Oak Ridge Tokamak Experimental Power Reactor Study—1976

Part 1 EPR Summary

MASTER

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Fusion Energy Division

OAK RIDGE TOKAMAK EXPERIMENTAL POWER

REACTOR STUDY - 1976

PART I

EPR SUMMARY

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This Summary is Part 1 of a six-part report which comprises the Oak Ridge Tokamak Experimental Power Reactor Study - 1976. The other parts of the report are:

- ORNL/TM-5573 Part 2. Plasma Engineering
- ORNL/TM-5574 Part 3. Magnet Systems
- ORNL/TM-5575 Part 4. Nuclear Engineering
- ORNL/TM-5576 Part 5. Engineering
- ORNL/TM-5577 Part 6. Research, Development, and Demonstration Needs

OAK RIDGE EPR DESIGN

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ABSTRACT

The reference concept for a tokamak experimental power reactor which produces tens of electrical megawatts and operates with a high duty cycle was used as the starting point for the FY 76 work. The design was established on the basis of the FY 75 EPR Scoping Study. The plasma size required to achieve ignition was determined with a dynamic plasma model which uses collisionality-dependent transport coefficients, including the trapped-particle modes. Highly efficient impurity control, the availability of Nb_3Sn toroidal field magnets with 11-T maximum field, and 50 MW of neutral injection heating were assumed. The minor radius is 2.25 m for a plasma aspect ratio of 3. A minimum spacing of 1.40 m was added to the plasma radius to accommodate the blanket, shield, and required access. The resulting horizontal bore of the toroidal field (TF) coil is 7.4 m; structural considerations result in a vertical bore of 9.7 m.

A wide range of plasma engineering topics has been addressed. The purposes for which these topics were investigated vary from providing design criteria based on our present understanding of various processes to initiating applied research in areas where criteria are needed but cannot presently be generated. Examples of the former category are the effects of magnetic field ripples, of neutral beam injection system characteristics, and of poloidal field system features. Studies of fueling, pellet injection, and start-up fall into the latter category.

The blanket converts energy into heat, breeds tritium in a few experimental modules, and protects the superconducting coils. The design includes high-pressure helium coolant, liquid lithium absorber, graphite reflector, and stainless steel structure. The intrinsic elements of the blanket design, the coils, the breaks, and the interfaces are developed to be compatible with remote handling; that is, they are based on

modular, repetitive construction. The technical evaluation of the blanket design was focused on thermal considerations and resulted in the choice of a structurally thin, separate, water-cooled wall facing the plasma. A study of the neutronics behavior, materials aspects, and feasibility of the design supported the choice of sizes and materials and resulted in specific design features such as modularity, replaceability, and repairability.

Starting from the basic Nb_3Sn , hollow-conductor, cryostatically stabilized coil concept, effort was concentrated on producing an improved design for the TF coil structure, conductor, and protection scheme. An improved understanding of the losses in the system has been attained through the use of the calculational tools developed during the year.

The innovative electromagnetic shielding scheme proposed last year to protect the TF coils and improve the poloidal field (PF) system has been studied to determine the need for this design and its effectiveness, requirements, and engineering impact. Our present knowledge indicates that currents internal to the TF coils provide major advantages and are, in fact, necessary when plasma shaping is used. The engineering problems associated with this type of system are complex, but can be resolved. Extrapolation of this system to the Demonstration Reactor Plant and beyond is of primary concern.

Evaluation of other parts of the EPR Reference Design, such as the vacuum, beam, and tritium handling areas and the balance of plant systems, also resulted in improved designs. Significant questions, such as the vacuum topology to be used, and its impact on remote maintenance, thermal isolation, and operational feasibility, must be pursued further.

From this technical evaluation has come a set of key findings. In the scientific area,

there is a pressing need for improved theoretical understanding of radial profile effects on equilibrium plasma performance, for a focused emphasis on experimental work that will guide us in developing scaling relations, and for a coordinated program on fueling for the necessary long pulses. Further, there is a need for detailed experiments in the transient phases of plasma operation to support generation of engineering criteria in terms of start-up, impurities, and electromagnetic requirements. In the technological area, rapidly expanded development work is required in the area of Nb₃Sn superconductor if this fundamental design element is to be available for the next step; related to this magnet requirement is the need for a dedicated effort in energy storage and transfer in the poloidal field circuit. The problem of plasma heating demands a combination of ingenious plasma-beam operation to make use of existing positive ion source technology and negative ion source development to permit a wider range of options. The strong incentive for higher power density operation increases the pressure for materials capable of withstanding higher wall loading-time products. In the engineering area, the clear finding is that the present designs are complex; engineering judgment about this complexity supports very strong

efforts to examine the basic design assumptions with the aim of simplifying the assembly and maintenance problems. This improvement can come by increasing aspect ratio and approaching the design from a reactor point of view.

The principal alternative methods of achieving higher power density, namely, higher beta, higher magnetic field, and greater plasma elongation, are all evaluated to determine more promising directions.

The conclusion reached at this stage in the iterative process of designing the next logical tokamak step is that the benefit/cost of this EPR Reference Design is too low. The critical plasma physics assumptions must be reexamined. The need to work on the system features which impact on plasma performance and plant size and which are primary cost determinants must be reemphasized. This work is now proceeding in the advanced studies program. The objectives for the next step after TFTR have evolved from this present study to be those of a fusion reactor core and a forcing function for technology. In this updated plan, which starts with ignition as the next step, two of the most demanding EPR objectives which must be achieved after ignition, high duty cycle/availability and the conversion of heat to electric power, have been reserved for the subsequent step, the EPR.

FOREWORD

The study of the Tokamak Experimental Power Reactor (EPR) has been performed by the Oak Ridge National Laboratory (ORNL) for the Energy Research and Development Administration Division of Magnetic Fusion Energy (ERDA-DMFE) over a two-year period. This second year's work builds integrally upon the earlier efforts. Therefore, the following format has been adopted for this report. Part 1, entitled "EPR Summary," contains a brief discussion of all facets of the work done this year. The technical areas discussed in Part 1 include the Reference Design itself; the underlying plasma engineering base; the supporting technical evaluations in the areas of magnet systems, nuclear engineering, and general engineering; and the major research, development, and demonstration needs that have been identified. Each of these areas is then discussed fully in separate technical memoranda, which have been issued as Parts 2-6. Where appropriate, the essential elements of previous interim reports have been included in order to provide a self-consistent, readable report. These earlier reports, *Basic Considerations and Initiation of Studies* (ORNL/TM-4853), *EPR Scoping Study* (ORNL/TM-5038), and *EPR Reference Design* (ORNL/TM-5042), can be consulted for a more complete discussion of the entire study.

The documentary output of the present study, which has been brought to an end, is principally represented by this six-part report. The report is intended to provide a full technical picture of the EPR Reference Design, its key features, and, through the technical evaluation, the principal remaining questions. Had the study continued on its original course, the results presented herein would have been synthesized into a completely new iteration of the design, in an attempt to approach an optimum set of characteristics. Since the findings of the study led to a redirection toward a higher benefit/cost course for the next step, the EPR study was brought to a close and the Reference Design was changed only where necessary from the standpoint of maintaining feasibility and subsequent self-consistency. This report is designed to contain, therefore, the technical material from which future designers can proceed toward the EPR or toward intermediate steps.

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ACKNOWLEDGMENTS

The work presented herein represents the concerted efforts of the many individuals whose names are listed in the six parts of the composite report. In addition to the participants explicitly identified in the separate parts as contributors to this report, many other people in various parts of the organization supported the work through their technical knowledge, advice, and criticism. In particular, close coupling to the Fusion Reactor Technology Program and integration with the Demonstration Reactor Plant Study have provided valuable support, and interactions with the Superconducting Magnet Development Program (SCMDP) and the Large Coil Program (LCP) have benefited the study greatly. John Clarke and O. B. Morgan have continued their encouragement and technical input throughout the entire study. George Kelley has provided encouragement, critical technical input, and review throughout all the advanced design studies.

Valuable discussions with members of the design teams from General Atomic Company (GAC), Argonne National Laboratory (ANL), and the University of Wisconsin, with interested ERDA-DMFE personnel, and with many other members of the fusion community have helped shape this study.

1. INTRODUCTION

This section contains, in a condensed form, a discussion of the events which led to an EPR study, the nature of the study, the design objectives for the device, and the results and conclusions of the study itself. The design is discussed further in Sect. 2, the evaluation of the design in Sect. 3, and the R, D, and D needs in Sect. 4. The findings, choices, and assessment of the study's implications are presented in Sect. 5. Parts 2, 3, 4, 5, and 6 of the composite report expand on Sects. 2, 3, and 4 in this part.

1.1 Historical Background

The history of the fusion program can be divided roughly into three periods, using the role of advanced designs as the major guide. In the first period, which lasted until the mid-1960's, plasma physics was the dominant interest. In the second period, lasting until the early 1970's, reactor studies were undertaken, but with little or no effort devoted to methods of devising intermediate fusion devices. In the third period, the present one, investigations are being carried out on the entire range of the fusion program sequence, i.e., present-day physics experiments, intermediate reactor steps, and commercial reactors; an iterative process is being used to improve each of the separate concepts.

The work at Oak Ridge reflects this evolution. The studies of intermediate reactor steps in this third period have led to the EPR activities. In early 1972, SCORE¹ (Superconducting Oak Ridge Experiment) was conceived at ORNL as the first D-T burning device, as part of a contribution to the energy study of the Federal Council of Science and Technology-Office

of Science and Technology. SCORE combined the scientific feasibility step (i.e., ignition level n , T , τ in hydrogen) with actual D-T burning. In 1973 and 1974 the SCORE concept was reconsidered² and was developed into the Feasibility Burning Experiment (ORMAK F/BX),³ a candidate for the Tokamak Fusion Test Reactor (TFTR) facility. In July 1974 the F/BX team turned to a consideration of the step after TFTR — the EPR.

In terms of the total iterative design process, the current EPR study represents the most extensive effort to date in advanced studies, though it is still only the present point on the evolutionary scale of intermediate fusion reactor designs. The results of the EPR study and their relationship to the next step to be taken after the TFTR will be discussed below.

1.2 Nature of the Study

There are five principal considerations that characterize the EPR study.

1. EPR is a major step in the U.S. fusion program.

As such, it is necessary for rational fusion program planning that a clear picture of the EPR's nature, magnitude, requirements, and options be generated as input to the overall plan. Bringing the EPR into sharp focus is at the heart of the study which has served to provide planning information to ERDA-DMFE.

2. This study is a forcing function for science and technology.

2. M. Roberts, internal memorandum (January 1973).

3. P. N. Haubenreich and M. Roberts, ORMAK F/BX: A Tokamak Fusion Test Reactor, ORNL/TM-4634, Oak Ridge (June 1974).

1. Thermonuclear Division staff, internal memorandum (March 1972).

In the iterative process of defining objectives, projecting scientific and technological understanding, undertaking engineering design, and then comparing costs and schedules with objectives, two critical benefits emerge:

- the technical issues are better addressed, because the implications that emerge from the extrapolation of current ideas are made clear to the community, and
- the key research, development, and demonstration needs become better defined.

3. The effort is conceptual in nature and not intended to result in a project proposal.

At this early stage, the emphasis is placed on defining what is necessary and feasible and not on preparing to build the device. The study aims at the identification of problem areas and the development of possible solutions using engineering-oriented reference design. With the resources available, the study has been further limited to the examination of the key reactor systems (plasma, magnets, blanket, beams). A secondary emphasis has been placed on the balance of the plant systems.

4. This study was conducted using a project orientation.

Given the manifold problems so closely linked to ongoing research and development, and the perhaps surprisingly tight schedule for the execution of a major project even ten to fifteen years in the future, the study was organized into a project environment. This approach was used to effectively couple the systems-oriented technical personnel with the research and development specialists and with the engineering design team. This permits technical judgments to be made on the basis of available information in time to meet project requirements.

5. With the funding limitations in FY 76, the principal focus of the study was technical evaluation.

The steps leading to the formal design and construction of a Capital Project may be stated as:

- (1) Scoping Study
- (2) Establishment of a Reference Design
- (3) Technical Evaluation of the Design
- (4) Pre-Conceptual Design
- (5) Conceptual Design.

In the program planning at the beginning of FY 76, cost projections were made for each of the phases, and a plan including both the technical evaluation and pre-conceptual design steps was developed; progress in the plan was determined in large measure by the funding level in FY 76. When the budget level was determined, a set of program objectives for FY 76 was established and adhered to throughout the year. These milestones emphasize project-oriented output supported by technical evaluation of the Reference Design. The pre-conceptual design, which specifically includes determination of cost and schedule estimates, was to be the focus of work in FY 76T, FY 77, and FY 78. The context in which the study was performed is more fully indicated in the Program and Budget Proposals (Form 189) for both FY 76 and FY 77,⁴ which outline the programs anticipated at the end of the Scoping Study and near the end of the technical evaluation.

1.3 Relationship to Other Work

This study is one part of the overall fusion program and is related to that program in the following ways:

1. Relationship to other EPR Studies

Two similar EPR studies have been conducted: one at Argonne National Laboratory (ANL),⁵ funded by ERDA-DMFE, and the second at General

4. Fusion Power Research and Development Program, Budget Activity No. ED 0204000 189d No. 00028, for FY 1975-1977 (prepared March 1975) and Budget Activity No. EE 0204000 189d No. 00037, for FY 1976-1978 (prepared April 1976).

5. W. M. Stacey, Jr. et al., *Tokamak Experimental Power Reactor Studies*, ANL/CTR-75-2, Argonne, Illinois (June 1975).

Atomic Company (GAC),⁶ funded by the Electrical Power Research Institute (EPRI). The studies differ most significantly in these ways:

- Net electric power is taken as an objective in the GAC study, though not in the ORNL and ANL studies,
- Circular or slightly noncircular plasmas are used by ORNL and ANL, and Doublet shape is used at GAC,
- Nb₃Sn magnets at 11 T were chosen at ORNL, and NbTi magnets were chosen at ANL and GAC, at both 8 T and 10 T in the ANL design.

2. Relationship to Advanced Studies

The EPR is envisaged as a step leading to successful operation of the Demonstration Reactor Plant (Demo) and the succeeding Commercial Power Plants (CPP). In view of this role, the EPR study has been conducted in parallel with a Demonstration Plant study where the principal concepts for EPR were extrapolated to the Demo and CPP and designs acceptable for all three applications were sought.

3. Relationship to Supporting Work

The nature of EPR as an early fusion reactor requires that the study be closely connected to all facets of other fusion work. This is the case at ORNL. Through the work of the plasma engineering personnel, the best information available from the experimental and theoretical plasma effort here and elsewhere is included in the EPR study. In the technology areas, the ongoing development work in large superconducting tokamak magnets, in high-power, long-pulse neutral beams, and in nuclear technology areas is reflected by the EPR study.

1.4 Objectives of the EPR

A clear view of the EPR is essential to the effective performance of the EPR study. The EPR

objectives were outlined by ERDA-DMFE; they are detailed here from the results of the first task in the ORNL study and form the base for the remainder of this Section.

1.4.1 Objectives in the Context of the Fusion Program

The EPR has a central role in the national tokamak program plan,⁷ as a key step in an orderly progression of major experimental devices culminating in the first Fusion Power Demonstration Reactor Plant (Demo). The design and construction of EPR will incorporate experimental information obtained from the Princeton Large Tokamak (PLT), the Poloidal Divertor Experiment (PDX), ORMAK, and Doublet III, as well as from any upgraded versions of these devices. It will also benefit from experience in engineering and technology development and construction derived from the Tokamak Fusion Test Reactor (TFTR), from other intermediate devices, and from the magnet, beam, and other auxiliary component development programs. The EPR is viewed as the first facility to generate significant amounts of fusion-derived power and employ a high duty cycle continuously.

The objective of the EPR program is to advance the science and technology required for commercial fusion power by providing on a timely basis:

- system operating experience and testing of components and subsystems for a larger demonstration plant,
- a focal point for research and development programs, and
- large-scale testing of plasma physics scaling including the effects of an ignited deuterium-tritium plasma.

This general objective requires that the EPR program qualifies for subsequent demonstration

6. GAC Fusion Engineering Staff, *Experimental Power Reactor Conceptual Design Study*, GA-A13534, San Diego (July 1975).

7. *Fusion Power by Magnetic Confinement*, ERDA-DMFE Planning Document, WASH-1290, Washington D.C. (February 1974) and subsequent yearly updates.

(in the Demo) all essential features of a safe, reliable, working fusion power plant.

1.4.2 General EPR Objectives

As stated in the national fusion program plan,⁷ the EPR must:

- generate tens of megawatts from fusion power,
- recover significant amounts of fusion energy in a circulating medium at a high enough temperature to produce electrical megawatts, and
- operate continuously on a high duty cycle.

Accomplishment of these objectives is necessary to provide the technical basis for the Demonstration Plant. The issue of assigning particular target values to each of the three objectives is more one of perceived cost-effectiveness than of technical merit. Similarly, the choice among net electrical power, electrical power, or the equivalent of electrical power is more one of perceived than of technical importance.⁸

These last two statements lead to the definition of the fourth design objective, namely, the EPR must:

- be able to accomplish the three technical objectives in a timely and economical manner.

"Timely" means that design, construction, and operation of the EPR fit smoothly with the preceding experiments. "Economical" means that the actual total program costs be a reasonable step up from the preceding experiment and, as importantly, that the extrapolation of the EPR costs both to larger power output and to the economics associated with scale be reasonable.

1.4.3 Specific EPR Objectives

Consideration of the general objectives has led^{8,9} to the development of a set of component objectives and associated characteristics that represent qualitative targets in particular areas. These are listed in Table 1.1.

1.5 Summary of Tasks

The principal activities in the EPR Program fall into three major phases: a scoping study, development of a reference design, and evaluation of the reference design.

1.5.1 FY 75 Scoping Study

The principal aim of the Scoping Study was to develop:

- a clearer definition of the magnitude, complexities, and uncertainties of the tasks associated with an EPR which was designed to be operational in the mid-1980's,
- a technical base from which subsequent design studies could proceed, and
- the requirements for research and development in support of the EPR.

The study addressed basic issues of the EPR: objectives, essential features, configuration and ranges of dimensions, research and development required to support the design, and alternatives that must be pursued until information for firm decisions becomes available. In each technology and design area, the scoping studies provided recommendations for a preliminary reference design to serve as a focus for subsequent design studies at ORNL. They also served to identify the principal research and development needs associated with the EPR program.

8. P. N. Haubenreich (ed.), *Tokamak Experimental Power Reactor: Basic Considerations and Initiation of Studies at Oak Ridge*, ORNL/TM-4853, Sect. 3.1, Oak Ridge (April 1975).

9. M. Roberts, *Oak Ridge Tokamak Experimental Power Reactor Study Scoping Report*, ORNL/TM-5038, Sect. 3 and App. B, Oak Ridge (to be published).

A summary of the investigations in the Scoping Study can be found in Sect. 6.1 of the Annual Progress Report of the Thermonuclear Division for 1975.¹⁰ Details of the Scoping Study are contained in ORNL/TM-5038.⁹

1.5.2 Reference Design

Based upon the findings of the Scoping Study, a reference design, pictured in Fig. 1.1, was established as the design focus for the EPR studies in FY 76. Choices in the reference design were based on the following considerations:

- high probability of providing a successful scientific experiment based on extrapolation of current plasma physics, and
- high probability of technological and engineering feasibility in the 1980's, derived from research, development, and demonstration programs in the late 1970's.

