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GA-C19596

Rev. 3

DIII-D TOKAMAK LONG RANGE PLAN

Prepared under
Contract No. DE-AC03-89ER51114
for the San Francisco Operations Office
U.S. Department of Energy

AUGUST 1992

GENERAL ATOMICS
MASTER

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LIST OF ABBREVIATIONS AND SYMBOLS

ACDP	actively cooled divertor program
AD	advanced divertor
ADP	Advanced Divertor Program
APS	American Physical Society
ARIES IV	an advanced reactor design
AT	advanced tokamak
B	magnetic field
BPX	burning plasma experiment
CDA	conceptual design activity
DEMO	fusion demonstration plant
DIII-D	tokamak fusion device located at General Atomics
DOE	Department of Energy
EC	European community
ECCD	electron cyclotron current drive
ECH	electron cyclotron heating
EDA	Engineering Design Activity
ELM	edge localized mode that limits energy storage in plasma
FPAC	Fusion Policy Advisory Committee
FWCD	fast wave current drive
FWH	fast wave heating
FY	fiscal year
GA	General Atomics
H-mode	a high regime of tokamak confinement
HTGR	high temperature gas-cooled reactor
I	plasma current
IAEA	International Atomic Energy Agency

IBW	ion Bernstein wave method of ion cyclotron heating
ICH	ion cyclotron heating
ICRF	ion cyclotron radio frequency
ITER	International Thermonuclear Experimental Reactor
ITER CDA	ITER conceptual design activity
ITG	ion temperature gradient
JAERI	Japan Atomic Energy Research Institute
JET	Joint European Tokamak
L-H Transition	transition from L-mode to H-mode
L-mode	a low regime of tokamak confinement
LHCD	lower hybrid current drive
LLNL	Lawrence Livermore National Laboratory
MFE	magnetic fusion energy
MFPP	magnetic fusion program plan
MIT	Massachusetts Institute of Technology
MSE	motional stark effect
NBCD	neutral beam current drive
NBI	neutral beam injection
O-mode	a mode of microwave propagation in which the microwave electric field is parallel to the toroidal magnetic field
OFE	Office of Fusion Energy
ORNL	Oak Ridge National Laboratory
P/A	power-per-unit area
P/R	power-per-unit major radius
P	plasma heating power
PPPL	Princeton Plasma Physics Laboratory
q	tokamak safety factor $q = 3$ means that a magnetic field line traverses the toroidal direction three times in one poloidal rotation
R&D	research and development
RDP	radiative divertor program
rf	radio frequency
SOL	scrape off layer

TAE	toroidal Alfvén eigenmode
TFTR	the Princeton Tokamak Fusion Test Reactor
X-mode	a mode of microwave propagation in which the microwave electric field is perpendicular to the toroidal magnetic field
X-point	the location where the magnetic field is purely toroidal (no poloidal component)
Z_{eff}	effective plasma charge measures impurity fraction
SNL	Sandia National Laboratory
STE	steady-state tokamak experiment
T_e	electron temperature
T_i	ion temperature
TPA	technical planning activity
TPX	tokamak physics experiment
TTF	Transport Task Force
UCLA	University of California, Los Angeles
VH-mode	very high mode
β	the ratio of plasma pressure to magnetic field pressure
Ω_H	hydrogen ion cyclotron frequency
α	minor radius of plasma
β_N	normalized beta, $\beta/(I/aB)$
β_p	beta poloidal
τ_E	plasma energy confinement time

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1. FOREWORD

This document presents the long-range plan for the DIII-D tokamak facility. This plan is updated annually as part of a five-year contract between General Atomics (GA) and the U.S. Department of Energy (DOE). It forms the background for the annual experimental plan and the field task proposals. This plan updates the previous long-range plan issued as GA-C19596, Rev. 2, May 1991, and an interim DIII-D plan presented in Appendix B of the DOE Office of Fusion Energy Confinement Systems Field Task Proposal, April 1991. The long-range plan presented here has undergone extensive changes in response to revisions in the national magnetic fusion program, and in preparation for the next five-year contract period (November 1, 1993 to October 31, 1998). Foremost in these changes has been the formation of the International Thermonuclear Experimental Reactor (ITER) engineering design activity and the cancellation of the Burning Plasma Experiment (BPX), and the emergence of a Tokamak Physics Experiment (TPX).

Accordingly, the DIII-D long-range program plan has been revised to reflect increased emphasis on divertor research and development in support of ITER and emphasis of advanced tokamak (AT) physics with rf current drive in support of TPX and future reactors. This new research plan also extends beyond that envisioned in last year's plan. The divertor program now includes a radiative divertor installation aimed at dispersing peak divertor heat loads and reducing divertor plasma temperatures with a concept that can be applied to ITER. The steady-state noninductive current drive element of the DIII-D program has been extended to planned employment of the rf current drive systems to sustain and optimize AT configurations that are now being produced in DIII-D with transient methods. Success in this DIII-D program element will provide the database for advanced reactor concepts such as ARIES-IV and would help speed the commercialization of magnetic fusion energy.

To carry out this program, GA provides a combination of extensive institutional experience in the field, a skilled and experienced staff of international repute, and a uniquely flexible facility — the DIII-D tokamak. GA has had an active fusion research program for the past three decades. It is the only industrial participant in the U.S. program with a major, integrated effort in all aspects of plasma physics and fusion research, from basic plasma theory to fusion reactor technology.

In addition, collaborations with other U.S. and international fusion programs are an essential feature of the DIII-D program. These collaborations assist and bring expertise to DIII-D. Principal among these efforts are the on-going cooperative

efforts with the Japan Atomic Energy Research Institute (JAERI), the Lawrence Livermore National Laboratory (LLNL), Oak Ridge National Laboratory (ORNL), University of California at Los Angeles (UCLA), Sandia National Laboratory (SNL), and Princeton Plasma Physics Laboratory (PPPL). Other foreign laboratory groups involved in cooperative programs on DIII-D include the Joint European Tokamak (JET); Culham Laboratory; Max-Planck-Institute für Plasmaphysik, Garching; Institute for Plasmaphysics KFA, Jülich; Centre d'Etudes Nucleaires de Cadarache; Kurchatov Institute of Atomic Energy, Moscow; and Troitsk Institute for Development of Thermonuclear Research. University collaborators include: Massachusetts Institute of Technology (MIT); Johns Hopkins University; the University of Maryland, the University of California at San Diego, Los Angeles, Berkeley, and Irvine; University of Wisconsin; University of Illinois; University of Washington; and Rensselaer Polytechnic Institute. In addition, the DIII-D program continues to provide facilities for training graduate students from UCLA, Berkeley, Irvine, San Diego, MIT, University of Illinois, University of Wisconsin, RPI, University of Maryland, and Johns Hopkins University. In addition to hosting scientists to participate in DIII-D experiments, we send GA scientists to participate in experiments at other laboratories. We also collaborate with other institutions on ITER, TPX, and the development of a physics database for the national transport initiative (TTF).

The annual update of this long-range plan is required by GA's five-year contract for operation of the DIII-D facility and related research with the U.S. DOE. The goal of this research program is to demonstrate high-beta plasma confinement with noninductive current drive. The original anticipated cost profile in Table 1-1 is compared with actual funding. In addition, the DIII-D program is augmented with collaborations from national laboratories and universities. The LLNL and ORNL support to DIII-D totals \$4.7M in FY92.

TABLE 1-1
DIII-D RESEARCH OPERATIONS COST PROFILES

	FY89	FY90	FY91	FY92	FY93	Total
Anticipated to GA	31.5	39.5	48.0	52.3	38.1	209.6
Actual to GA	32.3	32.1	31.0	32.1	33.5*	161.0

*Anticipated.

The lower-than-anticipated funding levels for FY90 through FY92, plus the increased national emphasis on ITER and BPX/TPX and transport, have had substantial impact on the long-range DIII-D plan. The net effect will be to delay the accomplishment of the five-year goal by several years. Consequently, the revision

of the long range plan described here extends to 1999 and includes new divertor and AT research.

The outline of this document is as follows. Section 2 provides an overview of the long-range DIII-D research program. Sections 3 and 4 describe the advanced tokamak and divertor research programs. The status and plan of the research facility in FY92 is described in Section 5, which also discusses the additional systems and facilities that we propose to add in the future. Section 6 provides the relationship of the DIII-D program to the U.S. magnetic fusion program.

2. INTRODUCTION AND SUMMARY

The DIII-D Tokamak Long Range Plan for controlled thermonuclear magnetic fusion research will be carried out with broad national and international participation. The plan covers: (1) operation of the DIII-D tokamak to conduct research experiments to address needs of the U.S. Magnetic Fusion Program; (2) facility modifications to allow these new experiments to be conducted; and (3) collaborations with other laboratories to integrate DIII-D research into the national and international fusion programs. The period covered by this plan is 1 November 1993 through 31 October 1998.

2.1. THE DIII-D MISSION

DIII-D is on the path to a commercial fusion reactor.

The mission of the DIII-D tokamak research program is to provide data needed by the International Thermonuclear Experimental Reactor (ITER) and to develop a conceptual physics blueprint for a commercially attractive electrical demonstration plant (DEMO) that would open a path to fusion power commercialization. The National Energy Strategy calls for the development of magnetic fusion as an energy option with operation of a DEMO by 2025. The DEMO will be based on nuclear technology demonstrated in ITER and the physics and engineering database established in magnetic fusion facilities during the next two decades. A path to a commercially attractive DEMO is illustrated in Fig. 2.1-1.

Applying recent experimental results to development of advanced tokamak modes and improved divertors will accelerate commercialization of fusion.

On the present path, based on extrapolation of current conventional operating modes, ITER is twice as large as Joint European Tokamak (JET), and DEMO, using the same logic, will be even larger. However, successful development of advanced tokamak (AT) operating modes could open the way for significantly improved confinement and stability, leading to a smaller, more commercially attractive DEMO, provided new divertor concepts are developed to handle the accompanying high divertor power density. A smaller and lower cost DEMO opens up the possibility that multiple nations, utilities, and industries could build DEMOs simultaneously and, therefore, rapidly optimize the tokamak for commercialization.

Results from experiments at DIII-D and other tokamaks indicate that plasma and divertor performance can be increased beyond the baseline conceptual design of ITER. A simultaneous demonstration of such improved tokamak plasma and divertor operation for steady state would open a path to an attractive DEMO, provide new operating options for ITER, and establish an advanced physics foundation for the tokamak physics experiment (TPX) program. The planned DIII-D program strives to incorporate new theoretical and technology developments to extend the tokamak experimental physics database toward steady state. This

Fig. 2.1-1. A tokamak path to electricity production. The results of DIII-D will provide data needed for ITER and a physics blueprint for a commercially attractive electrical demonstration plant (DEMO) to follow ITER and precede a fusion power plant such as ARIES IV.

research program will continue to provide increased understanding in many areas of fusion science and technology.

2.2. DIII-D OBJECTIVES AND GOALS

Based on the broad DIII-D mission described above and nearer-term needs of the U.S. fusion program, we have identified relevant DIII-D programmatic objectives and quantitative goals. The objectives and goals are based on extensions of DIII-D results that can be tested with increases to the DIII-D tokamak capability. To assure that our near-term research results will benefit ITER and steady-state machines, we aim to conduct our experiments at sufficiently high temperatures with appropriate dimensionless plasma parameters in an integrated fashion so that all features are demonstrated simultaneously with sufficient theoretical understanding and appropriately developed technologies.

The DIII-D program objective is to carry out an integrated long-pulse demonstration of a well-confined high- β plasma with noninductive current drive in an advanced plasma and divertor configuration. Our reference scenario has an ITER-relevant quantitative goal: to sustain a 2-MA plasma with 5% β for 10 sec. This goal is a step toward a steady-state tokamak with improved divertor operation. It is the long-range programmatic goal of the present DIII-D contract. Present results and understanding indicate that this goal can be achieved with the new capabilities identified in this, and the previous, plan. These new capabilities will also allow new understanding in AT and divertor physics design to establish a path via ITER to significantly improved DEMO.

To establish quantitative DIII-D AT goals we are guided by the ARIES reactor study, which has quantified tokamak performance parameters that could lead to an attractive reactor concept. These figures of merit are confinement quality, stability factor, bootstrap current fraction, and divertor peak heat flux reduction factor. We will use these parameters as targets for DIII-D AT goals, to provide a basis for specifying facility capability requirements, and to chart technical progress. Target parameters for DIII-D research are specified in Table 2.2-1. Our goal is to simultaneously achieve and sustain such parameters in DIII-D for times longer than the tokamak current relaxation time (5 to 10 sec).

To achieve these target parameters and goals we have developed a research plan with two parallel but interrelated programs; the AT program and the divertor development program. The AT program will primarily address the achievement of the first three target parameters, combined with the study of long-pulse 2 MA noninductive current drive. The divertor development program will address the fourth target parameter. These two programs are closely interrelated since achievement of success in one area will affect achievement of success in the other.

TABLE 2.2-1
TARGET PARAMETER GOALS FOR DIII-D ADVANCED TOKAMAK OPERATION

	ITER CDA Design	Best DIII-D Already Achieved (Independently)	Goal DIII-D Target (Simultaneously)
Confinement quality ($H = \tau_E / \tau_{ITER-89P}$)	2	3.4	4
Stability factor [$\beta_N = \beta / (I/aB)$]	3	6.5	6
Bootstrap current (fraction)	0.3	0.5-0.8	0.5
Divertor plate heat load dispersal (reduction factor)	1	5	10

2.3. DIII-D RESEARCH PLAN

DIII-D is a flexible
and capable facility.

The DIII-D tokamak is well suited to carry out the challenging AT and divertor objectives outlined above. DIII-D is relatively modern; in 1986 it was outfitted with a new vacuum vessel, new outer poloidal field coils, and 5 sec pulse length neutral beams. The DIII-D poloidal magnetic field system allows an extremely wide range of plasma shapes to be generated and a real-time digital computer control system allows a broad range of experiments to be efficiently carried out. The neutral beam system operates reliably with power levels up to 20 MW. The maximum DIII-D magnetic field is 2.1 T. The magnet set provides good access for heating and diagnostic systems. The plasma diagnostic set is among the best in the world, particularly in the plasma edge and divertor region and plasma profile information. Thus, the existing DIII-D facility is well suited for the planned research and, as described later, complements research being carried out at other tokamak facilities. DIII-D has operated reliably and is ready to carry out the planned, higher-power longer-pulse divertor and AT experiments to address steady-state tokamak issues.

AT results and
improved divertor
have already
been obtained
transiently from
DIII-D.

Transient current profile control experiments have shown that plasma stability and confinement can be improved. Bootstrap current fractions as high as 50% to 80% have been achieved in DIII-D experiments. Fast wave electron heating and current drive experiments at the 1 MW level as well as 110 GHz ECH microwave heating experiments at the 0.3 MW level show efficient power coupling. These results all bode well for the planned AT programs. AT and improved divertor results have already been obtained transiently from DIII-D ensuring that prospects for the

planned research are very promising. Advanced divertor (AD) experiments, which have demonstrated high pressures in the divertor plenum chamber, bode well for particle control with divertor pumping. Simple gas puffing in the present divertor region has transiently reduced peak divertor heat loads by a factor of five from an ITER-relevant level of 5 MW/m^2 , making plausible that a ten-fold reduction could be sustained with a more optimally designed radiative divertor configuration.

