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Oak Ridge TNS Program: Technical Needs Assessment

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FUSION ENERGY DIVISION

OAK RIDGE TNS PROGRAM: TECHNICAL NEEDS ASSESSMENT

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ABSTRACT

This document highlights the technical requirements of the key subsystems of the Reference Design for The Next Step (TNS) and presents a preliminary assessment of the adequacy of the technical capabilities available to satisfy these requirements. Additional information on the Reference Design and the FY 1978 TNS Program activities can be found in the associated technical memoranda, ORNL/TM-5720, ORNL/TM-6721, and ORNL/TM-6723 — ORNL/TM-6733.

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INTRODUCTION

The Next Step (TNS) represents that phase of fusion energy development in which the major emphasis would be directed toward engineering testing and demonstration. The objective of the TNS studies (initiated by the Department of Energy's Office of Fusion Energy) has been to define the characteristics and requirements of a facility dedicated to the engineering testing phase of fusion power development. For this reason, the TNS study results are providing a basis for defining an Engineering Test Facility (ETF). Because the scientific basis required for a TNS/ETF will first be available for the tokamak concept, the reactor core of the facility has been based on the tokamak concept. However, the commitment to an ETF with a reactor core based on the tokamak concept does not represent a commitment to tokamaks as the ultimate power reactor concept.

The TNS studies at Oak Ridge National Laboratory (ORNL) were initiated in FY 1977. During FY 1977 the Oak Ridge effort pursued scoping studies in three broad areas: plasma engineering, systems modeling, and program planning. Based upon the findings of the FY 1977 efforts, it was judged that continued activities in the Oak Ridge TNS program should be directed toward preconceptual design with particular emphasis placed on engineering feasibility. As a point of departure for the FY 1978 activities, a Baseline Design was selected, based on the systems modeling effort of FY 1977. The primary objective of our FY 1978 TNS effort has been to evolve the Baseline Design toward a preconceptual design. However, it is emphasized that the FY 1978 effort was not intended to lead to a completed preconceptual design. Therefore, the design resulting from this year's effort is referred to as a Reference Design, rather than as a preconceptual design.

This document highlights the requirements of the key subsystems of the TNS Reference Design and presents a preliminary assessment of the adequacy of the technical capabilities available to satisfy these requirements. Additional advancements in both physics and technology are necessary to establish the technical basis for the design of TNS. The major areas requiring special research and development (R&D) emphasis are discussed for each subsystem.

1. SUMMARY

The technical needs of the key systems of the Reference Design are summarized. The goal of this continuing assessment effort¹ is the identification of potential barriers and technical gaps that may impede the successful implementation of the TNS program. This assessment employs both a rollback and rollforward determination of the needs for specific systems. The evaluation for each system is based on three questions:

- (1) Will the present and expected R&D effort satisfy the requirements of the TNS program?
- (2) Where are the major gaps in this R&D area?
- (3) How and when must these gaps be closed?

The level of effort directed toward the definition of the technical needs of the various systems was limited, and therefore a balance was made between depth and breadth of coverage for this year. Ten areas were addressed: physics needs, divertor system, pellet fueling system, poloidal field (PF) system, neutral beams, mechanical systems, blanket, radio frequency (rf) system, toroidal field (TF) system, and instrumentation and controls system. The technical needs and the R&D advancements required in each area are summarized below.

It is noted that these assessments tend toward generic issues which will gain in specificity as the studies go forward. Two continuing themes appear in these assessments: (1) an increased emphasis on technology and engineering developments appears to be warranted to provide reliable, cost-effective components for TNS on a timely basis, and (2) an expansion of the efforts to integrate the various physics results into a systems transient scenario and description is required to provide better guidance to R&D needs.

1.1 PHYSICS R&D NEEDS

Recent advances in tokamak confinement and heating experiments have provided optimism concerning the prospects for producing a controllable fusion core based on the tokamak concept. Nevertheless, uncertainties still exist in essentially all of the physics assumptions adopted in the Reference Design, although these assumptions are considered reasonable. This section briefly summarizes and discusses the physics R&D needs as suggested by the required improvements in understanding. Highlighted are five physics areas in which even a rudimentary level of understanding is considered seriously lacking. These areas are ordered according to increasing level of uncertainty: plasma confinement, plasma heating, ignition and burn, operation and control, and impurity control.

1.1.1 Plasma Confinement and Optimization

This area deals with the plasma size, the TF strength, the plasma density, and the plasma temperature required for ignition and burn in the Reference Design. These parameters are chosen in the context of their influence on plasma confinement. Substantial understanding in this area is already available, although surprises in future experiments cannot be ruled out. Continued studies to improve tokamak confinement via improved stability by optimization of the plasma shape, profile, and beta are also being actively pursued. The evolution of high beta plasmas over long pulses is one of the topics on which work is still needed.

1.1.2 Plasma Heating

Heating with 150-keV, 50-MW deuterium beams for 6 sec is assumed in the Reference Design. Recent neutral beam injection in the Princeton Large Torus (PLT), heating the plasma to near ignition temperatures, has increased confidence in these assumptions. Experimental corroboration is needed for the theoretical calculations showing that ignition can be reached with a neutral beam energy level of only 150 keV using the low density or small radius heating scenario in the Reference Design. The

physics of rf heating is presently less well understood, although vigorous experimental programs are in progress. The optimal combination of rf and neutral beam heating also needs to be explored experimentally.

1.1.3 Fusion Ignition and Burn

Achieving $Q = 1-2$ in the Tokamak Fusion Test Reactor (TFTR) will only provide a demonstration of the approach to ignition, while the Reference Design assumes ignition ($Q = 5$) and burn ($Q > 5$). An experimental understanding of the plasma physics of ignition and burn is needed. The study of the dependence of the burn process on the plasma parameters, the fusion alpha particle power deposition profile, the fueling rate, the deuterium-tritium (D-T) composition, and the method of fueling will provide the basis for burn control over long pulse times ($\sim 10^2$ sec). Detailed numerical simulations, consistent with the proper confinement assumptions and the possible improvements in TFTR achievements to $Q > 2$, are needed to reduce the gap between present estimates and the actual demonstration of ignition and burn. An experimental test of ignition and burn still seems necessary to verify our assumptions.

1.1.4 Plasma Operation and Control

The desired plasma parameters, profiles, and cross sections through the phases of startup, heating, ignition, burn, and shutdown should be elucidated through studies in plasma confinement and optimization, heating, and ignition and burn. An understanding of the requirements of the plasma control along the desired path of operation during the pulse is presently lacking. Studies of methods to avoid fusion thermal runaway and plasma disruption and to reduce pulsed fields at the TF coils should be actively pursued.

1.1.5 Impurity Control

The plasma and impurity transport processes through the plasma edge/scrape-off region and in the divertors are presently the weakest area of physics understanding. The use of poloidal divertors will be

studied in the Poloidal Divertor Experiment (PDX). However, the physics of the bundle divertors chosen for the Reference Design (because of their potential engineering advantages) and the impact of the divertors on plasma confinement must be studied at a level comparable to that of PDX before a clear choice can be made for the TNS class of device. The impurity production process involves interactions between the plasma and the wall/collector plates under large thermal power loads. The long pulse (~ 10 - 10^2 sec) behavior of these interactions and their effects on the impurity control processes also need to be studied because presently successful methods using titanium gettering may be limited to short pulses (~ 1 sec). Nondivertor impurity control schemes should also be explored.

1.1.6 Summary of Physics Needs

On the basis of these discussions we can identify critical physics issues for the Reference Design. They include:

- (1) Long pulse behavior of impurity production and transport under high thermal power in a diverted plasma.
- (2) Long pulse evolution of high beta MHD equilibria through substantial flux diffusion, influenced by fusion alpha particle heating and various fueling sources.

We can also identify areas where some understanding can be developed via theoretical analyses and numerical simulations with some confidence (although experimental studies in these areas would increase our confidence in the fusion plasma operation in the Reference Design). They include:

- (1) The physics of fusion ignition and burn ($Q > 5$) for a time scale much longer than the energy confinement time.
- (2) The physics of helium ash accumulation and transport in diverted plasmas.

Finally, the physics/engineering studies during FY 1978 have suggested schemes that could have a favorable impact on the reactor engineering and technology requirements. These studies point to the following desirable experimental verifications:

- (1) Experiments to optimize the neutral beam injection process (such as the low density and small radius scenarios) that directly or indirectly corroborate the process of heating to ignition with beam energies of ~ 150 keV.
- (2) Fueling experiments with partial pellet penetration.
- (3) Experiments with a flexible PF coil system and superconducting TF coils to study ways to minimize the pulsed field during startup and disruption.
- (4) Experiments with the use of plasma heating sources before and during current startup to increase the startup time and to reduce cost.
- (5) Experiments with variable field perturbations (such as ripples in the toroidal field) as a method of controlling the runaway electrons and the plasma edge condition and improving access to the plasma core.

1.2 DIVERTOR SYSTEM

In the Reference Design, a bundle divertor was adopted for impurity control. The concept chosen for continuing study incorporates a liquid lithium jet system to absorb the large amount of diverted power and particles with a reasonable cross-sectional area. The assessment of the technical needs for the bundle divertor shows two areas of needed emphasis:

- (1) Bundle divertor coil configurations that address
 - the incorporation of larger coils that permit adequate scrape-off areas and allow room for adequate shielding,
 - approaches to equalize the energy variation expected across the face of the lithium collection jets, and

- methods of ceramic insulation of the coils that would ease the shielding requirements.
- (2) The liquid lithium "shower" concept for ion/energy collection that addresses
- the expansion of the temperature range beyond the present $\sim 350^{\circ}\text{C}$ limit where data are available on the collection of hydrogen isotope ions by liquid lithium,
 - the ability of the "shower" to retain embedded alpha particles, and
 - the ability of liquid lithium to pump neutral hydrogen isotopes.

1.3 PELLET FUELING SYSTEM

The fueling technique adopted in the Reference Design is injection of pellets of solid deuterium and tritium. Developments in this area have provided encouraging results both in laboratory experiments and on the Impurity Study Experiment (ISX-A) tokamak.² Continuing physics calculations and experiments are needed to better define the injection requirements. The technical advancements required for TNS implementation are highlighted by three needed actions:

- (1) Development of the potentially promising concepts of pellet injection by gas, mechanical, and electromagnetic methods of acceleration.
- (2) Expansion of the data base on the physical properties of deuterium and tritium pellets.
- (3) Intensification of efforts to achieve equipment design innovations and simplifications that will lead to enhanced equipment reliability, fabricability, and cost-effectiveness.

1.4 NEUTRAL BEAMS

The demonstrated success of neutral beam heating, in particular the recent PLT results, has provided a growing confidence that such systems can be used in future devices. However, the complexities of existing

experimental beam lines raise issues concerning cost, reliability, and maintenance in the TNS context. The chief technical needs of neutral beam systems for the Reference Design are as follows:

- (1) The capacity for 12.5 MW per beam line for 5 sec at energy levels up to 150 keV.
- (2) Simple, more reliable systems that can obtain the necessary performance.
- (3) Simple means of operating and controlling multiple beam lines.
- (4) Design and fabrication concepts that will minimize the difficulties of maintaining the beam lines in a reactor environment.

1.5 POLOIDAL FIELD SYSTEM

The PF system consists of seven sets of coils, each requiring separate power supplies. A combination of copper and superconducting coils (inside and outside the TF coil, respectively) is used; the interior copper coils permit a close coupling to the plasma. The key R&D needs are:

- (1) Pulsed superconducting coil technology to meet the Reference Design needs (e.g., $\dot{B} \sim 2$ T/sec, $B_{\max} \sim 8$ T).
- (2) Simple electrical storage and supply circuitry for cost and reliability improvements.