Description of the September 1975 Reference

Design. A set of basic machine parameters for the EPR Reference Design were chosen which would produce a reactor-grade plasma with significant neutron flux. These parameters included the following: minor radius of 2.25 m, major radius of 6.75 m, magnetic field on axis of 4.8 T (requiring use of Nb₃Sn conductor with 11 T at the winding), and plasma current of 7.2 MA. Despite uncertainties in plasma scaling, it was possible to predict that the neutron flux will be produced under both beam-driven and ignition conditions. The neutral beam power needed was 100 MW, with an accelerating voltage of 200 keV. The safety factor [$q(a)$] was 2.5. Burn time was 100 sec, with a duty cycle equal to 50%. The overall height was 15 m, and the overall diameter was 22 m. The output power for a driven system was ~400 MW (t) and for a system under ignition conditions was ~200 MW(t), for the specific operating modes examined.

10. *Thermonuclear Division Annual Progress Report for Period Ending December 31, 1975*, ORNL-5154, Oak Ridge (June 1976).

A complete description of the Reference Design is presented in ORNL/TM-5042.¹¹

1.5.3 Evaluation of the Reference Design

The bases upon which the Reference Design was established varied widely from explicit calculations to management judgments. Consequently, the major thrust of the work in FY 76 was to perform a technical evaluation of the Reference Design and to upgrade it where necessary. A number of technical issues were addressed, and where possible "solutions" were developed. That is to say, in a number of critical areas, those ideas which upon serious consideration appeared to be feasible design concepts have been proposed and examined. In the current reference design, for example, the following have been added to the original reference design:

- A fuller description of an ignited plasma.
- A new modular, maintainable, workable blanket design.
- A new TF magnet design more credible in terms of fabrication. Additional analytic tools have been developed and used to define more closely the stability margin and the ac losses.
- In-depth study of the ohmic heating and electromagnetic shielding systems driving the plasma current, protecting the TF coils, and reducing the energy requirements. Critical questions have been identified.
- Establishment of the basis for the overall structural system.
- A first pass at the balance of plant.

The conceptual development of some of the key technical problems has provided a clearer view of the technical problems and has identified some very difficult judgmental issues.

11. M. Roberts and E. S. Bettis, *Oak Ridge Tokamak Experimental Power Reactor Study Reference Design*, ORNL/TM-5042, Oak Ridge (November 1975).

Fig. 1.1. EPR reference design September 1975.

The unresolved technical problem areas requiring expanded study include:

- impurity control
- fueling
- heat removal at a constant rate
- thermal cyclic effects
- behavior of irradiated materials
- tritium recovery
- remote maintenance
- outage impact
- leak detection
- accident considerations
- effects of penetrations.

The judgmental issues requiring a wide discussion and having substantial impact on the overall planning can be stated as:

- assessment of benefit/cost for a low power, high cost design
- complexity of mechanical system

- extrapolation of plasma physics parameters
- simultaneous extension of many component technologies.

1.6 Principal Findings and Underlying Scientific and Technological Considerations

On the basis of this technical evaluation, a number of key findings became clear. Considerations of these findings and an appreciation of the difficult judgmental issues raised leads to a reexamination of the fundamental assumptions used in the study. Each of these three topics is discussed here.

1.6.1 Principal Findings

The principal findings impacting the technical program can be divided into three convenient categories.

Table 1.1. Specific technical objectives for EPR

Scientific – generation of fusion by D-T plasma reactions in a tokamak configuration.

Requires demonstration of adequate understanding and achievement in:

- plasma dynamics (start-up – burn – shutdown)
- fueling or burn maintenance (long pulse effects, impurity control)
- size scaling (transport/size extrapolatable to Demo)
- electromagnetic coupling of plasma and system (control of the magnetic fields throughout the device).

Technological – providing the environment needed for fusion power generation with the capability of extrapolation to operation with net electrical power output.

Requires demonstration of adequate understanding and achievement in:

- superconducting magnets (toroidal field and perhaps the poloidal field set)
- neutral beam heating
- energy conversion at acceptable efficiencies
- tritium breeding and extraction in experimental blanket modules
- energy storage and transfer in the ohmic heating system
- radiation shielding of equipment and personnel
- definition of neutron, neutral, and charged particle environment for a focused radiation damage program.

Engineering – demonstration of the ability to achieve safe, reliable, and cost-effective plant operation.

Requires demonstration of adequate understanding and achievement in:

- system integration
- reliable, high duty cycle operation (both high availability and duty cycle, >50%)
- remote maintenance (measured both in outage time and in personnel and equipment costs)
- environmental behavior (with regard to health, safety, and environmental standards, public acceptance and support).

Scientific

- In the area of scaling, continually improved models are available for performance prediction, but significant additions to the experimental data base and improvements to the analysis are needed before confidence can be generated that, for example, the scaling relations are appropriate and that profile effects have been properly considered.
- In the area of dynamics, detailed experiments are needed to provide hard information on startup/burn (fueling)/shutdown for engineering criteria.
- Similarly, in impurity effects and plasma electromagnetics, experimental information is sorely needed for guidance of design efforts.

Technological

- Demonstrations, at the appropriate scale, are required for each of the key technological components in the EPR design: TF and PF coils, tritium handling, heating techniques and hardware, etc.
- Nb₃Sn superconductor development must be emphasized and expanded if the critical technological flexibility of higher magnetic field is to be available for any next step.
- A dedicated poloidal field coil design and development effort is required to address this crucial and newly appreciated area.
- Development of a nonconducting structural material capable of meeting the stringent thermal, mechanical, electrical, and nuclear requirements of the OH coil support structure is essential to the feasibility of a superconducting PF system.
- Energy storage, transfer, and control associated with the PF system require a dedicated, intensive effort to make this aspect feasible; cost reduction is a secondary objective here.
- Plasma heating requirements demand an

uncertain combination of ingenious plasma operating scenarios to make use of the available positive ion beam technology and creative negative ion source development for flexibility in the design.

- Exploitation of the indications of extended material lifetimes under irradiation is a fundamental necessity in the drive for economical systems in which replacement costs are a dominant factor in the plant economics.

Engineering

Given that a number of significant engineering issues have been addressed, and addressed successfully (e.g., workable design solutions in the blanket, TF coil, and remote maintenance areas), the net result is still an extraordinarily complex device.

This complexity can be appreciated by consideration of the following four areas of design requirements:

- mechanical complexity of nested tori, wide temperature ranges, large structural loads on large elements, and intertwined coil sets
- many operating pulses
- high auxiliary power demands
- remote handling combined with high tritium inventory.

The key finding here is not this list of complexities but rather the appreciation of means of reducing the complexity.

- Increasing the aspect ratio relieves the first problem considerably.
- Lengthening the burn time relieves the second.
- Increasing the thermal power output reduces the impact of the third.
- Ingenuity and serious design can relieve the fourth.

The finding here, then, is a set of requirements for both engineering and plasma physics that can lead to an economically viable power plant.

1.6.2 Judgmental Issues

The difficult judgmental issues can be stated simply, although their resolution is inherently difficult.

- Summary judgment — The output of this design — nominally fusion power — is rather low for a rather high cost. This judgment is subject to error and is clearly a current judgment about future events.
- Scientific judgment — How far can the crucial plasma parameters, like density and impurity level, and the characteristics of the plasma regime, like collisionality, be extrapolated from present experience using theory and experimental indications?
- Technological judgment — How credible is a plan in which many of the essential technologies must be developed from small existing data bases on a short time scale and simultaneously?
- Engineering judgment — At present, the mechanical system is extremely complex. Even though a case can be made for the successful execution of any one of the many severe engineering problems, the successful execution of all of them simultaneously seems doubtful.

Answers to these issues are not formulated here in terms of rebuttals. The "answers" lie in additional work based upon the perceptions contained in the issues themselves. ❧

1.6.3 Technical Considerations

With these two factors (findings and judgmental issues) clearly in mind, it is instructive to reexamine the underlying scientific and technological considerations that have led to this situation.

- Size and cost. The size of the EPR plasma and its blanket and shield determine the size of the superconducting coil which, in turn, is closely related to overall cost. The plasma size

is determined by the scaling laws used to specify the energy containment of the plasma. All of the EPR designs have used the so-called trapped ion mode scaling at densities of $\sim 0.7 \times 10^{20} \text{m}^{-3}$ to specify the energy containment time. This has led to large plasma sizes, which are necessary in order to attain an $n_e \tau_E$ near the ignition condition. This large plasma size, combined with the necessity for a blanket and shield between the plasma and the superconducting coils, has resulted in EPR superconducting coils with sizes of roughly 7 m (horizontal) $\times$ 10 m (vertical) weighing several hundred tons each.

- Low aspect ratio. The EPR designs are of low aspect ratio machines. Once an aspect ratio has been chosen (low aspect ratios are favored in the present regimes of the models), the magnetic field in the plasma is determined by the maximum allowable field at the conductor. There is thus strong motivation to utilize the more expensive high field superconductors. However, the use of high field superconducting magnets in a low aspect ratio torus results in extreme asymmetric forces on the coils; these forces must be minimized by the fabrication of asymmetric coils. Unfortunately, the fabrication of such asymmetric coils is complicated by the fact that they must be of large size, in accord with the EPR designs. Simultaneously, the general design and assembly problems are made most difficult by the cramped nature of a low aspect ratio device.
- High fusion power density. The fusion power density produced in a magnetic confinement system varies as $\beta^2 B^4$; therefore, the attainment of high beta and high magnetic field has been emphasized in the EPR studies. In the ORNL EPR studies, β_p has been limited to ~ 4 . As a result, the attainable total beta in the EPR designs is $\lesssim 3\%$.

Elongation of the plasma may permit an increase in the attainable value of beta, but uncertainties exist in the equilibrium and stability behavior to be expected in such cases, and economic and engineering difficulties are associated with producing a high degree of elongation. The incentive for elongated plasmas is clear, but the actual advantage remains to be determined.

Although the fusion power varies as the fourth power of the magnetic field, the importance of achieving the highest possible value of beta follows from the difficulty of obtaining high magnetic fields. The strength of the magnetic field, B , which can be utilized in a magnetic confinement device is constrained by technology. There is a limit on magnetic field for the two types of superconductor, NbTi and Nb₃Sn, which are available in commercial quantities. In addition, practical engineering considerations and economic constraints may prevent the use of fields much in excess of 12 T at the surface of the superconducting coil.

Many of the difficulties perceived to exist in the technology and engineering areas result from the plasma physics scaling laws and beta relationships discussed above. If the power density and the aspect ratio could be increased, many of these problems would be less severe. The concept of flux conservation through intense neutral beam heating appears to have potential as a way to accomplish both requirements.^{12,13} The exploration of the implementation of this more attractive set of conditions forms the basis for the positive tone of the conclusions presented below.

12. J. F. Clarke, *EPR and TNS*, ORNL/TM-5418, Oak Ridge (June 1976).

13. J. F. Clarke, *High Beta Flux Conserving Tokamaks*, ORNL/TM-5429, Oak Ridge (June 1976).

1.7 Conclusions

1.7.1 About EPR

- EPR should produce a significant amount of power.

The step in the fusion program that encompasses thermonuclear power production at a high duty cycle will be large, difficult, and costly. Since this step appears to be inherently very expensive, it should be accompanied or characterized by production of an amount of power that represents a significant fraction of the output of a commercial power plant.

- The assumption and requirements used in the EPR study have resulted in a large device and the need for major technology extrapolations.

The inherent difficulty and associated cost follow directly from the need for a large device. The size is dictated by the plasma scaling assumptions used in the study and by the blanket/shield needs to satisfy the requirement of continuous, high duty cycle operation at hundreds of thermal megawatts. The toroidal field magnet size compatible with this system is of itself very large and is a major extrapolation from present experience. Similarly, the neutral beam power requirements and ohmic heating system characteristics far exceed those of present devices.

- The current EPR device is uncomfortably large and the power output is uncomfortably small.

For a device of the size and complexity of the current EPR design, the cost appears to be in the range of \$1 billion. The fusion power density depends on a number of variables and can potentially be improved by technical advances. Therefore, effort must first be directed toward developing improved performance in a device of reduced size.

- EPR, as presently conceived, is not the next logical step.

Since the magnitude of the EPR task is such a large increment beyond the present state of the art and since the benefit is uncertain, the EPR is not the next logical step to take in the fusion program. Following the line of argument above, efforts are needed to conceive, develop, and demonstrate means of increasing the fusion power density before an EPR is undertaken.

1.7.2 About the Next Step

The outcome of the EPR study indicates that intermediate steps will be required to support development of an EPR,^{8,10} and that an improved plasma performance is required for an acceptable benefit/cost for both EPR and a fusion reactor economy. A potential means of providing the needed improvement in plasma performance is now being outlined.⁸ In order to define the step to be taken in the program following TFTR, efforts are being directed toward the following:

- Exploration of the technical basis for projection of a stable high beta equilibrium in which the plasma power density is sufficiently high, the material wall lifetime sufficiently long, and the concomitant requirements on all other components sufficiently achievable for extrapolation to an economically viable fusion reactor.
- Identification of useful, economical upgrades of the major facilities over the period before the design of the next step is to be fixed. This means evaluation of the benefit/cost of extending the pulse length, increasing the field strength, and otherwise expanding the experimental scope of TFTR, Doublet III, PDX, PLT, etc.
- Identification and testing of the specific characteristics of the scientific and technological steps which must be demonstrated in this next device. For example, scientific techniques for achieving and maintaining higher densities must be developed and evaluated in the next few years, using facilities such as ORMAK Upgrade, to establish a firm basis for the investment in the next device; reasonable extrapolations from qualified magnets, beams, and plasma technologies must be made in conjunction with development programs to delimit the design possibilities.
- Identification, refinement, evaluation, and detailed development of a feasible reference design must be accomplished in order to prepare for actual implementation of the next step at the earliest prudent time.

2. BRIEF DESCRIPTION OF THE EPR REFERENCE DESIGN

As a result of an intensive evaluation of the blanket and magnet systems of the September 1975 Reference Design, along with a continuing examination of the plasma engineering basis, an up-dated Reference Design has been generated. Although significant improvements have been made in the key systems, this updated design does not represent a full iteration of the design; in particular, there is no change in plasma shape or impurity control, both of which would be changed in a new Reference Design. A full description of the Reference Design is contained in ORNL/TM-5576, Part 5 of this report.

2.1 Summary

The EPR is an ignited, D-T burning, power-producing, air core tokamak with superconducting toroidal field (TF) and a combination of superconducting and normal poloidal field (PF) coils. The duty cycle is 87%, with a burn time of 100 sec. The machine has a major radius of 6.75 m, and a circular plasma radius of 2.25 m (aspect ratio of 3.0). A blanket is provided to transfer a nominal 410 MW(t) from the burning plasma to a steam power cycle. For design purposes, the nuclear aspects of the EPR are based on a neutron wall loading of ~ 1 MW/m², which is equivalent to a power output of 800 MW(t). Operation at this power level is considered an optimistic upper limit. A neutral beam system is provided to inject 50 MW of power into the plasma to achieve ignition. Figures 2.1 and 2.2 illustrate the elevation and plan views, respectively. Figure 2.3 presents a composite chart of each of the key preparatory steps before operation and the key events during an operating cycle. An initial plant layout is shown in Fig. 2.4, identifying 20 facility areas necessary for the reactor and balance of the plant.

2.2 Key Systems

2.2.1 Toroidal Field System

The toroidal field of the EPR is generated by 20 superconducting coils. These coils are

arranged on a stainless steel structure to form a torus, with the noses of the coils contacting a cylindrical center bucking column. The coils have an oval shape with a minor (horizontal) bore diameter of 7.4 m and a major (vertical) bore diameter of 9.7 m. These coils produce a maximum field at the winding of 11 T and an on-axis field of 4.8 T at the plasma. In the high field area, the conducting elements are composite wires composed of superconducting Nb₃Sn filaments in a CuSn matrix; in the lower field portion of the coil (<6.5 T), superconducting NbTi filaments in a copper matrix are used. The conductor coolant is supercritical liquid helium used in a forced-flow mode.

2.2.2 Blanket and Shield System

The blanket and shield are located inside the toroidal field coils. The blanket is composed of 60 circular wedge-shaped segments which, when assembled, make up the torus. Each segment is further divided into 12 modules which constitute the basic building blocks of the blanket (see Fig. 2.5). The first wall of the blanket, which intercepts the non-nuclear energy, is cooled with low-pressure water. The liquid metal absorber portion of the blanket receives the high-energy neutrons from the plasma and is cooled by high-pressure helium.

The shield is located outside the blanket. It is segmented and is composed of stainless steel and lead. The shield is cooled by borated water and forms a thermal barrier between the high-temperature blanket and the cryogenic TF coils. Neutron flux coming through the blanket is attenuated by a factor of 2×10^3 . The shield segments serve as the mechanical support for the blanket, and these segments themselves rest on tripod support columns.

2.2.3 Poloidal Field System

The poloidal magnetic system is composed of the ohmic heating (OH) coil, the shield-vertical field (S-VF) coil, the decoupling coil, and the trim-vertical field (T-VF) coil. A major portion

ORNL-DWG 76-8745

Fig. 2.1. EPR elevation view.

ORNL-DWG 76-8744

① AIR CORE PRIMARY WINDING	⑤ T.F. COILS (CONDUCTOR, BOBBIN, DEWAR)
② DECOUPLING WINDINGS	⑥ SHIELD
③ V.F. TRIM WINDINGS	⑦ BLANKET
④ SHIELDING/V.F. WINDINGS	⑧ VACUUM CONTAINMENT

Fig. 2.2. EPR plan view.

Fig. 2.3. EPR operating chart.

of the ohmic heating and decoupling windings are located inside the center bucking column of the structure. The remainder of these windings, and all of the trim-vertical field windings, are located outside the TF coils, and their mounting structure connects to the TF coils.

The OH coil serves to generate the toroidal electromagnetic driving force on the plasma. With the expected plasma impedance, a flux change of 130 W in the central solenoid results in a 7.2 MA plasma current. The coil current is driven from fully positive to fully negative values. The S-VF coil delimits the magnetic fields from the plasma current, thus shielding the TF coils. The S-VF coil also produces the

major portion of the vertical field maintaining plasma column equilibrium. The decoupling coil, closely coupled to the OH coil, is connected in series opposition with the S-VF coil to cancel the electromagnetic force generated by the OH coil in the S-VF circuit. The T-VF coil provides an auxiliary vertical field supplementing the S-VF coil. Table 2.1 presents some key distinguishing features of these four related coil systems.

2.2.4 Beam, Vacuum, and Tritium Systems

The neutral beam injection system deposits 50 MW of power into the plasma at an energy level of 200 keV for 5 sec each cycle, to

ORNL-DWG 76-4350

Fig. 2.4. EPR plant layout.

Table 2.1. Distinguishing engineering features of the poloidal field system

<u>Coil</u>	<u>Conductor Type</u>	<u>Number of Windings</u>	<u>Location Key on Fig. 2.1</u>	<u>Sense Relative to Plasma Current</u>
OH	superconducting	1220	1	opposite
S-VF	copper	16	4	opposite
Decoupling	superconducting	16	2	same
T-VF	superconducting	16	3	not applicable (creates vertical field only)

Fig. 2.5. Schematic of blanket construction.

contribute to the start-up heating of the plasma and to raise the plasma temperature to the ignition level. Six beam lines, with two sources per line, penetrate the plasma through six 40-cm-diameter apertures. Each beam line includes a negative ion source, a neutralizer region, a bending magnet to deflect ions which are not neutralized, a vacuum system, and a drift tube to connect the beam line to the torus.

The primary vacuum system penetrates the torus through two vacuum ports located between each TF coil; it is a cryosorption system in which helium is used to maintain the

cryosorption panel at 4.2 K. The forty 1-m-diameter cryopanel provide a total net pumping speed of 1,100,000 liters/sec. The roughing vacuum system, which is used for initial pump-down of the torus, consists of three 5000-cfm Roots Blowers coupled to three 500-cfm mechanical pumps.

