only 5 - 8%.

A promising alternative heating procedure involves the coupling of electromagnetic energy to the plasma. By rf excitation near the lower hybrid resonance frequency, many problems inherent to beams can be circumvented. Furthermore, scaling of lower hybrid heating up to reactor regimes appears to be more palatable technologically and more attractive economically. A systematic analysis has been carried out (Vol. III) of the major questions associated with lower hybrid heating of reactor grade plasmas, namely the questions of coupling of the energy from the launching structure to the plasma, propagation of the wave in the plasma, and the absorption of the wave energy by the plasma. Just as with neutral beams, the penetration of hot, dense plasmas turns out to be the most difficult matter and the adequacy of the lower hybrid heating approach is found to be marginal. However, since the physics of rf heating has not been as thoroughly explored as that of neutral beam heating, there is room for considerable optimism that innovation in design can make lower hybrid heating significantly more attractive.

1.3.3. BURN PHASE

The primary ingredients that determine the plasma conditions at burn are the relevant transport scaling laws, MHD stability, and the degree of impurity contamination. We have assumed that the plasma temperature is MHD-limited rather than transport-limited so that the scaling laws do not play a major role in the analysis.

From Doublet IIA experimental evidence (Ref. 1.3-1), it is concluded that there is much to be gained from employing an elongated plasma cross section, at least for ohmically heated discharges. As reactor conditions are approached, the efficiency with which the applied fields confine the plasma, i.e., the β of the plasma, takes an overriding importance. While high β values have not yet been achieved in tokamaks, the theoretical evidence (Ref. 1.3-2) is overwhelming that vertical elongation significantly extends the β limits. The question of the optimal elongation must be answered by MHD analyses. Three relevant MHD considerations are illustrated in Fig. 1.3-4, which displays the results of a number of MHD code runs for doublet shaped plasmas. Note that the abscissa is a reverse scale for κ, but a forward scale for plasma radius. The vertical height and the radial position of the inside surface of the plasma were fixed. The function a'-a, where a' is the distance from the center of the plasma to the F-coil cage, measures the allowable separation between F-coils and plasma surface. This is a very design dependent parameter and will be discussed in more detail below. For the moment, we note that a'-a increases as elongation is decreased. The quantity β_c is the beta limit set by a quasi-analytical formulation of stability to ballooning modes; it reaches a maximum for $\kappa \simeq 3.0$ but the maximum is a very broad one. Stability to interchange modes was also calculated, but it was found to be non-limiting in all cases. The parameter δ is a measure of current peaking which can be tolerated before an active adjustment of field-shaping coils is required. This parameter plays a very important role in quantifying the versatility of the field-shaping coil system; specifically, it determines the extent to which the plasma can be shaped adequately. Uncertainties in the underlying plasma dynamics can be tolerated without compromising the plasma shaping

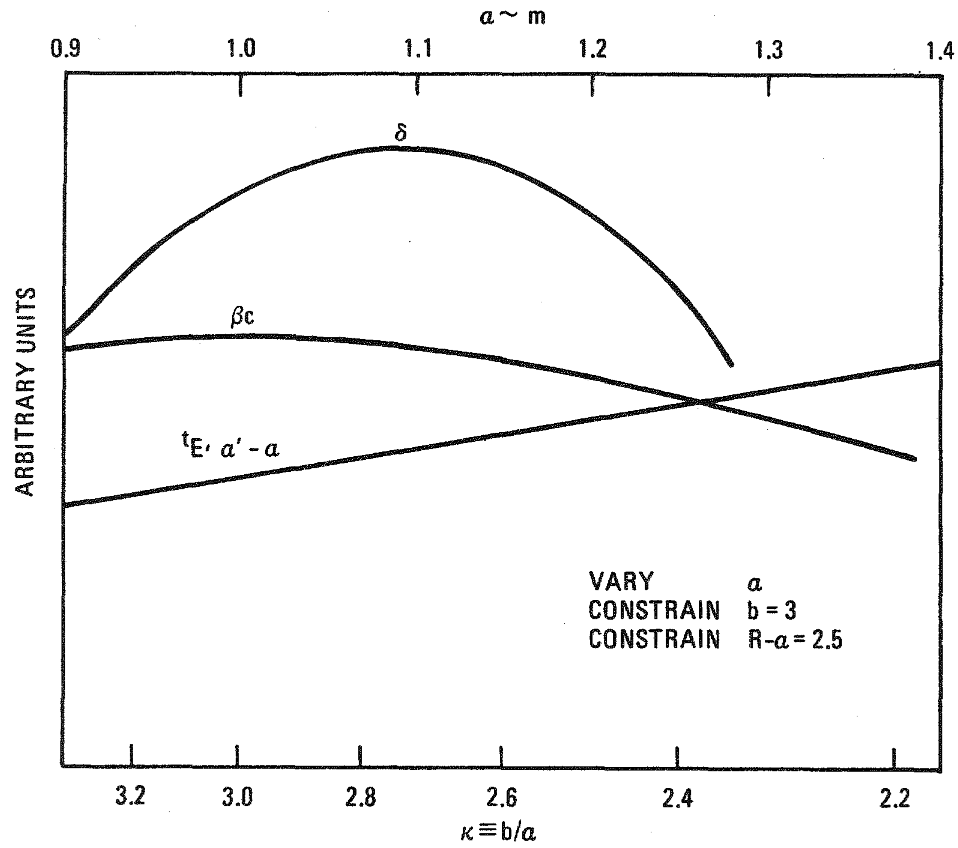


Fig. 1.3-4. A typical elongation study

capability. The most favorable value of δ occurs at $\kappa = 2.7$. In the devices to be described in this report, the maximum of the δ curve was used to define the height-to-width ratio of the plasma.

Now return to the function (a'-a); this variable has a considerable impact on the device design. In principle, the most effective way to shape a plasma is to use a close fitting set of F-coils which completely surround the plasma. Operationally, however, access must be provided through these coils to permit evacuation and filling of the chamber, diagnostics, maintenance and auxiliary heating. Our experience with coil placement indicates these coils must be placed such that a'/a does not exceed values of 1.3 to 1.5 depending on the location, the waist of the doublet being the most critical location. This requirement imposes design constraints on the device subject to the specified mission. An alternate coil system design has been developed* (Ref. 1.3-3) which is in many ways superior to the single layer coil approach adopted, for example, in Doublet III (Ref. 1.3-2); this superiority is particularly evident when examined in light of the requirements

*Developed by General Atomic Company, February 1977, under private funding.

of a D-T burning device. The configuration involves use of a double set of coils (F and S) as shown in Fig. 1.3-5. Conventional F-coils are placed a considerable distance from the plasma and a new set of small trim coils with a small number of turns are placed immediately outside the vacuum