1.6 MECHANICAL SYSTEMS

The mechanical systems for the Reference Design include the machine structure, the secondary vacuum enclosure (SVE), the remote servicing equipment, and the shielding. The technology data base is largely available to meet the mechanical systems requirements of the Reference Design. However, the specific implementation of the design requirements requires significant expansion of data base, especially in two areas:

- (1) Material properties in a high temperature, high neutron flux environment.

- (2) Remote servicing equipment and methods for replacement and repair of highly activated device components.

1.7 BLANKET

The Reference Design includes concepts for the first wall and bulk shield functions but does not address the requirements of a breeding blanket. The blanket that must breed fuel, recover heat, and provide some shielding is being studied in depth in the ORNL Tokamak Reactor Blanket Study,³ the objective of which is to develop a blanket concept based on near-term technology that satisfies reliability and lifetime needs for a commercial reactor. A promising blanket concept has been advanced and preliminary analyses support its potential for reactor application. This technical area is of particular value to the engineering test phase of TNS, and a continuing blanket development effort is necessary. The blanket requirements in terms of surface and volumetric heat generation rates coupled with the large number of power on/off cycles are challenging — even if the severe neutron irradiation effects are ignored. The following R&D activities would be of particular benefit:

- (1) Development of an experimental basis for promising blanket concepts via model testing that includes some limited neutron environment. MHD effects can and should be tested for blankets employing liquid lithium.
- (2) Characterization of the requirements imposed on the blanket (i.e., particle loading and heat loads) in the event of an abnormal or disruption situation as well as for a normal cycle.

1.8 TOROIDAL FIELD SYSTEM

The Reference Design specifies an on-axis magnetic field of 5.3 T, which is to be produced by 12 superconducting TF coils with a horizontal bore of 6.2 m and a vertical bore of 9.8 m. This combination of number of coils and coil size yields a magnetic field ripple of 1.5% at the plasma edge. The TF coils are assumed to be a pure tension D-shape

design with a trapezoidal cross section. The centering forces on the coils are reacted through a bucking cylinder. The major technology needs associated with the TF system may be summarized as follows:

- (1) Production of Nb_3Sn conductor and manufacture of coils without strain degradation of properties.
- (2) Designs for coil cooling that will avoid collections of helium bubbles and/or vapor in the liquid helium, which would result in degradation of the heat transfer.
- (3) Means of coil protection, including the detection of small normal zones, quench pressure in forced flow channel, or quench voltage in bath-cooled coils and the control of coil currents or fault forces.

1.9 RF SYSTEM

The use of rf heating around 120 GHz has been selected to aid plasma breakdown. Experimental substantiation of this assumption is critical and should be vigorously pursued because of the large potential cost benefits. Microwave power on the order of 1 MW is suggested for deposition in the plasma region for ~ 2 sec. Relatively efficient waveguides and flush-mounted arrays can be designed to accomplish the power transmission. The major R&D areas that should be pursued include:

- (1) Development of power tubes to achieve 200 kW cw per tube at 120 GHz.
- (2) Development and component testing of the elements required to transfer the rf power from the sources to the plasma, including oversize waveguides, modal filters, materials that can tolerate the reactor environment, pressurization windows, and antennas.

1.10 INSTRUMENTATION AND CONTROLS SYSTEM

This system is required to automatically control the device through all its phases of operation. Instrumentation is provided for controlling functions and for diagnostics. The data from the instrumentation are

processed and either displayed or stored for later on-call display. The philosophy of operation is that of semiautonomous subsystem controls and diagnostics operating under the supervision of a central computerized control. A significant emphasis in the development of this system must be the necessary designs for fail-safe operations and degraded mode options.

Table 1.1 is a list of major parameters that may need to be measured, along with some sensing concepts that should be studied for use in the reactor environment.

Table 1.1. Parameters to be measured
and candidate sensing concepts

Parameters	Candidate sensing concepts
Plasma density as a function of time and plasma radius	Thomson scattering, microwave absorption
Plasma location	Inductive pickup loops
Average plasma current	Rogowski coils
Plasma ion and electron temperatures as functions of time and plasma radius	Thomson scattering, soft x-ray detectors
Magnetic field strengths	Inductive pickup loops
Neutron flux at first wall	Activation foils, collimating tube and detectors

Future investigations must include more study of the problems of disruptions, aborts, and shutdown and of the difficulties of defining sensors and instrumentation systems that can survive the reactor environment and meet the accuracy and response demands of the controls and diagnostics. Thus, the primary R&D needs include:

- (1) Development of a hybrid computer instrumentation and controls (I&C) simulation, benchmarked with experiments and expansion to long pulse calculations and experiments.
- (2) Development of instrumentation that can operate in the expected reactor environment.

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2. S. L. Milora et al., *Hydrogen Pellet Fueling Experiment on the ISX-A Tokamak*, ORNL/TM-6496, Oak Ridge, Tennessee (October 1978).
3. P. B. Mohr et al., *Tokamak Blanket Design Study: FY 78 Summary Report*, ORNL/TM-6847 (1979).

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2. PHYSICS R&D NEEDS

This section discusses the R&D needs in fusion plasma physics to ensure successful plasma operations in a device like the TNS Reference Design. This discussion points to the additional knowledge and understanding required to reduce the uncertainties in our physics assumptions, extrapolations, and theoretical calculations to a level acceptable for economy and risk prevention. The needs discussed here are not compared with specific present and planned physics experiments in the magnetic fusion community. However, some preliminary evaluation is included to highlight those physics areas where serious gaps are considered most likely to exist.

Recent advances in tokamak plasma confinement and heating experiments have resulted in reactor level temperatures and densities in separate devices. These have provided optimism concerning the prospects for producing a controllable fusion core based on the tokamak concept. It should be emphasized that uncertainties still exist in essentially all of the physics assumptions adopted in the Reference Design. Our discussion is concerned with the presently conceived Reference Design and approaches to plasma operation. However, uncertainties of a more general kind also exist; because our experiences in tokamak operations are limited to physics oriented experimental devices, it should not be surprising that an engineering oriented reactor device may dictate approaches drastically different from our present conceptions.

For convenience, the physics issues are divided into five categories, ordered according to increasing levels of uncertainty: plasma confinement and optimization, plasma heating, fusion ignition and burn, dynamic scenarios of operation and control, and impurity and particle control.

2.1 PLASMA CONFINEMENT AND OPTIMIZATION

This area deals with the achievable plasma temperature, density, and confinement times, and provides estimates of the plasma size, TF strength, and heating power required for ignition and burn. Two areas are of interest here: confinement scaling and stability optimization.

2.1.1 Confinement Scaling

Recent tokamak confinement experiments have indicated that the ion heat conduction is nearly neoclassical, while the electron heat conduction seems to scale empirically like $n^{-1}T^{-\alpha}$, where α apparently ranges from 0 to 1. Particle diffusion seems to behave similarly to the empirical electron scaling. Without an understanding of the processes leading to the empirical scaling laws, the extrapolation of these laws to the TNS Reference Design would involve a significant level of risk. The uncertainties, although they may not be critical, may result in a wide envelope of device parameters. Continued experimental and theoretical studies of the tokamak confinement processes are called for to ensure an economical design with acceptable risks.

2.1.2 Stability Optimization

The concerns about the effects of trapped particle instabilities have decreased because of recent results on PLT that indicate nearly neoclassical ion energy confinement in reactor level collisionality regimes. MHD instabilities due to the presence of minute plasma resistivity have been observed in many tokamak experiments. They are seen to have sensitive dependences on the profiles of current and safety factor. At high levels of instability, they lead to degradation of plasma confinement and eventually to sudden disruptions of the plasma current. While significant progress has been made in understanding the properties and consequences of resistive MHD instabilities, the scaling to reactor level plasmas is by no means understood.

It is generally assumed that the pressure driven, ideal MHD instabilities will place a stringent limit on the attainable plasma beta. The magnitude of beta has a dramatic impact on the size, the field strength, and hence the cost of a reactor. Understanding the dependence of the beta limit on parameters such as cross section shape, plasma profiles, and aspect ratio is of great importance to the Reference Design. The assumed value of $\beta = 7\%$ is consistent with the latest calculations and is presently considered adequate for marginal reactor economy. A significant increase in the beta limit is highly desirable.

The designed pulse length of 500 sec is expected to be comparable to the plasma resistive time scale. The evolution of high beta plasmas over this time scale is one of the topics on which work is still needed.

2.2 PLASMA HEATING

Neutral beam heating with 50 MW of 150-keV deuterium beams for 6 sec is calculated to be adequate to ignite a clean plasma in the Reference Design. The recent success of the PLT experiments in reaching near ignition ion temperatures has increased the confidence in this assumption. However, the success of using 150-keV beams depends on the success of the heating scenarios proposed: expanding radius heating or low density heating. Early experimental verification of these heating scenarios is desirable to determine whether higher energy beams are needed.

In comparison to neutral beam heating, the physics of rf heating is presently less well understood but has the potential for more economical and convenient applications to a reactor grade tokamak; rf heating may also be used to supplement neutral beam heating for increased reliability and reduced cost. The optimal combination of rf heating and neutral beam heating needs to be explored experimentally.

2.3 FUSION IGNITION AND BURN

Achieving $Q = 1-2$ in TFTR will provide a demonstration of the approach to ignition when the fusion alpha particles become the dominant heating source near the center of the plasma. The information obtained, while extremely valuable for understanding the ignition process, may not be adequate for reducing the uncertainties in extrapolating to ignition ($Q \sim 5$) and burn ($Q \rightarrow \infty$) to an acceptable level. The process of ignition and burn may depend strongly on the electron and ion temperatures, the plasma density, the fusion alpha particle power deposition process, and the D-T composition. Experimental study of near ignition processes should not only help confirm the confinement scaling laws and the heating scenarios, but also provide information on optimal regions of plasma parameters for ignition.

A difficult question has to do with the level of near ignition to be achieved before the obtained information can be considered adequate for extrapolation to ignition and burn. If demonstration of full ignition is not required for this, it may become desirable to improve the performance of TFTR and the Joint European Torus (JET) to approach $Q \sim 5$ for as long a pulse as possible. Detailed numerical modeling and simulation of the experimental results are needed to bridge the gap between present estimates and the actual demonstration of ignition and burn.

2.4 PLASMA OPERATION AND CONTROL

It may not be overly optimistic to expect that an adequate understanding of the three categories discussed above will be obtained in the early 1980s. However, it is possible that plasma operation through the phases of startup, heating, ignition, burn, and shutdown in a reactor grade tokamak will be drastically different from our present conceptions, which are based on physics oriented devices. Desirable operating scenarios may be discovered through a concerted study of the reactor engineering, technology, and physics concepts, but the scenarios thus proposed must be experimentally verified before they are applied to reactor grade plasmas.

Closely coupled to the plasma operation are the plasma control requirements along the desired path in the parameter space during the pulse. Critical areas include fusion thermal runaway, plasma disruption, and the pulsed fields at the TF coils. Lack of clear definition of the requirements in these areas may present overly stringent requirements on engineering and technology, making a reactor grade tokamak unacceptably costly and unreliable.