A tritium handling system used to process approximately 2 kg of tritium per day is provided. The tritium handling system includes equipment to (1) evacuate the plasma vessel and the neutral beam injectors and maintain a vacuum there, (2) purify and recycle tritium and deuterium for the plasma fuel cycle, (3) recover

tritium from experimental breeding modules, and
(4) provide tritium containment and atmospheric
clean-up.

2.3 Key Parameters

A tabulation of overall system parameters
and their values for the EPR is shown in Table
2.2.

Table 2.2. Overall system parameters for the EPR

A. General Characteristics

1. Requirements

	<u>Symbol</u>	<u>Value</u>
Device lifetime	t_L	10 years
Duty cycle	dc	87%
Availability — normal	--	80%
Thermonuclear power (ignition)	P_0	410 MW

2. Other Features

Plasma radius	a	2.25 m
Major radius	R_0	6.75 m
Aspect ratio (plasma)	A	3.0
Plasma edge to winding distance	Δ	1.40 m
Plasma volume	V	675 m ³
Overall height	H	15 m
Overall diameter	D	23.3 m

B. Plasma Parameters

Burn time	t_B	100 sec
Current	I_p	7.2 MA
Confinement measure	$N_e \tau_E$	$2.7 \times 10^{20} \text{ sec m}^{-3}$
Plasma temperature		
Ion temperature	T_i	12.2 keV
Electron temperature	T_e	13 keV
Beta poloidal	β_p	1.9
Beta (total)	β	0.03
Safety factor	q	2.5
Electron density	N_e	$7.4 \times 10^{19} \text{ m}^{-3}$
Fuel ion density	N_i	$7.0 \times 10^{19} \text{ m}^{-3}$
Impurity level	Z_{eff}	1.34
Injection power into plasma	P_b	50 MW
Injected deuteron energy	E_b	200 keV
Particle confinement time	τ_p	9.6 sec
Energy confinement time	τ_E	3.7 sec

Table 2.2. (continued)

	<u>Symbol</u>	<u>Value</u>
<u>C. Electromagnetic Parameters</u>		
Maximum toroidal field	$B_{\max}$	11 T
Toroidal field on axis	B_T	4.8 T
Number of toroidal coils	N_c	20
Magnetic field ripple at plasma edge	ΔB	2.2%
Conductor configuration	--	composite cable in a square aluminum conduit
Superconductor		
in the high field region	--	Nb_3Sn in CuSn matrix conductor
in the moderate and low field region	--	NbTi in Cu-matrix conductor
Average poloidal field at plasma edge	B_p	0.64 T
Poloidal field coil core type	--	air
Poloidal coil volt-seconds	VS	165 Wb
<u>D. Thermal</u>		
Maximum power rating of blanket	P_R	800 MW
Blanket coolant	--	helium
Outlet temperature of blanket coolant	T_{out}	370°C
Blanket coolant pressure	P_{He}	70 atm
Shield coolant	--	borated water
Coil coolant (superconducting)	--	helium
<u>E. Nuclear Parameters</u>		
Neutron wall loading (operating point)	P_w	0.55 MW/m ²
Neutron wall loading (design point)	P_w	1.07 MW/m ²
Neutron flux on first wall (operating point)	ϕ_w	2.4×10^{19} n/m ² /sec
Neutron flux on first wall (design point)	ϕ_w	4.7×10^{19} n/m ² /sec
Tritium breeding ratio (in an experimental module)	BR	1.15

3. EVALUATION OF THE REFERENCE DESIGN

3.1 Plasma Engineering¹

3.1.1 Introduction

Fusion-power system design requires the application of techniques from many disciplines of physics and engineering. There is a wide variation in the level to which the disciplines have been developed. Plasma Engineering involves the use of plasma physics both to design the plasma system itself and to establish the criteria needed to permit design engineering to proceed self-consistently. In some cases techniques are available and only need to be utilized. In other cases, however, physics research must be carried out, and in fact the design process is made difficult by the fact that some of the most critical aspects of a fusion power device or its behavior are not well understood at present. The procedure that must be used here is one of making plausible estimates and retaining a flexibility in the design work so that it can always accommodate change.

In this section, several of the main topics and the logic used to determine fusion system parameters are outlined. Most of the information was developed in support of the EPR and Demo studies at ORNL. However, since much of this information is evolving and represents the state of the art, concrete results are not always available. In these cases the features which are important in analyzing a given physics or engineering area are described. The object is to provide an understanding of the bases for fusion design and the way in which the bases are reflected in appropriate design parameters.

Critical decisions related to a number of topics must be made for the execution of a reactor design. These topics include:

- choice of major tokamak dimensions
- choice of magnetic field strength
- choice of burn time
- provisions for impurity control and wall materials
- position and current waveforms of the poloidal field system
- external pulsed magnetic fields
- number and shape of toroidal field coils
- plasma ionization and breakdown requirements
- refueling of a burning plasma
- plasma heating power and energy requirements
- vacuum pumping
- low-neutron-yield testing
- prospects for clean fuels.

3.1.2 Plasma Simulation Models

The plasma simulation models used to predict the behavior of the system include both time-independent and time-dependent codes. The time-independent model, POINT, is used to scan the wide range of parameters and generally determine the characteristics of a device suitable to meet a specified objective. The time-dependent model, SYSTEMS, is used to examine the dynamics and behavior of the plasma system during operation. It includes the profile effects of impurities and charge-exchange neutrals and is used to find the equilibrium operating points for a specific machine. Physics models are included for:

- Maxwellian and beam-plasma fusion events
- fast-ion heating
- refueling and recycling
- conduction and convection
- bremsstrahlung, synchrotron, and line radiation
- neutron, ion, neutral, chemical, and self-sputtering
- charge exchange.

1. An extensive treatment of the topics outlined here is contained in ORNL/TM-5573, Part 2 of this report.

Table 3.1 shows the nominal operating conditions for the ORNL EPR which were determined using these codes.

In Fig. 3.1, the thermal fusion power output for the ORNL EPR is shown as a function of plasma density, a factor to which it is highly sensitive.

Using 50 MW of neutral beam heating, low-density cases do not ignite for the scaling model assumed.

The reference design does ignite, and for $N_e = 7.4 \times 10^{19} \text{ m}^{-3}$ and $\beta_p = 1.9$, P_{th} equals 410 MW during the burn at nominal conditions.

Table 3.1. EPR parameters and power balance

Parameter	Symbol	Value
Major radius	R_o	6.75 m
Plasma radius	a	2.25 m
Field strength on axis	B_T	4.8 T
Field strength at winding	B_{max}	11.0 T
Wall area	S	600 m ²
Plasma volume	V	675 m ³
Safety factor	q	2.5
Plasma current	I	7.2 MA
Beam power	P_b	50 MW
Beam energy	E_b	200 keV
Electron density	N_e	$7.4 \times 10^{19} \text{ m}^{-3}$
Electron temperature	T_e	13 keV
Ion temperature	T_i	12.2 keV
Energy confinement time	τ_E	3.7 sec
Particle confinement time	τ_p	9.6 sec
Poloidal beta	β_p	1.9
Total beta	β	0.03
Thermal power	P_{th}	410 MW
Neutron wall loading	P_w	0.55 MW/m ²
Impurity level ($t = 50$ sec)	Z_{eff}	1.34
Neutral density ($r = a$)	N_o	10^{15} m^{-3}
Characteristic length	λ	5 cm
Alpha power	P_α	+82 MW
Transport power	P_{tr}	-66 MW
Bremsstrahlung power	P_{br}	-9 MW
Line radiation power	P_l	-6 MW
Synchrotron power	P_{synch}	-1 MW
$\sum_k P_k = 0$ (ignition)

Fig. 3.1. EPR steady-state operating points, P_{thermal} vs N_e .

3.1.3 Impurity Modeling and Effects on System Dynamics

The SYSTEMS code is used to model time-dependent behavior of impurity and alpha-particle build-up. The impurities result from sputtering due to a variety of processes: the dominant sputtering term for an edge temperature of 100 eV is due to charge exchange. The charge-exchange flux is computed from radially dependent neutral and plasma densities. Line radiation from the sputtered impurities is computed from radially varying profiles of impurities, density, and temperature.

Plasma behavior has been computed for a variety of assumptions about impurity diffusion, divertor (or general impurity control) efficiency, neutral density, and wall material. The plasma ion temperature is shown as a function of time in Fig. 3.2 for various assumptions. Note that if a stainless steel wall is used and the impurity confinement time is equal to that of the plasma

fuel ions, a steady-state burn requires 90% divertor efficiency and 90% shielding of incoming impurities. In the case of a low-Z wall material (taken to be carbon), a 100-sec burn is attained if the impurity confinement time is on the order of the ion confinement time. Impurity accumulation limits the burn to ~50 sec. The results are sensitive to the neutral densities assumed and the sputtering model. In the absence of wall impurities, alpha-particle build-up ($\tau_\alpha = \infty$) could limit the burn to ~120 sec.

3.1.4 Poloidal Field System Design

The poloidal field system and magnetic shielding for the ORNL EPR are described in Part 2 of this report. The system is consistent with a long-pulse, high- β_p tokamak with superconducting TF coils. It is nearly an optimal system, in the sense that it minimizes the volt-seconds required to set up the plasma current and permits an increase in pulse length and duty cycle. In Fig. 3.3(a), the magnetic flux lines are shown for the EPR, for a plasma equilibrium with MHD parameter $\beta_J = 1.33$ ($\beta_J \approx \beta_p$). In Fig. 3.3(b), the flux plot which results from a more conventional poloidal field system is shown under similar conditions. The EPR system in Fig. 3.3(a) minimizes the pulsed field at the superconducting TF coils (reduces it by a factor of six) in order to permit the highest toroidal field to be attained with the least risk of superconduction quenching. It has equilibrium field coils coupled closely to the plasma so that they will respond rapidly to large increases in plasma β . An intrinsic advantage of such a system is a substantial reduction of power supplies (by 41%) and stored energies (by 34%) as compared to the designs without magnetic shielding. In view of the uncertainties in the development of large, superconducting, high-field coils (TF) and pulsed high-field coils (OH-primary), the fact that the magnetic field shielding system makes available the options of using simpler superconductor coil designs makes it a likely candidate for future tokamak reactors.

1. REFERENCE CASE: DIVERTOR EFF. = 90%, $\tau_{imp} = \tau_D$,
 $N_0(\text{edge}) = 1 \times 10^{15}/\text{m}^3$, STAINLESS STEEL WALL
2. NO DIVERTOR
3. DIVERTOR EFF. = 70%, $\tau_{imp} = \text{INFINITY}$
4. $N_0(\text{edge}) = 5 \times 10^{15}/\text{m}^3$
5. NO DIVERTOR, CARBON LINER
6. NO DIVERTOR, CARBON LINER, $\tau_{imp} = \text{INFINITY}$

Fig. 3.2. Time-dependent ion temperature for several conditions of impurity control.

3.1.5 Toroidal Field Ripple

In the design of tokamak devices, the choice of the number of TF coils depends on the toroidal field ripple that can be tolerated according to plasma physics considerations and on the space needed for access to the device. When the effect of magnetic ripples on transport is analyzed, enhanced heat conduction is found to be the limiting consideration. Figure 3.4 illustrates the radially dependent heat conduction coefficient from the dissipative trapped-particle modes, neoclassical scaling, pseudoclassical scaling, and ripple-trapping. For a particular set of parameters, typical of the plasma size of the ORNL EPR, the calculated overall enhanced heat loss due to ripples will be less than the total heat conduction loss due to neoclassical and trapped-particle modes at all radii provided that the magnetic field ripple at the plasma

boundary is less than about 2.5%. Fast-ion losses due to ripple-trapping have no significant effect on power balance, but they still need to be recognized.

Additionally, the magnetic field ripple has the beneficial effect of producing a viscous drag on toroidal rotations, except near the center of the plasma. The momentum input can be made small near the plasma center by balancing co- and counter-injected beams. In this regard, a higher magnetic field ripple near the plasma center might be useful.

In the ORNL EPR Reference Design, which has 20 TF coils, a maximum ripple of about 2.2% at the outer edge of the plasma column is found to be tolerable on the basis of plasma physics considerations; the actual ripple generated by the coil system is somewhat less than that.

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Fig. 3.3. "Snapshots" of poloidal field due to currents in shielding system ($\beta_j = 1.33$) (a) with shielding coils and (b) without shielding coils.

Fig. 3.4. Heat conduction coefficients versus plasma radius for the ORNL EPR design parameters.

3.1.6 Start-up: Breakdown, Pre-ionization Effects, Runaway Electron Production, Coupled Rate Equations

The plasma start-up problem has been considered on the basis of several effects. For very low pre-ionization, the breakdown of the plasma proceeds as an exponential increase in plasma density, and is described in terms of the gas discharge parameters E/p , α/p , V_D , and T_e . The volt-second consumption depends critically upon the pre-ionization level. At 50 V and $p = 4.4 \times 10^{-4}$ torr in EPR, a density increase by a factor of 10^{12} requires 10.5 V-sec, whereas an increase by a factor of 10^2 only requires 1.75

V-sec. During this period, runaway electrons may be produced. In this regime electron-neutral collisions are dominant. Massive runaway production can be avoided by pre-ionization levels greater than 1%. One method of pre-ionization is by resonant electron cyclotron heating; the power required by this method is in excess of 160 kW.

After this pre-ionization phase is completed, the plasma start-up during the period of partial to full ionization is described by a set of coupled rate equations for the densities and temperatures of H_1^0 , H_2^0 , H_1^+ , H_2^+ , e , or isotopes thereof, and plasma current. The numerical simulations show that wall effects are very important during start-up. The high reflection coefficient of energy at low temperature prevents the plasma from being cooled by charge exchange. For a voltage of 50 V in EPR, rapid ionization occurs only for initial n_{20} (= density of H_2^0) below $0.6 \times 10^{19} \text{ m}^{-3}$, and in this case a density far below the desired level is produced. Time-dependent results for a filling density of $0.3 \times 10^{19} \text{ m}^{-3}$ are shown in Fig. 3.5. Small radius or low-density start-up procedures may be required in the EPR and future power devices.

3.1.7 Fueling

In the most general terms, fueling a fusion reactor means (1) maintaining the density of fuel ions in the reactor core under operating conditions, while (2) removing the ash created by fusion and, simultaneously, (3) keeping a clean core, where $Z_{\text{eff}} \approx 1$.

The first topic is addressed directly, the second not at all at present, and the third indirectly. There are two methods of accomplishing task (1) in the EPR. A high edge plasma density can be maintained and dependence can be placed on diffusive mixing to maintain the central density. Alternatively, neutral atoms can be injected directly into the core. The first approach appears to be the simpler of the two methods, but it seems that violation of the "clean core criterion" casts doubt on the viability of a fueling system based

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Fig. 3.5. EPR start-up characteristics. Time-dependent density and temperature of H_1^0 , H_2^0 , H_1^+ , H_2^+ , e, and current. T_{10} and T_{20} are variable. $N_{20} = 0.3 \times 10^{19} \text{ m}^{-3}$.

on this approach. Neutral atom injection is assumed to be a feasible means of fueling, and the physics of the pellet-plasma interaction is being studied.

The existence of a dense gas ablation shielding layer which forms around a frozen fuel pellet has been analyzed and its properties and effects are being examined. This work is in progress and represents a newly developed area. In Fig. 3.6 the radial and time behavior of the ablatant shield density of a pellet is shown as the pellet penetrates into the system. The existence of such a layer could prolong pellet lifetime, enhance the penetration depth possible for a given velocity, and ease the requirements for development of pellet acceleration devices. The oscillatory nature of the ablatant shield shown in Fig. 3.6 results from the following steps: (1) creation of the cold dense plasma due to energy flux which impinges on the pellet, (2) attenuation of the energy flux by the shield until the latter is heated and expanded, and (3) repeated ablation of the pellet surface. Research is also presently being directed toward the development of pellet acceleration devices.

3.1.8 Neutral Beam Injection System

The neutral beam injection system provides the supplemental heating required during start-up and during the ignition phase of plasma operation. The magnitude of the injected power, 50 MW, makes the efficiency of the system for converting primary electric power into 200 keV of neutral beam power of great importance. Four possible injection systems have been studied and their associated efficiencies calculated: (1) positive ions, $\eta = 0.15$, (2) positive ions with direct recovery of the unneutralized ion energy, $\eta = 0.36$, (3) negative ions, $\eta = 0.50$, and (4) negative ions with direct recovery of the unneutralized ion energy, $\eta = 0.69$. The negative ion system without recovery appears to offer moderate electrical efficiency, as well as requiring fewer injectors and beam lines than the positive ion systems. It has been adopted as the reference injection system. The positive

ion system designs are considered backups and will be pursued further, since the schedule for the development of the negative ion source remains uncertain.

3.1.9 Vacuum Systems

The vacuum systems must pump the plasma exhaust, impurities, beam lines, and outgassing products. This is accomplished by a combination of cryopanel, cryosorption pumps, Roots Blowers, and mechanical pumps. In the absence of a divertor, the most critical uncertainty is the pumping of plasma exhaust during operation and the boundary neutral density. The reference system should achieve a base pressure of 2×10^{-10} torr against outgassing and should reduce the D-T pressure in the chamber by 10^3 during the 5 sec between pulses.

In the beam line, enormous pumping speeds are required to achieve low densities downstream from the stripping cell; this requires a large array of cryopanel, with attendant problems of neutron and gamma ray heating.

The main vacuum system in the EPR includes a roughing system composed of three 5000-cfm Roots Blowers coupled to three 500-cfm mechanical pumps. The high-vacuum system will consist of 40 cryosorption pumps attached to the chamber via 0.8-m^2 ducts. The overall pumping speed of the system is 1.12×10^6 liters/sec for D-T. The beam vacuum system will require 20 m^2 of cryopanel for each 8.5-MW beam.

3.1.10 Advanced Fuel Prospects

D-T is the "classical" fusion fuel for ignition experiments and first-generation fusion reactors. The advanced fuels D^3He and DD offer important prospects for exploratory plasma studies in EPR before DT ignition. These studies will permit essential evaluations of plasma behavior and plasma space parameters. Follow-on studies of the properties of an ignited DT system ($T_0 = 6\text{ keV}$, $N\tau > 8 \times 10^{20}\text{ m}^{-3}\text{ sec}$) may reveal less severe losses than the dissipative trapped-ion mode losses ($N\tau \propto T_e^{-11/2}$), thermal runaway to high ion temperatures ($T_i > 50\text{ keV}$), and large $N\tau$ values ($2 \times 10^{21}\text{ m}^{-3}$). If such high

NOTE: THE LARGE ELECTRON TEMPERATURES AT t_1 AND t_2 ARE FOLLOWED QUICKLY BY A LARGE DENSITY EXCURSION WHICH RELAXES IN $\sim 1 \mu\text{sec}$.

Fig. 3.6. Dynamic ablation of a pellet whose shield cannot keep up with the changes in plasma conditions.

temperatures were attained, the feasibility of the advanced fusion fuel reactors utilizing DD or D^6Li in a catalyzed burner mode would be demonstrated.

3.2 Magnet Systems²

3.2.1 Introduction

This section summarizes the technical evaluation of the magnet systems. These systems include the toroidal field (TF) coils, the poloidal field (PF) coils, and the associated power supplies and cryogenics. Because of the presence of huge forces (on the order of 100 MN) and energies (tens of gigajoules), special attention must also be paid to structural considerations and magnet protection schemes. The comprehensive work³ of the Superconducting Magnet Development Program at ORNL, which is aimed directly at producing the

technology needed for the EPR coils, has provided the base from which much of this study has evolved.