The AT program builds on recent DIII-D progress in plasma physics and rf current drive.

The objective of the planned research in AT development is to establish improved tokamak operation through significant improvements in both the stability factor and confinement quality and bootstrap current fraction using localized radio frequency (rf) current profile control, rf and neutral beam heating for pressure profile control, as well as control of plasma rotation and of the plasma boundary conditions. The first phase of this program has two parallel parts: (1) building on recent experimental results achieved under transient and dynamic conditions which show AT modes (including VH-mode, high inductance H-mode, second stable core, and high bootstrap configurations); and (2) experimental validation of the physics of rf current drive and efficiency optimization, in order to develop and prove the tools needed for AT optimization. The second phase entails combination of these efforts for active control and optimization of the AT modes, leading to a third phase of demonstration of fully noninductive, high β , actively controlled operation. The planned rf current drive systems (8 MW of fast wave power plus 10 MW of ECH/LH microwave power) will provide the local deposition and control needed for this research program. Supplemented by the DIII-D neutral beam system, the rf systems will provide full noninductive operation for 10 sec at the target plasma parameter of 2 MA at 5% β . In addition, the current profile control systems will provide the flexibility to optimize AT configurations and carry out fundamental tokamak physics research.

A three-step divertor development program is planned: (1) AD, (2) radiative divertor, (3) actively cooled divertor.

The planned divertor program concentrates on divertor power and particle control under conditions relevant to ITER and future devices. The initial divertor research program will be carried out with the existing AD configuration that includes particle pumping, electrical bias capability steady state and pellet fueling. Experiments will concentrate on hydrogen pumping, helium removal, impurity control, edge plasma, and divertor physics. Divertor predictive code modeling and supporting theory will be benchmarked with detailed diagnostic measurements. This will provide a reliable model for design of future divertors. In 1995, a new configuration, called a radiative divertor, will be installed. It is designed to disperse the divertor power over a larger surface area so that the peak divertor power flux is reduced by up to an order of magnitude when compared to present conventional divertors. The first radiative divertor will be passively cooled and operate for up to 10 sec. A later actively cooled configuration will allow experiments up to 60-sec duration to investigate particle control issues. The rf power levels and plasma conditions for the 2 MA, 5% β , 10 sec integrated AT demonstration will also provide a relevant test of divertor performance.

In order to apply the DIII-D research to next-step fusion facilities (e.g., ITER, TPX, and DEMO), we will apply and develop predictive models to validate theories on stability, transport, current drive, and divertor operation. In this way, progress in achieving the DIII-D objectives will focus ITER and TPX operations on the most productive operating modes, provide a database to support an effective divertor for ITER, and focus the DEMO concept toward smaller, more cost-effective, and more attractive commercial designs.

The DIII-D Long Range Plan is schematically illustrated in Fig. 2.3-1. The AT and divertor program elements both consist of three phases. At all times, these two elements are coupled as illustrated to form a coherent integrated program.

The more detailed experimental plan is shown in Fig. 2.3-2. Each year's operating campaign is characterized by simple physics objectives that will provide key information to next-generation tokamaks (ITER, TPX, DEMO, Reactor). The cross-hatched periods indicate the periods of tokamak operations. The experimental research objective for the Advanced Tokamak and Divertor Development programs will be carried out with close coupling between theory, experiment, and technology development to advance the magnetic fusion program in critical areas such as transport, stability, plasma-wall interaction, toroidal Alfvén eigenmode suppression, bootstrap current optimization, divertor physics, rf heating and current drive, and disruption control.

2.4. PROGRAM IMPLEMENTATION

DIII-D is operated
with many
collaborators.

GA will manage and be responsible for the conduct of the work described in this plan. GA will operate the planned DIII-D research program as a national program with three major U.S. partners: LLNL, ORNL, and UCLA. GA and these partners will develop and refine long- and near-term research plans with DOE to most effectively utilize the joint resources of the four partners. In addition, several other laboratories and universities collaborate on specific areas that are essential to the success of the DIII-D program. Presently, half of the scientific staff working on DIII-D are from other laboratories. The DIII-D facility operations staff is almost entirely from GA. A summary of current DIII-D collaborators is given in Table 2.4-1. It is anticipated that other collaborators will be added in the future.

In order to encourage and accommodate increased national participation in the DIII-D program, a high-speed computer-linked remote experimental station is being developed by LLNL in collaboration with GA. Such remote experimental stations will allow experts at other laboratories to participate in real-time diagnostic and physics operation, while reducing travel costs, reducing burdens on the limited computer capability at DIII-D and increasing real-time data analysis.

International collaborators participate in the DIII-D program and DIII-D staff participate in experiments at international laboratories. These international

Fig. 2.3-1. The DIII-D Long Range Plan is a coherent integrated steady-state program.

collaboration programs allow the DIII-D program to test ideas at other facilities and helps ensure that important new ideas successfully tested elsewhere are quickly incorporated into the DIII-D program. In return, the international programs receive timely benefits of DIII-D research. Such programs are now active between European Community (EC), Japan, and Russia with the tokamaks listed in Table 2.4-1. Strong collaborations on divertor and current drive research exist between DIII-D and the JT-60U, JET, Tore Supra, ASDEX, Textor, and T-10 tokamaks.

An essential input to DIII-D program planning are the needs of future national and international projects in which DIII-D staff can provide valuable inputs to specific DOE-authorized collaborative activities such as ITER and TPX.

GA operates the DIII-D facility with highest environment, safety, and health standards. As a California company, GA falls under the laws, regulations, and purview of federal, state, county, and city authorities. As a DOE contractor, GA also complies with DOE orders regarding environment, safety, and health practices.

Fig. 2.3-2. DIII-D Long-Range Program Plan, program objectives and facilities capabilities.

TABLE 2.4-1
COLLABORATORS PRESENTLY AT DIII-D

International Tokamaks	National Laboratories	Universities
JT60 U (Japan)	LLNL	UCLA
JET (EC)	ORNL	UCSD
ASDEX (Germany)	SNLA	UCI
TEXTOR (Germany)	SNLL	UCB
Tore Supra (France)	ANL	MIT
T-10 (Russia)	PPPL	RPI
TSP (Russia)	INEL	Cal Tech
COMPASS (England)		Johns Hopkins Un.
		Un. of Maryland
		Un. of Illinois
		Un. of Paris
		Un. of Washington
		Un. of Wisconsin

3. ADVANCED TOKAMAK DEVELOPMENT AND RESEARCH

The objective of the planned research is to establish advanced tokamak (AT) operation through significant improvements in stability and confinement limits, and bootstrap current fraction using localized radio frequency (rf) current profile control, rf and neutral beam heating for pressure profile control, and control of plasma rotation and of the plasma boundary conditions. The development and research goals are: (1) to develop physics understanding of the formation and sustainment of AT configurations; (2) to establish experimental validation of the physics of rf current drive and efficiency optimization; and (3) to combine these two to provide a demonstration of optimized, long-pulse AT operation with simultaneous improved confinement, enhanced stability, and fully noninductive current drive at high β .

3.1. OVERVIEW

3.1.1. INTRODUCTION

Design of fusion reactors on the basis of conservative extrapolation of presently accepted plasma performance leads to large devices, with high magnetic fields and high plasma currents, requiring large recirculating power for noninductive current drive. Results from recent experiments and extrapolations of plasma theory have identified regimes of enhanced plasma performance called advanced tokamak (AT) operation. The goal of the planned DIII-D research program is to demonstrate AT operation under conditions directly applicable to the design of the fusion demonstration plant (DEMO) facility and to the operation of next generation experiments such as ITER and TPX.

Demonstration of AT regimes requires control of j , p , and v profiles, for times approaching steady state.

AT regimes have been demonstrated transiently in several devices, but application of these regimes to reactors necessitates they be sustained for several current relaxation times in order to demonstrate steady-state feasibility. In order to implement steady-state AT scenarios, special "tools" are required, particularly rf systems for controlling the current density and pressure profiles, coils and neutral beams for controlling the plasma rotation, and particle fueling. The thermal and particle control requirements for operation in the AT scenarios rely on effective divertor operation, so the AT program will be closely coupled to the divertor program.

This section provides an overview of the development of the DIII-D AT objective from the ultimate fusion objective of an attractive reactor. From this we develop specific quantitative goals and a program plan of research and implementation of new tools to allow us to reach those goals. Subsequent sections discuss in further

detail the present experimental evidence for AT operation, the several scenarios we have examined which show promise for sustained and optimized AT demonstration, and the improvements to the DIII-D rf systems and facility needed to carry out the planned research program.

3.1.2. RESEARCH GOALS AND QUANTITATIVE TARGETS

Simultaneous improvement in confinement and stability is needed for fusion reactors to take advantage of AT operation. Successful AT operation could lead to a smaller and more attractive fusion reactor.

The DIII-D AT program has three essential research goals: (1) to develop physics understanding of the formation and sustainment of AT configurations; (2) to establish experimental validation of the physics of active rf current drive and efficiency optimization, and (3) to combine these two to provide a demonstration of optimized, long-pulse AT operation with simultaneous improved confinement, enhanced stability, and fully noninductive current drive at high β .

The first goal is development of an understanding of the physics underlying the formation and sustainment of AT configurations. This includes exploration and expansion of the second stability regime, evaluation and enhancement of the bootstrap current, and thorough understanding of the dependence of stability and confinement on plasma shapes and profiles. In addition, the AT program will be supported by a continuing effort in related tokamak physics studies, such as development of an improved understanding of basic plasma transport mechanisms, disruption avoidance and mitigation, and fast ion toroidal Alfvén eigenmode (TAE) instability control. This effort will be greatly aided by our continuing development and implementation of advanced digital control systems in DIII-D.

The second goal is the experimental validation of the physics of noninductive current drive with fast waves and electron cyclotron waves and optimization of the current drive efficiency. The use of noninductive current drive for the AT work ensures compatibility with future steady-state operating scenarios. Fast wave current drive (FWCD) was chosen as the primary current drive technique for DIII-D, although it is supplemented by electron cyclotron current drive (ECCD) and bootstrap currents. The physics of FWCD must be demonstrated in the initial phase of these experiments; current drive efficiencies approaching the theoretically expected values will scale to attractive reactor scenarios when combined with optimized bootstrap current.

The decisive goal is to combine the noninductive current drive and AT operation to demonstrate optimized, long pulse AT operation at high β . We will first reproduce AT regimes with fully noninductive current drive, then optimize the results to achieve the highest possible simultaneous values for confinement and stability with the minimum amount of auxiliary power. Detailed transport studies and modeling will be carried out.

Development of optimized designs of tokamak reactors has illustrated the need for simultaneous improvement in confinement and stability. Experiments have shown that confinement and stability are linked — the spatial profiles that provide good

stability also have good confinement properties. If confinement alone is improved, then stability limits performance, and the resulting reactor must still be large in order to get sufficient wall loading. Conversely, if stability alone improves, large size is needed to provide adequate confinement. Improving both confinement and stability can lead to an attractive, small reactor with high bootstrap fraction and low recirculating power.

Targets for our AT objective are to double present accepted scaling rules for confinement and stability.

Improvement in both parameters would allow reactor operation at lower magnetic field and/or plasma current. Lower field relieves the need for high stress magnets using exotic materials. Lower plasma current operation has the particular benefit of allowing higher values of the safety factor (q), greatly decreasing the likelihood of disruption, even at high normalized β , and reducing the current drive power requirement (by increasing the bootstrap fraction).

Improvement in stability is measured in terms of the Troyon scaling, i.e., the ratio of the achieved value of β to the normalized plasma current, I/aB [$\beta_N \equiv 100 (\beta)/(I/aB)$]. Generally accepted tokamak design guidelines set a limit of $\beta_{Ncrit} \approx 3$, on the basis of stability calculations with ohmically constrained current profiles and the overall tokamak database. Confinement improvement can be measured by the ratio between achieved energy confinement time and the ITER89-P scaling law. For L-mode plasmas, $H \approx 1$, and for "standard" H-mode, $H \approx 2$.

Our target for establishing AT operation by simultaneous improved confinement and enhanced stability is to double the present generally accepted tokamak performance criteria to achieve $H = 4$ and $\beta_N = 6$, sustained for several current relaxation times (>10 sec). We expect to obtain a bootstrap fraction of over 50% in these regimes. This dramatic improvement is illustrated in Fig. 3.1-1 which shows the expansion of the operating space and reduction in power to achieve high β fully noninductive operation

In addition to the specific targets for confinement and stability improvement, we plan an integrated demonstration of sustained noninductive plasma operation at parameter levels directly applicable to the design of next generation experiments. This entails operation with 2 MA fully noninductive plasma current, in a 5% β plasma, for 10 sec. This was a stated goal of the present DIII-D contract, but has been deferred due to funding limitations. The value of 2 MA corresponds to $q \approx 3$ in a full size DIII-D plasma at full toroidal field, which should yield the best plasma parameters under conditions of present good H-mode operation ($H = 2$ and $\beta_N = 3.2$).

Modeling indicates that 18 MW of rf power (nominal) will be required to drive this current. The β value of 5% corresponds to reaching the stability limit under the same conditions of power, current, and enhancement factors. This lies well within the range of reactor interest. Successful AT operation, reaching $H = 4$ and $\beta_N = 6$ (with the same profiles), would allow DIII-D to reach 9.5% β with essentially the same power, but using less than a third of the power for current drive.

Fig. 3.1-1. AT requires improvement in both stability and confinement.

3.1.3. RESEARCH PROGRAM PLAN

The AT research plan is based on experimentally achieved operating modes, including the high internal inductance, VH-mode, high bootstrap (high $\epsilon\beta_p$), and second stable core discharges.

The research plan for achieving these goals is presented in Fig. 3.1-2. The first phase of the plan involves development of basic understanding of stability and confinement through experiments using transient techniques to modify and control profiles and operating regimes. In parallel we will validate the current drive and heating techniques needed for AT operation at moderate rf power levels. In the second phase, we will develop the techniques required to sustain AT modes and work on optimization of plasma current, pressure, and rotation profiles using rf and neutral beam systems. Finally, we will proceed to demonstration of an integrated, fully noninductive, high current and high β plasma, continuing the effort to optimize performance with active control of long-pulse discharges.

The recent demonstration of several possible new tokamak operating modes is key to providing a strong foundation for our AT program, including the high internal inductance H-mode, VH-mode, high bootstrap (high $\epsilon\beta_p$), and second stable core discharges. Current density profiles for the four AT experimental regimes discovered in DIII-D are graphically summarized in Fig. 3.1-3 and discussed in detail below.

Fig. 3.1-2. DIII-D AT program plan with five experimental layering.

Fig. 3.1-3. Steady-state operation of AT regimes requires current profile control.

Similar improved confinement and stability results, have been obtained in other tokamaks.

At present in DIII-D these improved regimes are reached transiently and dynamically through manipulation of global engineering parameters such as power, current, and plasma shape. In the planned future AT studies, these conditions will be achieved and sustained through local control of the plasma current, flow velocity, pressure, and density profiles, as well as through control of the plasma boundary conditions.