2.5 IMPURITY AND PARTICLE CONTROL

This is presently the weakest area of physics understanding. The area deals with impurity production at the plasma-wall interface and with impurity transport through the plasma edge/scrape-off region and in the divertor chamber. Planned experiments will study various impurity

control methods both with and without divertors. Significant improvements in our understanding of this area are expected. However, these experiments are limited to short pulse times (~ 1 sec) at relevant power levels (~ 100 W/cm²) and are not likely to provide adequate information for a design to operate in a clean, long pulse, high beta, D-T fusion plasma, as assumed in the Reference Design.

Bundle divertors have been chosen for the Reference Design because of their potential engineering advantages, the desire to scope the requirements of bundle divertor technology, and the lack of quantifiable estimates of impurity control requirements. This has led to the design assumption in the TNS study that the bundle divertors offer sufficient flexibility to permit the evolution of a workable reactor grade impurity control method. The bundle divertor may extract a relatively small flux bundle from the plasma edge. This can be an attractive feature in the event that only fractional efficiency in particle unloading, impurity screening, and heat unloading is required for reactor impurity control and ash removal. However, the bundle divertor may introduce field ripples that degrade the confinement properties of the tokamak.

Nondivertor impurity control approaches should also be pursued actively because of their intrinsic simplicity. However, these approaches are likely to be inflexible and may not apply to long pulse operations because of helium and impurity buildup. It may be premature to rule out the possible usefulness of impurity control without divertors, but we cannot rule out the need for some divertor oriented impurity control method for long pulse, reactor grade plasmas. Research and development of divertor physics and technology should be pursued actively to provide the data base needed for proper reactor design and operation of impurity control schemes.

A closely related topic of interest is fueling. Recent pellet fueling experiments on ISX-A have demonstrated the feasibility of this technique, although vigorous efforts are needed before reactor application can become a reality. Combined with gas puffing, pellet injection may offer the needed flexibility in particle control in diverted reactor plasmas.

2.6 CRITICAL ISSUES

On the basis of these discussions we can identify critical physics issues for the Reference Design. They include:

- (1) Long pulse behavior of impurity production and transport under high thermal power in diverted and nondiverted plasmas.
- (2) Long pulse evolution of high beta MHD equilibria through substantial flux diffusion and influenced by fusion alpha particle heating and various fueling sources.

We can also identify areas where some understanding can be developed via theoretical analyses and numerical simulations of low Q experimental results. However, experimental studies in these areas would increase our confidence in the fusion plasma operation in the Reference Design. They include:

- (1) The physics of fusion ignition and burn ($Q > 5$) for a time scale much longer than the energy confinement time.
- (2) The physics of helium ash accumulation and transport in diverted plasma.

2.7 ISSUES SPECIFIC TO THE REFERENCE DESIGN

Finally, the TNS physics/engineering studies during FY 1978 have suggested schemes that could favorably impact the reactor engineering and technology requirements. These studies point to the following desirable experimental verifications:

- (1) Experiments to optimize the neutral beam injection process (such as the low density and small radius scenarios) that directly or indirectly corroborate the process of heating to ignition with beam energies of ~ 150 keV.
- (2) Fueling experiments with partial pellet penetration.
- (3) Experiments with a versatile PF coil system and superconducting TF coils to study ways to minimize the pulsed field during startup and disruption.

- (4) Experiments with the use of plasma heating sources before and during current startup to increase the startup time and to reduce costs.
- (5) Experiments with variable field perturbations (such as ripple in the toroidal field) as a method of controlling the runaway electrons and the plasma edge condition and of improving access to the plasma core.

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3. DIVERTOR SYSTEM

3.1 TECHNICAL NEEDS ASSESSMENT

Study in this area has concentrated on the bundle divertor option. (The poloidal divertor approach was studied in some depth last year.) Bundle divertors have been emphasized primarily because they remove the ion and energy collection functions to a region outside the space bounded by the TF coils.

In the consideration of previous bundle divertor concepts, a review of the data on solid ion absorbers (titanium and zirconium) suggests that the use of such absorbers in reactors is unlikely because of (1) the anticipated degradation of thermal and mechanical properties with cyclic saturation and regeneration and (2) the excessive times and temperatures to regenerate these absorbers. The bundle divertor concept proposed for the Reference Design is a system involving liquid lithium droplets traveling through the volume of interest, which can obviate these problems. The proposed concept is described in Sect. 3.2.

The technical needs that must be addressed for TNS are summarized below.

3.1.1 Bundle Divertor Coils

The use of larger coils is desirable, but magnetic field perturbations near the central region of the plasma must be minimized. Studies of configurations to fulfill these conditions are needed.

The plasma guided by the diverted flux bundle is expected to produce sharp peaking of surface heat loads. Methods must be found to reduce this peaking in the target region as well as to expand the field generally.

Ceramic insulation must be developed for water-cooled copper coils, based on studies showing that the shielding for superconducting coils and/or the organic insulation for regular coils tend to close up the ion flow passage through the bundle divertor coils.

3.1.2 Ion/Energy Collection

The present limit on the temperature range where data are available on the collection of hydrogen isotope ions by liquid lithium is $\sim 350^\circ\text{C}$. This must be extended to higher lithium temperatures.

The ability of the proposed concept to retain embedded alpha particles must be addressed. This will involve defining the concept for pumping the collected droplets and measuring the collection and removal of ions. Also, the ability of the liquid lithium to pump neutral hydrogen isotopes must be ascertained.

An approach that uses cold walls to retard backstreaming of lithium into the plasma region has been suggested; the merits of this approach must be determined.

Experiments should be conducted on the passage of droplets through nonuniform magnetic fields.

Finally, the data on solid absorbers should be extended for possible use in TNS, where full reactor requirements would not be imposed.

3.1.3 Computational Scheme

A computational model should be developed to optimize divertor coils so that perturbations of the magnetic fields can be minimized while the necessary nulling and reversal of the toroidal magnetic field are accomplished. This might be an interactive scheme involving cathode ray tube (CRT) display or mathematical minimization or maximization schemes.

3.2 PROPOSED BUNDLE DIVERTOR SYSTEM

3.2.1 Function

The divertor system (1) minimizes the interaction between plasma particles and the walls of the plasma containment vessel and (2) diverts and collects impurities emanating from the walls. As a consequence, it is necessary to collect the diverted particles and to handle the heat load associated with these diverted particles. For design purposes, it

has been assumed that the upper limit on the heat load to the divertor is equivalent to the alpha particle energy. The actual heat load to the divertor is estimated to be 50% of the alpha particle energy, with the remaining 50% absorbed by the first wall. The divertor system parameters are listed in Table 3.1.

Table 3.1. Divertor system parameters

Geometric	
Number of bundle divertors	1 operating, 1 or 2 on standby or being cleared of deposited lithium
Divertor opening	0.4 m
Shield thickness	0.4 m
Flux bundle expansion ratio	90
Performance	
Ripple at plasma centerline (due to divertor)	3%
Plasma edge particle loss rate (energy transport)	3×10^{23} particles/sec
Plasma fraction recycled at edge as neutrals	85%
Maximum heat load (design value)	226 MW
Actual heat load	113 MW
Electromagnetic	
Coil type	Water-cooled copper
Ampere-turns	12×10^6
Thermal	
Heat flux to divertor collector	10 MW/m ²
Divertor collector surface area	22.6 m ²
Lithium temperature rise	114°C
Divertor coil coolant	Water
Divertor coil ohmic heating	70 MW
Shield coolant	Water
Particles pumped by divertor	4.6×10^{22} particles/sec

3.2.2 Design Description

The bundle divertor shown schematically in Fig. 3.1 is the type chosen for the Reference Design. It is composed of two coils, side by side, with opposite current flows that provide a strong short-range magnetic field, which is superimposed on the toroidal field in the plasma to produce zero field close to the external surface of the plasma. The resulting field distortion allows a bundle of field lines to be extracted and form a loop external to the plasma along which charged particles trapped on the lines can travel.

Fig. 3.1. Bundle divertor.

In a reactor environment, the divertor will be subjected to a high level of radiation, and some shielding (on the order of 1 m for superconducting coils) will be required to protect conductors and insulation from cumulative damage. This shielding requirement leads to oversize divertor coils, which create excessive perturbation of the field lines in the plasma. However, divertor coils of the proper size cannot be adequately shielded. A shield thickness of 0.4 m appears to be a

feasible compromise. This thickness precludes the use of superconducting and normal cryogenic coils but permits the use of water-cooled copper with a shortened life.

The divertor will expand the diverted magnetic field to reduce particle and energy flux. For the purpose of design, it is assumed that the divertor must absorb all the power associated with the alpha particles (226 MW). This power is assumed to leave the plasma through a particle loss rate at the plasma edge of 3×10^{23} particles/sec. It is further assumed that 85% of these particles recycle as neutral gas. The number of plasma particles that the divertor is required to pump is therefore 4.6×10^{22} particles/sec.

For a solid absorber, the getter material must either constitute or be a part of a heat exchanger, which requires material having good strength and thermal conductivity, since it will be subjected to at least 10^5 cycles of absorption and desorption. Consideration of data on hydrogen embrittlement of structural materials (for one absorption cycle) suggests that this will seriously degrade the material. Also, the times and temperatures for regeneration of getters ($700-1000^\circ\text{C}$ for several hours) tend to indicate infeasibility. Therefore, these functions will be accomplished by an array of lithium jets, shown schematically in Fig. 3.2. For the absorption of a heat flux of 10 MW/m^2 , there must be 22.6 m^2 on which the expanded, diverted magnetic field impinges. The expansion requirement leads to a system that has much in common with the direct conversion approaches proposed for advanced fuel concepts.

Fig. 3.2. Lithium jet energy collection system for bundle divertor.

4. PELLETT FUELING SYSTEM

4.1 TECHNICAL NEEDS ASSESSMENT

Pellet injection has been chosen as the fueling technique for the TNS Reference Design. The requirements for pellet injection systems result from the plasma characteristics and their control over the penetration depth, along with the attendant particle spatial and temporal deposition. Studies and R&D needs in these areas are discussed in Sect. 2.

Specific requirements for the pellet composition, velocity, and injection rate will be established by the burn scenario. Thus, a primary area of concern for the pellet injector system is the pellet characterization, which includes pellet formation, size, composition, and thermal and mechanical properties. Early definition of these requirements is important so that their impact on injector design can be taken into account and the ability of the candidate ideas to meet these needs can be evaluated.

The pellet injector concepts that now offer the most promise of meeting the needs of a fusion reactor are the pneumatic gun, the pressurized fluid injector, and the centrifugal slinger. It seems desirable to investigate all three concepts in increasing depth, since none has been advanced much beyond laboratory studies and tests.

The pneumatic gun concept was used in a short feasibility test on ISX-A in which pellets were injected into the plasma with encouraging results. Experiments are planned on ISX-B to assess the injection performance more fully. The recent study and development test results strengthen the consideration of the pneumatic gun as a candidate injection system. The fluid injector ideas need a different perspective in some respects, because the pellet formation is a more integral part of the actual injection technique. The centrifugal slinger approach has been selected for the TNS Reference Design, but it should be noted that the recommended R&D effort includes consideration of the other leading candidates as well, because all developments are in such an early stage that firm choices cannot be made with confidence.

These concepts have been studied in light of recent results from fueling calculations that have increased confidence about finding scenarios that require pellet velocities of ≤ 1000 m/sec. All three concepts show promise of being able to meet these velocity requirements.