3.2.2 Design Considerations for the TF Coil System

The function of the TF coil system is to provide a steady toroidal field for plasma confinement and stability. The higher the field is the better will be the confinement and stability of the plasma. Furthermore, fusion power density is proportional to the fourth power of the field. Therefore, from the plasma engineering point of view, there is a great incentive to go to higher and higher fields; this incentive is counterbalanced, however, by the increasing difficulty of achieving and maintaining higher fields.

For significant power output, long plasma burn time (on the order of minutes), and minimal external power requirements, it is necessary to use superconducting magnets. Considering existing and developing superconductors, a maximum field of 11 T was chosen as the design reference point. This provides a field of 4.8 T at the plasma axis (this specific value is determined by the major radius and the blanket/shield design). The required plasma size and blanket/shield thickness dictate a magnet bore of

2. An extensive treatment of the topics outlined here is contained in ORNL/TM-5574, Part 3 of this report.

3. H. M. Long and M. S. Lubell, *Program for Development of Toroidal Superconducting Magnets for Fusion Research*, ORNL/TM-5401, Oak Ridge (January 1976).

≥ 7 m for the TF coil. This size and the non-axial symmetric field of a torus make the coil subject to large body forces; it is also subjected to pulsed fields and neutron radiation. Therefore, it is a nontrivial undertaking to design a TF coil under such severe environments. In deciding on the number and size of the coils, the field ripple at the plasma and the access for neutral beams and blanket/shield maintenance must also be taken into consideration.

3.2.3 Design Considerations for the PF Coil System

The function of the PF coil system is to create, pre-heat, and develop the plasma current, and to maintain and stabilize the plasma column. In present tokamak device design, the system consists of two different sets of coils: the ohmic heating (OH) coils, which provide the necessary flux swing, and the equilibrium vertical field (VF) coils, which serve a stabilizing function (although the latter usually also supplies significant volt-seconds).

An important design consideration is the position of the VF coils: inside or outside of the TF coils. Placing the VF coils outside the TF coils is the engineer's choice, as it makes assembly and maintenance jobs simpler. On the other hand, placing the VF coils inside could lower the ampere-turn and energy storage requirements of the PF coil system; by a proper design, this alternative could also reduce pulsed fields on the TF coils and provide an option of shaping the plasma. The physics advantages were judged to be strong enough to take the latter approach. Following the PF system design outlined in Sect. 3.1.4, a set of shield-vertical field (S-VF) coils is placed inside the TF coils, a counter-wrap winding of decoupling coils is placed at the same locations as the OH winding, and a set of trim-vertical field (T-VF) coils is used for the fine-tuning of the vertical fields. Figure 3.7 shows the locations of the different sets of PF coils. Because of the engineering difficulties anticipated in the construction, the S-VF coils are room-temperature, normal conductor windings.

Fig. 3.7. Poloidal field system schematic with magnetic shielding.

The rest of the vertical field coils and the OH coils are superconducting coils, as the power and refrigeration requirements for these coils are found to be prohibitively high if they are non-superconducting.

3.2.4 Key Features of the September 1975 Reference Design

Scoping studies⁴ for EPR, which considered the basic factors and possible choices involved with EPR, resulted in the recommendation of a September 1975 Reference Design.⁵ The attempt there was to achieve a workable design instead of an optimized one. The Reference Design embodied advanced design concepts and traditional methods aimed at obtaining a system that meets the various requirements. Although considerable development will be required before such a system can be reliably and economically built, the new ideas introduced deserve closer examination. The key features of the magnet systems in that design are:

4. M. Roberts, *Oak Ridge Tokamak Experimental Power Reactor Study Scoping Report*, ORNL/TM-5038, Oak Ridge (to be published).
5. M. Roberts and E. S. Bettis, *Oak Ridge Tokamak Experimental Power Reactor Study Reference Design*, ORNL/TM-5042, Oak Ridge (November 1975).

- (1) The TF coil has an "oval" shape, which minimizes the bending moment.
- (2) The TF coil structure, made of 304 stainless steel, is in a honeycomb configuration, with a distributed solid frame that permits no accumulated forces on the conductor.
- (3) The conductor uses multifilamentary Nb₃Sn for a maximum design field of 11 T.
- (4) NbTi was recommended for use at the low field regions.
- (5) The conductors are layer-wound to avoid joints in every pancake.
- (6) The conductor is in cable configuration, with helium flowing in strand interstices.
- (7) Each conductor cable is housed in a stainless steel hollow conduit which serves both as a helium-tight jacket and as a support structure.
- (8) The conductor is cooled by forced-flow supercritical helium.
- (9) Cryostability is obtained through a combination of a large surface area in contact with helium and a modest heat transfer coefficient.
- (10) A central column and sets of plate/beams serve as the centering force and out-of-plane loading structure.
- (11) A hybrid dewar system is designed such that the inner legs of the TF coils are enclosed in a common dewar.
- (12) The PF coil system gives special attention to reducing the stored magnetic energy and the pulsed field on the TF coils. An S-VF coil set closely coupled to the plasma current is a unique feature of this system.
- (13) The conductor cable for the PF coils (except the S-VF coils) uses NbTi in Cu and a mixed-matrix of Cu-Ni to reduce the ac coupling loss.
- (14) A fiber-reinforced insulating matrix is used for structural support of the OH coils, to avoid the eddy current loss.

The scheme of cryostabilizing a bundled cable housed in a hollow conduit with forced-flow supercritical helium is a new concept, first developed at Francis Bitter National Magnet Laboratory at MIT. Computer analysis and preliminary experimental results⁶ are encouraging. Because of the large cooled surface area and the high heat transfer coefficient of the forced-flow helium, the conductor can recover from a normalcy by depositing the Joule heating into the helium. The single-phase helium will not form bubbles as the result of heat absorption. Therefore, the behavior of the conductor can be better predicted than would otherwise be the case.

In addition to providing high fields, the use of Nb₃Sn has the advantage of significantly larger temperature differential between the bath temperature and the critical temperature. For example, at 8 T and 4.2 K the temperature margin for NbTi (carrying current) is about 1 K, whereas for Nb₃Sn it is about 6 K. Hence a higher heat input stability limit can be obtained with Nb₃Sn. The main difficulty in utilizing Nb₃Sn (and all the other potential high-field superconducting compounds) is its inherent brittleness. The use of conductor cable in hollow conduit configuration may serve to remove this disadvantage; because the conductor is not intimately bound to the structure, it does not undergo a large strain as the structure is taking the forces. The cable conductors, in contrast to large monolithic conductors, can also be supplied in very long lengths. This, plus the process of layer winding, would keep the number of undesirable superconductor splicings to a minimum.

6. M. O. Hoenig et al., "Cryostabilized Single Phase Helium Cooled Cabled Conductors for Large High Field Superconducting Magnets," paper presented at the 6th Symposium on Engineering Problems of Fusion Research, San Diego, California, November 18-21, 1975.

3.2.5 Investigations of Critical Design Features

Many of the concepts in the magnet systems, as well as in the other systems of the ORNL EPR Reference Design, were employed for the first time in any conceptual tokamak design. The working guideline since the establishment of the Reference Design has been to evaluate the original design, rather than to start a new design. Parametric studies, sensitivity checks, and detailed calculations were made on several selected subsystems. Special attention was also paid to the self-consistency and the extrapolatability of the systems to larger demonstration and commercial power reactors. The major modifications and refinement work are discussed in detail in Part 3 of this report. These areas are outlined below.

Conductor Design

- The TF winding scheme has been simplified by using "L"-shaped structural segments. Each segment encloses multiple conductor conduits. The number of welds has been greatly reduced so that significant savings in time and cost can be achieved. Figure 3.8 shows the TF coil cross section, with a detail of the structure-conductor arrangement.
- With several TF conductors running in parallel, the series current is up from 20 kA to 78 kA. This not only eases the protection problem by reducing the discharge voltage, but it also shortens the length of the cooling path.
- The TF conductor cable composition is optimized. Better understanding of the thermal behavior and better analytic tools have led to an optimization of the conductor in the forced-flow scheme. It is found that there are optimum values for the metal-to-helium ratio inside the conduit and the copper-to-superconductor ratio of the metal. Figure 3.9 plots an example of the limit of sudden heat input, ΔH_S , from which the conductor can still recover to its superconducting state as a function of the normal metal (cu) fraction (F_n) of the conductor. Starting with circular strands in the cable, the metal occupies 77% (F_{co}) of the cable space. This percentage can be changed by fluting grooves on the strands, as shown by the different F_{co} values. For each F_{co} , ΔH_S goes through a maximum as a function of F_n . This maximum first increases as F_{co} decreases and then decreases. Therefore, depending on the temperature, field, current density, and the operating heat transfer coefficient (h), there are optimum choices for the F_{co} and F_n . Also shown in the figure is a triple strand configuration case, i.e., a strand which consists of three wires twisted together, which indicates a better margin of stability than the fluted strands.
- Stability of the force-cooled cable was thoroughly analyzed. The quantitative stability limit for sudden heat input and gradual heat input can now be identified for a conductor at a specific temperature and field environment. Figure 3.10 plots the sudden heat input limit, ΔH_S , as a function of the pumping power requirement (P_p). At a given temperature and field, ΔH_S increases rapidly only in a certain range of P_p and then saturates. The pumping power (and the helium flow velocity) can thus be limited to a value close to the top of this rapid variation range, without losing the stability capability of the conductor. The limit of P_p approaching infinity gives the gradual heat input limitations, ΔH_G , of the conductor. This is the amount of heat that is required to raise the temperature of helium (plus the conductor) to the current-sharing threshold of the superconductor. ΔH_S applies to unpredictable heat sources such as flux jump and conductor slippage, while ΔH_G applies to ac losses, neutron heating, and thermal

Fig. 3.8. TF coil winding and conductor configuration.

radiation. It is the present practice to design for a near maximum ΔH_s and arrange the cooling paths to stay comfortably below ΔH_G .

- AC losses on the TF coils are evaluated. Because the pulsed fields have components both perpendicular and parallel to the conductor, it is found that an optimum choice of filament twist pitch exists. Figure 3.11 shows the contribution of the two components as functions of the twist pitch parameter, χ_w . The loss due to perpendicular pulse field decreases with tighter twist, while that due to parallel pulse field increases. Thus, depending on the relative field pulse values, an optimum value for the twist pitch can be found.
- The ac loss calculation in the PF coils has been upgraded to take into account the detailed behavior of the magnetic field in the coil inside the windings.

Coil Design

- The shape of the TF coils has been carefully reexamined. It is found that, with a minor modification to the radius of curvature, the "oval" shape is a viable optimum choice.
- An evaluation of the conflicting requirements of the plasma engineering, remote maintenance, and other engineering concerns about the S-VF coils resulted in a decision to maintain them inside the TF coils.
- The detailed arrangement of the PF coils has been changed (see Fig. 3.7) to decrease further the stored energy and ampere-turn requirements.
- Several possible connecting schemes for the turns in the OH coil are discussed. Consideration of alternatives is necessary because of differing requirements on current, voltage, and inductive coupling

Fig. 3.9. Sudden heat input stability limit for fluted and triplex strand configuration.

in the various portions of this coil. For example, one can operate the upper-half-plane turns and the lower-half-plane turns in two parallel circuits. With peak current in the conductor being 50 kA, this would only require one power supply rated at 35 kV and 100 kA. One can also divide the OH turns into five more or less symmetric groups (see ORNL/TM-5574, Sect. 4.4, for details). Then the voltage rating of the power supplies ranges from 13 kV to 15 kV.

Cryogenic Design

- The TF pumping power requirement has been minimized, with no sacrifice of stability. This is achieved by operating the helium flow at lower speeds for lower field layers, while

retaining the same stability limit (see Fig. 3.10).

- Specific cooling passages and power consumptions are identified for the OH coil central solenoids.
- It is recommended that the refrigeration cycle for the forced flow include a cold pump loop. Some experimental data on forced-flow cooling for bundled cable are available to incorporate in the refrigeration load update.

Protection

- A protection scheme which relies on external dump resistors and active switchings is proposed and examined in detail. Satisfactory results on thermal excursions, terminal voltage, and mechanical loads can be achieved by

Fig. 3.10. Field variation of the sudden heat input limit dependence on the pumping power required. Location of some Reynold's numbers, R_e and helium velocities, V , are indicated.

connecting the TF coils in several parallel loops and by isolating and discharging only the quenched coils. (Figure 3.12 shows the circuit arrangement and the operation sequence of the switches for the TF coils.) The protection problems of the PF coils are also examined.

Energy Handling

- Discussions and recommendations are made on the energy storage and transfer scheme for the OH coils. However, a satisfactory scheme cannot be identified without more extensive studies.

Materials

- The resistance of the TF coil copper matrix increases as a result of

radiation damage by as much as 20% in a year; this increase can be relieved by a warm-up to 77 K. Uncertainty concerning the effects of radiation damage on the organic insulators is a principal concern; insufficient data exists here to develop design lifetimes.

- The key features in the materials area of the original reference design are judged to be viable concepts. The integrated honeycomb coil structure continues to be the backbone of the TF coils. It is worth noting that the modified structure can also be used for a pool-boiling or convection flow-cooled conductor. Forced-flow supercritical helium-cooled conductor cable is still recommended for both the TF and PF coils.

Fig. 3.11. Hysteresis and coupling losses in superconducting wire. Both the functional dependence on the twist pitch parameter x_w and the limiting equations are shown on the graph.

No insurmountable problems in the magnet systems or their protection structures were identified.

3.3 Nuclear Engineering⁷

3.3.1 Introduction

The nuclear engineering studies on the EPR are focused on the blanket and shield design. Considerations include:

- material selection
- first radiation wall composition
- mechanical layout and design
- neutronics
- heat transfer and thermal hydraulics
- tritium handling.

Assembly/disassembly and remote maintenance requirements have significantly influenced the mechanical design and are covered as appropriate in that area.

The reference EPR concept which has been the focus of this investigation was evolved through the Scoping Study⁴ and the incorporation of the resulting blanket/shield concept into an EPR Reference Design.⁵ Since the time the Reference Design report was issued, effort has continued on the EPR reference concept.

The reference concept remains basically the same. However, a number of changes have occurred in many of the key areas. It should be emphasized that the primary intent of this effort has been to evolve and evaluate concepts that appear

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1. FOUR SYMMETRIC GROUPS
2. NORMAL OPERATION:
 S_1 AND S_1' CLOSED; S_2 , S_2' , S_3 AND S_3' OPEN
3. DISCHARGE:
 (A) CLOSE S_2' AND S_3' , THEN OPEN S_1' FOR THE QUENCHED COIL
 (B) OPEN S_3' TO DISCHARGE THE QUENCHED COIL THROUGH ITS R_D
 (C) THE POWER SUPPLY CAN BE ISOLATED THROUGH S_1 , S_2 , AND S_3 , IF DESIRED

Fig. 3.12. TF coil protection circuitry. Only one of the four groups is shown. The switching sequence for discharge is also indicated.

feasible; detailed, quantitative design analyses will follow. If a decision is made at some future date to pursue these feasible concepts through standard conceptual, preliminary, and final design phases many additional questions will certainly surface, some of which may prove to be fundamental in nature. In addition, many in-depth quantitative assessments would be required in areas which so far have only been examined qualitatively.

7. An extensive treatment of this topic is found in ORNL/TM-5575, Part 4 of the composite report.

3.3.2 Design Considerations for the Nuclear Systems

The essential functions of the nuclear systems of the EPR are to:

- absorb >90% of the available plasma energy and convert this energy to heat at meaningful temperatures
- attenuate the plasma neutrons and provide radiation shielding to protect other system components, including the windings of the toroidal field coils
- provide a system where tritium breeding could be demonstrated in at least a few experimental modules.

A minimum requirement imposed on the design is that the first structural wall handle safely the cyclic heat loads associated with power levels of several hundred megawatts, while holding impurity influx to the plasma at a rate that permits fusion pulses of at least moderate length (many tens of seconds). A design objective is to handle the anticipated heat loads safely and reliably, using configurations and materials that can be adapted to higher power reactors. The blanket must absorb a fairly high fraction of the incident plasma energy and the penetrating neutron energy. The absorption of this energy must raise the temperature of the blanket high enough to permit efficient use of the blanket coolant to produce electrical power. Heat generated in the blanket must go predominantly into the coolant, with only a small fraction escaping to the components adjoining the blanket.

The blanket will quickly become highly radioactive due to neutron reactions; thus, reliable, tested, completely remote means of assembly and disassembly of the blanket and shield must be provided. For the size devices under consideration, the blanket will be massive, and modular construction will be essential. Design of the blanket modules must facilitate a high degree of remote assembly of these units.

Insofar as possible, consistent with reliability and economy, the design of the EPR

blanket should be extrapolatable to the Demonstration Reactor Plant and should meet full power station requirements. A number of assumptions have been made in developing this blanket/shield design. These are:

- Materials should be fully qualified and commercially available.
- Coolant technology should be fully qualified and commercially available.
- Design must assure totally remote assembly/disassembly/maintenance.
- Materials should not be resource-limited.
- Design must assure adequacy of heat transfer, stress levels, and coolant pumping power requirements.
- Design must be compatible with demonstration of tritium breeding in selected experimental modules.
- Design should be based on materials, concepts, and technology that permit extrapolation to a demonstration reactor.

In addition, the following conditions have been assumed for design purposes only:

- ignition conditions
- fusion power during the burn: 800 MW(t)
- duty cycle: variable, but from 50% to 95%
- impurities controlled, but method of impurity control to be determined.

3.3.3 Materials Considerations

The reference material for the EPR blanket and shield structures is type 316 stainless steel. The design data base for this material is more extensive than for any other candidate material, and some data are available on its high-temperature properties after irradiation to high levels of both displacement damage and helium contents.

The major concern at this time is fatigue properties. No fatigue tests have been conducted on material irradiated to high dpa levels and high helium contents. The most important parameter in determining low-cycle fatigue behavior is ductility. Tensile and creep-rupture tests show that the ductility of 316 stainless steel is reduced at nearly all temperatures. It

is thus expected that its fatigue life will also be adversely affected. With this in mind, the burn time/downtime ratio was increased in the present EPR design to reduce thermal-stress cycling. In addition, an effort was made to reduce design stresses as much as possible. Additional design studies are required, as well as fatigue studies on irradiated materials, before the magnitude of the fatigue problem will be understood.

It would be highly desirable to maximize the ductility and minimize swelling of the structural material. Previous work has shown that in austenitic stainless steels the swelling rate is extremely sensitive to composition, especially the collective influence of minor elements such as carbon, nitrogen, sulphur, manganese, and silicon.⁸ Preliminary evidence⁹ indicates that small additions of titanium can improve the ductility of a material with both a high dpa level and a high helium content. These results lead us to believe that an optimum alloy similar to type 316 stainless steel can be developed that will perform satisfactorily for the life of the EPR, provided stress levels are reduced to a minimum in the design.

3.3.4 First Radiation Wall

In the design studies which have been performed in the past year, effort has been devoted to gaining an understanding of impurity control, as discussed in Part 2. No effort has been devoted, however, to designing mechanical impurity control features into the ORNL EPR design. Although it is recognized that a divertor may prove to be necessary, one was not included in the reference concept because:

- simpler impurity control measures may prove adequate,
- the problems and effectiveness of such divertors remain to be explored experimentally, and
- incorporation of a divertor into the design would entail a significant increase in complexity and size which at this stage of design investigation

should not be incorporated.

It is recognized and expected that as theoretical and experimental results of divertor studies become available they may indicate that incorporation of a divertor into the EPR design is required and potentially preferable to what are today considered to be simpler impurity control measures.