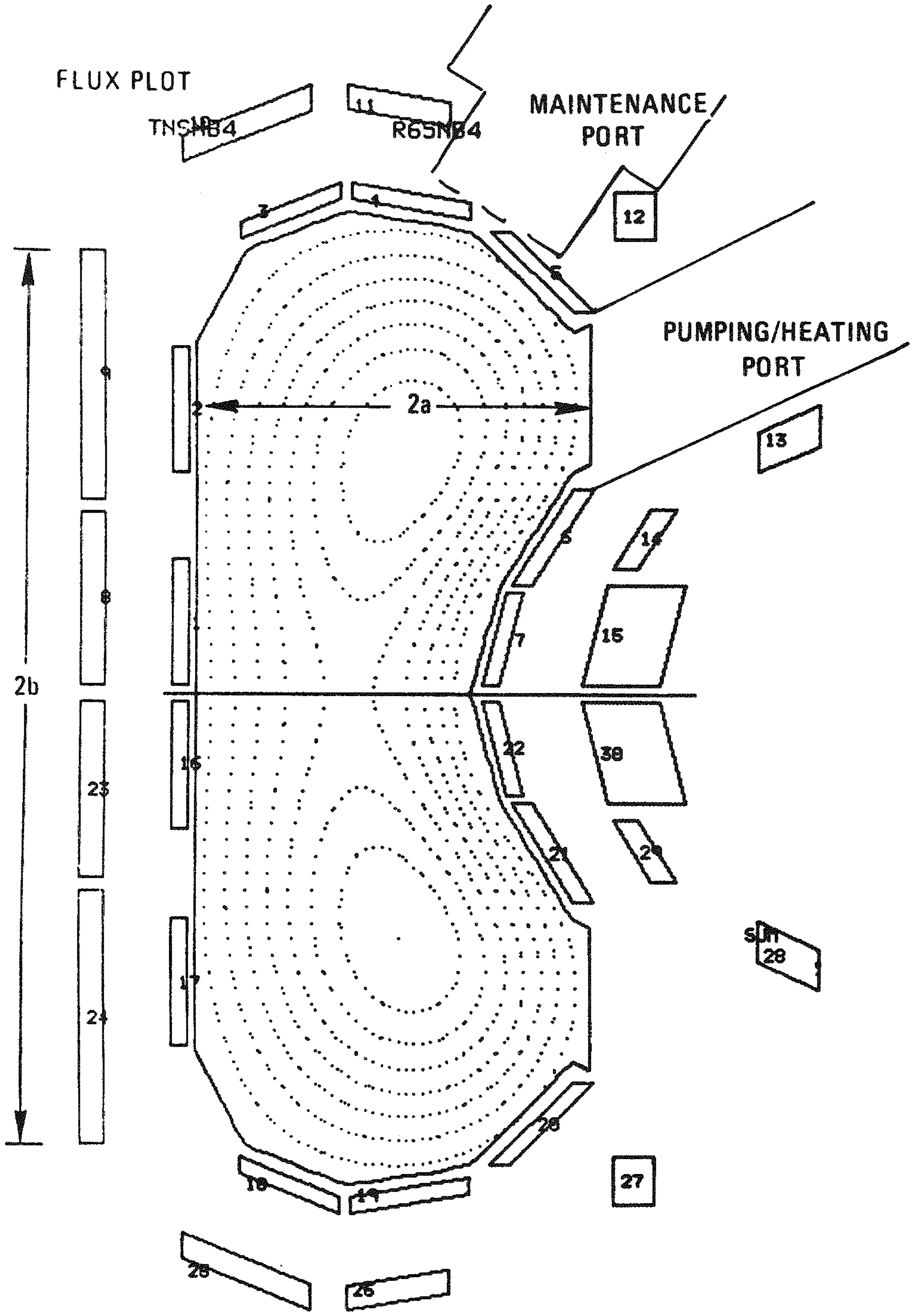


Fig. 1.3-5. A double layer field-shaping coil system

vessel. The close fitting S-coils respond very efficiently to plasma distortions and allow for large access porting as shown. In addition to augmenting the plasma-shaping function of the outer F-coils, the S-coils provide the requisite plasma stabilization (hence "S - stabilizing coils"). Since these coils are small, they are nearly transparent to neutrons and allow for a blanket to be introduced for the generation of sensible heat. The decision to use the additional set of trim coils is then one of mission requirements. If sensible heat is to be produced, the F-coils should not intercept a large fraction of the neutron energy. This consideration, in fact, defines the major design variation in the devices considered for TNS.

Finally, turn to the issue of impurities, the most serious obstacle to the attainment of the long burn times necessary for net power production. The allowed concentrations of a variety of impurity species are computed on the basis of recent impurity radiation calculations. This defines the extent of the impurity control problem and, in particular, forms the basis for an assessment of the adequacy of passive impurity control. Two impurity control schemes, impurity flow reversal and divertors, have been examined in some detail (Ref. 1.3-3). Estimates have been made for the required up-down assymmetric gas source necessary to give significant impurity pumping. Divertors have been shown to be fully compatible with the doublet design; in fact, the field-shaping coil system adopted lends itself to divertor designs which require very little current in the divertor coil and which are highly controllable.

Perhaps the most practical solution to the high-Z impurity problem is to exclude such materials totally from the vacuum chamber walls. These metals would be replaced by low-Z substitutes, which presumably would be less troublesome to plasma performance. However, the basic drawback thus far has been in determining which low-Z material best satisfies a rather demanding set of requirements. These criteria include not only how well such materials hold up to intense radiation bombardment but also how their exposed surfaces behave when in contact with the gaseous impurities expected to be present in the plasma chamber. For example, the adsorption of impurities such as oxygen during the dwell time and the subsequent liberation during start-up can frustrate the most technologically sophisticated pumping system.

The problems related in choosing a proper first wall or "limiter" material and the restrictions that this selection can place on a vacuum pumping system have been examined (Vol. V). The wall protection concept adopted is a combination of beryllium-on-copper limiters to shape normal discharges and graphite tiles strategically located on the inboard wall to absorb the enormous energy loads associated with disruptions and runaway dominated discharges. In the outboard region, a carbon coating is applied directly to the chamber wall.

REFERENCES

1.3-1. Freeman, R., *et al.*, "Plasma Physics and Controlled Fusion Research," Proc. of the 6th Intl. Conf., Berchtesgaden, 1976, IAEA, Vienna, Vol. 2, p. 387.

1.3-2. Chu, M. S., *et al.*, *ibid.*, Vol. 1, p. 317.

1.3-3. Ohyabu, N., "Double Layer Field Shaping Coil System," General Atomic Report GA-A14434, to be published in *Nuclear Fusion*.

1.4. REACTOR ENGINEERING

Reactor engineering studies were carried out on the five reactor concepts presented in Section 1.1. Emphasis was placed on the studies of the 3.8 m ITR, and, to the extent appropriate, the results were extended to the other concepts. The reactors differed from one another in many features, however, necessitating some conceptual design activities unique to each reactor.

Argonne National Laboratory collaborated with General Atomic in the design of neutral beam systems, vacuum pumping and tritium systems, superconducting E-coil, penetration shielding systems, and power supplies. ANL also performed studies of radiation effects on materials in support of the design effort (Vol. V).

Studies of the ohmic heating power supply were carried out by Los Alamos Scientific Laboratory (Vol. XI).

1.4.1. IGNITION TEST REACTOR

A conceptual design of an ignition test reactor (ITR) was developed to meet the design objectives of a high beta machine to study ignition and burn dynamics. This machine (Fig. 1.4-1), with a major radius of 3.8 m, utilizes the doublet plasma concept, and is designed for an expected lifetime of at least ten years. Burn times are moderate (30 sec) and the maximum duty factor is 0.1; no attempt is made to extract high grade heat or generate power. Component and system designs were developed sufficiently to show feasibility with emphasis given to integration of the design to yield an overall self-consistent device. A more complete description of the design is presented in Volume IV.

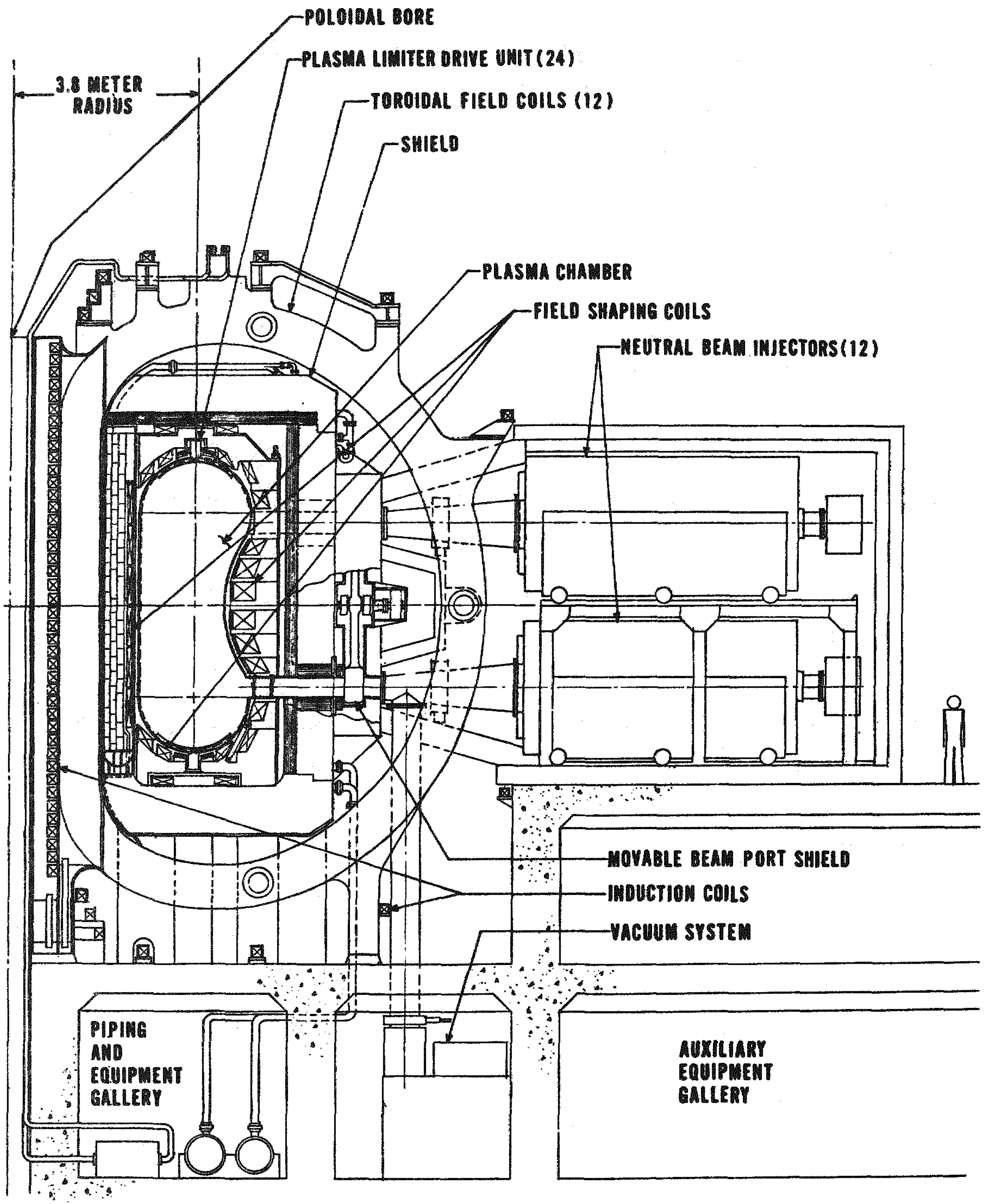


Fig. 1.4-1. Ignition Test Reactor (ITR)