High Internal Inductance H-Mode. Transient experiments using plasma current and elongation ramps have produced high-internal-inductance (or peaked-current-density-profile) plasmas. Under different conditions, discharges in DIII-D show confinement enhancement, reaching $H = 3.6$, and stability improvement to $\beta_N = 6.5$. The energy confinement time is proportional to ℓ_i under most conditions. Furthermore, there are experimental and theoretical indications that the stability limit (β_{Ncrit}) is also proportional to ℓ_i . Thus, increasing ℓ_i appears to have the same effect on plasma performance as a similar increase in plasma current.

High Bootstrap Current Fraction. Plasmas with high bootstrap fraction were produced with low current (high q) and strong neutral beam heating in order to maximize β_p . Under stationary, ELMing H-mode conditions, half of the current was provided by neutral beam current drive (NBCD) and half by the bootstrap effect,

with confinement reaching $H = 2$. Under ELM-free conditions, up to $H = 3$ and $\beta_N = 4$ were reached. Ballooning stability calculations indicate that from half to all of the plasma volume is in the second stability regime [for the experimental range $1.3 < q(0) < 2.0$].

VH-Mode. This mode was first seen following boronization in DIII-D. The confinement improved to $H = 3.5$. However, when β_N reaches ~ 3 , the VH-mode terminates when a global MHD event. The ELM-free VH-mode has been sustained for up to 1 sec with feedback control of the input power was used, to keep the plasma below just the β limit. The VH-mode plasmas show an enlarged region of high electric field shear, at the edge relative to H-mode and also a very large edge bootstrap current that puts the plasma edge region into the second stability regime. Either or both of these may account for the improved confinement.

Second Stable Core. Plasmas with second stable core were produced by simultaneous elongation and current ramps and full power neutral beam heating, programmed to reach the $n = 0$ and $n = 1$ stability limits at the same time (at $\beta_N \approx 3.5$). The highest values of β obtained in any tokamak were reached in this regime [$\langle \beta \rangle = 11\%$; $\beta(0) = 44\%$]. The core of these plasmas is in the second stability regime, as the result of negative shear near the axis. Thus far, the confinement is characteristic of L-mode ($H \approx 1$), but this can be expected to improve in stationary discharges.

3.1.4. REFERENCE SCENARIOS

Based on the experimentally achieved plasma configurations described in the previous section, we have developed five reference plasma scenarios for the DIII-D AT research program. These are:

- High Internal Inductance H-mode.
- High Bootstrap Fraction.
- VH-Mode.
- Second Stable Core.
- High Performance Steady State Demonstration.

The scenarios describe the conditions required for improved plasma operation, and specify the facility subsystems needed to achieve them. The scenarios have been modeled using zero-dimensional scaling laws to determine the required power. In the case of the second stable core, one-dimensional transport simulations have been carried out to extend the zero-dimensional scaling studies. The parameters for the AT scenarios for DIII-D are summarized in Table 3.1-1. The table gives parameters for the scenarios that put the most extreme demands on the rf systems — high internal inductance and second stable core, as well as the 5% β , 2 MA demonstration.

TABLE 3.1-1
ADVANCED TOKAMAK SCENARIOS FOR DIII-D

Scenario	High ℓ_i [Low $q(0)$]	Second Stable Core	VH-Mode	High β_p	Standard
β_N	6	6	6	6	3.38
H	4	4	4	4	2
B (T)	2.08	2.08	2.08	2.08	2.08
I (MA)	2	1.79	1.79	1.17	2
κ	1.8	2.1	1.8	1.8	1.8
$q(0)$	0.56	1.5	1	2	1
q_ψ	3.92	3.92	5	7.62	3.92
I_{ECCD} (MA)	0	0.18	0	0.10	0.48
P_{EC} (MW)	5.9	6.0	5.2	0.7	7.7
I_{FWCD} (MA)	1.06	-0.03	0.05	-0.52	0.62
P_{FW} (MW)	7.0	0.1	7.0	5.2	7.0
I_{NBCD} (MA)	0.10	0.23	0.11	0.21	0.15
P_{NB} (MW)	2.0	6.0	2.0	4.0	4.0
I_{LHCD} (MA)	-0.46	-0.21	0	0	0
P_{LH} (MW)	2.5	1.4	0	0	0
P_{total} (MW)	17.4	13.5	14.2	9.9	18.7
n_e ($\times 10^{20} \text{ m}^{-3}$)	0.64	0.78	0.57	0.50	0.64
T_{e0} (keV)	21.1	20.2	17.0	16.8	10.7
$p(0)/\langle p \rangle$	3.2	4.2	2.5	4.1	3.2
$I_{bootstrap}$ (MA)	1.30	1.61	1.62	1.39	0.75
t_{relax} (sec)	12.4	4.7	7.6	4.3	3.6

The goal for this scenario is operation at $\beta_N=6$ and four times L-mode confinement simultaneously.

High Internal Inductance. The goal for this scenario is simultaneous operation at $\beta_N = 6$ and $H = 4$. We have developed three approaches to this case at 1 MA plasma current. At this current, we expect that the density control requirement will not be severe. Note that, in the 0-D modeling of this case, the bootstrap fraction is estimated to be close to 1. However, significant rf current drive will be needed in order to establish the more peaked current profile needed. The first case (shown as scenario 1A in Table 3.1-1) entails exceeding the desired plasma current with central current drive, and creating a region of very low or negative current at the edge with the ohmic transformer and sustained with LHCD. This scenario requires a source

power of about 10 MW of EC and 8 MW of FW; 3 MW of LHCD appears adequate to modify the edge current density. The second case (scenario 1B) creates low central shear by driving some current off-axis (hence the lower ECCD efficiency). This calls for about 10 MW of EC, 2 MW of FW (for heating only), and 5 MW of NB. The third case examined is to try for very high ℓ_i by using central current drive combined with ion cyclotron minority heating to suppress sawteeth. This case has parameters very similar to 1B, but uses 7 MW of FW with no NB power.

The goal of this scenario is to demonstrate high bootstrap fraction with enhanced confinement in the second stable range.

High Bootstrap Fraction. Operation at high $\epsilon\beta_p$ can allow access to the second stability regime across the entire plasma. This can be achieved by keeping $q(0) \geq 2$, with a very broad current profile. This is also consistent with very high bootstrap fraction ($>70\%$). Since this profile is very sensitive to the radial distribution of the current drive, it is difficult to model using 0-D simulations. This case will also be run at ≤ 1 MA, so the heating and current drive systems specified for the other scenarios are more than adequate to implement this type of discharge. The development of this scenario will involve the use of counter-FWCD and/or ECCD to maintain high $q(0)$. The use of electron cyclotron heating (ECH) to control the pressure profile will be an important tool in determining the magnitude and location of the bootstrap current.

The VH-mode scenario uses LH-enhanced edge current drive and radial electric field shear to create a second stable edge region.

VH-Mode. The VH-mode scenario involves creation of a region of second stability at the plasma edge by producing a high edge current and controlling its profile. RF current drive at the plasma edge can be inefficient because of the relatively low edge temperature ($I/P < 0.05$ A/W near the edge for typical DIII-D scenarios). However, modeling lower hybrid waves near the edge can produce the required control of the edge current density. In addition, control of plasma rotation and shear in the radial electric field provides another path to development of the VH-mode scenario. We have identified several possible means of affecting the shear, including rotating helical magnetic fields generated by external coils, a combination of co- and counter-injected neutral beams, or the use of ion cyclotron radio frequency (ICRF) to enhance trapped ion orbit losses.

RF profile control and pressure profile control yield a fully noninductive driven second stable core.

Second Stable Core. The goal of this scenario is to achieve second stability in the central part of the plasma by creating a hollow current profile with $q(0)$ just below 2, and with a region of negative shear near the axis. This can be realized in two ways: by driving current off-axis, or by hollowing the profile with on-axis counter-current drive (scenarios 2A and 2B). Both schemes require significant power, but 1-D simulation of this case indicates the current drive power needed is reduced, because the bootstrap current provides $\sim 50\%$ of the total current drive with a profile consistent with the desired total current profile. This scenario will require active plasma density control by divertor pumping.

The high performance steady state drive demonstration scenario will be a benchmark for evaluating the success of AT operation.

High-Performance Steady State Demonstration. This scenario has the goal of the demonstration of 2 MA of fully noninductive current at 2 T. It will provide a benchmark for evaluating the success of the AT scenarios. Assuming only "standard" H-mode operation ($H = 2$, $\beta_N = 3.2$), this case requires 18 MW of rf power for current drive and an additional 2 to 3 MW of neutral beams. This plasma will reach $\beta = 5\%$. Note that the heating power required to reach the β limit ($\beta_N = \beta_{Ncrit}$) is roughly proportional to $(\beta_{Ncrit}/H)^2$. Thus, successful achievement of the AT goals ($\beta_{Ncrit} = 6$, $H = 4$) will mean that, with 20 MW, the plasma will also reach the β limit, but at $\beta = 9.4\%$. In addition, because of the higher plasma temperature, only about a third of the power will be needed for current drive, the remainder of the power is for heating.

3.1.5. HARDWARE REQUIREMENTS

Extension of today's experimental results to steady-state relevant regimes will lay the foundation for the design of the next generation of advanced tokamaks.

The tools for creating and controlling the current and pressure profiles required are fast wave heating (FWH) and fast wave current drive (FWCD), ECH and electron cyclotron current drive (ECCD), and lower hybrid current drive (LHCD). Fast wave antenna development and fabrication is being executed in collaboration with ORNL. The fast waves are expected to have good current drive efficiency, so they will provide the bulk current drive power and control of the central profiles by co- and counter-operation. Central electron cyclotron current drive efficiency is comparable to FWCD, although trapped particle effects are predicted to reduce the efficiency off-axis. However, the ability to localize the ECH power for heating and current drive is a major asset for AT studies, including bootstrap optimization and disruption avoidance studies. Finally, the LHCD will be used to control the edge currents since the efficiency can be reasonably good even in the colder outer regions, particularly if local seed currents can be created. The control of edge currents is particularly important for VH-mode discharges and for all high bootstrap cases.

The ability to drive the total plasma current noninductively has been demonstrated in several experiments (DIII-D, TFTR, JT-60, JET, Tore Supra). In DIII-D, experiments have shown neutral beam driven current at the level of 350 kA. The FWCD and ECCD currents have reached the 100 kA range. The extension of today's experimental results to steady-state relevant regimes can lay the foundation for the design of the next generation of ATs. This motivates our plan to extend the pulse duration capability of DIII-D to provide a pulse length reliable operation at full field and power to the 10 to 20 sec range. This will provide of at least twice the time constant for relaxation of the plasma current in fully noninductive discharges (and 10 time constants for internal profile relaxation), even under full power, high performance conditions.

The AT hardware plan is shown schematically in Fig. 2.3-2. The DIII-D facility presently has a 2 MW ICRF. This will be increased to 6 MW in FY94 to 8 MW in FY96. The present 2 MW of ECH power will be increased to 4 MW in FY95, and to

10 MW in FY96. This schedule for the ECH system is predicated on the successful development of a 1 MW long-pulse gyrotron by the end of FY94. In addition, we will modify and upgrade the DIII-D facility, particularly power supplies and cooling systems in order to accommodate the increased rf power, and to provide the capability of reliably extending the DIII-D pulse to the 10 to 20 sec range in FY96, and up to 60 sec in FY98 for long pulse divertor and particle control studies. These enhancements of the DIII-D are coordinated with the research programs in both divertor and AT development.

In addition to the rf systems, which provide pressure and current profile control, we plan to implement plasma rotation control as well as improved diagnostics, better sensors with higher spatial and temporal resolution, and digital control systems for application to shape and profile control. Recent detailed H-mode and VH-mode studies have shown the importance of plasma rotation. The most likely technique for the control of plasma rotation at the present time involves the use of toroidally and poloidally nonsymmetric, time-dependent coil arrays or counter-injected neutral beams, although further research may yield more attractive options.

3.1.6. EXPERIMENTAL STUDIES AND CAMPAIGNS

Three experimental campaigns will be conducted; culminating with demonstrations of noninductive current drive at high β and optimized AT discharges.

The research goals for the DIII-D advanced tokamak development program are (1) to develop an understanding of the physics of advanced performance tokamak regimes, (2) to establish and validate the physics of efficient noninductive current drive and current profile control, and (3) to use this information for active control and optimization of the DIII-D plasma to develop and demonstrate stationary, advanced tokamak operating regimes.

On the basis of these goals, we have developed a research campaign plan which incorporates the availability and installation plans for new DIII-D hardware in both the divertor and advanced tokamak programs. The research campaigns cover three major phases: near-term (1993–1994), medium-term (1995–1996), and long-term (1997–1998).

I. Physics Understanding and Validation Phase (1993–1994)

The first campaign features detailed studies of the experimentally achieved transient AT discharges.

The first phase of the planned research program will be devoted to developing an understanding of the physics of the improved operating regimes and to demonstration and validation of the physics of rf current drive and the bootstrap current.

The research program in this campaign period will develop a better understanding of AT physics by studying in more detail the scenario discharges achieved by transient methods during the previous operations on DIII-D. For example, the recently demonstrated feedback control of the total stored energy in VH-mode discharges allows extending the duration of VH-modes. Parametric studies of all AT scenarios

will be enhanced by increased experimental time, installation of new diagnostics such as the Motional Stark Effect (MSE) System, and implementation of plasma density control tools, such as the Advanced Divertor Program (ADP) cryopump and the C-coil. We expect that much improved understanding can be achieved during this campaign while the hardware required for the active control becomes operational.

Several issues remain to be addressed experimentally. The shape dependence of the AT discharges has not been explored in any systematic manner, although there exists fragmented information pointing to differences between single and double null divertor and variation with triangularity. This information will significantly impact the radiative divertor design. The existence and expansion of the second stable core discharges will be explored; at higher current the possible reduction of transport in the second stable region is an important issue and will be studied. The role of the bootstrap current in these configurations and optimization of the bootstrap current are topics for further exploration for achieving both higher β_N and H. Simultaneous combination of different AT features will be attempted to further improve plasma performance.

The first campaign will also validate the theoretically predicted physics and CD efficiencies of FWCD, ECCD, and bootstrap currents.

During this campaign, the physics of FWCD will be studied to validate the theoretically predicted current drive efficiency and its parametric scaling. A clear validation of the FWCD will require sufficient power to drive all of the plasma current to eliminate effects due to the residual inductive electric field. For DIII-D this requires driving a minimum of 0.3-0.5 MA of non inductive current. The ability of FWCD to modify the current density profile in response to changes in the target plasma, with ECH for example, and with the strap-to-strap phasing on the antenna are topics of interest during this campaign.

Studies of ECCD are also planned to address the issue of the trapped particle degradation of the ECCD efficiency. A definitive conclusion may require a fully ECCD driven plasma to accurately assess the effects of trapped electrons. A related issue is the current drive efficiency as a function of radius; ECCD is expected to depend on radius and since the ECCD is the most easily localized of all of the current drive methods, the effectiveness at various locations in the profile will be determined experimentally. The results will be incorporated into our AT scenario modeling.