It is recognized that from a plasma point of view considerable uncertainty remains on both the qualitative and quantitative nature of the impurity problem in tokamak fusion devices the size of EPR and Demo. Although theoretical models can and are being constructed to simulate postulated behavior of particles, both those derived from the plasma and those sputtered from the first radiation wall, there is a large uncertainty as to what the actual behavior might be and this uncertainty is likely to remain large until experimental results become available. It is recognized that calculated plasma burn characteristics for EPR-size plasmas are strongly influenced by assumptions made about impurity behavior. Still, insufficient knowledge exists to make a decision as to whether a divertor is necessary, or if such a device can be designed to work. Whether a "liner" or "coating" is or is not needed, and whether it would be adequate or inadequate, remain to be determined. Similarly, gas blankets might provide the solution, but it is still not clear whether they might work and how to incorporate such a feature.

As briefly discussed here, a wide range of possibilities is recognized and developments in these areas are being followed within the fusion program. However, the judgment was made that, in view of the current limited understanding in this area, design efforts could more profitably be directed toward other systems. Accordingly,

the first radiation wall in the design continues to be a thin carbon "liner"; this has been retained in the design in performing the neutronics and heat transfer analyses.

3.3.5 Mechanical Design

The original blanket design consisted of 60 blanket segments forming an annulus around the torus (i.e., each blanket segment was designed to be continuous in the poloidal direction, but 60 such segments were used in the toroidal direction). In the revised design, each of the 60 blanket segments is subdivided into 12 blanket modules in the poloidal direction. Each of the 12 modules has its own autonomous cooling circuit (using helium to cool the bulk of the blanket). The 12 coolant circuits in each blanket segment are in parallel, and each is connected to a helium inlet and outlet header. These 12 modules are bolted to a skeletal framework which consolidates the modules into a self-contained blanket segment. The skeletal framework is designed to support the blanket modules so that they project toward the plasma with support entirely from behind, thereby minimizing the exposure of the framework to neutrons.

The modularization of the blanket design is such that symmetry exists above and below the equatorial plane. As a result, there are a total of six different-size blanket module designs in each blanket segment, plus a filler module design.

The vacuum boundary has been relocated from the outboard side of the blanket, as it was in the reference concept, to the inboard side. In addition, this vacuum wall is designed to be cooled separately from the blanket.

Recognizing the difficulty sometimes associated with visualizing the way these pieces might fit together, simplified three-dimensional scale models were employed during the design process to facilitate the development of feasible assembly and holding schemes. Also, throughout this design process it was important that the entire procedure of assembly/disassembly

8. E. E. Bloom et al., "Austenitic Stainless Steels with Improved Resistance to Radiation-Induced Swelling," *Scripta Met.* 10: 303-308 (April 1976).

9. E. E. Bloom (ORNL), personal communication to J. L. Scott (ORNL), May 1976.

be devised with careful attention to the provision of totally remote maintenance capability.

Figure 3.13 is a photograph of the current model, disassembled with all pieces lying on their sides. For convenience, only one of the blanket modules was made separately. The remaining modules are contained in one unit, with lines showing where the individual modules would be separated.

A side view of a typical module is shown in Fig. 3.14. There are basically three regions in each module. These are (1) a helium-cooled liquid metal (potassium or lithium) region as the principal neutron absorber (breeder), (2) a graphite reflector, and (3) a gamma shield region made from concentric cylinders of stainless steel. The annuli between these cylinders are filled with liquid metal. As indicated, each blanket module is cooled autonomously. The coolant is helium at 70 atm, which flows in numerous $\frac{1}{8}$ -in. stainless steel coolant tubes. The tubes traverse the inside front surface of the module and then are directed as indicated until they exit in an outlet header near each corner. The three principal regions of each module and the coolant tubes are con-

tained in the stainless steel module structure, as indicated. Each module is supported by the outer support frame, as also shown in this figure. All of the internal structure of each blanket module is supported from the top plate. Each blanket module has the same thickness (i.e., inboard to outboard) of ~ 51 cm and the same arc length (~ 1.18 m). However, the width varies depending on the poloidal location of the blanket module in the blanket segment.

Each blanket segment has mounted on its inboard surface an independently cooled, thin vacuum vessel. The wall of the vessel is a composite stainless steel structure; as envisioned for EPR it would be about 1 cm thick, with a honeycomb interior enclosed between two stainless steel surface plates, each about 0.16 cm thick. The present plan would be to flow water through the interstitial spaces of the honeycomb in the interior of the wall, and to remove the heat generated by the energy deposited on the surface.

The shield consists of a "can" structure made of 316 stainless steel. This structure contains stainless steel spheres which are cooled with circulating borated water. The borated water is circulated through a heat exchanger to remove

Fig. 3.13. Model of blanket segment (disassembled).

Fig. 3.14. Blanket module (side view).

the heat generated in the shield. Adjacent to and bolted to the outside shield container wall of the shield is a 6.2-cm layer of lead.

3.3.6 Neutronics

The neutronics and photonics calculations were performed using the one-dimensional discrete-ordinates code ANISN, a P_3 scattering expansion, and an S_{12} angular quadrature. The transport cross sections were taken from the coupled 100 n-21 γ cross-section library¹⁰ prepared for EPR calculations and compiled from the latest available ENDF/B-IV data.

Table 3.2 presents a detailed comparison of results for the present design with the water-cooled vacuum vessel. Tabulated are values of fractional heat (in percent) in the various regions for three absorber materials: potassium (35 vol%) and stainless steel (65 vol%), pure

potassium, and lithium. The cases with potassium represent non-breeding blanket modules. The case with lithium represents blanket modules where tritium breeding would occur. These cases indicate the distribution of heat fraction in the major regions of the blanket and in the toroidal field coils.

Also given in the table are estimates of displacements and gas production in the stainless steel first structural wall. The data are presented for a neutron wall loading of 1.0 MW/m². The atomic displacement rates of 10-11 dpa/year are consistent with values obtained for earlier EPR blanket designs investigated at ORNL and are also consistent with estimates obtained by other designers. The gas production values of ~400 appm for hydrogen and ~130 appm for helium are also consistent with earlier estimates.

The tritium breeding ratio for the case where lithium is the absorber is also presented. Note that the value of 1.15 does not account for the adverse effect of penetrations on the breeding; it is estimated that penetrations will amount to about 5% of the blanket volume. During the course of the study, work has been

10. D. M. Plaster, R. T. Santoro, and W. E. Ford, III, *Coupled 100-Group Neutron 21-Group Gamma-ray Cross Sections for EPR Calculations*, ORNL/TM-4872, Oak Ridge (April 22, 1975).

Table 3.2. Fractional heating rate in the current design as a function of blanket absorber material

	Absorber Material		
	K + SS-316	Li	K
Total heating rate (W/n/sec)	3.18×10^{-12}	2.73×10^{-12}	2.83×10^{-12}
MeV per neutron	19.84	17.04	17.66
<u>Fractional heating rate in reactor components (%)</u>			
Graphite curtain	1.57	1.56	1.61
Vacuum vessel with water coolant	5.32	5.52	5.30
Blanket			
First structural wall	11.89	8.72	10.78
Absorber compartment	65.89	49.59	30.79
Reflector ^a	6.19	15.72	24.84
Gamma-ray shield	<u>6.02</u>	<u>12.27</u>	<u>16.56</u>
Total	89.99	86.20	82.97
Shield	3.11	6.61	10.03
TF coil			
Inner SS-316 structure	5.88×10^{-4}	3.37×10^{-3}	5.68×10^{-3}
TF coil windings	8.90×10^{-4}	5.31×10^{-3}	8.67×10^{-3}
Outer SS-316 structure	1.05×10^{-5}	6.45×10^{-5}	8.00×10^{-5}
<u>Radiation damage</u>			
Displacements per atom/year ^a	10.9	10.7	10.1
Gas production (appm/year)			
H	424	452	421
He	129	134	129
Tritium breeding ratio	--	1.15	--

^aBased on an effective displacement energy of 40 eV.

initiated to develop a three-dimensional calculational capability to determine the effects of penetrations on the estimates made to date.

3.3.7 Heat Transfer and Thermal-Hydraulics

Although alternative coolants have been considered, as discussed in Part 4 of this report, the heat transfer and thermal hydraulic design have been performed assuming low-pressure, low-temperature water as the coolant in the vacuum vessel, and helium gas at 70 atm as the coolant for the bulk of the blanket. The thermal-hydraulic parameters for the vacuum vessel coolant system (assuming water) and for the blanket (assuming helium) are as follows.

In the vacuum vessel system, the ΔT is

$\sim 80^\circ\text{C}$, with an inlet temperature of 38°C and an outlet temperature of 121°C . The water flows at ~ 50 psia. The surface heat flux is ~ 27 W/cm², and the neutron and gamma energy heat load that must be handled by the water coolant system is about 7% of the total. This system appears capable of handling the heat loads associated with the assumed power levels (i.e., ~ 1 MW/m²) and also appears capable of handling much higher heat loads.

Numerical calculations of the temperature distributions in the blanket were performed using a generalized heat conduction computer program designed to solve steady-state and transient heat conduction problems. Transient cyclic calculational results for selected duty

Table 3.3. Thermal cycling results for selected blanket locations

Times within the cycle		Average output power [MW(t)]	Vacuum vessel front surface		Blanket structure front surface		Lithium peak values	
Burn (sec)	Down (sec)		Temp (°C)	Var (°C)	Temp (°C)	Var (°C)	Temp (°C)	Var (°C)
100	100	400	131	±47	278	±35	423	±27
100	50	533	130	±35	297	±29	490	±17
100	10	727	136	±14	317	±7	555	±4
200	10	762	137	±13	318	±7	571	±3

cycle cases are presented in Table 3.3. Results are given for three cases with a plasma burn time of 100 sec and downtimes varying from 10-100 sec. Results for a case with the plasma burn time increased to 200 sec and a 10-sec downtime are also given. The duty cycle selected for EPR consists of a plasma burn of 100 sec and a dwell time of 15 sec. These data suggest the following noteworthy observations:

- Temperatures are generally reduced in the current design compared to the earlier design of September 1975.
- The vacuum vessel coolant system has the capability of maintaining cool metal temperatures at the front surface.
- The cyclic temperature variation can be a problem, unless the cycle downtime is kept very small (or unless some modulation in the coolant system is employed).
- Cycle downtimes on the order of 10 sec with any reasonable length of burn time result in modest cyclic temperature variations.

In addition, the temperature variations due to the presence of metal structures in the current design are lower than those of the original design. The front plate of the vacuum vessel experiences gradients of $\leq 20^\circ\text{C}$, depending on the duty cycle. The temperature gradients in the stainless steel structure members of the blanket itself are $\leq 15^\circ\text{C}$.

The temperature distribution through the blanket during a given cycle of the 100/10 (burn/downtime) duty cycle is presented in Fig. 3.15. The figure contains a number of temperature

distributions giving the temperature distribution at the end of the plasma burn (maximum temperatures), at the end of the downtime (minimum temperatures), and for several intermediate times. These results indicate that all temperatures remain below acceptable upper limits at all locations in the blanket. The results also show the magnitude

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Fig. 3.15. Temperature distributions in blanket for plasma burn of 100 sec, duty cycle 0.91. The band of data represents different times in the cycle.

of the temperature variation to be expected during the cycle in the different regions of the blanket.

As indicated by these results, temperature variations are for the most part modest in all blanket regions. However, in the separately cooled vacuum vessel significant cyclic temperature variations do occur under the design conditions evaluated, and additional design effort would be warranted to identify changes that could permit operation with reduced cyclic temperature variations.

3.3.8 Tritium Handling Systems

The Experimental Power Reactor (EPR) operating at the design point of 800 MW(t), derived from fusion reactions, would consume 115 g of tritium per day. If the fraction burned were 6% and no recycling took place in the plasma, this would correspond to a tritium process rate of 2 kg (20 MCi) per day.

In addition to the equipment used to evacuate the plasma vessel and the neutral beam injectors and maintain a vacuum there, the tritium handling systems include equipment for:

- purification of the deuterium-tritium mixtures for recycle to feed systems
- tritium storage systems
- separation of hydrogen isotopic impurities from deuterium-tritium mixtures
- plasma fueling and injector feed systems
- preparation of wastes containing tritium for disposal.

The vacuum system considerations were summarized above in Sect. 3.1.9. The remainder of the tritium handling systems are covered below.

The purification technique chosen for the EPR reference design involves uranium beds, perhaps with additional palladium membrane purification on the exit stream. A daily processing rate from the plasma of 2 kg T and 1.3 kg D could be handled by a 160-liter uranium bed (with 50% excess capacity).

Storage of tritium as either a gas or a solid tritide (probably uranium tritide) is considered feasible and safe. There is consid-

erable experience with both methods, and there appears to be little reason for choosing one over the other on the basis of safety. The daily volume of deuterium-tritium from the plasma would occupy 16,500 liters, at 1 atm and 300 K. Surge storage containers of 300-400 liters may be desirable for feeding deuterium-tritium mixtures at 1 atm to either the isotope separation or plasma fueling systems, but the bulk of the fuel storage will be on low-volume uranium beds.

One or more isotope separation methods will be required to prepare deuterium for the injectors and to remove the trace of hydrogen produced by D-D reactions. The system designed for the first purpose will be the larger, but the two operations could be carried out using the same equipment if this were judged desirable.

Requirements for plasma fueling and neutral beam injector feed are not yet sufficiently well defined to allow conclusions concerning tritium processing.

Tritium containment and cleanup systems are provided to minimize leakage of tritium to the reactor building and the environment. These systems consist of enclosures which surround all components and piping carrying significant amounts of tritium, equipment to clean up the atmospheres in these enclosures and to clean up waste streams from the tritium processing system, an atmospheric cleanup system for the reactor building, and a system for the disposal of tritiated water. The total volume of the secondary enclosures will be minimized; only rough estimates of the volume can be made now.

The reactor building volume is expected to be about 4,000,000 ft³. A recirculating building-atmospheric cleanup system with a flow rate of 16,700 scfm and a decontamination factor of 10² for tritium was determined to be adequate for this volume. If the stack height is 100 m and the site boundary distance is 200 m, the whole body dose at the site boundary during 2 hr of fumigation conditions following a release of 5 x 10⁷ Ci of tritium to the reactor building would be approximately 0.62 rem, which is

appreciably lower than the reference limit of 25 rem set forth in 10 CFR 100.¹¹

The system for the disposal of tritiated water consists of a large tank for handling the regeneration load from the reactor building system, a smaller tank for handling regeneration loads from the enclosure and process stream cleanup systems, and appropriate filters, valves, sampling equipment, and pumps. A 10,000-gal tank will be sufficient for the regeneration load associated with 3 weeks' operation of the reactor building atmospheric cleanup system.

The blanket recovery system is one of the least understood items in the tritium processing system, and considerable experimental work will be required to determine the most promising system and to provide the necessary design data.

Since a molten lithium blanket appears to be the most promising means of achieving tritium breeding for tokamak fusion reactors, that is the most likely material to be tested in EPR. Three recovery techniques have been seriously considered at ORNL for the removal of tritium from liquid lithium:

- extraction with a molten salt
- permeation through a niobium window
- sorption on a hydrogen-gettering metal.

3.4 Engineering¹²

3.4.1 Introduction

Engineering concerns, analysis, and design clearly pervade the entire EPR study. As defined in this chapter, engineering refers specifically to EPR system-wide studies of power conversion, structural design, remote maintenance, and project engineering topics. Engineering work forming an intimate part of other activities, such as blanket systems, tritium handling, etc., is discussed separately in sections dealing with those areas.

11. "Reactor Site Criteria," *Code of Federal Regulations*, Title 10, Part 100 (10 CFR 100), U.S. Office of Federal Registry (April 1975).

12. An extensive treatment of the topics outlined here is contained in ORNL/TM-5576, Part 5 of this composite report.

3.4.2 Power Conversion System and Heat Balance

The EPR has a nominal thermonuclear power output of 410 MW(t) at a duty cycle of 87% [equivalent to an average power input of 356 MW(t)], all of which was initially assumed to be transferred from the blanket to the power conversion system. Twenty percent of the power is transferred there by the process of water-cooling the first wall, and the remaining 80% is transferred by the process of helium-cooling the liquid metal absorber portion of the blanket. The determination of the initial power conversion system and associated heat balance was based on blanket coolant temperatures of 38°C inlet and 121°C exit for the water and 204°C inlet and 371°C exit for the blanket.

An analysis based on these parameters indicates that utilization of the thermal energy from the pressurized water loop of the first wall as input to a power conversion cycle would not be practical. Thus, this energy was set up in the design to be dissipated to the environment by evaporative cooling provided by mechanical draft towers. For the 285 MW(t) of energy removed from the blanket by the helium loop, the compressor power needed for circulating the helium was estimated to be 6.7% of the blanket thermal energy, or 19 MW. Thus 304 MW(t) is available as heat input to the steam generator. An overall energy schematic of the thermal energy from the blanket is given in Fig. 3.16.

The cyclic operation of the reactor will result in variations in the temperature of the helium coolant. A thermal energy storage system is required to minimize variations in the rate of delivery of thermal energy to the power conversion cycle. An intermediate heat transfer and energy storage loop would introduce an additional temperature difference between the heat source and the fluid in the steam generator, and would consequently result in a decreased thermal efficiency. An alternative considered in this study, which avoids the penalty of an additional temperature difference, is a helium branch loop for energy storage, shown schematically in Fig. 3.17. This concept would produce helium at essentially a constant temperature at the steam

Fig. 3.16. Overall energy schematic for thermal energy from EPR blanket.

Fig. 3.17. Schematic diagram of helium coolant loop with branch loop for energy storage.

generator by varying the helium flow rates through the branch and parallel flow paths during the operating cycle of the reactor. The necessary helium storage capacity of the branch loop is estimated to be 934 m³.

The net thermal efficiency of the reference cycle is 28.4%, based on a generator output of 86.3 MW(e) and a thermal input to the steam generator of 304 MW(t). However, if the power input to the helium compressor [19 MW(e)] is deducted from the generator output, the remaining power output is only 67 MW(e), and the net efficiency [based on the average total blanket output of 356.5 MW(t)] is 18.8%.

An alternate power conversion system was determined which utilizes both the high-temperature thermal energy in the blanket and the low-temperature, first-wall thermal energy in the water loop. Efficient use of the low-temperature energy requires an increase of the maximum temperature of the primary helium coolant to 430°C and of the pressurized water to 207°C. Although this alternate power conversion system design increases the net plant efficiency from 18% to 28%, a design evaluation has not been made to determine the impact of the high-temperature coolant operation.

3.4.3 Structural Design

The structural mounting elements of the EPR do the following:

- react to electromagnetically induced coil forces
- support gravity loads
- support inertial load induced by seismic excitation
- provide an interface between systems to facilitate assembly
- provide a mechanism for alignment control.

The mounting concept for each major component is listed in Table 3.4.

The primary considerations for the definition of the structural support components were (1) mechanical loadings, (2) environmental loading, and (3) codes and standards.

Mechanical loading is defined here as any loading arising from the operation of the device: i.e., electromagnetic, pressure, or thermal. Only electromagnetic loads were used in this study, since they are the major loading on the support systems. A sketch of a typical coil, showing the direction of current and the resulting electromagnetic forces, is shown in Fig. 3.18. Figure 3.19 gives the intensity of the

Table 3.4. Structural component mounting concepts

<u>Components</u>	<u>Mounting concept</u>
Blanket supports	Hanger rods attached to shield
Shield supports	Cantilevered posts attached to base of machine
Toroidal field supports	
In-plane supports	Cylindrical center support hub
Out-of-plane supports for torsional forces	Box beam and plates at top and bottom of TF coil which transfer torsional loads to center support hub
Poloidal coil supports	Built-up box beam configuration for outer coil supports which transfers their forces to the TF coils and TF coil supports
Machine-to-ground supports	Posts between concrete foundation and machine to support gravitational and seismically induced forces

Fig. 3.18. Electromagnetic system current and force directions.

out-of-plane electromagnetic forces imposed on the coils and, hence, on the coil supports that were considered in this design.