1.4.1.1. Plasma Chamber

The doublet shaped plasma chamber, designed by McDonnell-Douglas (Vol. VI), is a compound curved continuous welded Inconel 625 structure assembled from twelve 30^{o} segments (Fig. 1.4-2). The toroidal vessel has a major radius of 3.8 m, a height of 6.04 m, and at its widest point, a width of 3.2 m. The mass of the 360 m^2 surface area vessel is about 42,000 kg, and the vessel toroidal electrical resistance is about 0.2 mΩ. The water cooled vessel wall consists of a thick inner skin which incorporates poloidal coolant passages, to which is attached a thin outer skin which seals the coolant channels. The wall thicknesses, which are controlled by plasma disruption loads and static buckling considerations, are 2.0 and

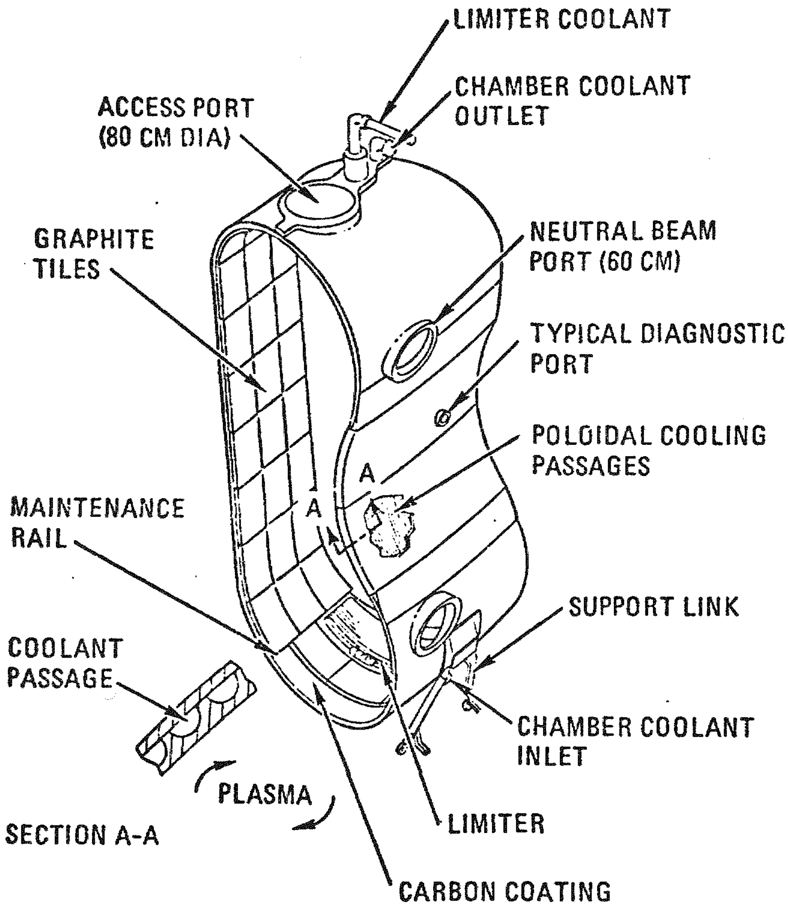


Fig. 1.4-2. Plasma chamber segment

4.8 cm for the inboard and outboard walls, respectively. Each segment has its own water supply and return manifolds located at the bottom and top of the vessel.

Each segment of the chamber incorporates an 80 cm diameter access penetration at the top, two neutral beam penetrations on the upper and lower lobes, and numerous diagnostic ports, all of which are actively cooled. Twelve pairs of water cooled tubular legs support the chamber. These legs are canted such that as the vessel is heated or cooled, the net upward and downward thermal expansions are identical, thereby maintaining the vessel midplane in the same vertical position.

The interior of the vessel is coated with ion vapor deposited carbon to provide a low Z surface. Carbon tiles, 0.5 m square, are located on the inboard wall to protect the chamber from plasma disruptions, neutral beam dumps, and neutral beam shine-through. Velcro (metal hooks and loops) is utilized to attach the tiles to the chamber wall, thereby permitting relatively simple and fast tile replacement. The tiles are cooled by radiation to the chamber walls.

Plasma positioning is accomplished with 24 movable limiters located at the top and bottom of the chamber. Each limiter consists of a bank of tightly spaced beryllium coated copper tubes utilizing high velocity water as the coolant. The limiter extends over a 90^{o} poloidal arc on the top and bottom lobes of the vessel and is 0.7 m wide to permit installation and removal through the access port. Concentric coolant supply and return lines which penetrate the vessel wall form the drive shaft connecting the limiter to the drive mechanism outside the vessel.

1.4.1.2. Toroidal Field Coils

The twelve identical toroidal field coils incorporate NbTi superconductor with copper stabilizer, and are maintained at 4.2 K with liquid helium bath cooling. The normal operating field is 8 tesla (4 tesla on axis), but the coils are designed to produce a peak field of 10 tesla, providing a margin to ensure that ignition is achieved. The pancake (helically) wound conductor is graded on the basis of magnetic field and bearing loads. In the low field inboard region of the coil where the radial bearing loads are highest, the conductors are cooled only on their edges, whereas in the higher field regions, progressively greater face cooling is provided. The 10,000 ampere conductor has an average current density of 2300 A/cm^2 and the overall helium vessel packing factor is 0.5. The radial bearing loads are transmitted through the helium vessel walls into the centerpost support cylinder, which is integrated with the induction coil. Cold braces, some of which are removable for overhead crane access, connect the coils together for stability. Superconducting-to-normal conducting transitions are located at the top of the coil with copper buses leading to the lower gallery where the dump resistors and switches are located.

1.4.1.3. Field-Shaping Coils

Plasma shaping and positioning is accomplished with the field-shaping coils (F-coils). These coils are situated as close to the plasma chamber as practical for enhanced plasma shaping and economical operation, as well as to minimize the loads imposed on the vacuum vessel during a plasma disruption. Normal conducting copper coils are utilized and removal of the substantial neutronic energy deposition and I^2R heat is accomplished with water cooling.

The inboard coils are located inside an annular steel tank as shown in Fig. 1.4-1. Forced external cooling of these six coils and the shield, which also occupies the tank, is provided by upward bulk flow of demineralized water.

The outboard coils are located symmetrically about the horizontal midplane and are cooled by forced water flow in internal channels. The coils are segmented toroidally with demountable joints so that portions may be removed for access. Thin, low power S-coils are situated between the F-coils and the plasma chamber at the very top and bottom of the device as well as in the indented midplane area. These coils, which supplement the plasma shaping and positioning functions of the F-coils, provide plasma stabilization.

1.4.1.4. Induction Coils

The induction coil (E-coil) system is comprised of a solenoid in the poloidal bore formed by the toroidal field coils, and upper, lower, and outboard coils arranged so as to minimize the error fields in the plasma region. NbTi superconductor, cryogenically stabilized by bath cooling, is utilized for all portions of the E-coil except for the outermost coils which are normal conducting and have their own power supplies. A plasma current inducing flux swing of 37 volt-sec is realized with this coil system.

1.4.1.5. Shields

The primary function of the inboard shield is to limit the nuclear heating in the superconducting toroidal field coils to one megajoule during the burn. This is accomplished in the ITR with the use of tungsten bricks which are stacked inside an annular steel tank which also contains the inboard field-shaping coils. Cooling is provided by water flowing upward through holes in the bricks. The stainless steel tank walls are 1 cm thick and support a portion of the weight of the upper shield slabs.