Pending a favorable assessment of LHCD, an existing 3 MW 2.45 GHz system will be mounted on DIII-D for edge current profile modification of AT configurations.

The physics of bootstrap current will be studied during this phase including an assessment of theoretically predicted rf enhanced bootstrap currents. Also, the ability to control the spatial location of bootstrap current by pressure profile modification with ECH will be studied.

II. Advanced Tokamak Active Control and Optimizat~~on~~ Phase (1995-1996)

The second campaign will feature active modification and control of current density and pressure profiles using current drive.

During this period the current drive tools will be available to provide some active control of the current density and pressure profiles, so the pulse length of some of the AT scenarios can be extended. In fact, the current drive power should be sufficient to drive substantial noninductive current (0.5 to 1 MA) in some conditions. The key feature during this campaign is expected to be the ability to actively modify and control the current density and pressure profiles, with a full complement of diagnostics to ascertain the causes and consequences on stability and confinement.

All four of the AT scenarios will be studied to explore the performance improvements which can be achieved with active control in addition to extending the discharge duration. These studies include, for example, attempts to cancel excessive bootstrap currents at the edge, improving the central peaking of pressure and current, and use of noninductive current drive to expand the second stable core region. With active current profile control, the current relaxation time will be prolonged and the enhanced performance regimes will be sustained for the full field (>10 sec) pulse capability of the DIII-D facility.

III. Demonstration of Fully Noninductive Current Drive at High β (1997-1998)

The third campaign will use all of the tools available to achieve the AT goal of $\beta_N = 6$, $H = 4$ in long-pulse conditions relevant to steady state to assess β limits and confinement properties.

The third campaign will use all of the tools available to achieve the AT goal of $\beta_N = 6$, $H = 4$ in long-pulse conditions relevant to steady state. This campaign assumes the current drive tools have been demonstrated and their application to advanced scenarios is understood. Optimization and demonstration will combine features of our four advanced scenarios to achieve the best result.

During this campaign, fully noninductive plasmas will be evaluated to assess their β limit. The goal of this campaign includes producing a fully noninductive current driven standard H-mode at 5% β at 2 T, for 10 sec.

The three campaigns described here progress in a logical fashion from short pulse length transient studies aimed at physics understanding, through noninductive current drive and profile control to separate the transient responses from steady-state properties, to finally using a combination of rf tools, neutral beams pumped and biased divertor, and particle fueling, and rotation control to sustain the AT scenarios for a significantly long duration. Ideally, we want to attain a quasi-stationary state lasting several current relaxation times (about 10 to 20 sec). This will ensure that the AT conditions are sustainable on the time scale of the confinement of plasma energy and particles and that there are no violent instabilities associated with the external modification of the current profile which remain hidden during the transient phase. This will provide critical information for the design and operation of new very-long-pulse or steady-state tokamaks.

4. DIVERTOR DEVELOPMENT AND RESEARCH PROGRAM

The objective of the planned Divertor Development and Research Program is to develop and implement a divertor design that solves the "divertor problem" by providing significant dispersal of plasma energy flow prior to it reaching the divertor surface to ease divertor heat removal and reduce surface erosion from excessively hot impinging plasma.

4.1. OVERVIEW

The DIII-D Divertor Research will proceed through three phases. The first phase, the Advanced Divertor Program (ADP), will complete the present on-going DIII-D plasma boundary and divertor research program. This phase will use the electrically biasable ADP divertor ring now installed in DIII-D and an in-vessel cryopump (to be installed in early 1993) to conduct studies of the effects of electrical biasing, baffling, pumping, and gas and impurity injection on divertor performance and tokamak particle and impurity control. The focus of these studies will be to obtain data to support the ITER divertor design and the design of the follow-on DIII-D radiative divertor and actively cooled divertor installations.

The second phase of the Divertor Development and Research will be the Radiative Divertor Program (RDP), in which the physics basis of radiative power dispersal and divertor plasma cooling will be demonstrated in an inertially cooled double-null divertor configuration that is simultaneously optimized for radiative power dispersal and advanced tokamak (AT) performance. The RDP will initially focus on divertor physics studies and obtaining data for the International Thermonuclear Experimental Reactor (ITER) design. However, the RDP, through density and recycling control, will contribute to and enhance the Advanced Tokamak Development Program described in Section 3.

The third phase of the Divertor Development and Research will be the Actively Cooled Divertor Program (ACDP), in which the radiative divertor solution will be demonstrated with actively cooled hardware suitable for steady-state tokamak operation. We anticipate that the hardware design will draw heavily upon the ITER Plasma-Facing Component Research and Development (R&D) effort in terms of material, cooling technology and fabrication methods development. This installation will enable DIII-D to extend its pulse length to 60 sec with up to 38 MW input power and conduct research on times scales much longer than the current relaxation time scale and at the particle-wall equilibration time scale.

4.1.1. CONTRIBUTIONS OF THE PROGRAM TO FUSION DEVELOPMENT

A solution to the power handling/plasma cooling "divertor problem" is urgently needed regardless of whether ITER follow-on devices are based upon conventional or advanced tokamak physics.

The Divertor Development and Research Program in DIII-D will contribute to the physics and technology basis for effective divertor implementations in ITER and in the DEMO-class tokamaks that will follow ITER.

The combination of an effective divertor and AT operation is the key to an attractive fusion power reactor. A possible sequence of device steps toward a commercial fusion reactor based upon success with the AT approach is identified in Fig. 2.1-1. The AT (as opposed to conventional) approach to DEMO may reduce the immediate severity of the divertor power handling and erosion problems encountered in a post-ITER design, the same key divertor problems identified during the ITER conceptual design activity (CDA). However, severe levels are encountered in a commercial AT reactor design such as ARIES IV. Thus a solution to the power handling/plasma cooling "divertor problem" is urgently needed regardless of whether ITER follow-on devices are based upon conventional or AT physics.

The design and operations problems that stand out above all others stem from the highly concentrated thermal power flow that emerges from the fusion core plasma.

While the divertor in a tokamak must perform many functions, the design and operations problems that stand out above all others stem from the highly concentrated thermal power flow that emerges from the fusion core plasma. The problem manifests itself at the divertor target in at least three major ways: (1) the engineering difficulties of high thermal power handling at the target, (2) the divertor target surface erosion problem stemming from the high plasma temperatures that develop across the divertor target plasma sheath, and (3) the poor disruption tolerance and potential for catastrophic failure that the thin target sections mandated by the high power loadings entail.

At the most basic level, the "divertor problem" is one of too high a concentration of power loading at the target surface. If a mechanism to disperse this power can be found, thermal power handling problems become more tractable, lower-melting-point target materials and more robust cooling channel geometries become possible, divertor plasma temperatures and surface erosion rates drop dramatically, and target designs can be made much more resistant to the unavoidable effects of ablative erosion during a plasma disruption. The basic goal of our research effort is to develop and implement a divertor design which solves the "divertor problem" by providing significant dispersal of plasma energy flow prior to it reaching the divertor surface.

The divertor power handling problem becomes more severe as we progress toward DEMO. In the approximation that the width of the plasma scrape-off layer (SOL) is independent of device major radius, the plasma-imposed demand on the divertor for handling power exhaust is set by the thermal power-per-unit major radius (P/R). Table 4.1-1 compares the P/R values of present tokamaks and candidate ITER, DEMO, and commercial reactor designs. ITER imposes a three-fold increase in plasma power handling requirement relative to present experience; an

advanced-performance commercial reactor (ARIES IV) imposes yet another two-fold increase. Even in ITER, and especially in DEMO, a new divertor approach is required to deal with these cumulative increases and the accompanying similar increase in the disruption erosion problem.

TABLE 4.1-1
PLASMA THERMAL POWER HANDLING DEMAND

Device	P (MW)	R (m)	P/R (MW/m)
DIII-D	20	1.7	12
JET	40	3.0	13
C-Mod	8	0.67	12
DIII-D*	38	1.7	23
ITER	200	6	33
DEMO**	75	4.5	17
ARIES IV**	550	7.0	77

*With heating power upgrade.

**With advanced tokamak performance.

The most immediate manifestation of the "divertor problem" is the power loading at the target surface. The engineering difficulty of the divertor ultimately derives from the power-per-unit area (P/A) on the divertor structures. To first approximation, this areal power loading is given by

$$P/A = (1 - f) (P/R) (2\pi\Delta_{\text{eff}})^{-1} ,$$

where f is the relatively constant (from machine to machine) fraction (~50%) of power radiated from the core and SOL plasmas, and Δ_{eff} is the effective poloidal length of divertor structure wetted by the power flow. Essentially, all the ingenuity in divertor physics and design is expressed in determining the distance Δ_{eff} . For example, in the ITER CDA design with $P/R = 33$ MW/m, the equivalent wetted length for worst-case SOL conditions is 0.13 m, resulting in a peak P/A on the divertor plate of 20 MW/m². This high peak power density poses difficult and possibly insoluble engineering problems and also forecloses solutions to other divertor problems that become tractable at lower power density.

A method of dispersing the power flow over a greater area of divertor structure is the key development need.

For example, the total divertor region surface area that is available in the ITER CDA design corresponds to $\Delta_{\text{eff}} \approx 5$ m, which would yield a power density on the divertor structure of 0.5 MW/m². While completely uniform dispersal of power over a 5-m length is not possible, dispersal scenarios with a peaking factor of 3 (i.e., with local peaks of up to 1.5 MW/m²) are conceivable, provided a physics means of spreading the power reasonably uniformly over the available structure can be found.

Our planned RDP in DIII-D will develop a radiative dispersal solution to the ITER divertor power handling problem. As we explain below, we believe that radiation from impurity, rather than hydrogenic species, is the most likely mechanism by which a satisfactory dispersal scenario can be obtained. We plan to base our activity on this "impurity radiative divertor" approach, rather than the more difficult approach of the hydrogenic "gas-target" divertor. We note, however, that there is a continuum of cases between these two extremes in which both impurity and hydrogen radiation and charge exchange play a role, and we intend to conduct design studies to explore the full range of possibilities.

A key principle guiding our program is the need to provide research flexibility to settle the configurational questions of single- versus double-null pumping of the inner and outer divertor legs and more detailed geometric issues.

The overall goal of the DIII-D RDP will be to demonstrate operation with peak heat fluxes less than 1 MW/m^2 with ITER-relevant divertor stress parameters $P/R \sim 20 \text{ MW/m}$. These parameters correspond to an order-of-magnitude reduction in peak heat flux.

The divertor solutions implemented in DIII-D will be compatible with and contribute to the enhanced confinement and β performance needed in DEMO and ARIES IV.

We will be able to conduct our tests at ITER-relevant plasma power loadings. With our planned upgrade of rf heating power, the total heating power available will increase to 38 MW and (P/R in DIII-D will be 23 MW/m) close to the ITER value. Even with the present heating power of 20 MW, giving $P/R = 12 \text{ MW/m}$, the power demand on DIII-D's divertor is comparable to levels that will initially be run in ITER. The overall goal of the DIII-D RDP will be to demonstrate operation with peak heat fluxes less than 1 MW/m^2 with ITER-relevant divertor stress parameters $P/R \sim 20 \text{ MW/m}$. These parameters correspond to an order-of-magnitude reduction in peak heat flux.

4.1.2. RELATION OF THE DIVERTOR PROGRAM TO ADVANCED TOKAMAK PERFORMANCE

The Divertor Development and Research Program will also support achievement of AT operation in DIII-D and future tokamaks. In particular, divertor improvement will play an important role in the general development of the AT, particularly in obtaining enhanced confinement and also in making it possible to successfully employ the significant amounts of noninductive current drive power that most of the planned AT scenarios require.

The relationship between divertor efficiency and improved confinement is harder to quantify than the power handling function. However, most enhanced confinement regimes produced worldwide have arisen from plasma edge and recycling control and wall conditioning. The most outstanding examples are the supershot in TFTR, produced by helium discharge wall conditioning; the H-mode in DIII-D, which depends on helium glow wall conditioning; and most spectacularly, the VH-mode in DIII-D produced by boronization. Present experiments in DIII-D provide strong evidence that an actively pumped divertor that can controllably reduce core plasma density and increase temperature will produce further advances in confinement. Our divertor designs will, therefore, be compatible with enhanced confinement and β operation.

A look at the DEMO devices (Fig. 2.1-1) that may follow ITER makes clear the need for development of a divertor solution that works with AT features like current profile control. In the small AT DEMO or pilot plant that could follow ITER, the demands on the divertor are reduced. In such a device, enhanced confinement and β performance (to be developed in DIII-D, see Section 3) are used to reduce the size, plasma current, and power output of the device in order to achieve a low-cost first fusion pilot plant that can be built as the starting point for commercial optimization. The result is a lower power density device with an associated lower P/R than ITER. Hence, the divertor solutions developed in DIII-D should suffice for an advanced DEMO device, especially since they will be developed in concert with the AT thrust in DIII-D.

The actively cooled divertor in DIII-D will demonstrate safe, long-pulse technology needed in ITER, DEMO, and ARIES IV.

In the commercial reactor as exemplified by ARIES IV, the enhanced confinement and β are used to maximize the power output and power density in a modest-sized device. Accordingly, the divertor stress P/R rises to 77 MW/m, about twice ITER. Given that the device is also similar in size to ITER, we can expect divertor heat fluxes, *even with the same radiative divertor physics*, to be 1.5 MW/m² average with peaks up to 4.5 MW/m².

These higher heat fluxes and the higher reliability requirements inherent in a commercial reactor will place more demands on the technology of divertor construction. The DIII-D ACDP will implement the radiative divertor physics with hardware designed for both higher heat fluxes and steady-state cooling technology. The goal of the ACDP will be to demonstrate 60 sec operation with peak heat fluxes up to, but controlled to, 3 MW/m². This design level approaches the requirements of a commercial reactor.

4.1.3. DIVERTOR ISSUES

The physics and technology R&D issues and requirements for divertor design and operation have been the subject of continuing study and searches for solutions since the earliest days of the fusion program. The comprehensive study of the physics and technology R&D requirements for ITER produced by the ITER CDA is the most recent example. The CDA study identified peak power handling, excessive erosion caused by too high a plasma temperature and poor disruption survivability as especially serious issues. The CDA study also highlighted major concerns about the reliability, maintainability, and safety aspects of the fragile tiles-brazed-to-tube target designs developed during the CDA.

A complete discussion of the complex and interrelated series of divertor issues and requirements is beyond the scope of this plan. However, in Table 4.1-2, we summarize the various functions/needs that a divertor research program should address and identify what we plan to do on each issue during the various phases of the DIII-D Divertor Development Program. An "I" means the subject will be investigated in DIII-D with relevant information gained and promising paths to solutions tested. A "D" means a solution to the problem will be demonstrated.

TABLE 4.1-2
DIVERTOR RESEARCH PROGRAM FUNCTIONS/NEEDS WITH EXPECTED LEVEL
OF DIII-D RESOLUTION OF ISSUES (D = DEMONSTRATED, I = INVESTIGATED)

Issue	Divertor Development Phase		
	ADP	RDP	ACD
Power Dispersal	I	D	D
Density Control	D	D	D
Impurity Control	I	D	D
Helium Exhaust	D	D	D
Divertor Physics	I	I	I
Model Validation	I	D	D
Erosion Mitigation	I	I	D
Safe Coolant Technology	—	—	D
Remote Maintainability	I	I	D ^(a)
Tokamak Compatibility	D	D	D

(a) Not remotely maintained, but design features demonstrated will make remote maintenance much easier in future devices.