In addition to the physics issues discussed in Sect. 2, the technical needs in pellet injection for TNS are highlighted by three areas of recommended action:

- (1) Evaluation, by calculation and experiment, of the potential of alternate concepts of pellet injection, such as gas and mechanical methods, to provide the frozen D-T pellets at a controlled velocity and rate of delivery to the plasma.
- (2) Expansion of the data base on pellet formation and handling and on mechanical and thermal characteristics under injection conditions.
- (3) Intensification of efforts to achieve equipment design innovations and simplifications that will lead to enhanced equipment reliability, fabricability, and cost effectiveness.

4.2 PROPOSED FUEL INJECTOR SYSTEM

4.2.1 Function

The fuel injector injects D-T fuel in the form of frozen pellets to replenish any fuel lost or consumed during a burn, thus maintaining the plasma characteristics consistent with the fusion process. The pellets must be properly sized and delivered with sufficient velocity to penetrate the hot plasma to an appropriate depth. The fuel injector parameters are given in Table 4.1.

4.2.2 Design Requirements

The injector must be shielded from electrical and magnetic fields in a manner that provides unimpaired operation during steady state or in transient environments. The unit electrical circuits must be protected from fault or induced electrical overloads by adequate fuses or circuit breakers.

Table 4.1. Fuel injector parameters

Steady-state burn	500 sec
Cycle time	560 sec
Fueling rate (during burn)	0.185 g/sec
Pellet shape	Spherical
Pellet diameter	~0.5 cm
Pellet velocity	1000-2000 m/sec
Repetition rate	16 pellets/sec.

Operating temperatures of units of the injector must be controlled by passive or active thermal control systems. The recommended cooling media are liquid nitrogen (LN₂) and liquid helium (LHe).

The injector must have adequate shielding to limit neutron activation of components. Modular designs for components or parts subject to radioactivity or tritium products will provide for ease in maintenance.

There must be no particles from moving mechanical assemblies, lubricants, or outgassing products that could migrate to the vacuum vessel. Vacuum pumps maintain the injector enclosure at a pressure equal to or lower than that of the vacuum chamber and exhaust evaporated D-T products and gaseous products.

4.2.3 Design Description

The fueling method chosen for the Reference Design is the injection of pellets of solid deuterium and tritium. Recent theoretical studies indicate that pellet penetration may be reduced to about 0.2 times the plasma radius, compared to earlier estimates requiring penetration to the plasma centerline. This may result in reductions of pellet velocity to as low as 1000 m/sec.

The fuel pellet injector mechanism provides the desired injection velocity by exerting centrifugal force on a fuel pellet. A rapidly rotating disc spinning on a vertical axis captures the fuel pellet, which is introduced to the upper face of the disc close to the central hub. The pellet is accelerated along a formed track mounted on the

disc's upper face by centrifugal force and leaves the disc through a tube connected to the plasma chamber. The rotational speed of the disc, the disc diameter, the track geometry, and the pellet characteristics govern the ejection velocity and trajectory. Principal subassemblies of the injector mechanism are the rotating disc with its drive, the device that forms the solid (frozen) pellets from deuterium and tritium gases, and the pellet cutter/deliverer that propels the D-T pellet to the impeller.

The entire assembly is housed in a vacuum-tight container. A vacuum-tight tube connects the main unit to the plasma chamber and serves as a clear passage for pellet entrance to the plasma chamber. While the pellets are being injected, the pellet injection device is subjected to direct thermal radiation and neutron streaming. The entire vacuum-tight housing and the interfacing subsystems are shielded from possible radioactivity in the vicinity of the device for personnel protection.

The pellet injector requires these supporting components and subsystems: vacuum pumping, liquid helium supply and plumbing, electrical power, instrumentation and controls, basic structure to support injector housing and passive shielding, and filtration and recycling for handling misfired pellets or gaseous tritium products. Figure 4.1 illustrates the functions of the system.

The injector and major subsystem components will be assembled as a single removable unit, with provisions for remote removal and capping of subsystem lines when the pellet assembly is removed for service. The injector will be designed in functional modules that facilitate disassembly and repair. The housing in which moving parts are enclosed must contain any component parts that fail and must protect tritium system and lines from damage or loss of tritium from the injector system.

Two or more pellet injector systems will be installed with one unit always in standby mode. Suitable instrumentation and controls will warn of malfunction in the operating unit and switch the standby unit into operation.

Fig. 4.1. Fuel pellet injector system.

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5. NEUTRAL BEAMS

5.1 TECHNICAL NEEDS ASSESSMENT

5.1.1 Introduction

Neutral particle injection is to be used in tokamak fusion devices to reach the high temperatures desired for physics studies in the vicinity of ignition and for ignition in reactors. Ohmic heating (OH) techniques are not expected to provide ignition temperatures in plasmas and must be assisted by neutral beams (and/or by rf heating, if it is shown to be feasible).

Previous considerations pointed to the expected need of high energy beams to penetrate to near the plasma center. Present calculations seem to support the systems advantages of using injection scenarios that call for moderate energy (100- to 150-keV) beams. This avoids the use of high power neutral beams with particle energies greater than 200 keV, which are not readily available. The efficiency of such high energy, positive ion source beam lines becomes quite low (<20%) and requires large quantities of power. Negative ion sources, which have the potential for greater efficiencies, are in the early stages of development and are five years or more behind the positive ion sources in possible applications. The system needs growing out of these data have provided the impetus to evolve injection scenarios which may avoid the need for high energy beams, such as small radius startup, low density, full radius startup, etc.

Direct recovery of the energy in the unneutralized particles in a beam line is being developed and is considered a part of the TNS Reference Design. Such use is expected to allow the achievement of power efficiencies over 50% in the 120- to 150-keV positive ion source beam lines. The TNS Reference Design neutral beam system is described in Sect. 5.2.

5.1.2 Status of System and Equipment Development and Availability

Five key elements of the neutral injection system have been selected. This discussion evaluates their status and readiness for fabricability and reliable, cost-effective operation.

The key elements (see Fig. 5.1) are:

- (1) the ion source,
- (2) the power supplies,
- (3) the vacuum components and pumping system,
- (4) the controls and diagnostics, and
- (5) the direct recovery system.

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Fig. 5.1. Status of key elements of neutral beam injection system.

The source is further divided into two options, a H^+D^+ source and a H^-D^- source, even though the TNS Reference Design now shows the positive ion sources. The power supplies can be divided into two groups: those for the arc and source and those for particle acceleration. The drift tube is considered with vacuum components and pumping. The area of controls and diagnostics is considered of equal importance with the other elements, especially in the context of multiple injection operation. Direct recovery, which will play a key role in the TNS neutral injection system, is also deemed important enough for separate evaluation.

The status descriptors used in the figure are defined as follows:

- Development refers to the state of scientific and technical feasibility.
- Producibility refers to how ready and suitable the design concept is for detailed design and quantity fabrication of a reliable device.
- Maintainability refers to how well the potential design concept would perform under the remote installation and maintenance procedures dictated by TNS operation.

These status descriptors are general and therefore somewhat overlapping. However, it is felt that they can serve as an aid to an orderly consideration of unsolved problems and to the determination of development needs.

The key technical needs that appear necessary from this evaluation and from other studies and considerations include:

- (1) Beam line development for higher currents (>100 A), longer pulses (>10 sec), and better species control (90%/10%) with positive ion sources at energies of ≤ 150 keV.
- (2) Direct recovery techniques with efficiencies of 80%.
- (3) Controls and diagnostics for multiple beam systems.
- (4) Developments for cost-effective, reliable operation and ease of fabrication and maintenance.

It seems evident, relative to the last item, that an increased effort or new effort is required to seriously consider how these elements and the

entire system can be developed to be less complicated and expensive and more reliable, producible, and maintainable. Such an activity would be very valuable in providing active feedback to the development and conceptual design efforts.

5.2 PROPOSED NEUTRAL BEAM HEATING SYSTEM

5.2.1 Function

The bulk heating system raises the tokamak plasma temperature from that achieved by rf assisted ohmic heating to the ignition level. For this purpose, neutral beam injection has been chosen. The system parameters are listed in Table 5.1.

5.2.2 Design Requirements

The design requirements for the bulk heating system call for a total injected power of 50 MW with a beam energy of 150 keV. The pulse length is ≤ 6 sec and the duty cycle is 1%. The system is designed to last for the 10-year lifetime of the machine.

5.2.3 Design Description

Because OH techniques alone are not adequate to provide ignition temperatures in plasmas, they must be assisted by neutral beams and/or by rf heating. The Reference Design uses four beam lines, with three sources per line, injecting through four rectangular apertures into the plasma. Each beam line includes a positive ion source, a neutralizer region, a direct energy recovery system, and a drift region to connect the beam line to the torus. The total electrical efficiency of the beam line is estimated to be 51%. For the Reference Design plasma, a full bore, low density startup is feasible. This scenario requires injecting 50 MW of 150-keV beams into the plasma for approximately 6 sec. Direct recovery of the energy in the unneutralized particles in a beam line is being developed. This technique should allow the achievement of efficiencies higher than 51% in the 150-keV positive ion source beam lines.

Table 5.1. Bulk heating system parameters

Structural/configurational	
Sources per beam line	3
Total sources	12
Number of beam lines	4
Size of beam apertures into torus	0.4 m x 1.2 m
Injection angle (perpendicular to plasma surface)	$\geq 16^\circ$ (in the direction of plasma current)
Performance	
Source current	100 A
Full energy power/source	3.9 MW
Half energy power/source	0.4 MW
Total power/source	4.3 MW
Total full energy injected power capability	46.8 MW
Total injected power capability	51.6 MW
Total power for particle acceleration	101 MW
Beam energy	150 keV
Full/half energy	0.90/0.10
Neutralization fractions (full energy/half energy)	0.33/0.70
Geometric transmission	0.88
Pulse length	≤ 6 sec
System power efficiency	51%

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6. POLOIDAL FIELD SYSTEM

6.1 TECHNICAL NEEDS ASSESSMENT

The PF system for the TNS Reference Design is described in Sect. 6.2. The selection of the systems approach and of the elements in the system was guided largely by the desire to minimize necessary developments. The key elements that make up this system are:

- (1) the rf preheating system,
- (2) the superconducting exterior OH and equilibrium field (EF) coils,
- (3) the interior copper EF coils,
- (4) the power supplies, including
 - transformers,
 - 12-pulse rectifier/inverter (thyristor bridges),
 - power supply controller, and
 - auxiliary equipment,
- (5) the motor-generator flywheel (MGF) sets, and
- (6) the electrical distribution systems.

The last three elements have many similarities to state-of-the-art equipment and to equipment being supplied to TFTR, Doublet III, and JET. Therefore, no significant R&D is considered necessary to ensure feasibility or to determine a proof of principle. However, this is not the case for the first two items.

The 120-GHz, 1-MW rf preheating system can be considered an extrapolation from the technology advancement of the 28-GHz system being developed by Varian Associates for the ELMO Bumpy Torus (EET) program at ORNL. The specific R&D needs of the TNS rf system are discussed in detail in Sect. 10.1. Theoretical physics calculations of the rf preheating option show a promise of reducing the required startup volt-seconds by 18 V-sec and the probability of increasing startup time from 2 to 6 sec. This has significant ramifications in both the cost and the complexity of the device. The necessary substantiation of these calculations is discussed in Sect. 2. Without rf preheating (or some other less well-defined physics approach) to allow the longer startup time,

the PF system cost is expected to increase by \$85 million ($\sim 17\%$ of total device cost), with the additional complication that an interruption scheme will be required. The capability for an extended burn will also be significantly impaired by the absence of rf as the startup requirements consume more volt-seconds. Finally, by permitting the use of a lower plasma loop voltage, rf preheating reduces the requirement of electrical breaks in the plasma vacuum chamber and further simplifies the device construction.