The outboard shield is designed such that the dose rate outside the shield is sufficiently low to allow direct personnel access one day after shutdown. It consists of a permanent bottom shield which also serves as a support base for the shield and all components inside of it, as well as top and side shield segments, all of which are water cooled. The bottom shield is a steel structure with distributed coolant passages comprising 30% of the volume. Lead panels 15 cm thick are attached to the exterior of the bottom shield to reduce the external gamma ray levels. The 24 side shield sections, which extend the full height of the shield, consist of a 30 cm thick inner layer of water cooled steel plates followed by an 80 cm section incorporating primarily lead and borated water with some imbedded steel structure and coolant lines. The upper shield is toroidally segmented into 24 sections also, and incorporates the same distribution of shielding materials as the side shielding. The shield segments fit relatively close and are sealed with epoxy at the outer surfaces to provide a secondary vacuum inside the shield.

1.4.1.6. Neutral Beams

Plasma heating is accomplished with the use of neutral beam injectors incorporating positive ion deuterium sources. Twelve injectors, each providing 5 MW of 150 keV D^o beam, are situated in two clusters on opposite sides of the reactor. Each beam cluster consists of six injectors stacked in pairs such that the beams are injected normal to the plasma at the middle of the top and bottom lobes in three adjacent spaces between B-coils. Beam injection is provided for about 2 seconds at the beginning of each cycle, and quick closing shutter shield devices which are mounted on the outside of the outboard shield are used at the end of the injection period to minimize neutron streaming up the beam lines.

1.4.1.7. Plasma Chamber Vacuum Pumping System

The principal function of the plasma chamber vacuum pumping system is the pumpdown of the vessel between discharges such that impurity concentrations are reduced to levels acceptable for the initiation of the next discharge. In the ITR, pumping is accomplished through three 0.5 m diameter ducts which branch off neutral beam lines downstream of the neutral beam shield plug. Hybrid cryocondensation/gas bearing turbopumps were selected for this design. Six pumps are included in the chamber evacuation system and are connected such that one set of three is being used while the other set is being regenerated. Large vacuum isolation valves are utilized during the four hour pump regeneration period as well as during maintenance operations performed at atmospheric pressure on other parts of the system.

1.4.1.8. Tritium Handling System

The tritium handling system was developed as a joint effort between Argonne National Laboratory and General Atomic Company (Vol. V). The multiple gas streams from the regeneration turbopumps of the plasma chamber vacuum pumping system are consolidated and passed from the reactor building into the tritium facility building, a lined containment structure. Further compression is achieved with bellows pumps in series followed by a diaphragm compressor. Chemical impurities are then removed by cryogenic trapping and hot gettering, while removal of helium is accomplished by a cryogenic stripping column. The fuel stream containing deuterium, tritium, and protium then enters a cryogenic distillation cascade where the protium is removed and tritium enrichment is achieved. Three output streams are produced from this cascade: a deuterium-tritium stream enriched in tritium for refueling, a high purity deuterium stream for the neutral beams, and a HD waste stream. The D-T stream is directed to the fuel blender where final D/T ratio adjustments are made utilizing D-T from storage. The gas stream leaves the tritium facility building through isolation valving and passes to either a small gaseous fuel delivery tank or through a liquefier to the pellet manufacture and injection subsystem. Low level gaseous, liquid and solid tritiated wastes are processed for disposal in small waste consolidation and disposal subsystems.

1.4.2. UPGRADABLE ITR - HELIUM COOLED

A conceptual design was developed for an upgradable ignition test reactor (UITR). This reactor (Fig. 1.4-3), which is based on the doublet plasma concept, is larger but similar in many respects to the 3.8 m ITR. In its ITR mode, the reactor operates in moderate length pulses (30 sec) with a duty factor of 0.1 (5 min cycles). For net power experimental power reactor (EPR) operating mode, the machine operates with a 90 sec pulse with a nominal dwell of 26 seconds, resulting in a duty factor of 0.78.

The plasma chamber is sized so as to produce sufficient thermal power to achieve net power in the EPR mode of operation. Likewise, a blanket is

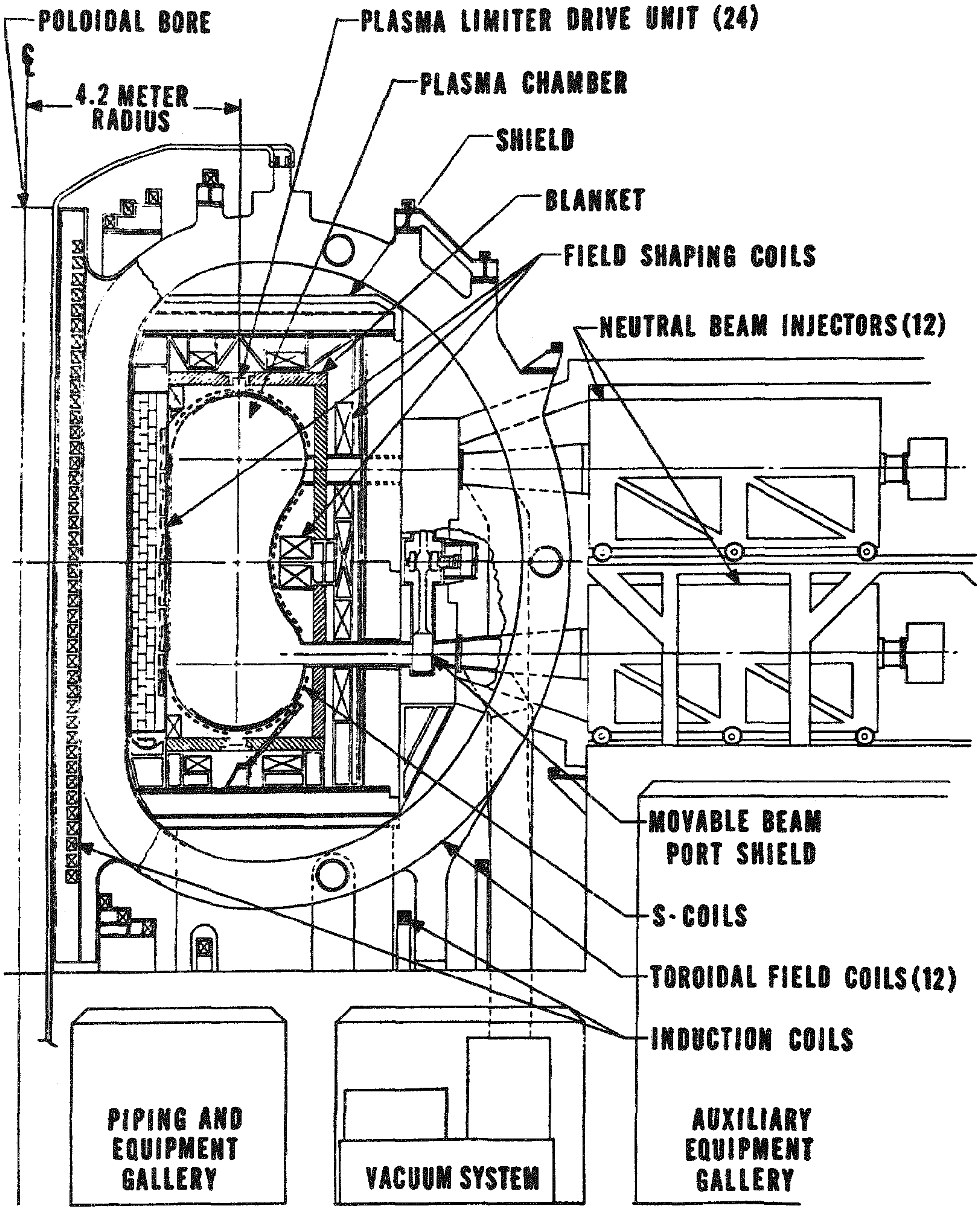


Fig. 1.4-3. Upgradable Ignition Test Reactor (UITR)

included and sufficient shielding is provided to permit ultimate operation with a high temperature coolant at high duty factor. Initially, however, no power conversion system is provided and only the power supplies and energy storage for operation in an ITR mode are installed. All the heat produced is rejected through cooling towers. After several years operation as a low duty factor ignition test reactor, the power conversion system would be installed and the power supplies and energy storage systems would be upgraded for EPR operation at high duty factor.

Table 1.2-1 gives the major design parameters for the UITR. Included in this table are the parameters for the two different modes of operation (e.g. ITR and EPR).

The plasma chamber for the UITR is a continuous toroidal shell structure of Inconel 625, similar to that used for the 3.8 m ITR. Pressurized water (2.5 MPa) is used as coolant to maintain the material temperatures in the chamber wall at low temperatures. The coolant water enters at 150^{o}C and exits the chamber at 200^{o}C absorbing approximately 215 MW(t). A large portion (65%) of this low grade heat is utilized by the power conversion cycle in the form of feedwater heating, thereby enhancing the power cycle performance.