Environmental loading is defined as those loads resulting from gravity, high winds, earthquakes, and other natural occurrences. With the exception of gravity, no environmental loading has been defined for an experimental fusion device; hence only minimum consideration of such loads has been given to the EPR design.

However, capital investment and safety requirements will dictate that these effects ultimately be considered.

As with environmental loading, no specific code or standard has been identified as being applicable without modification to an experimental fusion device. General guidance may be obtained by referring to the Code of Federal Regulations as it applies to fission reactors and related apparatus. The EPR program will make

necessary a code and standards effort for structural components, including their design, fabrication, and operation, similar to that which has proceeded in the fission reactor area. Work is under way to establish the character of such standards through projects like the Large Coil Program (LCP) in the Superconducting Magnet Development Program (SCMDP) at ORNL, and the Magnet Safety Studies of Brookhaven National Laboratory.

3.4.4 Structural Analysis

Results. The detailed analysis presented in Part 5 of this composite EPR report represents an extension of the work reported in the EPR Scoping Study for FY 75. New areas considered this year were the effects on the TF coils of out-of-plane loads resulting from vertical fields, and the problem of a fault condition. The relatively small vertical field was found to

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Fig. 3.19. Distribution of out-of-plane force on TF coil (vertical field of 0.1 T).

have a major impact on the overall structural design. The TF coils which are optimized for the large in-plane radial forces have relatively little resistance to out-of-plane loading. The fault condition (zero current in a single TF coil) also resulted in substantial deformation and stress in the reference design structure. To provide adequate restraint against out-of-plane and fault loadings, additional TF coil support must be considered. Three alternatives have been examined:

- additional structure which will essentially close the access between coils
- additional structural material as part of the TF coil
- external support structures like the Princeton Large Torus (PLT) torque frame.

Techniques. The "Reflective Symmetry" method was used to simplify the three-dimensional finite-element analysis. The primary EPR structural components are shown in Fig. 3.20, and the full finite-element model developed for the study is shown in Fig. 3.21. The entire model was generated using plate and beam elements. The plate elements used were either triangular or quadrilateral bending plate elements. The beam elements were simple bending, torsion, and axial force members which included the effect of transverse shear.

A new capability has recently been obtained by ORNL for the NASTRAN computer program. The new capability is called "Cyclic Symmetry." The analysis described above required that one-fourth of the structure be modeled, while the cyclic symmetry approach would have required modeling of only one-twentieth of the structure. The savings in both manpower and computer time is significant, and this technique will be utilized in future design analysis.

3.4.5 Remote Maintenance

Remote handling is a necessary and critical factor in the design. The EPR device will quickly become highly radioactive due to neutron reactions in materials; this situation dictates that remote maintenance methods be used for

TOP VIEW

STRUCTURAL DETAIL

Fig. 3.20. Toroidal field coil structural system.

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Fig. 3.21. Exploded view of finite element model showing major components.

essentially all components. The practical feasibility of some remote maintenance tasks remains to be developed and demonstrated, because of complex configuration, the large size of individual components, the need for accurate positioning and reliable joining, and the limited space available for assembly/disassembly operations. Preliminary design studies have emphasized the important influence of the remote maintenance requirements on reactor design and the critical role of remote maintenance in determining plant availability and overall economics.

System operating costs are highly sensitive to remote handling factors. The work of Kulcinski and Young¹³ was extended to investigate the sensitivity of electrical generation costs to the outage time assumed for blanket module replacement. For the UWMAK-1 design, which has 12 modules, the Wisconsin study assumed three days per module for replacement time and five days per outage for cooldown and heat-up. The module replacement time appears optimistic: this evaluation examined the costs for nine and 15 days per module, with five days for cooldown and heat-up. The evaluation indicates a high sensitivity to the assumed replacement time for modest first-wall lifetimes (0-5 years) (see Fig. 3.22). Such considerations reinforce the need for substantial effort to develop design concepts which include the consideration of remote maintenance.

Key assembly questions have been examined for feasibility of the design concepts. The basic approach to the Reference Design has been to assume that the center compression hub and the large TF coils would be semi-permanent installations, and that all other components could be removed and replaced without moving them. Modular construction of the vacuum vessel, blanket, and shield units has been proposed to facilitate the design and development of remote handling equipment.

13. G. L. Kulcinski and J. L. Young, *The Influence of First Wall Lifetime on the Cost of Electricity in UWMAK Type Fusion Reactors*, UWDFM-140, University of Wisconsin (October 1975).

Fig. 3.22. Effect of module replacement time on electricity cost.

Another example of the impact of remote maintenance is in the design of the connecting joints of the shield-vertical field windings. Sixteen conductors recessed into the inner shield surface form this coil, which lies in a toroidal direction around the plasma. These coil segments are made of 2- x 2-in. copper extrusions, and they have a ½-in. hole in the center for water-cooling. Nine of these conductors are potted in a square to make up the conductors of the coil segments (see Fig. 3.23). These segments are bolted into the shield segments before the shield segments are installed in the reactor. When the shield segments are put in place, each of the 16 coil segments extends beyond the edge of the shield on each side. The coil segments in one shield match the corresponding coil segments in the adjacent shield. A square bus with a special terminal fixture is situated on each end of the segments, and a special clamping clevis engages adjacent coil terminals, completing a mechanical and electrical joint between coil segments.

Fig. 3.23. Mechanical joints for shield coil winding.

To check the assembly feasibility of the Reference Design, a step-by-step sequence was tabulated (see Part 5 of this composite report). In addition, a scale model was constructed to verify the assembly process (see Figs. 3.24 and 3.25).

Dewar topology remains a principal area of importance, difficulty, and uncertainty. A major design alternative still requiring further study is a secondary vacuum enclosure around the entire tokamak. Use of this approach would greatly simplify the remote welding required for

high-vacuum seals in the plasma vacuum chamber. In addition, a secondary vacuum vessel would eliminate the need for individual vacuum dewars around the TF coils and structural components; this change would be highly desirable, since such individual dewars pose a major design problem and complicate remote maintenance. Questions of access for diagnostics, beams, repairs, etc., must be answered before a decision is made. Design trade studies are recommended to provide a basis for a final design selection.

Fig. 3.24. Disassembly process with a scale model.

Based upon the design synthesis, a program for remote handling qualification has been identified. The review conducted this past year of the mechanical design of the EPR has led to a better understanding of the types of operations which will have to be performed in effecting remote maintenance. There are six categories of areas for which development programs must be inaugurated in order to solve the maintenance problems. These are:

- transportation and positioning of large components
- remote welding and cutting
- mechanical clamping, connecting, and fastening
- viewing and inspecting
- instrumentation
- waste disposal.

3.4.6 Project Engineering — Development of the Preliminary Conceptual Engineering Plan

At the outset of FY 76, it was assumed that the EPR was to be constructed for operation in

Fig. 3.25. Assembly process with a scale model.

Fig. 3.26. EPR conceptual design manpower schedule generated at outset of FY 76.

the mid-1980's. A preliminary engineering job plan consistent with the planning assumptions of early FY 76 was established to define the overall scope of the conceptual design in terms of manpower, cost, and schedule. A detailed work breakdown was used as a basis for the cost estimate. In addition, a survey was conducted of some recent Oak Ridge area ERDA projects in an attempt to establish an additional basis for the overall cost of the conceptual design.

As a result of this effort, it was recommended that DMFE plan for an 18-month pre-conceptual design at a cost of \$3 million.¹⁴ This plan provided for the submission of a Schedule 44 Data Sheet and conceptual design report in September 1978, and provided for the undertaking of the capital project in FY 80. The elements of the pre-conceptual design activity were defined as follows:

- Establish a system for management and control of the project within the general framework of proven standards.

- Develop a top-level schedule.
- Establish overall project objectives and individual system objectives.
- Provide design criteria for each system and subsystem.
- Establish a reference design through an evaluation of system alternatives and trade-off studies.
- Identify research and development needs.
- Develop a working cost estimate and financial schedule.
- Provide an orderly manpower build-up for the conceptual design.

The required staff for the recommended program is shown in Fig. 3.26. A total cost of \$6 million was estimated for the conceptual design effort which would produce a Schedule 44 Data Sheet and a preliminary conceptual design report in September 1978.

A listing of program milestones developed for planning purposes is given in Table 3.5 for the recommended funding level for the pre-conceptual design. Also shown is an alternate funding (Level B) which assumed that the FY 76 funding was held at \$500,000 to allow the majority

14. Letter from M. Roberts (ORNL) to F. E. Coffman (ERDA-DCTR), October 9, 1975.

of the pre-conceptual design to be accomplished in an 18-month period starting in July 1976. This would have produced the preliminary conceptual design report in September 1979.

Although the specific personnel and cost levels and dates are no longer applicable to the plan for the next step after TFTR, the elements, milestones, and sequences do form a basis for subsequent project engineering planning.

Table 3.5. EPR program milestones generated at the outset of FY 76 for pre-conceptual design, assuming two funding levels

<u>Item</u>	<u>Completion date</u>		<u>Remarks</u>
	<u>Funding Level A^a</u>	<u>Funding Level B^b</u>	
1. EPR project plan summary			
1.1 Project plan logic diagram	12/5/75	12/5/75	For each EPR system this will include an identification of the requirements for the design and qualification of the system, and an integration of all the systems into the EPR. In most instances steps required will be generic in definition and dates will be estimates.
1.2 Reference schedule	5/28/77	1/28/78	The best definition of effort required to place EPR into initial operation will be determined and included in this schedule. This will occur near the end of the pre-conceptual design.
2. Pre-Conceptual design information			
2.1 Identification of EPR systems	10/31/75	10/31/75	This will be an identification of major functional pieces of EPR, such as TF coils, vacuum pumping, tritium processing, which will be used as the basis for developing design descriptions.
2.2 Identification of EPR subsystems	1/29/76	1/29/76	This will be an identification of the significant elements comprising each of the systems designated in Item 2.1.
2.3 Identification of components	2/26/77	12/3/77	This will be a compilation of major hardware items comprising EPR which will be useful in an early assessment of materials requirements, potential technology developments, schedule, and cost estimates.
2.4 Identification of materials requirements	4/31/77	12/31/78	This will be an evaluation of the type and amount of materials required for EPR. This identification will permit assessment of the availability of basic resources, available production capability, etc.
2.5 Draft of system descriptions	6/4/77	2/11/78	This will consist of documentation by system of information similar to that identified in Sects. 1 and 2 of the SDD standards, supported by refinements in the current design.

Table 3.5. (CONTINUED)

<u>Item</u>	<u>Completion date</u>		<u>Remarks</u>
	<u>Funding</u> <u>Level A^a</u>	<u>Funding</u> <u>Level B^b</u>	
3. Assessment of technology			
3.1 Identification of major R & D requirements	12/26/75	12/26/75	This will be an identification of technology development required for EPR. It will be an ongoing process in which additional needs become apparent as the design evolves. The date has been selected recognizing the value of early identification based on existing information.
3.2 Technology program reallocation recommendations	4/3/76	4/3/76	This will be an identification of priorities in the FED technology development program as indicated by EPR design requirements and probable Demo needs. These will be determined from refinements in the reference design and considerations developed for the Demo plant.
4. Cost assessments			
4.1 Assessment of component costs, including material requirements	6/4/77	2/4/78	This will follow from the completion of the items in Item 2.
4.2 Assessment of yearly operating costs for EPR	9/18/76	4/31/77	
4.3 Manpower and resource requirements for EPR	6/4/77	2/4/78	This will be an assessment of the staffing levels required as a function of time, for the duration of the project.
5. Design report	7/3/76 6/4/77	7/3/76 6/4/77 2/4/78	These reports will summarize refinements made on the reference design through advances occurring in the intervening periods. Specific parts of the design will be improved as needed; these will be incorporated into the periodic design updates.

^aLevel A — This assumes \$1 million spent in FY 76 (a decision effective 12/75) and \$2 million spent in the next 12 months.

^bLevel B — This assumes \$500,000 spent in FY 76, \$750,000 spent in the following six months, and \$2 million spent in the 12 months following that.

4. SUPPORTING RESEARCH, DEVELOPMENT, AND DEMONSTRATION NEEDS

One of the principal aims and benefits of the EPR study is the improved coupling between the basic fusion program and advanced projects. Much of this coupling comes about as a result of the close interaction between base program staff and advanced project staff; this interaction manifests itself through continuous refinement of both parts of the match and is, therefore, difficult to measure easily. A clearly visible result of the coupling is the identification of key research, development, and demonstration needs for EPR. These identified needs (presented fully in Part 6 of this composite report (ORNL/TM-5577) serve as guidance to fusion program planners for maintaining and improving the applicability of the base programs.

4.1 Scope of the R, D, and D Needs Coverage

The identification of major research, development, and demonstration (R, D, and D) needs has followed from the technical evaluation of the ORNL EPR Reference Design. In the preparation and subsequent evaluation of the Reference Design, the key tokamak device systems were studied rather closely and the remaining, mostly peripheral, systems were given only a cursory coverage. As a result, the major needs identified represent serious consideration of the following areas:

- plasma, in terms of basic properties, performance, and source of criteria for other systems
- overall system integration
- nuclear engineering, as represented by the blanket/shield concerns
- magnets, separated into TF and PF systems
- beam heating.

Additionally, several areas of concern were identified in the brief look at all the other systems. As the EPR and related designs progress, it is expected that other, more detailed, R, D, and D needs will be defined.

4.2 Content of Each Needs Statement

The R, D, and D needs identified are organized by system in the Systems Design Breakdown developed to characterize the engineering concerns of the EPR. A listing of major topics is shown in Table 4.1. Part 6 of this composite report describes these topics in a standard format containing the following items:

- Brief statement of basic need, containing a definition of the problem and an indication of principal applications.
- More detailed statement of the technical problem.
- Classification of the type of program development needed.
 - A - applied research and development required before program can be laid out,
 - B - existing information sufficient to permit development of the relevant program,
 - C - a program exists now, although not necessarily at optimum support.
- Priority judgment.
 - 1 - a critical go or no-go issue for EPR,
 - 2 - an important cost/schedule determinant,
 - 3 - a necessary, though postponable, task.
- References - specific references for additional information on this particular task.

4.3 Example of R, D, and D Needs Identification: The PF Coil System

One of the best ways of determining the R, D, and D needs is to examine an important design area from a systems point of view and catalog all the problems encountered in the resolution of interface trade-offs. Table 4.2 presents a listing by disciplinary area of the constituent R, D, and D problems in one of the most difficult design areas: the PF system. As the various contributing elements of the design task were

Table 4.1. Index to the listing of major R, D, and D needs^a

A. PLASMA PHYSICS AND ENGINEERING			
1. Macroscopic Equilibrium	4. Plasma Heating	7. Fueling	
2. Macroscopic Stability	5. Impurity Control	8. Plasma Dynamics	
3. Plasma Transport	6. Plasma Edge	9. Diagnostics	
		10. Atomic and Molecular Cross Sections	
B. OVERALL PLANT			(1.0) ^b
1. Long Pulse, High Duty Factor, System Integration	2. System Modeling		
C. BLANKET AND SHIELD SYSTEM			(2.0)
1. Nuclear Data Base	6. Liquid Lithium Circulation		
2. Neutronics Methods	7. Coolant/Containment Compatibility		
3. Thermal Hydraulic and Heat Transfer Analysis	8. Qualification of Design		
4. Poloidal Field Effects	9. Structural Materials		
5. Electrical Insulation	10. Tritium Recovery		
(See also I and M for tritium-related areas.)			
D. TOROIDAL MAGNET SYSTEM			(2.0)
1. High-Field Conductor	5. AC Loss Analysis and Measurements		
2. High-Current Cable	6. Stress Analysis		
3. Cryostabilizing Cooling	7. Radiation Damage		
4. Conductor Joint Technique	8. Protection		
	9. Large Coil System Test		
E. POLOIDAL FIELD COIL SYSTEM			(2.2)
1. Mechanism of Carrying High Current in Rapidly Changing Fields	5. Pulsed Conductor Fabrication		
2. Cryostabilizing Cooling	6. Structural Materials		
3. Pulsed Coil Protection	7. PF System Analysis		
4. High-Voltage Capability	8. Remotely Maintainable Shielding Coil Joints		
F. STRUCTURAL SYSTEM			(2.3)
1. Structural Design Criteria	2. Thermal Insulation		
IMPURITY CONTROL SYSTEM ^c			(2.4)
ELECTRICAL ENERGY STORAGE AND DISTRIBUTION SYSTEM ^d			(3.0)
G. ELECTRICAL ENERGY STORAGE AND PULSE SYSTEMS			(3.1)
1. Ohmic Heating Energy Storage and Transfer	2. Neutral Injection Power Supplies and Components		
INSTRUMENTATION AND CONTROL SYSTEMS ^d			(4.0)
DIAGNOSTIC SYSTEM ^d			(4.1)
H. PLASMA HEATING SYSTEM			(5.0)
1. In-situ System Application	3. Mechanical Components		
2. High-Performance, High-Energy Neutral Beams	4. High-Power, High-Frequency Microwave Tubes		
I. TRITIUM PROCESSING SYSTEM			(6.0)
1. Tritium Processing			
TRITIUM STORAGE SYSTEM (see M.1) ^d			(6.1)
PLASMA FUELING SYSTEM ^c			(6.2)
J. VACUUM PUMPING SYSTEM			(6.3)

Table 4.1. (Continued)

1. Vacuum Systems/Cryosorption Pumps	2. Large Valves	
LITHIUM PROCESSING SYSTEM ^d (see C.6, C.10)		(6.4)
PRIMARY COOLANT SYSTEM ^d		(7.0)
SECONDARY COOLANT SYSTEM ^d		(7.1)
HEAT CONVERSION/ELECTRICAL GENERATOR SYSTEM ^d		(7.2)
WASTE HEAT PROCESSING SYSTEM ^d		(7.3)
K. CRYOGENIC COOLING SYSTEM		(7.4)
1. Large 4-K Hardware		
REACTOR BUILDING COMPLEX ^d		(8.0)
L. MAINTENANCE AND ASSEMBLY SYSTEM		(8.1)
1. Adaptation of Remote Handling Techniques		
SAFETY AND FIRE PROTECTION SYSTEM ^d		(8.2)
LIQUID AND GAS STORAGE SYSTEM ^d		(8.3)
BALANCE OF PLANT SYSTEM ^d		(8.4)
SITE CRITERIA/IMPROVEMENT ^d		(8.5)
M. TRITIUM CONTAINMENT SYSTEM		(9.0)
1. Tritium Containment in Metals		
2. Tritium Behavior in the Environment		
RADIATION WASTE HANDLING SYSTEM		(9.1)

^aIn each area listed, considerable additional conceptual design effort is required to develop criteria and, from them, to determine the full extent of research and development necessary to support advanced design. This list represents only the major needs perceived at this early stage of the design process; the list will be updated as the design proceeds.

^bNumbers in parentheses refer to the hardware-oriented Systems Design Breakdown dated 10/31/76, issued by the ORNL EPR Program Office.

^cBasic information inadequate for determination of hardware-related R & D needs.

^dDesign insufficient to determine any hardware-related R & D needs.

considered and solutions were iterated in order to come to a common solution, each of the problems stated here appeared.