Los Alamos Scientific Laboratory (LASL) is planning the development of a 20-MJ superconducting pulsed energy storage coil to operate in a full bipolar mode and swing from 7 T to -7 T in 1 sec. This field swing rate ($\dot{B} \sim 14$ T/sec) is much more stringent than the $\dot{B} \sim 1.33$ T/sec specified in the TNS Reference Design. Although the coil energy of the LASL coil is significantly lower (20 MJ vs 750 MJ), the energy is a reflection of the coil size and should not impact the feasibility of the concept. The remaining coil parameters, such as eddy current loss, refrigeration, total current, and mechanical stresses, will also be addressed by the LASL program. Considerations involving the generation of design data on cryogenic properties of structural and insulation materials will be developed in the course of the work being performed on both the LASL program and the ORNL Large Coil Program (LCP).

The ISX-B program being performed at ORNL incorporates the same coil use philosophy and is considered to be the experiment from which the overall PF system approach will be demonstrated and the proof of principle substantiated.

6.2 PROPOSED POLOIDAL FIELD SYSTEM

6.2.1 Function

The PF system (1) induces a current in the plasma, (2) maintains the plasma columns in equilibrium both horizontally and vertically, and (3) protects the superconducting TF coils from magnetic flux changes. The parameters of the PF system are shown in Table 6.1.

Table 6.1. PF system parameters

Primary OH coil

General

Life, t_L	10 years
Duty cycle, dc	96%
Inside radius of solenoid, R_I	0.875 m
Outside radius of solenoid, R_{OS}	1.13 m
Length of central core, L_S	10 m
Number of turns (total), N_{oh}	2771

Mechanical

Maximum hoop stress, σ_S	20,000 psi
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Electromagnetic

Ampere-turns	64 MAT
Maximum field at winding, B_{Smax}	8 T
Field at coil axis, B_{Sax}	8 T
Induced plasma current (by OH system), I_p	4 MA
Maximum rate of flux change in the OH coil, $\dot{B}$	1.3 T/sec
Coil swing time (full forward to zero current), t_s	6 sec
Coil volt-seconds (OH system)	59 V-sec
Stored energy, W	1.1 GJ
Coil voltage (maximum), V_C	25 kV
Maximum coil current/turn, I_C	23 kA
Conductor	NbTi
Current density overall, J_{OA}	2.5 kA/cm ²
Self-inductance (single turn), L_{oo}	0.37 μ H

Thermal

Conductor coolant	Helium
Total refrigeration required, P_{ref}	TBD ^a

Table 6.1 (continued)

EF coil		
General		
Lifetime, t_L		10 years
Duty cycle, dc		94%
Number of coil systems, N_{sh}		6
Electromagnetic		
Ampere-turns		41 MAT
Coil volt-seconds		24 V-sec
Stored energy, W		2.6 GJ
Current/turn, I_{ac}		23 kA
Conductor material		
Normal conductor		Cu
Superconductor		NbTi

^aTBD = to be determined.

6.2.2 Design Requirements

The PF system has a 10-year design lifetime for the main components and allows remote maintenance and replacement of the coils. Operating with rf preheating, the OH and EF coil systems will be capable of supplying 53 V-sec during startup, inducing a plasma current of up to 4 MA. Without rf preheating, 70 V-sec will be required. The OH system will also have sufficient volt-seconds capacity to provide a relatively long burn pulse, which would be adequate for viable reactor operation. The EF coils will supply sufficient vertical field to hold the plasma current in equilibrium during the burn phase.

6.2.3 Design Description

An air core, hybrid PF system incorporating rf assisted startup was selected for the FY 1978 Reference Design. The PF system consists of a superconducting OH coil, interior (inside the TF coil bore) normal copper EF coils which carry 35% of the EF current, and exterior (outside the TF coil bore) superconducting EF coils which carry 65% of the EF

current. The name "hybrid" is used to identify the mix of interior and exterior EF coils.

The size of the poloidal field bore for the Reference Design was taken from system studies which identified the geometric characteristics (i.e., the plasma size and major radius) for a minimum cost ignited tokamak reactor.¹ The OH system was sized for maximum utilization of the poloidal bore and for operation at a maximum field of 8 T. The PF system will supply 83 V-sec when operating with a half-biased OH system swinging from +8 T to -8 T, which is adequate for startup and 500 sec of burn.

Placement of the EF coils is important from the standpoint of maintaining plasma equilibrium and shape during the heating and burn phase, influencing power consumption, protecting the TF coils against plasma disruption, and servicing the coils. In the Reference Design, the EF coil system consists of six coil systems, named primarily for their locations with respect to the TF coil bore and their rough proximity to the plasma: interior inside coils (II), interior D-shape coils (ID), interior outside coils (IO), exterior inside coils (EI), exterior D-shape coils (ED), and exterior outside coils (EO). The PF system is shown in Fig. 6.1. As noted, the EF coils inside the TF coil bore are normal copper and those outside are superconducting. This arrangement of interior and exterior coils can maintain the plasma position and D-shape for a wide range of beta values.

Because of the intrinsic engineering difficulties with the assembly and repair of interior coils in the radioactive environment of a D-T tokamak reactor, an effort was made to reduce the number of turns of these coils. The 65%-35% split between exterior and interior EF coils offered a reasonable balance between coil maintenance, TF coil protection, and power requirements.

The time behaviors of the EF and OH coil currents are shown in Fig. 6.2, based on rf preheating during startup. The rf assisted startup was chosen because a substantial reduction is realized in the startup loop voltage and resistive volt-second losses, along with an increase in the startup time relative to operating without a rf preheating system. The rf preheating is supplied by five 200-kW gyrokystrons, which deliver 1 MW to the plasma during the 6-sec startup period.

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Fig. 6.1. PF coil locations.

Fig. 6.2. PF coil current waveforms.

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7. MECHANICAL SYSTEMS

7.1 TECHNICAL NEEDS ASSESSMENT

7.1.1 Introduction

One of the TNS objectives, as stated in Ref. 1, is to provide a near-term means of focusing the efforts of the national fusion program, which seeks an economically viable fusion reactor concept. In this light, the assessment of the TNS mechanical systems has four considerations:

- (1) What are the tasks to be done?
- (2) Are those tasks in place; that is, does a technology base exist?
- (3) What are the time frames?
- (4) Does this represent a go or no-go situation regarding the total device feasibility?

The following generalizations can be made about mechanical systems as we examine each of the four considerations.

Tasks to be done

There are four major tasks to be accomplished:

- Define the mechanical/structural loads on all major components.
- Establish the operating environment of the major components.
- Define the useful life of the total machine and of its components.
- Design the mechanical systems to fit these criteria.

Technology base

Based on the current understanding of tokamaks and advanced reactor design studies to date, the technology base for these tasks does exist. Although it may be argued that the materials technology is lacking or that certain dynamic load phenomena are not fully understood, these objections really represent design obstacles which prevent an optimum

configuration: they force the configuration to be compromised in terms of useful life (i.e., number of pulses, neutron fluence, operating temperature, etc.).

Time frames

The time frames to overcome these design obstacles appear to be in place. A rigorous fusion materials program is under way. A variety of results from Doublet III, ISX-B, PDX, PLT, and TFTR, as well as data from the Tritium Systems Test Assembly (TSTA) and LCP, will be available in time for a TNS start date in 1984.

Go or no-go situation

A go situation exists for the start of TNS from a mechanical systems point of view because a viable technology base is in place. Optimization of the tokamak configuration is a mechanical design goal.

Again, these comments are generalizations for the R&D needs of mechanical systems. The following discussions for the vacuum vessel and limiters, machine structure, remote servicing/device disassembly, and shielding discuss in detail the four R&D considerations.

7.1.2 Toroidal Vacuum Vessel

A sufficient technology base exists for the design of the plasma chamber, and existing fabrication technology is adequate to construct such a vessel. Several areas, however, are not well understood and will result in conservative design criteria, detracting from an optimum reactor design. They are:

- plasma disruptions,
- allowable impurity levels,
- limiter materials, and
- materials behavior in a high temperature, high neutron flux environment.

Plasma disruption phenomena are just beginning to be studied for reactor-like power levels. The solution appears to be in controlling these dynamic thermal and mechanical loads through instrumentation. Other areas in need of further understanding or design experience have technology programs in place. They include the following.

- Data for radiation damage and mechanical properties at elevated temperatures for candidate materials are expected from fission reactor materials programs, and specific data will be generated from the fusion materials test programs, including TNS.
- Data for verification of materials performance under plasma conditions will be generated from ISX, PDX, P.T, and TFTR.
- Selection and subsequent demonstration of structural dielectric materials are expected from TFTR.
- Design and development experience for mechanical seals in lieu of welding for vessel penetrations (e.g., diagnostics) is expected from TFTR.

7.1.3 Machine Structure

The technology base exists for the design and fabrication of the overall machine structure. However, several areas require further development in order to achieve design optimization. These are:

- The mechanical properties of materials at cryogenic temperatures. This area is to be addressed by the National Bureau of Standards (NBS) at Boulder, Colorado.
- Radiation effects on materials properties at elevated temperatures. This area is being studied in materials test programs sponsored by the Department of Energy's Office of Fusion Energy and in fission reactor programs.
- Development of analytical methods for induced transient loading on structure and components. Liaison with the TFTR final design is required.

- Structural design criteria. While not necessarily a development need, these criteria require consideration before and during conceptual design. Definition of loading conditions, margins of safety, environmental safety, etc., will form the basis for design. Lack of established criteria will penalize the design with costly conservatism. Primary structural tests are not envisioned, provided the configuration evolves into a structurally determinant arrangement. National safety standards and criteria do not exist for these fusion systems but should be developed in the process of engineering TNS.

For a TNS to be built in the next 12 years, there are adequate structural materials. Because of this near-term goal, development and production of any new materials or alloys are unrealistic. Thus, the materials choices are austenitic stainless steels and perhaps some aluminum alloys.

7.1.4 Remote Servicing

The technology base for remote servicing concepts and equipment exists and is sufficient to support a TNS design. This technology is derived from the nuclear fission industry and the aerospace community and requires some adaptation for the unique problems of tokamak servicing. In particular, further development of remote handling concepts and equipment is needed to optimize device availability.

Several inputs are expected to support TNS, chiefly the TFTR development and demonstration during the TNS conceptual design period and the JET design experience through 1980-81. It is essential that TNS establish its own program of remote systems verification after the TNS conceptual design is completed. This can be accomplished with a series of mock-up systems, to be completed before a final design review. Specific areas to be investigated are handling of large components, remote cutting and welding, leak detection, and viewing systems. This TNS development program, coupled with liaison to ongoing TFTR development and demonstration, will provide a sound basis for the device design.

7.1.5 Shielding

The existing funded R&D programs for TNS provide an adequate base to initiate conceptual design. Uncertainty in the shield system design criteria will principally affect device cost (size), not schedule. However, if active impurity control requires a divertor, development of acceptable shielding for the divertor penetration could affect the present schedule. TNS final design will require some modifications and changes in emphasis; however, the basic R&D programs are in place. The development of a well-defined nuclear data base, substantiated with CTR-focused cross-section data experiments, suitable computational tools (including codes and input data), and an experimental program to validate calculated results, is essentially under way to provide a basis for optimizing shield design.