The reference design has a 25 cm stainless steel blanket cooled by helium pressurized at 50 atmospheres. The blanket surrounds a major portion of the plasma chamber (Fig. 1.4-3) and has the capability of generating high grade heat and sustained operation at a high duty factor.

Like the 3.8 m ITR, the shielding is divided into two regions: the inboard shielding which limits the nuclear heating of the superconducting toroidal field coils, and the outboard region which limits the activation of external components and provides protection for personnel so that contact maintenance can be performed 24 hours after shutdown. As with the 3.8 m ITR, tungsten is used in the inboard region and a combination of stainless steel, borated water, and lead is used in the outboard region.

Twelve constant tension shaped coils employing NbTi/Cu superconductor are used to produce the toroidal field. While the normal maximum operating field is 8 tesla, they are designed to generate a peak field as high as 10 tesla (5 tesla at the plasma center).

The E-coil system supplies the poloidal magnetic flux change (volt-seconds) through the tokamak bore which induces and sustains the toroidal plasma current. The E-coils, which are similar to those designed for the 3.8 m ITR, utilize NbTi/Cu conductors and are cryogenically stabilized by bath cooling. Peak current is 50 KA in the solenoid region.

Six inboard F-coils are located within the inboard shield tank, similar to the installation of the inboard F-coils in the 3.8 m ITR. Due to the imposition of the blanket in the outboard region, however, the arrangement for the outboard F-coils of the UITR is different from the arrangement in the 3.8 m ITR. The two equatorial coils, which are critical to the shaping of the doublet plasma, are located immediately outside the plasma chamber. The other outboard coils are located outside the blanket modules.

Because the blanket is placed between the plasma chamber and the massive outer F-coils, the F-coils are supplemented by low power and relatively thin stabilization coils (S-coils), disposed closely around the chamber (inside the blanket).

The plasma is heated by 12 neutral beam injectors similar to those described for the 3.8 m ITR. A total of 56 MW of monoenergetic 150 keV D^o beam power is required for about 2 seconds.

With the high duty factor and shorter downtime, a much greater capacity plasma chamber vacuum pumping system is required for the UITR. A total of 24 cryosorption pumps are used — two connected to each of 12 pumping ducts — so that one set of 12 can be regenerated while the other set is on line. The cryopumps are connected to the plasma chamber by means of twelve 70 cm ducts which branch off the neutral beam injector ducts just outside the shield.

The tritium handling system is similar to the system used for the 3.8 m ITR.

The superheat/reheat steam power conversion system (Fig. 1.4-4) is similar to the cycle utilized in the helium cooled HTGR steam plant. The hot helium from the blanket is routed to a steam generator which produces steam for turbine inlet conditions of 450°C/12.4 MPa. From the high pressure steam turbine, the exhaust flow is routed to the helium circulator turbine drive. The expanded steam is then reheated to its maximum temperature before being expanded through the intermediate and low pressure turbines. Partial flow extracted from the IP turbine is utilized for the main feed pump turbine drives. Heated cooling water from the plasma chamber is utilized for feedwater heating.

Operation at a duty factor of 0.78 results in a blanket power level of 250 MW(t) and a gross electrical power output of 103 MW(e). This corresponds to a gross cycle efficiency of approximately 41%. Allowing for a total parasitic power of 88 MW(e), the levelized net plant output is 15 MW(e).

1.4.3. UITR - PRESSURIZED WATER COOLED BLANKET

System studies were conducted in order to define the major design parameters for a UITR with a blanket cooled by pressurized water. The reactor is initially operated as an ignition test device for the first few years of operation. Then power conversion equipment, addition power supplies, and energy storage systems are installed for operation at a high duty factor and net power production.

The power conversion system is based on the Rankine power cycle utilized in nuclear pressurized water reactors (PWR). High pressure cooling water is circulated through the reactor blanket, absorbing its thermal energy and transferring it to a once-through steam generator. The pressurized cooling water enters the blanket at 250°C and exits at approximately

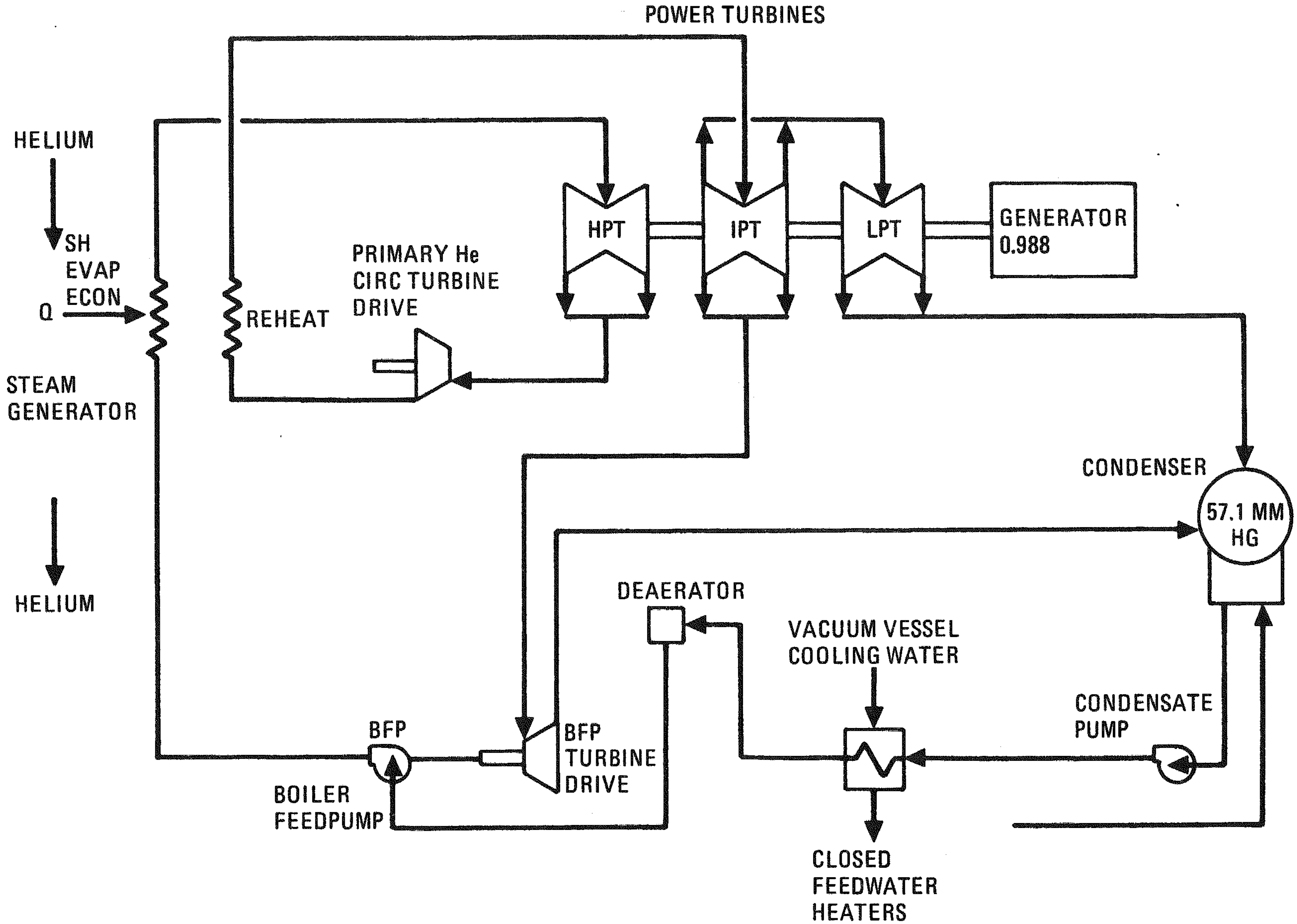


Fig. 1.4-4. Steam power conversion cycle flow diagram

$350^{o}C/17.5$ MPa. Steam cycle conditions result in a plant efficiency of approximately 30%. The water used in the vacuum vessel cooling system is again utilized for feedwater heating in the power conversion cycle, however additional turbine extraction is necessary to achieve the reasonable cycle efficiencies with the low temperature input from the reactor blanket.

Because of the lower steam cycle efficiency associated with the pressurized water-cooled blanket, the reactor size was increased to a major radius of 4.5 m in order to achieve approximately the same net power production as its helium cooled counterpart (Table 1.2-1).

The blanket is identical to the 4.2 m UITR design except the coolant passages are sized for pressurized water flow. Maximum material temperatures are limited to approximately $450^{o}C$. The blanket modules are slightly larger in size due to the increased plasma chamber dimensions associated with the 4.5 m machine.