4.1.4. PROGRAM GOALS AND GUIDELINES

The Divertor Development Program will pursue the following goals and be implemented with the following guidelines:

1. Demonstrate radiative divertor power dispersal to lower the peak heat flux to $\sim 1 \text{ MW/m}^2$, from ITER levels of ~ 5 to 10 MW/m^2 .
2. Implement divertor configuration consistent with the overall AT mission of DIII-D and TPX.
3. Provide research flexibility to settle the questions of single- versus double-null, pumping of the inner and outer divertor legs, and more detailed geometric issues.
4. Implement the testing of AD concepts as part of the AT mission of DIII-D.
5. Demonstrate density control in enhanced confinement regimes.
6. Demonstrate impurity retention and entrainment in the divertor.
7. Demonstrate sufficient helium exhaust for a reactor.

8. Provide a validated model of divertor performance that can be used to design future machines.
9. Show erosion reduction by power dispersal.
10. Implement safe helium cooling technology.
11. Take advantage of heat flux reduction to implement and demonstrate safe and easily maintainable divertor hardware.
12. Engineer all divertor systems in all phases to be fully compatible with tokamak requirements, e.g., temperature ranges, thermal expansion and contraction, halo currents, disruption forces, arcing hazards, etc.

4.1.5. DIVERTOR DEVELOPMENT PROGRAM PLAN

The Divertor Development Program Plan is shown in Fig. 4.1-1. The figure identifies the three program phases and their main themes and a schedule of anticipated research results. The plan shows the integration of experimental research, hardware development, and code model development.

Phase I: The Advanced Divertor Program (ADP)

The ADP is primarily a divertor and SOL physics R&D program.

Begun in 1988, the ADP employs an electrically biasable baffled divertor ring that is now installed in the lower outside divertor channel of DIII-D. The ADP is primarily a divertor and SOL physics R&D program and is presently focused to provide both the key physics data needed for ITER divertor design and the feasibility demonstrations needed for the follow-on Radiative and Actively Cooled phases of the DIII-D Divertor Program.

The two main present thrusts of the ADP are the development of divertor biasing and divertor pumping. The principal objective is density control. The biased divertor ring ADP hardware has been in operation in DIII-D for about two years and promising results have already been obtained. The cryopump for divertor pumping will be installed in the fall of 1992. We expect this program to successfully demonstrate its objective of density control during 1993.

In addition, the ADP effort has also enabled an attack to be initiated on a broad front on many of the divertor physics and operation issues identified in Table 4.1-2. Investigations of the feasibility of radiative power dispersal by puffing fuel and impurity gases are presently under way. Studies of the physics of impurity entrainment will soon begin. Programs are also in place for helium transport and exhaust and erosion and redeposition studies. An extensive program of divertor and SOL measurements is in progress with the dual objectives of supplying a database for ITER and supporting development and validation of codes such as LEDGE, NEWT1-D, DEGAS, and DIVIMP. Initial emphasis in this code development activity is on ensuring that the right physical effects are in the codes.

DIII-D DIVERTOR DEVELOPMENT PROGRAM PLAN

Fig. 4.1-1. DIII-D Divertor Development Plan.

The DIII-D ADP is the most collaborative program element in the U.S. fusion program.

The ADP has also had as a goal the development of pathways and methods of inter-institutional collaborations. Major collaborators in the program are Lawrence Livermore National Laboratory (LLNL), Oak Ridge National Laboratory (ORNL), Sandia National Laboratory (SNL), and the University of California at Los Angeles (UCLA). All these institutions have defined roles in hardware and research tasks. Over half of the boundary physics and technology staff on DIII-D are from collaborating institutions. This strongly collaborative approach will be continued in the ADP work covered by this plan and carried through the RDP and ACDP phases.

Phase II: The Radiative Divertor Program (RDP)

The radiative divertor will be built with concepts and technology proven in the ADP, but implemented in a geometry optimized for radiative power dispersal.

The RDP will demonstrate power dispersal and reduction of heat flux to the divertor structure. It will draw on techniques investigated in the ADP: divertor bias and pumping, gas fueling, and impurity entrainment; but will implement them in a geometry that is optimized for radiative power dispersal and which also allows these techniques to be applied to all four channels of double-null diverted plasmas or both channels of single-null diverted plasmas. The RDP will rely on atomic physics processes, radiation and charge exchange for the power dispersal.

We will implement the radiative divertor with inertially cooled (rather than actively cooled) divertor surfaces. The reasons for this choice are lower cost, faster schedule, simplicity, and greater flexibility to accommodate studies of the effects of plasma geometry variations. Even if radiative dispersal is only moderately successful, an inertially cooled RDP will be adequate for 10- to 20-sec pulse lengths and is thus well matched to the pulse length capability to be provided during the RDP time frame by other DIII-D systems.

The RDP hardware installation will be based on hardware developments already proven by present DIII-D operation or soon to be proven in the ADP. DIII-D's graphite tile mounting systems are well developed and very reliable. We already know how to engineer a successful divertor bias system; the present system has been trouble-free since its last modification in December 1991. Operating experience with in-vessel cryopump operation will be available in 1993, in time for the RDP hardware design.

Because we believe that geometry optimization is a critical aspect of the overall DIII-D divertor and AT programs, we will not commit to the plasma shape and specific divertor configuration until late in FY93 (see major milestone in Fig. 4.1-1). Based on present data, we expect future studies to confirm that best confinement and β performance is obtained in highly triangular double-null plasmas. We also expect the ADP to show that the confinement in low triangularity single-null plasmas can be improved by divertor pumping and density reduction. We will then be strongly motivated to combine these two routes to improved performance by implementing a highly triangular, double-null radiative divertor of the type planned herein. However, we are also evaluating the slot divertor gas target approach, which requires low triangularity. The probable end decision may be a compromise that

trades off triangularity for pumping access to the inboard leg of the divertor. Since geometry optimization is the key element of the RDP, and since the inertially cooled design will allow at least one fairly major adjustment of the divertor geometry during the operational period, we show such a possible modification in early 1997. An adjustment in the tightness of the divertor baffling is the most likely candidate modification.

A vigorous program of code development is part of the RDP.

Input to these design decisions at the end of FY93, and general support for the physics feasibility of the radiative divertor will be obtained largely from existing divertor and SOL modeling codes. The neutrals package in the two-dimensional code LEDGE with a noncoronal impurity radiation post-processor, will be upgraded to confront the gas target problem (fall 1992). DEGAS can address the important problem of maintaining high neutral pressure in the divertor and five orders of magnitude less neutral pressure near the main plasma. NEWT1-D is being used to access the process of impurity entrainment, albeit in one dimension. We are acquiring access to the two-dimensional Monte Carlo impurity code DIVIMP.

The DIII-D divertor research facility will be the most complete in the world for sorting out the details of divertor design.

While the above approaches will yield enough support of the basic physics to warrant proceeding with the radiative divertor project, they fall well short of an integrated treatment in one code of the plasma flow, neutral transport, and impurity transport processes in 2-D. Our LLNL collaborators are developing the LEDGE code for this purpose. It should be available in late CY94. The results, coming just when construction of the RDP hardware begins, will provide some design confirmation. This version of LEDGE will be available for the last year of ADP operation for calibration to those results. Hence, we will be in the strong position of starting the radiative divertor operation with a developed, tested, and completed model to compare to the experimental results. LEDGE will also help to guide the geometry change at the start of CY97.

We expect the radiative divertor to succeed at power dispersal while maintaining density control, impurity control, and helium exhaust. We also plan to make use of the plasma shaping flexibility of DIII-D to address the issue of single- versus double-null operation and details of strike point positioning on the divertor structure. We will design the RDP system to enable settling such questions as how much pumping and field line length are needed on the inner versus outer divertor legs. We will be able to assess and manipulate SOL $E \times B$ flows with the biasing system. The DIII-D divertor research facility will be the most complete in the world for sorting out the details of divertor design.

Phase III: The Actively Cooled Divertor Program (ACDP)

The implementation of an actively cooled divertor structure will also allow us to investigate divertor structure plasma-surface effects under conditions where the divertor surface temperature reaches nominal steady state.

The ACDP will provide ITER with a test of the radiative divertor implemented with steady-state technology and will also support operation with pulse lengths in excess of 60 sec. The implementation of an actively cooled divertor structure will also allow us to investigate plasma-surface effects on the divertor structure under conditions where the divertor surface temperature reaches nominal steady state after only a few seconds of operation (as opposed to the continuing surface temperature rise inherent with the inertially cooled RDP divertor structures).

The research goals for the ACDP will focus more on technology development and long-pulse physics and operational issues. The most important technology development goal will be to verify that the helium cooling or advanced water cooling system works satisfactorily in an actual tokamak operational environment. We also expect to obtain valuable operational experience concerning the response of the actively cooled target design to the real-world combination of normal plasma operation and the electromagnetic and thermal effects of occasional plasma disruptions. Although remote maintenance capability will not be a part of our design, we anticipate that success in the RDP program will allow us to engineer and test an actively cooled system with features amenable to easy remote maintenance.

The ACDP capability will also allow us to begin to study long-pulse physics and plasma-surface interaction effects. The longer 60 sec DIII-D pulses allow the study of density, impurity, and helium control on a quasi-steady-state basis. The rapid thermal equilibration time (~2 sec) of the actively cooled divertor target will also simplify interpretation of plasma-surface interaction effects (e.g., hydrogen and impurity implantation at the target surface). Finally, the long pulses possible with the ACDP hardware will allow meaningful total system erosion measurements in an actual, not simulated, tokamak environment.

The ACDP will interact strongly with ITER EDA Technology R&D.

The ultimate choice of the divertor surface material, heat removal structure construction and cooling technology that we will implement for the ACDP hardware remains open. To make this choice, we plan to work closely with the ITER EDA Technology R&D effort and the ITER Joint Central Team. This collaboration will allow us to derive maximum benefit from the extensive plasma-facing-component R&D effort anticipated for (and funded by) the ITER EDA.

At the present time, tiles brazed on substrates are the most credible technology (albeit with serious concerns about cost and reliability), and we have laid out our cost and schedule for our actively cooled divertor on this basis. However, we feel braze technology will likely not provide adequate reliability for ITER and, thus, expect that alternate developments will be pursued in the context of the ITER Engineering Design Activity (EDA). If the radiative divertor works to lower peak heat fluxes below 3 MW/m^2 , then monolithic beryllium structures or even beryllium or graphite tiles clamped to cooling structures become feasible options. The clamped arrangement would be very attractive for remote maintenance since only the tiles and

not the whole cooling structure should require the most frequent replacement. We also favor the use of helium gas cooling technology in the actively cooled divertor, based on the well-known safety and operational reliability features of helium cooling, as well as GA's extensive experience with high temperature gas-cooled reactor (HTGR) technology.

4.1.6. SCHEDULE AND RUN TIME CONSIDERATIONS

The hardware schedule for the RDP and ACDP divertor installations is shown in Fig. 4.1-2. The schedule is funding-constrained in FY93. The critical path element in the radiative divertor system is the carbon-fiber-composite procurement and machining.

In Fig. 4.1-3, we show a year-by-year estimate of the run time necessary to carry out this ambitious and broadly based program of divertor research. Although the program has the major thrusts identified previously, we expect to carry a broad agenda of physics and technology studies through the various phases of the program in order to assure our divertors fulfill all the functions necessary in ITER and future

Fig. 4.1-2. Schedule of divertor hardware construction, installation, and operation.

Fig. 4.1-3. Schedule of estimated DIII-D run time for the divertor development program.

machines. As in the present ADP program, we also anticipate that serendipity may lead to modifications of the experimental plan and/or focus as studies proceed. Years with low-run weeks are the years in which in-vessel divertor construction takes place.

5. OPERATIONS AND FACILITY IMPROVEMENTS

The research program planned here will require (1) continued safe and reliable operation of the tokamak in new plasma configurations, (2) meeting the needs of expanding numbers of collaborators, (3) upgrading the capability of the facility for longer pulse operation, (4) installation of new advanced tokamak rf current profile control systems, (5) new divertor configurations to disperse the peak power incident on the divertor and reduce the incident plasma temperature, (6) a higher level of plasma operations than heretofore achieved.

5.1. INTRODUCTION

Reliable operation of the DIII-D facility in new tokamak regimes is essential to completing the proposed research.

The research programs described in this plan depend on the efficient reliable operation of the DIII-D facility and requires that certain subsystems be improved in order to conduct the planned research program. An extensive on-going effort is needed to provide reliable operation of the DIII-D tokamak and its auxiliary systems. To meet the operations requirements of the research programs, a highly skilled operations staff has been assembled and a comprehensive maintenance program has been developed that has allowed the DIII-D facility to become one of the most productive tokamak facilities in the world. With the addition of new subsystems and the emphasis on integrated approaches to resolving tokamak issues, reliable operation of the DIII-D facility is increasingly important. This effort has become increasingly more demanding due to the increasing complexity of the facility, increasing numbers of diagnostics, and increasing participation by collaborating scientists. The combination of these changes and stagnant funding levels has resulted in decreased operations and reduced emphasis on development of the facility. A critical element in this plan is the safe, sustained, and reliable operation of the DIII-D facility.

DIII-D is a major fusion research facility.

DIII-D is the second largest magnetic fusion facility in the U.S., nearly a city block in size. A plan view of the entire DIII-D facility is shown in Fig. 5.1-1. Here a brief overview of the facility is given before a discussion of the operations requirements is undertaken. At the center of the DIII-D facility is the DIII-D tokamak, a large, noncircular cross-section tokamak designed to allow a large range of plasma configurations to be established in the same device. Figure 5.1-2 is a photograph taken inside of the plasma chamber. A number of the port openings that allow access for diagnostics and auxiliary heating are visible in the picture. Armor tiles, designed to protect the vacuum vessel wall, and the advanced divertor baffle plates, a fast wave current drive (FWCD) antenna, and two 110 GHz electron cyclotron heating (ECH) antennas can also be seen. The tokamak is located inside a radiation shield (shown in Fig. 5.1-3) along with the neutral beam injectors and other

Fig. 5.1-1. Plan view of the DIII-D facility, which spans a half city block in size.

Fig. 5.1-2. DIII-D plasma chamber.

Fig. 5.1-3. DIII-D tokamak.

auxiliary equipment that need to be close to the tokamak. The radiation shield and the tokamak are located near the center of the figure, surrounding the radiation enclosure (in the same building) are the neutral beam and rf power supplies, the rf sources and computer and control rooms. Further from the tokamak (in separate buildings) are the motor generators, utility substations, diagnostic laboratories, and neutral beam transformers. In this section, we discuss the requirements for operating the DIII-D facility and upgrades to the facility required to support the planned research programs.