The CTR data base currently being assembled is generally focused on the needs of the fusion community and should be assessed with regard to specific TNS requirements. Radiation transport computer codes exist but will require some modification for application to toroidal geometry and for solution of radiation streaming. No suitable code exists for the calculation of accurate distributions of the radioactivity and after-heat; an evaluated activation data library and codes must be developed to access these data efficiently. These calculations are presently performed in many disjointed steps, and increased efficiency and shield optimization will result from an automated procedure. Integral experiments to study the materials and configuration of the TNS shield are funded and under way at ORNL. Future experiments to measure neutron streaming through penetrations in the shield are planned, and these studies will be consistent with the TNS design.

Timely judgments of both physics and technology understanding will be required throughout the design phase to provide design criteria for the shield system. Physics understanding will be required to provide design criteria for total plasma power, power distributions, and shield eddy currents. Impurity control R&D programs could identify additional requirements for the shield system if a divertor (with attendant penetrations) is found to be necessary. This physics input is closely

interfaced to our understanding of plasma transport phenomena and scaling laws and will not be forthcoming from a single experiment. Rather, evaluations of all key physics experiments and analytic developments will be an ongoing process and will be factored into the shield R&D program; such evaluations will be a topic for each of the scheduled design reviews.

Technology understanding from other R&D programs will be required to provide design criteria for TF coil heat input, heat removal capability, and vacuum vessel penetrations. The Large Coil Segment (LCS) and LCP R&D programs are scheduled to provide relevant data on TF coil heat transfer characteristics. Consideration of the impact of penetrations on shield effectiveness will be closely interfaced with the neutral beam and diagnostics programs.

7.2 PROPOSED MECHANICAL SYSTEMS

7.2.1 Limiter System

Function

Limiters provide physical protection to the first wall during uncontrolled plasma startup or shutdown. In addition, they can provide plasma shaping as a scrape-off mechanism. Limiters are either fixed or movable and either actively or passively cooled.

Design description

The limiter system is made up of two sets of poloidal limiters, located 180° apart. Each set consists of four fixed units, as shown in Fig. 7.1. Due to the irregular shape of the torus cross section and the fact that the limiters are for first wall protection only, and not plasma shaping, fixed units were chosen.

A passively cooled limiter is assumed. An actively cooled system is avoided because of (1) the added complexity of supplying coolant to the limiter and (2) the possibility of coolant leaking to the plasma in the event of a limiter coolant tube burnout.

Fig. 7.1. Poloidal limiter.

Graphite is the limiter material; its low-Z characteristics are desirable in regard to limiter contributions to plasma impurities. Graphite's thermal properties (high melting point, shock resistance, and low expansion) are all suitable for the operating environment. The graphite panels, which are 1-2 cm thick, are installed between adjacent first wall modules and extend 20 cm from the first wall toward the plasma. It is intended that they last for the life of the first wall. Their replacement would be part of the scheduled maintenance of the first wall. Limiters can be removed only by disassembling the torus.

7.2.2 First Wall

Function

The primary function of the first wall, in conjunction with the divertor, is to absorb and remove the energy associated with the alpha particles, which makes up approximately 20% of the total plasma energy.

The first wall is designed to take all the alpha particle energy. However, the actual heat load to the first wall is estimated to be equivalent to 50% of the alpha particle energy with the remaining 50% handled by the bundle divertor. Material damage associated with particle-surface interactions (e.g., sputtering, embrittlement, etc.) is taken by the first wall. The parameters of the first wall are listed in Table 7.1.

Table 7.1. First wall parameters

Construction	2.5-cm-diam stainless steel tubes
Maximum heat load (design value)	226 MW
Coolant	Water
Inlet temperature	40°C
Exit temperature	~150°C
Actual heat load	113 MW
Neutron wall loading, L_w	2.38 MW/m ²
Maximum wall loading (surface heat)	0.60 MW/m ²
Actual wall loading (surface heat)	0.30 MW/m ²
Lifetime	4.5 x 10 ⁵ cycles (10 years operation)

Design description

Type 316 stainless steel tubes, 2.5 cm in diameter, are brazed to type 315 stiffened sheet steel to make up modular first wall panels (see Fig. 7.2). Each of the faceted inner surfaces of the plasma chamber is lined with an independently cooled panel. The panels are mechanically fastened to the chamber so that induced thermal stresses in the panels are minimal. Type 316 stainless steel was chosen because its physical properties are adequate to meet the design requirements and because its use presents an opportunity to expand the nuclear/thermal data base for this material.

Fig. 7.2. First wall.

The total first wall installation consists of 6 panels for each of the 16 sectors. Sectors may be removed from the device for replacement or repair.

The first wall is expected to last for the proposed 10-year life of the facility (based on the mission description in Ref. 2). Adherence to this schedule requires approximately 4.5×10^5 pulses. This number is within the low cycle fatigue limit of stainless steel even if the total alpha particle energy is deposited in the first wall. The actual heat load to the first wall is estimated to be 50% of the alpha particle

energy (113 MW) with the remaining 50% handled by the bundle divertor. However, the coolant flow rate through the tubular first wall is adequate to transfer all the alpha particle energy (226 MW) at a temperature rise of approximately 110°C.

7.2.3 Shield

Function

The functions of the shield are to:

- (1) reduce nuclear heating in the TF coils to acceptable levels,
- (2) minimize activation of reactor components and auxiliary systems,
- (3) attenuate radiation from activated components during maintenance shutdown periods,
- (4) reduce radiation streaming around and through penetrations that view the plasma, and
- (5) transfer the heat generated by attenuation of neutron and gamma fluxes by means of an active cooling system.

The shield parameters are listed in Table 7.2.

Table 7.2. Shield parameters

Shield material	Stainless steel balls and borated water
Major radius, R	5.0 m
Shield minor radius	~1.5 m
Shield thickness	0.6 m
Shield thickness (around penetrations)	0.30 m
Number of sectors	16
Coolant	Low pressure borated water
Coolant heat load	1220 MW
Neutron wall loading, L_w	2.4 MW/m ²

Design requirements

The shield between the first wall and the TF coils will be thick enough so that the total time-averaged nuclear heating rate in the superconducting TF coils will be ≤ 20 kW (about one-third of the total pulse average heating rate in the TF coils).

Design description

The D-T plasma operation will produce large quantities of 14.1-MeV neutrons. These high energy particles, plus the secondary radiations they generate in the form of gamma rays and low energy neutrons, will cause damage to materials and represent a hazard to personnel. The shielding system of a D-T fusion device normally includes four components: (1) blanket, (2) bulk shield, (3) penetration shield, and (4) biological shield. The primary function of the blanket and bulk shield is to convert the kinetic energy of fusion neutrons and secondary gamma rays into heat. The additional function reserved for the blanket is breeding tritium; since TNS, at this time, does not include fuel generation, we are describing only a bulk shield having a dual purpose, energy conversion and reactor component protection by neutron attenuation. The penetration shielding surrounds the ducts for neutral beam injectors, divertors, vacuum pumps, fueling, and diagnostics. It provides virtually no attenuation for direct streaming but will protect components (such as the TF coils) that are adjacent to the penetrations. The biological shield is generally considered to be the concrete reactor cell, which provides the last barrier for neutron attenuation during fusion and limits activation levels outside the building to less than 100 mrem/year.

The bulk shield material is stainless steel in the form of 1-cm-diam balls with borated water acting as the coolant for each module. The minimum outer shield thickness (i.e., the shielding furthest from the bucking cylinder) will limit the PF coil organic insulation to a radiation level of 10^{10} rads over the life of the tokamak. The minimum inner shield thickness will limit the total time-averaged nuclear heating rate in the superconducting TF coils to an acceptable rate. A 5-cm thickness of lead is part of the bulk shield.

The total device shielding thickness was evaluated relative to the size and number of penetrations. The total effectiveness of the shielding in the reactor cell is greatly influenced by the area of the penetrations into the torus. The penetration shield around the neutral beam duct is 30 cm thick and the concrete thickness for the reactor cell is 2 m thick, assuming ordinary concrete.

7.2.4 Machine Structure

Function

The machine structure provides the structural support for the components that make up the tokamak device: the torus plasma chamber, the TF and PF coil supports, the dewar structure, and peripheral components that interface with the torus, such as the neutral beam injectors, the rf heating system, the pellet fueling device, and the divertor.

Design requirements

The machine structure will be designed in compliance with accepted ASME safety codes to ensure a reasonable degree of conservatism and consistency in applying the design rules. Deviations will be permissible in areas where the codes do not apply and where revisions or extensions of the codes are warranted. The machine structure will be designed to complement the remote handling characteristics of the overall device, have a 10-year minimum design life, and incorporate materials that minimize neutron activation and have low magnetic permeability and high electrical resistance.

Design description

The major components that make up the machine structure are listed with corresponding design descriptions (refer to Fig. 7.3 to establish the location of each component within the tokamak device).

Torus plasma chamber. The torus plasma chamber surrounds the plasma, contains the bulk shield material, and provides support for the

Fig. 7.3. Major components of the machine structure.

first wall. The first wall structure is mechanically supported off the inner wall of the sectors. The bulk shield material, consisting of stainless steel balls and borated water, is located in the interspace of the sector walls. The torus is made up of 16 sectors which are mechanically sealed and bolted at sector interfaces. Each 22.5° sector has a trapezoidal cross section and consists of three inner and outer, single-curvature-formed, stainless steel sheets separated by internal stiffeners and closed at each end. Four sectors form a quadrant (90°) of the torus and are supported from below by a curved beam attached to a truss structure connected to ground. The curved beam and truss structure are configured to allow a torus quadrant to be moved radially outward a small distance to provide initial intersector clearance during sector removal operations.

With all 16 sectors connected, a rigid structure is formed. Electromagnetically induced, radially outward forces are relatively small and are reacted by the intersector bolts. The interior D-shape coils of the poloidal field system, located at the top and bottom of the torus, act as bucking rings to support any electromagnetically induced centering forces. The torus itself is designed to withstand the 1-atm pressure differential required for shakedown operations.

Coil support structures. Central TF coil support column: The central support column is a cryogenically cold (5K) structure which (1) transfers the gravity forces of the TF coils and intercoil structure to a cryogenically isolated machine support base and (2) supports a portion of the magnetically induced centering and overturning forces of the TF coils based on the relative stiffness between the support column and the intercoil structure. The support column is a 12-sided polygon with a circular inner diameter. The structure has a vertical groove along the full height of each face to react a portion of the TF coil torsional forces. It is electrically segmented to reduce eddy currents.

Intercoil support structure: The intercoil support structure, which reacts a portion of the magnetically induced TF coil forces, is a box beam construction located between coils and mechanically fastened to the TF coil structure along the upper and lower portions of the coils. The torsional load is balanced at the machine midplane by differential bending in the outer portion of the TF coil and by keying action at the bucking cylinder. A structural insulating shim of glass-reinforced epoxy acts as an interface between the TF coils and the intercoil structure to interrupt the eddy current path. Fault condition requirements for the failure of a single TF coil and the associated loads have not been defined; therefore, the extent of the intercoil structure required for this condition has not been addressed. The design shown indicates that out-of-plane fault loads generated along the unsupported section of the TF coil are beamed to the existing intercoil structure. If the stresses generated on the TF coil under this condition are unacceptable, it appears feasible to add local demountable supports to reduce the stresses without compromising the remote maintenance features of the device.

EF coil support: The intercoil forces associated with the EF coils are supported by an interconnecting structure between the EF coils. The upper support structure and machine floor support the gravity forces and any imbalance in the coil system.

TF coil lateral support: The TF coil lateral support member is a ring located at the top of the OH solenoid and connected by pins to the

TF coils to support the imbalance in the TF coil gravity loads during initial assembly.