Discussion of the other machine components for pressurized water cooled UITR machine parallels the descriptions of the 4.2 m machine. The inboard shield design is again made up of tungsten bricks encased in a large stainless steel tank. Cooling is accomplished by gross water flow up through the brick coolant holes. The shield limits the nuclear heat deposition in the toroidal field coil to approximately 1000 KJ. The toroidal field coil system consists of twelve NbTi superconducting constant tension coils cooled by liquid helium to 4.2 K.

With a plasma power of 1035 MW(t), operation at a duty factor of 0.78 results in a blanket power level 342 MT(t) and a gross electrical output of 106 MW(e). Allowing for total parasitic power of 86 MW(e), the levelized net plant output is 20 MW(e).

1.4.4. ALL COPPER COIL ITR

An ITR based on the scale-up of Doublet III, which utilizes only copper coils, was investigated. With no superconducting B- or E-coils that require shielding, a smaller machine is possible than for the other designs considered. Design points with up to 30 second burn and 10% duty factor were surveyed. The results showed that a machine in the range of 3.0 m major radius, 1.0 m minor radius is feasible (Fig. 1.4-5). The characteristics of this reactor are summarized in Table 1.2-1.

The all copper coil ITR design differs significantly from one utilizing superconducting coils. Remote handling was assumed totally remote at all times. Accordingly, the entire assembly of the ITR is shielded.

The 16 B-coils are clustered inside a cylindrical shield wall which is lined with highly cooled stainless steel plates. Radial space between the B-coils and the shield is minimized to reduce the length required for neutral beam injector drift ducts and plasma diagnostics to penetrate the reactor. E-coils are placed inside the toroidal bore of the B-coils in order to reduce poloidal bore size and increase coupling with the plasma.

The plasma chamber and F-coils are essentially the same as for the 3.8 m ITR except for the inboard side.

Access to the reactor is from the top. The upper dome-shaped section of shield is removable (approximately 250 tonnes in one piece, but could be made in two pieces) with an overhead crane. A remote manipulator which operates on its own trolley can then be extended down into the opening to inspect or to make repairs.

Neutral beams are housed in separate chambers with the drift ducts penetrating the shield. A movable shield at this penetration prevents unnecessary streaming up the drift duct and into the neutral beam injector. The top of the chamber is removable for access to and removal of the neutral beam injectors.

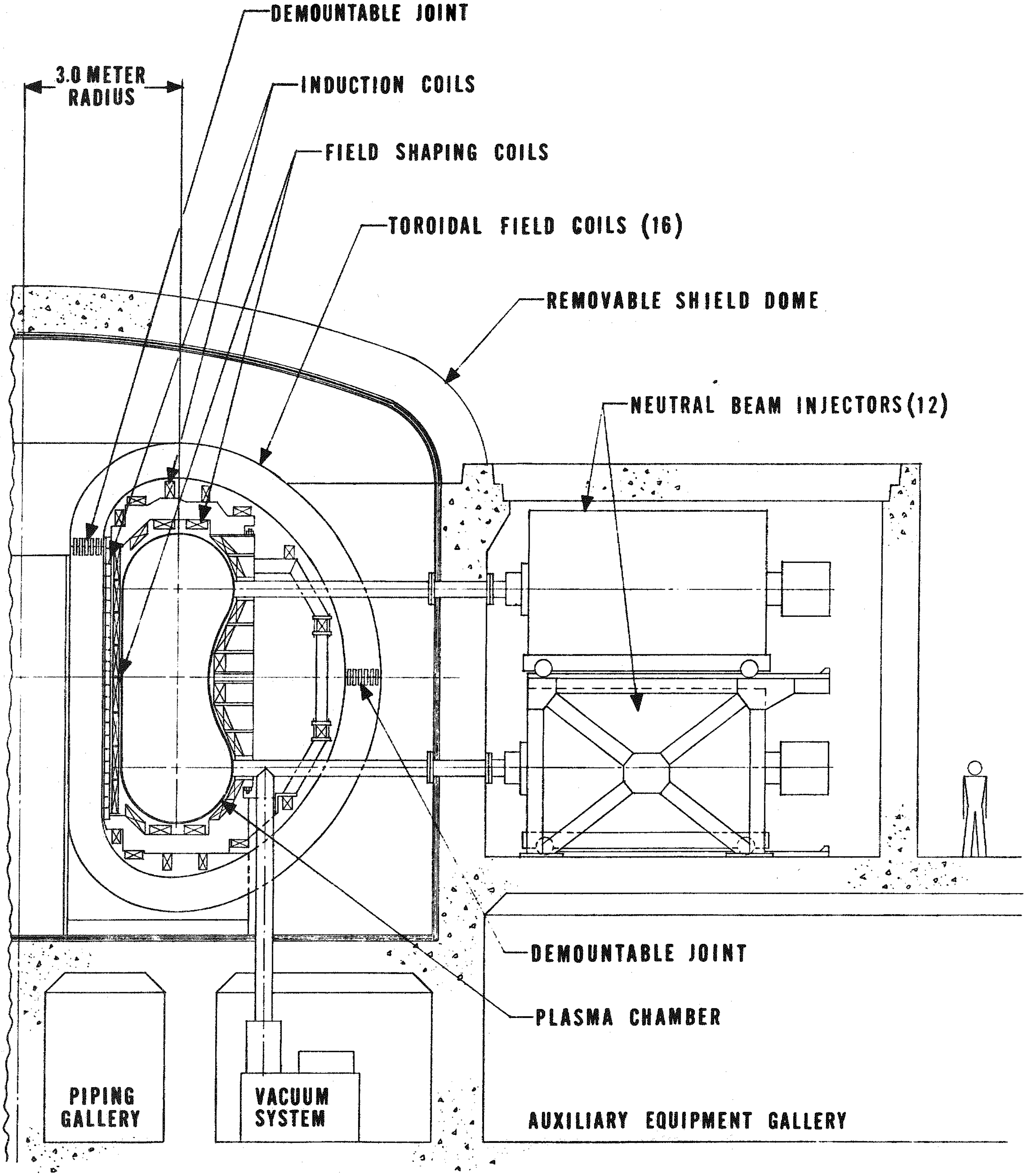


Fig. 1.4-5. 3.0 m copper coil ITR

1.4.5. PROTOTYPE EXPERIMENTAL POWER REACTOR (PEPR)

A Prototype Experimental Power Reactor (PEPR), which would be an ignition machine capable of extended burn times like the ITR and also be able to demonstrate certain operational characteristics typical of an experimental power reactor was also examined. The PEPR could operate at a moderately high duty factor of 0.75 and could demonstrate the extraction of high grade heat suitable for power conversion by incorporating a blanket with a high temperature (500^{o}C) cooling system. Unlike the UITR, however, this high grade heat would be dissipated in a heat rejection system and would not be used in a power conversion facility to generate electricity. Therefore, the objective of the PEPR, beyond that of achieving ignition and a sustained burn, is to demonstrate high duty factor operation.as well as extraction of high grade heat from a blanket.

A PEPR machine was sized so that comparisons could be made with other TNS candidate devices. The burn time was fixed at 90 sec as for the UITR and a blanket was incorporated on the top, bottom and outboard sides of the plasma chamber. Some of the major parameters are shown in Table 1.2-1. The major radius of 4.0 m is 20 cm greater than that of the reference ignition test reactor. Because of the longer burn time, the E-coil must supply about 25% more volt-seconds, necessitating an increase in the coil radius of about 8 cm. Again, because of the longer burn time, an additional 8 cm thickness of inboard shielding is needed to limit the heating of the toroidal field coil to 1 MJ. And finally, because the addition of the extra shielding reduces the on-axis field somewhat for a fixed maximum field at the coil, a slightly larger minor radius is required for ignition.

The PEPR components and systems are quite similar in design and performance to those described above for the ITR and the UITR, differing primarily in size. Inboard of the plasma chamber, the only differences are the slightly increased size of the shield and induction coil. Outboard of the chamber, the design resembles that of the UITR, with a 25 cm thick stainless steel helium cooled blanket provided to capture the plasma thermal power. The hot helium which leaves the blanket is routed through a heat dump heat exchanger where the thermal energy is transferred to water in a heat rejection loop.

1.5. ENGINEERING SUPPORT STUDIES

Engineering support studies were carried out in the areas of program planning, maintenance, facility design, safety assessment, regulatory considerations, design criteria, quality assurance, and site selection. The main purposes of these studies were to develop cost estimates and identify significant program implications.