Special attention is given to operating the DIII-D facility in a safe and responsible manner. DIII-D has an extremely well regarded safety program which is supported by the excellent GA safety program. These programs provide supervision and training for all hazardous activities. The DIII-D safety program is also reviewed and audited by a wide range of regulatory and insurance agencies including OSHA, the NRC, the EPA, the San Diego Fire Department, American Nuclear Insurers, and seventeen other agencies. Radiation related issues are managed through an on-site staff member and the General Atomics Health Physics program. Radiation levels to the public have been maintained well within the guidance of DOE consistent with ALARA (as low as reasonably achievable) principles. Radiation to personnel at the DIII-D facility is carefully monitored and maintained well within regulatory guidelines.

Future DIII-D operations are expected to result in increased hazardous activities in several areas. Increased electrical power and energy levels will be required to meet program goals; this can be managed within the present hazardous work authorization and power dispatching procedures. Increased radio frequency power will be required and this can be monitored through existing hazardous work authorization and monitoring efforts. The device is expected to produce more neutrons than presently produced so it is planned to increase the radiation levels at the site boundary (remaining within existing regulations) and to increase the radiation shielding.

5.2. OPERATING CAPABILITIES AND REQUIREMENTS

The DIII-D tokamak is a robust device with flexibility to address a wide range of plasma physics and magnetic fusion issues. A poloidal cross section of DIII-D is shown in Fig. 5.2-1. The extensive poloidal coil set and associated power supplies, coupled with graphite armor on the vacuum vessel, provide unique plasma shape flexibility. Examples of the types of plasma shapes that have been produced, or could be produced in DIII-D, are shown in Fig. 5.2-2, including: aspect ratio ranging from 2.5 to 5.3, circular cross sections to elongations up to 2.5, and small shifted or large indented plasmas. Access for both diagnostics and heating systems on DIII-D is excellent and has allowed the installation of substantial and varied auxiliary heating systems along with an extensive diagnostic set. The ability to bake the vessel to high temperatures (400°C) and an aggressive wall conditioning

Fig. 5.2-1. DIII-D capabilities allow a wide range of research and technology issues to be addressed.

program have produced excellent vacuum and wall conditions. The flexibility of the device has allowed us to make substantial contributions to the understanding of tokamaks. Table 5.2-1 lists some of the maximum achieved DIII-D parameters.

A number of substantial systems are needed to operate the tokamak.

In order to maintain, develop, and effectively utilize these unique capabilities of the DIII-D tokamak, a large number of separate but interlinked systems have to be kept operational. Figure 5.2-3 is a representation of the DIII-D facility showing the DIII-D tokamak, its auxiliary systems, and the principle interconnections between those systems. Since DIII-D operates in a pulsed manner, input power is supplied by both utility substations directly, and by utility driven motor generators (flywheel storage). This power is converted to direct current using a set of 25 power supplies and a similar number of precision regulators. This power is used to supply both the coil system and the heating systems. The heating systems have 27 MW of installed power and up to 40 MW of power is envisioned under this contract. A substantial 125 l/hr cryogenics system is needed to supply the needs of the heating systems (vacuum pumping and superconducting magnets) and the research program (advanced divertor cryopump). Most of the power supplied to the experiment is deposited by joule heating in the coils or by heating of the vessel walls, and all of this energy must be removed by the water-cooling system in the period between discharges. High quality vacuum and wall conditions are maintained using the

Fig. 5.2-2. DIII-D plasma shape flexibility.

vacuum pumping system and related hardware for baking, discharge cleaning and boronization of the chamber. A set of over fifty plasma diagnostics described in Volume II is used to characterize the plasma. Control of the principle operating systems, the heating systems, and the key diagnostics is done by computer. Another extensive set of computers collect the data produced by the plasma diagnostic set. A third set of computers are used to analyze the data. A substantial amount of the data analysis is done between plasma discharges with the results used by the operators to help determine the direction of the on-going experiment. More detailed analysis is

TABLE 5.2-1
MAXIMUM DIII-D PARAMETERS ACHIEVED
TO DATE 6/19/92, NOT SIMULTANEOUS

R	Nominal	1.67 m
a	Nominal	0.67 m
κ	Maximum	2.5
B_T	Range	0.5–2.2 T
I	Divertor and limiter	3.0 MA
q_{95}	Lowest	1.75
I/aB	Maximum	3.3
$\beta_T(0)$		44%
$\langle \beta_T \rangle$		11%
β_p		5.2
$\epsilon \beta_p$		1.8
n_e		$1.4 \times 10^{20} \text{ m}^{-3}$
$T_e(0)$		7 keV
$T_i(0)$		17 keV
W (5% β full field)		3.6 MJ
τ_E		0.44 sec
H-mode duration		10.3 sec
Z_{eff}	Lowest	1.3
P_{NBI}		20 MW
P_{ECH}		1.7 MW
P_{ICH}		2.0 MW
$n\tau_E$		$0.4 \times 10^{20} \text{ m}^{-3} \text{ sec}$
$n\tau_E T_i$		$2 \times 10^{20} \text{ m}^{-3} \text{ keV sec}$

done on a much longer time scale and the results are disseminated to the fusion community and used to further develop the DIII-D research programs.

As the capability of the DIII-D facility has increased, greater demands have been placed on the the reliability of each of its subsystems. In many cases, the research programs require simultaneous operation of most of the facility, further increasing the reliability required of both the overall facility and of the individual subsystems.

Fig. 5.2-3. A wide variety of hardware system are needed to operate the DIII-D facility.

In order to meet these increasing operating demands, we have undertaken a facility wide program of preventive maintenance. This program tracks and reports all maintenance activities, identifies problem areas, tracks staff training and aids in staff planning. Furthermore, a program of hardware refurbishment is on-going; this is of particular importance to the DIII-D project due to the age of the facility and of much of the equipment (the facility was commissioned in 1978). The program of refurbishments will provide needed flexibility and reliability, particularly in the area of system controls, and it will reduce maintenance and labor costs.

An experimental run period (typically two to three weeks followed by two weeks of general maintenance and experimental modification) requires a coordinated effort from the groups operating the subsystems in order to be successful. The goals of the research programs that are carried out at the DIII-D facility are implemented by the staff and detailed experimental plans are developed.

The staff required to operate, maintain, repair, and refurbish the various subsystems in the DIII-D facility in FY92 is shown in Fig. 5.2-4 along with the labor expended to develop new hardware and to perform data analysis and interpretation. This effort

Fig. 5.2-4. Diagram of allocated staff requirements, numbers are in full-time equivalents (FTE) for FY92.

TABLE 5.2-2
TENTATIVE ALLOCATION OF DIII-D OPERATIONS TIME
IN SINGLE-SHIFT WEEKS FOR DIFFERENT RESEARCH AREAS

Research Area	FY92	FY93	FY94	FY95	FY96	FY97	FY98
Divertor	2.2	7	11	8	10	4	9
RF CD	2.2	3	6	2	5	2	1
Advanced tokamak physics and transport	8.0	1	13	8	15	12	18
Plasma development	1.6	1	3	3	2	2	3
Unallocated	0	0	3	3	4	4	5
Total	14.0	12	36	24	36	24	36

allowed 14 weeks of single-shift operation. In FY94, FY96, and FY98, we assume 18 weeks of double-shift operation (36 single-shift weeks). We have used the research scenario given in Table 5.2-2 for our planning purposes. FY93 operations are severely budget constrained. In FY95 and FY97, the research operations are reduced to 24 single-shift weeks to allow installation of new divertor configurations. A brief description of the responsibilities of each of the operations related groups shown in Figs. 5.2-3 and 5.2-4 is summarized in Table 5.2-3.

The DIII-D project has been productive due to the flexibility of the tokamak coupled with the ability to maintain safe, effective, and reliable operation for the facility. We plan to continue our efforts aimed at providing safe reliable operation of the DIII-D facility.

5.3. FACILITY DEVELOPMENT

New and improved research tools are required.

Continued progress in addressing tokamak issues is dependent on continued improvements in the tools used by researchers. In this section, we discuss improvements to the facility necessary to support the planned divertor and advanced tokamak (AT) research programs, upgrades, and additions to subsystems to increase the overall quality and quantity of research, and subsystems required for longer pulse operation.

New diagnostics are needed to provide relevant data.

The increase in overall research output is illustrated by the increase of plasma diagnostic instruments and physics data output. Figure 5.3-1 shows how the amount of data collected has grown in the past five years and the projected data collection through 1998. Increases in the amount of data collected come primarily from increases in the number and sophistication of plasma diagnostics (Fig. 5.3-2 plots diagnostic systems versus time as reference) and increases in the pulse duration. A description of the AT and divertor diagnostics is listed in Table 5.3-1 and are discussed in the respective earlier sections.

Facility improvements are also needed to provide new research capabilities; such as longer pulse length and rf current profile control power. The peak utility power demand is plotted in Fig. 5.3-3 as a function of time from 1979 (first operation of the Doublet III tokamak) through 1998 (the end of the planned five-year contract). The power usage has increased at a fairly constant rate, reflecting an increase in the amount of power used for experimental research and, more importantly, reflecting an increase in the amount of power that can successfully handled by the tokamak. The projected increases in power through 1998 are primarily due to expanded capability of the rf auxiliary heating systems and long-pulse operation. Table 5.3-2 summarizes the changes to the facility necessary to carryout the planned research. The first five: divertor improvements and AT rf heating and current drive systems, as well as the required diagnostics, were described in previous chapters. Detailed descriptions of the remaining planned upgrades and improvements are discussed in the remainder of this section.

TABLE 5.2-3
RESPONSIBILITIES OF DIII-D OPERATIONS TEAM

Tokamak operations and machine control

- Plan and carry out all facets of device operations.
- Integrate all DIII-D research hardware and operational activities.
- Provide mechanical and electrical engineering and design
- Conduct radiation monitoring.
- Carry out operations related research including:
 - * Disruption studies.
 - * Digital plasma control.
 - * Start up and shutdown studies.
 - * Plasma wall heating and impurity influx.
 - * Neutron diagnostics.
- Coordinate all facility related collaborator activities.

• Neutral Beam Operations

- Operate and maintain:
 - * Four neutral beamlines.
 - * Eight ion sources.
 - * Neutral beam power systems controls.
 - * Instrumentation and beamline diagnostics.
 - * Perform source conditioning.

• RF Operations

- Operate and maintain:
 - * Seven 60 GHz gyrotrons, waveguides, and antennas.
 - * One 110 GHz gyrotron, waveguide, and antenna.
 - * Transmitters, transmission lines, and antenna for 30 to 60 MHz system

• Power Systems

- Develop, operate and maintain device electrical systems including:
 - * Motor generators.
 - * Utility substations.
 - * Coil power systems and regulations.
 - * Heating power systems.
 - * Instrumentation and control systems.

• Fluid Systems

- Operate and maintain cryogenic systems, including:

TABLE 5.2-3 (CONTINUED)

- * He liquefier and recovery system.
- * LHe distribution systems.
- * LN₂ system.
- Operate and maintain water cooling systems for:
 - * Tokamak vacuum vessel and magnetic coils.
 - * Auxiliary heating systems.
 - * Power systems.
 - * Diagnostics.
- **Vacuum Systems**
 - Operate and maintain:
 - * Tokamak primary vacuum systems.
 - * Auxiliary vacuum systems for diagnostics and other systems (including 90 separate vacuum pumps).
 - Develop and implement wall conditioning:
 - * Machine baking.
 - * Boronization and carbonization glow cleaning.
 - * Glow cleaning.
 - Ensure the integration of the various vacuum systems:
 - * Vacuum leak checking.
 - * Monitor and control tokamak primary vacuum work.
- **Diagnostics Operations**
 - Operate and maintain plasma diagnostic systems.
 - Implement improvements on diagnostics.
 - Coordinate installation of new diagnostic systems.
- **Computer Systems**
 - Provide computer and programming support for all DIII-D activities.
 - Develop and operate computers for control of tokamak and heating systems.
 - Develop and operate computers for data acquisition, storage, and analysis.
 - Provide network services.
 - Implement and overview personal computer interfaces.
- **Safety**
 - Ensure that the entire facility is operated in a safe manner.
 - Provide on-site supervision of all radiation producing activities.
 - Provide explicit overview of all hazardous tasks.

Fig. 5.3-1. Growth of collected data in the past five years and the projected data collection through 1998.

Fig. 5.3-2. Plot of diagnostic systems versus time.

TABLE 5.3-1
NEW DIAGNOSTICS NEEDED TO CARRY OUT
THE ADVANCED TOKAMAK AND DIVERTOR RESEARCH PROGRAM

Advanced Tokamak Diagnostics

Core plasma diagnostics

Central Thomson

Superheterodyne electron cyclotron emission

Fluctuation diagnostics

High frequency magnetic probes

Beam emission spectroscopy (edge and central)

Improved microwave reflectometry, scattering, PCI

Current profile diagnostics

Motional Stark effect radial channel upgrade

X-ray spectrometer radial array

Improved soft X-ray imaging

Divertor Diagnostics

Toroidal and poloidal asymmetries

Infrared and visible TV cameras

Upper divertor Langmuir probes and pressure gauges

Inner strike point pressure gauge

Impurity transport and radiation

Multichannel divertor spectroscopy

Impurity gas injector and laser blowoff

Erosion and redeposition

Divertor real time interferometry

Ion and sheath physics

Time of flight analyzer

Ion energy analyzer

Basic scrape off layer parameters

Reciprocating divertor probe

Divertor interferometer

Divertor Thomson scattering

The pulse duration requirements for advanced tokamak and divertor experiments can be met by modest extensions of present DIII-D systems.

5.3.1. TOKAMAK MAGNET SYSTEMS PULSE EXTENSION

In order to carry out the planned research, the pulse length capabilities of DIII-D must be extended. Extending the pulse length at full field and current to 10 sec would allow experiments in the AT program, and in particular experiments with non-inductive current drive to be carried out for durations comparable to the inductive current equilibrium timescale. The ability to conduct experiments with

Fig. 5.3-3. Peak utility power demand.

60 sec duration, albeit at half plasma current (1 MA) and toroidal field (1 T), would allow divertor and confinement experiments to be carried out on a timescale comparable to the wall equilibrium time. The hardware needed to meet these requirements is in many cases similar.