In the plasma area, basic criteria are necessary, though they are not yet confidently available at this time. In each of the other areas, particular problems were identified that are included in the relevant needs statement whose label is given in the table. This kind of system analysis is being continued and extended to other key systems.

4.4 Summary of the R, D, and D Needs Identification

Part 6 of this composite EPR report contains the detailed needs statements for each of the 56 problem areas established. An overview of the situation can be produced by categorizing each of the needs by its program status/priority assignment. This categorization has been done and is shown in Table 4.3. Based on the table, there are a number of observations that can be made:

Table 4.2. Research, development, and demonstration needs in the PF coil system

	R, D, and D Need Number
1. <u>Plasma</u>	
• Plasma Dynamics (Start-up, Burn, Shutdown Cycle) for $\bar{B}$, V, V-sec, and ampere-turn distribution requirements	A-8
2. <u>Conductor</u>	
• Basic properties of high-current, small filament wire	E-1
• Reliable manufacture of mixed-matrix conductor	E-5
• Behavior of cable in conduit at 4 K	E-2
3. <u>Material</u>	
• Mechanical properties of plastic at 4 K for non-conducting structure	E-6
• Insulator, structure, and conductor radiation damage effects	D-7
• Fatigue properties of structure	E-6
• Construction of non-electrically conducting, helium-tight conduit liner	E-6
4. <u>Cryogenics</u>	
• Forced-flow system characteristics	E-2
• Impact of multiple coolant leads	E-7
5. <u>Electrical</u>	
• High-voltage strength of helium	E-4
• Economical power systems	G-1
6. <u>Mechanical</u>	
• Demonstration of remotely maintainable joints	E-8

- There are many questions yet to be answered.
- The majority of those needs deemed either critical or important are the subject of ongoing programs.
- There are 19 critical or important needs that could be addressed in programs for which adequate information exists.
- There are two critical areas, fueling and OH system energy handling, for which an inadequate technical base exists for prudent development of R, D, and D programs. Increased emphasis in these two areas is sorely needed.

4.5 Work to be Done

Three principal tasks remain unfinished. These tasks, which should be carried on during the further stages of the advanced design work, are the following:

- (1) Extension of the evaluation of the Reference Design to all the peripheral systems and the concomitant identification of R, D, and D needs therein.
- (2) The continuing process of screening the findings of the design work, as it becomes more and more specific, for additions or modifications to the R, D, and D list.
- (3) Development of specific recommendations about the applicability of ongoing

Table 4.3. Summary table of R, D, and D needs

	Program Status			
	A		B	
	Requires more fundamental definition of problem		Could be developed	
Priority	C		Ongoing	
1 Critical	A-7 ^a (Fueling) G-1 (OH Energy Storage and Transfer)		A-4	A-1 D-3
			A-5	A-2 D-6
			A-6	A-3 D-8
			A-8	A-4 E-1
			A-9	A-5 E-2
			B-1	C-7 E-3
			C-8	C-9 E-6
			D-9	C-10 E-7
			M-2	D-1 H-1
				I-1
				M-1
2 Important			C-4	A-10 D-5
			C-5	B-2 E-4
			D-7	C-1 E-5
			E-8	C-2 H-2
			F-1	C-3 H-4
			F-2	C-6
			G-2	D-2
			H-3	D-4
			J-1	
			L-1	
3 Necessary			J-2	
			K-1	

^aAlpha-numeric code (e.g., A-7) refers to ordering of needs statements presented in Table 4.1.

R, D, and D programs and about the structure of new R, D, and D programs that should and could now be undertaken.

These recommendations should include basic direction, specific tasks and milestones, and necessary auxiliary facilities.

5. CONCLUSIONS

5.1 About the Role and Objectives for EPR

One of the most challenging problems in the EPR work has been the development of the proper set of achievable objectives. Usually there is a dilemma: setting the objectives a reasonable increment beyond the capabilities of our present knowledge is not fully satisfying, and yet setting the objectives at the level of performance commonly ascribed to a power reactor is very costly and fraught with uncertainties. The solution to this dilemma is suggested by the study; namely, treat the EPR and Demonstration Plants for what they are commonly recognized to be, fixed points near the end of development, and then develop the plan for the appropriate intermediate device(s).

Another way of viewing this solution is to recognize that the ratio of benefit to cost must be acceptably high for an EPR. The action steps needed for this task are to be found in reserving for EPR those significant objectives that cannot be achieved in any lesser facility. Similarly, all those objectives that can be achieved in lesser facilities should be scheduled in that way. Table 5.1 contains a selection of specific technical objectives that make definite the philosophical argument stated above. The purpose of this list is to emphasize the point that a sequence of facilities and objectives can be set up in such a way as to maximize the benefit/cost of each one. In this scenario TFTR and the next step after TFTR, the latter just begun at the conclusion of this present study, perform most of the functions of the originally conceived EPR, while EPR performs the most difficult and full-reactor-like functions, those associated with nearly continuous operation and power conversion.

5.2 Findings of the Study

5.2.1 Scientific Advances and Uncertainties

Using analytical tools that have been developed and sharpened during the course of

this study, more comprehensive descriptions of and projections about plasma behavior have been generated than were possible earlier. Even though many large strides have been taken in plasma physics in recent years, a significant number of scientific uncertainties still remain to be addressed. This is true of each general scientific area described below.

Scaling. In this general area, characterized by the terms "scaling," "transport," and "performance," many parametric studies have been undertaken which take into account the sensitivities of the performance of the plasma to uncertainties in the functional form and magnitude of scaling coefficients. Nevertheless, the experimental base from which a confident judgment can be made as to which coefficient or exponent to use is still lacking. Extrapolation on the basis of trapped-particle mode scaling, even using a judgmental improvement by a factor of ten, still points to a situation that results in low benefit/cost for EPR. Also, some performance parameters are very sensitive to radial profiles of density and temperature. These effects have not been taken into account in present studies, partly because some of the relevant physics information is incomplete. This indicates a greater demand for selectivity and creativity in the theory. Moreover, those experiments in the next years which should provide the needed confirmation of new ideas and existing models can be identified: namely, ORMAK Upgrade and PLT for β and τ_E limitations, and Doublet III for elongation.

Dynamics. This area, defined as "dynamics," "fueling," "start-up," and "operations," is a step closer to actual operating conditions than is scaling; it is, therefore, less developed than the previous area. The behavior of the plasma at plasma initiation and current build-up was initially defined by best estimate. Now, at the end of the study, a consistent description

Table 5.1. Classification of demonstration facilities for EPR features

Requirement in EPR	Demonstration in		
	TFTR ^a	Intermediate System Step ^b	EPR Only
<u>Plasma Physics</u>			
Driven D-T operation of large plasma	Yes	Optional	
<u>Long-Pulse Operation</u>			
Superconducting toroidal field system (steady)	No	Yes	
Superconducting plasma driving system (pulsed)	No	Yes ^c	
Steady-state plasma fueling	No	Yes	
Steady-state operation of high-efficiency neutral beam injectors	No	Yes ^c	
Steady-state gas removal	No	Yes	
Control of impurity build-up in long pulse	Limited	Yes	
Steady-state cooling of first wall	No	Yes	
<u>Other Advanced Plasma Engineering</u>			
Noncircular plasma	No	Yes	
Divertor	No	Optional	
Magnetic shielding of TF coils	No	Yes	
High-density regime	Yes ^c	Yes ^c	
<u>Fusion Reactor Technologies</u>			
Tritium feed and recovery	Yes	Yes	
Tritium enrichment and recycle	No	No	Yes
Radiation shielding	Yes	Yes	
Maintenance of radioactive systems	Yes	Yes	
Containment of radioactive gases	Yes	Yes	
Control of fusion reaction near ignition	No	Yes	
Recovery of neutron energy in high-temperature coolant	No	No ^c	Yes
High duty cycle operation/availability	No	No ^c	Yes
Conversion of fusion heat to net electrical power	No	No ^c	Yes

^aFinal conceptual design [Tokamak Fusion Test Reactor: Conceptual Design Report, TFTR-TR-001, Princeton Plasma Physics Laboratory and Westinghouse Electric Corporation Fusion Power Systems Department (October 1975) Draft].

^bPreliminary design objectives.

^cUncertain.

of the start-up process is being developed for subsequent use. Uncertainty in the need for, requirements for, and feasibility of fueling makes any conceptual design effort in this area tentative at best, until additional experimental evidence is available. Fueling analysis and fueling technology have both been initiated in a serious way during the course of the study; preliminary models describing present pellet injection experiments and concepts of future pellet injectors have been prepared. An overall time line sequence has been constructed using the best estimates available for each element of the start-up/burn/shutdown process; but it remains for detailed experiments in the Impurities Study Experiment (ISX) and other new machines to demonstrate the applicability of many assumptions regarding transient phenomena.

Impurities. This area represents a step even further into reality; it includes "impurity control," "limiters," and "faults," conditions representing what can perhaps be called "off-nominal" situations. Strong impurity effects are shown clearly in model calculations; even modest impurity concentrations are enough to prohibit ignition in these analyses. Nontrivial assumptions concerning impurity control are built into the design criteria; no consistent scientific basis can be put forth, however, to substantiate the assumptions made, other than the implications of the recent Alcator Z=1 experiments and the unacceptable results of higher Z. The development of a basis for choosing between a point limiter and an all-enclosing "limiter surface" awaits ISX and new hot-plasma experiments. The range of fault conditions which have a severe impact on wall loading and local damage has not yet been delimited. Here again, the next generation of experiments is looked to for provision of this kind of operational experience.

Electromagnetics. Running across the various areas of plasma phenomena is the set of electromagnetic effects linking the plasma to the outside

environment. This area, most aptly described as plasma magnetics, has been the scene of much innovation and progress, which has resulted in large part from focus on EPR. Careful, detailed distributions of ampere-turns providing and matching appropriate plasma conditions have been calculated. Conception and development of the TF coil shielding idea find basis in an understanding of plasma magnetics. Investigation of this scheme leads to a comprehensive calculation of volt-second demands, but even here experimental evidence is sorely needed to verify or to guide choices in volt-second requirements in the transient states. A remaining undetermined factor is the actual magnitude of the changing magnetic fields at the TF coils which may lead to significant hardware demands on both refrigeration and ohmic heating supplies.

5.2.2 Technological Challenges

Fusion power is clearly characterized by requirements for simultaneous application of many complex technologies. Three principal areas stand out now, as they have for some time: namely, magnets, beams, and materials.

Magnets. The need for large, high-field superconducting magnets for fusion reactors has been assumed for some time, but it has remained for the EPR studies to make graphic the largeness of "large" and the magnitude of "high." Straightforward projections from current plasma physics models dictate these large coil sizes, which not long ago were thought to approximate reactor sizes. The order of magnitude increases in linear dimension from the TF coil of the first (as yet unfinished) superconducting tokamak, T-7, to that of the EPR has come to embody the size of the technology step required to support EPR. Application of large, high-field superconductors in the EPR, here viewed as the qualifying step before Demo, is certainly required for the TF coil and for at least some portions of the PF system. In making this application, which is the principal rationale for the superconducting magnet

development programs at ORNL and other laboratories, three basic problem areas stand out: conductor design, structure, and energy storage and transfer. Throughout the course of the design study, increasing attention to these key elements has supported the case for a large coil demonstration before the EPR designs are fixed for fabrication.

The specific design remarks in the sections below on conductor design and structure apply equally well to the use of Nb_3Sn and NbTi superconductors, both of which are used in the Reference Design. One of the study's clear findings is that Nb_3Sn development work is necessary if Nb_3Sn conductor is to be qualified for EPR use. Much less clear is the necessary level of support, since much of the effort¹ in conductor design and structural analysis aimed toward NbTi coils is also applicable to Nb_3Sn coils.

Conductor design: The central areas of effort are: (1) calculation of ac losses, (2) determination of cooling methods to ensure cryogenic stability, and (3) development of a specific conductor design which meets the established technical criteria and can still be fabricated economically. Although precise wave shapes of the instantaneous electromagnetic field at both the TF and PF coils are not yet available, considerable progress in the loss calculations was made through the development of analytical tools. Codes are now available for the determination of losses from the various components of spatially and time-varying fields in conductors approximating the real coil geometry.

The original Reference Design depended upon the cryogenic stability of forced-flow cooling in a hollow conductor. While this technique is still seen as an exciting possibility,

considerable experimentation is required in the development process. During this experimentation period the design has been examined in order to improve it and still retain the basic features of rigid support structure and forced-flow cooling. The winding now consists of several conductors in parallel, supported by shaped stainless steel structural forms, that are installed as the winding proceeds. This technique reduces the number of welds, increases the series current, and reduces the path length of the coolant.

The PF conductor design has remained the same, while effort has been placed upon strengthening the support for the design. The loss, heat transfer, and stability calculations worked out and performed during the study do support the original design.

Structure: The principal areas of structural concentration in this study have been: (1) the optimization of the TF coil shape, (2) the consideration of normal and abnormal, in-plane and out-of-plane forces and torques, and (3) involvement with the overall thermal isolation question. Based on our studies of active loading of the TF coil, we are now able to determine the technically optimum coil shape, leaving only considerations of actual fabrication cost of a given desired internal configuration (plasma, vacuum chamber, coils, etc.) yet to be included in a full optimization.

One key requirement recognized in the study of losses is the need for a nonconducting structure with acceptably low eddy current heating to support the central OH coil. Although the need can be clearly specified from a technical point of view, the kernel of the work still remains: namely, provision of such a material.

Another key requirement, which was recognized earlier in the study of losses, is the need to provide thermal isolation for the $\sim 4\text{-K}$ conductors while retaining usable access.

Energy storage and transfer: These electrical engineering problems are potentially among the most difficult in the EPR; they are of

1. See H. M. Long and M. S. Lubell (eds.), *Program for Development of Toroidal Superconducting Magnets for Fusion Research*, ORNL/TM-5401, Oak Ridge (April 1976).

greatest concern now, because so small a base of experience exists. Various circuits and power system options were examined for possible consideration. The principal finding is that a large, dedicated effort is required to assess this area; this finding is now being implemented at Los Alamos Scientific Laboratory (LASL).

High Power Beams. The complex area labeled "beams" actually includes neutral beams, their sources, mechanical and electrical coupling, power supplies, vacuum systems and, in some measure, tritium handling problems. Neutral beams and tritium handling are mature technologies for which one can state development probabilities and demonstration needs. The vacuum system area poses the greatest potential problems and uncertainties.

Neutral beams: At this present stage of plasma heating, the need for neutral beam injection as a heating technique is clear. As rf heating is developed in the near future other options may open, but in this study our full concentration has been on injection. The Reference Design was based upon the use of negative ion sources, in an effort to attain high-efficiency heating systems at high energies. The controlling element along this route is negative ion source development. In the present context, however, the priorities for development effort in this area are completely dominated by the pressing need to pursue positive ion source accomplishments on a schedule tightly coupled to the confinement systems experiments. Therefore, little new work was done on negative ions. The work done in the positive ion beam area, an area which takes on greater significance as we consider only an ignited plasma rather than an ignited or driven plasma as originally conceived, has led to the following finding: a combination of new, more effective penetration schemes and new, more efficient beam line components is needed to make positive ion, TFTR-type beam systems attractive for EPR use, especially with regard to overall efficiency calculations.

Vacuum: Coupled as strongly as they are to the uncertainties of particle accountability,

the vacuum technology needs are neither clear, well-defined, nor capable of straightforward achievement. The pumping requirements for the bare EPR chamber of assumed conditioning have been determined, along with pump selection and specifications. The three central remaining problems are: (1) the handling of the large gas loads transiently imposed upon the torus from the beam lines, (2) the behavior of the plasma with respect to density maintenance, and, especially, (3) the role of impurity particles. In each of these problem areas the level of difficulty to be assigned to the hardware is not yet known, because of the undetermined plasma engineering basis on which the hardware would eventually be designed.

Tritium handling: Since the neutral beam system is intimately tied to the D-T handling vacuum system, tritium handling becomes a necessary ingredient in vacuum system consideration. Standard doubly enclosed tritium handling techniques are used throughout the design.

Materials. At the study's inception, material requirements, though not quantitatively understood, were believed to be severe. By the study's completion, the net outcome of the iterations and technical evaluation were the perceived needs to increase plasma fusion power density, with an accompanying significant increase in neutron wall loading, and to reduce the off-time, with an accompanying significant decrease in cyclic thermal fatigue. The overall change in severity that results from this increase and decrease in difficulty cannot be determined simply.

In the former case (i.e., the impact of radiation damage), it now appears necessary to consider neutron wall loadings in the range of 2-5 MW/m² in economic reactor regimes. These wall loadings imply materials requirements beyond the characteristics of conventionally available candidates such as standard stainless steel. Important findings in radiation-resistance properties have led to the belief that the class of titanium-modified stainless steels can meet these severe requirements. Continued development is

required to move from an empirical method of dealing with this class of materials to a basic understanding sufficiently deep to permit optimization of the radiation-resistance properties.

In the latter case above (i.e., the impact of cyclic thermal fatigue), the problem is basically one of design, rather than of development. The original Reference Design duty cycle was set at 50%. Examination of the implications of this assumption showed severe cyclic thermal stresses arising from the long (100 sec) downtime. When this downtime was reduced by an order of magnitude, the size of the temperature swing was also reduced accordingly, and the stress level dropped. An associated material requirement, static thermal stress, was found to present severe difficulties in the original design; introduction of the water-cooled first wall reduced that requirement satisfactorily.

5.2.3 Engineering Considerations

The most conspicuous result of the EPR study is the value of the benefit/cost ratio. The benefit perceived is principally scientific and the cost principally one of engineering, in the last analysis. Enough engineering effort has been invested during the study to be able to support the following three statements.

1. "A number of issues were addressed successfully during the design process."

During the course of the design study, a number of engineering issues were addressed in some depth with successful conceptual development. "Successful conceptual development" here is used to mean the establishment of criteria, the development of potentially acceptable concepts, the successful pursuit of key design features, and synthesis of all systems into an acceptable whole. As examples of issues addressed, one can point to the blanket, the TF coil, the coil structure, parts of the PF system, and remote maintenance.

Blanket: In this vital functional area, effort was directed at a comprehensive consideration of all the principal mechanical design questions. The following were considered in the

evolution of the Reference Design:

- steady-state and transient thermal analysis
- workability and life properties of materials
- initial fabrication
- remote assembly and repair
- neutronic functional performance.

The net output of these efforts was a conceptual design of this principal component that is judged to be workable. Many significant engineering problems certainly remain, but there is now a technically evaluated and improved reference design that can serve as a secure basis for future work.

TF coil: At the beginning of the study, the TF coil design was established as an advanced, and hence somewhat speculative, idea based on forced-flow cooling in hollow conduit structures. During the study considerable attention was focused on the structural, superconducting, electrical, and mechanical aspects of the design (see Sects. 5.2.2 and 3.2 above). As a result, there now exists a conceptual design with more nearly optimized characteristics and better supported analyses than at first.

Remote maintenance: Throughout the study, the engineering question remained as to whether remote maintenance could be accomplished feasibly. In the area of the TF coil system, a technique was developed for installation, assembly, and alignment using an air pad transporter and overhead cranes. The blanket and shield were modularized with a standard set of pieces. In the area of the intertwined superconducting coils, detailed consideration was given to a demountable joint that could be assembled from afar. Finally, in terms of the tokamak system as a whole, the question of remote maintenance resulted in a minimum spacing between plasma and TF coil of 1.40 m, including a necessary 15 cm for access.

2. "Given that a number of issues have been addressed, the net result is still an extraordinarily complex device."