Dewar structure. The dewar structure forms a vacuum enclosure for the cryogenically cooled components that make up the tokamak device: the TF coils and intercoil structure, the OH solenoid, the interior EF coils, and the central support column. The dewars minimize tritium collection on the surfaces of these components and enable them to maintain their cryogenic condition when the vacuum building is repressurized, eliminating the dependence of the tokamak device on their cooldown and warmup cycles.

The dewar structure consists of two continuous stiffened shells, which enclose the upper and lower portions of the TF coils; a central cylindrical dewar, which encloses the inboard legs of the TF coils; and individual rectangular dewars, which enclose the outboard legs of each TF coil. The secondary vacuum environment provided by the vacuum building allows the use of mechanical connections to join the dewar sections.

REFERENCES

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The TNS Reference Design includes design concepts for the first wall and methods for shielding. Although complete requirements for a blanket are to be addressed in later design studies, it was decided that these technical needs assessments, since they deal with the future program needs and provide guidance to the R&D effort, should include all the critical needs of the blanket and shield.

The blanket functions envisioned for a tokamak fusion power reactor include the very important ones of absorbing the energies from the energetic neutrons and from the nonneutron plasma energy deposited on the first wall, removing the heat via compatible fluids for steam and electric power generation, and breeding and handling tritium.

The impingement of energy on the blanket and its deposition in the blanket are processes that require considerable iterative consideration of composition of materials, nonuniformity of energy flux, cooling approaches and capacities, and the structural properties of the materials. In addition to the thermal and mechanical stresses due to the normal duty cycle operation, transients, major disruptions, and off-design point performance must be considered so that the design integrity is maintained. The best estimates to date of operation of blanket concepts point to useful lifetimes shorter than those of other major device elements. Thus, replacement of the total blanket or of weakened modules will be required during the useful life span of the fusion power reactor. The many interlocking concerns in selecting a blanket design approach have been categorized into nine areas.

8.1 NUCLEAR DATA BASE

The establishment of a well-defined nuclear data base for all materials under consideration is required to provide the means to obtain the desired accuracy in the associated analyses and to permit accurate optimization of the breeding and shielding design.

The nuclear data base presently available is incomplete and contains large uncertainties. As the evaluations of the designs and their performance mature, the analyses required to support the design will become substantially more sophisticated. The need to know and reduce the level of uncertainty in these analyses will increase.

The main problem concerns the accurate description of the nuclear heating and radiation damage in the blanket, shield, magnets, and structures. In particular, shielding of superconducting coils at penetrations in the reactor blanket will be a sensitive area requiring accurate data to avoid overconservative and hence costly design. Other problems include the determination of tritium breeding capability in the experimental breeding modules, the description of induced activity in the various structural materials in the reactor, and the requirements for nuclear data on the fuel cycle reaction process and the radiation shielding evaluations.

8.2 NEUTRONICS METHODS

The development of neutronics methods for accurate description of the nuclear performance of finite-sized, realistically configured blankets/shields is required, along with bench-scale tests, to provide a demonstrated nuclear design capability.

Generalized numerical techniques exist to describe sophisticated space-time nuclear behavior; the Monte Carlo technique is one example. However, methods must be developed on the basis of such techniques and then tested against bench-scale experiments to provide the calculational capability needed for detailed analyses of future designs. This includes the capability to describe the system under analysis in all dimensions, including the effect of penetrations and the proper neutron source term for the toroidal fusing plasma geometry.

8.3 THERMAL-HYDRAULIC AND HEAT TRANSFER ANALYSIS

The development of computational methods to quantify the thermal fluid behavior of promising coolants is required to provide the thermal-hydraulic and heat transfer bases for the design.

The geometry of the coolant system, and of the blanket it is cooling, is likely to be complicated, requiring sophisticated multidimensional calculational methods to describe accurately the thermal fluid and heat transfer behavior of the blanket for normal pulsed operation and for postulated off-design and accident conditions. The impact of the thermal cycling resulting from pulsed operation is recognized as a potentially severe problem. Accuracy in assessing the thermal stress conditions will be required to define materials development needs.

8.4 PULSED MAGNETIC FIELD EFFECTS

A determination of how the design and performance of the pulsed magnetic field systems are affected by the proposed mechanical aspects of the blanket and shield must be made to provide a basis for the electromagnetic aspects of the blanket/shield design.

The specific blanket and shield design selected may use many materials of varying conductivities arranged in a complicated configuration. The problem of induced eddy currents (and the resultant forces and fields) has been recognized, and some work has been done in this area. However, electromagnetic field perturbation criteria must be generated, and methods must be developed to describe accurately the electromagnetic effects of nonsinusoidal time variations on the complex blanket and shield designs being considered. Calculations of losses in the blanket are also needed to determine the effect of the blanket on the air core volt-second requirement.

8.5 ELECTRICAL INSULATION

Requirements must be established for the electrical insulation of the blanket and shield to provide the basis for the design of any electrical breaks in the blanket.

The blanket and shield are located in strong electromagnetic fields that undergo large, rapid changes presently uncertain in magnitude by much more than a factor of two. Accordingly, criteria for the electrical insulation of the blanket and shield must be developed. Any special materials required to satisfy such criteria would have to be identified and approved, their behavior in a plasma radiation environment determined, and their performance under cyclic thermal loads established.

8.5 LIQUID LITHIUM CIRCULATION

Experimental demonstration of the acceptability of circulating liquid lithium in the configuration environment is necessary to the qualification of the blanket design.

Liquid lithium has been identified for use in tokamak power reactors.¹ An understanding of the performance of circulating liquid lithium in the magnetic field environment must be developed, so that acceptable performance can be realized.

8.7 COOLANT/CONTAINMENT COMPATIBILITY

The compatibility between candidate coolants and containment materials must be determined to develop a basis for prediction of component lifetime.

8.8 STRUCTURAL MATERIALS

Supporting R&D are needed to support the selection of the structural material (including both composition and metallurgical condition) which

has optimum end-of-life properties for reactor application. This material will form the basis for design of the first wall and other blanket structure.

Radiation effects in the various fusion reactor materials in the first wall and blanket must be known so that a balance can be achieved between conservative (and hence costly) design requirements and shortened working life of the reactor components. The 14-MeV fusion neutrons generated by the thermonuclear reaction will produce significant radiation effects in the first vacuum wall and the structural walls of the blanket. Helium produced by the (n,α) reactions will embrittle most conventional structural materials, such as type 316 stainless steel (20% cold worked), which is presently considered the best materials choice for the first experimental reactors. Moreover, radiation induced damage increases with increasing wall temperature, with a drastic reduction in the creep rupture strength properties of type 316 stainless steel above 550°C. Uncertainties exist in both irradiation conditions and requirements. Indications from reactor studies consistently support the economic desire for neutron wall loadings high enough to require significant developments.

8.9 TRITIUM RECOVERY

A feasible scheme for recovery of tritium from the blanket must be identified and developed for application to provide a basic design for this essential function in the breeding modules. Significant tritium-related questions must be answered before full-scale environmental considerations can be addressed.

The fusion power reactor blanket has been studied in depth in the ORNL Tokamak Blanket Design Study,² the objective of which was to develop a blanket concept based on near-term technology which appears to satisfy practical reliability and lifetime needs. An interesting concept has been advanced and preliminary analyses show its promise for application. This very challenging engineering area, impacted by machine complexities and activation that make repairs and changeouts difficult, can be and is being addressed with logical engineering design and analysis. This area

is of particular value to the engineering test phase, and a continuation of the development is necessary. The requirements in terms of surface and volumetric heat generation coupled with the large number of power on-off cycles are challenging — even if the severe neutron irradiation effects are ignored. The following R&D activities should be of particular benefit:

- (1) Development of an experimental basis for the concept via model testing that includes some limited neutron environment. MHD effects can and should be tested in a lithium blanket fluid.
- (2) Characterization of the requirements imposed on the blanket (i.e., particle loading and heat loads) from an abnormal or disruption situation as well as a normal cycle.
- (3) Continued investigation of alternate concepts on a relatively lower level.

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9. TOROIDAL FIELD SYSTEM

9.1 TECHNICAL NEEDS ASSESSMENT

The major technology needs which affect the status of the TF system in meeting the proposed TNS Reference Design and schedule are discussed here. The system is described in detail in Sect. 9.2.

The Large Coil Program was initiated to advance the design and manufacturing capabilities of organizations that can provide superconducting TF coils of a size and field in the range predicted for fusion needs. A TF coil capable of operating at 10.9 T extends the capabilities presently being addressed in the LCP; this indicates the need for supplementary emphasis in order to meet the requirement. The tentatively planned development and test programs for conductors suitable for 12-T fields would provide necessary data for the Reference Design. It is expected that the combined results of the LCP coil development and test program and of the 12-T conductor development program would generate sufficient information to provide a proof of principle for the design of the TNS-size TF coil. With the implementation of the 12-T conductor program, it appears that the major R&D needs associated with a TNS TF coil are being planned in a timely manner.

The application of data from these developments must include many considerations to achieve the desired performance and must establish the confidence necessary to embark on the project design and fabrication. The key objective is to demonstrate reliable operation of a realistic magnet system at selected design conditions without expensive design margins. Success depends on the adequacy of design problem solutions and on the reduction of uncertainties. The specific R&D emphases needed include:

- (1) Production of Nb_3Sn conductor and manufacture of coils without strain degradation of properties.
- (2) Stabilization by boiling helium (degradation of heat transfer flux by vapor blanket).

- (3) Means of coil protection, including the detection of small normal zones, quench pressure in forced flow channels, and quench voltage in bath-cooled coils and the control of coil currents or fault forces.

9.2 PROPOSED TOROIDAL FIELD SYSTEM

9.2.1 Function

The toroidal field system generates a toroidal magnetic field that combines with the pulsed poloidal magnetic field to confine the plasma. The system parameters are listed in Table 9.1.

9.2.2 Design Requirements

The TF coils must generate a magnetic field on axis (B_T) of 5.3 T and must have a magnetic field ripple at the plasma edge (δ_a) of <2.5% (peak to average). The duty cycle is 100%, and the coil lifetime is 10 years.

9.2.3 Design Description

The required on-axis magnetic field of 5.3 T at a plasma major radius of 5.0 m is produced by 12 superconducting TF coils with a horizontal bore of 6.2 m and a vertical bore of 9.8 m. This combination of number of coils and coil size achieves a magnetic field ripple of 1.5% at the plasma edge, which is less than the maximum value of 2.5%. However, these larger TF coils, relative to the size required for 2.5% ripple, allow the PF coils inside the bore of the TF coils to be raised to parking orbits during torus segment removal and also allow space for the removal of a torus segment (consisting of one-sixteenth of the torus) between stationary TF coils.

The TF coils are a pure tension D-shape design with a trapezoidal cross section. The centering forces on the coils are reacted through a bucking cylinder and an intercoil support structure.