The DIII-D tokamak is now capable of operating short pulses (2.5 sec) at full field (2.1 T) and currents of 2.5 MA over a wide range of plasma configurations. The machine has been tested to plasma currents of 3 MA in restricted configurations. It is also capable of operating for 10-sec pulses, but at low plasma current (0.7 MA) and low auxiliary plasma heating power (3 MW). The magnetic coils are capable of longer (10 to 60 sec) pulse operation as illustrated by Fig. 5.3-4. The fundamental limitations of both the poloidal field coil system and the toroidal field, along with many of the power system components is joule heating. This occurs because most of the heat is removed between pulses and the joule input determines the temperature rise. Up to 2 MA at 2 T can be sustained for 10 sec and operation up to 60 sec can be sustained at half field and current. In order to operate for these longer pulses, several subsystems need to be upgraded. These systems are magnet power supplies, magnet

TABLE 5.3-2
SUMMARY OF PROPOSED DIII-D FACILITY IMPROVEMENTS

Sybsystem	Improvement	Reason
Radiative divertor	To disperse the peak divertor heat load	To demonstrate viable divertor physics concept for ITER
Actively cooled divertor	Active cooling for long pulse operation	To demonstrate viable divertor engineering concept for ITER
Fast wave	Increase power to 8 MW	Advanced tokamak current drive and divertor research
Microwave ECH	Increase power from 2 to 10 MW	Advanced tokamak current profile control and divertor research
Lower Hybrid	Adds 3 MW LHCD power	Edge current profile control
Diagnostics	Thirty-one new diagnostic systems	Support for divertor and advanced tokamak research
Tokamak facility	Upgrade magnet power supplies and cables for 10 and 60 sec operation	Conduct divertor and advanced tokamak research
	Add two additional 84 MVA utility transformers	Provide higher power for divertor research
	400 GPM of deionized water for vessel cooling; add 4 MW of water cooling capacity for ECH	Long-pulse operation and 1 MW ECH gyrotrons
Neutral beam	Upgrade internal heat handling components and power supplies	Long-pulse operation (10 or 60 sec)
Computer systems	Replace data acquisition and control computers	Old systems obsolete, unreliable, and lack capacity
Remote experimental site	To provide interface to link with RES developments at LLNL	To provide users effective nationwide access to DIII-D facility
Radiation shielding	Increase in neutron shielding	Increased higher performance operations with longer pulse
Building modifications	Increased space will be provided by GA	To accommodate the above improvements

buswork (including toroidal field coil belt bus), cooling water system capacity, and electrical utility systems.

The power supplies that are used in the field shaping coil system must be upgraded for pulse lengths in excess of 5 sec flattop duration. These upgrades are principally modifications to increase the energy dissipation ratings of resistors, varistors, and snubber components used in the existing high-current switch regulator system that consists of 36 modules. In addition to these improvements, it is necessary to add 10 modules to provide sufficient shaping coil current for the long-pulse discharges. One additional line-commutated ac/dc converter power supply is also needed to supply additional input power. This upgrade will provide the needed capability to support discharges of up to 60 sec in duration.

Fig. 5.3-4. The magnetic coils are capable of longer pulse operation.

The toroidal field coils are capable of long-pulse operation as shown in Fig. 5.3-2. However, the toroidal electrical belt buses that connect the 24 toroidal field coil bundles together and the return bus have a thermal limit that limits the full field pulse length to 4.5 sec. In order to increase the pulse width, we plan to replace the belt buses with new bus work that will match the capability of the toroidal field coils.

Some of the cables used in the poloidal field-shaping system also have a thermal limit that restrict the plasma pulse length. To extend the plasma pulse length, a careful evaluation of each of these cables is needed, and we expect that a significant number of them will have to be replaced or reconfigured.

5.3.2. RF POWER SYSTEMS

Additional ICRF and ECH power is needed to control the current profile.

The AT Program calls for the upgrade of the 110 GHz ECH system to 10 MW, the upgrade of the FW system to 8 MW, and the addition of a 3 MW 2.45 GHz LH system. These systems are described in detail in Section 3.1.5.

5.3.3. ELECTRICAL UTILITY SYSTEMS

In order to support the upgrades and new systems planned here, we must augment the existing electrical utility systems at the DIII-D site. The primary power demand for the new rf systems is approximately 51 MVA. The upgraded neutral beam power systems will represent an input power demand of 114 MVA. The total auxiliary heating power demand will thus grow to approximately 165 MVA.

The power for the auxiliary heating systems is currently derived from a 138 kV utility line from San Diego Gas and Electric (SDG&E). Preliminary discussions with engineers at SDG&E indicate that the 165 MVA loading will be acceptable with a possible minor upgrade of the transmission system.

Additional substation capability is needed to support new power demands.

The 138 kV substation currently consists of one 138 kV to 12.47 kV step-down transformer, which is 1 sec pulse rated at 84 MVA. The 10 sec pulse rating for this transformer is 56 MVA. We plan to upgrade this substation to include two additional transformers to achieve a total 10 sec pulse power rating of 168 MVA. The power rating for a 60 sec pulse will be 130 MVA which appears adequate. Switchgear and distribution system for the new rf system loads are included in their respective system cost projections. Procurement of the new transformer will begin in FY94 and be completed in FY95.

The additional transformers also provide backup for the existing transformer in the event of a failure, thus preventing a major shutdown of the project over several months should this kind of a mishap occur.

5.3.4. NEUTRAL BEAM SYSTEMS

Active cooling and power supply upgrades are the key to long-pulse neutral beam operation.

In order to carry out long-pulse (10 to 60 sec) divertor experiments, several improvements are needed in the neutral beam system. These changes would allow operation with a total deuterium beam power of 20 MW or more from eight ion sources. The DIII-D neutral beam systems presently operate routinely at power levels near 20 MW for 3.5 sec pulses. The ion sources are capable of 60 sec operation. The following subsystems require upgrades in order to extend the beam pulse length: beamline internal heat handling components, beamline drift ducts, beam diagnostics and interlock system, water cooling system, and power supply system.

Existing internal heat handling components (beam dumps, various collimators, etc.) that use inertial or water cooling and were designed for 5 sec, 14 MW hydrogen beam operations, will be replaced by components with active water cooling. The gap between the pole faces of the residual ion bending magnet will be enlarged to allow installation of better protection for the magnet poles.

Damage to the beamline drift ducts by the high power deuterium beams with pulse length less than 5 sec has been observed. This has been attributed to the bending and focusing of the reionized energetic particles within the drift duct region by the DIII-D toroidal and poloidal magnetic fields. Since the reionization rate is directly proportional to the residual gas density in the region, additional pumping will be installed. Because the residual gas density is highly dependent on the cleanliness of the drift duct surface, duct surface conditioning using short pulses will be required prior to long-pulse operation.

The existing beam diagnostics need minor improvement for longer pulse operation. A new diagnostic to monitor the temperature rise of the ion source accelerator grids for source protection will be installed. In addition, the monitoring and interlocking system for beamline internal components protection, vacuum and cryogenic status, and water cooling system, etc., will be upgraded based on the needs of the new system and the experience obtained from operation of the existing system.

Four of eight high-voltage transformer rectifier units will have to be upgraded in order to allow full-power neutral beam operation for long pulses. In the near future two of these units will have to be refurbished in any case, and as part of that refurbishment, will be upgraded to allow longer pulse operation.

Additional energy throughput requires additional water cooling.

5.3.5. WATER SYSTEM

The water system will also need to be upgraded to provide the additional cooling for the tokamak walls, field coils and their power supplies, tokamak auxiliary power system and their power supplies, and the motor generators called for in the plan.

To support a 10-sec pulse with 5 pulses-per-hour will require an upgrade of the primary pumping capacity for deionized water, both in volume and delivery pressure, to deliver about 400 GPM for vessel wall cooling and 300 GPM for neutral beam internal cooling. In addition, a 4 MW increase in cooling tower capacity (from 15 MW to 19 MW) is needed. Finally, an upgrade of the ECH deoxygenated, high resistivity, cooling system to accommodate the 1 MW gyrotrons will also be required. This ECH cooling system upgrade would about double the existing pumping capacity. The gyrotron vault would be reconfigured to accept the much larger water distribution headers and manifolds. Experiments with 60-sec pulses with 15 min repetition rates would be accommodated by the use of chillers to provide cooling water at 20°C.

5.3.6. COMPUTER, DATA, AND INSTRUMENTATION SYSTEMS MODERNIZATION

The outdated DIII-D computer system will be replaced with modern computers to meet expanded demands.

As more and more national collaborators focus their efforts on DIII-D, the DIII-D computer capabilities will need to be expanded accordingly to accommodate more users, more and longer plasma shots, and multiple experiments with digital plasma control on each shot. Thus, we plan to provide a new computer and data acquisition systems. This is also appropriate because the current systems are at the end of their lifetime and advances in the electronics and software field now make available to us totally new capabilities.

The planned system will be easily expandable, modular, and based on open system concepts. This means that changes and additions can be made to subsystems without affecting the global system. As new computer hardware and electronics becomes available, they can easily be incorporated. Distributed computing modules that are networked at high speeds will perform many different functions. This networked aggregate will provide all the control, data acquisitions, archival storage and analysis for each tokamak pulse.

This new system will provide many capabilities that are not presently available. Faster CPUs, buses, and instrumentation will allow many calculations to be done during the pulse and displayed for the experimentalists. These calculations could include temperature and density calculations, global power, MHD parameters, and flux lines and shapes. Many "events" that occur during a shot could be sensed and trigger additional actions. This could include changing data acquisition rates for certain signals, starting up additional diagnostic experiments asynchronously, and modifying operating parameters. The large amount of data generated could be examined as it is collected. Much data would be reduced to smaller amounts automatically and non-meaningful data discarded. The system could handle up to 500 Mbytes of data if required, compared to the 60 Mbytes we presently acquire.

New computers will help meet the demands of collaborators.

The new computer systems will be designed with remote capabilities in mind, including remote computers, subsystems, subsystem control, data handling, and other features that will allow enhanced collaborative efforts that would allow collaborators to carry out some activities from off the GA site. These efforts will focus on a collaboration between LLNL and GA as a test bed. The remote experimental site concept being developed by LLNL is described in the next section.

New computers will be phased in to minimize disruptions to on-going work.

We plan to replace the computers using a phased approach to minimize the impact on the ongoing DIII-D program and to efficiently use the existing staff. The data acquisition system would be replaced first. All CAMAC data transfer activities of the three existing data acquisition computers would be moved to one perhaps two new computers. The functions of the control system would be moved over to a new computer approximately one year later. The four neutral beam systems would be moved to two new computer systems during the last year. At this time (two to three years after initiation), all of the existing computers will have been replaced.

The data acquisition hardware — principally analog to digital convertors, timing and logic control systems — will be modernized in order to reduce maintenance costs, improve reliability, and enhance the flexibility of the way we collect data. The improved flexibility will accommodate longer pulse discharges and ease the use of the new sophisticated diagnostics that will be installed on DIII-D. The new sophisticated diagnostics that will be installed will take the form of dedicated workstations with memory coupled directly to analog to digital convertors. These workstations could then in real time make decisions on how the data is sorted, which data is collected and at what rate. After the plasma discharge, these workstations would transfer the sorted data to the main data acquisition computers described above. A new timing system will be developed that will coordinate the timing for these systems and will distribute timing information generated anywhere in the facility.

5.3.7. ADVANCED PLASMA CONTROL

Advanced Plasma Control provides the framework for integrated plasma control and improved reliability.

The development of advanced plasma control concepts is crucial to carrying out the research program planned herein. An advanced plasma control system provides precise control of the plasma position throughout the duration of long pulse discharges, whereas present-day systems are limited by the time independent nature of the programming algorithms. This system also provides the capability of controlling a wider range of plasma parameters such as the plasma energy or reaction rate, the plasma current profile, and the coupling of rf power to the plasma. The use of a fully digital control system provides considerable flexibility in changing from one plasma configuration to another, both to allow the efficient utilization of long pulse discharges and to permit the rapid change from one discharge type to

another to support the very broad experimental program. The incorporation of advanced logical concepts, such as neural networks, into this framework would allow even more complex control functions to be envisioned and could also greatly reduce the effort needed to preprogram the control system.

GA is a leader in precision plasma control.

General Atomics (GA) is a leader in developing precise plasma position control capabilities. This has allowed us to utilize a wide range of plasma configurations in our research program and to precisely evaluate their features. The first stages of an advanced control system were recently installed on DIII-D, and this has allowed us to productively expand our capability. The use of time-independent digital algorithms allowed us to sharply increase our ability to switch from one configuration to another even during a single discharge. This system also allowed some important new capabilities to be developed. Control of the plasma energy was accomplished through modulation of the neutral beam input power, and control of the coupling of fast wave power to the plasma was accomplished by controlling the antenna loading by varying the distance between the plasma edge and the antenna.

Plasma control must now integrate control of shaping, current profiles, and disruption avoidance.

The objective of future work on DIII-D is to learn how to implement integrated control of discharge shape, position, density profiles, temperature profiles, current profiles, pressure profiles, and disruptions. The present and future capability and flexibility of the DIII-D tokamak in shaping, heating, and fueling systems make it the ideal facility for the testing of a wide variety of control algorithms. Control of these profile quantities requires both the diagnostic capability to measure the profile in real time and the ability to modify this profile with an external system. The planned rf systems are crucial assets in obtaining these objectives. We anticipate implementing expert systems technology in the development of methods for reliable discharge performance and automated operation.

5.3.8. REMOTE EXPERIMENTAL STATION OPERATION

Remote experimental stations make participation by collaborators more effective.

Future fusion energy research experimental facilities are likely to be much larger and much more costly than current facilities and, thus, there will likely be few of them. Under present operating scenarios, moving personnel from existing facilities to newer facilities is a necessity if those people are to be involved in the experiment, but this can be costly and also disruptive to the families involved. A much different approach is to develop remote experimental stations to allow participation of certain experienced individuals at distant sites. In this way, the major research facility will benefit from the participation of people from many different sites. This concept would allow for real-time data acquisition and analysis of experiments from a remote location, including remote operation of experimental systems such as diagnostics.

LLNL and GA are pioneering the development of a remote experimental station concept at DIII-D. GA is planning to provide the interface hardware and on-site

support at the DIII-D facility. The goal is to enhance the usefulness of, and access to, the DIII-D experiment and to provide a demonstration of mechanisms to be used for future experiments such as ITER or TPX. Full development of the remote participation concept also creates the capability of developing and testing a diagnostic at a remote location before moving it to the primary facility. Then it could continue to be operated from the remote station, so long as primary control and safety functions were maintained at the main site.

Full participation in on-going operation at a central site from a remote station requires three main ingredients: (1) the ability to transfer a significant portion of the newly acquired device data to a remote site on a time-scale comparable to the rate at which it is being collected, (2) the ability to operate hardware systems from the remote station, and (3) the capability for close interactive personal communication between those at the remote station and those at the main site. All of these capabilities currently exist. The challenge is to bring these capabilities up to a level sufficient for full remote participation simultaneously.

The required high-speed transfer of data can be accomplished by the use of existing data transfer technology. High-speed data transfer lines (so-called T3 capability) are expected to be installed between DIII-D and many of the collaborating laboratories in 1993. Using such lines, it is possible to transfer all of the data, if necessary, and carry out a significant amount of interactive computing on a fully interactive basis. It could, however, require a significant fraction of the capability of the line and would be costly. We also plan to do work to implement the storage of data in a universal format and provide the algorithms needed for dealing with data in a wide variety of different formats generated by user hardware and software. Algorithms will also be developed to catalogue the location of various components of the data on different computers within the user network and to expedite speedy location and transfer of the data. Particular attention must be paid to maintaining archival storage of the data on a distributed system of computers to ensure that there is reliable backup and long-term storage.

Operation of hardware systems from a more distant site requires the installation of the necessary data transfer capability, a more extensive system for error checking, and the installation of additional sensors to compensate for the lack of any direct contact. It also requires overriding duplicate controls for the operation from the main site of all primary and safety-related functions. Human interpersonal interaction is the most difficult feature of the operation of a facility to achieve amongst a number of remote sites. Complete teleconferencing between any number of sites is available commercially, although it appears to be far from ideal when used to communicate between more than two points. It is also somewhat expensive in its present form. A productive first step will be to implement an electronic bulletin board that is rapidly updated with current operational information.