As presently conceived, the tokamak EPR concept is a complex assemblage of difficult engineering conditions, including:

- (1) nested tori consisting of vacuum chamber, energy conversion system, and magnetics systems
- (2) adjacent regions of greatly differing temperatures (hot shield, cold magnets)
- (3) structural loads of great magnitude, in pieces of large size and low tolerance to movement
- (4) intertwined sets of superconducting coils
- (5) pulsed operation with more than 10^5 shots
- (6) electrical loads, pulsed and steady, all of which must be optimally designed to produce net power
- (7) handling of $>10^3$ g of tritium
- (8) maintenance by remote means only.

It is our judgment that the specific EPR design as conceived during this study is so complex that successful construction and operation of an EPR in this manner is doubtful. It is difficult to quantify this statement, but the essence of the argument is the following: one or more of these complex areas can be attacked successfully, but the chances of successful solution of all problems simultaneously seems questionable with our current level of understanding. Experience with other large, complex projects leads one to the belief that there is a critical mass of problems, coupled with basic uncertainties, that become too great. It must be pointed out, however, that these remarks apply to the design as conceived. The result of this effort is to say: let us modify those elements preventing us from proceeding to our goal and then proceed with an EPR design that can be successfully implemented.

While the list of eight difficulties above reads like the chapter headings of a tokamak design manual, it is not what it seems. The first four certainly represent difficult design problems, but they are all made more difficult by the customary demand for low aspect ratio. Relaxation of that demand would permit a wider choice of design solutions. The fifth item relates directly to our ability to produce long-lived plasmas; it should serve as an incentive to produce even longer pulses rather than to bemoan the engineering

difficulties attendant on short pulses. The sixth item speaks basically to the low level of the power output; a key finding of the study certainly is to seek ways of increasing this power. Items seven and eight are inherent to the system but are amenable to ingenious design efforts.

3. "Extrapolation to the Demo is an important design criterion."

Another theme present throughout the EPR study has been the "extrapolation to Demo" perspective that has served as a criterion for acceptance or rejection. In the plasma engineering portions of the study, in the magnet and beam technology area, and in most of the engineering areas, a reasonable case could be made for acceptable extrapolation. In at least two instances, however, a serious doubt remains about the appropriateness of the EPR scheme for Demo service and, therefore, for fusion use. One instance is the OH driving coils, which at present are located within the shield yet outside the blanket. The total question of feasibility under Demo conditions has not yet been fully investigated. At present the perceived need to locate these coils close to the plasma competes with the engineering preference for access, protection of the insulation, and reduction of power losses through the use of superconductors rather than normal conductors. The second instance where there is some question of extrapolation to the Demo is the set of implications arising from the use of a complex impurity control scheme. In the EPR Reference Design only passive techniques, such as wall conditioning, were seen to be necessary and describable. As this process unfolds, more serious consideration of the EPR/Demo/CPD sequence may force a different philosophy of impurity control in EPR designs.

5.2.4 Project Planning

In this study, which emphasized technical evaluation of the Reference Design, project planning activities were limited to the following topics:

- Direct extension and use of the technical work in the preparation of a project

format. This activity included:

- (1) system breakdown
 - (2) parameter listings
 - (3) R, D, and D identifications.
- Abstraction of the technical work in order to provide a basis for early consideration of schedules. The types of schedules were:
 - (1) elementary schedule based upon experience in fission work
 - (2) logic schedule for the overall and blanket systems.
 - No cost information was developed at this time. The program plans indicated that cost and schedule development would occur in FY 77.

5.3 Principal Areas of Choice

The fusion power density, which is of great concern in the development of an economical reactor, can be written in the following way:

$$\frac{dP}{dV} = n^2 \frac{\langle \sigma v \rangle}{4} E_f, \quad (1)$$

where n is plasma density, $\langle \sigma v \rangle$ is the cross section for fusion, and E_f is energy per fusion event.

The normalized plasma pressure, β^2 , is a measure of the efficiency with which the magnetic field is used and is defined as $nT/(B^2/4\mu_0)$ (where T is total plasma temperature and B is total magnetic field). When β is used, then Eq. 1 becomes:

$$\frac{dP}{dV} = \frac{\beta^2 B^4}{16\mu_0^2} \frac{\langle \sigma v \rangle}{4T^2} E_f. \quad (2)$$

The plasma parameter β can also be written approximately as

$$\beta = \frac{\beta_p}{q^2 A^2} C^2 = \beta_0 C^2, \quad (3)$$

where

- β_p = ratio of plasma pressure to poloidal magnetic field pressure
- q = plasma stability safety factor
- A = aspect ratio $\equiv R/a$ (ratio of plasma major radius to minor radius)

C = variable representing the noncircularity of the plasma shape.

When Eq. 2 is evaluated with a plasma in a pure state (i.e., where the effective composition normalized to hydrogenic species is $Z=1$), then we have

$$\left. \frac{dP}{dV} \right|_{Z=1} = \frac{\beta_0^2 C^4 B^4 \langle \sigma v \rangle}{16 \mu_0^2 T^2 4} E_f. \quad (4)$$

Serious consideration has been given to increasing dP/dV through improving β_0^2 , C^4 , and B^4 while $\langle \sigma v \rangle T^{-2}$ and E_f are kept fixed and $Z=1$ is achieved and maintained.

Sections 5.3.1-5.3.4 below present discussions of the arguments associated with proceeding along alternative paths.

Modifications to the design, such as varying the thickness of the blanket in order to place the plasma in a higher field region, also affect the power density. This kind of improvement is viewed as a design optimization rather than as a principal alternative that has fundamental impact on the fusion program's research and development effort. Design optimization would be done in the subsequent conceptual design phase.

It must be kept clearly in mind that the final cost of electrical fusion power is almost certainly not directly proportional to the basic fusion power density. Therefore, while the scientifically based incentive for increased power density places a premium on high B and, with lesser certainty, high C , plant economics must be considered before a firm value is established for each quantity. Definitive plant cost estimates, including development costs, schedule slippage, etc., have not yet been done; therefore, all these analyses must still stand the test of engineering cost evaluations.

5.3.1 Higher Magnetic Field

Since dP/dV is proportional to B^4 , the incentive for increasing B is very strong. Three arguments, presented in the sections below, can be developed to place a limit on the magnetic

field strength used. For the purpose of this discussion, a base field strength of 11 T maximum at the coil (the Reference Design case) is assumed.

Direct Magnet System Cost. As a starting point, the cost of the TF magnet system for which preliminary costing procedures are being developed can be considered. In the absence of any definite design for the EPR magnets, the cost of the TF magnet system can be estimated using a technique developed for the Toroidal Test Assembly (TTA) superconducting magnet system.² This technique of scaling labor and material costs with size and field strength as appropriate in each of the component areas is judged to be better than simple scaling of cost with stored magnetic energy.

A point cost calculation indicates that the relative costs for nominally equal power outputs of a postulated 16-T and the reference 11-T EPR magnet systems are 1.5/1. The principal component of cost in both designs is the high field Nb₃Sn wire; the ratio holds whether the costs used in the calculation are today's wire costs or future cost,³ which are expected to be lower. In this calculation, which is based on the use of NbTi where the field is 8 T or less and the use of Nb₃Sn from 8 T up, the large difference in the costs is caused by the steeply rising price of using and producing Nb₃Sn wire at higher fields in the range of 11-16 T. As the peak field increases from 11 T to 16 T, the allowable current density drops and the amount of wire per ampere goes up. The cost in the 16-T case is thereby significantly higher than in the 11-T case, both because more of the coil must be Nb₃Sn wire (16-8)/16 vs (11-8)/11 and because more wire per ampere must be used at the higher field level.

Indirect Magnet System Cost. The total costs associated with the use of advanced technologies such as those implied by 16-T magnets, and the associated compatible high wall loading materials, 0.5-0.75 MeV neutral beams, and high-performance OH circuits are not the only capital construction costs; a financial assessment must also include the development cost and the cost-equivalent associated with the risk of failure in the development and delay in the schedule. In each case, it is not now possible to put a dollar figure on the additional cost required in each technical area, but certainly each of these already difficult technological problems is made more difficult by the higher demands on field strength. Therefore, it can be concluded that power density is not the lone primary cost determinant, but rather some combination of power density and technology demand probably yields a better approximation of the costs. The near equivalence of the magnet system costs in our calculations and in the work by Wisconsin⁴ seems to support this idea of constancy of cost for given output. Based on the 1.5/1 cost ratio for 16-T and 11-T magnet sets, it can be estimated that any reductions in total plant cost coming from smaller sized blanket, shield, and structural materials in coils of reduced size will be offset by the use of additional, more expensive, high-field superconducting material. Further support for this argument comes from capital cost comparison for the Light Water Reactor (LWR) and the Liquid Metal Fast Breeder Reactor (LMFBR), which differ by an order of magnitude in power density. The LMFBR, with the higher power density, is slightly more expensive for a given output because of the more complex technology used.

2. P. B. Thompson, *Procedures for Parametric Studies of Costs of Superconducting Toroidal Test Assemblies*, ORNL/TM-4821, Oak Ridge (May 1976).
3. J. Z. Z. Stekly and W. F. B. PUNCHARD, *Impact of High Critical Temperature Materials in Fusion Toroidal Magnets*, MCA-TR-AEC-1.1, Magnetic Corporation of America, Waltham, Massachusetts (June 1975).

4. R. W. Boom, R. W. Conn, and J. W. Lue, "Apparent Difficulties Associated with High Magnetic Field, Large Tokamak Power Reactor Design," paper presented at the 6th Symposium on Engineering Problems of Fusion Research, San Diego, California, November 18-21, 1975.

Relation of Magnet System Cost to Overall Plant Cost. As a third component of this consideration, the relationship between the highly visible part of the cost, that of the nuclear "island," or tokamak device, and the total plant cost has also been considered briefly. Fission reactor experience and fusion reactor studies were both used. In the cases of both the LMFBR and the LWR, the nuclear island, including escalation, contingencies, etc., accounts for no more than 20-25% of the total cost. Similarly, in the TFTR and UWMAK designs, the cost of the "reactor" portion is much less than half of the total. In this light, the overall cost optimization process is not directly proportional to changes in power density.

Recommendation. The specific arguments presented here support the conclusion that additional work is needed in the costing area, and that a specific 16-T point calculation should not be used as a general criticism of fields high relative to 11 T. The optimum practical field value may well be in the range of 11-14 T, where increasing gains in plasma performance are not outweighed by rapidly increasing magnet cost.

5.3.2 Plasma Shape

The incentive for increasing the noncircularity⁵ of the plasma cross section is of equal magnitude with that for magnetic field strength, namely $dP/dV \sim C^4 B^4$. The set of factors to be considered in the range of possible value, engineering costs, and theoretical basis is also remarkably similar, with an important difference in the coupling to plasma behavior.

From circular to highly elongated plasma cross sections, C can vary over an order of magnitude (from 1 to 10). Similarly, B can theoretically range from 2.5 T to ~ 25 T. In both cases, the

upper half to two-thirds of the range is only of academic interest at this time.

Large values of the elongation C require high currents in shaping windings near the vertical plasma boundaries. Maintenance of the currents, against electromagnetic forces in present designs, and against high radiation and heat loading in reactor designs, is a difficult task; the problem is intrinsic to currents close to the plasma, as evidenced in the electromagnetic shielding scheme described earlier. The magnitudes of the tasks associated with ever larger values for C are difficult to gage quantitatively without underlying design studies.

A principal point of difference between the considerations involved in increasing C and those involved in increasing B lies in the certainty of the incentive. The functional dependence of dP/dV upon B is firmly grounded in theory and experiment. Many carefully controlled experiments have verified the improvement of thermal insulation and other plasma properties with increasing magnetic field; field increase also, by its nature, makes no qualitative changes in the basic structure of the plasma. The role of noncircularity, on the other hand, is not fully understood either from theory or experiments. Although the direct functional dependence of dP/dV on C may be written as $[(1 + C^2)/2]^2$, it is the indirect dependences that are in question. What impact is there on the basic physical processes when $C > 1$? What change, if any, is there in allowable q ? Additional experimental results and continued iteration of the theory are needed to develop further the end result of changing shape.

The net result of these uncertainties is the decision to determine and aim for the highest practical value for C . Our calculations indicate that $C \sim 1.6$ represents a practical maximum, in that the following hold true: (1) a considerable benefit accrues from this modest elongation, (2) the nominal TF coil shape adequately includes this noncircularity, and (3) the necessary ampere turns seem to be possible, although rather difficult from an engineering sense.

5. T. Ohkawa and T. H. Jensen, "Parameter Studies for Tokamaks and Doublets," *Plasma Phys.* 12:789-797 (1970).

5.3.3 Impurity Control

Achievement of the power density indicated by the dP/dV relationship is very much related to the results of impurity control. If present indications hold, then, as the impurity level rises from $Z=1$, various plasma characteristics worsen. In particular, if ignition can be achieved at all, the plasma size required for ignition becomes much larger, the burn pulse is shortened, and the beam energy required to reach ignition increases significantly. Three general categories exist to describe the range of possible outcomes of the impurity control problem.

The first, and preferred, approach is that impurity control is not a problem, because the plasma is self-healing — i.e., particle flows and wall interactions are such that impurities do not accumulate, are handled without any upset of equilibrium, and have no effect on plasma behavior. This ideal situation seems unlikely at this time.

The second, and acceptable, approach is that impurity control is necessary and can be achieved by some combination of "passive" means such as wall coating, configuration and conditioning, gas blankets, or magnetic shaping. Each of these techniques involves a minimum perturbation to the "self-healing" situation, and therefore invokes probably only a moderate economic penalty. Each technique will be tested in the next few years, as will the need for impurity control itself.

The third, and least desirable, alternative is the need for active plasma diversion and reconditioning. Probably the most complex and costly of the three, active impurity control is not, at this time, known to be either necessary, feasible, or of certain cost.

In the face of the many uncertainties in impurity control; a course has been adopted that builds upon the passive techniques and takes into account the impact of the active techniques, while trying to exploit any self-healing tendencies. In particular, a combination of the passive techniques is included in the Reference Design. Analyses about diversion requirements

have also been carried out, and engineering studies of the impact of including active techniques are being started. With respect to the natural plasma processes, the possibility of reducing the ampere-turn requirements for flux line diversion is being investigated, in conjunction with the postulated high-density operations.⁶ In this mode of operation, there appear to be boundary conditions compatible with small negative currents that serve to direct flux lines outside the usual boundary.

5.3.4 Higher Density

The n^2 dependence of dP/dV represents only part of the incentive for higher density operation. The other consideration is that as the density is raised scaling becomes more favorable, regardless of the transport model used, i.e., whether an empirical or trapped-particle mode is chosen. Given the desirability of operating at higher densities, specifically at densities of $2-3 \times 10^{20} \text{ m}^{-3}$ rather than $0.5-1 \times 10^{20} \text{ m}^{-3}$, what is the cost of doing so? Three factors must be considered in answering this question:

- (1) What is to be the operational method used to raise the density?
- (2) What requirement is placed on technology to support this method?
- (3) What is required to maintain the high density?

To answer the first question requires some new insight, because this high-density regime has not been available heretofore. It does appear that through the mechanism of magnetic flux conservation,⁷ a path can be found that leads from low-density (low-beta) plasmas to high-density (high-beta) plasmas. This possibility must be pursued to determine whether a high-beta equilibrium does exist, whether it is possible to traverse the path to that equilibrium, and whether that equilibrium is stable.

6. J. F. Clarke, *EPR and TNS*, ORNL/TM-5418, Oak Ridge (June 1976).

7. J. F. Clarke, *High Beta Flux-Conserving Tokamaks*, ORNL/TM-5429, Oak Ridge (June 1976).

Reexamination of both the beta limitations in the theory and the high-beta operations in rapidly pulsed toroidal plasmas (screw pinch, etc.) lend credence to this possibility.⁷

A major technological requirement that is made more difficult by a density increase is the one on neutral beam energy. As density rises, so does the beam energy required for penetration. A careful look must therefore be given to optimizing the density, while keeping both plasma performance and beam penetration energy in mind. Possibilities for lowering the required effective beam energy, such as in ripple trapping and time-dependent density build-up, must be examined to see if simplifications can be made.

At the same time, the configuration of the high-beta plasma appears to be more asymmetric poloidally and more time varying than a low-beta plasma. Therefore, the demands on the PF electrical system to provide a time-varying, asymmetrically arranged ampere-turn distribution may also become considerably greater than in the low-beta case.

For the third question, more basic plasma physics understanding is necessary to provide the basis for maintenance of the high-density conditions over the burn pulse. A knowledge of fueling and density control, as well as of fundamental particle transport, is necessary before one can postulate a credible density maintenance program.

The next iteration of the advanced design work will be based upon a high-density plasma, with a concerted effort being made to understand and verify the basic hypothesis.

5.4 Recommendations for the Next Tokamak Step

As the EPR study progressed and the findings at each of the EPR study laboratories became more definite, while remaining similar, attention was focused on the assessment of the implications of the findings of the study.⁶ These final sections discuss the Oak Ridge judgments concerning this assessment, and the actions to be taken.

5.4.1 Clarification of the EPR Objectives

Examination of the set of EPR objectives indicates (see Table 5.1) that many of the objectives subordinate to the ultimate EPR goal of net electric power can be achieved in earlier, lesser facilities. Those objectives that can only be achieved in an EPR are related to the high duty cycle and availability of the plant and to the full energy conversion and tritium breeding in the blanket. With the renewed assignment of these most advanced goals to the EPR, satisfaction of the subordinate objectives can be achieved at a benefit/cost ratio and cost level seen to be acceptable.

5.4.2 Rethinking the Basic Physics Extrapolation

In the assessment of the EPR situation it is quite clear that the basic plasma engineering assumptions concerning minimum size and field for ignition play a dominant role in setting the overall size and difficulty of EPR. Recent advances, both experimental and conceptual, have led to a rethinking of these basic assumptions. These advances indicate (1) that it may be possible to achieve higher β values than had previously been supposed, and (2) that operation at high density should provide improved performance. In setting up the basis for the next tokamak step, use will be made of the benefits of high-density plasma operation, and close attention will be given to a better understanding of its achievement. More data on the scientific basis for the postulated⁶ high-density, high-beta operation will be forthcoming from the next generation of large experiments now under construction. Along with a rethinking of the high-density operation comes the rethinking of the possible performance of the many large existing and planned facilities. Just as the EPR benefits from a rethinking of the physics base, so does any other facility in which a high-density plasma can be generated. The possible achievement of ignition and/or other significant subordinate goals of the EPR can only lead to a stronger base for the next step.

5.4.3 Relationship to Supporting Technologies

One of the most significant programmatic questions that arose during the study was the availability of the critical technologies for EPR design on a very tight schedule. For this reason, the technology needs must be matched to the planned output of the principal program: namely, superconducting TF coils and positive ion beams. In these two cases care will be taken to ensure that subsequent designs are reasonable extrapolations from the designs which will be qualified in the DMFE Large Coil and TFTR Beam Programs at ORNL. Close examination of the connections between other supporting technology programs and the system design will be required as well.

5.4.4 Development of the Design for the Next System

On the basis of the general programmatic and technical points enumerated above, the design

of the next system should proceed through four stages of increasing intensity and total effort:

- (1) Establishment of a Reference Point.
Selected on the basis of the EPR and prior experience, this nominal design point will serve as the center point for the next phase.
- (2) Exploratory, sensitivity, and trade-off studies. Given the Reference Point as a nominal center point, the required studies will be undertaken to find the optimum design points and to give better support to the decisions made.
- (3) Establishment of the Reference Design. This will be done by a synthesis of all the work performed beforehand.
- (4) Performance of the Conceptual Design. In this largest, most general category there are various subparts, which have been described above as technical evaluation, pre-conceptual design (seeking alternative solutions), and the full conceptual design.

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