Table 9.1. TF coil system parameters

Geometric	
Number of coils, N_c	12
Coil shape	D
Horizontal opening, d_{tf}	6.2 m
Vertical opening, h_{tf}	9.8 m
Mean coil circumference, C	29.5 m
Plasma radius, a	1.2 m
Major radius, R	5.0 m
Electromagnetic	
Maximum field at winding, B_{max}	10.9 T
Field on axis, B_T	5.3 T
Total ampere-turns	132 MAT
Stored energy, W_c	18 GJ
Number of turns per coil	733
Structure-to-conductor area ratio	2.14
Current per turn, I_t	15 kA
Conductor	Nb ₃ Sn superconductor with copper matrix
Overall average conductor current density, JAC	37.5 MA/m ²
Structural	
Strain in conductor matrix, ϵ_m	<0.2%
Centering force per coil, F_c	470 × 10 ⁶ N
Structural material	300 series stainless steel
Weight of each coil, W_{coil}	220,000 kg
Thermal	
Conductor coolant	Supercritical helium
Average helium inlet temperature, T_{in}	5.0K
Helium flow rate/coil, m	4 kg/sec
Total refrigeration required at 5.0K (includes pumping power), P_{ref}	48 kW

The superconducting elements are composed of an insulated, cabled conductor, using Nb_3Sn filaments in a copper stabilizing matrix. The cables are encapsulated by a thin stainless steel jacket and cooled by forced flow supercritical helium, which runs through the interstices of the strands. The conductors are pancake wound around a stainless steel coil support bobbin. There are eight slots in a bobbin with three Nb_3Sn superconductors in a slot. The TF coil configuration is shown in Fig. 9.1.

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Fig. 9.1. TF coil configuration.

The superconducting TF coils are designed to be cryostable. The cryogenic recovery capability of the conductor design is defined as the ability to recover to a fully superconducting state from an initially imposed normal zone temperature (driven to that temperature by a sudden heat deposition in the conductor) over a given normal zone length. The present stability requirement ground rule specifies that superconductors shall be able to recover to their fully superconducting state after 10^5 J/m^3 of conductor plus helium volume is suddenly deposited in half a turn of the conductor winding. Figure 9.2 indicates that the Nb_3Sn design can meet the requirement with a very small flow rate. However, in order to maintain a turbulent flow in the Nb_3Sn conductor channels, a flow mass velocity of 2 g/sec-cm^2 per conductor was selected. At this mass velocity, the stability margin of the Nb_3Sn superconductors is >3 .

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Fig. 9.2. Recovery capability of the TF coil superconductor.

10. RF SYSTEM

10.1 TECHNICAL NEEDS ASSESSMENT

The use of rf energy to assist during the startup phase of the tokamak is incorporated into the TNS Reference Design. The potential advantages of this system include PF system cost reductions and equipment simplification. The major technical need for rf assisted startup appears to be proving that the plasma physics of this approach is favorable (see Sect. 2.2). The key technology issues are discussed below and depicted schematically in Fig. 10.1.

A target power output of 200 kW per tube appears to be a reasonable objective at this time. It follows that a tube development program should be started for devices operating at 120 GHz, using the technical fallout from the Varian Associates 28-GHz tube development efforts.

The transfer of rf power from the source to the plasma poses several problems in the areas of high power waveguide components and coupling of rf heating to the plasma. A development/test program should be instituted to investigate the power handling of oversize waveguides and the design of mode filters. Optimum materials for waveguides and vacuum seals able to resist radiation must also be developed. Some recommended tasks which will provide needed engineering information are outlined below.

10.1.1 Quasi-Optical Oversize Waveguide

The characteristics of various oversize circular and rectangular waveguides should be investigated. Tapers from larger to smaller and from circular to rectangular guides are of special interest. Breadboard models should be built and used to measure attenuation, voltage standing-wave ratio (VSWR), mode characteristics, and the effects of bends, smoothness, material types, and circularity tolerances. Tests would be conducted at low power at a frequency in the 120-GHz range.

Fig. 10.1. R&D needs for rf heating system.

Oversize waveguide directional couplers, mode filters, and miter bends should be designed and breadboard models built to measure attenuation, VSWR, and mode purity. Again, tests would be conducted at low power and 120 GHz.

Once these investigations are complete, high power testing of waveguide straight sections, bends, mode filters, and directional couplers should take place with 200 kW at 120 GHz with a 4% duty cycle, under both pressurized and unpressurized conditions. The development of

the microwave window vacuum seals would tie in with this work (see Sect. 10.1.2).

10.1.2 Pressurization Window

Investigations should include evaluation of materials (such as beryllium oxide) for their loss characteristics, power handling, radiation resistance, and strength and evaluation of possible window locations in the waveguide run. Tests of samples in the waveguide should measure transmission loss and strength by pressurization of the guide at low power. Power handling could be evaluated by subjecting the waveguide and window to 200 kW (peak) with a 4% duty cycle, both with and without pressurization. The radiation resistance characteristics should be confirmed either by analysis or through testing.

10.1.3 Antenna Pattern

A partial full-scale screen mock-up of the torus could be used to investigate antenna designs and determine those capable of coupling the rf radiation into the plasma. Antenna designs could be breadboarded, tested for free space pattern integrity and VSWR, and then mounted on the mock-up for further pattern and VSWR measurements.

10.2 PROPOSED RF HEATING SYSTEM

10.2.1 Function

The rf heating system supplies high power microwave energy to the plasma to initiate ionization and heating during startup. The rf assisted startup is expected to significantly reduce OH power requirements and cost. System parameters are shown in Table 10.1.

Table 10.1. RF heating system parameters

Structural/configurational	
Number of gyroklystron tubes	5
Tube length	1.22 m
Tube diameter	0.91 m
Performance	
Total injected power	1 MW
Pulse length	2 sec
Frequency	120 GHz
Injected power per tube	200 kW
Efficiency (gyroklystron device)	30%
Total electrical demand (from grid)	7.5 MVA

10.2.2 Design Requirements

The design requirements for the rf heating system call for 1.0 MW of injected power delivered at a frequency of 120 GHz. The pulse length is 2 sec and the duty cycle is 0.4%. The system's lifetime is 10 years.

10.2.3 Design Description

The use of microwave heating in the Reference Design is considered to be a promising approach to preionization and may significantly lower the power requirements and cost of the OH power supply systems. Preionization by rf heating can substantially reduce the OH power supply energy needed for large loop voltage breakdown. Power transmission is accomplished by relatively efficient waveguides. The proposed system is based on a configuration of five 200-kW gyroklystrons which will deliver 1 MW to the plasma for approximately 2.0 sec.

Five power amplifiers driven by a single master gyrotron oscillator generate the rf power. The gyrotron is a new type of microwave vacuum tube based on the interaction between an electron beam and microwave fields where coupling is achieved by the cyclotron resonance condition. This type of coupling allows the beam and microwave circuit dimensions

to be large compared to a wavelength. Power density problems encountered in conventional traveling wave tubes and klystrons at millimeter wavelengths are not as severe in the gyrotron. Each gyroklystron amplifier tube has a minimum gain of 30 dB and produces 200 kW. The single master oscillator operates at the fundamental frequency, produces 1 to 2 kW, and drives the individual power amplifiers through a 5:1 divider network. Each power amplifier tube is 1.22 m long by 0.92 m in diameter, requires several electrical power supplies and oil, water, and fluorocarbon coolants and has a superconducting focusing magnet. The rf power is transmitted through oversize waveguides (i.e., the waveguide diameter is approximately 10 times the wavelength) to penetrations in the torus.

Five shielded canisters, each containing a high power rf source, are to be mounted on a support platform on top of the tokamak.

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11. INSTRUMENTATION AND CONTROLS SYSTEM

11.1 TECHNICAL NEEDS ASSESSMENT

The present concepts for the controls and instrumentation of the TNS Reference Design are described in Sect. 11.2. To date, there has been little study of the specific controls needs for tokamak reactors, and the projected transient operation of a reactor has only recently been a subject for controls studies. This area needs the attention that would result from an expanding, in-depth investigation into TNS needs and potential solutions. The results from such studies will undoubtedly play a significant role in the early definition of potential problems and the delineation and application of solutions which will shape the transient operation or burn scenario for TNS.

The major controls R&D requirement to meet the needs of TNS includes both physics and technology; it may be stated as definition and test of the controls concepts suitable for tokamak operation. The tokamak controls R&D effort throughout the fusion community has been limited to very preliminary studies and tests of plasma position control. The major areas of concern (startup, burn, refueling, shutdown, and abort) have hardly been considered. The approach to controlling a long burn tokamak has only recently been addressed in any significant fashion.

In support of the key technical need discussed above, additional instrumentation and controls R&D needs include:

- (1) Determination of reactor-relevant instrumentation that can adequately survive the projected environment and provide the data for control.
- (2) Development of a hybrid computer simulation, bench marked with ISX-B and later machines, to provide a correlated and necessary controls analysis tool.
- (3) Assessment of subsidiary systems (those having no direct interface with the plasma system) and of their controls needs, problems, and solutions. These would include heat removal and tritium handling systems, power supplies, etc.

- (4) Definition of the diagnostic system approach and equipment matrix to provide the required data handling for experimenters and operators on both a real-time and an archival basis.

11.2 PROPOSED INSTRUMENTATION, CONTROLS, AND DATA HANDLING SYSTEM

11.2.1 Function

The instrumentation, controls, and data handling (ICDH) system automatically controls the tokamak plasma and systems through all phases of operation. Semiautonomous operation of subsystem controls and diagnostics is carried out under the supervision of a central computerized control. Data from the controls and diagnostics instrumentation are processed and then displayed or stored in an archival system for later on-call display. Fail-safe operations and degraded mode options will receive significant emphasis in the design of the ICDH system.

11.2.2 Design Requirements

The ICDH system will consist of a central operations station and several semiautonomous controls substations. The system is centralized in that the machine can be started, operated, and secured from the central operations station and in that the data relevant to machine performance are stored in a central data pool. Upon direct assignment from the central operations station, the semiautonomous controls substations may operate a particular subsystem. However, the responsibility of monitoring all operations and the ability to override commands from substations will remain with the central operations station.

Individual subsystem operations, such as testing and checkout, will be prevented from affecting other elements (operating or dormant) by a system of interlocks and overrides. As the environment permits, each controls substation will be located in proximity to the equipment it monitors and controls to facilitate testing and checkout.

The ICDH system will be constructed with normal industry practices, modified as necessary for shielding from detrimental effects of nuclear

radiation and magnetic fields. The equipment will incorporate redundancy, diversity, and fail-safe features to minimize maintenance needs. Only monitoring devices and communications links and lines will require remote maintenance. With the exception of certain monitoring devices that may require periodic replacement because of exposure to plasma operation, the system has a design life of 10 years.

11.2.3 Design Description

The ICDH system has four major divisions, each with its own monitoring and data handling.

Operations control

This subsystem controls all machine operations during all phases of the burn cycle and the immediate preparation for the next burn cycle. The supervisory control and the central operations station form this division.

Instrumentation and monitoring

This subsystem measures all pertinent variables and processes the data through appropriate algorithms. The instrumentation used for this normal control and monitoring will use multiplexing microwave and light transmission to facilitate maintenance and to reduce the space needed for penetrations to the machine from the operating stations.

It is desirable to monitor the following parameters:

- (1) spatial and temporal variations in plasma density,
- (2) plasma electron and ion temperatures,
- (3) neutron flux at the first wall,
- (4) tritium levels at selected locations,
- (5) divertor temperatures and coil currents,
- (6) poloidal magnetic field strength and coil currents,
- (7) toroidal magnetic field strength and coil currents,
- (8) neutral beam parameters and status,

- (9) rf heating system parameters and status, and
- (10) fueling rate.

Auxiliaries control

This subsystem controls and monitors the elements that are less frequently commanded, including the TF power system, the vacuum pumping systems, the refrigeration systems, and the electrical substation.

Abort control

The abort control subsystem is a fail-safe, scram-type system that prevents machine damage in the event of any expected disruption or operation anomaly. The instrumentation for this subsystem will receive the strongest emphasis on reliability and redundancy, and variables related to this area will be continually monitored at all stations.

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