5.3.9. RADIATION SHIELDING INCREASE

Increased neutron production at the machine will require increased shielding and result in increased radiation levels of the site boundary.

The current DIII-D radiation production at the site boundary is less than 6 mRem/year, a third of the present DOE guideline limit of 20 mRem/yr. However, the planned 18 weeks of double-shift operation and AT operating modes with pulse lengths of 10 to 60 sec would produce radiation at the site boundary exceeding this guidance with the present shielding. We, therefore, plan in the next contract period to increase the radiation shielding. In addition, we will request an increase in the site boundary limit imposed by DOE from 20 to 50 mRem/yr, which would still be significantly below the limit imposed by the State of California of 500 mRem/yr and the anticipated limit of 100 mRem/yr in 1994.

This increase in the neutron production would also increase the on-site radiation, but we would still be able to operate the facility within GA and governmental guidelines for on-site personnel. Some changes in operating procedures would be required. Entry into the vessel during vents would have to be delayed until the activation has decreased to an acceptable level, and access to the machine after a run period would have to be restricted to keep the radiation level to personnel below the accepted levels. This would affect repair and construction near the machine.

During the first year, the radiation pattern will be evaluated and the conceptual design of the increased shielding will be initiated. During the second year, the engineering of the shielding will be completed, and the construction will take place during the third and fourth year so that it will be in place when most of the auxiliary long-pulse heating power becomes operational.

5.3.10. BUILDINGS

GA will provide needed building modifications.

GA will modify its buildings in order to carry out the planned research work. We have evaluated the space and power demands of all the planned upgrades to ensure that they can be accommodated. Our plan is to expand the main tokamak building by 10,000 ft² and to increase the control room by approximately 25%. Figure 5.3-5 shows the planned layout of the machine building.

The following changes need to be made to the DIII-D site:

1. Five power supply modulator regulator and crowbar systems for ECH will be located in the area outside the ECH vault.
2. Two additional power supplies for the lower hybrid heating system will be located in the existing pipe shop area.
3. Six Klystron (4 × 8 ft) cabinets for the lower hybrid system will be located in the water skid area.
4. Ten gyrotrons (1 MW) will be placed in the present ECH vault or in a new extension on the north end of the machine building.

Fig. 5.3-5. Proposed building machine and layout.

5. Two fast wave transmitters will be located either in existing machine building office areas or in a north building extension.
6. An area for seven MFTF transformer, rectifier, main switchgear subsystems, and four fast-wave power supplies will be sited outside the building at the northwest site boundary.

7. The existing ICH transmission line and phase shifter. system will be moved to a new mezzanine to be constructed over the vacuum laboratory, along with two new sets of phase shifters and related components for the new antennas.
8. Three new ICH antennas will be located in the vacuum vessel at 0° , 180° , and $285^\circ/300^\circ$, with a possible lower hybrid antenna at $255^\circ R0$.
9. A pellet injector will be located just outside the tokamak at $135^\circ R0$.
10. Three power supplies for the correction coil (C-coil) will be located in the current chopper test-stand area of the E-power supply building. The switch gear for the unused opening switch in the ohmic heating system may also be repackaged for these supplies.
11. A preliminary design for C-coil located at the machine midplane is currently being reviewed.
12. A water system upgrade (including a new cooling tower) will require more space outside the machine building.
13. Two transformers will be added to the 138 kV substation in an existing area.
14. Additional neutron shielding will either be added to existing shielding or a new shield will be installed.
15. Additional office space will be required for people displaced by the annexation of office areas for new equipment and those needed to support additional capabilities.
16. Additional space is required for the relocation of the pipe shop and technician work areas, possibly requiring an additional building.
17. The ICRF dummy loads will be located in the north end lower level.
18. There will be 27 new diagnostics installed on the machine. Rearrangement and consolidation of some existing diagnostic systems will be required.
19. The vacuum lab, storeroom, clean room, and offices will be renovated and a mezzanine added.

6. RELATIONSHIP TO THE U.S. PROGRAM

DIII-D first began operation in 1986. The DIII-D Tokamak Long Range Plan of July 1987 has been continually modified based on technical findings and in response to changing national magnetic fusion energy program strategy. The relation of the DIII-D program to the dynamic magnetic fusion strategy is summarized each time the DIII-D Tokamak Long Range Plan is modified.

6.1. RELATION TO THE NATIONAL ENERGY STRATEGY

The recently developed National Energy Strategy [6-1] lays out goals and a strategy for the development of magnetic fusion energy. This strategy, based on recommendations of the Fusion Policy Advisory Committee (FPAC) [6-2], recommends that the United States develop fusion as a potential energy source and that the fusion program take maximum advantage of international collaboration and increase participation by industry. The National Energy Strategy for Magnetic Fusion Energy (MFE) is shown in Fig. 6.1-1. The ITER engineering design activity is being formed. Due to national budgetary constraints, the BPX experiment has been canceled and the U.S. national program is now focused on the steady-state experiment.

The General Atomics (GA) fusion program has been a major contributor for more than three decades to the significant programs in developing innovative fusion concepts, in extending plasma parameters to reactor conditions, in increased understanding and predictability of reactor plasma regimes, in advancing fusion technology, and in the refinement of magnetic fusion reactor concepts. Today, DIII-D is the central element of the GA fusion program and has the capability of being a principal contributor to the development of magnetic fusion in the years to come. A summary of the broad range of physics issues that are under research in DIII-D is given in Table 6.1-1 and Fig. 6.1-2 (from the FPAC report [6-2]).

In relation to the national fusion energy development strategy of Fig. 6.1-1, the DIII-D program is a primary element of the core program. It will be a key facility supporting the ITER objectives throughout the 1990s. Its primary mission will be to support the ITER and steady-state advanced tokamak TPX mission. The high β confinement and radio frequency (rf) programs support the steady-state advanced tokamak TPX mission. The divertor physics and technology programs support both ITER and TPX and provide non-nuclear divertor data to the materials program. The Fusion Energy Advisory Committee (FEAC) recently recommended that a strong DIII-D program be an element of the U.S. Magnetic Fusion Energy Program.

Fig. 6.1-1. Magnetic fusion energy development strategy (February 1991).

6.2. RELATION TO THE MAGNETIC FUSION PROGRAM PLAN

The long-range DIII-D program forms an integral part of the total generic long-term U.S. fusion research effort. This effort is based on the key technical issues and strategic objectives defined by the Magnetic Fusion Program Plan (MFPP) [6-3], and the programmatic implementation of the MFPP defined by the final report of the Technical Planning Activity (TPA) [6-4]. While this plan is out-dated by research progress and programmatic events, it still provides a framework for fusion research planning.

The U.S. fusion program will undertake a number of major steps on the way to the assessment of the viability of controlled fusion as a commercial power source. The TPA has developed a methodology for proceeding to an evaluation of the tokamak reactor. The MFPP identified four key technical issues to be addressed:

- **Magnetic Confinement Systems.** Developing an understanding of the plasma science concepts suitable for commercial application of fusion energy.
- **Properties of Burning Plasmas.** Understanding the effects introduced when the plasma is internally heated by the fusion reaction.

TABLE 6.1-1
PRINCIPAL FACILITIES CONTRIBUTING TO TOKAMAK PHYSICS ISSUES (FPAC)

	HIGH CURRENT TOKAMAKS									
	TFTR	DIII-D	C-MOD	JET (EUROPE)	JT-60U (JAPAN)	T-15 (USSR)	ASDEX-U (FRG)	TORUS (FRANCE)	FT-U (ITALY)	
GOAL	Break-even with D-T fusion physics	High-B/good confinement with non-inductive current drive	RF Heated High-B/plasma physics	Break-even with D-T fusion physics	Break-even physics D-D	Large tokamak physics	Impurity control with reactor relevant divertor	Physics & technology of long pulse operation	High field/high n physics	
Operation	1982	1982	1982	1982	1982	1982	1982	1982	1982	
APPROACH	Circular, Superhot, High power/short pulse	U-shaped, H-mode, ECH, Divertor	D-shaped, Divertor, High B, ICRF	D-shaped, Pumped divertor, High power/long pulse	D-shaped, Diverted, High power/long pulse	Circular, ECH power	Shaped, Divertor	Circular, Ergodic magnetic divertor, Long pulse	Circular, with high power LH	
CURRENT	3MA	3.5MA	3MA	7MA	6MA	1.4MA	1.6MA	1.7MA	1.6MA	
CONFINEMENT	High-Temp. Fluct. diagn.	H-mode physics Fluct. diagn.	High-dens. High-field	High temp.	High temp.	---	---	---	High-dens. High-field	
STABILITY	1982 High- β	First stab. High elong. High- β	---	First stab.	---	---	First stab. coils	---	---	
POWER HANDLING & PARTICLE EXHAUST	High power density	High power density divertor with Bias and Pumped	Open/closed divertor High-Z	Pumped divertor	Divertor	---	Open divertor 1994	Ergodic & pumped limiter Long pulse (30 sec)	---	
ADVANCED HEATING	ICRF	ECH	ICRF	ICRF	Neg. ion beam (?)	ECH	ICRF	ICRF	LH	
CURRENT DRIVE	Neutral beam Bootstrap	ECCD/FWCD NBI	---	Lower hybrid (edge)	Lower hybrid (core) 1993	Electron cyclotron	---	Lower hybrid Long pulse 1994	Lower hybrid	
ALPHA-PARTICLE PHYSICS	Q = .5-1 Alpha instab. 1993-94	Beam Simulation	---	Q = .5-1 Alpha instab. 1993-94	---	---	---	---	---	

U.S. FACILITIES

FOREIGN FACILITIES

Items in Bold = Major Contributions

- **Fusion Nuclear Technologies.** Developing nuclear technologies unique to the commercial application of fusion energy.
- **Fusion Materials.** Developing materials that will enhance the economic and environmental potential of fusion.

This process is summarized in a set of logic diagrams for the tokamak program, shown in Fig. 6.2-1. The squares labeled M are milestones, and the diamonds are decision points. The DIII-D program directly addresses many of these issues. The GA steady-state tokamak experiment (STE) study addressed the definition of the advanced geometry and steady-state experiments. This steady-state activity has now been taken on as the central theme of the U.S. national fusion program by the new initiative and tokamak physics experiment (TPX) process.

The TPX and ITER designs include a strongly shaped plasma, with an elongation of about 2 and a poloidal divertor. The divertor is needed for impurity control and assure H-mode confinement. Both TPX and ITER depend on the results from

Fig. 6.1-2. A FPAC summary of the broad range physics issues under research in DIII-D.

DIII-D to demonstrate the required high confinement at high β .

A U.S. National Review of ITER was conducted in March 1991 (Baldwin Committee). The committee arrived at six high-priority issues to be resolved (see Table 6.2-1). Indicated in Table 6.2-1 is the DIII-D program response. A checkmark indicates that the activity is funded. An underline indicates an unfunded activity for which supplemental funding has been requested by GA.

Both TPX and ITER would benefit from improved understanding of transport. The DIII-D program has an active program in transport, particularly in the high β H-mode and VH-mode physics.

The DIII-D program is carried out with DOE milestones. The DIII-D milestones are given in Table 6.2-2.

Fig. 6.2-1. Level 2 logic diagram for tokamak, research plan (TPA).

REFERENCES

- 6-1. National Energy Strategy, First Edition, 1991/1992.
- 6-2. Fusion Policy Advisory Committee Final Report, September 1990.
- 6-3. Magnetic Fusion Program Plan, U.S. Department of Energy, Office of Fusion Energy, DOE/ER-0214, February 1985.
- 6-4. Technical Planning Activity Final Report, Argonne National Laboratory Report ANBL/FPP-87-1, January 1987.

TABLE 6.2-1
CRITICAL ITER PHYSICS R&D ISSUES CAN BE ADDRESSED
WITH DIII-D CURRENT DRIVE AND ITER R&D DIVERTOR PROGRAMS

Critical Physics Issue (Identified by U.S. National Review of the ITER CDA, March 1991 — Baldwin Committee)	DIII-D FTP Response (✓ = base program; — = <u>incremental program</u>)
<ul style="list-style-type: none"> • Disruptions <ul style="list-style-type: none"> — Characterize and model — Active control e.g. high power rf heating and current drive • Divertor head loads <ul style="list-style-type: none"> — Develop understanding — Emphasize current drive with divertor operation • Helium ash removal from H-mode <ul style="list-style-type: none"> — Deep helium implantation — Edge helium removal • Alpha particle predicted instability <ul style="list-style-type: none"> — Simulate with neutral beams • Aspect ratio H-mode scaling data <ul style="list-style-type: none"> — Experimental data needed • Efficient current drive and plasma profile control <ul style="list-style-type: none"> — More studies of neutral beam and bootstrap current drive — Experiments with adequate rf current drive power 	<ul style="list-style-type: none"> ✓ — Dedicated staff and runtime — <u>High power rf necessary</u> ✓ — Advanced divertor program (ADP) — <u>High power rf and ITER R&D divertor necessary</u> ✓ — Helium beam operation — ADP pumping and <u>ITER R&D divertor necessary</u> — <u>High power ECH</u> and ADP extends operating space ✓ — Runtime and JT-60U collaboration — <u>High power ECH</u> extends operating space — <u>High power rf necessary</u>

TABLE 6.2-2
DIII-D MILESTONE SCHEDULE

	Task Description	Approved (Target) Date	Completed Date
70.0	Inject pellets into DIII-D H-mode plasma (with ORNL)	8/93	TBD
76.0	Begin DIII-D 110 GHz ECH experiments with 4 gyrotron system	5/92	
78.0	Report on DIII-D ECH and fast wave current drive using 110 GHz ECH system for ITER	11/92	
79.0	Report on divertor characteristics, including pumping for ITER	12/92	
80.0	Report on results of divertor bias experiments	5/92	5/92
81.0	Report on results of 110 GHz ECH experiments	6/92	TBD
82.0	Report on turbulence studies with the lithium beam diagnostic (TTF)	12/92	
83.0	Install ITER related multi-element ORNL fast wave current drive antenna	9/93	
84.0	Complete FMIT modification of 2 MW FMIT transmitter (ITER related)	9/93	
85.0	Provide initial analysis of characterization of vertical displacement events during disruptions for ITER	11/91	11/91
86.0	Generate boundary physics database for ITER	11/91	11/91
87.0	Evaluate plasma performance over a range of aspect ratios for ITER	3/92	5/92
88.0	Characterize ITER related operational limits	8/92	8/92
89.0	Complete initial ITER related TAE mode studies	8/92	8/92
90.0	Complete ITER related characterization of plasma and divertor performance with long pulse ELMy H-mode operation	TBD	
91.0	Complete in-vessel cryopump and outer wall tile installation	2/93	
92.0	Report on plasma operation using outer wall tiles	5/93	
93.0	Report on results of divertor cryopump	7/93	
94.0	Complete report on DIII-D ITER R&D tasks	10/93	
95.0	Complete report on preliminary radiative divertor designs	12/93	
96.0	Complete checkout of second 2 MW fast wave transmitter	2/94	
97.0	Report on results of FWCD additional 2 MW system	7/94	
98.0	Complete report on DIII-D ITER R&D tasks	10/94	

(b) Awaiting OFE to approve new milestone and/or date change/or completion.