1.5.1. PROGRAM PLANNING

An overall program plan for the TNS program was developed by the Ralph M. Parsons Company based on construction of the 3.8 m ITR (Vol. X). As reflected in the schedule (Fig. 1.5-1), it is expected that the TNS program will involve 3 years for site studies and conceptual design and 7 years for preliminary design, final design, construction, and start-up. The controlling work item is the procurement, fabrication and installation of the tokamak system which is expected to take some 5 years. In the tokamak system, the fabrication and installation of the toroidal field coils are on the critical path.

1.5.2. MAINTENANCE

A review of basic approaches to the maintenance of the ITR was performed by Battelle Columbus (Vol. VIII). While consideration was given to concepts involving low activation chamber and coil materials, primary emphasis was placed on comparing concepts involving total remote maintenance with the concept which, with the use of a heavy lead-sheathed shield, permits contact maintenance outside the shield envelope. Because maintenance can be performed much more quickly, it was concluded that the concept with the highly effective shield located close to the plasma chamber is the best concept for the ITR.

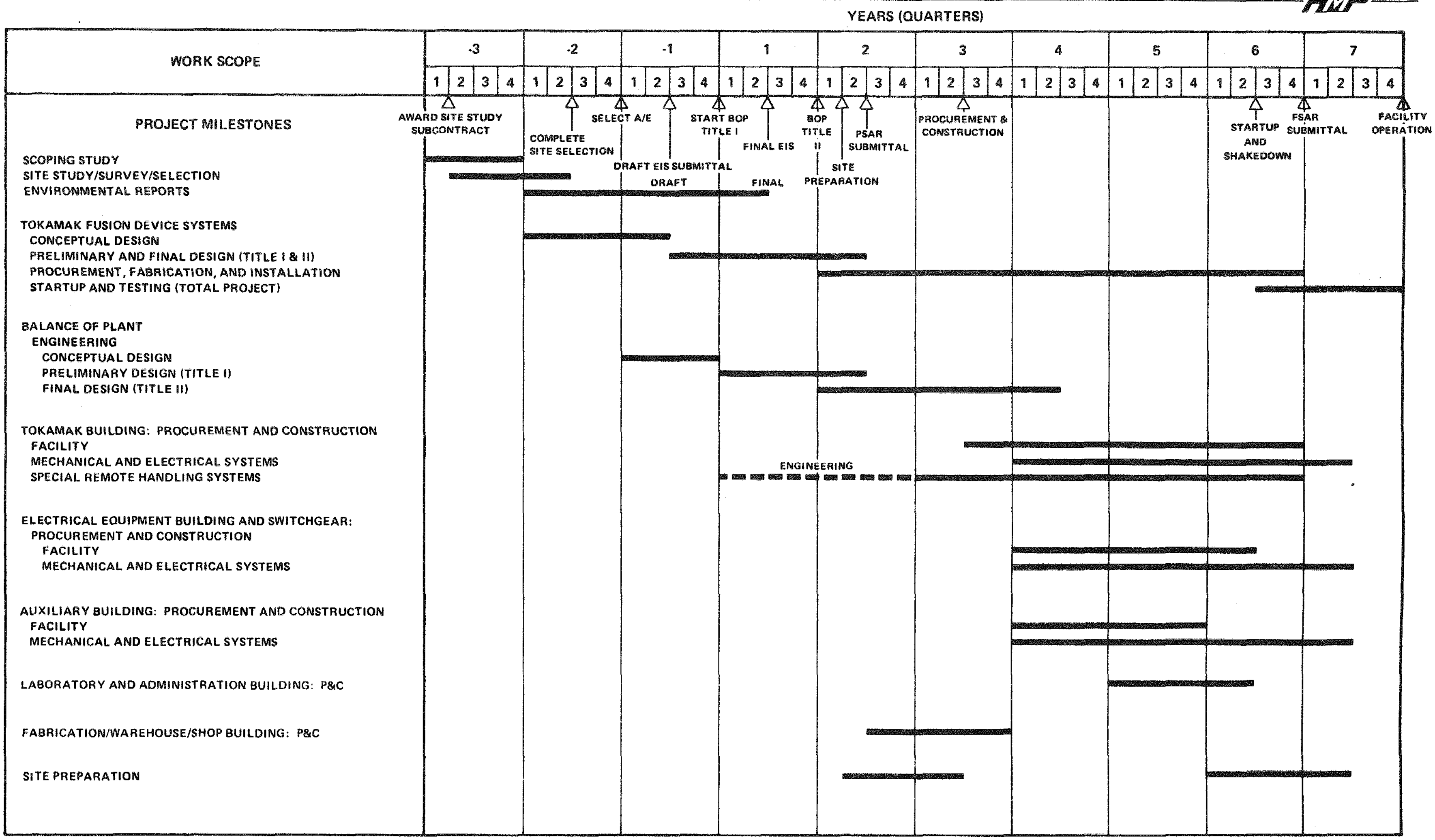


Fig. 1.5-1. 3.8 m ITR schedule

Designs were developed for equipment for a remote maintenance system by Aerojet Manufacturing Company (Vol. VII). In addition to a general purpose bridge mounted shielded cab and a vehicle mounted shielded cab, special purpose equipment was designed for plasma chamber maintenance (Fig. 1.5-2), replacement of the shield modules for maintenance inside the shield, and neutral beam maintenance. Hot cell equipment conceptual designs were also developed. Operating procedures for this equipment were generated and cost estimates for the equipment were developed.

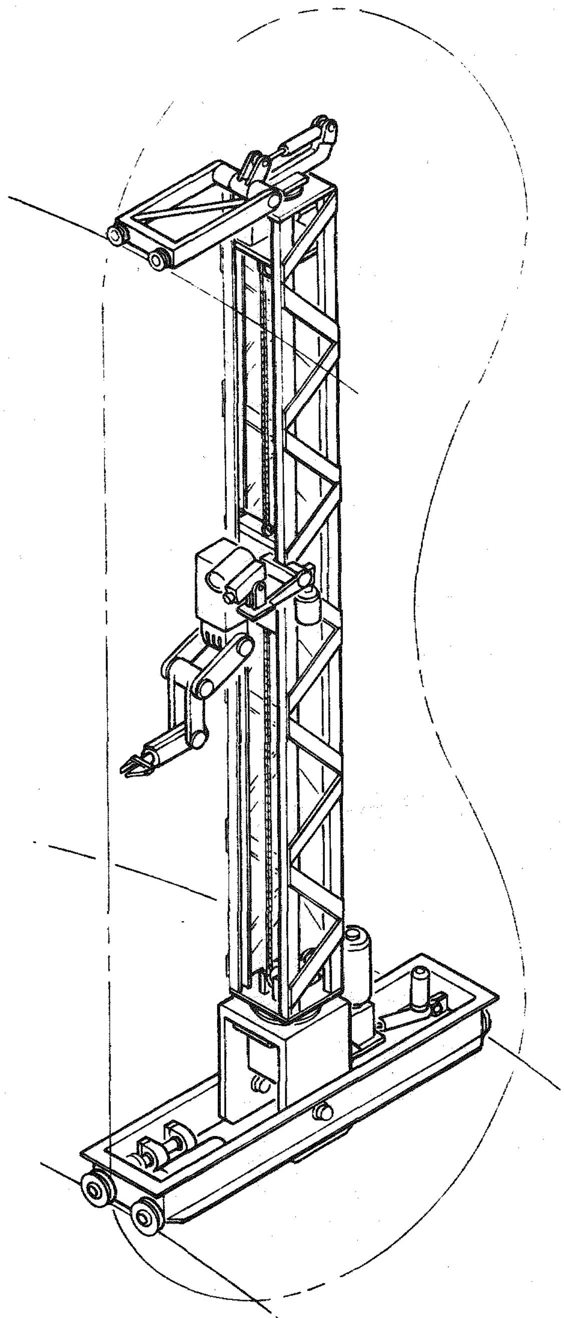


Fig. 1.5-2. In-vessel rail mounted maintenance machine

1.5.3. ITR FACILITY

The balance of plant and an overall facility was designed for the ITR by the Ralph M. Parsons Company (Vol. X). In addition to the reactor building, this facility (Fig. 1.5-3) encompasses a hot cell, an auxiliary building, an electrical building, a shipping and receiving building, and a laboratory and administration building. Provisions are also made for helium storage, electrical substation and cooling towers. The reactor building is designed to operate at subatmospheric pressure so that in the event of a tritium spill, the tritium can be removed with the tritium clean-up system without significant loss through the building structure.

1.5.4. SAFETY ASSESSMENT

An initial safety assessment for the 3.8 m ITR was carried out by Nuclear Services Corporation (Vol. IX). The design basis events previously identified for fission reactors were examined. From these, a total of 11 design basis accidents for fusion were identified for future study. In addition, radioactive materials handling, activation, and biological hazards were examined. It was concluded that while appropriate attention must be given to these aspects of a fusion reactor, the analysis involved is straightforward and the risks are minimal.

1.5.5. REGULATORY CONSIDERATIONS

The regulatory considerations appropriate for an ITR were also examined by Nuclear Services Corporation (Vol. IX). Existing regulations and standards were examined with regard to protecting personnel and the environment, protecting the economic investment in the ITR, and providing a source of good engineering practice. It was concluded that fusion-pertinent regulations should be established in order to place the appropriate emphasis on the safety implications of each hazard and to facilitate the development of fusion devices by avoiding unnecessary and costly constraints on various phases of fusion projects.

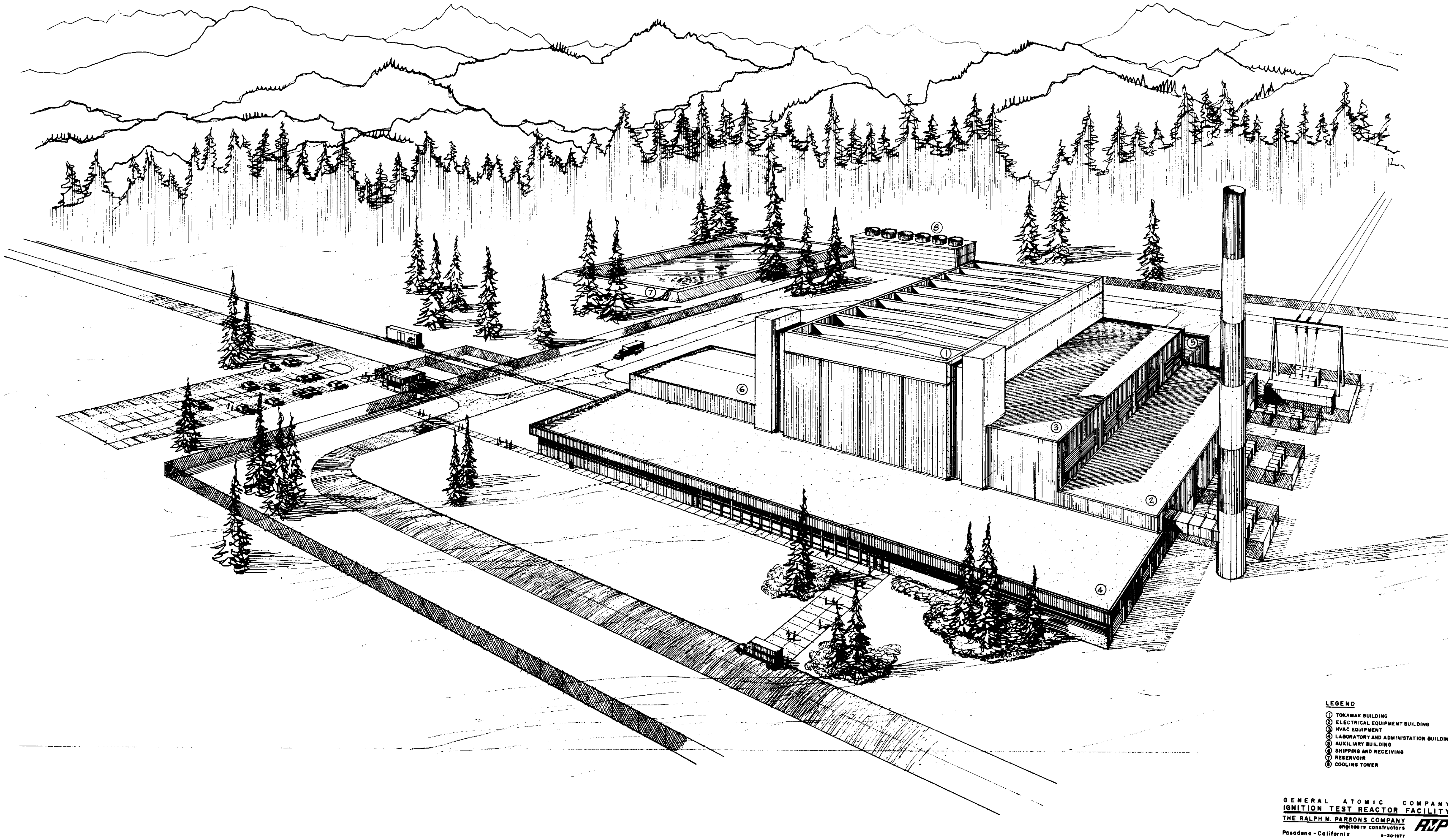


Fig. 1.5-3. ITR facility

1.5.6. DESIGN CRITERIA

The design criteria appropriate to the TNS program were also considered by Nuclear Services Corporation (Vol. IX). It was concluded that the only major safety accident of concern is the release of tritium and its oxidation in its pathway to those areas accessible to the public. The consequences of such an accident can be limited by inventory minimization, confinement, and establishing adequate site boundaries. Since the total tritium inventory at the site is limited (70 gms for the ITR), the dose at the site boundary would be acceptable with even a nominal exclusion radius.

The other design base accidents are of the type in which plant operability, on-site protection, and protection of plant investment are of concern. The design criteria to minimize these effects must be weighed against possible restrictions to the execution or cost of the experimental program caused by more stringent design and operating requirements.

1.5.7. QUALITY ASSURANCE

The Ralph M. Parsons Company developed an initial approach to the quality assurance program for the ITR (Vol. X). In considering quality assurance philosophy for fusion plants, in general, it was recognized that fusion does not represent the same scope or depth of safety problems that one attributes to fission plants. It may not be necessary to reference a quality assurance program such as 19 CFR 50 Appendix B specifically; on the other hand, it was accepted that criteria such as 19 CFR 50 Appendix B, ANSI N45.2, ASME Code, etc., do have certain quality assurance elements within them that one would want to incorporate into any project which embodied a sophisticated technology and in which reliability or maintenance of onstream performance would be extremely important objectives.

From this framework, a set of quality assurance requirements was developed for ITR. The requirements were based on specific criteria that should be adapted in fusion plants with the degree and depth of application determined by (1) the type of fusion facility involved, (2) the importance of safety, (3) the environment and reliability, and (4) the nature of the organization and interfaces providing the design and construction activities for a given facility.

1.5.8. SITE SELECTION

Site selection criteria for the TNS were examined and certain specific siting topics were considered by Nuclear Services Corporation (Vol. IX). It was concluded that the criteria for fusion facilities are no more restrictive than those used for fission, and in some instances less so, notably in the case of demographic criteria.

The advantages and disadvantages of locating on private or on Government land were identified. It was found that it is difficult to assign a definite advantage to either approach.

1.6. CONCLUSIONS

The TNS studies indicate that a range of technically feasible options is available for selecting the next major tokamak facility beyond TFTR. The actual selection is also dependent upon a number of other factors, including the cost, schedule implications, physics results from ongoing and currently planned experiments, state-of-the-art developments in technology programs, and objectives ultimately established for the TNS program. The latter, in turn, is highly dependent upon the program plan for the overall magnetic fusion program.

A prime candidate for the TNS program is the 3.8 m ITR with superconducting toroidal field coils. This machine appears to offer the least expensive route to achieving ignition and sufficiently long burns to permit study of fueling techniques and burn dynamics. It also incorporates many features, such as superconducting toroidal field coils and induction coils which are recognized as important to later commercial reactors.

Another prime candidate is the 4.2 m UITR with a helium cooled blanket. While its cost is substantially more than the cost of the 3.8 m ITR, it has the potential of not only achieving the objectives of the ITR but also can be upgraded to operate at a high duty cycle. The UITR is designed to produce net power and therefore, assuming the plasma physics performance is at least as favorable as expected, the plant would fully meet the objectives of the EPR. This would obviate the requirement to build a separate EPR and thereby would result in substantial overall program cost and schedule savings. Even if the plasma physics results were not as favorable as expected, a power conversion system could be installed to demonstrate power output capability, thereby essentially achieving the objectives of an EPR.

If the risk of the consequences of not generating the desired power level in the UITR is judged to be unacceptable, a reactor of the PEPR type could be built for the TNS program. With this approach, the reactor is designed for a high duty factor and a blanket designed for a high temperature coolant is installed. The heat generated in the blanket is dissipated in a cooling tower.

A variation of the PEPR approach involves the installation of a power conversion system to utilize the heat transported by the blanket coolant to generate a nominal amount of gross electrical power [e.g. 20 MW(e)]. Such a power generating fusion reactor (PGFR) could be built for a modest additional investment and would serve to accomplish most of the objectives considered for the EPR. A successful program based on the PGFR would lead directly to the construction of a demonstration power reactor and then to a lead commercial plant.